September 27, 2002

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 11 - AP1000 DESIGN CERTIFICATION REVIEW (TAC NO. MB4683)

Dear Mr. Corletti:

By letter dated March 28, 2002, Westinghouse Electric Company (Westinghouse) submitted its application for final design approval and standard design certification for the AP1000.

The Nuclear Regulatory Commission (NRC) staff is performing a detailed review of your design certification application to ensure that the information is sufficiently complete to enable the NRC staff to reach a final conclusion on all safety questions associated with the design before the certification is granted.

The NRC staff has determined that additional information is necessary to continue the review. The topics covered in the requests for additional information (RAIs) contained in Enclosure 1 include the areas of reactor systems, radiological consequence of postulated design-basis accidents, and reliability and risk assessment. These RAIs were sent to you via electronic mail on September 20 and September 24, 2002. You agreed that Westinghouse would submit a response to these RAIs by December 2, 2002. Receipt of the information by December 2, 2002, will support the schedule documented in our letter dated July 12, 2002.

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

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

Sincerely,

/RA/

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

Docket No. 52-006

Enclosure: As stated

cc: See next page

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Request for Additional Information (RAI)
AP1000 Standard Design Certification
Series 440 - Reactor Systems
Series 451 - Meteorology
Series 470 - Radiological Impact
Series 720 - Reliability and Risk Assessment

Series 440 - Reactor Systems

Reference: WCAP-15833,"WCOBRA/TRAC AP1000 ADS-4/IRWST Phase Modeling," Revision 1, July 2002

440.149

Please clarify the following:

(a) In the Equation Nomenclature, on page xii of WCAP-15883, "Re" is defined as a Reynolds number. However, on page 2-10, it is used as an entrainment source. Please revise the Nomenclature as appropriate.

(b) On page 2-11 of WCAP-15833, Equation (2-22) appears to be missing an "=" sign. Please indicate the correct expression.

440.150

Section 2.2.1.1 documents the Side Offtake Orientation model for predicting the onset of liquid entrainment. Describe under what circumstances this model is applied in the analysis of the AP1000 or in the associated code validation. If this model is used, provide suitable justification by comparing the correlation to experimental data and state the valid range of thermal-hydraulic conditions over which the validation is applicable.

440.151

Section 2.2.1.5 states that the quality (*x*) entering the automatic depressurization system stage 4 (ADS-4) branch line from the hot leg is calculated using,

$$x = R^{3.25(1-R)^2} \qquad (1)$$

where, R = (h/h_b). Here, h is the distance between the top of the main pipe and the liquid surface, and h_b is the critical distance for entrainment onset. The onset of entrainment is obtained from an expression based on the gas phase Froude number (*Fr*) at the branch line inlet,

$$Fr_g = \frac{U_g}{\sqrt{\frac{gd\Delta\rho}{\rho_g}}} = C_1\left(\frac{h_b}{d}\right)^{C_2} \qquad (2)$$

Various values of C_1 and C_2 have been proposed by different investigators, several of which are listed in Table 1.

Table 1

Reference	C_1	C_2
Anderson and Benedetti [1]	0.35	2.5
Rouse [2]	5.67	2.0
Schrock, et al. [3]	0.395	2.5
Smoglie [4]	0.355	2.5
Maciaszek and Micaelli [5]	1.75	1.5

Equations (1) and (2) are based on experiments in which the branch line diameter is small in comparison to the diameter of the horizontal pipe. In general, the ratio of the branch line to the main pipe was less than 0.1 in the development of these correlations.

The coefficients used in WCOBRA/TRAC-AP are those of Anderson and Benedetti. The TRAC-M code also uses Equations (1) and (2) but with the coefficients by Smoglie. However, the coefficients by Smoglie and by Anderson and Benedetti are nearly identical, and the WCOBRA/TRAC-AP and TRAC-M models generate the same results. Figure 1 shows the predicted variation of branch line quality as a function of the branch line Froude number assuming the liquid level in the main pipe is at the midplane (h/D = 0.5) for two cases in which the ratio of the branch line to main pipe diameter is large compared to the database used to determine the coefficients in Table 1. ATLATS is a separate effects test facility with dimensions scaled to those of the AP600 that was used to investigate phase separation at the junction between a small branch line and a main pipe. The ratio of the branch line to the main pipe in ATLATS is d/D = 0.33, which is less than the AP1000 ratio of d/D = 0.47. Calculations for both the ATLATS test facility geometry and the AP1000 hot leg and ADS-4 junction are also shown. As the ratio (d/D) increases, the model predicts lower branch line qualities. However, in neither case are the predicted branch line qualities in reasonable agreement with the experimental data shown in Figure 1. These data are from ATLATS tests, with the equilibrium level at approximately h/D = 0.5, consistent with the calculations.

The comparison suggests that the model and coefficients of Equations (1) and (2) and Table 1, grossly overpredict the ADS-4 quality for conditions expected in the AP1000. Please provide suitable justification for Equations (1) and (2) and their coefficients for the large d/D ratios in the AP1000 design. Provide justification that the phase separation equations used in the WCOBRA/TRAC-AP code are appropriate for the AP1000 ADS-4 analysis in light of these data.

References

[1] Anderson, J. L., and Benedetti, R. L., “Critical Flow Through Small Pipe Breaks,” EPRI/NP-4532, 1986.

[2] Rouse, H., “Seven Exploratory Studies in Hydraulics,” J. Hydr. Div. Proc. ASCE, HY4, pp (1038) 1-35, August, 1956.

[3] Schrock, V. E., Revankar, S. T., Mannheimer, R., and Wang, C-H., “Small Break Critical Discharge -- The Roles of Vapor and Liquid Entrainment in a Stratified Two-Phase Region Upstream of the Break,” NUREG/CR-4791, 1986.

[4] Smoglie, C., “Two-Phase Flow Through Small Branches in a Horizontal Pipe with Stratified Flow, Ph. D. Dissertation, Univ. of Karlsruhe, 1984.

[5] Maciaszek, T., and Micaelli, J. C., “The CATHARE Phase Separation Model in Tee Junctions,” SETh/LEML-EM/89-159, 1986.

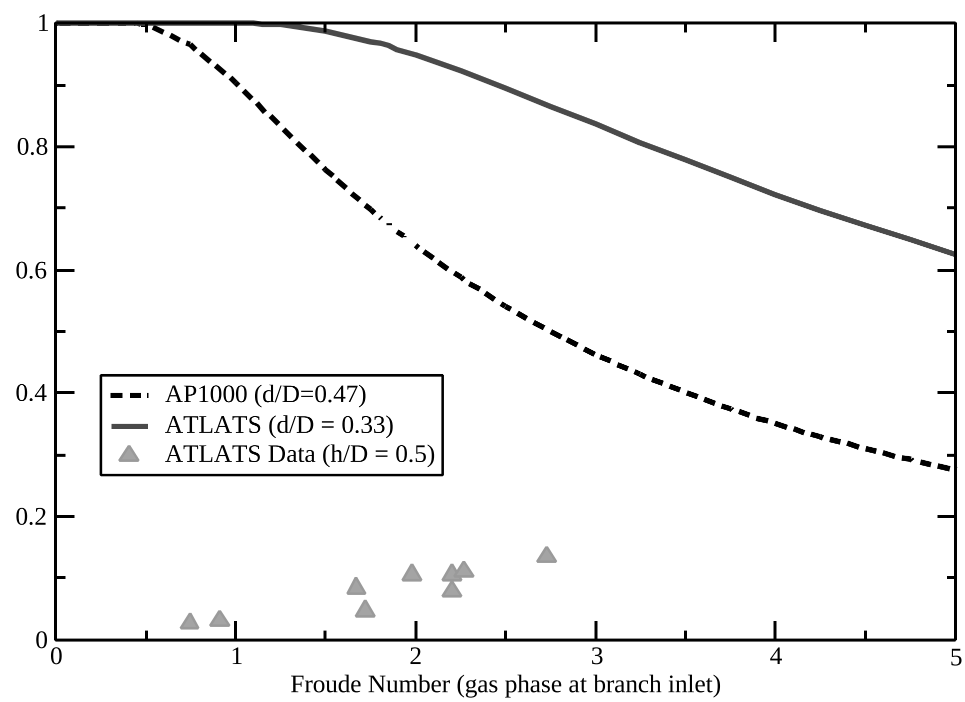

Figure 1. Branch line quality for liquid level at main pipe midplane for ATLATS and AP1000.

440.152

In Section 2.2.1.5 application of Entrainment/Vapor Pull-through Model is described. Under the "Model as Coded" subsection Step 6 is a branch line void fraction calculation. Please describe the slip or other models at the branch line junction to obtain a void - quality relationship. Since WCOBRA/TRAC-AP also calculates entrainment from a horizontal stratified flow using the models described in Section 2.2.1.4, describe how the liquid flow rate at the junction is determined from the entrained and continuous liquid fields.

440.153

In Section 2.2.1.5, the "Model as Coded" subsection describes calculation of the branch line flow quality using the Entrainment/Vapor Pull-through model when the main pipe flow is stratified. How is the branch line quality determined when the flow pattern is different than stratified?

440.154

Section A.4.4 presents results of a sensitivity study using the Top Offtake Orientation model. In general, the onset for entrainment for a top offtake configuration is:

$$Fr_g = \frac{U_g}{\sqrt{\frac{gd\Delta\rho}{\rho_g}}} = C_1\left(\frac{h_b}{d}\right)^{C_2} \qquad (1)$$

where d is the diameter of the branch line.

In the sensitivity study, the values of C_1 and C_2 were varied from their reference values of C_1=0.355 and C_2=2.5. The results were intended to show that the AP1000 performance for the Inadvertent ADS case has little sensitivity to the Top Offtake Orientation correlation.

(a) Since hot leg entrainment is sensitive to steam velocities in the hot leg, show why the Inadvertent ADS case is sufficient to characterize or bound the AP1000 performance for other accident scenarios such as a small cold leg break or a direct vessel injection (DVI) line break.

(b) When either the coefficient C_1 is increased or the exponent C_2 is decreased in the sensitivity study, the minimum in-vessel inventory increases and the start of in-containment refueling water storage tank (IRWST) injection is delayed compared to the Reference case. However, when C_1 is decreased, the vessel inventory minimum again increases and the IRWST injection is delayed. Please explain why both increasing and decreasing the onset of entrainment results in this sensitivity.

