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Washington, D.C. 20555

ATTENTION: R. W. BORCHARDT

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of April 15, 1994, May 23, 1994, June 1, 1994, June 8, 1994 and June 15, 1994. In addition, a revision of a response previously submitted are provided. A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A.

These responses are also provided as electronic files in WordPerfect 5.1 format with Mr. Kenyon's copy.

If you have any questions on this material, please contact Mr. Brian A. McIntyre at 412-374-4334.

Nicholas J Liparulo

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory And Licensing Activities

/nja

Enclosure

cc: B. A. McIntyre - Westinghouse
T. Kenyon - NRR

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NTD-NRC-94-4291
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED SEPTEMBER 2, 1994

RAI No.	Issue
220.008R01	Contaniment residual stresses
440.070	Containment closure/reduced inventory operation
440.170	Multiple SGTR analysis
440.178	ADS success criteria
440.194	Recovery factors for LOOP
440.201	Non-coherence in very small break LOCA event tree
440.204	Mission times used in event tree analyses
440.207	Transfers from other event trees
440.217	ADS MOV failure repairs at power
440.222	Sequence 6 of S2 event tree
440.224	Sequences 23 & 29 of ATWS event tree
440.229	Modeling of SG PORVs & SG safety valves
440.237	Dependancy on CCW system in PRA
440.242	No credit for RNS cooling of IRWST
440.243	No RNS dependency on chilled water subsystem
450.010	Main control room habitability system
480.079	Fuel-coolant interaction parameters

Question 220.8

Residual stresses are known to reduce buckling capacity. Describe how residual stresses were incorporated into the analysis (Section 3.8.2.4.2.2).

Response: (Revision 1)

The effect of residual stresses and imperfections on the buckling capacity is included in a single capacity reduction factor, which is intended to address all variables that reduce the theoretical buckling capacity. As described in SSAR Subsection 3.8.2.4.2.2 ~~(see proposed revised text in response to Q220.9)~~, the capacity reduction factor is based on a correlation of theoretical and experimental results~~ and~~. It is equal to 1.0 in the BOSOR-5 analysis performed using an elastic-perfectly plastic bi-linear stress-strain curve. This estimated the buckling capacity of 174 psig. (The buckling capacity was calculated to be 206 psig using an alternate method based on BOSOR-4 elastic analysis and a combined capacity and plasticity reduction factor of 0.385. The 0.385 factor was determined from the lower bound curve of the test data of several ellipsoidal and torispherical heads). The test results and analyses show that the pressures at which the first buckles occurred in the test heads described in SSAR references 22 and 23 were greater than the calculated theoretical buckling pressures determined by BOSOR-5 analyses. These references also confirm that there was substantial capacity in the test heads beyond the occurrence of the first buckle.

Reference 220.8-1 states that buckling occurs in a region of the head where the residual manufacturing stresses are low. This is one reason why the test pressure at which the first buckle occurs is relatively insensitive to residual stresses.

An additional BOSOR-5 analysis using the model described in Figure 220.6-1 and stress-strain curves given in Figure 220.8-1 was performed. The stress-strain curve 'OABC' was used for the segments representing the knuckle region where circumferential stresses are compressive and meridional stresses are tensile due to internal pressure. This curve is suggested to account for the effects of residual stresses on buckling of cylindrical shells due to axial compression and/or external pressure (Reference 220.8-2). For other segments where both the stresses are tensile, bi-linear elasto-perfectly plastic stress-strain curve was used. The point 'A' was selected at 0.00112 in/in ($\approx$0.55 times the yield strain, ϵ_y, ϵ_y = yield strength, σ_y/modulus of elasticity, E), whereas the point 'B' was at 0.006 in/in (about three times ϵ_y).

The point "A" was selected on the basis of ASME Code Case N-284. The nature of results would indicate that the selection of point "B" will not have any significant impact on the overall conclusions. The equation for the curve "AB" was used as $\sigma = E^*\epsilon + K^*(\epsilon\text{-}0.00112)^n$, where constants "K" and "n" were determined from the conditions $\sigma = \sigma_y$ and $d\sigma/d\epsilon = 0$ at "B", respectively. This analysis concluded that there was no potential buckling due to internal pressure. The reason for this was the development of plastic (softer) region over a larger meridional length. The failure mode was an axisymmetric plastic collapse resulting from excessive vertical displacements at the pole. The pressure-maximum displacement curve is given in Figure 220.8-2. Therefore, the values reported in SSAR are considered to be appropriate.

SSAR Revision: NONE

References

220.8-1: M. J. Kemper, "Buckling of Thin Dished Ends under Internal Pressure," Proceeding of the Symposium on Vessel under Buckling Conditions, Institution of Mechanical Engineers, London, 1972, pages 23-32.

220.8-2: C. D. Miller and R. B. Grove, "Current Research Related to Buckling of Shells for Offshore," presented at 15th Annual Offshore Technology Conference in Houston, Texas, May 2-5, 1983.

