RAI Number: 440.058

Question:

Table 15.0-4a indicates that the setpoints assumed in the analysis for the safeguards ("S") signal and steamline isolation on low steamline pressure are 405 psia and 535 psia for the cases with and without an adverse environment assumed, respectively.

A. Identify specific transients that rely on these low pressure signals (at both conditions with and without an assumed adverse environment) for consequence mitigation and specify the assumed conditions (such as temperature, pressure, moisture and radiation levels) for which the low pressure signal actuation setpoints are applicable.

B. Provide a discussion of the results of transient analysis and demonstrate that the analytical results are within the applicable range of the low pressure actuation signal setpoints.

C. Provide a TS that correctly reflects the setpoints of these low steamline pressure actuation signals with inclusion of the total allowance for measurement uncertainties.

Westinghouse Response:

A. Accident Protected by the Low Steam Line Pressure Signal

The table below indicates, for each transient that relies on the Low Steam Line Pressure Signal, the setpoint used in the safety analyses for Chapter 15.

EVENT	Low Steamline Pressure	
	Without Environmental Errors 535 psia	With Environmental Errors 405 psia
Inadvertent Opening of a SG Relief or Safety Valve[1]	X	
Steam System Piping Failure		X
Feedwater System Pipe Break		X
Steam Generator Tube Rupture	X	

[1] AP1000 analyses show that the steam line isolation is actuated following a low-2 Tcold "S" signal.

The values of the low steam line pressure setpoint are conservative Safety Analysis Limit (SAL) values and their adequacy has been verified by performing the analyses listed in the table above. The two values (i.e., with and without environmental errors) differ by the environmental error (see point B. below).

The definition of the low steam line pressure set-point value (i.e., with or without environmental error) to be used for each analysis, is based, at the present stage of the design, on the following guidelines:

- All the accidents, potentially resulting in a mass and energy release in the containment system or auxiliary building are analyzed assuming the setpoint with environmental errors.
- Accidents that do not cause mass and energy release or for which the M&E is directly released to the external environment (i.e., steam release through safety valves) are analyzed by assuming no environmental errors on the safety analysis setpoint.

The above approach, also based on previous evaluation for other Westinghouse PWR plants, is expected to be bounding. Once the instrumentation is selected, the above assumptions can be fully verified with an environmental condition analysis (e.g., evaluation of pressure, temperature and radiation levels) and, as discussed in Section 16.1.1, the Combined License applicant will replace the preliminary information with final plant specific values (see answer to RAI 440.103).

B. Instrumentation Range and Accident Related Considerations

The low steam line pressure signal is lead/lag compensated according to the following equation:

$$\frac{1+\tau_1 s}{1+\tau_2 s}$$

Where: τ_1 = Manually adjustable lead time constant (seconds)

τ_2 = Manually adjustable lag time constant (seconds)

s = Laplace Transform

The low steam line pressure compensated signal anticipates the actual pressure behavior so that the actual pressure value, at the time at which the setpoint is reached, is higher than the setpoint itself.

For the low steam line pressure signal, the instrumentation channel range is from 0 to 1200 psig with a channel statistical allowance of 3% (DCD Chapter 7, Table 7.3-4). Under extreme conditions, typical maximum specified allowance is 10.5% for outside containment high energy line breaks (for a typical period of 5 minutes) following: a) double-ended or small steam line rupture, b) feedline rupture or c) seismic event.

Based on the above, the environmental error is 10.5% of 1200 psig, that is 126 psig (rounded to 130 psig) that corresponds to the difference between the two setpoints used for safety analysis.

Following both the steamline and feedline breaks accidents, the low steam line pressure setpoint (405 psia including environmental errors) and the actual pressure during the accident are well inside the instrumentation range. Moreover, the low steamline pressure setpoint is expected to be reached, for both steam line and feedline breaks events, within a couple of minutes from accident initiation (i.e., less than 80 seconds for the feedline break and less than 2 seconds for the main steam line break accident) and, hence, well inside the time range typically specified for the instrumentation. While for medium and small breaks the time to the low steam line pressure signal is expected to be longer, the lower M&E releases assure milder environmental conditions. In addition, as also demonstrated in Section 15.1.4, for breaks comparable to the Inadvertent opening of a safety or PORV valves, the steam line and feedline isolation signal is generated on the Low-2 Tcold Temperature.

C. Safety Analysis Setpoints and Technical Specifications

Plant Technical Specifications (DCD Chapter 16) report the nominal trip setpoint and Allowable Value data only. Moreover, at the present time, the values specified for the trip setpoint are conservative Chapter 15 safety analysis values (SAL) or typical values reported for information only (see reviewer note in Table 3.3.2-1 and answer to RAI 440.103).

As described in DCD Section 7.1.6, Combined License applicants referencing the AP1000 certified design will provide a calculation of setpoints for protective functions consistent with the setpoint methodology described in WCAP 14606 (WCAP 14606 is an AP600 document that describes a methodology that is applicable to the AP1000). Allowable values will be calculated in accordance with the setpoint methodology and specified in the Allowable Value column (e.g., Table 3.3.2-1 Chapter 16 of the Design Control Document). The setpoint calculation will reflect the design basis and incorporate NRC accepted setpoint methodology.

The low steamline pressure actuation signal nominal setpoint, the total allowance for measurement uncertainties and the channel statistical allowances for environmental conditions will be determined in the plant specific setpoint calculation, based on the actual plant instrumentation selected by the Combined License applicant.

Following completion of a plant specific setpoint study, the values reported in Table 3.3.2-1 will be replaced by the actual trip setpoint (Nominal Trip setpoint).

Design Control Document (DCD) Revision:

None. As discussed in Section 16.1.1, the Combined License applicant will replace the preliminary information in brackets with final plant specific values (see also answer to RAI 440.103).

PRA Revision:

None.

RAI Number: 440.059

Question:

Table 15.0-6 lists plant systems and equipment that are available for transient and accident conditions. The table is incomplete and inconsistent with the information provided in the various sections of Chapter 15. For example, it does not include (1) an increase in RCS inventory due to CVS malfunction event, and (2) small line breaks outside containment event, which are analyzed and discussed in Sections 15.5.2 and 15.6.2, respectively. Table 15.0-6 indicates that for the loss of external load and turbine trip events, the reactor trips available for the consequence mitigation are the trip signals from high pressurizer pressure, overtemperature delta T, overpower delta T, and the manual trip signal. The information is inconsistent with Section 15.2.3 (page 15.2.6), which indicates that for the turbine trip event, trip signals are expected due to high pressurizer pressure, overtemperature delta T, low RCP speed, high pressurizer water level, and low SG water level. The inconsistent information related to the available trip functions, engineered safety feature (ESF) actuation functions and available ESF exists for other Table 15.0-6 events such as inadvertent opening of a steam generator safety valve, steam line break (SLB), loss of normal feedwater flow, feedwater system pipe break, and uncontrolled reactive rod control cluster assembly (RCCA) bank withdrawal from a subcritical, low power conditions, or at power conditions.

Verify the accuracy of the information provided in Table 15.0-6 and revise the table to be consistent with applicable Sections of Chapter 15.

Westinghouse Response:

Table 15.0-6 has been revised to be consistent with applicable sections of Chapter 15.

As a general consideration related to the consistency between the table and the Chapter 15 sections, it should be noted that according to the initial conditions and analysis assumptions, several trip functions and equipment may be available for mitigating the consequences of transient and accidents.

Whilst Table 15.0-6 reports the typical functions and equipment assumed available in the safety analyses, the information provided in the various sections of the Chapter 15 provide a more complete description of the accident scenario and of the mitigating features that can be actuated, according to the plant conditions, control systems availability and specific analysis assumptions.

As a typical example, consider the Turbine Trip/Loss of Load accident for which, as reported in Table 15.0-6, reactor trip is expected to occur on a high pressure signal or overtemperature/overpower ΔT. In the analysis it is assumed that following the turbine trip, with

3 seconds delay, ac power is lost resulting in RCP's coastdown. However, a conservative and unrealistic assumption, that no reactor trip occurs as consequence of the loss of ac power and reactor trip gear de-energization, is taken in the analysis. Hence, a delayed reactor trip on low reactor coolant pump speed is modeled. It is clear that, even if the reactor trip on low RCP speed is assumed in the analysis, it should not be considered as a trip signal for the turbine trip event mitigation.

Design Control Document (DCD) Revision:

Replace Table 15.0-6 with the revised Table

Table 15.0-6 (Sheet 1 of 4)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.1			
Increase in heat removal from the primary system			
~~Feedwater system malfunctions that result in an increase in feedwater flow~~	~~Power range high flux, overtemperature ΔT, overpower ΔT, manual~~	~~High-2 steam generator level produced feedwater isolation and turbine trip~~	~~Feedwater isolation valves~~
Feedwater system malfunctions that result in an increase in feedwater flow	**High-2 Steam Generator Level, Power range high flux, overtemperature ΔT, overpower ΔT, manual**	**High-2 steam generator level produced feedwater isolation and turbine trip**	**Feedwater isolation valves**
Excessive increase in secondary steam flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	–	–
~~Inadvertent opening of a steam generator safety valve~~	~~Low pressurizer pressure, manual "S"~~	~~Low pressurizer pressure, low T_{cold}, low-2 pressurizer level~~	~~Core makeup tank, feedwater isolation valves, steam line stop valves~~
Inadvertent opening of a steam generator safety valve	**Power range high flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, "S", manual**	**Low pressurizer pressure, low compensated steam line pressure, low T_{cold}, low-2 pressurizer level**	**Core makeup tank, feedwater isolation valves, steam line stop valves**
~~Steam system piping failure~~	~~"S," low pressurizer pressure, manual~~	~~Low pressurizer pressure, low compensated steam line pressure, high-1 containment pressure, low T_{cold}, manual~~	~~Core makeup tank, feedwater isolation valves, main steam line isolation valves (MSIVs), accumulators~~

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Steam system piping failure	Power range high flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, "S", manual	Low pressurizer pressure, low compensated steam line pressure, high-1 containment pressure, low T_{cold}, manual	Core makeup tank, feedwater isolation valves, main steam line isolation valves (MSIVs), accumulators
~~Inadvertent operation of the PRHR~~	~~Overpower ΔT, power range high neutron flux, low pressurizer pressure, "S," manual~~	~~Low pressurizer pressure, low~~ $\cancel{T_{cold}}$~~, low-2 pressurizer level~~	~~Core makeup tank~~
Inadvertent operation of the PRHR	Power range high flux, overtemperature ΔT, overpower ΔT, Low pressurizer pressure, "S", manual	Low pressurizer pressure, low T_{cold}, low-2 pressurizer level	Core makeup tank
Section 15.2			
Decrease in heat removal by the secondary system			
~~Loss of external load/turbine trip~~	~~High pressurizer pressure overtemperature ΔT, overpower ΔT, manual~~	–	~~Pressurizer safety valves, steam generator safety valves~~
Loss of external load/turbine trip	High pressurizer pressure , high pressurizer water level, overtemperature ΔT, overpower ΔT, Steam generator low narrow range level, low RCP speed, manual	–	Pressurizer safety valves, steam generator safety valves

Table 15.0-6 (Sheet 2 of 4)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.2 (Cont.)			
Loss of nonemergency ac power to the station auxiliaries	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low startup water flow, steam generator low wide range level	PRHR, steam generator safety valves, pressurizer safety valves
Loss of normal feedwater flow	Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, manual	Steam generator low narrow range level coincident with low startup water flow, steam generator low wide range level	PRHR, steam generator safety valves, pressurizer safety valves
~~Feedwater system pipe break~~	~~Steam generator low narrow range level, high pressurizer pressure, manual~~	~~Steam generator low wide range level, low steam line pressure, high-1 containment pressure~~	~~PRHR, core makeup tank, MSIVs, feedline isolation, pressurizer safety valves, steam generator safety valves~~
Feedwater system pipe break	**Steam generator low narrow range level, high pressurizer pressure, high pressurizer level, manual**	**Steam generator low narrow range level coincident with low startup feedwater flow, Steam generator low wide range level, low steam line pressure, high-1 containment pressure**	**PRHR, core makeup tank, MSIVs, feedline isolation, pressurizer safety valves, steam generator safety valves**
Section 15.3			
Decrease in reactor coolant system flow rate			
Partial and complete loss of forced reactor coolant flow	Low flow, underspeed, manual	–	Steam generator safety valves, pressurizer safety valves

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
~~Reactor coolant pump shaft seizure (locked rotor)~~	~~Low flow, manual, high pressurizer pressure~~	–	~~Pressurizer safety valves, steam generator safety valves~~
Reactor coolant pump shaft seizure (locked rotor)	Low flow, high pressurizer pressure, manual	–	Pressurizer safety valves, steam generator safety valves
Section 15.4			
Reactivity and power distribution anomalies			
~~Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition~~	~~Power range high flux (low setpoint), source range high flux, intermediate range high flux, manual~~	–	–
Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	Source range high neutron flux, intermediate range high neutron flux, power range high neutron flux (low setting), power range high neutron flux (high setting), high nuclear flux rate, manual	–	–

Table 15.0-6 (Sheet 3 of 4)

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.4 (Cont.)			
~~**Uncontrolled RCCA bank withdrawal at power**~~	~~**Power range high flux, overtemperature ΔT, high pressurizer pressure, manual**~~	–	~~**Pressurizer safety valves, steam generator safety valves**~~
Uncontrolled RCCA bank withdrawal at power	**power range high neutron flux, high power range positive neutron flux rate, overtemperature ΔT, overpower ΔT, high pressurizer pressure, high pressurizer water level, manual**	–	**Pressurizer safety valves, steam generator safety valves**
RCCA misalignment	Overtemperature ΔT, manual	–	–
Startup of an inactive reactor coolant pump at an incorrect temperature	Power range high flux, low flow (P-8 interlock), manual	–	–
Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant	Source range high flux, overtemperature ΔT, manual	Source range flux doubling	Low insertion limit annunciators
Spectrum of RCCA ejection accidents	Power range high flux, high positive flux rate, manual	–	Pressurizer safety valves
Section 15.5			
Increase in reactor coolant inventory			
Inadvertent operation of the ECCS during power operation	High pressurizer pressure, manual, "safeguards" trip, high pressurizer level	High pressurizer level, low T_{cold}	Core makeup tank, pressurizer safety valves, chemical and volume control system isolation, PRHR

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Chemical and volume control system malfunction that increases reactor coolant inventory	**High pressurizer pressure, "safeguards" trip, high pressurizer level, manual**	**High pressurizer level, low T_{cold}**	**Core makeup tank, pressurizer safety valves, chemical and volume control system isolation, PRHR**
Section 15.6			
Decrease in reactor coolant inventory			
Inadvertent opening of a pressurizer safety valve or ADS path	Low pressurizer pressure, overtemperature ΔT, manual	Low pressurizer pressure	Core makeup tank, ADS, accumulator

Table 15.0-6 (Sheet 4 of 4)

PLANT SYSTEMS AND EQUIPMENT
AVAILABLE FOR TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	ESF and Other Equipment
Section 15.6 (Cont.)			
Failure of small lines carrying primary coolant outside containment	------	Manual isolation of the Sample System or CVS discharge lines	Sample System isolation valves, Chemical and volume control system discharge line isolation valves
Steam generator tube rupture	Low pressurizer pressure, overtemperature ΔT, safeguards ("S"), manual	Low pressurizer pressure, high steam generator level, low steam line pressure	Core makeup tank, PRHR, steam generator safety and/or relief valves, MSIVs, radiation monitors (air removal, steamline, and steam generator blowdown), startup feedwater isolation, chemical and volume control system pump isolation, pressurizer heater isolation, steam generator power-operated relief valve isolation
LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low pressurizer pressure, safeguards ("S"), manual	High-1 containment pressure, low pressurizer pressure	Core makeup tank, accumulator, ADS, steam generator safety and/or relief valves, PRHR, in-containment water storage tank (IRWST)

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.060

Question:

List all systems or components that are considered in the transients and accidents analyses for determination of the limiting single failure events and discuss the rationale of selecting the worst single-failure event for each event listed in Table 15.0-7.

Westinghouse Response:

The most limiting single active failure is selected in accordance with the guidance provided in 10 CFR 50 Appendix A as described in DCD Section 3.1. Westinghouse considers the limiting single failure of the AP1000 safety-related equipment as identified in each analysis description provided in Chapter 15. The listing of and safety classification of the AP1000 systems and components that are considered as potential single failures in the analysis are provided in DCD Section 3.2. The consequences of their failure are described in each Chapter 15 analysis section. In some instances, because of redundancy in protection equipment, no single failure that could adversely affect the consequences of the transient is identified. The failure assumed in each analysis is listed in Table 15.0-7. The limiting single failure assumed in the AP1000 analyses are consistent with the limiting single failures selected for the AP600. This subject is discussed in DCD section 15.0.12.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.064

Question:

Section 15.1.3 presents the results of analysis for the excessive steam flow event initiated from rated load. Section 15.1.3 also indicates that two cases are analyzed: one for minimum reactivity feedback and the other for maximum reactivity feedback. Both cases are evaluated with and without automatic rod control.

A. Discuss how the event with the initial power level at rated power bounds the cases initiated from lower power conditions.

B. Provide the values of the moderator temperature and Doppler feedback coefficients assumed in the analysis for the minimum and maximum reactivity feedback, and confirm that the analytical values are bounded by the TS values.

Westinghouse Response:

A. Initial Power Level

The Excessive Load Increase Accident historically is not a limiting transient for Westinghouse PWR plants. The AP1000 is designed to accept a step load increase or decrease of 10 percent between 25 and 100 percent power without reactor trip or steam dump system actuation provided the rated power level is not exceeded. Such a step load increase is a Condition 1 event and plant control system is designed accommodate this transient.

The excessive load increase analysis is performed to show that following a 10% step load increase, even assuming that no reactor trip is actuated, DNBR remains above the DNBR limit. The event analyzed for the DCD assumes a step load increase to occur starting from 100% power. This is the most adverse condition for a 10% step load increase since, even with the plant in automatic rod control, the rated power can be significantly exceeded.

The analysis methodology for this transient assumes that no reactor trip is actuated. The plant will then reach an equilibrium condition at a higher power level. The minimum DNBR reached depends on the equilibrium power level and power overshoot.

For the plant in manual control, there is essentially no overshoot. The equilibrium power level for the minimum feedback case is a few percent above the initial value. For the maximum feedback case, the power level reaches about 110% with essentially no overshoot. Core average temperature decreases to provide the reactivity balance. Should the transient start from a lower power level, the power increase, both for minimum and maximum feedback, will be comparable to that occurring from 100% power for the two cases respectively. This means that the equilibrium power level and the core average temperature, would be lower than that evaluated in the analyses presented in the DCD.

With the plant in automatic power control, the equilibrium power level is consistent with the overall load demand (i.e., 110% of rated power). For the minimum feedback cases there is also a moderate power overshoot while for the maximum feedback case there is almost no overshoot since the moderator feedback smoothes the power transient and helps to compensate the load demand.

Also for the automatic rod control cases, should the accident start from a lower power level, the equilibrium level would be consistent with the overall load demand and hence, the resulting power level would be lower than for the cases starting from full power. Moreover, considering the larger steam generator secondary mass and thermal inertia at lower power level, a smoother transient and lower power overshoot is expected.

Based on the above, it is concluded that excessive load increase from intermediate power levels are bounded by the analysis performed starting from the plant rated power.

Finally, from the HZP conditions, the transient is bounded by the analysis reported in DCD Section 15.1.4 "Inadvertent Opening of a Steam Generator Relief or Safety Valve." This analysis assumes a steam flow of 520 lbm/sec (> 10% of nominal steam flow).

B. Core Physical Parameters

The analysis is performed assuming two different core physics sets of parameters. The two sets do not correspond to a particular time in life of the core (e.g., BOL, EOL...) but are generated by mixing core parameters to get a minimum and a maximum reactivity feedback respectively.

The values used in the analysis are reported in the following table and are consistent with the data reported in Table 15.0-2 (sheet 1 of 5) of the AP1000 DCD:

Reactivity parameter	LOFTRAN parameter	Maximum feedback	Minimum feedback
Moderator density coefficient $\Delta K/gr/cm^3$	AROCW	Most positive 0.47	Least positive 0.0
Doppler-only power Defect (0 to 100% power) ΔK	DKQ	Most negative -1.6%	Least negative -0.843%
Doppler temperature coefficient	ATF	Most negative	Least negative

Reactivity parameter	LOFTRAN parameter	Maximum feedback	Minimum feedback
pcm/°F		-3.5	-1

The values reported above are bounding values and include uncertainties. Technical Specification prescribes operation within the limits sets in the Core Operating Limits Report. The COLR limits are set to assure plant operation remains within the limiting conditions assumed in the Chapter 15 accident and transients analyses.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.065

Question:

It states on page 15.1-9 of Chapter 15 that the CMT actuation on a "S" signal is from one of the four signals including low pressurizer level signals. This statement implies that the low pressurizer level signal is an "S" signal. This is inconsistent with item 1 of TS Table 3.3.2-1, which indicates that "S" signals are from manual initiation, high containment pressure, low pressurizer pressure, low steamline pressure and low RCS cold leg temperature. According to Item 1 of TS Table 3.3.2-1, the low pressurizer level signal is not an "S" signal.

Clarify the inconsistency between the TS and the Chapter 15 analyses related to the definition of an "S" signal for the low pressurizer level signal.

Westinghouse Response:

In addition to actuation from a safeguards ("S") signal, the CMTs are actuated from two out of four low pressurizer level signals. The text in Chapter 15.1.4.1 will be modified, as indicated below, to correctly identify the signals which actuate the CMTs.

Design Control Document (DCD) Revision:

From DCD Section 15.1.4.1:

- Core makeup tank actuation ~~on a safeguards ("S") signal~~ from one of the following ~~four~~ signals:

  - Safeguards ("S") signal

    - Two out of four low pressurizer pressure signals
    - ~~Two out of four low pressurizer level signals~~
    - **Two out of four high-2 containment pressure signals**
    - Two out of four low T_{cold} signals in any one loop
    - Two out of four low steam line pressure signals in any one loop

  - **Two out of four low pressurizer level signals**

PRA Revision:

None

RAI Number: 440.068

Question:

It is stated on page 15.1.15 of Chapter 15 that for the SLB analysis, "the maximum overall fuel-to-coolant heat transfer coefficient is used to maximize the rate of cooldown."

Discuss methods for calculations of overall fuel-to-coolant heat transfer coefficient and demonstrate that the coefficient used in the analysis is a maximum value expected during an SLB event.

Westinghouse Response:

LOFTRAN calculates a parabolic fit of the overall fuel-to-coolant heat transfer (UA) using three input points. Each point provides the overall heat transfer (UA) versus fuel average temperature (TUA).

Maximum fuel UAs, based on the minimum temperatures, result in the minimum fuel stored energy and fast thermal response, whereas the minimum fuel UAs, based on maximum fuel temperatures, result in maximum fuel stored energy and a slow thermal response.

The overall fuel to coolant heat transfer coefficients provided in input to the LOFTRAN code are evaluated on the basis of the PAD 4.0 code calculation.

PAD 4.0, described in WCAP-15063-P-A Rev. 1, "Westinghouse Improved Performance analysis and Design Model (PAD 4.0)", calculates minimum and maximum expected fuel temperatures as a function of the linear power and fuel burnup (DCD Section 4.2.3.3).

The overall fuel heat transfer is expressed as follows:

$$UA_i = \frac{(Q_i)(N)(L)(C)}{TUA_i - TAVG_i} \qquad [1]$$

Where:

TUA_i = Fuel average temperature for the point i

Q_i = Rod power at TUA_i, kW/ft (see figure 1)

- ➔ For minimum UA coefficients use maximum fuel average temperature curves plus uncertainties
- ➔ For maximum UA coefficients use minimum fuel average temperature curves minus uncertainties

N = Total number of fuel rods in the core

L = Active fuel length, ft

$TAVG_i$ = RCS vessel average temperature at a core power level equal to $\frac{Q_i}{Q_{nom}}$

and Q_{nom} is the average rod power (kW/ft) at 100% power.

In particular, for the SLB event, the maximum overall fuel-to-coolant heat transfer coefficient is evaluated by using the minimum fuel average temperatures minus uncertainties, evaluated by the PAD code, in equation [1].

The heat transfer coefficients, as evaluated above, are bounding values for any time in life of the core, since they are calculated on the basis of the bounding fuel temperatures (fuel temperatures are calculated between 0 and 70,000 MWD/MTU) and evaluated for full RCS flow condition by the PAD code.

During a steam line break event, as shown in section 15.1.5, a "S" signal on low steam line pressure is reached in less than two seconds and the reactor coolant pumps trip in less than 6 seconds from the generation of the "S" signal. From this time on, due to the lower core flow rate, the heat transfer coefficient is expected to drop. This phenomenon is not simulated by the LOFTRAN code and hence the overall heat transfer coefficient is overestimated from the time of RCPs trip that occurs earlier than the time at which criticality is attained (28 seconds). The use of a constant and conservatively high heat transfer coefficient, maximize the rate of fuel cooldown and results in a faster reactivity insertion and return to criticality.

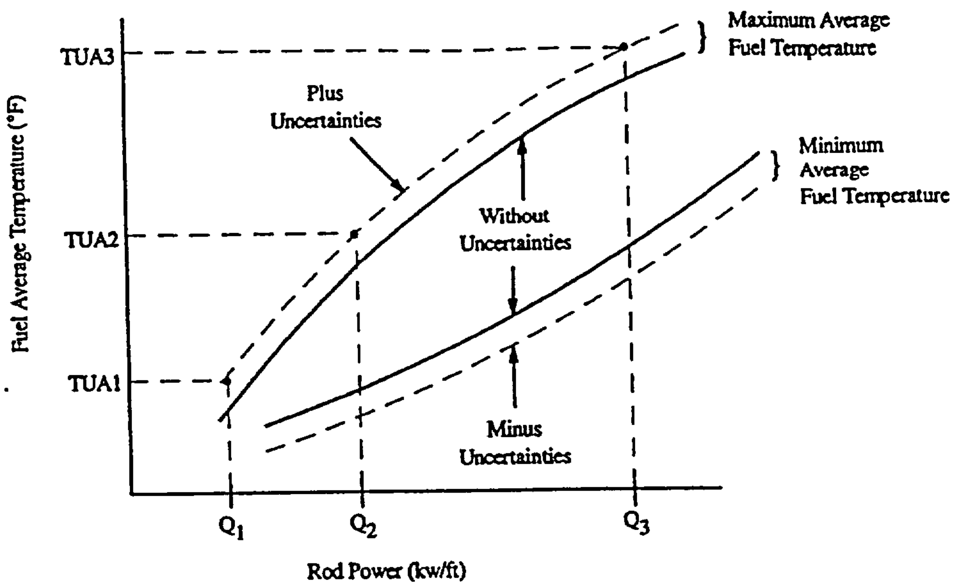


Figure 1 – Average Fuel Temperatures vs. Rod Power

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.071

Question:

It states on page 15.1-22 of Chapter 15 that for the analysis of an inadvertent operation of the PRHR heat exchanger, "a negative moderator coefficient corresponding to the end-of-life rodded core" is used.

A. Discuss the methods used for the moderator coefficient determination and address the acceptance of the methods and computer codes used.

B. Provide values of the calculated negative moderator coefficients and associated uncertainties, and address their acceptability for use in the analysis of an inadvertent operation of the PRHR heat exchanger.

Westinghouse Response:

A. The inadvertent operation of the PRHR results in a radially asymmetric power distribution. The value of the moderator coefficient to be used in the LOFTRAN code for the analysis of inadvertent operation of the PRHR heat exchanger is determined via an iterative process.

Step 1) A very conservative moderator temperature coefficient (0.47 $\Delta K/g/cm^3$), was used, in the LOFTRAN code, to evaluate the core boundary conditions (i.e., vessel inlet temperature, RCS pressure, and loops flow rates) following the inadvertent operation of the PRHR.

Step 2) Core boundary conditions from the LOFTRAN conservative run above, were provided for a 3D ANC nuclear model analysis. An assembly wise core inlet temperature distribution has been evaluated on the basis of LOFTRAN temperature and flow rates and a power search was performed in the 3D ANC nuclear model, at the end of the equilibrium cycle (21081 MWD/MTU), and assuming a core soluble boron concentration of 0 ppm.

Step 3) Since 3D ANC power evaluations were significantly lower than that evaluated by LOFTRAN at the first step, LOFTRAN moderator density reactivity coefficient was re-evaluated, via iteration runs, to provide a peak power level consistent with the 3D ANS evaluation.

Step 4) New statepoints from LOFTRAN were re-analyzed according to Step 2) method.

Step 5) Since core power, as evaluated by 3D ANC, was only slightly lower than the LOFTRAN one, as evaluated at the step 3), the moderator density coefficients for the LOFTRAN code were confirmed and final runs were performed.

B. The moderator temperature coefficient (0.105 $\Delta K/g/cm^3$), calculated as described above, provides a power level in excess of the 3D ANC calculated values (120.5% versus 119.1%) for the worst time in life condition (i.e., EOL, rodded core, 0 ppm boron).

Moreover, it should be noted that conservative assumptions are taken in the LOFTRAN analyses to define the input conditions for the 3D ANC evaluation. These assumptions include no credit for high power range neutron flux and overpower ΔT reactor trips and conservatively high PRHR performance.

This analysis is similar to the analysis provided for the AP600, and makes use of the same analytical methods already approved for AP600 licensing.

LOFTRAN-AP computer code and its applicability to the Westinghouse plant design are discussed in WCAP-14234, Rev.1 "LOFTRAN and LOFTTR2 AP600 Code Applicability Document".

The 3D ANC code permits the introduction of advanced fuel designs and heterogeneities, such as axial blankets and part length burnable absorbers, and allows such features to be modeled explicitly. The three dimensional nature of this code provides both radial and axial power distribution. The 3D ANC code is described in "ANC: Westinghouse Advanced Nodal Computer Code", WCAP-10965-P-A September, 1986.

In addition to the above, as described in the LOFTRAN users guide "AP1000 Analysis Methodology Summary for Events Using the LOFTRAN Code Family" which was submitted to the staff during the AP1000 pre-certification review, FACTRAN (DCD Section 15.0.11.1) and VIPRE code (DCD Section 4.4) are used in conjunction with LOFTRAN to evaluate DNBR.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.073

Question:

The analysis of the turbine trip event (Section 15.2.3) identifies that the most limiting DNBR case is the case with a minimum reactivity feedback and without pressurizer spray in combination of an LOOP. However, Section 15.2.3 does not provide calculated DNBRs.

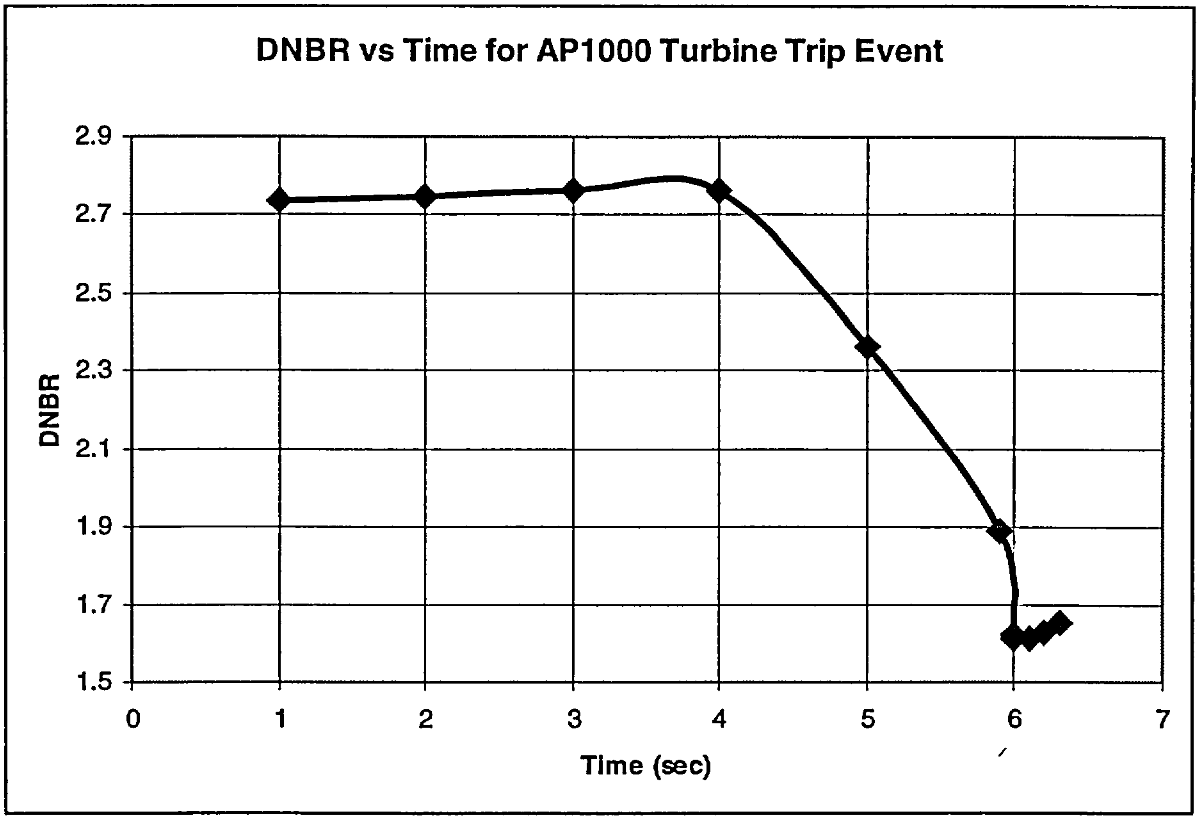
Provide a figure to show the calculated DNBRs for the limiting turbine trip case.

Westinghouse Response:

The following table and figure provide DNBR as a function of time for the AP1000 Turbine Trip Event.

TABLE 1
DNBRs as a Function of Time for
AP1000 Turbine Trip Event

Time (seconds)	DNBR
0.0	2.7371
1.0	2.7491
2.0	2.7621
3.0	2.7641
4.0	2.3611
5.0	1.8931
5.9	1.6231
6.0	1.613 ← Minimum DNBR
6.1	1.613 ← Minimum DNBR
6.2	1.625
6.3	1.653



Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.075

Question:

In consideration of the measurement uncertainty effects, various combinations of initial temperature and pressure are assumed in the analyses. The assumed initial temperatures and pressure are: 7°F below and 50 psi above the normal values for the loss of ac power event (page 15.2-10 of Chapter 15); 7°F and 50 psi below the normal values for the loss of normal feedwater flow (page 15.2-14); and 6.5°F above and 50 psi below the normal values for the feedwater line break event (page 15.2-18). For all three events, the RCS and steam generator pressures will increase during the events. However, the measurement uncertainties are assumed in different directions (above and below the normal value) for the three events.

