DCP/NRC1535
Docket No. 52-006

November 26, 2002

Attachment 3

"AP1000 Design Certification Review –
Response to Request for Additional Information"

RAI Number: 210.011

Question:

Reference, Volume 6, Section 3.9.2.5, Dynamic System Analysis of the Reactor Internals Under Faulted Conditions, Pgs. 3.9-38 and -39:

The application of leak-before-break criteria is discussed as a means of defining those pipe breaks which are used in the faulted condition analysis of the reactor internals, but the specific breaks postulated for the design basis analysis are not specified.

Please identify those pipe breaks which, after application of leak-before-break criteria, do not qualify for elimination of postulation and post-rupture dynamic analysis. Also, further identify those postulated pipe breaks analyzed to determine the maximum dynamic response of the AP1000 RPV internals.

Westinghouse Response:

As was the case for AP600, for AP1000 nominal pipe sizes of 6" and larger are qualified for elimination of post-rupture dynamic analysis through application of leak-before-break criteria. Therefore, the largest break analyzed to determine the dynamic response of the AP1000 reactor vessel internals is that of a 4" pipe connected to the reactor coolant system components or loop piping. These lines are the pressurizer spray lines and the first stage automatic depressurization system line.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number 210.012

Question:

Reference, Volume 6, Section 3.9.2.5, Pg. 3.9-41, last sentence of Section 3.9.2.5.2:

This section describes the analytical methods used to calculate stresses and deflections in the RPV internals due to the combined loads from postulated pipe rupture and the safe shutdown earthquake. The last sentence in this section states the final conclusion that the reactor internals components are within acceptable stress and deflection limits. This significant conclusion is stated without providing, or referencing, any supporting stress and deflection data from the actual analyses (which presumably have been done in order to reach this conclusion).

Please provide a results summary of analytical data, including comparison to appropriate allowable values, which demonstrates that stress, deflection, and stability criteria for the RPV internals design have been met when subjected to the combined effects of the limiting postulated pipe break, and the safe shutdown earthquake.

Westinghouse Response:

The design of the core supports and reactor vessel internals in the AP1000 is based on the AP600 and other previously licensed plants and is structurally very similar to those designs.

The design analysis for Service Level D load combinations for the AP600 core supports and reactor internals used a combination of 1 square foot hot leg break and the safe-shutdown seismic event as an enveloping Service Level D load combination.

As was the case for AP600, for AP1000 nominal pipe sizes of 6" and larger are qualified for elimination of post-rupture dynamic analysis through application of leak-before-break criteria. Therefore, the limiting design basis break analyzed to determine the dynamic response of the AP1000 reactor vessel internals is that of a 4" pipe (pressurizer spray line, first stage ADS line).

Loads at key locations in the AP1000 reactor vessel internals resulting from the Service Level D load combinations have been evaluated from preliminary seismic analyses and utilization of enveloping LOCA loads. The enveloping LOCA loads used in the AP1000 evaluation are the loads from the 1 square foot break developed for the AP600. A comparison of the loads on the AP600 reactor vessel internals resulting from the 1 square foot LOCA with those from a 4" pipe break show that the loads from the 1 square foot LOCA are about an order of magnitude higher. Since the AP1000 design basis LOCA is a 4" pipe break, the AP600 one square foot break loads are a conservatively high bound for the AP1000.

For Service Level D conditions the seismic and LOCA loads are combined as the square root of the sum of the square of each load (See AP1000 DCD Table 3.9-5). The table below compares the AP600 design basis loads (pounds) at key locations in the internals with the loads calculated for the AP1000.

AP600/AP1000 Reactor Vessel Internals Faulted (Service Level D) Loads, lbs						
	AP600			AP1000		
Component	**LOCA**	**Seismic**	**Combination**	**LOCA**	**Seismic**	**Combination**
Upper Core Plate Pin	5.73E06	1.16E05	5.73E06	5.73E06	1.25E05	5.73E06
Outlet Nozzle	5.21E05	8.84E03	5.21E05	5.21E05	4.41E04	5.23E05
Lower Radial Key	2.28E07	4.61E05	2.28E07	2.28E07	4.06E05	2.28E07
Reflector/ Shroud Pins	4.85E06	2.41E05	4.86E06	4.85E06	5.35E05	4.88E06

The table illustrates that the AP1000 combined LOCA and seismic loads (Level D Service Conditions) at key reactor vessel internals locations are essentially the same as the AP600 loads. This is assuming conservative LOCA loads for the AP1000. Analyses of the AP600 reactor vessel internals using the above Service Level D loads showed that stress, buckling, and deflection limits were met.

Based on the above load comparison and the results of the AP600 reactor vessel internals stress analyses, the AP1000 reactor vessel internals components are expected to be within acceptable stress and deflection limits for the postulated pipe rupture combined with the safe shutdown earthquake condition. Final stress analyses, which are the responsibility of the Combined License Applicant, will confirm the acceptability of the AP1000 reactor vessel internals. The commitment for the Combined License Applicant to perform the analyses and produce a reactor vessel internals design report which will be available for NRC audit is included in DCD section 3.9.8.2.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 210.019

Question:

Reference, Volume 6, Section 3.9.5.2.3, Pg. 3.9-79:

Please identify the limiting, worst case pipe break used for the faulted condition design basis analysis of the AP1000 RPV internals prototype design. Also provide a summary of analysis results which demonstrate that the appropriate faulted condition allowable stress criteria have been satisfied. (See RAI 210.11)

Westinghouse Response:

As was the case for AP600, for AP1000 nominal pipe sizes of 6" and larger are qualified for elimination of post-rupture dynamic analysis through application of leak-before-break criteria. Therefore, the design basis break analyzed to determine the dynamic response of the AP1000 reactor vessel internals is that of a 4" pipe (pressurizer spray line, first stage ADS line).

See discussion of the analysis of the Service Level D (Faulted) Conditions in the response to RAI 210.012.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 220.007

Question:

AP1000 DCD Subsection 3.8.3.3.1, "Passive Core Cooling System Loads," describes the pressure and thermal transients associated with operation of the passive core cooling system which are used to evaluate structures inside containment. Some of the water temperature transients have changed from the AP600 design and it is not clear how these have affected the analysis and design of the structural modules. Therefore, please address the following issues:

A. The transient temperature was revised from 240°F reached in 5.5 hours (AP600) to 250°F reached in 3.5 hours (AP1000). Provide a discussion to explain the change and how this change was considered in the analysis and design of the AP1000 modules?

B. The extreme transient starting temperature used for the structural design was revised from 50°F (AP600) to 70°F (AP1000). This would seem to be less extreme than the 50°F case in the AP600 design. Provide the basis for this change and explain how this change was considered in the analysis and design of the AP1000 modules?

Westinghouse Response:

The thermal transients associated with operation of the passive core cooling system have been revised. The extreme transient starting temperature used for the structural design has been revised to 50°F and is now the same as was used for the AP600.

The AP1000 containment model was used to determine the IRWST water temperature as a function of time. Decay heat is added to the IRWST using a heater component to represent the PRHR heat exchanger. The following assumptions are used in this model:

- ANSI-79 decay heat with 2σ uncertainty
- IRWST water volume = 75,300 ft^3
- IRWST initial temperature = 50°F
- Complete mixing in the IRWST (no stratification)
- No heatup of IRWST walls or other structures (all decay heat deposited in IRWST)
- Full reactor power of 3400 MWt prior to simultaneous scram and PRHR actuation.
- Containment pressure conditions as calculated by WGOTHIC.

The IRWST temperature transient is shown in the attached Figure 3.8.3-7, which will be added in the DCD. The IRWST begins steaming approximately 5 hours after the event is initiated, and reaches a maximum temperature of 259°F in approximately 11 hours. The containment atmosphere heats up once the IRWST steams. This transient will be used to determine the thermal loads on the IRWST walls. There are no changes in the analysis or design criteria for the structural modules for thermal effects. The critical loading for the concrete filled steel plate modules above the containment internal structure basemat is the thermal gradient across the wall. Close to the containment internal structure basemat the critical loading is the maximum temperature difference between the steel plate and the basemat concrete. The design of the structural modules for the AP1000 IRWST transient will be available for review during the NRC audit of critical sections.

Design Control Document (DCD) Revision:

Revise second bullet in subsection 3.8.3.3.1:

- ADS_2 – This automatic depressurization system transient considers heatup of the water in the in-containment refueling water storage tank. This may be due to prolonged operation of the passive residual heat removal heat exchanger or due to an automatic depressurization system discharge. **For structural design an extreme transient is defined starting at 50°F since this maximizes the temperature gradient across the concrete filled structural module walls.** Prolonged operation of the passive residual heat removal heat exchanger raises the water temperature from an ambient temperature of ~~120~~50°F to saturation in about ~~2~~5 hours, increasing to about 2~~5~~60°F within about 11~~3.5~~ hours. Steaming to the containment atmosphere initiates once the water reaches its saturation temperature. The temperature transient is shown in Figure ~~19.E.4.10-4~~3.8.3-7. Blowdown of the primary system through the spargers may occur during this transient and occurs prior to 24 hours after the initiation of the event. Since the flow through the sparger cannot fully condense in the saturated conditions, the pressure increases in the in-containment refueling water storage tank and steam is vented through the in-containment refueling water storage tank roof. The in-containment refueling water storage tank is designed for an equivalent static internal pressure of 5 psi in addition to the hydrostatic pressure occurring at any time up to 24 hours after the initiation of the event.

PRA Revision:

None

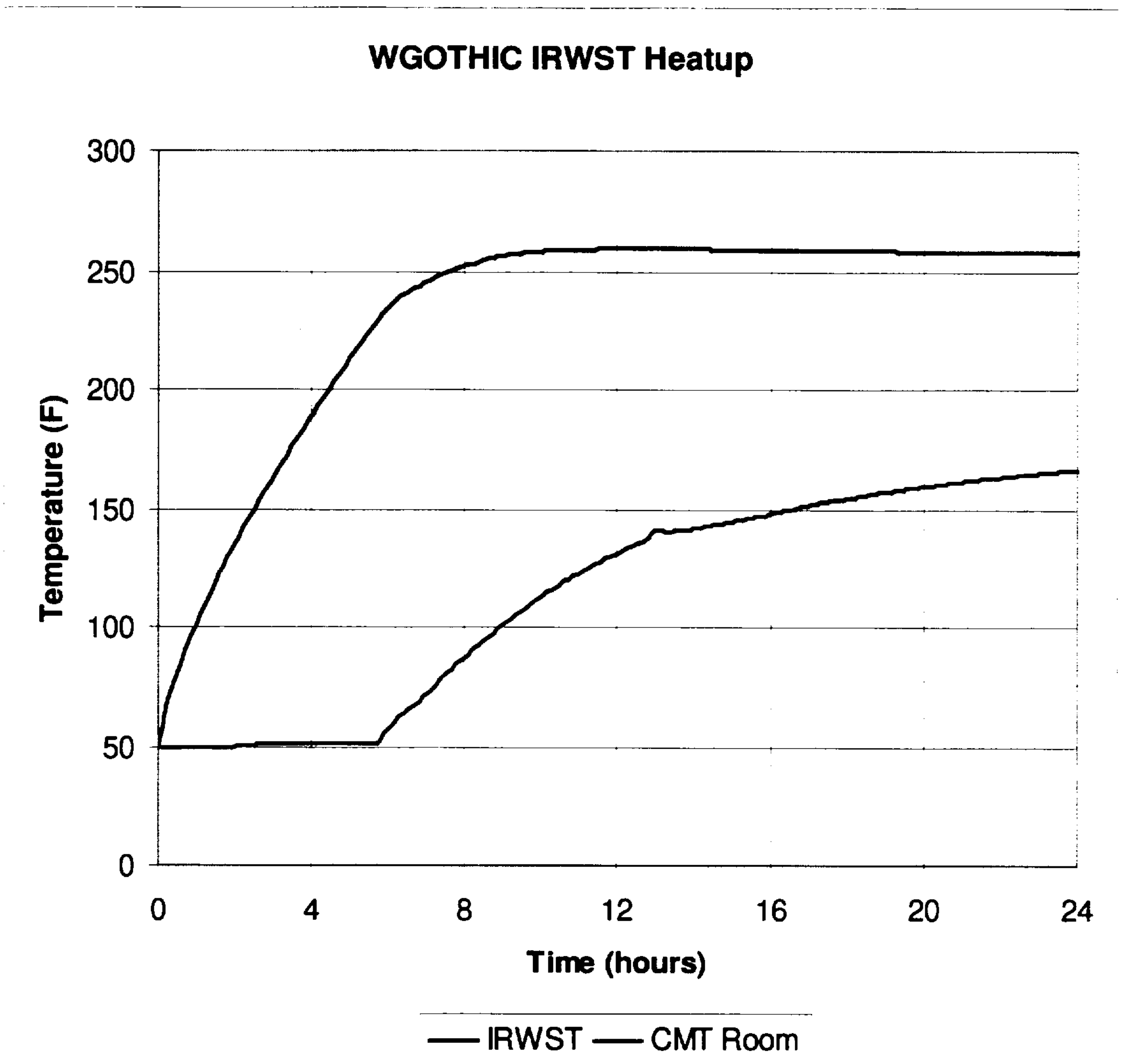


Figure 3.8.3-7
IRWST Temperature Transient

RAI Number: 251.001

Question:

It was stated in Section 10.2.2 that "(t)he rotor design, manufacturing, and material specification and the inspections recommended for the AP1000 provide an acceptable very low probability of missile generation." Subsection 10.2.2 further explains that "(t)he probability of destructive overspeed condition and missile generation, assuming the recommended inspection and test frequencies, is less than 1×10^{-5} per year."

Provide the source of this value (e.g., from WCAP-15783, "Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines"). List all modifications that Westinghouse has made to the Monte Carlo simulation methodology since 1984, the year when an early version of the methodology described in WSTG-3-P, "Analysis of the Probability of a Nuclear Turbine Reaching Destructive Overspeed," was approved by the Nuclear Regulatory Commission (NRC). (DCD Section 3.5.1.3)

Westinghouse Response:

The probability of missile generation in the AP1000 DCD comes from WCAP-15783 and WCAP-15785, WSTG-3-P does not apply to AP1000. Monte Carlo simulation methodology is not used in the AP1000 evaluation of turbine missile generation.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.002

Question:

Address questions 2.a to 2.f if the modifications mentioned in question 251.001 affect the subject of each of these six questions. (DCD Section 3.5.1.3)

a. Provide the probability distributions for both undetected and reported indications for the probabilistic burst and missile analysis and the bases for the selection. How are they used in the Monte-Carlo simulations?

b. Explain how stress corrosion crack growth and fatigue crack growth were considered in your turbine missile analysis. Assess the impact of crack growth due to low cycle fatigue (associated with turbine unit startups and shutdowns) on your missile probability analysis. Also, please demonstrate that the vibratory stresses of the turbine disks due to various excitations are negligible.

c. Was the stress corrosion crack growth rate independent of the level of stress intensity factor? Plot data from tests and operating plants to support your conclusion. If any variables related to stress corrosion crack initiation were included in your current turbine missile probability analysis, please provide detailed information regarding the use of the stress corrosion crack initiation parameters in your analysis.

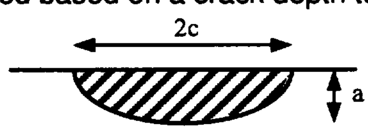
d. Provide all random variables, their distributions, and suggested number of standard deviations that were used in your Monte-Carlo simulations. Explain the need for distributions other than the normal distribution for any of the variables, and justify the use of the suggested number of standard deviations for all variables. Comment on the convergence of the calculated P_1 value for your Monte-Carlo simulation involving this many random variables. Further, how do were the mean values for these variables determined, especially for the mass of bladed disk fragments and fragments from other rotating parts? How do they correlate with industry experience on turbine missile events? Are values for these variables dependent upon the specific design or model of a turbine? Please use a typical turbine model to be used in the AP1000 application as an example and illustrate how these values were established. How were the degree of blade crushing, blade bending, and deformation of stationary blades considered in your calculation of the probability of casing penetration?

e. Assess the contributions due to these modifications to the turbine missile probability reported in the submittal.

Westinghouse Response:

a. In WCAP15783, the missile generation probability was not analyzed utilizing Monte Carlo simulation nor did it use the distributions for undetected and reported indications. The size of the initial crack depth, a_{i_max}, is assumed to be the maximum one which can not be detected by inspection.

Since the inspection procedures used for fully integral rotor forgings will reliably detect a flaw length more than 1.50 mm [0.0591in], a maximum crack depth of 0.375mm [0.0148in] was assumed based on a crack depth to length ratio of 1:4 .



a_{i_max}:	0.375 mm	[0.0148in]
$c_{\_max}$:	1.50/2 mm	[0.0591/2in]

b. Stress corrosion crack growth and fatigue crack growth are included in the turbine missile analysis as explained in WCAP 15783. Note that cracks due to stress corrosion cracking (SCC) begin from the rim of the disk and cracks due to low cycle fatigue (LCF) begin at the surface of the center bore under the disk of the last blades. Because the shrunk-on disk bore has been eliminated, the rotor center bore surface will never get in contact with steam in the AP1000 LP turbine rotor structure. Crack initiation by SCC and LCF begin at different positions, so the interaction between SCC and LCF is not considered in our turbine missile analysis.

c. Yes, the stress corrosion crack growth rate is independent of the level of stress intensity factor. Figure 251.002-1 shows the crack growth rate test results using pre-cracked specimens in actual PWR steam conditions. The crack propagation data shows a plateau with some scatter but the upper limit of crack propagation is almost constant and independent of the level of stress intensity factor. The probability of reaching a critical crack size was analyzed by the modified Clark equation accompanied by uncertainty term, ε. The crack growth rate, γ, model used is as follows:

$$\ln \gamma = a + \frac{b}{T} + c \cdot \sigma_{ys} + \varepsilon$$

The stress corrosion crack initiation is conservatively assumed to exist when the rotor is installed.

d. Monte Carlo simulation is not used in calculating missile generation probability.

e. Monte Carlo simulation is not used in calculating missile generation probability.

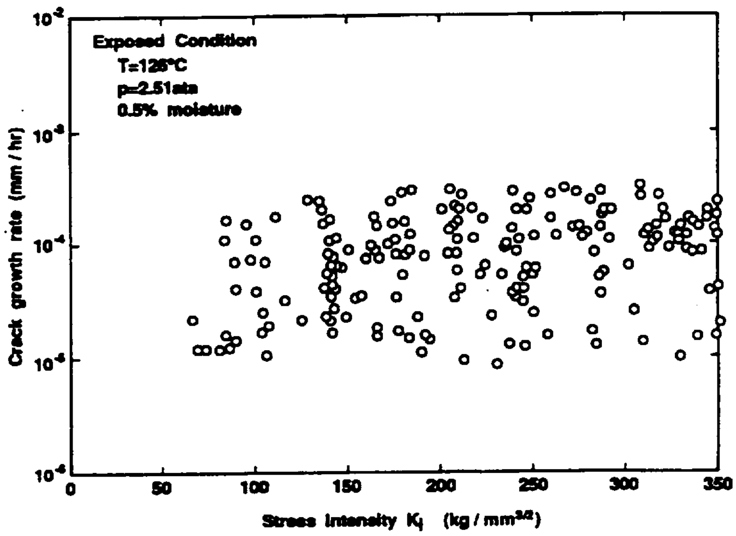
Design Control Document (DCD) Revision:

None

PRA Revision:

None

Figure 251.002-1: Crack Growth Rates of Specimens



PWR-Vol.21 ASME 1993; T. Endo, H. Itoh, Y. Kondo, "MATERIAL ASPECT FOR THE PREVENTION OF ENVIRONMENTALLY-ASSISTED CRACKING IN LOW PRESSURE STEAM TURBINE" Fig.14

RAI Number: 251.003

Question:

How was the probability (10^{-7} per year) determined for the high-trajectory missile to impact safety-related areas of the AP1000? (DCD Section 3.5.1.3)

Westinghouse Response:

As described in DCD Section 3.5.1.3, the turbine generator is located north of the nuclear island with its shaft oriented north-south. This is considered a favorably oriented turbine. The NRC has studied the probability of turbine missiles striking important equipment. The probability of a turbine missile striking and damaging important equipment can be assumed to be 1E-3 per year for favorably oriented turbines per Reference 251.003-1. Based on the conservative missile generation analysis in WCAP-15783 and the missile ejection probabilities in WCAP-15785, the probability of generating a missile is less than 1E-4 per year. Therefore, the probability of generating a turbine missile, which damages safety-related equipment, is less than 1E-7 per year.

References:

251.003-1 NRC letter from C.E. Rossi to J.A. Martin (Westinghouse Electric Corp.), February 2, 1987.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.021

Question:

In the AP600 review, RAIs 251.2 through 251.23 pertain to RCP flywheel integrity. In addition, WCAPs-13734 and 13735, "Structural Analysis Summary for the AP600 Reactor Coolant Pump Flywheel," were submitted as supplemental information for the revised response to question 251.11. Confirm that these responses and the WCAPs are applicable to the AP1000 application as it pertains to RCP flywheel integrity. Should aspects of these responses or reports not be applicable, provide updated information to address the AP600 RAIs as applicable to AP1000 RCP flywheel integrity. (Section 5.4.1)

Note: AP600 RAIs 251.2 through 251.23 were issued by the NRC on October 1, 1992 (NUDOCS Accession No. 9210090123). Westinghouse provided its responses to these RAIs in letters dated January 14, May 24, and May 28, 1993 (NUDOCS Accession Nos. 9301250260, 9306020387, and 9306020220, respectively).

Westinghouse Response:

Responses to AP600 RAIs 251.2 through 251.23 specific to the AP1000 design are given below. The format is to repeat the AP600 question and provide a response specific to the AP1000 design. WCAP-15994-P (Proprietary), WCAP-15994-NP (Non-Proprietary), "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel", November 2002, has been issued to supplement some of the responses given below. In the responses below, this WCAP is referred to as Reference 1.

When the responses below refer to other AP600 RAI responses, the reference is to the AP1000 response to the AP600 RAI as given here in the overall response to this AP1000 RAI.

AP600 RAI 251.2
Westinghouse proposes to use a depleted uranium alloy casting in an Inconel alloy welded enclosure to construct the pump flywheel. These materials are not addressed in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity". Provide technical justifications for the use of these materials (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.2
As noted in Subsection 5.4.1 of the DCD, the AP1000 canned motor reactor coolant pump uses a fundamentally different approach to demonstrate safe operation of the flywheel than the design approach for which Section 5.4.1.1 of the Standard Review Plan and Regulatory Guide 1.14 were developed. Of prime importance in the consideration of flywheel integrity is minimizing the potential for generation of missiles from the flywheel in conformance with the

requirements of General Design Criteria 4. The AP1000 approach is to demonstrate that fragments from a postulated flywheel fracture do not penetrate the surrounding pressure boundary and thus do not become missiles. See the AP1000 response to AP600 RAI 251.11 for additional information on the analysis of the retention of flywheel fragments. This basis of containing postulated fragments is the same as for the rotor and other rotating parts in previous shaft seal pump designs. The approach behind the recommendations of Section 5.4.1.1 of the Standard Review Plan and Regulatory Guide 1.14 is to minimize the potential for a flywheel fracture by extensive testing and inspection of the flywheel.

Although conformance with the recommendations in the regulatory guide is not the only means to demonstrate safe operation of the pump, many of the recommendations are followed in the design and fabrication of the flywheel assembly for operational reliability. Since the AP1000 design does not rely on flywheel material integrity to minimize the potential for the generation of missiles, the quality assurance requirements inherent in the use of ASME Code pressure boundary quality material as suggested by the Standard Review Plan are not required. The design requirements for the flywheel assembly materials are selected to provide a high level of operational reliability. The basis for the design requirements for the flywheel assembly materials is outlined below.

The flywheel assembly is a uranium-alloy casting or forging surrounded by a nickel-chromium-iron alloy enclosure. The material strength used for the analyses that demonstrate flywheel integrity is based on the material specification outlined in AP1000 DCD Table 5.4-2. The material toughness is demonstrated by the yield strength and elongation. See the AP1000 response to AP600 RAI 251.3 for additional information on the fracture toughness properties of the uranium alloy. Since the uranium alloys to be used in the flywheel were not developed for use as pressure boundary materials, ASME Code material specifications do not exist. See the AP1000 response to AP600 RAI 251.23 for additional information on the material specification. Nevertheless, quality assurance practices can confirm that the minimum material requirements are met. The nickel-chromium-iron Alloy 690 material used in the enclosure is a commonly used material with established material specifications.

The uranium alloy does not come in contact with the reactor coolant. The Alloy 690 enclosure material has been shown to be compatible with reactor coolant in other applications. The operating temperature of the coolant surrounding the flywheel assembly is substantially less than the reactor coolant system operating temperature, so stress corrosion cracking of the Alloy 690 is not expected to be an issue. See the AP1000 response to AP600 RAI 251.21 for additional information on the resistance to stress corrosion cracking of the flywheel enclosure.

AP600 RAI 251.3

Westinghouse indicates that the fracture toughness guidelines in Section 5.4.1.1 of the SRP and Regulatory Guide 1.14 are not applicable to depleted uranium alloy castings. Provide information on the fracture toughness properties for this material and propose fracture toughness requirements with technical justifications (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.3
The fracture toughness of the uranium alloy casting is approximately 50 ksi √in. between 100°F and 200°F based on available data. Over the same temperature range the minimum impact energy (Charpy V-notch) is 10 foot-pounds. The material specification for the flywheel material includes a requirement for this minimum impact energy. The material specification does not include a fracture toughness requirement, but the properties and processing specified define a material that meets the 50 ksi √in. minimum.

Calculation of the critical flaw sizes (Reference 1) is based on the 50 ksi √in. fracture toughness. The minimum critical flaw size is greater than 1 inch for a full-length axial crack on the inner diameter. This flaw size was calculated for assembly plus design conditions (125% overspeed).

AP600 RAI 251.4
Provide information on the fabrication process and resulting quality for the depleted uranium alloy casting (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.4
The melting of the depleted uranium alloy for the flywheel casting or forging billet is done under vacuum or inert atmosphere to provide a high quality product. The vacuum or inert atmosphere prevents reaction of the uranium with air and minimizes the potential for the formation of voids. Because of the density of the uranium alloy, slag and other impurities tend to float to the top of the molten metal and porosity in the cast material is not a problem. The molds for the casting are treated to minimize the contamination of the uranium with carbon. The rest of the manufacturing process is controlled to minimize the contamination of the uranium alloy with carbon and hydrogen. Excessive carbon reduces the ductility of the uranium alloy. Hydrogen contamination may induce delayed cracking. Because of the thickness of the flywheel, the final heat treatment is a solution anneal in a vacuum furnace followed by a slow cooling. Other heat treatments such as annealing followed by water quenching and aging hardening are not appropriate for a thick uranium alloy flywheel. See the AP1000 response to AP600 RAI 251.22 for additional discussion of the heat treatment.

AP600 RAI 251.5
Section 1A of the SSAR indicates that the AP600 design meets the guidelines of Regulatory Position 1.d in Regulatory Guide 1.14. However, the flywheel, including the enclosure welds, will not be inspected. Discuss how the flywheel design meets Regulatory Position 1.d.

Westinghouse AP1000 Response to AP600 RAI 251.5

The uranium alloy flywheel is not subject to welding operations, including repair welding, or any other finishing operations that use thermal methods. The component parts of the enclosure are connected together with flexible, full-penetration welds. These welds are inspected following fabrication by ultrasonic testing and liquid penetrant testing. ASME Code, Section III criteria for structural welds are used as guidelines to establish welding and inspection requirements.

See the AP1000 response to AP600 RAI 251.14 for additional information on the analysis and inspection of the enclosure flexible welds. The enclosure represents only a small fraction of the energy in a rotating flywheel assembly. The locations of the flexible welds are such that there is minimal effect on the fracture analysis.

AP600 RAI 251.6
Regulatory Positions 2.c, 2.d, and 2.e in Regulatory Guide 1.14 recommends that an analysis be submitted for staff review. Provide the analysis with appropriate technical justifications. Further, because no inservice inspection for the flywheel is being proposed, describe the flaw size assumed in its analysis (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.6
Regulatory Positions 2.c., 2.d., and 2.e. in Regulatory Guide 1.14 recommend that analyses be conducted to predict the critical speed for ductile failure, nonductile failure, and excessive deformation of the reactor coolant pump flywheel. As noted in Subsection 5.4.1 of the DCD and the AP1000 response to AP600 RAI 251.2, the approach to demonstrate safe operation of the AP1000 canned motor reactor coolant pump flywheel differs from the design approach for which Regulatory Guide 1.14 was developed. The AP1000 design approach of demonstrating that postulated flywheel fragments are contained by the pump structure limits the significance of the analysis of critical flywheel failure speeds.

The analysis completed for the flywheel structure evaluates the stress intensity levels at the normal speed and at the design speed of 125 percent of normal. The calculated stress levels are evaluated against ASME Code, Section III, Subsection NG stress limits and the recommended stress limits in Positions 4.a. and 4.c. of Standard Review Plan 5.4.1.1. of one-third and two-thirds of yield stress for normal speed and design speed, respectively. The margin inherent in these limits provides an appropriate degree of margin to failure at the normal and design speeds. See Reference 1 for additional information on the evaluation of stress in the flywheel assembly.

The flaw size assumed in the evaluation of fracture toughness is described in the AP1000 response to AP600 RAI 251.3.

AP600 RAI 251.7
Section 1A of the SSAR indicates conformance with Regulatory Position 2.f in Regulatory Guide 1.14. Provide information to support this statement.

Westinghouse AP1000 Response to AP600 RAI 251.7
As noted in Subsection 5.4.1.3.6.3 of the AP1000 DCD and in the AP1000 response to AP600 RAI 251.8, the design speed (125 per cent of normal speed) envelopes all expected and postulated overspeed conditions including overspeeds due to postulated pipe ruptures. See the AP1000 response to AP600 RAI 251.8 for a discussion of the size of postulated pipe ruptures also. This limitation on the potential overspeed along with the design approach of demonstrating that postulated flywheel fragments are contained by the pump structure limits the significance of

the analysis of critical flywheel failure speeds. The analysis completed for the flywheel stress report evaluates the stress intensity levels at the normal speed and the design speed of 125 percent of normal. The calculated stress levels satisfy the ASME Code, Section III, Subsection NG stress limits. The calculated primary stress levels are less than the recommended stress limits in Positions 4.a. and 4.c. of Standard Review Plan 5.4.1.1 of one-third and two-thirds of yield stress for normal speed and design speed, respectively. See Reference 1 and the AP1000 responses to AP600 RAIs 251.16, 251.17, 251.18, and 251.19 for additional information on the evaluation of stress in the flywheel assembly.

The flywheel structural analysis verifies that the failure modes outlined in Positions 2.c, 2.d, and 2.e of Regulatory Guide 1.14 do not occur at the design speed. The flywheel stress evaluation noted above demonstrates an appropriate margin against these failure modes. In addition, the design of the canned motor pump mitigates the effects of hypothetical failures by these modes, as outlined below.

The AP1000 response to AP600 RAI 251.11 discusses the containment of fragments from a postulated flywheel fracture. The mode of failure, ductile or nonductile, would not alter the capacity of the surrounding pump structure to absorb the energy of the fragments and prevent the generation of missiles from the flywheel assembly.

Regulatory Guide 1.14 defines excessive deformation as any deformation that could cause separation of the flywheel from the shaft. Because of the restriction of the lateral movement of the flywheel assembly by the surrounding structure and axial movement by the thrust bearings, the loss of shrink fit would not be expected to result in substantial movement of the flywheel assembly or significant separation of the assembly from the shaft. This restriction in movement of the flywheel assembly and the adjacent location of the journal bearing to the flywheel assembly minimize the potential for a structural failure of the shaft during a hypothetical overspeed transient sufficient to result in excessive deformation.

Neither separation of the flywheel assembly from the shaft nor structural failure of the shaft would result in a loss of safety-related function of the canned motor pump during an overspeed transient. That safety-related function is the maintenance of the primary pressure boundary. Neither separation of the flywheel assembly nor structural failure of the shaft would degrade the pressure boundary of the pump. The safety-related function of providing flow during coastdown of the pump is not germane during an overspeed transient.

AP600 RAI 251.8

Section 1A of the SSAR indicates conformance with Regulatory Position 2.g in Regulatory Guide 1.14, relating to the flywheel overspeed due to postulated pipe rupture. Section 5.4.1.3.6.3 of the SSAR appears to assume the application of leak-before-break (LBB) for all high-energy piping 10 cm (4 in) in diameter or larger. Since the outcome of the staff's review of the application of LBB to the AP600 design is uncertain, the staff recommends that Westinghouse discuss how the flywheel conforms with RG 1.14 if the criteria of Section 3.6.2 and BTP MEB 3-I is used to determine pipe break size.

Westinghouse AP1000 Response to AP600 RAI 251.8
As was the case for AP600, for AP1000 nominal pipe sizes of 6" and larger are qualified for elimination of post-rupture dynamic analysis through application of leak-before-break criteria. Therefore, the largest break analyzed to determine the dynamic response of the AP1000 reactor coolant pump is that of a 4" pipe (e.g. pressurizer spray line, first stage ADS line).

The overspeed analysis of the AP1000 reactor coolant pump flywheel is based on the design speed of 125 per cent of normal speed. The AP1000 pipe rupture overspeed is expected to be enveloped by the design speed since the reactor coolant main loops and all of the branch line piping with a nominal diameter of 6 inches and greater are being qualified for LBB. The pipe rupture overspeed is expected to be substantially less than any of the calculated critical flywheel failure speeds.

As noted in the AP1000 response to AP600 RAI 251.2, the approach used to demonstrate the safe operation of the flywheel is containment of the fragments from a postulated fracture by the surrounding pump structure. For a postulated flywheel fracture at the flywheel design speed there is a large amount of margin in the calculated capability of the pump structure to contain flywheel fragments. Thus even in the event of a postulated failure of a flywheel during a hypothetical break of a reactor coolant loop pipe, it is not expected that additional breaks in the reactor coolant pressure boundary would be created nor would missiles be generated by the flywheel.

AP600 RAI 251.9
Section 1A of the SSAR indicates that Westinghouse is taking exception to Regulatory Position 4.a in Regulatory Guide 1.14. Propose an alternative to this position with appropriate technical justifications.

Westinghouse AP1000 Response to AP600 RAI 251.9
A spin test is done on the flywheel assembly after the enclosure is welded closed. Inspection of the flywheel inside the assembly is not practical. Because of the density of the uranium, radiographic examination is also not a practical option.

The uranium alloy flywheel is ultrasonically inspected following final machining and prior to assembly of the enclosure around the flywheel. The ultrasonic inspection conforms to the requirements of the ASME Code, Section III, paragraph NB-2574, for ferritic steel castings, including the use of the procedures outlined in SA-609 (ASTM-A-609). See the AP1000 response to AP600 RAI 251.13. Machined surfaces of the uranium flywheel undergo liquid penetrant inspection prior to final assembly. The liquid penetrant inspection conforms with the requirements of the ASME Code, Section III, paragraph NB-2576, including the use of the procedures outlined in SA-165 (ASTM-A-165).

In-process controls during the assembly of the enclosure onto the flywheel are used to provide for the quality of the completed assembly. The spin test of the completed assembly confirms the quality of the flywheel assembly. Since the basis for safe operation of the flywheel assembly is the retention of the fragments from a postulated fracture by the structure of the pump, inspection subsequent to the spin test is not necessary for safe pump operation.

AP600 RAI 251.10

Performance of inservice inspection of the flywheel should be considered. If the ISI procedures in Section 5.4.1.1 of the SRP are not applicable to uranium flywheels, propose alternative inservice inspection procedures with appropriate technical justifications (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.10

Inservice inspection of the uranium alloy flywheel would be very labor intensive and involve significant radiation exposure. Since the surrounding structure of the pump would contain flywheel fragments even in the worse case fracture, inservice inspection would do little to increase the safety of pump operation. The technical justification of no inservice inspection is the analysis that shows that the fragments of a fractured flywheel would not penetrate the pressure boundary of the pump to become missiles (see the AP1000 response to AP600 RAI 251.11). On this basis a flywheel fracture is an operational reliability consideration rather than a safety-related consideration. The use of inspections and in-process controls during fabrication of the flywheel assembly and a spin test of the completed assembly also provide verification of the initial quality of the assembly. The use of vibration monitoring of the pump during operation provides an indication of rotating part stability and thus integrity. This allows any necessary maintenance to be performed as needed for operational reliability.

As noted in the AP1000 response to AP600 RAI 251.2 the design approach to the flywheel in the AP1000 canned motor reactor coolant pump is fundamentally different than that for previous shaft seal reactor coolant pump designs. The canned motor pump design was selected for several safety related and operational reasons. Inherent in the design of a canned motor reactor coolant pump is the location of the flywheel assembly within a pressure housing and the flywheel enclosure in contact with reactor coolant. To make the flywheel readily accessible for an inservice inspection of marginal utility, many advantages of the canned motor pump would have to be foregone. Routine inservice inspection of the flywheel is neither recommended nor advantageous.

AP600 RAI 251.11

Section 1A of the SSAR states that a flywheel rupture will be contained within the stator shell. Provide an analysis and technical justifications supporting this statement.

Westinghouse AP1000 Response to AP600 RAI 251.11

The canned motor reactor coolant pump has an outer shell that comprises the pressure boundary. The shell is analyzed to demonstrate that in the event of a postulated flywheel fracture, the surrounding pump structure is sufficient to prevent missiles from leaving the pump. The analysis considers that portion of the shell, including the flange, and motor end cap around the flywheel assembly between the top and bottom elevations of the assembly as the barrier to missile generation. The structural analysis summary is documented in Reference 1 and is outlined below.

The analysis of the capacity of the surrounding pump structure to contain the fragments of a postulated flywheel failure is done using the energy absorption equations of Hagg and Sankey (Hagg, A. C., and Sankey, G. O., "The Containment of Disk Burst and Fragments by Cylindrical Shells," ASME Journal for Power, April 1974, pp. 114-123). The containment of missile-like metal disk fragments is by a two-stage process. Stage 1 involves inelastic impact and transfer of momentum to include an effective target mass. To show that the fragments do not perforate the surrounding structure, the energy dissipated in plastic compression and shear strain and the local impact area must be sufficient to account for the loss in kinetic energy of the system. For the nonperforation case the process enters Stage 2, which involves dissipation of energy in plastic tension strain over extended volumes of shell material. For containment, the energy dissipated in plastic strain in Stage 2 must account for the residual kinetic energy on the system. In predictive calculations it is more conservative to consider Stage 2.

For the AP1000 reactor coolant pump analysis, the uranium insert in the flywheel assembly is assumed to fracture at the design speed of 125 percent of normal speed. The worst-case scenario of fragment size and number was derived analytically, using methods from Hagg and Sankey to determine the mass and velocity combination that would produce the most severe impact on the surrounding pressure boundary components. The following conservative assumptions are also made:

1. End plates and welds of the flywheel enclosure and the coolant surrounding the flywheel assembly have negligible energy-absorbing capability.
2. Only the mass in the stator shell and flange and the motor end cap between the elevation of the top and bottom of the flywheel assembly are considered to absorb energy.
3. Closure bolts and joint effects were not considered to be affected.
4. The minimum material properties were used.

The analysis results show that the fragments impact the surrounding pump structure with a kinetic energy of less than 15 percent of the tensile energy-absorbing capability of the surrounding pump structure. Thus the components around the flywheel contain the flywheel fragments using only a small portion of the energy-absorbing capability. The energy absorbed by the flywheel enclosure is small compared to the surrounding pump structure and was not considered in the calculation of flywheel fragment containment within the pump pressure boundary.

AP600 RAI 251.12

Section 1A of the SSAR indicates that a "small" flywheel rupture or leak in the enclosure will not result in stresses in the pressure boundary to cause a break. Provide information to clarify what is the intent of the term "small" flywheel rupture. The staff is concerned with the rupture of the flywheel into large fragments of high energy.

Westinghouse AP1000 Response to AP600 RAI 251.12
The canned motor pump design is evaluated for a spectrum of postulated uranium flywheel fractures. A fracture that ruptures the flywheel enclosure is bounded by the analysis of the worst-case fracture (Reference 1) that shows that the fragments are contained as noted in AP1000 DCD section 5.4.1.3.6.3. A fracture that deforms the enclosure enough to bring it in contact with the surrounding structure is bounded by the analysis described in AP1000 DCD section 5.4.1.3.6.2. A small fracture in the context of the DCD discussion is one that may unbalance the assembly, but any resulting fragment is contained by the enclosure without sufficient deformation to result in interference with the surrounding structure. The discussion of these faults on the low end of the spectrum are included for completeness of the discussion of postulated flywheel fractures.

AP600 RAI 251.13
Section 5.4.1.3.6.3 of the SSAR indicates that ultrasonic inspection of the uranium following final machining will be based on ASTM A388 as modified for uranium. Identify any modifications to the application of ASTM A388 to the AP600 design with appropriate technical justifications. In addition, demonstrate that this preservice inspection is equivalent to that in Section III of the ASME Code.

Westinghouse AP1000 Response to AP600 RAI 251.13
ASTM A388, which is a standard for use of ultrasonic inspections on steel forgings, is not given as the standard for ultrasonic inspection of the uranium following final machining in AP1000 DCD section 5.4.1.3.6.3. ASTM A609, which is a standard for use of ultrasonic inspections on ferritic steel castings, will be used as the standard for ultrasonic inspection of the uranium flywheel. Changes to the practices specified in the standard to account for use on uranium include the use of uranium reference blocks and potential additional restrictions on the couplants used. The size and frequency of transducers may also be different than the standard, although the inspection of a prototype flywheel casting was done with a transducer size and frequency in the range designated in the standard. Areas of the standard that are not dependent on the type of material inspected, such as personnel qualification requirements, surface conditions, procedures, and data reporting should not have to be modified. See the AP1000 response for AP600 RAI 251.9 for additional discussion of inspection of the uranium flywheel.

It is not the intent that the inspection of the uranium alloy flywheel be equivalent in every respect to inspections required of components built to the requirements of the ASME Code, Section III. The requirements for the flywheel are chosen to provide high operational reliability. There are no pressure boundary functions associated with the flywheel assembly that require the use of the ASME Code.

AP600 RAI 251.14
Demonstrate that the construction of the flywheel enclosure meets Section III of the ASME Code, including inspection (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.14
Since the flywheel enclosure is not a pressure boundary and is not relied upon to contain fragments from a postulated flywheel fracture, there is no requirement to meet the requirements of the ASME Code, Section III for construction of the enclosure. Additionally, the enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The function of the enclosure is to isolate the uranium alloy from the reactor coolant circulating in the reactor coolant pump. A leak in the enclosure could result in an out-of-balance condition for the flywheel assembly or, over the long term, the possible introduction of depleted uranium into the reactor coolant. Neither of these events represents a catastrophic failure and both would be addressed by other systems. Sensors in the pump detect vibration of the pump and the chemical and volume control system includes provisions to reduce contaminants in the reactor coolant. The uranium would be detected by periodic sampling of the reactor coolant by the primary sampling system.

The ASME Code, Section III criteria for structural welds are used to establish welding requirements and inspection requirements for the enclosure. As noted in the AP1000 DCD section 5.4.1.3.6.3, the welds are subject to dye penetrant and ultrasonic tests. The ASME Code Subsection NG stress limit criteria are used as guidelines to evaluate the stress in the enclosure components and the flexible welds for normal and design speeds. The use of the ASME Code, Section III to establish design, fabrication, and inspection requirements was selected to provide operational reliability and availability.

AP600 RAI 251.15
Demonstrate that the design overspeed of the flywheel is at least 10% above the highest anticipated overspeed (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.15
The requirement for the AP1000 is that the design speed (125 percent of normal speed) be greater than or equal to anticipated overspeed conditions due to electrical faults and overspeed conditions due to postulated pipe breaks. Anticipated overspeed conditions are those due to electrical faults including turbine overspeed events. Because of design of the turbine control system (see AP1000 DCD section 10.2.2), reactor coolant pump overspeed resulting from an electrical fault is expected to be less than the design speed. See the AP1000 response to AP600 RAI 251.8 for a discussion of flywheel overspeed due to postulated pipe rupture.

Since the basis for safe operation of the pump with respect to flywheel integrity is the containment of flywheel fragments by the pump structure rather than the prevention of fracture (see the AP1000 responses to AP600 RAIs 251.2 and 251.11), a 10% margin between calculated overspeed and the design speed is not necessary to assure safe operation.

AP600 RAI 251.16
Show that the combined stresses for the uranium flywheel at the normal operating speed, due to centrifugal forces and the interference fit of the wheel on the shaft, is less than 1/3 of the minimum specified yield strength (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.16
The flywheel structural analysis verifies that the primary stresses in the uranium due to centrifugal forces at the normal operating speed are less than one-third of the minimum yield strength. The combination of primary and secondary stresses is evaluated using stress limits in the ASME Code, Section III, Subsection NG. The secondary stresses are due to the interference fit of the uranium on the shaft. The allowable stress values developed applying ASME Code, Section III, factors (Appendix III) to the mechanical properties of uranium are satisfied for analyzed stresses at normal operating speed. See Reference 1 for analysis details.

AP600 RAI 251.17
Discuss how the limit in AP600 RAI 251.16 is met for the flywheel enclosure and associated welds (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.17
The evaluation of the flywheel enclosure does not use the limit of one-third of minimum yield strength as a criterion for normal operating speed. The flywheel enclosure prevents contact of coolant with the uranium flywheel. No credit is taken in the analysis of the flywheel missile generation for the retention of the fragments by the enclosure, and the flywheel enclosure contributes only a small portion of the energy in a rotating flywheel assembly. The evaluation of the stress in the flywheel enclosure components and the flexible welds connecting the components for normal and design speeds uses the criteria in Subsection NG of the ASME Code as a guideline. The ASME Code limits are satisfied for analyzed stresses of the flywheel enclosure at the normal operating speed (see Reference 1).

The AP600 and AP1000 flywheel enclosure designs are very similar. The stresses in the AP1000 flywheel enclosure at normal operating speed are similar to those in the AP600 flywheel enclosure. The maximum radial displacement at the AP1000 flywheel enclosure welds is also similar to that for the AP600 flywheel enclosure welds. The AP600 flywheel enclosure welds were shown to meet the ASME Code limits during operation at the normal operating speed. Therefore, it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits during normal operating conditions.

AP600 RAI 251.18
Show that the combined stresses for the uranium flywheel at the design overspeed, due to centrifugal forces and the interference fit, is less than 2/3 of the minimum specified yield strength (Section 5.4.1).

Westinghouse AP1000 Response to AP600 251.18
The flywheel structural analysis verifies that the combined stresses in the uranium flywheel due to centrifugal forces and the interference fit at the design speed of 125 percent of normal speed are less than the limit of two-thirds of the minimum yield strength. The combination of primary and secondary stresses is also evaluated using stress limits in the ASME Code, Section III, Subsection NG. The secondary stresses are due to the interference fit of the uranium on the shaft. The allowable stress values developed applying ASME Code, Section III, factors

(Appendix III) to the mechanical properties of uranium are satisfied for analyzed stresses at the design speed (see Reference 1).

AP600 RAI 251.19
Discuss how the limit in AP600 RAI 251.18 is met for the flywheel enclosure and associated welds (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.19
The evaluation of the flywheel enclosure does not use the limit of two-thirds of minimum yield strength as a criterion for design speed conditions. The criteria in the ASME Code, Section III, Subsection NG, are used as a guideline for stress limits. The ASME Code limits are satisfied for analyzed stresses in the flywheel enclosure at the design speed. See Reference 1 and the AP1000 response for AP600 RAI 251.17.

The AP600 and AP1000 flywheel enclosure designs are very similar. The stresses in the AP1000 flywheel enclosure at design speed are similar to those in the AP600 flywheel enclosure. The maximum radial displacement at the AP1000 flywheel enclosure welds is also similar to that for the AP600 flywheel enclosure welds. The AP600 flywheel enclosure welds were shown to meet the ASME Code limits during operation at the design speed. Therefore, it is expected that the AP1000 flywheel enclosure weld stresses will also meet the ASME Code limits during operation at the design speed.

AP600 RAI 251.20
Demonstrate that the shaft and the bearings supporting the flywheel will be able to withstand any combination of loads from normal operation, anticipated transients, the design basis loss-of-coolant accident, and the safe shutdown earthquake (Section 54.1).

Westinghouse AP1000 Response to AP600 RAI 251.20
The containment of fragments from a postulated fracture of the flywheel is not dependent on the support of the shaft and flywheel by the bearings. Postulated failures of the bearings and shaft would result in the rotating assembly being slowed to a stop. Bearing or shaft failures would be indicated by vibration or temperature sensors. A postulated failure of a bearing or shaft that allowed excessive lateral movement would result in contact between one or more rotating parts and the surrounding structure thereby slowing the rotation. A postulated failure of a bearing or shaft that allowed excessive axial movement would not remove the restriction provided by the pump internals, including the impeller and suction adapter. Thus a failure that would allow axial movement would not result in significant movement of the flywheel assembly.

Based on this information, the effect of these loads on the shaft and bearings is of interest with regard to operational reliability but not with regard to safe operation. The shaft and bearing supports are evaluated for loads due to seismic events.

AP600 RAI 251.21

Identify the materials for the flywheel enclosure and associated welds. Provide technical justification to show that the flywheel enclosure and associated welds are resistant to stress corrosion cracking, especially if Inconel 600 or 182 materials will be used (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.21

The material of construction of the flywheel assembly enclosure is nickel-chromium-iron Alloy 690. The material for the welding filler metal is nickel-chromium-iron Alloy 52. Since the coolant surrounding the flywheel assemblies is normally at a relatively low temperature (approximately 165 F) and Alloy 690 has shown good resistance to stress corrosion cracking in applications at the higher reactor coolant system temperatures, primary water stress corrosion cracking in the flywheel assembly would not be expected.

AP600 RAI 251.22

Demonstrate that the uranium flywheel is resistant to stress corrosion cracking or other potential degradation mechanisms in a reactor coolant environment (Section 5.4.1).

Westinghouse AP1000 Response to AP600 RAI 251.22

The uranium alloy flywheel is sealed in the nickel-chromium-iron alloy enclosure and is not in contact with the reactor coolant or other fluid. See the AP1000 response to AP600 RAI 251.14 for additional discussion of the enclosure flexible welds. The uranium alloy flywheel is heat treated by solution annealing in a vacuum furnace and slowly cooled. This heat treatment minimizes the potential for residual stresses. The heat treatment process also removes hydrogen from the material to reduce the potential for hydrogen embrittlement. Since the depleted uranium alloy is not in contact with reactor coolant or any other fluid and operates at a relatively low temperature, degradation of the material is not expected.

AP600 RAI 251.23

Table 5.4-2 in the SSAR lists the flywheel material specifications. Provide the technical basis for these specifications.

Westinghouse AP1000 Response to AP600 RAI 251.23

The material specification information including ultimate tensile strength and yield strength provided in AP1000 DCD Table 5.4-2 is based on material testing by the material supplier. The composition of the alloys, including the limits on the constituent elements, is also based on the experience of the material supplier. The production of the uranium flywheel is controlled to minimize the formation of voids or other defects. The heat treatment process is controlled to provide the required material properties. See the AP1000 response to AP600 RAI 251.22 for a discussion of the heat treatment. Quality assurance testing of the material verifies that the material supplied conforms to the material specification. Ultrasonic and liquid penetrant inspections are performed on the uranium flywheel to verify the absence of unacceptable defects. See the AP1000 responses to AP600 RAIs 251.9 and 251.13 for a discussion of the ultrasonic and liquid penetrant inspections.

Design Control Document (DCD) Revision:

See the response to AP1000 RAI 440.040.