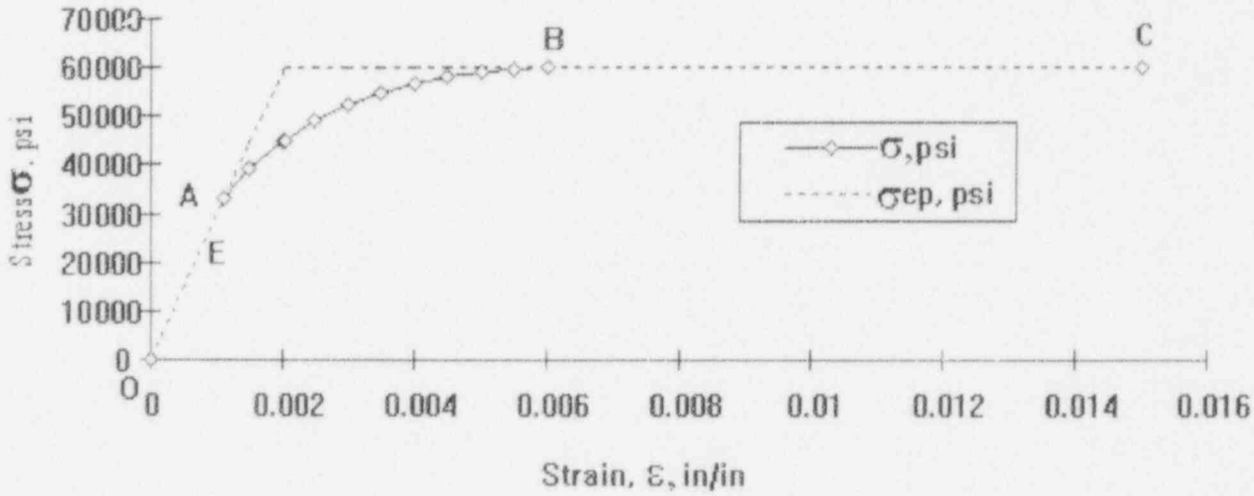

Figure 220.8-1

Stress-Strain Curve

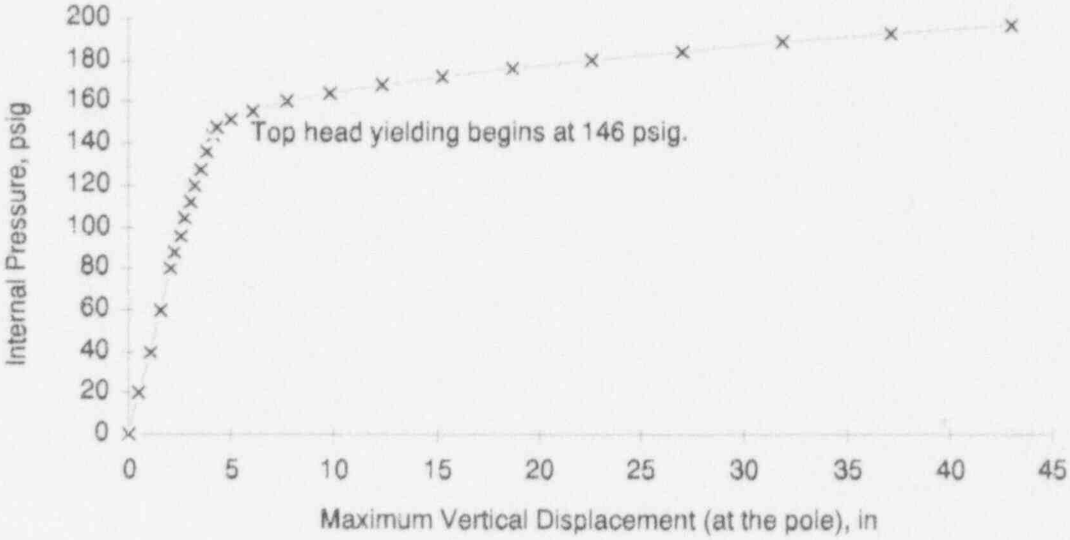

Figure 220.8-2

Internal Pressure - Maximum Displacement Curve

Question 440.70

a. NUREG-1269 addressed the containment closure issue resulting from the Diablo Canyon occurrence on April 10, 1987, and identified the need for procedures to reasonably ensure the capability for containment closure in the event of progression of an accident to core damage conditions. Address this containment closure issue. The discussion should include design considerations such as the need for removal of the equipment hatch and improvements in the AP600 design that facilitate rapid replacement of the hatch should the need arise. Similarly, address other containment penetrations and potential bypass paths.

b. NUREG-1269 states that "the design of the nuclear steam supply system did not appear to provide detailed provisions for mid-loop operation." Identify and discuss each of the design changes in the AP600 design that establishes the adequacy of the AP600 design for reduced RCS inventory operation.

Response:

a. Establishment of containment integrity during reduced inventory operations is provided by several design features and administrative controls.

   - The equipment hatches are required to be closed by Technical Specifications during reduced inventory operations (Technical Specification 3.6.3). Should it be necessary to bring equipment into containment or remove equipment from containment via the equipment hatches the activity will be performed in mode 5 with the loops full or in mode 6.

   - The personnel hatch design includes the following provisions to re-establish or provide for containment integrity:
     - Each personnel hatch has two double-gasketed doors.
     - The doors are mechanically interlocked to prevent simultaneous opening of both doors and to allow one door to be completely closed before the second door can be opened. Technical Specifications permit the interlock to be bypassed only during plant shutdown (restricted to mode 5 with the loops filled and mode 6 - See Technical Specification 3.6.2) by using special tools and procedures.
     - Personnel hatches are designed for manual closure without power assistance. Closure is independent of the availability of power.

   - Technical Specifications require containment pressure boundary integrity and/or operability for the containment penetrations during modes 1, 2, 3, 4 and 5 with loops not full consistent with the need to prevent release of radioactivity following an accident and to mitigate a loss of RCS inventory or loss of RCS cooling event. In mode 5 with loops full and mode 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these modes and large inventory of coolant. Therefore, containment air locks are not required to be operable in mode 5 with loops full and mode 6 to prevent leakage of radioactive material from containment.

b. The AP600 incorporates features to address midloop operations as discussed in SSAR section 5.4.7.2.1. These features were specifically incorporated as a result of the industry experience discussed in NUREG 1269, NUREG 1449 and Generic Letter 88-17. These features include:

- Loop piping offset that allows midloop operations to be performed at a much higher RCS water level
- Step nozzle connection that significantly lowers the allowable RCS level during midloop
- No RHR throttling is required during midloop, even with the RCS at saturated conditions
- Self-venting suction line that eliminates local high points and allows for immediate restart of the RHR pumps after air-binding
- Redundant, remote hot leg level readout in the main control room
- Improved RCS draining procedures

Additional details are provided in Sections 1.9.5 and 5.4.7.2.1 and in response to Questions 440.53, 440.55, 440.56, 440.58, 440.71 and 440.72.

SSAR Revision: NONE

Question 440.170

Section 6.3.3.3.1 of the SSAR discusses the scenario for a single steam generator tube rupture, and states that if the non-safety-related systems fail to start, the core makeup tanks and the PRHR heat exchangers automatically actuate. The section states that during these events, the plant conditions are stabilized without actuating the ADS. Because various stages of the ADS are actuated based on the CMT levels and subsequent delay time, will the CMT maintain its level so that the ADS remain unactuated for a multiple tube rupture? If the ADS actuates, will the primary system over-depressurize, which would result in back flow of the secondary unborated water into the primary system with a subsequent reactivity increase? Although the staff agrees with the position stated in the January 22, 1993 response to Q440.27 that the multiple steam generator tube rupture (SGTR) scenario should not be a design basis event and the licensing design basis is for consideration of a single SGTR, the staff is concerned with a multiple SGTR scenario for a passive PWR. Therefore, provide a realistic analysis of the rupture of 3 to 5 steam generator tubes.

Response:

A MAAP4 realistic analysis of a multiple steam generator tube rupture will be provided to support the Final Safety Evaluation Report. The analysis will be provided by the end of October, 1994.

SSAR Revision:NONE
PRA Revision: NONE

Question 440.178

The ADS is credited for in many sequences of the event trees in the PRA. The staff is concerned with whether or not it is capable of performing the intended function in the scenarios. Provide the following information (see also Q440.177):

a. Table C5-2 of the June 26, 1992 PRA lists the success criteria of the ADS under different boundary conditions. For each entry of the table, indicate the thermal hydraulic analysis or tests performed that verify the success criteria used. Refer to sections, tables, and figures of Chapter 15 of the SSAR or Appendix J of the PRA. If no thermal hydraulic analysis or test corresponds to the entry of the table, indicate the analysis or test that provides bounding justification of the success criteria used. Discuss why the analysis or test justifies the success criteria.

b. For each thermal hydraulic analysis or test, specify the IE, initial condition, and the status of systems such as the CVS, the PRHRS, and the CMT. Provide a discussion on the behavior of the systems and how it affects the scenario. Demonstrate that the operation of the CVS pump will not adversely affect gravity injection and the ADS success criteria.