Address the acceptability of the initial temperatures and pressures with associated uncertainties assumed in the analyses for these pressurization events.

Westinghouse Response:

The definition of the initial conditions for the loss of normal feedwater, loss of ac power and feedline rupture accidents is based on previous sensitivity studies performed on Westinghouse passive plants (e.g., AP600) and specific sensitivity analyses performed for the AP1000 plant.

In particular, sensitivity analyses have been performed to evaluate the impact of the initial average temperature and pressurizer pressure. The different initial conditions for the loss of ac power event and the loss of normal feedwater events are due to the different assumptions and transient behavior for the two events, both of which are, however, concerned mainly with the capability of the PRHR to remove the decay heat (i.e., avoid boiling in the hot legs and avoid pressurizer overfilling) and assure the natural circulation and core cooling in the long term.

For the loss of normal feedwater event, the ac power is assumed available after reactor trip and the CVS makeup pump is assumed to operate. A lower initial pressurizer pressure results in a slightly higher CVS flow rate (CVS flow rate is calculated, by the LOFTRAN code, as function of the RCS pressure) that conversely results in a slightly higher pressurizer water level peak. This is why lower pressure is conservative for this transient. Sensitivity studies show the impact of the pressurizer initial pressure on the plant behavior is minimal.

The analysis of loss of ac power also assumes the loss of normal feedwater as the initiating event. The loss of ac power is modeled at the time of reactor trip (that occurs on low SG water level – narrow range). The analysis is performed to evaluate the PRHR heat transfer performance in natural circulation conditions. The CVS makeup pumps are not factored in the analysis since they are energized by the offsite ac power. Hence, for this analysis, the most critical issue for the overfilling is related to the mass stored in the RCS. This explains the reason that the assumption related to the maximum initial pressurizer pressure is conservative for this transient. Sensitivity studies show the impact of the pressurizer initial pressure on the plant behavior is minimal.

For both of these analyses, minimum average RCS temperature is assumed since this assumption results in the maximum mass stored in the RCS and in the maximum pressurizer water level peak during the transient.

The feedline rupture event is a Condition 4 event. The analysis focuses on the core cooling and the fuel integrity both in short and in the long term and the RCS and SGS integrity. To conservatively assure meeting the core cooling and fuel integrity criteria, the analyses demonstrates that no bulk boiling occurs in the primary coolant system following a feedline rupture prior to the time that the heat removal capability of the steam generators and PRHR exceeds the NSSS heat generation. Sensitivity analyses clearly show that the margin to bulk boiling is minimized by assuming maximum initial RCS average temperature and minimum initial pressurizer pressure. This explains why the assumption related to the maximum initial temperature for the feedline break event is conservative.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.076

Question:

The feedwater line break (FLB) event is analyzed for a double-ended rupture of the largest feedwater line, which was previously identified in WCAP-9230,"Report on the Consequences of a Postulated Main Feedline Rupture," as the limiting case, resulting in a highest peak RCS pressure. Considering the plant design differences in the AP1000 and currently operating PWRs, the results of FLB analysis may be different and the limiting FLB case may be different from the one previously identified for currently operating PWRs.

Address the applicability of WCAP-9230 to the AP1000 design for determination of the limiting FLB case, and confirm that the double-ended rupture is the limiting break for AP1000.

Westinghouse Response:

WCAP-9230 was developed to assess the consequences of a feedline break accident for Westinghouse plants, in agreement with the criteria set forth in the Standard Review Plan, section 15.2.8. WCAP-9230 reports the results of a sensitivity study which determines those system parameters that have the most effects on the results of a system transient following a feedline break accident and defines the worst case initial conditions and assumptions, with and without offsite power available.

To address adverse environmental effects (control system interaction) applied to feedline rupture analyses, a subsequent methodology was needed. Specifically, it was postulated that the adverse steam-filled environment caused by the feedline rupture could have detrimental effects on the response of the feedwater control system.

For this particular accident scenario, the main feedwater control system equipment is assumed to be exposed to an adverse environment, following a feedwater line rupture between the steam generator and the feedline check valve. As a result of this adverse environment, the main feedwater control system malfunctions such that prior to reactor trip, the main feedwater flow in the faulted loop is equivalent to the fluid spilling out the break and no feedwater is being injected into the non-faulted steam generators. This results in no net feedwater flow out the faulted SG. The steam generator water levels will decrease as steam continues to be supplied to the turbine until a reactor trip on the low-low water level setpoint occurs. After reactor trip, the main feedwater control system is assumed to be lost and a full double-ended rupture blowdown is modeled in the faulted loop.

Assuming no blowdown prior to reactor trip is conservative because the assumption of a double-ended break at the initial time would result in an almost immediate reactor trip followed by the safety systems actuation (including steam lines and feed lines isolation). In this case, the intact steam generator inventory would be almost completely preserved and would provide a significant heat sink in the post trip part of the transient and, in particular, in the initial portion of the accident, during which PRHR heat transfer capacity is lower than core decay heat.

Smaller breaks can result in longer time to reactor trip and to safeguard actuations and hence less secondary side inventory would be available at the time of reactor trip and feed line isolation. Of course, in this case the blowdown phase would be slower than for a full double-ended rupture and, more time would be available for the safeguard systems to operate (lower system performances would be required in the short term). The resulting transient would be less limiting than the double ended guillotine rupture of the larger feedline. This is confirmed by the results of sensitivity studies reported in WCAP-9230 and by preliminary runs performed to analyze the behavior of the AP1000 plant.

Thus, the AP1000 feedline break accident is analyzed with the following conservative assumptions:

1) The accident is initiated by a break whose size is such that all the feedwater is spilled out of the break and no secondary fluid is discharged or supplied to the steam generators due to the feedwater control interaction.

2) In the first part of the accident, the plant behavior is the same as a loss of normal feedwater. The secondary system inventory, in both the steam generators, decreases at the same rate and due to the loss of subcooling in the steam generator, a moderate RCS heatup occurs.

3) At the time at which the Low-Low SG Water Level (Narrow Range) is reached (in both the steam generators at the same time), a trip signal is generated and a full double ended rupture of the main feed line is modeled.

4) The blowdown flow rate is evaluated as a critical discharge rate (L/D=0) for the cross sectional area of the main feedline. The fast blowdown results in a very fast drop of the secondary side inventory and complete loss of heat sink in the ruptured steam generator. During the blowdown from the ruptured steam generator, steam is also supplied from the intact steam generator to the ruptured one, until the steam lines are isolated following a low steam line pressure "S" signal. This also results in a decrease of the intact steam generator inventory.

5) A conservative blowdown quality is assumed in the analysis, as described in WCAP-9230 and further discussed in the answer to RAI 440.078.

Based on this, it can be concluded that the analysis methodology followed for the AP1000 and AP600 plant, covers the whole spectrum of feed line breaks. In fact, the first part of the transient is analyzed as loss of normal feedwater event. This results in a heatup of the RCS in the initial portion of the transient. In addition, in the post trip phase of the transient, a full double ended break of the main feed water is assumed. The plant conditions, at which the break occurs, are the worst possible ones in terms of steam generator residual mass, since both the steam generators are at the low-low steam generator water level and the minimum secondary inventory is available, in the intact steam generator, to provide, in conjunction with the PRHR operation, the required heat sink to remove the core decay heat.

Adoption of this analysis approach for both the AP600 and AP1000 conservatively addresses the main feedwater control system environmental interaction issue, the small and intermediate break sizes, and the operation of the feedwater control system following a small feedline rupture.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.077

Question:

The guidance for a LOOP assumed for the FLB analysis is provided in SRP 15.2.8. Specifically, item b of the acceptance criteria states that "Assumptions as to whether offsite power is lost and the time of loss should be make conservatively. Offsite power may be lost simultaneously with the concurrence of the pipe break, the loss may occur during the accident, or offsite power may not be lost."

Discuss the determination of the time of an LOOP assumed for the limiting FLB analysis and address the compliance with the SRP guidance related to the time of an LOOP.

Westinghouse Response:

For the AP1000 plant, the main effects due to the loss of offsite power are related to the loss of forced reactor coolant flow and coincident opening of the trip breakers resulting in the plant shutdown. The passive safety systems do not rely on safety-related ac power, and therefore their operation is not affected by the loss of offsite power. The initiation of the passive systems requires the alignment of safety-related valves energized by safety Class 1E dc power system. Actuation times are essentially the same both with and without ac power available, except for the PRHR actuation that, if ac power is available, is delayed by about 45 seconds with respect to the time at which the low SG water level (narrow range) setpoint is reached. However, for this analysis the PRHR is started on a low SG water level (wide range) signal or following the low steamline pressure "S" signal that follows immediately after.

For the AP1000 feedline break analysis, the loss of ac power is conservatively assumed to occur at the time of reactor trip (rod motion). According to the accident methodology followed to simulate the accident, the full double ended break opens in the feedline (control interaction assumption, refer to answer to RAI 440.076) at the time at which the low SG level NR reactor trip setpoint is reached. This means that rod motion follows the time at which the double ended break open by two seconds, and, hence two full power seconds are input to the coolant, during the blowdown phase, before the rods start to fall in the core.

Should the loss of ac power occur at the time of the break, control rods will be inserted sooner (on the loss of ac power) and hence a lower RCS heat up and pressurization would be experienced by the plant.

The assumption of the loss of ac power at the time of reactor trip (rod motion) is conservative since it results in the maximum possible power input to the RCS and also in an immediate drop of the RCS forced flow and hence of the heat transfer between primary and secondary systems.

Should a loss of ac power occur any later in the transient, the energy stored in the RCS will be lower than for the case analyzed due to the better heat transfer with the steam generator secondary side.

Finally, should the offsite power be available, an automatic reactor coolant pump trip signal will occur on a low steam line pressure "S" signal. This signal occurs quite soon for a double ended feedline break event (about 15 seconds from the reactor trip) but could occur with a significant delay for small feedline breaks. In the later case, PRHR performance will be significantly higher due to the forced flow through the loop and overall RCS conditions will be less limiting. In this case the assumption of loss of ac power at the time of the trip is conservative.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.078

Question:

The Semiscale test data (Section 4.3.3.1 of NUREG/CR-4945) show that the steam generator heat transfer capacity remain unchanged until the steam generator liquid inventory is nearly depleted. This is followed by a rapid reduction to zero percent heat transfer with little further reduction in the steam generator liquid inventory.

Discuss the steam generator heat transfer model used in the FLB analysis and verify that the heat transfer model is conservative as compared to the Semiscale test data.

Westinghouse Response:

The LOFTRAN code evaluates the SG heat transfer as a function of the primary flow, heat flux and secondary system pressure and mass, initialized to match nominal input conditions. The overall heat transfer coefficient during a transient is computed as:

$$UA = U^*A = 1/R * A = (UA)_{nom} * A/A_{nom} \ / \ R/R_{nom} \qquad [1]$$

Where: U = Total primary to secondary thermal conductivity, BTU/sec-ft^2-°F
A = effective heat transfer area, ft^2
R= 1/U = Primary to secondary thermal resistance , (sec-ft^2-°F)/BTU
nom = indicates nominal values

The term R is evaluated by summing up the tube metal resistance, primary film resistance, and secondary film resistance. In particular, the primary film resistance is evaluated using the Dittus-Boelter correlation for forced convection and hence it is sensitive to primary mass flow behavior. On the secondary side, the effective heat transfer area is evaluated as follows:

$$A/A_{nom} = MIN(1.0, VSSGW / VSTB) \qquad [2]$$

Where : VSSGW is the water volume (ft^3) in the steam generator secondary side;
VSTB (ft^3) is the water volume required to cover the steam generator tubes.

In the feedline break analysis, the UA remains constant up to the reactor coolant pump coastdown, that occurs on the loss of offsite power, at the time of reactor trip on low SG Water Level (NR) setpoint. The pumps coasting down results in an immediate drop of the primary flow and hence the primary film heat transfer drops according to the $Re^{0.8}$.

It should be noted that, at the time the pumps start their coast down, the steam generator residual mass is less than 30% of the initial and, due to the large break size and to the conservative assumptions on exit quality, the steam generator empties in less than 10 seconds from the time of the reactor trip.

Based on the above, it can be concluded that, in the analyses reported in the DCD, the overall SG UA, in the broken steam generator starts decreasing for a secondary inventory of about 30% of the initial and hence conservative with respect to the Semiscale tests. The initial decrease in SG UA is mainly due to the coastdown of the RCPs. A few seconds after the pumps coastdown the overall steam generator UA drops to very low values, since the steam generator liquid inventory is nearly depleted.

Even if the feedline break accident analyses are not immediately comparable to the Semiscale experimental results, due to the different assumptions on RCPs operation, the LOFTRAN results are consistent and conservative with respect to the Semiscale test results. The initial drop in overall heat transfer, evaluated by the LOFTRAN code, is driven by the loss of forced primary loop circulation. This assumption is conservative since it results in more energy being stored in the primary system.

Should the analyses be performed assuming no primary pumps coastdown, the overall heat transfer coefficient would have been maintained for lower residual SG masses (in the order of about 15%) and then linearly drop down to zero according equation [2].

In addition to the above, analyses performed for the AP600 (Westinghouse letter DCP/NRC 0990, dated August 14, 1997, in response to comment 22) have shown that the LOFTRAN calculated SG heat transfer is within the range of the Semiscale test data. In particular, a sensitivity study was performed, using the LOFTRAN code, to quantify the effects of different steam generator UA values. Three cases were run:

1. FLB with the SG UA set to normalized heat transfer from Semiscale 14.3 percent break
2. FLB with the SG UA set to normalized heat transfer from Semiscale 50 percent break
3. FLB with the SG UA set to normalized heat transfer from Semiscale 100 percent break

The calculated peak pressure of cases 1, 2, and 3 are 18.09 MPa (2624 psia), 18 MPa (2612 psia), and 17.36 MPa (2518 psia), respectively. The case with the heat transfer from the Semiscale 14.3 percent break is, as expected, the limiting case since the overall heat transfer starts decreasing at higher steam generator residual mass. The resulting peak pressure, for the limiting case, is 18.09 MPa (2624 psia). Compared to the peak pressure of 18.06 MPa (2620 psia) for the AP600 SSAR case. The limiting case with the Semiscale heat transfer shows a small increase of 27.6 kPa (4 psi) in the peak pressure.

In NUREG-1512 (AP600 FSER), the NRC concluded that since the LOFTRAN calculated SG heat transfer is within the range of the Semiscale data and that the sensitivity studies showed that the effects of the heat transfer data differences between the Semiscale data and LOFTRAN model on the peak pressure are small, that therefore the SG heat transfer model used in the FLB is acceptable.

During the AP1000 pre-certification review, Westinghouse documented the applicability of the LOFTRAN code to the AP1000 and addressed the limitations of their use in WCAP-15644 "AP1000 Code Applicability Report." The NRC documented their review in the NRC letter "Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design" dated March 25, 2002 and found these codes acceptable for use for the AP1000, with limitations identified. For LOFTRAN, the staff found that the LOFTRAN code was applicable to the AP1000, and could be used for AP1000 Design Certification. The SG heat transfer model used in the AP1000 analyses is the same as that which has been approved for both the operating plant version of LOFTRAN, as well as the version used for both the AP600 and AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.079

Question:

Page 15.2-26 of Chapter 15 indicates that for the loss of ac power event, the CMT actuates on the low RCS cold leg temperature (T_{cold}) "S" signal at 4753 seconds followed by the closure of the steam line isolation 12 seconds later at 4765 seconds. Page 15.2-27 shows that for the loss of normal feedwater flow event, the steam line isolation occurs on the low T_{cold} "S" signal at 1160.6 seconds followed by the CMT actuation 11 seconds later at 1171.6 seconds. Even though both the CMT and steam line isolation are actuated on low T_{cold} "S" signals, the sequence of the CMT and steam line isolation actuations and the delay time between them are different for the loss of ac power and loss of normal feedwater flow events.

Provide reasons for the differences in the event sequences and delay times.

Westinghouse Response:

For both loss of normal feedwater and loss of ac power events the steam line isolation occurs 12 seconds after the Low Tcold "S" signal.

For the Loss of normal feedwater event, the CMT actuation and RCP trip are assumed to be actuated with the maximum delay from the Low Tcold signal (i.e., 17 seconds and 6 seconds respectively). However, the time of CMT actuation and the time of steam line isolation have been switched in the loss of normal feedwater event sequence (typographical error). The correct timing is as follows:

Event	Time (seconds)
Low Tcold signal reached	1154.6
RCPs trip on low Tcold	1160.6
Steam line isolation on low Tcold	1166.6
CMT actuation on low Tcold	1171.6

As it can be noted the following time delay have been used:

	Delay (seconds)
Steam line isolation	12
CMT actuation	17
RCPs Trip	6

CMT actuation and RCP trip delays are not critical parameters to the results of this analysis. Sensitivity analysis for the loss of normal feedwater, performed assuming the minimum safeguard data for CMT actuation (5 seconds delay instead of 17 seconds), provides the same maximum pressurizer water volume. The reason for that is that the pressurizer water volume peak occurs at about 5 hours from the CMT actuation, so that the effects from a small delay on CMT actuation have no effect on the margin to pressurizer overfill. The assumptions used in this AP1000 analysis are the same as those for the AP600 analysis presented in the AP600 DCD.

For the loss of ac power, no delay has been assumed for the CMT actuation following the Low Tcold Signal. However, as is the case for the loss of normal feedwater, the delay time associated with the CMT actuation has no effect on conclusions from this analysis. For this case, the minimum margin to overfill is evaluated at about 6 hours in the transient. Moreover, as shown by the sensitivity analyses, lower margin to overfilling was evaluated for earlier CMT actuations (for example modifying the Low Tcold setpoint). Based on these sensitivity runs, a zero time delay has been assumed for CMT actuation.

Design Control Document (DCD) Revision:

Correct Table 15.2-1 (Sheet 6 of 7) as indicated on the attached page

PRA Revision:

None

Table 15.2-1 (Sheet 6 of 7)

TIME SEQUENCE OF EVENTS FOR INCIDENTS WHICH RESULT IN A DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

	Accident	Event	Time (seconds)
III.	Loss of normal feedwater flow	Feedwater is lost	10.0
		Low steam generator water level (narrow range) reactor trip reached	70.4
		Rods begin to drop	72.4
		Steam generator safety valves open	80.0
		Pressurizer safety valves open	-
		Maximum pressurizer pressure reached	-
		Pressurizer safety valves reclose	-
		PRHR heat exchanger actuation on low steam generator water level (narrow range coincident with low start up feeedwater flow rate)	132.4
		Steam generator safety valves reclose	144
		Cold leg temperature reaches Low T_{cold} setpoint	**1154.6**
		Reactor coolant pump trip on low T_{cold} "S" signal	1,160.6
		Steam line isolation on low T_{cold} "S" signal	1,166.6
		Core makeup tank actuation on low T_{cold} "S" signal	1,171.6
		Pressurizer safety valves open	3,500
		Pressurizer safety valves reclose	17,702
		Passive residual heat removal heat exchanger extracted heat matches decay heat	~ 17,620
		Maximum pressurizer water volume reached	19,548

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.083

Question:

As indicated in Section 15.4.6, minimum reactor coolant water volumes are used in the boron dilution event analysis. The values of the RCS water volumes are 8126 cubic feet (ft^3) for Modes 1 and 2; 7300 ft^3 for Mode 3; 2805 ft^3 for Mode 4 and 2402 ft^3 for Mode 5.

Specify the water volumes in the reactor vessel, steam generators and RCS pipes used to calculate the minimum RCS water volumes and justify that the RCS water volumes used in the analysis for each mode are conservative and acceptable.

Westinghouse Response:

Background

According to LCO 3.4.4 at least two reactor coolant loops must be OPERABLE and in operation (4 pumps running with variable speed controller bypassed) during Mode 1 and 2 and for Mode 3, 4 and 5, wherever the reactor trip breakers are closed.

According to LCO 3.4.9 at least one RCS pump must be in operation with a total flow through the core of at least 10,000 gpm, in Mode 3, 4 & 5 operation, whenever the reactor trip breaker are open.

If the above condition is not met the operator is required to isolate the possible sources of unborated water.

Mixing Volume Calculation

Mode 1 & 2

In both of these modes of operation the RCPs are running (LCO 3.4.4); therefore, most of the RCS volume is included in the mixing volume for dilution. Mixing volume assumed in the analysis includes all the RCS volumes except the pressurizer and surge line. Steam generator tube volume is evaluated accounting for 10% tube plugging.

Mode 3

For mode 3 operation, whenever the trip breakers are closed, all the pumps are in operation (LCO 3.4.4). If trip breakers are open, LCO 3.4.9 requires at least one pump running. The flow rate driven by the pump is sufficient to provide sufficient mixing. The only difference from the active mixing volume between Modes 1 & 2 and Mode 3, is that Mode 3 mixing volume conservatively does not include the volume of the upper head. Also in this case, considering that the Technical Specifications require the operation of the reactor coolant pumps and that there is almost one hour to dilute to criticality, it can be concluded that the active volume used for the analysis is conservatively calculated since it is expected that a significant dilution would also occur in the upper head.

Steam generator tube volume is evaluated accounting for 10% tube plugging.

Mode 4

Under certain circumstances, it can be postulated that mixing volume for Mode 4 operation is as high as Mode 3 active mixing volume due to the RCP operation. Nevertheless, the analysis performed for the DCD, makes use of a very conservative mixing volume that only includes the reactor vessel volume minus the upper head and only one of the two RNS trains. Also in this case significant RCS portions potentially undergoing mixing (technical specification require at least one pump working at 25% speed) and the loop (in particular the hot leg) volume at which the PRHR is connected have not been considered in the active volume for dilution.

Mode 5

LCO 3.4.9 requires, for mode 3, 4 and 5 operation, whenever the trip breaker are open, at least one pump operating at 25% speed to assure a minimum flow rate of 10,000 gpm through the core. If no pumps are in operation LCO requires the operator to isolate all the sources of unborated water.

Based on the above, dilution during Mode 5 operation with the plant drained to mid-loop is to be considered precluded by administrative procedures and the minimum volume for dilution should be assumed the same as for Mode 4 operation.

Nevertheless, the minimum volume for Mode 5 operation has been evaluated considering the plant drained to middle loop position in the hot leg, while the RNS is operating. Also in this case, RV volume has been conservatively evaluated.

The table below reports the active mixing volumes for dilution and the splitting between the main RCS volumes.

	Mode 1 / 2	Mode 3	Mode 4	Mode 5
V_{RV} (Reactor Vessel Volume excluding Upper Head)	2548.50 ft^3	2548.50 ft^3	2548.50 ft^3	2145.84 ft^3 (drained)
V_{UH} (Reactor Vessel Upper Head Volume)	825.811 ft^3	------ ft^3	------ ft^3	------ ft^3

	Mode 1 / 2	Mode 3	Mode 4	Mode 5
V_{HL} (Hot leg Piping Volume, 2 pipes)	235.096 ft^3	235.096 ft^3	------ ft^3	------ ft^3
V_{SG} (Steam Generator Volume , 10 % Tube Plugging, 2 SG)	1947.865 ft^3	1947.865ft^3	------ ft^3	------ ft^3
V_{CL} (Cold leg Piping Volume, 2 Loops, Including pumps)	621.344 ft^3	621.344 ft^3	------ ft^3	------ ft^3
V_{RNS} (Normal Residual Heat Removal, 1 Train)	------ ft^3	------ ft^3	257 ft^3	257 ft^3
Total Volume	8126.48 ft^3	7300.67 ft^3	2805.50 ft^3	2402.84 ft^3

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.084

Question:

Page 15.4-31 of Chapter 15 indicates that the system overpressure analysis for the rod ejection event is performed with a "plant transient computer code."

Reference the associated NRC acceptance letters and confirm the acceptance of the "plant transient computer code" for licensing calculations.

Westinghouse Response:

The plant transient computer code used for system overpressure analysis is the LOFTRAN computer code. The rod ejection calculation is performed using the TWINKLE code, and the resulting nuclear power transient is then input to LOFTRAN to conservatively calculate the RCS pressure vs. time.

Methods used in the analysis are documented in WCAP-7588, Revision 1A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," which the staff has previously reviewed and accepted. This report demonstrates that the model used in the accident analysis is conservative with regard to a three-dimensional kinetics calculation. In this analysis, Westinghouse considered four cases including beginning-of-cycle at full-power and zero-power, and end-of-cycle at full-power and zero-power. For all cases, the calculated radial average fuel enthalpy is less than 182 calories per gram, which is less than the acceptance criterion of 280 calories per gram specified by RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs." In addition, the calculated pressure surge resulting from the rod ejection does not exceed the reactor coolant system emergency limits (Service Level C) and, thus, satisfies the guidance of RG 1.77.
Transient and Accident Analyses.

The LOFTRAN code has been reviewed and approved by the NRC as noted in the acceptance letter from C. O. Thomas (U.S. NRC) to E. P. Rahe, Jr. (Westinghouse) titled "Acceptance for Referencing of Licensing Topical Report WCAP-7907 (P)/(NP), LOFTRAN Code Description", dated July 29, 1983. In addition, LOFTRAN was approved for use in the AP600 (NUREG-1512), and its applicability to the AP1000 has been approved by the NRC during the pre-certification review of AP1000 letter ("Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design" dated March 25, 2002). Please see the response to 440.054 for a discussion of the applicability of the analysis codes to the AP1000.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.085

Question:

Page 15.5-3 of Chapter 15 indicates that in the over-pressure analysis for the CMT inadvertent operation event, "PRHR heat transfer capacity has been minimized."

Describe the model that is used to calculate a minimized PRHR heat transfer capacity and demonstrate its conservatism by comparing with heat transfer test data applicable to the AP1000 PRHR heat exchanger.

Westinghouse Response:

PRHR heat transfer capability has been minimized by modeling conservatively high pressure drops through the PRHR loop. This limits the PRHR flow rate and hence the calculated value of the primary side heat transfer coefficient that is internally evaluated according to the Dittus-Boelter correlation. In addition, maximum technical specification value for PRHR tube plugging (53 tube plugged) and minimum effective heat transfer area (based on a minimum effective length per tube of 454.64 in instead of 465.64 in) have been assumed.

The model used to calculate the heat transfer coefficients and their applicability to the Westinghouse passive plants accident analysis is described and justified in the WCAP-14234, "LOFTRAN & LOFTTR2 AP600 Code Applicability Document", Rev. 1, and WCAP-15644, "AP1000 Code Applicability Report".

More information on the same issue is reported in the Answer to RAI 440.074.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.086

Question:

As indicated on pages 15.5-3 and 15.5-7 of Chapter 15, the assumed initial temperatures and pressure are: 7°F and 50 psi below the normal values for the CMT malfunction event, and 6.5°F and 50 psi above the normal values for the CVS malfunction event. Both events are analyzed to predict over-pressure during the transients. The assumed initial temperatures and pressures show that measurement uncertainties are assumed in different directions (below and above the normal values) for these events.

Discuss the criteria used to select initial temperatures and pressures for the analyses and justify that the temperatures and pressures so selected are conservative and acceptable.

Westinghouse Response:

Initial conditions for CMT Malfunction Event and for the CVS malfunction events have been defined through a significant number of sensitivity runs. In particular, plant initial conditions assumed in the above analysis corresponds to the worst combination of initial plant parameters for the given transient.

The basis for the plant initial conditions, and in particular the initial RCS temperature and the pressurizer pressure used in the transient analyses presented in Chapter 15 is specific to each accident scenario. The definition of the initial conditions for each accident analysis is based on past Westinghouse operating plant analyses, specific AP600 analyses, and/or specific sensitivity analyses that have been performed.

In the case of CMT Malfunction, analysis is performed assuming the spurious operation of one CMT (both valves are assumed to open even if this assumption is unrealistic unless a "S" signal is generated). In addition it is assumed that no reactivity effects are associated to the CMT boron injection (please note that this corresponds to the assumption of rod control operation with the capability to compensate very large reactivity changes) so that the reactor operates at full power and, full power temperature is maintained until reactor trip is reached on Hi pressurizer water Level setpoint.

Initial RCS mass is maximized by assuming that the average coolant temperature is at the minimum value. This corresponds to a higher coolant density with respect to nominal conditions. This assumption assures that the RCS active mass is maximized at the time of the reactor trip and that at the time of the reactor trip the CMT coolant expansion has occurred (from CMT initial conditions, 120°F, to RCS average temperature). Sensitivity analyses have shown larger margins to pressurizer overfill assuming nominal initial temperature and maximum initial temperature.

The CVS malfunction event is a quite different event since the initial mass input is provided by the CVS system. The analysis is performed, with very conservative assumption, in terms of CVS boron concentration, that result in a plant cooldown that causes the operation of CMTs after a significant mass input from the CVS has already occurred.

Starting the transient from higher coolant average temperature requires more time to reach the Low-Tcold "S" signal setpoint. This, in turn, results in more water being added from the CVS to the RCS, as seen in sensitivity runs, and finally results in lower margins to acceptance criteria.

For both events, initial pressurizer pressure is not as significant as initial RCS temperature. Nevertheless, sensitivity studies have been performed to define the worst combination of initial plant conditions.

The reason for the assumptions on initial pressurizer pressure can be justified as follow:

1) The CVS malfunction event is characterized by an initial cooldown and pressure would decrease without the operation of the heaters. Pressurizer control system operates to keep the pressure to the reference value (initial pressurizer pressure) and hence the assumption of a maximum initial pressurizer pressure results in a slightly higher coolant density at the time of the reactor trip and CVS isolation (that occur as a consequence of the Low Tcold signal). More water is then added by the CVS before the CMT actuation. This results in slightly lower margins to acceptance criteria.

2) The CMT malfunction event is characterized by a pressurizer pressure increase that is counteracted by the pressure control system. Initiating the transient from an higher pressure would result in an higher pressure during the pre-trip phase and hence more water should be added by the broken CMT before the reactor trip. This would result in a lower injection capability in the post trip phase and hence higher margins to the acceptance criteria.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.087

Question:

Section 15.6.2 indicates that the break flow rates are limited to 100 gallons-per-minute (gpm) and 130 gpm for the CVS charging line break and the RCS sample line break, respectively.

Discuss the models used to calculate these break flow rates and show that the break flow rates are overpredicted when the temperature and pressure conditions and break sizes are considered in the break flow rate calculation.

Westinghouse Response:

As mentioned in Section 15.6.2 of the DCD, the two lines carrying primary coolant outside containment are the CVS letdown line and the reactor coolant system sample line.

Flow through the CVS letdown line is limited by the letdown orifice, located inside containment and upstream of the containment isolation valve. This orifice is sized to limit flow to 100 gpm at normal letdown conditions (approximately 130°F, 2200 psia) , and this value will be confirmed during pre-operational testing. This flow limiting orifice will be a multi-stage design and procured from an orifice manufacturer; standard orifice sizing techniques are used.

The 130 gpm cited for the RCS sample line break is very conservative; it is based on the historic Westinghouse practice of using a 3/8" flow restricting orifice in connections to the primary loops.

For the sample line, assuming 200 feet (including the N^{16} delay coil) of 0.25 inch OD / 0.12 inch ID drawn tubing, Darcy's formula indicates that the actual unrestricted flowrate through this tubing to the outside of containment will be less than 1.5 gpm.

Therefore, the assumption used in offsite dose calculations of 130 gpm leakage for 30 minutes is conservative.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.088

Question:

Page 15.6-10 of Chapter 15 indicates that the modified LOFTT2 code described in WCAP14234 is used for the steam generator tube rupture (SGTR) analysis.

Reference the associated NRC acceptance letters and confirm the acceptance of the modified LOFTTR2 code for the AP1000 licensing calculations. Also, verify the use of the computer code for the SGTR analysis is within the applicable range of the NRC-approved code.

Westinghouse Response:

The LOFTTR2 code, as documented in WCAP-10698-P-A, and supplemented by WCAP-10759-A and WCAP-11002 is the NRC-approved code used to analyze an SGTR event for conventional Westinghouse PWRs. LOFTTR2 is a modified version of LOFTRAN with a more realistic break flow model, a two-region steam generator secondary side, and an improved capability to simulate operator actions during an SGTR event. The version of LOFTTR2 applied to the AP600 SGTR analyses incorporated the LOFTRAN changes to simulate passive safety features for the AP600 design. These changes are documented in WCAP-14234. The staff has reviewed and accepted the application of the modified LOFTTR2 code to the AP600 for SGTR analyses and provided its evaluation in Section 21.6.1 of NUREG-1512 – The AP600 Final Safety Evaluation Report.

The applicability of the LOFTTR2 analysis code to the AP1000 was addressed as part of AP1000 pre-certification review. Westinghouse submitted WCAP-15644, "AP1000 Code Applicability Report." The applicability of the LOFTTR2 code is addressed in the section on LOFTRAN-AP, which refers to the LOFTRAN family of codes used for AP600 and AP1000. The use of the LOFTTR2 analysis code for AP1000 is within the applicable ranges of the approved code. The NRC Letter "Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design" dated March 25, 2002 provides the results of the NRC staff's review of the LOFTRAN/LOFTTR2 analysis codes for AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.089

Question:

Page 15.6-11 of Chapter 15 indicates that in the SGTR analysis, the ruptured SG power-operated relief valve (PORV) is assumed to fail open when the low-2 pressurizer level "S" signal generates, while page 15.6-12 and Table 15.6.3-1 show that the failure of the PORV occurs on the low pressurizer pressure "S" signal.

Provide the rationale for selection of the time of the SG PORV to fail and show that the selected PORV failure time results in a maximum RCS mass release and is conservative for the SGTR analysis. Correct any inconsistencies in Chapter 15 for the PORV failure time.

Westinghouse Response:

The most probable time for the PORV to fail open would be at reactor trip, when the valve initially opens to relieve steam due to the loss of steam dump to condenser associated with the assumed loss of offsite power. In analyses for operating plants performed with the approved methodology of Supplement 1 to WCAP-10698, the failure is conservatively delayed until the time when the operators is assumed to isolate the ruptured steam generator. Delaying the failure allows for accumulation of activity in the secondary system (and in the primary system in the case with the accident initiated iodine spike). The delayed failure also results in higher integrated releases by delaying the actions that lead to break flow and steam release termination. For the AP1000 the analysis does not credit operator actions, so a different time was selected. The assumption of a delayed failure was retained resulting in the buildup of activity and higher integrated releases seen in operating plant analyses. Once an "S" signal is generated the passive safety systems act to reduce the atmospheric releases by reducing heat transfer to the steam generators, and reducing RCS pressure. Since the PORV failure itself would result in an "S" signal being generated (on low steamline pressure, or low pressurizer pressure or level) the failure was delayed such that the failure would not provide any benefits in actuating safety systems.