PRA Revision:

None

RAI Number: 251.023

Question:

Provide a definition of fracture appearance transition temperature (FATT) and discuss the relationship between FATT and nil-ductility transition (NDT) temperature and reference nil-ductility temperature (RT_{NDT}). (Section 10.2.3)

Westinghouse Response:

The fracture appearance transition temperature (FATT) is defined as the temperature where a 50% ratio of the ductile fracture surface area appears for a Charpy impact test.

In general, the nil-ductility transition (NDT) temperature and reference nil-ductility temperature (RTNDT) are used to assess the nuclear vessel steel. These temperatures are not applied to evaluate the probability of turbine missile generation for the AP1000 LP rotors.

Figure 251.023-1 shows an example of the relationship between FATT and nil-ductility transition (NDT) temperature. The reference nil-ductility temperature (RTNDT) is obtained by conducting Charpy impact tests and drop-weight tests and is related to the fracture toughness.

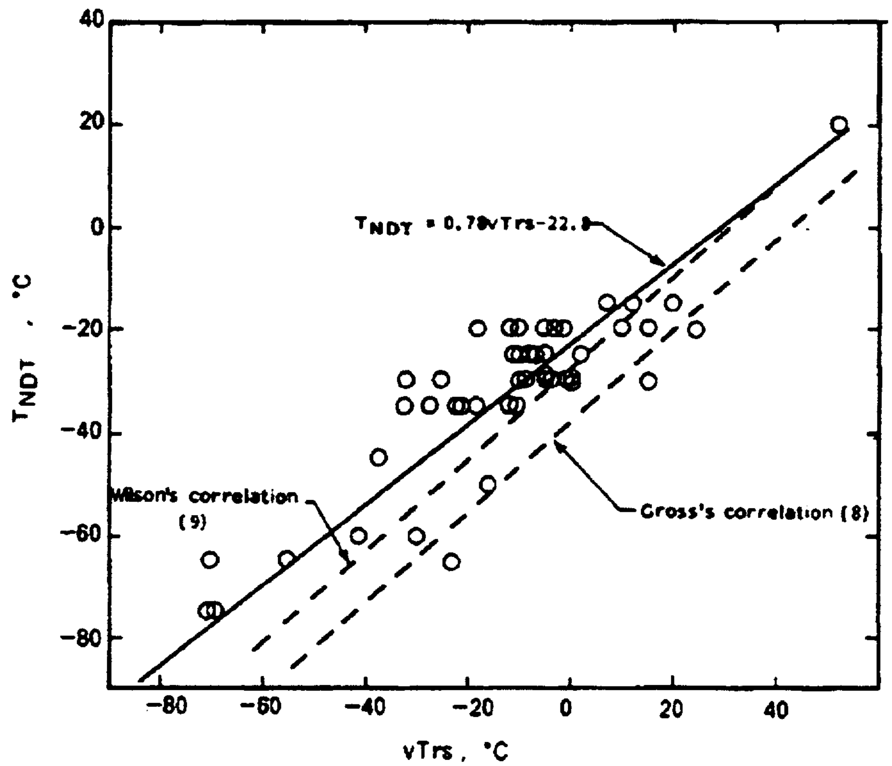
Design Control Document (DCD) Revision:

None

PRA Revision:

None

Figure 251.023-1: Relationship between FATT (vTrs) and NDT



AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number 251.024

Question:

The second paragraph in Section 10.2.3.2 contains the statement, "(t)he ratio of material fracture toughness, K_{IC} (as derived from material tests on each rotor) to the maximum tangential stress for rotors at speeds from normal to design overspeed, will be at least 200 ksi x $\sqrt{in}$ (or at least 2) at minimum operating temperature." This sentence is not clear and should be revised. Confirm that you are trying to suggest that fracture toughness will be at least 200 ksi x $\sqrt{in}$ and the ratio of fracture toughness to the maximum applied stress intensity factor for rotors at speeds from normal to design overspeed will be at least 2. (Section 10.2.3)

Westinghouse Response:

The DCD will be modified as indicated below.

Design Control Document (DCD) Revision:

10.2.3.2 Fracture Toughness

Suitable material toughness is obtained through the use of materials described in Subsection 10.2.3.1 to produce a balance of material strength and toughness to provide safety while simultaneously providing high reliability, availability, and efficiency during operation. The restrictions on phosphorous, sulphur, aluminum, antimony, tin, argon, and copper in the specification for the rotor steel provides for the appropriate balance of material strength and toughness. The impact energy and transition temperature requirements are more rigorous than those given in ASTM 470 Class 6 or 7.

Bore stress calculations include components due to centrifugal loads and thermal gradients where applicable. **Fracture toughness will be at least $220MPa \cdot \sqrt{m} = 200ksi \cdot \sqrt{in}$ and the ratio of fracture toughness to the maximum applied stress intensity factor for rotors at speeds from normal to design overspeed will be at least 2.** ~~The ratio of material fracture toughness, K_{IC} (as derived from material tests on each rotor) to the maximum tangential stress for rotors at speeds from normal to design overspeed, will be at least 200 ksi x $\sqrt{in}$ (or at least 2) at minimum operating temperature.~~ Material fracture toughness needed to maintain this ratio is verified by mechanical property tests on material taken from the rotor.

The rotor is evaluated for fracture toughness by criteria that include the design duty cycle stresses, number of cycles, ultrasonic examination capability and growth rate of potential flaws. Conservative factors of safety are included for the size uncertainty of potential or reported ultrasonic indications, rate of flaw growth (da/dN versus dK) and the duty cycle stresses and number.

Reported rotor forging indications are adjusted for size uncertainty and interaction. A rotor forging with a reported indication that would grow to critical size in the applicable duty cycles is not accepted. The combined rotation and maximum transient thermal stresses used in the applicable duty cycles are based on the brittle fracture and rotor fatigue analyses described below.

Maximum transient thermal stresses are determined from historical maximum loading rates for nuclear service rotors.

PRA Revision:

None

RAI Number: 251.025

Question:

It was mentioned in the third paragraph of Section 10.2.3.2 that conservative factors of safety are included for the size uncertainty of potential or reported ultrasonic indications, rate of flaw growth, and the duty cycle stresses and number. Provide these factors of safety, and comment on how they are determined. (Section 10.2.3)

Westinghouse Response:

The values of initial crack size, flaw growth rate, centrifugal and thermal stress and cycle number used to evaluate the failure mechanism of low cycle fatigue are explained in Section 4.3 of WCAP-15783.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 251.026

Question:

It was mentioned in the first paragraph of Section 10.2.3.2.1 that the maximum rotor stress is determined from rotation, steady-state thermal loads, and transient thermal loads from startup and load change. Provide the operating speed and the first and second critical speeds for the rotor. If any of the rotor critical speeds are below the operating speed, explain why you do not need to consider rotor vibratory stresses when passing through critical speeds during startups and shutdowns. (Section 10.2.3)

Westinghouse Response:

The operating speed of the turbine is provided in Table 10.1-1 and Section 10.2.2.1 of the DCD. The vibratory stress when passing through critical speeds during startups and shutdowns is not included in the evaluation of low cycle fatigue. This is because the bending stress for this condition is greatest on the surface of the rotor and negligibly small on the rotor bore surface, which is the point where maximum stress of low cycle fatigue appears. The low cycle fatigue evaluation is discussed in Section 4.3 of WCAP-15783.

Stress amplitude by the rotation is estimated by centrifugal forces at rated speed and 120% overspeed at every start up and shut down cycles.

The numbers of cycles used for evaluation are assumed to be 600 cycles (20 times per year for 30 years operating), and 1500 cycles to assess the sensitivity of the analysis to this assumption.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 251.027

Question:

Provide the K_{IC} value and the factor of safety that was used to generate the allowable initial flaw area from an initial flaw area. Discuss the appropriateness of the assumption that a crack would originate from the centerline for rotors without bores. (Section 10.2.3)

Westinghouse Response:

The fracture toughness value and the factor of safety for the AP1000 rotor evaluation are given in WCAP 15783. The initial flaw size was taken as the minimum flaw that would be detected during pre-service inspections (see WCAP-15783 and the response to RAI 251.002). The low cycle fatigue stress for rotors without bores is not discussed, because the stress of no-bore rotor is lower than for a bore rotor.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 251.029

Question:

It was mentioned in Section 10.2.3.6 that the maintenance and inspection program plan for the turbine assembly and valves is based on turbine missile probability calculations reported in WCAP-15783, operating experience of similar equipment, and inspection results. Provide the calculated turbine missile probability results that were used for this purpose and explain how they were used to determine the inspection intervals of 10 years for low-pressure (LP) turbines and 8 years for high-pressure (HP) turbines, the inspection intervals of 3 years for a variety of valves, and the quarterly testing frequency for valves. (Section 10.2.3)

Westinghouse Response:

The turbine inspection interval of assembly and valves is determined based on not only the probability of turbine missile generation but also operating experience of similar equipment and inspection results.

The maintenance and inspection program plan that applies to turbine missile generation is described in WCAPs 15783 and 15785. It is concluded that turbine missile generation probability is low enough even without inspection or maintenance for more than 30 years. According to operating experience of similar equipment and inspection results, the inspection intervals is established at 10 years for LP turbines, 8 years for HP turbines and 3 years for a variety of valves. The quarterly testing frequency for valves is based on avoiding the potential for destructive overspeed conditions due to valve failures as discussed in WCAP-15785.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 252.001

Question:

Recent NRC generic communications, including NRC Bulletins 2001-01, 2002-01 and 2002-02, have addressed issues related to cracking of vessel head penetration (VHP) nozzles and degradation of the reactor pressure vessel (RPV) head in operating PWRs. Describe how this operational experience has been incorporated into the AP1000 design. Specifically, address the differences in the AP1000 design compared to the current fleet of PWRs, including the following specific items:

a. geometry of the VHP nozzle weld joint,

b. processes used for fabrication of the nozzle base material,

c. accessibility for inspection of the VHP nozzles and the RPV head - describe any impediments or limitations in the AP1000 design,

d. materials used for both the nozzle base material and the welds, and

e. operating conditions, including the operating temperature of the RPV head, provisions for bypass flow to cool the head, etc. (Section 4.5.1)

Westinghouse Response:

In relation to the Alloy 600 issues of the current fleet of PWRs, operational experiences in materials, fabrication processes, and inspection methods were considered in the design of AP1000 RPV head. A comparison of some key design and fabrication features in the AP1000 compared to the current fleet of Westinghouse PWRs are as follows:

a. The geometry of the AP1000 vessel head penetration nozzle weld joint is the same as in current Westinghouse PWRs.

b. The main process used for fabrication of the AP1000 nozzle base material is the automatic welding process. It will be combined with manual welding processes as necessary. In current Westinghouse PWRs the main welding processes were manual.

c. The AP1000 design contains the following penetrations through the reactor vessel head: CRDM penetrations, top-mounted in-core instrumentation and a vent line. There are no top-mounted in-core instrumentation penetrations in the current fleet of Westinghouse-designed PWRs. Accessibility to the AP1000 penetrations for inspection is the same as that for current PWRs, e.g. under the head. Inspection accessibility has been increased to the ID surface of the CRDM penetrations in that the thermal sleeves have been eliminated.

Thus the small gap access is eliminated and substituted for an open access tube. The top-mounted in-core instrumentation penetrations and vent line are also open access tubes. Open access tubes allow for easier insertion of inspection probes/end effectors into the penetration and greater flexibility in the implementation of a multitude of inspection approaches.

The AP1000 design has an integrated head package permanently attached to the reactor vessel head. This acts to reduce access to the top of the vessel head for inspection as compared to the current fleet of PWRs. However, the integrated head package has doors just above the vessel head that allow inspection access. Vessel head insulation configuration and access ports through this insulation allow for the implementation of visual inspection approaches across the vessel head.

d. Alloy 690 is used for both the base material and weld filler metal in the AP1000. Alloy 600 was used in current Westinghouse PWRs.

e. The operating temperature for the AP1000 reactor vessel head is approximately 560 F, which is between Tcold (537 F) and Thot (610 F). This temperature is in the colder range of current Westinghouse PWR plants that run at various temperatures between Tcold and Thot. The bypass flow to cool the vessel head is provided through spray nozzles similar to current plants. The AP1000 nozzle area has been sized to provide approximately 1.5% of the total flow to the head.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 410.005

Question:

(Section 5.2.5) AP1000 DCD, Section 5.2.5.3 indicates that the N_{13}/F_{18} radioactivity monitor can detect an 0.5 gallon-per-minute (gpm) leak when the plant is above 20 percent power. The detection sensitivity (0.5 gpm) is a function of the primary coolant radioactivity. Section 11.1 of the DCD discusses two source terms for the primary coolant: a conservative design-basis source term and a realistic source term. Please clarify which source term was assumed to determine the detection sensitivity of 0.5 gpm for the N_{13}/F_{18} radioactivity monitor. How can the assumption be verified with respect to the actual operating primary coolant radioactivity to assure the detection sensitivity of 0.5 gpm.

Position C.6 of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," states that the response time of each leakage detection system should be adequate to detect a leak rate of 1 gpm, or its equivalent, in less than one hour. What is the response time for the N_{13}/F_{18} radioactivity monitor? Demonstrate the adequacy of this response time in meeting RG 1.45, Position C.6, and in supporting leak-before-break (LBB) for the AP1000.

Westinghouse Response:

The production of $^{13}N/^{18}F$ in the reactor coolant is a predictable function of core power, and is independent of primary coolant source term. ^{18}F and ^{13}N are produced by the radiolysis of oxygen atoms of the water of the reactor coolant, through the reactions:

$^{16}O(p, \alpha)\ ^{13}N$, and
$^{18}O(p, n)\ ^{18}F$

The leakage detection monitor continuously pumps a stream of the containment atmosphere through its detectors and then returns that stream to the containment. Since the amount of ^{18}F and ^{13}N produced in the coolant is a direct function of core power, and other parameters are fixed (e.g., containment volume), the level of ^{18}F and ^{13}N in the containment atmosphere will be a direct function of the leakage of coolant into the containment.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 410.006

Question:

(Section 5.2.5) TS 3.4.10, "RCS [reactor coolant system] Leakage Detection Instrumentation," specifies that (a) one containment sump level channel and (b) one containment atmosphere radioactivity monitor shall be operable for Modes 1, 2, 3, and 4. However, there are two NOTES allowing these two leakage detection instrumentation systems to not be required during certain conditions. The first note states that the containment atmosphere radioactivity monitor is only required to be Operable in Mode 1 with RTP [rated thermal power] > 20 percent. The second note states that containment sump level measurements cannot be used for leak detection if leakage is prevented from draining to the sump, such as by redirection to the in-containment refueling water storage tank (IRWST) by the containment shell gutter drains.

During Modes 1, 2, 3, and 4, if any one of these two conditions is satisfied and the instrument system is not used, will the RCS inventory balance (determined in accordance with surveillance requirement [SR] 3.4.8.1) or any other actions be required to compensate for the loss of diversity in TS 3.4.10? If not, justify the adequacy of this TS. When both conditions in these two notes are satisfied, and without compensatory actions, there will be no TS requirement for RCS leak detection such that LBB and small break loss-of-coolant accident (LOCA) may not be detected.

Westinghouse Response:

When the conditions in both notes are satisfied, there are compensatory actions required to monitor for RCS leakage, as discussed below.

The containment atmosphere radioactivity monitor is not required to be operable any time plant power is less than 20 percent of rated thermal power, and there are no additional compensatory leakage monitoring actions required when this instrument is not required to be operable.

However, the containment sump level instrument is required to be operable in Modes 1, 2, 3, and 4 to provide RCS leakage detection, whether the containment radioactivity monitor is required or not.

The second note for the sump instrument does NOT eliminate the operability requirements for at least one containment sump level instrument channel in Modes 1, 2, 3, and 4 when the gutter drains are closed.

The second note is intended to inform the operator that although the sump level instrument(s) may be operational, if the drain path for the containment shell gutter to the containment sump is closed (which returns condensate to the IRWST instead of to the containment sump), then the sump level measurement cannot perform its leak detection function. No condensate can return to the containment sump when the drain path is closed. (Condensate is able to drain to the sump as long as both series drain path isolation valves are open.)

In the case with the drain path closed, the containment sump level instruments do NOT meet the TS definition of operable. Therefore, when the drain path is closed, both channels are inoperable (even though both may be operating) and Condition A must be entered. The compensatory action is to perform SR 3.4.8.1 (RCS inventory balance) more frequently, once every 24 hours instead of once every 72 hours.

In addition, at least one containment sump channel must be restored to operable status within 72 hours. This means that both gutter drain path isolation valves must be opened. Once both series isolation valves are open, then condensate will drain to the sump and the available containment sump level instrument is considered to be operable.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 410.010

Question:

(DCD, Tier 2, Chapter 6, Section 6.4, and Chapter 9, Sections 9.4.1 through 9.4.3 and 9.4.6 through 9.4.11) The HVAC systems (VAS, VBS, VCS, VFS, VHS, VRS, VTS, VXS, AND VZS) were designed for a nominal 600 MW(e) plant. The same systems are credited for the 1000 MW(e) plant. Discuss the adequacy of these systems for the AP1000 design.

Westinghouse Response:

The design of HVAC systems is not directly a function of plant power rating. The AP1000 and the AP600 plant designs are similar or identical in many ways, which allow the same HVAC systems to be utilized in both designs. For example, they share the same indoor and outdoor ambient design temperature ranges and utilize essentially the same building structures except for the containment structure, which is taller; and the turbine building, which is larger. Similarly, much of the equipment in the plants is the same with the largest changes again being in the containment and in the turbine building. Where the aforementioned items are the same, the AP1000 and AP600 HVAC systems are the same. Where the items are not the same, Westinghouse has further considered the performance of the AP600 HVAC systems in the AP1000 application. It should be noted that the AP600 HVAC systems are very conservatively sized and include design margins that are in significant excess of those traditionally applied by industry to HVAC designs. In AP1000, some of this excess conservatism is reduced. This allows the equipment to remain identical in most of the HVAC system designs of both plants, while maintaining traditional industry design margins.

In the following sections, Westinghouse provides a brief description and evaluation of the adequacy of the AP1000 HVAC systems as compared to those in AP600.

- Radiologically Controlled Area Ventilation System (VAS)

  The VAS serves the fuel handling area of the auxiliary building, and the radiologically controlled portions of the auxiliary and annex buildings, except for the health physics and hot machine shop areas which are provided with a separate ventilation system (the health physics and hot machine shop HVAC system [VHS].)

  The VAS has two subsystems, they are the auxiliary/annex building subsystem and fuel handling area ventilation subsystem.

  The AP1000 and AP600 ambient design temperature ranges, building designs and equipment served by VAS are the same. As the equipment in these areas are essentially the same, the VAS HVAC design heat load limits are also the same. The AP1000 does have a larger maximum spent fuel pool heat load in this area, which is removed by the spent fuel pit cooling system (SFS) and not by the VAS. The AP1000 hardware designs for the VAS subsystems are therefore designed to be the same as those in AP600.

The AP1000 VAS fuel handling area ventilation subsystem and auxiliary/annex building ventilation subsystem are also capable of maintaining airborne radioactivity in the access areas at safe levels for plant personnel.

Thus, the AP1000 VAS, which is identical to the AP600 VAS, is adequate to perform its intended function.

- Nuclear Island Nonradioactive Ventilation System (VBS)

  The VBS serves the main control room (MCR), technical support center (TSC), Class 1E dc equipment rooms, Class 1E instrumentation and control (I&C) rooms, Class 1E electrical penetration rooms, Class 1E battery rooms, remote shutdown room, reactor coolant pump trip switchgear rooms, adjacent corridors, and the passive containment cooling system (PCS) valve room during normal plant operation.

  The VBS consists of the main control room/technical support center HVAC subsystem, the Class 1E electrical room HVAC subsystem and the passive containment cooling system valve room heating and ventilation subsystem.

  The AP1000 and AP600 ambient design temperature ranges and building designs are the same. The AP1000 and AP600 equipment served by the VBS are very similar. As the equipment in these areas are very similar, it is possible to maintain the AP600 VBS HVAC design heat load limits for AP1000. The AP1000 hardware designs for the VBS subsystems are therefore the same as those in AP600.

  Thus, the AP1000 VBS, which is identical to the AP600 VBS, is adequate to perform its intended function.

- Containment Recirculation Cooling System (VCS)

  The VCS controls building air temperature and humidity to provide a suitable environment for equipment operability during normal operation and shutdown.

  The AP1000 VCS hardware design is the same as AP600's. The AP1000 and AP600 VCS are designed for the same ambient design temperature ranges. The AP1000 containment volume, however, is larger than that of the AP600 and the primary equipment in the containment is different. Consistent with the first paragraph of the response to this RAI, the AP600 VCS design is conservative and incorporates at least 15% design margin over the calculated heat load. A review of the AP1000 design established that the AP1000 VCS heat load is less than 15% greater than that of the AP600, thus allowing application of the AP600 VCS equipment to AP1000 without change.

The AP1000 VCS is adequate to perform its intended function.

- Containment Air Filtration System (VFS)

In the containment area, the VFS conditions and filters outside air supplied to the containment for compatibility with personnel access during maintenance and refueling operations. The system supplies air between 50° and 70°F. The air is distributed and conditioned within the containment by the VCS.

In radiologically controlled areas outside containment, the VFS provides filtration of exhaust air from the fuel handling area, auxiliary, or annex buildings to maintain these areas at a slightly negative pressure with respect to the adjacent areas when the VAS detects high airborne radioactivity or high pressure differential.

Although, the AP1000 containment volume is larger than that of the AP600, the AP1000 VFS hardware design is the same as AP600's. The AP1000 VFS design provides less containment air changes per hour than AP600. The number of air changes is identified in the subsection 9.4.7.2.3. Although the reduction in containment air changes will require that the AP1000 VFS operate longer to achieve the same result, the system is still capable of performing its design functions of:

- Providing intermittent flow of outdoor air to purge the containment atmosphere of airborne radioactivity during normal plant operation, and continuous flow during hot or cold plant shutdown conditions to provide an acceptable airborne radioactivity level prior to personnel access
- Providing intermittent venting of air into and out of the containment to maintain the containment pressure within its design pressure range during normal plant operation
- Directing the exhaust air from the containment atmosphere to the plant vent for monitoring, and provides filtration to limit the release of airborne radioactivity at the site boundary within acceptable levels
- Monitoring gaseous, particulate and iodine concentration levels discharged to the environment through the plant vent

Thus, the AP1000 VFS is adequate to perform its intended function.

- Health Physics and Hot Machine Shop HVAC System (VHS)

The VHS serves the annex building stairwell, S02; the personnel decontamination area, frisking and monitoring facilities, radiation monitor calibration area, and health physics facilities on the 100'-0" elevation of the annex building and the hot machine shop on the 107'-2" elevation of the annex building.

The AP1000 and AP600 ambient design temperature ranges, building designs and equipment served by VHS are the same. The AP1000 VHS hardware design is therefore the same as that in AP600.

Thus, AP1000 VHS is adequate to perform its intended functions of:

- Providing conditioned air to work areas to maintain acceptable temperatures for equipment and personnel working in the areas
- Providing air movement from clean to potentially contaminated areas to minimize the spread of airborne contaminants
- Collecting the vented discharges from potentially contaminated equipment in the area
- Providing for exhaust from welding booths, grinders and other miscellaneous equipment located in the hot machine shop
- Providing for radiation monitoring of exhaust air prior to release to the environment
- Maintaining the access control area and hot machine shop at a slight negative pressure with respect to outdoors and the clean areas of the annex building to prevent unmonitored releases of radioactive contaminants
- Providing humidification to maintain a minimum of 35 percent relative humidity

- Radwaste Building HVAC System (VRS)

  The VRS serves the radwaste building, which includes the clean electrical/mechanical equipment room and the potentially contaminated HVAC equipment room, the packaged waste storage room, the waste accumulation room, and the mobile systems facility.

  The AP1000 and AP600 VRS ambient design temperature ranges, building designs and design HVAC heat load limits are the same. The AP1000 VRS hardware design is therefore the same as that in AP600.

  Thus, AP1000 VRS is adequate to perform its intended functions of:

  - Providing conditioned air to work areas to maintain acceptable temperatures for equipment and personnel working in the areas
  - Providing confidence that air movement is from clean to potentially contaminated areas to minimize the spread of airborne contaminants
  - Collecting the vented discharges from potentially contaminated equipment
  - Providing for radiation monitoring of exhaust air prior to release to the environment
  - Maintaining the radwaste building at a negative pressure with respect to ambient to prevent unmonitored releases from the radwaste building

- Turbine Building Ventilation System (VTS)

The turbine building ventilation system consists of the general area heating and ventilation subsystem, the electrical equipment and personnel work area HVAC subsystem, and the local area heating and ventilation subsystem.

The general area heating and ventilation subsystem of AP1000 is not identical to that of AP600. The designs are conceptually the same, but capacities differ. For example, both designs incorporate roof ventilators, but the AP1000 design incorporates additional roof ventilators to maintain the turbine building at the design temperatures identified in the DCD. Similarly, both designs incorporate hot water unit heaters, but the AP1000 design incorporates additional hot water unit heaters. As the VTS is a nonsafety-related system and performs no defense-in-depth function, the number of these ventilators and unit heaters is not specified in the DCD.

The electrical equipment portion of the VTS electrical equipment and personnel work area HVAC subsystem serves the turbine building switchgear rooms 1 and 2, the electrical equipment room and the variable frequency drive (VFD) power converter room. In AP600, the VFD power converter room contains the feedwater pump VFD power converter. In AP1000, the same room contains the reactor coolant pump RCP VFD power converter. The AP600 feedwater pump VFD operates continuously during normal plant operations while the AP1000 RCP VFD normally operates only during plant startup and cooldown. The RCP VFD power converter is cooled by component cooling water. This different operational requirement, combined with a reduction of excess conservatism allows the electrical equipment portion of the VTS electrical equipment and personnel work area HVAC subsystem to be applied to AP1000 without change.

Thus, the AP1000 VTS is adequate to perform its intended function.

- Annex/Aux Building Nonradioactive Ventilation System (VXS)

The VXS serves the nonradioactive personnel and equipment areas, electrical equipment rooms, clean corridors, the ancillary diesel generator room and demineralized water deoxygenating room in the annex building, and the main steam isolation valve compartments, reactor trip switchgear rooms, and piping and electrical penetration areas in the auxiliary building.

The VXS consists of the following independent subsystems:

- General area HVAC subsystem
- Switchgear room HVAC subsystem
- Equipment room HVAC subsystem
- MSIV compartment HVAC subsystem
- Mechanical equipment areas HVAC subsystem

- Valve/Piping penetration room HVAC subsystem

The hardware designs of the AP1000 VXS subsystems are the same as those in AP600.

The AP1000 and AP600 design temperature ranges, building designs and design HVAC equipment served by the VXS subsystems are similar except that the ancillary diesel generators are larger. The larger ancillary diesel generators will not affect the VXS mechanical equipment areas HVAC subsystem sizing however, because doors are opened to the outdoors when the generators are operated as stated in the last sentence of the last paragraph of DCD subsection 9.4.2.2.1.6, which states "Ventilation and cooling for the room when the ancillary diesel generators operate is provided by means of manually operated dampers and opening doors to allow radiator discharge air to be exhausted direct to outdoors."

Another plant difference that could affect the VXS design is the larger feedwater and steam piping that pass through the MSIV compartment. This area of the plant is served by the MSIV compartment HVAC subsystem. Although AP1000 will have a higher actual heat load in this area of the plant as compared to AP600 due to the larger piping, there is sufficient conservatism in the AP600 heat load design such that no change is required for the AP1000 MSIV compartment HVAC subsystem hardware.

Thus, the AP1000 VXS design, which is identical to the AP600 VXS design, is adequate to perform its intended function.

- Diesel Generator Building Heating and Ventilation System (VZS)

The diesel generator building heating and ventilation system serves the standby diesel generator rooms, electrical equipment service modules, and diesel fuel oil day tank vaults in the diesel generator building and the two diesel oil transfer modules located in the yard near the fuel oil storage tanks.

The AP1000 VZS hardware design is the same as that in AP600. The AP1000 equipment served by VZS are identical in AP1000 and AP600. The AP1000 and AP600 design temperature ranges, building design, and design HVAC heat load limits are therefore also the same.

Thus, the AP1000 VZS design, which is identical to the AP600 VZS design is adequate to perform its intended function.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 410.016

Question:

The second paragraph of Section 9.1.3.4.3 "Abnormal Conditions," states that, in the unlikely event of an extended loss of normal spent fuel pool (SFP) cooling, a water level is maintained above the spent fuel assemblies for at least 7 days and that the amount of makeup required to provide the 7 day capability depends on the decay heat level of the fuel in the SFP and is provided "when the calculated decay heat level in the spent fuel pool is less than 2.3 MWt [megawatts thermal], no make up is needed to achieve spent fuel pool cooling for at least 7 days."

Please describe the mechanism that the operator will use to determine the thermal power level in the SFP. For example, if the calculated decay heat level in the SFP is 2.4 MWt, how would the operator know? Are means provided for the operator to read this value in the control room? Will the operator be required to perform the calculation?

Westinghouse Response:

Calculated power levels in the spent fuel pool, as provided in the AP1000 DCD, are used to determine the bases for availability requirements for spent fuel pool makeup water sources (LCO 3.7.9 in DCD Chapter 16, Technical Specifications). These power levels in conjunction with minimum water level/volume in the makeup tanks are used in calculations to show that sufficient on-site water is available to maintain the spent fuel pool water level above the spent fuel assemblies for at least 7 days. Calculation of spent fuel pool heat load is only required when taking makeup water sources out of service. The heat load in the spent fuel pool is calculated with the output from the temperature and flow instrumentation in the spent fuel pool cooling system. This heat load is calculated by the plant control system and is available to the operator in the control room.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 410.018

Question:

Section 9.1.3.4.3.4, "Station Blackout," states that water vapor that evaporates from the surface of the SFP is vented to the outside environment through an engineered relief panel. This vent path maintains the fuel handling area at near atmospheric pressure conditions. Activity releases due to pool boiling are analyzed. Please discuss the method of analyzing the releases and provide details describing exactly how the activity is captured for analysis.

Westinghouse Response:

In DCD section 9.1.3.4.3.4 the sentence "Activity releases due to pool boiling are analyzed" is referring to an analytical calculation to predict the doses resulting from boiling of the spent fuel pool.

See the response to RAI 470.007 for the inputs and assumptions that are used in this calculation of expected doses resulting from boiling of the spent fuel pool. For each design basis accident in which doses are calculated and a loss of off-site power is assumed, there is a DCD section in which the doses resulting from both the accident and from boiling of the spent fuel pool are given. For example, see DCD section 15.1.5.4.6 for the calculated doses resulting from the steam line break and spent fuel pool boiling.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 420.019

Question:

420.19 (DCD Figures 7.1-3, 7.1-5, 7.1-6, 7.1-8, and 7.1-9)

DCD Section 7.1 states that DCD Chapter 7 for the AP1000 has been written to permit the use of either the protection system hardware described in the AP600 DCD or the Common Q described in Reference 8. Justify the deletion of Figures 7.1-3, 7.1-5, 7.1-6, 7.1-8, and 7.1-9 in AP1000 DCD. These figures are related to the integrated protection cabinets, the ESF actuation cabinets, the protection logic communication cabinets, the qualified data processor, and the protection logic cabinet architecture. Since Reference 8, "CENPD-396-P, Rev.1 - Common Qualified Platform," is a proprietary document, the equivalent information should be provided for both the AP600 design and the Common Q design in the AP1000 DCD.

Westinghouse Response:

Figures 7.1-3A & B, 7.1-5, 7.1-6, 7.1-8A & B, 7.1-9A & B and 7.1-11 will be added to the DCD as shown below.

Design Control Document (DCD) Revision:

7.1 Introduction

The instrumentation and control systems presented in this chapter provide protection against unsafe reactor operation during steady-state and transient power operations. They initiate selected protective functions to mitigate the consequences of design basis events. This chapter relates the functional performance requirements, design bases, system descriptions, and safety evaluations for those systems. The safety evaluations show that the systems can be designed and built to conform to the applicable criteria, codes, and standards concerned with the safe generation of nuclear power.

Because of the rapid changes that are taking place in the digital computer and graphic display technologies employed in a modern human system interface, design certification of the AP1000 focuses upon the process used to design and implement instrumentation and control systems for the AP1000, rather than on the specific implementation. The design specifics provided here are included as an example for illustration.

DCD Chapter 7 for the AP1000 has been written to permit the use of either the Eagle protection system hardware described in the AP600 DCD or the Common Qualified Platform (Common Q) described in References 8 and 13 and accepted in References 11 and 14. The I&C functional requirements of the AP600, which has received Design Certification, have been retained to the maximum extent compatible with the Common Q hardware and software and the ~~AP600 DCD~~Eagle hardware and software.

The terminology used for Chapter 7 is intended to be independent of any product, but when this is not possible, Common Q terminology is used.

7.1.2 Protection and Safety Monitoring System

The protection and safety monitoring system is illustrated in Figure 7.1-2. The functions of the protection and safety monitoring system are implemented in separate processor-based subsystems. Each subsystem is located on an independent computer bus to prevent propagation of failures and to enhance availability. In most cases, each subsystem is implemented in a separate card chassis. Subsystem independence is maintained through the use of the following:

- Separate dc power sources for redundant subsystems with output protection to prevent interaction between redundant subsystems upon failure of a subsystem.
- Separate input or output circuitry to maintain independence at the subsystem interfaces.
- Deadman signals: A device, circuit, or function that forces a predefined operating condition upon the cessation of a normally dynamic input parameter to improve the reliability of hard-wired data that crosses the subsystem interface.
- Optical coupling or resistor buffering between two subsystems or between a subsystem and an input/output (I/O) module.

WCAP-13382 (Reference 2) provides a description of the Eagle hardware elements which comprise the protection and safety monitoring system configuration for the AP600. WCAP-14080 (Reference 4) provides a description of the Eagle software architecture and operation for the AP600. The Eagle hardware and software described for the AP600 may be used for the AP1000; alternatively, the AP1000 protection and safety monitoring system may be based on the Common Qualified Platform described in References 8 and 13 and accepted in References 11 and 14.

7.1.2.1 Plant Protection Subsystems

The plant protection subsystems contain the necessary equipment to perform the following functions:

- Permit acquisition and analysis of the sensor inputs required for reactor trip and ESF actuation calculations.
- Perform computation or logic operation on variables based on these inputs.
- Provide trip signals to the reactor trip switchgear and ESF actuation data to the ESF coincidence logic, as required.
- Permit manual trip or bypass of each individual automatic reactor trip function and permit manual actuation or bypass of each individual automatic ESF actuation function.
- Provide data to external systems.
- Provide redundancy for the reactor trips and ESF actuations.
- Provide isolation circuitry for control functions requiring input from sensors which are also required for protection functions.

Figure 7.1-3A illustrates the plant protection subsystems for the Eagle I&C architecture. Figure 7.1-3B illustrates the plant protection subsystems and the engineered safety features coincidence logic for the Common Q architecture.

7.1.2.2 Engineered Safety Features Coincidence Logic

The ESF logic functions are also performed in two subsystems per division for more reliable accident mitigation. The primary functions of the ESF coincidence logic are to process inputs, calculate actuations, combine the automatic actuation with the manual actuation and manual bypass data, and transmit the data to the ESF actuation subsystems. To perform the ESF logic calculations, the subsystems require data from the plant protection subsystems, and also use manual inputs from the main control room and the remote shutdown workstation.

The ESF coincidence logic performs the following functions:

- Receives bistable data supplied by the four divisions of the plant protection subsystems and performs two-out-of-four voting on this data.
- Implements system-level logic and transmits the output to the ESF actuation subsystems for ESF component actuation.

- Processes manual system-level actuation commands received from the main control room and remote shutdown workstation.

Figure 7.1-5 illustrates the engineered safety features coincidence logic for the Eagle I&C architecture. Figure 7.1-3B illustrates the plant protection subsystems and the engineered safety features coincidence logic for the Common Q architecture.

7.1.2.3 Engineered Safety Features Actuation Subsystems

The ESF actuation subsystems provide a distributed interface between the plant operator and the nonmodulating safety-related plant components. Nonmodulating control relates to the opening or closing of solenoid valves and solenoid pilot valves, and the opening or closing of motor-operated valves and dampers. The ESF actuation subsystems implement criteria established by the fluid systems designers for permissive and interlock logic applied to the component actuations. It also provides the plant operator with information on the equipment status, such as indication of component position (full closed, full open, valve moving), component control modes (manual, automatic, local, remote) or abnormal operating condition (power not available, failure detected).

The ESF coincidence logic performs the appropriate voting operation on the bistable signals and generates the system-level ESF logic commands including the system-level manual commands. These system-level actuations are then sent to the ESF actuation subsystems. The ESF actuation subsystems decode the system commands and actuate the final equipment through the interlocking logic specific to each component. Component-level actuation signals are sent from the main control room to the ESF actuation subsystems over redundant data highways. Component status is transmitted from the ESF actuation subsystems to the main control room over the same redundant data highways. Those components used for safe shutdown can also be controlled from the remote shutdown workstation.

Figure 7.1-6 shows this redundant data highway for a single safety division for the Eagle I&C architecture. Figure 7.1-3B includes the communication between the engineered safety features coincidence logic and the engineered safety features actuation logic for the Common Q architecture. Figure 7.1-9A illustrates the engineered safety features actuation logic for the Eagle I&C architecture. Figure 7.1-9B illustrates the engineered safety features actuation logic for the Common Q architecture.

7.1.2.5 Qualified Data Processing Subsystems

The qualified data processing (QDPS) subsystems are a redundant configuration consisting of sensors, QDPS hardware, and qualified displays.

The qualified data processing subsystems perform the following functions:

- Provide safety-related data processing and display
- Provide the operator with sufficient operational data to safely shut the plant down in the event of a failure of the other display systems
- Provide qualified and nonqualified data to the real-time data network for use by other systems in the plant
- Process data for main control room display, and to meet Regulatory Guide 1.97 requirements
- Provide data to the main control room, the remote shutdown workstation, the plant computer, other nonsafety-related devices, and nonqualified emergency response facilities in conformance with NUREG-0696

The QDPS hardware consists of safety-related modular data gathering units. The QDPS receives inputs from process sensors and safety-related digital systems. The QDPS consolidates the input data, performs conversions to process units, and formats the data for data link transmission.

Figure 7.1-8A illustrates the qualified data processing subsystem for the Eagle I&C architecture. Figure 7.1-8B illustrates the qualified data processing subsystem for the Common Q architecture.

7.1.2.11 Test Subsystem

The test subsystem provides a means of testing the operation of the protection and safety monitoring system and verifying that the plant protection system setpoints are within the system requirements. Each redundant subsystem is tested individually.

Testing from the sensor inputs of the protection and safety monitoring system through to the actuated equipment is accomplished through a series of overlapping sequential tests with the majority of the tests capable of being performed with the plant at full power. Where testing final equipment at power would upset plant operation or damage equipment, provisions are made to test the equipment at reduced power or when the reactor is shut down.

Each division of the protection and safety monitoring system is furnished with a test subsystem. The test subsystem provides for verification of the accuracy of setpoints and other constants, and verification that proper signals appear at other locations in the system.

Verification of the signal processing algorithms is made by exercising the test signal sources (either by hardware or software signal injection) and observing the results up to, and including, the attainment of a channel partial trip or actuation signal at the power interface.

When required for the test, the tester automatically places the voting logic associated with the channel function under test in bypass.

The overlapping test sequence continues by inputting digital test signals at the output side of the threshold functions, in combinations necessary to verify the voting logic. Some of the input combinations to the coincidence logic cause outputs such as reactor trips and ESF initiation. The reactor trip circuit breakers are arranged in a two-out-of-four logic configuration, such that the tripping of the two circuit breakers associated with one division does not cause a reactor trip. This circuit breaker arrangement is illustrated in Figure 7.1-7. To reduce wear on the breakers through excessive tripping, and to avoid a potential plant trip resulting from a single failure while testing is in progress, the test sequence is designed so that actual opening of the trip breakers is only required when the breaker itself is being tested.

The test subsystem does not test the ESF actuators. This portion of the test may be accomplished by using component-level actuation signals. For those final devices that can be operated at power, without upsetting the plant or damaging equipment, the test is performed by actuating the manual actuation control which causes the device to operate. Position switches on the device itself send a signal back to the ESF actuation subsystem, where it is transmitted to the main control room for display purposes. The display verifies that the manual command is successfully completed, thus verifying operability of the final device. For those devices which cannot be tested at power without damage or upsetting the plant, continuity of the wiring up to the actuation device is verified. Operability of the final equipment is demonstrated at reduced power or at shutdown, depending on the equipment.

In addition to the testing function, the tester subsystem monitors the failure and diagnostic information from the subsystems during normal operation, thus enhancing system maintenance of the protection system.

The test subsystem provides the operator interface used for testing and maintenance.

Figure 7.1-5 includes the test subsystem for the Eagle I&C architecture. Figure 7.1-11 illustrates the test subsystem for the Common Q architecture.

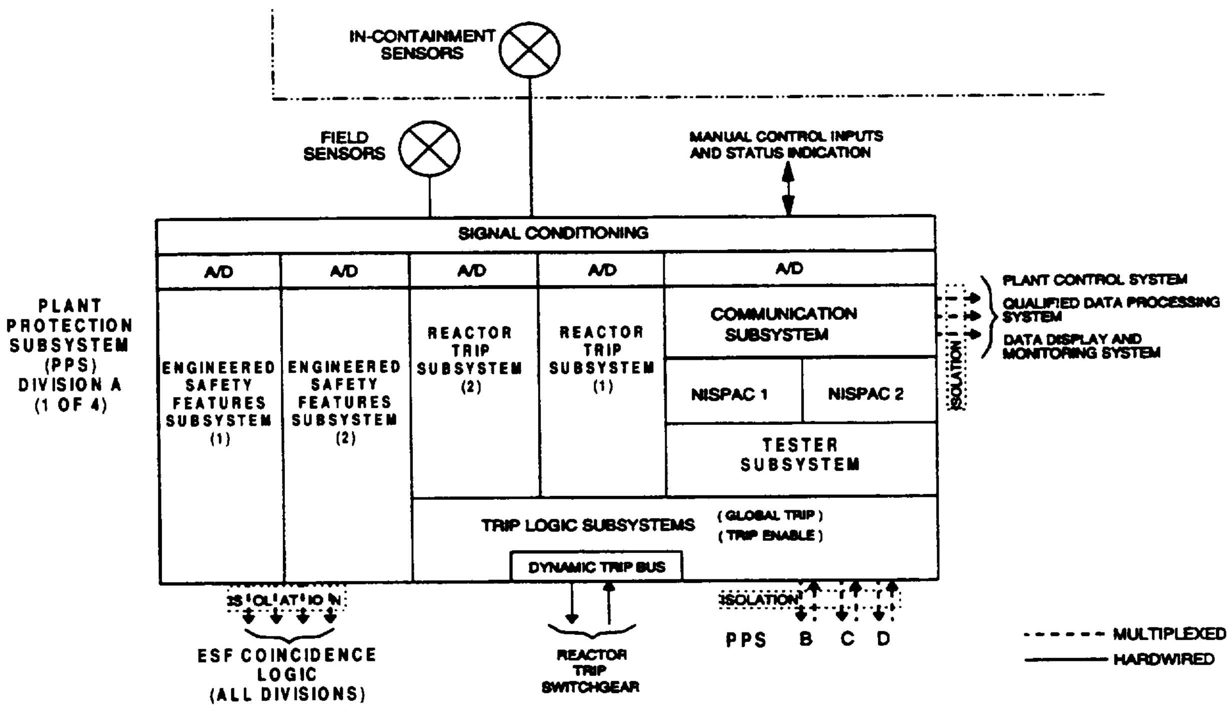


Figure 7.1-3A

Plant Protection Subsystem (Eagle Platform)

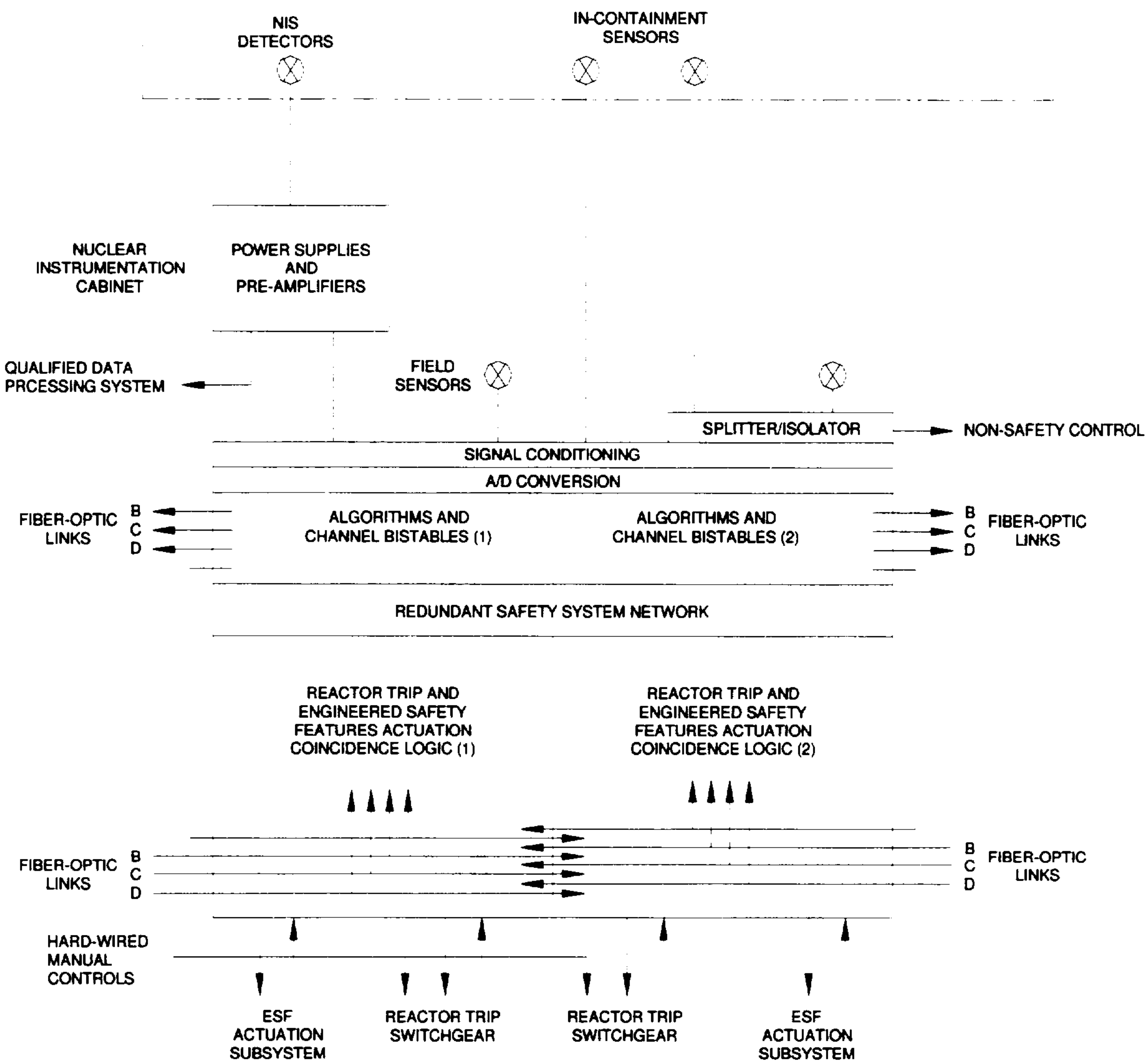


Figure 7.1-3B

Plant Protection Subsystem and Engineered Safety Features Coincidence Logic (Common Q Platform)

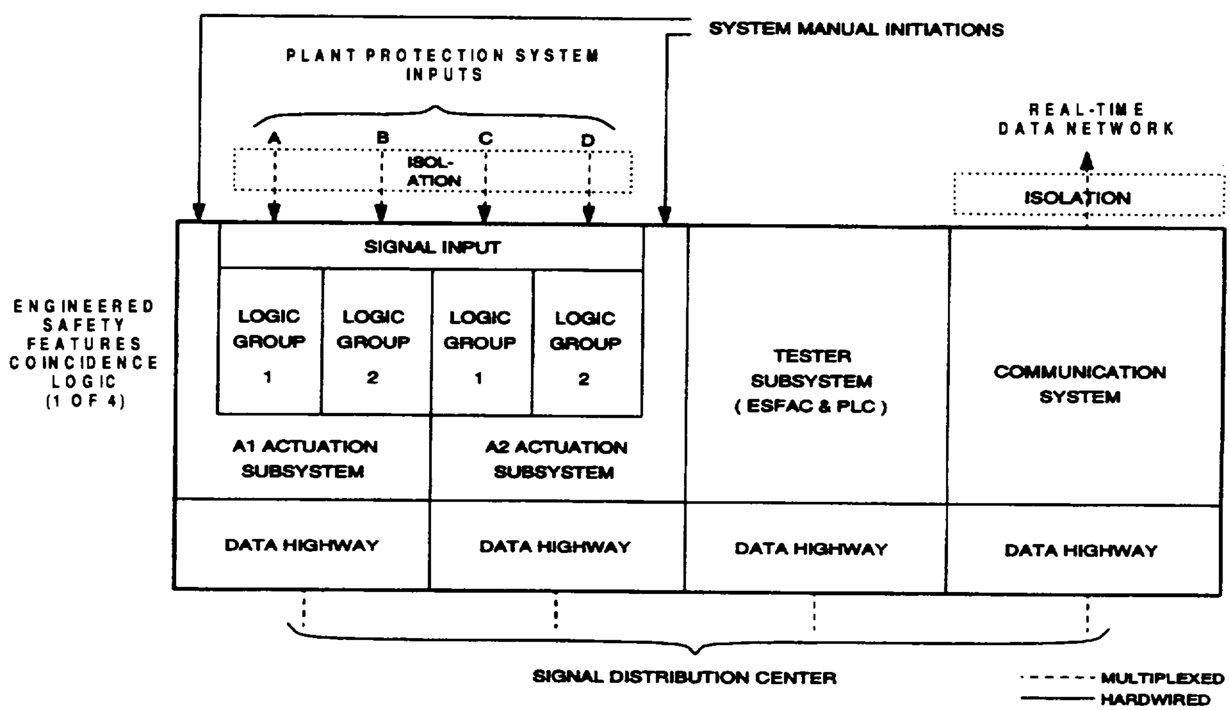


Figure 7.1-5

Engineered Safety Features Coincidence Logic (Eagle Platform)

