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Nuclear Systems

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DCP/NRC1486
Project 711

September 21, 2001

Document Control Desk
U. S. Nuclear Regulatory Commission
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

ATTENTION: Mr. Alan Rae, NRC, MS 12E15

SUBJECT: Transmittal of Westinghouse Responses to Requests for Additional Information (RAIs) for AP1000 Pre-Certification Review

Dear Mr. Rae:

Attached please find the Westinghouse responses to the following Requests for Additional Information (RAIs) related to the pre-certification review of the AP1000:

P23	P49	P60
P29	P50	P62
P33	P51	P63
P42	P52	P65
P45	P53	P67
P48	P54	

Please contact me if you have questions related to these responses.

Very truly yours,

M. M. Corletti
Passive Plant Projects & Development

/Attachments
1) Responses to NRC Request for Additional Information

D063

Rec'd from NRC 02/27/02

2764alf.doc

bcc:	J. W. Winters	EC E3-08	1L	1A
	W. E. Cummins	EC E3	1L	1A
	C. B. Brinkman	Rockville	1L	1A

Attachment 1

"Westinghouse Responses to Requests for Additional Information (RAIs) for AP1000 Pre-Certification Review"

(Westinghouse Non-Proprietary Class 3)

AP1000 PRE-CERTIFICATION REVIEW

REQUEST FOR ADDITIONAL INFORMATION

RAI: P023

Question:

The PRHR for AP1000 will be 22% larger than for AP600, but the average heat flux is expected to be 42% greater. Justify that the expected heat flux is bounded by test conditions. Justify that the heat transfer correlations in NOTRUMP and LOFTRAN codes are adequate to model the greater heat flux.

Westinghouse Response:

The PRHR heat exchanger has been designed to remove core decay heat and sensible heat following loss of heat sink events. The limiting design basis functional requirement is to perform this function with natural circulation flow from the reactor coolant system. The PRHR heat exchanger takes suction from an RCS hot leg, and discharges to the steam generator channel head (cold leg side). Typically, the PRHR heat exchanger is actuated on low steam generator water level (either low narrow range level and low startup feedwater flow, or low wide range water level); however, the PRHR heat exchanger is also actuated on a Safeguards Actuation signal, which also actuates the CMTs, and trips the reactor coolant pumps. If PRHR is actuated while the reactor coolant pumps are operating, the flow through the PRHR heat exchanger is higher, and the calculated heat flux is higher than under natural circulation conditions. This typically results in the reactor coolant system temperature being reduced to the low Tcold signal actuating the Safeguards actuation signal, thus tripping the RCPs, and actuating the CMTs. Again, following the Safeguards actuation signal, the PRHR operates in natural circulation mode.

Westinghouse developed a heat transfer correlation to calculate the boiling heat transfer coefficients on the outside of the PRHR heat exchanger tubes. This correlation is in the form of the Rohsenow heat transfer correlation, and is described in Reference 1. This correlation was based over a large range of conditions that covered the expected natural circulation (low flow) conditions that are the most critical for the calculation of the heat transfer performance of the PRHR heat exchanger. The tests also covered very high flow, high heat flux conditions that are prototypical of a forced flow condition through the PRHR heat exchanger. As described above the system response during the majority of the transient events is dominated by the natural circulation heat transfer through the PRHR heat exchanger.

Figure 1 from Reference 1 shows the range of heat flux conditions that were covered by the AP600 PRHR heat exchanger test. The tests covered heat fluxes as high as 250,000 BTU/hr-ft^2. In LOFTRAN, the PRHR heat transfer correlation is applied, unless the calculated heat flux is greater than Critical Heat Flux (CHF). The correlation is therefore applied up to CHF. Figure 2 from Reference 2 shows the calculated heat flux for the AP600 and AP1000 PRHR heat exchangers under both natural circulation, and forced flow conditions. For all cases, the calculated heat flux, as predicted using the PRHR correlation, is much lower than

the CHF limit. Therefore, the boiling heat transfer correlation developed from the AP600 PRHR tests can be used to predict boiling heat transfer for the AP1000 PRHR heat exchanger.

References

1. WCAP-14727, AP600 Scaling and PIRT Closure Report, July 1997.

2. WCAP-15613, AP1000 PIRT and Scaling Assessment, February 2001.

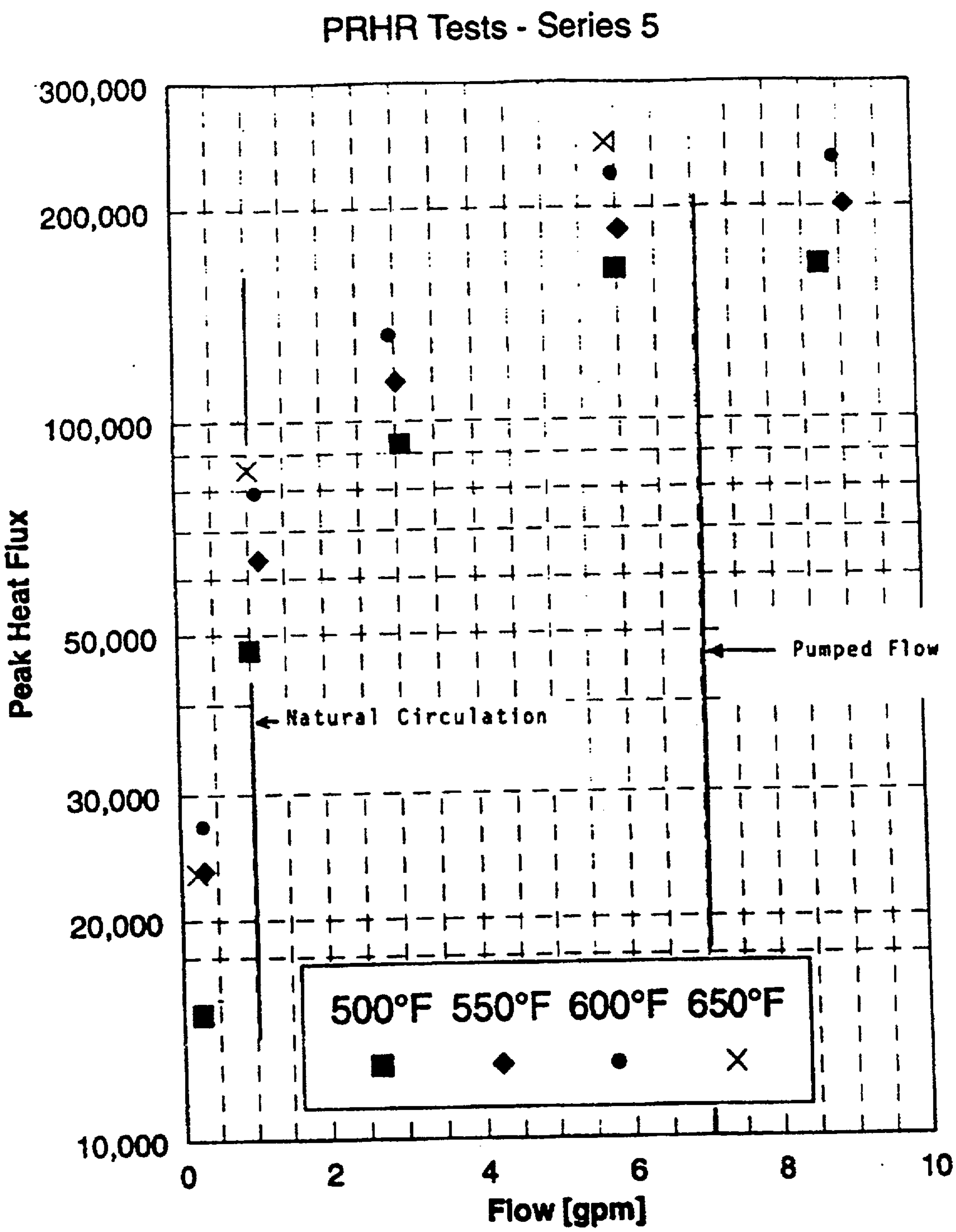


Figure 1: PRHR Component Test Results (Ref. 1)