The inconsistencies in Chapter 15 for the PORV failure time will be corrected as shown.

Design Control Document (DCD) Revision:

DCD section 15.6.3.2.1.2 and Table 15.6.3-1 will be revised as follows:

The reactor is assumed to be operating at full power at the time of the accident, and the initial secondary mass is assumed to correspond to operation at nominal steam generator mass minus an allowance for uncertainties. Offsite power is assumed to be lost and the rods are assumed to be inserted at the start of the event because continued operation of the reactor coolant pumps has been determined to reduce flashing of primary-to-secondary break flow and, consequently, lower offsite radiological doses. Maximum chemical and volume control system flows and pressurizer heater heat addition are assumed immediately (even though offsite power is not available) to conservatively maximize primary-to-secondary leakage. The steam dump system is assumed to be inoperable, consistent with the loss of offsite power assumption, because this results in steam release from the steam generator power-operated relief valves to the atmosphere following reactor trip. The chemical and volume control system and pressurizer heater modeling is conservatively chosen to delay the low pressurizer pressure "S" **and the low-2 pressurizer level** signal and associated protection system actions.

The limiting single failure is assumed to be the failure of the ruptured steam generator power-operated relief valve. Failure of this valve in the open position causes an uncontrolled depressurization of the ruptured steam generator, which increases primary-to-secondary leakage and the mass release to the atmosphere.

It is assumed that the ruptured steam generator power-operated relief valve fails open when **either** the low-2 pressurizer level **signal or the low pressurizer** "S" signal **are** ~~is~~ generated. This results in the maximum integrated flashed primary-to-secondary break flow.

Table 15.6.3-1

STEAM GENERATOR TUBE RUPTURE SEQUENCE OF EVENTS

Events	Time (seconds)
Double-ended steam generator tube rupture	0
Loss of offsite power	0
Reactor trip	0
Reactor coolant pumps and main feedwater pumps assumed to trip and begin to coastdown	0
One chemical and volume control pump actuated and pressurizer heaters turned on	0
Low~~-pressurizer pressure "S" signal~~-**2 pressurizer level signal** generated	2,498
Ruptured steam generator power-operated relief valve fails open	2,498
Core makeup tank injection and PRHR operation begins (following maximum delay)	2,515
Ruptured steam generator power-operated relief valve block valve closes on low steamline pressure signal	2,979
Chemical and volume control system isolated on high-2 steam generator narrow range level setpoint	12,541
Break flow terminated	24,100

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.090

Question:

According to the description on Section 7.3.1.2.4, the ADS-4 consists of four parallel paths. The four paths are divided into two redundant groups with two paths in each group. Within each group, one path is designed to be substage A and the second path is designed to be substage B. Therefore, there are two paths for each of Stage-4A and Stage-4B ADS. Table 15.6.5-7, AP1000 ADS parameters, indicates that the number of paths for Stage-4A ADS is 1 out of 2.

Explain the inconsistency in Table 15.6.5-7 for the number of paths for Stage-4A ADS.

Westinghouse Response:

As described in Section 7.3.1.2.4, the AP1000 ADS Stage-4 design consists of a total of four parallel flow paths. Two flow paths are designated Stage-4A and two flow paths are designated Stage-4B. For some design basis accidents, a single failure of one of the four installed ADS-4 flow paths is postulated to occur. Specifically, a failure of one of the ADS Stage-4A flow paths is postulated to occur. This is reflected in Table 15.6.5-7 where the number of intact Stage-4A flow paths is shown to be "1 out of 2".

Design Control Document (DCD) Revision:

None required.

PRA Revision:

None required.

RAI Number: 440.093

Question:

Please include the units in Table 15.6.5-1

Westinghouse Response:

Table 15.6.5-1 refers to the releases of activity from the core during a postulated large break Loss-of-Coolant Accident. The releases are listed as fractions of the original core inventory. Table 15.6.5-1 will be revised to make this clear.

Design Control Document (DCD) Revision:

The following revision to DCD Table 15.6.5-1 will be incorporated.

Table 15.6.5-1

CORE ACTIVITY RELEASES TO THE CONTAINMENT ATMOSPHERE

Nuclide	**Gap Release Released over 0.5 hr. (0.167 - 0.667 hr)[1]**	**Core Melt In-vessel Release (0.667 - 1.967 hr)[1]**
Noble gases	0.05	0.95
Iodines	0.05	0.35
Alkali metals	0.05	0.25
Tellurium group	–	0.05
Strontium and barium	–	0.02
Noble metals group	–	0.0025
Cerium group	–	0.0005
Lanthanide group	–	0.0002

Notes:
1. **Releases are stated as fractions of the original core fission product inventory.**
2. Dash (–) indicates not applicable.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.094

Question:

Tables 15.6.5-10 and -11. Accumulator injection start; is it out of sequence or is there a misprint in the initiation time?

Westinghouse Response:

This is not a misprint. However, the sequence of events tables in DCD Section 15.6.5 will be modified so that they are in sequential order as indicated below.

Design Control Document (DCD) Revision:

DCD Section 15.6, Table 15.6.5-9:

Table 15.6.5-9

2-INCH COLD LEG BREAK IN CLBL LINE SEQUENCE OF EVENTS

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	54.7
Steam turbine stop valves close	60.7
"S" signal	61.9
Main feed isolation valves begin to close	63.9
Reactor coolant pumps start to coast down	67.9
ADS Stage 1	1334.1
~~Accumulator injection starts~~ADS Stage 2	~~1405~~1404.1
~~ADS Stage 2~~Accumulator injection starts	~~1404.1~~1405
ADS Stage 3	1524.1
Accumulator empties	1940.2
ADS Stage 4	2418.6
Core makeup tank empty	2895
IRWST injection starts	3280

DCD Section 15.6, Table 15.6.5-10:

Table 15.6.5-10

DOUBLE-ENDED INJECTION LINE BREAK SEQUENCE OF EVENTS

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	13.1
Steam turbine stop valves close	19.1
"S" signal	18.5
Main feed isolation valves begin to close	20.5
Reactor coolant pumps start to coast down	24.5
~~Accumulator injection starts~~	~~251~~
ADS Stage 1	182.7
Intact accumulator injection starts	251
ADS Stage 2	252.7
ADS Stage 3	372.7
ADS Stage 4	492.7
Intact accumulator ~~Accumulator~~ empties	598.4
Intact loop core makeup tank empties	2006
Intact loop IRWST injection starts*	2076

Note:
*Continuous injection period

DCD Section 15.6, Table 15.6.5-11:

Table 15.6.5-11

DOUBLE-ENDED INJECTION LINE BREAK SEQUENCE OF EVENTS

Event	AP1000 Nominal Containment Time (seconds)	AP1000 w/25 psi Back-presure Time (seconds)
Break opens	0.0	0.0
Reactor trip signal	13.1	13.1
Steam turbine stop valves close	19.1	19.1
"S" signal	18.5	18.6
Main feed isolation valves begin to close	20.5	20.6
Reactor coolant pumps start to coast down	24.5	24.6
~~Accumulator injection starts~~	~~251~~	~~255~~
ADS Stage 1	182.7	182.4
Intact accumulator injection starts	251	--
ADS Stage 2	252.7	252.4
Intact accumulator injection starts	--	255
ADS Stage 3	372.7	372.4
ADS Stage 4	492.7	492.4
Intact accumulator ~~Accumulator~~ empties	598.4	600.7
Intact loop IRWST injection starts*	--	1440
Intact loop core makeup tank empties	2006	2350
Intact loop IRWST injection starts*	2076	~~1440~~ --

Note:
*Continuous injection period

DCD Section 15.6, Table 15.6.5-12:

Table 15.6.5-12

10-INCH COLD LEG BREAK IN SEQUENCE OF EVENTS

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	5.2
"S" signal	6.4
Main feed isolation valves begin to close	8.4
Steam turbine stop valves close	11.2
Reactor coolant pumps start to coast down	12.4
Accumulator injection starts	85.
Accumulator 1 empties	418.2
Accumulator 2 empties	425.5
ADS Stage 1	750.0
ADS Stage 2	820.
ADS Stage 3	940.
ADS Stage 4	1491.
Core makeup tank 2 empty	1800.*
IRWST injection starts	~1800
Core makeup tank 1 empty~~IRWST injection starts~~	1900.* ~~1800~~

Note:
*The CMTs never truly empty although they cease to discharge at these times.

PRA Revision:

None

RAI Number: 440.095

Question:

It is not evident that the choice of the DEDVI for the demonstration of long-term cooling is the most conservative case. The case of a DVI break achieves IRWST injection early with relatively high decay heat. However, a great deal of water is injected through the DVI break after the sump water level achieves the break elevation.

If the transient did not involve a DVI break, would there be sufficient water to keep the core covered through the IRWST injection?

Westinghouse Response:

The IRWST injection capability following a DEDVI break is significantly degraded from that available for any other postulated break location. This is demonstrable from a review of the long-term cooling transient scenarios.

Any other break location will receive delivery of IRWST water through two intact direct vessel injection lines throughout long-term cooling. Referring to the DEDVI break analysis provided in AP1000 DCD section 15.6.5.4C, IRWST injection occurs only through the intact DVI line (Figure 15.6.5.4C-14) at the initiation of long-term cooling. The broken DVI line does not provide any flow (Figure 15.6.5.4C-13) into the reactor vessel until the compartment liquid level exceeds the broken pipe elevation. The intact DVI line flow rate (at 3300 seconds) before water delivery through the break location begins is less than the sum of the injection rates through the two DVI lines during the remainder of IRWST injection. Therefore, 3300 seconds represents the limiting point in time during IRWST injection for the DCD DEDVI break case. Furthermore, it also represents a more limiting condition during IRWST injection than exists for any/ all other break locations during long-term cooling. For any other break location postulated, both intact injection lines will deliver IRWST water at a rate that is very close to the Figure 15.6.5.4C-14 rate for the intact DVI line throughout IRWST injection. The long-term cooling phase of other LOCA events begin at a later time after reactor trip and with higher containment water level. Overall, the DEDVI break case predicts the minimum IRWST injection rate among all post-LOCA long-term cooling scenarios and does so at a time when core decay heat is high and is generating a large steam flow to be vented through the ADS-4 flow paths.

The above discussion confirms that the DEDVI break is the most conservative long-term cooling IRWST injection ECCS performance case for AP1000. Any other postulated break location would be less limiting in its ability to provide the water necessary to keep the core covered during IRWST injection.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.107

Question:

TS 5.6.5 lists WCAP-14807, "NOTRUMP Final Validation for AP600," and WCAP-15644, "AP1000 Code Applicability Report," as the approved analytical methods used to determine the heat flux hot channel factor. However, the NRC review of WCAP-15644 has identified possible deficiencies for the AP1000 application in the NOTRUMP entrainment models at the time of ADS4 actuation, as described in a letter from James Lyons to W. E. Cummins, "Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design," dated March 25, 2002.

Appropriate approved reports should be listed when the final resolution of the application of the AP600 codes to the AP1000 design is reached.

Westinghouse Response:

The NOTRUMP code is used to perform the Small Break Loss of Coolant Accident (SBLOCA) analyses. These SBLOCA analyses verify that the design heat flux hot channel factor is such that the SBLOCA does not result in a violation of the limits specified in 10 CFR 50.46 Appendix-K.

Please see the Westinghouse response to RAI 440.054 for a discussion of the open items contained in the NRC letter from James Lyons to W. E. Cummins, "Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design," dated March 25, 2002.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.110

Question:

Section 19E.2.1.2.6 indicates that the design pressure of 40 psia for the SG nozzle dam is selected to withstand the RCS pressures that can occur during a loss of shutdown cooling event.

Discuss the analysis of the loss of shutdown cooling event to show that the calculated peak RCS pressure is within the design pressure of 40 psia for the nozzle dam.

Westinghouse Response:

The steam generator is equipped with permanently mounted nozzle dam brackets, which are designed to support nozzle dams during refueling operations. The design pressure of the nozzle dam bracket and nozzle dam is selected to withstand the RCS pressures that can occur during a loss of shutdown cooling.

A loss of shutdown cooling analysis was performed for AP1000 using the NOTRUMP code. The analysis is performed consistent with the analysis approach used for the AP600 loss of shutdown cooling analysis presented for the AP600 in the "AP600 Shutdown Evaluation Report," WCAP-14837 Rev. 3. The initial conditions are assumed to be Mode 5 with the RCS open through the ADS Stage 1-3 valves. Following the loss of the normal residual heat removal system cooling, the RCS pressure increases as shown in Figure 1. Manual ADS-4 actuation is assumed to occur at 4800 seconds, once the vessel inventory is reduced to the elevation of bottom of the hot leg. The operators base this action on the installed hot leg level instrumentation. For this transient, the maximum RCS pressure is 44 psia. The design pressure for the SG nozzle dam is specified to be 50 psia.

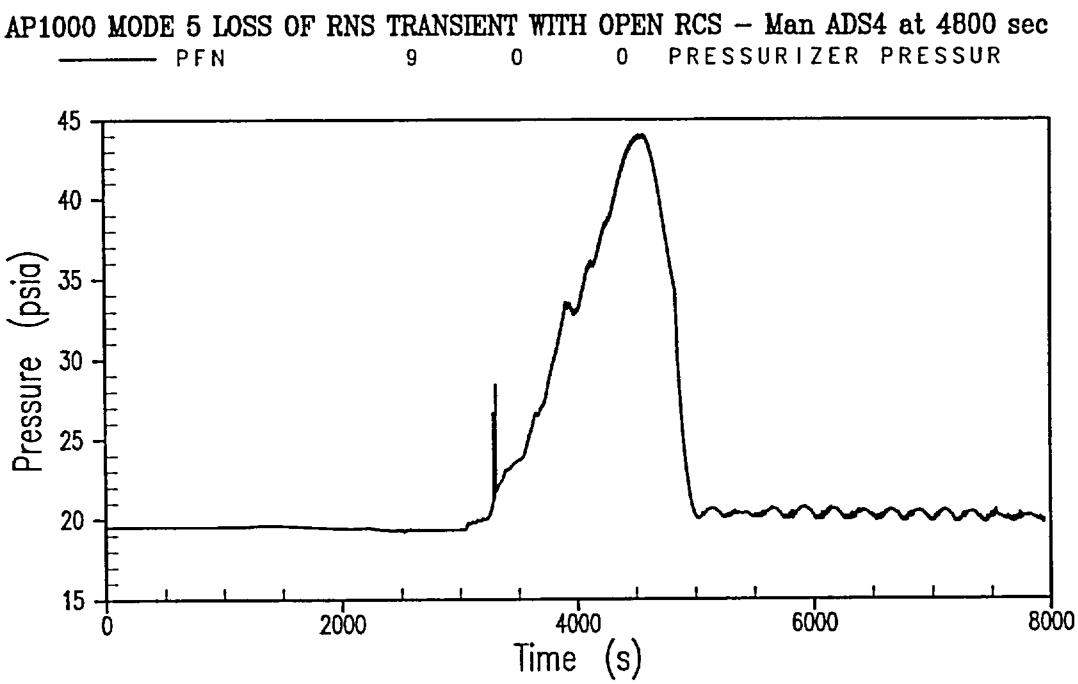


Figure 1 - RCS Pressure for Mode 5 Loss of RNS Event

Design Control Document (DCD) Revision:

Appendix 19E will be revised as follows:

The steam generator is equipped with permanently mounted nozzle dam brackets, which are designed to support nozzle dams during refueling operations. The design pressure of the nozzle dam bracket and nozzle dam is selected to withstand the RCS pressures that can occur during a loss of shutdown cooling. The nozzle dam design pressure is at least ~~40~~ 50 psia.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.111

Question:

As stated in Section 19E.2.2.2.2, the AP1000 safety-related actuations include the signal to isolate the main steam line on a high negative rate of change in steam pressure. This safety-related signal is provided to address a steam line break (SLB) that could occur in Mode 3 or 4. Item c(2) of TS Table 3.3.2-1 specifies that the steam line isolation signal (SLIS) on a high negative steam line pressure is required to be operable for only Mode 3 conditions.

Explain why the applicability of the TS requirements for the SLIS does not include Mode 4 conditions.

Westinghouse Response:

The text of Section 19E.2.2.2.2 is inconsistent with itself and with the Technical Specifications. See the correction to DCD Section 19E.2.2.2.2 listed below. Similar to the AP600 and current Westinghouse operating plants, the high negative rate of change in steam pressure signal is not required to be operable in Mode 4 and below.

The emergency safeguards features used to mitigate steam line breaks include borating the reactor coolant system and closing the main steam line isolation valves (MSIVs). The RCS is borated by activating the core makeup tanks. Borating the RCS attenuates post reactor trip criticality concerns and ensures the integrity of the reactor core. Closing the main steam isolation valves terminates the blowdown of the steam generators or if the steam line break is between the steam generator and the MSIV, closing the MSIVs limits the blowdown to a single steam generator.

In Modes 1 and 2 a Safeguards Actuation ("S") signal actuates the CMT tanks. A safeguards actuation signal is generated on a low RCS cold leg temperature setpoint, a low steam line pressure setpoint, a low pressurizer pressure setpoint, or the high-2 containment pressure setpoint. A low steam line pressure signal also actuates closure of the main steam isolation valves. These same protection functions remain active in the upper portion of Mode 3.

In Mode 3, the AP1000 Technical Specifications require the RCS boron concentration to be increased to meet the shutdown margin requirements at an RCS temperature of 200 F prior to depressurizing the RCS below the P-11 permissive (on the order of 1970 psig). Below P11, the low steam line pressure signal may be blocked. When the low steam line pressure signal is disabled, the high negative steam pressure rate signal is automatically enabled.

In the lower portion of Mode 3 with the RCS borated to meet shut down margin requirements at 200 F, there is no longer a concern that a steam line break can cause a return to power. Therefore, there is no longer a need for the CMTs to actuate for the purpose of borating the

RCS during a cool down event. With the RCS heavily borated in Mode 3, the primary concern is the impact of an uncontrolled blowdown of both steam generators on the integrity of the containment.

In the lower portion of Mode 3, the primary signal for closure of the main steam isolation valves is the high-2 containment pressure signal. As an alternate diverse signal, the high negative steam pressure rate signal continues to be available in the lower portion of Mode 3 to protect containment integrity in the case of a main steam line break. The high-2 containment pressure signal limits the blowdown to at most a single steam generator if the break is inside containment between the steam generator and the MSIV.

As the plant enters Mode 4, the high-2 containment pressure signal continues to be the primary signal to close the main steam line isolation valves and alleviate the steam release to containment. The high-2 containment pressure signal is available throughout Mode 4, and may only be blocked if the MSIVs are first closed. However, in Mode 4 and below, the amount of stored energy is reduced, and the consequence of a main steam line break is not expected to challenge containment integrity. Therefore, in Mode 4 and below, the high negative steam pressure rate isolation function may be blocked.

Design Control Document (DCD) Revision:

19E.2.2.2.2 Safety-Related Actuation in Shutdown Modes

The AP1000 has safety-related actuations associated with the SGS that are operable during shutdown modes. These include the PRHR HX actuation on low steam generator level during shutdown modes, and this is discussed in subsection 19E.2.3 of this appendix. Also included is the isolation of the main steam line on a high (large) negative rate of change in steam pressure. This safety-related signal is provided to address a steam line break that could occur in Mode 3~~ or 4~~. If actuated, this signal causes the MSIVs to close to terminate the blowdown of the SGS following a steam line break. This signal is placed into service below the setpoint that disables the low steam line pressure signal (P11) that actuates steam line isolation as discussed in Section 7.3. When the operator manually blocks the low steam line pressure signal, the steam line high pressure-negative rate signal is automatically enabled.

This signal is operable during Mode 3 when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In Modes 4, 5, and 6, this function is not needed for accident detection and mitigation. Subsection 19E.4.2.3 discusses steam line break events that could occur in shutdown modes. Operability of this actuation logic is discussed in the AP1000 Technical Specifications (Section 16.1).

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.112

Question:

As indicated on page 19E-10, for non-LOCA transients, the passive core cooling system (PXS) in conjunction with the passive containment cooling system (PCS) has the capability to establish safe shutdown conditions, cooling the RCS to less than 420 °F within 36 hours, with or without the RCPs operating. The staff finds that none of the analyses for the non-LOCA events has demonstrated that the RCS is cooled down to less than 420°F (see results on pages 15.2-58, -58, -69,-70, -79 and -80 etc..)

Provide a discussion of the PXS analysis to show that the PXS can be used to cool down the RCS to less than 420°F within 36 hours as designed for non-LOCA transients.

Westinghouse Response:

The NRC Commission guidance regarding the ability of the passive safety systems to bring the plant to a safe shutdown temperature of 420F within 36 hours is outlined in SECY-94-084, Item C, Safe Shutdown. To address this issue, an analysis is presented in Appendix 19E.4.10.2 of the AP1000 DCD. This analysis is for a loss of ac power event, and the results presented demonstrate the ability of the AP1000 passive safety systems to reduce the temperature of the RCS to less than 420F within 36 hours. This analysis is similar to the analysis submitted to the NRC in WCAP-14837 Revision 3, "AP600 Shutdown Evaluation Report," March 1998, and which was included as Appendix 19E of the AP600 DCD. As is the case for AP1000, the purpose of the AP600 analysis was to demonstrate compliance with the guidance outlined in SECY-94-084 Item C Safe Shutdown. The accident analysis results presented in Chapter 15 are not intended to demonstrate compliance with SECY-94-084, but rather, are intended to demonstrate compliance with the requirements of 10CFR 50, and more specifically, the analysis of accidents stipulated in the Standard Review Plan.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.113

Question:

As indicated in Section 19E.2.4.2.1, a large net positive suction head (NPSH) provides the RNS pump the capacity during mid-loop operations with saturated fluid in the RCS without throttling the RNS flow.

Discuss the required NPSH for this configuration.

Westinghouse Response:

In Chapter 5, section 5.4.7.2.1 of the AP1000 DCD it states that some throttling of a RNS flow control valve is necessary when the RCS is at saturated conditions and mid-loop level in order to maintain adequate net positive suction head for the RNS pump. This change from the AP600 design was mistakenly not reflected in Section 19E.2.4.2.1. The text in Section 19E.2.4.2.1 will be modified to reflect the current AP1000 design.

The required NPSH for this configuration is approximately 10 feet.

Design Control Document (DCD) Revision:

From DCD page 19E-12:

19E.2.4.2.1 RNS Pump Elevation and NPSH Characteristics

The AP1000 RNS pumps are located at the lowest elevation in the auxiliary building. This location provides the RNS pumps with a large available NPSH during all modes of operation including RCS mid-loop and reduced inventory operations. The large NPSH provides the pumps with the capability to operate during **most** mid-loop conditions ~~with saturated fluid in the RCS~~ without throttling the RNS flow. **If the RCS is at mid-loop level and saturated conditions, some throttling of a flow control valve is necessary to maintain adequate net positive suction head for the RNS pumps.** ~~This allows for~~ T~~t~~he RNS pumps ~~to~~ **can** be restarted and operated with ~~saturated~~ RCS conditions that might occur following a temporary loss of RNS cooling.

The plant piping configuration, piping elevations and routing, and the pump characteristics allow the RNS pumps to be started and operated at their full design flow rates **in most conditions** without the need to reduce RNS pump flow to meet pump NPSH requirements. This ~~eliminates~~ **reduces** the potential failure mechanism that exists in current PWRs, where failure of an air-operated control valve can result in pump runout and cavitation during mid-loop operations.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.116

Question:

As indicated on page 19E-27, in Modes 2 through 4, the transient response to a loss of condenser vacuum or inadvertent MSIV closure is bounded by the turbine trip analysis from full power because the power mismatch is low.

Explain why the power mismatch is lower for a loss of condenser vacuum or inadvertent MSIV closure as compared to the turbine trip event at full power conditions.

Westinghouse Response:

The severity of events is increased if the primary to secondary power mismatch is increased. Therefore, the most severe results occur if the plant is initially operating in Mode 1 at maximum-rated plant power conditions rather than lower power conditions.

Thus, a loss of condenser vacuum or inadvertent MSIV closure from Modes 2, 3 or 4 will be bounded by the turbine trip analysis, which is, initiated from Mode 1 full power conditions when the power mismatch is the highest.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.118

Question:

As stated on page 19E-33, WCAP–10698-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," concludes that for standard Westinghouse PWRs, zero power and low mode SGTR overfill analyses are not limiting, based on more rapid operator responses expected in those conditions. It further states that when operator actions are credited for AP1000 SGTR mitigation, the plant behaves in a manner comparable to a standard Westinghouse PWR and the conclusions of WCAP-10698-A apply to AP1000.

Discuss a comparison of applicable analyses to demonstrate that the AP1000 plant behaves in a manner comparable to a standard Westinghouse PWR.

Westinghouse Response:

In the event of a SGTR, the operators can diagnose the accident and perform recovery actions to stabilize the plant, terminate the primary-to-secondary leakage, and proceed with orderly shutdown of the reactor. The operator actions for SGTR recovery are provided in the plant emergency operating procedure. The major operator actions for both a standard Westinghouse PWR and the AP1000 plant are compared in the following table.

Operator Action	Standard Westinghouse PWR	AP1000
Identify the ruptured steam generator	Detected by an unexpected increase in steam generator narrow range level or a high radiation indication from any main steam line monitor, steam generator blowdown line monitor, or steam generator sample.	Same
Isolate the ruptured steam generator	Isolate steam flow from and stop the feedwater flow to the ruptured steam generator.	Same
Cooldown of the reactor coolant system	Offsite power available – using the normal steam dump system to the condenser. Offsite power not available – using the intact steam generator PORV	Same. Additionally the PRHR can be used.
Depressurize the reactor coolant system to restore reactor coolant inventory	Reactor coolant pumps running – normal spray. Reactor coolant pumps not running – pressurizer PORV or auxiliary pressurizer	Reactor coolant pumps running – normal spray. Reactor coolant pumps not running – auxiliary

Operator Action	Standard Westinghouse PWR	AP1000
	spray.	pressurizer spray or ADS valves.
Termination of the injection flow to stop primary to secondary leakage	Emergency core cooling system injection flow is stopped to terminate primary-to-secondary leakage.	If operating, the chemical volume control system makeup flow is stopped to terminate primary-to-secondary leakage.

Since the major actions are similar and the equipment available is comparable, the AP1000 would behave in a manner comparable to a standard Westinghouse PWR.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.121

Question:

Page 19E-4 documents information related to RCS temperature detectors for shutdown conditions.

Describe how these detectors are used during shutdown, mid-loop, and accident conditions.

Westinghouse Response:

The RCS has two safety-related, wide-range, thermowell-mounted temperature detectors, one in each hot leg. These detectors are mounted below the mid-plane of the hot leg piping to provide information to the operator during all operating modes including mid-loop operation. They are wide range detectors so that they can indicate the full range of RCS temperatures from shutdown through power operation.

For shutdown conditions including mid-loop operations, temperature indication from the Normal Residual Heat Removal System (RNS) would be used; the RCS wide-range detectors provide a backup to the RNS temperature indication.

In the event of RNS failure the RNS detectors would be ineffective, so these hot-leg detectors could be used as an indication of core conditions under some circumstances.

Although these RCS detectors are safety-related and are connected to the Qualified Data Processing System, they do not form the basis for any post-accident operations.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.123

Question:

Section 6.7 of NUREG-1449, "Shutdown and Low Power Operation at Commercial Nuclear Plant in the United States," describes instances in which the failure of temporary RCS boundaries (such as freeze seal, which is used to temporarily isolate fluid systems and, temporary plugs for neutron instrument housing) can lead to a rapid non-isolable loss of reactor coolant.

Address this concern with respect to failure of temporary boundaries in the AP1000.

Westinghouse Response:

The AP1000 passive safety-related systems provide the safety-related means for protecting the plant during all modes of operation including shutdown and refueling. The AP1000 includes design features to mitigate accidents that occur at low pressures. The passive safety-related systems are designed to either automatically mitigate events that occur during shutdown, or are available for manual actuation. The AP1000 technical specifications identify when the various portions of the passive safety-related systems are required to be available.

Portions of the passive systems are required to be operable during modes of operation until mode 6, when the refueling cavity has been flooded, and the upper internals have been removed. At this point in time, the stored heat capacity of the water in the refueling cavity is sufficient to maintain the reactor in a safe condition for an extended period of time without need for operator recovery actions in case of a loss of decay heat removal. In all other modes, the availability of the passive safety-related systems is maintained via technical specifications and the safety-related systems' redundant design features.

The availability of the passive safety-related systems during shutdown as described above significantly reduces the risk associated with failures of temporary RCS boundaries. In addition, the following AP1000 design features reduce the risks associated with temporary RCS boundaries:

- Steam Generator nozzle dams - the AP1000 steam generator nozzle dams are classified as AP1000 Equipment Class C so that the design, manufacture, installation, and inspection of this boundary (when installed) are controlled by the following requirements: 10CFR21; 10CFR50, Appendix B; Regulatory Guide 1.26 Quality Group C and ASME Boiler and Pressure Vessel Code, Section III, Class 3. In addition, this pressure boundary is classified as Seismic Category I so that it is protected from failure following a safe shutdown earthquake.

- Elimination of temporary plugs for nuclear instrumentation - The AP1000 does not contain bottom mounted instrumentation that require temporary plugging during shutdown and refueling. The AP1000 utilizes a fixed incore system with penetrations through the top head, rather than the bottom head.

- Current plants remove the excore detectors from above the excore housings through the floor of the refueling cavity. During refueling operations, these holes are plugged to facilitate flooding of the refueling cavity. The AP1000 has eliminated these temporary plugs by designing the excore instrumentation to be inserted from below the excore housings.

- Reduced reliance on freeze seals - The AP1000 has reduced the potential applications for freeze seals by reducing the number of lines that connect to the RCS and by providing the ability to perform operability tests on many valves that connect to the reactor coolant pressure boundary. This improved inservice testing reduces the requirements for disassembly of reactor coolant pressure boundary valves to test their operability. The use of freeze seals during a forced outage will typically occur in cold shutdown (Mode 5). During Mode 5 the passive core cooling system (PXS) is required to be available, and therefore the PXS could respond to at loss of coolant through a failed freeze seal.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.124

Question:

Current plants use temporary reactor cavity seals to flood the refueling cavities. Failure of these seals can divert water to the reactor pit, and subsequently to the reactor drains, and may result in a loss of shielding and fuel cooling during spent fuel assembly movement.

Address the ability to quickly move and safely store fuel assemblies during a seal failure event.

Westinghouse Response:

While current plants incorporate a pneumatic type of seal between the vessel flange and the refueling cavity floor, the AP1000 has incorporated a permanently welded seal ring in this location. This refueling cavity seal is part of the refueling cavity and is seismic Class I, and is illustrated in AP1000 DCD / PRA Figure 19E.2-3. The cavity seal is designed to accommodate the thermal transients associated with the reactor vessel flange.

The AP1000 permanent seal eliminates the failure mechanism that exists with temporary seals for some current plants.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.126

Question:

NRC Information Notice (IN) 92-54, "Level Instrumentation Inaccuracies Caused by Rapid Depressurization," describes the issue related to potential problems of noncondensible gases in hot-leg level instrument lines. The applicant indicates on page 19E-3 that the IN 92-54 issue has been addressed in the layout of the instrument lines.

Identify the features in the level instrument lines that are designed to address the IN 92-54 issue and show the issue is satisfactorily resolved.

Westinghouse Response:

The concerns raised in the NRC Information Notice deal with level errors resulting from a rapid depressurization of the reactor coolant system which causes the dissolution of non-condensible gasses from the liquid into the instrument lines. To minimize issues related to non-condensible gases, the hot-leg level instrument lines are downward sloping from the hot leg, the length of the lines are minimized, and the lines do not include large condensing pots (where non-condensible gases would concentrate). Also, the hot leg instrumentation is provided primarily for shutdown operations when the RCS is already at low pressure. During these conditions, there are low levels of dissolved gases in the fluid in the instrument lines, and therefore the quantity of non-condensible gases which could be released is small. Therefore, the accuracy of the hot leg level measurement is not significantly affected by non-condensible gases during the periods of their intended use.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.132

Question:

Figure 5.1.2-37 of WCAP-14252 provides IRWST injection rates for OSU test SB18. Please provide the corresponding predictions for IRWST injection initiation times and flow rate from WCOBRA/TRAC. Were IRWST flow rates predicted by the code or input from the test data?

Westinghouse Response:

The purpose of the OSU Test SB18 simulation presented in Section 2.3 of WCAP-15833 is to predict the depressurization of the test facility from the initiation of ADS Stage 4 flow until the time at which IRWST injection began during the test. The WCOBRA/TRAC simulation is terminated at the time at which IRWST injection occurred in Test SB18, and flow from the IRWST is not modeled. The WCOBRA/TRAC simulation of Test SB18 provides a reasonable prediction of the ADS-4 depressurization behavior observed during the test.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.134

Question:

Section 3.1.14 states that ADS 1-4 and broken pipe boundaries are described using WCOBRA/TRAC BREAK components. The WCOBRA/TRAC Users Manual is referenced. The Users Manual describes BREAK components as a pressure boundary condition and provides a number of user options. Please describe and justify all the options used. Provide a comparison of the methodology used by WCOBRA/TRAC to that used by NOTRUMP. Provide rationale for the greater ADS-4 depressurization calculated by WCOBRA/TRAC in comparison to NOTRUMP. See Figures 3-9 and 3-18.

Westinghouse Response:

In the AP1000 WCOBRA/TRAC methodology, BREAK components are used to provide the appropriate boundary conditions for the ADS-4 IRWST initiation phase transient. [

.]^{a,c}

BREAK components using break table option 0 provide the containment boundary conditions at the downstream junctions of the ADS Stage 4 valve components and the DEDVI break junction. The parameters specified are essentially constant during the ADS-4 IRWST initiation phase, so break table option 0 may be used.

BREAK components using break table option 0 provide the boundary conditions at the downstream junctions of the ADS Stage 1-3 valve components. The parameters specified are essentially constant during the ADS-4 IRWST initiation phase, so break table option 0 may be used. The pressure at this location is specified at the value equal to the containment pressure plus the static head of liquid inside the IRWST above the sparger elevation at the time of ADS Stage 4 actuation.