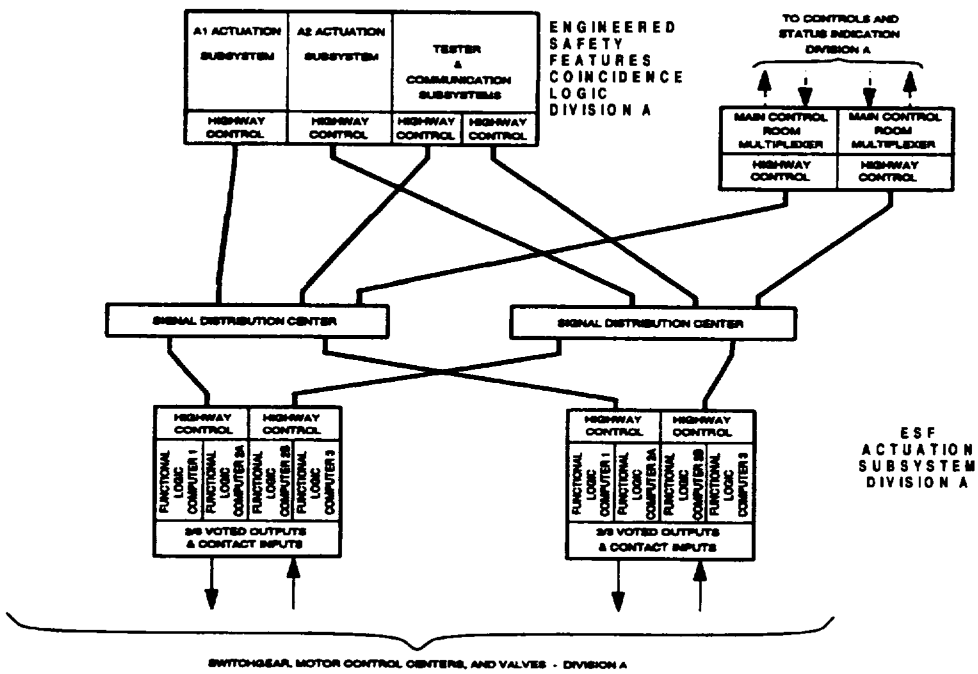


Figure 7.1-6

Protection Logic Communication Diagram (Eagle Platform)

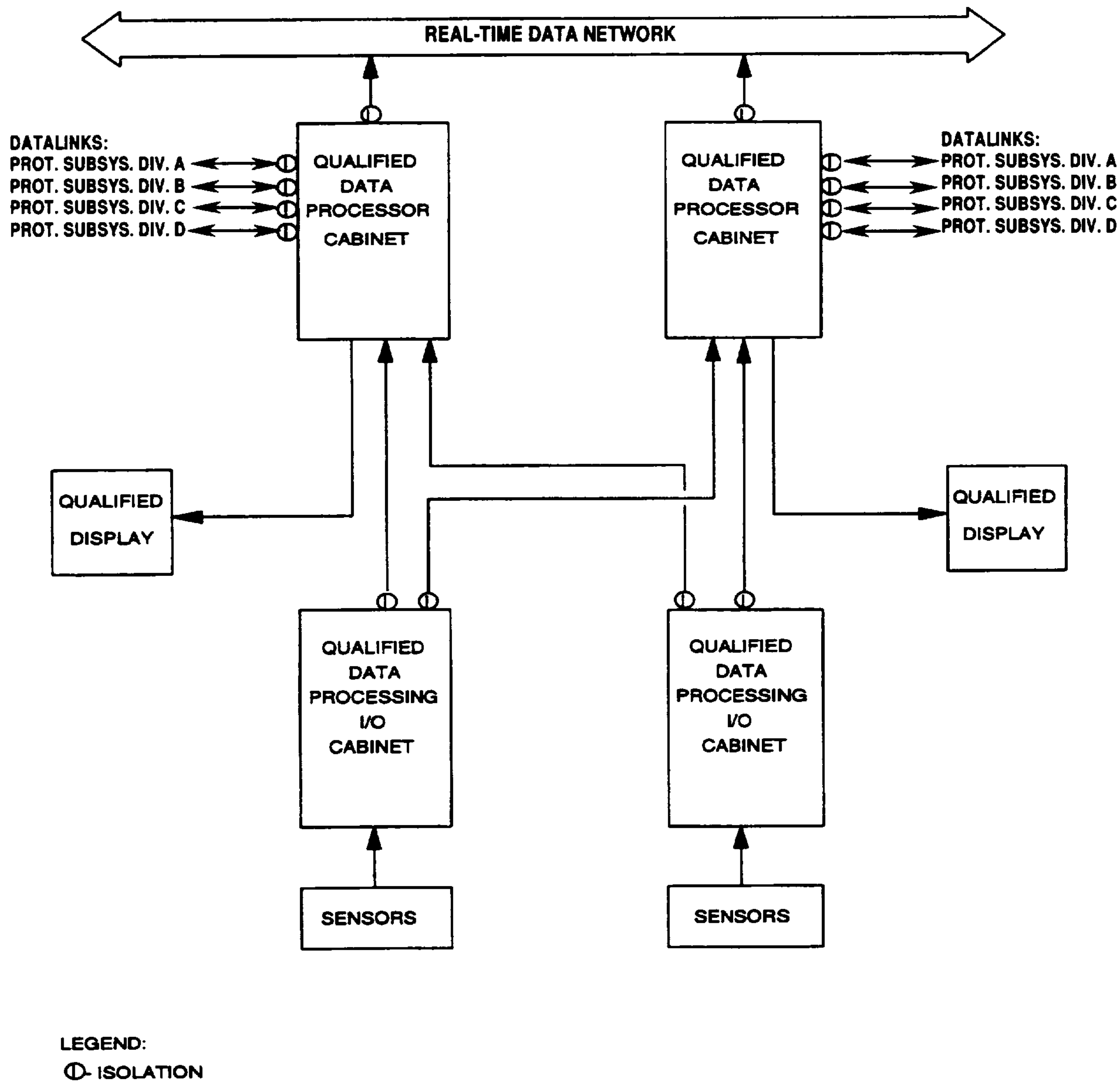


Figure 7.1-8A

Qualified Data Processing Subsystem (Eagle Platform – Channels B&C only)

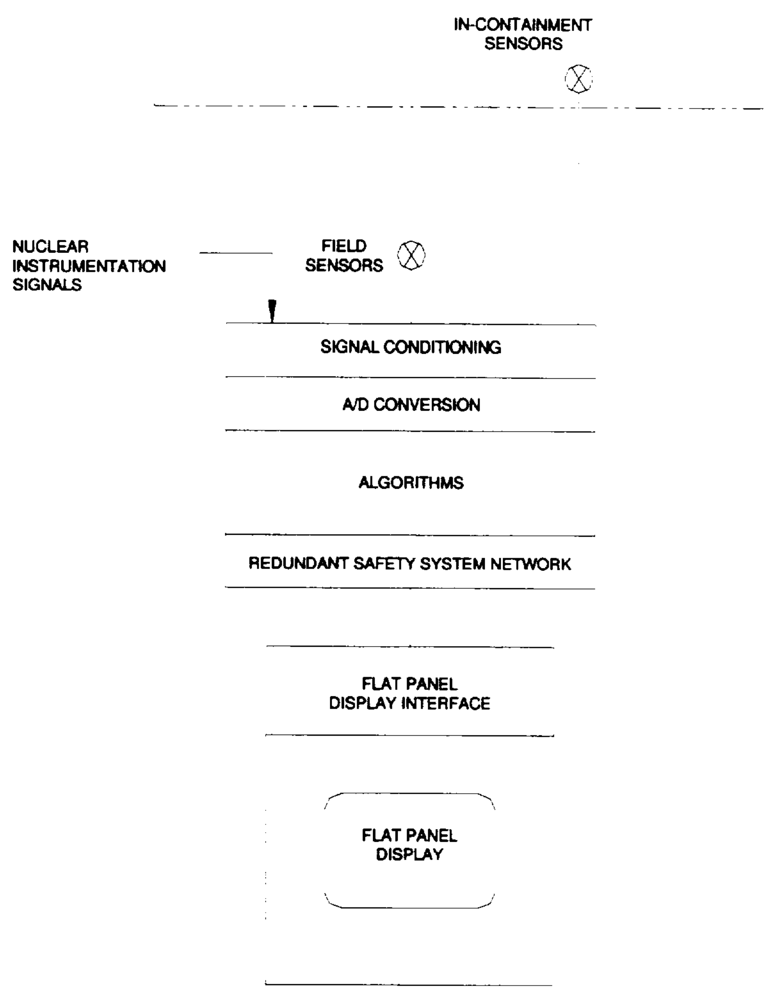


Figure 7.1-8B

Qualified Data Processing Subsystem (Common Q Platform - Channels B&C only)

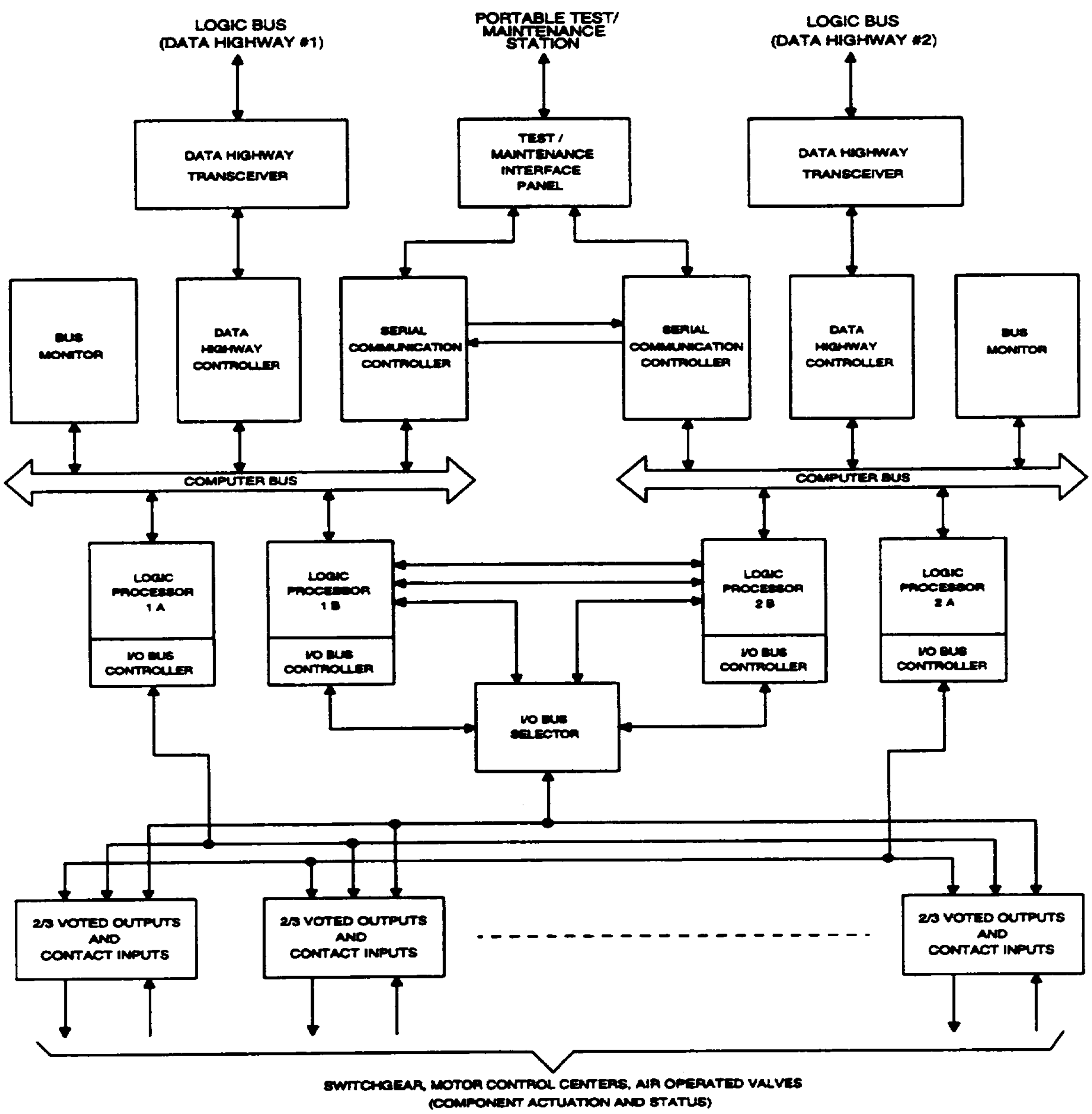


Figure 7.1-9A

Engineered Safety Features Actuation Subsystem (Eagle Platform)

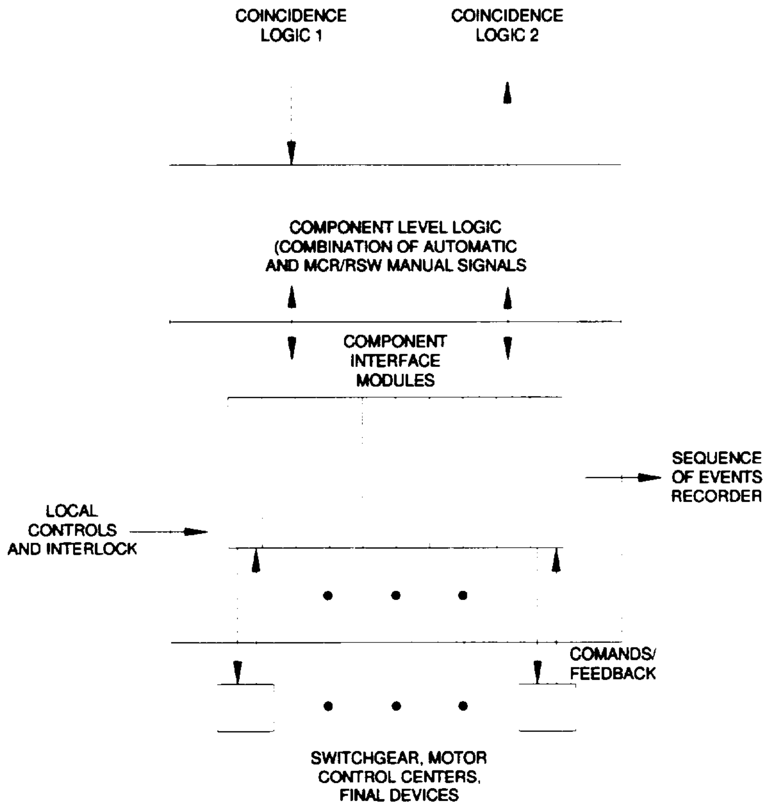


Figure 7.1-9B

Engineered Safety Features Actuation Subsystem (Common Q Platform)

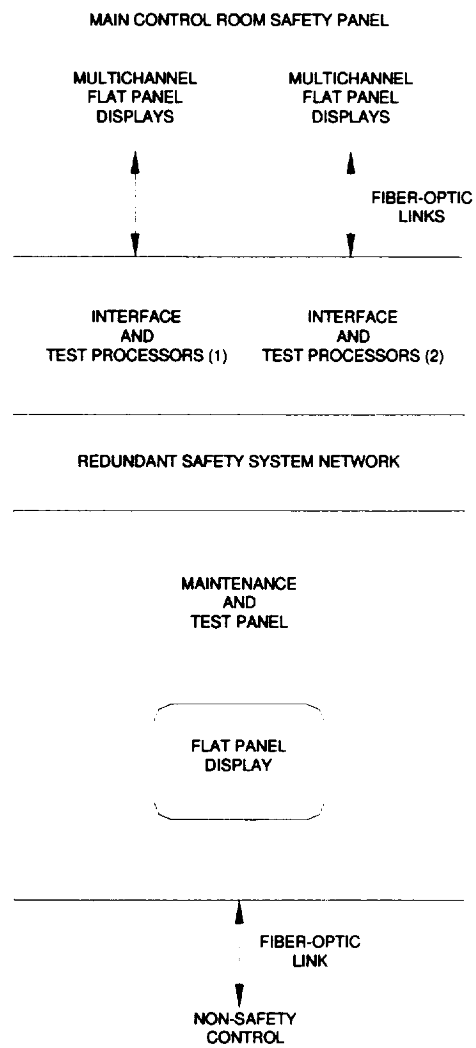


Figure 7.1-11

Maintenance and Test Subsystem (Common Q Platform)

PRA Revision:

None

RAI Number 420.028

Question:

420.28 (DCD 7.1.7 item 11)

In the NRC's safety evaluation report (SER) on Common Q design (Reference 11 on DCD 7.1.7), the staff identified the following plant-specific action items for licensee to follow when the Common Q design is implemented in a specific plant. Please address each of these items applicable to the AP1000 Design Certification. If the action item can not be implemented during the design certification stage, then the action should be included in the Tier 1 Material (ITAAC).

1. Each licensee implementing a specific application based upon the Common Q platform must assess the suitability of the S600 I/O modules to be used in the design against its plant-specific input/output requirements. (See Common Q SER Section 4.1.1.1.2)

2. A hardware user interface that replicates existing plant capabilities for an application may be chosen by a licensee as an alternative to the Flat-Panel Display System (FPDS). The review of the implementation of such a hardware user interface would be a plant-specific item. (See SER Section 4.1.2)

3. If a licensee installs a Common Q application that encompasses the implementation of FPDS, the licensee must verify that the FPDS is limited to performing display and maintenance functions only, and is not to be used such that is required to be operational when the Common Q system is called upon to initiate automatic safety functions. The use of the FPDS must be treated in the plant-specific FMEAs. (See SER Section 4.2.1.2)

4. Each licensee implementing a Common Q application must verify that its plant environment data (i.e., temperature, humidity, seismic, and electromagnetic compatibility) for the location(s) in which the Common Q equipment is to be installed are enveloped by the environment considered for the Common Q qualification testing, and that the specific equipment configuration to be installed is similar to that of the Common Q equipment used for the tests. (See SER sections 4.2.2.1.1, 4.2.2.1.2, and 4.2.2.1.3)

CENP configured the Common Q test specimen for seismic testing using dummy modules to fill all the used rack slots. As part of the verification of its plant-specific equipment configuration the licensee must check that it does not have any unfilled rack slots. (See SER Section 4.2.2.1.2)

5. On the basis of its review of the CENP's software development process for application software, the staff concludes that the Software Program Manual (SPM) specifies plans that will provide a quality software life cycle process, and that these plans commit to documentation of life cycle activity that will permit the staff or others to evaluate the quality of the design features upon which the safety determination will be based. The staff will review the implementation of the life cycle process and the software life cycle process design output for specific application on a plant-specific basis. (See SER 4,3,2)

6. When implementing a Common Q safety system (i.e.,PAMS, CPCS, or DPPS), the licensee must review CENP's timing analysis and validation tests for that Common Q system in order to verify that it satisfies its plant-specific requirements for accuracy and response time presented in the accident analysis in Chapter 15 of the safety analysis report. (See SER Sections 4.1.1.4, 4.4.1.3, 4.4.2.3, and 4.4.3.3)

7. The Operator's Module (OM) and the Maintenance Test Panel (MTP) provide the human machine interface for the Common Q platform. Both the OM and the MTP will include display and diagnostic capabilities unavailable in the existing analog safety systems. The Common Q design provides means for access control to software and hardware such as key switch control, control to software media, and door key locks. The human factors considerations for specific applications of the Common Q platform will be evaluated on a plant-specific basis. (See SER Sections 4.4.1.3, 4.4.2.3, 4.4.3.3, and 4.4.4.3.6)

8. If the licensee installs a Common Q PAMS, CPCS, or DPPS, the licensee must verify on a plant-specific basis that the new system provides the same functionality as the system that is replaced, and meets the functionality requirement applicable to those systems. (See SER Sections 4.4.1.3, 4.4.2.3, and 4.4.3.3)

9. Modifications to plant procedures and/or technical specifications due to the installation of a Common Q safety system will be reviewed by the staff on a plant-specific basis. Each licensee installing a Common Q safety system shall submit its plant-specific request for license amendment with attendant justification. (See SER Sections 4.4.1.3, 4.4.2.3, 4.4.3.3, and 5.0)

10. A licensee implementing any Common Q application (i.e., PAMS, CPCS, or DPPS) must prepare its plant-specific model for the design to be implemented and perform the PMEA for that application. (See SER Sections 4.4.1.3, 4.4.2.3, 4.4.3.3, and 5.0)

11. If a licensee installs a Common Q PAMS, CPCS, DPPS or integrated solution, the licensee shall demonstrated that the plant-specific Common Q application complies with the criteria for defense against common-mode failure in digital instrumentation and control systems and meet the requirements of HICB BTP-19 in the NRC Standard Review Plan Chapter 7. (See SER Sections 4.1.6, 4.4.2.3, 4.4.3.3, 4.4.4.3.3, and 5,0)

12. A licensee implementing a Common Q DPPS shall define a formal methodology for overall response time testing. (See SER Section 4.4.3.3)

13. The analysis of the capacity of the shared resources to accommodate the load increase due to sharing. (See SER Section 4.4.4.3.1)

14. The licensee must ascertain that the implementation of the Common Q does not render invalid any of the previously accomplished TMI action items (See SER Section 5.0)

Westinghouse Response:

On August 11, 2000; the NRC issued a safety evaluation report regarding Topical Report CENPD-396-P, Rev. 01, Common Qualified Platform, Appendices 1,2,3,4, Rev. 01 and CE-CES-195, Rev. 01, Software Program Manual for Common Q Systems. Section 6 of the safety evaluation report identified 14 plant specific action items (PSAIs). The applicability of each PSAI to the AP1000 Design Certification is discussed below.

PSAI 6.1
Each licensee implementing a specific application based upon the Common Q platform must assess the suitability of the S600 I/O modules to be used in the design against its plant-specific input/output requirements.

Discussion: The suitability of all new components is assessed to meet applicable requirements in accordance with the Quality Assurance Program. Performance requirements for these components are assured, for example, by specifying them in purchase contracts, observing vendor testing and analysis, reviewing vendor documentation, performing design reviews by the engineering department, and by performing validation tests after installation. The Quality Assurance Program is described in DCD Chapter 17.

The Input/Output subsystem has been designed to fully meet the functional requirements for Common Q safety systems. The Input/Output subsystem will be deemed capable of performing its design function by successful completion of testing, culminating in a Factory Acceptance Test (FAT) to be performed by the vendor. Acceptance criteria will be based on the system requirements specification.

PSAI 6.2
A hardware user interface that replicates existing plant capabilities for an application may be chosen by a licensee as an alternative to the FPDS. The review of the implementation of such a hardware user interface would be a plant-specific action item.

Discussion: AP1000 safety systems will utilize the Flat Panel Display System (FPDS) as developed by Westinghouse for Common Q safety systems. An alternative hardware interface will not be used. Therefore, this action item is not applicable.

PSAI 6.3
If a licensee installs a Common Q application that encompasses the implementation of FPDS, the licensee must verify that the FPDS is limited to performing display and maintenance functions only, and it is not to be used such that it is required to be operational when the Common Q system is called upon to initiate automatic safety functions. The use of the FPDS must be treated in the plant specific FMEAs.

Discussion: The FPDS' to be implemented in AP1000 I&C safety system are not required to be operational when the safety system is called upon to initiate automatic safety functions. The plant-specific FMEA will address the loss of the FPDS (see item 6.10 below). Additionally, the NRC has stated that this action item has been resolved and is considered closed. Therefore, no further evaluation is required.

PSAI 6.4
Each licensee implementing a Common Q application must verify that its plant environmental data (i.e., temperature, humidity, seismic, and electromagnetic compatibility) for the location(s) in which the Common Q equipment is to be installed are enveloped by the environment considered for the Common Q qualification testing, and that the specific equipment configuration to be installed is similar to that of the Common Q equipment used for the tests.

Westinghouse configured the Common Q test specimen for seismic testing using dummy modules to fill all the used rack slots. As part of the verification of its plant-specific equipment configuration the licensee must check that it does not have any unfilled rack slots.

Discussion: Common Q safety equipment will be located in the Auxiliary Building in a mild (non-harsh) environment. Therefore, age related degradation is expected to be insignificant for temperature and humidity.

The AP1000 temperature and humidity conditions for qualification of protection and safety monitoring system equipment are presented in DCD Appendix 3D. Temperature and humidity qualification of the protection and safety monitoring system equipment is covered by DCD Tier 1 (ITAAC) 2.5.2, item 4.

The protection and safety monitoring system seismic Category I equipment will be tested or analyzed to confirm that it can withstand seismic design basis loads without loss of safety function. The seismic qualification of the protection and safety monitoring system seismic Category I equipment is covered by DCD Tier 1 (ITAAC) 2.5.2, item 2.

The protection and safety monitoring system equipment will be tested or analyzed to confirm that it has electrical surge withstand capability (SWC), and can withstand the electromagnetic interference (EMI), radio frequency interference (RFI), and electrostatic discharge (ESD) conditions that would exist before during and following a design basis accident without loss of safety function for the time required to perform the safety function (DCD Tier 1 (ITAAC) 2.5.2, item 3).

PSAI 6.5
On the basis of its review of the Westinghouse software development process for application software, the staff concludes that the SPM specifies plans that will provide a quality software life cycle process, and that these plans commit to documentation of life cycle activities that will permit the staff or others to evaluate the quality of design features upon which the safety determination will be based. The staff will review the implementation of the life cycle process and the software life cycle process design outputs for specific applications on a plant specific basis.

Discussion: In accordance with the Quality Assurance Program, administrative control procedures are used to establish software quality assurance and configuration management for process computer software, firmware and associated software development, computer systems, and associated documentation. They ensure that the integrity of a process software product is known and preserved throughout its life cycle (from development to retirement). These controls also apply to the development tools and systems used to develop and test process software. The Quality Assurance Program is described in DCD Chapter 17. The software life cycle process is covered by DCD Tier 1 (ITAAC) 2.5.2, item 11.

PSAI 6.6
When implementing a Common Q safety system (i.e. PAMS, CPCS, or DPPS), the licensee must review Westinghouse's timing analysis and validation tests for that Common Q system in order to verify that it satisfies its plant specific requirements for accuracy and response time presented in the accident analysis in Chapter 15 of the safety analysis report.

Discussion: The accuracy and response time of the AP1000 safety systems will be commensurate with the Chapter 15 Safety Analysis. The Combined License applicant is responsible for the setpoint analysis (DCD Section 7.1.6). The setpoint analysis (including accuracy and time response) is covered by DCD Tier 1 (ITAAC) 2.5.2, item 10.

PSAI 6.7
The OM and the MTP provide the human machine interface for the Common Q platform. Both the OM and MTP will include display and diagnostic capabilities unavailable in the existing analog safety systems. The Common Q design provides means for access control to software and hardware such as key switch control, control to software media, and door key locks. The human factors considerations for specific applications of the Common Q platform will be evaluated on a plant-specific basis.

Discussion: This is a human factors issue. The human factors engineering program is described in DCD Chapter 18. The human factors program is the responsibility of the Combined License applicant and is covered by DCD Tier 1 (ITAAC) 3.2.

PSAI 6.8
If the licensee installs a Common Q PAMS, CPCS or DPPS, the licensee must verify on a plant-specific basis that the new system provides the same functionality as the system that is being replaced, and meets the functionality requirement applicable to those systems.

Discussion: The AP1000 is a new plant safety system installation; therefore, this action item is not applicable.

PSAI 6.9
Modifications to plant procedures and/or TS due to the installation of a Common Q safety system will be reviewed by the staff on a plant-specific basis. Each licensee installing a Common Q safety system shall submit its plant-specific request for license amendment with attendant justification.

Discussion: The AP1000 is a new plant safety system installation; therefore, this action item is not applicable. The Combined License applicant is responsible for plant procedures (see DCD Section 13.5).

PSAI 6.10
A licensee implementing any Common Q applications (i.e., PAMS, CPCS, or DPPS) must prepare its plant specific model for the design to be implemented and perform the FMEA for that application.

Discussion: An FMEA will be performed for each AP1000 safety system. The FMEAs will confirm that no single failure of a safety system component will defeat more than one of the four protective channels, assuring proper protective action at the system level. DCD Table 1.8-2 and Section 7.2.3 will be modified as shown below to add this item.

PSAI 6.11
If a licensee installs Common Q PAMS, CPCS, DPPS or Integrated Solution, the licensee shall demonstrate that the plant-specific Common Q application complies with the criteria for defense against common-mode failure in digital instrumentation and control system and meets the requirements of HICB BTP-19.

Discussion: The response to RAI Number 420.014 provides a discussion on AP1000's compliance with BTP HICB-19.

PSAI 6.12
A licensee implementing a Common Q DPPS shall define a formal methodology for overall response time testing.

Discussion: A formal methodology will be defined for response time testing of AP1000 safety systems. This methodology is covered by DCD Tier 1 (ITAAC) 2.5.2, item 10.

PSAI 6.13
The analysis of the capacity of the shared resources to accommodate the load increase due to sharing.

Discussion: The shared resource issue relates to multiple Common Q based systems using the same resources such as the AF100 bus or an Operator Module. An analysis will be performed to ensure that the capacity of shared resources for AP1000 safety systems is commensurate with anticipated loads. This issue will be addressed as part of the design process that is covered by DCD Tier 1 (ITAAC) 2.5.2, item 11.

PSAI 6.14
The licensee must ascertain that the implementation of the Common Q does not render invalid any of the previously accomplished TMI action items.

Discussion:
Section 7.1.4.1 of the DCD describes conformance of the AP1000 safety system Instrumentation to applicable safety criteria. The safety-related system instrumentation described in subsection 7.1.1 is designed and built to conform to the applicable criteria, codes, and standards concerned with the safe generation of nuclear power. Applicable General Design Criteria are listed in Section 3.1, NRC Regulatory Guides in subsection 1.9.1, and Branch Technical Positions in subsection 1.9.2. Industry Standards are cited as references.

Design Control Document (DCD) Revision:

Table 1.8-2 (Sheet 3 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
7.1-1	Setpoint Calculations for Protective Functions	7.1.6
7.2-1	FMEA for protection system	7.2.3

7.2.3 Combined License Information

Combined License applicants referencing the A1000 certified design will provide an FMEA for the protection and safety monitoring system.~~This section has no requirement for information to be provided in support of the Combined License application.~~

PRA Revision:

None

RAI Number 420.029

Question:

420.29 (DCD 7.1.7, item 11)

In the NRC's safety evaluation report on Common Q design (Reference 11 in DCD 7.1.7), the staff identified 10 Generic Open Items. Provide a discussion on the resolution on each of those open items. If a specific item can not be resolved during the design certification stage, then the action should be included in the Tier 1 Material (ITAAC).

Westinghouse Response:

On August 11, 2000; the NRC issued a safety evaluation report regarding Topical Report CENPD-396-P, Rev. 01, Common Qualified Platform, Appendices 1,2,3,4, Rev. 01 and CE-CES-195, Rev. 01, Software Program Manual for Common Q Systems. Section 7 of the safety evaluation report identified 10 generic open items (GOIs). Each GOI and its resolution are discussed below.

GOI 7.1
Westinghouse (formerly CENP) "has committed to develop a new I/O module or re-design some of those already considered for use in the Common Q platform in order to meet the performance requirements of EPRI TR-107330."

Resolution: A new I/O module, the AI685, has been developed and qualified. The previous resistance-temperature detector (RTD) and thermocouple (T/C) modules did not have adequate sampling time for inputs required for protection. The AI685 can be configured for use as a voltage, RTD, or T/C analog input and has been qualified for environmental, seismic and EMC conditions. The design and qualification is documented in, "Summary Qualification Report of Hardware Testing for Common Q Applications", 00000-ICE-37764, Revision 00. This report was submitted to the NRC in August 2002. An NRC Meeting was held October 2, 2002 to review the Common Q August submittal. The NRC is currently evaluating the information submitted and is working to provide Westinghouse with a supplemental safety evaluation report that is expected to close GOI 7.1.

GOI 7.2
Westinghouse (formerly CENP) "has not yet finalized the selection of the Common Q power supplies."

Resolution: The Common Q Power Supply System has been developed and qualified. This is documented in, "Summary Qualification Report of Hardware Testing for Common Q Applications, 00000-ICE-37764, Revision 00. This report was submitted to the NRC in August 2002. An NRC Meeting was held October 2, 2002 to review the Common Q August submittal. The NRC is currently evaluating the information submitted and is working to provide Westinghouse with a supplemental safety evaluation report that is expected to close GOI 7.2.

GOI 7.3

Westinghouse (formerly CENP) "has not submitted information on the design or dedication of the hardware watchdog timer and it has not yet been subjected to testing for environmental qualification."

Resolution: The internal PM646A watchdog timer meets the requirements for this on-line monitoring tool for Common Q Systems. Environmental qualification testing of the PM646A has been completed. This is documented in, "Summary Qualification Report of Hardware Testing for Common Q Applications, 00000-ICE-37764, Revision 00. This report was submitted to the NRC in August 2002. A revision to the Common Q Topical Report was also submitted that describes the use of the internal PM646A watchdog timer. An NRC Meeting was held October 2, 2002 to review the Common Q August submittal. The NRC is currently evaluating the information submitted and is working to provide Westinghouse with a supplemental safety evaluation report that is expected to close GOI 7.3.

GOI 7.4

Westinghouse (formerly CENP) "has committed to arrange a value-added reseller agreement with QSSL that is similar to BA AUT-99-ADVANT-00, the value-added reseller agreement it has with ABB products. A value-added reseller agreement is needed to satisfy the configuration control and incoming inspection requirements of EPRI TR-106439."

Resolution: On June 22, 2001 the NRC issued, "Safety Evaluation by the Office of Nuclear Reactor Regulation related to the Closeout of Several of the Common Qualified Platform Category 1 Open Items." This report states that the staff has reviewed the value-added reseller agreement with QNX Software Systems Limited (QSSL), the vendor for the flat panel display system (FPDS) operating system and display system, and concludes that it satisfies the configuration control and incoming inspection guidance of EPRI TR-106439. The reseller agreement is, therefore, acceptable. This closes GOI 7.4.

GOI 7.5

Westinghouse (formerly CENP) "will perform additional EMC tests and measurements on the PM646."

Resolution: The PM646 processor module has been modified to the PM646A. This modification involved the removal of an internal terminating resistor for the High-Speed Data links (HSLs). The link termination resistor is now external to the module, permitting several high-speed data links to be connected in parallel. Additional EMC tests and measurements were performed using the PM646A. These tests are documented in, "Summary Qualification Report of Hardware Testing for Common Q Applications", 00000-ICE-37764, Revision 00. This report was submitted to the NRC in August 2002. A revision to the Common Q Topical Report was also submitted that describes the modification of the PM646 to the PM646A. An NRC Meeting was held October 2, 2002 to review the Common Q August submittal. The NRC is currently evaluating the information submitted and is working to provide Westinghouse with a supplemental safety evaluation report that is expected to close GOI 7.5.

GOI 7.6
Westinghouse (formerly CENP) "has not yet conducted seismic and environmental qualification testing on the non-AC160 hardware components. Items not yet tested include the FPDS, watchdog timer, and power supply modules."

Resolution: Seismic and environmental qualification testing on the non-AC160 hardware components has been completed. These components include the FPDS and the power supply modules. The external watchdog timer is no longer required. The internal PM646A watchdog timer meets the requirements for this on-line monitoring tool for Common Q Systems. The seismic and environmental testing is documented in, "Summary Qualification Report of Hardware Testing for Common Q Applications", 00000-ICE-37764, Revision 00. This report was submitted to the NRC in August 2002. A revision to the Common Q Topical Report was also submitted that describes the use of the internal PM646A watchdog timer. An NRC Meeting was held October 2, 2002 to review the Common Q August submittal. The NRC is currently evaluating the information submitted and is working to provide Westinghouse with a supplemental safety evaluation report that is expected to close GOI 7.6.

GOI 7.7
"The staff has reviewed the information in the SVVP about software module testing and finds that the information provided is not sufficient for the staff to arrive at a conclusion about the adequacy of the scope of the tests for validating a software module."

Resolution: On June 22, 2001 the NRC issued, "Safety Evaluation by the Office of Nuclear Reactor Regulation related to the Closeout of Several of the Common Qualified Platform Category 1 Open Items." This report states that Westinghouse submitted additional information indicating in which sections of CE-CES-195, Rev. 01, "Software Program Manual for Common Q Systems", and topical report CENPD-396-P, Rev. 1, "Common Qualified Platform," the staff would find the Westinghouse Nuclear Automation (WNA) procedures for performing software module testing. The staff has reviewed the indicated sections and concludes that the procedures specified therein satisfy the software verification and validation program (SVVP) requirements of IEEE Std 7-4.3.2-1993 with regard to testing of software modules and are, therefore, acceptable. This closes GOI 7.7.

GOI 7.8
Westinghouse (formerly CENP) "needs to provide in future submittals the design information for the loop controllers to support their diversity from the Common Q components."

Resolution: This GOI relates to the "level 3 loop controllers" referenced in the Integrated Solution Appendix. The level 3 controllers provide component control based on signals from the ESFAS. This component will require a future submittal for NRC staff review; therefore GOI 7.8 will remain an open item. The Combined License applicant will be responsible for resolution of this GOI. DCD Table 1.8-2 and Section 7.1.6 will be revised accordingly. DCD Tier 1 Section 2.5.2 (ITAAC) does not need to be revised because it adequately covers the functional and design process requirements for the protection system.

GOI 7.9
"The staff has reviewed the approach for the integrated solution of using the ITPs and the AF100 buses to provide separation of safety and nonsafety signals and finds that there is not sufficient detail to permit an evaluation against the independence requirements set forth in IEEE Std 7-4.3.2. This must be the subject of a future {Westinghouse (formerly CENP)} submittal."

Resolution: On June 22, 2001 the NRC issued, "Safety Evaluation by the Office of Nuclear Reactor Regulation related to the Closeout of Several of the Common Qualified Platform Category 1 Open Items." This report states that Westinghouse has revised Appendix 4, "Common Qualified Platform Integrated Solution," to provide additional information on the use of the interface and test processors (ITPs) and the AF100 buses to provide separation of safety and non-safety signals. The staff has reviewed the revised information in Appendix 4, Rev. 2 on the use of the ITPs and the AF100 buses to provide separation of safety and non-safety signals and finds that the conceptual approach as presented therein is consistent with the independence requirements set forth in IEEE Std 7-4.3.2. The staff, therefore, concludes that this conceptual approach may be used for guidance for the anticipated application-specific and plant-specific designs involving the integration of multiple Common Q digital instrumentation and control (I&C) upgrades. This closes GOI 7.9 as far as the conceptual approach is concerned, but the evaluation of each forthcoming design remains a plant-specific action item because the staff finds that the forthcoming details of the actual designs may require an evaluation against the independence requirements for safety systems in specific nuclear power plants. The Combined License applicant will be responsible for resolution of this plant-specific action item.

In the AP1000 I&C configuration the interface between the safety system and the non-safety system will be implemented via the maintenance and test panel (MTP). The function of this interface is to provide communications between the AF100 bus of the Common Q safety system and the data highway of the non-safety system. The physical means of transferring data between the safety system and non-safety system is a Fast Ethernet data link using fiber optic media. One node of the Ethernet data link is the MTP flat panel display system. The other node of the Ethernet data link is a non-safety workstation, which is a drop on the non-safety system data highway.

GOI 7.10
"The evaluation of the design for the multichannel operator station control for the integrated solution requires detail beyond the scope of the present submittals."

Resolution: On June 22, 2001 the NRC issued, "Safety Evaluation by the Office of Nuclear Reactor Regulation related to the Closeout of Several of the Common Qualified Platform Category 1 Open Items." This report states that Westinghouse has revised Appendix 4 to provide additional information on the multi-channel operator station control. Westinghouse does not intend for Appendix 4, Rev. 2 to contain sufficient detail to permit the staff to evaluate any plant-specific multiple upgrade. However, Westinghouse requested the staff to evaluate the concept of the multi-channel operator station with regard to potential adverse multi-channel interaction.

The staff reviewed the design concepts presented in Appendix 4, Rev. 2, and found that the multi-channel operator stations can control most Class 1E components directly and use the ITPs and the AF100 buses discussed in GOI 7.9 above, to provide separation of safety and non-safety signals. In the AP1000 design the Class 1E signals will override any erroneous signal that may originate from non-safety equipment; therefore, manual release switches are not needed.

The AP1000 control room uses the compact workstation (CW) variation of the non-safety, multi-channel operator station. In the CW design, the multi-channel operator stations are part of the safety system. They utilize the Common Q FPDS, which the staff has approved as a Class 1E HMI device, subject to its successful performance during qualification testing. Westinghouse has determined that each CW console meets all qualification and single failure requirements for display and control of Class 1 E functions. As a result, the Common Q channelized operator modules and channelized post accident monitoring instrumentation and safe shutdown displays, which serve as credited backup HMI devices for the non-safety multi-channel operator station design, would not be required in the AP1000 CW version of the implementation of the multi-channel operator station. Therefore, the AP1000 MCR soft control display units are not classified as Class 1E and are not required to be environmentally qualified. These components will be removed from DCD Table 3.11-1.

The staff reviewed the revised information in Appendix 4, Rev. 2 with regards to the use of the multi-channel operator station including the Integrated Solution CW variation and found the conceptual approach to be free of potential for adverse multi-channel interaction and is consistent with the Class 1E independence requirements set forth in IEEE Std 7-4.3.2. The staff, therefore, concluded that the conceptual approach employing multi-channel operator stations may be used for guidance for the anticipated application-specific and plant-specific designs involving the integration of multiple Common Q digital I&C upgrades. This closes GOI 7.10 as far as the conceptual approach is concerned. The evaluation of each forthcoming design remains a plant-specific action item because the staff finds that the forthcoming details of the actual designs may require an evaluation against the independence requirements for safety systems in specific nuclear power plants. The Combined License applicant will be responsible for resolution of this plant-specific action item. DCD Table 1.8-2 and Section 7.1.6 will be revised accordingly. DCD Tier 1 Section 2.5.2 (ITAAC) does not need to be revised because it adequately covers the functional and design process requirements for the protection system.

Design Control Document (DCD) Revision:

Table 1.8-2 (Sheet 3 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
7.1-1	Setpoint Calculations for Protective Functions	7.1.6
7.1-2	Resolution of generic open items and plant-specific action items.	7.1.6

Table 3.11-1 (Sheet 17 of 45)

ENVIRONMENTALLY QUALIFIED ELECTRICAL AND MECHANICAL EQUIPMENT

Description	AP1000 Tag No.	Envir. Zone (Note 2)	Function (Note 1)	Operating Time Required (Note 5)	Qualification Program (Note 6)
~~MCR Soft Control Display Unit~~	~~PMS-JY-SCSA01~~	~~3~~	~~ESF~~	~~24 hr~~	~~E~~
~~MCR Soft Control Display Unit~~	~~PMS-JY-SCSA02~~	~~3~~	~~ESF~~	~~24 hr~~	~~E~~
~~MCR Soft Control Display Unit~~	~~PMS-JY-SCSB01~~	~~3~~	~~ESF~~	~~24 hr~~	~~E~~
~~MCR Soft Control Display Unit~~	~~PMS-JY-SCSB02~~	~~3~~	~~ESF~~	~~24 hr~~	~~E~~
~~MCR Soft Control Display Unit~~	~~PMS-JY-SCSC01~~	~~3~~	~~ESF~~	~~24 hr~~	~~E~~
~~MCR Soft Control Display Unit~~	~~PMS-JY-SCSC02~~	~~3~~	~~ESF~~	~~24 hr~~	~~E~~
~~MCR Soft Control Display Unit~~	~~PMS-JY-SCSD01~~	~~3~~	~~ESF~~	~~24 hr~~	~~E~~
~~MCR Soft Control Display Unit~~	~~PMS-JY-SCSD02~~	~~3~~	~~ESF~~	~~24 hr~~	~~E~~

7.1.6 Combined License Information

Combined License applicants referencing the AP1000 certified design will provide a calculation of setpoints for protective functions consistent with the methodology presented in Reference 5. Reference 5 is an AP600 document that describes a methodology that is applicable to AP1000. AP1000 has some slight differences in instrument spans.

Combined License applicants referencing the A1000 certified design will provide resolution for generic open items and plant-specific action items resulting from NRC review of the I&C platform.

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 420.045

Question:

420.45 (DCD Tier 1, Section 2.5.2)

Describe the method of safety analysis that will be performed on the PMS and its components to ensure the PMS will perform as specified and that failure have been identified.

Westinghouse Response:

Several activities will be performed to validate that the protection and safety monitoring system (PMS) will meet the stated functional requirements. These include the following:

a. Subjecting the PMS to environmental and seismic testing or referencing the results of applicable tests that have already been conducted, on the product line (ITAAC 2.5.2, items 2 and 4),
b. Subjecting the PMS to Electromagnetic Interference (EMI) testing or referencing the results of applicable tests that have already been conducted on the product line (ITAAC 2.5.2, item 3),
c. Performing a system level Failure Mode and Effects Analysis (FMEA) to demonstrate the PMS architecture is not susceptible to postulated single random failures (COL item, see response to RAI 420.028),
d. Use of a structured software life cycle process, per IEEE Std. 7-4.3.2, to minimize the possibility of software common mode failure errors (ITAAC 2.5.2, item 11), and
e. Subjecting the PMS to a Factory Acceptance Test (FAT) and Site Acceptance Test (SAT) to validate the system meets the stated functional requirements (ITAAC 2.5.2, items 5, 6, 8, and 9),
f. Performing preoperational tests as described in DCD subsection 14.2.9.1.12,
g. Performing startup tests as described in DCD subsections 14.2.10.1.8 and 14.2.10.1.10.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.036

Question:

Section 5.2.2.1 states that a relief valve in the residual heat removal system (RNS) provides low-temperature overpressure protection (LTOP) for the RCS, and that the valve is sized to prevent overpressure based on the following design basis events with a water solid pressurizer:

(1) the limiting mass input event of the makeup/letdown flow mismatch, and
(2) the limiting heat input event of inadvertent start of a reactor coolant pump (RCP).

Provide the safety analyses of both the limiting mass-input and heat-input overpressure events to support the adequacy of the RNS relief valve relieving capacity and set pressure specified in Table 5.4-17 for the LTOP. The description should include:

A. The applicable RCS pressure-temperature limits (LCO 3.4.3) with corresponding neutron fluence values of the reactor vessel, or the effective full power years.

B. The analysis methodology and assumptions, including consideration of limiting single failure assumption, the instrumentation uncertainties of pressure and temperature measurements, the relief valve set pressure and accumulation, the dynamic head effect of the reactor coolant flow, and the static head between the pressure tap and the limiting vessel locations, and pressure overshoot.

C. The analysis results.

D. The determination of the LTOP enable temperature of 275°F (Technical Specifications LCO 3.4.15).

Westinghouse Response:

The normal residual heat removal system (RNS) relief valve mitigates the low temperature overpressure transients and is sized to prevent the RCS pressure from exceeding the lower of either the applicable pressure-temperature (P/T) limit or 110% of the RNS system design pressure. The limiting mass and energy input transients assumed for the sizing analysis are as follows:

- Mass Input: Injection of water into the RCS from the operation of both makeup pumps due to makeup/letdown flow mismatch. The maximum flow mismatch is 177 gpm. The makeup flow is limited by the cavitating venturi located in the discharge header of the chemical and volume control system makeup pumps.

- Energy Input: During an RCS cooldown, the reactor coolant pumps are tripped at an RCS temperature of approximately 160 F. Below this temperature, the RNS continues to cool down the RCS, while the steam generators may remain at or near 160 F. It can be postulated that a 50 F differential temperature can be developed between the RCS and the steam generators under this condition. Subsequent restart of one reactor coolant pump under these conditions results in the limiting energy input cold overpressure transient. This transient is postulated to occur over a range of reactor coolant temperatures between 100 F and 200 F because an administrative requirement has been imposed in the Technical Specifications that does not allow a reactor coolant pump to be started while the RCS is water solid and the RCS temperature is above 200 F.

A. The nominal steady-state P/T limits applicable up to 54 effective full power years (EFPY) are given in DCD Figures 5.3.2 and 5.3.3. The lowest Appendix G limit from these curves is 1023 psig. The RNS system design pressure is 900 psig, and therefore the system pressure limit is 990 psig. Therefore, the lowest of the two pressure limits (990 psig) is used as the limit in the sizing of the RNS relief valve.

B. & C. The energy input transient is the limiting event for an RCS temperature above 100 F. Below 100 F, the mass input transient is more limiting. The energy input transient is analyzed using a specialized version of the LOFTRAN computer code (Reference 1), which has the capability to model the RNS relief valve. The peak pressure in the RNS system is calculated using the methodology as described in Reference 2 except that the RNS relief valve instead of the pressurizer PORV is used to mitigate the energy input transient.

Based on the energy input transient, the minimum RNS relief valve capacity of 750 gpm has been calculated at an RCS pressure equivalent to the valve setpoint of 636 psig plus 10% accumulation (700 psig). With this setpoint and capacity, the relief valve mitigates the limiting LTOP transient while maintaining the RCS pressure less than 110% of RNS design pressure. Since the relief valve is located on the RNS pump suction line, the set pressure must account for the RNS pump head to maintain the RNS discharge piping below the system design pressure. Pressure losses in the flow path and the static pressure difference between the RNS suction piping and the relief valve are also considered in establishing the relief valve set pressure. The peak pressure at the discharge of the RNS pump for the energy input transient is no higher than 979 psig. The peak pressure at the inlet to the RNS relief valve is 779 psig.

The minimum required capacity of the RNS relief valve based on the energy input transient is 750 gpm. Since the maximum flow rate for the mass input transient is 177 gpm, the RNS relief valve will be adequate to mitigate the mass input transient without overpressurizing the RNS system. The peak pressure at the inlet to the RNS relief valve will be no higher than the RNS relief valve full open pressure of 700 psig. The peak pressure at the discharge of the RNS pump will be no higher than 900 psig.

Single active failure is not considered for passive valves such as the RNS self-actuated spring relief valve. Therefore, the analysis does not consider a single failure of this valve. Also, no single active failure can occur in the RNS that could prevent the RNS suction relief valve from performing its function.

The 10% setpoint accumulation includes a 3% setpoint uncertainty. No other uncertainties are explicitly modeled in the analysis.

D. The LTOP enable temperature is based on utilizing the pressurizer safety valves for RCS overpressure protection when the RCS temperature is above 275 F (Technical Specification LCO 3.4.15). Once the RCS temperature reaches 275 F the RCS pressure can exceed the pressurizer safety valve set point pressure (2500 psig) and still be in the acceptable operating range according to the pressure/temperature curves (DCD Figures 5.3-2 and 5.3-3). The RCS pressure transients described in DCD section 15.2.3 confirm that the pressurizer safety valves are adequately sized to provide RCS overpressure protection.

DCD Table 5.4-17 will be revised to reflect the RNS relief valve design parameters given above.

Reference 1: WCAP-7907-PA, "LOFTRAN Code Description," April 1984.

Reference 2: WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1998.

Design Control Document (DCD) Revision:

From DCD page 5.4-40:

5.4.7.1.2.5 Low Temperature Overpressure Protection

The normal residual heat removal system provides a low temperature overpressure protection function for the reactor coolant system during refueling, startup, and shutdown operations. The system is designed to limit the reactor coolant system pressure to the lower of either the limits specified in 10 CFR 50, Appendix G, or 110% of the normal residual heat removal system design pressure.

From DCD page 5.4-61, Section 5.4.9.3:

The relief valve on the normal residual heat removal system has an accumulation of 10 percent of the set pressure. The set pressure is the lower of the pressure based on the design pressure of the residual heat removal system and the pressure based on the reactor vessel low temperature pressure limit. The pressure limit determined based on the design pressure includes the effect of the pressure rise across the pump.

The set pressure in Table 5.4-17 is based on the ~~reactor vessel low temperature pressure limit~~design pressure of the residual heat removal system. The lowest permissible set pressure is based on the required net positive suction head for the reactor coolant pump.

From DCD page 5.4-93:

Table 5.4-17

PRESSURIZER SAFETY VALVES - DESIGN PARAMETERS

Number	2
Minimum required relieving capacity per valve (lb/hr)	750,000 at 3% accumulation
Set pressure (psig)	2485 ±25 psi
Design temperature (°F)	680
Fluid	Saturated steam
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	500
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100

Residual Heat Removal Relief Valve - Design Parameters

Number	1
Nominal relieving capacity per valve, ASME flowrate (gpm)	~~650~~750
Nominal set pressure (psig)	~~818~~636*
Full-open pressure, with accumulation (psig)	~~900~~700*
Design temperature (°F)	400
Fluid	Reactor coolant
Backpressure	
Normal (psig)	3 to 5
Expected maximum during discharge (psig)	200
Environmental conditions	
Ambient temperature (°F)	50 to 120
Relative humidity (percent)	0 to 100

* See text (5.4.9.3) for discussion of set pressure

PRA Revision:

None

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.045

Question:

Section 5.4.7.2.2 describes the AP1000 normal residual heat removal system (RNS) design features addressing intersystem LOCA issue described in SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990. Also, Section 1.9.5.1.7 addresses AP1000's compliance with the NRC position regarding the inter-system LOCA issue. It states that AP1000 has similar fluid system design to the AP600; therefore, the conclusions of topical report WCAP-14425, "Evaluation of the AP600 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," dated July 1995, are applicable to the AP1000.

