In addition, the trip of main feedwater pumps provides an another level of safety. Although the MFW pump trip is not safety-related it is expected to be reliable. The use of the trip of the main feedwater pumps as a backup to the MFIV for the AP1000 should be acceptable for the following reasons;

- The same approach is used for AP600 and most operating plants
- The MFW pump trip is only required in unlikely case that a MFCV opens excessively and
  - MFIV fails to close
  - The failure that initially opened the MFCV was such that the safety signal could not override the failure and close the valve
- The trip of the MFW pumps is achieved by opening the breaker associated with its electric motor drive. The opening of a simple breaker is expected to be reliable.
- Safety actuation signals, separate from the MFIV signals, are used to trip the MFW pumps
- Technical Specification limiting conditions for operation are specified for the MFW pump trip in the DCD Technical Specifications Table 3.3.2.

**Design Control Document (DCD) Revision:**

*Revise minimum DNBR value in the Chapter 15.1.2.2.2.*

Transient results show the increase in nuclear power and ΔT associated with the increased thermal load on the reactor (see Figures 15.1.2-1 and 15.1.2-2). A new equilibrium condition is reached and all the plant parameters, except for the SG water level, remain almost constant. Following the turbine trip, the consequential loss of offsite power produces the reactor coolant system flow coastdown shown in Figure 15.1.2-3. The minimum DNBR is predicted to occur before the reactor trip and the reactor coolant pump coastdown caused by the loss of offsite power. The minimum DNBR predicted is 2.14~~2.21~~ using the WRB-2 equation, which is well above the design limit described in Section 4.4. Following the reactor trip, the plant approaches a stabilized and safe condition; standard plant shutdown procedures may then be followed to further cool down the plant.

*Correct time of minimum DNBR in Table 15.1.2-1 (Sheet 1 of 2).*

Table 15.1.2-1 (Sheet 1 of 2)

**TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT RESULT IN AN INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM**

| Accident | Event | Time (seconds) |
|---|---|---|
| Excessive increase in secondary system | | |
| – Manual reactor control (minimum moderator feedback) | 10-percent step load increase | 0.0 |
| | Equilibrium conditions reached (approximate time only) | 250.0 |
| – Manual reactor control (maximum moderator feedback) | 10-percent step load increase | 0.0 |
| | Equilibrium conditions reached (approximate time only) | 70.0 |
| – Automatic reactor control (minimum moderator feedback) | 10-percent step load increase | 0.0 |
| | Equilibrium conditions reached (approximate time only) | 125.0 |
| – Automatic reactor control (maximum moderator feedback) | 10-percent step load increase | 0.0 |
| | Equilibrium conditions reached (approximate time only) | 50.0 |
| Feedwater system malfunctions that result in an increase in feedwater flow | One main feedwater control valve fails fully open | 0.0 |
| | Turbine trip/feedwater isolation and reactor trip on high steam generator level | 201.9 |
| | Rod motion begins | 203.9 |
| | Loss of offsite power occurs | 204.9 |
| | Minimum DNBR occurs | 205.8~~112.2~~ |

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 440.072

***Question:***

As stated on page 15.2-15 of Chapter 15, in the analysis of the turbine trip event, the initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-operation. The use of initial plant parameters without inclusion of measurement uncertainties may not predict a highest peak RCS pressure and may be non-conservative from consideration of over-pressurization point of view.

Address acceptability of the turbine trip analysis in determination of the peak RCS pressure.

**Westinghouse Response:**

The text in Section 15.2.3.2.1 is incorrect with respect to the initial conditions assumed in the analyses and will be corrected as shown below. Two sets of turbine trip analyses are performed for Section 15.2.3 of the DCD. One set of analyses is performed for DNB purposes and a second set is performed for peak pressure purposes. The set of cases performed for evaluating minimum DNBR use initial conditions without the inclusion of measurement uncertainties. The cases used for evaluating peak RCS pressure have measurement uncertainties included.

Table 440.072-1 summarizes the eight turbine trip event cases analyzed and presented in Section 15.2.3 of the DCD. Cases A1, A2, B1, & B2 are analyzed for the evaluation of minimum DNBR and measurement uncertainties are not included in the initial conditions. Table 440.072-2 summarizes the initial conditions used for the minimum DNBR cases. Uncertainties in initial conditions are included in the DNBR limit.

Cases C1, C2, D1, and D2 as identified in Table 440.072-1 are analyzed for the evaluation of peak RCS pressure. These cases use initial conditions with the inclusion of measurement uncertainties. Table 440.072-3 summarizes the initial conditions and uncertainties assumed in these cases. It should be noted that the initial condition uncertainties assumed in the peak pressure cases can be observed in the transient figures presented in the DCD. For example, DCD Figure 15.2.3-15 shows the transient nuclear power for Cases C1 and C2 and DCD Figure 15.2.3-21 shows the transient nuclear power for Cases D1 and D2. From the DCD figures it can be observed that the power at Time=0.0 is 102% and that measurement uncertainty has been included.

The limiting pressure case identified in Table 440.072-1 is Case C2. This case uses initial conditions with measurement uncertainties included. The peak pressure obtained is 2694. psia and the pressure is maintained within 110% of design pressure (2748.5 psia).

Table 440.072-1 Summary of DCD Section 15.2.3 Turbine Trip Analysis Cases

| Case Description | | | | | Initial Conditions Measurement Uncertainty Inclusion | Peak RCS Pressure, psia | DCD Figures |
|---|---|---|---|---|---|---|---|
| Case Purpose | ID | Pressurizer Pressure Control | Reactivity Feedback | Offsite Power Availability | | | |
| Minimum DNBR Cases | A1 | With | Minimum | With | No (See Table 440.072-2) | 2671. | 15.2.3-1 to 7 |
| | A2 | With | Minimum | Without | No (See Table 440.072-2) | 2650. | 15.2.3-1 to 7 |
| | B1 | With | Maximum | With | No (See Table 440.072-2) | 2615. | 15.2.3-8 to 14 |
| | B2 | With | Maximum | Without | No (See Table 440.072-2) | 2533. | 15.2.3-8 to 14 |
| Peak RCS Pressure Cases | C1 | Without | Minimum | With | Yes (See Table 440.072-3) | 2680. | 15.2.3-15 to 20 |
| | C2 | Without | Minimum | Without | Yes (See Table 440.072-3) | 2694. | 15.2.3-15 to 20 |
| | D1 | Without | Maximum | With | Yes (See Table 440.072-3) | 2626. | 15.2.3-21 to 26 |
| | D2 | Without | Maximum | Without | Yes (See Table 440.072-3) | 2590. | 15.2.3-21 to 26 |

| Table 440.072-2 Initial Conditions Used for Minimum DNBR Analyses | |
|---|---|
| Initial Condition | Value Used |
| Initial Power | 100% of 3415 MWt |
| Initial RCS Average Temperature | nominal value (573.6 °F) |
| Initial Pressurizer Pressure | nominal value(2250 psia) |
| Initial Vessel Total Flow | 301670. gpm,<br>Minimum Measured Flow corresponding to 10% steam generator tube plugging |

| Table 440.072-3 Initial Conditions Used for Peak RCS Pressure Analyses | |
|---|---|
| Initial Condition | Value Used |
| Initial Power | 102% of 3415 MWt |
| Initial RCS Average Temperature | nominal value plus uncertainty<br>573.6 °F +6.5 °F = 580.1 °F |
| Initial Pressurizer Pressure | nominal value ± uncertainty (see Note 1 below) |
| Initial Vessel Total Flow | 296000. gpm,<br>Thermal Design Flow corresponding to 10% steam generator tube plugging |
| (1) Nominal pressurizer pressure is 2250 psia. An uncertainty of ± 50 psi is assumed. Cases that trip on high pressurizer pressure assume the pressure is initially at 2200 psia. Assuming the initial pressurizer pressure is low delays reaching the high pressurizer pressure trip. Cases that do not trip on high pressurizer pressure assume the initial pressurizer pressure is 2300 psia. | |

## Design Control Document (DCD) Revision:

### Initial Operating Conditions

~~The accident is analyzed using the revised thermal design procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397-P-A (Reference 5).~~

Two sets of initial operating conditions are used . Cases performed to evaluate the minimum DNBR obtained are analyzed using the revised thermal design procedure. Initial core power, reactor coolant temperature, and pressure are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397-P-A (Reference 5).

Cases performed to evaluate the maximum calculated RCS pressure include uncertainties on the initial conditions. Initial core power, reactor coolant temperature, and pressure are assumed to be at the nominal full power values plus or minus uncertainties. The direction of the uncertainties is chosen to maximize the RCS pressure.

## PRA Revision:

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number. 440.097

***Question:***

The documentation of the large break LOCA (LBLOCA) analysis methodology and results in Section 15.6.5.4A is totally inadequate.

Please include additional information comparable in content and detail to the small break LOCA (SBLOCA) and the long-term cooling.

**Westinghouse Response:**

The large break LOCA results described in Section 15.6.5.4A will be expanded as shown in the attached draft markup.

**Design Control Document (DCD) Revision:**

DCD Chapter 15 will be modified as shown in the attachment.

**PRA Revision:**

None

**15.6.5.4A Large-break LOCA Analysis Methodology and Results**

Westinghouse applies the WCOBRA/TRAC computer code to perform best-estimate large-break LOCA analyses in compliance with 10 CFR 50 (Reference 5). WCOBRA/TRAC is a thermal-hydraulic computer code that calculates realistic fluid conditions in a PWR during the blowdown and reflood of a postulated large-break LOCA. The methodology used for the AP1000 analysis is documented in WCAP-12945-P-A and WCAP-14171, Revision 2 (References 10 and 11).

The NRC staff has reviewed and approved the best-estimate LOCA methodology documented in Reference 10 for estimating the 95th percentile PCT (Reference 8) for three-loop and four-loop Westinghouse pressurized water reactors (PWRs). In the methodology approved for three- and four-loop Westinghouse PWRs, three major components of uncertainty are considered. The initial conditions uncertainty component addresses variations and uncertainties in the initial fluid conditions in the reactor coolant system and the emergency core cooling system boundary conditions. The power distribution uncertainty component addresses variations and uncertainties in power-related parameters, such as peaking factors and axial power distributions. The model uncertainty component addresses uncertainties in the code models that affect the overall system transient ("global" models), as well as those which affect the hot rod only ("local" models). The WCOBRA/TRAC code is used to calculate the effects of initial conditions, power distributions, and global models, and the HOTSPOT code is used to calculate the effects of local models. Biases and uncertainties due to the assumption that the initial conditions, power distribution, and model uncertainty components can be linearly combined are quantified and accounted for.

In addition to the code uncertainty estimates quantified in the model uncertainty component, a separate code uncertainty has been estimated based on direct comparisons of WCOBRA/TRAC predictions to experimental data. This estimate is considered in the appropriate step of the methodology used to develop the overall uncertainty. Finally, the uncertainty of the experimental data has been quantified. This uncertainty is also considered in the appropriate step of methodology used to develop the overall uncertainty.

A simplification of this methodology was approved for the AP600 in Reference 3. The parameters important to the initial conditions and power distribution uncertainty components are set to bounding values established by sensitivity studies. The model uncertainty component is quantified in the same way as for three- and four-loop plants, in cases where the other parameters are set to their bounding values. The code uncertainty estimate based on direct comparisons with data and the uncertainty in the experimental data itself, is also considered in the overall uncertainty estimate. A discussion of the AP600 large-break LOCA uncertainty methodology is given in WCAP-14171, Revision 2 (Reference 11). As stipulated in the Reference 3 approval, a PCT bias is included in the 95th percentile blowdown and reflood PCT results to account for the sensitivity to eliminating the operation of the CMT and PRHR from the WCOBRA/TRAC calculation.

