AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.100

Question:

AP1000 TS Surveillance Requirement (SR) 3.1.4.3 requires verification of each control rod drop time, from the fully withdrawn position, to be ≤ [2.7] seconds from beginning of decay of stationary gripper coil voltage to dashpot entry, with $T_{avg} \geq 500°F$, and all reactor coolant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

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

Westinghouse Response:

The control rod drop time of [2.7] seconds in AP1000 Technical Specification Surveillance Requirement (SR) 3.1.4.3 should be consistent with the accident analysis assumptions described in Chapter 15 and shown in Figure 15.0-5.1 as 2.47 seconds.

Therefore, the control rod drop time in SR 3.1.4.3 will be changed to 2.47 seconds to be consistent with the safety analysis. The existing value resulted from a typographical error while updating the original value of [2.4] to the new value of [2.47] shown in Figure 15.0.5-1.

Design Control Document (DCD) Revision:

DCD Chapter 16.1, Technical Specification 3.1.4, SR 3.1.4.3, pg 3.1-10

> Verify rod drop time of each rod, from the fully withdrawn position, is ≤ [~~2.7~~**2.47**] seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with:
>
> a. $T_{avg} \geq 500°F$, and
>
> b. All reactor coolant pumps operating.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.102

Question:

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

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

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

Westinghouse Response:

The purpose of the "Reviewer' s Note" in TS 5.6.5 is to recognize that additional NRC review and approval of BEACON application methodology with fixed incore detectors will be completed prior to generation of a plant specific TS 5.6.5. As noted, Westinghouse intends to seek NRC review and approval of an addendum to the BEACON topical report (WCAP-12472-P-A) covering the use of AP1000 fixed incore detectors in conjunction with use of similar detectors at operating plants. This addendum will be completed as part of the Combined Operating License (COL) as stated in DCD section 4.3.4. Discussion of uncertainty components relative to use of BEACON methodology with fixed incore detectors is provided in AP1000 DCD subsection 4.3.2.2.7.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.103

Question:

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

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

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

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

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

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

Westinghouse Response:

The treatment for the [bracketed] information in Tables 3.3.1-1 and 3.3.2-1 is discussed in Section 16.1.1 and in the notes on page 1 of each table. This treatment is the same as that approved by NRC in the AP600 Design Certification.

As discussed in Section 16.1.1, the Combined License applicant will replace the preliminary information in brackets with final plant specific values.

The note on page 1 of each table, and included below, provides a clear description of the basis for each of the parameters in the table, that they come either directly from safety analysis assumptions, or that they represent typical values for the function where the parameter is not credited in the safety analysis.

The intention of the AP1000 Technical Specifications is to present the safety analysis limit in the Trip Setpoint column, and where the Technical Specification value does not match the safety analysis limit in Chapter 15, including information in Table 15.0-4a, the Technical Specification value will be corrected.

The setpoint for the reactor trip on low steam generator narrow range level will be revised so that the trip setpoint value for Item 13 in Table 3.3.1-1 is 95,000 lbm, and matches the corresponding value in Table 15.0-4a. (This error was also identified in RAI 440.117.)

The setpoint for the High 2 steam generator narrow range water level in Technical Specification Table 3.3.1-1, Item 14 is incorrect, and will be revised to match the value of 100 percent of narrow range level for this trip setpoint in Table 15.0-4a.

Tables 3.3.1-1 and 3.3.2-1 of NUREG-1431, Revision 2, include columns for allowable value and nominal trip setpoint values for the reactor trip system and ESFAS instrumentation, with a note that unit specific implementations may contain only allowable values, depending on the setpoint methodology used by the unit. The AP1000 Technical Specifications approach is to include both columns, but to NOT include allowable values until provided in the Combined License application, based on the plant specific setpoint calculation, as discussed below.

DCD Section 7.1.6 references the setpoint methodology for protection systems that is applicable for the AP1000. In addition, the note on page 1 of Tables 3.3.1-1 and 3.3.2-1 describes the overall approach for providing the trip setpoint value for these two tables, including a discussion on implementation of the AP1000 setpoint methodology.

The note on page 1 of each table states the following:

> Reviewer Note: The values specified in brackets in the Trip Setpoint column are the Chapter 15 safety analysis values and are included for reviewer information only.
>
> The values specified in brackets followed by " * " in the Trip Setpoint column are typical values for the Function. No credit was assumed for these Functions (typically diverse trips/actuations) in the Chapter 15 safety analyses and no safety analysis value is available.

In all cases, the values specified in brackets must be replaced, following the plant-specific setpoint study, with the actual Trip Setpoints. Upon selection of the plant specific instrumentation, the Trip Setpoints will be calculated in accordance with the setpoint methodology described in WCAP-14606. (WCAP-14606 is an AP600 document that describes a methodology that is applicable to AP1000. AP1000 has some slight differences in instrument spans as a result of the higher power level.) Allowable Values will be calculated in accordance with the setpoint methodology and specified in the Allowable Value column. The plant specific setpoint calculations will reflect the latest licensing analysis/design basis and may incorporate NRC accepted improvements in setpoint methodology.

A. The first sentence in the note immediately after "Reviewer Note:" directly states that "the values specified in brackets are the Chapter 15 safety analysis values and are included for reviewer information only." No additional information is needed to describe how the trip setpoint value is determined since it comes directly from the safety analysis. For those items not resulting from safety analysis assumptions, typical values are provided until the setpoint methodology value can be calculated for a plant specific application.

B. The plant specific setpoint calculation will provide the allowance for instrumentation uncertainties (once the actual plant instrumentation is selected) and the trip setpoint can then be calculated and inserted into the tables by the Combined License applicant.

C. As stated in the third paragraph of the note, the "Allowable Values will be calculated in accordance with the setpoint methodology...and the plant specific setpoint calculations will reflect the latest licensing analysis/design basis and may incorporate NRC accepted improvements in the setpoint methodology" at the time when the plant specific setpoint calculation is provided by the Combined License applicant.

D. The setpoint methodology described in WCAP-14606 provides the process for the combination of instrumentation uncertainties, calibration device accuracy, measurement uncertainties, and other parameters considered in the setpoint calculation.

Design Control Document (DCD) Revision:

DCD, Technical Specification Table 3.3-1, Item 13, Steam Generator (SG) Narrow Range Water Level – Low, TRIP SETPOINT:

[≥ ~~45,000~~95,000 lbm]

DCD, Technical Specification Table 3.3-1, Item 143, Steam Generator (SG) Narrow Range Water Level – High 2, TRIP SETPOINT:

[≤ ~~95%~~100%]

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.104

Question:

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

Explain the inconsistency, and make corrections if necessary.

Westinghouse Response:

The treatment for the [bracketed] information in Tables 3.3.1-1 and 3.3.2-1 is discussed in Section 16.1.1. The Combined License applicant can replace the preliminary information in brackets with final plant specific values. This approach is consistent with AP600 (see NUREG-1512).

The pressure, temperature, and flow values for TS 43.4.1 and the temperature value for TS 3.4.2 should be bracketed for both the LCO and the surveillance requirements, which is consistent with the treatment in NUREG-1431. These are typographical errors in the Technical Specifications and will be corrected.

Design Control Document (DCD) Revision:

DCD TS 3.4.1

SR 3.4.1.1	Verify pressurizer pressure is > [2185 psig].
SR 3.4.1.2	Verify RCS average temperature is <[578.1°F].
SR 3.4.1.3	Verify RCS total flow rate is > [301,670 gpm].

DCD TS 3.4.2

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be $\geq$ (551 °F).

PRA Revision:

None

RAI Number: 440.105

Question:

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

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

Westinghouse Response:

The two paragraphs in the Bases for LCO 3.4.1 in NUREG-1431, Rev. 2 related to feedwater venturi fouling error have been intentionally deleted from the AP1000 Technical Specifications since they are not applicable due to feedwater flow measurement equipment design differences.

A leading edge flow meter (LEFM) is installed in the plant to provide increased accuracy in the measurement of feedwater flow. The LEFM is expected to be used for the plant calorimetric procedures when high accuracy is desired in the reactor power calculation, such as when calculating RCS loop flow. Therefore, the note on venturi fouling error is not applicable for calculation of RCS loop flow in TS 3.4.1 with the implementation of the LEFM. Deletion of this note is also consistent with the approach for currently operating plants that have implemented the LEFM for similar feedwater flow measurement applications.

The leading edge flow meter can also be used to periodically cross-calibrate the venturi feedwater flow meters, thereby compensating for venturi fouling effects during the fuel cycle. This reduces the cumulative error that could otherwise exist in the venturi feedwater flow instrumentation with a longer calibration interval. The cross-calibration maintains the continuous day-to-day accuracy of the venturi feedwater flow measurement. The venturi flow measurement is used continuously by the steam generator water level control system, and can be monitored by the operators as an independent calculation of total reactor power when the leading edge flow meter may not be operating to perform high-precision plant calorimetric procedures.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.106

Question:

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

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

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

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

Westinghouse Response:

A. NUREG-1431, Rev. 2, Technical Specifications include specific requirements for RCS flow during shutdown MODES to provide adequate heat removal and boron dilution event mixing assumptions. The minimum RCS flow requirements in these approved Technical Specification are satisfied by operation of a single RHR pump.

   The operation of one AP1000 RCP in the specified reduced speed operation, with an RCS flow rate of at least [10,000 gpm], provides significantly greater RCS flow than the single RHR pump flow in current plants. Since AP1000 and current plants have the same boron dilution event design basis and minimum RCS flow requirements for boron mixing, the AP1000 RCS flow is signficantly more than required to achieve the required boron mixing.

The flow mixing assumptions for boron dilution analyses for both current plants and for AP1000 are based on NUREG/CR-2733, "Experimental Data Report for LOFT Boron Dilution Experiment L6-6," June 1982, conducted by Idaho National Engineering Laboratory for the U.S. NRC. This testing modeled the Trojan Nuclear Power plant design, assuming a base case RHR flow of 3000 gpm, and a second case with twice the flow of the base case. As stated in Section 3 of EGG-LOFT-5867, "Quick-Look Report on LOFT Boron Dilution Experiment L6-6," May 1982, stated that for both flow cases "the close agreement between the measurement and the core criticality value implies that the reactor vessel volume was well mixed." The Quick-Look Report abstract states that "the results of the boron dilution simulations [for both flow rates] showed that the direct flow path volume was well mixed and the boron concentration as a function of time was characterized by the perfect mixing model."

B. The AP1000 RCPs are described in DCD Sections 5.1.3.3 and 5.4.1. The RCPs are single-stage, canned motor centrifugal pumps. A variable frequency drive provides speed control to reduce RCP speed and motor power requirements during pump startup from cold conditions below 450°F. The variable speed controller is only operated in Mode 5 with the reactor trip breakers open. During other plant conditions including power operation, the variable frequency drive is isolated from the RCP so that the RCP operates at a constant (full) speed.

As discussed in the Bases for LCO 3.4.9 and for SR 3.4.9.1, the minimum flow requirement of [10,000 gpm] assures adequate mixing of the RCS in the event of a boron dilution event. SR 3.4.9.1 requires confirming RCS flow for the RCP combination and speed specified (one RCP operating at 25% speed), although the minimum flow is satisfied for the various pump combinations and speeds discussed in the Bases for SR 3.4.9.1.

As indicated in Table 5.4-1, the best estimate RCP design flow (during constant, full-speed operation) is 78,750 gpm per pump, or a total reactor vessel flow of 315,000 gpm with all four pumps operating. This flow can be used to calculate the pump flow at other lower operating speeds.

For a variable-speed centrifugal pump, the flow rate change is directly proportional to the pump rotational speed, and the head produced by the pump is proportional to the square of the pump speed change. Therefore, if the pump speed is reduced to the speeds indicated in the table below from Bases for Surveillance Requirement 3.4.9.1, the flow can be calculated based on the proportional change in pump speed. The table below calculates flow changes considering only changes in RCP speed, and assuming 4 RCPs continue operating. Calculating pump flow this way is conservatively low since it ignores the significant reduction in RCS system flow resistance when RCPs are stopped, and it also simplifies the approximation of RCS flow for this RAI response.

Number of RCPs	% Rated Speed	Calculated Flow (gpm, based on 4 RCPs running)
1	25%	19,688 x 1 = **19,688***
2	20%	15,750 x 2 = 31.500
3	15%	11,813 x 3 = 35,438
4	10%	7,875 x 4 = 31,500

* The first RCP combination is used as the flow value for SR 3.4.9.1 since it is the normal minimum pump flow combination expected during plant cooldown prior to securing RCPs at about 160°F.

The SR test condition provides significantly more flow than the required minimum flow for boron mixing (as discussed in Item A above) and meets the LCO requirements, even with no benefit from the reduction in RCS system flow resistance. The RCP flow provides mixing in the reactor vessel and core, the operating loop, and the idle loop.

As shown in the table above, the large RCS flow rates for the other three operating conditions discussed in the Bases for SR 3.4.9.1 significantly exceed both the boron dilution minimum flow mixing requirements and the specified LCO 3.4.9 flow requirements. These flows are also conservatively calculated without consideration of the RCS system flow resistance reduction.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.114

Question:

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

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

Westinghouse Response:

Table 19.E.2-1 provides the length of time necessary to uncover the core following an assumed loss of cooling during mid-loop operations at a time 28 hours after shutdown. During mid-loop and reduced inventory operations, the operators closely monitor the conditions in the RCS. During this condition, the normal residual heat removal system (RNS) is aligned to provide core cooling. The operators closely monitor the operation of the RNS pumps, and monitor the RCS temperature and pressure with installed instrumentation in both the RNS and reactor coolant system (RCS). In addition, the level in the RCS is monitored, using either the pressurizer level or the hot leg level or both. The pressurizer level instrumentation span used for shutdown spans from the top of the pressurizer to the bottom of the hot leg, to provide a continuous level reading during the transition from the time the water level is in the pressurizer to when the water level is reduced to within the span of the hot leg level instrumentation.

The operators are alerted to a loss of RCS cooling by a number of various indications. Primarily the RCS temperature is monitored to determine that core decay heat removal is sufficient to maintain the RCS temperature within the desired range. Direct measurement of the RCS temperature is provided by installed instrumentation in the RCS (hot leg and cold leg wide range temperature), as well as RNS heat exchanger inlet and outlet temperatures. An increase in RCS temperature would be detected during all modes of operation, and especially during reduced inventory conditions when the operators are closely monitoring the RCS parameters.

Additional indications are provided to allow the operators the ability to diagnose the cause of the loss of RCS cooling during shutdown and reduced inventory conditions. Examples include:

- RNS heat exhanger inlet and outlet temperature
- RNS pump running indication (i.e. motor current)

- Component cooling water system (CCS) temperatures, heat exchanger ΔT, and pump running indication
- Service water system (SWS) temperatures, heat exchanger ΔT, and pump running indication

The AP600 Emergency Response Guidelines (ERG) were submitted to the NRC during the AP600 review in Reference 1. The AP600 ERG provide shutdown safety status trees and shutdown guidelines that will be used to develop the detailed Emergency Operating Procedures for the AP1000. Please see the Westinghouse response to RAI 440.109 for a related discussion of the AP1000 procedure development and the applicability of the AP600 ERG to the AP1000. The following description refers to the AP600 Shutdown Response Guideline (SDG) where the described operator actions are provided in the AOP600 ERG.

As outlined in reference 1, if RCS cooling would be lost during shutdown and reduced inventory conditions, the operators would take action to re-establish RCS cooling, based on the cause of the loss of cooling. For example, if the RNS pump were not functioning, the operators would take action to restore cooling of the core using the RNS pumps (SDG-2). The operators would continue to attempt to restore RCS heat removal and would monitor the RCS hot leg level indication. Once the RCS becomes saturated, hot leg level would begin to decrease due to the boiling of the inventory in the hot leg. If the operators are unable to restore core cooling, the EOPs would direct the operator to increase coolant inventory (SDG-1). The first line of defense would be to use the CVS makeup pumps to increase inventory in the primary side. This action will temporarily delay steaming of the RCS coolant, and will provide more time for the operators to re-establish core cooling via the RNS.

If the CVS makeup pumps were unavailable, the operators would take action to establish passive safety injection via the IRWST. The operators would manually open the IRWST injection isolation valves and ADS stage 4 valves as necessary (SDG-1).

Given that the operators closely monitor RCS conditions during reduced inventory operations, and that clear indication of important parameters are provided that would indicate a loss of RCS cooling during mid-loop, sufficient time exists for the operator to take actions to prevent core uncovery following a loss of shutdown cooling. In addition, automatic features are provided to automatically actuate IRWST injection flow when the water level in the reactor vessel is reduced to below that of the hot leg.

DCD sections 18.9 and 13.5 include the commitments for the COL applicant to develop emergency procedures. Please see the Westinghouse response to RAI 440.109 for a related discussion of the AP1000 procedure development.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.115

Question:

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

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

Westinghouse Response:

The discussion of spurious openings of a SG PORV and of steam line breaks in chapter 19E provides an evaluation of consequences of these events during different shutdown conditions relative to the plants conditions analyzed in the DCD (MODES 1 and 2). This discussion addresses both stuck open SG PORVs and pipe breaks during other shutdown MODES (3,4,5, and 6); it does not say that steamline breaks larger than the SG PORVs are non-credible. The evaluation of these events concludes that the consequences of the conditions analyzed in the DCD are more severe than those that would occur at shutdown conditions. The reasons for this conclusion include:

- The lower the initial RCS temperature, the less mass / energy that can be released and the slower the resulting cooldown will be during a SLB
- When the RCS is being cooled by the RNS, the SGs can not affect the RCS temperature very much
- In lower MODES the RCS is borated sufficiently that it can not return to power as a result of a SLB

These reasons apply to both stuck open SG PORVs and steam line breaks. Note that Table 19E.4.1-1 should indicate that Steam System Piping Failure is evaluated; the table currently shows a blank for this accident.

Design Control Document (DCD) Revision:

Revision to DCD Table 19E.4.1-1:

Table 19E.4.1-1

AP1000 ACCIDENTS REQUIRING SHUTDOWN EVALUATION OR ANALYSIS

Tier 2 Section	Titles	Evaluation or Analysis Required
15.1	Increase in Heat Removal from the Primary System	
15.1.1	Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature	E
15.1.2	Feedwater System Malfunctions that Result in an Increase in Feedwater Flow	E
15.1.3	Excessive Increase in Secondary Steam Flow	E
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	E
15.1.5	Steam System Piping Failure	E
15.1.6	Inadvertent Operation of the Passive Residual Heat Removal Heat Exchanger	E
15.2	Decrease in Heat Removal by the Secondary System	

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.117

Question:

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

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

Westinghouse Response:

The reactor trip setpoint value of 45,000 lbm for the low steam generator narrow range level in DCD Technical Specifications, Table 3.3.1-1, Item 13, is incorrect. The correct value is 95,000 lbm and the DCD will be revised. Therefore, Sections 19E.4.3.3 and 19E.4.3.4 both correctly state that reactor trip on low steam generator narrow range level occurs before PRHR actuation.

As discussed below, the PRHR actuation signal is different for the two events, but the PRHR actuation signal is not identified in either of these sections.

Section 15.2.7 of the AP1000 DCD provides a discussion of the loss of normal feedwater event (and is referenced in Section 19E.4.3.3) and Section 15.2.8 discusses the feedwater system pipe break event (and is referenced in Section 19E.4.3.4).