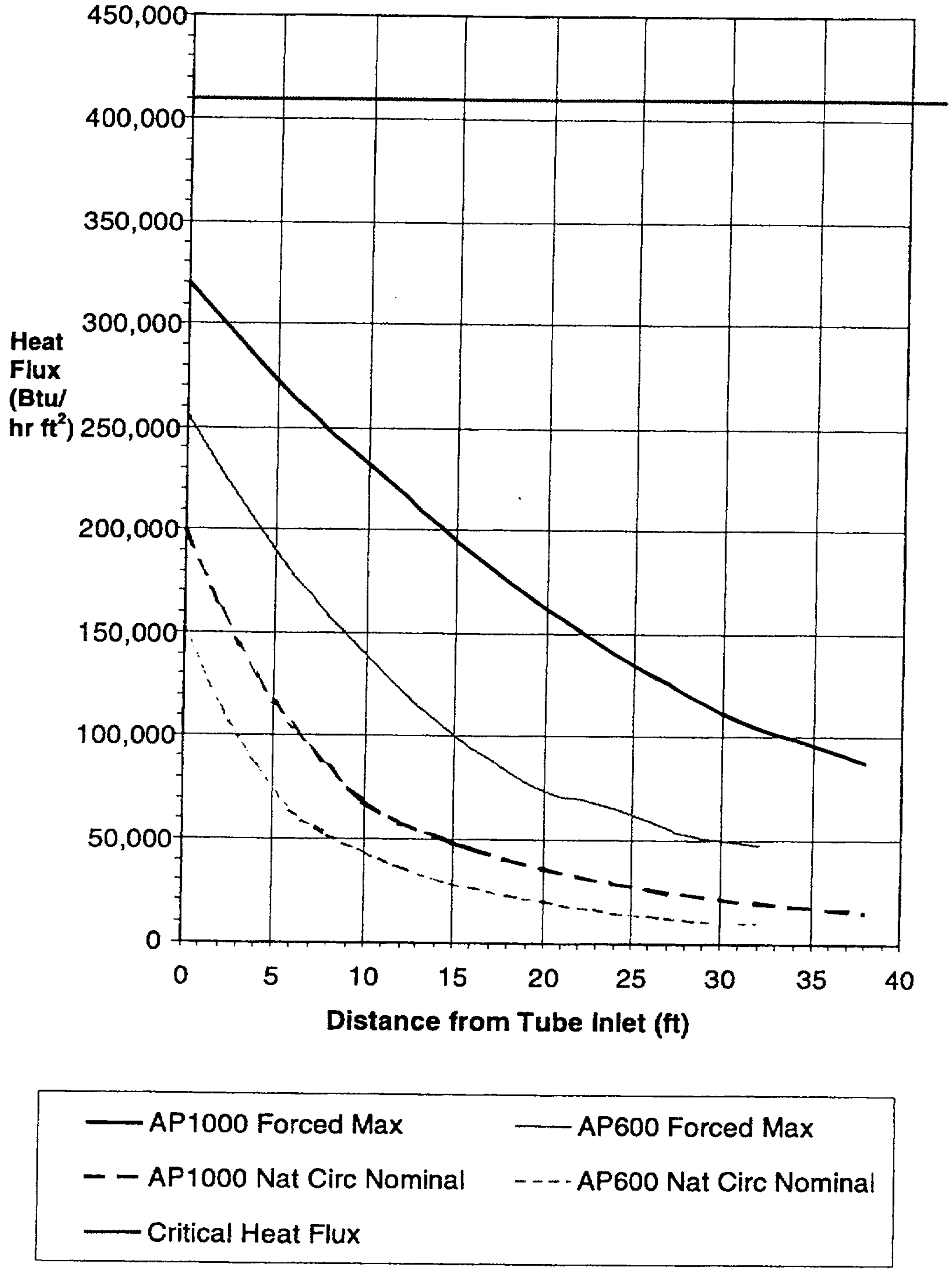


Figure 2: PRHR Calculated Heat Flux for AP600 and AP1000 Natural Circulation and Pumped Flow Conditions

REQUEST FOR ADDITIONAL INFORMATION

RAI: P029

Question:

LOFTRAN was benchmarked against SPES main steam line break test SO1512 . Justify that SPES properly scales the larger steam generators for AP1000.

Westinghouse Response:

Comparison of LOFTRAN simulations with the SPES-2 MSLB test data demonstrated that LOFTRAN accurately predicted the overall system performance of the MSLB tests at SPES. LOFTRAN provided good predictions of CMT and PRHR behavior. Reference 1 provides the summary of the verification and validation of LOFTRAN to the SPES tests.

With regards to the main steam line break accident, the system cooldown is dominated by the rate of the steam flow through the break. This is controlled by the steam line venturi located in the steam generator outlet. For both the AP600 and the AP1000, the venturi has the same effective flow area (i.e. 1.4ft^2), such that the rate of depressurization and rate of the cooldown for both plants will be similar.

LOFTRAN has been used to model steam generator heat transfer for a wide range of steam generator size and configuration for operating plants for the purposes of performing conservative, Chapter 15 accident analyses. The ability of the steam generator to transfer heat depends upon four factors: the primary fluid convective film coefficient, the SG U-tube conductive resistance, the secondary-fluid convective film coefficient, and the extent to which the SG U-tube bundle is covered with secondary fluid. These phenomenon and processes are the same for the operating plant steam generators, the AP600 (Δ75) steam generators, or the AP1000 (Δ125) steam generators. LOFTRAN has been used extensively for the purposes of performing Chapter 15 accident analyses for operating plants, and the SPES tests were not used specifically to validate the steam generator heat transfer models in LOFTRAN. In the AP600 Design Certification, testing was used for the validation of the LOFTRAN PRHR model, CMT model, and overall system performance predictions related to the operation of the passive safety systems.

Reference:

1. WCAP-14307Revision 1, "AP600 LOFTRAN-AP and LOFTR2-AP Final Verification and Validation Report.

AP1000 PRE-CERTIFICATION REVIEW

REQUEST FOR ADDITIONAL INFORMATION

RAI: P33

Question:

Describe the decay heat model that will be used in LOFTRAN for AP1000 analysis. Justify that this model is conservative for transient and accident analysis. Provide comparisons of the model with the American Nuclear Society Standard 5.1.

Westinghouse Response:

The LOFTRAN code models decay heat through a series of exponential terms as described in WCAP-7907-P-A (Reference 1).

$$\mathbf{FDH * DH (f.o.n)} = \sum_{i=1}^{n} BETADH(i) * e^{-LAMDADH(i)*T}$$

Parameter data for the coefficient and exponential terms are provided to the code through user input (LAMDAD, BETADH). A multiplier of the decay heat can also be provided as input to the code (FDH).

Exponential expansion terms are evaluated to conservatively reproduce the reference decay heat curve for safety analysis.

Typical LOFTRAN code decay heat model inputs use a five term exponential formula that conservatively reproduces decay heat releases for more than 10,000 seconds from shutdown.

The LOFTRAN version for AP1000 safety analyses, LOFTRAN-AP Version 1.9, features a ten term exponential expansion that conservatively reproduces with good accuracy decay heat releases for more than 1,000,000 seconds from shutdown.

With respect to the reference decay heat, Westinghouse developed a generic bounding decay heat curve based upon ANSI/ANS-5.1-1979 methods. The methods utilized the following assumptions:

1. Power History: Constant full power conditions to achieve a variety of burnup levels (effective region average burnup levels) in the fuel. Burnups levels that were evaluated ranged from 1000MWD/MTU to 60,000 MWD/MTU). The upper limit on burnup varied dependent upon the assumed initial enrichment of the fuel.
2. Fission Fraction: Fission fractions were based upon actual fission fraction which would occur for initial load enrichment levels varying from 1.5% to 5%.
3. Energy per Fission of Each Isotope: 200 Mev with a 1.5 uncertainty for all isotopes.
4. Neutron Capture in Fission Products by Use of a Multiplier: The methods use the equation 11 of ANSI/ANS-5.1-1979 for shutdown times less than 10,000 seconds and Table 10 of ANSI-ANS-5.1-1979 for shutdown times greater than 10,000 seconds.

5. Production Rate for 239 Isotopes: The equations for U239 and Np239 contributions are based on equation 14 and 15 of ANSI/ANS-5.1-1979. The value of "R" (atoms of U239 produced per second per fission per second evaluated for reactor composition at the time of shutdown) is determined as a function of the initial enrichment, burnup, and H/U ratio.
6. Activation Decay Heat Other than 239: This refers to the activation of the structural materials. With the use of zirconium as the major structural material in the core region as opposed to steel, this term is negligible and it is ignored in this calculation.
7. Uncertainty Parameters: A two-sigma uncertainty is applied to these calculations. The methods defined in ANSI/ANS-5.1-1979 section 3.3 are used and incorporated these uncertainties into the calculation of the decay heat levels. The uncertainties associated with the total power level of a plant are addressed in the initial power level of the plant assumed in the specific transient analysis.
8. Delayed Fission Kinetic Modeling: ANSI/ANS-5.1-1979 does not address this aspect of decay heat production. LOFTRAN provides for the production of decay heat precursors as a part of the decay heat model for both prompt and delayed neutron fission source.

The generic decay heat curves developed above bound Westinghouse standard and optimized fuel design with 15x15, 16x16, 17x17 fuel assembly configurations. The attached figures show a comparison plot of LOFTRAN calculated decay heat curve using nine exponential terms against the bounding generic ANSI/ANS-5.1-1979 decay heat model. The exponential terms may be input depending upon conservatism desired in the analyses or time duration of the event analyzed.

Reference:

1. Burnett, T. W. T., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Nonproprietary), April 1984

Figure P33-1

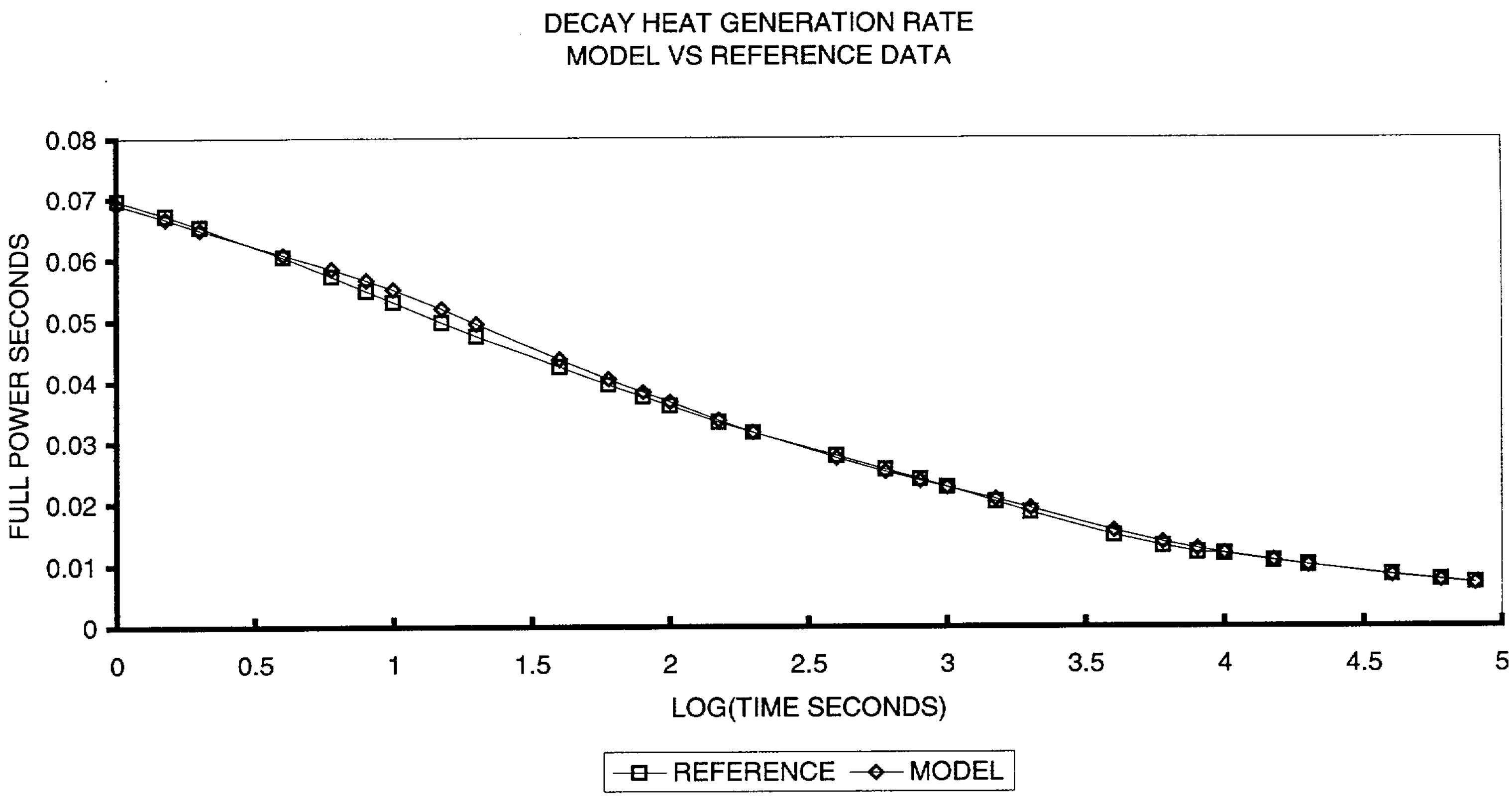
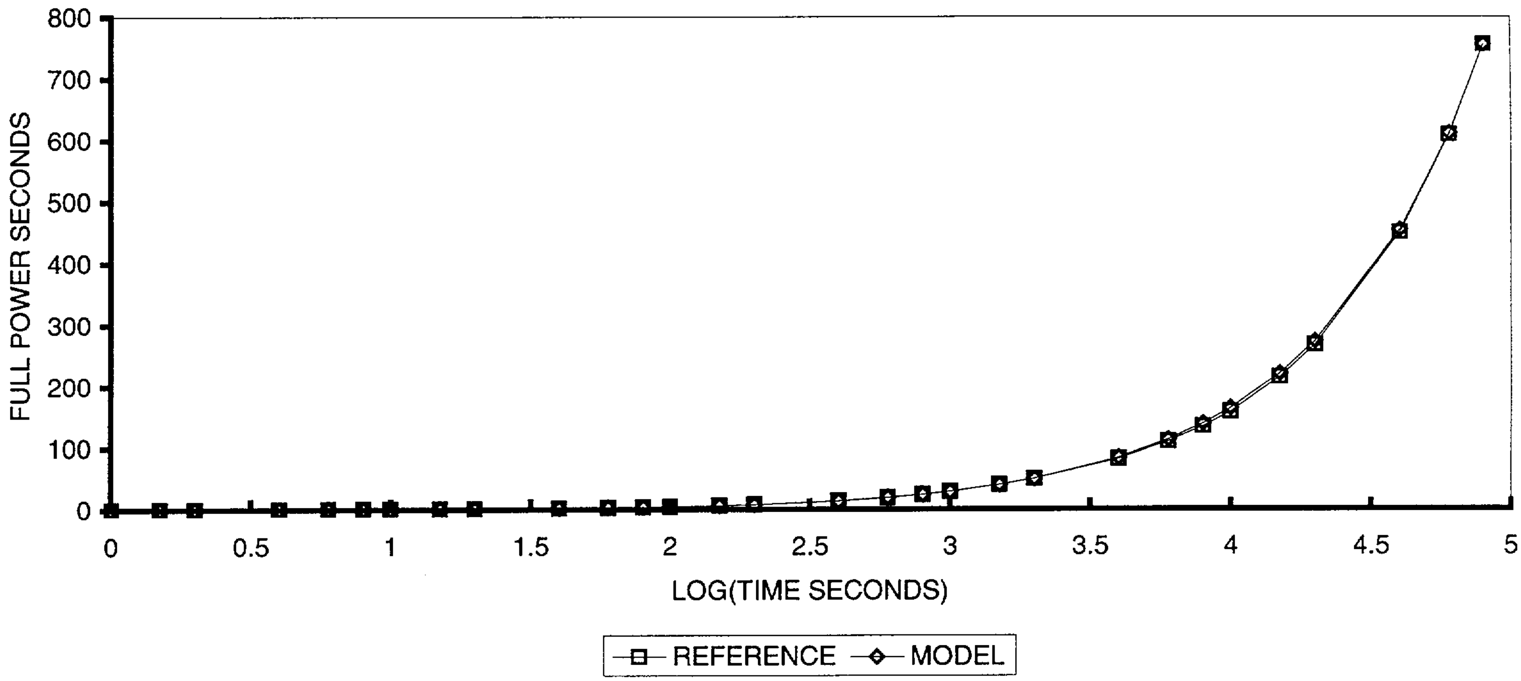

DECAY HEAT GENERATION RATE
MODEL VS REFERENCE DATA
FULL POWER SECONDS
0.08
0.07
0.06
0.05
0.04
0.03
0.02
0.01
0
0
0.5
1
1.5
2
2.5
3
3.5
4
4.5
5
LOG(TIME SECONDS)
REFERENCE
MODEL

Figure P33-2

INTEGRATED DECAY HEAT
MODEL VS REFERENCE DATA



REQUEST FOR ADDITIONAL INFORMATION

RAI: P042

Question:

Provide a comparison of user options described in response to RAI P41 for AP1000 modeling with the NOTRUMP models for SPES, OSU, and ROSA (RAI P21). Provide a comparison of system noding between the AP1000 model and the NOTRUMP input models used to describe the 3 integral system tests.

Westinghouse Response:

A comparison of the NOTRUMP noding for AP600, SPES and OSU are documented in Section 1.16 of Reference 1. Noding changes in the preliminary analyses performed to reflect the AP1000 design analyzed with NOTRUMP are as follows:

- The addition of two additional core nodes to reflect the added core length,
- The removal of two nodes in each of the DVI lines to reflect a piping simplification performed.

Should additional changes be warranted, they will be described appropriately. The user options are compiled in the form of Safeguards Engineering Standards for the AP1000 plant design which will be transmitted to the NRC under a separate transmittal letter.

References

1. WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, August 1998, R. L. Fittante, et. al.

REQUEST FOR ADDITIONAL INFORMATION

RAI: P045

Question:

In AP1000 the power density (and the decay heat) is higher than that of AP600. It was also indicated in your presentations that the actual AP1000 power may be 1080 MWe or may be even higher. Please indicate the upper limit of the decay power (and the corresponding reactor power level) which can be supported with natural circulation during the LTC phase.

Westinghouse Response:

The AP1000 thermal core full power has been set at 3400 MWt. The safety systems are designed to provide acceptable core cooling for accident events with core having this full power rating, in accordance with the applicable regulations and guidelines. The preliminary AP1000 LTC analyses indicate acceptable core cooling at this power level.

The nominal net electrical power of 1080 is dependent on the circulating water temperatures which is a site specific condition.

REQUEST FOR ADDITIONAL INFORMATION

RAI: P48

Question:

The scaling rationale presented in Section 4.1.2.2 (page 4-15) of WCAP-15613 claims that two-phase natural circulation and passive residual heat removal (PRHR) heat transfer are high ranked Phenomena Identification and Ranking Table (PIRT) phenomena for a small break loss-of-coolant accident (LOCA). Table 2.4-2, "PIRT for AP1000 Small Break Accident," however, does not list any highly ranked process for the PRHR, and only "Pool Level" and "Gravity Draining" for the In-containment Refueling Water Storage Tank (IRWST). Clarify where in the PIRT the processes natural circulation and/or PRHR heat transfer are given high rankings.

Westinghouse Response:

WCAP-15613 is inconsistent. The PIRT process ranked the two-phase natural circulation and PRHR heat transfer as low for SBLOCA. However, these phenomenon are important for transients, and therefore an assessment of the scaling parameters for Natural Circulation was provided for completeness. WCAP-15613 will be clarified at the next revision.

REQUEST FOR ADDITIONAL INFORMATION

RAI: P49

Question:

On page 3-68 of WCAP-14727, Rev. 2, reference is made to Appendix B, Section B.1 which lists calculations for single loop Π groups derived for various periods of a small break LOCA. This information, however, was not included in Appendix B. Appendix B contains only the multi-loop Π group calculations. In order to evaluate the OSU, SPES, and ROSA tests for applicability to the AP1000, please provide these calculations and/or a list of values used in the single-loop Π groups or verify that the information contained in Appendix E is that which applies.

Westinghouse Response:

Appendix E of WCAP-14727, Rev. 2 contains the SPES, OSU, and AP600 reference values that support the single loop scaling groups for various periods of a small break LOCA. Appendix B of WCAP-14727, Rev. 2 contains the calculations that support the multi-loop scaling groups.

REQUEST FOR ADDITIONAL INFORMATION

RAI: P50

Question:

Provide the Automatic Depressurization System Stage 4 (ADS-4) vapor phase flow rate for the 2-inch cold leg break to accompany Figure 3.3.1.4-31of WCAP-15612, and a figure or table providing the water level in the hot legs for this transient. Also provide the core exit vapor flow rate.

Westinghouse Response:

Please see the attached figures and table. The results provided correspond to the analysis of 2-inch cold leg break presented in WCAP-15612, "AP1000 Plant Description and Analysis Report."

Note that ADS4-2 is atop Hot Leg 1, and ADS4-1 is atop Hot Leg 2 in the legends of the plots.