Identify design differences between the AP1000 and AP600, in terms of the design and design pressure of the primary or secondary systems and subsystems that directly or indirectly interfacing the RCS, that could affect the inter-system LOCA conclusions. For each of these differences identified, justify why the conclusions of WCAP-14425 are applicable to the AP1000.

Westinghouse Response:

There are no significant differences between AP1000 and AP600 in terms of intersystem LOCA related features. However, we have determined that a new WCAP should be issued, with AP1000-specific descriptions and illustrations. Therefore, WCAP-15993, Revision 0, "Evaluation of AP1000 Conformance to Inter-System Loss-of-Coolant Acceptance Criteria" is being provided.

Design Control Document (DCD) Revision:

Update DCD Section 1.9.5.1.7 to reflect the new WCAP, as shown on the attached pages.

PRA Revision:

None

Table 1.6-1 (Sheet 2 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
1.5	WCAP-14169 (P) WCAP-14170	Phase IVa Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report, September 1994
	WCAP-14091 (P) WCAP-14092	Phase IVb Wind Tunnel Testing for the Westinghouse AP600 Reactor, July 1994
	WCAP-12980 (P) WCAP-13573	AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report, Revision 3, April 1997
	WCAP-14371 (P) WCAP-14372	AP600 Low Flow Critical Heat Flux (CHF) Test Data Analysis, May 1995
	WCAP-14252 (P) WCAP-14253	AP600 Low Pressure 1/4 Height Integral Systems Tests - Final Data Report, Revision 1, November 1998
	WCAP-14309 (P) WCAP-14310	AP600 Design Certification Program, SPES-2 Tests Final Data Report, Revision 2, May 1997
	WCAP-12648 (P) WCAP-13322	AP600 Incore Instrumentation System Electromagnetic Interference Test Report, Revision 1, April 1992
	WCAP-13298 (P) WCAP-13299	RCP Air Model Test Report, August 1991
	WCAP-12668 (P) WCAP-13321	AP600 High Inertia Rotor Testing - Phase I, Test Report, March 1990
	WCAP-13319 (P) WCAP-13320	AP600 High Inertia Rotor Testing - Phase 2 Report, August 1991
	WCAP-13758 (P) WCAP-13759	High Inertia Rotor Test - Phase 3 Report, June 1993
	WCAP-15613 (P) WCAP-15706	AP1000 PIRT and Scaling Assessment Report, March 2001
	WCAP-15612	AP1000 Plant Description and Analysis Report, December 2000
1.9	WCAP-~~14425~~15993	Evaluation of the ~~AP600~~ AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria, ~~July 1995~~December 2002

(P) Denotes Document is Proprietary

subsection 8.3.1. The diesel generator reliability is modeled in the PRA. The reliability assurance program is discussed in Section 16.2.

B-61 Allowable ECCS Equipment Outage Periods

Discussion:

Generic Safety Issue B-61 addresses surveillance test intervals and allowable equipment outage periods in the technical specifications for safety-related systems. This task involves the NRC development of analytically based criteria for use in confirming or modifying these surveillance intervals and allowable equipment outage periods.

AP1000 Response:

The AP1000 surveillance test intervals and allowable outage times help to meet plant safety goals while maximizing plant availability and operability. In determining these limits for the AP1000 technical specifications, a combination of NUREG-1431 precedent, system design, and safety-related function is considered.

B-63 Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary

Discussion:

Generic Issue B-63 addresses the adequacy of the isolation of low-pressure systems that are connected to the reactor coolant pressure boundary. The NRC staff requires that valves forming the interface between high- and low-pressure systems associated with the reactor coolant boundary have sufficient redundancy to prevent the low-pressure systems from being subjected to pressures that exceed their design limits.

AP1000 Response:

The AP1000 includes interconnections between high- and low-pressure systems. Each of these systems interfaces contains appropriate isolation provisions. Valves at the interface between high- and low-pressure systems have redundancy to prevent low-pressure systems from being subjected to pressures that exceed their design limits. The AP1000 design meets the provisions of the Standard Review Plan, Section 3.9.6.

The normal residual heat removal system interface is addressed in subsection 5.4.7. WCAP-~~14425~~ 15993 (Reference 56) provides an evaluation of the ~~AP600~~ AP1000 conformance to intersystem loss-of-coolant accident regulatory criteria. ~~The conclusions of WCAP-14425 are applicable to the AP1000 since the fluid system design for the AP1000 is the same as the AP600.~~

B-66 Control Room Infiltration Measurements

Discussion:

Generic Safety Issue B-66 addresses the adequacy of control room area ventilation systems and control building layout to ensure that plant operators are adequately protected against the effects of and the final actuation devices is accomplished by the physical separation of the various sensors and components using existing containment walls as barriers.

- The in-containment fire area contains reduced combustible material due to the use of canned reactor coolant pump motors that do not use oil lubrication and due to strict combustible material limitations.

Main Control Room:

- Functionality requirements dictate that the main control room be a single fire zone. Features are included in the main control room to:

  – Reduce the probability of fire initiation
  – Reduce the likelihood of fire spreading
  – Increase the probability of fire detection
  – Effectively mitigate the effects of a fire

- In the event of main control room evacuation, safe shutdown conditions are established and maintained using the remote shutdown workstation.

See Appendix 9A.3 for information on the main steam tunnel and the passive containment cooling system valve room. See subsection 9.5.1 and Appendix 9A for additional information.

1.9.5.1.7 Intersystem LOCA

NRC Position:

Overpressurization of low-pressure piping systems due to reactor coolant system boundary isolation failure could result in rupture of the low-pressure piping outside containment. This may result in a core melt accident with an energetic release outside the containment building that could cause a significant offsite radiation release.

It is the NRC position that designing interfacing systems to withstand full reactor pressure is an acceptable means of resolving this issue. The Staff Requirements Memorandum to SECY-90-016 (Reference 31) added that consideration should be given to all elements of the low-pressure system (such as instrument lines, pump seals, heat exchanger tubes, and valve bonnets). For interfacing systems not designed to withstand full reactor coolant system pressure, it is necessary to provide leak testing capability for the pressure isolation valves, main control room position indication for de-energized reactor coolant system isolation valves, and high pressure alarms to alert control room operators when increasing reactor coolant system pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

AP1000 Response:

The AP1000 has incorporated various design features to address intersystem loss-of-coolant accident challenges. These design features result in very low AP1000 core damage frequency for intersystem loss-of-coolant accidents compared with operating nuclear power plants. The design features are primarily associated with the normal residual heat removal system and are discussed in Section 3 of WCAP-15993 ~~14425~~ (Reference 56) as well as DCD subsection 5.4.7. WCAP-15993 ~~14425~~ was prepared to document the evaluation of the ~~AP600~~ AP1000 for conformance to the intersystem loss-of-coolant accident regulatory criteria identified in various NRC documents. ~~The AP1000 has similar fluid system design to the AP600 therefore, the conclusions of WCAP-14425 are applicable to the AP1000. As a result of the evaluation documented in WCAP-14425, additional design features were incorporated into the AP600. These design features are included in the AP1000 and are documented in other portions of the DCD.~~ See ~~WCAP-14425~~that document for additional information on conformance to intersystem loss-of-coolant accident regulatory criteria.

1.9.5.1.8 Hydrogen Generation and Control

NRC Position:

It is the NRC position that the likelihood of early containment failure from hydrogen combustion should be reduced. Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, advanced light water reactors should be designed to:

- Accommodate hydrogen equivalent to 100 percent metal-water reaction of the fuel cladding
- Limit containment hydrogen concentration to no greater than 10 percent

Further, because hydrogen control is necessary to preclude local concentrations of hydrogen below detonable limits, and given uncertainties in present analytical capabilities, advanced light water reactors should provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents. Additional advantages of providing hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) includes the lessening of pressure and temperature loadings on the containment and essential equipment.

AP1000 Response:

The AP1000 design includes mechanisms for monitoring and controlling hydrogen inside the containment. The containment hydrogen control system maintains hydrogen concentrations below 10 percent following the reaction of 100 percent of the zircaloy cladding.

Passive autocatalytic hydrogen recombiners control hydrogen concentration following design basis events. Nonsafety-related hydrogen igniters control rapid releases of hydrogen during and after postulated events with degraded core conditions or with core melt.

Sufficient vent area is provided for each subcompartment in the containment to prevent high local concentrations of hydrogen.

The containment air filtration system provides a capability to purge the containment atmosphere.

See subsection 6.2.4 for additional information.

55. Not used

56. WCAP-~~14425~~15993, "Evaluation of the ~~AP600~~ AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," ~~July 1995~~December 2002.

57. SECY-95-172, "Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," June 30, 1995.

58. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," L. Soffer, et al., February 1995.

59. SECY-94-302, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs," December 19, 1994.

60. SECY-94-194, "Proposed Revisions to 10 CFR Part 100 and 10 CFR Part 50, and New Appendix S to 10 CFR Part 50," July 27, 1994.

61. Not used.

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

63. WCAP-15799, "AP1000 Compliance with SRP Acceptance Criteria."

64. NCRP Report No. 116, Limitation of Exposure to Ionizing Radiation, March 31, 1993.

65. WCAP-15800, "Operational Assessment for AP1000."

66. SECY-98-161, "The Westinghouse AP1000 Standard Design as it Relates to the Fire Protection and the Spent Fuel Pool Cooling Systems," July 1, 1998.

67. SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994.

68. SECY-95-172, "Key Technical Issues Pertaining to the Westinghouse AP1000 Standardized Passive Reactor Design," June 30, 1995.

69. WCAP-14477, "The AP600 Adverse Systems Interactions Evaluation Report", Revision 1, April 1997.

AP1000 DESIGN CERTIFICATION REVIEW

Response to Request For Additional Information

RAI Number: 440.055

Question:

Table 15.0-2, listing a summary of initial conditions and computer codes used in the accident analysis, is incomplete and inconsistent with the safety analyses in various sections of Chapter 15. For example, Section 15.1.2.2.1 indicates that the computer codes used for an increased feedwater event are LOFTRAN, FACTRAN and VIPRE, while Table 15.0-2 lists only LOFTRAN as the code used for the analysis. Section 15.4.8.2 indicates that for the rod ejection accident analysis, three codes are used: TWINKLE for the calculations of physics parameters; FACTRAN for the fuel rod temperature calculations, and THINC for the DNBR calculations. The use of THINC is neither discussed in Section 15.0-11, "Computer Codes Used," nor included in Table 15.0-2. Also, Table 15.0-2 does not include information for the increase in reactor coolant inventory due to chemical and volume control system (CVS) malfunction event that is analyzed and discussed in Section 15.5.2.

Verify the accuracy of the information provided in Table 15.0-2 and revise the table if necessary to be consistent with applicable Sections of Chapter 15.

Westinghouse Response:

Table 15.0-2 will be revised, as indicated below, to be consistent with applicable Sections of Chapter 15.

For AP600 the THINC computer code was utilized for DNBR calculations. For AP1000 the VIPRE-01 computer code is used for DNBR calculations. There are no AP1000 specific analyses, which utilize the THINC computer code, and thus the reason it is not explicitly discussed in DCD Chapter 15.0 or listed in Table 15.0-2. However, as discussed in DCD 15.4.8.2.1.8 an analysis has been performed (and documented in WCAP-7588) for Westinghouse fuel using the THINC computer code, which is applicable to AP1000, to define the percentage of rods in DBD for use in the fission product release analyses.

Design Control Document (DCD) Revision:

From DCD Chapter 15, Table 15.0-2:

Table 15.0-2 (Sheet 1 of 5)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Moderator Temperature (pcm/°F)	Doppler	Initial Thermal Power Output Assumed (MWt)
15.1	Increase in heat removal from the primary system					
	Feedwater system malfunctions causing a reduction in feedwater temperature	Bounded by excessive increase in secondary steam flow	–	–	–	–
	Feedwater system malfunctions that result in an increase in feedwater flow	LOFTRAN, FACTRAN, VIPRE-01	0.470	–	Upper curve of Figure 15.0.4-1	0 and 3415
	Excessive increase in secondary steam flow	LOFTRAN, FACTRAN, VIPRE-01	0.0 and 0.470	–	Upper and lower curves of Figure 15.0.4-1	3415

	Inadvertent opening of a steam generator relief or safety valve	LOFTRAN, VIPRE-01	Function of moderator density (see Figure 15.1.4-1)	–	See subsection 15.1.4.	0 (subcritical)
	Steam system piping failure	LOFTRAN, VIPRE-01	Function of moderator density (see Figure 15.1.4-1)	–	See subsection 15.1.5	0 (subcritical)
	Inadvertent operation of the PRHR heat exchanger	LOFTRAN, FACTRAN, VIPRE-01	See subsection 15.1.6.2.1	–	Upper curve of Figure 15.0.4-1	3415

Table 15.0-2 (Sheet 2 of 5)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

			Reactivity Coefficients Assumed			
Section	**Faults**	**Computer Codes Used**	**Moderator Density ($\Delta k/gm/cm^3$)**	**Moderator Temperature (pcm/°F)**	**Doppler**	**Initial Thermal Power Output Assumed (MWt)**
15.2	Decrease in heat removal by the secondary system					
	Loss of external electrical load and/or turbine trip	LOFTRAN, FACTRAN, VIPRE-01	0.0 and 0.470	–	Lower and upper curves of Figure 15.0.4-1	3415 & 3483.3 (a)

	Inadvertent closure of main steam isolation valves	Bounded by turbine trip event	–	–	–	–
	Loss of condenser vacuum and other events resulting in turbine trip	Bounded by turbine trip event.	–	–	–	–
	Loss of nonemergency ac power to the plant~~station~~ auxiliaries	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)
	Loss of normal feedwater flow	LOFTRAN	0.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)
	Feedwater system pipe break	LOFTRAN	~~0.4~~700.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)
15.3	Decrease in reactor coolant system flow rate					
	Partial and complete loss of forced reactor coolant flow	LOFTRAN, FACTRAN, COAST, VIPRE-01	0.0	–	Lower curve of Figure 15.0.4-1	3415
	Reactor coolant pump shaft seizure (locked rotor) & reactor coolant pump shaft break	LOFTRAN, FACTRAN, COAST, VIPRE-01	0.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)

Table 15.0-2 (Sheet 3 of 5)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density ($\Delta k/gm/cm^3$)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.4	Reactivity and power distribution anomalies					
	Uncontrolled RCCA bank withdrawal from a subcritical or low power startup condition	TWINKLE, FACTRAN, VIPRE-01	–	0.0	Coefficient is consistent with a Doppler defect of -0.67%Δk	0
	Uncontrolled RCCA bank withdrawal at power	LOFTRAN, FACTRAN, VIPRE-01	0.0 and 0.470	–	Upper and lower curves of Figure 15.0.4-1	10%, 60%, and 100% of 3415
	RCCA misalignment	~~See Section 4.3~~ LOFTRAN, VIPRE-01	NA	–	NA~~Not applicable~~	3415
	Startup of an inactive reactor coolant pump at an incorrect temperature	NA	NA	–	NA	NA
	Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant	NA	NA	–	NA~~Not applicable~~	0 and 3415

Table 15.0-2 (Sheet 4 of 5)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

Section	Faults	Computer Codes Used	Reactivity Coefficients Assumed: Moderator Density (Δk/gm/cm^3)	Reactivity Coefficients Assumed: Moderator Temperature (pcm/°F)	Reactivity Coefficients Assumed: Doppler	Initial Thermal Power Output Assumed (MWt)
15.4	Inadvertent loading and operation of a fuel assembly in an improper position	~~See Section 4.3~~ ANC	NA	–	NA~~Not applicable~~	3415
	Spectrum of RCCA ejection accidents	TWINKLE, FACTRAN	Refer to subsection 15.4.8	Refer to subsection 15.4.8	Coefficient consistent with a Doppler defect of -0.90% ΔK at BOC(b) and -0.87% ΔK at EOC (b)	0 and 3483.3 (a)
15.5	Increase in reactor coolant inventory					
	Inadvertent operation of the emergency core cooling system during power operation	LOFTRAN	0.0 ~~and 0.470~~	–	Upper ~~and lower~~ curves of Figure 15.04-1	~~3415~~3483.3 (a)

	Chemical and volume control system malfunction that increases reactor coolant inventory	LOFTRAN	0.0	–	Upper curve of Figure 15.04-1	3483.3 (a)
15.6	Decrease in reactor coolant inventory					
	Inadvertent opening of a pressurizer safety valve and inadvertent operation of ADS	LOFTRAN, FACTRAN, VIPRE-01	0.0	–	Lower curve of Figure 15.0.4-1	3415

Table 15.0-2 (Sheet 5 of 5)

SUMMARY OF INITIAL CONDITIONS AND COMPUTER CODES USED

			Reactivity Coefficients Assumed			
Section	**Faults**	**Computer Codes Used**	**Moderator Density ($\Delta k/gm/cm^3$)**	**Moderator Temperature (pcm/°F)**	**Doppler**	**Initial Thermal Power Output Assumed (MWt)**
15.6	Steam generator tube failure	LOFTTR2	0.0	–	Lower curve of Figure 15.0.4-1	3483.3 (a)
	A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate containment	NA	NA	–	NA	NA

LOCAs resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary	NOTRUMP WCOBRA/ TRAC	See subsection 15.6.5 references	–	See subsection 15.6.5 references	3468.0

Notes:
a. 102% of rated thermal power
b. BOC – Beginning of core cycle
EOC – End of core cycle

PRA Revision:

None

RAI Number 440.066

Question:

The statements on page 15.1-15 and 15.1-22 indicate that a comparison of the results from "the detailed core analysis" with the LOFTRAN predications confirms the overall conservatism of the methodology used for analyses of an SLB event and an inadvertent operation of the PRHR heat exchanger event.

Describe the methods and computer codes used for "the detailed core analysis," reference the associated NRC acceptance letters to confirm the acceptance of the methods and codes for licensing calculations, and demonstrate that the use of the acceptable methods and codes are within in the applicable ranges.

Westinghouse Response:

The analysis methodology summary of a Steamline Break event has been provided in Section 3.3 of Reference 440.066-1: (Section 3.3 Secondary Side Depressurization Events Due To Steam System Piping Failure Or Inadvertent Opening Of A Steam Generator Relief/Safety Valve).
Similarly, the analysis methodology summary of a PRHR heat exchanger event has been provided in Section 3.4 of Reference 440.066-1: (Section 3.4 Inadvertent Operation Of The Passive Residual Heat Removal System).

Analytical methods and computer codes used in the evaluation of these accidents remain primarily unchanged from those previously reviewed and accepted for AP600 and identified in Reference 440.066-2. The one exception is the replacement of the THINC code (WCAP-7956-P-A – February 1989 and WCAP-8054-P-A – February 1989) with the VIPRE-01 code (WCAP-14565-P-A – October 1999).

Core analysis codes PHOENIX-P/ANC (WCAP-11596-P-A – June 1988 and WCAP-10965-P-A – September 1986) have been approved for core reactivity and power distribution determinations including the more severe power distributions resulting from off-normal conditions such as ejected-rod, dropped-rod, and stuck-rod configurations (similar to that which would be assumed in a Steamline Break event).

One primary goal of the detailed core analysis is to provide confirmation that the reactivity response as predicted by LOFTRAN is representative of that which would be predicted by the detailed 3D core analysis code ANC. The consistency of the reactivity behavior between LOFTRAN and ANC is demonstrated below for analysis of a SLB event and an inadvertent operation of the PRHR heat exchanger event.

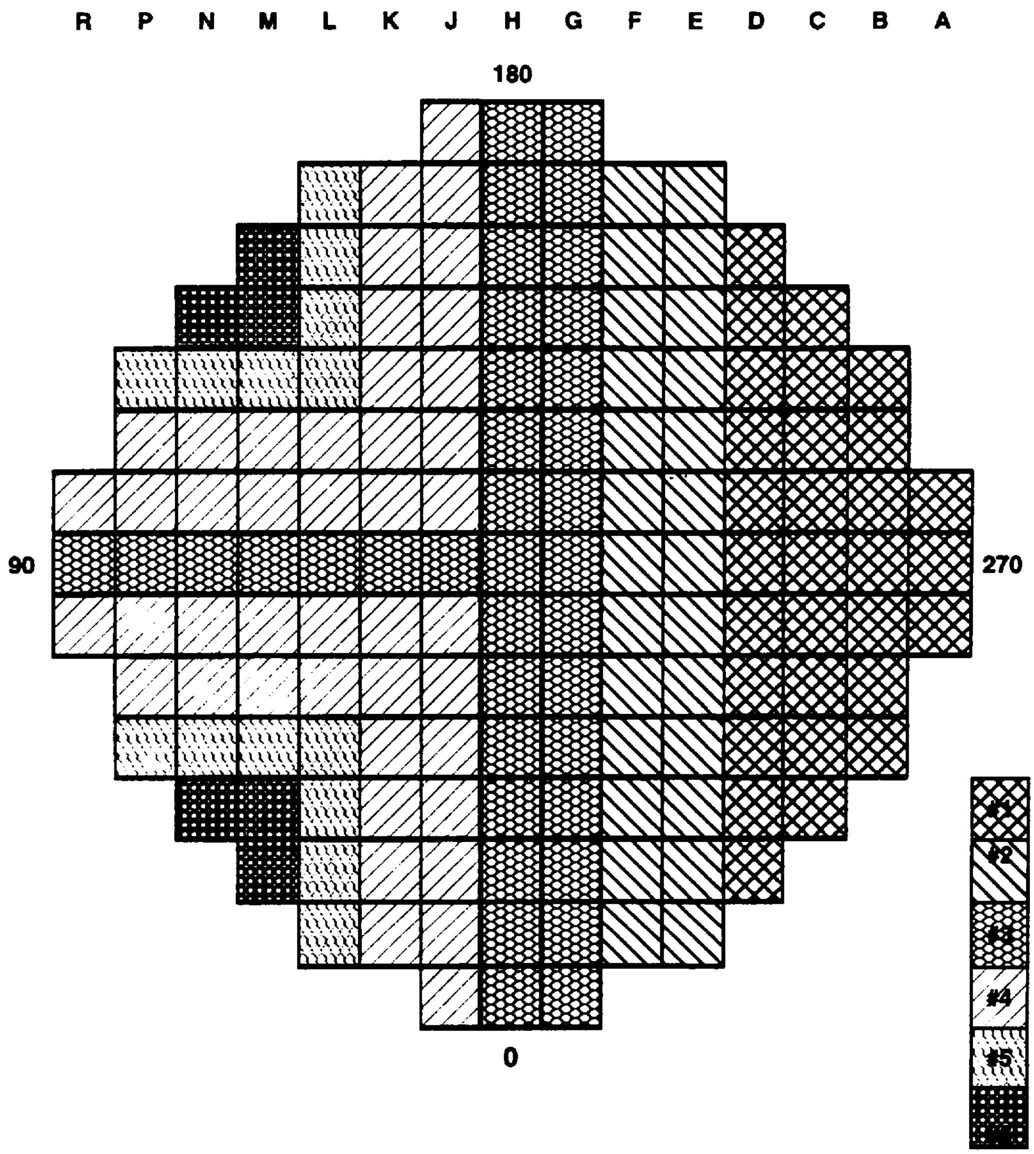
With regard to analysis of an SLB event, refer to the response given for RAI 440.070. With regard to the analysis of inadvertent operation of the PRHR heat exchanger event, the following provides a brief description of the core analysis:

1. The end of cycle life (EOL) timestep of a representative 18 month Equilibrium Cycle design 3D ANC model is unfolded from a quarter core model to a full core (157 discrete assembly) model. Use of an Equilibrium Cycle model results in the maximum reactivity insertion for a cooldown event since the moderator density coefficients are greatest for this model. The ARO core soluble boron concentration is conservatively set to zero ppm.

2. Using LOFTRAN calculations initiating from Hot Full Power, an estimate of the variable loop cooldown impact on the core inlet temperature distribution is determined for a number of statepoints. A typical distribution is illustrated in Figure 440.066-1. This asymmetry in temperature distribution results in a radially asymmetric power distribution, which is captured by the 3D ANC model.

3. The variable inlet temperature option in ANC is utilized with the unfolded ANC model to perform power searches for the LOFTRAN calculated inadvertent PRHR statepoints. The 3D ANC calculated power is compared to the LOFTRAN point kinetics calculated power. An iterative process follows which adjusts the moderator density coefficient used in LOFTRAN so that a power match with the 3D ANC model is obtained. It is worthwhile to note that ANC yielded a maximum power level of 119.1 % power versus the LOFTRAN prediction of 119.6 % power and the results are in excellent agreement.

4. Detailed power distribution data is then used in the formal VIPRE-01DNB evaluations.



Tin (1) = 535.3
Tin (2) = 529.0
Tin (3) = 520.7
Tin (4) = 510.3
Tin (5) = 502.1
Tin (6) = 498.3

Figure 440.066-1
AP1000 Mixing Regions

References:

440.066-1 "AP1000 Analysis Methodology Summary for Events Using the LOFTRAN Code Family", APP-GW-GSR-010, Transmitted via Letter DCP/NRC 1488, Oct. 31, 2001, to A.C. Rae (NRC) from M. M. Corletti (Westinghouse) as "Transmittal of Westinghouse Report"

440.066-2 NUREG-1512 "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design Docket No. 52-003, September 1998.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.102 (Response Revision 1)

Question:

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

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

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

Westinghouse Response (Revision 1):

While the original BEACON topical report was based on the use of moveable incore detectors, use of BEACON with fixed incore detector systems has been addressed through addendums to the BEACON Topical Report. These addendum were approved by the NRC. BEACON using fixed incore detectors signals has been used in two plants, one internationally and one non-tech spec surveillance version in the US.

Addendum 1 of WCAP-12472 -P-A was approved by the USNRC and issued as an approved version in January 2000. Addendum 1 addressed two key areas for the use of fixed incore detectors. First, it presented the methodology for using fixed incore detectors and described the uncertainty analysis methodology to be used in plants with fixed incore detectors. Data from different NSSS vendors plant types were used to demonstrate this methodology. The second area is the methodology used to model rhodium fixed incore detectors. The BEACON monitoring methodology is based on the ratio of the measured to predicted detector currents. Since Westinghouse plants had not used fixed incore detectors, the methodology for BEACON to predict a fixed incore detector current had not previously been presented to the NRC. This methodology is based on previsouly approved Westinghouse core physics methods.

Because of customer interest in a long-lived fixed incore detector material, the methodology to predict incore detector currents was extended in Addendum 2 to WCAP-12472. This addendum extended the fixed incore detector methods to include fixed detectors made of vanadium or platinum material. This addendum was reviewed and approved by the USNRC and the approved version of the topical report was released in May 2002.

Westinghouse does not believe that additional licensing of the BEACON system is required to support the AP1000. The specific effort will be determining the measurement variability of the selected fixed incore detector design and, using the methodology described in the BEACON Topical report including the addendums, to determine appropriate measurement uncertainties appropriate for the AP1000.

Design Control Document (DCD) Revision:

DCD Chapter 16, TS 5.6.5, pg. 5.0-19:

6. WCAP-12472-P-A, "BEACON Core Monitoring and Operations Support System," August 1994, ~~and~~ Addendum 1, May 1996 (Westinghouse Proprietary), **and Addendum 2 March 2001 (Westinghouse Proprietary).**

   (Methodology for Specification 3.2.5 - OPDMS - Monitored Power Distribution Parameters.)

~~REVIEWER'S NOTE~~

~~Additional power distribution control and surveillance methodologies (for MSHIM and OPDMS monitoring) are currently under development and will be added upon NRC approval. An NRC approved addendum to WCAP-12472-P-A covering AP1000 power distribution control and surveillance methodologies must be in place prior to generating an AP1000 specific version of this technical specification.~~

DCD Chapter 4, Sections 4.3.4 and 4.3.5:

4.3.4 Combined License Information

This section contains no requirement for additional information to be provided in support of the combined license. Combined License applicants referencing the AP1000 certified design will address changes to the reference design of the fuel, burnable absorber rods, rod cluster control assemblies, or initial core design from that presented in the DCD.

~~The Combined License applicant will reference an NRC approved addendum to WCAP-12472-P-A (Reference 4) covering AP1000 fixed incore detectors.~~

4.3.5 References

1. Bordelon, F. M, et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A, July 1985.

[*2. Davidson, S. L. (Ed.), "Fuel Criteria Evaluation Process," WCAP-12488-P-A (Proprietary) and WCAP-14204-A - (Nonproprietary), October 1994.*]*

3. ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

4. Beard, C. L. and Morita, T., "BEACON: Core Monitoring and Operations Support System," WCAP-12472-P-A Proprietary) and WCAP-12473-P-A (Nonproprietary), August 1994, ~~and~~ Addendum 1, May 1996, **and Addendum 2, March 2001.**

DCD Chapter 1.6, Table 1.6-1 (Sheet 6 of 20):

Table 1.6-1 (Sheet 6 of 20)

MATERIAL REFERENCED

DCD Section Number	**Westinghouse Topical Report Number**	**Title**
4.2	WCAP-8691	Fuel Rod Bow Evaluation, Revision 1, July 1979
	WCAP-9500-P-A (P) WCAP-9500-A	Reference Core Report 17x17 Optimized Fuel Assembly, May 1982
	WCAP-8236 (P) WCAP-8288	Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident, December 1973
	WCAP-9401-P-A (P) WCAP-9402-A	Verification, Testing, and Analysis of the 17x17 Optimized Fuel Assembly, August 1981
	WCAP-9283	Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events, March 1978
	WCAP-15063-P-A (P)	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), Rev. 1, July 2000
	WCAP-8377 (P)	Revised Clad Flattening Model, July 1974
4.3	WCAP-9272-P-A	Westinghouse Reload Safety Evaluation Methodology, July 1985
	[*WCAP-12488-P-A (P)* [*WCAP-14204-A*]*	*Fuel Criteria Evaluation Process, October 1994*]*
	WCAP-12472-P-A (P) WCAP-12473-P-A	BEACON: Core Monitoring and Operations Support System, August 1994, ~~and~~ Addendum 1, May 1996,

Table 1.6-1 (Sheet 6 of 20)

MATERIAL REFERENCED

DCD Section Number	Westinghouse Topical Report Number	Title
		Addendum 2, March 2001
	WCAP-8330	Westinghouse Anticipated Transients Without Reactor Trip Analysis, August 1974

DCD Chapter **1.6, Table 1.6-1 (Sheet 6 of 20)**:

Table 1.8-2 (Sheet 3 of 6)

SUMMARY OF AP1000 STANDARD PLANT COMBINED LICENSE INFORMATION ITEMS

Item No.	Subject	Subsection
4.3-1	Changes to Reference Reactor Design	4.3.4
~~4.3-2~~	~~Fixed Incore Detectors~~	~~4.3.4~~
4.4-1	Changes to Reference Reactor Design	4.4.7
5.2-1	ASME Code and Addenda	5.2.6.1

PRA Revision:

None

RAI Number: 440.147

Question:

In Tier 1 ITAAC Table 2.3.6-4, Item 9.d specifies a design commitment of the RNS to provide heat removal from the IRWST with the acceptance criteria that each RNS pump provides at least 925 gpm to the IRWST.

Describe how this acceptance criteria is established.

Westinghouse Response:

The criterion for the RNS pump flow rate required to cool the IRWST is based on investment protection, not safety. The criterion is to prevent the IRWST from boiling during reactor decay heat removal by the passive residual heat removal heat exchanger. Nominal conditions with both RNS pumps and heat exchangers operating are assumed in the calculation of the required flow rate. The specified RNS flow rate provides sufficient heat removal capability to maintain the bulk IRWST temperature below 212 F. The total RNS flow rate required to meet this criterion in the AP1000 is 2200 gpm.

The minimum flow rate in Tier 1 Table 2.3.6-4, Item 9.d will be revised from 925 gpm per pump to 2200 gpm total RNS flow with both pumps operating.

Design Control Document (DCD) Revision:

From DCD Tier 1 page 2.3.6-14, Table 2.3.6-4:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
9.d) The RNS provides heat removal from the in-containment refueling water storage tank (IRWST).	Testing will be performed to confirm that the RNS can provide flow through the RNS heat exchangers when the pump suction is aligned to the IRWST and the discharge is aligned to the IRWST.	~~Each~~ Two operating RNS pumps provides at least ~~925~~ 2200 gpm to the IRWST.

PRA Revision:

None

RAI Number: 440.155

Question:

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

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

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

Reference

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

Westinghouse Response:

(a) The AP1000 WCOBRA/TRAC model shown in Figure 3.1 of WCAP-15833 indicates that Gaps 23 and 26 are located between ADS4 ports and the Steam Generator inlet plenum where an oscillating plug may be present. These gaps are connected to the hot leg bend channels, 23 and 26, where the downstream flow connection is vertical. It has been pointed out that the slugging transition by Taitel-Dukler flow regime map may be less limiting or inadequate for counter-current flows at the bend (see for example Reference [2]). In addition, as the reviewer commented, a particular flow configuration may exist at the ADS4 port and downstream of ADS4 where liquid may be trapped.

In a bend like channels 23 and 26, WCOBRA/TRAC tends to predict liquid buildup since the noding scheme allows liquid to build up by the lateral force balance (lateral momentum equations) at three vertical cells and the presence of interfacial drag vertically. Figure 155.1 shows the integrated liquid flow into (ADS4 side) and out of (Steam Generator side) Channel 26, and the liquid fraction in Channel 26 during the time period between opening of the first

ADS4 (at 11 sec) and opening of the second ADS4 (at 70 sec). As seen in the figure, liquid accumulated quickly in the bend region in the first 20 seconds, and its liquid fraction reached to a semi-constant value of 0.8. The liquid fraction in the bend then decreased to 0.5 about 10 seconds before opening of the second set of ADS4. The figure also shows that the flow reversed occasionally and trapped liquid flowed back into the pipe section connected to the ADS4. When this discharge occurred at approximately 22, 36 and 43 seconds, the flow quality into ADS4 was reduced significantly and the total flowrate increased as seen in Figure 155.2.

The potential underprediction of counter current flooding at the bend may have an impact on this transient since the discharge of liquid from the bend to ADS4 tends to lower the quality and increase the flowrate. Figure 155.3 shows the integrated liquid flow immediately downstream of ADS4 port (solid) together with the integrated vapor flow (dash). The figure indicates that the vapor flow was mostly positive with occasional flow reversal or stagnation while the liquid repeated periods of forward and backward flows. The liquid discharge back from the bend at 22, 36 and 43 seconds occurred when the vapor flow is either small positive (the flow is counter-current) or negative (the flow is co-current back flow). Assuming that the flow pattern in the bend region is similar to the hydraulic jump known to occur in the horizontal leg connected to a riser, the next figure examined the predicted flows at the bend against the known correlation which considers such flow patter at the bend. Figure 155.4 shows the WCOBRA/TRAC prediction for counter current flow at gap 26 in Jg*-Jl* with Ohnuki's correlation for AP1000 geometry at the bend [3]. Figure 155.5 shows the same parameters after the second ADS4 opening. Both figures indicate that the counter-current flows predicted by WCOBRA/TRAC for the bend in AP1000 were reasonable assuming that the correlation for the CCFL at the bend is applicable to downstream of ADS4.

The modeling assumption of the entrainment at the ADS4 port is that the ADS4 port sees liquid in the channel connected to the branch line as the only source of entrainment. This assumption does not consider the liquid accumulated in the bend region as the direct source of liquid for entrainment. However, this liquid does become entrained whenever it flows back into the port section as shown in Figure 155.2.

If we allow a direct entrainment of liquid that is trapped in the bend region back into ADS4 line, we would have a higher entrainment and higher mass flow in ADS4 line, at least for the period between opening of the first and second set of ADS4. However the high mass flow could not continue since a limited amount of liquid is available in the bend region and also in the hot leg. Figure 155.6 shows the same parameters as Figure 155.1 after the opening of second set of ADS4. As seen in the figure, liquid accumulation in the bend region is limited and the liquid fraction remains at around 0.5.

(b) The Martinelli parameter X2=|(dp/dz)L|/|(dp/dz)G| loses meaning for use in Taitel-Dukler map when liquid is stagnant or counter-current. Because of this, in this version of WCOBRA/TRAC-AP, a stratified flow regime is assumed when a stagnant or counter-current flow is predicted. This treatment is thought to be adequate since it has been shown that WCOBRA/TRAC is able to predict the counter-current flooding in a horizontal pipe without a bend without using Taitel-

Dukler map. The predicted flooding curve is comparable to that predicted by Taitel-Dukler's slugging transition boundary [4].

Additional References:

[2] H. Siddiqui, S. Banerjee, and K. H. Ardron, "Flooding in an Elbow between a Vertical and a Horizontal or Near-Horizontal Pipe, Part I: Experiment," Int. J. Multiphase Flow, Vol. 12 No. 4, pages 531-541, 1986.
[3] A. Ohnuki, "Experimental Study of counter-current two phase flow in horizontal tube connected to inclined riser," Journal of Nuclear Science and Technology, Volume 23 (1986), pages 219-232.
[4] K. Takeuchi, S. M. Bajorek, L. E. Hochreiter, and R. M. Kemper, "Horizontal Stratified Flow in Hot and Cold Legs at a Small Break LOCA of PWR," 93-HT-1, presented at ASME National Heat Transfer Conference, August 8-11, 1993.

Figure 155.1: Liquid Flows Upsteam and Downstream of Bend

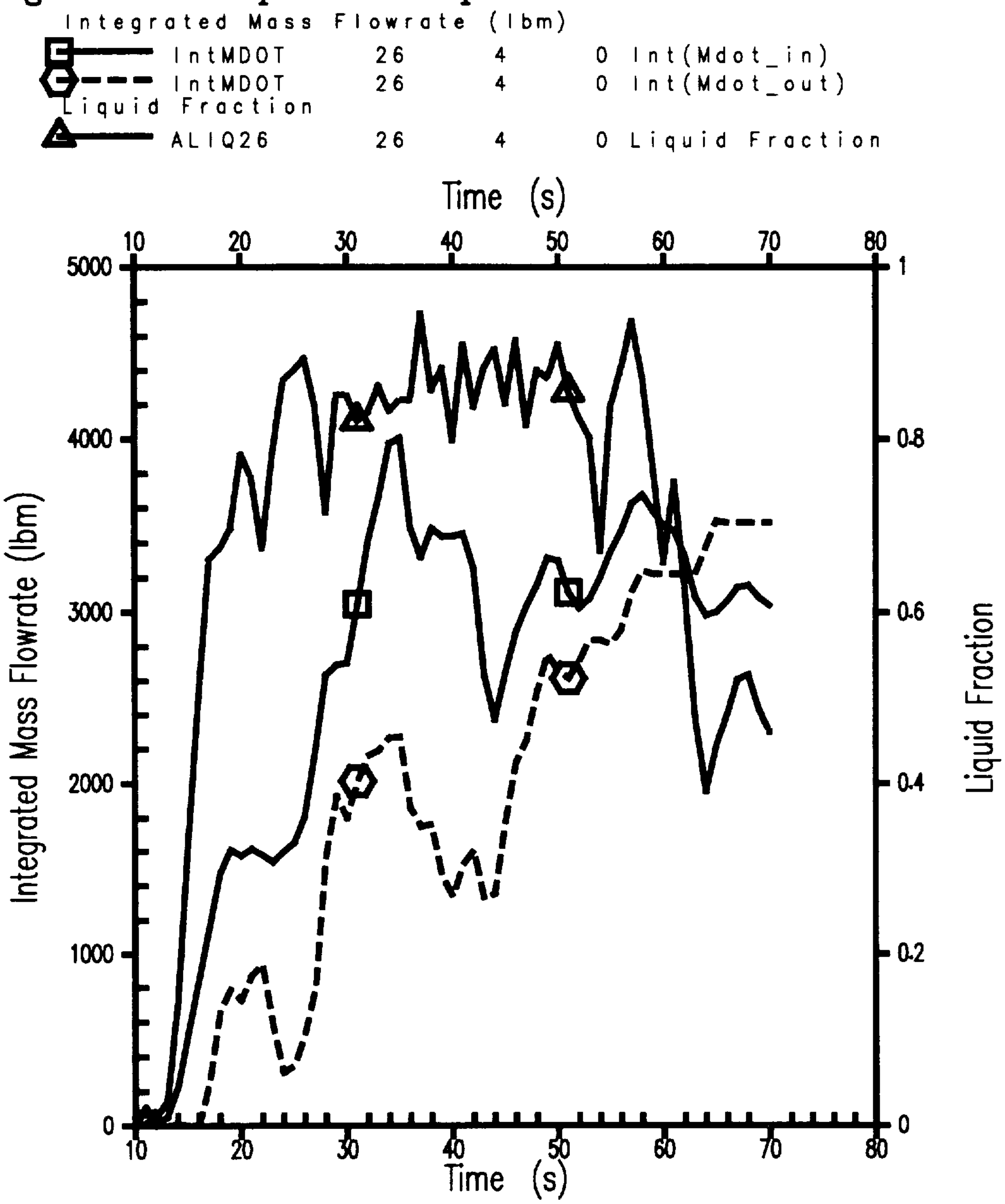


Figure 155.2: ADS Flow Quality and Mass Flowrate

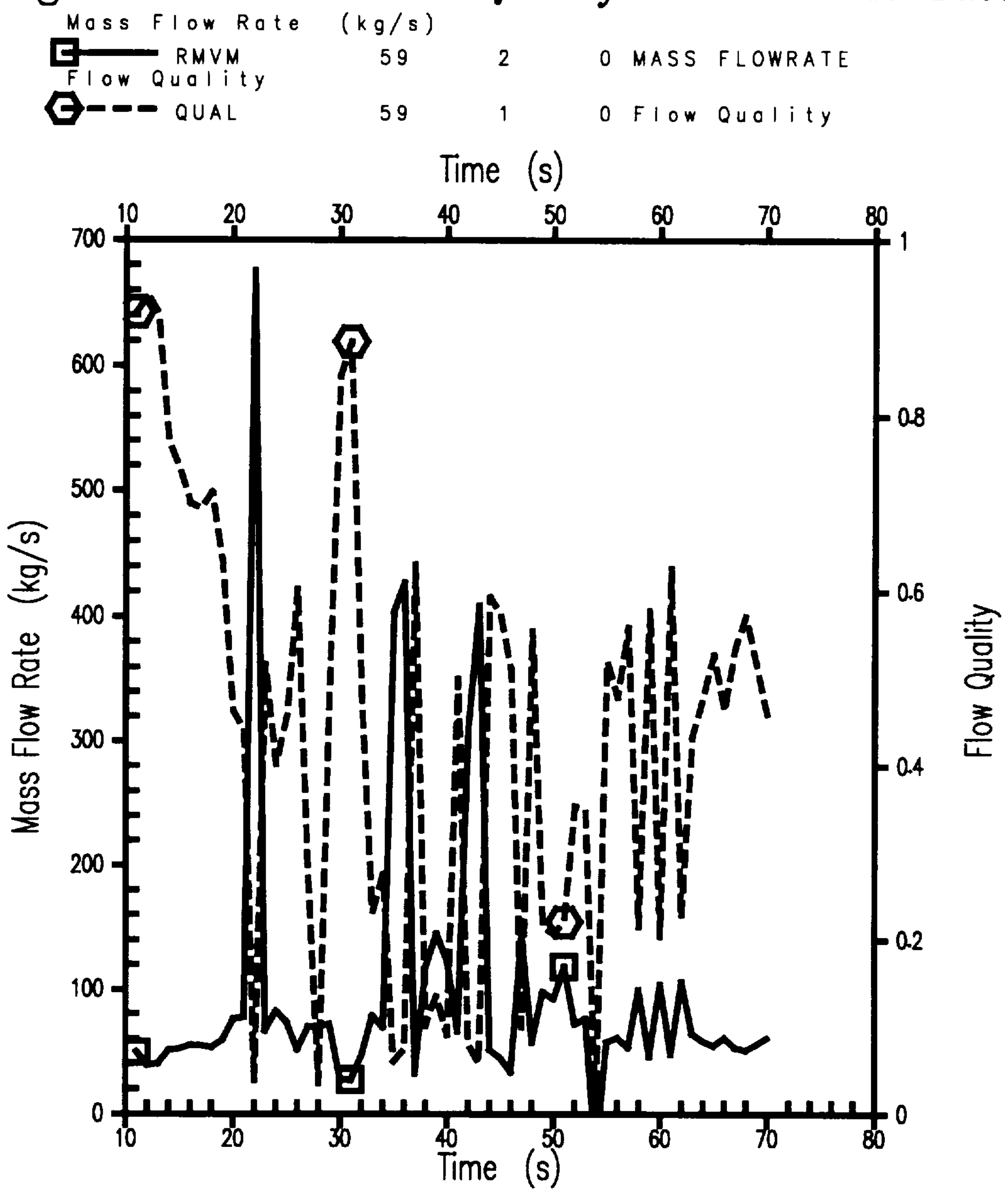


Figure 155.3: Liquid and Vapor Flow Upstream of Bend after the first ADS4 Opening

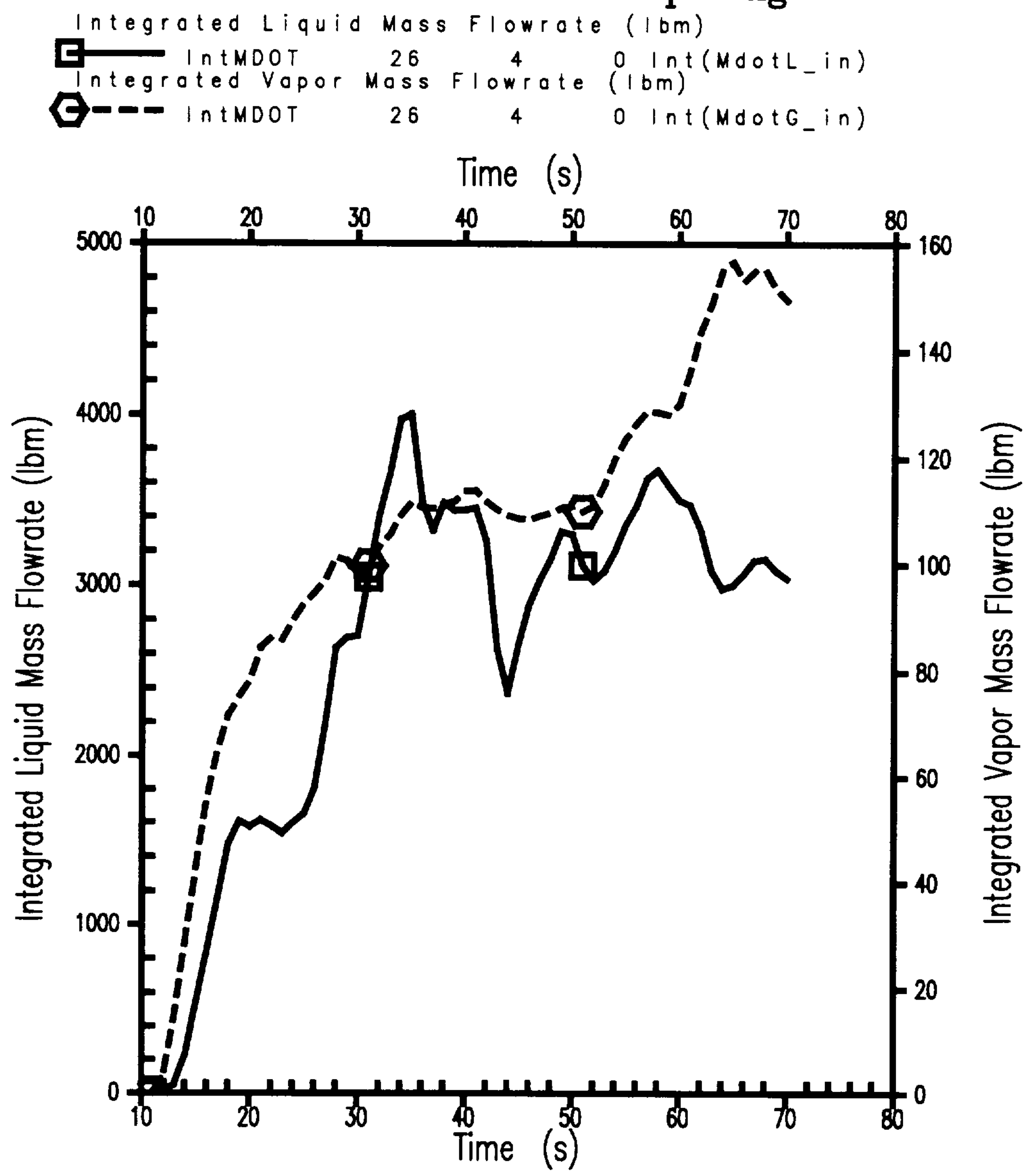


Figure 155.4: CCFL Curve and WCOBRA/TRAC Prediction

CCFL at Bend Comparison Between AP1000 and Ohnuki Correlation after opening of First ADS4

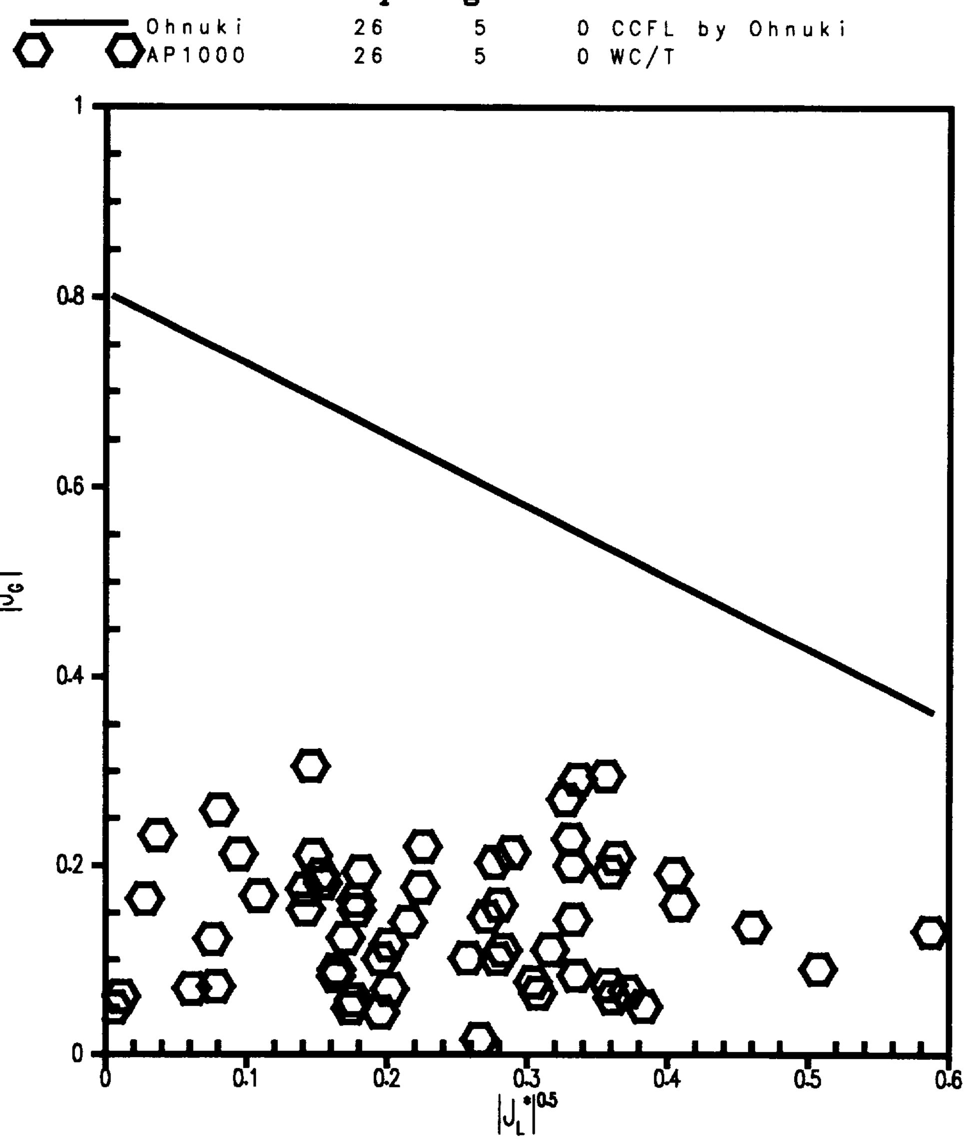


Figure 155.5: CCFL Curve and WCOBRA/TRAC Prediction
CCFL at Bend Comparison Between AP1000 and Ohnuki Correlation after opening of Second Set of ADS4