Table 15.2-1 provides the time sequence of events for each accident, Sheet 6 of 7 for the loss of normal feedwater event and Sheet 7 of 7 for the pipe break event. These tables identify the reactor trip and PRHR actuation signal for the associated event.

For both events, the reactor trip results from a low steam generator narrow water level and the PRHR heat exchanger actuates after the reactor trip occurs.

The PRHR heat exchanger can actuate due to either low steam generator narrow range level, coincident with low startup feedwater flow, or due to low steam generator wide range level. The wide range signal occurs at a lower steam generator mass than the narrow range signal.

As shown in DCD Figure 7.2-1 (Sheet 7 of 20), the narrow range level actuation signal is the same signal (and setpoint value) as the reactor trip low steam generator narrow range level signal. However, as shown in DCD Figure 7.2-1 (Sheet 8 of 20), a time delay is added to this signal when it is used for PRHR actuation. The time delay allows startup feedwater to attempt to actuate before initiating PRHR cooling. If startup feedwater fails to start (which is assumed for the design basis case), then PRHR actuates once the time delay is satisfied.

Sheet 8 shows that the PRHR heat exchanger also actuates on a low steam generator wide range level signal.

For a loss of normal feedwater event, PRHR actuates on the low narrow range level coincident with a low startup feedwater flow signal. The low steam generator narrow range level signal to the PRHR actuation logic exists at the time of the reactor trip, and the delay timer starts when the trip signal is generated. For this event, the time delay is satisfied before steam generator mass is reduced to the low steam generator wide range setpoint. For a feedwater line break, the higher mass loss results in reaching the low wide range level setpoint before the narrow range time delay is satisfied.

The Protection and Safety Monitoring System setpoints assumed in the safety analyses are provided in Table 15.0-4a, which shows that the reactor trip on low steam generator narrow range level setpoint is 95,000 lbm, while the PRHR actuation on low steam generator wide range level setpoint is 55,000 lbm. As discussed in the response to RAI 440.103, the reactor trip setpoint in Technical Specification Table 3.3.1-1, Item 13 is incorrect and will be changed from 45,000 lbm to 95,000 lbm to be consistent with Table 15.0-4a.

An additional revision will be made to correct two minor editorial errors related to the name of the PRHR actuation signals in Section 15.2.7

Section 15.2.7.1 incorrectly states that the PRHR heat exchanger is also actuated due to low-low steam generator water level (wide range). This will be corrected to read low steam generator water level (wide range).

Section 15.2.7.2.1 incorrectly states that the PRHR heat exchanger is actuated due to low-low steam generator water level narrow range signal, coincident with low startup feedwater. This will be corrected to read low steam generator water level narrow range signal coincident with low startup feedwater flow.

Design Control Document (DCD) Revision:

DCD, Section 15.2.7.1, fourth bulleted item:

If startup feedwater is not available, the PRHR heat exchanger is actuated on either a low steam generator water level (narrow range), coincident with a low startup feedwater flow rate signal or a low-~~low~~ steam generator water level (wide range) signal. The PRHR heat exchanger transfers the core decay heat and sensible heat to the IRWST so that core heat removal is uninterrupted following a loss of normal and startup feedwater (see Section 15.2.6).

DCD, Section 15.2.7.2.1, last paragraph under third bulleted item:

The PRHR heat exchanger is actuated by the low-~~low~~ steam generator water level narrow range signal, coincident with low start up feedwater flow.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.120

Question:

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

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

Westinghouse Response:

DCD Appendix 19E 2.1.2.2 describes the RCS instrumentation to accommodate shutdown operations and includes the two hot leg level instrumentation channels.

DCD Figure 7.2-1 (Sheet 16 of 20) provides the functional diagram for the RCS hot leg level actuation functions which include ADS stage 4 actuation, IRWST injection actuation, and CVS letdown isolation. DCD Section 7.3 provides additional information on the actuation functions from the hot leg level instrumentation.

DCD Technical Specifications Table 3.3.2-1 (page 13 of 13) identifies the ESFAS instrumentation requirements for the hot leg level channels.

See the response to RAI 440.103 for a discussion of the approach for identifying the allowable values and the trip setpoints for Technical Specifications Table 3.3.2-1, and for calculating setpoint uncertainties for engineered safety feature actuation systems.

As discussed in the response to RAI 440.103, the AP1000 Technical Specifications only provide the trip setpoint value, and do not include the allowable value. Measurement uncertainties for hot leg level instrumentation cannot be determined until the plant specific setpoint calculation is completed by the Combined License applicant.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.122

Question:

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

Westinghouse Response:

The test program and data analysis of Reference 4 of Appendix 19E (Letter, Westinghouse to NRC, DCP/NRC0124, APWR-0452, "AP600 Vortex Mitigator Development Test for RCS Mid-loop Operation," July 6, 1994) resulted in a correlation between the Froude number for the normal residual heat removal (RNS) flow conditions in the RNS step nozzle and the critical vortexing water level in the hot leg with respect to the bottom of the hot leg inside diameter. The critical vortexing level is that hot leg level below which the vortex will cause air to be entrained in the water flowing to the pump. This correlation was found to be valid over a range of Froude numbers.

Since the range of the scaled RNS flow rates tested bound the AP1000 RNS flow rate, the Froude number resulting from the AP1000 RNS flow rate is within the valid range for the correlation, and the correlation can be applied to the AP1000. The correlation shows margin between the predicted critical vortexing water level for the AP1000 and the normal mid-loop hot leg operating level.

Additionally, the tests demonstrated that the RNS pumps can continue to operate when the water level in the hot leg drops below the critical vortexing water level. The step nozzle prevents the vortex from being drawn down to the pump. When the water level drops below the critical vortexing level, the vortex is changed into a spill and fill type operation – the water spills over into the step nozzle which acts as a holdup tank. This spilling action results in some air entrainment, however the average percentage volume of air drawn into the RNS pump suction is less than 5%, which has been shown experimentally to not significantly affect pump operation.

Therefore, full flow operation of the AP1000 RNS pumps at mid-loop conditions is not expected to result in air entrainment that could result in cavitation of the pump.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.125

Question:

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

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

Westinghouse Response:

The AP1000 Technical Specifications include the requirements of LCO 3.4.14 for minimum required RCS vent paths during MODE 6 conditions with the upper core internals in place. These requirements are intended to specifically address the concerns identified in NUREG/CR-5820 related to potential loss of core cooling and pressurization of the RCS following the loss of normal residual heat removal system (RNS) cooling in this refueling condition.

LCO 3.4.14 requires that Automatic Depressurization System (ADS) stage 1, 2, and 3 flow paths are open and that two paths of ADS stage 4 are operable in MODE 6 with the upper internals in place. This ADS venting requirement is also applicable in Mode 5 with the RCS pressure boundary open or with pressurizer level less than 20 percent.

As discussed in the bases for LCO 3.4.14, in DCD Technical Specifications B3.4.14, the ADS venting requirements assure that sufficient vent area is available to support injection from the in-containment refueling water tank (IRWST) to mitigate events requiring RCS makeup, boration, or core cooling.

When the refueling cavity is flooded during MODE 6 by transferring the IRWST water to the refueling cavity, the ADS vent path allows refueling cavity water to flow down through the upper internals into the core. The open ADS flow paths in the pressurizer vent steam generated in the core following a loss of RNS heat removal in this condition. With the ADS vent path available, counter-current flow through the upper internals is not required for core cooling. Therefore, the hydraulic resistance of the upper internals does not prevent the refueling cavity water from draining down into the core since steam generated in the core region does not have to pass upward through the upper internals at the same time the refueling cavity is draining.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.127

Question:

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

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

Westinghouse Response:

As with AP600, the AP1000 provides the RVLIS function with reactor vessel water level indication for a range spanning from the bottom of the hot leg to approximately the elevation of the reactor vessel mating surface. This reactor vessel water level indication complies with the requirements of TMI Action Item II.F.2 as described in NUREG-1513 (AP600 FSER), Chapter 20, Issue II.F.2. The AP1000 design features, actuation logic, and operator responses are described below.

BACKGROUND

TMI Action Item II.F.2 describes the requirements for plants to incorporate new instrumentation to monitor inadequate core cooling. The requirements of TMI Action Item II.F.2 are provided in Table 440.127-1. These requirements resulted in current Westinghouse PWRs incorporating a reactor vessel level indication system (RVLIS) that provides indication of reactor coolant system void fraction when the reactor coolant pumps are operating, and reactor vessel water level when the reactor coolant pumps are tripped. This instrumentation was designed specifically for PWRs that rely on operator action to trip the reactor coolant pumps following a LOCA.

Prior to TMI, the accepted philosophy for PWRs was that during a LOCA, operation of the reactor coolant pumps, if available, was always desirable in that they provided improved core cooling. However, during the TMI scenario, the reactor coolant pumps continued to operate during a loss of coolant accident caused by a stuck open pressurizer PORV. Although pressurizer level remained high (which was interpreted as an indication of adequate coolant inventory), the continuous loss of reactor coolant and subsequent inadequate core cooling caused the reactor coolant to become highly voided. As long as the reactor coolant pumps continued to operate, core cooling was maintained.

However, eventually the reactor coolant pumps were tripped, and due to the high void content in the coolant, the water level dropped to below the top of the core causing core damage. Therefore, the RVLIS systems were designed to provide the operators with an unambiguous indication of void content in the reactor coolant when the reactor coolant pumps are operating. Void content indication is used to manually trip the reactor coolant pumps following a LOCA, to avoid the possibility of core uncovery later during the event, if the RCPs were tripped when the coolant was highly voided.

In current Westinghouse PWR's, RVLIS is also used to provide the measurement of reactor vessel level during a LOCA event after the RCPs are tripped. For these plants, water level in the vessel is an indication of inadequate core cooling and coolant inventory. This measurement is used to prioritize operator recovery actions, and is used to instruct the operator to:

- Establish / re-establish safety injection flow,
- Manually depressurize the RCS.

AP1000 REACTOR VESSEL LEVEL INDICATION

The AP1000 design complies with the requirements of TMI Action Item II.F.2. The requirements for reactor vessel level indication are provided by redundant, safety-related reactor vessel level instrumentation. As shown in DCD Figure 5.1-5, these instrument channels (LT-160 and LT-170) have one level tap that connects to the bottom of a hot leg, and one level tap that connects to the top of the hot leg bend that connects to the steam generator. This instrumentation is used to provide reactor vessel water level during an accident, and also is used to provide hot leg level during shutdown operations including mid-loop. This instrumentation provides indication of reactor vessel water level for a range spanning from the bottom of the hot leg to approximately the elevation of the reactor vessel mating surface. This instrumentation is temperature compensated and provides accurate level measurement during all modes of operation. This instrumentation complies with the requirements of TMI Action Item II.F.2 as a result of the AP1000 design features, actuation logic, and operator responses described below.

AP1000 Design Features:

<u>Automatic Depressurization</u>

The AP1000 passive safety-related systems operate in conjunction with automatic depressurization system (ADS) valves that automatically reduce the pressure in the RCS in response to a loss of coolant accident. Following a small break LOCA, the AP1000 core makeup tanks (CMTs) inject water into the RCS. After the CMTs have injected approximately one-third of their inventory, the ADS valves receive a signal to open. The ADS valves are comprised of four stages of valves, three stages connected to the pressurizer and the fourth stage connected to the hot legs. The first stage opens on the low CMT level signal, and the second and third stages open sequentially on a time delay. The fourth stage ADS valves open when the CMTs are approaching empty. See DCD Subsection 5.4.6 for a description of the ADS valves.

Opening of the ADS valves will cause a significant disturbance of the water level in the RCS. Initially water and steam will be discharged via the first three stages of ADS valves connected to the pressurizer. Water level in the RCS will vary greatly during the transient, and water level will not be a reliable indication of inadequate core cooling during ADS. Eventually the CMTs will reach a low level and the fourth stage ADS valves connected to the hot legs will open. This will reduce the pressure in the RCS sufficiently to enable gravity injection from the IRWST. Once the fourth stage ADS valves are opened and IRWST injection is established, the water level in the RCS will remain at a relatively constant level, and its measurement will provide a reliable indication.

Passive RHR Heat Exchanger

The AP1000 passive RHR heat exchanger provides safety-related core cooling during accident events including loss of heat sink accidents such as loss of normal feedwater and feedline break accidents. The passive RHR heat exchangers provide sufficient core cooling during these events, even if RCS subcooling is not maintained. Unlike current plants that rely on the steam generators (in conjunction with safety-related auxiliary feedwater pumps) that require RCS subcooling be maintained in the hot legs to ensure core cooling during these events, the AP1000 relies on the passive RHR heat exchanger that provides sufficient core cooling without requiring RCS subcooling. This eliminates the need to use vessel head steam space indication as a means of detecting a loss of RCS subcooling. Since passive core cooling does not require maintaining subcooling, reactor vessel level indication above the mating surface is not an indication of inadequate core cooling.

AP1000 Protection and Safety Monitoring System Actuation Logic:

Automatic Reactor Coolant Pump Trip

The AP1000 does not require the operators to manually trip the RCPs following a LOCA because of the automatic trip of the RCPs on a safeguards actuation signal (i.e., CMT actuation). This eliminates the need for a RVLIS to provide a safety-related measurement of coolant void fraction to be used to manually trip the RCPs following a safeguards actuation signal. RCS subcooling is continuously monitored and is determined by the safety-related measurements of RCS pressure (wide range or pressurizer pressure) and RCS temperature (hot leg wide range, narrow range, and core exit thermocouples). The AP1000 instrumentation described provides an unambiguous indication of inadequate core cooling and provides an advanced warning of an approach to inadequate core cooling.

AP1000 Operator Responses:

In current PWRs, RVLIS is used to provide the measurement of reactor vessel level during a LOCA event after the RCPs are tripped. For these plants, water level in the vessel is an indication of inadequate core cooling and coolant inventory. This measurement is used to prioritize operator recovery actions, and is used to instruct the operator to perform the following actions:

- Establish / re-establish safety injection flow,
- Manually depressurize the RCS.

The AP1000 operator responses do not rely on reactor vessel level indication. Operator actions are not required to establish safety injection flow or to depressurize the RCS. Following a safeguards actuation, the operators will monitor plant parameters that provide indication of successful operation of the passive safety-related systems. These parameters are:

- PXS valve position indication (CMT, accumulator and IRWST discharge valves)
- CMT water level and temperature
- IRWST level
- RCS pressure, temperature
- Reactor vessel water level
- RCS core exit thermocouples
- ADS valve position indication

Upon actuation of the CMTs, the water temperature in the CMT will increase, and water level will eventually begin to decrease. Eventually CMT level will be reduced and the operators will be alerted when ADS actuation is to occur. Valve position indication of the safety injection discharge isolation valves will indicate that these valves have opened. The operator will monitor these parameters to determine if the CMTs have been initiated. If these parameters indicate failure of the CMTs to actuate, the operator will manually actuate the CMTs. Failing this, the operators would manually depressurize the RCS using the ADS valves.

If the CMTs actuate but ADS fails to actuate when required, the operator will manually actuate the ADS valves. The operator will have unambiguous indication that ADS is to occur, and unambiguous indication of its failure to occur. The use of RVLIS for the AP1000 is not required to alert the operator to depressurize the RCS to mitigate accidents.

Following a safeguards actuation signal, the operators are not required to perform manual operations on the passive safety-related systems. Operators will monitor the key parameters described above. In addition, the operators will control operation of the nonsafety-related CVS makeup pumps and RNS pumps to provide high pressure and low pressure RCS makeup based on the instrumentation listed above, and the additional instrumentation provided with each system (RNS flow rate, CVS flow rate). The CVS makeup pumps will operate automatically based on pressurizer water level. The RNS pumps will be manually started upon a safeguards signal. In the event of a small LOCA where the RCS pressure is not reduced immediately to below the RNS pump shut-off pressure, the RNS pumps will operate on miniflow. As the pressure in the RCS is reduced (due to operation of the ADS) to below the RNS pump shut-off pressure, the RNS pumps will provide makeup flow via the direct vessel injection lines. For these events, the operator will not rely on any RCS parameter (such as reactor vessel level, pressurizer level, RCS pressure) to start the RNS pumps and heat exchangers, but will use the safeguards actuation signal.

Following ADS, the water level in the vessel will be maintained within the range of the reactor vessel water level instrumentation. A high level reading will be used to confirm adequate safety injection and core cooling. A low water level (below the range of the reactor vessel water level instrumentation), when combined with core exit thermocouple readings, will provide an indication of inadequate core cooling.

SUMMARY

The AP1000 has been designed to provide the operators with an unambiguous indication of inadequate core cooling before, during, or after a loss of coolant accident. Indication of inadequate core cooling is provided by the various RCS instrumentation such as hot leg and core exit temperature. Indication of RCS inventory and reactor vessel level are provided by the reactor vessel instrumentation. Indication of RCS subcooling is provided by the RCS wide range pressure and core exit temperature. Indication of safety injection operation and ADS operation are provided by the various parameters discussed above. Therefore the AP1000 design complies with the requirements of TMI Action Item II.F.2. Table 440.127-1 summarizes the compliance of the AP1000 design with the ten requirements contained in TMI Action Item II.F.2

Table 440.127-1 Summary of AP1000 Compliance to TMI Action Item II.F.2

TMI Action Item II.F.2	AP1000 Compliance
(1) *Design of new instrumentation should provide an unambiguous indication of inadequate core cooling. This may require new measurements or a synthesis of existing measurements which meet design criteria (item 7)*	Indication of inadequate core cooling is provided by the following parameters: - Core exit and hot leg temperature - Reactor vessel water level - RCS wide range pressure - Pressurizer level In addition, the following instrumentation provides indication of operation of the safety-related and nonsafety-related systems that perform core cooling: - Safety-related valve position indication - CMT water level and temperature - IRWST water level - PRHR flow rate, temperature - RNS flow rate temperature - CVS flow rate
(2) *This evaluation is to include reactor-water-level indication.*	The AP1000 provides reactor vessel water level indication from the bottom of the hot leg to approximately the mating surface of the reactor vessel flange.
(3) *Licensees and applicants are required to provide the necessary design analysis to support the proposed final instrumentation system for inadequate core cooling and to evaluate the merits of various instruments to monitor water level and to monitor other parameters indicative of core-cooling conditions.*	The instrumentation that monitors the plant parameters described above are designed and qualified in accordance with their safety classification as described in DCD section 7.5.
(4) *The indication of inadequate core cooling must be unambiguous in that it should have the following properties:* a) *It must indicate the existence of inadequate core cooling caused by various phenomena (i.e. high-void fraction-pumped flow as well as stagnant boil-off); and,* b) *It must not erroneously indicate inadequate core cooling because of the presence of an unrelated phenomenon.*	a) High void fraction pumped flow is not an issue because the AP1000 RCPs are tripped on a safeguards actuation signal. RCS subcooling is determined by the core exit thermocouples in conjunction with RCS pressure. Stagnant boil-off is monitored with the reactor vessel water level instrumentation. b) Unrelated phenomenon does not compromise the indication of inadequate core cooling.
(5) *The indication must give advanced warning of the approach of inadequate core cooling.*	An advanced warning of inadequate core cooling is provided by the instrumentation that monitors the parameters described in (1).
(6) *The indication must cover the full range from normal operation to complete core uncovery. For example, water-level instrumentation may be chosen to provide advanced warning of two-phase level drop to the top of the core and could be supplemented by other indicators such as incore and core-exit thermocouples provided*	Reactor vessel water level provides an advanced warning of two-phase level drop to the top of the core. Core exit thermocouples are provided and can be used to indicate core uncovery.