Table 50-1: Sequence of Events for AP1000 2-in. Cold Leg Break	
Event	**Time (seconds)**
Break opens	0.0
Reactor Trip signal	58.3
Steam turbine stop valves close	59.3
"S" signal	64.9
Main feed isolation valves begin to close	69.9
Reactor coolant pumps start to coast down	81.1
ADS Stage 1	2719.6
Accumulator injection starts	2760
ADS Stage 2	2719.6
ADS Stage 3	2909.6
Accumulator Empties	3183
ADS Stage 4	3941.4
Core makeup tank empties	4240
IRWST injection starts	4500

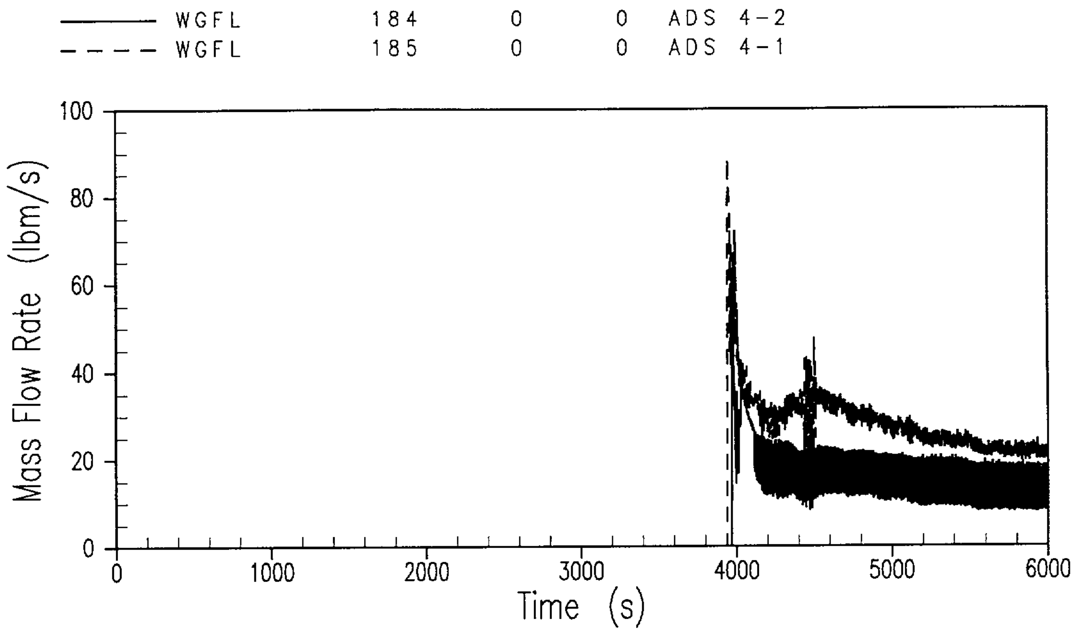


Figure P50-1: ADS-4 Steam Flow – 2inch Cold Leg Break

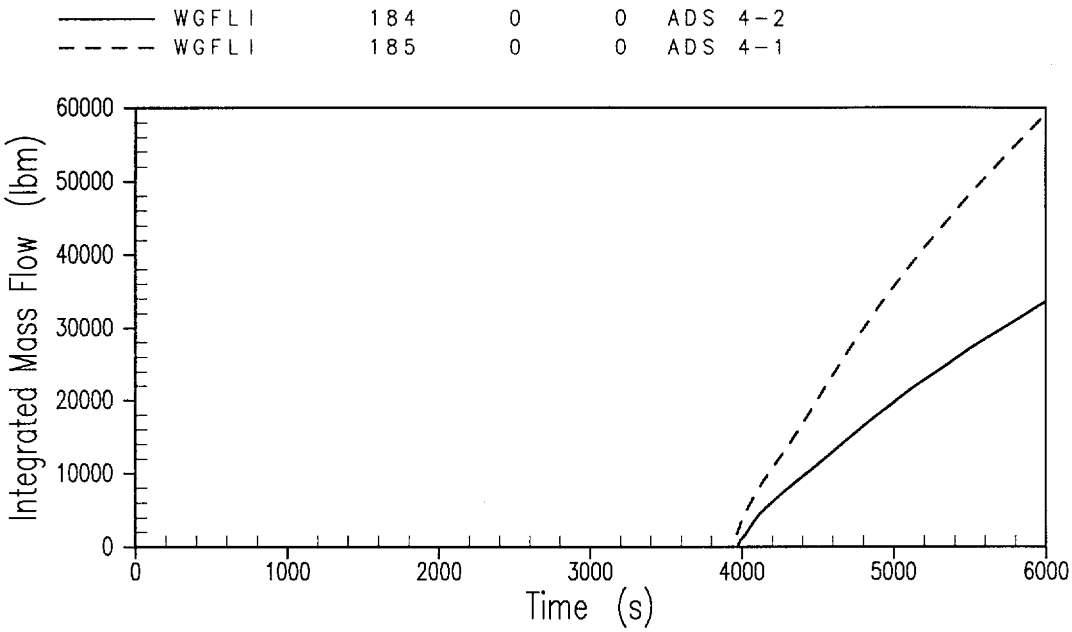


Figure P50-2: Integrated ADS-4 Steam Flow – 2-inch Cold Leg Break

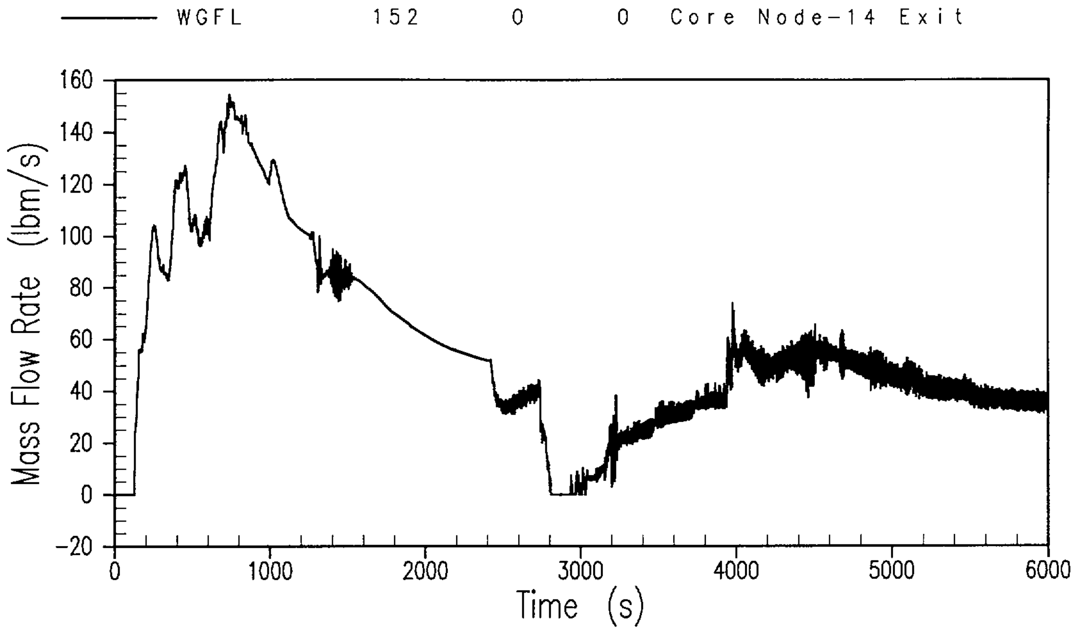


Figure P50-3: Core Exit Steam Flow – 2 inch Cold Leg Break

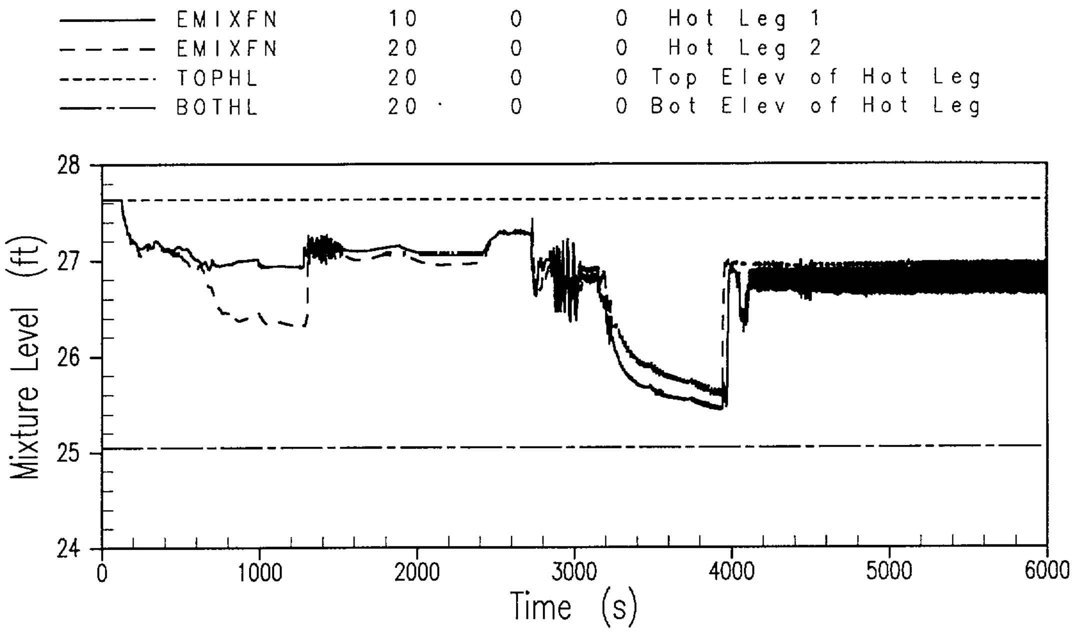


Figure P50-4: Hot Leg Level – 2inch Cold Leg Break

REQUEST FOR ADDITIONAL INFORMATION

RAI: P51

Question:

Provide the expected ADS-4 vapor phase flow rate for the DEDVI (double-ended direct vessel injection line) break, and the water level in the hot legs for this transient.

Westinghouse Response:

Please see the attached figures and table. The results provided correspond to the analysis of DEDVI presented in WCAP-15612, "AP1000 Plant Description and Analysis Report."

Note that ADS4-2 is atop Hot Leg 1, and ADS4-1 is atop Hot Leg 2 in the legends of the plots.