For the AP1000 large-break LOCA analysis, the best-estimate LOCA analysis methodology approved for AP600 by the NRC Staff is applied as described in Reference 11. The plant boundary conditions for WCOBRA/TRAC, including the initial operating conditions and the core power distribution, are bounded in a conservative manner based on initial sensitivity studies investigating the range of AP1000 possible values. The modeling bias and uncertainty is then evaluated. This component accounts for uncertainties in the ability of the WCOBRA/TRAC code

to accurately predict important phenomena affecting the overall system response ("global" parameters) and the local fuel rod response ("local" parameters). The code and model bias is the difference between the reference transient PCT, which assumes nominal values for the global and local parameters, and the average PCT, taking into account the possible values of global and local parameters.

**15.6.5.4.1A General Description of WCOBRA/TRAC Modeling**

**WCOBRA/TRAC is the best estimate thermal-hydraulic computer code used to calculate realistic fluid conditions in the PWR during blowdown and reflood of a postulated large-break LOCA.**

**The WCOBRA/TRAC Code Qualification Document (Reference 10) contains a complete description of the code models and justifies their applicability to PWR large break LOCA analysis.**

**Table 15.6.5-4 lists the AP1000-specific parameters identified for use in the large-break LOCA analysis. WCOBRA/TRAC studies were performed for AP600 to establish sensitivities to parameter variations. A spectrum of large-break LOCA sensitivity cases considered different values of the AP600 initial condition and power distribution parameters; ranges of parameters in the studies performed for the AP600 are reported in Reference 7. Some of these parameter studies were performed again for AP1000 to evaluate the effect of changes in key initial plant conditions over their expected range of operation. These studies included effects of ranging $T_{avg}$, steam generator tube plugging, core burnup, and hot assembly location. The calculated results were used to identify bounding conditions which are then used in the AP1000 reference transient.**

**The WCOBRA/TRAC vessel modalization is developed from plant design drawings to divide the vessel into 10 vertical sections. The bottom of section 1 is the inside vessel bottom, and the top of section 10 is the inside top of the vessel upper head. In addition to the major downcomer and core flow paths, the modeled bypass flow paths are the upper head cooling spray, guide thimbles, and core bypass. After defining the elevations for each section, a noding scheme is defined for the WCOBRA/TRAC model as shown in WCAP-14171, Revision 2 (Reference 11). WCOBRA/TRAC assumes a vertical flow path for vertically stacked channels, unless specified otherwise in the input. Positive flow for the vertically connected channels (and cells) is upward. Several of the 10 sections are divided vertically into 2 or more levels; these levels are referred to as cells within a channel.**

**The WCOBRA/TRAC loop model represents the major primary, secondary, and passive safety systems components. Both loops are explicitly modeled, including the hot leg, steam generator, and the two cold legs and associated pumps. The loop designated "2" has the pressurizer and the PRHR system connections, and loop "1" cold legs have the core makeup tank pressure balance line connections. The reactor coolant pump models contain the AP1000 homologous curves together with appropriate two-phase head and torque multipliers and degradation data. AP1000 values for pump coastdown characteristics are also applied. The passive safety features are modeled using design data for elevations,**

liquid volumes, and line losses. Because the ADS is not actuated until long after the time of PCT in large-break LOCA events, it is not modeled in detail.

15.6.5.4.2A Steady-state Calculation

A WCOBRA/TRAC LOCA calculation is initiated from a point at which the flows, temperatures, powers, and pressures are at their approximate steady-state values before the postulated break occurs. Steady-state WCOBRA/TRAC calculations are run for a brief time period to verify that the calculated conditions are steady and that the desired reactor conditions are achieved.

The values used to set the steady-state plant conditions reflect the AP1000 parameters for reactor coolant pump flows, core power, and steam generator tube plugging levels. The fuel parameters provide the steady-state fuel temperatures, pressures, and gap conductances as a function of fuel burnup and linear power. The calculated fuel temperatures from WCOBRA/TRAC are adjusted to match the specified fuel data by adjusting the gap heat transfer coefficient between the pellet and the cladding. Once the vessel fluid temperatures, flows, pressures, loop pressure drop, and core parameters are in agreement with the desired values and are steady, a suitable initial condition is achieved.

15.6.5.4.3A Signal Logic for Large-break LOCA

The reactor trip signal occurs due to compensated pressurizer pressure within the first second of the large-break transient. Because control rod insertion is not modeled in WCOBRA/TRAC, no effects on reactivity ensue. A safeguards "S" signal occurs due to containment high pressure at 2.2 seconds of large-break LOCA transients.

As a consequence of this signal, after appropriate delays, the PRHR and core makeup tank isolation valves open and containment isolation occurs. The rapid depressurization of the primary system during a large-break LOCA leads to the initiation of accumulator injection early in the large-break transient. The accumulator flow diminishes core makeup tank delivery to such an extent that the core makeup tank level does not approach the ADS Stage 1 valve actuation point until after the accumulator tank is empty. The accumulator empties long after the blowdown portion of the large-break LOCA transient is complete. Actuation of the ADS on CMT water level does not occur until long after the AP1000 PCT is calculated to occur.

15.6.5.4.3A Transient Calculation

Once the steady-state calculation is found to be acceptable, the transient calculation is initiated. The semi-implicit pipe break model is added to the desired break location. The containment back-pressure is specified consistent with WCAP-14171, Revision 2 (Reference 11) methodology. The steady-state calculation is restarted with the above changes to begin the transient.

The calculation is continued until the fuel rods are quenched. Passive safety injection flow into the vessel from the accumulators is larger than the break flow for as long as the accumulators discharge.

Table 15.6.5-5 shows a general sequence of events following a large cold leg break LOCA and the relationship of these events to the blowdown and reflood portion of the transient.

15.6.5.4.4A Large-break LOCA Analysis Results

For the AP1000 large-break LOCA analysis, the best-estimate LOCA analysis methodology approved for AP600 is applied as follows. The plant boundary conditions for WCOBRA/TRAC, including the initial operating conditions and the core power distribution, are bounded in a conservative manner based on the sensitivity studies that investigated the range of AP600 possible values. Studies were reperformed for AP1000 to establish the bounding values for the AP1000 reference transient.

Conceptually, the following equation defines the effect on the reference transient PCT of the uncertainties due to global model parameter variations:

$$PCT_i = PCT_{REF,i} + \Delta PCT_{MOD,i}$$

where,

$PCT_{REF,i}$ = Reference transient PCT: The reference transient PCT is calculated using WCOBRA/TRAC at the bounding initial conditions and distribution for blowdown (i = 1) and reflood (i = 2).

$\Delta PCT_{MOD,i}$ = Model bias and uncertainty: This component accounts for uncertainties in the ability of the WCOBRA/TRAC code to accurately predict important phenomena affecting the overall system response ("global" parameters) and the local fuel rod response ("local" parameters). The code and model bias is the difference between the reference transient PCT, which assumes nominal values for the global and local parameters, and the average PCT, taking into account the possible values of global and local parameters. The global model matrix for AP1000 is presented in Reference 11.

Reference 3 indicated the application restrictions on the AP600 methodology. The AP1000 large break LOCA analysis has complied with those restrictions. The global model matrix of calculations and the final 95 percent uncertainty calculations have been performed for AP1000. The reference transient was reanalyzed to address the sensitivity to the modeling of the CMT and PRHR. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the reference transient PCT. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the reference transient result.

Figures 15.6.5A-1 through 15.6.5A-12 present the parameters of principal interest for the Reference Case DECLG break analysis. Values of the following parameters are presented:

The highest calculated clad temperature at any elevation for the five fuel rods modeled

- Hot rod cladding temperature transient at the limiting elevation for PCT
- Core fluid mass flows at top of core for the fuel assemblies modeled in WCOBRA/TRAC
- Core pressure
- Break flow rates
- Core and downcomer collapsed liquid levels
- Accumulator water flow rate
- Intact loop core makeup tank flow rate

Cold leg breaks are analyzed because the hot leg break location is nonlimiting in the large-break LOCA best-estimate methodology. The DECLG break was shown to be more limiting than the limiting size split break.

In all cases analyzed, the bounding core design values of Fq (2.60) and FdH (1.65) are applied to the hot rod, and 102 percent of nominal core power is assumed.

15.6.5.4.5A Description of AP1000 Large-Break LOCA Transient

A description of the reference transient DECLG break with bounding initial and boundary conditions follows. The sequence of events is presented in Table 15.6.5-6. The break was modeled to occur in one of the cold legs in the loop containing the core makeup tanks. Shortly after the break opens, the vessel rapidly depressurizes and the core flow quickly reverses. The hot assembly fuel rods dry out and begin to heat up (Figures 15.6.5A-1 and 2) during the flow reversal (Figure 15.6.5A-3). In Figure 15.6.5A-3, FGM is the vapor flow rate at the top of the hot assembly, FEM is the entrained liquid drop flow rate at that location, and FLM is the continuous liquid flow rate at that location.

In Figure 15.6.5A-1, Rod 1 refers to the hot rod at the maximum allowed linear heat rate, Rod 2 represents the average rod in the hot assembly which contains the hot rod, Rod 3 represents the open hole/support column rod, Rod 4 represents the guide tube rod, and Rod 5 represents the peripheral fuel assembly rod.

The steam generator secondaries are assumed to be isolated immediately at the inception of the break to maximize their stored energy. The massive size of the break causes an

immediate, rapid pressurization of the containment. At 2.2 seconds of the transient, credit is taken for receipt of an "S" signal due to high-2 containment pressure. Applying the pertinent signal processing delay means that the valves isolating the core makeup tanks from the direct vessel injection line and the PRHR begin to open at 4.2 seconds into the transient. The reactor coolant pumps are presumed to trip immediately following the break. Core shutdown occurs due to voiding; no credit is taken for the control rod reactivity effect.

The system depressurizes rapidly (Figure 15.6.5A-4) as the initial mass inventory is depleted due to break flow. The pressurizer drains completely approximately 25 seconds into the transient, and accumulator injection commences 15 seconds into the transient (Figure 15.6.5A-5). Accumulator actuation shuts off core makeup tank flow (Figure 15.6.5A-6), which has been delivering since the isolation valve opened. The core makeup tank (CMT) liquid level remains well above the ADS Stage 1 actuation setpoint throughout the AP1000 DECLG LOCA cladding temperature excursion, even though CMT injection begins again at 215 seconds.

The dynamics of the reference transient are shown in terms of the flow rates of liquid, vapor, and entrained liquid at the top of the core (Figures 15.6.5A-7 through 9) for the peripheral, open hole/support column average power interior, and guide tube average power interior assemblies (the corresponding figure for the hot assembly is Figure 15.6.5A-3). The variables plotted are the same as those in Figure 15.6.5A-3 for the respective assemblies.

Figures 15.6.5A-8 and 9 illustrate the impact of upper head drain through the guide tubes and upper core plate holes, respectively, on core flow. While liquid remains in the upper head above the top of the guide tubes, the guide tubes (Figure 15.6.5A-8) are the preferred path for draining of liquid into the upper plenum. Top of core liquid flow is relatively stagnant for the first few seconds; once the upper head begins to flash, liquid drains directly down the guide tubes and that fraction that is able to penetrate into the core does so, at a maximum flow rate exceeding 2000 lbm/sec of total liquid flow between 5 and 18 seconds. At that point, the flow entering the guide tubes in the upper head is largely steam; residual liquid is supplied to the guide tube fuel assemblies at a constant or decreasing rate out to 42 seconds.