BREAK components provide the pressure and temperature of IRWST water at junctions entering the DVI piping in the AP1000 simulations. The BREAK component pressure input to WCOBRA/TRAC equals containment pressure plus the static head of liquid present in the IRWST for these locations. A BREAK component using break table option 1 is used to specify the boundary pressure for the IRWST at the junction with the intact DEDVI piping for the DEDVI break. The variable pressure capability of option 1 is used to model the reduction in IRWST level that occurs due to spilling of the tank through the broken pipe during the course of the ADS-4 IRWST initiation phase of the DEDVI break transient. For the Inadvertent ADS scenario, the IRWST level change is minimal during the ADS-4 IRWST initiation phase transient, so break table option 0 is used to provide constant boundary conditions.

Figure 3-8 of WCAP-15833 presents the WCOBRA/TRAC nodalization used to initialize ADS-4 IRWST initiation phase simulations to desired initial conditions. A BREAK component using break table option 0 provides the boundary conditions for component 93 in the initial WCOBRA/TRAC runs; the values used in this BREAK component input are adjusted in order to establish the desired conditions within WCOBRA/TRAC at the time of ADS-4 actuation.

The NOTRUMP computer code uses "boundary fluid nodes" to accomplish the same purposes that the BREAK components do in WCOBRA/TRAC. Interior fluid nodes are connected by flow links to boundary fluid nodes when the imposition of a given boundary condition on the problem is desired. [

]^{a,c}

The ADS-4 depressurization predicted by WCOBRA/TRAC is more effective than that predicted by NOTRUMP because it is based on a best estimate prediction of the flow through the ADS-4 flow paths, whereas the NOTRUMP prediction is biased to predict conservatively low flow rates. During initial operation of ADS Stage 4, which begins at approximately a pressure of 100 psia in both the DEDVI break case and the Inadvertent ADS actuation scenario, the two codes predict very similar depressurization rates in the referenced figures until a pressure of approximately 50 psia is achieved; at that point the predicted pressures deviate. The WCOBRA/TRAC computation uses the critical flow model as described in Section 4.3 of Reference 440.134-1. It merges into the TRAC subcritical flow model when upstream fluid conditions warrant; the ADS Stage 4 flow prediction is a best estimate representation for either flow condition.

In contrast, the NOTRUMP calculation of the ADS Stage 4 flow rate is intentionally conservative at pressures where critical flow approaches the subcritical. The effect of this has been observed in the overprediction of OSU APEX facility pressures reported in Reference 440.134-2 at the time directly preceding and following actuation of ADS Stage 4. In the AP1000 simulation, the deviation between the two codes begins when [

,]^{a,c} the resistance in NOTRUMP is increased as described in the AP1000 DCD to accommodate the lack of a momentum flux model in the code, and to compensate for the lack of entrainment models in NOTRUMP. The NOTRUMP bias to low flow prediction through the ADS-4 flow paths reduces the calculated AP1000 depressurization capability and allows the system mass inventory to continue to deplete longer than predicted by WCOBRA/TRAC.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

References:

440.134-1: WCAP-14776, Revision 4, *WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report,* 1998.

440.134-2: WCAP-14807, Revision 5, *NOTRUMP Final Validation Report for AP600,* 1998.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.135

Question:

Section 3.1.15 states that the initial conditions used in the WCOBRA/TRAC simulations of AP1000 are the NOTRUMP values at the time of ADS-4 initiation. Figures 3-16 and 3-25 show that WCOBRA/TRAC was initialized at larger values of reactor vessel mass than predicted by NOTRUMP. Please discuss the reasons for this discrepancy.

Westinghouse Response:

The initialization of WCOBRA/TRAC was performed with a tolerance of difference between the calculated and desired values similar to the WCOBRA/TRAC established practice for large break LOCA analysis steady-state cases. The values of reactor vessel mass inventory are in fact very close at the time of ADS-4 actuation. For the Inadvertent ADS scenario, the NOTRUMP reactor vessel mass inventory at the time ADS-4 actuates is 110,000 lbm, and the WCOBRA/TRAC calculated value equals 109,400 lbm. For the DEDVI line break case, the NOTRUMP reactor vessel mass inventory at the time ADS-4 actuates is 93,250 lbm, and the WCOBRA/TRAC calculated value equals 94,600 lbm. Thus, the WCOBRA/TRAC calculated value is within a few per cent of the desired value from NOTRUMP in each case, and this is judged to be adequately close agreement.

What the referenced figures indicate is an adjustment in mass inventory introduced in plotting to facilitate the comparison between WCOBRA/TRAC and NOTRUMP predicted mass inventories at the time of minimum inventory on an equivalent basis. The NOTRUMP input methodology [

]^{a,c} values to obtain a plotted comparison that indicates the equivalent mass present in the vessel providing margin to core uncovery at the time of minimum inventory during the AP1000 ADS-4 IRWST initiation phase transient.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.136

Question:

Section 1.3, "Figure Legend," of Tier 1 Information provides a list of conventions used in the Tier 1 figures, which are somewhat different from the conventions described in Section 1.7 of Tier 2 Information for the Tier 2 Piping and Instrumentation Diagrams (P&ID). Also, some Tier 1 figures actually use Tier 2 conventions which are not defined in Tier 1 Section 1.3. For example, the conventions for the air-operated valve or pneumatic operator are different between Tier 1 and Tier 2; and, in Figure 2.3.6-1, "Normal Residual Heat Removal System (RNS)," the RNS pump mini-flow air-operated isolation valves, RNS-PL-V057A and -V057B, use the Tier 2 convention, which is not defined in Tier 1.

Explain why is it necessary to use different conventions for Tier 1 and Tier 2 information, respectively, and make corrections to the Tier 1 figures to ensure they are consistent with the Tier 2 conventions.

Westinghouse Response:

The conventions are different for the AP1000 DCD Tier 1 figures than those for the DCD Tier 2 figures because the level of detail which is to be committed to the ITAAC process is significantly less than that provided in the Tier 2 information. As a result, the Tier 1 figures are generally sketches intended to show important component design features or general system arrangements compared to more detailed component and system drawings in Tier 2. Therefore, the conventions are different in order to provide only the needed information for Tier 1. For example, in Tier 1 we did not intend to distinguish between the various types of pneumatically operated valves.

Tier 1 Figure 2.3.6-1 will be revised to incorporate the Tier 1 conventions.

Design Control Document (DCD) Revision:

From DCD Tier 1 page 2.3.6-17, Figure 2.3.6-1:

See attached figures showing changes to the RNS pump mini-flow isolation valve operators.

PRA Revision:

None

Figure From AP1000 DCD Revision 2 Tier 1

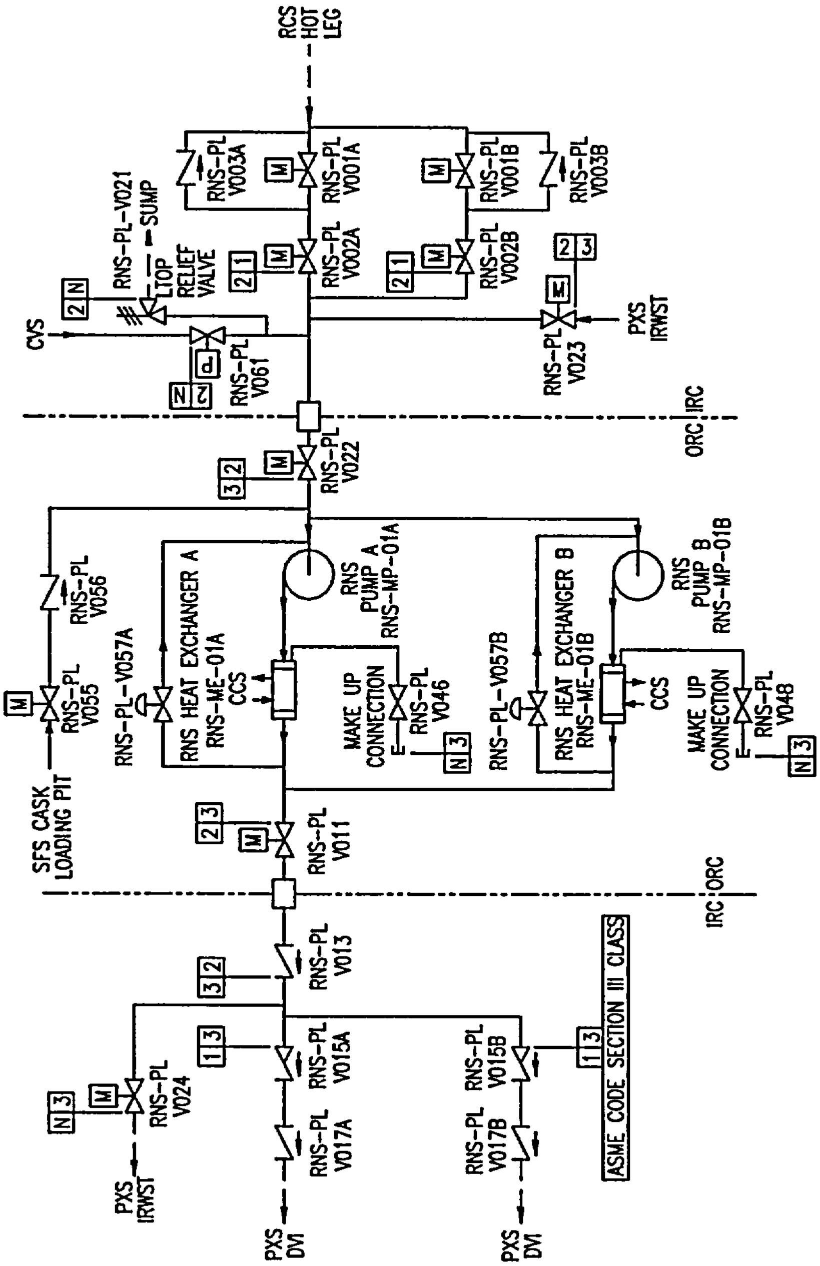


Figure 2.3.6-1

Normal Residual Heat Removal System

Corrected Figure

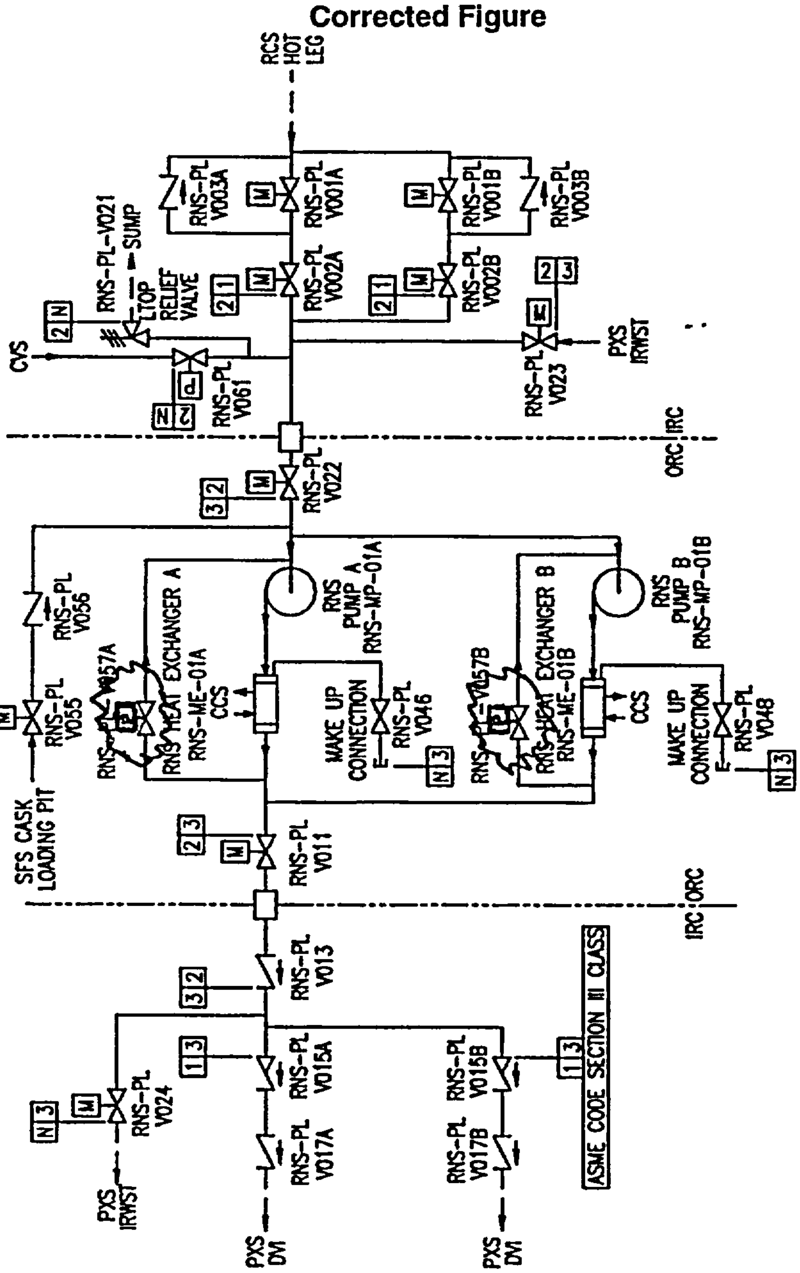


Figure 2.3.6-1

Normal Residual Heat Removal System

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.137

Question:

Many figures in Tier 1 Information show valves status inconsistent with the normal positions of the design arrangement shown in the Tier 2 P&IDs. For example, in Tier 1 Figure 2.1.2-1, "Reactor Coolant System," the pressurizer safety valves, the pressurizer spray valves, the reactor vessel heat vent valves, the ADS stages 1, 2, and 3 isolation valves and depressurization control valves, and the ADS-4 squib valves, which are all normally closed, are indicated as open. In Figure 2.2.3-1, "Passive Core Cooling System," the air-operated valves on the PRHR outlet line and the CMT injection lines, and the squib valves on the ADS stage 4 discharge lines, the IRWST Injection lines, and containment recirculation lines, which are all normally closed, are shown as open.

Explain why the Tier 1 Information figures do not show the valves' normal positions, and make revisions as necessary to be consistent with the Tier 2 P&IDs.

Westinghouse Response:

The Tier 1 information presents only that information that is necessary for the COL applicant to demonstrate to the NRC that the plant, as constructed, is consistent with the design that was approved for Design Certification. The identification of the normal position of valves (i.e. open or close) is not necessary to determine that the plant has been properly constructed. The format of the AP1000 Tier 1 information is the same as the format of the AP600 Tier 1 information, and is similar in format to the other Certified designs. There is no intention to show valve position on figures provided in Tier 1.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.139

Question:

The reactor vessel head vent system (RVHVS) valves, described in Tier 2 Section 5.4.12, are used to remove noncondensable gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or impaired natural circulation through the steam generators resulting from the accumulation of noncondensable gases in the RCS. The design of the RVHVS is in accordance with the requirements of 10 CFR 50.34(f)(2)(vi).

Explain why there is no design description in Tier 1 Information, Section 2.1.2, and design commitment in the ITAAC Table 2.1.2-4 regarding the RVHVS valves.

Westinghouse Response:

The safety related function of the reactor vessel head vent system valves is included in the Design Description of Section 2.1.2 as item 8e (DCD Tier 1 page 2.1.2-2). The function of the valves is to provide emergency letdown following an accident. As ASME Code Section III components, the valves are also included in Table 2.1.2-1 (Tier 1 page 2.1.2-7). The head vent valves are shown in the RCS flow schematic in Figure 2.1.2-1 (Tier 1 page 2.1.2-30).

The design commitment for the reactor vessel head vent system is included in Table 2.1.2-4 as commitment 8e (Tier 1 page 2.1.2-24). The design commitment is that the capacity of the head vent system is sufficient to pass not less than 8.2 lbs/sec at an RCS pressure of 1250 psia.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.140

Question:

In ITAAC Table 2.1.2-4, Item 8d, the acceptance criteria for the automatic depressurization system (ADS) design are as follows:

- The calculated ADS piping flow resistance from the pressurizer through the sparger with all ADS Stages 1-3 valves of each group open is $\le$ 2.92E-6 ft/gpm^2, and the calculated flow resistance for each group of Stage 4 valves and piping is $\le$ 1.71E-7 ft/gpm^2.
- The effective flow areas through Stage 1, 2, 3, and 4 valves are $\ge$ 4.6 square inches (in^2), $\ge$ 21 in^2, $\ge$21 in^2, and $\ge$67 in^2, respectively.

A. Describe how the ADS piping flow resistance acceptance criteria are determined and explain if they are consistent with the assumptions in the design basis accident analyses.
B. Describe how the ADS effective flow areas acceptance criteria are determined, and explain if they are consistent with the ADS valve design of 4", 8", 8", and 14", for Stages 1, 2, 3, and 4 valves, respectively, as well as consistent with the safety analysis input.

Westinghouse Response:

(A) The ADS flow resistances are determined by calculations based on the AP1000 line routings and valve L/D requirements. The same maximum flow resistances are used for both the DCD safety analysis and the ITAAC acceptance criteria.

(B) The ADS effective flow areas were established through discussions with valve vendors; the discussions included valve size and valve body type. These flow areas were confirmed to be sufficient by the DCD safety analysis. The same effective flow areas are contained in the valve specifications that will be used to procure these valves.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.141

Question:

ITAAC Table 2.1.2-4, Item 9.a shows the acceptance criteria for the RCS flow as "the calculated post-fuel load RCS flow rate ≥ 296,000 gpm," with the flow measurement uncertainties accounted for. As shown in Tier 2 Table 5.1-3, 296,000 gpm is the RCS thermal design flow with 10 percent SG tube plugging. Without SG tube plugging, the RCS thermal design flow is 299,880 gpm.

Provide a clarification in the ITAAC RCS flow rate acceptance criteria with respect to the SG tube plugging condition.

Westinghouse Response:

The ITAAC RCS flow rate acceptance criteria is consistent with the 10% SG tube plugging value identified in Table 5.1-3. This allows for the possibility that the plant, as built, may have some SG tubes plugged. This value is consistent with the minimum RCS flow rate assumed in the Chapter 15 accident analysis. This approach is consistent with the approach taken for the AP600.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.142

Question:

The AP1000 reactor core consists of 157 fuel assemblies (FA), 53 rod control cluster assemblies (RCCA), 16 gray rod cluster assemblies (GRCA) which are used in load follow maneuvering, and 69 control rod drive mechanisms (CRDM). Tier 1 Information Section 2.1.3 provides the design description and the Design Commitment (in Table 2.1.3-2) of the FAs , RCCAs and CRDMs. Tier 1 Tables 2.1.3-1and 2.1.3-3 list the tag numbers, American Society of Mechanical Engineers (ASME) Code Section III classification, Seismic categories, and locations of the FAs, RCCAs, and CRDMs.

A. Explain why the GRCAs are not included in the Tier 1 Table 2.1.3-2 design commitment to be designed and constructed in accordance with the principal design requirements.

B. Explain why the tag numbers and locations for the GRCAs are not included in Tier 1 Tables 2.1.3-1 and 2.1.3-3.

Westinghouse Response:

A. GRCA information will be included in Tier 1 Table 2.1.3-2 in Revision 3 of the AP1000 DCD.

B. GRCA information will be included in Tier 1 Tables 2.1.3-1 and 2.1.3-3 in Revision 3 of the AP1000 DCD.

Design Control Document (DCD) Revision:

From DCD Tier 1 Table 2.1.3-1, page 2.1.3-4:

Rod Cluster Control Assemblies (RCCAs) (minimum 53 locations)	RXS-FR-B06/B10/C05/C07/C09/C11/D06/D08/D10/E03/E05/E07/E09/E11/E13/F02/F04/F12/F14/G03/G05/G07/G09/G11/G13/H04/H08/H12/J03/J05/J07/J09/J11/J13/K02/K04/K12/K14/L03/L05/L07/L09/L11/L13/M06/M08/M10/N05/N07/N09/N11/P06/P10	No[1]	Yes	-	-
Grey Rod Control Assemblies (GRCAs) (16 locations)	**RXS-FG-A07/C03/C11/E05/E07/E09/G01/G05/G09/G13/J05/J07/J09/L03/L11/N07**	No[1]	Yes	-	-
Control Rod Drive Mechanisms (CRDMs) (69 Locations)	RXS-MV-11B06/11B08/11B10/11C05/11C07/11C09/11C11/11D04/11D06/11D08/11D10/11D12/11E03/11E05/11E07/11E09/11E11/11E13/11F02/11F04/11F06/11F08/11F10/11F12/11F14/11G03/11G05/11G07/11G09/11G11/11G13/11H02/11H04/11H06/11H08/11H10/11H12/11H14/11J03/11J05/11J07/11J09/11J11/11J13/11K02/11K04/11K06/11K08/11K10/11K12/11K14/11L03/11L05/11L07/11L09/11L11/11L13/11M04/	Yes	Yes	No/No	No

Note: Dash (-) indicates not applicable.
1. Manufacture standard, but uses ASME Section III guidelines

From DCD Tier 1 Table 2.1.3-2, page 2.1.3-6:

2.b) The ~~rod cluster~~ control **assemblies (rod cluster and grey rod)** and drive rod arrangement is as shown in Figure 2.1.3-2.	Inspection of the as-built system will be performed.	The as-built RXS will accommodate the ~~rod cluster~~ control assemblies **(rod cluster and grey rod)** and drive rod arrangement shown in Figure 2.1.3-2.

From DCD Tier 1 Table 2.1.3-3, page 2.1.3-10:

Rod Cluster Control Assemblies (RCCAs) (minimum 53 locations)	RXS-FR-B06/B10/C05/C07/ C09/C11/D06/D08/D10/E03/ E05/E07/E09/E11/E13/F02/F04/ F12/F14/G03/G05/G07/G09/ G11/G13/H04/H08/H12/J03/ J05/J07/J09/J11/J13/K02/K04/ K12/K14/L03/L05/L07/L09/ L11/L13/M06/M08/M10/N05/ N07/N09/N11/P06/P10	Containment
Grey Rod Control Assemblies (GRCAs) (16 locations)	**RXS-FG-A07/C03/C11/E05/ E07/E09/G01/G05/G09/G13/J0 5/J07/J09/L03/L11/N07**	**Containment**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.143

Question:

In Tier 1 ITAAC Table 2.2.3-4, Item 8.c, states that a low-pressure injection test and analysis for each CMT, accumulator, IRWST injection line, and containment recirculation line, as well as the CMT cold leg balance line, will be conducted. The acceptance criteria for each injection line or pressure balance line flow resistance (R) from each source are specified as follows:

CMT Injection Line:	$1.80 \times 10^{-5} \leq R \leq 2.26 \times 10^{-5}$ ft/gpm^2
Accumulator Injection Line:	$1.47 \times 10^{-5} \leq R \leq 1.83 \times 10^{-5}$ ft/gpm^2
IRWST Injection Line A:	$5.52 \times 10^{-6} \leq R \leq 9.21 \times 10^{-6}$ ft/gpm^2
IRWST Injection Line B:	$6.20 \times 10^{-6} \leq R \leq 1.04 \times 10^{-5}$ ft/gpm^2
Containment Recirculation Line A:	$R \leq 1.12 \times 10^{-5}$ ft/gpm^2
Containment Recirculation Line B:	$R \leq 1.04 \times 10^{-5}$ ft/gpm^2
CMT Cold Leg Balance Line:	$R \leq 7.22 \times 10^{-6}$ ft/gpm^2

Describe how these acceptance criteria are derived, and how they are consistent with the assumptions used in the safety analyses.

Westinghouse Response:

The flow resistances of these lines are determined by calculations based on the AP1000 line routings and valve L/D requirements. The line resistances including flow tuning orifices (where applicable) were shown to be acceptable with the DCD safety analysis. The same minimum (where applicable) and maximum flow resistances are used for both the DCD safety analysis and the ITAAC acceptance criteria.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.144

Question:

For Item 8.b in Tier 1 ITAAC Table 2.2.3-4, regarding the passive residual heat removal (PRHR) heat exchanger (HX) capability, the "Inspections, Tests, Analyses" column describes the conditions under which the PRHR HX heat removal test will be conducted, i.e., the hot leg temperature initially at ≥ 540°F with the reactor coolant pumps stopped, and continues until the hot leg temperature decreased below 420°F. The acceptance criteria for the PRHR HX heat transfer rate is identified as "TBD Btu/hr with 520°F HL Temp and 120°F IRWST temperatures."

A. Since the IRWST water temperature can increase to the boiling temperature during the PRHR HX operation, why is it that the acceptance criteria for the PRHR HX heat transfer rate is specified for only the initial phase of the test, but not the end of the test condition? Why is one criterion sufficient to demonstrate the PRHR HX heat transfer capacity.

B. When will the PRHR HX heat removal acceptance criteria be determined?

Westinghouse Response:

(A) The purpose of the PRHR HX ITAAC is to verify that the capacity of the PRHR HX installed in the plant is consistent with the capacity of the HX used in the safety analysis. Showing that the HX can remove a minimum amount of heat at one typical safety operating point verifies that the basic HX characteristics have been built and installed properly. Once it has been shown that the HX can remove the required amount of heat at one operating point, its operation at other conditions can be calculated.

This approach is the same as is used for many other system parameters such as critical flow capability of a valve.

(B) The PRHR HX heat removal acceptance criteria is shown in the current revision of the DCD as follows:

> A report exists and concludes that the PRHR HX heat transfer rate with the design basis number of PRHR HX tubes plugged is:
>
> ≥ 1.78×10^8 Btu/hr with 520°F HL Temp and 80°F IRWST temperatures.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.145

Question:

In Tier 1 ITAAC Table 2.2.4-4, the acceptance criteria for the main steam safety valve (MSSV) (Item 8.a) is shown as 8,300,000 lb/hr per SG, and the acceptance criteria for the main steam power-operated relief valve (PORV) (Item 9.b) is 300,000 lb/hr at 1106 psia ± 10 psi.

Explain how these acceptance criteria are consistent with the design data shown in Tier 2 Table 10.3.2-1, where the PORV design capacity is shown to be 70,000 lb/hr at 100 psia inlet pressure, and 1,020,000 lb/hr at 1200 psia inlet pressure, and Tier 2 Table 10.3.2-2, where the MSSV relieving capacity is shown to be 8,340,000 lb/hr per steam line at 110 percent design pressure.

Westinghouse Response:

The acceptance criteria for the MSSV valve, item 8.a of Tier 1 ITAAC Table 2.2.4-4, which is presently shown as "8,300,000" lb/hr per SG, should be "8,340,000". The ITAAC will be revised so that it is consistent with the value shown in Tier 2 Table 10.3.2-2 (and Table 14.3-2).

The acceptance criteria for the PORV valve, item 9.b of Tier 1 ITAAC Table 2.2.4-4, which is shown as "300,000" lb/hr at 1106 psia ± 10 psi is consistent with the value shown in Tier 2 subsection 10.3.4.1.1. As stated in subsection 10.3.4.1.1, a relief capacity of at least 300,000 lb/hr at 1106 psia is necessary in order to satisfy the non-safety related function of decay heat removal. The valve also performs the function of minimizing challenges to the MSSV valve, which requires the capacity of the PORV hardware to be larger than the value needed to satisfy the non-safety decay heat removal function. Tier 2 Table 10.3.2-1 reflects the capacity of the PORV hardware. The function of protecting the main steam line from over pressure is included in the ITAAC by the MSSV capacity.

Design Control Document (DCD) Revision:

Revise Tier 1 ITAAC Table 2.2.4-4 item 8.a Acceptance Criteria i) as follows:

8.a) The SGS provides a heat sink for the RCS and provides overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	i) Inspections will be conducted to confirm that the value of the vendor code plate rating of the steam generator safety valves is greater than or equal to system relief requirements. ii) Testing and analyses in accordance with ASME Code Section III will be performed to determine set pressure.	i) The sum of the rated capacities recorded on the valve vendor code plates of the steam generator safety valves exceeds ~~8,300,000~~ **8,340,000** lb/hr per steam generator. ii) A report exists to indicate the set pressure of the valves is less than 1305 psig.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.146

Question:

Tier 1 Table 2.3.6-1 provides a list of normal residual heat removal system (RNS) components and quality requirements, including, among others, the RNS heat exchanger (HX) -A channel head drain valve RNS-PL-V046.

A. Explain why Table 2.3.6-1 does not include the HX-B channel head drain valve RNS-PL-V048.

B. Explain why the cask load pit isolation valve RNS-PL-V055, and RNS pump mini-flow air-operated isolation valves RNS-PL-V057A and -V057B do not have valve position indication.

Westinghouse Response:

A. If containment leakage occurs after an accident, makeup to the containment is necessary to maintain core cooling in the long term. The RNS heat exchanger channel head drain line (including drain valve RNS-PL-V046) is a safety-related flow path which can be used to provide makeup water to containment using temporary or portable equipment. A single flow path is sufficient to provide the makeup water necessary to maintain core cooling, and therefore, Tier 1 Table 2.3.6-1 includes only the flow path associated with heat-exchanger A, and not heat exchanger-B channel head drain valve RNS-PL-V048.

B. Tier 1 Table 2.3.6-1 indicates only safety-related displays. The cask loading pit isolation valve and the RNS pump mini-flow air-operated isolation valves do not perform safety-related functions and therefore their position indication is not included under the safety-related display column. Position indication for these valves would be available in the main control room through the Plant Control System and Data Display and Processing System.

DCD Tier 1 Table 2.3.6-3 will be revised to show that position indication for valves RNS-PL-V055, RNS-PL-V057A, and RNS-PL-V057B will be displayed in the main control room.

Design Control Document (DCD) Revision:

From DCD Tier 1 page 2.3.6-9:

Table 2.3.6-3			
Equipment Name	**Tag No.**	**Display**	**Control Function**
RNS Pump 1A (Motor)	RNS-MP-01A	Yes (Run Status)	Start
RNS Pump 1B (Motor)	RNS-MP-01B	Yes (Run Status)	Start
RNS Flow Sensor	RNS-01A	Yes	-
RNS Flow Sensor	RNS-01B	Yes	-
RNS Suction from Cask Loading Pit Isolation Valve (Position Indicator)	**RNS-PL-V055**	**Yes**	**-**
RNS Pump Miniflow Isolation Valve (Position Indicator)	**RNS-PL-V057A**	**Yes**	**-**
RNS Pump Miniflow Isolation Valve (Position Indicator)	**RNS-PL-V057B**	**Yes**	**-**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.149

Question:

Please clarify the following:

(a) In the Equation Nomenclature, on page xii of WCAP-15833, "Re" is defined as a Reynolds number. However, on page 2-10, it is used as an entrainment source. Please revise the Nomenclature as appropriate.

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

Westinghouse Response:

Response to part (a):

The nomenclature on page xii of WCAP-15833 will be revised so that "E" (instead of "Re") will represent the entrainment flux in Equation (2-19). "Re" will remain as the nomenclature for Reynolds number.

Response to part (b):

Equation (2-22) of WCAP-15833 that represents the terminal velocity of droplets in turbulent flow contains typographical errors. The equation will be corrected to include an equal sign ("=") and a numerical constant of "1.7" as follows:

$$V_{s,2} = 1.7\sqrt{\frac{D_e(\rho_l - \rho_v)g}{\rho_v}}$$

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

WCAP-15833 will be revised as shown in the attached.

EQUATION NOMENCLATURE

E	Entrainment Flux
A	Area
C_o	Distribution parameter
d	Offtake diameter
D	Pipe diameter
D_H	Hydraulic diameter
E_{fg}	Entrainment ratio
F_r	Froude Number
g	Gravitational acceleration
G	Mass flux
h	Enthalpy, or height
h_L	Mixture level
H	Heat Transfer coefficient
i	Interfacial
j	Superficial velocity
k	Thermal conductivity, or unit vector indicating flow direction
K	Interfacial friction factor
N_μ	Viscosity number
P	Pressure
P_f	Friction perimeter
P_H	Heated perimeter
P_r	Prandtl number
q''	Heat flux
R	Resistance, hydraulic
R_e	Reynolds number
T	Temperature
V_s	Settling velocity
x	Vertical direction, Cartesian coordinates, or flow quality
X	Phasic pressure drop ratio in two-phase flow
U	Vertical velocity component, Subchannel coordinates
V_{gj}	Drift velocity
z	Transverse direction, Cartesian coordinates, or elevation
Z	Transverse direction, Subchannel Coordinates

Greek

ρ	Density
δ	Angle
ν	Kinematic viscosity
α	Void fraction
σ	Surface tension
μ	Viscosity
τ	Interfacial shear stress
Δ	Delta, or difference
ϕ	Phase
Φ	2-phase multiplier

Subscripts

b	Bubble
g	Saturated vapor
ℓ	Liquid field, or subcooled liquid
v	Vapor field
f	Saturated liquid
fo	Liquid only
m	Mixture
sat	Saturated
sub	Subcooled
w	Wall

Superscripts

e	Entrained field
k	Continuous phase
x	Vertical direction, Cartesian Coordinate
S	Superficial
*	Indicates dimensionless quantity
-	Indicates average quantity
^	Indicates vector quantity

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2.2.1.4 Entrainment in Horizontal Stratified Flow

Model Basis

When horizontal stratification is identified, the Ishii-Grolmes (Reference 2) criteria are checked; if the criteria are satisfied, the calculation of entrainment off of the horizontal surface is enabled.

Ishii and Grolmes describe entrainment in horizontal cocurrent flow as the stripping of drops from the top of waves. They describe four mechanisms, but the shearing off of the top of roll waves by turbulent gas flow is expected to be significant for the ADS-4 IRWST initiation. Ishii and Grolmes state that this mechanism is valid for liquid Re > 160 in horizontal concurrent flow. For roll wave entrainment, Ishii and Grolmes provide two correlations based upon Re:

For Re > 1635:

$$\frac{\mu_\ell U_g}{\sigma}\sqrt{\frac{\rho_g}{\rho_\ell}} \geq N_\mu^{0.8} \text{ for } N_\mu < \frac{1}{15}$$

$$\frac{\mu_\ell U_g}{\sigma}\sqrt{\frac{\rho_g}{\rho_\ell}} \geq 0.1146 \text{ for } N_\mu < \frac{1}{15}$$

For Re < 1635:

$$\frac{\mu_\ell U_g}{\sigma}\sqrt{\frac{\rho_g}{\rho_\ell}} \geq 11.78 N_\mu^{0.8}\, Re_\ell^{-1/3} \text{ for } N_\mu < \frac{1}{15}$$

$$\frac{\mu_\ell U_g}{\sigma}\sqrt{\frac{\rho_g}{\rho_\ell}} \geq 1.35 Re_\ell^{-1/3} \text{ for } N_\mu < \frac{1}{15}$$

Re is based upon liquid film thickness, U_g is the minimum gas velocity for entrainment to occur, and N_μ represents viscosity number.