(c) Provide justification that the variation of the Top Offtake Orientation model in Section A.4.4 is sufficient to account for inaccuracies in the model when compared to experimental data. Figure 2 shows the variations considered in the sensitivity study. When data from the ATLATS tests are compared to the correlation and its variations in the sensitivity study, it can be seen that the branch line quality is grossly over-estimated for the two cases in which the offtake quality was reduced compared to the Reference. For example, at Froude numbers less than one the data suggests very low branch line quality while the correlation and its variations predict very high quality (*x* >0.75).

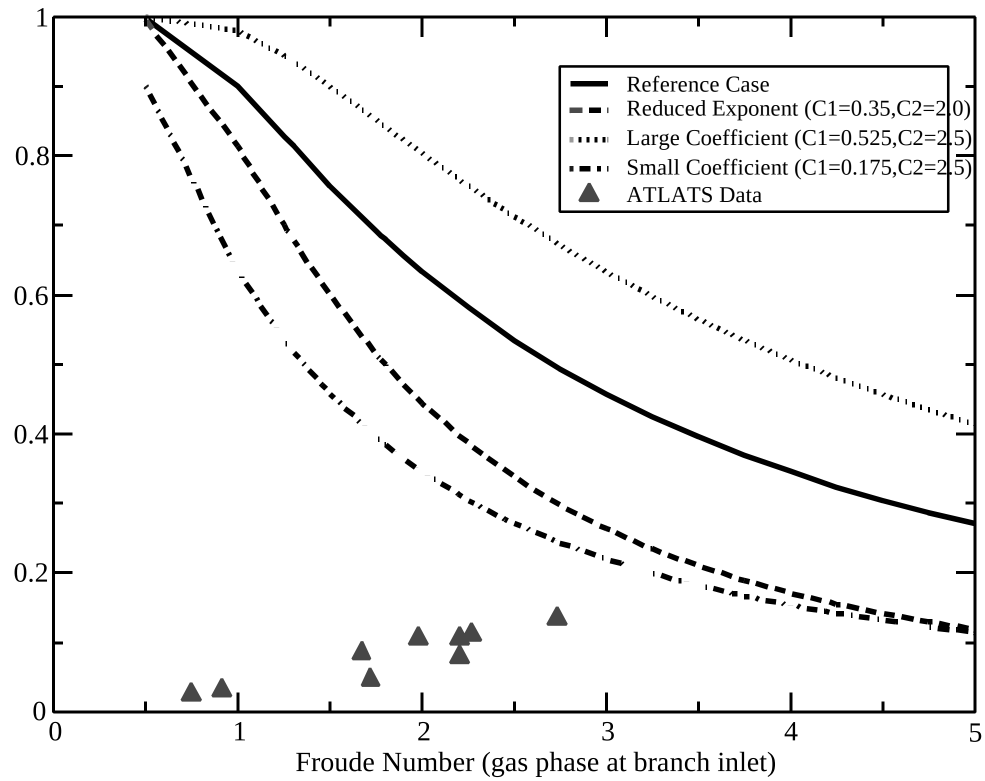

Figure 2. Variation in Branch Line Quality of Entrainment Onset Correlations in Westinghouse Sensitivity Study.

440.155

Section 2.2.1.2 of WCAP-15833 describes the Horizontal Flow Regime Map in WCOBRA/TRAC-AP that is used elsewhere in the report to predict the flow pattern in the hot legs of the AP1000 and the APEX test facility. The flow pattern map is that proposed by Taitel and Dukler [1], which is frequently used in reactor safety analysis.

(a) While this flow pattern map has been successful in predicting the flow patterns for co-current horizontal flow, it has not been established that this map is appropriate to characterize the patterns that may develop in the region between the ADS-4 branch line and the steam generator (SG) inlet plenum. In this region, liquid may stagnate and become trapped. An oscillating plug of liquid may occur in this region because of insufficient gas flow to sweep the liquid through the SG or into the branch line. Justify the use of the Taitel-Dukler flow pattern map in this region.

(b) The Taitel-Dukler flow pattern map uses the Martinelli parameter in the flow pattern selection. Describe how the pattern is selected if the liquid flow rate becomes zero, or the flow is countercurrent.

Reference

[1] Taitel, Y., and Dukler, A. E., 1976, "A Model for Predicting Flow Regime Transitions in Horizontal and Near Horizontal Gas-Flow," AIChE Journal, Vol. 22, No. 1, pp. 47-55.

440.156

Section 2.2.1.4 describes the equations used to determine the onset of entrainment in horizontal stratified flow. Equations (21) and (22) in the original reference by Ishii and Grolmes [1] correlate the onset using the superficial gas velocity, not the local velocity. Please verify that the as-coded expressions for entrainment listed (un-numbered) on page 2-10 of WCAP-15833 use superficial velocity for U_g and not the local velocity (as the Nomenclature suggests).

Reference

[1] Ishii, M., and Grolmes, M., 1975, "Inception Criteria for Droplet Entrainment in Two-Phase Concurrent Film Flow," AIChE Journal, Vol. 21, No. 2, pp. 308-318.

440.157

As described in Section 2.2.1.4 for horizontal stratified flow, the correlation by Tatterson, et al. [1] is used to determine the size of entrained droplets. In the original reference, it was recommended that the "volume median diameter" be approximated as,

$$\frac{D_d}{D_t}\left(\frac{\rho_g U_g{}^2 f_s D_t}{2\sigma}\right)^2 = 0.016 \qquad (1)$$

where D_d is the volume median diameter, D_t is the channel hydraulic diameter, and f_s is the friction factor for a smooth interface.

(a) The coefficient in Equation (2-20) of WCAP-15833 does not appear to be correct. Please demonstrate that Equation (2-20) is equivalent to the expression recommended by Tatterson, et al. [1].

(b) Provide justification for using the hydraulic diameter of the gap above the mixture elevation as the characteristic length instead of the channel hydraulic diameter as recommended by Tatterson, et al.

(c) For the velocities and conditions in the hot legs of the AP1000 and the APEX tests for which Equation (2-20) is applied, provide justification that the predicted drop diameters are in reasonable agreement with experimental data. Note that Tatterson, et al. did in fact list drop diameter for horizontal flows for an air-water system and the flow rates.

Reference

[1] Tatterson, D. F., Dallman, J. C., and Hanratty, T. J., "Drop Sizes in Annular Gas-Liquid Flows," 1977, AIChE Journal, Vol. 23, No. 1, pp. 68-76.

440.158

Specify the steam flow rate for the liquid flows shown in Figure 2-7 and provide a key for the curves in this figure. (For example, there are two curves with open triangles and solid lines, and there are two curves with "X" and a solid line.)

440.159

What is Figure 2-8 attempting to show? Are the points in Figure 2-8 the liquid level at the 1.233 meter (m) axial position in Figure 2-7?

440.160

Please provide Figures 2-9, 2-11, and 2-13 that include a key, or clearly indicate what the individual curves represent. In addition, explain why there are periodic oscillations in the steam flow rate (see dashed curves in Figure 2-13) although the pressure differentials along the channel remain relatively constant (see Figure 2-11).

440.161

Please provide Figure 2-15 with a key, or clearly indicate what each of the three curves represent. In addition, clarify the as-coded expression for the condensation heat transfer coefficient. (On page 2-15, it is claimed that for condensation heat transfer the "dependence on x is neglected" in the WCOBRA/TRAC-AP model, but on 2-21 it is stated that condensation heat transfer "rises with distance proportional to $x^{0.1}$.")

440.162

Section 2.3.2 provides an assessment of WCOBRA/TRAC-AP using APEX Test SB18. This is a small cold leg break with a simulated failure of one of the ADS-4 lines. Please provide information for the following:

(a) For the comparison of predicted and measured pressurizer levels shown in Figure 2-29, justify the claim that the WC/T level agrees "extremely well" with the data through 1150 seconds, although WC/T clearly underpredicts the level for most of this period and does not capture the oscillations in level that are seen in the data.

(b) The predicted collapsed liquid levels in the downcomer, core, and upper plenum for Test SB18 are shown in Figures 2-31, 2-32, and 2-33, respectively. On page 2-52 the claim is made that the relatively constant code predicted levels are "consistent with the test data." However, no test data are presented in these three Figures. Please provide a meaningful comparison of predicted and measured results to validate this claim.

(c) Page 2-52 describes a “detailed comparison of vessel mass inventory with the test inventory” to show that the WCOBRA/TRAC prediction is in “excellent” agreement with the measured mass reduction during the ADS-4/IRWST initiation phase. There are no Figures comparing the predicted and measured inventories for Test SB18. Please provide this comparison.

(d) Section 2.3.2 concludes that the WCOBRA/TRAC prediction is in reasonable agreement with Test SB18 data and the code can be used in the AP1000 calculations. This conclusion is reached with only three comparisons between the predicted and measured results; pressurizer level in Figure 2-29, integrated liquid flow in Figure 2-30, and downcomer pressure in Figure 2-34. Since the system pressure is primarily set by input to the BREAK Components in the model, Figure 2-34 may not be a true indication of code performance. Section 2.2.2 in WCAP-15833 showed that condensation heat transfer is underpredicted and steam flow rates in the hot leg are overpredicted. An overprediction of steam velocities in the hot legs for Test SB18 would result in an overprediction of ADS-4 flows. Thus, the apparently reasonable agreement in Figure 2-30 ADS-4 flow may be right for the wrong reasons. It remains to be shown therefore, that the simulation of Test SB18 is reasonable in comparison to experimental data and free of compensating errors. Please provide sufficient comparisons between predicted and measured results to demonstrate adequate simulation of Test SB18. Included in the comparisons and evaluation of code performance should be ADS-4 steam and liquid flows (not just the total integral) ADS-4 quality, hot leg levels, upper plenum two-phase level, and fluid temperatures throughout the system. Provide information sufficient to characterize how WCOBRA/TRAC predicted entrainment in the upper plenum and hot legs during the simulation of Test SB18.

440.163

Sections 3.2.1 and 3.2.2 of WCAP-15833 describe WCOBRA/TRAC simulations of a Double-Ended DVI (DEDVI) Break and an Inadvertent ADS Actuation in the AP1000. Both simulations show significantly higher liquid flows through the ADS-4 at the ADS-4/IRWST initiation phase of the transients. No information is provided however, on the specific cause of the predicted low ADS-4 flow quality. Please provide information to detail entrainment processes as predicted by WCOBRA/TRAC in this simulation. Of particular interest are steam and liquid flows at the core exit, lateral flows of continuous liquid, entrained liquid, and steam from the upper plenum to the hot legs, lateral flow of each field in the hot legs just upstream of the ADS-4/hot leg junction, hot leg collapsed water level, collapsed liquid level at the SG inlet (Channels 26-28), the predicted hot leg flow pattern, and axial void distribution in the hot leg.