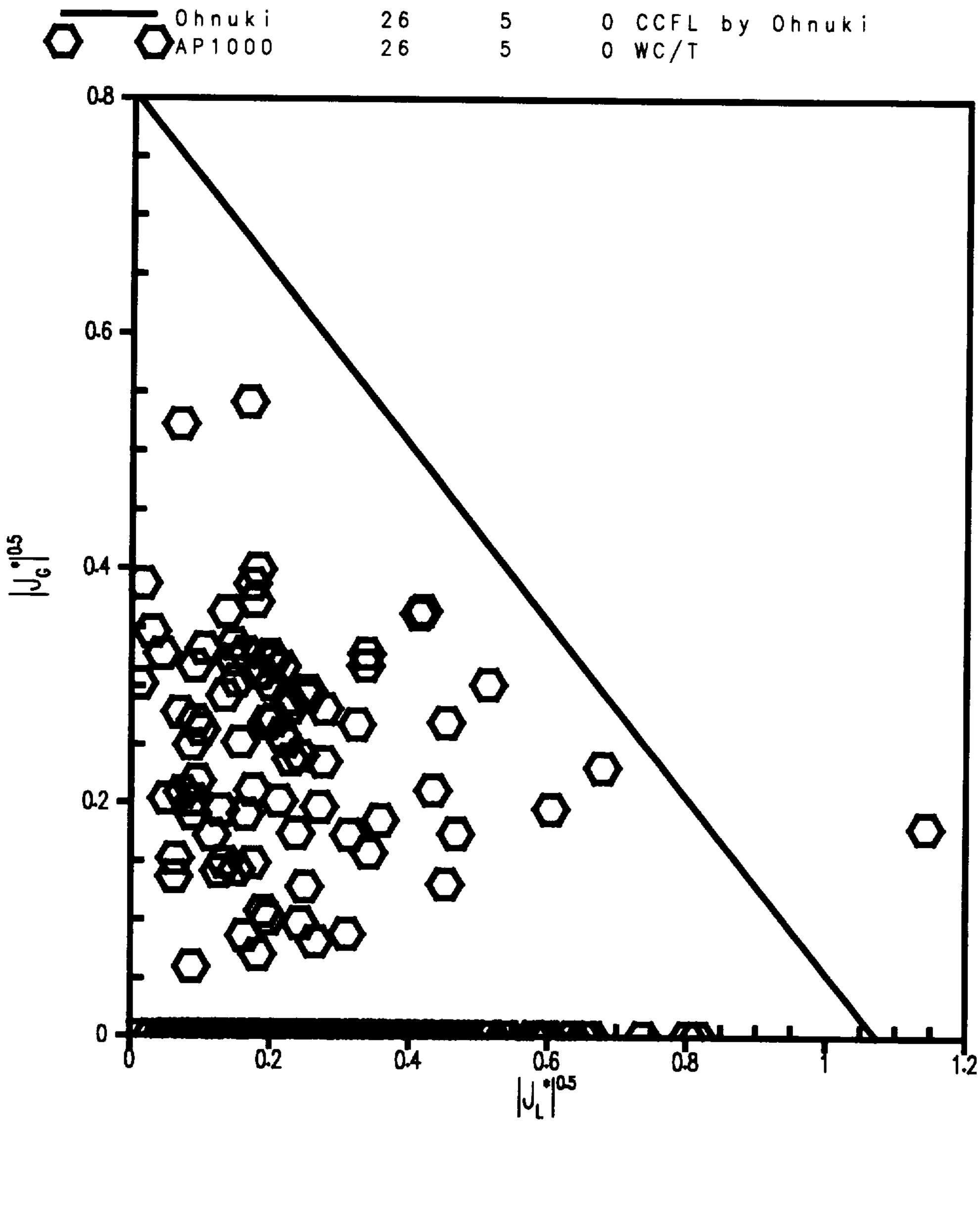
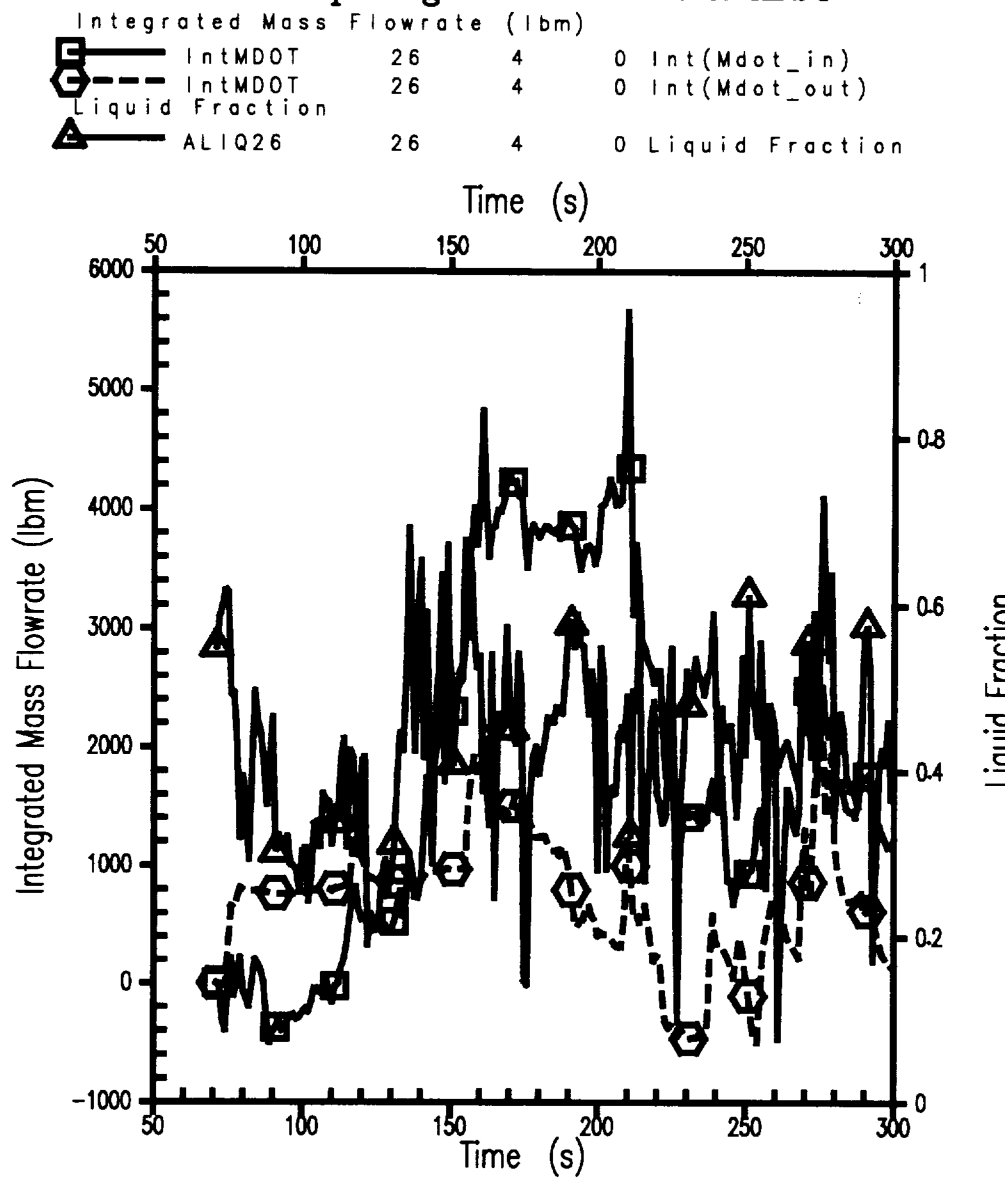


Figure 155.6: Liquid Flow Upstream and Downstream of Bend after Opening of Second Set of ADS4



Design Control Document (DCD) Revision:

None

PRA Revision:

None

RAI Number: 440.157

Question:

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

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

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

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

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

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

Reference

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

Westinghouse Response:

In section 2.2.1.4 of WCAP-15833 it was incorrectly stated that the size of the entrained droplets for the horizontal stratified flow is determined by Tatterson's model. The correlation used in WCT (Eq. 2-20 of WCAP-15833) to determine the characteristic size of the entrained droplet within the hot leg is not the same or equivalent to Tatterson's correlation.

Nevertheless the following discussion shows that the model used to determine the droplet size for the horizontal stratified flow regime is similar to the Tatterson's model. Moreover, for the expected conditions in the AP1000, the droplet size under question is a secondary effect such that any bias does not affect the system response significantly.

Within the code the droplet size is used to determine the interfacial area concentration source term based on Equation 4-125 of the CQD:

$$A'''_{i,E} = \frac{6S_E}{\rho_l D_{d,E}} - \frac{6S_{DE}}{\rho_l D_d}$$

After the solution of the interfacial area transport equation (Equation 3-72 CQD), the local droplet size is back calculated from the local interfacial area concentration as follows:

$$D_d = \frac{6\alpha_e}{A'''_i}$$

As a result the droplet size $D_{d,E}$ affects the local characteristic droplet size D_d depending on how significant is the source of entrainment in the horizonthally stratified hot leg relative to the net entrainment flow from the upper plenum. At the inlet boundary of the hot leg, the entrained field is characterized by a droplet size, which is calculated based on the entrainment process occuring in the upper plenum. Inside the hot leg the local droplet size can vary because of the entrainment process occuring within the horizontal stratified section of the hot leg. The interfacial area transport equation attempts to represent both population of droplets with a characteristic droplet size calculated from the solution of the interfacial area transport equation. When the entrainment flow rate at the boundary of the cell is dominant over the local source of entrainment and de-entrainment the effect of the droplet size $D_{d,E}$ is small.

Figure 1 shows the entrainment flow rate at the inlet of one of the hot legs during a time window after the opening of ADS 4. Figure 2 shows both the sources of entrainment and de-entrainment in Channel 24. Channel. 24 is the first hot leg channel and is located upstream of Channel. 25 which is connected to the ADS 4 line. The comparison in Figure 3 shows clearly that based on the WCT predictions the entrainment flow from the upper plenum is the dominant component. Predictions show that most of the liquid is in the dispersed form. Figure 4 shows that the continuous liquid field volume fraction (film) is very small through the transient. As a result, the effect of entrainment and de-entrainment in the hot leg is expected to be small, and the characteristic size of the droplet is also entrained within the hot leg is very small.

Measurements of drop size in horizontal two-phase flow are sparse and typically results for a vertical configuration are used. The droplet size predicted by Eq. 2-20 (Dwct) of WCAP-15833 was compared with the original Tatterson (Dtat) and two other recent correlations suggested for annular flow by L. Pan and Hanratty (2002) (D25 and D26). The calculation is provided in the attachment. In the calculation it was assumed that Dg~Dh. The droplet size calculated with Eq.

2-20 of WCAP-15833 is about 30% smaller than the one predicted with Tatterson. On the other hand the two correlations suggested by Pan and Hanratty (2002) give droplet size that are significantly smaller than Tatterson.

In conclusion, for the conditions expected in AP1000, the droplet size predicted with Eq. 2-20 of WCAP-15833 is in reasonable agreement with test data. Moreover any bias has a very small effect on the transient response because the source of entrainment within the hot leg is expected to be small, compared to the entrainment flow from the upper plenum is the dominant component.

Reference:

[2] Lei Pan, T. Hanratty, "Correlation of entrainement for annular flow in horizontal pipes", Int. J. of Multiphase Flow 28 (2002) 385-408.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None

ATTACHMENT

Entrainment in Horizontal Stratified Flow
- Droplet Size -

Inputs:

$wg := 50 \cdot \frac{lb}{s}$

Fluid Properties :

$p := 550000\ Pa$

$\rho g := 2.91 \cdot \frac{kg}{m^3}$ $\quad \rho f := 912 \cdot \frac{kg}{m^3}$ $\quad \mu g := 0.0000142 \cdot Pa \cdot s$ $\quad \mu f := 0.000176 \cdot Pa \cdot s$

$\sigma := 0.047 \cdot \frac{N}{m}$

Geometry:

$Dh := 0.74 \cdot m$

$\alpha := 0.8$ $\quad Ag := \alpha \cdot \left(\pi \cdot \frac{Dh^2}{4} \right)$ $\quad Dg := Dh$

Calculation:

$ug := \frac{wg}{\rho g \cdot Ag}$ $\quad ug = 22.652 \frac{m}{s}$

$fi := 0.046 \cdot \left(\frac{\rho g \cdot ug \cdot Dg}{\mu g} \right)^{-0.2}$ $\quad fi = 2.268 \times 10^{-3}$

$Dwct := 0.0112 \left(\frac{Dg \cdot \sigma}{0.5fi \cdot \rho g \cdot ug^2} \right)^{0.5}$ $\quad Dwct = 1.605 \times 10^{-3} m$

$Dtat := 0.016 \cdot Dh \cdot \left(\frac{\sigma}{0.5fi \cdot \rho g \cdot ug^2 \cdot Dh} \right)^{0.5}$ $\quad Dtat = 2.293 \times 10^{-3} m$

$D25 := \left(0.0091 \frac{\sigma \cdot Dh}{\rho g \cdot ug^2} \right)^{0.5}$ $\quad D25 = 4.604 \times 10^{-4} m$

$D26 := \left[0.14 \cdot \left(\frac{\sigma}{\rho f \cdot g} \right)^{0.5} \cdot \frac{\sigma}{\rho g \cdot ug^2} \right]^{0.5}$ $\quad D26 = 1.005 \times 10^{-4} m$

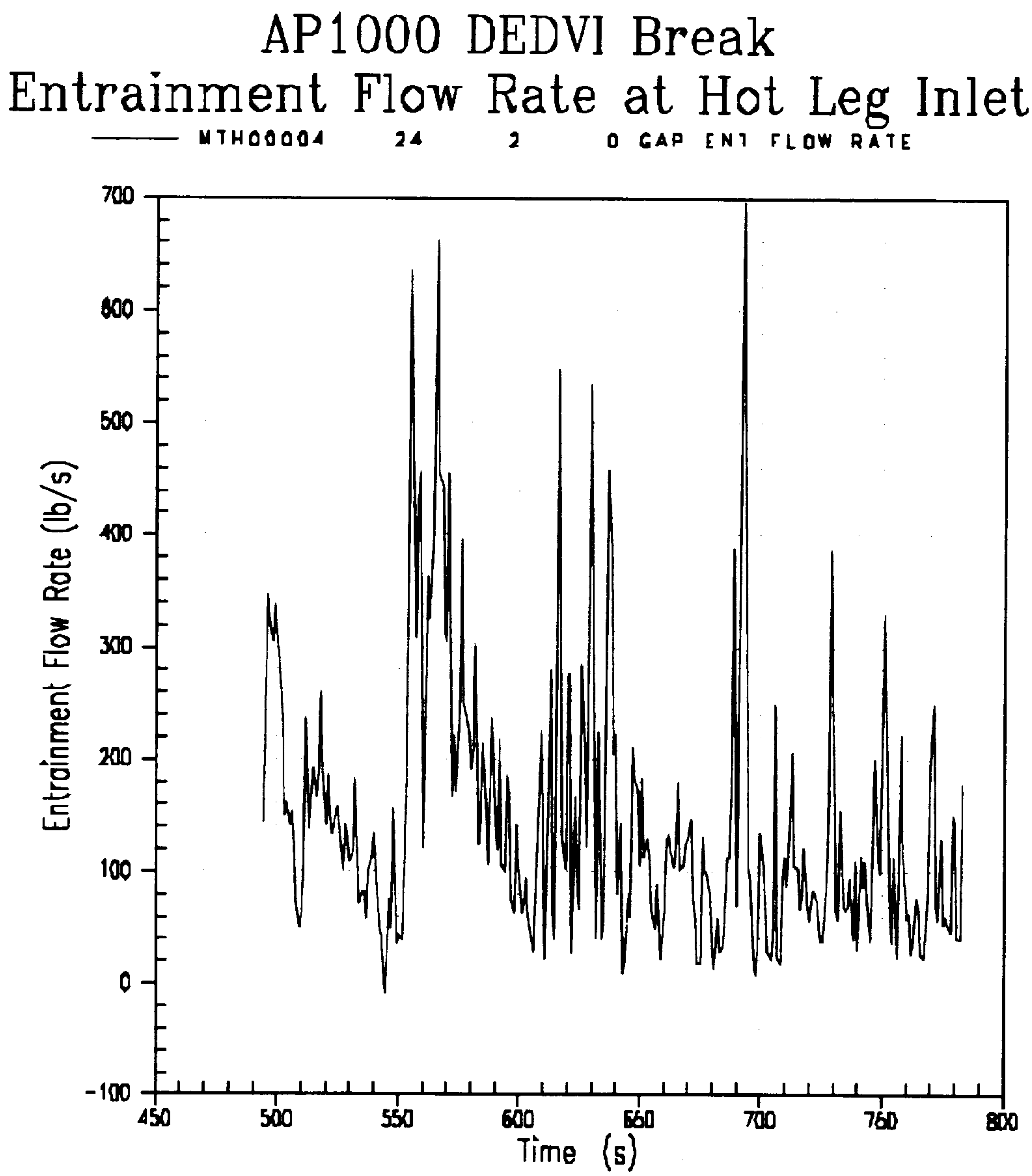

AP1000 DEDVI Break
Entrainment Flow Rate at Hot Leg Inlet
MTH00004 24 2 0 CAP ENT FLOW RATE
700
600
500
400
300
200
100
0
-100
Entrainment Flow Rate (lb/s)
450
500
550
600
650
700
750
800
Time (s)

Figure 1

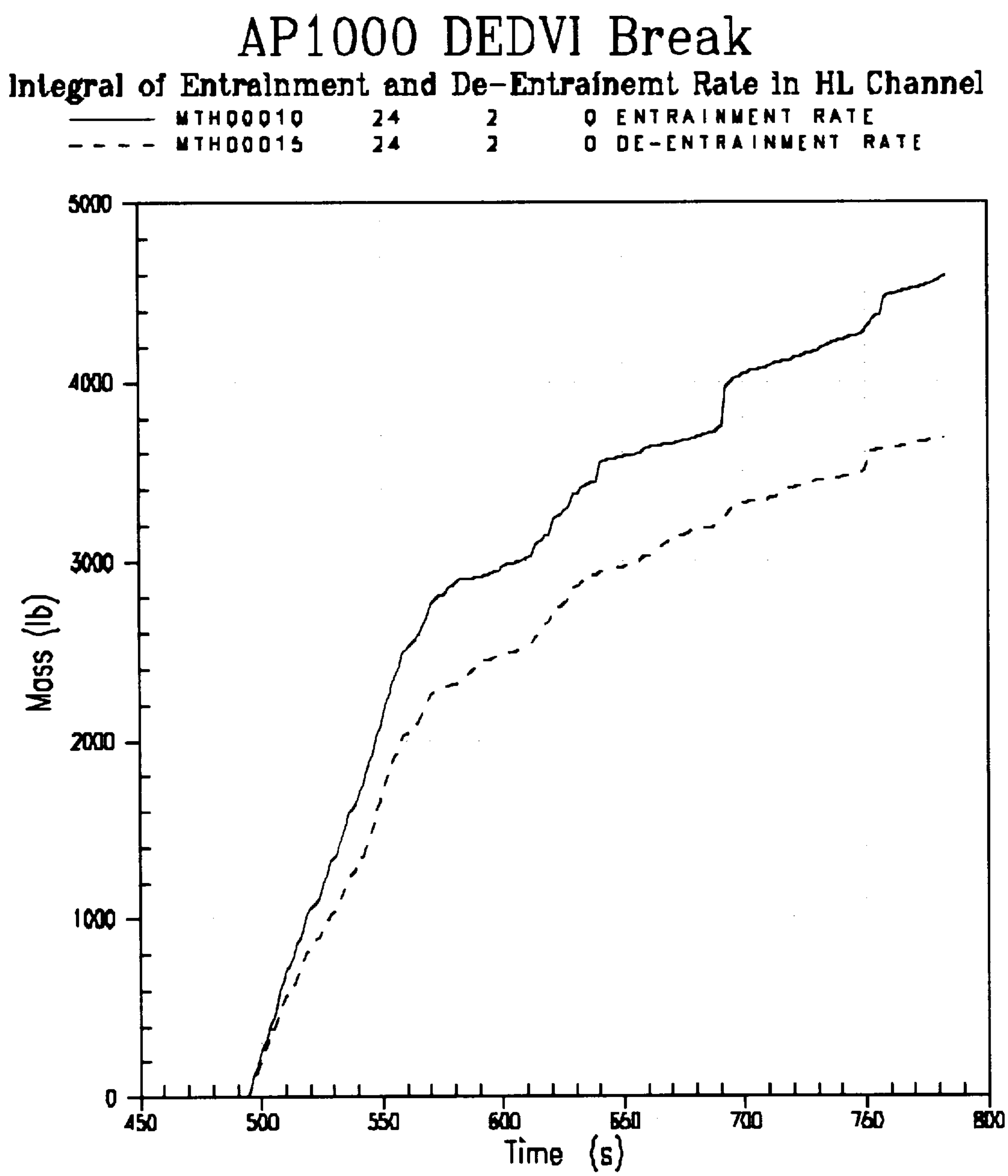
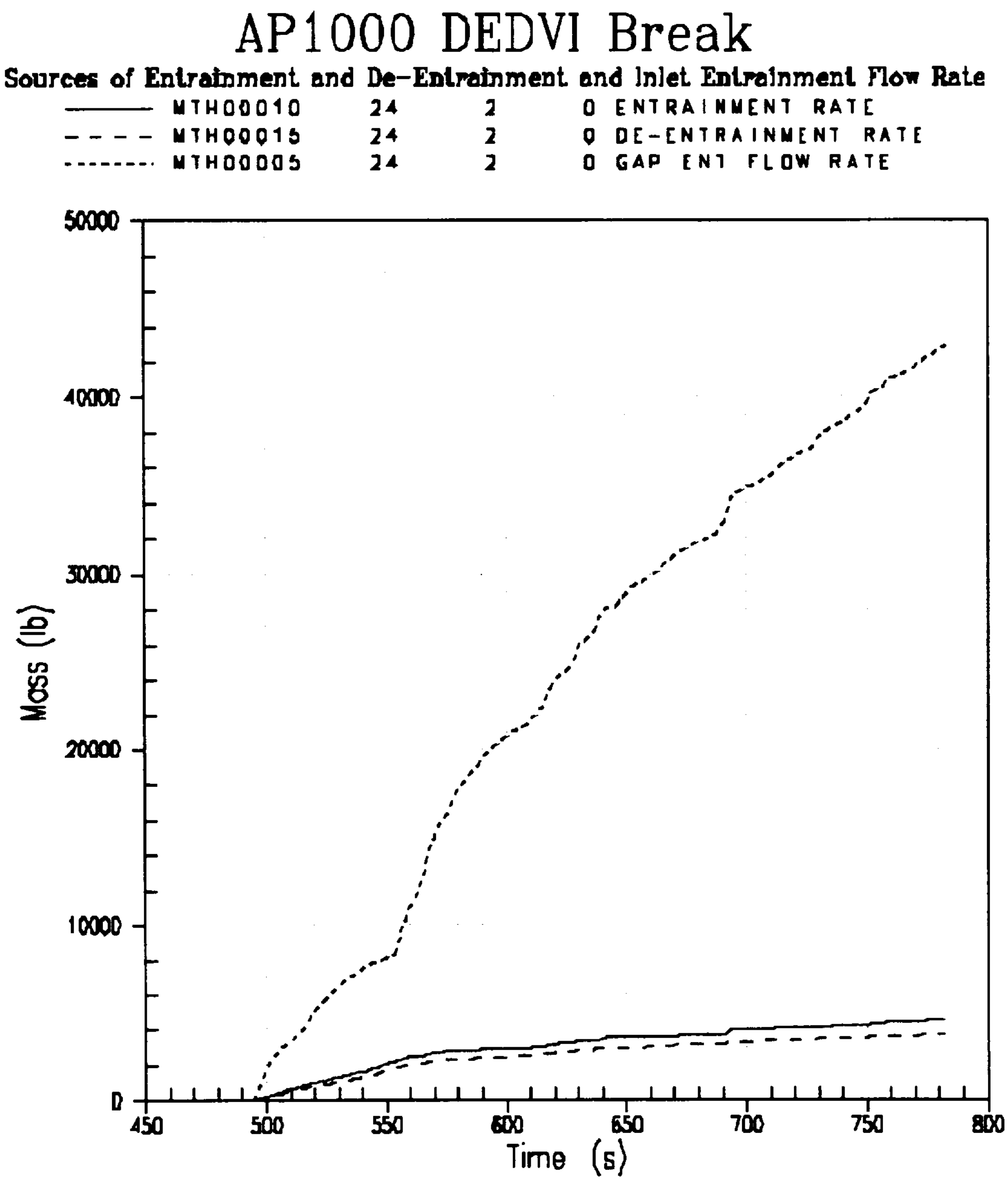
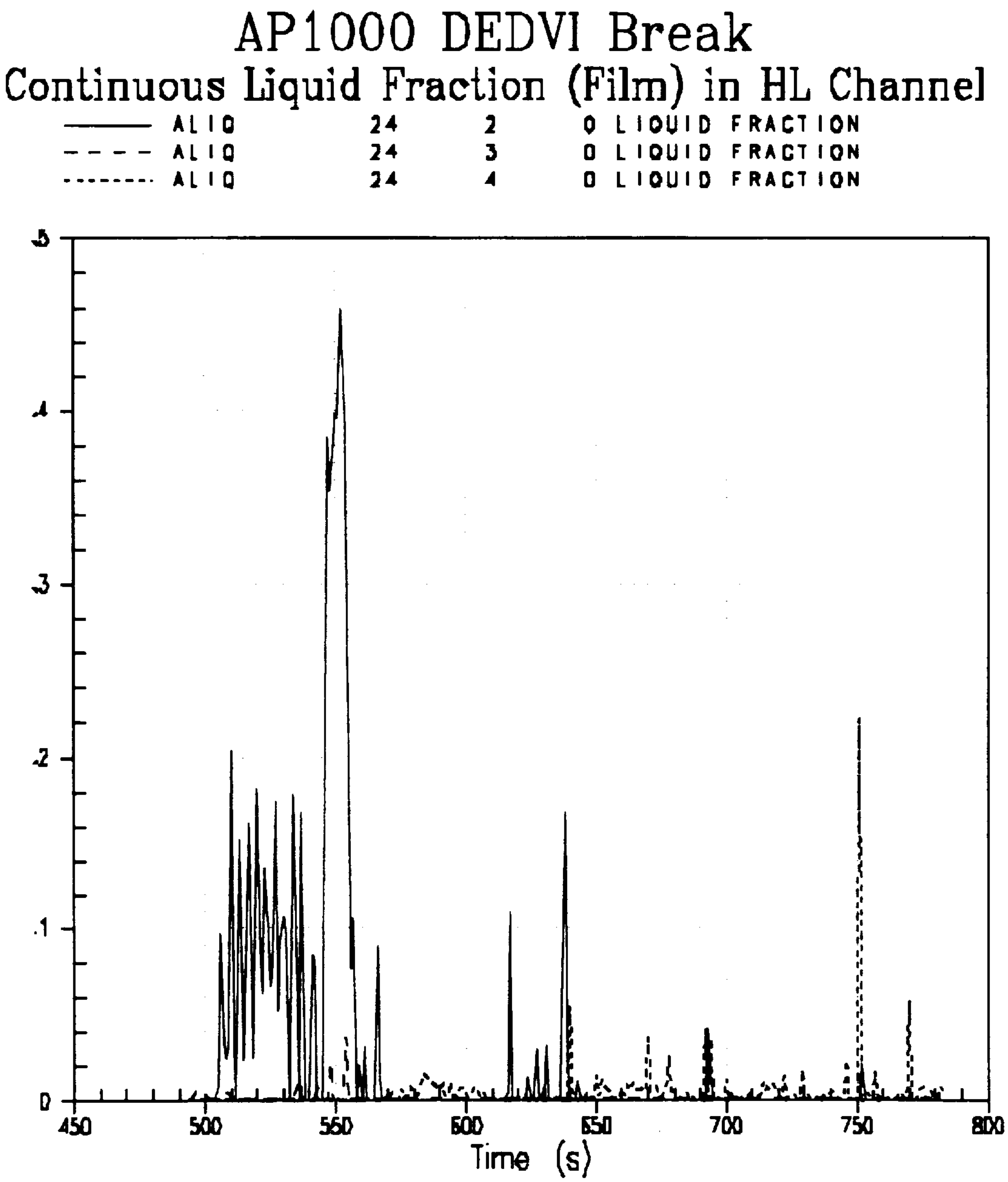

AP1000 DEDVI Break
Integral of Entrainment and De-Entrainemt Rate in HL Channel
MTH00010 24 2 0 ENTRAINMENT RATE
MTH00015 24 2 0 DE-ENTRAINMENT RATE
Mass (lb)
Time (s)
0
1000
2000
3000
4000
5000
450
500
550
600
650
700
750
800

Figure 2



AP1000 DEDVI Break
Sources of Entrainment and De-Entrainment and Inlet Entrainment Flow Rate
MTH00010 24 2 0 ENTRAINMENT RATE
MTH00015 24 2 0 DE-ENTRAINMENT RATE
MTH00005 24 2 0 GAP ENT FLOW RATE
Mass (lb)
50000
40000
30000
20000
10000
0
450
500
550
600
650
700
750
800
Time (s)

Figure 3



AP1000 DEDVI Break
Continuous Liquid Fraction (Film) in HL Channel
ALIQ 24 2 0 LIQUID FRACTION
ALIQ 24 3 0 LIQUID FRACTION
ALIQ 24 4 0 LIQUID FRACTION
.5
.4
.3
.2
.1
0
450
500
550
600
650
700
750
800
Time (s)

Figure 4

RAI Number 440.163

Question:

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

Westinghouse Response:

Consider first the DEDVI break simulation of Section 3.2.1 in WCAP-15833. Figure 440.163-1 shows that the steam flow entering the upper plenum from the core is affected somewhat by the opening of ADS-4 flow paths. The flow rate is 100 lbm/sec at 500 seconds, shortly after the first ADS-4 valve opens. It then diminishes until 553 seconds, when two additional ADS-4 valves open; at that time it increases to approximately 100 lbm/sec once again, after which it levels off at a rate of 70-80 lbm/sec. Overall, there is a significant steam flow entering the upper plenum throughout this phase of the DEDVI break transient.

Figure 440.163-2 shows the continuous liquid flow from the core into the upper plenum, and Figure 440.163-3 presents the entrained droplet flow into the upper plenum. During the initial part of the ADS-4 IRWST initiation phase, the continuous liquid flow fluctuates in direction at the core / upper plenum interface, with the flow integral upward in direction. After the latter two ADS valves open at 553 seconds, the continuous liquid field flow becomes uniform in direction, from the core into the upper plenum. Figure 440.163-3 shows the entrained liquid flow to be upward into the upper plenum most of the time during the ADS-4 IRWST initiation phase transient. Entrained liquid flow down into the core occurs on several occasions when the upper plenum liquid is at an increased collapsed level.

The steam entering the upper plenum leads WCOBRA/TRAC to predict entrainment according to the models detailed in WCAP-12945-P-A, Section 4-6-2. A discussion of the entrainment prediction in the upper plenum is provided in WCAP-15833, Appendix A3. Figure 440.163-4 shows that the entrainment rate predicted in the upper plenum in the initial portion of the ADS-4 IRWST initiation phase is very high. Liquid is carried by steam flow from the upper plenum into each hot leg. Figure 440.163-5 shows the lateral flow of steam from the upper plenum to the hot leg to which the pressurizer is connected. Figure 440.163-6 shows the lateral flow of continuous liquid from the upper plenum into this hot leg, and Figure 440.163-7 shows the lateral flow of entrained liquid from the upper plenum into this hot leg. Prior to 553 seconds in the transient, there is no ADS-4 flow path available in this hot leg; however, flow does proceed through this hot leg into the pressurizer and out through the open ADS Stage 1-3 valves. The composition of the flow entering this hot leg changes after 553 seconds, when the ADS-4 valve opens in this loop and a second ADS-4 valve opens in the other loop hot leg. When the additional ADS-4 flow paths open, both of the liquid flow fields, which have been small, increase in magnitude, then diminish again by 600 seconds when IRWST injection begins. The behavior at the entrance of this hot leg is essentially preserved in the lateral flows of each field in the hot leg just upstream of the ADS-4/hot leg junction. Figures 440.163-8 through 10 show the lateral flows of steam, continuous liquid and entrained liquid at this location.

Figure 440.163-11 shows the lateral flow of steam from the upper plenum to the hot leg in which an ADS-4 valve opens at 493 seconds. Figure 440.163-12 shows the lateral flow of continuous liquid from the upper plenum into this hot leg, and Figure 440.163-13 shows the lateral flow of entrained liquid from the upper plenum into this hot leg. Prior to 553 seconds, only this hot leg has an open ADS-4 valve. In contrast with the other loop, the composition of the flow entering this hot leg shows significant continuous and entrained liquid flows both before and after 553 seconds. When the second ADS-4 valve opens in this loop, the entrained flow shows a large increase. This is associated with the increase in steam flow entering this hot leg. As observed for the other hot leg, the behavior at the entrance of this hot leg is essentially preserved in the lateral flows of each field in the hot leg just upstream of the ADS-4/ hot leg junction. Figures 440.163-14 through 16 show the lateral flows of steam, continuous liquid and entrained liquid at this location.

Figure 440.163-17 presents the overall collapsed liquid level in each of the hot legs over the entire horizontal length at the vessel exit elevation during the ADS-4 IRWST initiation phase of the DEDVI break transient. The solid curve is the collapsed level in the hot leg to which the pressurizer is connected, and the dashed curve is the level in the hot leg in which an ADS-4 valve opens at 493 seconds. After a liquid level of approximately 0.6 feet is established in the pressurizer loop hot leg prior to the ADS-4 valve opening at 553 seconds, the collapsed liquid level reestablishes itself above 1.0 feet after the ADS-4 valve has opened. The behavior in vertical cells of the hot leg horizontal channels is shown in the void fraction plots of Figures 440.163-18 through 20. Figure 3-1 of WCAP-15833 provides the WCOBRA/TRAC VESSEL Component nodal network. Figure 440.163-18 shows that the liquid present in the bottom cell 2 of the hot leg channel 21 adjacent to the vessel upper plenum prior to the ADS-4 valve opening essentially disappears when the valve first opens, then later is reestablished with fluctuations. Figure 440.163-20 shows that while there is some liquid prior to the ADS-4 valve opening in the horizontal hot leg channel 23 that ultimately connects into the pressurizer, the overall channel 23 liquid fraction is about 0.6 after the ADS-4 flow path becomes available. The bottom cell of Channel 23 is eventually almost solid liquid. WCOBRA/TRAC is predicting a length-wise gradient in the pressurizer loop hot leg in which liquid is pushed against the boundary at the end of the horizontal pipe segment.

The dashed curve of Figure 440.163-17 is the collapsed level in the hot leg in which an ADS-4 valve opens at 493 seconds. The collapsed liquid level behavior in the vertical cells of hot leg horizontal channels is shown in the void fraction plots of Figures 440.163-21 through 23. These figures show a similar behavior while one ADS-4 valve is open from 493-553 seconds to that observed in the channels of the other loop hot leg after an ADS-4 valve opens there at 553 seconds; WCOBRA/TRAC predicts a length-wise gradient in which the liquid in the hot leg is pushed against the boundary at the end of the horizontal pipe segment. Once the second ADS-4 valve has opened, the void profile in the hot leg channels changes such that there is roughly half as much total liquid there, although the bottom cell of Channel 26 returns to largely liquid. After the second ADS-4 flow path becomes available, Figure 440.163-21 shows there is very little liquid in the hot leg channel adjacent to the reactor vessel, while the middle hot leg channel in Figure 440.163-22 exhibits an average void fraction that is intermediate between the hot leg channels present on either end. Figure 440.163-24 presents the void fraction in the vertical, angled hot leg pipe length adjacent to Channel 26. The bottom cell of Channel 27 has a significant liquid content while only one ADS-4 valve is open, but the top cell is almost all vapor. Once two ADS-4 flow paths are open at 553 seconds, there is little liquid content in Channel 27 until after IRWST injection commences. Channel 28, which is located above Channel 27 and connects into the steam generator plenum, contains almost no liquid through the time at which IRWST injection commences during the ADS-4 IRWST initiation phase, as shown in Figure 440.163-25.

Figures 440.163-26 and 27 present graphically the flow regimes predicted by WCOBRA/TRAC in the pressurizer loop hot leg and in the other hot leg, respectively, for the lateral flow through the vessel side of the channel connected to the ADS-4 flow path. Figure 440.163-26 indicates that there is significant intermittent countercurrent flow in the pressurizer loop hot leg during the ADS-4 IRWST initiation phase of the DEDVI break prior to the time at which the ADS-4 valve opens, even though the pressurizer does not drain. Once the ADS-4 valve opens, horizontal stratified flow alone is indicated until IRWST injection has begun. Figure 440.163-27 shows horizontal stratified flow is predicted in the other hot leg during almost all of the ADS-4 IRWST initiation phase; at least one, and ultimately two ADS-4 flow paths, are open in this loop at all times depicted.

Consider next the Inadvertent ADS actuation simulation of Section 3.2.2 in WCAP-15833. Figure 440.163-1a shows that the steam flow entering the upper plenum from the core is affected somewhat by the opening of ADS-4 flow paths. The flow rate is 60 lbm/sec at 1750 seconds, shortly after the first ADS-4 valve opens. It is relatively constant until 1806 seconds, when two additional ADS-4 valves open; then it increases to over 80 lbm/sec and subsequently diminishes. Overall, there is a significant steam flow entering the upper plenum throughout this phase of the transient.

Figure 440.163-2a shows the continuous liquid flow from the core into the upper plenum, and Figure 440.163-3a presents the entrained droplet flow into the upper plenum. During the initial part of the ADS-4 IRWST initiation phase, the continuous liquid flow fluctuates in direction at the core / upper plenum interface. After the latter two ADS valves open at 1806 seconds, the continuous liquid field flow becomes essentially uniform in direction, from the core into the upper plenum. Figure 440.163-3a shows the entrained liquid flow to be upward into the upper plenum most of the time during the transient. There are occasions of entrained liquid flow down into the core when the upper plenum liquid is at an increased collapsed level.

The steam entering the upper plenum leads WCOBRA/TRAC to predict entrainment according to the models detailed in WCAP-12945-P-A, Section 4-6-2. A discussion of the entrainment prediction in the upper plenum is provided in WCAP-15833, Appendix A3. Figure 440.163-4a shows that the entrainment rate predicted in the upper plenum in the initial portion of the ADS-4 IRWST initiation phase is very high. Liquid is carried by steam flow from the upper plenum into each hot leg. Figure 440.163-5a shows the lateral flow of steam from the upper plenum to the hot leg to which the pressurizer is connected. Figure 440.163-6a shows the lateral flow of continuous liquid from the upper plenum into this hot leg, and Figure 440.163-7a shows the lateral flow of entrained liquid from the upper plenum into this hot leg. Prior to 1806 seconds in the transient, there is no ADS-4 flow path available in this hot leg; however, flow does proceed through this hot leg into the pressurizer and out through the open ADS Stage 1-3 valves. The composition of the flow entering this hot leg changes after 1806 seconds, when the ADS-4 valve opens in this loop and the vapor flow rate into the hot leg increases. When the additional ADS-4 flow paths open, the continuous liquid flow field, which has intermittently flowed into and out of the upper plenum, mainly proceeds from the upper plenum into the hot leg beyond the inception of IRWST injection at 1890 seconds. The magnitude of the entrained liquid flow rate increases along with the vapor flow rate through the time at which IRWST injection begins. The behavior at the entrance of this hot leg is essentially preserved in the lateral flows of each field in the hot leg just upstream of the ADS-4/hot leg junction. Figures 440.163-8a through 10a show the lateral flows of steam, continuous liquid and entrained liquid at this location.

Figure 440.163-11a shows the lateral flow of steam from the upper plenum to the hot leg in which an ADS-4 valve opens at 1746 seconds. Figure 440.163-12a shows the lateral flow of continuous liquid from the upper plenum into this hot leg, and Figure 440.163-13a shows the lateral flow of entrained liquid from the upper plenum into this hot leg. Prior to 1806 seconds, only this hot leg has an open ADS-4 valve. In contrast with the other loop, the composition of the flow entering this hot leg shows a more established continuous and entrained liquid flow rate before 1806 seconds. When the second ADS-4 valve opens in this loop, the entrained flow shows a large increase. This is associated with the increase in vapor flow entering this hot leg. As observed for the other hot leg, the behavior at the entrance of this hot leg is essentially preserved in the lateral flows of each field in the hot leg just upstream of the ADS-4/ hot leg junction. Figures 440.163-14a through 16a show the lateral flows of steam, continuous liquid and entrained liquid at this location.

Figure 440.163-17a presents the overall collapsed liquid level in each of the hot legs over the entire horizontal length at the vessel exit elevation during the ADS-4 IRWST initiation phase of the DEDVI break transient. The solid curve is the collapsed level in the hot leg to which the pressurizer is connected, and the dashed curve is the level in the hot leg in which an ADS-4 valve opens at 1746 seconds. The liquid level in the pressurizer loop hot leg remains similar both before and after the ADS-4 valve opens at 1806 seconds. The behavior in vertical cells of hot leg horizontal channels is shown in the void fraction plots of Figures 440.163-18a through 20a. Figure 3-1 of WCAP-15833 provides the WCOBRA/TRAC VESSEL Component nodal network. Figure 440.163-18a shows that the liquid present in the bottom cell 2 of the hot leg channel 21 adjacent to the vessel upper plenum prior to the ADS-4 valve opening decreases once the valve has opened, then later fluctuates from almost all vapor to almost all liquid. Otherwise, the opening of the ADS-4 valve has little impact on the void fractions in the horizontal hot leg channel cells of the pressurizer loop; WCOBRA/TRAC predicts a lesser length-wise liquid gradient in the hot leg horizontal pipe segment than in the DEDVI break case.

The dashed curve of Figure 440.163-17a is the collapsed level in the hot leg in which an ADS-4 valve opens at 1746 seconds. The collapsed liquid level behavior in vertical cells of hot leg horizontal channels is shown in the void fraction plots of Figures 440.163-21a through 23a. These figures show a similar behavior to the corresponding figures from the DEDVI break case; WCOBRA/TRAC predicts a length-wise gradient in which the liquid in the hot leg is pushed against the boundary at the end of the horizontal pipe segment. When the second ADS-4 valve has opened, the void profile in the hot leg channels changes such that there is roughly half as much total liquid there. Figure 440.163-24a presents the void fraction in the vertical, angled hot leg pipe length adjacent to Channel 26. Neither the bottom cell nor the top cell of Channel 27 has a significant liquid during the entirety of the ADS-4 IRWST initiation phase. Channel 28, which is located above Channel 27 and connects into the steam generator plenum, contains virtually no liquid during the entire ADS-4 IRWST initiation phase, as shown in Figure 440.163-25a.