Response:

Information in response to this question will be provided in Sections 6 and 7 of the AP600 PRA, Rev. 2.

SSAR Revision: NONE

PRA Revision:

The information requested will be provided in Revision 2 of the PRA.

Question 440.194

Regarding the loss of offsite power (LOOP) event tree (event T_E, Figure F-9 of the PRA) for RTNSS, how were the recovery factors handled? Recall that in an extended station blackout, the system automatically actuates the ADS at 24 hours.

Response:

There are two recovery factors modeled for the loss of offsite power event tree. They are for offsite power recovery in 1/2 and 24 hours, event tree top events R05 and R24, respectively. In the RTNSS evaluation of the loss of offsite power (LOOP) event documented in Reference 440.194-1, the probabilities for failure to recover offsite power for both time periods were set to 1.0.

The PRA does not model ADS actuation at 24 hours. The inclusion of automatic ADS actuation at 24 hours in the fault tree models is not expected to impact core damage frequency results.

Reference:

440.194-1 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Summary Report," September 1993.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.201

It appears that in the very small-break LOCA event tree, sheet 2 of Figure F-18 of the PRA, sequences 1 to 8 give an example of non-coherence (i.e., the top event frequency may either increase or decrease with increasing failure probability, depending on the values assumed by other variables) in the logic. In this example, successful partial depressurization can lead to core damage, while its failure leads to "OK." Is this an artifact of the mission time assumption? Does event PP (stuck-open pressurizer valve) ever help or does it always hurt? (see the event tree for secondary to primary mismatch; PP turns the scenario into an S1). If depressurization is needed, is the conditional probability of CD lower given event PP, or higher? Do sequences ending in T_{SOV} on the T_E tree have a lower conditional probability of CD, or higher, as a result of stuck open valve?

Response:

The very small LOCA event tree is coherent. For the very small LOCA event, when the chemical and volume control system and the passive residual heat removal system is operating, more than six hours are available to the operator in cases where depressurization fails. It is assumed that cold shutdown is possible within this time frame using normal shutdown systems, such as startup feedwater and normal residual heat removal after reactor coolant system pressure reduction. These sequences of events assume minimal reactor coolant system inventory loss through the break. This is reflected in Sequences 8 and 26 on sheet 2 of Figure F-18 resulting in "OK" end states. However, when depressurization succeeds, reactor coolant system inventory needs to be replenished by either gravity injection, when full automatic depressurization actuates, or by normal residual heat removal in injection mode when automatic depressurization actuation partially depressurizes the reactor coolant system. In the very small LOCA event tree, sequences where partial depressurization occurs and normal residual heat removal fails will eventually lead to core uncovery and subsequently to core damage because reactor coolant system inventory is lost through the automatic depressurization system valves without a sufficient makeup capability.

During certain transient initiating events and/or the loss of feedwater, the pressurizer safety valves open to limit the reactor coolant system pressure. Top event PP models the failure of at least one valve to reclose, given successful opening. The PP event, in these cases is always routed to the medium loss of coolant accident event tree. The automatic depressurization system failure probability used during quantification of the medium LOCA event tree is not conditioned on the PP event. In other words, the opening of a pressurizer safety valve does not affect the success criteria for the automatic depressurization system.

The probability that a safety valve fails to reclose, given successful opening, is not conditioned upon the initiating event. Therefore, the same failure probability is used regardless of the accident scenario.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.204

Define the mission time of the event tree analyses and discuss its effect on how the "OK" sequences are terminated. For example, in the case of sequence 37 of event tree F-9 for a loss of offsite power, the PRHRS is in operation and can continue for 72 hours without the condensate returned to the IRWST. Here a mission time of 72 hours appears achievable. However, the mission time used in calculating the valve failure probability is probably 24 hours. In those cases in which gravity injection is used, how long can gravity injection last? When is the containment sump recirculation needed? Is it before 24 hours?

Response:

For equipment required to remain in operation or "running" for successful core cooling after an initiating event, a mission time of 24 hours was used. Sequences are terminated with an "OK" end state when the sequence does not lead to core damage within the 24 hour mission time frame. A core damage sequence is one in which the peak clad temperature exceeds 2200 °F.

For large and medium LOCA events, switching to containment sump recirculation is required. For all other events gravity injection will last beyond 24 hours, however, switching to recirculation before 24 hours is conservatively modeled in the event trees.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.207

In the event trees of the PRA, some sequences lead to a transfer to event trees of other IEs. These sequences have some pre-defined conditions that may be different than that assumed for the event trees they transfer to. For example, sequence 21 of the loss of instrument air event tree is a transfer to a medium-break LOCA. How do you account for the fact that the instrument air is lost?

Response:

When the loss of component cooling water, service water, or instrument air systems event trees transfer to another event tree, such as the medium LOCA or the main steam line stuck open safety valve event trees, the PRA does not explicitly account for the loss of these systems in the other event tree. Therefore, systems in the "downstream" event tree did not account for "upstream" system failures because a simplified quantification process was employed. Changes to the quantification process could account for the upstream system failures, however, the changes are not expected to have significant impact on the PRA results. For the specific event tree described, the loss of these systems impacts only the mitigation contribution from the normal residual heat removal system injection and does not affect the passive, safety-related systems and components that also provide mitigation for this event. For the affected trees, additional event tree quantification analysis will be performed to evaluate the impact of this modeling approach when the PRA is revised.

SSAR Revision: NONE

PRA Revision:

The impact discussed above will be evaluated for Revision 2 of the PRA.

Question 440.217

What fraction of ADS MOV failures can be repaired at power?

Response:

A review of the ADS system description and the valve P&ID for the ADS system was performed to determine the fraction of ADS MOV failures that can be repaired at power. These valves are normally closed valves and are not isolable. They will also be located in a high radiation zone area inside containment (Zone 4 or 5); therefore, it will not be possible to perform maintenance on these valves while at power.

SSAR Revision: NONE

Question 440.222

Provide the following information regarding sequence 6 of the S2 event tree of the PRA:

a. The PRHRS provides depressurization and there is some pressurizer relief, but not enough for full depressurization to succeed without PRHR. What is the flow pattern in the PRHRS under these conditions?

b. In sequence 6 of the S2 event tree, is one HX enough to support credit for (PRHRS + Stage 4) operation of full depressurization, or, if one HX is isolated, does this success option become marginal?