440.164

Figure 3-16 shows the vessel mass inventory for the DEDVI Break and Figure 3-25 shows the vessel mass inventory for the Inadvertent ADS Actuation scenario as predicted by WCOBRA/TRAC. Since there is less mass in the vessel, there is the possibility that there is less mass in the core and void fractions at the top of the active core may be high. In previous NOTRUMP calculations, the void fraction at the top core was found to exceed 0.90. Therefore, please provide figures showing the axial void gradient in the AP1000 predictions at the time of minimum vessel inventory. Also provide void fraction variation with time for several hydraulic cells in Channel 10, including the top cell.

440.165

Both the DEDVI and Inadvertent ADS Actuation calculations in Sections 3.2.1 and 3.2.2 show flow qualities of roughly 0.20 (plus/minus 0.1) in the ADS-4 lines for the brief period simulated. Provide justification that Test SB18, which was used to demonstrate that WCOBRA/TRAC can adequately predict ADS-4 flows, has approximately this ADS-4 flow quality.

460.166

The comparisons between WCOBRA/TRAC and NOTRUMP in Section 3 show very large differences in results between the two codes. Why should it be expected that the starting conditions for the WCOBRA/TRAC simulations be the same as those from NOTRUMP? The large deviations suggest that if WCOBRA/TRAC were used to predict the initial parts of each transient, the initial conditions for the WCOBRA/TRAC simulation would be significantly different. If WCOBRA/TRAC were used to simulate the entire transient, why should it not be expected that there be considerably less vessel mass at the start of ADS-4/IRWST transition given the much higher ADS-4 flow rates predicted by WCOBRA/TRAC?

440.167

Section A.2.1 discusses scaling for entrainment in the hot legs of the AP1000. The scaling ratio is apparently based on the entrainment onset correlation listed in WCAP-15613, “AP1000 PIRT and Scaling Assessment,” February 2001, as Equation 4-90:

$$\frac{U_g\sqrt{\rho_g}}{\sqrt{g\left(\rho_f-\rho_g\right)Lg}} \geq 5.7\left(\frac{L_g}{d_{offtake}}\right)^{3/2} \qquad (1)$$

This expression is based on the work of Rouse as described by Zuber [1].

In Sections 2.2.1.4 and 2.2.1.5 for WCOBRA/TRAC however, Westinghouse decided to use different expressions for hot leg entrainment, presumably because those expressions are more accurate. Using Equation 2-24 and the coefficients in Equation 2-25 of WCAP-15833 will lead to a different conclusion on hot leg onset scalability. Note that the exponent in the above expression becomes important in the scaling ratio Westinghouse defined in Equation 4-92 of WCAP-15613. Starting with Equation 4-92, one could alternately write the scaling ratio as,

$$\left[\frac{U_g}{\sqrt{L_g}}\right]_R\left[\left(\frac{d_{offtake}}{L_g}\right)^m\right]_R = 1.0 \qquad (2)$$

or,

$$\left[\frac{U_g\, d_{offtake}^m}{L_g^{(m+0.5)}}\right]_R = 1.0 \qquad (3)$$

Using Equation 4-94 of WCAP-15613 for U_g, this becomes,

$$\left[\frac{q_{core}\, d_{offtake}^{m-2}}{L_g^{m+0.5}}\right]_R = 1.0 \qquad (4)$$

or, if $d_{offtake} = d_{ADS}$ and $L_g = D_{HL}$ then the scaling ratio is,

$$\left[\frac{q_{core}\, d_{ADS}^{m-2}}{D_{HL}^{m+0.5}}\right]_R = 1.0 \qquad (5)$$

Scaling the AP1000 and the APEX facility then,

$$\Pi_R = \frac{\left[\frac{q_{core}\, d_{ADS}^{m-2}}{D_{HL}^{m+0.5}}\right]_R}{\left[\frac{q_{core}\, d_{ADS}^{m-2}}{D_{HL}^{m+0.5}}\right]_R} = \left(\frac{q_{core,APEX}}{q_{core,AP1000}}\right)\left(\frac{d_{ADS,APEX}}{d_{ADS,AP1000}}\right)^{m-2}\left(\frac{D_{HL,AP1000}}{D_{HL,APEX}}\right)^{m+0.5} \qquad (6)$$

Using an exponent of 2.0, which is consistent with the expression in WCOBRA/TRAC, and using parameters for the APEX and the AP1000 designs, this gives,

$$\Pi_R = \left(\frac{\frac{1}{96}}{1.75}\right)\left(\frac{1.61}{14.438}\right)^0\left(\frac{31}{5}\right)^{2.5} = 0.57 \qquad (7)$$

This is just within the range of acceptability as claimed, but not as good as with the smaller exponent.

A revised argument for hot leg entrainment scaling should be provided. In addition, provide justification on why one correlation is more appropriate for calculation of entrainment onset while a different one is more appropriate for scaling unless the correlations are made consistent in Sections 2.2.1.5 and A.2.1.

Reference

[1] Zuber, N., NUREG-0724, “Problems in Modeling of Small Break LOCA,” 1980.

440.168

Please provide suitable comparison between measured and predicted results to support the claim that for a Froude number of 3.5 the difference between data and the Craya prediction is only approximately 1 percent. Soliman and Sims [1] report that for Fr < 10 there are "significant deviations," and Figure 5 of their paper does not appear to support such close agreement. In addition, please explain why it can be concluded that a Top Offtake Orientation model should be independent of offtake diameter based on a study for side orifice offtake. Can the same conclusion be reached if the assumption of Soliman and Sims that viscous and surface tension forces remain negligible is relaxed?

Reference

[1] Soliman, H. M., and Sims, G. E., "Theoretical and Analysis of the Onset of Liquid Entrainment for Orifices of Finite Diameter," Int. J. Multiphase Flow, Vol. 18, No. 2, pp. 229-235, 1992.

440.169

Section A.3 describes the Kataoka-Ishii model for pool entrainment, which made use of data from several small-scale facilities. Provide justification that the Kataoka-Ishii model is appropriate for full-scale reactor upper plenum geometries.

440.170

Section A.3.2 discusses calculation of upper plenum pool entrainment and presents the results for two WCOBRA/TRAC calculations. Please provide the WCOBRA/TRAC model (nodalization and input, or appropriate description). Indicate if the AP1000 upper plenum (including guide tubes, lateral connections, heat slabs, etc.) was used, or if a simple stand alone model was used. Do the entrainment rate results represent an axial flow to the hot leg level, or do they represent the net flow from the upper plenum to the hot legs?

440.171

Figures A.3-4 and A.3-5 show a prediction of WCOBRA/TRAC upper plenum entrainment versus time for conditions simulating a DVI line break. Please provide the WCOBRA/TRAC collapsed liquid level for this simulation.

440.172

Figures A.3-9 and A.3-10 show a prediction of WCOBRA/TRAC upper plenum entrainment versus time for conditions simulating an Inadvertent ADS Actuation. Please provide the WCOBRA/TRAC collapsed liquid level for this simulation. In addition, please provide an explanation for the "spikes" that reduce the entrainment rate to near K-I Region 3 rates even though the steam velocity remains greater than 10 feet-per-second (ft/sec).

440.173

Please provide additional information on the simulations reported in Section A.4 on sensitivity studies on upper plenum entrainment rate and interfacial drag. In particular, show:

(a) Axial continuous liquid and steam flow rates into the upper plenum cell(s).

(b) Void fraction in each of the upper plenum cell(s).

(c) Collapsed liquid level in the upper plenum.

(d) Lateral flows (WLM, WEM, and WGM) into each hot leg from the upper plenum.

(e) Axial continuous and entrained liquid flows, and steam flow in the upper plenum at the bottom of the hot legs, and also the entrained drop size.

440.174

On page 2-4 WCOBRA/TRAC is stated to use the Taitel and Dunkler flow regime map for horizontal flow. Five flow regimes are considered: Slug and plug, stratified smooth, stratified wavy, dispersed bubble, and annular/annular dispersed liquid flow. For the case of the DEDVI line break case in Appendix A to WCAP-15833, Revision 1, please provide the liquid and steam velocities and identify the flow regime for the hot legs between the reactor vessel (RV) and the ADS-4 takeoff tees. Provide this information as a function of time between ADS-4 activation and IRWST injection.

440.175

On page 3-2 of WCAP-12945, Code Qualification Document for Best Estimate LOCA Analysis," Volume 1, the WCOBRA/TRAC flow regimes for vertical flow are described as small bubble, small to large bubble, churn-turbulent and film/drop. For the case of the double ended DVI line break case in Appendix A to WCAP-15833, Revision 1, please provide the liquid and steam velocities and identify the flow regime for the core exit, upper plenum to the hot leg elevation and in the ADS-4 piping. Provide this information as a function of time between ADS-4 activation and IRWST injection.

440.176

Figures 2-14 and 2-15 of WCAP-15833, Revision 1 indicate that condensation heat transfer was initially under predicted by WCOBRA/TRAC for the Lim test section and then over predicted. For the lengths and geometries of the AP1000 hot legs, provide a discussion of the conservatism of the WCOBRA/TRAC model for hot leg condensation for the AP1000 based on the correlation of the Lim data.

440.177

Section A.4.2 indicates that WCOBRA/TRAC results are relatively insensitive to assumptions made for entrainment rate in the upper plenum. The sensitivity to entrainment rate in the core is not addressed. Provide analyses similar to those of Section A.4.2 in which the core entrainment rate is varied consistently with that of the upper plenum. What is the uncertainty in the WCOBRA/TRAC predictions for core entrainment rate?

440.178

Section A.4.3 indicates that WCOBRA/TRAC results are relatively insensitive to assumptions made for interfacial drag in the upper plenum. The sensitivity to interfacial drag in the core is not addressed. Provide analyses similar to those of Section A.4.3 in which the core interfacial drag is varied consistently with that of the upper plenum. What is the uncertainty in the WCOBRA/TRAC predictions for core interfacial drag?