Figure 15.6.5A-9 presents the open hole/support column assembly top of core flow behavior. In contrast to the guide tubes, flow of liquid down into the core open hole/support column locations does not become significant until about 9 seconds of the reference transient. Between 11 and 18 seconds, the combined flow of continuous and entrained liquid is 600 to 1500 lbm/sec; the entrained liquid flow continues to be significant until 30 seconds. After 10 seconds of transient, the downflow pattern in the open hole/support column locations and the guide tubes is established to the extent that vapor downflow is also predicted. Thus, there exists good flow of liquid into the top of the core at these locations from before 10 seconds to after 20 seconds. The flow in the open hole and guide tube assemblies is sufficient to quench the fuel in each respective assembly (Rod 3 and Rod 4 respectively in Figure 15.6.5A-1).

Liquid downflow is delayed into the hot assembly. By 10 seconds into the transient, liquid that has built up in the global region above the upper core plate begins to flow through the plate at the hot assembly location and then proceeds into the core (Figure 15.6.5A-3). Significant flow of continuous and/or entrained droplet liquid into the hot assembly exists from 10 to 22 seconds. The liquid flow is not enough to quench the hot rod and hot assembly rod at all elevations (Figure 1), although effective cooling is achieved.

Figure 15.6.5A-7 demonstrates that liquid downflow exists through the top of the peripheral core assemblies from 2 seconds through about 26 seconds in the reference DECLG transient. The power of the fuel in this region is almost identical to that in the open hole and guide tube locations, so the clad temperature profiles are similar.

About 15 seconds into the transient, the accumulator begins to inject water into the upper downcomer region, most of which is initially bypassed to the break. At approximately 25 seconds, accumulator water begins to flow into the lower plenum. Break flow rates through the loop (Figure 15.6.5A-10) and vessel (Figure 15.6.5A-11) sides of the break diminish as the transient progresses. At approximately 70 seconds, the lower plenum fills to the point that water begins to reflood the core from below. The void fraction at the core bottom begins to decrease and as time passes, core cooling increases substantially. The cladding temperature begins to decrease once the core water level has risen high enough in the core. Figure 15.6.5A-12 presents the collapsed liquid levels in the core referenced to the bottom elevation of the active fuel (solid line) and downcomer (dashed line) referenced to the bottom of the reactor vessel.

15.6.5.4.6A Global Model Sensitivity Studies and Uncertainty Evaluation

The global model run matrix developed for the approved best-estimate LOCA methodology was analyzed to evaluate the effect of broken loop resistance, break discharge coefficient, and condensation rate on the PCT for the guillotine break. These parameters were varied singly and in combination to obtain a data base that could be used for response surface generation. The run matrix and ranges of the break flow parameters are described in WCAP-14171, Revision 2 (Reference 11). The bounding power shape and initial conditions identified for AP1000 by sensitivity study are used.

Further, studies of split breaks with areas equal to between 1.4 and 1.8 times the cold leg area were performed to identify the limiting split break area. The calculated results from these additional split breaks are summarized in Table 15.6.5-7.

The calculated results were used to develop a response surface by regression analysis. This is then used in the uncertainty evaluation to predict the PCT uncertainty component resulting from uncertainties in global model parameters, $\Delta PCT_{MOD,i}$. An uncertainty evaluation is performed solely for the global model parameters because the initial condition and power distribution parameters are bounded.

**The PCT equation as presented before requires evaluation of the element of uncertainty associated with the $\Delta PCT_{MOD,i}$ term. Separate initial PCT frequency distributions are constructed as follows for the guillotine break and the limiting split break size:**

1. **Generate a random value of the ΔPCT element**

2. **Calculate the resulting PCT**

3. **Repeat the process many times to generate a histogram of PCTs**

**The results of this assessment indicated the DECLG (guillotine) break is limiting for the AP1000.**

**To account for the uncertainty due to statistical approximation methods, several global model runs are imposed to identify the PCT-based code uncertainty for a final Monte Carlo simulation of the guillotine break PCT. Results obtained in the determination of the 95th percentile PCT are presented in Table 15.6.5-8.**

**15.6.5.4.7A Large Break LOCA Conclusions**

~~Table 15.6.5-4 lists the AP1000-specific parameters identified for use in the large-break LOCA analysis. A spectrum of large-break LOCA sensitivity cases considered different values of the AP600 initial condition and power distribution parameters. Ranges of parameters in the studies performed for the AP600 are reported in Reference 7. Some of these parameter studies were performed for AP1000 to evaluate the effect of change in key initial plant conditions over their expected range of operation. These studies included effects of ranging $T_{avg}$, steam generator tube plugging, core burnup, and hot assembly location. The calculated results were used to identify the bounding conditions which are then used in the AP1000 reference transient.~~

In accordance with 10 CFR 50.46, the conclusions of the best-estimate large-break LOCA analysis are that there is a high level probability that the following criteria are met.

1. The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature (PCT)) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (i.e., maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling.

Note that criterion 4 has historically been satisfied by adherence to criteria 1 and 2, and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. Criteria 1 and 2 are satisfied for best-estimate large-break LOCA applications. The approved methodology specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low power channel as defined in the WCOBRA/TRAC model. This situation has not been calculated to occur for the AP1000. Therefore, acceptance criterion 4 is satisfied.

5. After successful initial operation of the emergency core cooling system (ECCS), the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

   Criterion 5 is satisfied if a coolable core geometry is maintained, and the core is cooled continuously following the LOCA. The AP1000 passive safety systems provide effective post-LOCA long-term cooling.

Table 15.6.5-8~~5~~ presents the calculated 50th and 95th percentile PCT, maximum cladding oxidation, maximum hydrogen generation, and core cooling results.

Based on the analysis, the Westinghouse Best-Estimate Large-Break LOCA methodology has shown that the acceptance criteria of 10 CFR 50.46 are satisfied for AP1000.

## 15.6.6 References

1. 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," and Appendix K to 10 CFR 50, "ECCS Evaluation Models."

2. American Nuclear Society Proposed Standard, ANS 5.1 "Decay Energy Release Rates Following Shutdown of Uranium-Cooled Thermal Reactors," October (1971), Revised October (1973).

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

4. Dederer, S. I., Hochreiter, L. E., Schwartz, W. R., Stucker, D. L., Tsai, C.K., and Young, M. Y., "Westinghouse Large-Break Best Estimate Methodology, Volume 1 Model Description and Validation, Volume 2, Revision 2, Application to Two-Loop PWRs Equipped with Upper Plenum Injection," WCAP-10924-P-A (Proprietary), December 1988.

5. "Emergency Core Cooling Systems; Revision to Acceptance Criteria," Federal Register, Vol. 53, No. 180, September 16, 1988.

6. DELETED

7. "AP600 Design Control Document," Revision 3, December 1999.

8. Letter from R. C. Jones, Jr., (USNRC), to N. J. Liparulo, (W), Subject: "Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse CQD for Best Estimate LOCA Analysis," June 28, 1996.

9. DELETED

10. Bajorek, S. M., ~~Hochreiter, L. E.,~~ Young, M. Y., Dederer, S. I., Nissley, M. E., ~~Tsai, C. K., Yeh, H. C., Chow, S. K.,~~ Takeuchi, K., ~~Cunningham, J. P.,~~ and ~~Stucker, D. L.~~ Ohkawa, K., "Code Qualification Document for Best Estimate Analysis," Volumes 1 through 5, WCAP-12945-P, Revision 2~~1~~ (Proprietary), March 1998.

11. Hochreiter, L. E., et al., "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident," WCAP-14171, Revision 2, March 1998.

12. Meyer, P. E., "NOTRUMP - A Nodal Transient Small-Break and General Network Code," WCAP-10079-P-A, (Proprietary) and WCAP-10080-A (Nonproprietary), August 1985.

13. Lee, N., Rupprecht, S. D., Schwarz, W. R., and Tauche, W. D., "Westinghouse Small-Break ECCS Evaluation Model Using the NOTRUMP Code," WCAP-10054-P-A (Proprietary) and WCAP-10081-A (Nonproprietary), August 1985.

14. Carlin, E. L., Bachrach, U., "LOFTRAN & LOFTTR2 AP600 Code Applicability Document," WCAP-14234, Revision 1 (Proprietary), August 1997.

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

16. Friedland, A. J., Ray S., "Revised Thermal Design Procedure," WCAP-11397-P-A (Proprietary) and WCAP-11398-A (Nonproprietary), April 1989.

17. Kemper, R. M., "AP600 Accident Analyses - Evaluation Models," WCAP-14601, Revision 2 (Proprietary), June 1998.

18. Hargrove, H. G., "FACTRAN - A FORTRAN-IV Code for Thermal Transients in a $UO_2$ Fuel Rod," WCAP-7908 (Proprietary) and WCAP-7337 (Nonproprietary), June 1975.

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

20. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

21. Lewis, R. N., et al., "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," WCAP-10698-P-A (Proprietary) and WCAP-10750-A (Nonproprietary), August 1987.

22. "NOTRUMP Final Validation Report for AP600," WCAP-14807, Revision 5, (P), August 1998.

23. Garner, D. C., et al., "WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report," WCAP-14776, Revision 4, April 1998.

24. "AP1000 Code Applicability Report," WCAP-15644 (Proprietary) and WCAP-15707 (Nonproprietary), May 2001.

25. "AP1000 PIRT and Scaling Assessment," WCAP-15613 (Proprietary) and WCAP-15706 (Nonproprietary), February 2001.

26. Kemper, R. M., "Applicability of the NOTRUMP Computer Code to AP600 SSAR Small-Break LOCA Analyses," WCAP-14206 (Proprietary), November 1994.

27. "WCOBRA/TRAC AP1000 ADS-4/IRWST Phase Modeling," WCAP-15833 (Proprietary) and WCAP-15834 (Nonproprietary) April 2002.

28. Zuber, et al., "The Hydrodynamic Crisis in Pool Boiling of Saturated and Subcooled Liquids," Part II, No. 27, International Developments in Heat Transfer, 1961.

29. Griffith, et al., "PWR Blowdown Heat Transfer," Thermal and Hydraulic Aspects of Nuclear Reactor Safety, ASME, New York, Volume 1, 1977.