Table P51-1: Sequence of Events for AP1000 Double-ended DVI Break	
Event	**Time (seconds)**
Break opens	0.0
Reactor Trip signal	15.3
Steam turbine stop valves close	16.3
"S" signal	20.6
Main feed isolation valves begin to close	25.6
Reactor coolant pumps start to coast down	36.8
Accumulator injection starts	280
ADS Stage 1	250.3
ADS Stage 2	320.3
ADS Stage 3	440.3
ADS Stage 4	560.3
Accumulator Empties	681
Intact loop core makeup tank empties	2066
IRWST injection starts	2340

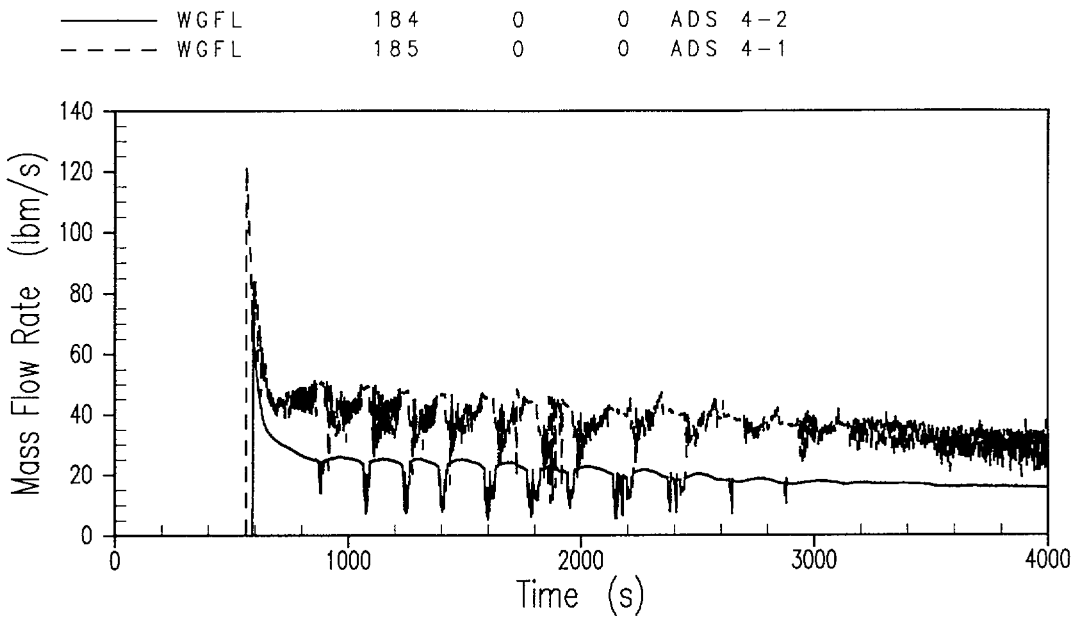


Figure P51-1: ADS-4 Steam Flow – Double-ended DVI Break

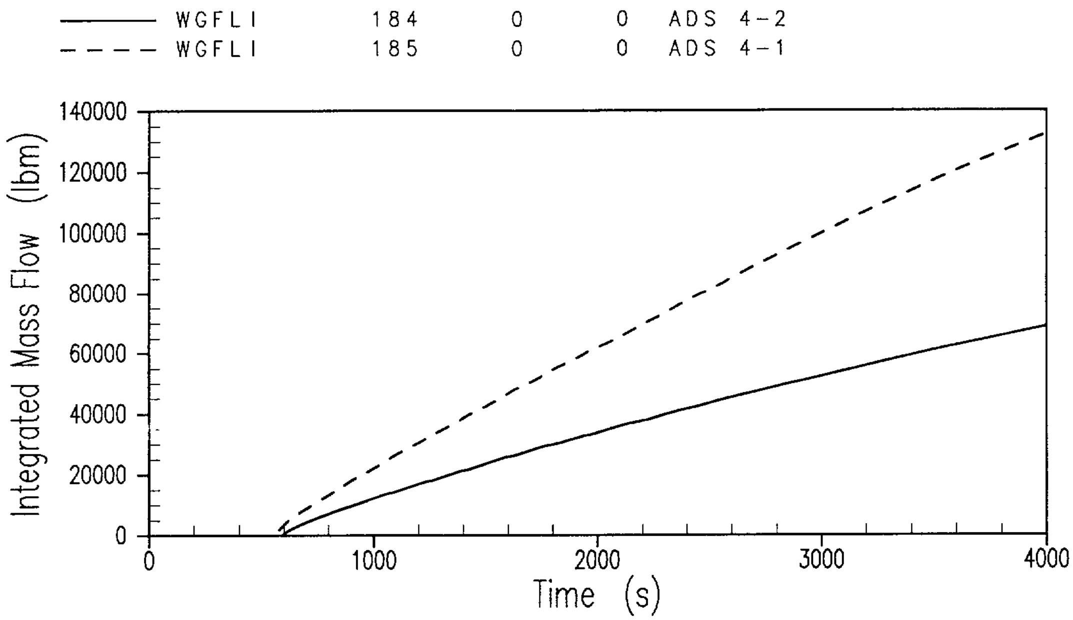


Figure P51-2: Integrated ADS-4 Steam Flow – Double-ended DVI Break

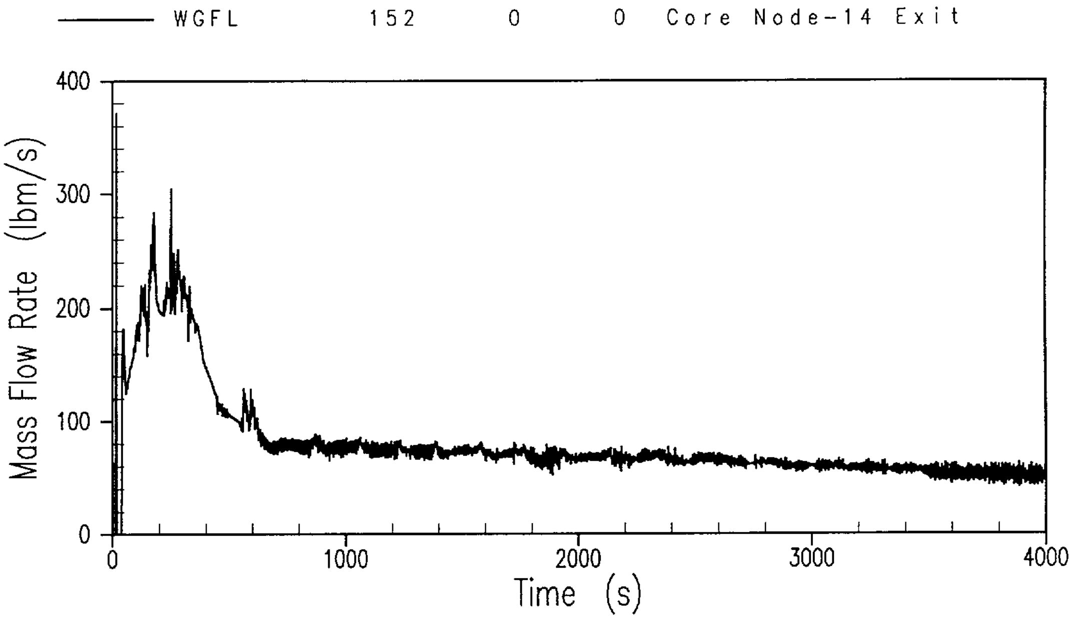


Figure P51-3: Core Exit Steam Flow – Double-ended DVI Break

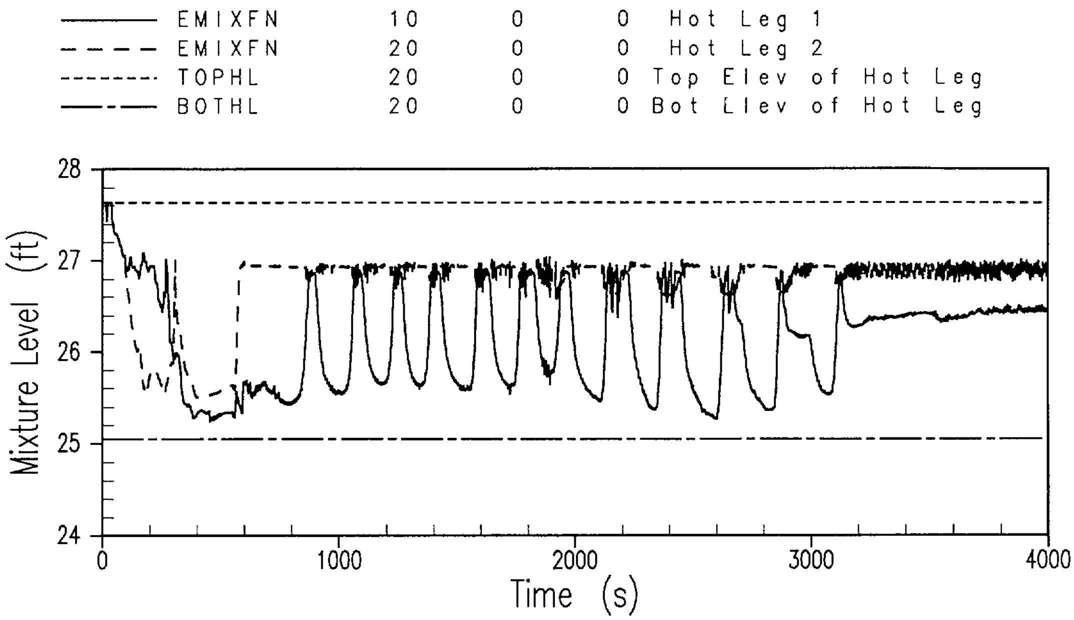


Figure P51-4: Hot Leg Level – Double-ended DVI Break

REQUEST FOR ADDITIONAL INFORMATION

RAI: P052

Question:

Higher vapor generation rates in the core may result in a lower inner vessel mixture level for some transients due to entrainment in the upper plenum. Please provide the axial flow area in the AP1000 upper plenum at an elevation just below the bottom of the hot legs. In addition, specify the net free volume between the top of the heated core and the bottom of the hot legs.

Westinghouse Response:

The requested safety analysis input for the AP1000 reactor vessel upper plenum design is as follows. The net free volume between the top of the heated core and the bottom of the hot legs is approximately 370 cu. ft. The axial flow area of the AP1000 upper plenum at an elevation just below the bottom of the hot legs is approximately 70 sq. ft. The values provided are exclusive of the flow area and free volume inside the control rod guide tubes.

REQUEST FOR ADDITIONAL INFORMATION

RAI: P53

Question:

In NUREG/CR-5541 it is reported that there were two important phenomena that were either distorted by, or not present in the three major integral effects test facilities (APEX, SPES, and ROSA). These were "flow inertia," or the ratio of inertia over pump forces during the initial depressurization, and "effect of reactor pressure vessel injection from the pressurizer" during the ADS-4 depressurization phase. Since:

(i) The flow inertia distortion was not considered important to reactor vessel inventory for the AP600. Verify that the flow inertia distortion continue to have no effect for the AP1000, taking into account the differences between the AP1000 and AP600 pump parameters.