Response:

a. Small LOCA (event S2) sequence 6 models the following: successful reactor coolant pump trip, core makeup tank injection, and passive residual heat removal operation, followed by the failure of full depressurization via the automatic depressurization system, but success of partial depressurization (via the automatic depressurization system) to the normal residual heat removal operating pressure, with successful normal residual heat removal operation. For this sequence, the success criterion for full depressurization required the functioning of either 3 out of 4 automatic depressurization system stages 2 and 3 valves, or 1 out of 2 automatic depressurization system stage 4 valves (both based on the automatic depressurization system design at the time PRA Rev. 0 was performed). Success of partial depresssurization required the operation of only one automatic depressurization system valve. Since passive residual heat removal succeeded in this sequence, the automatic depressurization system success criteria did not require operation of automatic depressurization system stage 1.

   For this sequence, the important factor is the operation of the automatic depressurization system. Passive residual heat removal allows reactor coolant system pressure to be reduced to the point where the stage 4 valves can open. After that point, passive residual heat removal operation is no longer required, and any change in passive residual heat removal flow patterns does not affect the chances of success. Since the stage 1, 2, and 3 automatic depressurization system valve connections are independent from those of the passive residual heat removal system, no interaction is expected. Once stage 4 automatic depressurization system valve opening is possible, passive residual heat removal is no longer required.

   Sequence 14 of the S2 event tree is an example of a success path equivalent to that of sequence 6, but without passive residual heat removal. For sequence 14, the success criteria for automatic depressurization are more restrictive, requiring the operation of at least one stage 1, 2, or 3 valve before stage 4 is credited. This is because passive residual heat removal is not available to reduce reactor coolant system pressure in this sequence.

   In addition to the above, the water injected from the core makeup tank is at a lower temperature than the reactor coolant system, and will also contribute to reactor coolant system pressure reduction.

b. Information in response to this question will be provided in Section 6 of the AP600 PRA, Revision 2.

SSAR Revision: NONE

PRA Revision: The information requested will be provided in Revision 2 of the PRA.

Question 440.224

Describe the scenarios of sequences 23 and 29 of the ATWS event tree of the PRA. In particular, assuming that startup feedwater failed, what will actuate the PRHRS? If the pressurizer safety valves are lifted, what is the flow pattern in the PRHRS? Is one PRHRS HX sufficient? What will actuate the CMT? Will the CMT level become low enough to actuate the ADS? If not, what will actuate the ADS? How is the success criterion for the ADS determined?

Response:

Sequences 23 to 29 contain the mechanical failure of control rods to insert. For these sequences it is assumed that the protection and safety monitoring system and the necessary actuation signals are available. For these sequences, the passive residual heat removal system actuates on the low steam generator level narrow range with low startup feedwater flow to any steam generator or on the low steam generator level wide range signals. Models for the signals are contained in the passive residual heat removal fault trees. The passive residual heat removal heat exchangers provide reactor coolant system heat removal while steam is venting to the pressurizer, such as during the operation of the automatic depressurization system valves or when the pressurizer safety valves are open. Successful passive residual heat removal cooling can be accomplished with 1 out of the 2 passive residual heat removal heat exchangers.

If boration by the chemical and volume control system is not achieved, the alternative means of core heat removal and boration can still be initiated automatically by the actuation of the core makeup tank system on low steam generator level coincident with high reactor coolant system temperature. The actuation of this system, in turn, produces depressurization of the reactor coolant system by opening the automatic depressurization system valves. The addition of borated water from the core makeup tanks provides the necessary boration to achieve subcriticality. The protection and safety monitoring system and the diverse actuation system also provide for manual actuation of the core makeup tanks and automatic depressurization system valves.

The automatic depressurization success criteria for the ATWS event is assumed to be the same criteria used for other transient events. AP600 success criteria are discussed in Appendix J of the AP600 PRA report.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.229

It is not clear whether or not secondary steam relief through the SG PORVs or safety valves was modelled in the PRA. The documentation discusses steam dump but not the SG PORVs and safety valves. Some event tree branches use "VAL2" or "VAL3." It is not known where they were documented. Clarify the above and provide the assumptions regarding steam relief.

Response:

For transient events following the loss of secondary heat sink, the main steam line safety valves and power-operated relief valves are expected to open and close. Failure to open is not modeled because of the large number of safety valves provided. The failure of safety valve, power operated relief valve, and block valve closure is modeled in the event trees as top event PM. Two cases are used for this top event, depending on the transient. Case PM assumes that the power-operated relief valve and one safety valve have opened to relieve pressure in each line. Case PM1 assumes that the power-operated relief valve and two safety valves have opened to relieve pressure in each line.

"VAL2" is the fraction of loss of feedwater events in which the condenser is unavailable. "VAL3" is the fraction of secondary to primary side power mismatch events in which the startup feedwater system is unavailable. Revision 2 of the PRA will utilize separate event trees for the various initiating event cases, thus eliminating the need for any "VAL" case names.

PRA Revision: Revision 2 of the PRA will include the changes discussed above.

SSAR Revision: NONE

Question 440.237

In the PRA, where is the dependency on the CCW system modeled? For example, the RNS is used to provide injection in many event trees. In these cases, is the dependency on the CCW modeled? Why does loss of the system in the September 24, 1993 focused PRA lead to a 99% increase in the core damage frequency?

Response:

The dependency on the component cooling water system is modeled in the fault trees of the systems that it supports. For example, fault tree RNR contains a reference to basic event SUB-CCT. The event SUB-CCT represents the fault tree for the component cooling water system.

The fault tree models for the normal residual heat removal system when operating in the injection mode include the dependency on the component cooling water system.

For the RTNSS evaluation, the loss of component cooling water initiating event frequency was combined with the loss of offsite power events and several other transients events to create one "general" loss of offsite power and transient initiating event. This simplified the quantification process since the event tree for these initiating events is the same for the focused PRA sensitivity study, as discussed in Subsection 2.1.1 of reference 440.237-1. The percent contribution to core damage frequency for this event increases due to the removal of credit for nonsafety-related system mitigation functions.

Reference:

440.237-1 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Summary Report," September 1993.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.242

Section 5.4.7.1.2.3 of SSAR states that the RNS system is initiated within 2 hours of actuation of the PRHR to remove heat from IRWST. This appears to not be modelled in the PRA. Why?

Response:

The normal residual heat removal system provides, as a nonsafety-related function, the capability to remove heat from the in-containment refueling water storage tank during passive residual heat removal operation. Such normal residual heat removal system operation precludes steaming and associated circulation of in-containment refueling water storage tank inventory through the containment (i.e., steaming, condensation on the containment vessel, and return to the in-containment refueling water storage tank, given appropriate valve alignment). However, as indicated on page F-2 of the AP600 PRA, the in-containment refueling water storage tank inventory is sufficient for continuous passive residual heat removal operation for more than 72 hours without any water replenishment. Inclusion of the normal residual heat removal system in the model would provide additional but unnecessary redundancy. The PRA therefore does not include this mode of normal residual heat removal system operation.