Table 15.6.5-4

**MAJOR PLANT PARAMETER ASSUMPTIONS USED IN THE BEST-ESTIMATE LARGE-BREAK LOCA ANALYSIS**

| | Parameter Value Selected | Other AP1000 Cases Analyzed/Selection Basis |
|---|---|---|
| Plant physical configuration | | |
| Steam generator tube plugging level | 0% (Bound) | 10% uniform |
| Hot assembly location | Under open hole (bound) | Under guide tube |
| Pressurizer location | In intact loop (bound) | In broken loop |
| Power-related parameters | | |
| Initial core power | 3400x1.02=3468 MW (Bound) | — |
| Core linear power | 5.977 kW/ft (core average) | — |
| Peak linear heat rate (FQ) | 2.6 (bound) | — |
| Axial power distribution | Power shape 3 (top skewed, bound) | Based on AP600 study |
| Hot rod assembly power (FΔH) | 1.65 (bound) | — |
| Hot assembly (FΔH) | 1.586 (bound) | — |
| Peripheral assembly power | 0.995 (upper bound) | 0.3 (lower bound) |
| Initial fluid conditions | | |
| Reactor coolant system average temperature | 573.6°F (bound) | Value 6.5°F higher |
| Pressurizer pressure | 2300 psia (bound) | Based on AP600 study |
| Pressurizer level (water volume) | 1000 $ft^3$ (nominal) | — |
| Accumulator temperature | 120°F (bound) | Based on AP600 study |
| Accumulator pressure | 651.7 psia (bound) | Based on AP600 study |
| Reactor coolant system boundary conditions | | |
| Containment pressure response | From WGOTHIC (lower bound) | — |
| Offsite power availability | Loss at time zero | Available[1] |

Note:

1. A reactor coolant pump automatic trip occurs at 8.2 seconds; due to an "S" signal.

| ~~Parameter~~ | ~~Value~~ |
|---|---|
| ~~Plant Physical Description~~ | |
| ~~Steam Generator Tube Plugging~~ | ~~≤ 10%~~ |
| ~~Plant Initial Operating Conditions~~ | |
| ~~Reactor Power~~ | ~~≤ 3400 MWt with 2% Calorimetric Uncertainty~~ |
| ~~Peaking Factors~~ | ~~Fq = 2.60, FΔH = 1.65~~ |
| ~~Axial Power Distribution~~ | ~~Bounding top-skewed~~ |
| ~~Fluids Conditions~~ | |
| ~~RCS Average Temperature ($T_{avg}$)~~ | ~~567°F to 580°F~~ |
| ~~Reactor Coolant Flow~~ | ~~≥ 75000 gpm/loop~~ |
| ~~Accumulator Pressure~~ | ~~≥ 652 psia~~ |
| ~~Accident Boundary Conditions~~ | |
| ~~CMT and PRHR Initiation Delay Time*~~ | ~~2.0 seconds~~ |
| ~~Offsite Power Availability Containment Pressure~~ | ~~Lost at Time of LOCA Event~~ |
| | ~~Lower bound value from WGOTHIC~~ |

~~Note:~~
~~*In accordance with the AP600 FSER restriction, cases are executed assuming the CMT and PRHR are inactive. They are non-limiting in calculated PCT compared to the reference case that models CMT and PRHR operation.~~

Table15.6.5-5
AP1000 LOCA Chronology

| | | |
|---|---|---|
| BLOWDOWN | | BREAK OCCURS |
| | | REACTOR TRIP (PRESSURIZER PRESSURE OR HIGH CONT. PRESSURE) |
| | | SI SIGNAL (HIGH CONT. PRESSURE) |
| | | CMT INJECTION BEGINS |
| | | ACCUMULATOR INJECTION BEGINS |
| | | END OF BLOWDOWN |
| | REFILL | BOTTOM OF CORE RECOVERY |
| REFLOOD | | CALCULATED PCT OCCURS |
| | | ACCUMULATORS EMPTY: CMT INJECTION COMMENCES AGAIN |
| LONG TERM COOLING | | ADS ACTIVATES ON LOW CMT LEVEL SIGNALS/IRWST ACTIVATES |
| | | IRWST EMPTY: COOLING CONTINUES VIA CIRCULATION OF SUMP WATER |

TABLE 15.6.5-6
REFERENCE TRANSIENT DECLG BREAK SEQUENCE OF EVENTS

| Event | Time (seconds) |
|---|---|
| Break occurs coincident with loss of offsite power | 0.0 |
| "S" signal occurs due to containment high-1 pressure | 2.2 |
| PRHR, core makeup tank isolation valves begin to open | 4.2 |
| Accumulator injection begins | 15 |
| End of blowdown | 47 |
| Calculated PCT occurs | 109.6 |
| Core quench occurs | 238 |

TABLE 15.6.5-7
DECL SPLIT BREAK RESULTS

| Discharge Coefficient | Blowdown PCT (°F) | Reflood PCT (°F) |
|---|---|---|
| 1.8 | 1557 | 1498 |
| 1.6 | 1529 | 1538 |
| 1.4 | 1482 | 1468 |

Table 15.6.5-~~58~~

**BEST-ESTIMATE LARGE-BREAK LOCA RESULTS**

| Parameter | Value | Criteria |
|---|---|---|
| Calculated 50th Percentile PCT (°F) (for time period of maximum 95th Percentile) | 1840 | N/A |
| Calculated 95th Percentile PCT (°F) | 2124 | 2200 |
| Maximum Cladding Oxidation (%) | < 17 | 17 |
| Maximum Hydrogen Generation (%) | < 1 | 1 |
| Coolable Geometry | Core Remains Coolable | Core Remains Coolable |
| Long-Term Cooling | Core Remains Cool in Long Term | Core Remains Cool in Long Term |