Figures 440.163-26a and 27a present graphically the flow regimes in the pressurizer loop hot leg and in the other hot leg, respectively, predicted by WCOBRA/TRAC for the lateral flow through the vessel side of the channel connected to the ADS-4 flow path. Figure 440.163-26a indicates that there is a similar amount of intermittent countercurrent flow predicted in the pressurizer loop hot leg during the ADS-4 IRWST initiation phase of this case as for the DEDVI break. Figure 440.163-27a shows that significantly more intermittent countercurrent flow is predicted in the other hot leg during the ADS-4 IRWST initiation phase for this case than for the DEDVI break case. This is likely a consequence of the lower hot leg steam flow rate observed in this case.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None

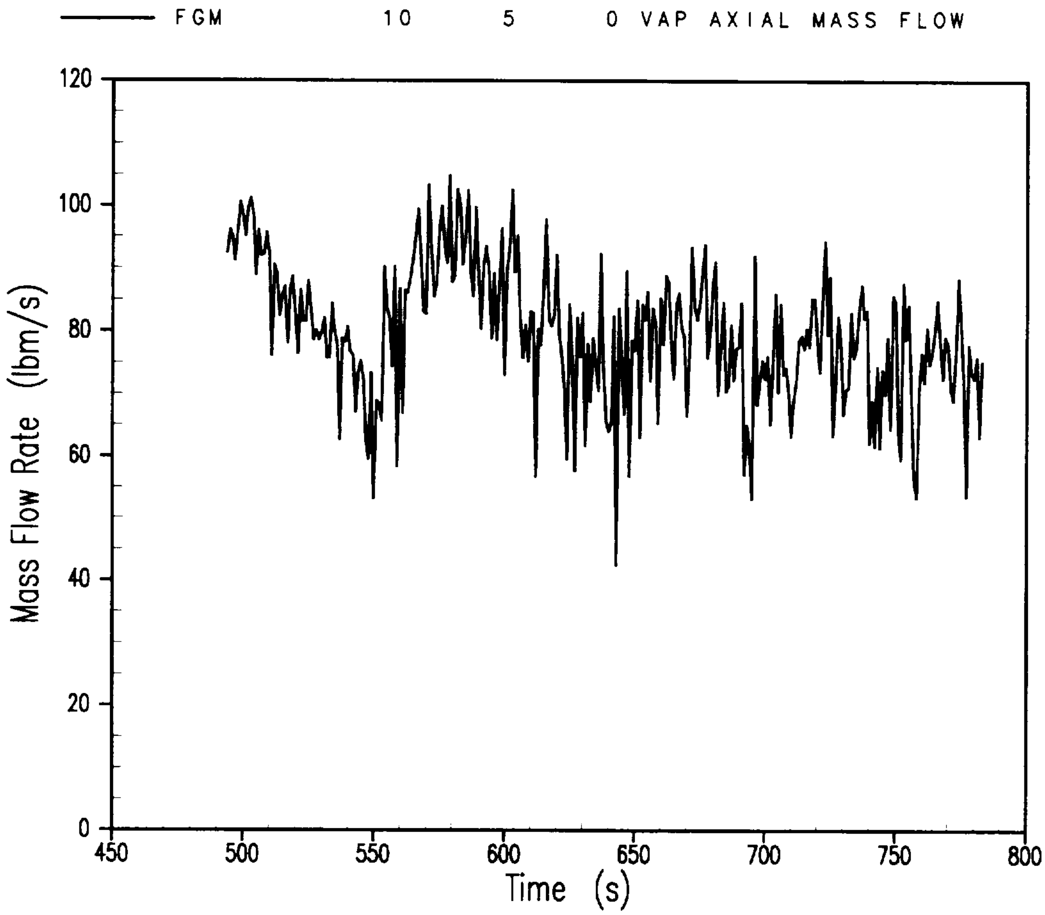


Figure 440.163-1
Core Exit Steam Flow Rate, DEDVI Break

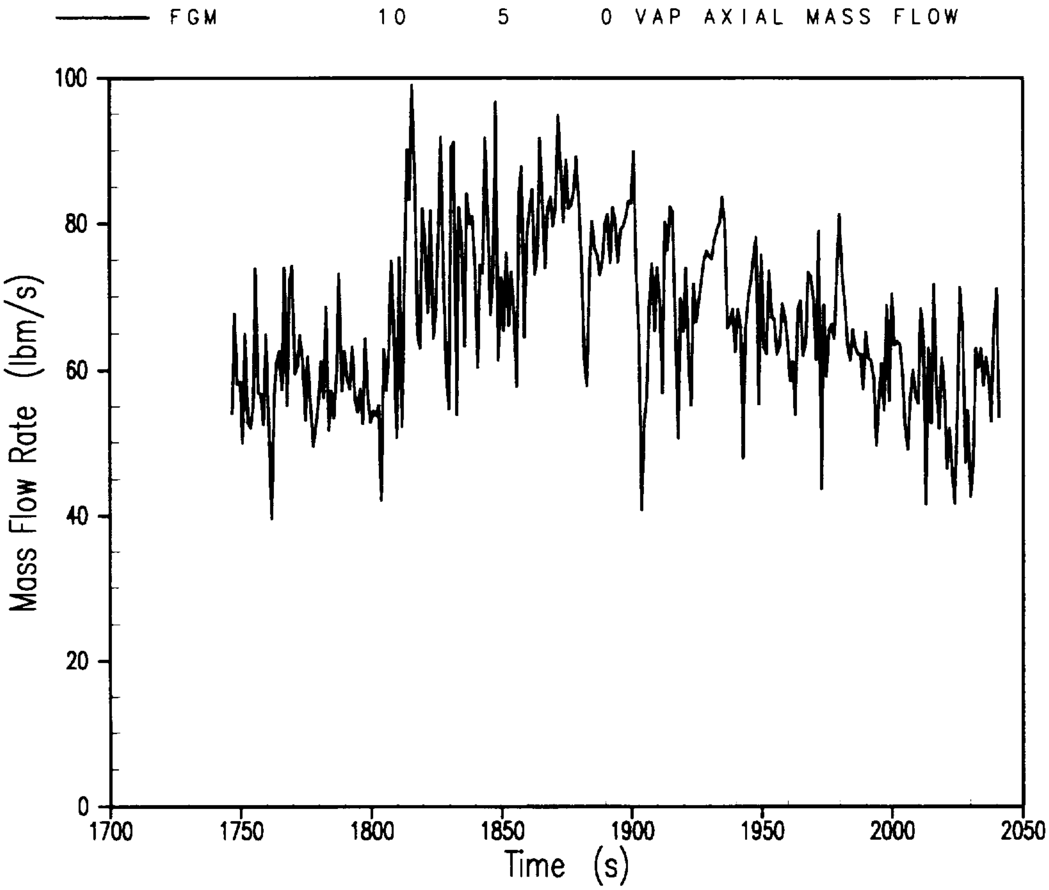


Figure 440.163-1a
Core Exit Steam Flow Rate, Inadvertant ADS Case

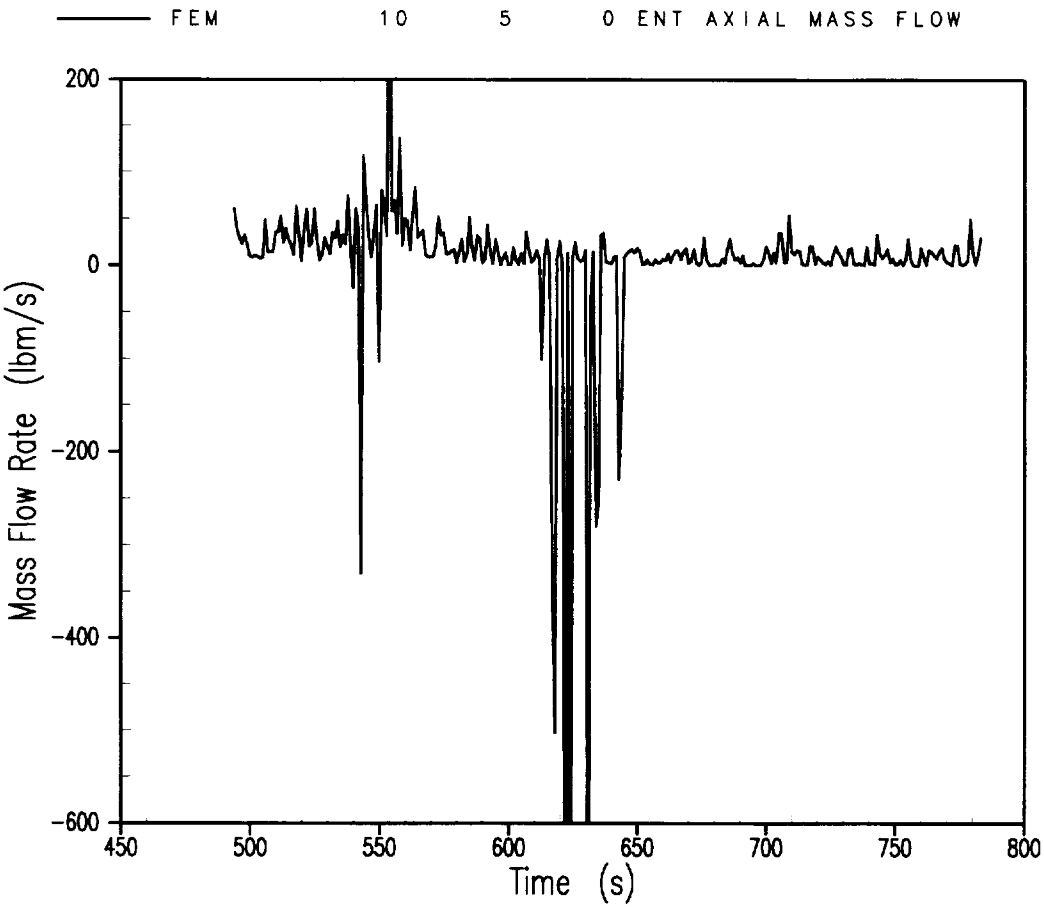


Figure 440.163-3
Core Exit Entrained Liquid Flow Rate, DEDVI Break

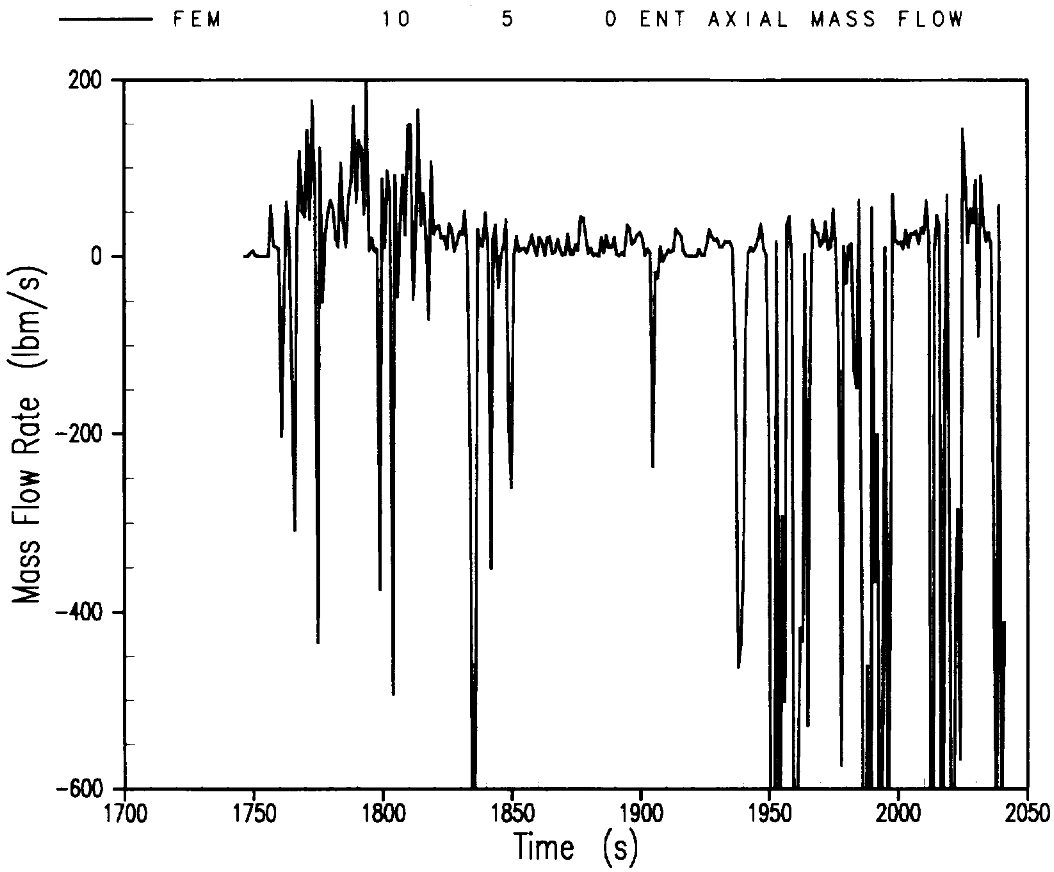


Figure 440.163-3a
Core Exit Entrained Liquid Flow Rate, Inadvertant ADS Case

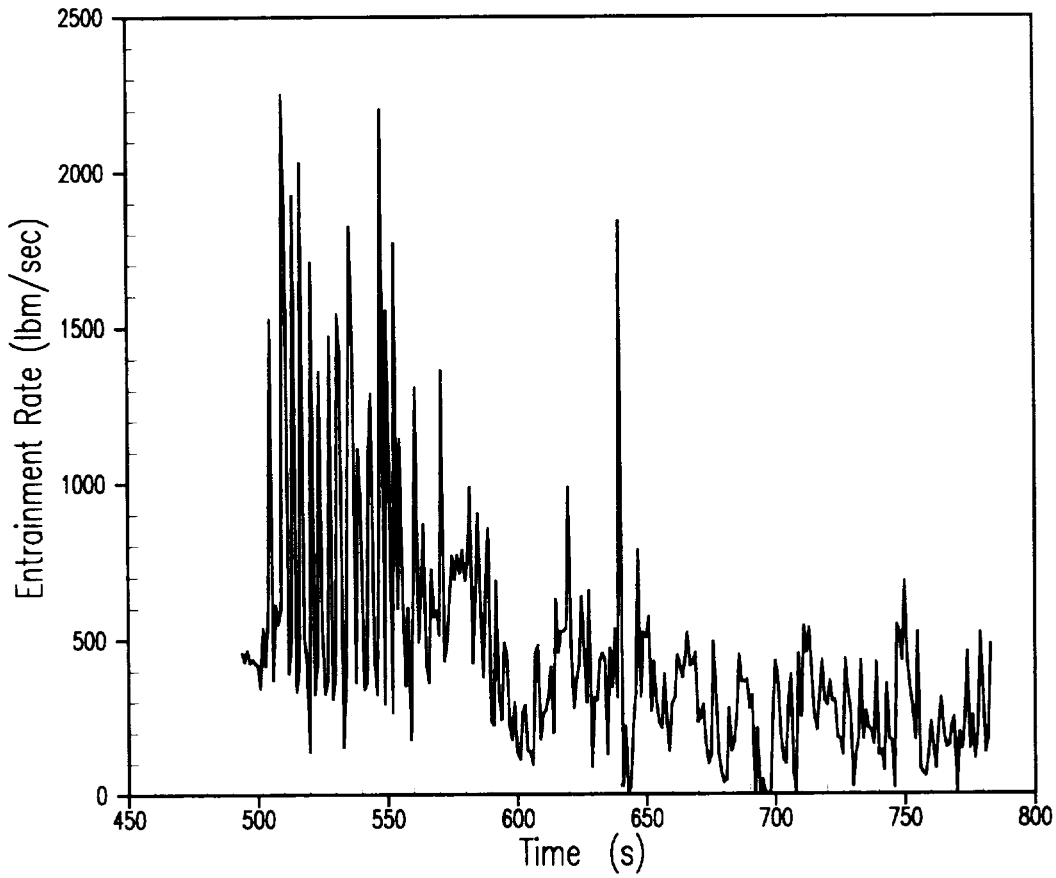


Figure 440.163-4
DEDVI Break Upper Plenum Entrainment Rate vs. Time

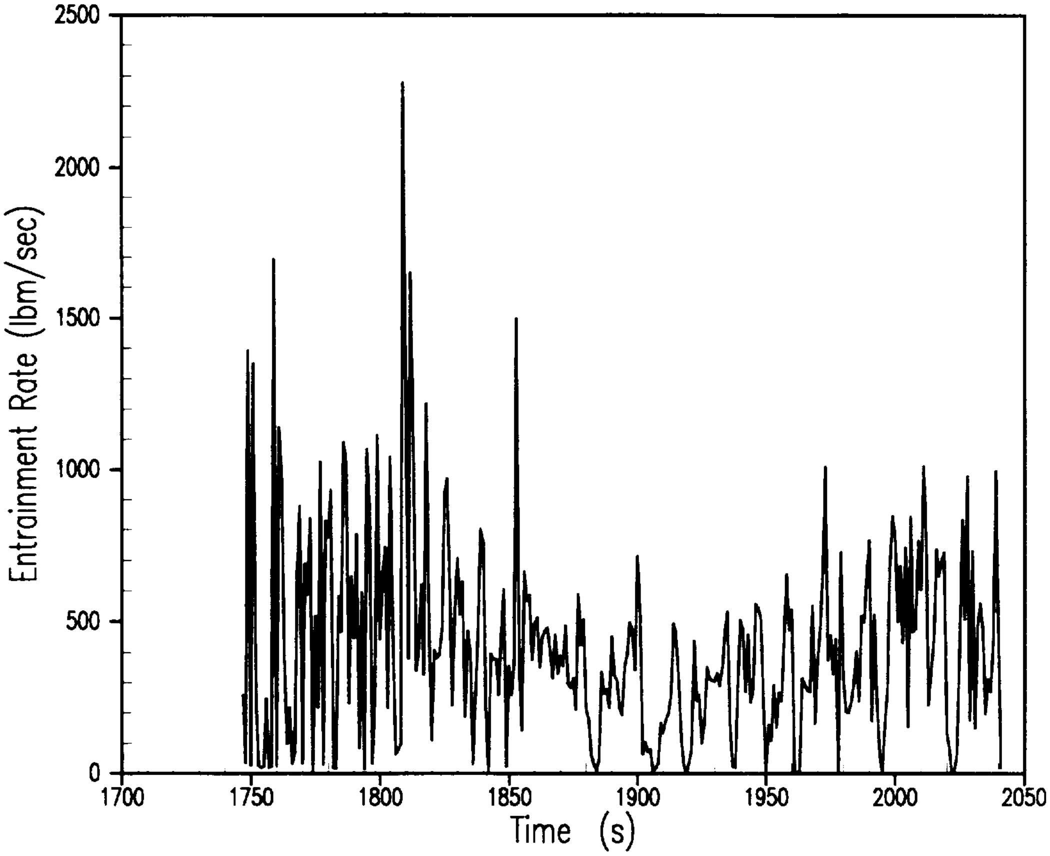


Figure 440.163-4a
Inadvertant ADS Case Upper Plenum Entrainment Rate vs. Time

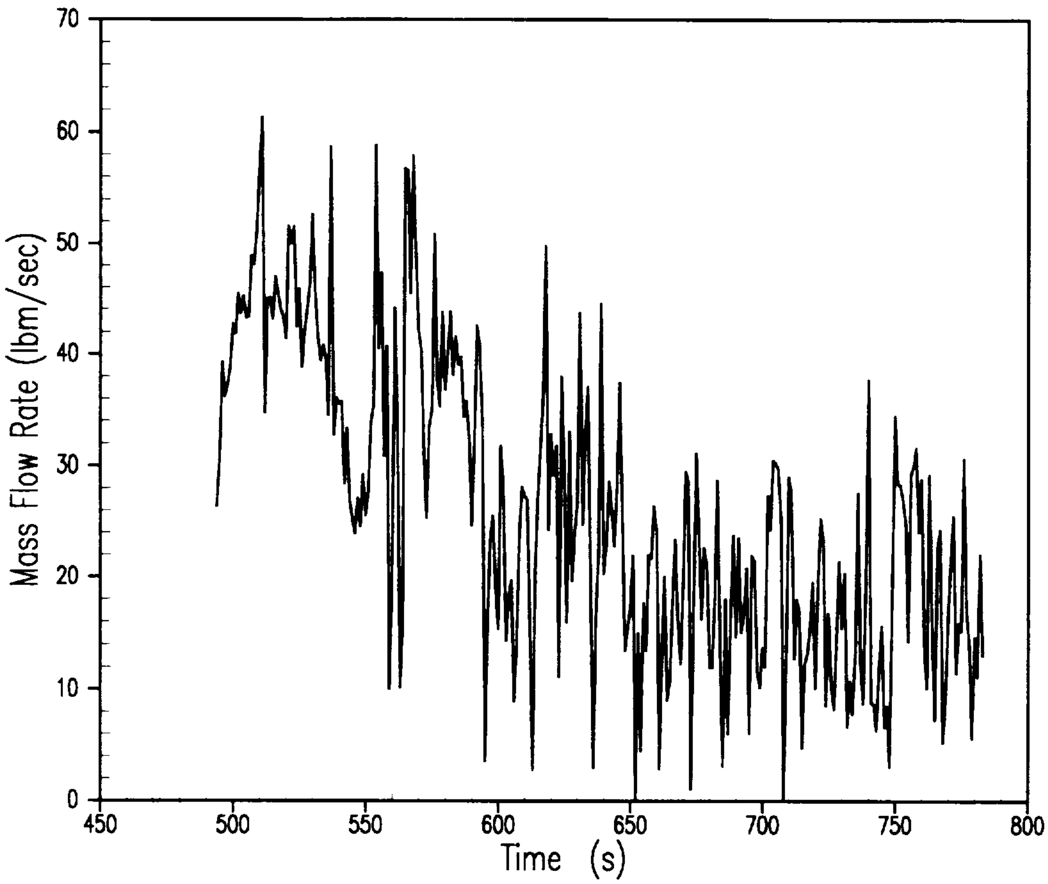


Figure 440.163-5
Vapor Flow Entering the Pressurizer Loop Hot Leg, DEDVI Break

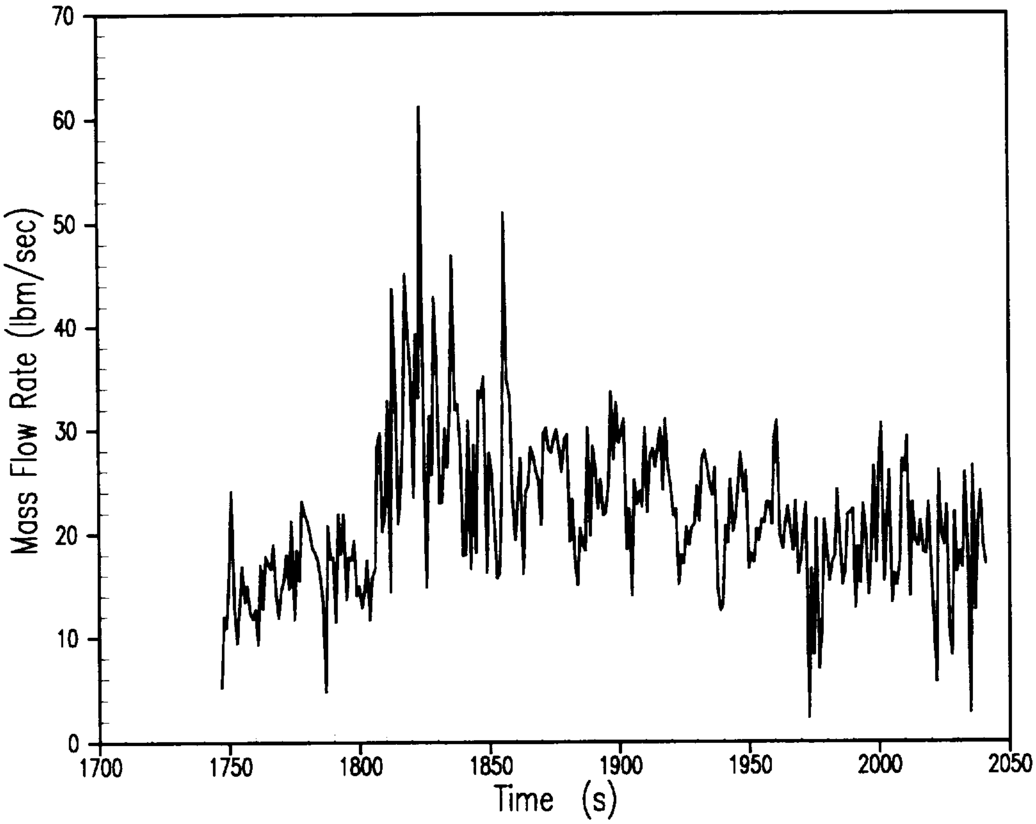


Figure 440.163-5a
Vapor Flow Entering the Pressurizer Loop Hot Leg, Inadvertant ADS Case

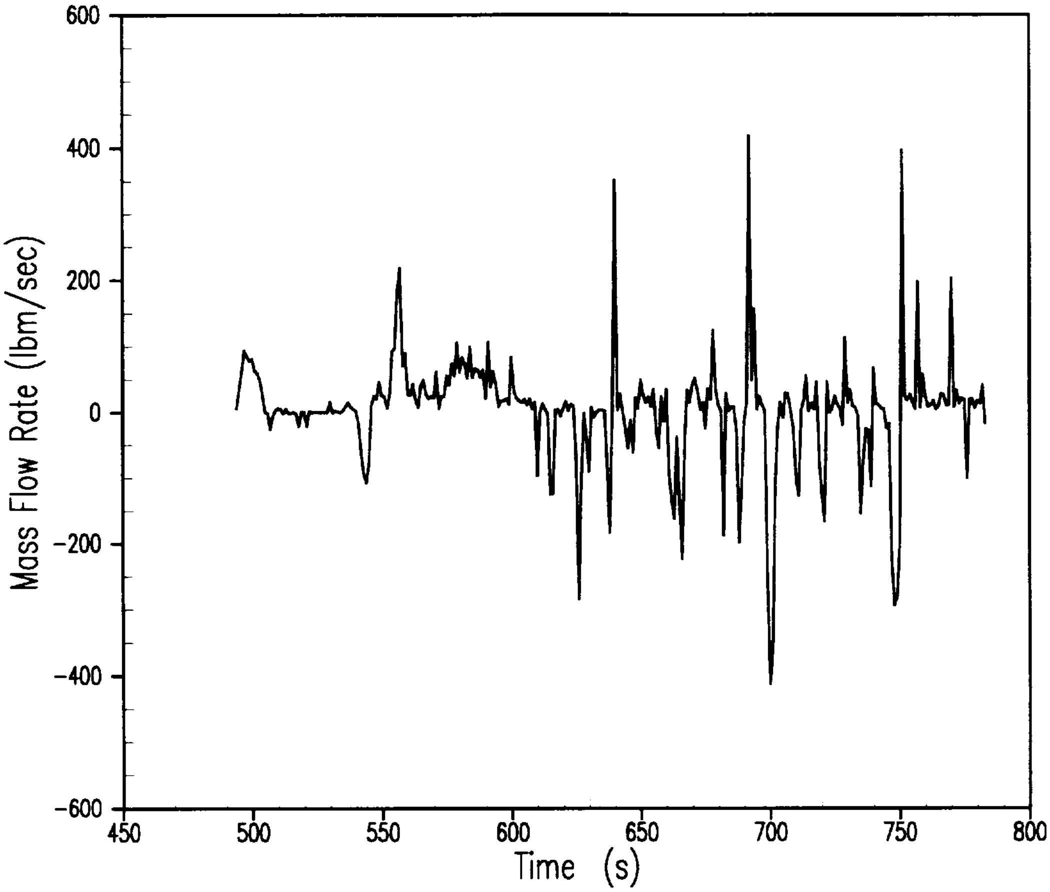


Figure 440.163-6
Continuous Liquid Flow Entering the Pressurizer Loop Hot Leg, DEDVI Break

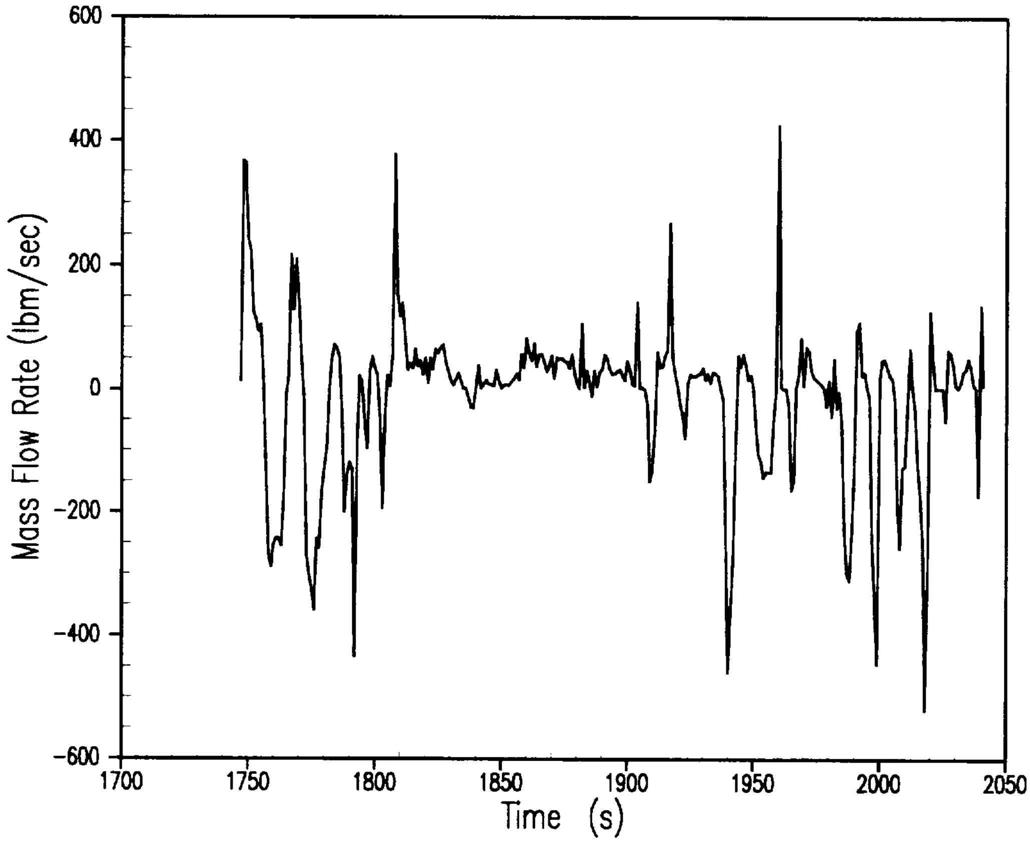


Figure 440.163-6a
Continuous Liquid Flow Entering the Pressurizer Loop Hot Leg, Inadvertant ADS Case

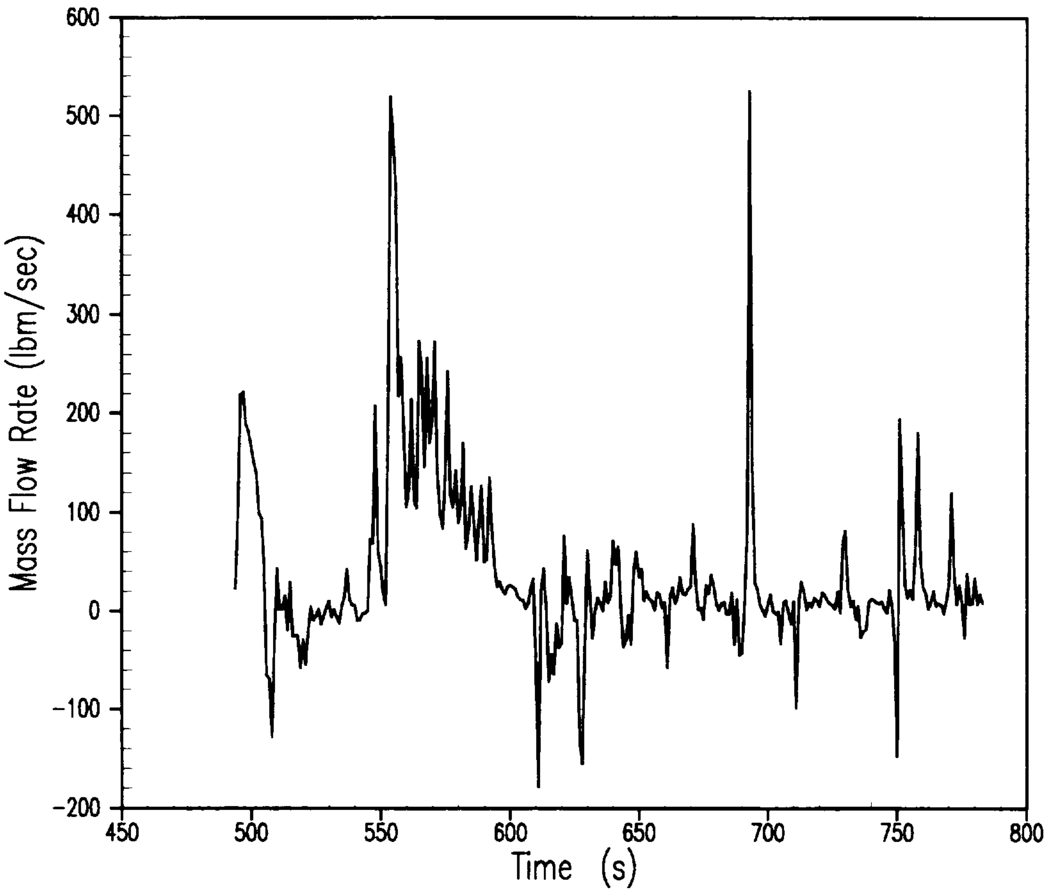


Figure 440.163-7
Entrained Liquid Flow Entering the Pressurizer Loop Hot Leg, DEDVI Break

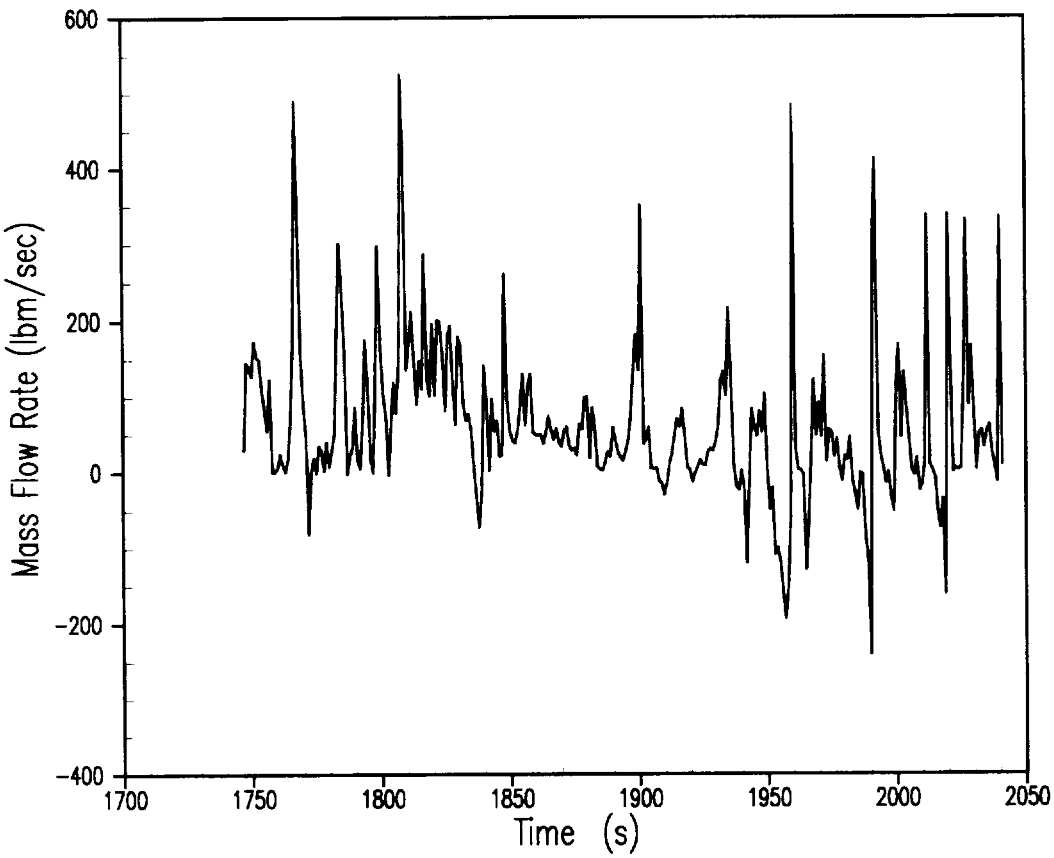


Figure 440.163-7a
Entrained Liquid Flow Entering the Pressurizer Loop Hot Leg, Inadvertant ADS Case

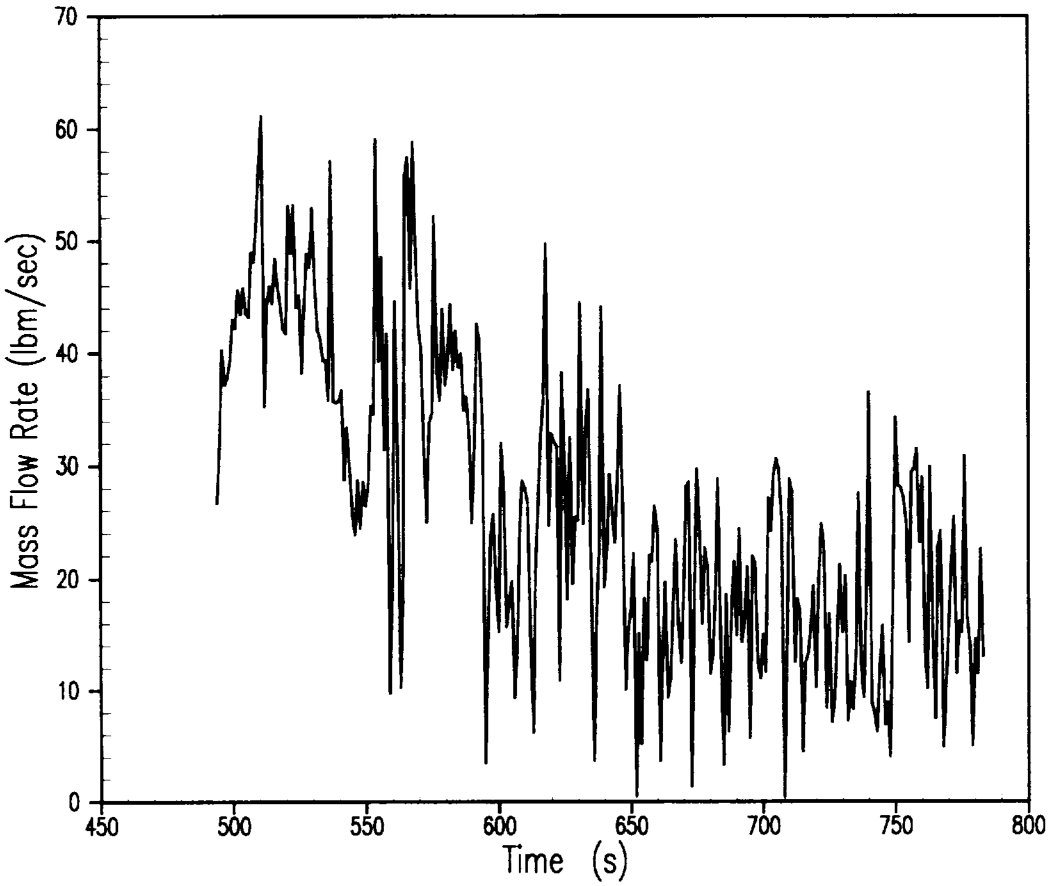


Figure 440.163-8
Vapor Flow Upstream of the Pressurizer Loop ADS-4 / Hot Leg Junction, DEDVI Break

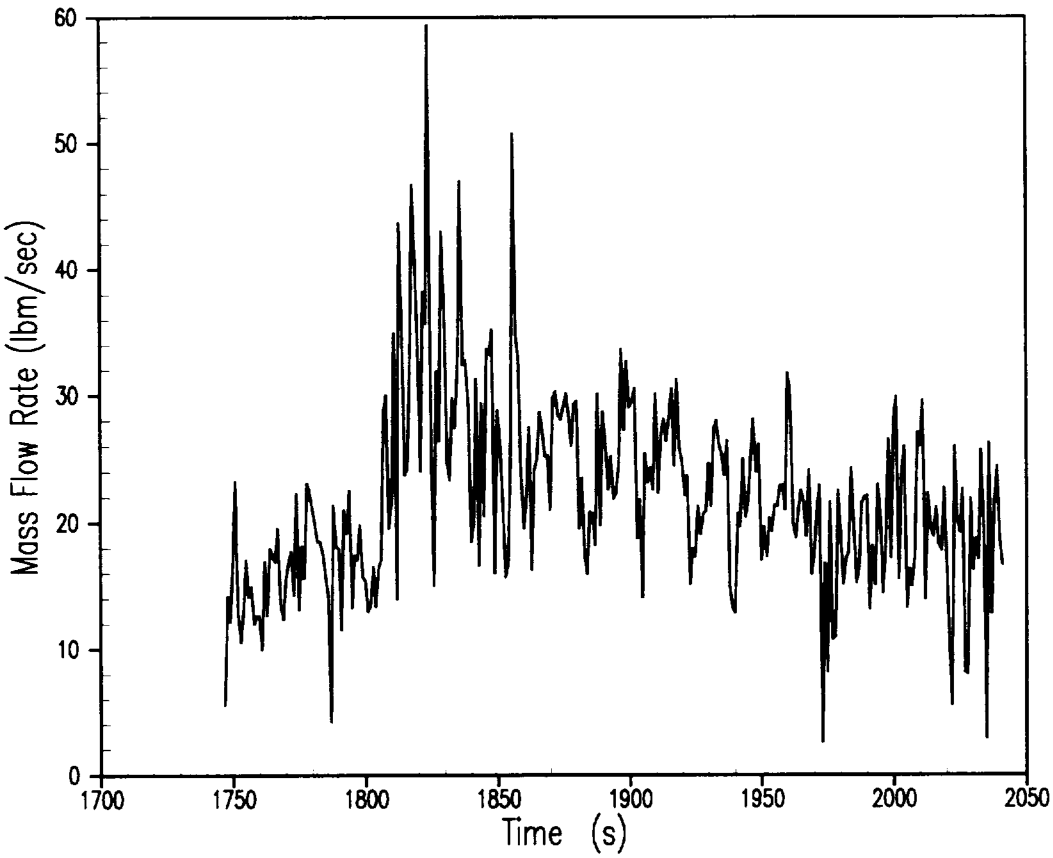


Figure 440.163-8a
Vapor Flow Upstream of the Pressurizer Loop ADS-4 / Hot Leg Junction, Inadvertant ADS Case

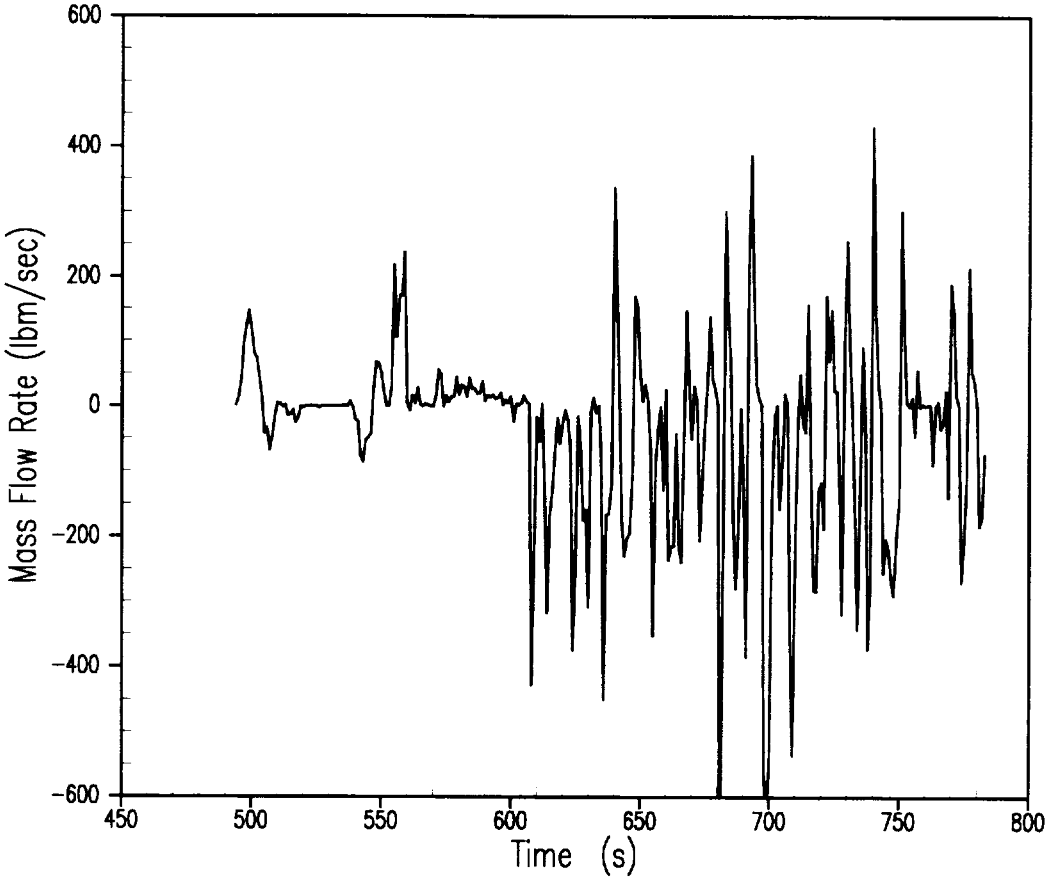


Figure 440.163-9
Continuous Liquid Flow Upstream of the Pressurizer Loop ADS-4 / Hot Leg Junction, DEDVI Break

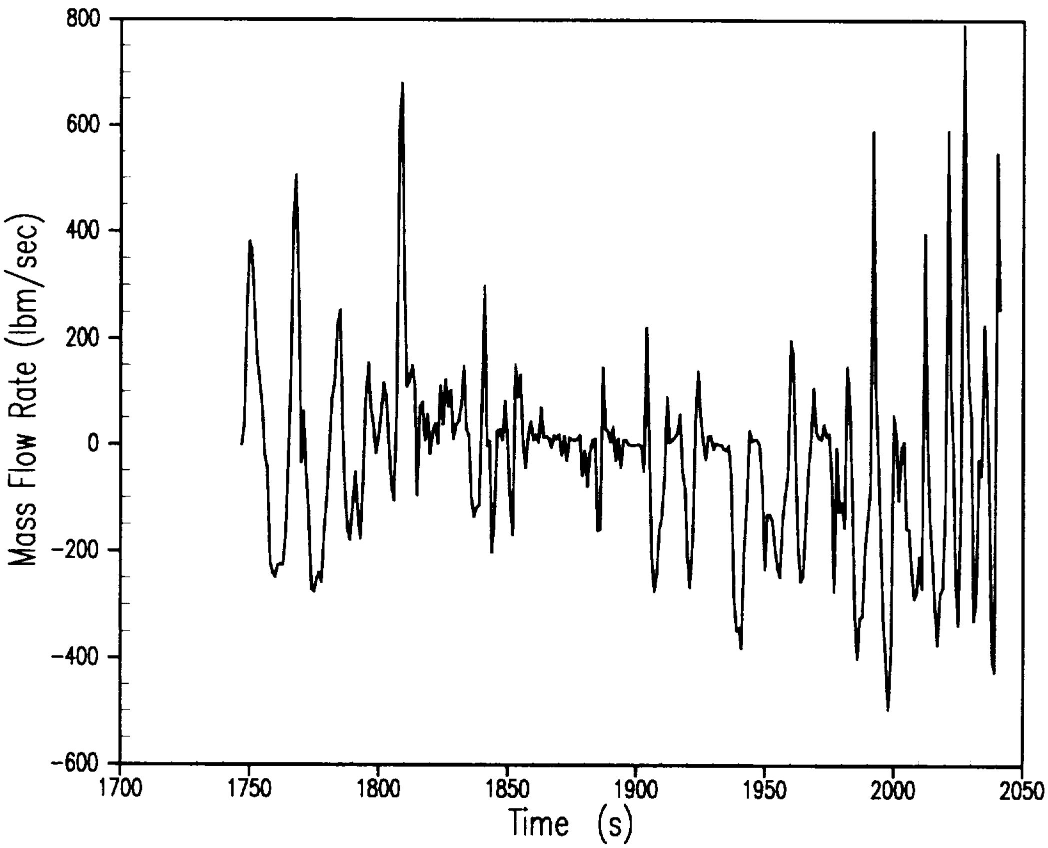


Figure 440.163-9a
Continuous Liquid Flow Upstream of the Pressurizer Loop ADS-4 / Hot Leg Junction, Inadvertant ADS Case

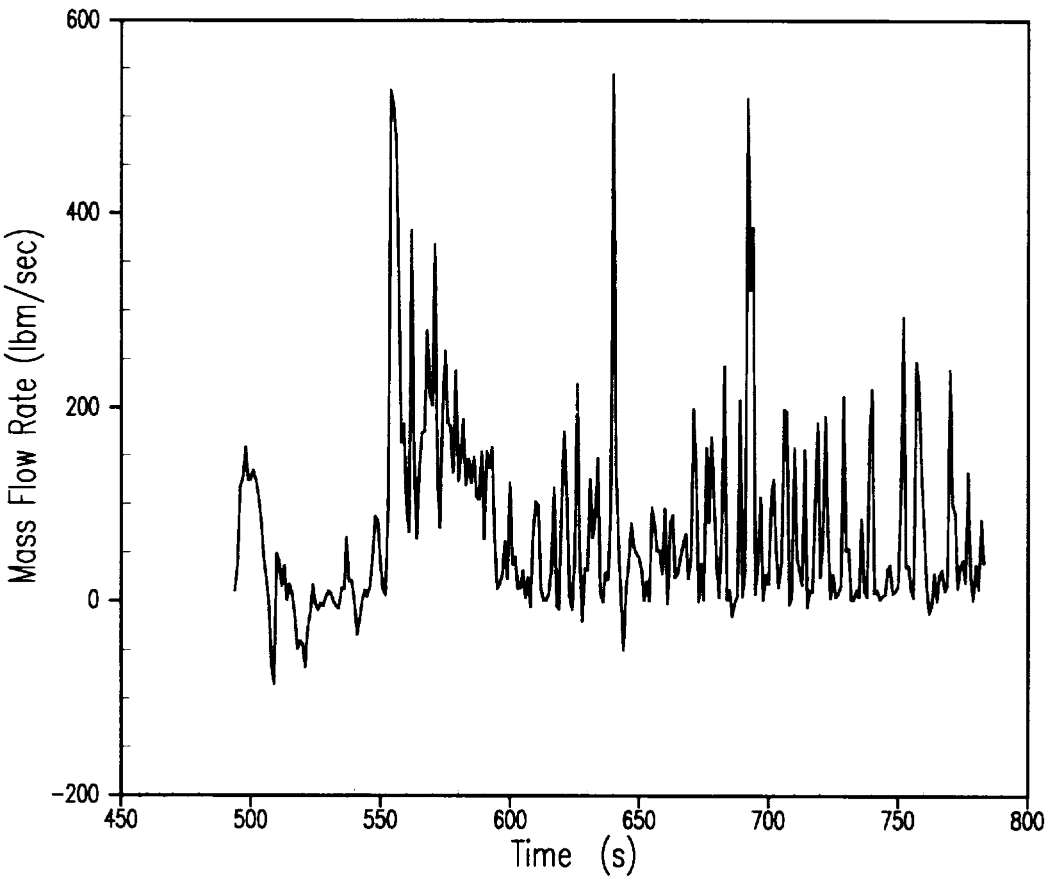


Figure 440.163-10
Entrained Liquid Flow Upstream of the Pressurizer Loop ADS-4 / Hot Leg Junction, DEDVI Break

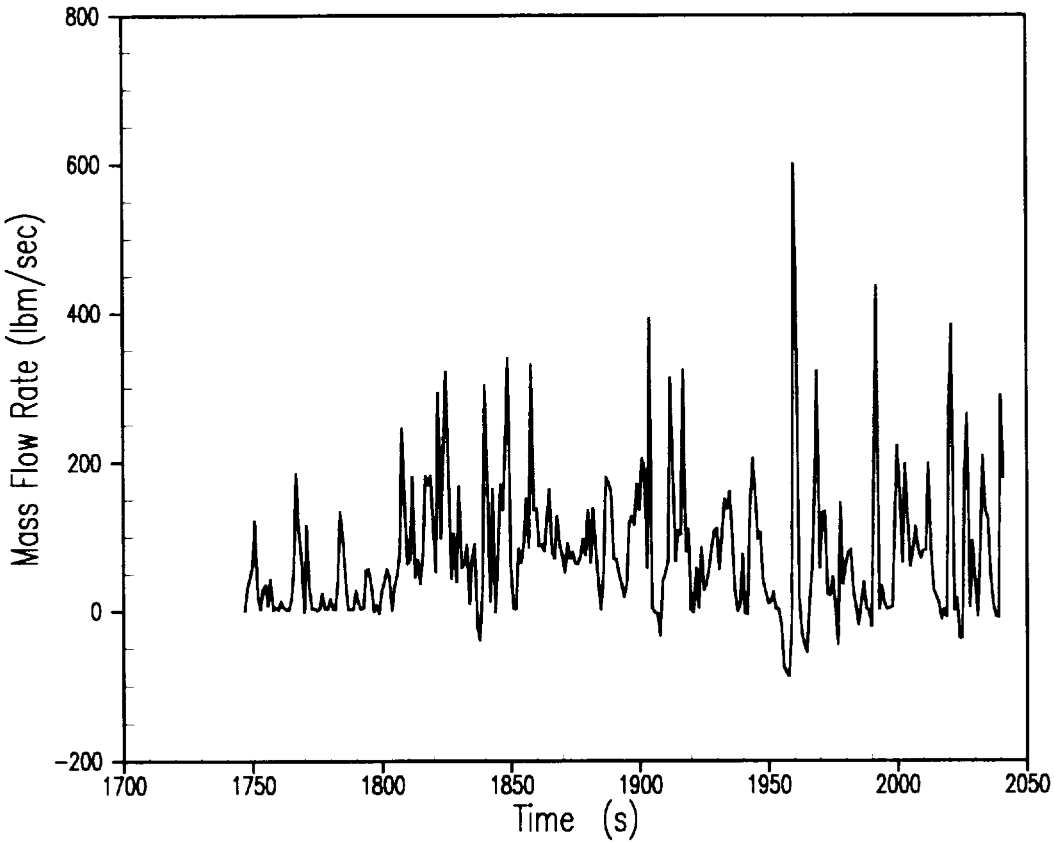


Figure 440.163-10a
Entrained Liquid Flow Upstream of the Pressurizer Loop ADS-4 / Hot Leg Junction, Inadvertant ADS Case

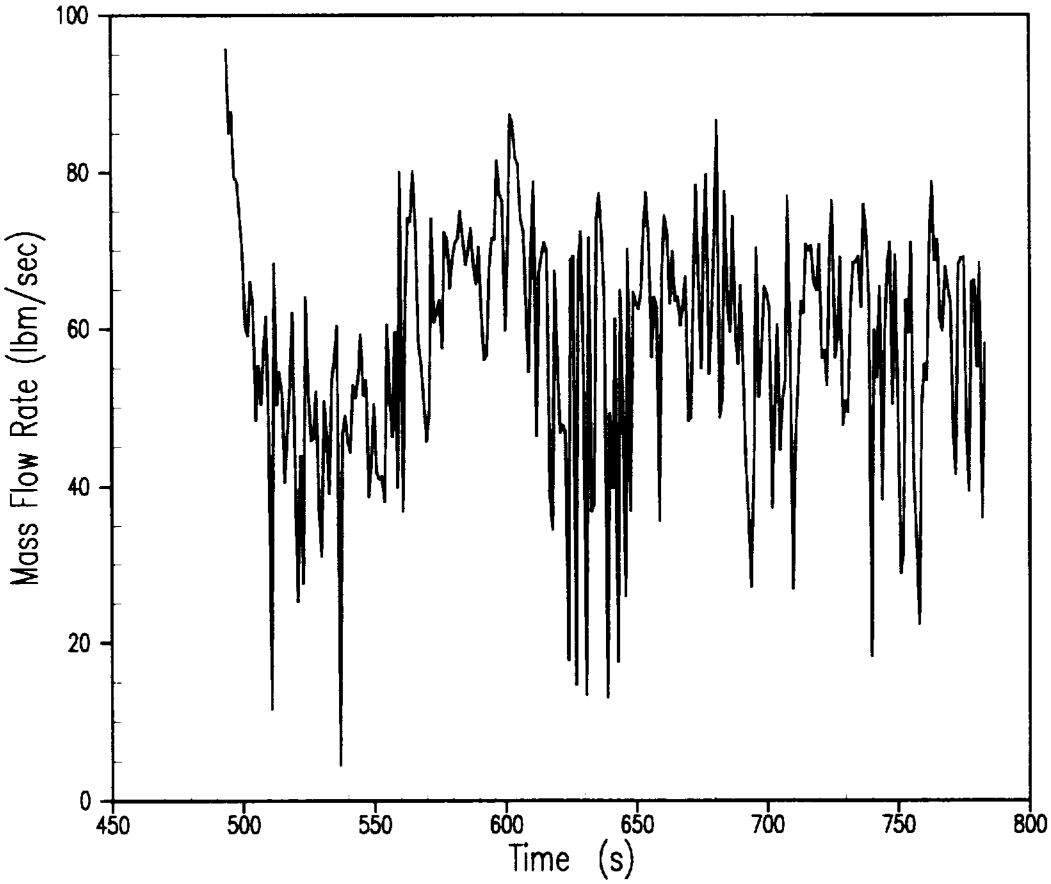


Figure 440.163-11
Vapor Flow Entering the 2*ADS-4 Flow Path Hot Leg, DEDVI Break

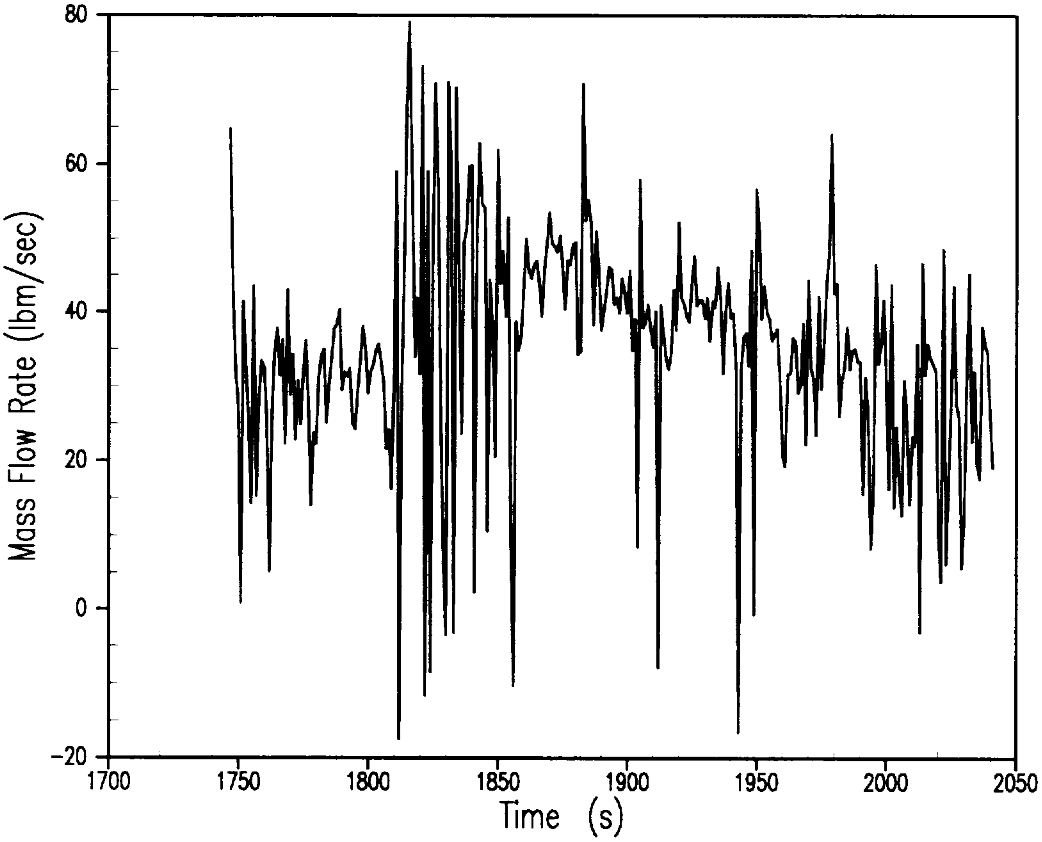


Figure 440.163-11a
Vapor Flow Entering the 2*ADS-4 Flow Path Hot Leg, Inadvertant ADS Case