The event tree models do include credit for normal residual heat removal system operation for decay heat removal, following failures of startup feedwater and passive residual heat removal, given successful reactor coolant system depressurization via the automatic depressurization system.

SSAR Revision: NONE

PRA Revision: NONE

Question 440.243

Section 9.2.7.2.4 of SSAR states that the low capacity chilled water subsystem provides cooling to both the makeup pump room and the RNS pump room. What is the reason that the dependency of the RNS is not modelled while that of the makeup pump is?

Response:

The low capacity chilled water subsystem supplies chilled water to the cooling coils of the chemical and volume control system makeup pump and normal residual heat removal pump room coolers. The chemical and volume control system makeup pump's motor and lube oil are air-cooled and require the chilled water system to keep room temperature within the pump's safe operating temperature range. The makeup pump's dependency on chilled water is modeled in the chemical and volume control system fault trees.

The normal residual heat removal pump seals are water cooled, with cooling water provided by the component cooling water system. The normal residual heat removal pump motor is air cooled. For the normal residual heat removal system pumps, preliminary analysis indicates that, on the loss of room cooling, the pumps can operate for a period of greater than the mission time of the PRA before the motor fails due to overheating. This analysis takes into account the layout and location of the pumps and the low heat load of the motors.

SSAR Revision: NONE

PRA Revision: NONE

Question 450.10

Address the following concerns regarding the main control room habitability system (VES):

a. Table 9.4.1-1 of the SSAR (page 9.4-40) shows inleakage of 10 SCFM through the MCR access doors and 36 SCFM through the MCR/TSC equipment ductwork (operating), and outleakages of 20 SCFM through the MCR structure and 443 SCFM through the MCR/TSC HVAC equipment and ductwork (operating). This indicates that in the emergency mode, there is a possibility of a total of 46 SCFM inleakage into the MCR envelope which is not filtered. Has this amount of inleakage been included in the dose calculations? What are the provisions for limiting unfiltered inleakage into the MCR?

b. The VES is designed for a 20 SCFM flow, which does not assure proper cooling needs for the limited number of MCR operators to function appropriately within human tolerance thresholds, including consideration of their survivability in a high stress environment and the need for MCR equipment operability during and beyond 72 hours. This conclusion is based on an MCR envelope initial temperature of 80 °F and initial temperature rise of 15 °F during the first 72 hours of a postulated LOCA scenario. Also, based on its experience with currently operating plants and the evolutionary plant designs, the staff believes that a five person limited MCR occupancy with long shift hours (12 hrs.) is not a realistic assumption. Address these issues, provide any revised assumptions, and show how the design of the AP600 provides the safety-related ventilation and cooling functions for the MCR equipment and its occupants during accident conditions.

c. The VES design provides safety-related connections to hook up non-safety-related portable equipment that will cool the MCR envelop after the initial 72 hours. Provide a detailed evaluation describing how this arrangement replaces the safety-related cooling functions during the entire duration of a given LOCA scenario for the MCR occupants and equipment operability. Also, provide the scope of responsibilities for Westinghouse and the COL applicant for the above arrangement, and revise the SSAR accordingly.

d. Describe how a positive pressure of 1/8-inch water gauge is maintained for given LOCA conditions, considering worst case unfiltered inleakages of 46 SCFM and outleakages of 463 SCFM, while providing only 20 SCFM bottled air supply inside MCR. Include this in the SSAR.

e. Westinghouse has indicated that a temperature increase of 15 °F over a 72 hour period is a bounding condition for the MCR design. Provide the rationale for the capability to maintain this threshold condition beyond 72 hours if accident conditions continue. Also, during the March 23, 1993, meeting, Westinghouse indicated that the MCR temperature profiles would be completed within 2 to 3 months using the WGOTHIC model. Provide a description of the assumptions and the results of the MCR modeling. Include this in the SSAR.

f. In the January 22, 1993, response to Q450.1, the acceptable CO_2 concentration level was evaluated based on only five persons inside the MCR envelope with a net MCR envelope volume of 42,260 ft^3. Based on its experience with currently operating plants and the evolutionary plant designs, the staff believes that a five person limited MCR occupancy with long shift hours (12 hrs.) is not a realistic assumption. Re-evaluate the allowable CO_2 concentration level that will provide a habitable environment for the MCR occupants based

on a more realistic maximum bounding number of people occupying the MCR envelope during accident conditions.

g. Describe in Section 1.9.3 of the SSAR how the AP600 conforms with the guidance of Sections 2.2.1 through 2.2.3 and 6.4 of the SRP, and Attachment 1 to NUREG-0737, TMI Task Action Plan Item III.D.3.4, to assure that control room operators are adequately protected against the effects of accidental releases of toxic and radioactive gases, and can safely operate the plant under normal conditions or shutdown the plant under design basis accident conditions.

Response:

a. The nuclear island nonradioactive ventilation system (VBS), is a nonsafety-related system that provides defense-in-depth for the protection of the operators following a postulated accident. In the supplementary filtration mode of operation the main control room is pressurized by filtered air inflow and there is recirculation cleanup of the air (this is more fully described in SSAR Subsection 9.4.1.2.4). The 56 cfm unfiltered inleakage into the main control room and technical support center presented in SSAR Table 9.4.1-1 is applicable <u>only</u> to the situation in which the VBS is operating in the supplementary filtration mode. In an accident situation in which there is a release of radioactivity, if the VBS is operational, the VBS is used in the supplementary filtration mode in preference to actuating the VES. If the VBS is not operable, the VES is actuated. During VES operation the only unfiltered inleakage path is that of the door to the main control room during ingress/egress and, as discussed in the response to RAI 450.2, the unfiltered inleakage during VES operation is 0.24 cfm.

The determination of doses to the operators in the main control room as a result of the postulated large break LOCA provided in SSAR Subsection 15.6.5.3 assumes that the VES is actuated at the very beginning of the accident and the analysis assumes an unfiltered inleakage rate of 0.3 cfm which bounds the 0.24 cfm value.

Additionally, in Revision 1 of the response to RAI 470.9, analysis of the doses resulting from the large break LOCA when using the source term from draft NUREG-1465 is presented. In this analysis the dose to the operators is calculated in two different ways: one with the VES in operation and the other with the VBS operating in the supplementary filtration mode. The 56 cfm unfiltered inleakage was included in the VBS main control room dose analysis. The list of assumptions provided in the response to RAI 470.9 states that unfiltered inleakage is 40 cfm. The remaining "unfiltered inleakage" is included in the filtered air inflow because the point of inleakage is the ductwork upstream of the air filter.