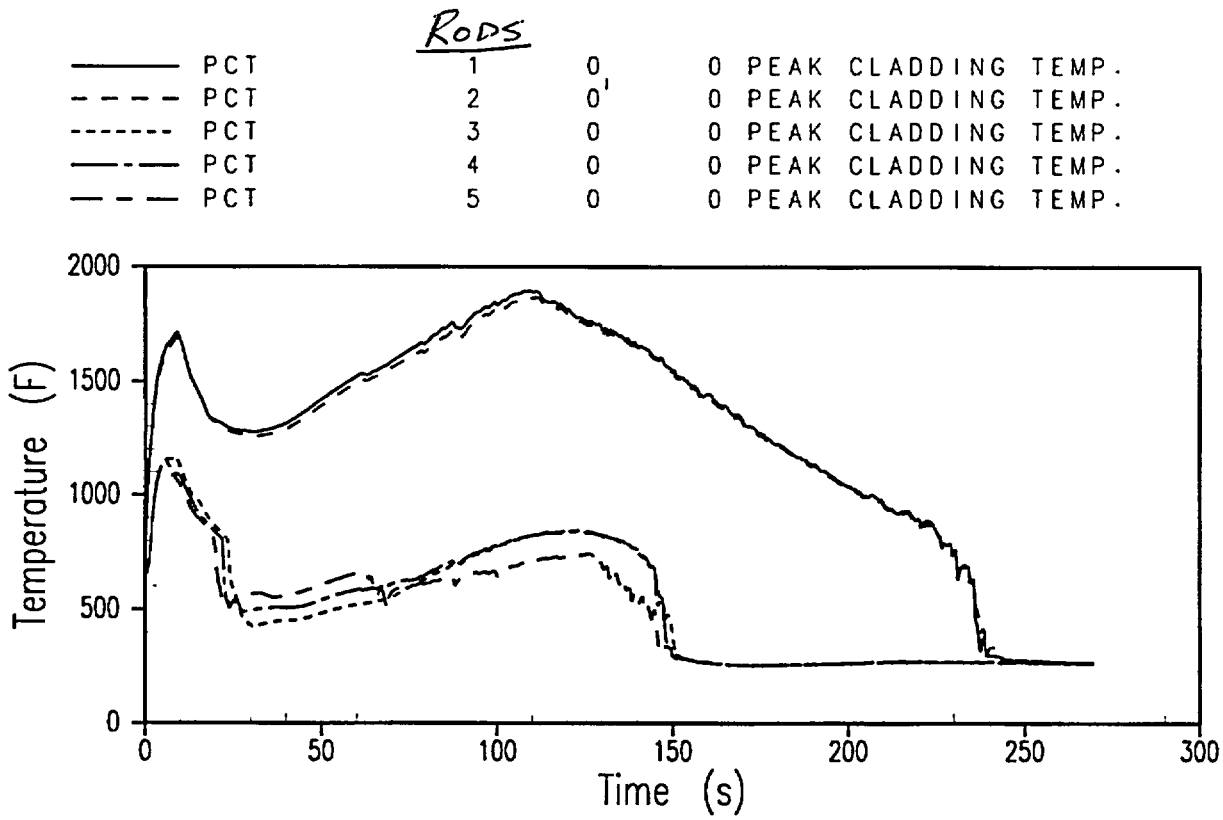



**Figure 15.6.5A-1 PCT Among All Elevations for Each Fuel Rod**

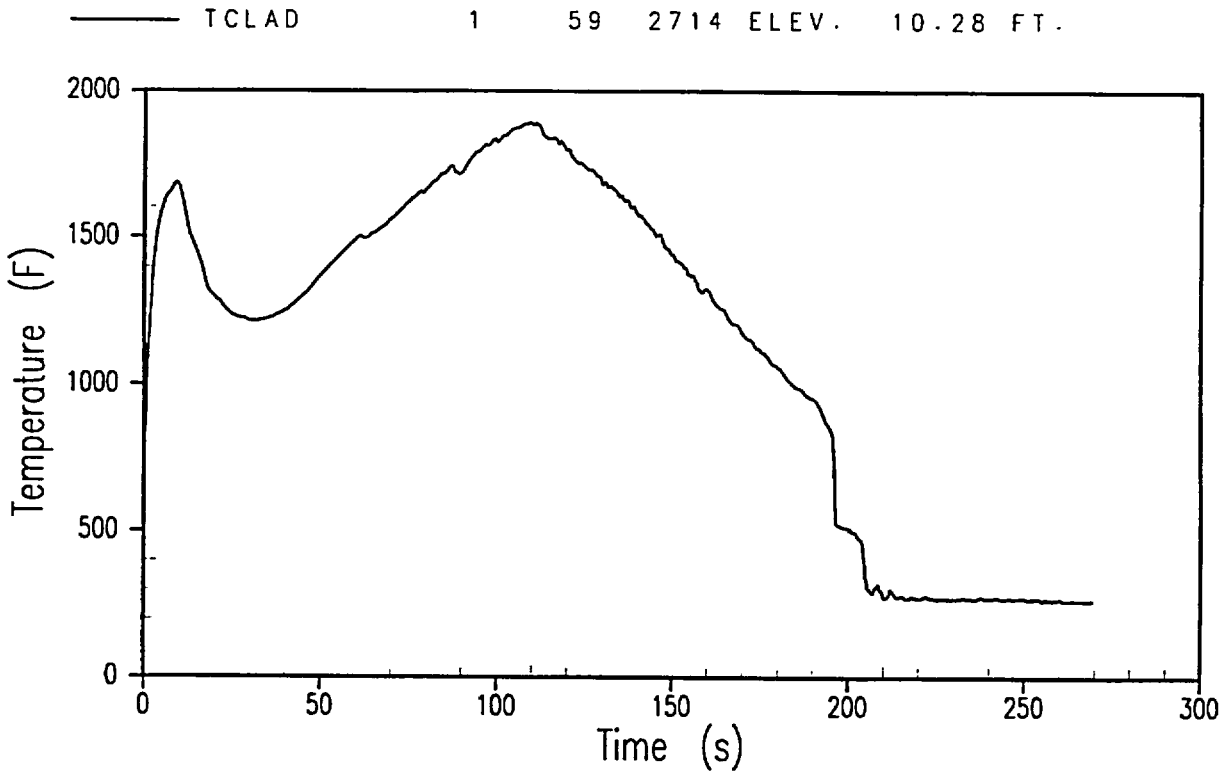



**Figure 15.6.5A-2 Hot Rod Cladding Temperature Transient, PCT Elevation**

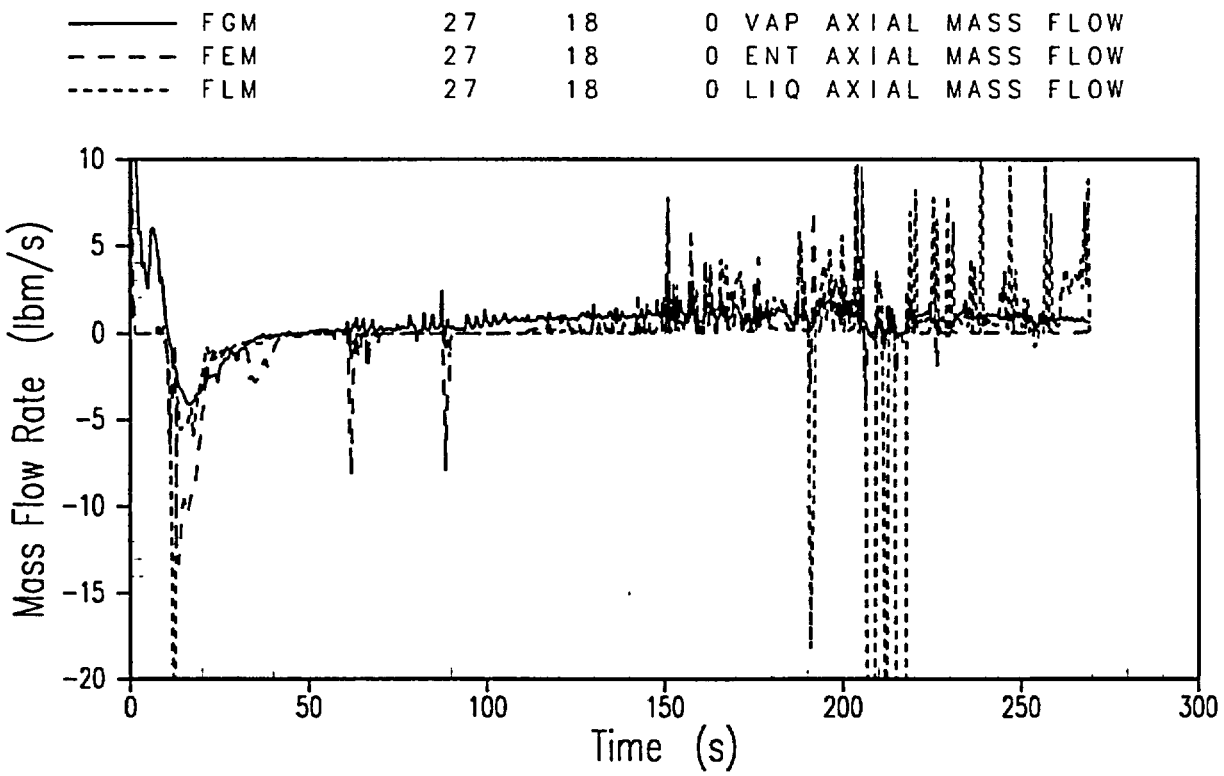



**Figure 15.6.5A-3. Hot Assembly Exit Vapor, Entrained Drop, Liquid Flow Rates**

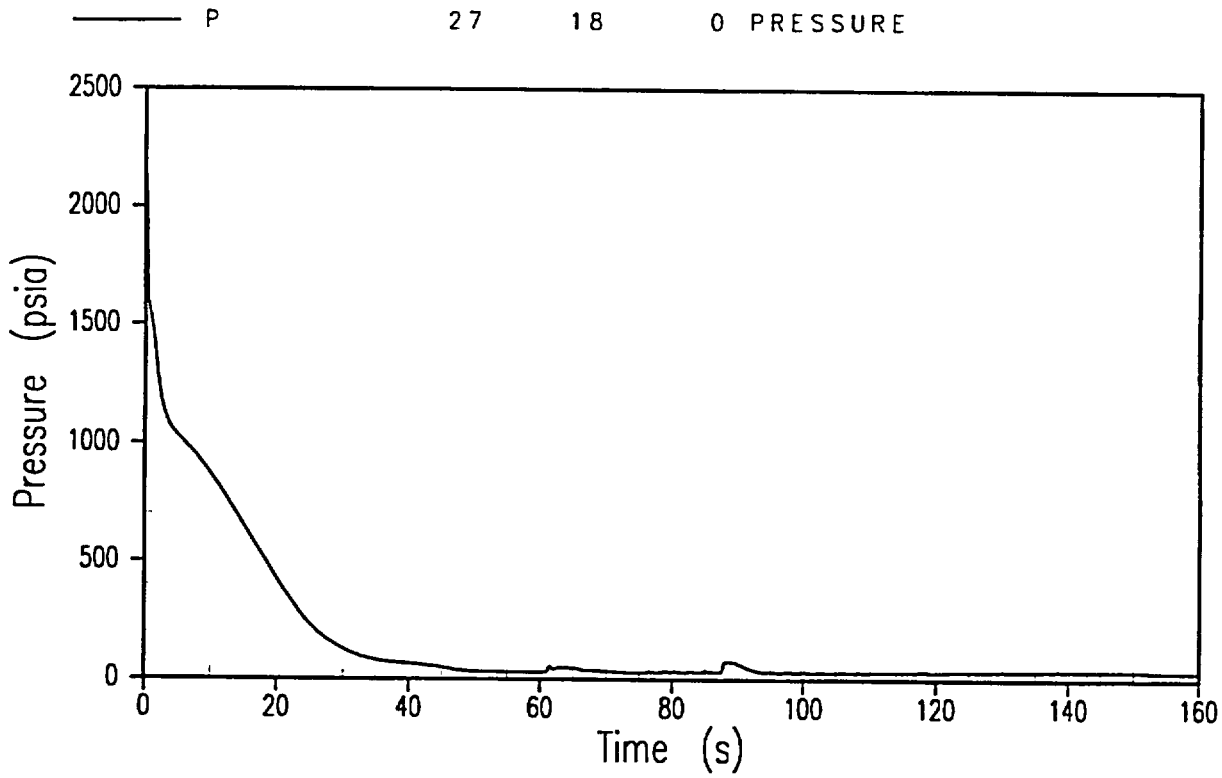



**Figure 15.6.5A-4 Core Pressure**

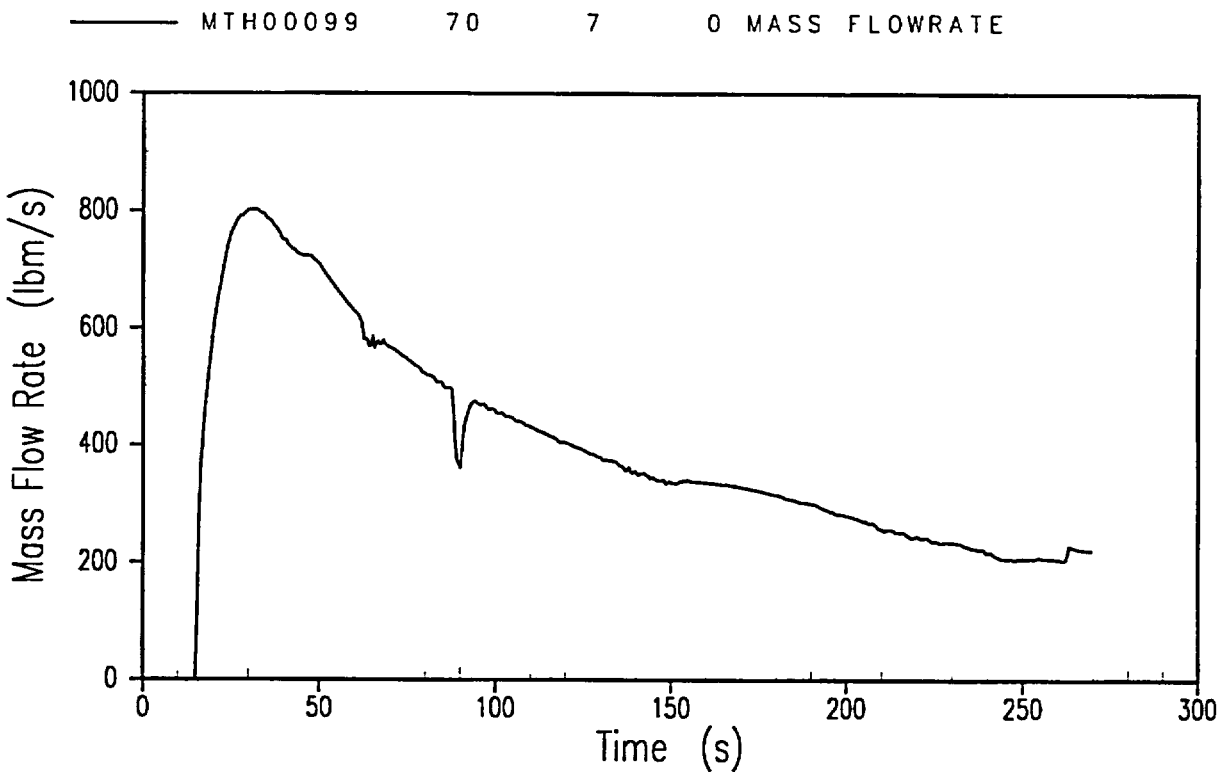



**Figure 15.6.5A-5 Accumulator Flow Rate**

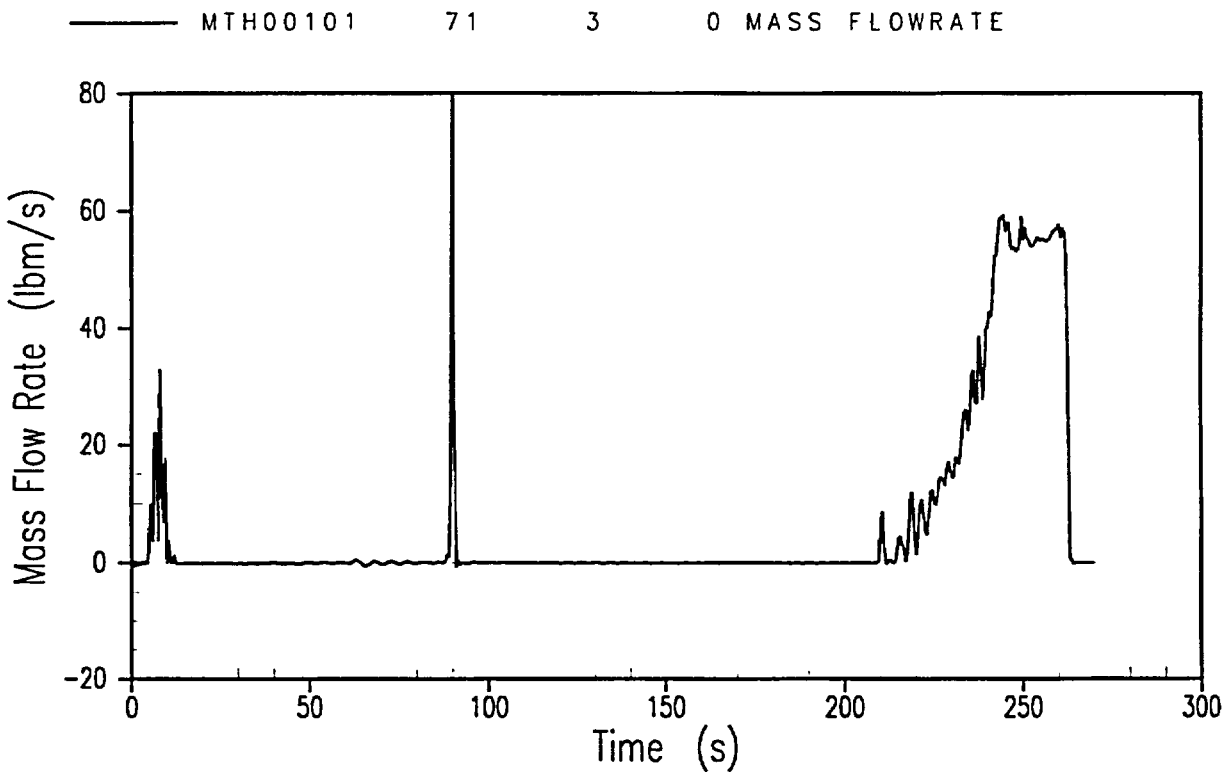



**Figure 15.6.5A-6 Intact Loop Core Makeup Tank Flow Rate**

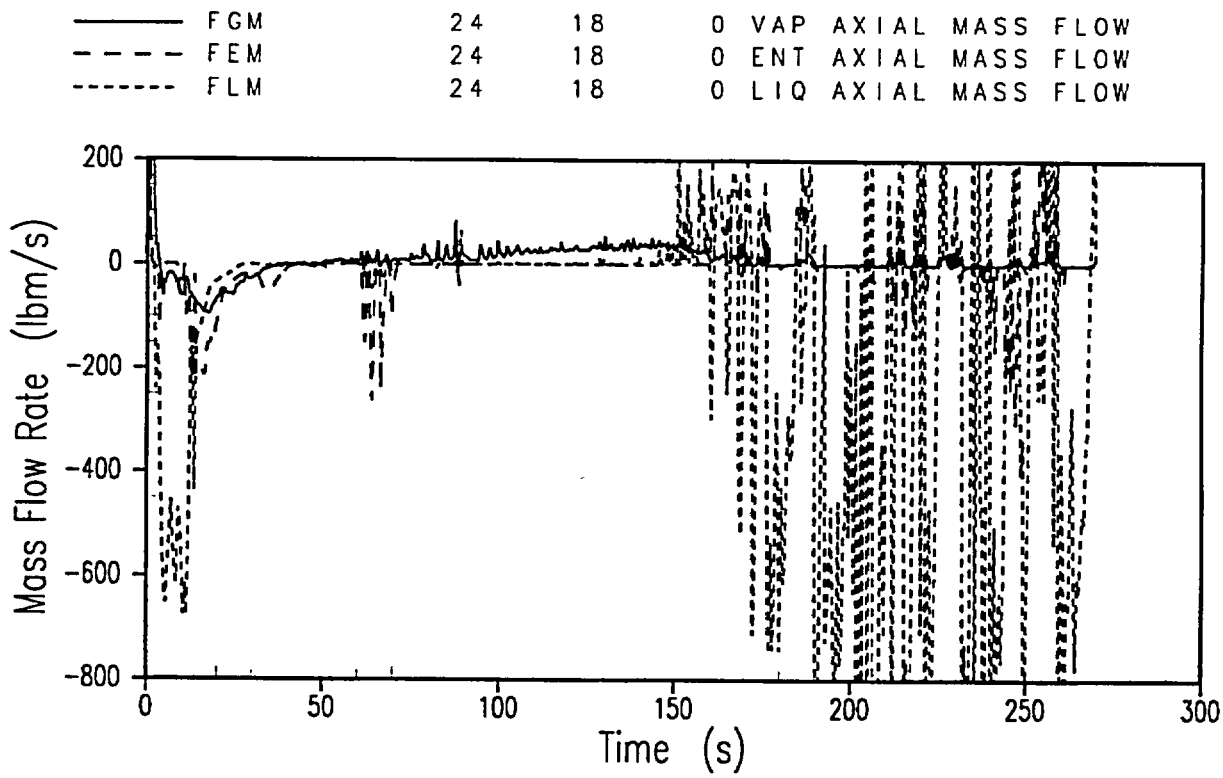



**Figure 15.6.5A-7 Peripheral Assemblies Exit Vapor, Entrained Drop, Liquid Flow Rates**

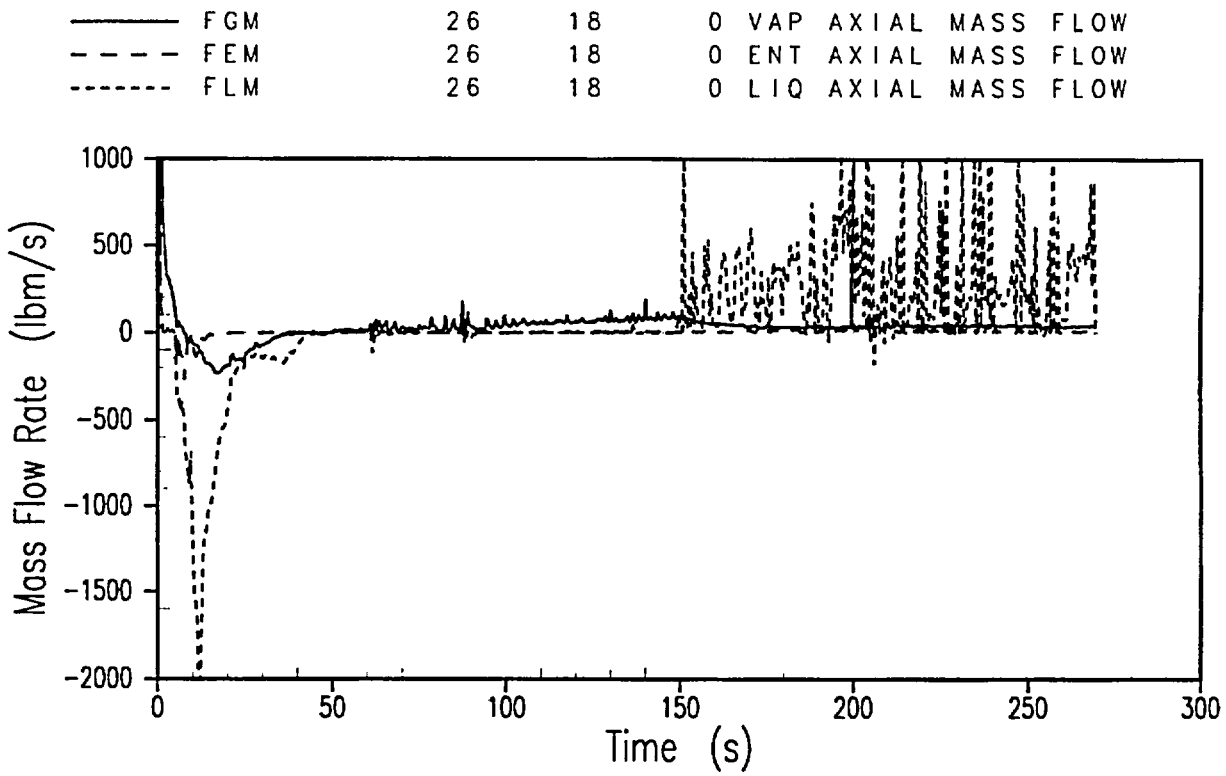



**Figure 15.6.5A-8 Guide Tube Assemblies Exit Vapor, Entrained Drop, Liquid Flow Rates**

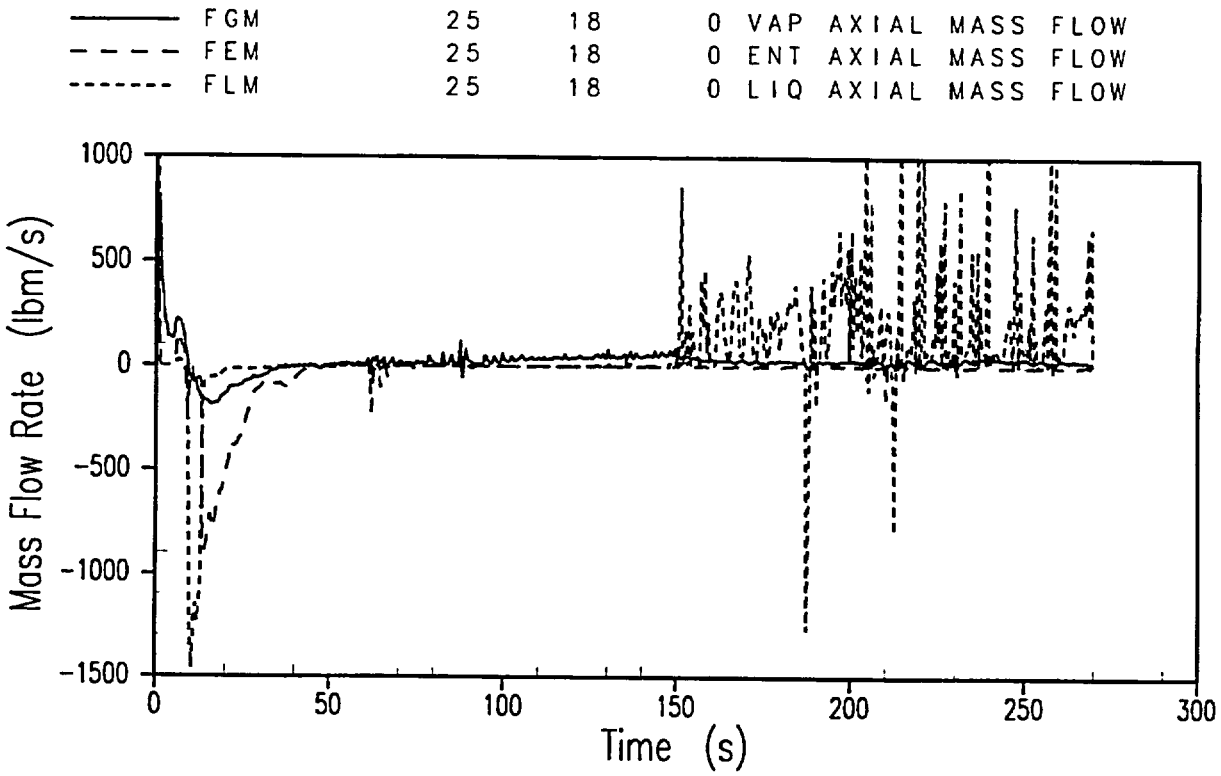



**Figure 15.6.5A-9 Open Hole Assemblies Exit Vapor, Entrained Drop, Liquid Flow Rates**

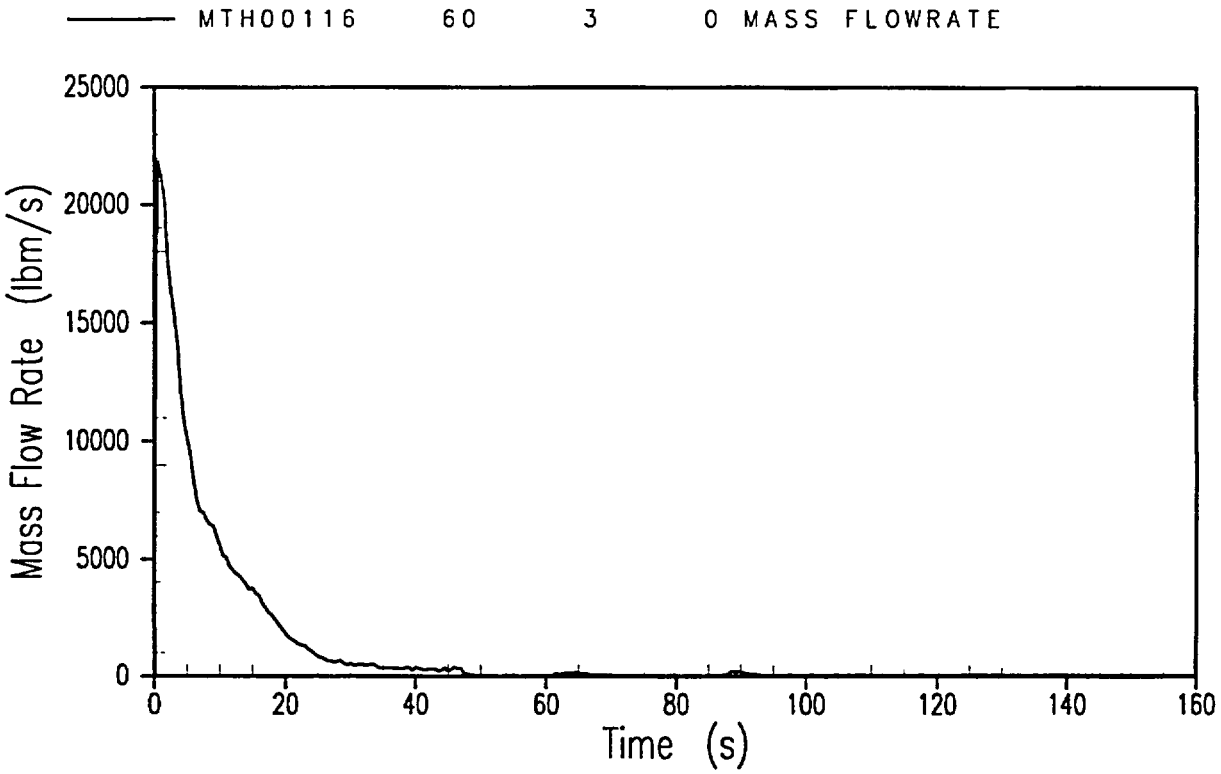



**Figure 15.6.5A-10 Steam Generator Side DECLG Break Flow Rate**

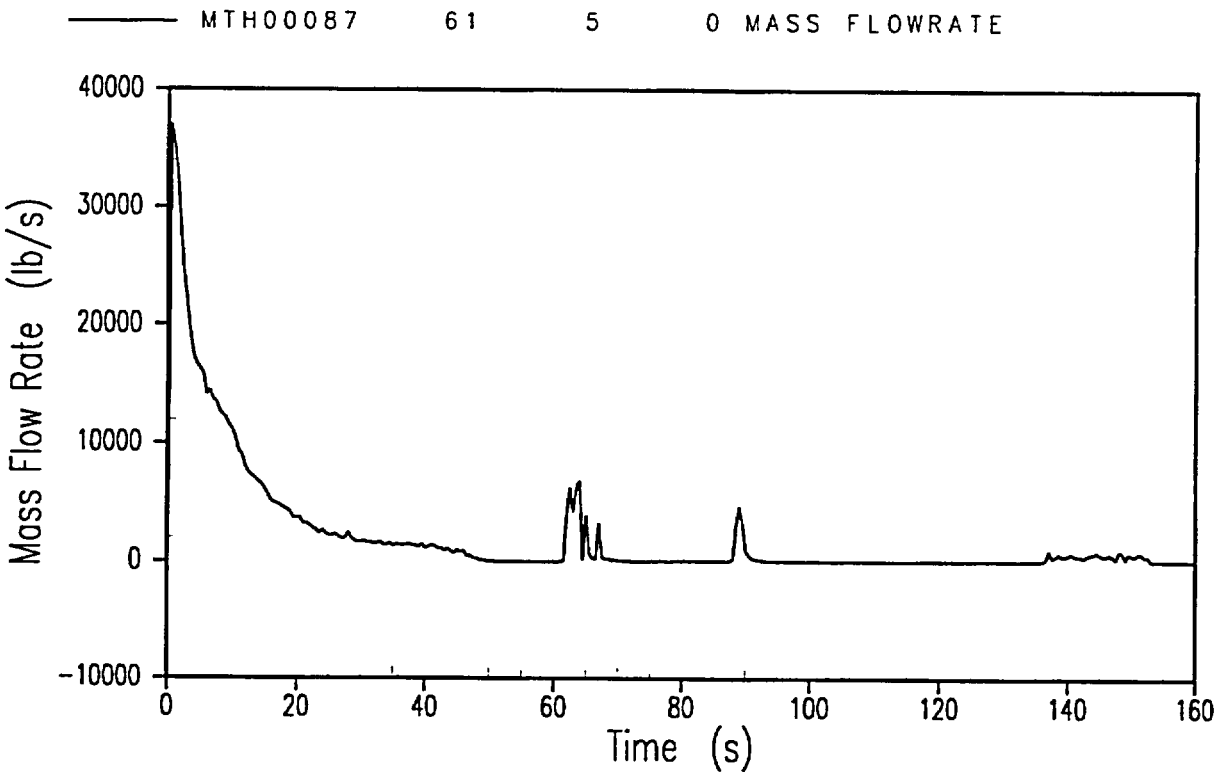



**Figure 15.6.5A-11 Vessel Side DECLG Break Flow Rate**

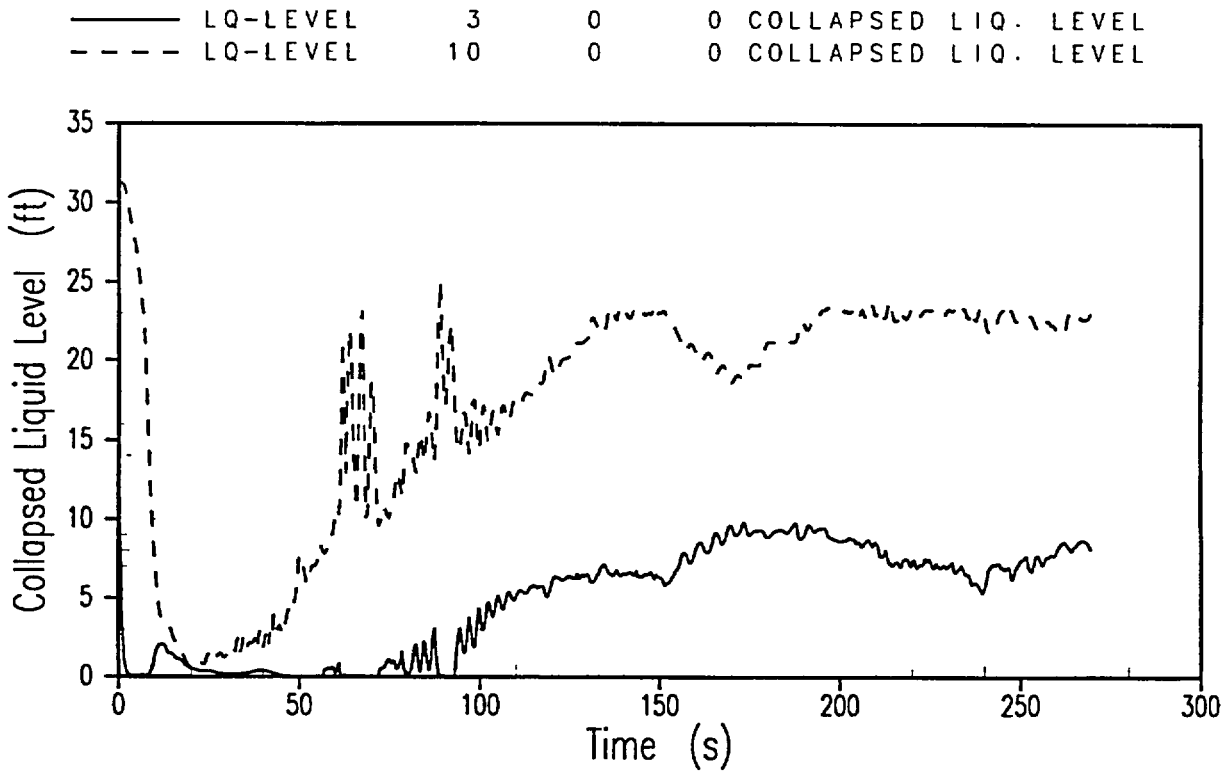



**Figure 15.6.5A-12 Core and Downcomer Collapsed Liquid Levels**

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 440.098

***Question:***

SBLOCA analyses are performed for inadvertent ADS actuation, a 2-inch break in a cold leg, the double-ended rupture of a DVI line and, a 10-inch cold leg break. Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.46 requires analysis of a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. Analyses described in Section 7 of WCAP-14869, "MAAP4/NOTRUMP Benchmarking to Support Use of MAAP4 for AP600 PRA Success Criteria Analyses," demonstrate that for AP600, small breaks in the hot leg lead to lower levels of reactor vessel inventory than do cold leg break locations. This is because for AP600 the cold legs are at a higher elevation than the hot legs. AP1000 has the same coolant loop arrangement as AP600. Please provide SBLOCA analyses for a complete spectrum of hot and cold leg break sizes to ensure that the most severe SBLOCAs are calculated. Please include postulated breaks in the CMT pressure balance lines.