TMI Action Item II.F.2	AP1000 Compliance
that the indicated temperatures can be correlated to provide indication of the existence of inadequate core cooling and to infer the extent of core uncovery. Alternatively, full-range level instrumentation to the bottom of the core may be employed in conjunction with other diverse indicators such as core-exit temperatures to preclude misinterpretation due to any inherent deficiencies or inaccuracies in the measurement system selected.	
(7) *All instrumentation in the final inadequate core cooling system must be evaluated for conformance to Appendix A, "Design and Qualification Criteria for Accident Monitoring Instrumentation" as clarified or modified by the provisions of items 8 and 9 that follow. This is a new requirement.*	The instrumentation that monitors the parameters described in (1) are included in DCD Table 7.5-1. The instrumentation provided is designed in accordance with their safety-related functions as described in DCD Section 7.5.
8) *If a computer is provided to process liquid-level signals for display, seismic qualification is not required for the computer and associated hardware beyond the isolator or input buffer at a location accessible for maintenance following an accident. The single-failure criteria of item 2, Appendix A, need not apply to the channel beyond the isolation device if it is designed to provide 99% availability with respect to functional capability for liquid-level display. The display and associated hardware beyond the isolation device need not be Class 1E but should be energized from a high-reliability power source which is battery backed. The quality assurance provisions cited in Appendix A, item 5, need not apply to this portion of the instrumentation. This is a new requirement*	The instrumentation provided is designed in accordance with their safety-related functions as described in DCD Section 7.5.
9) *Incore thermocouples located at the core exit or at discrete axial levels of the inadequate core cooling monitoring system and which are part of the monitoring system should be evaluated for conformity with Attachment, "Design and Qualification Criteria for PWR Incore Thermocouples," which is a new requirement.*	The incore instrumentation is described in DCD Subsection 4.4.6.1. Core exit thermocouples are safety-related as described.
10) *The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:* a) *the use of this information by an operator during both normal and abnormal plant conditions,* b) *integration into emergency procedures*	The instrumentation provided to monitor inadequate core cooling are displayed in the main control room. DCD Chapter 18 discusses the AP1000 Human Factors Engineering.

TMI Action Item II.F.2	AP1000 Compliance
c) integration into operator training, and *d) other alarms during emergency and need for prioritization of alarms.*	

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.130

Question:

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

Westinghouse Response:

The referenced figure in WCAP-14252 presents the upper plenum liquid level relative to the bottom of the upper core support plate from instrument LDP-139 as "RV Upper Plnm Lvl - WR." Figure 2-33 in WCAP-15833 presents the WCOBRA/TRAC-predicted collapsed liquid level in the reactor vessel inside the core barrel relative to the bottom of the reactor vessel lower plenum, so no direct comparison exists with Figure 5.1.2-12. For more discussion of the upper plenum prediction, refer to the response to RAI 440.131.

The most meaningful comparison that can be made concerning the WCOBRA/TRAC reactor vessel prediction is to compare the predicted mass inventory reduction during the ADS-4 IRWST initiation phase to the Test SB18 result. The initial WCOBRA/TRAC reactor vessel core, downcomer and upper plenum inventories at the time of ADS-4 actuation are set to the respective NOTRUMP values at that time in both the Test SB18 simulation and the AP1000 plant calculations; NOTRUMP was assessed in Reference 440.130-1, Section 8.4.3, to underpredict the system mass for the OSU tests from the time of ADS-1 actuation onward. Further, the WCOBRA/TRAC-predicted vessel mass inventory reduction during the ADS-4 IRWST initiation phase transient agrees well with the test result from Figure 5.2.2-40 of Reference 440.130-2 during the time interval from the opening of the ADS Stage 4 flow paths until the initiation of IRWST injection. Overall, the agreement of the WCOBRA/TRAC prediction of the Test SB18 reactor vessel mass inventory reduction during the ADS-4 IRWST initiation phase with the test result provides assurance that the AP1000 reactor vessel mass inventory WCOBRA/TRAC predictions in WCAP-15833, which have been obtained using the same method of initialization from NOTRUMP values at ADS-4 actuation, are reasonable.

References:

440.130-1: WCAP-14807, Revision 5, "*NOTRUMP Final Validation Report for AP600,*" August 1998.

440.130-2: WCAP-14292, Revision 0, "*AP600 Low-Pressure Integral Systems Test at Oregon State University Test Analysis Report*," September 1995.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.131

Question:

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

Westinghouse Response:

Figure 5.1.2-4 presents a simple translation of the Test SB18 measured collapsed liquid levels into steam percentages, as indicated by the title line "based on indicated level channels." This figure does not indicate two-phase mixture void fractions in the upper plenum, and it does not correspond to the WCOBRA/TRAC-predicted mixture void fractions in the upper plenum channels.

Since Figure 5.1.2-4 does not contain any additional information to that available from a liquid level comparison, it is more straightforward to compare levels directly. The attached figure is Figure 2-33 from WCAP-15833, but with the Test SB18 data for upper plenum collapsed level (relative to the bottom of the lower plenum) superimposed. The level predicted by WCOBRA/TRAC inside the core barrel shows reasonable agreement with the Test SB18 test data.

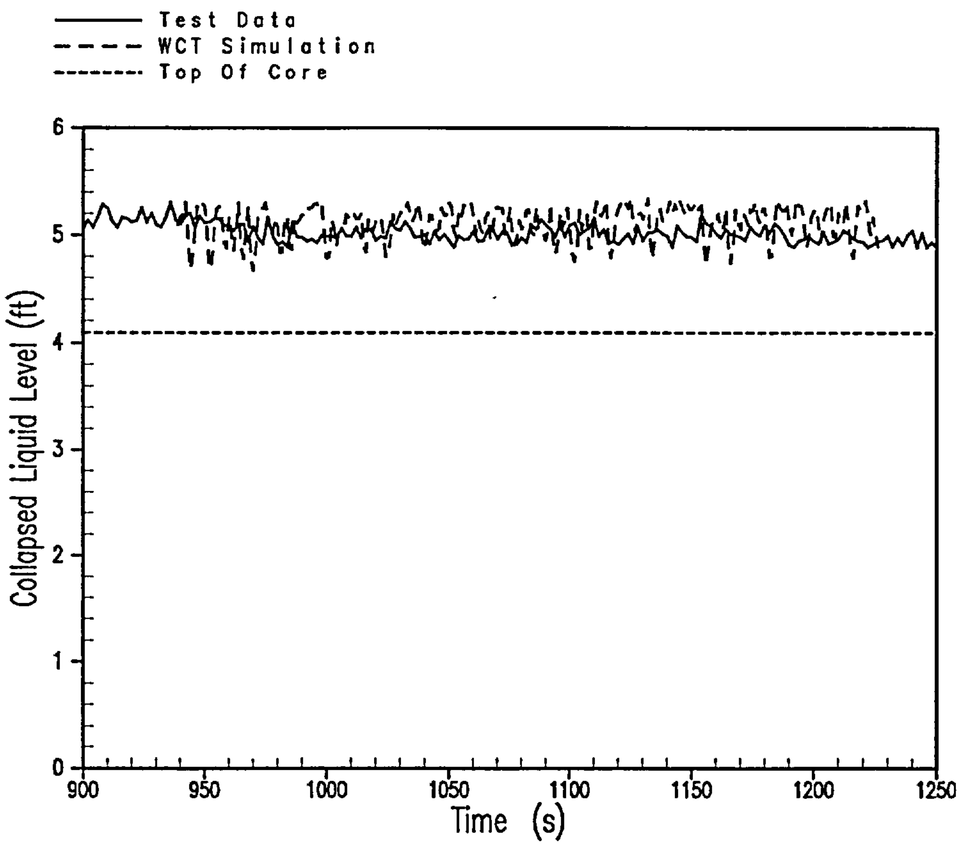
Design Control Document (DCD) Revision:

None

PRA Revision:

None

Figure 440.131-1: OSU Test SB18 ADS-4 IRWST Initiation Phase Upper Plenum Collapsed Levels (Relative to Bottom of Lower Plenum)



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.138

Question:

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

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

Westinghouse Response:

The safety-related function of the pressurizer safety valves is to protect the reactor coolant system from transient events that cause an increase in system pressure. The safety valve capacity is selected to relieve sufficient steam from the pressurizer so that the RCS pressure following any Condition I and II event remains below 110% of RCS design pressure in accordance with the ASME code requirements. This safety-related function is included in the AP1000 Tier 1 material for the reactor coolant system, and is identified in Tier 1 Table 2.1.2-4, Item 8.a.

The statement that is referred to in the DCD section 5.4.9.3 describes the behavior of the pressurizer safety valves during some postulated transients, where the rate of pressurization in the long term (i.e. > 1 hour) is significantly less than the design basis pressurization rate. This valve behavior is modeled in the transient analysis provided in Chapter 15, as are many other operating characteristics of the passive safety systems. Due to specific design characteristics of different vendor's safety valves, some design options may or may not exhibit such behavior. For instance, spring loaded safety valves are typically assumed to open at 3% accumulation pressure above the valve set pressure, and reclose within 5% of the opening pressure. Sensitivity studies were performed for AP600 that demonstrated that if such valve behavior were assumed in the Chapter 15 analysis during this long term operation of the passive safety systems, the acceptance criteria of no pressurizer overfill would still be met. For AP1000, such valve behavior also yields acceptable transient results. Thus the AP1000 safety analysis does not depend on avoiding such valve behavior.

A graded approach to the level of detail that is necessary to be included in Tier 1 is applied. Therefore, the design basis overpressure function of the pressurizer safety valves is included in Tier 1. The description of the safety valve behavior described in section 5.4.9.3 is not included in Tier 1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.148

Question:

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

- Table 2.5.1-1 indicates that the in-containment refueling water storage tank (IRWST) gutter isolation valve closure is actuated on low wide-range steam generator water level or on high hot leg temperature, whereas Tier 2 Section 7.7.1.11 indicates the IRWST gutter isolation is actuated on the high hot leg temperature signal only.

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

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

Westinghouse Response:

- DCD section 7.7.1.11 states that a low wide-range SG water level "initiates PRHR heat removal". PRHR heat removal includes closing the gutter isolation valves. Figure 7.2-1 shows that any signal that opens the PRHR HX discharge isolation valves also closes the IRWST gutter isolation valves.

- The DAS Tier 1 design description 2.a states that "DAS provides an automatic reactor trip on low wide-range steam generator water level or on low pressurizer water level separate from the PMS". In addition, Table 2.5.1-4, item 2.a, describes an ITAAC that verifies that DAS provides a automatic reactor trip. Table 2.5.1-1 will be revised to make it consistent with the rest of Tier 1 and Tier 2 DAS information.

Design Control Document (DCD) Revision:

Revision to Tier 1, Table 2.5.1-1:

Table 2.5.1-1 Functions Automatically Actuated by the DAS
1. Reactor and Turbine Trip on Low Wide-range Steam Generator Water Level or Low Pressurizer Water Level
2. Passive Residual Heat Removal (PRHR) Actuation and In-containment Refueling Water Storage Tank (IRWST) Gutter Isolation on Low Wide-range Steam Generator Water Level or on High Hot Leg Temperature
3. Core Makeup Tank (CMT) Actuation and Trip All Reactor Coolant Pumps on Low Pressurizer Water Level
4. Isolation of Selected Containment Penetrations and Initiation of Passive Containment Cooling System (PCS) on High Containment Temperature

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.152

Question:

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.

Westinghouse Response:

The available correlations specify the branch line quality as a function of main line liquid level, and the level at onset of entrainment. The branch line quality then is defined as,

$$X_{BR} = \frac{\rho_g V_g \alpha_{BR} A}{\rho_g V_g \alpha_{BR} A + \rho_l V_l (1-\alpha_{BR}) A}$$

Solving for α_{BR} as,

$$\alpha_{BR} = \frac{\rho_l V_l X_{BR}}{\rho_g V_g \left(1 - X_{BR}\right) + \rho_l V_l X_{BR}}$$

Where X_{BR} is the branch line quality, α_{BR} is the branch line void fraction, and V_g and V_l are phase velocities. In the actual coding these velocities were taken from the previous time step.

The collapsed liquid level is used for the prediction of branch line quality. The entrained droplets that may be present in the stratified flow are added to the actual level in the main line.

If the predicted branch line void fraction is less than the donor cell void fraction (in this application, the top cell of the main line channel). The branch line void fraction is set to the void fraction in the top cell of the main line channel where the 1D component is attached.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.153

Question:

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?

Westinghouse Response:

If the main pipe flow is not stratified, the main pipe cell connected to the branch line becomes the donor cell and the void fraction and phasic properties are convected from this cell to the branch line. In ADS-4 application, the top cell of the main pipe channel is connected to the branch line so that, when the stratified flow is not predicted, the content of the top cell is convected to the ADS-4 line.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.156

Question:

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 Ug 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.

Westinghouse Response:

The code uses the local vapor phasic velocity while the original paper used the superficial velocity in their final correlation. There are two reasons to use the vapor phasic velocity instead of the superficial velocity. (1) Ishii-Grolmes used a simplifying assumption in order to arrive at the resultant correlation which uses the vapor superficial velocity. Their simplifying assumption is that the liquid film thickness compared to the diameter of the pipe is small enough such that the vapor superficial velocity is close to the vapor phasic velocity and that the contribution of film thickness to the friction factor may be neglected. With this assumption, the relative velocity was replaced with the vapor superficial velocity in their work [1]. (2) In order to use the Ishii-Grolmes correlation to a stratified flow with a relatively deeper liquid (which would be considered an extrapolation), JAERI test [2] was studied. This test which used a deep liquid layer (~0.5 m) in a rectangle conduit (0.1 m wide, 0.7 m high and 28.3 m long) showed that the vapor phasic velocity at the onset of entrainment matched that of Ishii-Grolmes. The test condition used in this test was j_l=0.89 m/s, and j_g=1.49 m/s while the calculated onset of entrainment according to Ishii-Grolmes was 17 m/s which was closed to the estimated local phasic velocity at the onset of entrainment.

It is felt that with these observations, the use of Ishii-Grolmes correlation (with the vapor phasic velocity) for a stratified flow with deeper water is reasonable.

[2] H. Nakamura, Y. Kukita, "An Evaluation of Bernoulli Effect on Slugging in Horizontal Two-Phase Flow," Jounal of Nuclear Science and Technology, Volume 31, No. 2, pp 113~121, February 1994.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.169

Question:

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.

Westinghouse Response:

Appendix A.3 of WCAP-15833 provides a comparison of the amount of liquid entrainment predicted to occur in the upper plenum of the AP1000. The comparison is between the results obtained with the WCOBRA/TRAC liquid film-type entrainment models described in WCAP-15833 and the results obtained with the Kataoka-Ishii pool-type entrainment model. It was concluded that the WCOBRA/TRAC film-type entrainment models produce entrainment in the same range as the Kataoka-Ishii pool-type correlation for this application. Applicability of the film type entrainment models is documented in Volume 1, Section 4 of the Westinghouse Code Qualification Document for Best Estimate LOCA. Sensitivity analyses with WCOBRA/TRAC show that core cooling is not sensitive to variations in the amount of liquid entrainment in the upper plenum. The Kataoka-Ishii pool-type entrainment model is not applied in WCOBRA/TRAC.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.181

Question:

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. Previously in support of the AP600 design certification review, 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." 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.

Westinghouse Response:

A. The RCCA Ejection Accident analysis results for the Beginning of Cycle (BOC), End of Cycle (EOC), Hot Full Power (HFP) And Hot Zero Power (HZP) cases are given in Table 15.4-3 of the AP1000 Design Control Document. The hot spot radially averaged peak fuel enthalpy (RAPFE) results for the four cases are:

1) BOC HFP 181 cal/g
2) BOC HZP 104 cal/g
3) EOC HFP 170 cal/g
4) EOC HZP 117 cal/g

These values of peak fuel enthalpy are well under the 280 cal/g limit specified in the NRC's Standard Review Plan Section 15.4.8 (NUREG 0800, July 1981) and Regulatory Guide 1.77 (May 1974). The calculated values are also under the Westinghouse-specified analysis limit of 200 cal/g. The HZP case results, which are of concern with respect to high burnup fuel, are only slightly above 100 cal/g at the peak hot spot in the core, which does not occur in high burnup fuel rods. It should be noted that these results were obtained using the approved Westinghouse 1-D analysis methodology, which is very conservative compared to a 3-D neutronics analysis method as stated in the Westinghouse report NTD-NRC-95-4438 (see below).

B. The Westinghouse report NTD-NRC-95-4438 was written in response to an NRC request to confirm the continued validity of any Westinghouse Topical reports which address fuel performance, in the light of the (at that time) recent experimental results (notably the CABRI Reactivity Insertion Accident results) which showed an unexpected fuel failure for a zero initial power event at a relatively low enthalpy rise. It was concluded in the Report that the radiological consequences would still meet the acceptance criteria for the rod ejection accident, even if additional high burnup fuel failures were assumed. (This conclusion was based on an assumed failure limit of 30 cal/g above 40,000 MWD/MTU burnup.) Since that time, additional zero power RIA testing in the CABRI reactor has not duplicated the low failure threshold of the first test. In fact, the test results indicate that fuel failures in high burnup rods are not expected for radially averaged peak fuel enthalpies below 100 cal/g. Consequently, the generic analysis and discussion presented in NTD-NRC-95-4438 can conservatively be applied to the AP1000 design.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 451.002

Question:

Section 2.3.4 of the AP1000 DCD discusses calculation of the bounding short-tem relative concentration (X/Q) values for use in the off-site design-basis accident dose assessments. Was this description provided for information only and it is 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 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?

Westinghouse Response:

As stated in DCD Section 2.3.4, the set of short-term relative concentration (X/Q) values (i.e., reference site values) was selected to envelope potential AP1000 construction sites. These values were not calculated but were selected to bound the values for a large percentage of existing nuclear power plant sites (also see the response to RAI 451.001).

DCD Section 2.3.6.4 currently states that the COL applicant will address the X/Q values specified in subsection 2.3.4 to confirm their applicability to a specific site. In order for the COL applicant to address this area, it is necessary to collect site-specific meteorological data and perform the appropriate analysis to calculate the atmospheric dispersion factors. As required for all Combined License information items, the specific methodology, inputs and assumptions used in this analysis will be provided as part of the COL application.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 451.004

Question:

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 and it is 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 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 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?

Westinghouse Response:

Section 2.3.5 reads as follows:

> The long-term diffusion estimates are site specific and will be provided by the Combined License applicant. The site boundary annual average X/Q shown in Table 2-1 is used to calculate release concentrations at the site boundary for comparison with the activity release limits defined in 10 CFR 20. The value specified is expected to bound atmospheric conditions at most U.S. sites. If a selected site has a X/Q value that exceeds this reference site value, the release concentrations reported in Section 11.3 would be adjusted proportionate to the change in X/Q.

As stated in Section 2.3.5, the long-term X/Q value was specified for the reference site, not calculated. The selection of the long-term X/Q was made with the expectation that the value would exceed the values calculated for specific sites. Section 2.3.6.5 currently states that the COL applicant will address long-term diffusion estimates and X/Q values specified in subsection 2.3.5 to confirm that they are applicable to the specific site being considered. In order for the COL applicant to address this area, it is necessary to collect site-specific meteorological data and perform the appropriate analysis to calculate the atmospheric dispersion factors. As required for all Combined License information items, the specific methodology, inputs and assumptions used in this analysis will be provided as part of the COL application.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 451.005

Question:

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. 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.