The entrainment source term in the continuity cell is evaluated when the Ishii-Grolmes criteria are satisfied for gap flow connections according to the model used by Hanratty (Reference 12):

$$E\,R_e = K_a U_v \sqrt{\rho_v \rho_\ell}\ (\text{lb/s-ft}^2) \qquad (2\text{-}19)$$

where:

$K_a = 0.2$ is currently used.

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The size of the entrained droplets is determined by Tatterson's (Reference 13) model:

$$D_e = 0.0112\left(\frac{D_g \sigma}{0.5 f_i \rho_v U_v^2}\right)^{1/2} \tag{2-20}$$

This correlation is for vertical annular flow, and the characteristic length is the pipe diameter. It will be implemented here by assuming that the characteristic length is the hydraulic diameter (D_g) of the gap above the mixture elevation.

De-entrainment onto the interface is assumed to be dominated by the terminal velocity of the droplets. The settling velocity (V_s) is the minimum of the Stokes flow solution Equation 9.13 (Wallis, Reference 14):

$$V_{s,1} = \frac{1}{18}\frac{D_e^2 g(\rho_\ell - \rho_v)}{\mu_\ell} \tag{2-21}$$

and the turbulent flow solution Equation ~~12.29 (Wallis):~~

$$V_{s,2}\sqrt{\frac{D_e(\rho_\ell - \rho_v)g}{\rho_v}} \tag{2-22}$$

$$V_{s,2} = 1.7\sqrt{\frac{D_e(\rho_\ell - \rho_g)g}{\rho_v}}$$

where:

D_e is the average diameter of the entrained drops in the vapor above the mixture. The net flux of droplets into the mixture is:

$$R_{de} = \rho_\ell \alpha_e \left(V_s - U_{v,ver}\right) \tag{2-23}$$

where:

$U_{v,ver}$ is the average vertical vapor velocity above the mixture and $V_s = \min(V_{s,1}, V_{s,2})$.

Model as Coded

As previously described, the horizontal stratified flow model is activated [for individual gap flowpath calculations by input. When the model is activated, the gap flow conditions are used][a,c] to identify the flow regime according to the Taitel-Dukler flow regime map. The parameters used in the determination of the horizontal flow regime are the total liquid superficial velocity, total vapor superficial velocity, gap average vapor density, gap average liquid density, the vapor viscosity, liquid viscosity, total gap void fraction, hydraulic diameter of flow channel, and mixture level.

Within the structure of WCOBRA/TRAC, entrainment must be treated [in a continuity cell. The source term for entrainment is accumulated over all of the gaps connected to a given channel.][a,c] The

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AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.150

Question:

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

Westinghouse Response:

The Side Offtake Orientation model for predicting the onset of liquid entrainment is not applied in the analysis of AP1000. Only the Top Offtake Orientation described in Section 2.2.1.1 is applied in the analysis of AP1000; specifically, the Top Offtake Orientation is applied to the ADS-4 paths that vent off the top of the hot legs.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.167

Question:

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

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

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

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

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

or,

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

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

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

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

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

Scaling AP1000 and the APEX facility then,

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

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

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

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

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

Reference:

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

Westinghouse Response:

The correlations used in WCAP-15613 and WCAP-15833 to describe the onset of liquid entrainment from a vertical offtake have the same form in that they contain a Froude number (Fr) and a geometric ratio ($L_g/d_{offtake}$). The only difference between the correlations is the coefficient and exponent associated with the geometric ratio. For scaling, use of either set of coefficients and exponents provides acceptable results for APEX relative to AP1000. Either correlation is acceptable for AP1000 scaling.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 471.001

Question:

Discuss what affects the increase in reactor coolant activity (as discussed in Chapter 11.1 of the DCD, Tier 2) for the AP1000 will have on both operating and shutdown dose rates as compared with the dose rates for the AP600.

Westinghouse Response:

The reactor coolant activities discussed in Chapter 11.1 are based on the highly conservative assumption that the plant is operating with a large amount of fuel cladding defects (i.e. activity released from fuel rods generating 0.25% of the total core power). This activity is considered in establishing the conservative sources used in plant shielding design as listed in DCD Section 12.2. These sources are orders of magnitude higher than that experienced in operating plants since the reactor coolant fission product sources are generally limited to "tramp" activity and/or cladding defects in only a few fuel rods.

Dose rates during operation inside containment are dominated by the short-lived N-16 in the primary coolant and the neutron levels. Since limited access is required inside the containment at power, very little ORE is accrued from these sources. After shutdown the radiation fields due to N-16 and neutron sources do not exist and the dominant sources are the corrosion product deposits that build up on system surfaces.

Outside the containment the dose rates during operation are generally dominated by corrosion product activities in the reactor coolant or corrosion products deposits on component surfaces - except for areas near components that process radioactive gases and/or in the event of significant fuel defects. After shutdown, the coolant activities are reduced to low levels by processing through demineralizers and filters. However, as stated above, the design bases values are intentionally conservative such that operational problems are limited and personnel radiation exposure can be maintained ALARA.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 471.002

Question:

In Table 11.1-2 for the AP1000, the design-basis coolant activities increased (over the comparable values listed for the AP600) for all radionuclides except for the corrosion products (Cr-51, Mn-54, Mn-56, Fe-55, Fe-59, Co-58, and Co-60). Justify your reasons for not showing an increase in corrosion product activities for the AP1000. Assuming that the corrosion product concentrations for the AP1000 will be higher than those for the AP600, 1) provide the expected corrosion product activity levels for the AP1000 for each of the affected corrosion products, 2) state which plant areas will be affected by the resulting increase in dose rates and 3) describe what affects this increase in dose rates will have on occupancy levels in these areas.

Westinghouse Response:

The activity concentrations for the corrosion products are known to be complex functions of many plant parameters - including the plant chemistry regimes at which the plant is operated. In general, plants currently operate with coolant chemistries that are based on recommendations issued in EPRI Chemistry Guidelines documents (e.g., see *PWR Primary Water Chemistry Guidelines: Volume 1, Revision 4*, EPRI, Palo Alto, CA: 1999. TR-105714-V1R4.) The guidelines are "living documents" that change as new approaches to the reduction of the corrosion product source term evolve; thus making it likely that advanced plants operating under improved chemistry regimes will have lower dose rates from corrosion products than those observed in current operating units.

The approach that Westinghouse has taken in arriving at the design bases corrosion product activity levels for both AP600 and AP1000 is to use a set of values that are reasonably conservative relative to current operating plant experience. These values (for the four most important nuclides) are included in the attached table that includes data from representative Westinghouse PWR plants and illustrates the degree of conservatism that has been considered. As noted in the attached table, the design basis values exceed the average four-loop plant (which will have more reactor coolant system surface area than AP1000) measured values by factors in the range of 2 to 7 for the major corrosion product nuclides. Based on future improvements in plant operating chemistry that are anticipated along with the advanced plant design improvements that are intended to reduce the corrosion product sources (e.g, reduced cobalt impurity levels in materials and reduced number of components that constitute a source of corrosion product activities, such as valves and pumps, etc.) the design basis values are expected to apply to both the AP600 and AP1000 plant. There is no need to increase the values for the AP1000 plant design.

Table 1
Median Coolant Activities During Operation *

Plant	Cycle	Chemistry	Activity Concentration, uCi/ml			
			Co-58	Co-60	Mn-54	Cr-51
Callaway	8	Modified	1.07E-03	4.23E-05	7.31E-05	4.25E-04
Callaway	9	Varied	1.76E-03	4.24E-05	4.93E-05	
Callaway	10	Varied	8.99E-03	1.27E-04	5.22E-04	2.14E-04
Callaway	11	Increased BOL pH	1.11E-03	2.38E-05		
Comanche Peak 1	6	Modified	2.12E-04	1.46E-05	3.49E-05	1.31E-04
Comanche Peak 1	7	Increased BOL pH	1.00E-04	1.21E-05	1.14E-05	8.66E-05
Comanche Peak 1	8	Increased BOL pH	1.14E-04	1.06E-05		8.59E-05
Comanche Peak 2	3	Modified	4.35E-04	1.68E-05		
Comanche Peak 2	4	Increased BOL pH	2.65E-04	1.75E-05	1.08E-04	2.93E-05
Comanche Peak 2	5	Increased BOL pH	1.50E-04	1.19E-05	1.08E-04	
Seabrook	4	Modified	1.10E-03	2.37E-05	8.12E-05	1.80E-04
Seabrook	5	Modified	1.07E-03	2.75E-05	9.01E-05	2.02E-04
Seabrook	6	Increased BOL pH	5.57E-04	2.38E-05	8.26E-05	1.97E-04
Seabrook	7	Increased BOL pH	8.01E-04	2.01E-05	7.60E-05	1.80E-04
South Texas 1	6	Modified	6.48E-04	3.20E-05	8.55E-05	1.40E-03
South Texas 1	7	Modified	3.56E-04	4.30E-05	8.50E-05	8.88E-04
South Texas 1	8	Increased BOL pH	2.82E-04	3.20E-05	6.80E-05	1.47E-03
South Texas 1	9	Increased BOL pH	1.39E-04	2.05E-05	5.55E-05	
South Texas 2	5	Modified	6.66E-04	4.40E-05	8.70E-05	
South Texas 2	6	Modified	6.91E-04	4.90E-05	1.02E-04	
South Texas 2	7	Increased BOL pH	6.02E-04	4.00E-05	9.10E-05	
South Texas 2	8	Increased BOL pH	2.27E-04	1.88E-05	6.04E-05	8.98E-05
Average Measured Concentration			9.70E-04	3.15E-05	9.85E-05	3.98E-04
AP1000 Design Value			1.90E-03	2.20E-04	6.70E-04	1.30E-03
Ratio – Design/Measured			2.0	7.0	6.8	3.3

** - Source: Evaluation of Cycle Length and Non-Standard Coolant Chemistries on PWR Radiation Fields, EPRI, Palo Alto, CA: 2001. 1003123. (Table 4-1)*

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 471.003

Question:

Discuss what effect the increase (over the AP600 levels) in the reactor coolant source terms contained in Table 11.1-8 (as well as the other plant source terms) will have on the estimated inhalation exposures to plant personnel (resulting from the expected increase in airborne activity levels in the occupied areas of the plant).

Westinghouse Response:

There is no separate determination of doses due to airborne activity. Past experience demonstrates that the dose from airborne activity is not a significant contributor to the total doses.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 471.004

Question:

In Section 11.5.6.4 (Fuel Handling Area Criticality Monitors) of the DCD, you state that criticality monitoring of the fuel handling and storage areas is performed in accordance with 10 CFR 70.24. 10 CFR 70.24 requires the use of two criticality monitors in each area where specified amounts of licensed special nuclear material is handled, used, or stored.

A. In Appendix 1A of the DCD for the AP1000 you have withdrawn your commitment to follow the Guidance of Regulatory Guide (RG) 8.12, "Criticality Accident Alarm Systems." Justify your reasoning for not following the guidance contained in this RG when you have committed to have criticality monitors in the Fuel Handling Area.

B. In Section 11.5.6.4 of the DCD you state that radiation monitors RMS-JE-RE012 and RMS-JE-RE020 will be used for criticality monitoring of the fuel handling and storage areas. In your response to AP600 RAI 471.024, you stated that one of these criticality monitors will be located in the southwest (SW) corner of the South Auxiliary Building on elevation 135'-3" (Figure 12.3-3 (sheet 8 of 16)) and the other will be located in the East end of this same building at roof elevation 153'-0" (Figure 12.3-3 (sheet 9 of 16)). Justify why this second criticality monitor does not appear on Figure 12.3-3 (sheet 9 of 16) in the AP1000 DCD.

C. Section 11.5.6.4 states that the area radiation monitoring in the vicinity of the spent fuel pool (provided by radiation monitors RMS-JE-RE012 and RMS-JE-RE020) will be augmented during fuel handling operations by a portable radiation monitor on the machine handling fuel. Specify the location(s) of the machine(s) which will be used to handle fuel (e.g., will this monitor be used when lifting new fuel from elevation 100'-0" using the one ton jib crane?).

Westinghouse Response:

A. It is our understanding that Regulatory Guide 8.12 has been withdrawn by NRC.

B. Figure 12.3-3 is not intended to show radiation monitors, and none of them are shown on it. Both RMS-JE-RE012 and RMS-JE-RE020 will be located in room 12562, in the southwest and east ends of the room respectively. This is the same arrangement as AP600.

C. The portable radiation monitor will be used on all fuel handling machines, and moved from machine to machine. This includes the main fuel handling machine (for transfers to and from the spent fuel pool and containment) and the jib crane for new fuel. As noted in section 11.5.6.4 of the DCD, criticality monitors are not required for the new fuel pool because the arrangement of the new fuel prevents accidental criticality.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 471.006

Question:

Section 12.3.1 (Facility Design Features) of the DCD for the AP1000 remains unchanged from the comparable section for the AP600 which was compiled in the early 1990s. Justify why this section for the AP1000 does not contain a description of any of the improvements in radiation protection design features made by industry over the past decade.

Westinghouse Response:

The specific design features described in Section 12.3.1 are the same for both AP600 and AP1000, which is why the section is unchanged. The AP600 received Design Certification in December 1999.

Westinghouse personnel keep abreast of current ALARA related design and operational improvements through involvement in the PWR ALARA Committee, and attendance and participation in many technical and EPRI sponsored meetings related to ALARA aspects of plant design and operation. For example, Westinghouse has been a leader in the development and application of improved coolant technologies to reduce deposited source terms.

Appropriate advances to technology will be applied to AP1000 as equipment and vendors are selected for the AP1000 plant design.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 471.008

Question:

In response to Question 471.002 for the AP600 (concerning vital area access routes), you provided a description of the access routes to vital areas during post-accident conditions. Verify that the response to this question for the AP600 is still valid for the AP1000.

Westinghouse Response:

The response to AP600 RAI 471.002 is not valid for the AP1000.

The post-accident vital access routes for AP1000 are shown graphically on the post-accident radiation zone drawings, Design Control Document Figure 12.3-2, sheets 2 through 15.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 471.010

Question:

Section 12.4.1.7 of the DCD indicates that the overall estimated annual personnel dose associated with the AP1000 is 67 person-rems, the same as the estimate for the AP600. Justify your reasoning for not increasing the annual collective dose estimate for the AP1000 in light of the fact that the reactor coolant activity source terms have increased.

Westinghouse Response:

The annual personal dose estimate of Section 12.4.1.7 was re-estimated for AP1000, not carried over from AP600. The fact that both plants show a value of 67 person-rem is coincidental.

The coolant activities do not have a significant effect on personnel dose after the plant is shut down, since the main source of dose rates and thus occupational radiation exposure (ORE) is activity deposited on the component surfaces. Dose estimates for AP600 were determined based on a review of plant operating experience with adjustments made to the data for key plant parameters which affect radiation fields and plant ORE. The effect of a larger core power level for AP1000 as compared to that for AP600 was taken into account in projecting the plant radiation fields. However, the AP1000 dose estimate also took into account lower cobalt input into the RCS due to use of low cobalt materials in the SG tubing and certain valves. The AP600 dose estimate did not take full credit for these factors.

If full credit for low cobalt materials was to be considered for the AP600, its annual ORE is estimated to be 47 rem, or about 30% less than that estimated for AP1000.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 472.001

Question:

Table 1.9-1 (Sheet 8 of 15) indicates that RG 1.101, "Emergency Planning and Preparedness" (Revision 3, August 1992), "...is not applicable to AP 1000 design certification." The staff agrees Revision 3 to RG 1.101, which endorsed NUMARC NESP-007 as an alternative methodology for developing emergency action levels, is not applicable to the AP1000 DCD and will be a combined license (COL) applicant item. However, Revision 2 to RG 1.101, which endorsed the NUREG-0654 criteria for the development of emergency plans, includes emergency response facilities, (Technical Support Center, Operational Support Center, Decontamination Facilities) should be included in the AP1000 DCD. Why doesn't Table 1.9-1 include RG 1.101, Revision 2 (appears to contradict Appendix 1A, page 1-A-35, were it is indicated that AP1000 position is to "conform" to Revision 3)?

Westinghouse Response:

Table 1.9-1 is a table that simply indicates where in the Design Control Document a Regulatory Guide is discussed if it applies to AP1000 Design Certification. As indicated in Table 1.9-1 and RAI 472.001, RG 1.101 does not apply to Design Certification and will be addressed during the Combined License (COL) application process. The revision referenced (3) in Table 1.9-1 is the latest revision issued by NRC. As with all other Regulatory Guides listed in Table 1.9-1, the revision listed is considered to include or supercede requirements in all earlier revisions. In the case of RG 1.101, Revision 3 does continue to include the NRC endorsement of NUREG-0654 similar to the endorsement of Revision 2. See Sections B and C of RG 1.101, Revision 3. In addition, both Appendix 1A, which delineates AP1000 conformance to Regulatory Guides, and Section 13.3.1, which describes the COL information item related to emergency planning, specifically call out NUREG-0654. The clarification paragraph in Appendix 1A for RG 1.101 provides a historical perspective on the NRC endorsement of NUREG-0654, but the overall entry for RG 1.101 cites Revision 3 as the current revision. In addition, Sections 18.8.3.5, 18.8.3.6 and 18.8.3.9 discuss the design of emergency response facilities. Section 18.2.6 is another COL information item that addresses design of emergency response facilities. Emergency response facilities will be properly and adequately addressed during the COL application process.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 472.002

Question:

Figure 1.2-18, shows the Hot Machine Shop. Section 1.2.5 indicates that the Hot Machine Shop includes decontamination facilities. However, no such facilities are depicted on the figure. Explain the portable Decontamination System including it's location, use, etc., as well as other equipment and facilities, such as showers, that would be used for decontamination operations.

Westinghouse Response:

Room 40358, the Hot Machine Shop, includes a variety of equipment for servicing radiological controlled area equipment, including a lathe, a power hacksaw and power band saw. Also included is a diked decontamination basin with a grating support floor, connected to the radioactive waste drain system. A contaminated component can be supported off the grating and washed down with demineralized water to effect sufficient decontamination. This decontamination basin is about 8 feet by 8 feet and its edges are located about 4 feet and 6 feet from the south and west walls of Room 40358 respectively. The decontamination basin is a permanent part of Room 40358.

The "portable decontamination system" is a cart with a source of water, tools, swabs, clothes, other water collection and cleaning devices. It, like the standard machine tools in Room 40358, are to be purchased by the Combined License holder to specifications of their choosing to be compatible with their operating and maintenance practices.

Note that decontamination of people will be performed in Room 40355, Decontamination room, which included two personnel showers and two sinks connected to the radioactive liquid waste system.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 472.003

Question:

Section 9.4.1.2.1.1 indicates that radiation monitors are located inside the main control room upstream of the supply air isolation valves and that these monitors isolate the main control room form the nuclear island non-radioactive ventilation system on high-high particulate or iodine radioactivity concentrations. Does this include isolating the technical support center as well?

Westinghouse Response:

No, only the main control room is isolated on a high-high signal. At that time, the main control room emergency habitability system is placed into operation to protect the main control room operators. Please refer to the "Abnormal Plant Operation" portion of DCD subsection 9.4.1.2.3.1, which provides details as to the operation of the main control room and technical support center HVAC subsystem during abnormal events involving high and high-high signals.

Also see DCD subsection 18.8.3.5 "Technical Support Center Mission and Major Tasks" for discussions of the technical support center (TSC) including habitability and evacuation during emergencies.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.001

Question:

DCD Rev. 0, pages Intro-8 and -9. The following documents are listed in Table 1-1 "Index of AP1000 Tier 2 Information Requiring NRC [Nuclear Regulatory Commission] Approval for Change": WCAP-14396, "Man-In-The Loop Test Plan Description," Rev. 2; WCAP-14401, "Programmatic Level Description of the AP-600 Human Factors Verification and Validation Plan," Rev. 2; WCAP-14701, "Methodology & Results of Defining Evaluation Issues for the AP600 Human System Interface Design Test Program," Rev. 1; and WCAP-14822, "AP600 Quality Assurance Procedures Supporting NRC Review of AP600 SSAR Sections 18.2 and 18.8," Rev. 0. The table lists the Tier 2 references for these documents as Chapter 18 and Table 1.6-1 of the DCD. However, DCD, Rev. 0, Table 1.6-1 does not cite these WCAPs nor does Chapter 18. In addition, DCD Rev. 0, Table 1.6-1 identifies materials referenced as Tier 2* that do not appear in Table 1-1. Please clarify and reconcile these discrepancies.

Westinghouse Response:

WCAP-14396 will be deleted from Table 1-1. The referenced report addressed concept test plans, per AP600 DCD Section 18.8.1.4 (deleted). Disposition of concept testing for AP1000 is addressed in RAI #620.008.

WCAP-14401 for AP600 has been replaced by WCAP-15860 for AP1000 (e.g. DCD Table 1.6-1). WCAP-15860 has been submitted to NRC by Reference 1. Table 1-1 will be revised accordingly.

WCAP-14701 will be deleted from Table 1-1. The referenced report addressed a design test program composed of concept testing and validation testing. Disposition of concept testing for AP1000 is addressed in RAI #620.008. Dispositions of testing issues identified in WCAP-14701 are addressed in RAI #620.034.

WCAP-14822 for AP600 has been replaced by WCAP-15847 for AP1000 (e.g. DCD Table 1.6-1). WCAP-15847 has been submitted to NRC by Reference 2. Table 1-1 will be revised accordingly.

References

1) Westinghouse Letter DCP/NRC1497, "Transmittal of WCAP-15860"Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan" Revision 0 dated April 15, 2002.

2) Westinghouse Letter DCP/NRC1500, "Transmittal of WCAP-15847, "AP1000 QA Procedures Supporting NRC Review of AP1000 DCD Section 18.2 and 18.8" Revision 0" dated April 15, 2002.

Design Control Document (DCD) Revision:

From DCD Table 1-1 (cont.) on p. Intro-8:

~~WCAP-14396, "Man-In-The-Loop Test Plan Description," Rev 2~~	~~No~~	~~Chapter 18 Table 1.6-1~~
WCAP-~~14401~~**15860**, "Programmatic Level Description of the AP~~600~~**1000** Human Factors Verification and Validation Plan," ~~Rev 3~~ **Rev 0**	No	Chapter 18 Table 1.6-1
WCAP-14651, "Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan," Rev 2	No	Chapter 18 Table 1.6-1
WCAP-14695, "Description of the Westinghouse Operator Decision Making Model and Function Based Task Analysis Methodology," Rev 0	No	Chapter 18 Table 1.6-1
~~WCAP-14701, "Methodology & Results of Defining Evaluation Issues for the AP600 Human System Interface Design Test Program," Rev 1~~	~~No~~	~~Chapter 18 Table 1.6-1~~
WCAP-~~14822~~**15847**, "AP~~600~~**1000** Quality Assurance Procedures Supporting NRC review of AP~~600~~**1000** SSAR Sections 18.2 and 18.8," Rev 0	No	Chapter 18 Table 1.6-1

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.002

Question:

DCD Rev. 0, Intro-9, Table 1-1, "Index of AP1000 Tier 2 Information Requiring NRC Approval for Change." Fourth item from the bottom, typographical error: should read "Main" not "Mail."

Westinghouse Response:

Typo will be corrected as indicated below.

Design Control Document (DCD) Revision:

From DCD Table 1-1 (cont.) on p. Intro-9:

Safety Parameter Display System HFE	No	18.8.2.5
~~Mail~~ MainControl Area Mission and Major Tasks	No	18.8.3.2
Remote Shutdown Workstation Mission and Major Tasks	No	18.8.3.4

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.003

Question:

DCD Rev. 0, Intro-9, Table 1-1, "Index of AP1000 Tier 2 Information Requiring NRC Approval for Change." Last item, "Human System Interface [HSI] Design Test Program." DCD Rev. 0 was changed to "Human Factors Engineering Verification and Validation [V & V]." Please reconcile this inconsistency.

Westinghouse Response:

The last entry in Table 1-1 will be changed to "Human Factors Engineering Verification and Validation," for consistency with DCD Section 18.11, as indicated below.

Design Control Document (DCD) Revision:

From DCD Table 1-1 (cont.) on p. Intro-9:

Technical Support Center Mission and Major Tasks	No	18.8.3.5
Human System Interface ~~Design Test Program~~ **Verification and Validation**	No	18.11

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.004

Question:

DCD Rev. 0, page 1.4-4, "List of Acronyms and Abbreviations." The abbreviation "RSR" is included in the list. However, in several places in the DCD (e.g., in Tier 2, Chapter 18), RSR has been deleted and replaced with the term "remote shutdown workstation" or the abbreviation "RSW." It is used in, for example, the General Arrangement Plan, Figure 1.2-7. Please clarify this inconsistency. Is the term "remote shutdown room" (and abbreviation "RSR") applicable to the AP1000 design? If so, how does it differ from the "remote shutdown workstation" and why has it been removed from certain sections of the DCD?

Westinghouse Response:

The terminology "remote shutdown room (RSR)" and "remote shutdown area" are used interchangeably throughout the AP1000 DCD, and refer to the area where the resources to bring and maintain the plant in a safe shutdown condition, after evacuation of the main control room, are provided. The item within the room referred to as the "remote shutdown workstation (RSW)" contains the resources (i.e., displays, controls, visual alerts) needed to establish and maintain safe shutdown conditions for the plant from a location outside of the MCR. For the purpose of discussing HFE, RSW is therefore a more specific reference to HSI resources, and general references to either the RSR or the remote shutdown area imply the RSW by inclusion.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.005

Question:

DCD Rev. 0, page 2.5.2-9, Table 2.5.2-5, "Minimum Inventory of Displays, Alerts, and Fixed Position Controls in the MCR (Main Control Room)" and page 2.5.4-3, Table 2.5.4-1, "Minimum Inventory of Controls, Displays, and Alerts at the RSW (Remote Shutdown Station)." Please explain why containment hydrogen concentration has been eliminated from the AP1000 inventory as a fixed position control.

Westinghouse Response:

Control of "Manual Containment Hydrogen Igniter (Nonsafety-related)" remains listed in the minimum inventory for the MCR (Table 2.5.2-5 on page 2.5.2-9; also Table 18.12.2-1 on page 18.12-10) and for the RSW (Table 2.5.4-1 on page 2.5.4-3). Only the MCR minimum inventory is described as "fixed position". DCD Section 18.12.3 states that the controls, displays, and alarms listed in Table 18.12.2-1 are retrievable from the remote shutdown workstation. These aspects of the AP1000 DCD are unchanged from the approved AP600 submittal.

Display of "Containment Hydrogen Concentration" was removed from Tables 2.5.2-5, 2.5.4-1, and 18.12.2-1. This is due to the long time available before excessive H2 can be generated (72 hours after fuel meltdown) and the corresponding operator response to fuel failure (starting the igniters) is required. Hydrogen Igniters are discrete-state devices and are not adjusted in response to H2 levels. Thus, the ERGs do not use containment H2 concentration as a cue either to initiate or control the Hydrogen Igniters. Instead, indication to start Hydrogen Igniters is based on Core Exit Temperature. Core Exit Temperature remains listed in Table 2.5.2-5 on page 2.5.2-8, and in Table 2.5.4-1 on page 2.5.4-3. Thus, since fixed position display of H2 concentration is not required for emergency operation, it has been removed as noted.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.006

Question:

DCD Rev. 1, page 3.2-1, "Human Factors Engineering," Design Description. Please explain why the term "remote shutdown room" and its associated abbreviation "RSR" was replaced by "remote shutdown workstation" and "RSW" in the first paragraph of the design description. In the third paragraph, "RSW" is now being used to describe a "facility and resources," which was previously described in the AP600 DCD as the remote shutdown room. (See previous question 620.004.)

Westinghouse Response:

It is correct that in this discussion the DCD text is referring to the facility, and therefore, it is appropriate to use remote shutdown room (RSR). The DCD will be modified as indicated below.

Refer to RAI 620.004 for a discussion of the RSR and RSW terminology used in the DCD.

Design Control Document (DCD) Revision:

From DCD Tier 1, page 3.2-1:

The ~~RSW~~ RSR provides a facility and resources to establish and maintain safe shutdown conditions for the plant from a location outside of the MCR. The RSW includes a minimum inventory of displays, controls, and visual alerts. Refer to item 2 and Table 2.5.4-1 of subsection 2.5.4 for this minimum inventory. As stated in item 8.b of subsection 2.5.2, the protection and safety monitoring system (PMS) provides for the transfer of control capability from the MCR to the RSW.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.007

Question:

DCD Rev. 0, page 3.2-2, Item 5. Please describe how these activities associated with the HFE V & V implementation plan meet the guidance contained in NUREG-0711, Rev. 1, May 2002. If this guidance was not utilized, please describe the guidance that was utilized and the basis for using this guidance. The activities should be modified if they are to remain in agreement with NUREG-0711, Rev. 1, May 2002.

Westinghouse Response:

Page 3.2-2, Items 5a - 5e comprise a list of general issues addressed by the AP1000 Verification & Validation Plan. The existing level-of-detail of the cited Tier 1 material remains appropriate for ITAAC Design Description (DCD Sec. 3.2). The mapping of these issues to old and new guidance is as follows:

DCD Section 3.2 Item	NUREG-0711 Section	NUREG-0711 Rev.1 Section
5a	11.4.2	11.4.2.2
5b	11.4.3	11.4.2.3
5c	11.4.4	11.4.3
5d	11.4.5	11.4.4
5e	11.4.6	12.4.6

The indicated HFE issues therefore remain valid, and the DCD material that addresses them is unchanged from the approved AP600 submittal.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.009

Question:

DCD Rev. 0, page 3.2-6, "Acceptance Criteria." (See previous question 620.007.)

Westinghouse Response:

Item 4 of Table 3.2-1 is the Tier 1 Acceptance Criteria for the AP1000 HFE V&V Implementation Plan. The existing level-of-detail of the cited Tier 1 material remains appropriate for ITAAC Acceptance Criteria (DCD Table 3.2-1). The mapping of these issues to old and new guidance is as follows:

DCD Table 3.2-1 Item 4 Acceptance Criteria	NUREG-0711 Section	NUREG-0711 Rev.1 Section
Task Support Verification	11.4.2	11.4.2.2
HSI Design Verification	11.4.3	11.4.2.3
Integrated System Validation	11.4.4	11.4.3
Issue Resolution Verification	11.4.5	11.4.4
Plant HFE/HSI Verification	11.4.6	12.4.6

The indicated HFE issues therefore remain valid, and the DCD material that addresses them is unchanged from the approved AP600 submittal.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.010

Question:

DCD Rev. 0, page 3.2-7, "Design Commitment" and "Acceptance Criteria." (See previous question 620.007.)

Westinghouse Response:

Item 5 of ITAAC Table 3.2-1 (DCD page 3.2-7) includes the Tier 1 Design Commitment and Acceptance Criteria for the AP1000 HFE Verification & Validation Program. Items 5a - 5e comprise the general issues addressed by the Program. The existing level-of-detail of the cited Tier 1 material remains appropriate for ITAAC (DCD Table 3.2-1). The mapping of these issues to old and new guidance is as follows:

DCD Table 3.2-1 Item	NUREG-0711 Section	NUREG-0711 Rev.1 Section
5a	11.4.2	11.4.2.2
5b	11.4.3	11.4.2.3
5c	11.4.4	11.4.3
5d	11.4.5	11.4.4
5e	11.4.6	12.4.6

The indicated HFE issues therefore remain valid, and the DCD material that addresses them is unchanged from the approved AP600 submittal.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.011

Question:

DCD Rev 0, page 3.2-10, Number 10, "Design Commitment." Please explain how this design commitment meets the guidance contained in NUREG-0711, Rev. 1, May 2002. If this guidance was not utilized, please describe the guidance that was utilized and the basis for using this guidance. (See previous question 620.007.)

Westinghouse Response:

It is correct that in this discussion the DCD text is referring to the facility, and therefore, it is appropriate to use remote shutdown room (RSR). This design commitment is unchanged from AP600 and the text will be returned, as indicated below, to the original text.

Refer to RAI 620.004 for a discussion of the RSR and RSW terminology used in the DCD.

Design Control Document (DCD) Revision:

From DCD Tier 1, page 3.2-10:

10. The ~~RSW~~ RSR provides a suitable workspace environment, separate from the MCR, for use by the RSW operators.	i) See Tier 1 Material, subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System. ii) See Tier 1 Material, subsection 2.6.5, Lighting System. iii) See Tier 1 Material, subsection 2.3.19, Communication System.	i) See Tier 1 Material, subsection 2.7.1, Nuclear Island Nonradioactive Ventilation System. ii) See Tier 1 Material, subsection 2.6.5, Lighting System. iii) See Tier 1 Material, subsection 2.3.19, Communication System.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.012

Question:

DCD Rev. 0, page 3.2-10. Please describe how the HFE design implementation process depicted in Figure 3.2-1 meets the guidance contained in NUREG-0711, Rev. 1, May 2002. If this guidance was not utilized, please describe the guidance that was utilized and the basis for using this guidance. Figure 3.2-1 should be modified if it is to remain in agreement with current NRC guidance, NUREG-0711, Rev. 1, May 2002.

Westinghouse Response:

Figure 3.2-1, taken from NUREG-0711, July 1994, will be changed as shown below. The changes are described as follows. The added element shown as "Design Implementation" represents the unchanged activities of Issue Resolution Verification and Plant HSI/HFE Verification in DCD Section 3.2 and Table 3.2-1. This is no actual change to the process (see also RAIs 620.007, 009, & 010). The added element shown as Human Performance Monitoring will be a COL Action Item. These changes have no further impact on Tier 1. Details of the changes are included in the response to RAI 620.018.