**Westinghouse Response:**

The following small-breaks were analyzed to complete the spectrum of hot and cold leg break sizes for AP1000:

1. 2-inch cold leg break in the PRHR Loop
2. 2-inch hot leg break in the loop without the pressurizer

In addition the following break was analyzed in the CMT pressure balance line:

3. Double-ended cold leg balance line break

**2-inch Cold Leg Break in Fluid Node 19 (Loop With the PRHR)**

This case models a 2-inch break occurring in the reactor cold leg at the bottom of the cold leg in the PRHR Loop. The reactor steady-state initial conditions assumed for this transient can be found in Table 440.098-1. The event times for this transient are given in Table 440.098-2.

The break opens at time zero, and the pressurizer pressure begins to fall as shown in Figure 440.098-1 as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 440.098-2. The break fluid flow is shown in Figure 440.098-3 and Figure 440.098-4. With the break located at the bottom of the cold leg, a mixture flow exits the break

for the majority of the transient because the mixture level stays high in the reactor vessel (Figure 440.098-5). The pressurizer pressure falls below the reactor trip set point, causing the reactor to trip (after the appropriate time delay) and causing isolation of the steam generator steam lines. The core makeup tank isolation valves on both delivery lines and the PRHR delivery line isolation valve open after an "S" signal occurs (with appropriate delays); the reactor coolant pumps trip after an "S" with a 6.0-second delay. The reactor coolant system is cooled by natural circulation with the steam generators removing the energy through their safety valves (as well as by the break) and via the PRHR. Once the core makeup tank isolation valves open, the core makeup tanks begin to inject borated water into the reactor coolant system as shown in Figure 440.098-6 and Figure 440.098-7. The core makeup tanks operate in a recirculation mode initially until they begin to drain (Figure 440.098-8 and Figure 440.098-9).