The provisions for limiting unfiltered inleakage into the main control room during VES operation have been discussed in RAI 450.2 Revision 1 regarding damper leakage and ingress/egress, and in RAI 450.1 regarding low-leakage construction. Additional discussion on this topic is contained in the response to item d) below.

b. The nuclear island non-radioactive ventilation system (VBS) limits the initial temperature range of the main control room to between 67 °F and 78 °F.

The 20 scfm air flowrate into the main control room during VES operation was determined in design evaluations to be sufficient to maintain the main control room at a slight positive pressure with respect to its surroundings. Testing will confirm this capacity during plant startup, as identified in ITAAC Table 3.2.6-1. Should normal HVAC not be available, cooling is provided by the passive heat sinks of the walls and ceiling of the main control room, which is designed to limit the temperature increase to a maximum of 15 °F. After 72 hours, if normal HVAC remains unavailable, portable cooling units are utilized for continued main control room cooling. This aspect of VES operation is described in SSAR Subsection 6.4.2.2.

The air flowrate of 20 scfm is sufficient to maintain acceptable levels of carbon dioxide concentration in the main control room for at least 11 occupants. Currently, Westinghouse is in the process of evaluating the passive cooling capability for scenarios in which more than five people will be present in the main control room during VES operation. The results of this evaluation will be submitted for NRC review by March 30, 1995.

c. The portable cooling units are brought into the main control room from offsite within 72 hours, and are connected to hookups for power and cooling water. The portable coolers are water cooled, and are connected to permanently installed piping which extends through the control room wall. Outside the control room wall, a temporary hose connection is made to temporary air cooled chilled water units. In addition to the coolers, portable free standing fans may be placed within the main control room to provide the distribution of air cooled by the portable coolers. The portable equipment is only used in the unusual circumstance in which the VBS is not available. The AP600 design includes the functional requirements and permanent safety related hookups for the portable cooling units. The Combined License holder is responsible for providing the equipment to the site within 72 hours. Additional discussion is contained in item e) below.

d. The following provides clarification of the values provided in Table 9.4.1-1 of the SSAR. The table provides values for operation of the nuclear island non-radioactive ventilation system (VBS). Specifically, the 46 cfm inleakage value cited in the table is for VBS operation in supplemental filtration mode. When the main control room emergency habitability system (VES) is actuated, the control room envelope is isolated from the non-radioactive ventilation system by automatic closure of the isolation dampers located in the ventilation system ductwork. In this scenario, the 46 scfm unfiltered inleakage and 463 scfm outleakage do not apply. This is discussed in both item a), above, and in SSAR Subsections 9.4.1.2.4 and 6.4.3.2.

For VES operation, the compressed air supply flowrate is 20 scfm for 1 train operating, or 40 scfm if both trains are operating. For conservatism, and considering the single failure criteria, one train has been assumed to fail for the pressurization calculation. In considering outleakage, calculations have been performed to determine the magnitude of leakage through the main control room envelope. From the

individual component leak rates computed, more than 95 percent of the total esimated leakage is attributable to the doors. The remaining leakage paths are insignificant. Therefore, 20 cfm of VES ventilation airflow is more than adequate to maintain a 1/8 inch water gage positive pressure differential inside the main control room envelope.

e. After 72 hours, portable cooling units will be brought in from offsite and used to provide cooling and ventilation needs for the main control room. The coolers are sized to accept the continued heat loads from the main control room, and to cool the heat sinks which will enable the room to return to a more moderate temperature.

As indicated previously in the response to item b) above, Westinghouse is in the process of performing evaluations for the main control room envelope with respect to ventilation, pressurization, and passive cooling capability. Evaluation of main control room occupancy for more than five persons is a subset of this workscope. Assumptions and results will be submitted for NRC review by March 30, 1995.

f. Based upon the response to Q450.1 and item b, above, the MCR CO_2 concentration level supports long term occupancy for at least 11 main control room occupants.

g. The AP600 main control room operator habitability requirements under accident conditions are provided by VES. The VES is designed to satisfy safety-related system design and seismic Category I requirements. The VBS is a nonsafety-related system, and does not provide a safety-related function except for the isolation of the main control room envelope. The supplemental air filtration subsystem of VBS operating during abnormal modes is a defense-in-depth function when VBS is operable and supporting systems are available. There is no credit taken for VBS filtration operation in the main control room habitability analysis under accident conditions in which the VES system is used. AP600 does not have onsite chlorine or other toxic chemicals storage facility, and offsite chlorine or toxic gas release is a site-specific issue. Listed below is a table demonstrating conformance with Attachment 1 to NUREG-0737, TMI Task Action Plan Item III.D.3.4:

COMPARISON OF AP600 DESIGN AND NUREG 0737, SECTION III.D.3.4
INFORMATION REQUIRED FOR CONTROL ROOM HABITABILITY EVALUATION

NUREG 0737, SECTION III.D.3.4, ITEM NO.	AP600 DESIGN

1. Control room mode of operation, i.e., pressurization and filter recirculation for radiological accident isolation or chlorine release.	The nonsafety-related VBS provides a pressurization and filtration mode of operation on detection of a high radioactivity level at the main control room (MCR) supply duct. VBS provides a first line of defense, and adequately protects MCR operators under design basis accident conditions if other nonsafety-related support systems (ie., chilled water and ac power) are available. The VES provides the safety-related requirements of pressurization and breathing air to shutdown the plant under design basis accident conditions if the nonsafety-related VBS is not available. The VBS provides a 100 % recirculation mode with no outside air supply on detection of smoke in the outside air intake. This mode of operation could also be applied to chlorine or toxic chemical releases; Because toxic chemical sources are site-specific, no detection devices have been provided in the AP600 design for chlorine or toxic chemicals. If VBS and its support systems are not operable, VES provides the safety-related function of control room habitability where protection from chlorine or toxic chemical release is required.
2. Control Room Characteristic	
a. Air Volume Control Room	• Refer to RAI 450.2, Revision 1.
b. Control room emergency zone (control room, critical files, kitchen, washroom, computer room, etc.)	The AP600 control room emergency zone includes the main control console area, clerk area (critical files area), shift supervisor area, operator area, tagging area, kitchen, and toilet.
c. Control room ventilation system schematic with normal and emergency air flow rates.	• Refer to Figure 9.4-1 for VBS system schematic. • Refer to Table 9.4.1-1 for VBS air flow rates. • Refer to Figure 6.4.-2 for VES system schematic. • VES air flowrate of 20 scfm for one train operating; 40 scfm for both trains operating.
d. Infiltration leakage rate.	Refer to RAI 450.2, Revision 1.
e. High efficiency particulate air (HEPA) filter and charcoal absorber efficiencies.	Refer to Table 9.4.1-1 for HEPA filter and charcoal adsorber efficiencies.