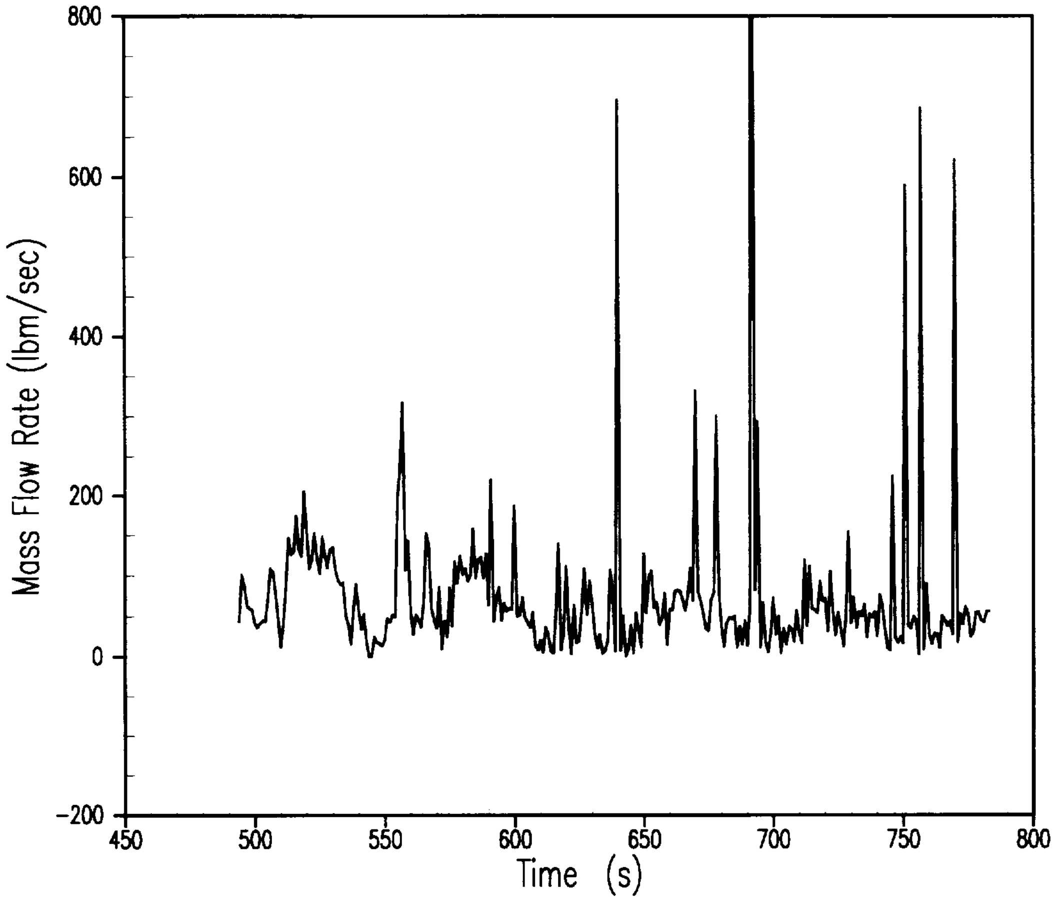


Figure 440.163-12
Continuous Liquid Flow Entering the 2*ADS-4 Flow Path Hot Leg, DEDVI Break

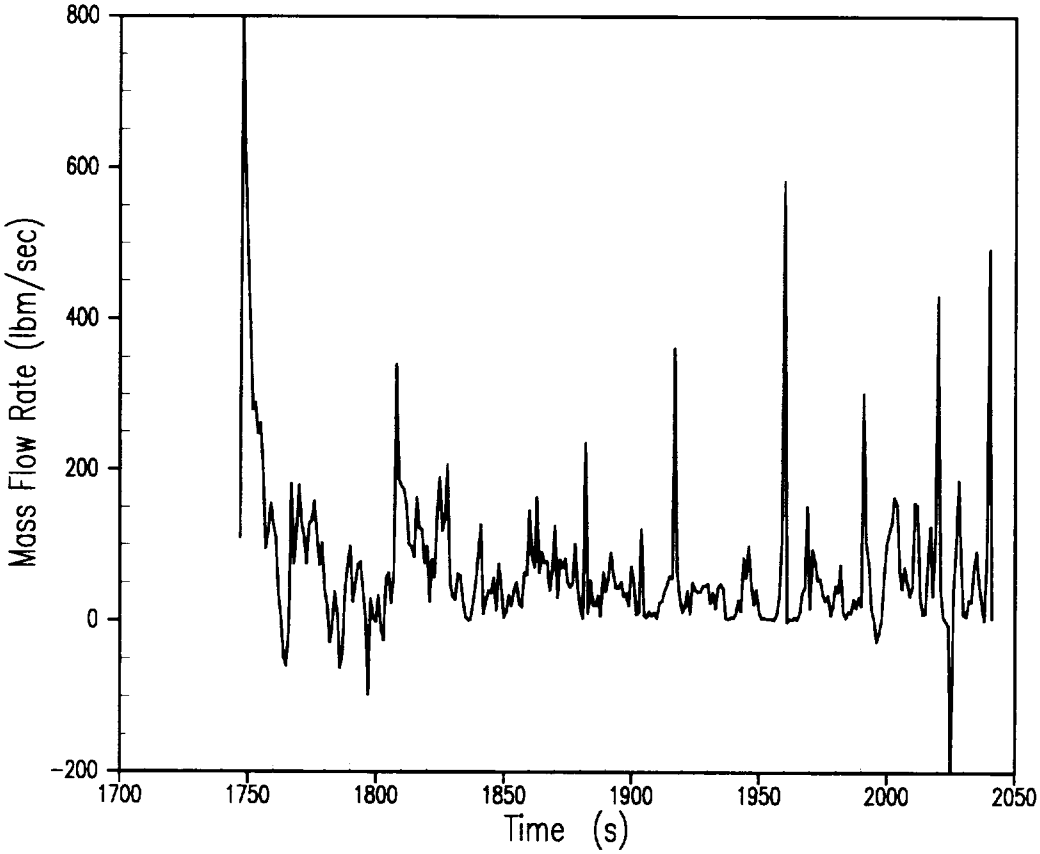


Figure 440.163-12a
Continuous Liquid Flow Entering the 2*ADS-4 Flow Path Hot Leg, Inadvertant ADS Case

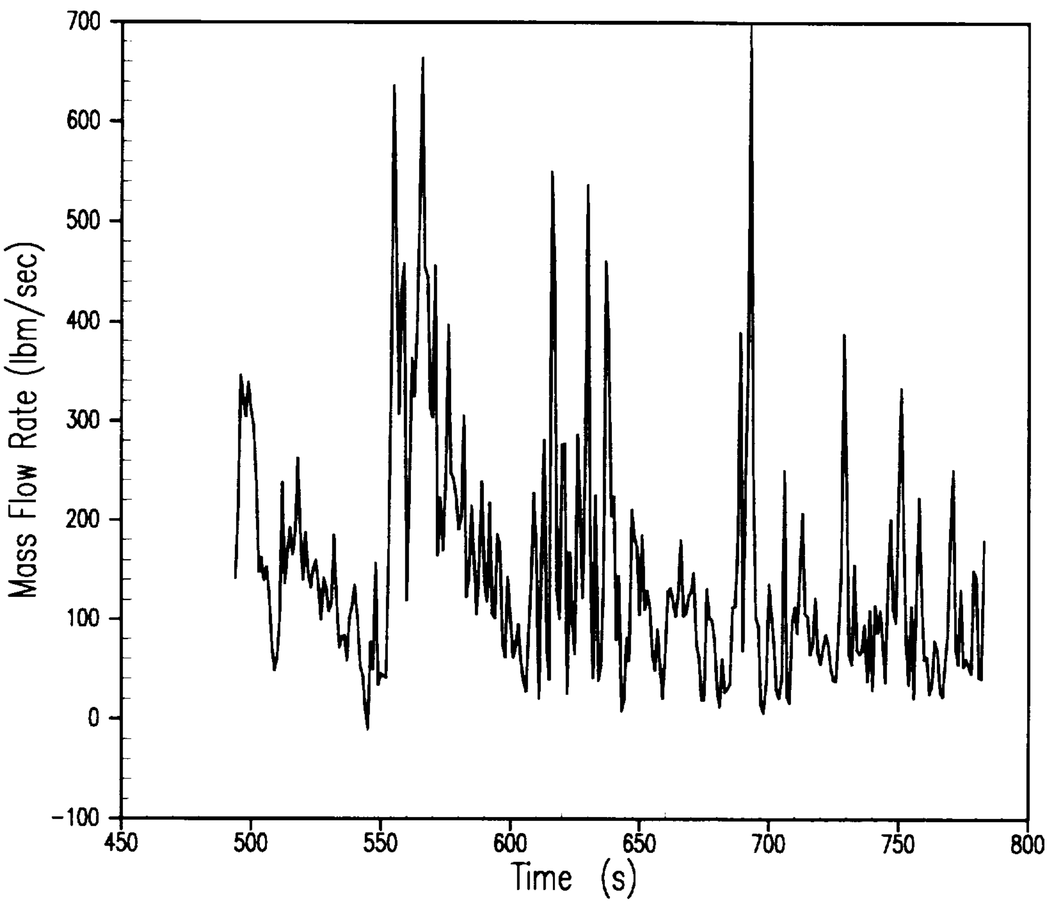


Figure 440.163-13
Entrained Liquid Flow Entering the 2*ADS-4 Flow Path Hot Leg, DEDVI Break

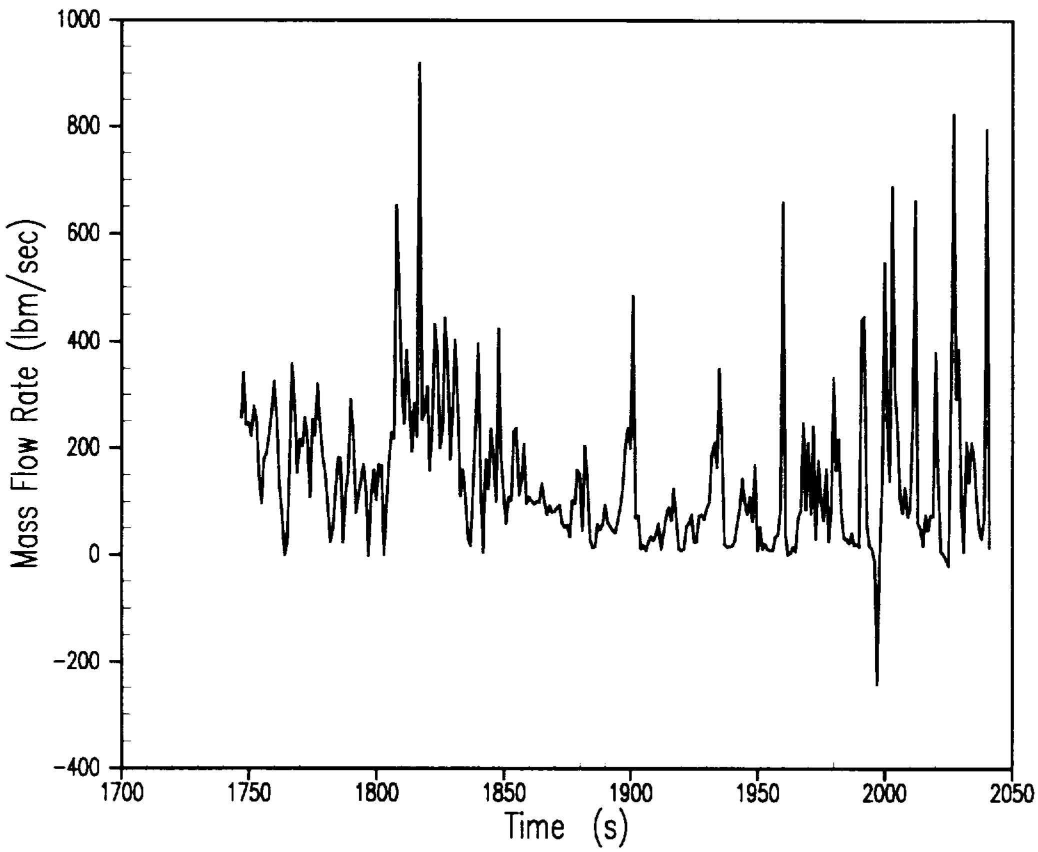


Figure 440.163-13a
Entrained Liquid Flow Entering the 2*ADS-4 Flow Path Hot Leg, Inadvertant ADS Case

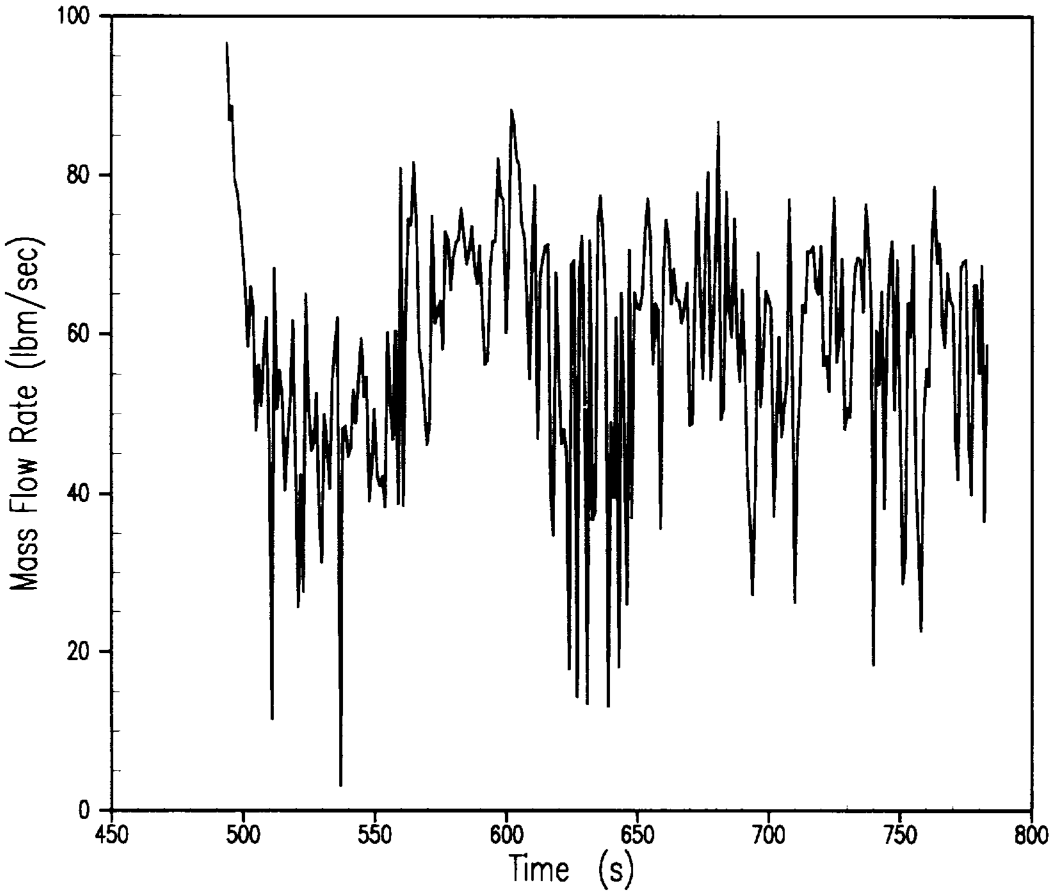


Figure 440.163-14
Vapor Flow Upstream of the 2*ADS-4 Loop ADS-4 / Hot Leg Junction, DEDVI Break

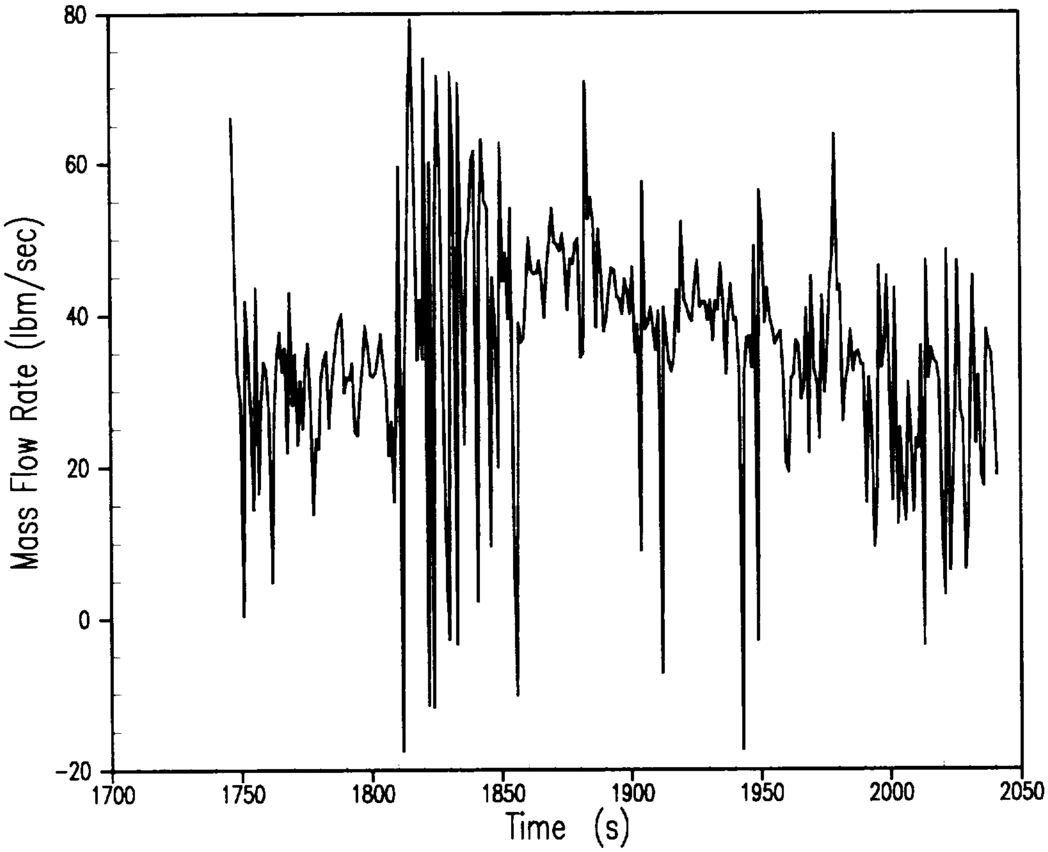


Figure 440.163-14a
Vapor Flow Upstream of the 2*ADS-4 Loop ADS-4 / Hot Leg Junction, Inadvertant ADS Case

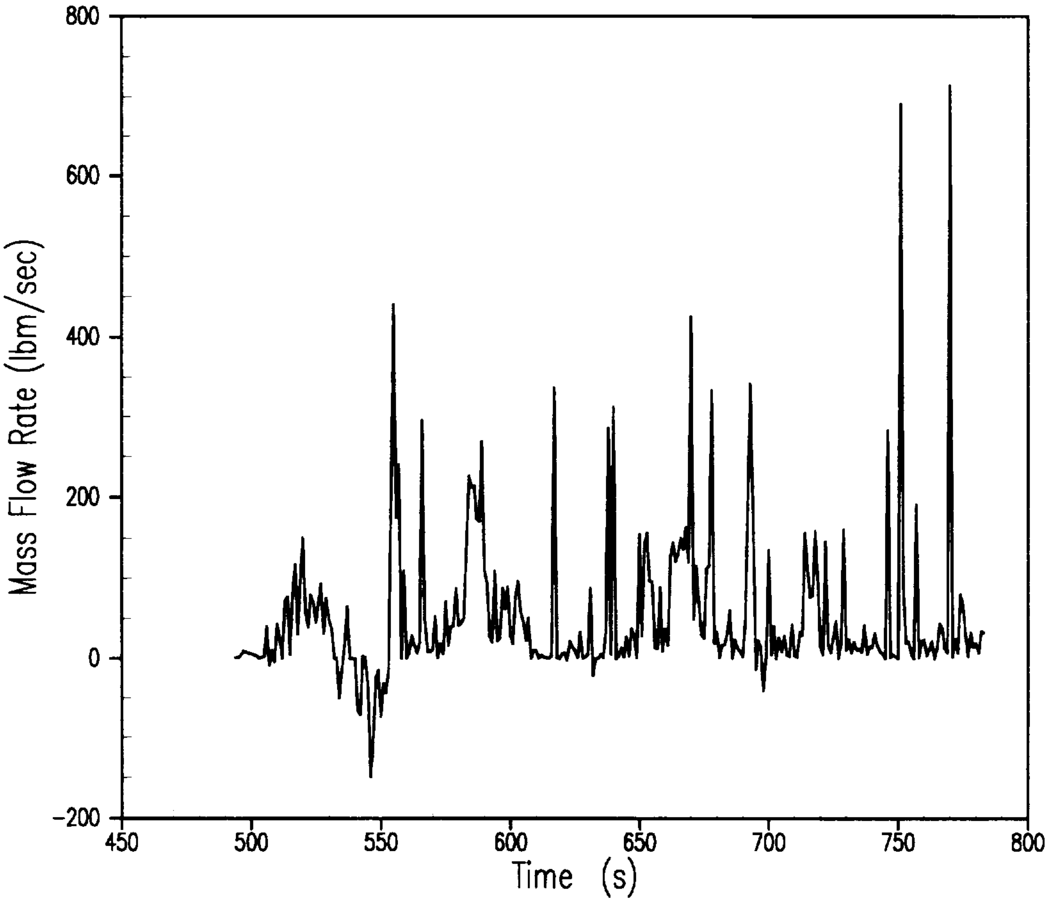


Figure 440.163-15
Continuous Liquid Flow Upstream of the 2*ADS-4 Loop ADS-4 / Hot Leg Junction, DEDVI Break

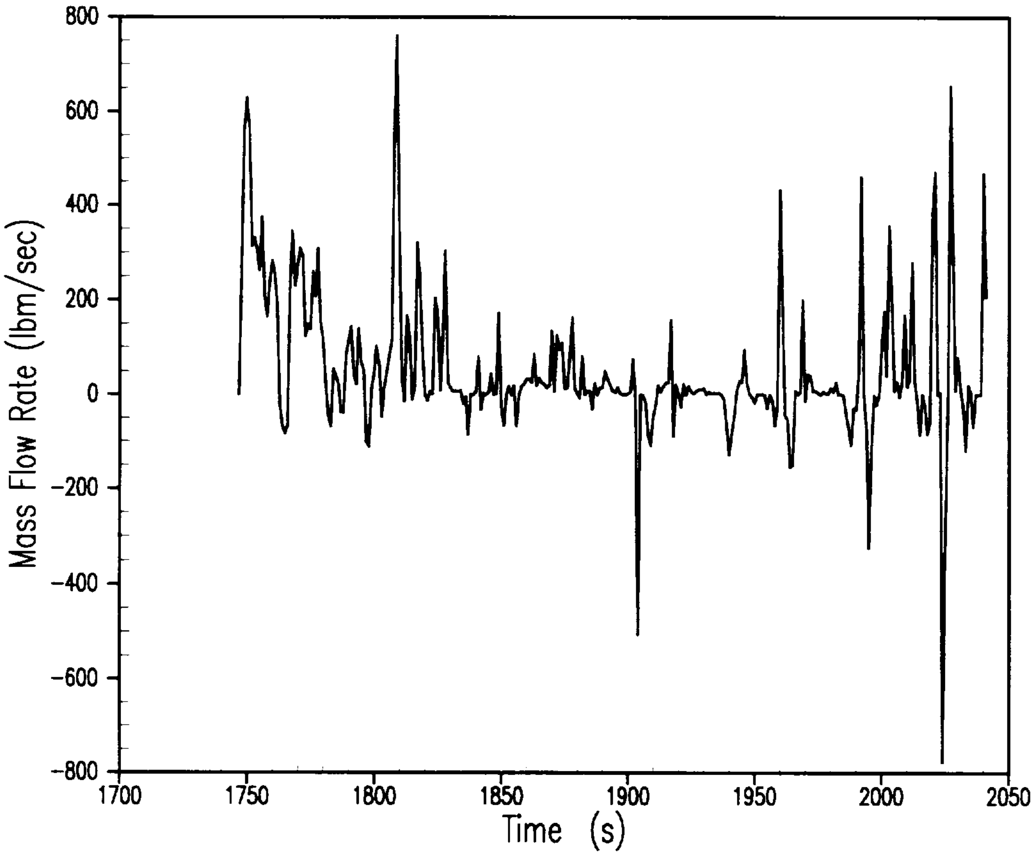


Figure 440.163-15a
Continuous Liquid Flow Upstream of the 2*ADS-4 Loop ADS-4 / Hot Leg Junction
Inadvertant ADS Case

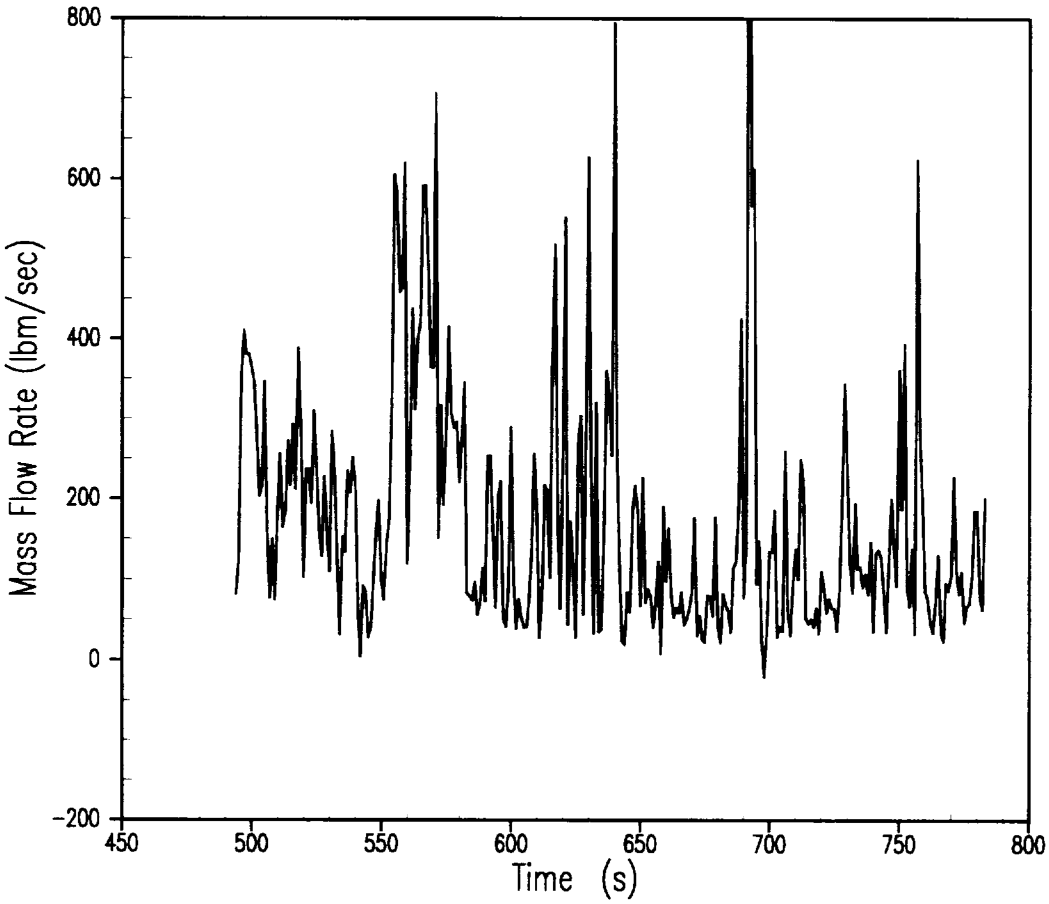


Figure 440.163-16
Entrained Liquid Flow Upstream of the 2*ADS-4 Loop ADS-4 / Hot Leg Junction, DEDVI Break

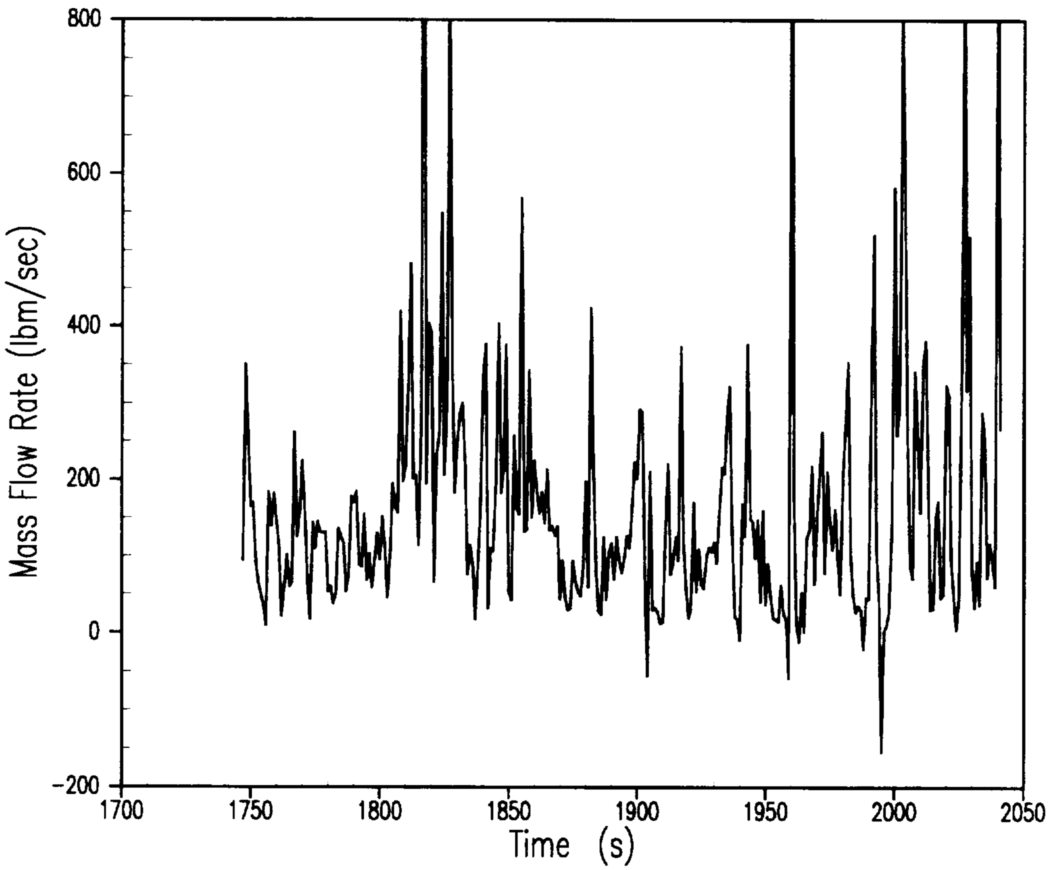


Figure 440.163-16a
Entrained Liquid Flow Upstream of the 2*ADS-4 Loop ADS-4 / Hot Leg Junction, Inadvertant ADS Case

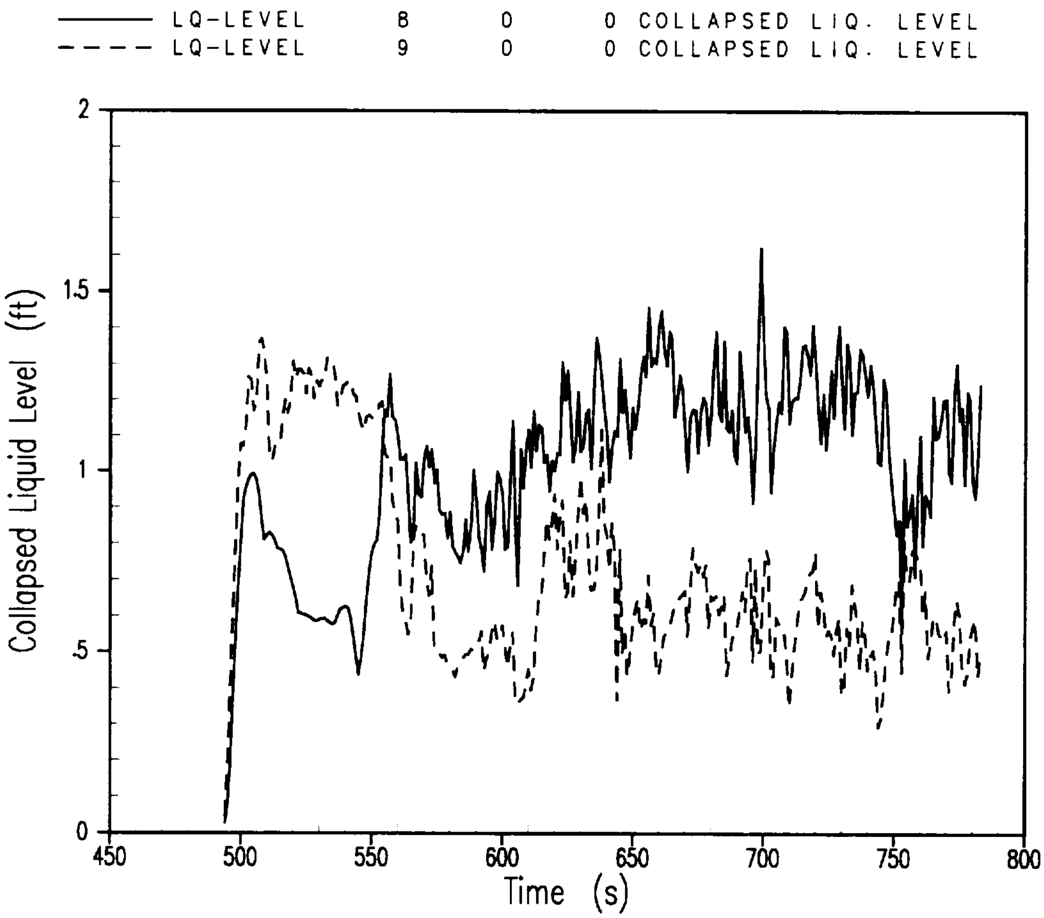


Figure 440.163-17
Overall Hot Leg Collapsed Liquid Levels, DEDVI Break

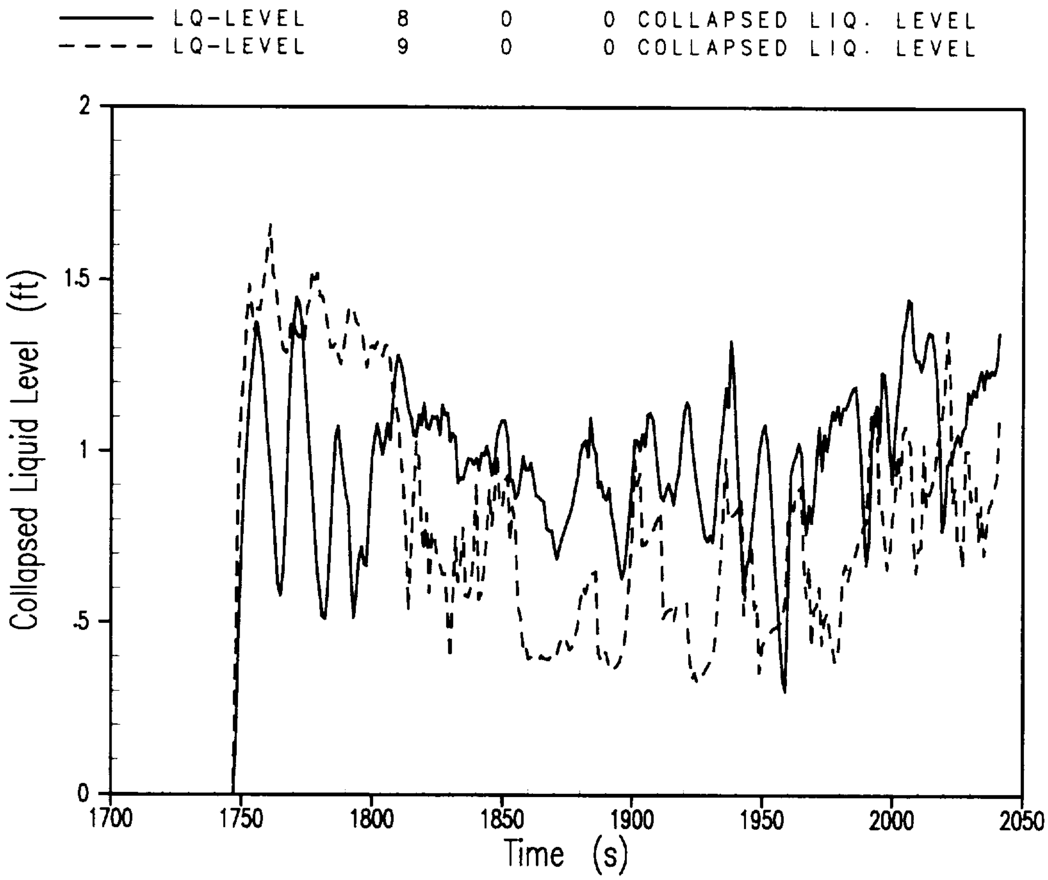


Figure 440.163-17a
Overall Hot Leg Collapsed Liquid Levels, Inadvertant ADS Case

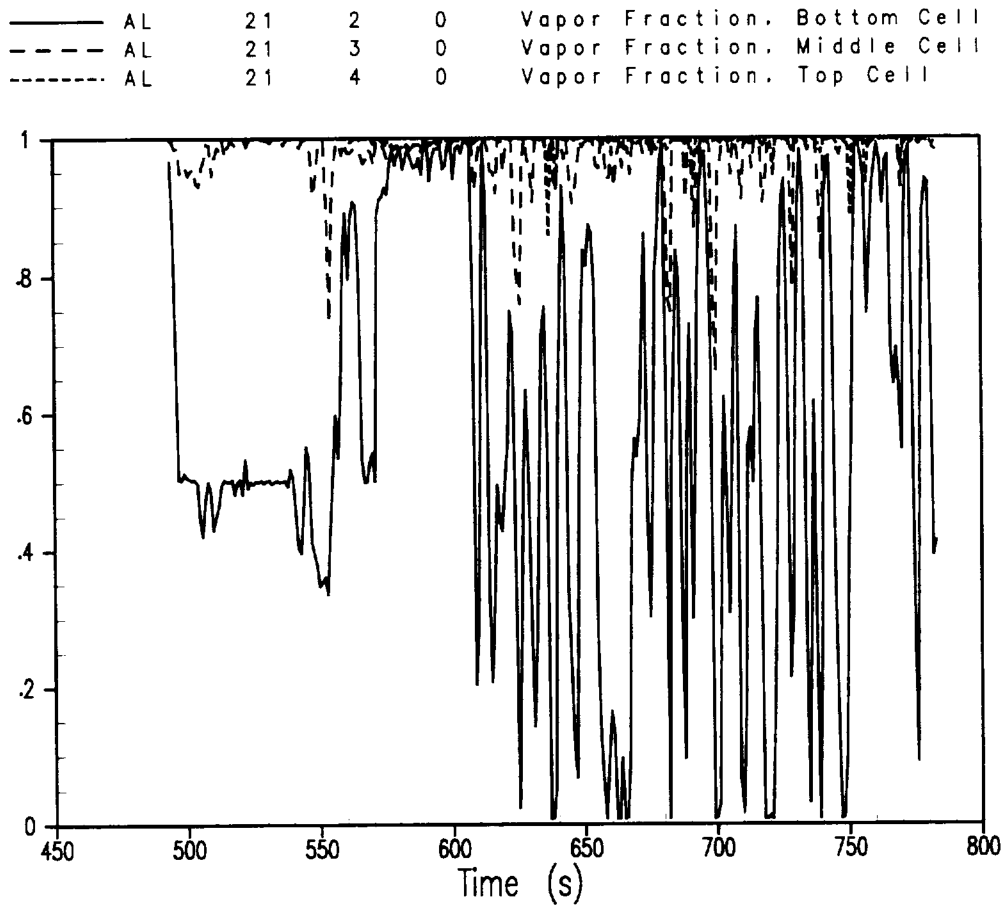


Figure 440.163-18
Initial Hot Leg Channel Void Profile, Pressurizer Loop, DEDVI Break

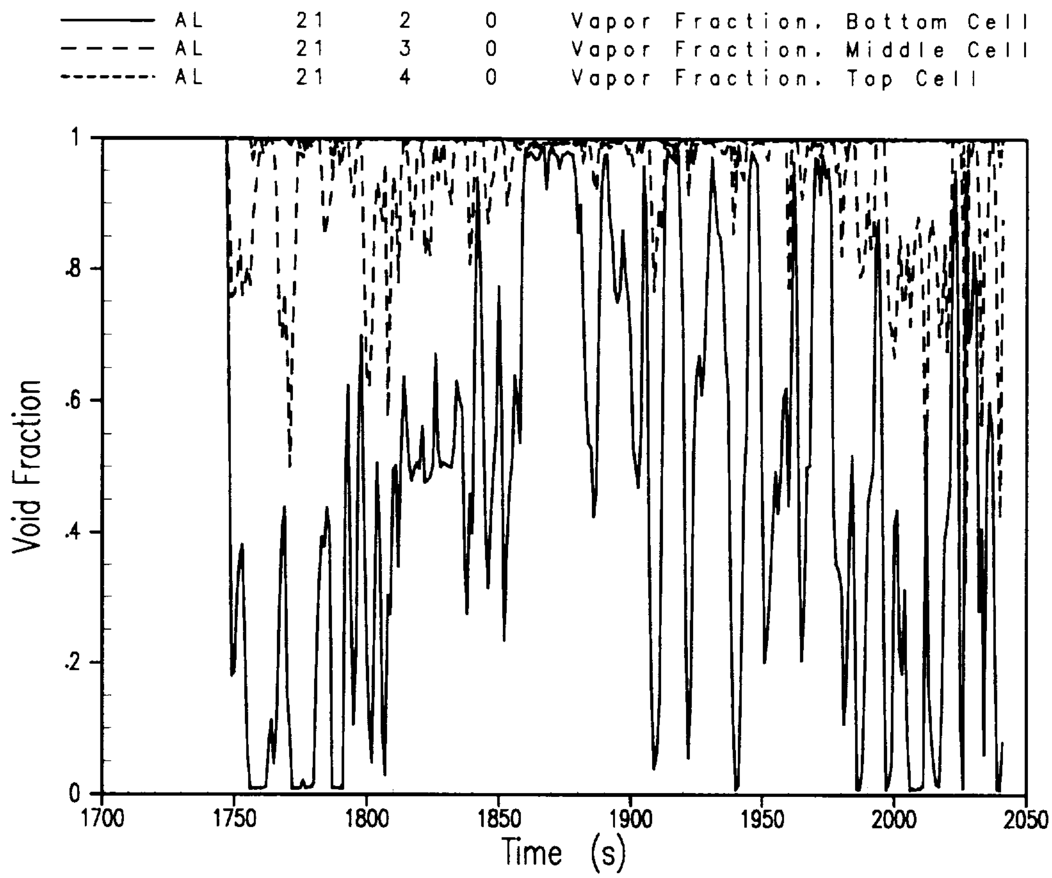


Figure 440.163-18a
Initial Hot Leg Channel Void Profile, Pressurizer Loop, Inadvertant ADS Case

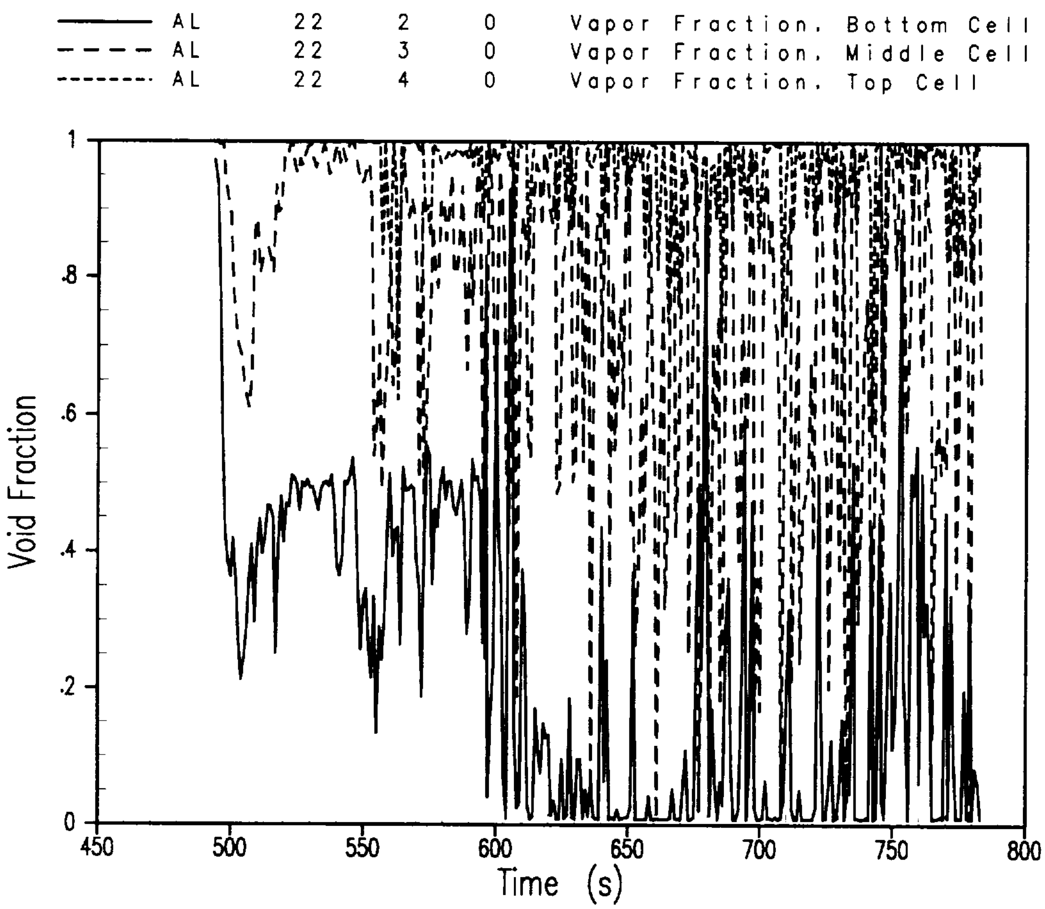


Figure 440.163-19
ADS-4 Offtake Channel Void Profile, Pressurizer Loop, DEDVI Break

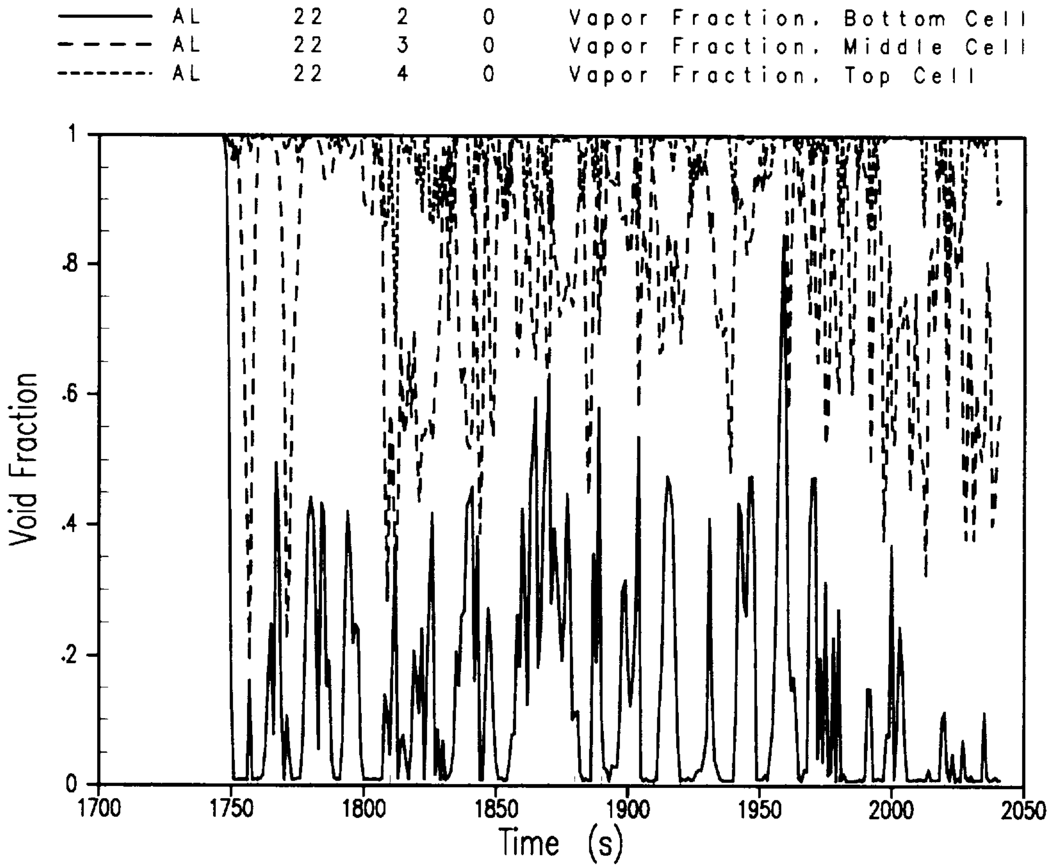


Figure 440.163-19a
ADS-4 Offtake Channel Void Profile, Pressurizer Loop, Inadvertant ADS Case