Westinghouse Response:

The first paragraph of 15A.3.3 will be modified, as indicated below, to specify that Section 2.3.4 provides the off-site short-term atmospheric dispersions factors (χ/Q) for the reference site.

Design Control Document (DCD) Revision:

From DCD Chapter 15, Section 15A.3.3:

15A.3.3 Atmospheric Dispersion Factors

Subsection 2.3.4 lists the off-site short-term atmospheric dispersion factors (χ/Q) for the reference site. Table 15A-5 (Sheet 1 of 2) reiterates these χ/Q values.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 451.007

Question:

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.

Westinghouse Response:

For AP1000, the environmental impact of heat dissipation systems will be evaluated as part of the COL application. The heat dissipation systems such as the discharge canal and the cooling tower are not safety-related.

DCD Tier 1, Chapter 5 and DCD Tier 2, Chapter 1, Section 1.8 define the key site parameters that are specified for the design of safety-related aspects of structures, systems, and components and those which require evaluation as part of the COL application.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.001

Question:

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 steam generators associated with the radiological consequences analysis of the MSLB?

Westinghouse Response:

A. In the analysis of radiological consequences of a MSLB the primary-to-secondary leakage is assumed to continue until the reactor coolant system (RCS) temperature (i.e., the temperature of the leakage) is less than 212°F. For the AP1000 the time to cool the RCS is calculated assuming the operators cool to the residual heat removal (RHR) system cut-in conditions by dumping steam from the intact steam generator(s) and then continue to cool with the normal RHR system. The basis for using 72 hours is the time needed to cool the RCS to 212°F.

B. Releases from the intact steam generator continue until the PRHR or the normal residual heat removal system (RHRS) is removing all decay heat. Releases from the faulted steam generator continue until the RCS has been cooled below 212°F. For the dose analysis presented in the DCD, releases from both steam generators were conservatively assumed to continue for 72 hours. Data was provided for the initial two hour interval since this was determined to be the limiting two hour interval for the site boundary dose for both the accident-initiated and pre-accident iodine spikes.

   It is assumed that the faulted steam generator releases all of its mass and initial activity in the first 10 minutes of the event and that the intact steam generator releases all of its mass and initial activity in the first hour. The initial mass of 3.03E5 lb per steam generator was conservatively calculated based on no-load conditions. All activity transferred from the RCS via the 175 lb/hr primary to secondary leakage to each steam generator is assumed to be released to the atmosphere with no retention in the secondary side of either steam generator. For the first 2 hours the primary system leakage of 350 lb per steam generator is added to the initial mass. Table 15.1.5-1 is being corrected to show the total steam release of 3.0335E5 lb from each of the steam generators in the 0-2 hour interval.

In the subsequent 70 hours of the event the leak of 175 lb/hr per steam generator transfers 1.225E4 lb into each of the steam generators and all of this mass is released to the atmosphere.

Design Control Document (DCD) Revision:

From DCD Chapter 15.1, Table 15.1.5-1:

Steam generator in intact loop	
– Primary to secondary leak rate (lb/hr)	175[(a)]
– Iodine partition coefficient	1.0
– Steam released (lb)	
0 - 2 hr	~~3.03 E+05~~3.0335E+05
2 - 72 hr	1.225 E+04
Nuclide data	See Table 15A-4

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.002

Question:

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 that 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% 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% 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% 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)?

Westinghouse Response:

A. The response to RAI Number 440.080 discusses the basis for the determination that no fuel is damaged as a result of the design basis locked rotor accident. The bounding dose analysis is performed assuming some fuel failure and 16 percent was selected based on preliminary conservative fuel failure assessments.

B. Safety-related decay heat removal is provided by the Passive Residual Heat Removal (PRHR) System. In the event of a locked rotor there may be no primary system signal that actuates the PRHR. In this case decay heat is removed by steaming off the secondary inventory until the low steam generator level signal actuates the PRHR. Eventually the PRHR system is removing all decay heat and heat transfer to the steam generators stops, thus terminating steaming. It was determined that this would occur within the first 1.5 hours.

   If startup feedwater is available, then steam releases would continue until the normal residual heat removal system (RNS) is operating and removing all decay heat, which occurs within eight hours of event initiation. This was determined to be a less limiting scenario since, with startup feedwater available, the iodine and alkali metal activity contained in the primary to secondary leakage that flashes is not directly released but is assumed to mix in the secondary liquid and be released with the steam, subject to partitioning.

C. As discussed above, steam releases stop when the PRHR is removing all decay heat. This occurs within the first 1.5 hours for the case with no startup feedwater available. The steam releases were calculated assuming the maximum initial steam generator water inventory. This maximizes both the time until the PRHR setpoint is reached and the mass of steam that is released until that time. The analysis also modeled minimum PRHR heat transfer capability.

D. The temperature of the hot leg following the locked rotor was used together with the secondary pressure to calculate the flashing fraction. Following PRHR actuation the primary temperature drops to the extent that the leak flow no longer flashes. This occurs before 1 hour after event initiation. The average flashing fraction over the hour was calculated to be less than 0.04 (i.e., less than 4% of the leakage flashes). Table 15.3-3 is being corrected to show the flashing as a fraction (0.04) with no units.

E. The reactor coolant noble gas activity used in the locked rotor analysis was 280 μCi/gm, dose equivalent Xe-133. The typographical error is being corrected.

F. The locked rotor analysis modeled Kr-85 with a gap fraction of 0.10. Kr-84 (which is stable) was not modeled. The typographical error is being corrected.

Design Control Document (DCD) Revision:

From DCD Chapter 15.3,Table 15.3-3:

Table 15.3-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A LOCKED ROTOR ACCIDENT

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 μCi/gm of dose equivalent I-131 (see Appendix 15A)[a]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 ~~m Ci/gm~~ μCi/gm dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of design basis reactor coolant concentrations at maximum equilibrium conditions
Fraction of fuel rods assumed to fail	0.16
Core activity	See Table 15A-3
Fission product gap fractions	
I-131	0.08
~~Kr-84~~ Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Reactor coolant mass (lb)	3.7 E+05
Secondary coolant mass (lb)	6.06 E+05
Condenser	Not available
Duration of accident (hr)	1.5 hr
Atmospheric dispersion factors	See Table 15A-5
Primary to secondary leak rate (lb/hr)	350[b]
Steam released (lb)	
0-1.5 hours[c]	6.48 E+05
Partition coefficient in steam generators for iodine and alkali metals	0.01
Leak flashing fraction~~(%)~~[d]	
0-60 minutes	0.04
> 60 minutes	0

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.003

Question:

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% 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?

Westinghouse Response:

A. The analysis included the maximum radial peaking factor of 1.65 in the calculation of activity released from failed/melted fuel. DCD, Chapter 15.4, Table 15.4-4 will be modified, as indicated below, to reflect this assumption.

B. As discussed below (Item C) the SBLOCA transient was used to provide transient data for the rod ejection radiological consequences analysis. The flashing fraction was calculated using the transient vessel average temperature from the SBLOCA analysis. The fraction of 0.04 (4% flashing) was chosen to bound the transient results. The analysis conservatively maintained this fraction for the initial 1800 seconds of the transient.

C. The design basis rod ejection transient results from a mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. This failure results in a loss of coolant accident with a possible reactivity insertion event. The steam generator steam releases to the environment, the time for the primary pressure to fall below the secondary pressure and the leak flashing fraction were chosen to bound those calculated for the 2 inch small break loss of coolant accident (SBLOCA). The 2 inch break is smaller than the flow area that results from a control rod mechanism pressure housing failure. The smaller break conservatively extends the steam releases and has a slower primary depressurization. This delays the time when the primary pressure drops below the secondary pressure and extends the time when the steam generators are steaming to remove decay heat.

   Figure 15.6.5.4B-17 of the DCD shows the primary pressure transient for the SBLOCA. The steam generator pressure is maintained at the safety valve setpoint until the reverse heat transfer to the primary system starts when the primary pressure falls below the secondary pressure. From Figure 15.6.5.4B-17 the primary pressure is below the secondary pressure well before the 1800 seconds assumed in the radiological consequences analysis.

   Heat transfer to the steam generators, and consequently steam releases from the steam generators, also stops well before 1800 seconds. The average steam flow rate until steam releases stop was calculated using the SBLOCA analysis results and conservatively increased to 60 lbm/sec for use in the radiological consequences analysis. The analysis conservatively maintained this rate for the initial 1800 seconds of the transient, resulting in a total steam release of 1.08E5 lbm.

D. The retention of particulate radionuclides such as alkali metals in the steam generators is limited by the moisture carryover from the steam generators consistent with the guidance of RG 1.183. The design full power moisture carryover fraction for the AP1000 is 0.001, and the moisture carryover would drop following reactor trip. The radiological consequences analysis conservatively maintained the full power value for the duration of the analysis.

Design Control Document (DCD) Revision:

Table 15.4-4 (Sheet 1 of 2)

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A ROD EJECTION ACCIDENT

Initial reactor coolant iodine activity	An assumed iodine spike that has resulted in an increase in the reactor coolant activity to 60 µCi/g of dose equivalent I-131 (see Appendix 15A)[a]
Reactor coolant noble gas activity	Equal to the operating limit for reactor coolant activity of 280 µCi/g dose equivalent ~~Xe-13~~ Xe 133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal activity	10% of reactor coolant concentrations at maximum equilibrium conditions
Radial peaking factor (for determination of activity in failed/melted fuel)	**1.65**
Fuel cladding failure	
– Fraction of fuel rods assumed to fail	0.1
– Fission product gap fractions	
Iodines and noble gases	0.1
Alkali metals	0.12
Core melting	
– Fraction of core melting	0.0025
– Fraction of activity released	
Iodines and alkali metals	0.5
Noble gases	1.0
Iodine chemical form (%)	
– Elemental	4.85
– Organic	0.15
– Particulate	95.0
Core activity	See Table 15A-3 in Appendix 15A
Nuclide data	See Table 15A-4 in Appendix 15A

Reactor coolant mass (lb)	3.7 E+05

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 470.005

Question:

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?

Westinghouse Response:

A. The flashing fraction was determined using a constant enthalpy process, based on the time-dependent temperature of the break flow and the time dependent ruptured steam generator secondary pressure. The primary system pressure shown in DCD Figure 15.6.3-2 and the hot leg temperature shown in DCD Figure 15.6.3-4 are used to calculate the enthalpy of the leakage. It is conservatively assumed that all leakage is at the hot leg inlet temperature. The hot leg temperature of the two loops is essentially the same in the period of interest. The secondary system saturation enthalpy is determined from the pressure shown in DCD Figure 15.6.3-3. Flashing stops before 3200 seconds, due to the combination of higher secondary pressure and colder primary temperature.

B. As indicated in DCD Table 15.6.3-1, break flow stops at 24,100 seconds (6.69 hours). It was assumed that the plant is cooled to the normal residual heat removal (RHR) system cut-in conditions by dumping steam from the intact steam generator within 6.5 hours of break flow termination. During this time the ruptured steam generator is also depressurized to the RHR cut-in pressure (to prevent re-establishing break flow) by dumping steam. Once it is put in service at 13.19 hours the RHR system removes all decay heat and no further steam is released.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.018

Question:

DCD Rev. 0, pages 18.1-2 and 18.1-3. The "annotated outline" titles of the chapter should be modified if they are to remain in agreement with current NRC guidance, NUREG-0711, Rev. 1, May 2002.

Westinghouse Response:

Sections 18.13, "Design Implementation", and 18.14, "Human Performance Monitoring", will be added to satisfy NUREG-0711 Rev 1.

Section 18.13 will be mapped to the existing activities denoted "Issue Resolution Verification" and "Final Plant HFE Verification" in Section 18.11. The applicable portion of NUREG-0711 Rev.1 Section 12 (specifically, 12.4.6) is equivalent to the combined Sections 11.4.5 and 11.4.6 of the original NUREG-0711. This input was previously reviewed and found acceptable; it therefore satisfies current guidance. This mapping approach will preserve consistency within the existing DCD document.

Section 18.14 will be incorporated as a COL action item, in similar fashion to Procedure Development (18.9) and Training Program Development (18.10).

Design Control Document (DCD) Revision:

[in DCD Section 18.1]

...

Section 18.9, Procedure Development - ~~and~~ Reference 7 provides input to the Combined License applicant for the development of plant operating procedures, including information on the AP1000 emergency response guidelines and emergency operating procedures.
Section 18.10, Training Program Development - ~~and~~ Reference 8 provides input from the designer on the training of the operations personnel who participate as subjects in the human factors verification and validation.
Section 18.11, Human System Interface ~~Design Test~~ Verification and Validation Program - ~~and~~ [*Reference 9 presents a programmatic level description of the human factors verification and validation.*]*

Section 18.12, Inventory - presents the minimum inventory of controls, displays, and alarms present in the main control room and at the remote shutdown workstation. The design basis and the selection criteria used to identify the minimum inventory are presented.
Section 18.13, Design Implementation – In accordance with Reference 1, this issue is addressed under Section 18.11 as "Issue Resolution Verification" and "Final Plant HFE Verification".
Section 18.14, Human Performance Monitoring – Human performance monitoring applies after the plant is placed in operation, and is a Combined License applicant responsibility.

[in DCD Section 18.2.5]

*[The human factors engineering program is performed in accordance with the human factors engineering process specified in NUREG-0711 (Reference 1).]** Figure 18.1-1 shows the ~~ten~~ elements of the AP1000 human factors engineering program. *[These elements conform to the elements of the Program Review Model specified in Reference 1, **as augmented by Reference** 7.]**

~~The first element,~~ Human factors engineering Program Management~~,~~ is addressed in Section 18.2. The remaining elements ~~2 through 10~~ are addressed in Sections 18.3 through **18.11, 18.13, and 18.14.**

These sections~~ corresponding to elements 2 through 10~~ address the activities conducted as part of the **corresponding** human factors engineering element, **including** the accepted industry standards, guidelines, and practices used as technical guidance, the inputs to the element, and the products, including documents~~ation~~ that are generated as output. The facilities, equipment, and tools employed~~-~~ are also addressed in the section corresponding to each element.

Operating Experience Review (~~element 2~~**Section 18.3**) and Functional Requirements Analysis and Function Allocation (~~element 3~~**Section 18.4**)~~,~~ are completed. Implementation plans are provided for ~~(element 4)~~ Task Analysis **(Section 18.5)**, ~~(element 6)~~ Integration of Human Reliability Analysis **(Section 18.7)** and ~~(element 7)~~ Human System Interface Design **(Section 18.8)**. Staffing (~~element 5~~**Section 18.6**), Procedure Development (~~element 8~~**Section 18.9**), ~~and~~ Training Development (~~element 9~~**Section 18.10**), **and Human Performance Monitoring (section 18.14)** are Combined License applicant responsibilities. A programmatic level description is provided for Human Factors Verification and Validation (~~element 10~~**Section 18.11**). **Human Factors Verification and Validation also addresses the activities identified under Design Implementation (Section 18.13).**

...

[in DCD Section 18.2.7]

...

[7. NUREG-0711, Rev.1, "Human Factors Engineering Program Review Model," U.S. NRC.]*

[in DCD Sections 18.13 and 18.14]

18.13 Design Implementation

This process element is added by Reference 2 to the Program Review Model specified in Reference 1. However, it mostly applies to plant modernization. The portions of the added element that apply to new plants were formerly addressed under the Verification and Validation element in Reference 1. Since these aspects of the Program Review Model are unchanged, AP1000 will continue to address them under Section 18.11 as "Issue Resolution Verification" and "Final Plant HFE Verification".

18.13.1 References

1. NUREG-0711, "Human Factors Engineering Program Review Model," U.S. NRC.
2. NUREG-0711, Rev.1, "Human Factors Engineering Program Review Model," U.S. NRC.

18.14 Human Performance Monitoring

Human performance monitoring applies after the plant is placed in operation, and is a Combined License applicant responsibility. Guidance and additional information on the objectives, scope, and methods of such programs are presented in Element 13 of Reference 1.

18.14.1 References

1. NUREG-0711, Rev.1, "Human Factors Engineering Program Review Model," U.S. NRC.

PRA Revision:

None

RAI Number: 620.025

Question:

DCD Rev. 1, page 18.2-19 (Figure 18.2-1). Please explain the basis for changing the human-system interface (HSI) design team's responsibility for building a "control room mock-up" to building a "partial mockup (as needed)."

Westinghouse Response:

This is not a change, but rather a clarification of the existing process. Mockups are inherently partial, which is acceptable for engineering tests because they are preliminary (i.e. based on evolving designs). Thus, a partial mockup that models important components of the AP1000 control room HSI will be provided to support design and testing activities. As a separate activity, final validation will be performed on a full scope, AP1000-specific simulator.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.026

Question:

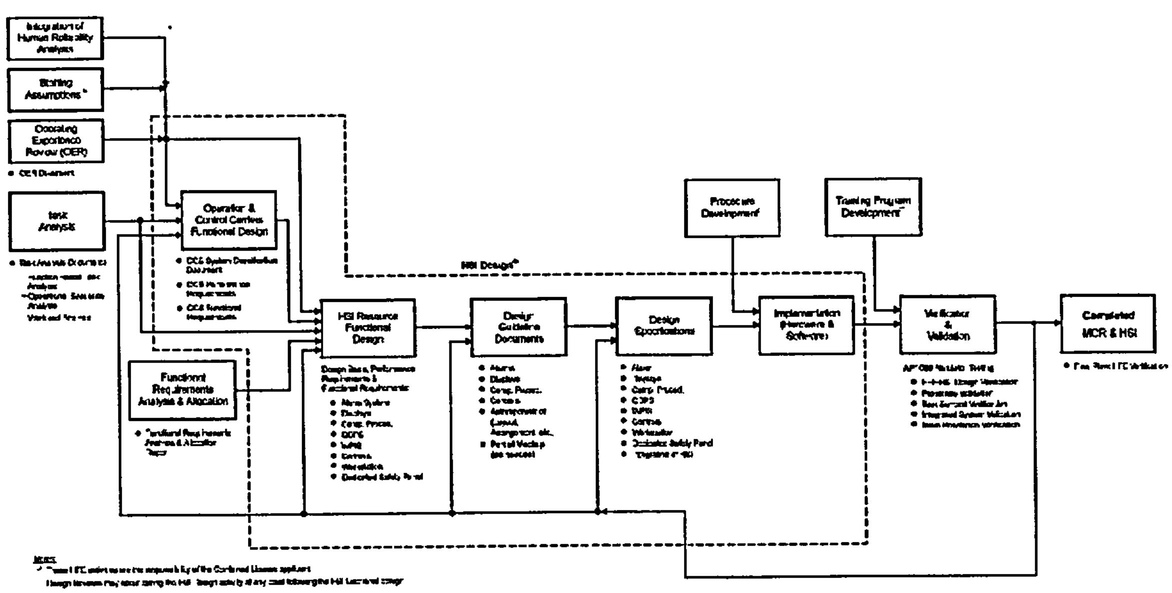
DCD Rev. 1, page 18.2-21, Figure 18.2-3. The figure should be modified if it is to remain in agreement with current NRC guidance, NUREG-0711, Rev. 1, May 2002. In addition, please provide the basis for all changes made to the HFE process from the process approved for the AP600 design.