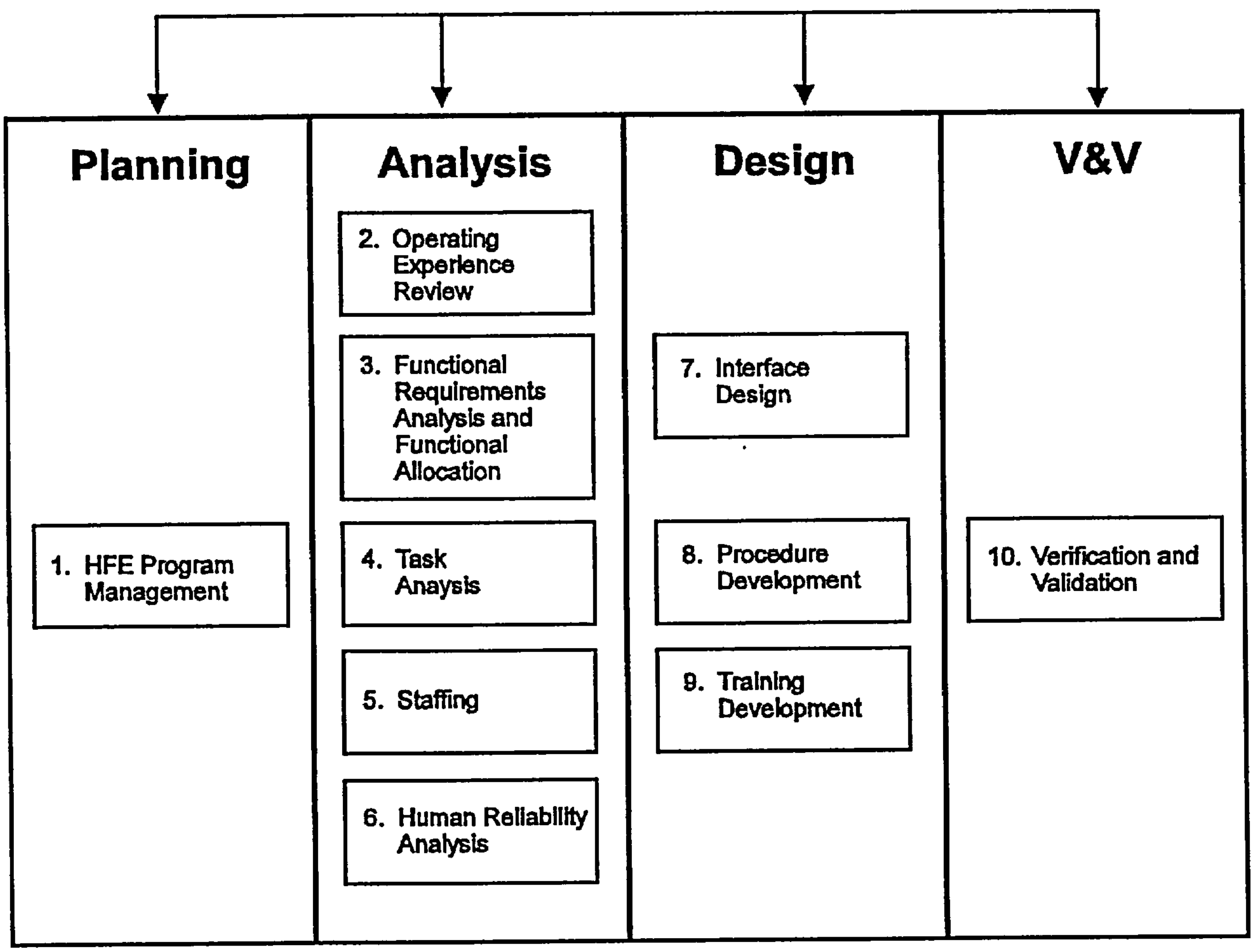
Design Control Document (DCD) Revision:

See RAI 620.018 for the Chapter 18 DCD changes resulting from addition of the COL Action Item "Human Performance Monitoring".

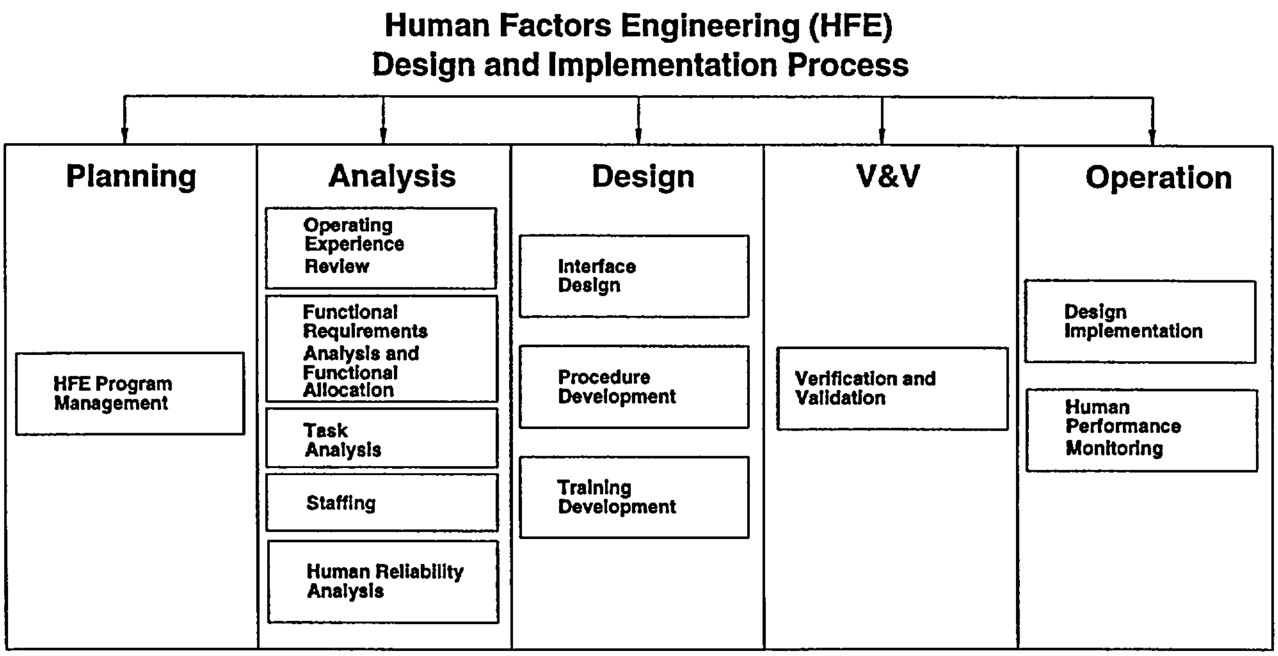
From DCD Tier 1, Section 3.2, Figure 3.2-1:

OLD FIGURE

Human Factors Engineering (HFE) Design and Implementation Process



NEW FIGURE



PRA Revision:

None

RAI Number: 620.013

Question:

DCD Rev. 1, page 1.6-19, Table 1.6-1, "Materials Referenced." WCAP 10170, "Emergency Response Facilities Design and V & V Process," WCAP-14695, "Description of the Westinghouse Operator Decision-Making Model and Function-Based Task Analysis Methodology," and WCAP-14651, "Integration of Human Reliability Analysis With Human Factors Engineering Design Implementation Plan" are referenced in the table but are not cited and do not appear as references in Chapter 18, Section 18.5. Also, WCAP-15847, "AP1000 Quality Assurance Procedures Supporting NRC Review of AP1000 DCD Sections 18.2 and 18.8," should be referenced as April, not March, 2002. WCAP-15847 does not appear in Table 1-1 (Intro-8). Please reconcile these inconsistencies.

Westinghouse Response:

WCAPs 10170, 14695 and 14651 are references 11, 12 and 13 of DCD section 18.5, respectively. They are cited in DCD section 18.5 as follows:

From 18.5, page 18.5-1, second paragraph:

This section describes the scope of the AP1000 task analysis activities and the task analysis implementation plan. In addition to Reference 1, References 2 through 12 are inputs to this plan. Execution and documentation of this task analysis implementation plan is the responsibility of the Combined License applicant.

From 18.5.1, page 18.5-2, first paragraph:

The human factors engineering program review model (Reference 1) indicates that task analysis should include tasks that are considered to be high-risk and tasks that require critical human actions. Reference 13 defines criteria for critical human actions and risk-important tasks and has identified a list of examples of AP600 tasks that meet these criteria. Reference 13 is applicable to AP1000.]*

From 18.5.2.1, page 18.5-2, last paragraph:

18.5.2.1Function-Based Task Analyses

Function-based task analysis is applied to each of the Level 4 functions. There are four components to a function-based task analysis. First, analysis is performed to identify the set of goals relevant to the function. Second, a functional decomposition is performed. This decomposition identifies the processes that, either individually or in combination, have a significant effect on the function. Third, a process analysis is performed by applying a set of questions derived from Rasmussen's model (References 6-9) analysis approach. [*The set of questions used and basis for the methodology is provided in Reference 12.*]* An example of a question from the process analysis is "Are the process data valid?" The results of the process analysis identify the indications, parameters, and controls that the operator uses to make decisions about the respective function. Finally, there is a verification that the indications and controls, identified in the process analysis are included in the AP1000 design.

The date for WCAP-15847 will be corrected to April 2002, as indicated below, in the next revision of the DCD.

WCAP-14822 for AP600 has been replaced by WCAP-15847 for AP1000 (e.g., DCD Table 1.6-1). Table 1-1 will be revised, as indicated below, in the next revision of the DCD.

Design Control Document (DCD) Revision:

From DCD, page 1.6-19, Table 1.6-1 (Sheet 18 of 20):

WCAP-14694	Designer's Input to Determination of the AP600 Main Control Room Staffing Level, July 1996
[*WCAP-15847*	*AP1000 Quality Assurance Procedures Supporting NRC Review of AP1000 DCD Sections 18.2 and 18.8, Rev. 0,* ~~*March 2002*~~ *April 2002*]*
WCAP-14644	AP600 Functional Requirements Analysis and Function Allocation, September 1996

From DCD Table 1-1 (cont.) on p. Intro-8:

WCAP-14701, "Methodology & Results of Defining Evaluation Issues for the AP600 Human System Interface Design Test Program," Rev 1	No	Chapter 18 Table 1.6-1
WCAP-~~14822~~15847 "AP~~600~~1000 Quality Assurance Procedures Supporting NRC review of AP~~600~~1000 SSAR Sections 18.2 and 18.8," Rev 0	No	Chapter 18 Table 1.6-1
Basis for Human Factors Engineering Program	No	18.1

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.014

Question:

DCD Rev. 1, page 1.6-20, Table 1.6-1, "Materials Referenced." WCAP-15860, "Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan," should be identified with an April 2002 date, not March (two citations). Also see previous comment re: WCAP-10170.

Westinghouse Response:

Date (April 2002) will be corrected as indicated below.

Design Control Document (DCD) Revision:

From DCD Table 1.6-1 (Sheet 17 of 20):

	[WCAP-15860	*Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan, Revision 0, ~~March 2002~~April 2002]**

From DCD Table 1.6-1 (Sheet 19 of 20):

18.8	*[WCAP-15860*	*Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan, Revision 0, ~~March 2002~~April 2002]**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.015

Question:

DCD Rev. 0, page 13-1. Title of Chapter 13 should be "Conduct of Operations," if the title is to be consistent with the guidance of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants."

Westinghouse Response:

As indicated below, the title chapter will be changed to "Conduct of Operations" to be consistent with SRP, NUREG-800 in the next revision of the DCD.

Design Control Document (DCD) Revision:

From DCD, page 13-1:

CHAPTER 13

CONDUCT OF ~~OPERATION~~ OPERATIONS

From DCD, Chapter 13 Table of Contents, page i:

TABLE OF CONTENTS

From DCD, Master Table of Contents, page xix:

TIER 2 MASTER TABLE OF CONTENTS (Cont.)

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.016

Question:

620.016, DCD, Rev. 0, page 13-3.

Please explain why reference to WCAP-14837, "AP600 Shutdown Evaluation Report," was eliminated. Is there a document comparable to this for AP1000? If not, please provide explanation?

Westinghouse Response:

Development of a comparable document to WCAP-14837, "AP600 Shutdown Evaluation Report" is not planned for AP1000. The relevant information from this WCAP is now included in DCD Appendix 19E.

As indicated below, a reference to Appendix 19E will be added to DCD page 13-3.

Design Control Document (DCD) Revision:

From DCD, page 13-3:

The Combined License applicant is responsible for the development of plant specific refueling plans <u>(DCD Appendix 19E provides input for refueling plans).</u>

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.017

Question:

DCD Rev. 0, page 18.1-2. The first paragraph refers to "remote shutdown rooms." Is there more than one remote shutdown room for the AP1000 design? Also, please see previous Tier 1 questions related to remote shutdown room vs. remote shutdown workstation (See previous question 620.004).

Westinghouse Response:

The referenced DCD paragraph is referring to the main control [room] and the remote shutdown [room], when quoting "rooms". There is only a single room in the AP1000 provided for the remote shutdown function. The DCD text has been clarified as noted below.

In this discussion the DCD text is referring to the facility, and therefore, it is appropriate to use remote shutdown room (RSR). Refer to RAI 620.004 for a discussion of the RSR and RSW terminology used in the DCD.

Design Control Document (DCD) Revision:

From DCD, page 18.1-2:

The layout and environmental design of the main control <u>room</u> and the remote shutdown <u>room</u> ~~rooms~~, and the supplementary support areas, such as the technical support center, are sites of application of the traditional disciplines of human factors engineering.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.019

Question:

DCD Rev. 0, pages 18.1-2, 18,1-3, 18.1.1, "References." NUREG-0711 should be cited as Rev. 1, May 2002.

Westinghouse Response:

Westinghouse understands that the Staff will review the AP1000 design submittal according to whatever is the latest NRC guidance. Changes to regulatory guidance may or may not impact the AP1000 design. In fact, AP1000 used the original NUREG-0711 as input, so that references to it as input remain valid. References to revised NRC guidance will be updated where corresponding changes or additions to the DCD are necessary to satisfy the revised guidance.

In DCD Section 18.1, the existing citation to NUREG-0711 in Section 18.4.2 is retained. A citation to NUREG-0711 Rev.1 (Reference 11) will be added. Added elements will be addressed in the Tier 2 DCD text; see response to RAI 620.018.

Design Control Document (DCD) Revision:

This chapter describes the application of the human factors engineering disciplines to the design of the AP1000. [*The basis for the human factors engineering program is the human factors engineering process specified in Reference 2.*]* Figure 18.1-1 illustrates the ~~10~~ elements of the human factors engineering program. These elements ~~conform~~**correspond** to the elements specified in Reference 2 **and Reference 11**. The organization of this chapter parallels these elements. In addition to the ~~10~~ elements of the program review model, this chapter includes a description of the minimum inventory of controls, displays, and alarms present in the main control room and at the remote shutdown workstation. The following provides an annotated outline of the chapter. A number of References are identified which were developed for the AP600 Design Certification. Since the AP1000 operating philosophy and approach are the same for AP600 and AP1000, the References identified below are applicable to AP1000.

[18.1.1 **References**]

11. NUREG-0711, Rev.1, "Human Factors Engineering Program Review Model", May 2002.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.020

Question:

DCD Rev. 0, page 18.1-5. The figure depicting the HFE design and implementation process should be modified if it is to remain in agreement with current NRC guidance, NUREG-0711, Rev. 1, May 2002.

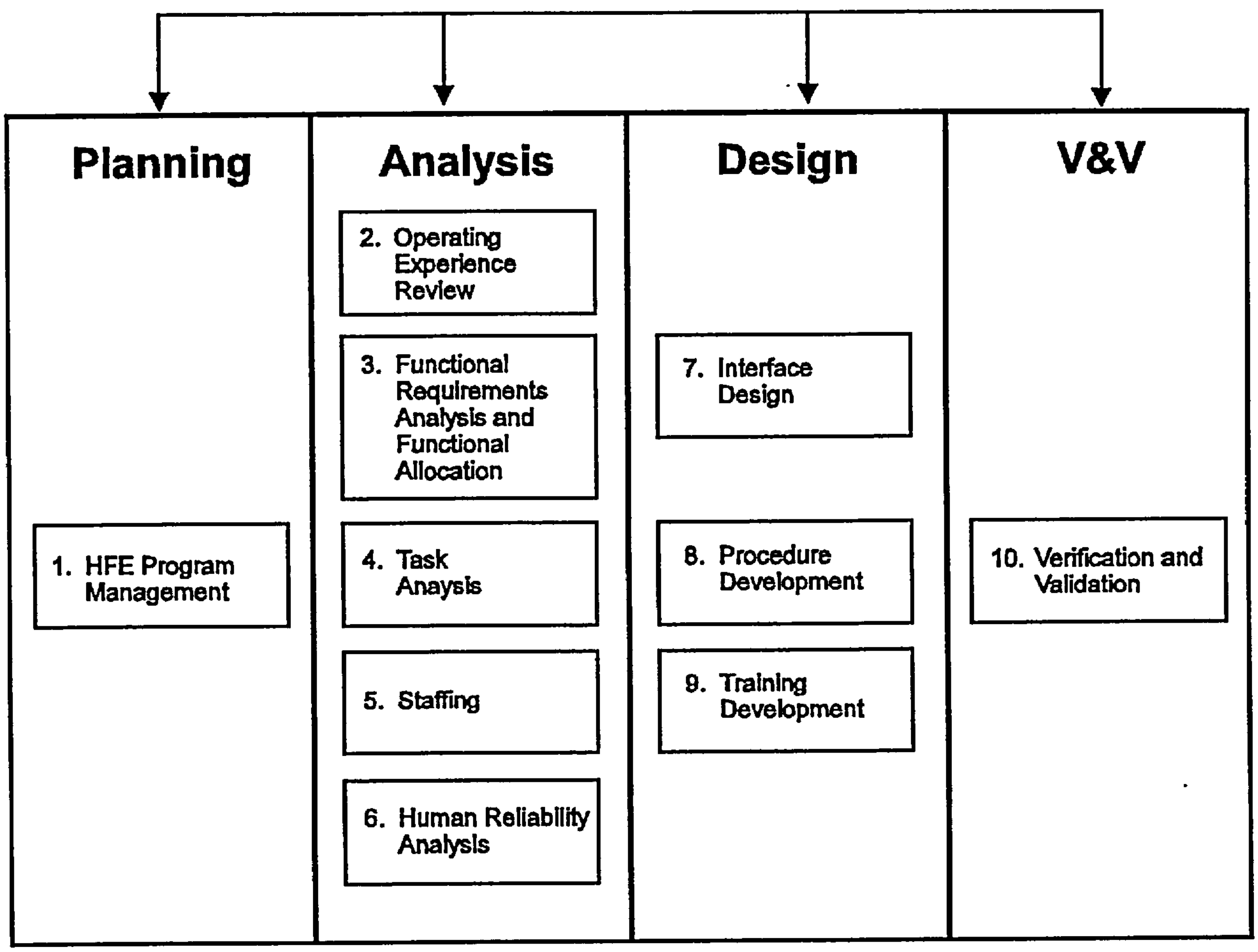
Westinghouse Response:

Figure 18.1-1, taken from NUREG-0711, July 1994, will be changed as shown below to satisfy NUREG-0711 Rev.1, May 2002. The details of the additions depicted in Figure 18.1-1 are described in the response to RAI 620.018.

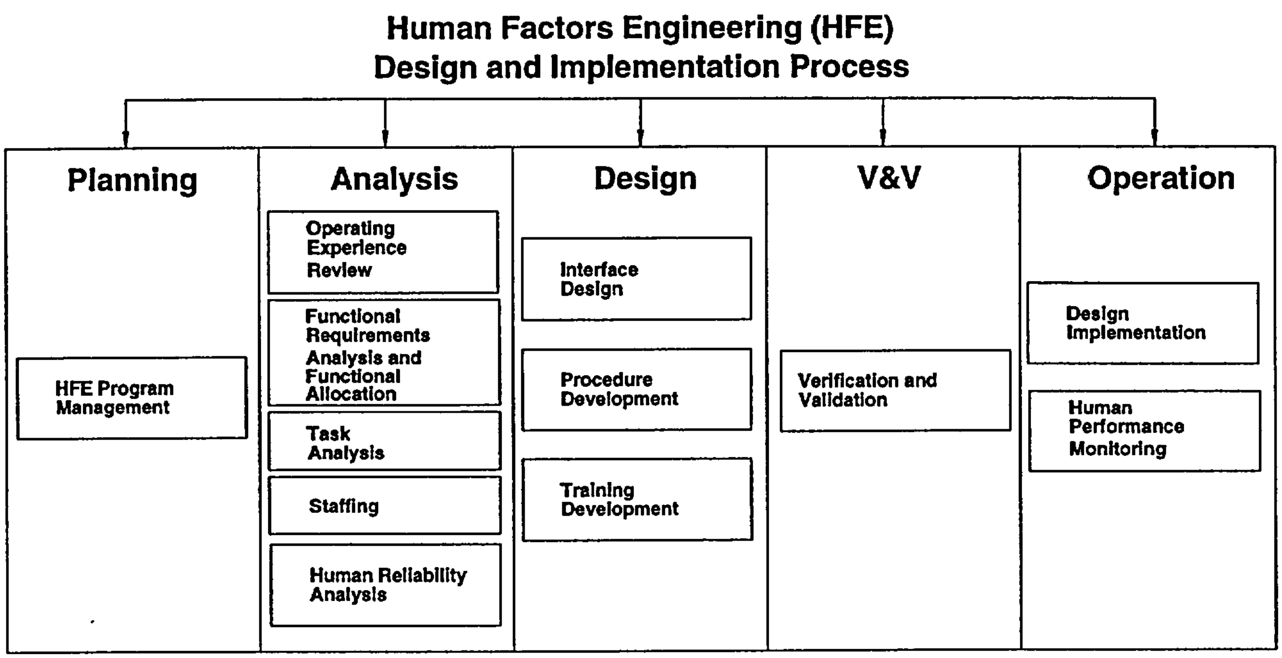
Design Control Document (DCD) Revision:

OLD FIGURE

Human Factors Engineering (HFE) Design and Implementation Process



NEW FIGURE



PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.021

Question:

DCD Rev. 1, page 18.2-2, "Regulatory Requirements." Due to the recent issuance of Rev. 1 to NUREG-0711, this guidance document no longer has an Appendix B. If NUREG-0711, Rev. 1 is referenced, NUREG/CR–6400, "HFE (Human Factors Engineering) Insights for Advanced Reactors Based on Operating Experience," (Higgins and Nasta) may cited instead of the previous Appendix B.

Westinghouse Response:

The second paragraph under "Regulatory Requirements" provided an example of AP1000 compliance with applicable regulatory requirements (e.g., NUREG 0711). Since this paragraph is unnecessary and references NUREG-0711, Appendix B, which no longer exists, the paragraph will be deleted from the DCD, as indicated below.

In addition, a date has been added to Reference 1 of DCD Section 18.2.7 as indicated below.

Design Control Document (DCD) Revision:

From DCD page 18.2-2:

Regulatory Requirements

One of the requirements for the AP1000 human factors engineering program is that it complies with applicable regulatory requirements. [*The human factors engineering process is designed to meet the human factors engineering design process requirements specified in NUREG-0711 (Reference 1).*]*

~~**As an example, the NRC Program Review Model includes a requirement to perform an operating experience review, and Appendix B of NUREG 0711 specifies particular classes of documents to be included as part of the operating experience review. DCD Section 18.3 and WCAP-14645 (Reference 2) document the operating experience review. The results of the operating experience review are used as input to the AP1000 design to establish requirements that the plant design, including the human system interfaces, adequately address the human factors issues raised by the operating experience review.**~~

From DCD page 18.2-2:

18.2.7 References

[*1. NUREG-0711, "Human Factors Engineering Program Review Model," U.S. NRC, July 1994.*]*

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.022

Question:

DCD Rev. 0, page 18.2-3, 18.2.1.3, "Applicable Facilities." The terms "remote shutdown room" and "emergency operations facility (EOF)" are used. See also page 18.2-12, last paragraph, for use of the term "remote shutdown room." Please clarify the proper use of terminology. (See previous questions 620.004, 620.006, 620.017.)

Westinghouse Response:

Refer to RAI 620.004 for a discussion of the RSR and RSW terminology used in the DCD.

The emergency operations facility (EOF) is a facility located outside the plant site, which provides a location for performing emergency response operations. The Combined License applicant is responsible for designing the EOF, including specification of a location, emergency planning, and associated communication interfaces among the main control room, the technical support center (TSC), and the EOF, in accordance with the AP1000 human factors engineering program. In the event the habitability of the TSC is challenged, due to lack of ventilation or a high radiation level resulting from a beyond-design-basis accident, the TSC personnel and the functions of the TSC are transferred to the EOF. EOF habitability is not dependent on plant systems and communication and data transfer links to the main control room to provide essential exchange of information.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.023

Question:

DCD Rev. 0, Section 18.2.3.5, page 18.2-16. As part of the AP1000 design certification application, Westinghouse submitted WCAP-15847, "AP1000 Quality Assurance Procedures Supporting NRC Review of AP1000 DCD Section 18.2 and 18.8." However, this topical report is not cited in Tier 2* material, Table 1-1, pages Intro-7 and -8. Please add this topical report to the documents listed in Table 1-1 or provide reason for not including.

Westinghouse Response:

WCAP-15847, *AP1000 QA Procedures Supporting NRC Review Of AP1000 SSAR Sections 18.2 and 18.8, Rev. 0,* will be added to DCD Table 1-1, as indicated below. This reference replaces WCAP-14822, see RAI 620.001.

Design Control Document (DCD) Revision:

From DCD Table 1-1, "Index of AP1000 Tier 2 Information Requiring NRC Approval for Change" on p. Intro-8:

WCAP-14701, "Methodology & Results of Defining Evaluation Issues for the AP600 Human System Interface Design Test Program," Rev 1	No	Chapter 18 Table 1.6-1
WCAP-~~14822~~**15847**, "AP~~600~~**1000** Quality Assurance Procedures Supporting NRC review of AP~~600~~**1000** SSAR Sections 18.2 and 18.8," Rev 0	No	Chapter 18 Table 1.6-1
Basis for Human Factors Engineering Program	No	18.1

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.024

Question:

DCD Rev. 1, page 18.2-18, 18.2.7, "References." Reference Number 6 (WCAP-15847) was submitted in April 2002 not March 2002 as indicated in the DCD. Please correct.

Westinghouse Response:

Date (April 2002) will be corrected, as indicated below, in the next revision of the AP1000 DCD.

Design Control Document (DCD) Revision:

From DCD, page 18.2-18, 18.2.7, "References":

[6. WCAP-15847, "AP1000 Quality Assurance Procedures Supporting NRC Review of AP1000 DCD Sections 18.2 and 18.8," Revision 0, ~~March~~ **April 2002.*]*****

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.027

Question:

DCD Rev. 0, page 18.4-2, 18.4.2, "References." If the most current revision to NUREG-0711 is utilized, the citation should be to NUREG-0711, Rev. 1, May 2002. See also, other instances within the DCD.

Westinghouse Response:

Westinghouse understands that the Staff will review future AP1000 design submittals according to whatever is the latest NRC guidance. Changes to regulatory guidance may or may not impact the AP1000 design. In fact, AP1000 used the original NUREG-0711 as input, so that references to it as input remain valid. References to revised NRC guidance will be updated where corresponding changes or additions to the DCD are necessary to satisfy the revised guidance.

In DCD Section 18.4, no change to "Functional Requirements Analysis and Allocation" has occurred to AP1000 (see page 18.4-1, paragraph seven). Since the existing citation to NUREG-0711 in Section 18.4.2 is correct, it will therefore be left as is.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.028

Question:

DCD Rev. 0, page 18.8-1. Use is again made of the terms "remote shutdown room" and "emergency operations facility." See previous questions related to the definition and use of these terms. In addition, there is an inconsistency in the use of the term "wall panel information system" and "wall panel information station." Please clarify.

Westinghouse Response:

Refer to RAI 620.004 for a discussion of the RSR and RSW terminology used in the DCD.

Refer to RAI 620.022 for a discussion of the emergency operations facility (EOF) terminology used in the DCD.

With regard to "wall panel information", the correct terminology is "wall panel information system (WPIS)". This will be corrected, as indicated below, in the next revision of the DCD.

Design Control Document (DCD) Revision:

The wall panel information ~~station~~ system presents information about the plant for use by the operators. No control capabilities are included. The wall panel information ~~station~~ system provides dynamic display of plant parameters and alarm information so that a high level understanding of current plant status can be readily ascertained. It is located at one end of the main control area at a height such that both operators and the shift supervisor can view it while sitting at their respective workstations. ~~This panel~~ It provides information important to maintaining the situation awareness of the crew and for supporting crew coordination. The wall panel information station provides a dynamic plant display of the plant. It also serves as the alarm system overview panel display. The display of plant disturbances (alarms) and plant process data are integrated on this wall panel information ~~station~~ system display. The wall panel information ~~station~~ system is a nonsafety-related system. It is designed to have a high level of reliability.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.030

Question:

DCD Rev. 1, page 18.8-4. Typographical error in the first sentence at the top of the page—should read, "local control **stations**."

Westinghouse Response:

Type will be corrected, as indicated below, in next revision of the DCD.

Design Control Document (DCD) Revision:

From DCD page 18.8-4:

Included in the operation and control centers system specification document are functional requirements and specifications for the AP1000 operation and control centers system, including the main control room, the technical support center, the remote shutdown room, and local control <u>stations</u>. In addition, functional requirement documents are generated for each of the individual human system interface resources. These documents are referenced by the operation and control centers system specification document.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.033

Question:

DCD Rev. 1, page 18.8-7, paragraph 18.8.1.5. Please explain the basis for using only a partial mock-up for the AP1000 design process.

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: 620.035

Question:

DCD Rev. 1, page 18.8-27. Reference to WCAP-14701 has been eliminated. However, this topical report was cited as Tier 2* material (information that requires NRC approval to change). Please correct or explain. (See previous question 620.034.)

Westinghouse Response:

Deletion of WCAP-14701 from DCD page 18.8-27 in Section 18.8.6 is intended. WCAP-14701 also will be deleted from Table 1-1 per response to RAI #620.001. The referenced report addressed a design test program composed of concept testing and validation testing. Dispositions of testing issues identified in WCAP-14701 are addressed in response to RAI #620.034.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.036

Question:

DCD Rev. 1, page 18.8-31. Please clarify the guidance utilized (Rev. 0 or Rev. 1 to NUREG-0711) as a basis for Figure 18.8-3, "Mapping of Human-System Interface to Operator Decision-Making Model." Please provide a discussion for using the respective guidance.

Westinghouse Response:

Figure 18.8-3 is an interpretation of a generic closed-loop model of controlled behavior (perception-cognition-action-feedback). This is based on Norbert Weiner's Cybernetic Model as extended by Jens Rasmussen and Dave Woods. NUREG-0711 does not specify the use of such a model. See WCAP-14695, Description of the Westinghouse Operator Decision-Making Model and Function-Based Task Analysis Methodology."

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.037

Question:

DCD Rev. 1, page 18.11-1. WCAP-15860, "Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan," was dated in April 2002. However, the date indicated in the DCD is March, 2002. Please correct.

Westinghouse Response:

Date (April 2002) will be corrected, as indicated below, in next revision of the DCD.

Design Control Document (DCD) Revision:

From DCD Table 1.6-1 (Sheet 17 of 20):

[WCAP-15860	*Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan, Revision 0, ~~March 2002~~April 2002]**

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.038

Question:

DCD Rev. 1, page 18.11-5. Please describe the guidance utilized in development of Figure 18.11-1 with respect to the HFE Verification and Validation Elements. If the guidance utilized is in NUREG-0711, Rev. 1, we recommend re-titling the figure to better represent the intent of the figure.

Westinghouse Response:

The five main activities specified for HFE Verification and Validation are recommended both by the original NUREG-0711, and by NUREG-0711 Rev.1. AP1000 DCD Section 18.13 maps the existing activities to the new guidance, to show that the process remains satisfactory (likewise, see responses to RAIs 620.007, 009, 010, and 018). Mapping allows the existing Figure 18.11-1 and supporting documents to remain correct, which exemplifies why mapping is desirable. To better represent the intent of the figure on page 18.11-1, it will be re-titled, "AP1000 HFE Verification and Validation".

Design Control Document (DCD) Revision:

[DCD p.18.11-5:]

Figure 18.11-1
AP1000 ~~Human Factors Engineering~~ Verification and Validation

[DCD Ch. 18 Table of Contents p.v:]

LIST OF FIGURES

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.039

Question:

DCD Rev. 1, pages 18.12-1, 18.12-2, paragraph 18.12.2. Please explain the basis for eliminating the terminology "soft controls" from the last sentence on page 18.12-1 and from the first sentence, second paragraph, on page 18.12-2.

Westinghouse Response:

In section 18.12.2, the term "soft controls" was deleted from the last sentence on page 18.12-1 in order to simplify the sentence, not to change its meaning.

In the second paragraph on page 18.12-2, the term "controls" (not "soft controls") was deleted from the first sentence because it was not relevant to the immediate discussion. This paragraph is now split in two, so that discussion of "Fixed-position controls" is clearly separated from discussion of "Fixed-position alarms and displays".

Design Control Document (DCD) Revision:

Fixed-position ~~controls,~~ alarms~~,~~ and displays are available at a fixed location and are continuously available, though not necessarily displayed, to the operator. Fixed-position displays can be accessed by the operator to monitor the plant status, based on indications from critical plant variables or parameters. Fixed-position alarms are designed to direct operator attention to the need to perform safety-related functions for which there is no automatic actuation function. **Although not continuously displayed, the fixed-position displays and alarms are quickly and easily retrievable.**

Fixed-position controls provide a means for manual reactor and turbine trip, and safety-related system/component actuation. Fixed-position controls are available to the operator to perform tasks in the operation of safety-related systems and components used to mitigate the consequences of an accident and to establish and maintain safe shutdown conditions following an accident. The fixed-position controls are a manual backup to the automatic protection signals provided by the protection and safety monitoring system.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.040

Question:

DCD Rev. 0, page 18.12-9. Please explain why containment hydrogen concentration has been eliminated from the AP1000 inventory as a fixed position control. (See previous question 620.005.)