f. Closest distance between containment and air intake.	Refer to Section 6.4.4.
g. Layout of control room, air intakes, containment building, and chlorine, or other chemical storage facility with dimensions.	• Refer to Figure 6.4-1 for the layout of main control room and Figure 1.2-10 for the location of the VBS outside air intake. • AP600 does not have onsite chlorine or other toxic gas storage.
h. Control room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc.	An analysis was performed that considered streaming from containment and the surrounding radioactive cloud shine, with the conclusion that the contribution to control room operator dose from external radiation effects, in addition to doses due to inleakage effects, was well within the GDC 19 allowable limits.
i. Automatic isolation capability damper closing time, damper leakage and area.	• MCR envelope isolation dampers closing time of 4-8 sec. • Refer to Section 9.4.1.2.4 for damper description. • The MCR envelope supply and return air isolation dampers are 24 in. diameter and the toilet exhaust isolation dampers are 6 in. diameter.
j. Chlorine detectors or toxic gas (local or remote)	Not applicable. AP600 does not have onsite storage of chlorine or other toxic gases. Therefore, chlorine or toxic gas detection system for onsite chlorine or toxic gas release is not provided.
k. Self contained breathing apparatus availability (number).	Refer to Section 6.4.2.2.2.
l. Bottled air supply (hours supply)	Refer to Section 6.4.3.2.
m. Emergency food and potable water supply (how many days and how many people.	Refer to Section 6.4.2.2.2.
n. Control room personnel capacity (normal and emergency).	Currently being evaluated.
o. Potassium iodide drug supply	Refer to Section 6.4.2.2.2.
3. Onsite storage of chlorine and other hazardous chemicals	AP600 does not have onsite storage of chlorine and other hazardous chemicals

a.	Total amount and size of container	Not applicable
b.	Closest distance from control room air intake.	Not applicable
4.	Offsite manufacturing, storage, or transportation facilities of hazardous chemicals.	Offsite manufacturing, storage, or transportation facilities of hazardous chemicals is a site-specific issue.
a.	Identify facilities within a 5 mile radius	Not applicable
b.	Distance from control room	Not applicable
c.	Quantity of hazardous chemical in one container.	Not applicable
d.	Frequency of hazardous chemical transportation traffic (truck, rail, and barge).	Not applicable
5.	Technical specifications (refer to standard technical specifications).	
a.	Chlorine detection system.	Not applicable. Chlorine detection system and Technical Specification requirements are not required for the standard plant.

b. Control room emergency filtration system including the capability to maintain the control room pressurization at 1/8 in. water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements.

- VBS filtration system is designed as two redundant 100% capacity 4,000 scfm units to maintain the control room pressurization at 1/8 in. water gauge during a high radiation mode when ac power is available as the first line of defense for VES.
- VBS MCR envelope isolation dampers require verification of operable status and will actuate upon demand (verification of isolation by test signals and damper closure times) per Section 3.7.6 of the Technical Specifications.
- VBS supplemental air filtration unit components which include HEPA filters and charcoal adsorbers are tested in accordance with ASME N509-1989 and ASME N510-1989.
- VES compressed air storage bottles require verification of sufficient pressurized volume.
- VES pressure control valve in each VES train requires verification of operability to ensure that a sufficient supply of air is provided, and that uncontrolled air flow into the MCR will not occur.
- VES isolation valves outside the MCR require verification of being locked open (or sealed open) to ensure air delivery pathway is open.
- VES isolation valves within the MCR envelope require verification of being operable.
- VES automatic startup test provides verification that both trains of equipment start on receipt of safety-related actuation test signals.
- VES functional test is required to verify that an air supply of 20 cfm will pressurize the MCR to the minimum value of 1/8" wg.

SSAR Revision:

1. Modify the 5th paragraph of Subsection 6.4.2.2 as follows:

In the unlikely event that power to the nuclear island non-radioactive ventilation system is unavailable for more than 72 hours, then cooling of the MCR ~~, instrumentation and control rooms, and/or dc equipment rooms~~ is provided by portable spot cooling units brought into the main control room from offsite within 72 hours. The portable coolers are water cooled, and are connected to permanently installed piping which extends through the control room wall. The piping penetrations are designated as Class C, seismic Category I. Outside the control room wall, a temporary hose connection is made to temporary air cooled chilled water units. In addition to the coolers, portable free standing fans may be placed within the main control room to provide the distribution of air cooled by the portable coolers. The portable equipment is only used in the unusual circumstance in which the VBS is not available. The portable units are standard commercial units and are sized to maintain the rooms at temperatures that allow for long term occupancy of personnel in the MCR. ~~and long term operation of equipment in the instrumentation and control rooms and DC equipment rooms. The MCR is provided with Class C, seismic Category I penetrations for heat rejection and power supply connections for the portable cooling units.~~

2. Replace the 1st paragraph of Subsection 6.4.2.4 with the following:

~~The MCR envelope is contructed so that the air flow supplied by the main control room emergency habitability system is sufficient to maintain a 1/8 inch water gauge positive pressure differential with respect to the adjacent areas. This prevents infiltration of contaminated air into the MCR envelope as long as the main control room emergency habitability system is operating.~~

The main control room envelope is designed for low-leakage. The MCR envelope main entrance is designed with an airlock-type double-door vestibule, and is designed with position indications that provide alarms in the MCR when not fully closed.. The emergency egress door is normally closed and will remain closed under design basis source term conditions.

The construction techniques used for the MCR envelope to minimize the leakage rate include the sequencing of concrete pours to minimize cold joints which could represent a leak path. Caulking and flashing will be provided at penetration inserts cast through walls or slabs. Coating the exterior or interior surfaces of the MCR envelope (walls, floor and ceiling) with low permeability paint/epoxy sealant will be performed as required.

During the MCR envelope isolation mode in an accident situation, there is no direct communication with the outside atmosphere, nor is there any communication with the normal ventilation system. Leakage from the MCR envelope will be the result of an internal pressure of 1/8-inch WG provided by VES operation.

3. Change SSAR Section on *Component Description* from Section 6.4.2.2.2 to 6.4.2.3

Question 480.79

The staff is assessing fuel-coolant interaction energetics and the resulting impulsive loads for the AP600 design. The staff's goal is to provide a best estimate scenario and a conservative variation from the best estimate. To this end, provide the following information on the AP600 design:

a. Core Inventory - Provide the amount of UO2, Zr, and steel in the core plus any additional steel from the upper or lower plenums (relatively thin structures) that could potentially melt and be added to the molten pool in the lower plenum prior to vessel breach. Also, what are the "best estimate" late in-vessel conditions given by MAAP (e.g., corium composition, and in-vessel pressure and temperature)?

b. Geometry - This information should include drawings of the cavity region at various elevations from Elevation 66'-6" to the top of the reactor pressure vessel at Elevation 107'-2". Westinghouse has provided the plan drawings at elevations 66'-6", 82'-6", and 96'-6". Are there any additional drawings at other elevations?