Westinghouse Response:

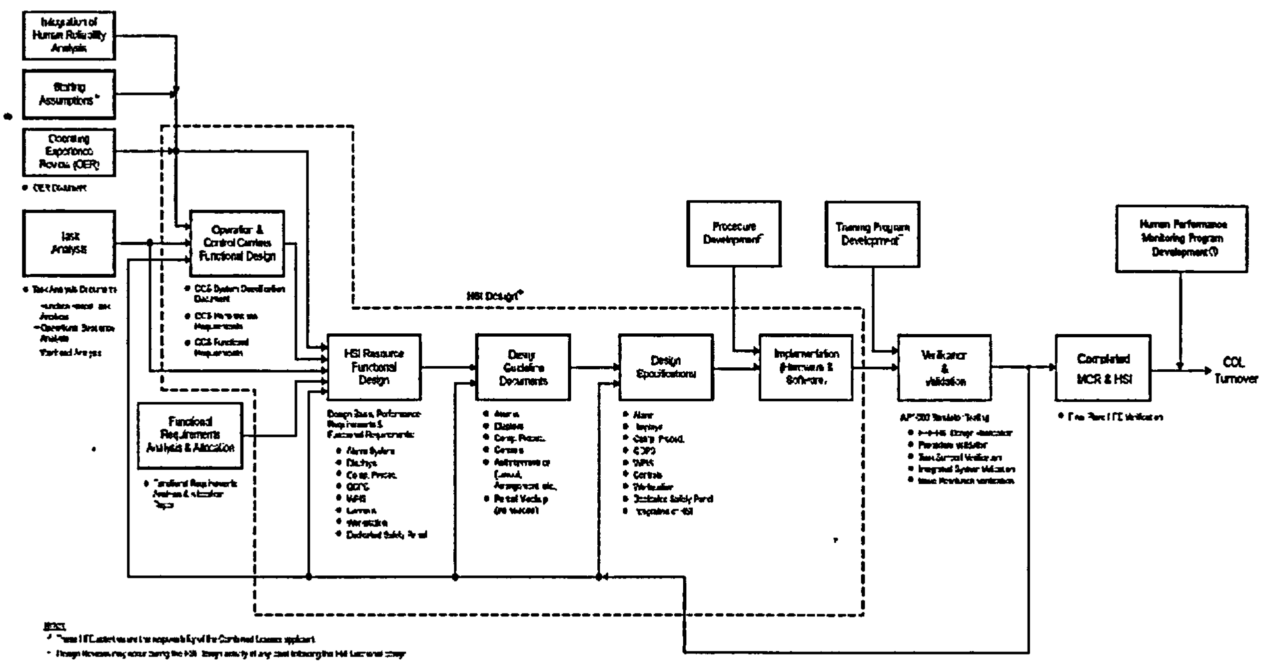
An activity block corresponding to DCD Section 18.14, "Human Performance Monitoring", will be added to Figure 18.2-3 to satisfy NUREG-0711 Rev 1. See also responses to RAIs 620.008 and 620.018.

Design Control Document (DCD) Revision:

OLD FIGURE:



NEW FIGURE:



PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.032

Question:

620.032, DCD, Rev. 1, page 18.8-5, paragraph 18.8.1.4

Please explain the basis for eliminating Man-In-The-Loop Concept Testing from the AP1000 HSI design process. (See previous Tier 1 question 620.008.)

Westinghouse Response:

See the response to RAI #620.008 that address removal of Concept Testing.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 630.004

Question:

(Section 16.1, TS Sections 1.2, 1.3, and 1.4) The last paragraph of Example 1.3-6 of the AP1000 TS Section 1.3 and in Section 16.1, page 1.3-11, is incorrectly placed after the title for Example 1.3-7. Also, each example in Sections 1.2, 1.3, and 1.4 should start on a new page, consistent with the STS format. Please revise the DCD accordingly.

Westinghouse Response:

The last paragraph of Example 1.3-6 of the AP1000 TS Section 1.3 (Section 16.1, page 1.3-11) will be revised as indicated in the attachment.

The other requested changes are format changes. As discussed in RAI 630.001, the AP1000 Technical Specification update to STS Revision 2 include technical changes but will not include formatting changes.

Design Control Document (DCD) Revision:

See attached AP1000 Tech Spec markup.

PRA Revision:

None

1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed, and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

~~EXAMPLE 1.3-7~~

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

EXAMPLE 1.3-7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND	
	A.2 Restore subsystem to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

(continued)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.002

Question:

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

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

Westinghouse Response:

During licensing of the AP600 Westinghouse and the NRC discussed the acceptability of small core uncoveries that had extended / slow recovery even though the PCT was well below 2200°F. In the early AP600 PRA, the ADS success criteria was 1 of 4 ADS stage 4 valves. With this criteria, there were some sequences with "extended uncovery" on the order of 1 hour or more that occurred during some transients and small break LOCA events. These uncoveries started 3 to 4 hours after reactor trip when decay heat is relatively low. The NRC was concerned that such a situation was an indication of low plant system performance margin, even though the core was being adequately cooled. Westinghouse agreed with the NRC that such situations should be reviewed carefully, however no specific criteria time was specified. Scenarios with extended core uncoveries (>1 hour) were eliminated from the AP600 when the ADS stage 4 success criteria was changed to increase the number of ADS valves required to 2 of 4 valves.

It is useful for this discussion to consider the final AP600 PRA T/H analysis cases that had core uncovery and were considered successful core cooling. Several AP600 MAAP4 / NOTRUMP benchmarking cases resulted in core uncovery. In all of these cases, the duration of core uncovery were not extended (i.e. the longest core uncovery was ~ 27 min) and they were considered to be successful core cooling. In addition, several AP600 T&H uncertainty cases resulted in core uncovery. In all of these cases, the duration of core uncovery was not extended (i.e. the longest core uncovery was ~ 37 min) and they were considered to be successful core cooling.

The AP1000 core uncoveries seen in the PRA T&H analysis are considered acceptable because the PCTs are well below 2200°F and the durations (< 18 min) are significantly less than the extended core uncoveries (> 1 hr) seen in the early AP600 PRA analysis. In addition, the AP1000 durations are also less than the core uncoveries (< 37 min) seen in the final AP600 PRA success criteria and T&H uncertainty analysis which were considered to be successful.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.003

Question:

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

Westinghouse Response:

As discussed in the response to RAI 720.002, the AP1000 core uncoveries are significantly shorter than the extended core uncoveries seen in the early AP600 that were greater than 1 hour.

In addition, the AP1000 core uncoveries are similar to or less severe than the corresponding core uncoveries for the AP600. As shown in Figures 3.2-7, 3.2-13 and 3.2-19 of the AP1000 PRA, the AP600 core uncovery is deeper and longer than the AP1000 for these success criteria analysis.

The T&H uncertainty analysis of the AP600 for the case UC5 (equivalent to AP1000 case 3) shows that the AP600 uncovers for 950 sec with a PCT of 1435°F. For the corresponding case, the AP1000 uncovery is very similar, with a duration of 1070 sec and a PCT of about 1570°F. In another AP600 T&H case (UC61) the analysis shows that the core uncovered for 2200 sec with a PCT of 1235°F. Both of these AP600 cases were considered successful core cooling.

As a result, the AP1000 T&H uncertainty case is not considered an extended core uncovery and should be considered successful.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.008

Question:

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

Westinghouse Response:

The A3.2-2 through A3.2-25 figures are from MAAP4 analysis for both AP600 and AP1000. There are several places in section 3.2 where MAAP4 and NOTRUMP analyses are mentioned. In each case, the MAAP4 and NOTRUMP analysis applies to AP600 with a reference to the AP600 MAAP4/NOTRUMP benchmarking document (ref. A-3).

For example, from section A.3.2.1, bottom of page A-15:

"In addition, the AP600 plant response to these cases was documented in Reference A-3, based on MAAP4 and NOTRUMP analysis."

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.009

Question:

Section A3.3.1 discusses success paths involving manual ADS4 leading to IRWST gravity injection. Analyses by both MAAP4 and NOTRUMP are referenced. Successful operation of the following equipment is credited in these success paths for hot leg break sizes of 3.5 to 8.75 inches: the PRHR HX, 1 of 2 accumulators, 3 of 4 ADS4 valves and, 1 of 2 IRWST injection paths. Containment isolation failure is assumed so that the containment remains at a low pressure throughout the event.

A. These success paths do not appear to be bounded by any of the analyses in Section A5.1 "T/H Uncertainty Cases for AP1000." Both Case No. 1 and Case No. 2 assume manual actuation of ADS4; however both Case No. 1 and Case No. 2 assume credit for 4 of 4 ADS4 valves and take credit for elevated containment pressure. Furthermore, Case No. 2 assumes credit for 2 of 2 accumulators. Please provide T/H uncertainty evaluation for the success paths in Section A3.3.1.

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

Westinghouse Response:

(A) During the licensing of the AP600, Westinghouse and the NRC discussed the approach that Westinghouse would use to address T&H uncertainty in the PRA. AP1000 uses the same approach as was used for the AP600. The cases selected for T&H uncertainty analysis (section A5) were determined to be the risk important, low margin sequences. It is not necessary or appropriate for the T&H uncertainty cases to have the same number of failures as in success criteria. In AP600 and AP1000 the success criteria cases tend not to be risk important because so many failures are included. The response to RAI 720.017 provides additional discussion of the risk important, low margin sequences in the AP1000.

(B) The analysis reported in PRA section 3.3.1 was performed with MAAP4. For the medium LOCA success criteria analysis the MAAP4 PRHR HX performance was compared with the NOTRUMP PRHR HX performance for a 3" HL LOCA. Figure 720.009-1 shows that the

MAAP4 PRHR HX removes a similar, but smaller amount of heat from the RCS as a function of pressure. Figure 720.009-2, -3, -4 show the resulting RCS pressure, accumulator mass and core mixture level calculated by MAAP4 and NOTRUMP. These parameters are very similar between the two.

The analysis reported in PRA section A.5.1 was performed with NOTRUMP. As stated in section 15.6 of the AP1000 DCD, the small break LOCA analysis performed for AP1000 that are presented in Chapter 15 of the DCD use the heat transfer penalty PRHR heat transfer that was identified for the AP600, for cases when the velocity in the PRHR tubes is greater than 1.5 ft/sec. For AP1000 DCD and PRA analysis, this penalty was applied for the entire transient, regardless of the velocity in the PRHR tubes. The following provides our justification for this penalty.

To evaluate the effect of the heat transfer correlation in NOTRUMP, a simple heat transfer model of the PRHR heat exchanger tube was constructed. This model was used to compare the Thom correlation which is used in NOTRUMP to the modified Rosenhow correlation which was developed from the AP600 PRHR component tests. These evaluations are shown in Attachment A to RAI 440.107. The results show that the Thom correlation slightly overpredicts the heat transfer relative to the modified Rosenhow correlation (~6% to 8%) depending on primary fluid inlet conditions. Reducing the heat transfer area by 50% and using the Thom correlation results in a reduction in the heat transfer relative to the modified Rosenhow correlation of 11% to 13% for the same conditions.

Therefore, as can be seen, the penalty on heat transfer for the PRHR as applied to the AP1000 small break LOCA analysis Cases No. 1 and No. 2 in PRA section A.5.1 is conservative. The Westinghouse response to RAI 440.107 contains additional information on the NOTRUMP PRHR HX model.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Figure 720.009-1
MAAP4 vs. NOTRUMP PRHR Heat Removal (AP1000)

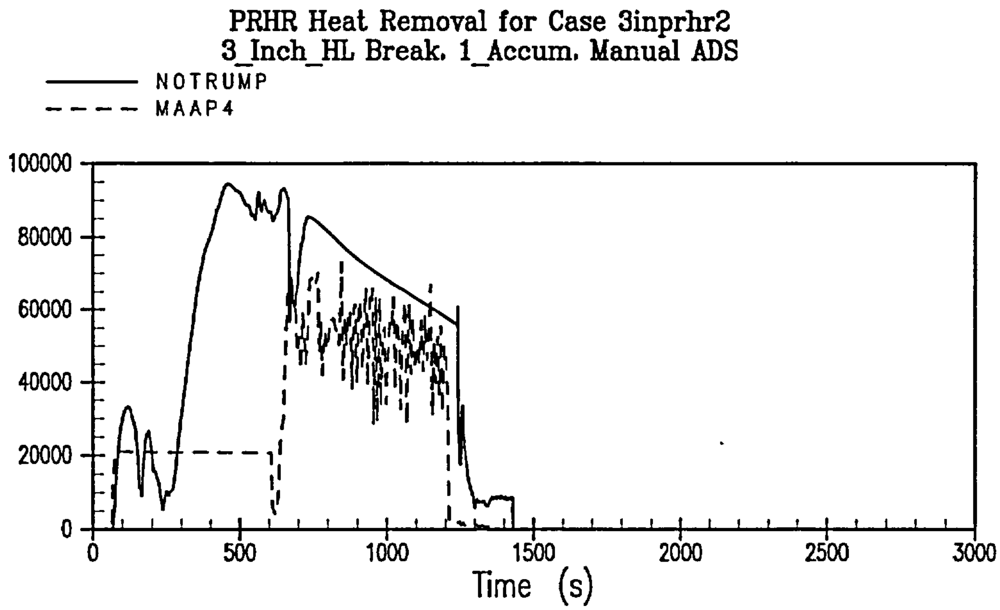


Figure 720.009-2
MAAP4 vs. NOTRUMP RCS Pressure with PRHR (AP1000)

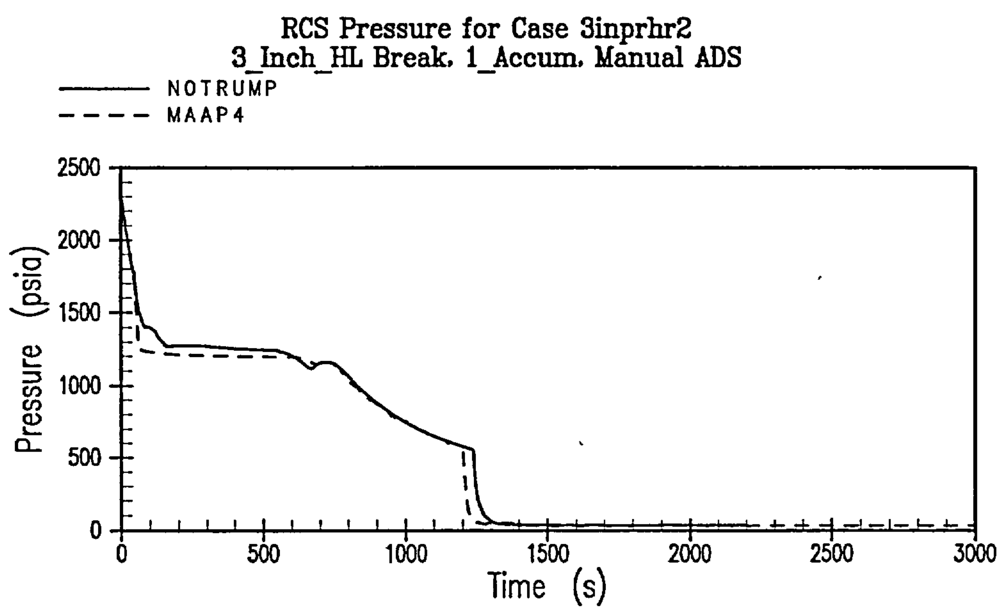


Figure 720.009-3
MAAP4 vs. NOTRUMP Accumulator Water Mass with PRHR (AP1000)

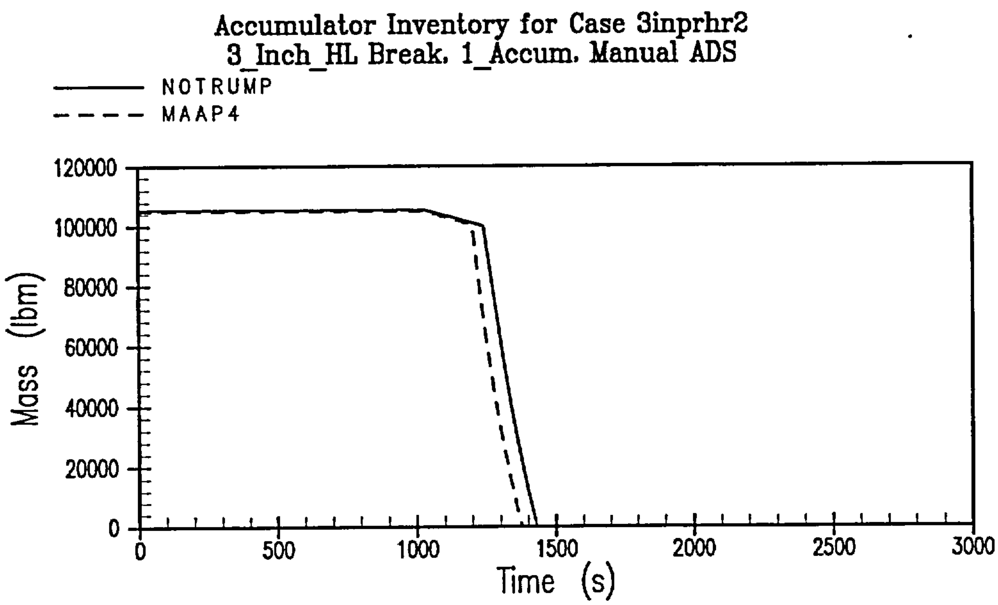
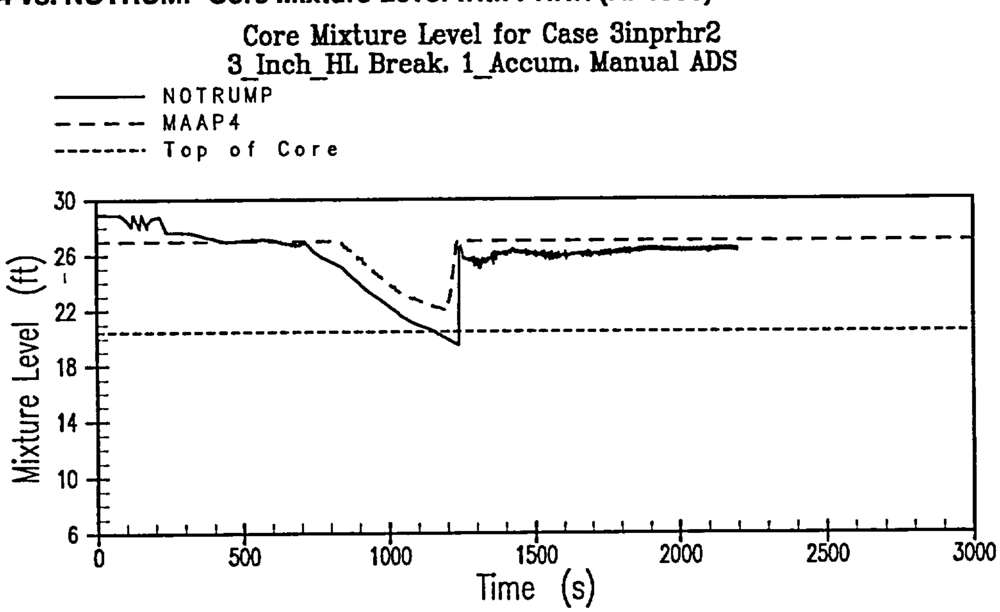


Figure 720.009-4
MAAP4 vs. NOTRUMP Core Mixture Level with PRHR (AP1000)



RAI Number: 720.010

Question:

Section A3.4 (also Section A5.1, Large-Break LOCA Case No. 1) discusses the AP1000 success criteria of one accumulator operating for the events of spurious opening of all four ADS4 valves. It indicates that the PCT for this equivalent of a large-break LOCA event caused by the spurious opening of all ADS4 valves with only one accumulator available is less severe than a double ended cold break LOCA with both accumulators available. It provides a hand calculation estimate of the PCT of 1739°F and an uncertainty of 251°F for a total of 1990°F.