440.179

As discussed in Section A.4.5, when the void fraction in WCOBRA/TRAC cell is less than 0.8 the cell is assumed to be liquid-continuous and when the cell void fraction is greater than 0.8 the cell is assumed to be vapor-continuous. Please justify this assumption by comparison to experimental level swell data for conditions corresponding to those calculated for the AP1000.

440.180

In Section A.4.5, it is stated that in WCOBRA/TRAC if the difference in void fraction between a cell center and the lower boundary is greater than 0.05 then a two-phase level exists in the bottom of the cell directly above; whereas if the difference in void fraction between a cell center and the upper boundary is greater than 0.05 then a two-phase level exists in the top of the cell directly above. Please justify these assumptions by comparison to experimental level swell data for conditions corresponding to those calculated for the AP1000.

440.181

A. As stated in Section 15.4.8.2.1.7 of the DCD Tier 2 information, four cases are analyzed for the rod ejection accident (REA).

Provide the calculated peak radial average fuel enthalpies for all four REA cases, including beginning-of-cycle at full-power and zero-power, and end-of-cycle at full-power and zero-power, and address the acceptability of the calculated fuel enthalpies for supporting the adequacy of the AP1000 REA analysis.

B. The generic analyses were performed by Westinghouse and documented in a Westinghouse report, NTD-NRC-95-4438, “Westinghouse Assessment of Topical Report Validity for Reactivity Insertion Accident with High Burnup Fuel,” previously in support of the AP600 design certification review. The generic analyses, which assumed a low enthalpy value for fuel failure, showed that the radiological consequences of the REA meet the acceptance criteria for the REA.

Discuss the applicability of the results in NTD-NRC-95-4438 to the AP1000 design and provide the value of the enthalpy assumed in the Westinghouse report for fuel failure during the REA.

440.182

A. AP1000 RAI 440.009 requested information on the steady-state pressure drops through the vessel and various regions and components in the primary loops of the AP1000 design at best estimate flow conditions for a 10 percent SG tube plugging level.

As part of the AP1000 design certification, please specify in Tier 1 inspection, test, analysis, and acceptance criteria (ITAAC) Table 2.1.2-4 the acceptable ranges of the resistance factors for the regions and components described in RAI 440.009. Describe the bases for the acceptance ranges.

B. Figure 2.1.2-1 in Tier 1 information provides the general layout of the reactor coolant system (RCS). Please specify in Table 2.1.2-4 the acceptable ranges of the pressurizer volume, the horizontal lengths of the hot legs and cold legs, the relative locations of the pressurizer and the ADS-4 offtake pipes from the RV, and the elevation of SG relative to the RV. Describe the bases for the acceptance ranges.

C. What is the acceptable range, and the basis, of the reactor coolant pump (RCP) head-capacity characteristics for the AP1000 design? Why is this not included in the Tier 1 ITAAC?

Series 451 - Meteorology

Meteorology

451.001

What are the references for the meteorological data used in the analyses resulting in selection of the site parameter values presented in Table 2-1?

451.002

Section 2.3.4 of the AP1000 DCD discusses calculation of the bounding short-term relative concentration (X/Q) values for use in the off-site design-basis accident dose assessments. Was this description provided for information only? Is it expected that the methodology and all inputs and assumptions selected by the combined operating license (COL) applicant will be evaluated at the time of the COL review? If the methodology and all inputs and assumptions will be evaluated during the COL review, that requirement should be explicitly stated in Section 2.3.6.4.

If a commitment will not be made that the methodology and all inputs and assumptions selected by the COL applicant will be evaluated at the time of the COL review, what specific inputs, and assumptions for use in the Regulatory Guide (RG) 1.145, “Atmospheric Dispersion Model for Potential Accident Consequence Assessment at Nuclear Power Plants,” methodology are proposed as part of the AP1000 Design Certification? Other than the site meteorological data, what specifically will be provided as part of the COL application?

451.003

What factors contributed to the reduction in the 0-2 hour exclusion area boundary (EAB) X/Q to 6.0E-4 sec/m^3 (seconds-per-cubic meter) proposed for the AP1000 from the value of 1.0E-3 sec/m^3 for the AP600?

451.004

Section 2.3.5 of the AP1000 DCD discusses calculation of the bounding long-term X/Q values for use in the off-site assessment. Was this description provided for information only? Is it expected that the methodology and all inputs, and assumptions selected by the COL applicant will be evaluated at the time of the COL review? If the methodology and all inputs and assumptions will be evaluated during the COL review, that requirement should be explicitly stated in Section 2.3.6.5.

If a commitment will <u>not</u> be made that the methodology and all inputs and assumptions selected by the COL applicant will be evaluated at the time of the COL review, what specific methodology, inputs and assumptions are proposed as part of the AP1000 Design Certification? Other than the site meteorological data, what specifically will be provided as part of the COL application?

451.005

The first paragraph of 15A.3.3 of the AP1000 DCD states that short-term atmospheric dispersion factors are listed in Section 2.3.4. This is correct for the off-site values, but not for the control room values (the control room values are available, but not in Section 2.3.4). Therefore, either the paragraph should be deleted, the first sentence should be modified to specify that off-site values are provided in Section 2.3.4, or the control room values should be inserted into Section 2.3.4.

451.006

Section 15A.3.3 of the AP1000 DCD discusses calculation of the bounding X/Q values for use in the control room design-basis accident dose assessments. Diagrams showing the site plan with release and intake locations are included. Was this description provided for information only? Is it expected that the methodology and all inputs, and assumptions selected by the COL applicant will be evaluated at the time of the COL review? If the methodology and all inputs and assumptions will be evaluated during the COL review, that requirement should be explicitly stated in Section 15A.

If a commitment will <u>not</u> be made that the methodology and all inputs and assumptions selected by the COL applicant will be evaluated at the time of the COL review, please address the following:

A. What specific methodology, inputs, and assumptions are proposed as part of the AP1000 Design Certification application? Other than the site meteorological data, what will be provided as part of the COL application?

B. Do Notes 4 through 7 on page 15A-15 simply list the design-basis accidents to which the X/Q values should be applied, or are the Notes stating that some other X/Q values have not been listed because they are less than, and therefore bounded by, the listed X/Q values? If only bounding values were provided, provide inputs and assumptions for the values that were not submitted in the AP1000 DCD.

C. Because distances, heights, building dimensions and assumptions have not been provided, it is difficult to judge these X/Q values. However, on page 15A-17 the main equipment hatch release location appears to be quite close to the control room heating, ventilation and air conditioning (HVAC) intake, yet the 0-2 hour X/Q value listed is, or is bounded by, 1.2E-3 sec/m^3. This value seems low given the short apparent distance relative to other postulated release locations. Further, it is the same value as for the equipment hatch at the staging area which is further away. This appears inconsistent. Please explain.

D. In the Table 15A-5 listing of X/Q values, some of the X/Q values are the same for two release locations and/or consecutive time periods. To what is this attributed? Are these occurrences the result of selecting bounding values or a result of other assumptions? Further, in the case of the ground level containment release points, the proposed values are different than those for the AP600. What factors contribute to the differences?

451.007

Will the environmental impact of heat dissipation systems such as the discharge canal and cooling tower be evaluated as part of the COL application? If so, that requirement should be explicitly stated in the appropriate section of the AP1000 DCD.

<u>Series 470 - Radiological Impact</u>

<u>Design-Basis Accident Radiological Consequences</u>

470.001

Please provide the following information with regard to the Main Steam Line Break (MSLB) as discussed in Chapter 15.1.5.4 and Table 15.1.5-1 of the AP1000 DCD:

A. What is the basis for assuming an accident duration of 72 hours for the MSLB? What assumptions lead to the determination of this time?

B. What assumptions were made in the determination of the values for the steam mass releases from both SGs associated with the radiological consequences analysis of the MSLB?

470.002

Please provide the following information with regard to the radiological consequences analysis of the design-basis Locked Rotor Accident (LRA) as discussed in Chapter 15.3.3 and Table 15.3-3 of the AP1000 DCD:

A. It is stated it was determined that as a result of the LRA no fuel is damaged such that the activity in the fuel-cladding gap is released, but that a conservative assumption of 16 percent of the core fuel rods failed was used in the radiological consequences analysis. How was it determined that no fuel is damaged? What is the basis for the assumption of 16 percent failed fuel?

B. What is the basis for the assumed accident duration of 1.5 hours for the LRA?

C. What assumptions were made in the determination of the steam mass release from the secondary system associated the radiological consequences analysis of the LRA?

D. What is the basis for the leak flashing fraction of 0.04 percent for the first 60 minutes of the LRA?

E. Table 15.3-3 lists the reactor coolant noble gas activity as equal to the operating limit of 280 milliCi/gm (milli-Curies-per-gram) dose equivalent Xe-133. Other accidents list this operating limit as 280 microCi/gm dose equivalent Xe-133. Please clarify the discrepancy (is this a typographical error)?

F. Table 15.3-3 lists a fission product gap fraction of 0.10 for Kr-84. The krypton isotope of concern with respect to gap fractions for non-LOCA design-basis accident dose analyses is Kr-85. Please clarify the correct isotope of Kr (is this a typographical error)?

470.003

Please provide the following information with regard to the radiological consequences analysis of the design-basis Rod Ejection Accident (REA) as discussed in Chapter 15.4.8.3 and Table 15.4-4 of the AP1000 DCD:

A. A fraction of the fuel rods are assumed to melt in the radiological analysis of the REA. Regulatory Position 3 of RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” states that, for design-basis accident events that do not assume melting of the entire core, radial peaking factors should be applied in determining the inventory of the damaged rods. This does not appear to have been done. Please either update your analysis to include the maximum radial peaking factor in the determination of the source term if it was not included, or provide a basis for why you did not do so.

B. What is the basis for the assumed leak flashing fraction of 4.0 percent in the radiological consequences analysis of the REA?

C. What assumptions were made in the determination of the steam mass release from the secondary system assumed in the radiological consequences analysis of the REA? What is the basis for the assumed release duration of 1800 seconds?

D. What is the basis for the alkali metal partition coefficient of 0.001 used in the REA radiological consequences analysis? What assumptions were made in the determination of the value?

470.004

With regard to the radiological consequences analysis of the design-basis Small Line Break Outside Containment as discussed in Chapter 15.6.2 and Table 15.6.2-1 of the AP1000 DCD, what is the basis for the assumed leak flashing fraction of 0.41?