As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 440.098-5, and the core remains completely covered as seen in Figure 440.098-10. The core makeup tank reaches the 67.5-percent level, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 440.098-2) because an opening has been created at the top of the pressurizer. After these valves open, a more rapid depressurization occurs as seen in Figure 440.098-1; the accumulator setpoint is reached and the accumulators begin to inject. The injection flow from the core makeup tanks is shown in Figure 440.098-6 and Figure 440.098-7, and from the accumulators, in Figure 440.098-11 and Figure 440.098-12. Shortly after, ADS Stage 2 and Stage 3 valves open increasing the system venting capability. Figure 440.098-13 and Figure 440.098-14 provide the liquid and vapor flow rates through ADS Stages 1-3.

As Figure 440.098-6 and Figure 440.098-7 indicate, when the accumulators begin to inject, the flow from both core makeup tanks is reduced, and the flow is temporarily stopped due to the pressurization of the core makeup tanks injection lines by the accumulators.

The ADS Stage 4 valves open when the core makeup tank water level is reduced to 20 percent. Figure 440.098-15 and Figure 440.098-16 indicate the instantaneous liquid and integrated total mass discharged from the ADS Stage 4 valves. After the ADS Stage 4 path opens, the pressurizer begins to drain mixture into the hot legs as seen in Figure 440.098-2. The Figure 440.098-17 mass inventory plot considers the primary inventory to be the reactor coolant system proper, including the pressurizer; the mass present in the passive safety system components is not included at time zero. Once the downcomer pressure drops below the IRWST injection pressure, flow enters the reactor vessel from the IRWST as shown in Figure 440.098-18 and Figure 440.098-19. The mixture level in the reactor vessel is approximately at the hot leg elevation as shown in Figure 440.098-10 throughout this transient; the core never

uncovers, and the peak cladding temperature occurs for this transient at the inception of the event. The 2-inch cold leg break case exhibits large margin-to-core uncovery.

**2-inch Hot Leg Break in Fluid Node 20 (Loop Without the Pressurizer)**

This case models a 2-inch break occurring in the reactor hot leg at the bottom of the hot leg in the loop without the pressurizer. The reactor steady-state initial conditions assumed for this transient can be found in Table 440.098-1. The event times for this transient are given in Table 440.098-3.