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

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

Westinghouse Response:

A COBRA-TRAC analysis has been performed of the spurious opening of all four ADS stage 4 valves with only one accumulator available. The scenario discussed in Section A3.4 has been analyzed on a best estimate basis using the WCOBRA/TRAC computer code and the AP1000 input from the DCD large break LOCA analysis. The peak cladding temperature (PCT) calculated by WCOBRA/TRAC is 833F. This result is less limiting than the corresponding DECLG break reference case result with both accumulators available that is presented in the RAI 440.097 response and is the base case of the AP1000 DCD Section 15.6 95th percentile PCT determination.

Figure A5.2.2-1 (see PRA change section) presents the PCT transient for the hot rod of the AP1000 core. Because the flow to the break location is in the normal operation flow direction upward through the fuel, there is no flow reversal immediately following the break to cause DNB to occur in the core. As a result, and due to the strong positive liquid flow through the core, there is no blowdown cladding heatup. Figure A5.2.2-2 shows the continuous liquid phase flow rates at the top (Dashed line) and bottom (Solid line) of the core until the cladding heatup has begun. Core pressure (Figure A5.2.2-3) is reduced to about 300 psia at the time the cladding temperature excursion begins.

Depletion of the reactor vessel mass inventory due to the flow through the open ADS-4 valves eventually leads to cladding heatup due to the lack of liquid flow through the core. Figures A5.2.2-4 and -5 present core and downcomer liquid levels, respectively, and show the loss in mass inventory that occurs through the time that the cladding temperature excursion begins as well as the subsequent increase in mass. The diminished liquid available leads to the low liquid flow rates through the core observed in Figure A5.2.2-2. Flow from the one accumulator that is assumed to be operable (Figure A5.2.2-6) is what causes the level increase observed in Figures A5.2.2-4 and -5. The accumulator initial conditions of water level, gas pressure, and discharge line resistance used in this calculation are the conservative values used in the AP1000 DCD Chapter 15 large break LOCA analysis.

The addition of an appropriate PCT uncertainty to the WCOBRA/TRAC best estimate PCT result will conservatively address thermal/ hydraulic analysis uncertainties. The AP1000 DCD Section 15.6 95th percentile PCT value of 2124F is 228F higher than the WCOBRA/TRAC reference case result. The addition of 228F to the current result should bound the thermal/ hydraulic uncertainty associated with this scenario. This is true because the most important component in the DCD analysis PCT uncertainty adder is the reflood phase PCT increase associated with the uncertainty of the code itself, as established during the licensing of the large break LOCA best estimate methodology. Because the code uncertainty term dominates, the PCT adder is not a strong function of the AP1000 DECLG calculated transient behavior. Moreover, the PCT sensitivity to variabilities in thermal/ hydraulics at the low calculated PCT of the spurious ADS-4 actuation case is judged to be lower in magnitude than the sensitivity that applies at the much higher cladding temperature level of the DCD large break LOCA analysis. Therefore, for the spurious ADS-4 actuation case the PCT with uncertainties considered can be conservatively equated to the PCT as calculated by WCOBRA/TRAC plus 228F, or 833 + 228 = 1061F.

The figures showing the resulting plant performance are attached to this RAI as part of the changes in the PRA.

Design Control Document (DCD) Revision:

None

PRA Revision:

Revisions to PRA Section A.3.4

A3.4 Large LOCA Success Criteria

There are two large break LOCA event trees used in the AP1000 PRA. One includes breaks of the hot leg (HL) or cold leg (CL) pipes, up to and including the double ended rupture of the main loop lines. The other includes the spurious opening of the ADS valves, up to and including opening of all 4 ADS stage 4 valves at the same time.

Large Break LOCA (LLOCA)

This LOCA is defined as a break sufficiently large such that injection from both accumulators is required. Operation of ADS valves is not required in order to depressurize the RCS to the RNS injection pressure. Operation of ADS valves is required in order to depressurize the RCS to allow gravity injection from the IRWST and containment recirculation. The corresponding break size is a break with an equivalent inside diameter of approximately 9 inches or larger.
The DCD Chapter 15 analysis covers this event since it also assumes operation of both accumulators. As a result, special PRA success criteria analysis is not required.

Spurious ADS LOCA (SPADS)

The opening of all four ADS stage 4 paths bounds this LOCA. Although this LOCA size is within the LLOCA size, the AP1000 success criteria is one accumulator because of the less severe plant response to HL LOCAs as compared with large CL LOCAs. Otherwise the mitigating system requirements are the same as for a LLOCA.

~~Note that for AP1000, this spurious ADS event is a new large LOCA PRA initiating event.~~ The upper bound of this event is the spurious opening of all 4 ADS-4 valves at the same time. Since the AP1000 PRA success criteria for this event is 1 accumulator, the design basis DCD analysis does not bound the PRA case. **The analysis of this accident shown in section A.5 provides a conservative evaluation of the response of AP1000 to this accident. It shows that the PCT is less than 1100 F even with uncertainties added.**

~~This LOCA is less severe because the HL break location does not result in the core experiencing an initial core flow stagnation/reversal. This flow stagnation/reversal causes a significant PCT increase, approximately 800°F. After the flow stagnation occurs, the flow reverses and reduces the PCT by 400°-500°F. As a result, the core temperatures at the end of blowdown will be significantly lower for a HL large LOCA, on the order of 400°-500°F instead of the 1100°-1200°F seen in a large CL LOCA. The following table shows a hand calculation that approximates the core reflood PCT rise. It assumes a relatively long core reflood time of 120 seconds that is consistent with operation of one accumulator. Note that the AP600 case is the T/H uncertainty analysis (Reference A-4).~~

	~~AP600~~	~~AP1000~~
~~Break Location~~	~~CL~~	~~HL~~
~~Break size~~	~~DECL~~	~~4 x ADS-4 valves~~
~~PCT at end of blowdown (°F)~~	~~1000~~	~~500~~
~~Number accumulator injecting~~	~~1~~	~~1~~
~~Core heatup time (sec)~~	~~106~~	~~120~~
~~Core linear power (kw/ft)~~	~~4.100~~	~~5.707~~
~~PCT increase (°F)~~	~~786~~	~~1239~~
~~PCT without uncertainty~~	~~1786~~	~~1739~~

~~(°F)~~		
~~PCT uncertainty (°F)~~	~~244~~	~~251~~
~~PCT with uncertainty (°F)~~	~~2030~~	~~1990~~

~~The estimate of the AP1000 is made by taking the ratio of the AP600 PCT increase by the increase in linear power and the increase in core heatup time. This should provide a reasonable approximation of the AP1000 PCT. Since there is more than 200°F margin to the 2200°F limit even with uncertainty, there is confidence that one accumulator will support successful core cooling for this spurious ADS event. As a result, the analysis of a spurious ADS-4 accident with one accumulator is not evaluated further.~~

Revisions to PRA section A.5.1

Large LOCA Case #1

This case was selected for AP600 because large LOCAs all have significant core uncovery and the design basis DCD analysis (Reference A-1) had injection from both accumulators. In the PRA, injection from only one accumulator is success. Use of one accumulator results in a longer core uncovery and higher PCT. The AP1000 PRA success criteria requires 2 out of 2 accumulators. As a result, the design basis DCD large LOCA analysis bounds the PRA success criteria.

Note that for AP1000, a new large LOCA PRA initiating event has been defined. This event is a spurious ADS event. As discussed in Section A3.4. the upper bound for this event is the spurious opening of all 4 ADS-4 valves at the same time. The AP1000 success criterion for this event is 1 accumulator. As a result, the design basis DCD analysis does not bound the PRA case. Since the ADS-4 valves are connected to the RCS HLs, this event is less severe than the same size break on the CL because the core does not experience the initial core flow stagnation/reversal.

~~As a result, the peak cladding temperatures at the end of blowdown will be significantly lower, on the order of 400°-500°F instead of 1100°-1200°F. Assuming a relatively long core reflood time (consistent with operation of one accumulator) of 120 seconds, the core PCT is estimated (Section A3.4) to be about 1740°F (without uncertainty). Even after adding uncertainty, the PCT is less than 2000°F.~~

A spurious opening of ADS stage 4 is analyzed for AP1000 with the same mitigating equipment.

Large LOCA Case #2

Same as large LOCA case #1 **except a split break instead of a double ended break. The ADS stage 4 valves do not have the same uncertainty with respect to break type, orientation.**

Large LOCA Case #3

Same as large LOCA case #1 except assumes failure of containment isolation. This case is more limiting than case #1, however since case #1 has a very large margin its PCT of 1060F and the safety limit of 2200F it is not necessary to analyze this case.

Addition of new PRA section A.5.2.2

A.5.2.2 WCOBRA/TRAC Analysis of Large-Break LOCA

Westinghouse applies the WCOBRA/TRAC computer code to perform AP1000 best-estimate large-break LOCA analyses in compliance with 10 CFR 50 (in the DCD). The methodology used for the AP1000 analysis is documented in References A-22 and A-23.

The acceptability of WCOBRA/TRAC computer code and methodology approved for AP600 Large-Break LOCA analyses for the AP1000 application is documented in Reference A-24.

A simplification of this methodology was approved for the AP600 in Reference A-25. The parameters important to the initial conditions and power distribution uncertainty components are set to bounding values established by sensitivity studies. The model uncertainty component is quantified in the same way as for three and four-loop plants, with the other parameters set to those bounding values. The code uncertainty estimate based on direct comparisons with data, the uncertainty in the experimental data itself, is also considered in the overall uncertainty estimate. A discussion of the large-break LOCA uncertainty methodology is given in Reference A-23

A.5.2.2.1 WCOBRA-TRAC Results for Spurious Stage 4 ADS Large LOCA

The scenario discussed in Section A5.1 has been analyzed on a best estimate basis using the WCOBRA/TRAC computer code and the AP1000 input from the DCD large break LOCA analysis. The peak cladding temperature (PCT) calculated by WCOBRA/TRAC is 833F. This result is less limiting than the corresponding DECLG break reference case result with both accumulators available that is presented in the AP1000 DCD and is the base case of the AP1000 DCD Section 15.6 95th percentile PCT determination.

Figure A5.2.2-1 presents the PCT transient for the hot rod of the AP1000 core. Because the flow to the break location is in the normal operation flow direction upward through the fuel, there is no flow reversal immediately following the break to cause DNB to occur in the core. As a result, and due to the strong positive liquid flow through the core, there is no blowdown cladding heatup. Figure A5.2.2-2 shows the continuous liquid phase flow rates at the top (Dashed line) and bottom (Solid line) of the core until the cladding heatup has begun. Core pressure (Figure A5.2.2-3) is reduced to about 300 psia at the time the cladding temperature excursion begins.

Depletion of the reactor vessel mass inventory due to the flow through the open ADS-4 valves eventually leads to cladding heatup due to the lack of liquid flow through the core. Figures A5.2.2-4 and A5.2.2-5 present core and downcomer liquid levels, respectively, and show the loss in mass inventory that occurs through the time that the cladding temperature excursion begins as well as the subsequent increase in mass. The diminished liquid available leads to the low liquid flow rates through the core observed in Figure A5.2.2-2. Flow from the one accumulator that is assumed to be operable (Figure A5.2.2-6) is what causes the level increase observed in Figures A5.2.2-4 and A5.2.2-5. The accumulator initial conditions of water level, gas pressure, and discharge line resistance used in this calculation are the conservative values used in the AP1000 DCD Chapter 15 large break LOCA analysis.

The addition of an appropriate PCT uncertainty to the WCOBRA/TRAC best estimate PCT result will conservatively address thermal/ hydraulic analysis uncertainties. The AP1000 DCD Section 15.6 95th percentile PCT value of 2124F is 228F higher than the WCOBRA/TRAC reference case result from the DCD. The addition of 228F to the current result should bound the thermal/ hydraulic uncertainty associated with this scenario. This is true because the most important component in the DCD analysis PCT uncertainty adder is the reflood phase PCT increase associated with the uncertainty of the code itself, as established during the licensing of the large break LOCA best estimate methodology (Reference A-22). Because the code uncertainty term dominates, the PCT adder is not a strong function of the AP1000 DECLG calculated transient behavior. Moreover, the PCT sensitivity to variabilities in thermal/ hydraulics at the low calculated PCT of the spurious ADS-4 actuation case is judged to be lower in magnitude than the sensitivity that applies at the much higher cladding temperature level of the DCD large break LOCA analysis. Therefore, for the spurious ADS-4 actuation case the PCT with uncertainties considered can be conservatively equated to the PCT as calculated by WCOBRA/TRAC plus 228F, or 833 + 228 = 1061F.

Revisions to PRA Section A6

A6 References

A-1 AP600 Standard Safety Analysis Report

A-2 AP600 Probabilistic Risk Assessment

A-3 MAAP4/NOTRUMP Benchmarking to Support the Use of MAAP4 for AP600 PRA Success Criteria Analysis, WCAP-14869, April 1997

A-4 AP600 PRA Thermal/Hydraulic Uncertainty Evaluation for Passive System Reliability, WCAP-14800, June 1997

A-5 AP600 Adverse Systems Interaction Report, WCAP-14477

A-6 AP600 Shutdown Report, WCAP-14837

A-7 "Evaluation of the AP600 Conformance to Inter-system LOCA Acceptance Criteria," WCAP-14425, July 1995

A-8 "Operational Assessment for AP1000," WCAP-15800

A-9 AP600 Human Factors Engineering Operational Experience Review Report, WCAP-14645

A-10 AP600 Test and Analysis Plan for Design Certification, WCAP-14141

A-11 AP600 Emergency Response Guidelines

A-12 AP600 Emergency Response Guidelines Background Information

A-13 AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, WCAP-13856

A-14 AP600 Passive System Reliability Roadmap, NSD-NRC-96-4996, 8/9/96

A-15 NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," September 1998

A-16 LOCTA-IV Program, Loss of Coolant Transient Analysis, WCAP-8301, June 1974 (Westinghouse Proprietary)

A-17 Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, WCAP-10054-P-A, August 1985 (Westinghouse Proprietary)

A-18 WCOBRA-TRAC OSU Long Term Cooling Final Validation Report, WCAP-14776, 11/96 (Westinghouse Proprietary)

A-19 MAAP4 Modular Accident Analysis Program, User's Manual, Rev. 0, May 1994

A-20 AP600 ATWS Analysis, SAE-APS-98-11, 1/22/98

A-21 "AP1000 PIRT and Scaling Assessment," WCAP-15613 (Proprietary) and WCAP-15706 (Non-Proprietary), March 2001

A-22 "Code Qualification Document for Best Estimate LOCA Analysis" WCAP-12945 (Westinghouse Proprietary), March 1998

A-23 "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-Of-Coolant Accident" WCAP-14171 (Westinghouse Proprietary), March 1998

A-24 "AP1000 Code Applicability Report" WCAP-15644 (Westinghouse Proprietary) and WCAP-15707 (Non-Proprietary), May 2001

A-25 "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design" NUREG-1512, September 1998.