(ii) The second distortion, "effect of reactor pressure vessel injection from the pressurizer," was present in all three integral effects test facilities and is due to distortion in the pressurizer surge line flow. The distortion in APEX was considered non-conservative, because of disproportionally low ADS-4 flow. The scaling parameter for "Effect of pressurizer injection" is:

$$\prod v_i w = \frac{(W_l)_0}{(W_{ADS4})_0}$$

Which represents the ratio of the vessel liquid inflow and outflow from the CMT and pressurizer and the flow out the ADS-4. Provide flows to determine $(W_l)_0$ and $(W_{ADS4})_0$ for the DEDVI line break

Westinghouse Response:

(i) As shown in WCAP-15613, "AP600 PIRT and Scaling Assessment," reactor coolant pump coastdown (i.e. flow inertia) is a low-ranked phenomena for the AP600 and AP1000 small break LOCA events. There are several reasons for this including:

- The passive plants trip the reactor coolant pumps upon actuation of the passive safeguards systems, and therefore pump coastdown occurs during the initial LOCA blowdown;
- The pump inertia does not affect the magnitude of the plant transient, only the RCS internal flow rate at the time at which the blowdown occurs. Essentially, following blowdown, the primary system conditions are independent of the flow inertia of the reactor coolant pumps;
- The blowdown phase of the small break LOCA transient is essentially the same for the passive plants as for current plants, and is not a factor related to the passive safety system performance.

Therefore, the flow inertia distortion is not considered important for either AP600 or AP1000.

(ii) The requested flow rates are quantities are shown in the attached figures.

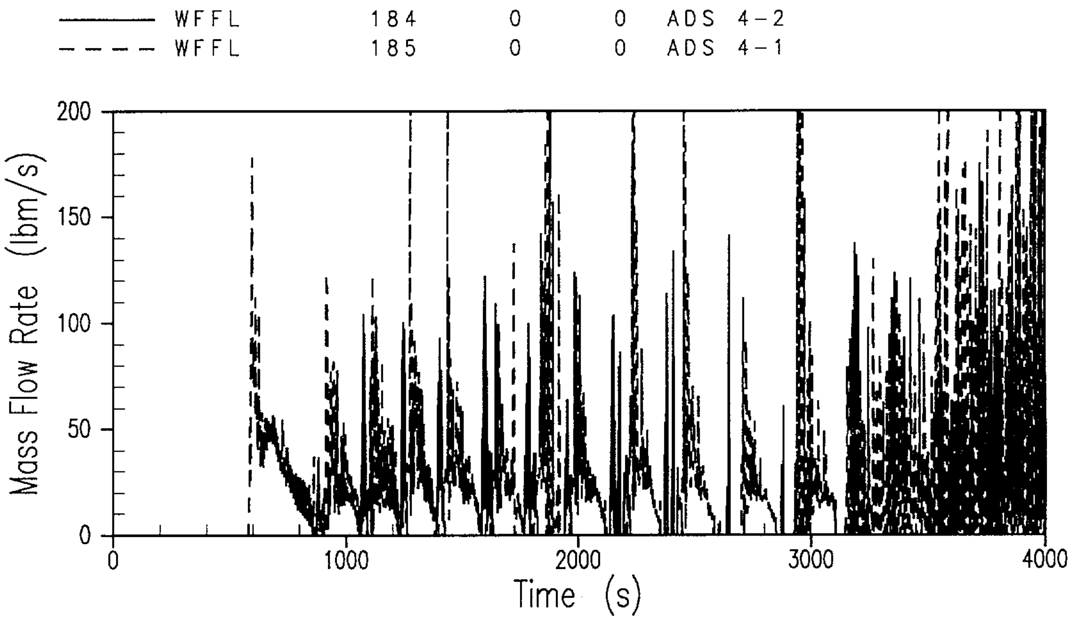


Figure P53-1: ADS-4 Liquid Flow – DEDVI Break

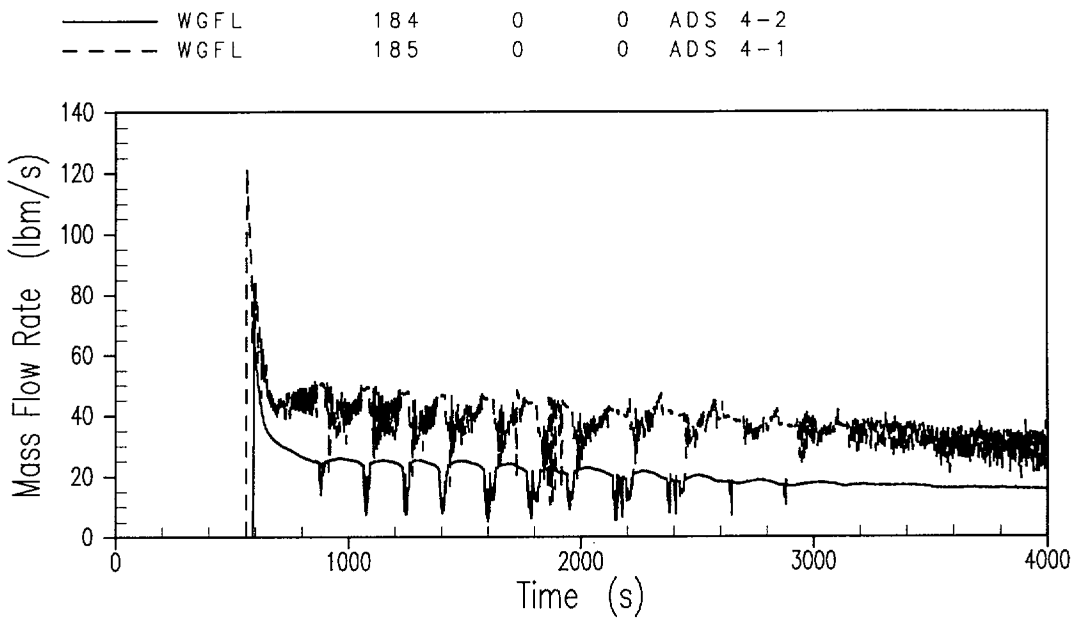


Figure P53-2: ADS-4 Vapor Flow – DEDVI Break

WFFL (50) is the broken loop CMT flow rate
WFFL (60) is the intact loop CMT flow rate