Westinghouse Response:

See response to RAI 620.005.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.041

Question:

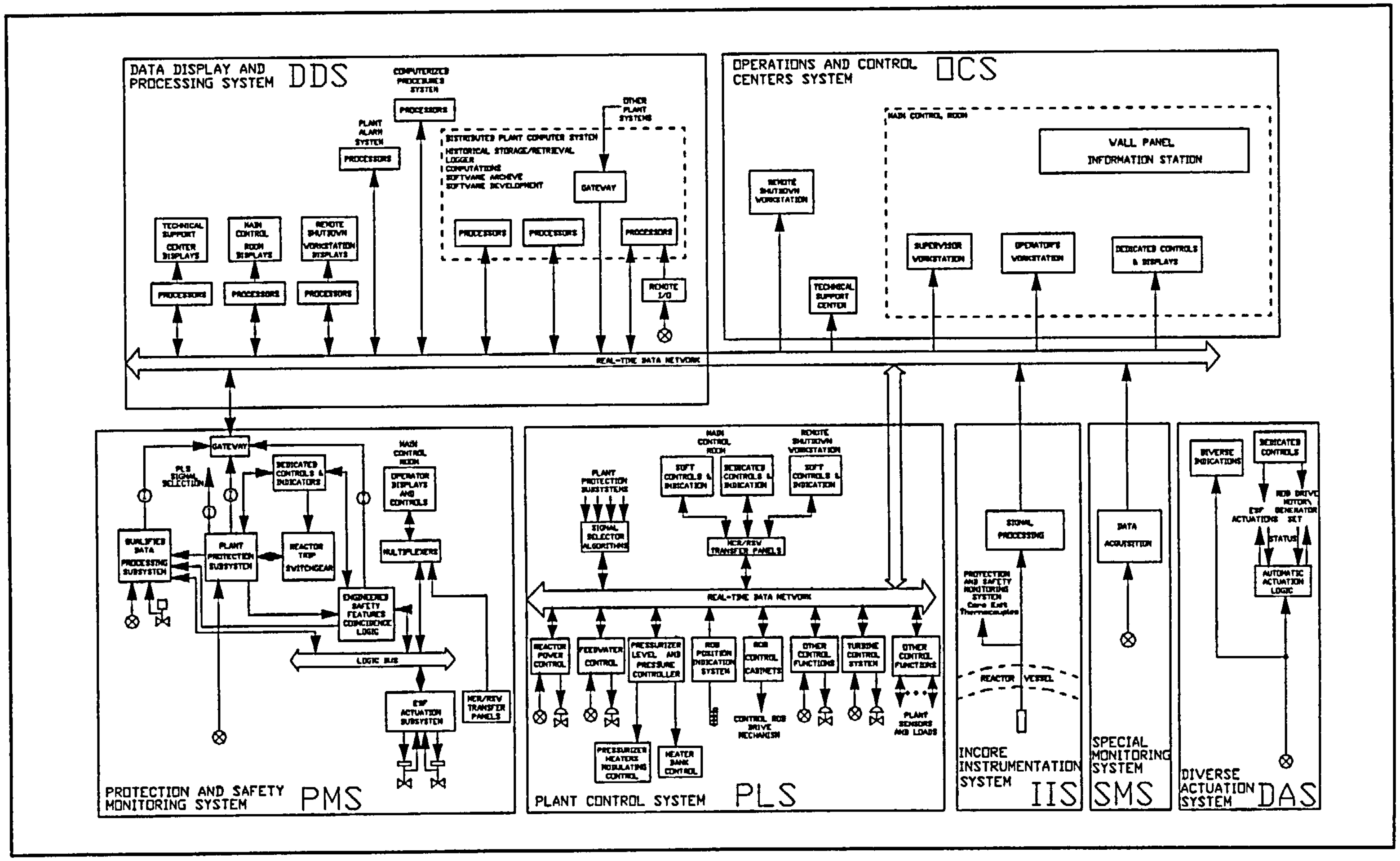
DCD Rev. 1, page 7.1-29. On page 7.1-4, both the emergency operations facility (EOF) and local control stations are identified as included in the operation and control centers system (OCS). Figure 7.1-1, "Instrumentation and Control Architecture," page 7.1-23, does not identify either the EOF or local control stations as included in the OCS. Please clarify/explain.

Westinghouse Response:

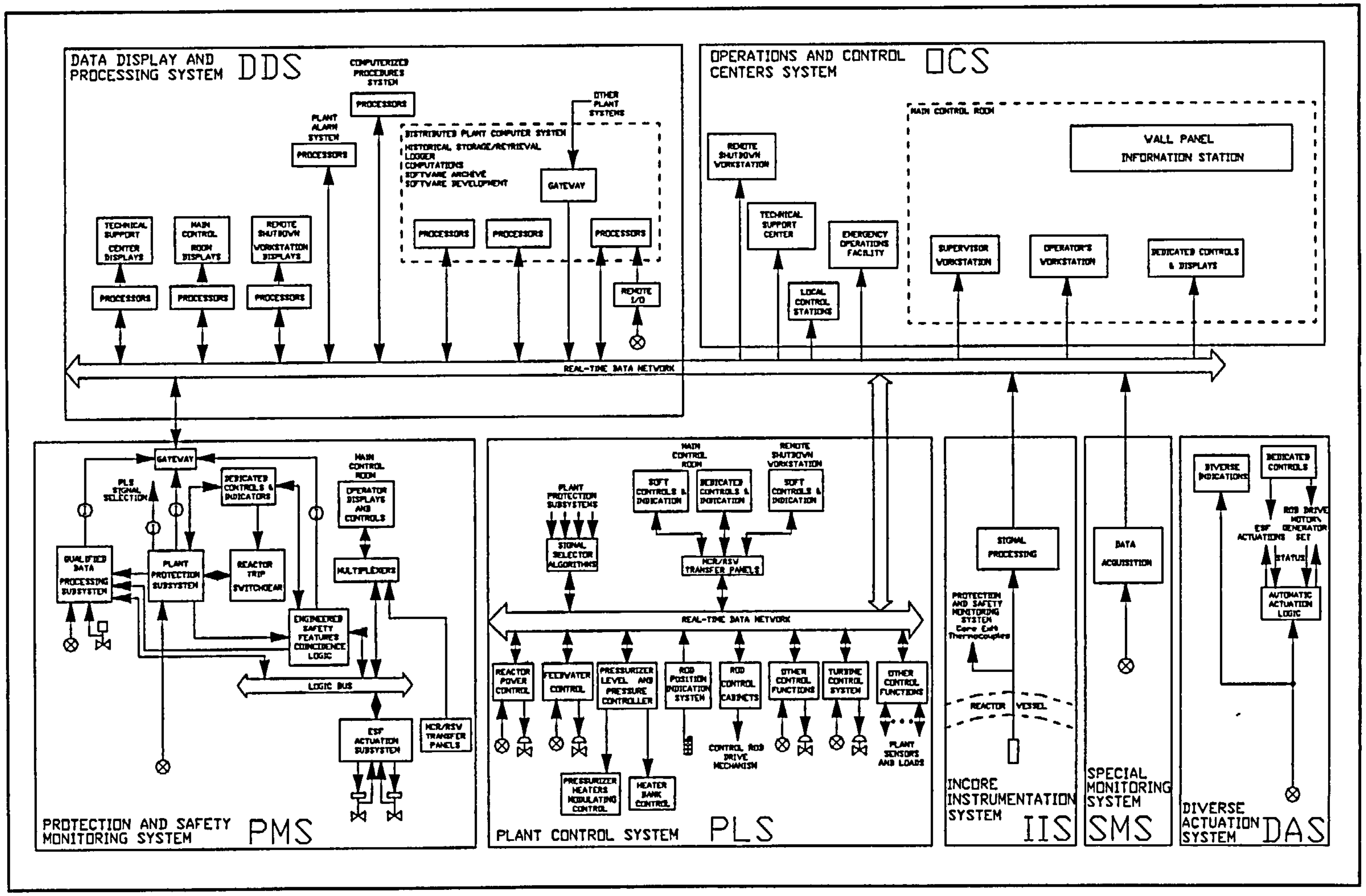
The EOF and local control stations are part of the operation and control centers system (OCS). In the next revision of the DCD Figure 7.1-1 will be updated, as indicated below, to include the EOF and the local control stations.

Design Control Document (DCD) Revision:

OLD FIGURE



NEW FIGURE



DATA DISPLAY AND PROCESSING SYSTEM DDS
COMPUTERIZED PROCEDURES SYSTEM
PROCESSORS
PLANT ALARM SYSTEM
PROCESSORS
OTHER PLANT SYSTEMS
DISTRIBUTED PLANT COMPUTER SYSTEM
HISTORICAL STORAGE/RETRIEVAL
LOGGER
COMPUTATIONS
SOFTWARE ARCHIVE
SOFTWARE DEVELOPMENT
GATEWAY
TECHNICAL SUPPORT CENTER DISPLAYS
MAIN CONTROL ROOM DISPLAYS
REMOTE SHUTDOWN WORKSTATION DISPLAYS
PROCESSORS
PROCESSORS
PROCESSORS
PROCESSORS
PROCESSORS
PROCESSORS
REMOTE I/O
REAL-TIME DATA NETWORK
OPERATIONS AND CONTROL CENTERS SYSTEM OCS
MAIN CONTROL ROOM
WALL PANEL INFORMATION STATION
REMOTE SHUTDOWN WORKSTATION
TECHNICAL SUPPORT CENTER
LOCAL CONTROL STATIONS
EMERGENCY OPERATIONS FACILITY
SUPERVISOR WORKSTATION
OPERATOR'S WORKSTATION
DEDICATED CONTROLS & DISPLAYS
GATEWAY
PLS SIGNAL SELECTION
DEDICATED CONTROLS & INDICATORS
MAIN CONTROL ROOM
OPERATOR DISPLAYS AND CONTROLS
QUALIFIED DATA PROCESSING SUBSYSTEM
PLANT PROTECTION SUBSYSTEM
REACTOR TRIP SWITCHGEAR
MULTIPLEXERS
ENGINEERED SAFETY FEATURES COINCIDENCE LOGIC
LOGIC BUS
ESF ACTUATION SUBSYSTEM
MCR/RSW TRANSFER PANELS
PROTECTION AND SAFETY MONITORING SYSTEM PMS
PLANT PROTECTION SUBSYSTEMS
SIGNAL SELECTOR ALGORITHMS
MAIN CONTROL ROOM
SOFT CONTROLS & INDICATION
DEDICATED CONTROLS & INDICATION
REMOTE SHUTDOWN WORKSTATION
SOFT CONTROLS & INDICATION
MCR/RSW TRANSFER PANELS
REAL-TIME DATA NETWORK
REACTOR POWER CONTROL
FEEDWATER CONTROL
PRESSURIZER LEVEL AND PRESSURE CONTROLLER
ROD POSITION INDICATION SYSTEM
ROD CONTROL CABINETS
OTHER CONTROL FUNCTIONS
TURBINE CONTROL SYSTEM
OTHER CONTROL FUNCTIONS
CONTROL ROD DRIVE MECHANISM
PLANT SENSORS AND LOADS
PRESSURIZER HEATERS MODULATING CONTROL
HEATER BANK CONTROL
PLANT CONTROL SYSTEM PLS
SIGNAL PROCESSING
PROTECTION AND SAFETY MONITORING SYSTEM Core Exit Thermocouples
REACTOR VESSEL
INCORE INSTRUMENTATION SYSTEM IIS
DATA ACQUISITION
SPECIAL MONITORING SYSTEM SMS
DIVERSE INDICATIONS
DEDICATED CONTROLS
ESF ACTUATIONS
ROD DRIVE MOTOR GENERATOR SET
STATUS
AUTOMATIC ACTUATION LOGIC
DIVERSE ACTUATION SYSTEM DAS

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.042

Question:

WCAP-15847, page 1-1. Does the WCAP contain all or some examples of the "pertinent procedures" applicable to the AP1000 design? If all the pertinent procedures are not included, how does Westinghouse propose to provide the remainder?

Westinghouse Response:

WCAP-15847 contains examples of all the pertinent procedures related to Sections 18.2 and 18.8 of the AP1000 DCD. The AP1000 Program Operating Procedures manual, prepared in accordance with our quality assurance program described in DCD Chapter 17, includes procedures in addition to those provided in WCAP-15847. The additional procedures in APP-GW-GAP-100 cover topics that are not relevant to Sections 18.2 or 18.8 of the AP1000 DCD. WCAP-15847 contains the revision of included procedures that were in force at the time of its issue. Westinghouse does not intend to revise WCAP-15847 for each revision of the AP1000 Program Operating Procedure manual. Therefore, the printed procedures in WCAP-15847 are examples. The AP1000 Program Operating Procedure manual is available for review by the NRC in Westinghouse offices.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 620.043

Question:

WCAP-15847, page 2-1. Reference 1 should be revised if it is to remain consistent with current NRC guidance.

Westinghouse Response:

WCAP-15847, page 2-1 will be revised, as indicated below.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

From WCAP-15847, page 1-1:

1 INTRODUCTION

Chapter 18 of the AP1000 Design Control Document (DCD) contains the AP1000 Design Certification information for Human Factors Engineering~~. The NRC is reviewing this information against NUREG-0711 (Reference 1)~~.

From WCAP-15847, page 2-1:

2.0 REFERENCES

1. <u>Reference Deleted.</u> ~~NUREG-0711, Human Factors Engineering Program Review Model, July 1994.~~
2. WCAP-12601 Revision 19, Westinghouse AP600 Program Operating Procedures Document.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.001

Question:

The US Nuclear Regulatory Commission (NRC) staff is reviewing your design certification request for the AP1000 standard plant design. Included in the staff's effort is the review of the severe accident management design alternatives. As part of its efforts in this area, the staff plans to use the MELCOR code to perform baseline analyses for risk significant sequences for the AP1000. The results will be used to develop input for the draft safety evaluation report (DSER), and final safety evaluation report (FSER).

The staff issued its design certification review schedule in its letter dated July 12, 2002. In this letter the staff documented its target date for issuance of the DSER by June 16, 2003. Efficient use of NRC resources is necessary to meet this target date. Timely preparation of the MELCOR input deck for the AP1000 design will facilitate efficient use of NRC resources. Timely receipt of the MAAP input decks will ensure completion of MELCOR calculation in support of the established schedule.

Also, comparison with the AP600 MAAP input deck is critical to performing a complete review of the AP1000 design. As such, the staff requests Westinghouse Electric Company to submit the MAAP code input decks for AP1000 and AP600 (electronic versions are preferred).

Westinghouse Response:

Westinghouse document WCAP-15938-P, "AP1000 MAAP4 Parameter File and Input Deck for Probabilistic Risk Assessment," Rev. 0 was submitted in Westinghouse letter DCP/NRC1522, dated September 10, 2002. This report is similar to WCAP-14729, "AP600 MAAP4 Parameter File and Input Deck for Probabilistic Risk Assessment," Rev. 0 that was submitted to the NRC as part of AP600 Design Certification.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.004

Question:

Section A.2.4 of Appendix A to the AP1000 PRA states that the MAAP4 code is used to justify post automatic depressurization system (ADS) success criteria, both for short-term and long-term core cooling. The MAAP4 code has not been submitted for the Nuclear Regulatory Commission (NRC) staff review, therefore the NRC staff must review those portions of the code relevant to each application including critical models, assumptions, code input, and experimental verification.

In order that the staff may exercise the MAAP4 code for AP1000 and evaluate the effect of the various user options selected by Westinghouse, please provide the parameter input file and the sequence input file used to describe the 3.5 inch hot leg break with manual automatic depressurization system stage-4 (ADS-4) actuation that was analyzed by Westinghouse using MAAP4 as described in Section A2.4.2 of Appendix A to the AP1000 PRA. Please provide this input in electronic form.

Westinghouse Response:

Westinghouse has used MAAP4 as a screening tool to identify PRA success sequences. It was also used for this purpose for the AP600 PRA. As part of AP600 Design Certification, the MAAP code was benchmarked against NOTRUMP, as a screening tool, to identify PRA success sequences. Westinghouse has used the MAAP4 code for AP1000 in the same application as was approved for the AP600. The AP1000 MAAP4 Parameter File, Revision 1 and Input Deck for the PRA analysis were submitted to the NRC as WCAP-15938-P. This WCAP presents the MAAP4 parameter file that has been specifically developed for the level 2 PRA success criteria analysis.

The attached Table 720.004-1 provides the input file for the AP1000 MAAP analysis of the 3.5-inch hot leg break level 1 success criteria analysis. The level 1 success criteria analysis was performed with revision 0 of the AP1000 MAAP Parameter File. Table 720.004-2 provides a listing of the differences with the MAAP Parameter file used for the 3.5-inch hot leg break calculations and the information presented in WCAP-15938-P. The differences are due to two reasons:

1. Additional modeling necessary to perform the Level 1 success criteria calculations.

2. Differences in the MAAP Parameter File that were incorporated between Revision 0 and Revision 1. The changes made from revision 0 to revision 1 were made based on level 2 PRA analysis requirements, and are related to post-core uncovery and containment severe accident modeling. As such, these revisions do not impact the level 1 PRA success criteria analysis.

A description of the changes to the parameter file incorporated in Revision 1 are outlined in WCAP-15938-P and are delineated below:

1. The core shroud baffle model is refined to properly model corium relocation in-vessel.
2. RCS cold leg sub-volumes are modified to better model post-core uncovery natural circulation for high pressure core damage sequences
3. The CMT Room water level vs. volume is updated to more accurately model containment flooding
4. The curbs of the PXS/CVS compartments are updated to reflect the AP1000 design. This provides a more accurate containment flooding evaluation
5. Added the sump drain to the refueling canal to more accurately model containment flooding
6. PCS friction values were updated in containment modeling
7. Added sump model to IRWST and adjusted associated elevation for gravity recirculation modeling in containment
8. PCS shell material properties updated in containment modeling
9. Updated fuel rod plenums and below active core masses for in-vessel core debris relocation modeling
10. Added non-safety spray modeling in containment

Design Control Document (DCD) Revision:

None

PRA Revision:

None

Table 720.004P-1 AP1000 MAAP4 Input for Success Criteria Run

a, c

a,c

a, c

a, c

a, c

,

Table 720.004-2 AP1000 MAAP4 Parameter File for 3.5" HL Break Case – Differences from WCAP-15938-P

a, c

a, c

a, c

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.007

Question:

Figure A3.2-1 is titled "Minimum Core Mixture Level for Spectrum of Break Sizes (with RNS [normal residual heat removal system] Injection)." Figure A3.2-26 is titled "Minimum Core Mixture Level for Spectrum of Break Sizes (with IRWST Injection)."

Are "RNS Injection" and "IRWST Injection" in these respective titles reversed?

Westinghouse Response:

The respective titles are reversed and will be corrected accordingly.

Design Control Document (DCD) Revision:

None

PRA Revision:

The figures A3.2-1 and A.3.2-26 with corrected titles will be included in the next revision (revision 1) of the PRA Report (see Attachment).

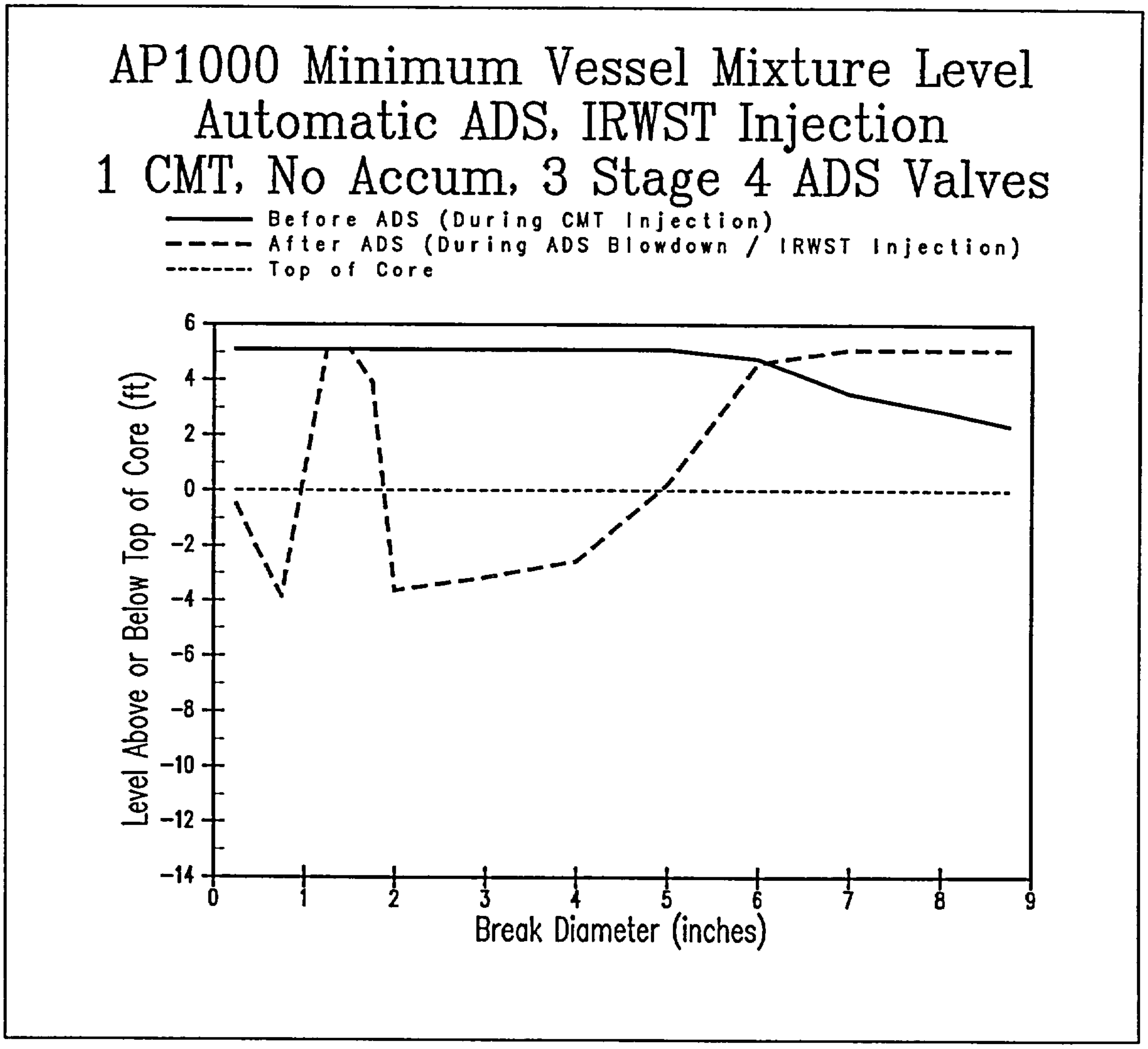


Figure A3.2-1

Minimum Core Mixture Level for Spectrum of Break Sizes (with ~~RNS~~ IRWSTInjection)

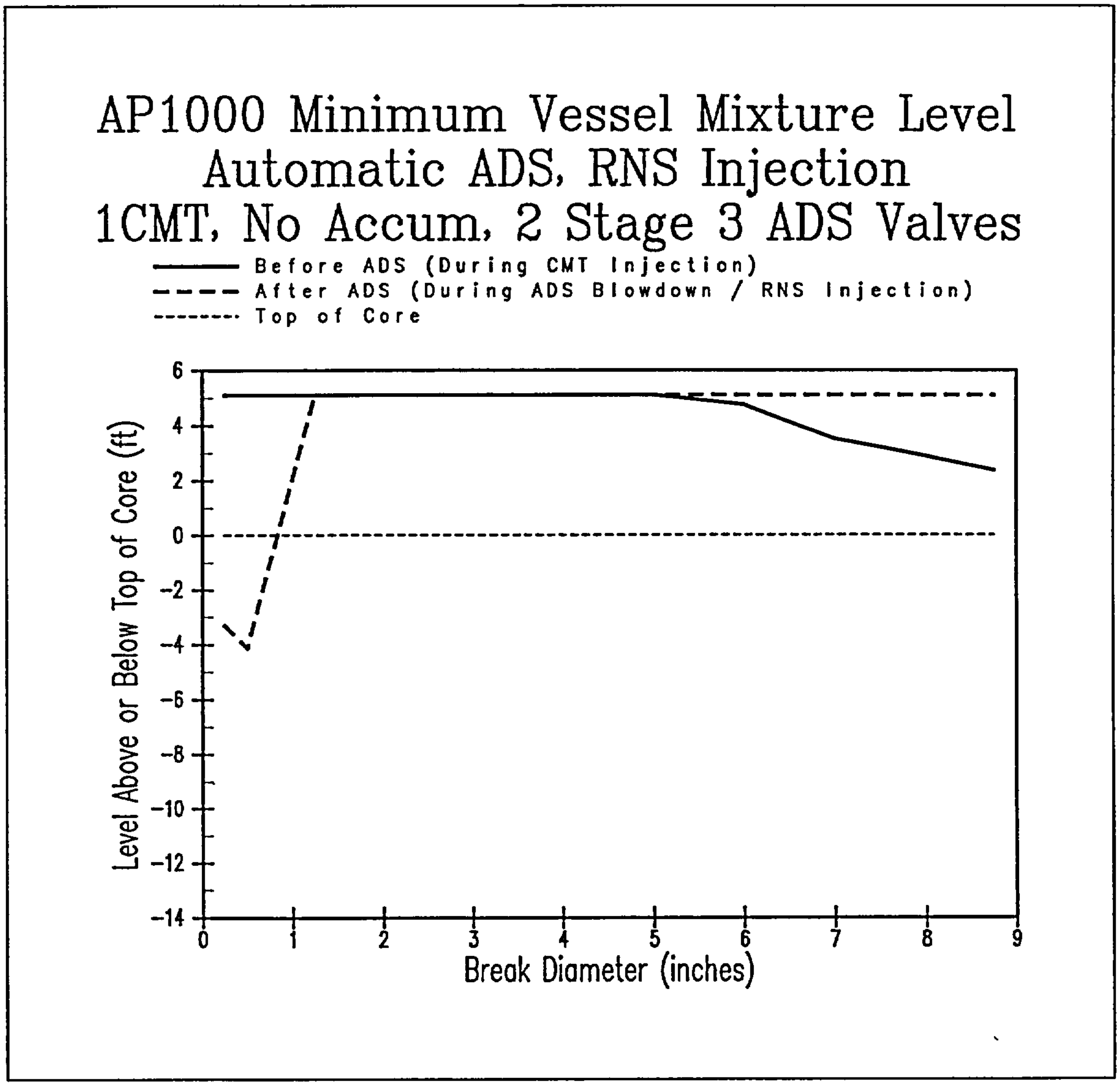


Figure A3.2-26

Minimum Core Mixture Level for Spectrum of Break Sizes (with ~~IRWST~~ RNS Injection)

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.011

Question:

In the anticipated transient without scram (ATWS) analysis described in Section A4, both Figures A4.2.1-2 and A4.2.2-2 (a typographic error in the figure) for the equilibrium cycles and the first cycle, respectively, show that the liquid volume in the pressurizer reaches 2200 cubic feet (ft^3) after about 105 seconds. However, the AP1000 pressurizer design has an internal volume of 2100 ft^3.

A. Explain the inconsistency in the pressurizer volume.

B. If the pressurizer volume of 2100 ft^3 was used in the ATWS analyses, would the peak RCS pressure exceed 3200 pounds-per-square inch absolute (psia) for the cases analyzed?

C. On Figures A4.2.1-4 and A4.2.2-4, explain why the nuclear power begins at 20% rated power.

Westinghouse Response:

A. The 2200 cubic feet (ft^3) volume refers to the pressurizer volume (2100 ft^3) and the surge line volume (99.7 ft^3). The figures A4.2.1-2 and A4.2.2-2 will be revised to indicate the pressurizer volume includes the volume of the surge line.

B. See above answer

C. The analysis is initiated for full power. The Figures A4.2.1-4 and A4.2.2-4 are only a cut-off of figures with 100% rated power (see figures in Attachment 1)

Design Control Document (DCD) Revision:

None

PRA Revision:

The revised figures A4.2.1-2 and A4.2.2-2 will be included in the next revision (revision 1) of the PRA Report (see Attachment 2).

Attachment 1

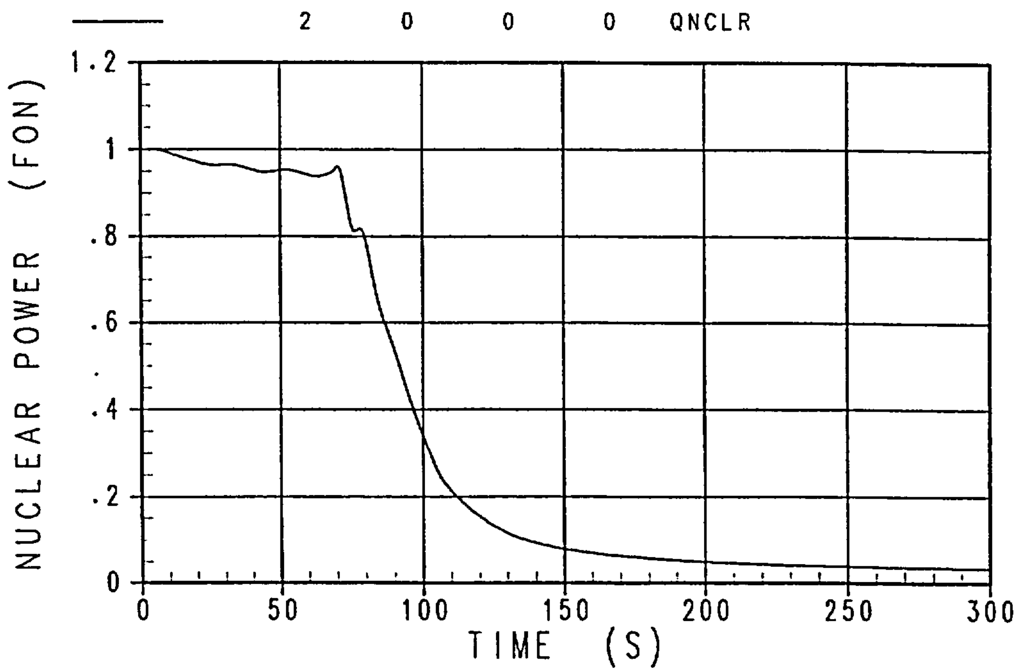


Figure 1 - ATWS Eq. Cycle: Nuclear power used for Figure A4.2.1-4

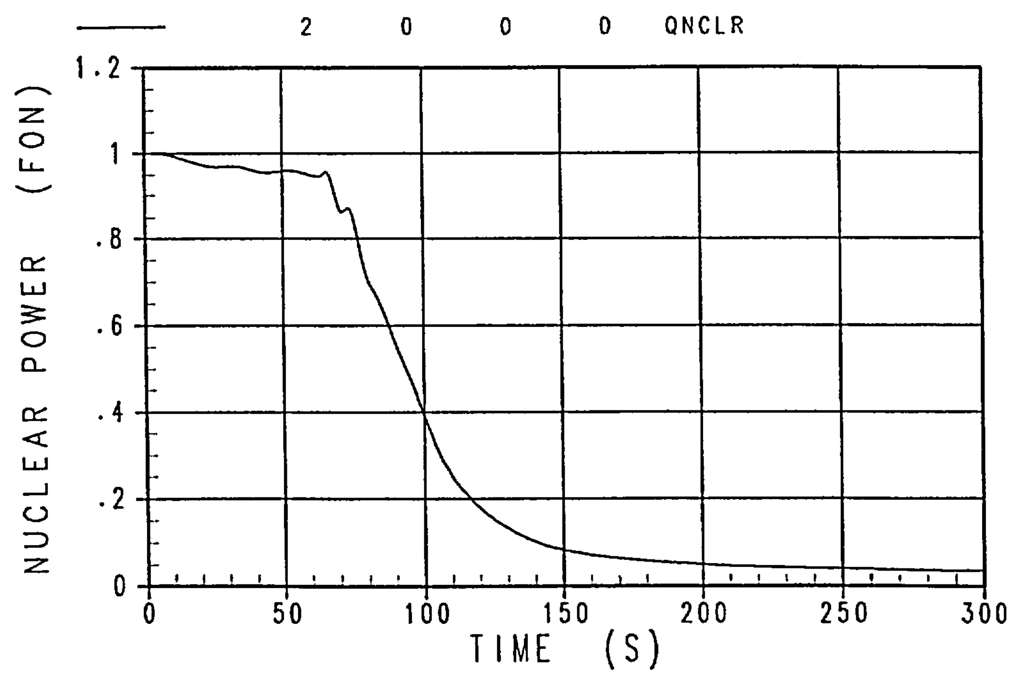


Figure 2 - ATWS First Cycle: Nuclear power used for Figure A4.2.2-4

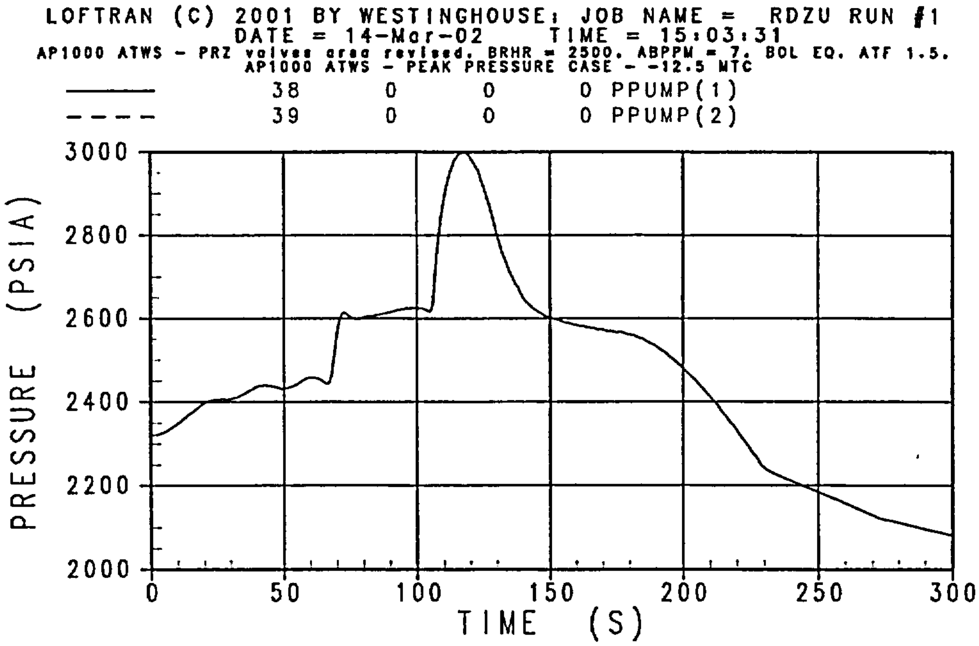
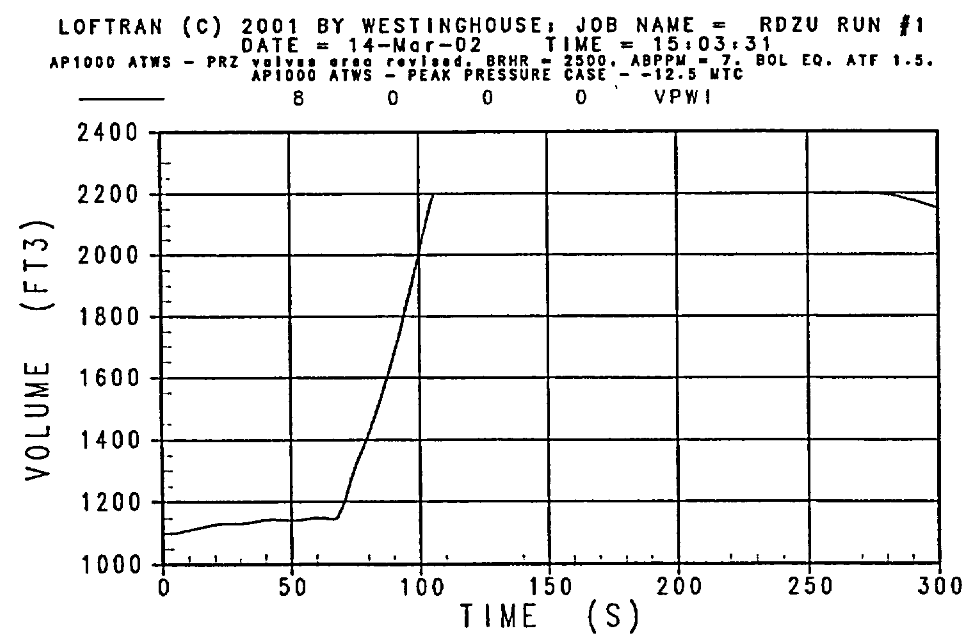