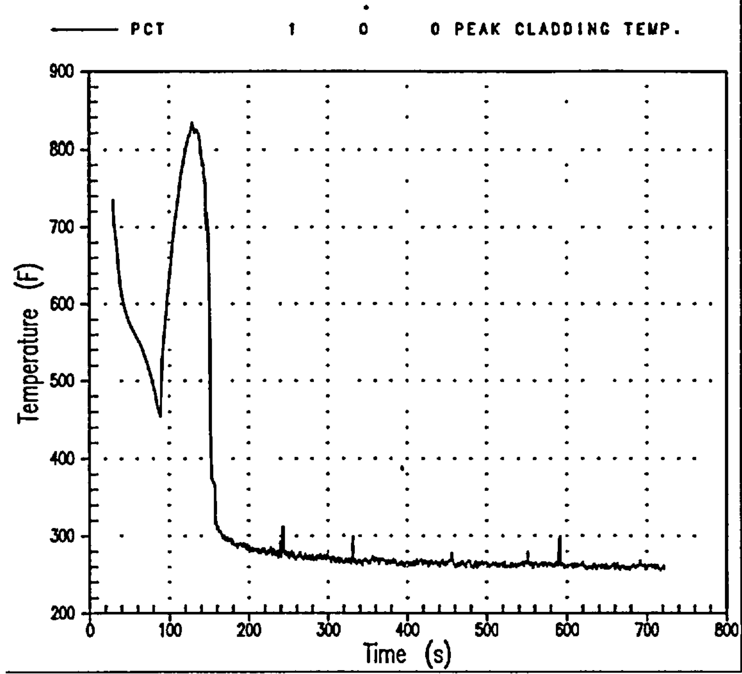


Figure A5.2.2-1 Peak Cladding Temperature

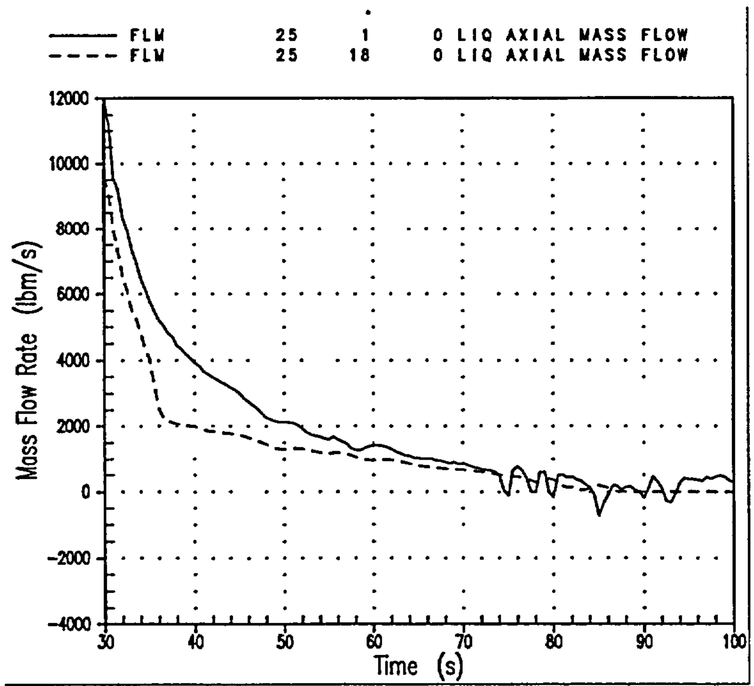


Figure A5.2.2-2 Core Liquid Entry / Exit Flow Rates

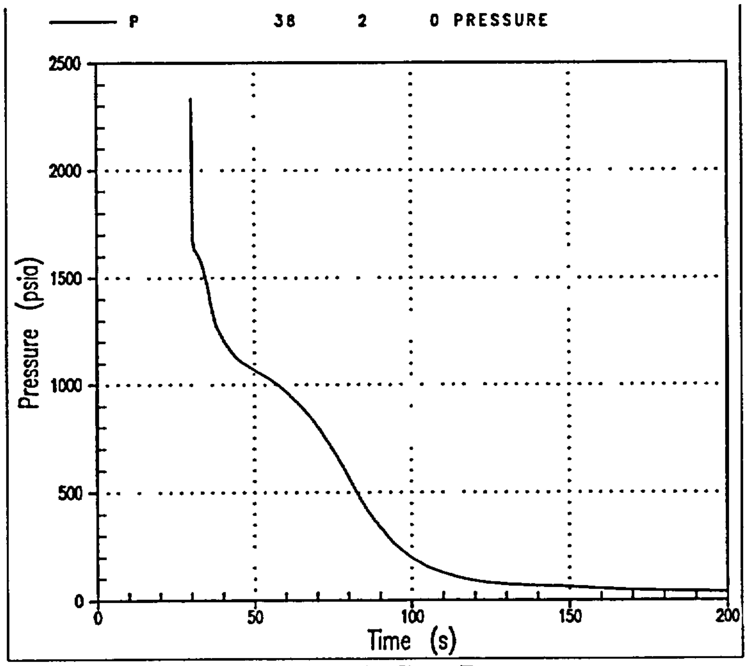


Figure A5.2.2-3 Core Pressure

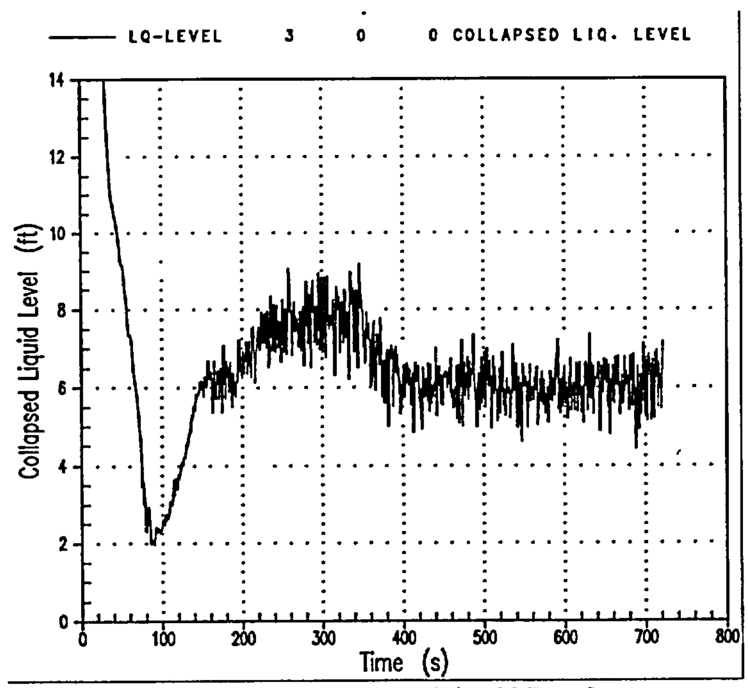


Figure A5.2.2-4 Core Liquid Level

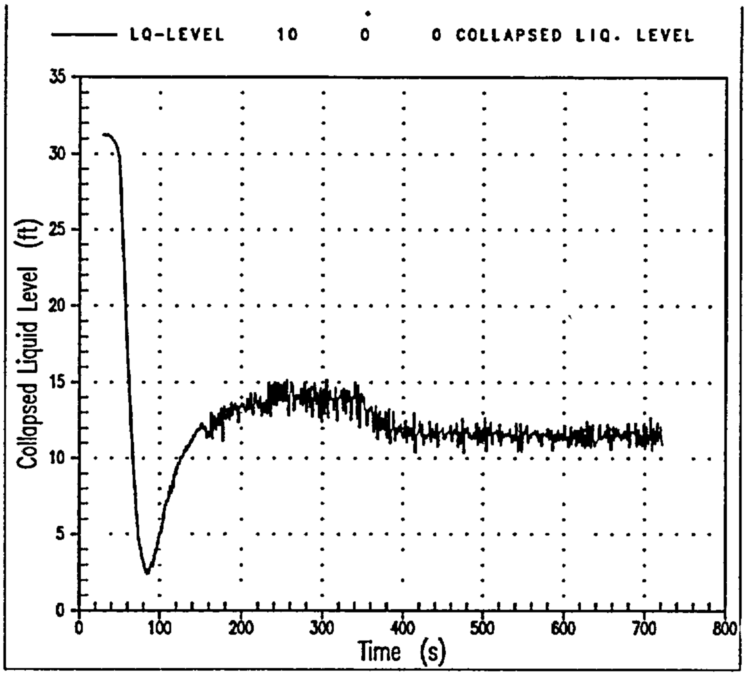


Figure A5.2.2-5 Downcomer Liquid Level

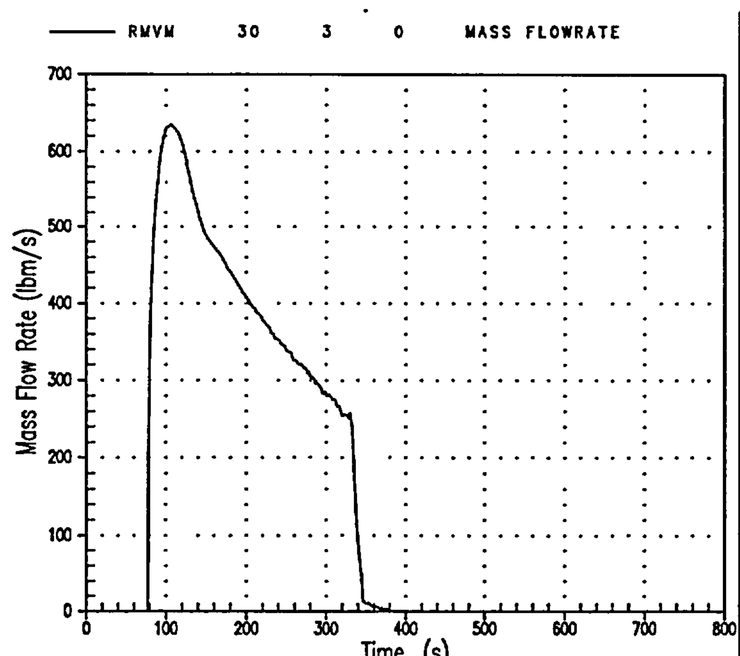


Figure A5.2.2-6 Accumulator Injection Flow Rate

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.022

Question:

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

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

Westinghouse Response:

Section A5 does not address success criteria; it addresses T&H uncertainty. As agreed with the NRC staff and discussed in the AP1000 PRA section A5, Westinghouse bounds the T&H uncertainty of the AP600/AP1000 by analyzing the high risk / low margin sequences using DCD analysis codes and methods. The cases analyzed for T&H uncertainty include just enough failures to make them low margin in order to maximize the risk. Including all of the failures from the success criteria would reduce the probability of the event to the point where it is not risk important.

The statement in PRA section 6.3.4 is correct and is consistent with Appendix A. The success criteria analysis performed in Appendix A (section A.3) assumes failure of the containment isolation. The T&H uncertainty analysis performed in section A.5 assumes failure of containment isolation when it is risk important as discussed in the first paragraph of this response.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.026

Question:

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

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

B. Are the sensitivity analyses performed for AP1000 design?

Westinghouse Response:

The impact of the Chemical and Volume Control System on the plant operation and the definition of the success criteria is the same for AP1000 as it was for AP600. The additional water from the CVS would have a beneficial impact on core cooling in a similar fashion as additional water from other sources such as accumulators or CMTs. As an example, the attached figures show the system response from MAAP4 analyses of an AP1000 2 inch hot leg break with 1 CMT (as was shown in Figures A.3.2-8 to A.3.2-13 of the AP1000 PRA) and with 2 CMTs. The additional water addition from a 2nd CMT slightly delays the system transient, and results in less core uncovery. Appendix A does not describe sensitivity analysis and its reference will be removed from PRA section 6.3.3.1.

Design Control Document (DCD) Revision:

None

PRA Revision:

Revision to section 6.3.3.1:

Impact of CVS operation on depressurization, gravity injection, and PRHR operation.

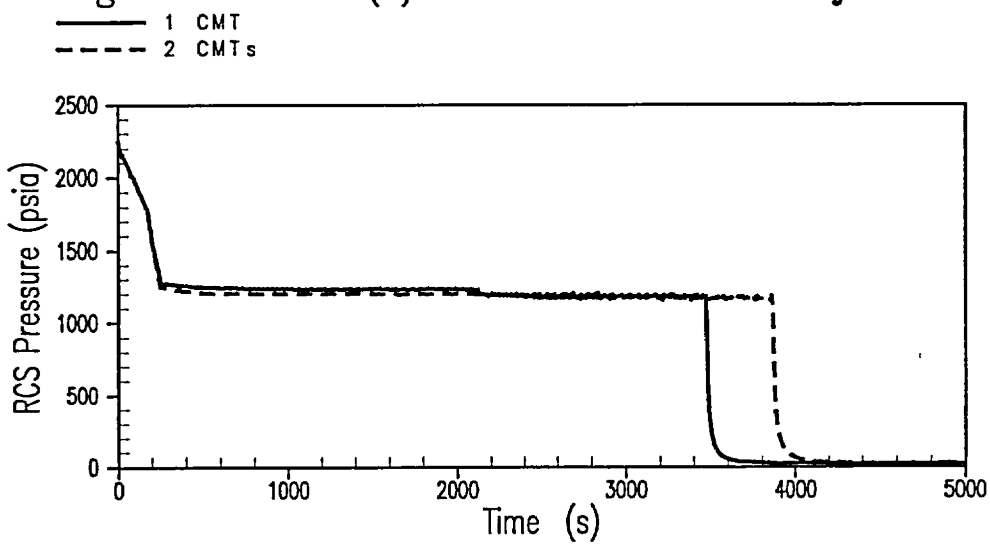
The status of the chemical and volume control system is not evaluated for events other than steam generator tube rupture, very small LOCA/RCS leak, and steamline breaks. Continuous or intermittent operation of the CVS is not expected to adversely affect depressurization or gravity injection because CVS injection flow is much smaller than that from the CMTs or accumulators.

In addition, CVS operation can only have a secondary impact on RCS pressure during depressurization because opening of the ADS valves provides large vent holes.
The PRA success criteria analysis sensitivity to numbers of CMTs and accumulators ~~(Appendix A)~~ bounds the impact of possible CVS operation.

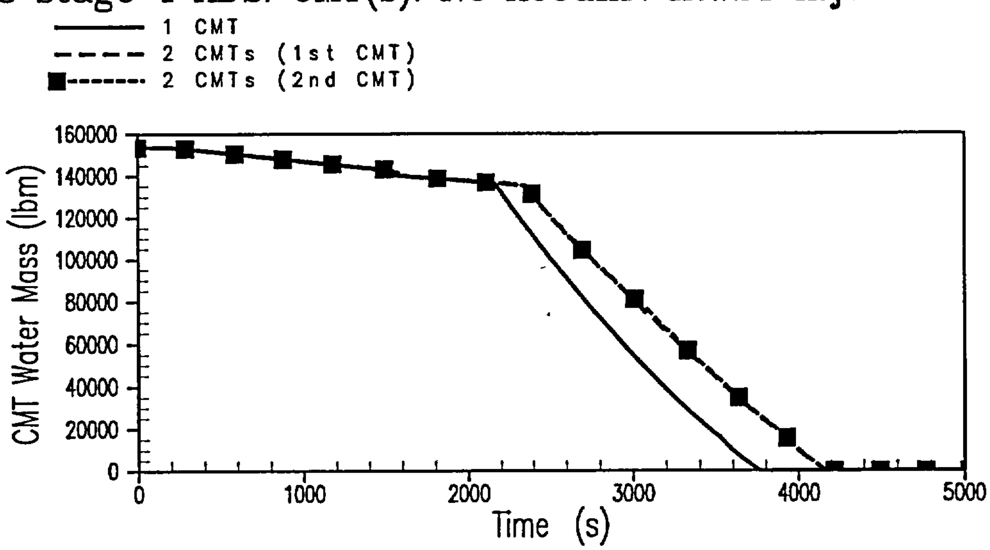
For steam generator tube rupture, CVS operation is modeled for auxiliary pressurizer spray, but normal CVS operation will not adversely affect event mitigation. Further, for SGTR, ADS operation is not required if CVS operation for sprays is successful, so there are no interactions of concern. Cooldown is accomplished via cooling from the intact steam generator or with PRHR operation.

Potential impacts of CVS operation on PRHR operation were investigated as part of the AP600 testing program. Those tests provide data to confirm analytical methods that support successful core cooling with CVS operation.

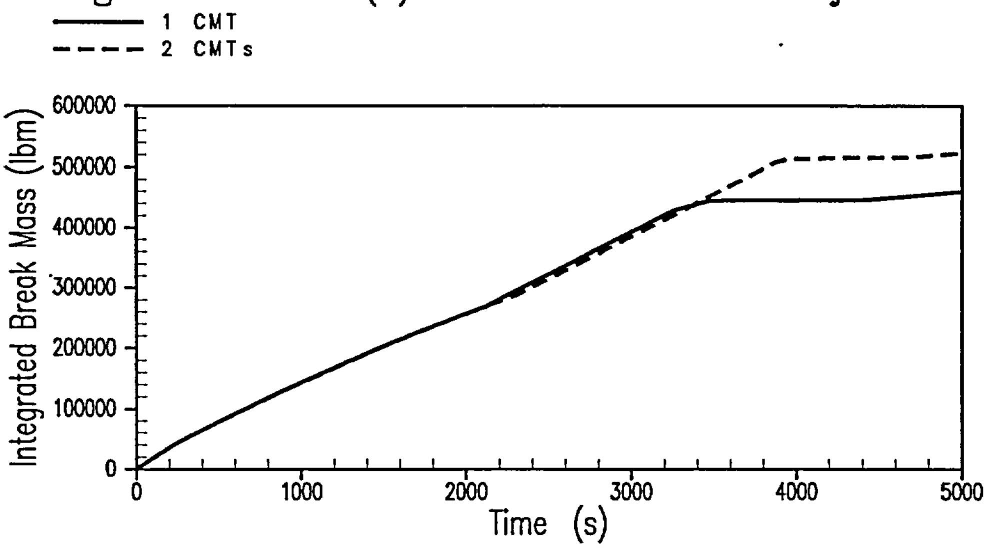
AP1000 MAAP4 Analysis of 2.0 Inch Hot Leg Break
3 stage 4 ADS, CMT(s), No Accum., IRWST Injection



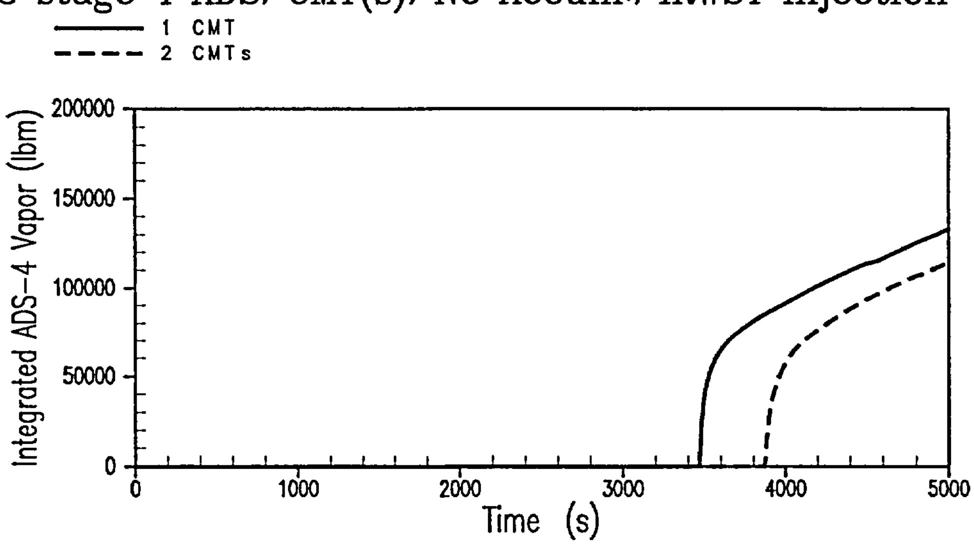
AP1000 MAAP4 Analysis of 2.0 Inch Hot Leg Break
3 stage 4 ADS, CMT(s), No Accum., IRWST Injection

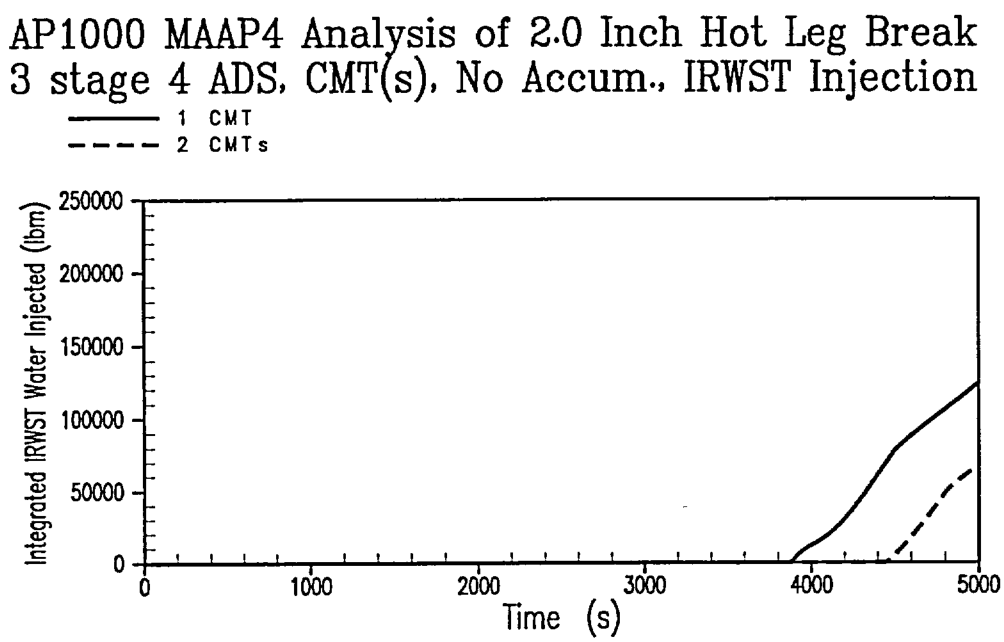


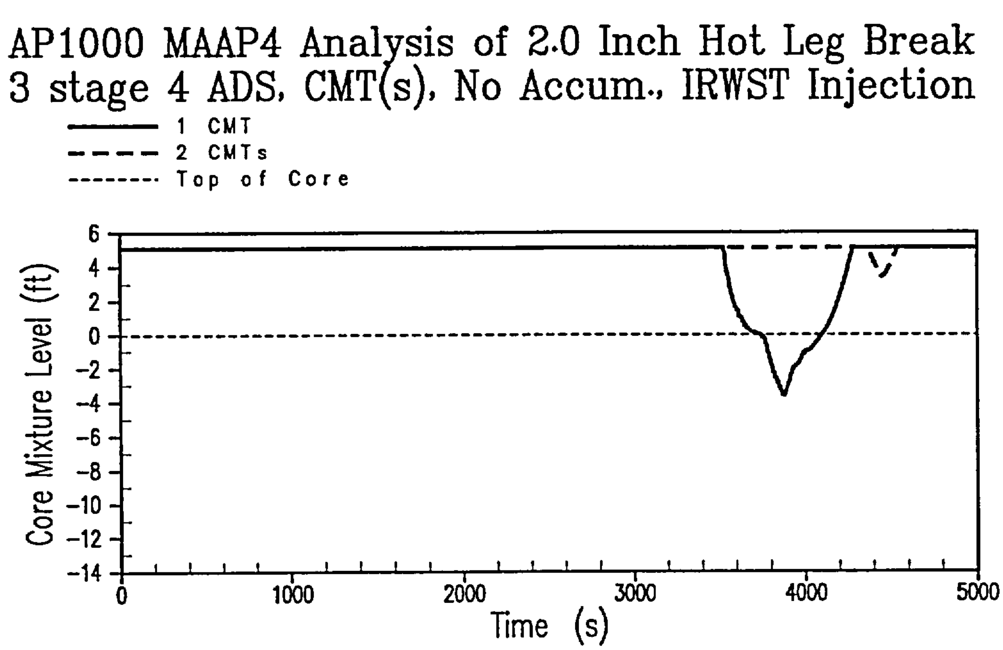
AP1000 MAAP4 Analysis of 2.0 Inch Hot Leg Break
3 stage 4 ADS, CMT(s), No Accum., IRWST Injection



AP1000 MAAP4 Analysis of 2.0 Inch Hot Leg Break
3 stage 4 ADS, CMT(s), No Accum., IRWST Injection




AP1000 MAAP4 Analysis of 2.0 Inch Hot Leg Break
3 stage 4 ADS, CMT(s), No Accum., IRWST Injection
1 CMT
2 CMTs
Integrated IRWST Water Injected (lbm)
250000
200000
150000
100000
50000
0
0
1000
2000
3000
4000
5000
Time (s)


AP1000 MAAP4 Analysis of 2.0 Inch Hot Leg Break
3 stage 4 ADS, CMT(s), No Accum., IRWST Injection
1 CMT
2 CMTs
Top of Core
Core Mixture Level (ft)
6
4
2
0
-2
-4
-6
-8
-10
-12
-14
0
1000
2000
3000
4000
5000
Time (s)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.036

Question:

In Appendix A (page A-29), several accident sequences, which were identified in the AP600 PRA as "low T/H margin, risk important" scenarios, are discussed with respect to their potential applicability to AP1000 design. While the first three cases discussed small LOCAs (SLOCAs), Appendix A states that "The PRHR HX is included for the AP1000 because the success criteria has been changed to require PRHR HX operation for MLOCAs with failure of CMTs." Please clarify and revise accordingly.