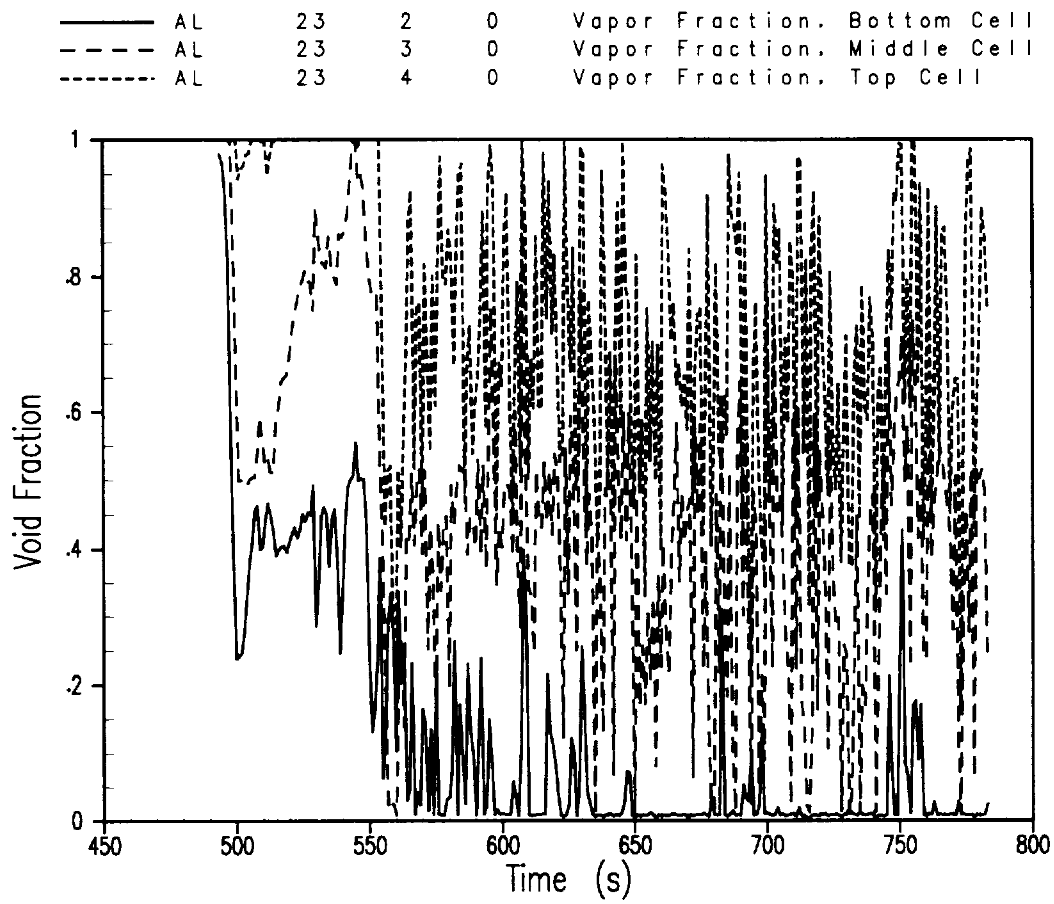


Figure 440.163-20
End Hot Leg Channel Void Profile, Pressurizer Loop, DEDVI Break

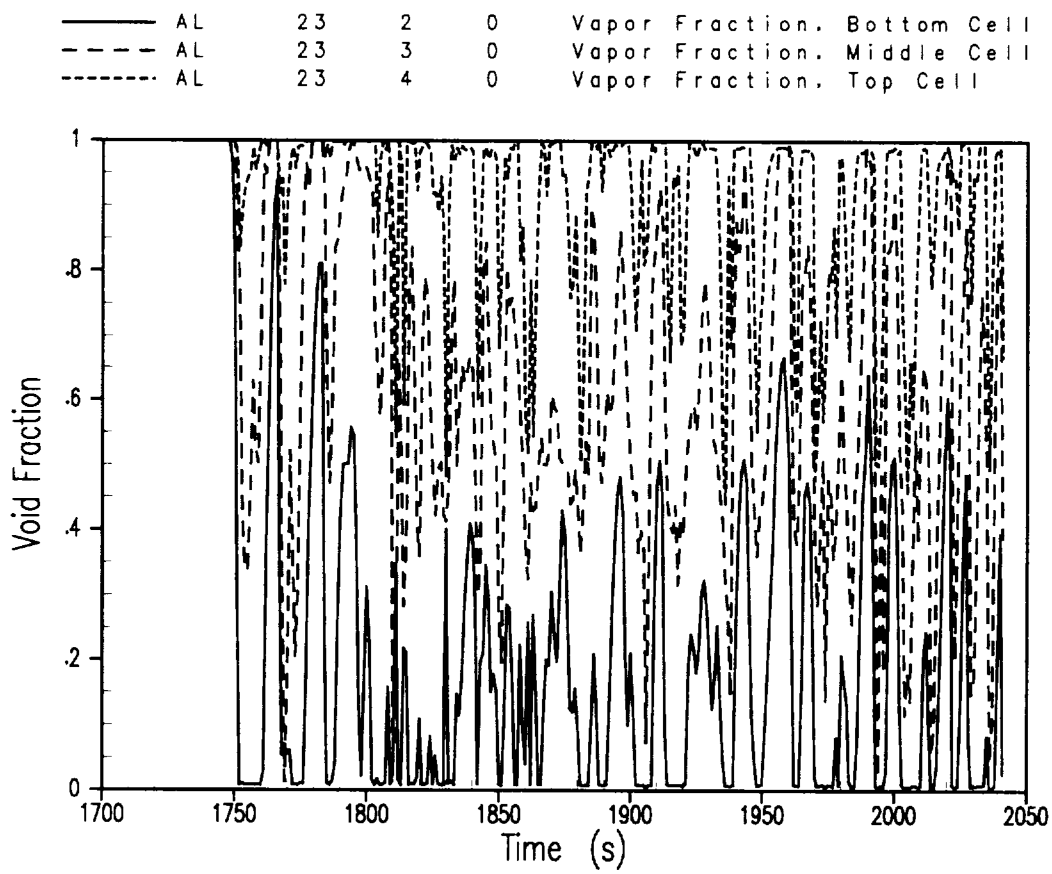


Figure 440.163-20a
End Hot Leg Channel Void Profile, Pressurizer Loop, Inadvertant ADS Case

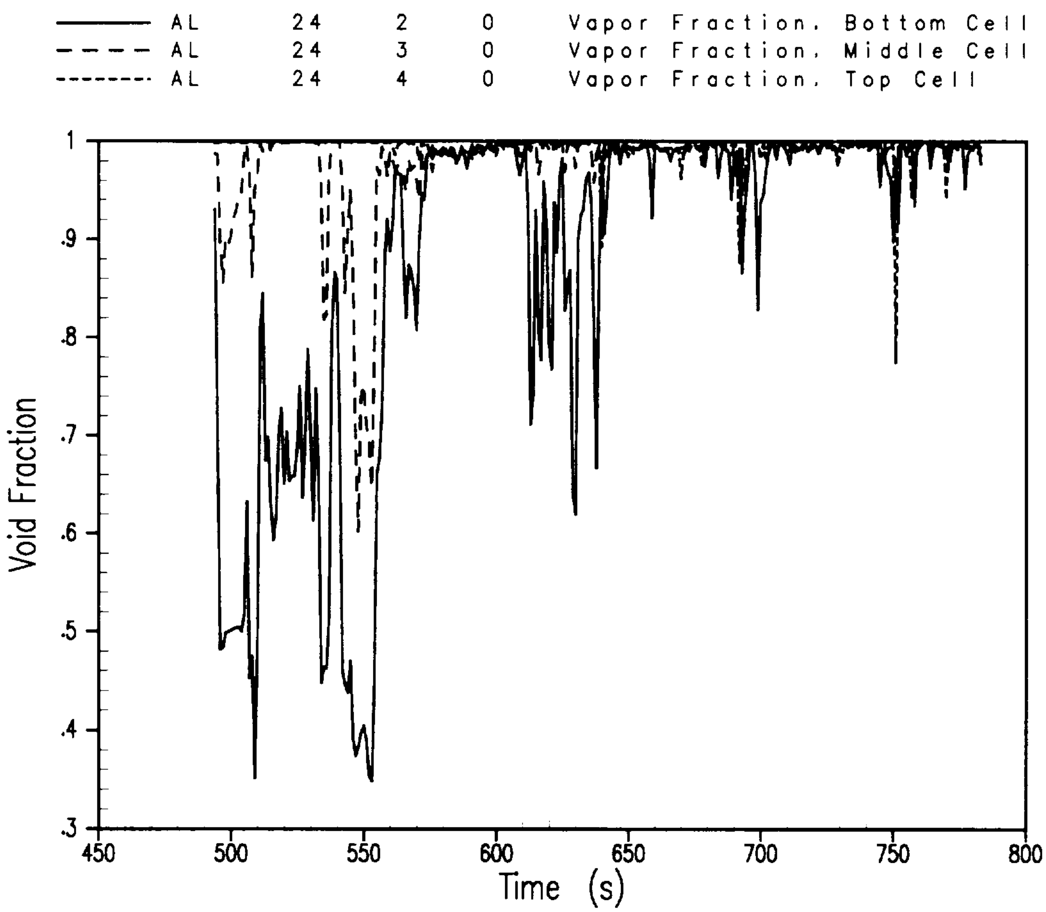


Figure 440.163-21
Initial Hot Leg Channel Void Profile, 2*ADS-4 Loop, DEDVI Break

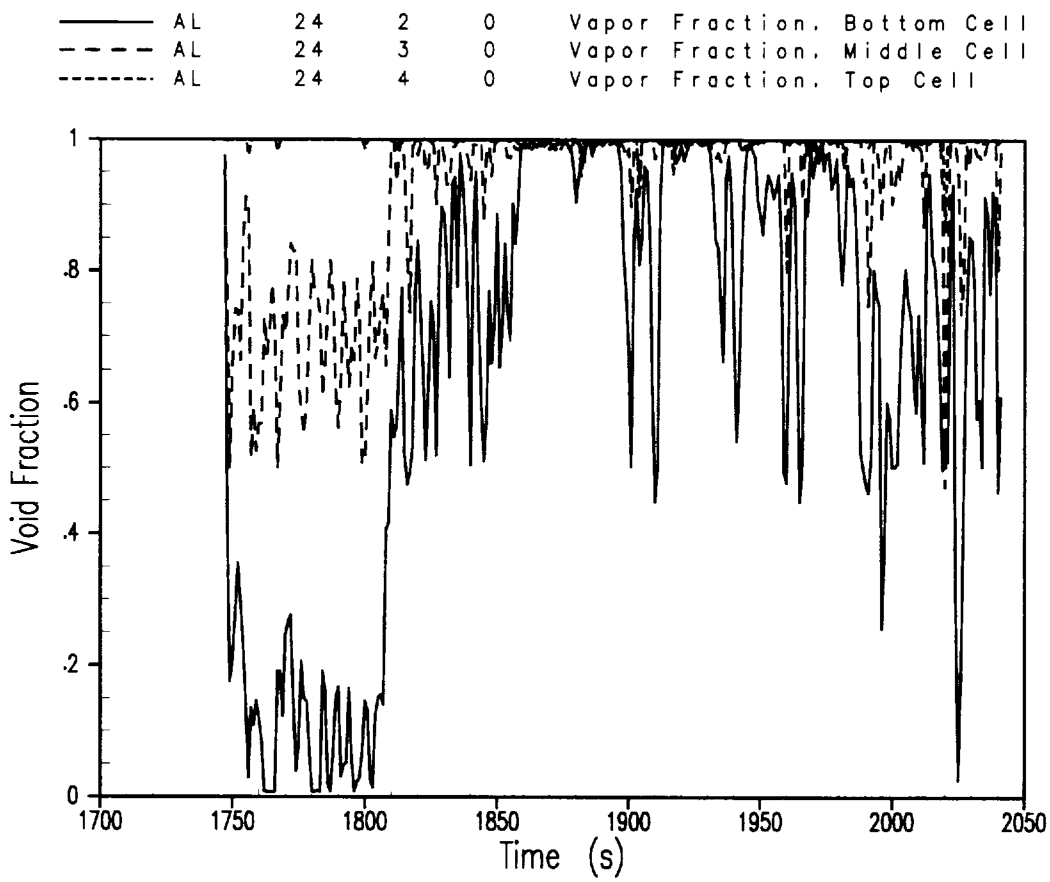


Figure 440.163-21a
Initial Hot Leg Channel Void Profile, 2*ADS-4 Loop, Inadvertant ADS Case

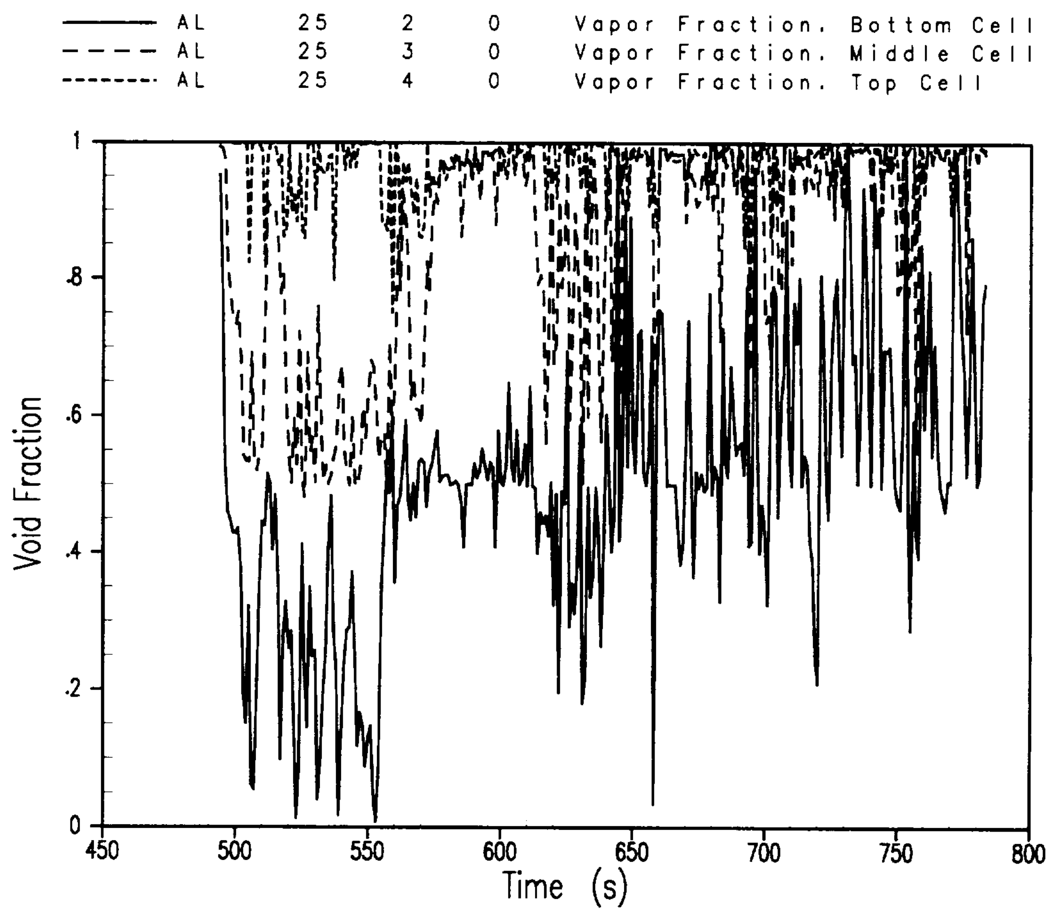


Figure 440.163-22
ADS-4 Offtake Channel Void Profile, 2*ADS-4 Loop, DEDVI Break

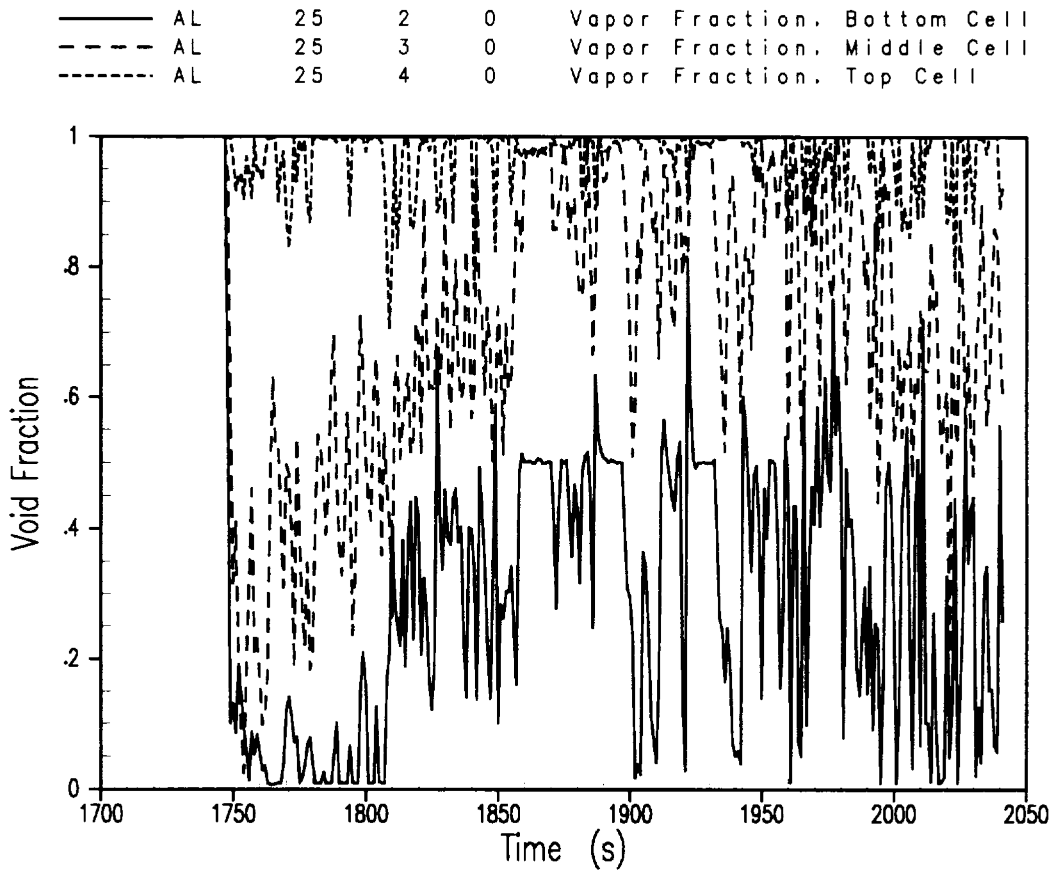


Figure 440.163-22a
ADS-4 Offtake Channel Void Profile, 2*ADS-4 Loop, Inadvertant ADS Case

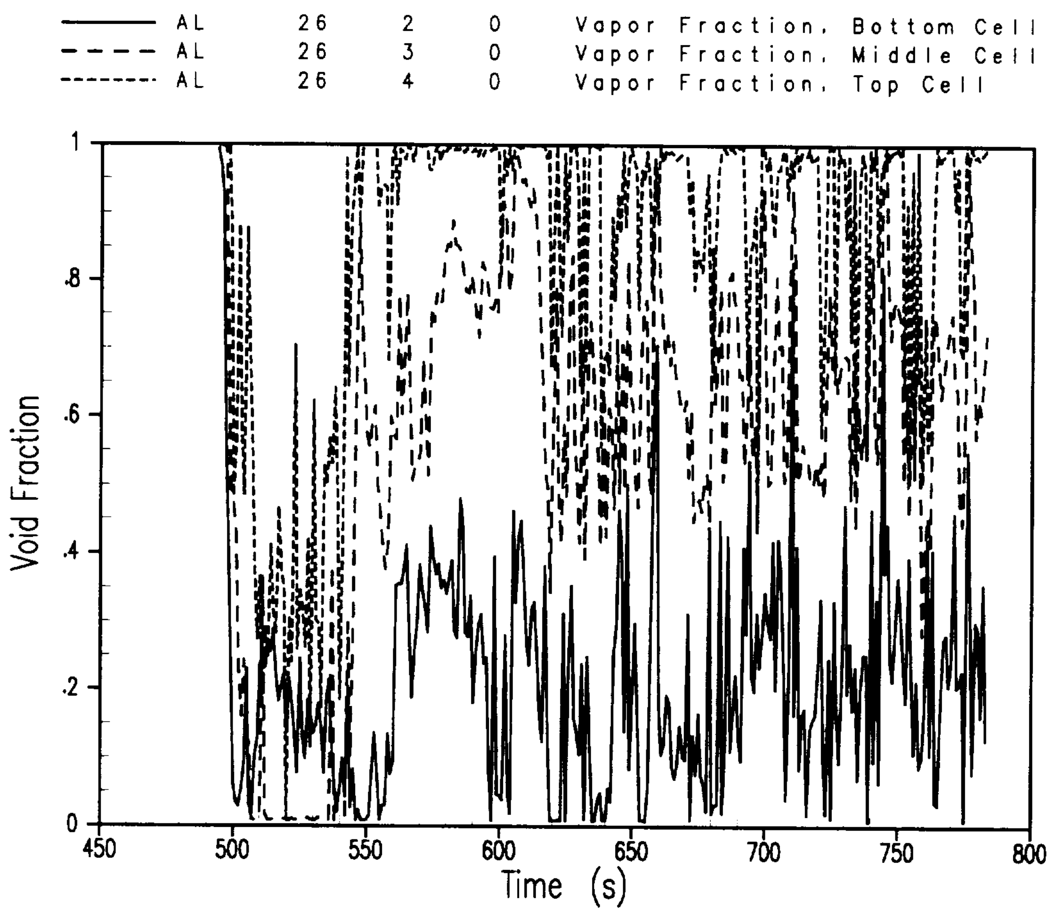


Figure 440.163-23
End Hot Leg Channel Void Profile, 2*ADS-4 Loop, DEDVI Break

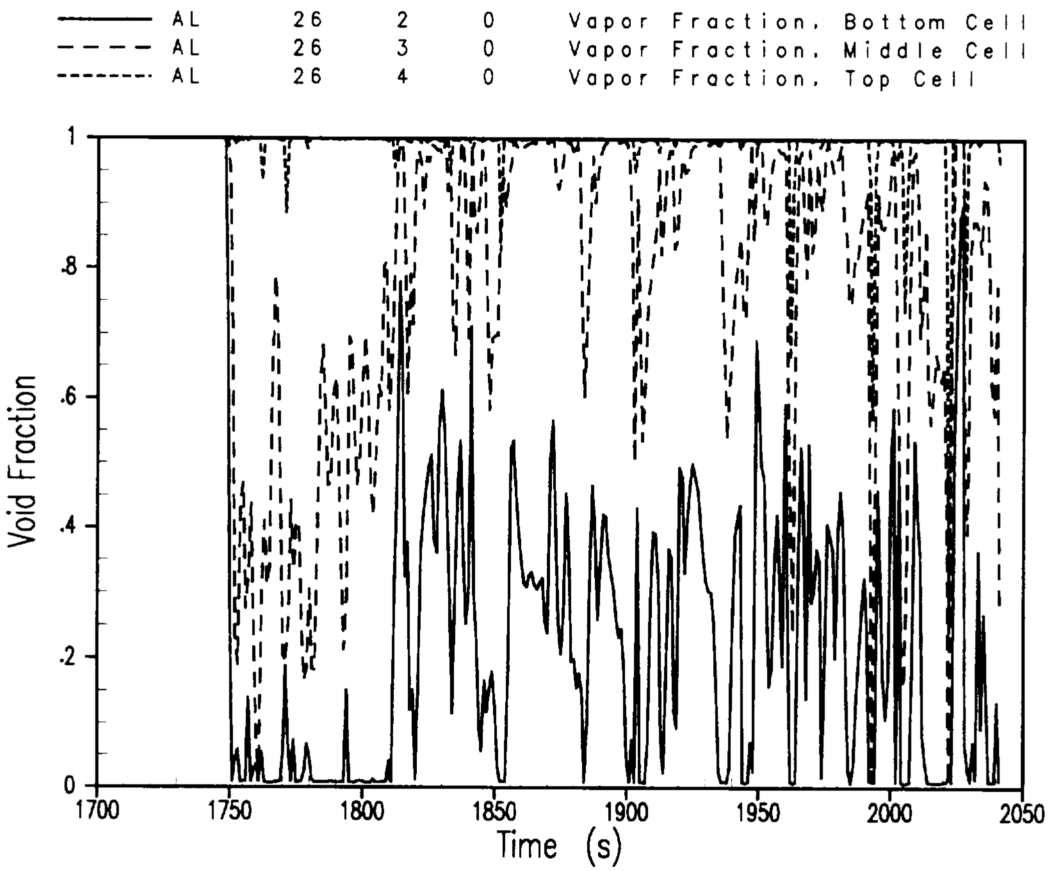


Figure 440.163-23a
End Hot Leg Channel Void Profile, 2*ADS-4 Loop, Inadvertant ADS Case

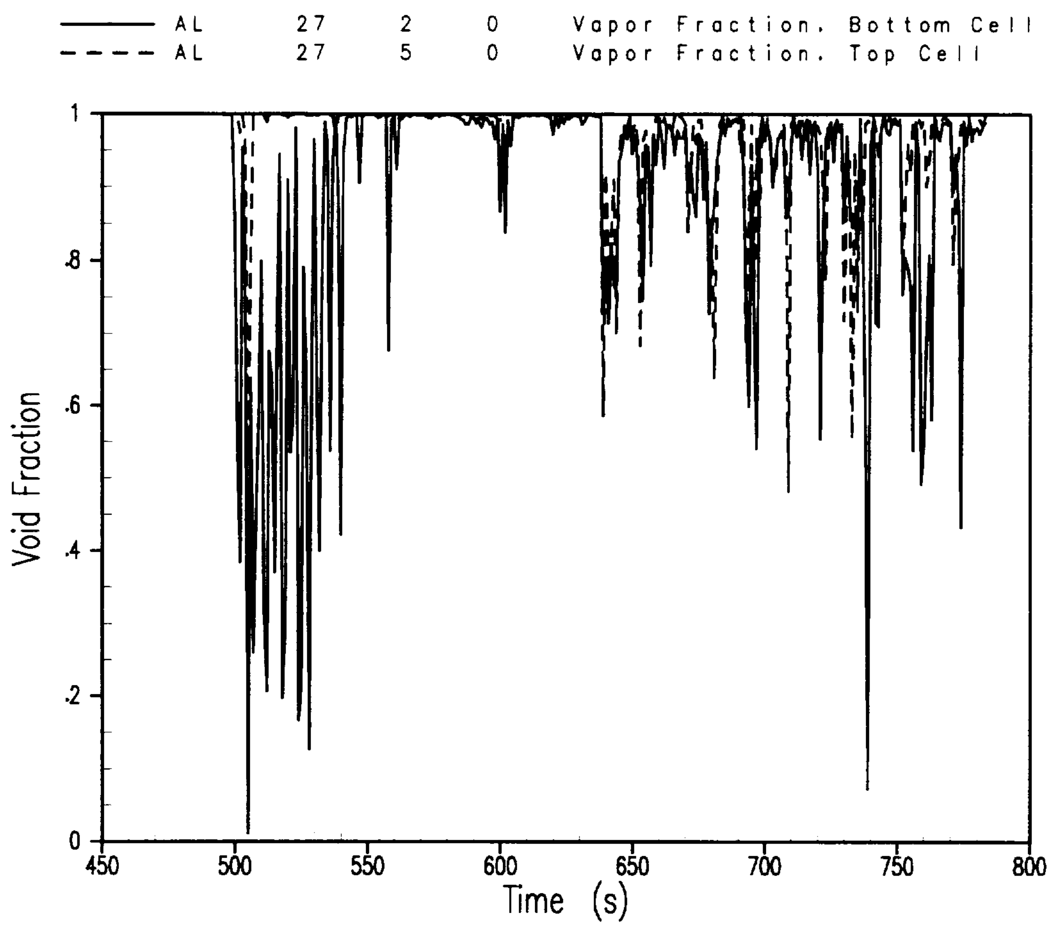


Figure 440.163-24
First Sloped Hot Leg Channel Void Profile, 2*ADS-4 Loop, DEDVI Break

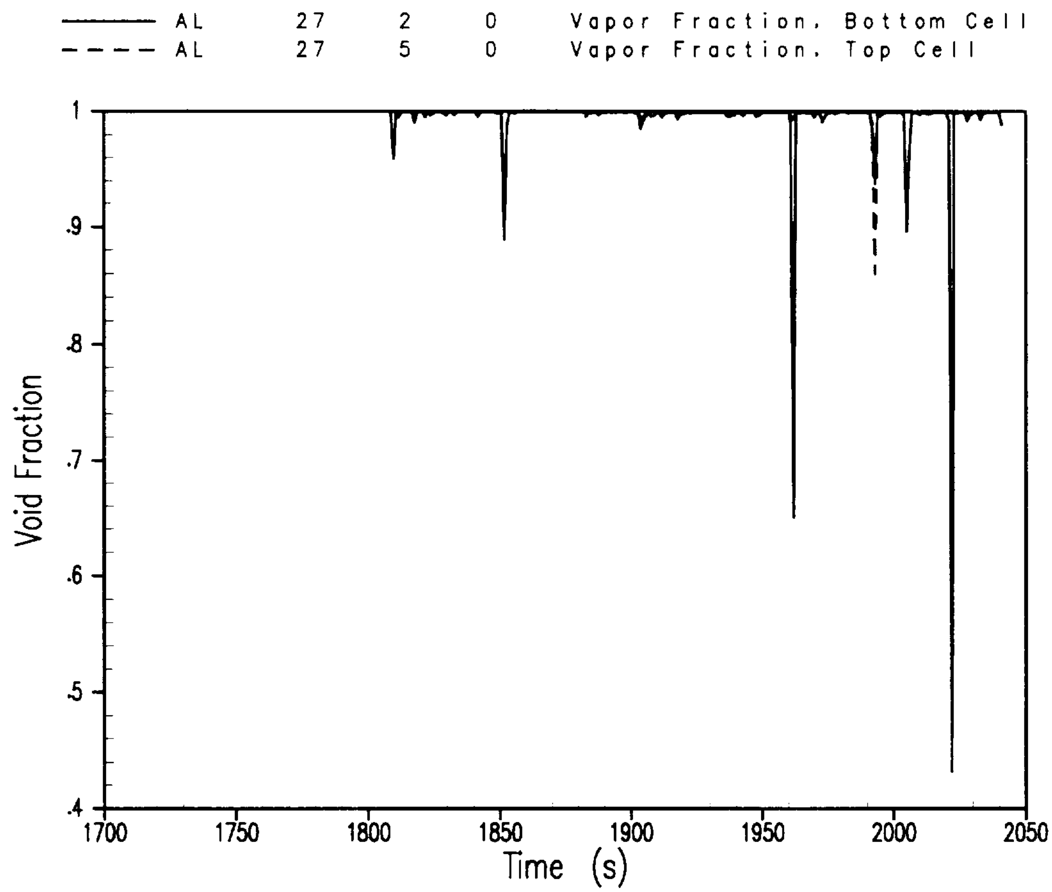


Figure 440.163-24a
First Sloped Hot Leg Channel Void Profile, 2*ADS-4 Loop, Inadvertant ADS Case

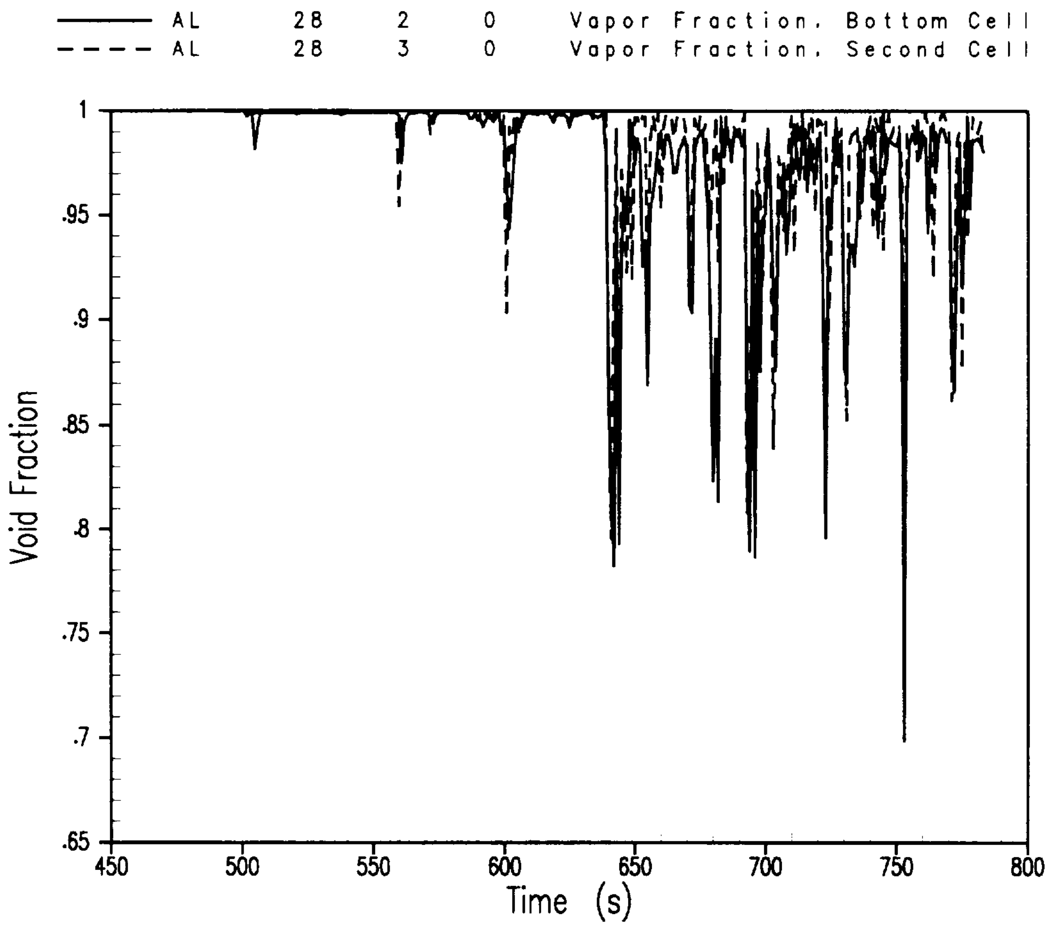


Figure 440.163-25
Second Sloped Hot Leg Channel Void Profile, 2*ADS-4 Loop, DEDVI Break

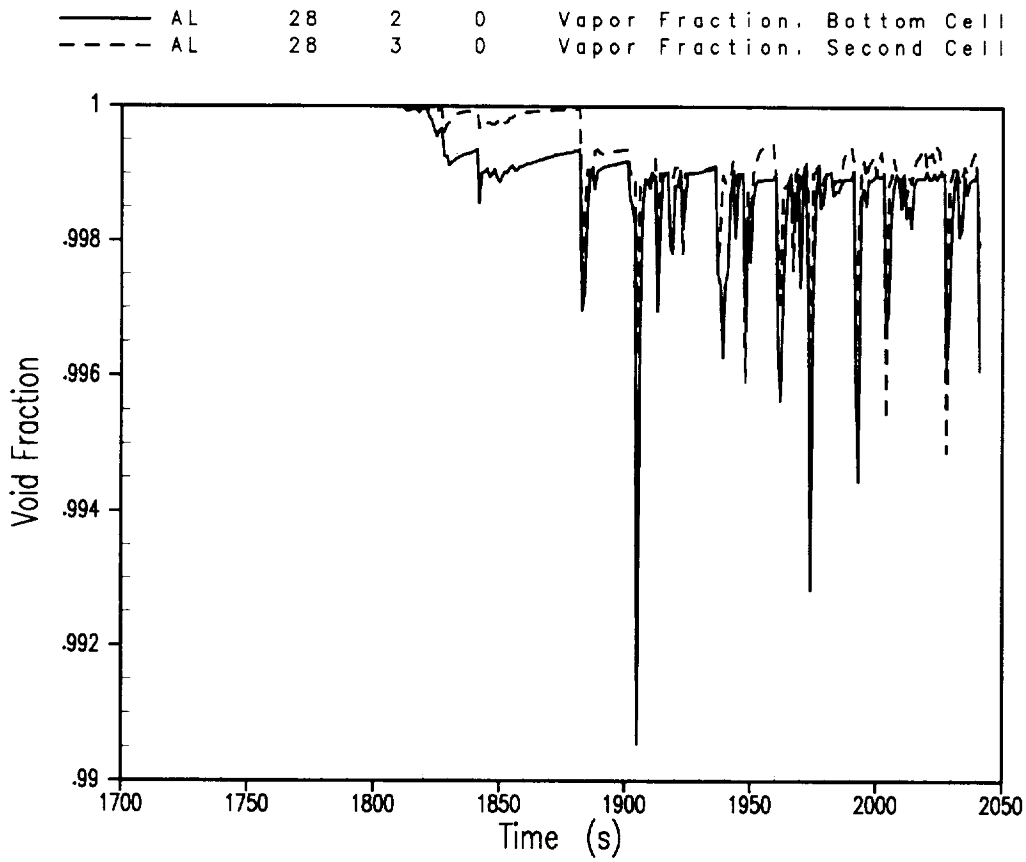


Figure 440.163-25a
Second Sloped Hot Leg Channel Void Profile, 2*ADS-4 Loop, Inadvertant ADS Case

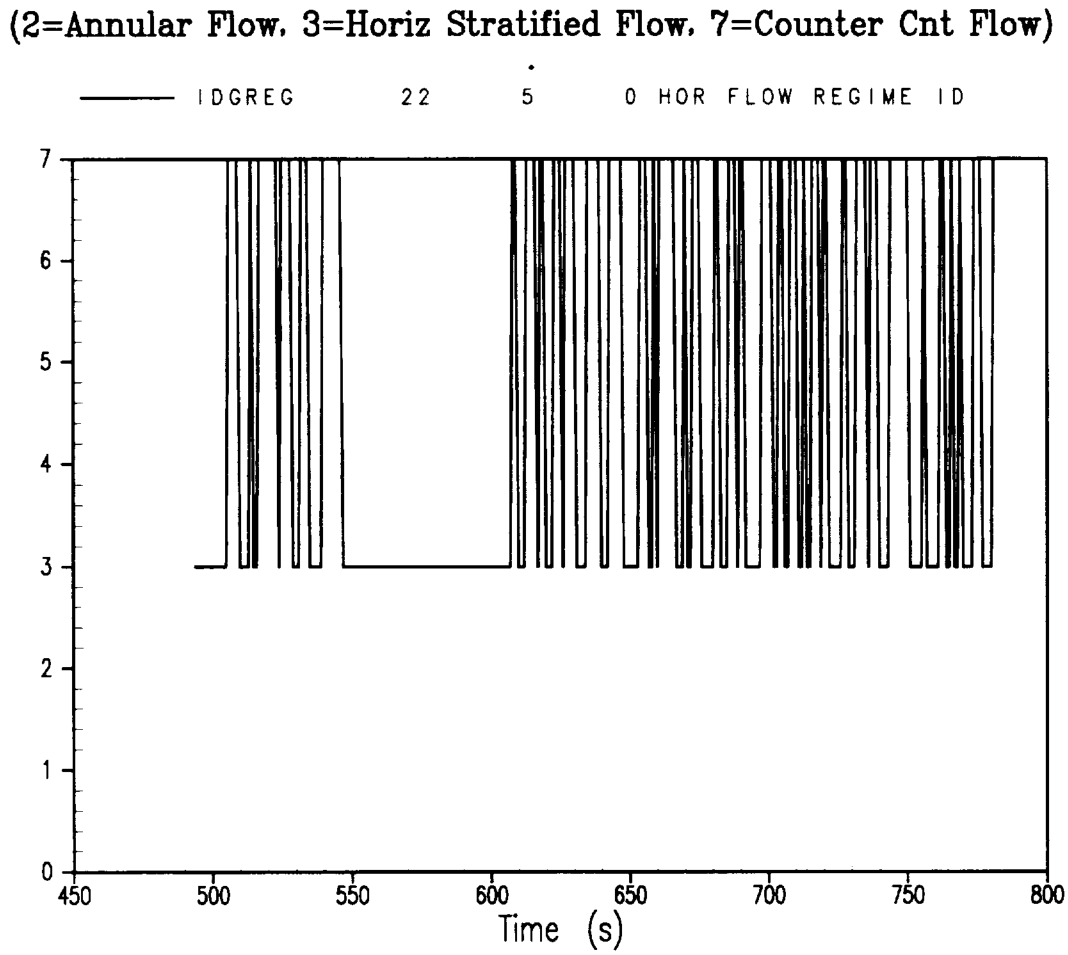


Figure 440.163-26
Horizontal Flow Regime, Pressurizer Loop Hot Leg, DEDVI Break

(2=Annular Flow, 3=Horiz Stratified Flow, 7=Counter Cnt Flow)

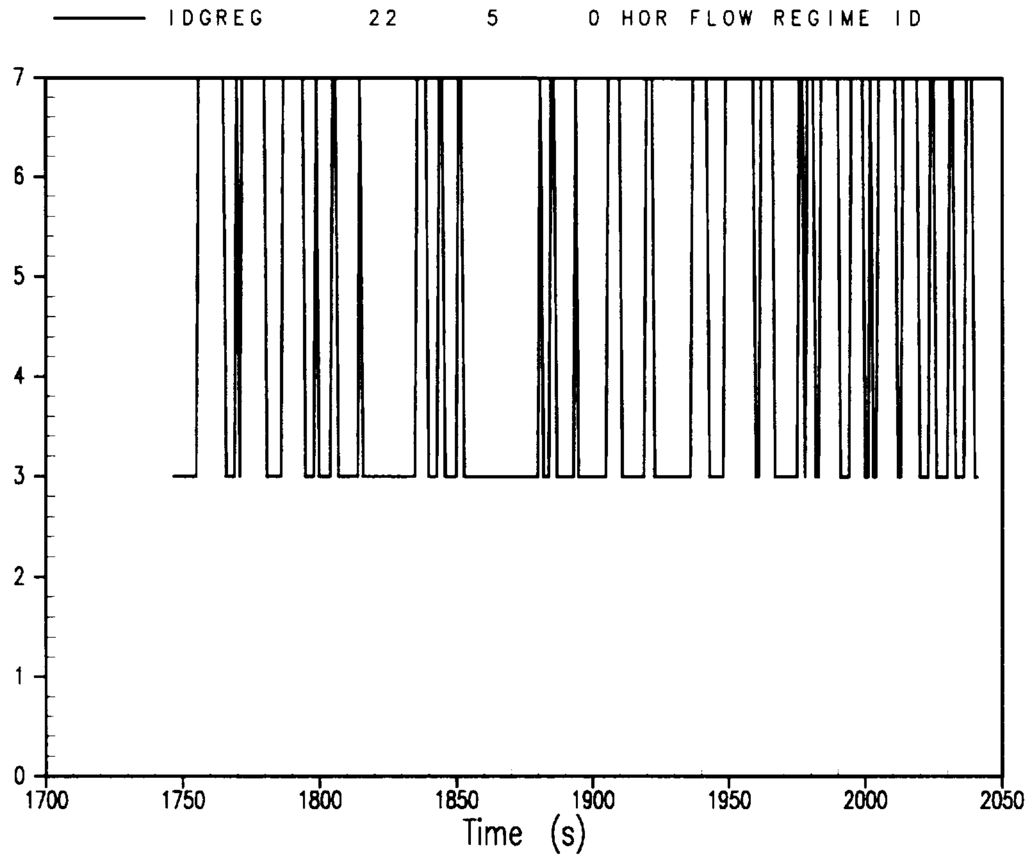


Figure 440.163-26a
Horizontal Flow Regime, Pressurizer Loop Hot Leg, Inadvertant ADS Case

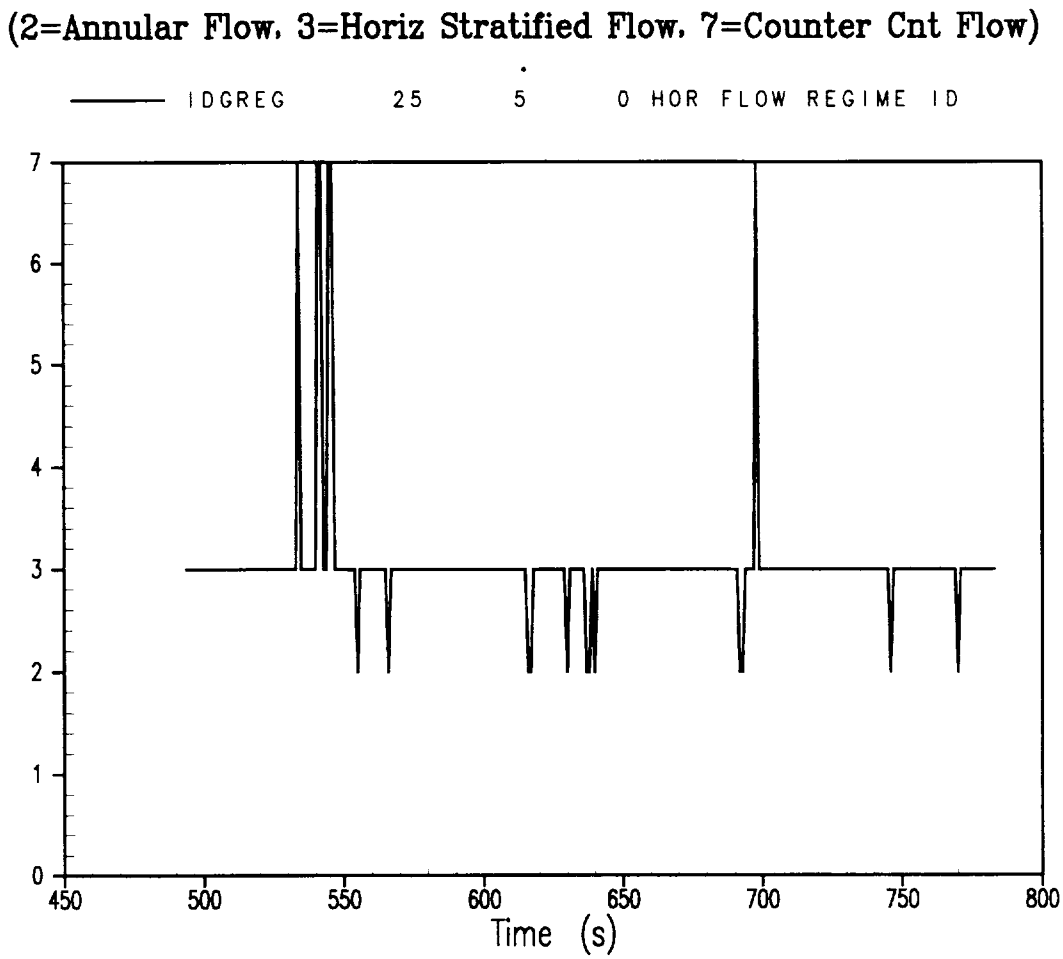


Figure 440.163-27
Horizontal Flow Regime, 2*ADS-4 Loop Hot Leg, DEDVI Break

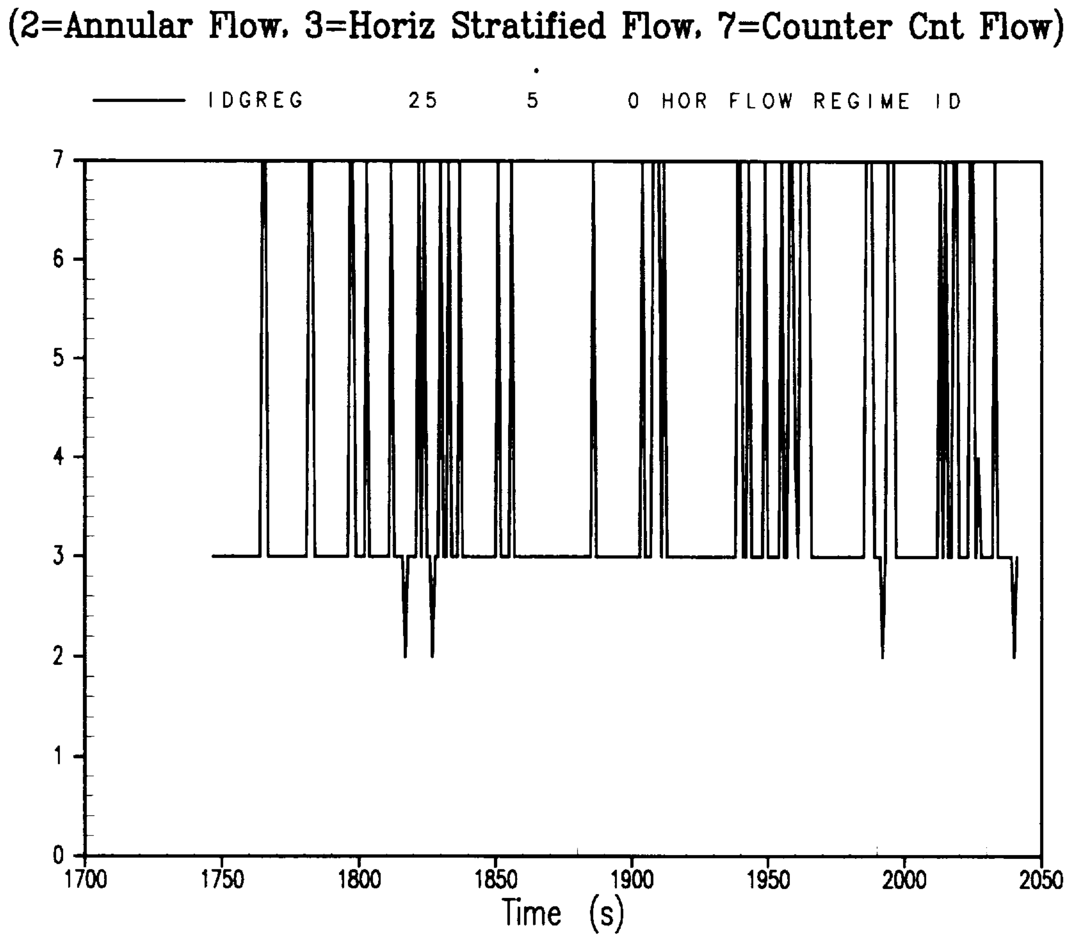


Figure 440.163-27a
Horizontal Flow Regime, 2*ADS-4 Loop Hot Leg, Inadvertant ADS Case

RAI Number: 440.168

Question:

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

Reference

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

Westinghouse Response:

Figure 5 of the paper by Soliman and Sims [1] shows that at a Froude number of 3.5, the data obtained by Armstrong et. al. [2] for the critical height (h) associated with the onset of liquid entrainment from a stratified fluid into a side-oriented offtake is within about 1% of that predicted by the Craya [3] model. In the range of Froude number for AP1000 (i.e. 2 to 4), Figure 5 also shows that the difference between the prediction of the critical onset height from the Craya model and the model suggested by Soliman and Sims is within 10% as stated in Section A.2.1 of WCAP-15833. Similarly, in Figure 3 [1], the predictions of h/d for the two models are in good agreement in the range of Froude number expected for AP1000. From Figures 3-5, the "significant deviations" between the two models seem to occur for Froude numbers less than about 1.

The main reason supporting the use of the top offtake orientation model is that while it may neglect viscous and surface tension forces and be independent of offtake diameter, it provides a reasonable prediction of the available data for liquid entrainment from stratified surfaces through vertical offtakes [4].

References

[2] Armstrong et al., "Theoretical and experimental study of the onset of liquid entrainment during dual discharge form large reservoirs", Int. J. Multiphase Flow, Vol. 18, pp. 217-227, 1992.

[3] Craya, A., "Theoretical research on the flow of non-homogeneous fluids", Houille Blanche 4, pp. 44-55, 1949.

[4] Yonomoto, T. and Tasaka, K., "Liquid and Gas Entrainment to a Small Break Hole from a Stratified Two-Phase Region", Int. J. Multiphase Flow, Vol. 17, pp. 745-765, 1991.

Design Control Document (DCD) Revision:

None

PRA Revision:

None

WCAP Revision:

None