   The staff also needs information regarding the cavity openings to other compartments (e.g., the annular region between the RPV and cavity boundary connecting the cavity to the upper compartments, or the openings in the vicinity of the hot leg or cold leg). For example, in the event of vessel breach, what are the possible pathways that steam/water/debris would be dispersed from the cavity?

   The following dimensions are also needed if they are not provided in the drawings:

   1. The characteristic dimensions of the cavity (the cavity has an octagonal shape).

   2. The internal diameter of the pressure vessel and the thicknesses of the pressure vessel walls (both the hemispherical lower head and the cylindrical region). Westinghouse provided values of 2 m, 0.152 m, and 0.2 m for the internal diameter of the pressure vessel, the thickness of the hemispherical region, and the thickness of the cylindrical region during a May 20, 1994 conference call. Verify this information.

c. Hot Leg/Surge Line - Provide design information for the hot leg and surge line, including the materials, lengths, diameters, and thicknesses. This information can be used to assess the likelihood of ex-vessel failure and primary system depressurization prior to vessel breach. Similar information pertaining to the Automatic Depressurization System is also desirable.

d. Cavity Conditions - Describe the more probable cavity conditions at the time of vessel breach. This should include the depth of the water pool, the range of pressures in the cavity or containment at vessel breach, and the temperature of the water pool under various accident sequences (this information could be provided from various plant-specific calculations).

   In addition, provide information regarding the source of water in the cavity. Westinghouse has identified the IRWST as the source of water in the cavity under flooded conditions. Verify that the IRWST is the source, and provide the flowrates and a description of the flowpaths from the IRWST to the cavity. This description may be in the form of a drawing and should include the valve arrangement and the diameter and length of the lines.

e. Lower Head - It is the staff's understanding that the lower head of the pressure vessel is smooth and free of any penetrations. If this is not true, provide the characteristic dimensions and location of these penetrations.

f. Structural Strengths - In order to adequately assess the impact of FCI energetics and the resulting impulsive loads on the containment design, the staff needs to understand the structural capability of the reactor cavity. What is the static and dynamic pressure capacity of the reactor cavity (ACI 349 provides one method for calculating these capacities)? A fragility curve of the cavity walls would be useful in determining the margin available to resist impulsive loads. Is the RPV supported only through the hot leg or cold leg? What is the structural capacity of the pressure vessel and steam line supports?

Response:

a. The inventory of in-vessel materials that could contribute significantly to the debris composition is as follows:

Mass of UO_2 = 75898. kg = 167325 lbm
Mass of Zr Cladding = 14380 kg = 31700 lbm
Mass of additional Zr in the vessel = 6930 kg = 15275 lbm
Mass of Stainless Steel Support Plate + Lower Nozzles of Fuel Assemblies = 23285 kg = 15335 lbm

For cases in which the AP600 reactor vessel is fully-flooded, the probability of reactor vessel failure is estimated to be in the range of 1E-2 to 1.E-4 in the AP600 PRA, revision 1. Therefore, the information provided for vessel failure cases is represented by case 3BE.cc0 from Appendix L of the PRA, revision 1 in which the vessel fails at low pressure into a partially flooded cavity.

The in-vessel debris composition as predicted by the MAAP4 code at the time of vessel failure for the 3BE.cc0 case:

Mass of UO_2 = 75898. kg = 167325 lbm
Mass of Zr = 15475. kg = 34115 lbm
Mass of ZrO_2 = 7880. kg = 17375 lbm
Mass of Stainless Steel = 23285. kg = 15335 lbm

In-Vessel Pressure = 1.5 bar = 21.8 psia
In-Vessel Debris Temperature = 2550°K = 4130°F

MAAP4 assumes that the entire core support plate mass is mixed into the debris when the core slumps into the lower head. This is a conservative bounding estimate of the mass of stainless steel in the debris, since only a portion of the support plate and reflector would be expected to be in the melt.

b. There are no additional general arrangement elevation drawings relevant to the reactor cavity dimensions.

The openings for the reactor coolant pipe include 54 inch minimum diameter openings for the hot legs and 48 inch minimum diameter openings for the cold legs into the steam generator compartment and 30 inch minimum diameter openings for the direct vessel injection lines into the accumulator compartments. The opening to the accumulators is sealed with an anchor or bellows pipe penetration. At the reactor vessel seal area (Elevation 107' 2") the reactor cavity has a 16' 6" diameter. Below the reactor coolant piping the reactor cavity has an octagonal cross section and is 17 feet across the flats. The major pathway out of the cavity for steam, water, and debris would be through a 8 foot by 5 foot blowout panel into the area containing the sump and the reactor coolant drain tank. This area communicates with the steam generator compartment through a 10 foot by 11 foot grating at the 83' elevation.

The dimensions for the reactor vessel are provided in SSAR Subsection 5.3.4.1. The internal diameter of the reactor vessel is 157 inches. The nominal thickness of the reactor vessel lower shell is 8 inches. The nominal thickness of the reactor vessel lower head is 6 inches. The response to RAI 210.101 provides additional reactor vessel key dimensions.

c. SSAR subsection 5.2.3 provides the material specifications for reactor coolant pressure boundary components and piping.

Hot Leg ID, inches (SSAR subsection 5.1.3.4)	31
Hot Leg wall thickness, inches	3.25
Hot Leg straight length, inches	151.68
Hot Leg elbow radius, degrees	55
Hot Leg radius of curvature, inches	56.25
Surge line pipe size, inches	18" Schedule 160
Surge line straight length	558 inches
Surge line elbow radii	3 elbows @ 89.1°, 42 inch radius 1 elbow @ 98.9°, 54 inch radius 2 elbows @ 89.65°, 54 inch radius 1 elbow @ 38.7°, 54 inch radius

Information on ADS fourth stage line sizes, routing and pipe lengths was provided in responses to RAIs 952.63 and 952.64.

d. For cases in which the AP600 reactor vessel is fully-flooded, the probability of reactor vessel failure is estimated to be in the range of 1E-2 to 1.E-4 in the AP600 PRA, revision 1. Therefore, the information provided for the most probable vessel failure cases is represented by case 3BE.cc0 from Appendix L of the PRA, revision 1 in which the vessel fails at low pressure into a partially flooded cavity. The pressure and temperature ranges include the effects of the success and failure of the passive containment cooling system water (cases 3BE.cc4 and 3BE.cc5).

Depth of Water Pool = 2.5 m = 8.2 ft
Containment Pressure = 1.5 to 2.5 bar = 21.8 to 36.3 psia

Cavity Water Temperature = 370°K to 380°K = 206°F to 224°F

e. As noted in last paragraph of Subsection 5.3.1.2 the reactor vessel has no penetrations below the core. This includes the lower head.

f. The capacity of the reactor cavity for impulsive loads is addressed in the response to RAI 720.208. The reactor vessel is supported on four supports, one below each of the cold legs.

SSAR Revision: NONE
PRA Revision: NONE