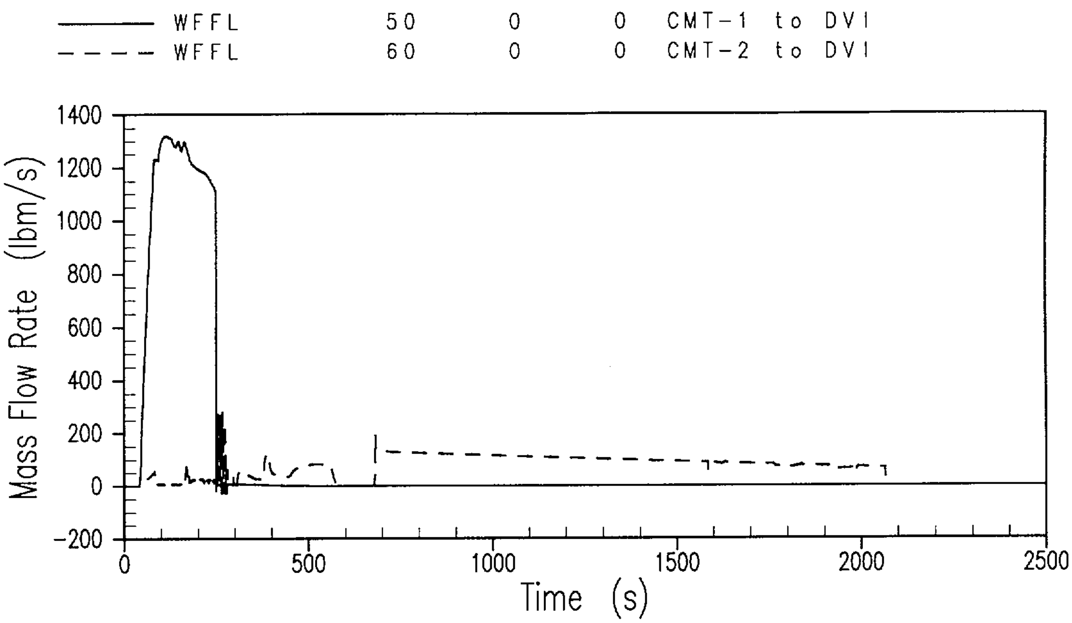


Figure P53-3: CMT Liquid Flow – DEDVI Break

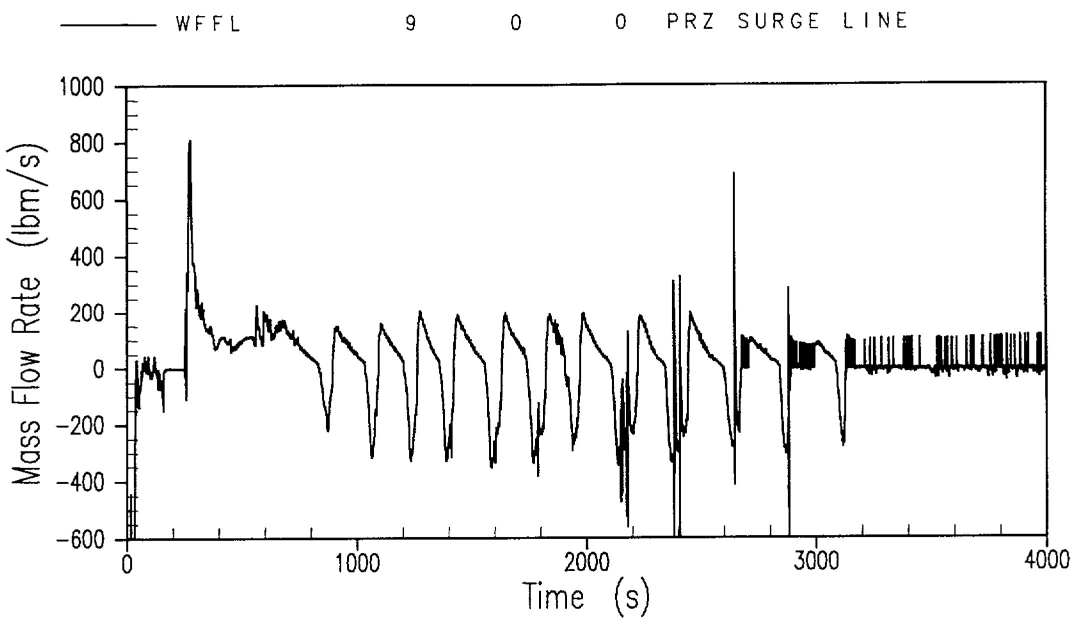


Figure P53-4: Pressurizer Surge Line Liquid Flow – DEDVI Break

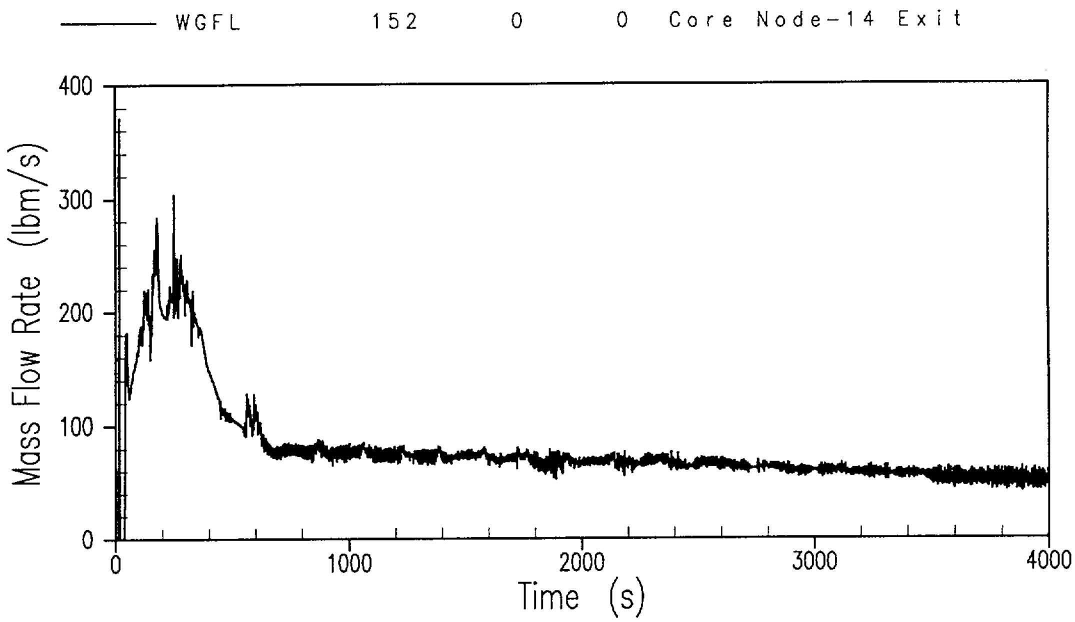


Figure P53-5: Core Exit Vapor Flow – DEDVI Break

REQUEST FOR ADDITIONAL INFORMATION

RAI: P54

Question:

The AP1000 PRHR Heat Exchanger is a C-shaped heat exchanger that transfers heat from the primary to the IRWST. Tests at Oregon State University (OSU) in the APEX facility found that the majority of heat transfer occurs in the upper part of the "C", where the tubes are primarily horizontal. In comparison to the AP600 PRHR heat exchanger, the horizontal section are longer in the AP1000. The PRHR tests, however, considered only the performance for vertical tubes. To assess the applicability of the AP1000 PRHR, please provide design information on the PRHR that includes:

(i) the lateral and transverse pitch to diameter ratios for the tube bank,

(ii) the heated lengths of the shortest and longest tubes in the horizontal span.

Westinghouse Response:

Design information for the PRHR heat exchanger is provided in section 11 of the AP1000 Plant Parameters. Revision 0 of the parameters was transmitted in Westinghouse letter DCP/NRC1484.

(i) The lateral and transverse pitch for the PRHR tubes is the same as that of the AP600. Specifically, the tube to tube pitch in the horizontal portion is 1.5 inches, tube centerline to centerline square pitch. The distance between each vertical row of tubes is 2.25 inches, tube centerline to centerline.

(ii) The heated lengths of the shortest and longest tubes in each horizontal span (top and bottom) is 74.11 inches and 158.11 inches respectively.

AP1000 PRE-CERTIFICATION REVIEW

REQUEST FOR ADDITIONAL INFORMATION

RAI: P60

Question:

Provide the design information that in the following Table, (noting that some of this information is included in information previously supplied by Westinghouse.)

Parameter	Unit	AP1000
Primary RCS volume	ft^3	*10,621 ft^3 (cold with water solid pressurizer; see APP-GW-G0-002 Sec. 9.1)*
*Pressurizer volume	ft^3	*2100 ft^3 (APP-GW-G0-002 Sec. 1.0)*
*Pressurizer length	ft	*50.6 ft*
*Pressurizer area	ft^2	*44 ft^2 cross section (90 inch diameter) (APP-GW-G0-002 Sec. 1)*
Pressurizer initial water level	%	*Minimum 21.1%* *Maximum 44%*
*Pressurizer heater power	kW	*1600 KW (APP-GW-G0-002 Sec. 1.0)*
*Pressurizer surge line volume	ft^3	*99.7 ft^3 (APP-GW-G0-002 Sec. 1.0)*
PRHR to core thermal center difference (middle of PRHR HX to mid-elevation of core)	ft	*30.2 ft (APP-GW-G0-002 Sec. 2.0 and Sec. 11.7*
PRHR hydraulic resistance	ft^{-4}	*See APP-GW-G0-002 Sec. 11.1 for resistance breakdown.*
*PRHR inlet temperature	°F	*567°F (APP-GW-G0-002 Sec. 11.1)*
*PRHR Outlet temperature	°F	*198.8°F (APP-GW-G0-002 Sec. 11.1)*

Mass of liquid in and above hot legs	lbm	*335,199 lbm (calculated from APP-GW-G0-002 Sec. 1.0, 9.1, and 11.7)*
*Accumulator water volume	ft^3	*1700 ft^3 (APP-GW-G0-002 Sec. 11.3)*
*CMT tank volume	ft^3	*2487 ft^3 (APP-GW-G0-002 Sec. 11.2)*
CMT tank height	ft	*20.51 ft (APP-GW-G0-002 Sec. 11.2)*
CMT tank ID	ft^2	*14.17 ft (APP-GW-G0-002 Sec. 11.2)*
CMT exit form loss (K/A^2)	ft^{-4}	*See APP-GW-G0-002 Sec. 11.2 for resistance breakdown.*
Lower plenum volume	ft^3	*381.6 ft^3 (APP-GW-G0-002 Sec. 9.1)*
RPV volume	ft^3	*3472 ft^3 (APP-GW-G0-002 Sec. 9.1)*
Elevation difference between the bottom of the CMT and the bottom of the core	ft	*20.5 ft (calculated from APP-GW-G0-002 Sec. 2.0, 11.2, and 11.7)*
Nominal sum of ADS-1+2+3 flow areas	ft^2	*See APP-GW-G0-002 Sec. 11.5 for various cases.*
Nominal sum of ADS-4 flow area	ft^2	*See APP-GW-G0-002 Sec. 11.5 for various cases.*
DVI line form loss (K/A^2)	ft^{-4}	*See APP-GW-G0-002 Sec. 11 for resistance breakdown.*
Elevation difference between DVI line and bottom of the core	ft	*17.9 ft (calculated from APP-GW-G0-002 Sec. 2.0 and 11.7)*

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Elevation difference between bottom of IRWST and bottom of core	ft	*21.3 ft (calculated from APP-GW-G0-002 Sec. 11.7)*
Total DVI path resistance	ft^{-4}	*See APP-GW-G0-002 Sec. 11 for resistance breakdown.*
Total ADS-4 path resistance	ft^{-4}	*See APP-GW-G0-002 Sec. 11 for resistance breakdown.*
Inertial length (L/A) for DVI line	ft^{-1}	*See APP-GW-G0-002 Sec. 11 for piping information.*
Inertial length (L/A) for ADS-4	ft^{-1}	*See APP-GW-G0-002 Sec. 11 for piping information.*
Maximum sump level (determined by curb height)	ft	*107.6 ft (8 ft above DVI nozzle centerline). See notes below.*

Note parameters indicated * already known to NRC.

Westinghouse Response:

Please see our responses in italicized text in the table above. As noted, most of this information is included in the Revision 0 of the AP1000 Plant Parameters (APP-GW-G0-002), which were submitted in Westinghouse letter DCP/NRC1484 dated 9/12/2001.

For a number of the parameters requested above (e.g.; ADS flow areas, which are affected by assumed failures) different answers could be given depending upon how the information is to be used. In those cases, a reference to the appropriate section of the Plant Parameters is made. The Plant Parameters provide detailed descriptions of the various assumptions that might be made so that NRC can select the most applicable values for their use.

Note on maximum sump level: This is the minimum safety basis number, assuming that a DVI line break floods PXS Valve Room B (the larger volume of the two PXS valve rooms). This number is based on passive system operation only; if the RNS operates the level will be higher. This number is appropriate for evaluating core cooling in the long term recirculation mode.

REQUEST FOR ADDITIONAL INFORMATION

RAI: P62

Question:

Section 4.1.2.1, "Blowdown Phase Scaling," of WCAP-15613, states that the blowdown phase will not be scaled because the blowdown phase behavior of the AP600 and AP1000 and the sensitivity of the plant behavior to core decay heat is similar to conventional PWR plants and the passive safety systems have virtually no influence on the blowdown phase. Given this:

(i) Without a scaling evaluation of the blowdown phase, how is it assured that the APEX, SPES, and ROSA test facilities depressurize as the AP600 and AP1000 plants?

(ii)What would be the consequence of the differences between the tests and the prototype in the subsequent phases in a SBLOCA?

Westinghouse Response:

(i) The reason a scaling evaluation was not performed for the blowdown phase of SBLOCA is that code validation for the phenomena in this phase already exists from application to conventional plants. As seen in the AP600 test facilities, the blowdown phase does not involve passive safety system components and regardless of the initial conditions, it typically ends with the RCS pressure and temperature approaching secondary side conditions. As this is essentially the same behavior as in a conventional plant, the AP600 integral effects tests are not needed for code validation for the blowdown phase of a SBLOCA.

(ii) The scaling evaluation of the subsequent phases of SBLOCA addresses the differences between the test facilities and the prototypes. Given that the RCS approaches secondary side conditions at the end of blowdown, the main difference between the AP600 tests and the AP600 or AP1000, if any, would involve the duration or timing of the blowdown phase. The duration of the blowdown phase would impact the decay heat level of the subsequent phases of a SBLOCA. However, the effect of decay heat level can be readily assessed or accounted for in the safety analysis codes via ranging decay heat. The thermodynamic conditions of the primary side are established by the decay heat and by the secondary side conditions.