470.005

Please provide the following in regard to the radiological consequences analysis of the design-basis Steam Generator Tube Rupture (SGTR) as discussed in Chapter 15.6.3.3 and Table 15.6.3-3 of the AP1000 DCD:

A. What is the basis for the assumed flashing fraction for the break flow, as documented in Figure 15.6.3-10? What assumptions lead to the determination of the time-dependent flashing fraction?

B. What is the basis for the assumed steam release duration of 13.19 hours? What assumptions lead to the determination of this time?

470.006

Please provide the following information in regard to the radiological consequences analysis of the design-basis LOCA as discussed in Chapter 15.6.5.3 and Table 15.6.5-2 of the AP1000 DCD:

A. How was the main control room activity level (2.0E-6 Ci/m^3 [Curies-per-cubic meter] of dose equivalent I-131) and time (0.2622 hours) at which the emergency habitability system is actuated determined? What assumptions were made in the determination of these values?

B. What is the basis for the control room unfiltered in-leakage assumption of 5.0 cfm (cubic feet-per-minute)?

C. What assumptions and inputs were used to calculate the LOCA doses in the control room due to radiation from adjacent structures and sky-shine?

470.007

All Chapter 15 design-basis accident radiological analyses include a discussion of additional radiological consequences of spent fuel pool boiling that may occur coincident with the accident. What assumptions and inputs were used to calculate the radiological consequences as a result of spent fuel pool boiling?

470.008

Table 15A-2 lists the iodine appearance rates used in design-basis accident radiological consequences analyses that include iodine spiking. How were the RCS iodine appearance rates calculated?

470.009

(Appendix 15B of the AP1000 DCD) The staff plans to perform an independent Monte-Carlo analysis of the uncertainties in prediction of the aerosol removal coefficients for the AP1000 design to determine if use of the previously determined AP600 values is acceptable. For use in this analysis, please provide:

A. containment geometry inputs for the AP1000 design (volume, upward facing surface area, etc.),

B. upper and lower values of the thermal-hydraulic containment parameters relevant to the aerosol removal, i.e., pressure, temperature, relative humidity, steam/air ratio, and steam condensation rate on the heat sinks, and

C. ranges of non-safety-related containment spray flow rate and spray droplets distribution.

470.010

(Appendix 15B, paragraph 15B2.4.2) Please provide references for the chosen aerosol size distribution parameters (i.e., radius of 0.22 micrometers and standard deviation of 1.81). Also, please justify the statement that "sensitivity of aerosol removal coefficient calculations to these values is small" since, in general, the opposite is true.

470.011

(Appendix 15B, paragraph 15B.2.6) The paragraph presents a qualitative discussion of the differences between the AP600 and the AP1000 designs concluding that the use of the AP600 removal coefficients is conservative. Please provide either a sample calculation, or an analytical justification for this conclusion. Also, one potentially important difference is omitted, i.e., the increased height of the AP1000 containment. It is known that the increased height decreases the rate of aerosol removal, which would be a non-conservative effect. Please discuss the significance of this issue.

470.012

In the AP1000 probabilistic risk assessment (PRA) (as in the AP600 PRA), a decontamination factor (DF) of 100 has been credited for SGTRs due to impaction on the SG tubes, based on the results of an unpublished paper by Jun Li, David Leaver, James Metcalf, entitled "Aerosol Retention During an Unisolated Steam Generator Tube Rupture Severe Accident Even." Please provide justification for this seemingly high DF estimate in light of more recent analyses and/or experiments, if available. In addition, please provide a copy of this paper. Also, since this release category is also used for inter-system LOCAs (ISLOCAs), please justify the use of such a high DF, as a general rule, for all bypass events.

470.013

Please justify why the AP600 release fractions, as calculated by MAAP, should be considered applicable to the AP1000. In addition, please justify ZERO release fraction for tellurium in all release classes (in both chemical forms, TeO_2 and Te_2)

Series 720 - Reliability and Risk Assessment

Reference: AP1000 Probabilistic Risk Assessment (PRA), Revision 0, submitted March 28, 2002

Late Containment Failure (No core damage)

720.041

A late containment failure (LCF) end state has been added to the Level 1 event trees. Table 4A-1 indicates that these events are assigned to plant damage state (accident class) 2. The footnote to Figure 4A-2 indicates that the LCF end state is not added to the core damage end states and is discussed in the Level 2 analysis. However, the Level 2 PRA does not include containment event trees (CETs) for accident class 2 (Chapter 35), and the assessment of events with failure of the passive containment cooling system (PCS) (Chapter 40) is limited to containment pressure response rather than core cooling. Please clarify whether/how the frequency associated with LCF endstates is addressed in the Level 2 PRA and CETs.

MAAP Analyses

720.042

Westinghouse states that the overall AP1000 plant response to severe accidents as well as the mass, composition, and superheat characteristics of the initial debris relocation is very similar to the AP600, and therefore: (1) the results and insights from the AP600 hydrogen generation and mixing analyses for each accident class are applicable to the AP1000, (2) the release fractions and timing for the AP1000 release categories would be approximately the same as for the AP600, and (3) the conclusions of the AP600 analyses regarding the challenge to the lower head integrity from core debris relocation into the lower plenum and the challenge from ex-vessel steam explosions are applicable to the AP1000. The premise for this conclusion is questionable given the large differences in core power and mass between the two designs and the substantial differences in melt progression timing indicated in the results for 1A/1AP sequences in Chapter 36 of the AP1000 PRA. Also, the dominant accidents within each release category, and their relative contribution, may be different for the AP1000 due to differences in the Level 1 PRA, and could lead to selection of different representative sequences for some of the release categories. Please provide the AP1000-specific analyses of core melt progression and fission product releases for the dominant accident sequences within each release category, and use this information to either define the AP1000-specific hydrogen releases and fission product source terms, or to substantiate the applicability of the AP600 hydrogen analyses and fission product releases to the AP1000. This should include a comparison of event timing, fraction of core melted, hydrogen generation rates and quantities, mass and superheat characteristics of debris relocating into the lower plenum, and fission product release histories for representative sequences in each accident class.

Operator Actions

720.043

Time windows available for operator actions in the AP1000 are shorter than for the AP600 (see Table 35-6 of the AP1000 PRA). For every human action in the Level 2 PRA, please describe the basis for the revised time estimates, and their impact on human error probabilities (HEPs) and containment performance (i.e., large release frequency).

720.044

Reactor cavity flooding success criteria has been modified to account for higher water depth and earlier flooding times required for the AP1000. Operator instructions to flood the cavity have been moved from the end of AFR.C-1 in the AP600 (before entering the Severe Accident Management Guidelines), to the entry to AFR.C-1 in the AP1000. Please confirm that moving this action does not adversely impact other operator actions that might be critical to core damage prevention or mitigation, or conflict with other objectives of AFR.C. (AFR.C-1 is the functional restoration guideline within the Westinghouse AP1000 emergency response guidelines for response to inadequate core cooling.)

Depressurization

720.045

For the AP1000, the thickness of the vessel wall that is conducting heat at the peak critical heat flux (CHF) is 36 times the minimum required to carry the dead load. Although a factor of 36 may appear substantial, 1 pound-per-square inch (gauge) [psig] of internal pressure within the reactor pressure vessel (RPV) would be roughly equivalent to the dead load for the AP1000. Thus, a pressure pulse of approximately 35 psi would be sufficient to eliminate this margin. This is more limiting than the 150 psig criteria used for determining whether the reactor is adequately depressurized to support in-vessel retention. (The analysis of in-vessel retention assumes all sequences with successful depressurization (DP) will have sufficiently low RCS pressure to prevent RPV failure by over-pressure/dead-load. For accident classes 3BE, 3BL, 3BR, and 3D/1D the primary system pressure by definition of the accident class is 150 psig or less, and success at DP is assumed.) Reflood of the damaged core may also produce RCS pressures in excess of the margin. Please address whether some portion of the sequences with success at DP or eventual reactor reflood can result in RV failure due to over-pressure.

Containment Isolation

720.046

The containment isolation fault tree success criteria tables (Tables 24-5a through-c and 24-8) do not include all of the isolation valves listed in Table 24-1 for the 12 penetrations analyzed (some of which are initially open). Please discuss why only a partial listing is provided.

In-Vessel Retention

720.047

Please discuss the implications of the most recently completed experimental work related to in-vessel retention of molten core debris on the reliability of the in-vessel retention strategy for the AP1000 design, including the work performed as part of the RASPLAV project and any available results from the Organization for Economic Cooperation and Development (OECD)-sponsored MASCA program at the Russian Research Center, and the SIMECO and FOREVER programs in Sweden. Specifically, address the implications of this work on the potential for debris bed stratification and chemical interactions between molten debris and the RV wall.

720.048

Please provide a quantitative assessment of the uncertainties in the reliability of in-vessel retention for the AP1000 design using the analytical approach and tools developed through the Idaho National Engineering and Environmental Laboratory (INEEL) assessment of in-vessel retention for the AP600 (i.e., J. L. Rempe, et al., "Potential for AP600 In-Vessel Retention Through Ex-Vessel Flooding," INEEL/EXT-97-00779, December 1997). This should include an assessment of the uncertainties in heat transfer, decay heat, and material property assumptions described in Appendix B of the report, and the implications of forming the alternate debris bed configurations described in Section 2.1.2 of the report. Please provide the AP1000-specific probability density function results for the final bounding state (comparable to Figures 3-5 through 11 in the report) and for each alternate debris configuration. Justify that the margins to failure are sufficient to support the lower head failure assumptions used in the AP1000 PRA.

720.049

Describe how the water/steam flow path and flow areas specified for the AP1000 in Chapter 5.3.5 of the AP1000 DCD were simulated in the ULPU-2000 experiments, including scaling effects.

Reactor Vessel Insulation

720.050

Westinghouse claims in Chapter 5.3.5.4 of the AP1000 DCD that the forces on the AP1000 RV insulation following core relocation and cavity flooding can be based on AP600 test results from the ULPU-2000 test program for Configuration III. Although this test data was used to develop the functional requirements for the AP600 RV insulation and support system, its suitability and applicability for the AP1000 has not been established, and is questionable given the substantial differences between the AP600 and the AP1000 insulation system designs and accident conditions. The AP1000 design would have higher heat fluxes from the vessel, higher water/steam flow rates and flow velocities through the insulation system, a considerably smaller gap between the insulation and RV, and closely-fitted hemispherically-shaped insulation panel (versus conically-shaped insulation with a substantial standoff distance from the RV in the AP600). Collectively, these differences could result in substantially different pressure loads and functional requirements for the AP1000 RV insulation and support system. Westinghouse needs to either: (1) establish the applicability of the ULPU-2000 Configuration III test results to the AP1000 considering the impact of each of the above factors, or (2) develop the AP1000-specific test data based on the prototypical insulation and flow conditions for the AP1000, i.e., ULPU-2000 Configuration IV. Note: Westinghouse also states in Chapter 5.3.5.4 that further evaluation of the forces on the RV insulation and supports is provided in the AP1000 PRA. Such information is not provided in the PRA, e.g., in Chapter 39 "In-Vessel Retention of Molten Core Debris."