Figure A4.2.1-1
ATWS Eq. Cycle – RCS Pressure



Note: The pressurizer volume includes the volume of the surge line

Figure A4.2.1-2
ATWS Eq. Cycle – PRZ Volume

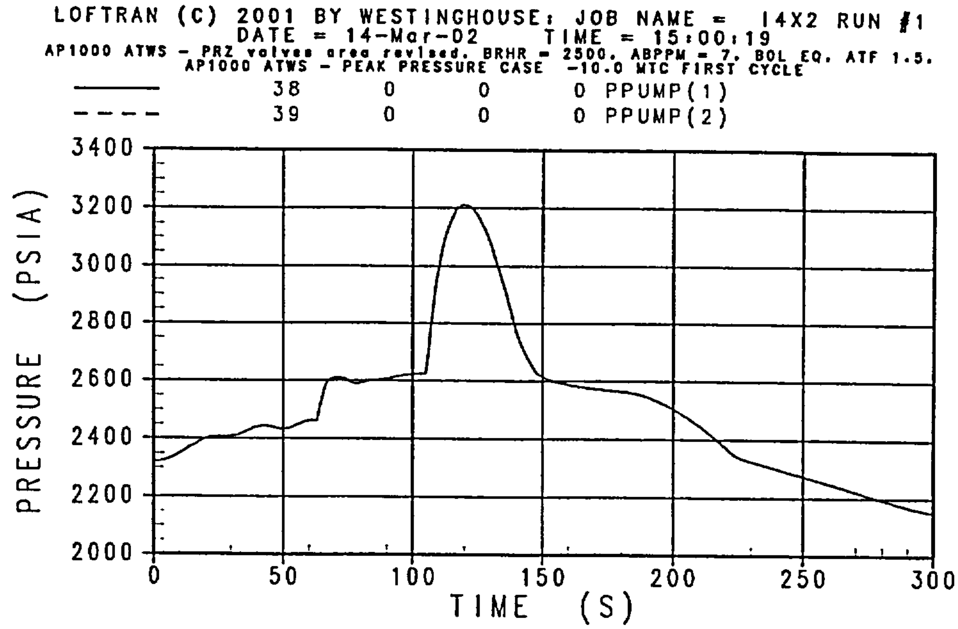
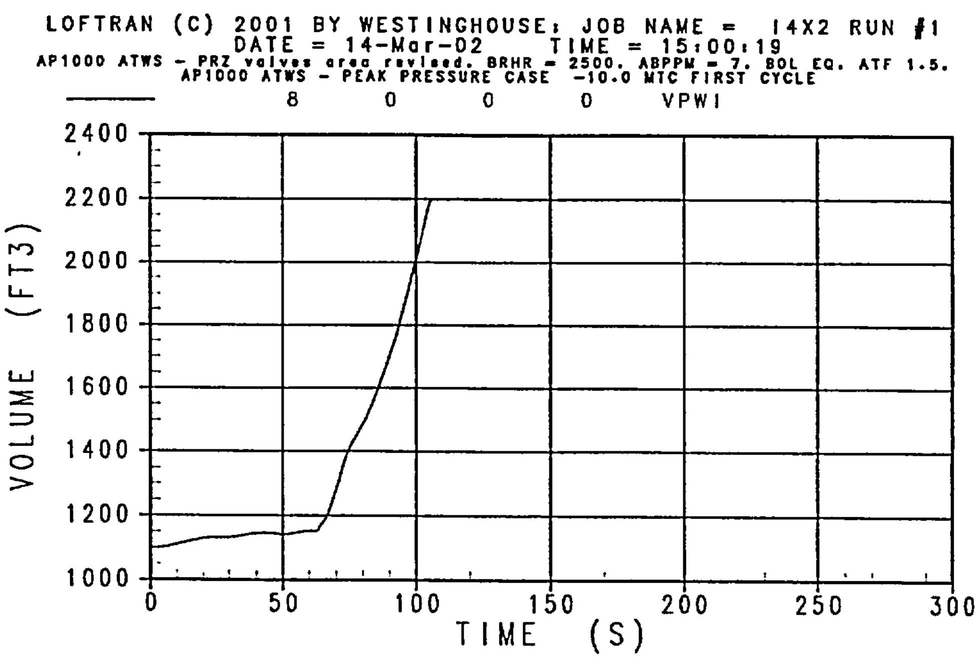


Figure A4.2.2-1
ATWS First Cycle – RCS Pressure



Note: The pressurizer volume includes the volume of the surge line

Figure A4.2.12-2
ATWS First. Cycle – PRZ Volume

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.014

Question:

Section A5.1 indicates that Westinghouse's purpose is to bound the T/H uncertainly for the various success paths rather than to quantify it. For case UC3 (a double ended direct vessel injection (DVI) line break for which the accumulators are assumed to have failed) two of the assumptions may not be bounding. Please supply supporting analyses to demonstrate that bounding assumptions have been made.

A. The CMT isolation valve on the broken DVI side is assumed to have failed closed. If the CMT valve was assumed to open as designed, earlier ADS4 actuation would occur from draining of the affected CMT but more coolant would be lost. Justify that to fail the isolation valve closed is bounding.

B. The IRWST injection valve on the broken DVI side is assumed to have failed closed. If the IRWST valve was assumed to open as designed, IRWST water would drain on the containment floor lowering the IRWST water level and delaying the time when water would begin to enter the reactor through the intact DVI line. Justify that to fail the IRWST injection valve closed is bounding.

Westinghouse Response:

A. A NOTRUMP calculation was made with the same assumptions as case UC3 except that the CMT isolation valve on the broken side is assumed to open as designed. The core mixture level is shown in Figure 1 for this case and for the original UC3 case with the failed CMT isolation valve on the broken side. The first curve shows that the level swells earlier due to the earlier ADS actuation. The core remains covered as the intact CMT injects, then uncovers after the CMT empties and before IRWST injection is established. The second curve is the original UC3 case with later ADS actuation.

   Core uncovery occurs earlier for the case where ADS is actuated on the broken side CMT, but the resulting depth of uncovery is less and the recovery is quicker. Figure 2 shows the peak clad temperature for both cases. The original UC3 case peak clad temperature was determined to be 1570F, and occurred at 2207 seconds after the start of the accident. For the case where ADS is actuated by the broken side CMT, the peak clad temperature is 1449F and occurs at 2080 seconds. Thus, the original assumption that the CMT isolation valve on the broken side fails is conservative for this case.

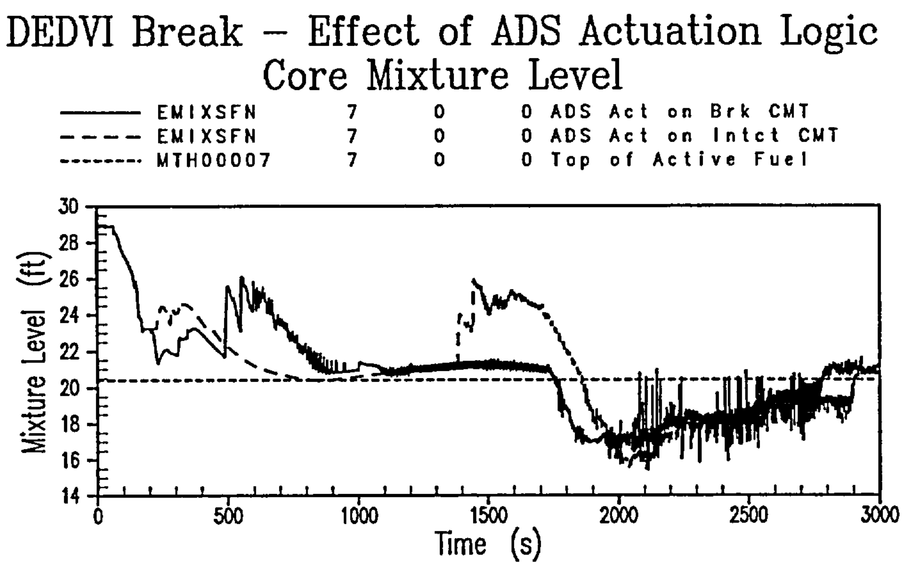


Figure 1: Core Mixture Level for DEDVI Break – Effect of ADS Actuation Time

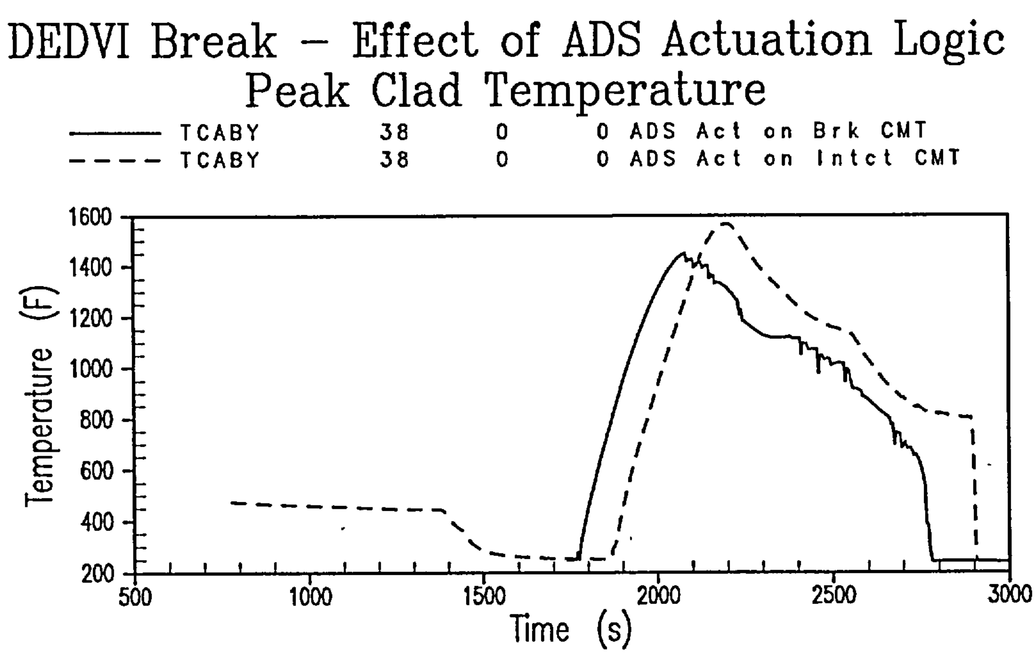


Figure 2: Peak Clad Temperature for DEDVI Break – Effect of ADS Actuation Time

B. The IRWST isolation valve is assumed to operate as designed for all of the cases analyzed. The IRWST flow and level are shown in Figure 3 for the double-ended DVI break (Case UC3). For the broken DVI line, flow from the IRWST starts after the CMT drains (~1400 seconds). This flow causes a reduction in the IRWST level from the initial elevation of 56.9 ft to about 55.6 ft at 1950 seconds. At this time, the IRWST begins to inject into the intact DVI line. The reduction in the IRWST liquid level results in slightly lower injection flow due to the reduction in static head. Thus, the analyzed case with the IRWST isolation valves open is the bounding case.

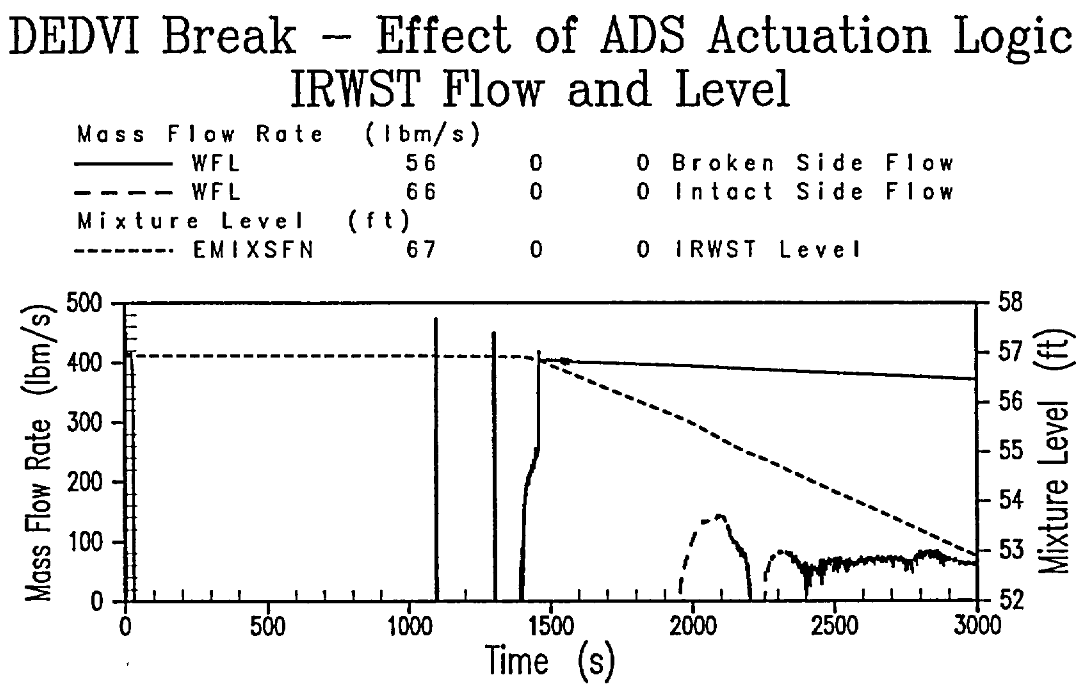


Figure 3: IRWST Flow and Level for DEDVI Break

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.016

Question:

Section A5.2.1.3.3 describes the results from T/H Uncertainly Case No. 3 (UC3) as an estimated cladding heat-up of well less than 2000°F. Section A5.2.1.2 states that the cladding heat-up calculations were made using the LOCTA code.

A. Describe the how this calculation was made and why the results are considered to be only estimated values as stated on page A-36.

B. Provide the PCT calculated by LOCTA as a function of time.

Westinghouse Response:

A. For this case in the PRA, a conservative calculation was performed assuming adiabatic heatup of the fuel rod for the time that NOTRUMP predicts that the core is uncovered. Thus, the peak clad temperatures are conservatively high since no steam cooling is assumed.

B. Figure 1 shows the peak clad temperature results as a function of time for case UC3 using the LOCTA code. PRA section A5.2.1.3.3 will be revised to include this figure and to include appropriate changes to the text.

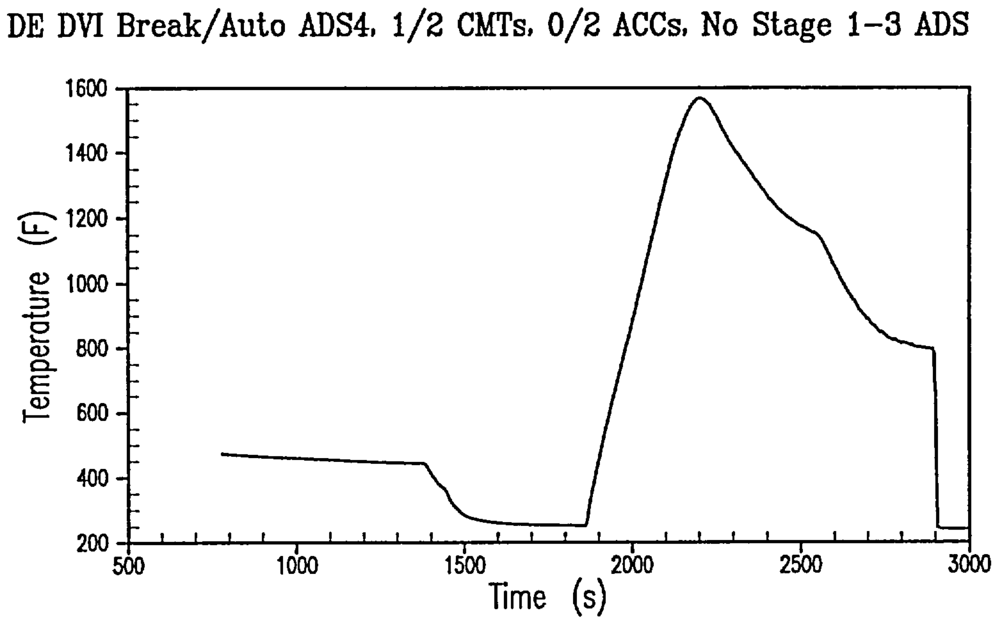


Figure 1: Peak Clad Temperature for Case UC3 as Calculated by LOCTA

Design Control Document (DCD) Revision:

None

PRA Revision:

The LOCTA figure will be included in the next revision of the PRA and the related section A5.2.1.3.3 will be updated accordingly (see Attachment).

A5.2.1.3.2 Case UC2 Results

Case UC2 is a double-ended rupture of an 8.0-inch CMT balance line (inside diameter of 6.8 inches). This break is very much like a break in the RCS cold leg. Both CMTs are assumed to fail. In addition, the break is assumed to be in a location that prevents the faulted CMT from draining. Therefore, operation action to actuate the ADS must be assumed.

- Credit for PRHR HX operation
- Credit for 2 out of 2 accumulators
- ADS stages 1, 2, and 3 fail to open
- Credit for 4 out of 4 ADS stage 4 at 20 minutes (1200 seconds)
- Only 1 of 2 IRWST lines is assumed to inject. Further, failure of 1 of the 2 parallel paths in the IRWST line to open is assumed
- Credit for containment isolation; containment pressure assumed to be 25 psia

Figures A5.2-14 through A5.2-25 provide plots of the plant response and Table A5.2-3 provides the sequence of key events. Figures A5.2-16 and A5.2-17 show the liquid and steam break flow rates that lead to depressurization of the RCS, as seen in Figure A5.2-14, and draining of the RCS pressurizer (Figure A5.2-15). Due to the large size of the break and lack of CMT injection, the RCS rapidly depressurizes and accumulator injection begins at around 290 seconds. Both accumulators continue to inject until around 1350 seconds, providing adequate injection to keep the core covered. At 20 minutes, the operator opens all 4 ADS stage 4 valves, which results in a further depressurization down to less than 50 psi. The depressurization brought on by the opening of ADS stage 4 is sufficient to allow for IRWST injection, which begins at 1450 seconds (250 seconds after opening ADS stage 4). The IRWST injection rate is sufficient to prevent core uncovery, stabilizing at about 150 lbm/sec, which matches the losses out of the break and ADS. Since core uncovery does not occur for case UC2B, the clad does not experience a heat-up, and a clad heat-up calculation is not performed.

A5.2.1.3.3 Case UC3 Results

Case UC3 is an double-ended rupture of the DVI line piping. On the vessel side, the break is limited to 4 inches in diameter by an orifice. On the passive injection side, the break is limited by the CMT discharge orifice. Additional assumptions are:

- The CMT on the intact loop provides injection to the RCS. The CMT isolation valve on the faulted loop is assumed to fail to open.
- Both accumulators fail to inject.
- ADS stages 1, 2, and 3 fail to open.

- Credit for 3 of 4 ADS stage 4, automatically actuated due to draining of the intact CMT.
- Only 1 of 2 IRWST lines is assumed to inject. Further, failure of 1 of the 2 parallel paths in an IRWST line to open is assumed.
- No credit for containment isolation; containment pressure assumed to be 14.7 psia.

Figures A5.2-26 through A5.2-36 provide plots of the plant response and Table A5.2-4 provides the sequence of key events. Figures A5.2-28 and A5.2-29 show the liquid and steam break flow rates on the vessel side of the broken DVI piping, which leads to RCS depressurization as seen in Figure A5.2-26, and draining of the RCS pressurizer (Figure A5.2-27). The intact CMT begins to recirculate at about 40 seconds and drain down starts at 280 seconds. The intact CMT drains, resulting in ADS stage 4 actuation at 1380 seconds. The actuation of ADS stage 4 results in a depressurization down to less than 50 psia. The depressurization brought on by the opening of ADS stage 4 is sufficient to allow for IRWST injection, which begins at 1960 seconds (580 seconds after ADS stage 4 opens). The IRWST injection rate is sufficient to recover the core, exceeding losses through the break and ADS stage 4 at 2890 seconds.

~~The cladding heatup is estimated to be well less than 2000°F,~~ **Figure A5.2-37 shows that the cladding heatup as calculated by LOCTA** is comfortably below the 2200°F acceptance criteria. Note that this case is a conservative case that bounds two different risk significant cases. The actual risk significant cases will have less core uncovery and lower PCTs.

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

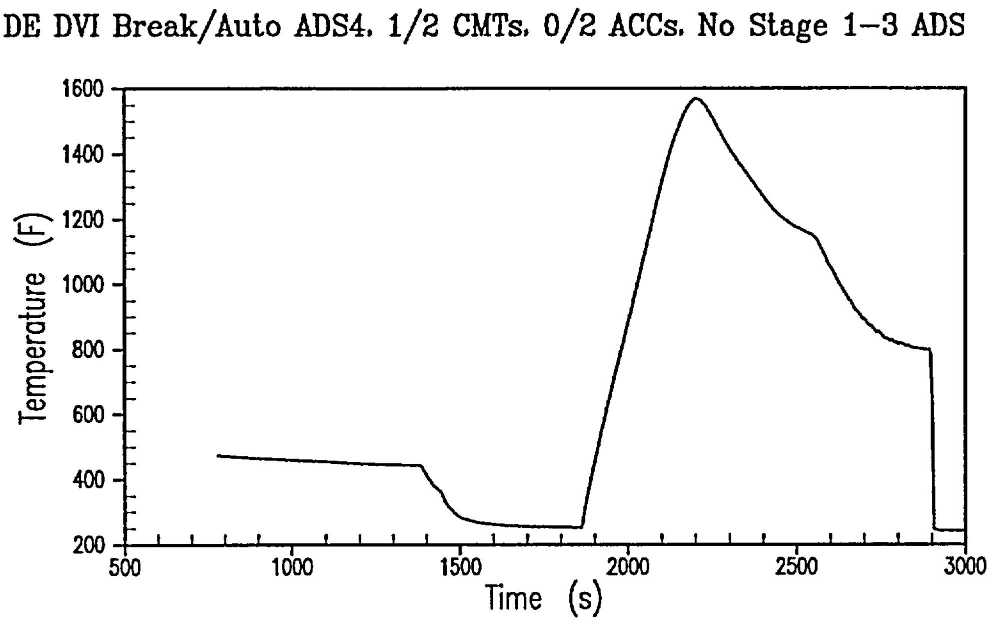


Figure A5.2-37

Case UC3 Peak Clad Temperature

RAI Number: 720.019

Question:

The T/H uncertainty evaluation in Section A5 utilized the NOTRUMP code to evaluate minimum sets of equipment needed to prevent core damage. The NOTRUMP code was qualified for AP600 by comparisons of code predictions to test data from integral and separate effects tests facilities as discussed in WCAP-14807. Following the pre-application review for AP1000 the NRC staff informed Westinghouse that "none of the integral effects facilities are sufficiently well scaled so that they provide an acceptable data base to validate T/H codes for the high rate of liquid entrainment that are expected to occur in the AP1000 during ADS4 and IRWST injection periods of a small break LOCA. --- The staff concludes that Westinghouse must either obtain entrainment test data applicable to the AP1000 steam flow rates for code verification or provide justification for the entrainment models to be used for the AP1000 applications."

Provide an evaluation showing the effect on core cooling for the various small-break LOCA success paths resulting from the uncertainty in ADS4 liquid entrainment.

Westinghouse Response:

The analyses presented in Appendix A5 of the PRA are the thermal-hydraulic uncertainty evaluations performed for AP1000. In PRA, success sequences are typically justified with best-estimate type analyses, where nominal assumptions regarding key factors such as decay heat, piping L/D, and system performance characteristics are made. In addition, for the AP600 and AP1000, Westinghouse has performed selected analyses to address thermal-hydraulic uncertainties in the passive systems as they relate to the PRA. These analyses are performed using the design basis analysis codes, with the purpose being to demonstrate that selected higher risk - lower margin scenarios are successful, even for the case where conservative bounding assumptions are made. The analyses in question are these thermal-hydraulic uncertainty evaluations. Results of these analyses show some core uncovery, however the calculated PCT values are still below 2200 F, and these cases are assumed success in the PRA.

WCAP-15833, Appendix A provides an evaluation of the effects of liquid entrainment in the AP1000 SBLOCA analysis. One of the major conclusions of this evaluation is that pool-type liquid entrainment (based upon work by Kataoka-Ishii) that occurs in the reactor vessel decreases significantly as the two-phase mixture level falls below the hot leg elevation. In fact, it decreases more than two orders of magnitude within the first ½ meter below the hot leg. This is due to the nature of the liquid entrainment phenomena whereby the entrainment of liquid droplets is not only a function of the gas velocity but also a strong function of height above the two-phase mixture surface. As the height above the two-phase mixture surface increases, the entrainment of liquid decreases significantly. This can readily be seen in WCAP-15833, Appendix A, Figure A.3-3 for DE-DVI Break and Figure A.3-8 for the Inadvertent ADS event.

So therefore, for the analyses presented in Appendix A5 of the PRA, the effects of entrainment on the calculated PCT is negligible. As demonstrated in WCAP-15833, the importance of the uncertainty associated with liquid entrainment in the reactor vessel decreases significantly in situations where the two-phase mixture level is below the hot leg since the liquid entrainment decreases significantly. The effect of the uncertainty in liquid entrainment from the reactor vessel in these situations does not significantly affect the calculated PCT for these cases, and the results are acceptable for the purposes of assessing the PRA success sequences.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.020

Question:

As described in Section A5.2.1.2, the LOCTA code is used for cladding heatup calculations in the PRA when noticeable core uncovery is predicted. The LOCTA code has been approved by the NRC staff for core heatup analyses for operating reactors. The high void fractions and low pressures predicted within the core of AP1000 following ADS4 activation may lie outside the range of the critical heat flux and transition boiling correlations contained in the code. In its March 15, 2002, letter to Westinghouse, the staff identified the need for the justification of the methodology used to calculate PCT in the event that the core becomes uncovered during a small-break LOCA.

A. Provide the pressure and void fraction limits for these correlations and demonstrate that the correlations are being used within these limits.

B. Since the AP1000 PRA small-break LOCAs result in core uncovery, provide further justification why the LOCTA code is applicable for the AP1000 PRA analyses.

Westinghouse Response:

The LOCTA code is the Westinghouse design basis analysis code used to perform fuel rod heatup calculations when core uncovery is predicted. The code has been approved by the NRC for use in LOCA analysis (both Appendix-K Small Break and Large Break LOCA) in accordance with 10CFR 50.46. For clarification, this question pertains to analysis that was performed in support of the AP1000 PRA success criteria.

A. For small break analysis, the LOCTA code takes boundary conditions from the NOTRUMP code. The boundary conditions passed from NOTRUMP to LOCTA are as follows:

1. Transient time
2. Pressure at the top of the core
3. Two-phase mixture level relative to the bottom of the active fuel region
4. Core inlet enthalpy
5. Normalized power fraction
6. Core average exit mass flowrate for all uncovered nodes

When core uncovery is predicted by NOTRUMP, LOCTA performs a single-phase vapor only cooling calculation for the uncovered portion of the fuel rod. As such, the LOCTA code as utilized for SBLOCA has no dependence on void fraction as only steam cooling heat transfer is modeled in the uncovered region.

B. The LOCTA code was used for similar calculations for AP600 PRA success criteria calculations, where core uncovery was predicted to occur. LOCTA has also been used to predict peak clad temperature for operating plants with cores similar to AP1000 including core designs with 14-foot fuel such as South Texas Units 1 & 2, Doel 4, and Tihange 3.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.034

Question:

The sensitivity of the PRA results to the unavailability of the non-safety related standby systems was investigated and the results are summarized in section 50.5.4 (sensitivity case no. 36). However, two different core damage frequency (CDF) numbers are reported in page 50-14. A CDF of 2.92E-5/year is reported at the end of Section 50.5.4 while in Section 50.6 (Results) a CDF of 7.345E-6/year is reported. Please explain and revise as necessary.

Westinghouse Response:

The correct value of the core damage frequency for sensitivity case no.36 is 7.41E-06/year (see result in Table 50-20). The Sections 50.5.4 and 50.6 will be corrected accordingly.

Design Control Document (DCD) Revision:

None

PRA Revision:

The corrected value of the core damage frequency for sensitivity case no.36 will be included in the next revision of AP1000 PRA Report (see Attachment)

Table 50-20 shows the contribution of the initiating events when no credit is taken for the above standby systems.

The output file contains 7269 cutsets. The top 50 of these cutsets are shown in Table 50-23.

This sensitivity analysis estimates that the CDF increases from 2.41E-07/year to ~~2.92E-05~~7.41E-06/year when no credit is taken for the standby systems CVS, SF~~W~~S, RNS, DAS, and DGs.

These results are limited by the way the sensitivity analysis is performed. Namely, if a CDF cutset does not appear in the CMTOT.OUT file due to cutoff probability, then it is resurrected in the present analysis.

50.6 Results

Importance and sensitivity analyses are performed on the core damage model for internal initiating events at power. The results for individual cases have been discussed in their respective sections, whenever needed.

The major conclusions of the sensitivity analyses are:

- If no credit is taken for operator actions, the plant core damage frequency is 1.37E-05/year (case 29). This compares well with the risk of existing plants where credit is taken for operator actions.
- For system importances the most important systems for core damage prevention are PMS, Class 1E DC, ADS, and IRWST. None of the non-safety systems have high system importance.
- The common cause failure basic events are important individually, as well as a group for plant core damage frequency. This is expected for a plant with highly redundant safety systems.
- There are no operator actions that provide a significant risk decrease if made to be more reliable.
- When no credit is taken for standby systems CVS, SF~~W~~S, RNS, DAS, and DGs, the plant core damage frequency increased by a factor of ~~25~~31. While this is a significant increase, the plant core damage frequency is still very low (~~7.345~~7.41E-06/year).

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 720.076

Question:

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

Westinghouse Response:

There are typographical errors. The correct Reference B-3 is "AP600 Probabilistic Risk Assessment", GW-GL-022, August 1998.

Design Control Document (DCD) Revision:

The Reference B-3 will be corrected in the next revision of the AP1000 DCD Appendix 19B (see Attachment).

PRA Revision:

The Reference B-3 will be corrected in the next revision of the AP100 PRA Appendix B (see Attachment).

pressure, which is 91 psig (0.73 MPa). This is well below the 50 percent containment failure probability value of 135 psig (1.03 MPa).

The results also show that, in all cases the containment does not pressurize to Service Level "C" containment challenge indicator value prior to the time that the core debris completely penetrates the containment basemat. Thus, for these cases there is no potential challenge to containment integrity due to overpressurization since: a) there is no longer a source of mass and energy input to the containment after the core debris penetrates the entire basemat, and b) basemat penetration assures that the containment will be depressurized through the basemat failure.

Based on these analyses, it can be concluded that it is not necessary to specify a concrete type for the containment basemat since containment overpressure failure due to non-condensable gas generation from core concrete interactions is not likely for any credible severe accident scenarios.

19B.5 Conclusions

The results of the limited deterministic analyses of ex-vessel severe accident phenomena presented in this section show that early containment failure is not a certainty if the reactor vessel fails. Based on the deterministic analyses, direct containment heating that might ensue from a high pressure melt ejection would not challenge the integrity of the containment. Ex-vessel steam explosions, assessed on a very conservative basis would not produce impulse loads that would challenge the integrity of the containment due to localized failures of the reactor cavity floor and walls. In addition, these analyses indicate that the ex-vessel steam explosion loads are not strong enough to displace the reactor vessel from its location inside the biological shield. Thus, there is no challenge to any containment penetrations connected to the reactor vessel or to the reactor coolant loops. In the case of a vessel failure at a low RCS pressure, the core concrete interactions analyses indicate that the containment integrity would not be challenged in the first 24 hours of the event and thus no significant releases of fission products are predicted in that time frame.

Thus, it is concluded that prevention of large fission product releases to the environment is not dependent on the integrity of the reactor vessel. If reactor vessel failure occurs, there may be challenges to the containment integrity, but these challenges are highly uncertain and the most likely challenge (containment failure by core penetration of the cavity basemat) would not occur in the first 24 hours of the accident. Thus, the AP1000 assumption that reactor vessel failure always leads to containment failure is a conservatism in the AP1000 risk profile.

19B.6 References

19B-1 "AP600 Phenomenological Evaluation Summaries," WCAP-13388 (Proprietary) Rev. 0, June 1992 and WCAP-13389 (Nonproprietary), Rev. 1, 1994.

19B-2 Theofanous, T. G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

19B-3 "AP~~1000~~600 Probabilistic Risk Assessment," GW-GL-02~~1~~2, August 1998.

vessel failure always leads to containment failure is a conservatism in the AP1000 risk profile.

B.6 References

B-1 "AP600 Phenomenological Evaluation Summaries," WCAP-13388 (Proprietary) Rev. 0, June 1992 and WCAP-13389 (Nonproprietary), Rev. 1, 1994.

B-2 Theofanous, T. G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

B-3 "AP~~1000~~600 Probabilistic Risk Assessment," GW-GL-02~~1~~2, August 1998.

B-4 Pilch, M. M., "Adiabatic Equilibrium Models For Direct Containment Heating," SAND-91-2407C, December 1992.

B-5 "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Design," SECY-93-087, dated April 2, 1993.