REQUEST FOR ADDITIONAL INFORMATION

RAI: P63

Question:

Section 4.1.2.2 of WCAP-15613 states that quality is a fundamental parameter of importance to be scaled during the two-phase PRHR natural circulation phase as RCS pressure is nearly constant. Also the scaling analysis in WCAP-14727 for AP600 assumed constant pressure during the natural circulation phase. However, as shown in see Figure 4.1-2 of WCAP-15613, during the PRHR natural circulation phase the RCS pressure decreases from almost 1000 psia to 650 psia before the ADS actuation for a 2-in cold leg break. Then:

(i) What is the basis for the assumption of constant RCS pressure in the scaling assessment?

(ii) What is the effect of this assumption on the scaling assessment result?

Westinghouse Response:

The basis for the assumption of constant RCS pressure during the natural circulation phase is in part supported by Figure 4.1-2 of WCAP-15613. Referring to Figure 4.1-2, which represents a typical primary side pressure transient for a SBLOCA, it can be seen that the pressure is close to secondary side pressure (~1000 psia) and is constant during the initial 500 second period of the natural circulation phase. RCS pressure does decrease to about 650 psia after about 500 seconds when CMT recirculation ends and the transition to CMT drain occurs, however, this change in pressure is on the order of 10 percent of the total pressure change during the SBLOCA transient. An assessment of the influence of lower RCS pressure (650 psia) indicates that it does not alter the conclusions regarding the scaling of the test facilities relative to the AP1000. The results show that the key scaling ratio of core exit quality-density is in a range from 0.7 to 1.5 over the range of RCS pressure during this phase (1000 to 650 psia). This is well within the scaling criterion of a factor of two and therefore does not change the conclusions that SPES is sufficiently scaled to AP1000 for the PRHR two-phase natural circulation phase.

It should be further noted that it is common practice to obtain scaling reference values from initial or boundary conditions, especially those that are well known or established. Steam generator secondary side pressure is an example of a well known reference condition. The scaling acceptance criteria allows for a range of results that are not typically very sensitive to modest changes or differences in initial conditions or boundary conditions.

REQUEST FOR ADDITIONAL INFORMATION

RAI: P65

Question:

There appears to be typographic errors in Equations 4-113, 4-114, and 4-115 of WCAP-15613 as they are inconsistent with Equations 4-111 and 4-112.

(i) Either confirm that these equations are correct as published, or make corrections if necessary.

(ii) What are the procedures used to assure the quality of the report?

Westinghouse Response:

Equations 4-111, 4-114, and 4-115 of WCAP-15613 contain typographical errors. The corrections shown in the attached will render equations 4-113, 4-114, and 4-115 consistent with equations 4-111 and 4-112. The equations will be corrected when WCAP-15613 is revised.

Westinghouse has prepared internal operating procedures covering document release in accordance with 10CFR Part 50 Appendix B. These procedures require appropriate technical review followed by responsible manager review and signoff.

Multiplying both sides of the above equation by $\left[x_e h_{fg} + \Delta h_{sub}\right]^2 \left[1 + x_{exit}\left(\dfrac{\Delta\rho}{\rho_g}\right)\right]$ and collecting like terms in x_e we obtain:

$$
x_e^3\left[\rho_f \cdot g\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right)h_{fg}^2\right] + x_e^2\left[h_{fg}\Delta h_{sub}\left\{\rho_f g\left[2\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right) - \left(Z_{\substack{core\\outlet}} - Z_{\substack{core\\inlet}}\right)\right]\right\}\right]
$$

$$
+ x_e\left[\Delta h_{sub}^2\left\{\rho_f \cdot g\left[\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right) - \left(Z_{\substack{core\\outlet}} - Z_{\substack{core\\inlet}}\right)\right]\right\} - \frac{q_{core}^2}{2\rho_f}\left\{\left(\sum\frac{R}{A^2}\right)_{DVI} + \Phi_{fo}^2\left(\sum\frac{R}{A^2}\right)_{ADS}\right\}\right]
$$

$$
-\frac{q_{core}^2}{2\rho_f}\left(\frac{\rho_g}{\Delta\rho}\right)\left[\left(\sum\frac{R}{A^2}\right)_{DVI} + \Phi_{fo}^2\left(\sum\frac{R}{A^2}\right)_{ADS}\right] = 0 \qquad \text{(4-110)}
$$

Dividing by $h_{fg}^2\, g\rho_f\left[Z_{IRWST} - Z_{\substack{core\\inlet}}\right]$ and performing some algebra, the following polynomial form of the two-phase natural circulation equation is obtained:

$$
x_e^3 + x_e^2\,\frac{\left[\Delta h_{sub}\left\{\cancel{g}\left[2\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right) - \left(Z_{\substack{core\\outlet}} - Z_{\substack{core\\inlet}}\right)\right]\right\}\right]}{h_{fg}\left[Z_{IRWST} - Z_{\substack{core\\inlet}}\right]}
$$

$$
+ x_e\left[\left(\frac{\Delta h_{sub}}{h_{fg}}\right)^2\frac{\left[\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right) - \left(Z_{\substack{core\\outlet}} - Z_{\substack{core\\inlet}}\right)\right]}{\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right)} - \frac{1}{2}\left(\frac{q_{core}}{\rho_f h_{fg}}\right)^2\frac{\left[\left(\sum\frac{R}{A^2}\right)_{DVI} + \Phi_{fo}^2\left(\sum\frac{R}{A^2}\right)_{ADS}\right]}{g\left[Z_{IRWST} - Z_{\substack{core\\inlet}}\right]}\right]
$$

$$
-\frac{\dfrac{q_{core}^2}{2\rho_f}\left(\dfrac{\rho_g}{\Delta\rho}\right)\left[\left(\sum\frac{R}{A^2}\right)_{DVI} + \Phi_{fo}^2\left(\sum\frac{R}{A^2}\right)_{ADS}\right]}{h_{fg}^2\,\rho_f \cdot g\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right)} = 0 \qquad \text{(4-111)}
$$

The third order polynomial in x_{exit} can be recast as:

$$
x_{exit}^3 + \phi_a x_{exit}^2 + \phi_b x_{exit} - \phi_c = 0 \qquad \text{(4-112)}
$$

where the coefficients are defined as:

$$\phi_a = \frac{\Delta h_{sub}}{h_{fg}} \frac{\left[2\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right) - \left(Z_{\substack{core\\outlet}} - Z_{\substack{core\\inlet}}\right)\right]}{\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right)} \tag{4-113}$$

$$\phi_b = \frac{\left(\frac{\Delta h_{sub}}{h_{fg}}\right)^2 \frac{\left[\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right) - \left(Z_{\substack{core\\outlet}} - Z_{\substack{core\\inlet}}\right)\right]}{\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right)} - \underline{\frac{1}{2}\left(\frac{q_{core}}{\rho_f h_{fg}}\right)^2 \left[\left(\sum \frac{R}{A^2}\right)_{DVI} + \Phi_{fo}^2 \left(\sum \frac{R}{A^2}\right)_{ADS}\right]}}{g\cancel{\rho_f}\left[Z_{IRWST} - Z_{\substack{core\\inlet}}\right]} \tag{4-114}$$

$$\phi_c = \frac{\frac{1}{2}\left(\frac{q_{core}}{\rho_f h_{fg}}\right)^2 \left(\frac{\rho_g}{\Delta\rho}\right) \left[\left(\sum \frac{R}{A^2}\right)_{DVI} + \Phi_{fo}^2 \left(\sum \frac{R}{A^2}\right)_{ADS}\right]}{g\cancel{\rho_f}\left[Z_{IRWST} - Z_{\substack{core\\inlet}}\right]} \tag{4-115}$$

The cubic root solution to the third order polynomial equation above does not result in a form that is easily used for scaling. Applying catastrophe theory similar to Reyes (Reference 1) in the OSU scaling report can circumvent this problem. Catastrophe theory holds that a scaling factor β exists such that the following relation can be made:

$$[x_{exit}]_R = \frac{1}{\beta} = [\phi_b]_R^{1/2} \tag{4-116}$$

$$[x_e]_R = \left[\left(\frac{\Delta h_{sub}}{h_{fg}}\right)^2 \frac{\left[\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right) - \left(Z_{\substack{core\\outlet}} - Z_{\substack{core\\inlet}}\right)\right]}{\left(Z_{IRWST} - Z_{\substack{core\\inlet}}\right)} - \frac{1}{2}\left(\frac{q_{core}}{\rho_f \cdot h_{fg}}\right)^2 \frac{\left[\left(\sum \frac{R}{A^2}\right)_{DVI} + \Phi_{fo}^2 \left(\sum \frac{R}{A^2}\right)_{ADS}\right]}{g \cdot \left[Z_{IRWST} - Z_{\substack{core\\inlet}}\right]}\right]_R^{1/2} \tag{4-117}$$

REQUEST FOR ADDITIONAL INFORMATION

RAI: P67

Question:

Section 1.0 of WCAP-15613 states that in the AP600, where scaling analyses of the tests identified that certain phenomena were not well scaled for the AP600 plant, conservatisms were applied to the analysis codes such that their predictions of the plant response were conservative with respect to safety, and proposes that such an approach also be used for the AP1000.

(i) Describe the phenomena which were not well scaled for AP1000.

(ii) Describe how the conservatisms are determined for these phenomena and applied to the analysis codes for the AP1000 analyses.

Westinghouse Response:

(i) The scaling studies performed for AP600 and AP1000 demonstrate that the phenomena associated with the passive core cooling system performance is well-scaled in at least one test facility. The blowdown transient performance of passive containment cooling is not considered to be well-scaled for modeling the transient condensation and convective energy transfer for either the AP600 or the AP1000.

The containment tests are sufficient to determine bounding steady-state heat transfer coefficients associated with passive containment cooling, and are sufficient to perform a bounding calculation of the transient phenomena associated with establishing water coverage on the outside of the containment shell.

(ii) A description of how each of the phenomena identified in the containment PIRT are implemented in the containment DBA evaluation model is provided in WCAP-14812, Rev. 1, "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System". The WGOTHIC containment DBA evaluation model is described in WCAP-14407 Rev.3, "WGOTHIC Application to AP600". Since there are no changes to the AP600 containment PIRT for AP1000, this same bounding approach discussed in WCAP-14812 is applicable to AP1000, and therefore will be applied to the AP1000 containment DBA evaluation model.