720.051

Details of the insulation design (e.g., specific dimensions or clearances as a function of angle) and the insulation support frame are not provided. The discussion in Chapter 39.10.2 indicates that a detailed mechanical analysis of the insulation and support frame will verify the specified structural aspects of the support frame and insulation panels. The discussion in Chapter 5.3.5.5 of the AP1000 DCD indicates that a structural analysis of the AP1000 reactor cavity insulation system demonstrates that it meets the functional requirements discussed in Chapter 5.3.5.4. Have these analyses been completed? If not, when will the analyses be completed? Please confirm the status of the insulation design. Provide the functional requirements and the structural analysis showing how these functional requirements are met by the AP1000 insulation system.

Passive Containment Cooling System (PCS)

720.052

The PCS was assumed to always be operable in the AP600, but its functionality is now modeled in the AP1000 with respect to its operational success or failure. The success criteria is 1 of 3 PCS water lines open or operator provides an alternate source of water to the containment shell. If the PCS operates, other challenges are considered downstream. If the PCS fails, paths downstream of PCS failure do not address hydrogen combustion. Westinghouse did not consider operator actions to use the non-safety-related containment spray system, even though such actions would be included within the severe accident management guidelines. Use of the sprays could have both positive (reduce containment pressure and source terms) and negative (de-inert containment atmosphere) impacts on accident progression. Please provide an evaluation and a Level 2 PRA sensitivity case addressing the net impact that spray operation would have on containment release frequency and magnitude.

Intermediate Containment Failure

720.053

The sequence used to quantify the intermediate containment failure probability given failures of the PCS and containment venting (Figures 40-5 and 40-6) appears to be a loss-of-coolant accident (LOCA) with full RCS depressurization and successful core cooling. The resulting failure probability (0.02) is applied to all accident classes. Please justify the applicability of this probability value for each accident class since events with core damage could result in higher containment pressures than the sequence on which the probability value is based.

Hydrogen Combustion

720.054

AP1000 addresses diffusion flames through a defense-in-depth philosophy in the design. As the last level of defense, Westinghouse claims that there is sufficient margin to failure even if design features fail and diffusion flames occur near the containment shell. Westinghouse bases the last statement on previous assessments for the AP600, where the quantities of hydrogen produced were lower than for the AP1000, and which did not include features (dampers) to preferentially direct hydrogen away from the shell. Failure of an in-containment refueling water storage tank (IRWST) vent damper could result in local thermal loads greater than if no dampers were present. Please provide additional justification for the application of AP600 insights on creep failure if this is to be considered an additional level of defense.

Direct Containment Heating

720.055

An assessment of direct containment heating (DCH) was performed for the AP600 using the methodology developed as part of the DCH issue resolution (i.e., NUREG/CR-6338, “Resolution of the Direct Containment Heating Issue for a Westinghouse Plants with Large Dry Containments or Subatmospheric Containments,” February 1996). Rather than update the assessment for the AP1000, Westinghouse (in Appendix B.3) provided a qualitative argument that the AP1000 design includes reactor cavity design features to decrease the amount of ejected core debris from reaching the upper compartment, as called out in SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Passive Advanced Light-Water Reactor Designs.” This qualitative argument provides an insufficient technical basis for addressing DCH, given the potential for a greater DCH pressure loading in the AP1000 (due to the larger core mass), and the more recent and technically-defensible methodology that is now available. Please provide the results of a deterministic assessment based on the methodology developed as part of DCH issue resolution.

Offsite Consequences

720.056

The offsite consequences for release categories CFI (intermediate containment failure) and CI (containment isolation) for the AP1000 are similar to or less than the corresponding values for the AP600. This is unexpected since the fission product inventories for the AP1000 are larger than for the AP600, and the same release fractions were assumed. Please explain the reasons for this inconsistency.

720.057

Sensitivity and importance analyses for large release frequency and sensitivity analyses for offsite dose risk were provided in Chapter 50 of the PRA for the AP600. Sensitivity analyses and top event importance analyses are provided in Chapter 43 for the AP1000. However, component and operator action importance analyses, and sensitivity analyses for offsite dose risk have not been included. Please provide this additional information for the AP1000.

Core-concrete Interactions

720.058

Westinghouse claims that the concrete penetration on the reactor coolant drain tank (RCDT) (sump) side of the cavity is minimal following a hinged failure mode of the RV, compared to the penetration on the RV side of the cavity. However, this is predicated on the core debris separating, with the oxide component (about 85-90 percent oxide) remaining on the RV side of the cavity, and a metallic component (about 75 to 85 percent metal) reaching the RCDT side of the cavity. This debris separation behavior is used by Westinghouse as the basis for concluding that core debris accumulation in the cavity sump would not be controlling for basemat melt-through. It is unclear whether this separation will actually occur given the large uncertainties in the configuration of molten core debris prior to vessel breach (i.e., mixed versus stratified), and the turbulence and mixing that would occur as the debris enters and spreads within the reactor cavity. Please confirm the robustness of your conclusion and the adequacy of the sump curb design by providing an assessment of the impact on basemat melt-through times and containment pressure (for both limestone and basaltic concretes) assuming that this oxide/metallic separation does not occur following a hinged failure of the RV, i.e., either a homogeneous melt or an oxide melt reaches the RCDT side of the reactor cavity and enters the sump.

720.059

Please provide a description of: (1) the physical characteristics of the door between the reactor cavity compartment and RCDT room, including its approximate size, construction, buoyancy, hinging arrangement (opening direction and jambs), and pressure retaining capability, (2) the expected response of the door during the flood-up period and following a postulated melt-through of the RV, and (3) the potential for the door to break free and block the inlets to the RV insulation system during flood-up, or remain in place and restrict debris spreading within the reactor cavity following a postulated melt-through of the RV. Also explain why these design details should not be included in the system design description in the DCD.

Severe Accident Mitigation Design Alternatives (SAMDAs)

720.060

In response to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34(f)(1)(i), Westinghouse provided an evaluation of potential AP600 design improvements (Severe Accident Mitigation Design Alternatives) in Appendix 1B of the AP600 standard safety analysis report (SSAR). The details of this evaluation, which included a design description and estimated risk reduction, costs for each alternative, and estimated offsite exposure for each of the major release categories, formed the basis for the staff’s review. A similar evaluation has not been provided in Appendix 1B of the DCD or in the PRA for the AP1000. In order to support the staff’s review of potential design improvements, please provide an AP1000-specific evaluation of SAMDAs similar in scope and content to that provided for the AP600. Please include the following within the response:

A. a summary of the risk-significant enhancements (i.e., impacting core damage frequency [CDF] and person-rem doses) incorporated subsequent to the AP600 design, such as the third PCS water injection line and the IRWST vent dampers,

B. the risk (CDF and population dose per year) associated with operation of an AP1000 at the reference site. Include the risk associated with internally and externally-initiated events, and events at shutdown to the degree that this can be inferred from the associated analyses,

C. the specific site characteristics, including population, meteorology, economic data, and evacuation assumptions on which the population dose estimates are based (provide these characteristics/interface assumptions in such a way that one can readily determine whether a potential new reactor site is enveloped by the same analysis),

D. the estimated dollar value of completely eliminating all severe accident risk for an AP1000 plant at the reference site, broken down by major cost category (i.e., public exposure, offsite property damage, occupational exposure, onsite cleanup and decontamination, and replacement power),

E. an explanation of how insights from the AP1000-specific PRA and supporting risk analyses for external and shutdown events, including importance analyses and cutset screening, were used to identify potential plant improvements, and

F. justification that the potential improvements identified through a systematic process as suggested in (e) are included within the set of 15 SAMDAs identified in Appendix 1B of the AP1000 DCD. Provide a supplemental analysis for those risk-significant improvements not included within the list of 15.

<u>Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)/Combined License (COL) Applicant Action Items</u>

720.061

The list of risk-significant structures, systems, and components (SSCs) within the scope of ITAAC (DCD Table 2.3.9-3) and design reliability assurance program (D-RAP) (DCD Table 17.4-1) includes the hydrogen igniters, but does not include the IRWST louvered vents. The design and testing of these vents should also be included since they are a key element of the defense-in-depth philosophy for hydrogen control, and are important for minimizing the potential for creep failure of the containment from diffusion flames.

720.062

The commitment to cover and lock closed access portals to the Passive Core Cooling System (PXS) that may be near the containment wall is important for limiting diffusion flames. Please discuss how this commitment will be addressed in ITAAC or COL action items.

720.063

Westinghouse added a check valve to the AP1000 refueling canal drain to permit the reactor cavity to flood more rapidly (AP1000 PRA page 39-2). This valve should be added to the system description in the DCD (e.g., Chapter 5.3.5.4) and included within the ITAAC related to the PXS (Chapter 2.2, Table 2.2.3-4).

Shutdown Risk Assessment

720.064

The following questions/comments pertain to: (1) the frequency of RCS-OD (4.4E-6 per year), the frequency of overdraining the RCS required for midloop operations, and (2) the design of the step nozzle for residual heat removal system (RNS) pump suction. Please address as requested.

A. The HEP designated as RCS-MANODS1 evaluates the probability of failure to observe failure of the hot leg level instruments and failure to close the air-operated chemical and volume control system (CVS) valves to preclude overdraining.

(i) This HEP is based on the availability of the pressurizer wide range level instrument which is not safety-related and is not in Technical Specifications (TSs), and therefore may not be available during midloop operations. The HEP assessment should not credit the availability of pressurizer wide-range level indication. Please revise the HEP assessment. (An alternative would be to evaluate the pressurizer wide-range level indication against the four criteria of 10 CFR 50.36(c)(2)(ii) for possible inclusion in the TSs.)