Westinghouse Response:

Appendix A page A-27 says, "The goal of the T/H uncertainty evaluation process was to demonstrate that the sets of equipment that have been credited as providing successful core cooling in the PRA (i.e., success criteria) are indeed successful, even with the consideration of T/H uncertainty." Therefore, by definition, the T&H uncertainty analysis only considers sets of equipment that are included in the success criteria of the AP1000.

For the AP1000, the success criteria was modified in a few cases to require additional equipment to be available. One of those additions was requiring the PRHR HX to be available for medium LOCAs when the CMTs were unavailable due to multiple failures. During such a sequence, there is no RCS makeup until the operators manually actuate ADS and the RCS pressure decreases to the accumulator injection pressure. For AP600, there was sufficient inventory in the RCS to limit the core uncovery during the 20 minutes allowed for operator action in the success criteria. For AP1000 there is less time available because of the higher core power compared to the RCS water volume. However with the operation of the PRHR HX, the RCS pressure is reduced and the loss of reactor coolant is reduced. The availability of the PRHR HX provides more than 20 minutes for operator action in the AP1000. Note that the AP1000 PRA event trees require the PRHR HX for such cases and if the PRHR HX is not available the sequence is counted as a core melt.

Since the AP1000 success criteria was changed to include the PRHR HX in these medium LOCAs, it must also be included in any T&H uncertainty case that is based on a medium LOCA and failure of both CMTs. The AP1000 T&H uncertainty cases #1 and #2 are such LOCAs and as a result the PRHR HX has been included.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 720.052

Question:

The PCS was assumed to always be operable in AP600, but is now modeled in 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 PCS operates, other challenges are considered downstream. If 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.

Westinghouse Response:

The total failure frequency of the PCS water cooling is conservatively calculated in Table 1. If all PCS failure sequences were assumed to have successful spray operation that resulted in early containment failure due to de-inerted hydrogen combustion, the increase in the CFE release category frequency (7.5E-9 reactor-yr^{-1}) would be negligible. If the sprays were assumed to produce intermediate containment failure (release category CFI), the impact would be equally negligible. Therefore, any negative impact of the AP1000 non-safety sprays on the PRA is negligible.

The benefit of successful sprays is also very small and has been conservatively neglected in the PRA.

Table 1 – Total PCS Failure Frequency

Accident Class	Frequency (reactor-yr^{-1})	PCS Water Failure Probability	PCS Failure Frequency (reactor-yr^{-1})
1A	5.0E-9	3.3E-6	1.7E-14
1AP	1.5E-9	0.0E-0	0.0E-0
3A	4.4E-9	7.0E-5	3.1E-13
3BE	8.1E-8	1.7E-6	1.4E-13
3BL	2.4E-8	1.0E-6	2.4E-14
3BR	4.6E-8	1.1E-6	5.1E-14
3C	1.0E-8	1.1E-6	1.1E-14
3D	6.0E-8	1.5E-6	9.0E-14
6	9.5E-9	1.0E-6	9.5E-15
		Total PCS Failure Frequency	6.2E-13

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.073

Question:

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?

Westinghouse Response:

In the AP600 PRA report, the companion reports to DOE/ID-10541 are only referenced in Chapter 39. The AP1000 Report, Chapter 39 will be updated to reference them directly in Chapter 39, as well.

Design Control Document (DCD) Revision:

The DOE/ID-10541 companion reports,

- Theofanous, T.G., et. al., "Premixing of Steam Explosions: PM-ALPHA Verification Studies," DOE/ID-10504, September 1996.
- Theofanous, T.G., et. al., "Propagation of Steam Explosions: ESPROSE.m Verification Studies," DOE/ID-10503, August 1996.
- Theofanous, T.G., Volume 1 – "Appendices E,F, and G to DOE/ID-10541," and Volume 2 – "Addenda to DOE/ID-10541, -10503, -10504," October 1997, and Volume 3 – "Addenda to DOE/ID-10503 and –10504," December 1997.

will be included in the next revision of the AP1000 DCD Chapter 19.39 (see Attachment 1)

PRA Revision:

The DOE/ID-10541 companion reports,

- Theofanous, T.G., et. al., "Premixing of Steam Explosions: PM-ALPHA Verification Studies," DOE/ID-10504, September 1996.
- Theofanous, T.G., et. al., "Propagation of Steam Explosions: ESPROSE.m Verification Studies," DOE/ID-10503, August 1996.
- Theofanous, T.G., Volume 1 – "Appendices E,F, and G to DOE/ID-10541," and Volume 2 – "Addenda to DOE/ID-10541, -10503, -10504," October 1997, and Volume 3 – "Addenda to DOE/ID-10503 and –10504," December 1997.

will be included in the next revision of the AP1000 PRA Chapter 39 (see Attachment 2).

Based on the ROAAM results, vessel failure in the AP600 was considered to be physically unreasonable, and a probability of zero was applied to vessel failure in the AP600 PRA (Reference 19.39-3) if the following conditions of the ROAAM analysis were met:

- The reactor coolant system was depressurized.
- The reactor vessel was submerged sufficiently to wet the heated surface.
- Reactor vessel reflective insulation and containment water recirculation flow paths allowed sufficient ingress of water and venting of steam from the cavity.
- The treatment of the lower head outside surface (painting, coatings, etc.) did not interfere with water cooling of the vessel.

19.39.3 Application of In-Vessel Retention to the AP1000 Passive Plant

To establish a strong basis for crediting in-vessel retention in the AP1000, the following steps are taken:

- Establish design measures to increase the capability of the water to remove heat from the external surface of the reactor vessel (increase critical heat flux).
- Demonstrate that the thermal failure remains the limiting failure over the structural failure for the AP1000.
- Demonstrate that the AP1000 in-vessel melt progression does not change from the AP600 melt progression in such as way as to challenge the vessel integrity during relocation.
- Demonstrate that the heat load correlation, as applied from the ACOPO program (Reference 19.39-4), scales appropriately to the AP1000.
- Quantify the thermal loads using probability distributions developed specifically for the AP1000.

These items are discussed in the following sections.

19.39.4 Reactor Vessel Failure Criteria

The conclusions of the structural analyses performed for the AP600 in Reference 19.39-1 can be extrapolated to the AP1000. Thus, for the AP1000, success of in-vessel retention can be based solely on the thermal success criterion.

19.39.5 In-Vessel Melt Progression and Relocation

The analysis of the challenge to the AP600 reactor vessel integrity due to jet impingement (Reference 19.39-1) and steam explosion (References 19.39-2, **19.39-7, 19.39-8 and 19.39-9)** demonstrated very large margin-to-failure from these phenomena. The AP1000 reactor vessel lower

19.39.12 Summary

In-vessel retention of molten core debris via external reactor vessel cooling can be accomplished in the AP1000.

- The reactor vessel insulation must provide a structurally sound baffle around the lower head and lower cylinder of the vessel to channel the flow between the vessel and insulation. An insulation design that provides the proper water inlet, steam venting and flow baffling is specified for the AP1000.
- The reactor cavity must be flooded to an elevation of at least 98 ft prior to the onset of the steady–state heat flux to the vessel wall from the debris to produce the driving head required to enhance the critical heat flux on the vessel surface. The operator action to flood the cavity has been moved to the first step of the emergency operating procedures to provide adequate flooding.

19.39.13 References

19.39-1 Theofanous, T.G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

19.39-2 Theofanous, T.G., et al., "Lower Head Integrity Under In-Vessel Steam Explosion Loads," DOE/ID-10541, June 1996.

19.39-3 AP600 PRA Report, GW-GL-021, August 1998

19.39-4 Theofanous, T.G., and S. Angeli, "Natural Convection for In-Vessel Retention at Prototypic Rayleigh Numbers, " Nuclear Engineering and Design, 200, 1-9 (2000).

19.39-5 Angelini S et. al., "The Mechanism and Prediction of Critical Heat Flux in Inverted Geometries," Nuclear Engineering and Design, 200, 83-94 (2000).

19.39-6 AP600 Emergency Response Guidelines.

19.39-7 Theofanous, T. G., et al., "Premixing of Steam Explosions: PM-ALPHA Verification Studies," DOE/ID-10504, September 1996.

19.39-8 Theofanous, T. G., et al., "Propagation of Steam Explosions: ESPROSE.m Verification Studies," DOE/ID-10503, August 1996.

19.39-9 Theofanous, T. G., Volume 1 - "Appendices E, F and G to DOE/ID-10541," and Volume 2 - "Addenda to DOE/ID-10541, -10503, -10504," October 1997, and Volume 3 - "Addenda to DOE/ID-10503, -10504,' December 1997.

The yield strength of carbon steel at temperatures of 900°K is 350 MPa (Reference 39-1). At the loading of $1x10^6$ N, the minimum area that would be required to carry this load would be:

$$\text{minimum required area} = \frac{\text{load}}{\text{yield strength}} = \frac{1.1x10^6\,\text{N}}{350x10^6\,\frac{\text{N}}{\text{m}^2}} = 3x10^{-3}\,\text{m}^2 \qquad (39\text{-}1)$$

The minimum area required to carry the dead load corresponds to a thickness of 0.022 cm of the AP1000 vessel wall at the outer radius of 2.1524 m.

The wall thickness that carries the loading is the portion of the wall with a temperature from the yield strength temperature (900°K) to the saturation temperature (373°K) at the peak critical heat flux obtainable by the AP1000 configuration. The peak critical heat flux is conservatively estimated to be 2000 kW/m^2 for this calculation. The thickness can be calculated from the thermal conductivity (32 W/m-K) through the vessel wall using the standard conduction rate equation and solving for the thickness:

$$x_v = \frac{K_v(T_{yield} - T_{sat})}{q_w} = \frac{32 * (900-373)}{2000x10^3} = 0.8\text{ cm} \qquad (39\text{-}2)$$

The AP1000 vessel wall, conducting heat at the peak critical heat flux, is 36 times thicker than the minimum thickness required to carry the dead load. The AP600 vessel is 73 times thicker than the minimum thickness required to carry the dead load (Reference 39-1). Therefore, the margin to structural failure is the same order of magnitude as the AP600.

It should be noted that the actual peak heat fluxes in the AP1000 are expected to be significantly lower than the critical heat flux limit of 2000 kW/m^2. Based on the calculation presented in Figure 39-1, the peak heat flux to the vessel wall is on the order of 1400 kW/m^2, which increases this margin to 50 times the thickness required to carry the load.

The conclusions of the structural analyses performed for the AP600 in Reference 39-1 can be comfortably extrapolated to the AP1000. Thus, for the AP1000, success of IVR can be based solely on the thermal success criterion.

39.5 In-Vessel Melt Progression and Relocation

In the AP600 ROAAM analysis of in-vessel retention (Reference 39-1) and in-vessel steam explosion (References 39-2, 39-10, 39-11 and 39-12), the melt progression and relocation to the lower plenum was analyzed. One of the conclusions from the AP600 analysis was that the reactor vessel lower internals, particularly the reflector situated inside the core barrel, significantly impacted the relocation such that:

- The debris relocation to the lower plenum occurs due to melt-through of the core barrel. The quantity of the initial mass of the molten debris that mixes with the lower plenum water is dictated by the failure size and includes only a small fraction of the debris.

- The relocation pathway to the lower plenum from the fuel region downward through the support plate is blocked by metal, which melts and re-freezes around the zirc plugs, lower fuel assembly nozzles and support plate.

- The relocation pathway to the lower plenum from the region between the reflector and the core barrel downward through the support plate is blocked by metal relocated from the failure of the core reflector.

- A fraction of the total molten UO_2 is needed to fill the lower plenum to contact the lower support plate, allowing the lower support plate and reflector to melt into the debris mass.

- The large metal mass from the melting of the reflector and support plate in the lower plenum produces a thick metal layer that mitigates any focusing effect of the metal layer and prevents a large heat flux from occurring at the top of the debris pool.

The AP1000 core and lower internals geometry has been changed from the AP600 geometry as a result of the higher power output. The core is made up of 157 fuel assemblies with a 14-foot active fuel length. To accommodate the larger reactor core, the thick stainless steel reflector has been replaced by a 7/8" thick core stainless steel shroud (Figure 39-2). The thick bottom plate of the shroud is mounted flush on the support plate. There are no former plates in the annulus between the shroud and the core barrel. The core barrel is 2" thick and hangs from the upper head flange. Cooling holes through the core shroud provide cooling flow to the shroud from the core flow.

The heat sink potential of the downward molten debris relocation path from the core region to the lower head is significantly greater than that of the sideward relocation pathway. Like the AP600, the downward relocation pathway through the AP1000 lower support plate to the lower plenum is expected to plug with molten metal debris frozen above the lower nozzles of the fuel assemblies. Molten oxidic debris that may melt sideward through the core shroud has no direct path from the core bypass region to the lower plenum. The cooling holes in the shroud direct the molten oxide debris back into the core region above the elevation of the downward blockage. The debris relocation to the lower head is not expected until a significant fraction of the core has melted.

Like the AP600, the initial debris relocation to the AP1000 lower plenum is expected to occur due to the melt-through of the core barrel. Only a small fraction of the debris is expected to participate in the initial relocation to the lower plenum as it pours through the failure in the core barrel. The debris relocation in the lower head is expected to be of similar mass flow rate and superheat as the AP600.

The analysis of the challenge to the AP600 reactor vessel integrity from impingement by the molten debris jet (Reference 39-1) and steam explosion (References 39-2, 39-10, 39-11 and 39-12) demonstrated very large margin-to-failure from these phenomena. The AP1000 reactor vessel lower head has the same geometry and thickness as the AP600. Given that the mass and superheat characteristics of the initial debris relocation to the lower plenum are expected to be of the same order of magnitude as the AP600, the conclusions of the AP600

39.13 Summary

In-vessel retention of molten core debris via external reactor vessel cooling can be accomplished in the AP1000, however enhancements over the AP600 IVR strategy are required to accommodate the uprated power level.

- The reactor vessel insulation must provide a structurally-sound baffle around the lower head and lower cylinder of the vessel to channel the flow between the vessel and insulation. An insulation design that provides the proper water inlet, steam venting and flow baffling is specified for the AP1000.

- The reactor cavity must be flooded to a higher elevation prior to the onset of the steady-state heat flux to the vessel wall from the debris to produce the two-phase natural circulation flow required to enhance the critical heat flux on the vessel surface. The operator action to flood the cavity has been moved to an earlier point in the emergency operating procedures to help accomplish adequate flooding.

The fault trees and scalar values linked for nodes IR and VF are summarized in Tables 39-2 and 39-3, respectively.

39.14 References

39-1 Theofanous, T. G., et al., *In-Vessel Coolability and Retention of a Core Melt*, DOE/ID 10460, July 1995.

39-2 Theofanous, T. G., et al., *Lower Head Integrity Under In-Vessel Steam Explosion Loads*, AP600 IVE ROAAM, June 1996.

39-3 *AP600 Probabilistic Risk Assessment Report*, GW-GL-021, Westinghouse Electric Corporation, August 1998.

39-4 Theofanous, T. G. and S. Angelini, "Natural Convection for In-Vessel Retention at Prototypic Rayleigh Numbers," *Nuclear Engineering and Design*, 200, 1 – 9 (2000).

39-5 Angelini, S., et al., "The Mechanism and Prediction of Critical Heat Flux in Inverted Geometries," *Nuclear Engineering and Design*, 200, 83 – 94 (2000).

39-6 Theofanous, T. G., "In-Vessel Retention as a Severe Accident Management Strategy," Plenary Lecture, *Proceedings of the Eighth International Topical Meeting on Nuclear Reactor Thermal Hydraulics, NURETH-8*, Kyoto, Japan (1997).

39-7 Churchill, S. W., and H. S. Chu, "Correlating Equations for Laminar and Turbulent Free Convection from a Vertical Plate," *International Journal Heat Mass Transfer* 18, 1323 – 1329 (1975).

39-8 Globe, S. and D. Dropkin, "Natural Convection Heat Transfer in Liquids Confined by Two Horizontal Plates and Heated from Below," *Journal Heat Transfer* 81, 24 (1959).

39-9 Theofanous, T. G., et al., *Quantification of Limits to Coolability in ULPU-2000 Configuration IV*, CRSS-01/11, Center for Risk Studies and Safety, Santa Barbara, CA (2001).

39-10 **Theofanous, T. G., et al., "Premixing of Steam Explosions: PM-ALPHA Verification Studies," DOE/ID-10504, September 1996.**

39-11 **Theofanous, T. G., et al., "Propagation of Steam Explosions: ESPROSE.m Verification Studies," DOE/ID-10503, August 1996.**

39-12 **Theofanous, T. G., Volume 1 - "Appendices E, F and G to DOE/ID-10541," and Volume 2 - "Addenda to DOE/ID-10541, -10503, -10504," October 1997, and Volume 3 - "Addenda to DOE/ID-10503, -10504,' December 1997.**