(ii) This HEP should be based on the time to drain the hot leg to the point of the low critical vortexing level based on the current step nozzle configuration, since a comparison is being made between the hot leg level instruments and the pressurizer wide range level instrument. The time window of 3 hours is not applicable for this HEP.

1. In the HEP assessment in the PRA and the DCD, document the time to drain the hot leg to the critical vortexing level given the current step nozzle configuration.

2. Revise the time window in the HEP assessment to reflect the time to drain the hotleg to the low critical vortexing level and revise the HEP assessment

B. The HEP designated as RCS-MANODS2 evaluates the probability of failure to detect the failure of automatic closure of the air-operated valves (CVS-V045 and CVS-V047). The estimated time window for the diagnosis and completion of the action appears to be five minutes. The HEP of 1.3 E-2 seems optimistically low considering the data provided by NUREG/CR-1278, “Handbook on Human Reliability Analysis with Emphasis on Nuclear Power Plant Operations,” Table 12-4 which suggests that the median joint HEP for diagnosis of an abnormal event annunciated closely time to be around 0.5. Please revise the HEP or document and justify the method used to arrive at the HEP.

C. In the Shutdown PRA (Chapter 19E of the AP 1000 DCD), on page 19E-5, it states that, should a vortex occur, the maximum air entrainment into the pump suction as shown experimentally will be no greater than 5 percent. This appears to be based on the results reported in Westinghouse letter to NRC, dated July 6, 1994, entitled "AP600 Vortex Mitigator Development Test for RCS Mid-loop Operation," (Reference 4 on page 19E-39 of the AP1000 DCD).

In its letter to P.G. Trudel, Sequoyah Project Engineer from B. J. Garry, Manager, TVA Sequoyah Plant Domestic Customer Projects, TVA-90-1050, September 27, 1990, entitled "Tennessee Valley Authority, Sequoyah Nuclear plant, <u>CCP Gas Issue</u>," Westinghouse states that “minimizing the gas accumulation does not preclude the possibility of initiating a longer term mechanism such as shaft fatigue, wear ring degradation, bearing wear or seal wear. Therefore, for the long term Westinghouse believes that any accumulation is detrimental to the pump reliability."

(i) If the RNS pump is continually subjected to vortexing during midloop conditions, discuss how RNS pump reliability would be impacted?

(ii) Please document in the DCD Chapter 19E and the AP1000 PRA at what level in the hot leg is the critical vortexing level reached given the current step nozzle configuration? Is this critical vortexing level above or below the hot leg level at which the CVS system isolates (hot leg level low 1)?

D. Failure of both hot leg level instruments was reported to be 2.9E-4 per demand/draindown on page 54-36 of the AP600 PRA (Revision 8, dated September 30, 1996). Please document in the AP1000 DCD (Chapter 19E) and the AP1000 Shutdown PRA (Chapter 54) the differences between the hot leg level instrumentation used in current Westinghouse plants versus the AP1000 design which supports using this value. The staff has observed problems with incorrectly installed RCS level indication which has resulted at a greater failure rate than 2.9E-4 per RCS draindown.

720.065

The reduced time to boiling during cold shutdown as compared to the AP600 is given in Table 54-3, of the AP1000 Shutdown PRA,

A. Please re-calculate and document the new containment closure failure probability and the revised AP1000 Shutdown large early release frequency (LERF) in Chapter 54 of the AP1000 Shutdown PRA, and

B. Please document in Chapter 19 of the AP1000 DCD and Chapter 54 of the AP1000 Shutdown PRA how containment recirculation would function if the containment could not be closed during cold shutdown/refueling, considering containment integrity is not required during these modes.

720.066

Please document in Chapter 19 of the AP1000 DCD and Chapter 54 of the AP1000 Shutdown PRA, how trash generated during outages in containment will impact containment recirculation.

720.067

In Figure 54-7 of the AP1000 Shutdown PRA, “Loss of Offsite Power During RCS Drained Condition Event Tree,” failure of the onsite alternating-current (AC) power through Diesel Generators top event is missing. Please correct the event tree and re-calculate the sequences adding the diesel generators.

720.068

The HEP designated as RHN-MAN05 represents the failure of the operator to initiate gravity injection from IRWST via the RNS suction line. The cues for this HEP include high core exit temperature; however, the core exit thermocouples are not required to be available/operable during Modes 5 and 6. Therefore, the HEP assessment should not credit the core exit thermocouples. Please revise the HEP assessment.

720.069

In the AP1000 DCD on page 19E-14, it is stated that “[i]n Modes 5 and 6, there is no potential for steam release into the containment immediately following an accident.” During the AP1000 drained operations with the SG channel manway covers removed, an extended loss of RNS will cause steam to be released in containment until gravity injection from the IRWST is manually initiated or automatically actuated. These statements appear to contradict each other. Please clarify the apparent inconsistency and revise the statement on page 19E-14, as necessary.

720.070

An assessment of shutdown risk, considering fires, internal floods, and seismic events has not been submitted. Please provide a shutdown assessment of these initiators considering (a) containment may be open, (b) fire/flood barriers may be breached for maintenance, and (c) transient combustibles for a given fire areas may be increased to support maintenance.

720.071

The staff noted that common cause failure (CCF) of 4 out of 4 IRWST injection squib valves was estimated as 2.6E-5 in the AP600 Shutdown PRA. The AP1000 Shutdown PRA documents a CCF of 2.6E-5 (the same value) for 6 high pressure squib valves. The staff would expect a different value. Please provide the analysis in the AP1000 Shutdown PRA that documents a CCF of 2.6E-5 for 6 high pressure (HP) squib valves. Please provide any data references.

720.072

Please provide the justification for not including low pressure (LP) squib valves 118A/B in the same common cause group as the 6 HP squib valves. This justification should include differences in operation, maintenance, and design.

Fuel-Coolant Interactions

720.073

Why are the companion reports to DOE/ID-10541, “Lower Head Integrity Under In-Vessel Steam Explosion Loads,” no longer referenced in Chapter 34 and 39 of the AP1000 PRA as they were for the AP600? Are these reports applicable to the AP1000?

720.074

Chapter 34.2.2.1 states that the mass flow rate, superheat and composition of debris is expected to be essentially the same as the AP600 and, therefore, it is reasonable to extend the results of the AP600 in-vessel steam explosion analysis to the AP1000. Please, provide either a sample calculation, an analytical justification, or an equivalent basis for this conclusion.

720.075

Chapter 34.2.2.2 states that initial debris mass, superheat, and composition are assumed to be the same as the AP600 and, therefore, the results of the AP600 ex-vessel steam explosion analysis are considered appropriate for the AP1000. Please, provide either a sample calculation, an analytical justification, or an equivalent basis for this conclusion.

720.076

It appears that there is a typographical error in Reference B-3. Shouldn’t Reference B-3 be the “AP600 Probabilistic Risk Assessment,” GW-GL-021, August 1998, as opposed to the AP1000 PRA?

Equipment Survivability

720.077

The AP1000 system equivalent to the AP600 post-accident sampling system is designated as the containment atmosphere sampling system. Are there any differences between the AP600 post-accident sampling system and the AP1000 containment atmosphere sampling system including differences in functions? If so, please describe.

720.078

SECY-93-087, dated April 2, 1993, and the associated Commission’s staff requirements memorandum (SRM), dated July 21, 1993, sets forth the policy regarding the criteria that the staff will use to evaluate an advanced light-water reactor (ALWR) vendor’s review an analysis of various severe accident scenarios and associated identification of the equipment needed to perform during a severe accident and the environmental conditions under which the equipment must function. The staff concluded in NUREG-1512, “Final Safety Evaluation Report [FSER] Related to Certification of the AP600 Standard Design,” that applicable environments described in Section 19.2.3.3.7 of the AP600 FSER met the above guidance. Table D.7-2 and Figures D.7-3 thru D.7-69 of the AP600 PRA were an integral part of defining that pressure, temperature, and event timing. These figures were the basis for defining the applicable environments described in Section 19.2.3.3.7 of the AP600 FSER. These figures do not appear to have been provided for the AP1000. Please provide similar figures for the AP1000 and/or additional information to support the position that the applicable environments meet the above-mentioned guidance for the AP1000.

720.079

The applicable environment for equipment survivability accepted by the staff as documented in Section 19.2.3.3.7 of the AP600 FSER included those late in-vessel and ex-vessel phase environments discussed in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants.” The severe accident radiation environment defined in Section D.7.1 of the AP1000 PRA appears to no longer include the late in-vessel and ex-vessel release phases. Please provide additional information to support the position that the equipment identified will survive late in-vessel and ex-vessel source terms for the AP1000.

720.080

Gamma and Beta Doses in Figures D-1 and D-2 of the AP1000 PRA are less than the corresponding figures for the AP600. Considering the power rating has gone up, one would expect these doses to increase, not decrease. Why are these doses less?

720.081

The AP600 FSER states that the combined license (COL) applicant referencing the AP600 certified design will perform a thermal lag assessment of the as-built equipment used to mitigate severe accidents to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns. This assessment is COL Action Item 19.2.3.3.7-1 and is discussed in Section 19.59.10.5 of the AP600 DCD. Where can this COL Action Item be found in the AP1000 DCD? How can the staff make a reasonable assurance finding, such as in Section 19.2.3.3.7, for the AP1000 without this COL Action Item?

Dominant Severe Accident Sequences

720.082

In order to obtain a clearer picture of severe accident progression in the AP1000, the staff plans to perform confirmatory MELCOR analyses for several sequences that are either dominant overall in terms of CDF or dominant within some set of risk-significant sequences. The AP1000 PRA, page 59-7, describes, in detail, the five sequences with the highest CDF. Please provide a similarly detailed description of the following additional sequences:

A. Sequence 13 (SGTR initiator, failure of core makeup tank [CMT] or RCP trip, success of passive residual heat removal [PRHR], failure of full and partial ADS). This is the highest-frequency SGTR-initiated core damage sequence reported.

B. Sequence 20 (Transient, failure of MFW/SFW/PRHR [main feedwater system/startup feedwater system/passive residual heat removal system], success of CMT and RCP trip, failure of full and partial ADS). This is the highest-frequency non-bypass sequence expected to be at high RCS pressure at the time that core damage begins.

In-vessel and Ex-vessel Steam Explosion

720.083

It is stated in Chapter 34 of the AP1000 PRA that the fuel-coolant interaction (FCI) cannot produce sufficient energy on a short time scale to produce a missile that would fail the AP1000 containment. This statement is based on direct extension of the AP600 assumptions and analysis for the AP1000. Please provide the justification that the results of the AP600 in-vessel steam explosion can be directly extended to containment failure in light of melt progression uncertainties in the AP1000.

720.084

The thick core reflector in the AP600 has been replaced by a core shroud in the AP1000 to allow for the additional fuel assemblies. Please justify that this change would not increase the size of the crucible failure at the shroud boundary, and thereby increase the initial pour rate into the lower plenum. In addition, please provide justifications for excluding other in-core crucibles that may be in more cylindrical forms that would support a failure at some mid-level location, instead of the hemispherical crucible with high heat fluxes near the upper surface of the pool. Under these conditions, the axial heat flux distribution may be somewhat different than that which would affect the initial pour rate into the lower plenum. Please demonstrate that your analyses consider the impact of melt progression uncertainties.

720.85

The AP600 in-vessel steam explosion analysis neglects the possibility of initially small FCIs (with little energetic potential) being a driver for larger melt crucible failures that would increase the melt pour rate. How were these events considered or bounded for the RPV survival in-vessel? Please elaborate.

720.086

The FCI during the premixing stage will attain an optimum set of conditions that would likely result in optimum energetics during the propagation phase. Such a scenario is contingent upon the choice of premixing time, trigger time, etc. Justify for the AP1000 that these parameters were selected to produce bounding values of premixtures and energetics.

720.087

There is no reference provided in the AP1000 PRA for the assessment of ex-vessel steam explosions except to say that the results of the AP600 are applicable to the AP1000. The AP600 ex-vessel steam explosion analysis considers the lifting of the RPV (see Chapter 19 in NUREG-1512), and concludes that the steel containment remains intact. Please provide the justification for neglecting the potential for failure of the containment penetrations as a result of violent movement of the reactor pressure vessel subject to steam explosions-impulse loads.

<u>In-vessel Retention</u>

720.088

In order to resolve the issue of in-vessel debris coolability due to lower head cooling for the AP1000, Westinghouse refers to a previous U. S. Department of Energy (DOE) study of the same issue for the AP600 (T. G. Theofanous, et al., “In-Vessel Coolability and Retention of a Core Melt,” DOE/ID-10460, October 1996), and states that these results are directly applicable to the AP1000. The following questions are relevant to the applicability of the DOE AP600 experiments and analyses to the AP1000:

A. Preliminary calculations performed for the AP1000 had shown that RV insulation based on the ULPU Configuration III gave very little margin to dryout, and, as a result, the AP1000 RV insulation was redesigned to more resemble ULPU Configuration IV (Figure 39-3 in the AP1000 PRA), for which the measured CHF was 20 percent to 30 percent higher.

(i) What are the main phenomenological reasons for the substantially higher critical heat flux that is measured for ULPU Configuration IV as compared with ULPU Configuration III?

(ii) In the DCD (e.g., Figure 5.3-7 in the AP1000 DCD), the insulation appears to resemble the Configuration III design. Please confirm the actual configuration (i.e., either Configuration III or Configuration IV) foreseen for the AP1000.

(iii) Please demonstrate the influence of boiling on the structural integrity of the reactor pressure vessel insulation.

(iv) What are the experimental uncertainties associated with the measured critical heat flux for the ULPU Configuration IV design as a function of angle?

B. Debris jet impingement on the lower head was shown in the DOE AP600 study to result in an upper-bound ablation depth of about 12.5 centimeters (cm) (as compared with 15.24 cm minimum total thickness), as per calculations in which ablation depth is directly proportional to pour mass, and in which one-third of the AP600 total core mass was used. As this sub-issue was not specifically discussed in the AP1000 supporting documents, please provide justifications for assuming that the lower head failures due to debris impingement in the AP1000 design should be extremely unlikely even with the 26 percent larger total core mass in the AP1000 as compared with the AP600.

C. What is the impact of the higher fuel enrichment on the melt progression and pour rates into the lower plenum?

D. The analysis performed by T. G. Theofanous (T. G. Theofanous, et al., “In-Vessel Coolability and Retention of a Core Melt,” DOE/ID-10460, October 1996) and Westinghouse, in portraying the molten pool state inside the lower head, does not consider:

(i) The potential focusing effects of a thin metal layer on top of the molten pool (with a high thermal conductivity) on the local heat flux on the vessel wall, especially considering the results of calculations for small molten steel masses documented in the study by INEEL (J. L. Rempe, et al., “Potential for AP600 In-Vessel Retention through Ex-Vessel Flooding – Technical Evaluation Report,” INEEL/EXT-97-00779, Idaho National Engineering Laboratory, December 1997).

(ii) The potential influence of impurities in the debris bed that could result in stratification of the oxide pool into layers of different composition (as observed in the OECD RASPLAV and MASCA programs).

(iii) The potential influence of zirconium, steel and uranium partitioning between oxide and metallic phases, on the downward heat transfer (as observed in the OECD MASCA experiments).

720.089

Please demonstrate that considering the above-mentioned phenomenological uncertainties, the AP1000 lower head integrity will not be challenged. In any re-analysis, please consider the uncertainties associated with the measured critical heat flux on the outside surface of the AP1000 lower head (see AP1000 RAI 720.88.A.iv).

<u>Core-Concrete Interaction and Core Coolability</u>

720.090

Since long-term core debris cooling on the cavity floor has not been demonstrated, please provide the implications of extended Core-Concrete Interactions (CCIs) on combustible gas generation and combustion-induced containment failure.

720.091

As part of the verification of the long-term integrity of the AP1000 containment during CCI, several calculations with MAAP and other supporting codes were carried out and discussed in Appendix B.4 of the PRA. These runs indicated that, absent water cooling from the Passive Containment Cooling Water Storage Tank (PCCWST), basemat melt-through would occur at between 2.8 and 4.5 days, and containment pressure at 24 hours would reach values between 20 psig and 39 psig. There are significant uncertainties regarding the coolability of ex-vessel core debris by an overlying water pool. In view of these uncertainties, please provide basemat erosion and containment pressurization results for MAAP scenarios identical to those examined in Appendix B.4 of the PRA, with the following differences: (1) the cavity is dry (so that credit is not given to debris quenching); and (2) water cooling from the PCCWST is assumed to be unavailable.

Hydrogen Generation, Transport, Mixing and Combustion

720.092

What is the philosophy of igniter placement in the upper compartment? The AP600 containment volume is 1.69×10^6 cubic feet (ft^3) as compared to 2.07×10^6 ft^3 for the AP1000. Presumably, the bulk of the increase in volume is in the upper compartment. Given that there is a certain volume of coverage for each igniter in the AP600, how does the AP1000 igniter scheme cover the increase in volume of 380,000 ft^3 with the same number of igniters?

720.093

How effective is the AP1000 igniter scheme in removing hydrogen with an increase in hydrogen generation rates over that of the AP600? The larger core of the AP1000 suggests that there may be a 30 percent increase in hydrogen generation rates. A MAAP4 calculation for a scenario with a conservative oxidization of 100 percent of reactive cladding would provide assurance that the AP1000 igniter scheme is as effective as that of the AP600.

720.094

What effect does the AP1000 hydrogen generation rate have on the mixture classes used in the Sherman-Berman methodology and how does this affect the probability of containment failure at node DTE in the containment event tree?

720.095

Why is the probability of random ignition assumed to be 1 during the intermediate time? The basis for this question is that it is not conservative to assume that ignition is guaranteed in the intermediate time when it comes to global detonations. Presumably the steam content in the uniformly mixed gases inside the containment decreases as the PCS is allowed to cool the containment shell. In the limit (dry mixtures), the concentration of hydrogen is about 14 percent in the AP1000, assuming 100 percent active cladding reaction, or about 19 percent assuming 100 percent reaction of all core zirconium. This mixture is becoming sufficiently sensitive to undergo a transition to detonation, especially if the entire containment is viewed as one confined compartment with a lot of clutter (individual compartments below the operating deck).

720.096

What would be the safety margin basis for containment performance if the uncertainty in the range of steam inerting concentrations was used? The safety margin is less than 1 psi when hydrogen produced from 100 percent active cladding reaction is mixed with air saturated with 55 percent steam. However, there is uncertainty in the steam-inerting limits, as measurements have ranged from 49 percent-63 percent (M. G. Zabetakis, “Research on the Combustion and Explosion Hazards of Hydrogen-Water Vapor-Air Mixtures,” AECU-3327, U. S. Atomic Energy Commission, September 1956.)

Level-1 Follow-up

720.097

(Related to AP1000 RAI 720.024) In the AP1000 PRA, Table 6-1, there are two bases designated as “M” and “P” for success criteria which are not defined. Please provide the missing definitions. Also, there are multiple success criteria basis provided for some of the event cases. Please provide the logic behind, and the decision making process for using a multiple criteria basis.

Note: AP1000 RAI 720.024 was issued on September 18, 2002, ADAMS Accession No. ML022610042.

HISTORY OF PREVIOUSLY-ISSUED
REQUESTS FOR ADDITIONAL INFORMATION

Letter No.	Date issued	ADAMS Accession No.	RAI Nos.	Date of response	ADAMS Accession No.
1	6/26/2002	ML021780568	440.001 - 440.008	7/24/2002	ML022110430
2	8/16/2002	ML022280379	720.001	9/10/2002	ML022560265
3	8/27/2002	ML022390103	420.001 - 420.046, 435.001 - 435.015		
4	9/3/2002	ML022460356	620.001 - 620.043		
5	9/4/2002	ML022470255	210.001 - 210.057		
6	9/5/2002 Reissued 9/18/2002	ML022480440, Reissued RAI ML022610042	440.009 - 440.148, 720.002 - 720.026	9/12/2002 (440.009)	ML022600097
7	9/19/2002	ML022620026	260.001 - 260.003, 261.001 - 261.010, 471.001, 471.010, 472.001, 472.003		
8	9/19/2002	ML022620319	220.001 - 220.019, 230.001 - 230.019, 240.001 - 240.004, 241.001 - 241.003		

Enclosure 2

Letter No.	Date issued	ADAMS Accession No.	RAI Nos.	Date of response	ADAMS Accession No.
9	9/24/2002	ML022620079	250.001 - 250.003, 251.001 - 251.029, 252.001 - 252.009, 281.001- 281.003		
10	9/25/2002	ML022620614	210.058 - 210.070, 261.011 - 261.013, 720.027 - 720.040		

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