The break opens at time zero, and the pressurizer pressure begins to fall as shown Figure 440.098-22 as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 440.098-23. The break fluid flow is shown in Figure 440.098-24 and Figure 440.098-25. With the break located at the bottom of the hot leg, a liquid flow exits the break for the majority of the transient because the mixture level stays high in the reactor (Figure 440.098-26). The pressurizer pressure falls below the reactor trip set point, causing the reactor to trip (after the appropriate time delay) and causing isolation of the steam generator steam lines. The core makeup tank isolation valves on both delivery lines and the PRHR delivery line isolation valve open after an "S" signal occurs (with appropriate delays); the reactor coolant pumps trip after an "S" with a 6.0-second delay. The reactor coolant system is cooled by natural circulation with the steam generators removing the energy through their safety valves (as well as by the break) and via the PRHR. Once the core makeup tank isolation valves open, the core makeup tanks begin to inject borated water into the reactor coolant system as shown in Figure 440.098-27 and Figure 440.098-28. The core makeup tanks operate in a recirculation mode initially until they begin to drain (Figure 440.098-29 and Figure 440.098-30).

As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 440.098-31, and the core remains completely covered as seen in Figure 440.098-26. The core makeup tank reaches the 67.5-percent level, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 440.098-23) because an opening has been created at the top of the pressurizer. After these valves open, a more rapid depressurization occurs as seen in Figure 440.098-22; the accumulator setpoint is reached and the accumulators begin to inject. The injection flow from the core makeup tanks is shown in Figure 440.098-27 and Figure 440.098-28, and from the accumulators, in Figure 440.098-32 and Figure 440.098-33. Shortly after, ADS Stage 2 and Stage 3 valves open increasing the system venting capability. Figure 440.098-34 and Figure 440.098-35 provide the liquid and vapor flow rates through ADS Stages 1-3.

As Figure 440.098-27 and Figure 440.098-28 indicate, when the accumulators begin to inject, the flow from both core makeup tanks is reduced, and the flow is temporarily stopped due to the pressurization of the core makeup tanks injection lines by the accumulators.

The ADS Stage 4 valves open when the core makeup tank water level is reduced to 20 percent. Figure 440.098-36 and Figure 440.098-37 indicate the instantaneous liquid and integrated total mass discharged from the ADS Stage 4 valves. After the ADS Stage 4 path opens, the pressurizer begins to drain mixture into the hot legs as seen in Figure 440.098-23. The Figure 440.098-38 mass inventory plot considers the primary inventory to be the reactor coolant system proper, including the pressurizer; the mass present in the passive safety system components is not included at time zero. Once the downcomer pressure drops below the IRWST injection pressure, flow enters the reactor vessel from the IRWST as shown in Figure 440.098-39 and Figure 440.098-40. The mixture level in the reactor vessel is approximately at the hot leg elevation as shown in Figure 440.098-26 throughout this transient; the core never uncovers, and the peak cladding temperature occurs for this transient at the inception of the event. The 2-inch hot leg break case exhibits large margin-to-core uncovery.

**Double-Ended Cold Leg Balance Line Break (Case 3)**

A double-ended break is analyzed in the pressure balance line connecting the cold leg to the core makeup tank. The reactor steady-state initial conditions assumed for this transient can be found in Table 440.098-1. The event times for this transient are given in Table 440.098-4.

This case models the double-ended rupture of one of the two cold leg pressure balance lines, at the top of balance line, where it is connected to the core makeup tank. The broken loop core makeup tank although modeled, is unable to deliver much flow because it has no means of pressure equalization and it's only driving force is static head. A break in the cold leg pressure balance line evaluates the ability of the plant to recover from a moderately large break with only one of two core makeup tanks operational.

The break opens at time zero, and the pressurizer pressure begins to fall rapidly (Figure 440.098-43), as subcooled liquid is discharged from the broken pressure balance line. The pressurizer mixture level initially decreases as given in Figure 440.098-44. The break fluid flow is shown in Figure 440.098-45 and Figure 440.098-46. The pressurizer pressure falls below the reactor trip set point, causing the reactor to trip (after the appropriate time delay) and causing isolation of the steam generator steam lines. The core makeup tank isolation valves on both delivery lines and the PRHR delivery line isolation valve open after an "S" signal occurs (with appropriate delays); the reactor coolant pumps trip after an "S" with a 6.0-second delay. The reactor coolant system is cooled by natural circulation with the steam generators removing the

energy through their safety valves (as well as by the break) and via the PRHR. Once the core makeup tank isolation valves open, the intact loop core makeup tank begins to inject borated water into the reactor coolant system as shown in Figure 440.098-47 and begins to form a mixture level (Figure 440.098-48).

As time proceeds, the mixture level drops in the upper plenum to the hot leg elevation (Figure 440.098-49). The upper parts of the reactor coolant system begin to drain and a mixture level forms in the downcomer (Figure 440.098-50). The core makeup tank reaches the 67.5-percent level, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 440.098-44) because an opening has been created at the top of the pressurizer. After these valves open, a more rapid depressurization occurs as seen in Figure 440.098-43; the accumulator setpoint is reached and the accumulators begin to inject. The injection flow from the accumulators is shown in Figure 440.098-51 and Figure 440.098-52. Accumulator injection into the downcomer causes the downcomer mixture level to rise. Flow from the intact loop core makeup tank replenishes the reactor coolant mass inventory both before and after ADS initiation. Shortly after, ADS Stage 2 and Stage 3 valves open increasing the system venting capability. Figure 440.098-53 and Figure 440.098-54 provide the liquid and vapor flow rates through ADS Stages 1-3. After the accumulator empties, stable but decreasing injection continues from the intact loop core makeup tank (Figure 440.098-47) as the reactor coolant system pressure declines slowly (Figure 440.098-43).

The ADS Stage 4 valves open when the core makeup tank water level is reduced to 20 percent. Figure 440.098-55 and Figure 440.098-56 indicate the instantaneous liquid and integrated total mass discharged from the ADS Stage 4 valves.

After the ADS Stage 4 path opens, the pressurizer begins to drain mixture into the hot legs as seen in Figure 440.098-44. Once the downcomer pressure drops below the IRWST injection pressure and after the intact loop core makeup tank empties, flow enters the reactor vessel from the IRWST as shown in Figure 440.098-57 and Figure 440.098-58. This flow results in a slow rise in the reactor coolant system inventory as shown in Figure 440.098-59. For the double-ended break in the pressure balance line, the minimum reactor coolant system inventory occurs shortly after IRWST injection begins. The double-ended cold leg balance line break exhibits large margin-to-core uncovery.

**Conclusions**

The results of the small breaks analyzed here to complete the small break LOCA spectrum as requested by the NRC, exhibit margin to core uncovery for the AP1000 design. The minimum

reactor coolant system mass inventory condition among the small-break LOCA cases analyzed here occurs for the 2-inch Hot Leg Break in Fluid Node 20.

The conclusions presented in the DCD section 15.6.5.4B also apply to the cases analyzed here and show that the passive safeguards systems in the AP1000 are sufficient to mitigate LOCAs. Specifically, it is concluded that:

- The primary side can be depressurized by the ADS to allow stable injection into the core.
- Injection from the core makeup tanks, accumulators, and IRWST prevents any cladding heatup for small-break LOCAs analyzed, including double-ended ruptures in the passive safeguards system lines.

The results of the complete spectrum of small-break LOCA analyses are summarized below:

| Break Location/Diameter | AP1000 Minimum RCS Inventory | Peak Cladding Temperature |
|---|---|---|
| Inadvertent ADS | 105,800 | (1) |
| 2-inch cold leg break in the loop without the PRHR | 106,620 | (1) |
| 10-inch cold leg break | 78,160 | <1370°F |
| DEDVI (Nominal Containment pressure) | 103,620 | (1) |
| DEDVI (25 psia Containment pressure) | 114,710 | (1) |
| 2-inch cold leg break in the PRHR Loop | 107,184 | (1) |
| 2-inch hot leg break in the loop without the pressurizer | 101,952 | (1) |
| DE cold leg balance line break | 110,091 | (1) |

(1) - No Core Uncovery

The 10-inch cold leg break remains the limiting minimum inventory condition that occurs during the initial blowdown period. However, the case that represents the limiting inventory condition at the onset of IRWST injection has been shifted to the 2-inch hot leg break. All breaks simulated in the break spectrum produce results that demonstrate significant margin to the 10 CFR 50.46 Appendix-K peak cladding temperature regulatory limits.

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

Table 440.098-1

**INITIAL CONDITIONS FOR AP1000 SMALL-BREAK LOCA ANALYSIS**

| | Calculation | | |
|---|---|---|---|
| **Condition** | 2-inch CL Break & Double-Ended CL Balance Line Break | 2-inch Hot Leg Break | **Nominal Steady-state** |
| Pressurizer pressure (psia) | 2302.8 | 2302.8 | 2300 |
| Vessel inlet temperature (°F) | 534.01 | 534.01 | 534.3 |
| Vessel outlet temperature (°F) | 612.82 | 612.82 | 612.9 |
| Vessel flow rate (lb/sec) | 31099 | 31087 | 31089 |
| Steam generator pressure (psia) | 806.4 | 806.4 | 788.5 |

Table 440.098-2

**2-INCH COLD LEG BREAK IN PRHR LOOP**

**SEQUENCE OF EVENTS**

| Event | AP1000 Time (seconds) |
|---|---|
| Break opens | 0.0 |
| Reactor trip signal | 55.3 |
| Steam turbine stop valves close | 61.3 |
| "S" signal | 62.6 |
| Main feed isolation valves begin to close | 64.6 |
| Reactor coolant pumps start to coast down | 68.6 |
| ADS Stage 1 | 1180.9 |
| ADS Stage 2 | 1250.9 |
| Accumulator injection starts | 1299 |
| ADS Stage 3 | 1370.9 |
| Accumulator empties | 1721.5 |
| ADS Stage 4 | 2296.4 |
| Core makeup tank 1 empty | 2654 |
| Core makeup tank 2 empty | 2670 |
| IRWST injection starts | 3100 |

Table 440.098-3

**2-INCH HOT LEG BREAK IN BALANCE LINE LOOP**

**SEQUENCE OF EVENTS**

| Event | AP1000 Time (seconds) |
|---|---|
| Break opens | 0.0 |
| Reactor trip signal | 92.1 |
| Steam turbine stop valves close | 98.1 |
| "S" signal | 99.9 |
| Main feed isolation valves begin to close | 101.9 |
| Reactor coolant pumps start to coast down | 105.9 |
| ADS Stage 1 | 1822.4 |
| Accumulator injection starts | 1869 |
| ADS Stage 2 | 1892.4 |
| ADS Stage 3 | 2012.4 |
| Accumulator empties | 2383.6 |
| ADS Stage 4 | 2840.3 |
| Core makeup tank 1 empty | 3253 |
| Core makeup tank 2 empty | 3314 |
| IRWST injection starts | 3505 |

Table 440.098-4

**DOUBLE-ENDED COLD LEG BALANCE LINE BREAK**

**SEQUENCE OF EVENTS**

| Event | AP1000 Time (seconds) |
|---|---|
| Break opens | 0.0 |
| Reactor trip signal | 9.75 |
| "S" signal | 13.71 |
| Main feed isolation valves begin to close | 15.71 |
| Steam turbine stop valves close | 15,75 |
| Reactor coolant pumps start to coast down | 19.71 |
| Accumulator injection starts | 259 |
| ADS Stage 1 | 746.1 |
| ADS Stage 2 | 816.1 |
| ADS Stage 3 | 936.1 |
| Accumulator empties | 1071.5 |
| ADS Stage 4 | 1675.2 |
| Core makeup tank 2 empty | 2040 |
| IRWST injection starts* | 2166 |

* - Continuous Injection Period

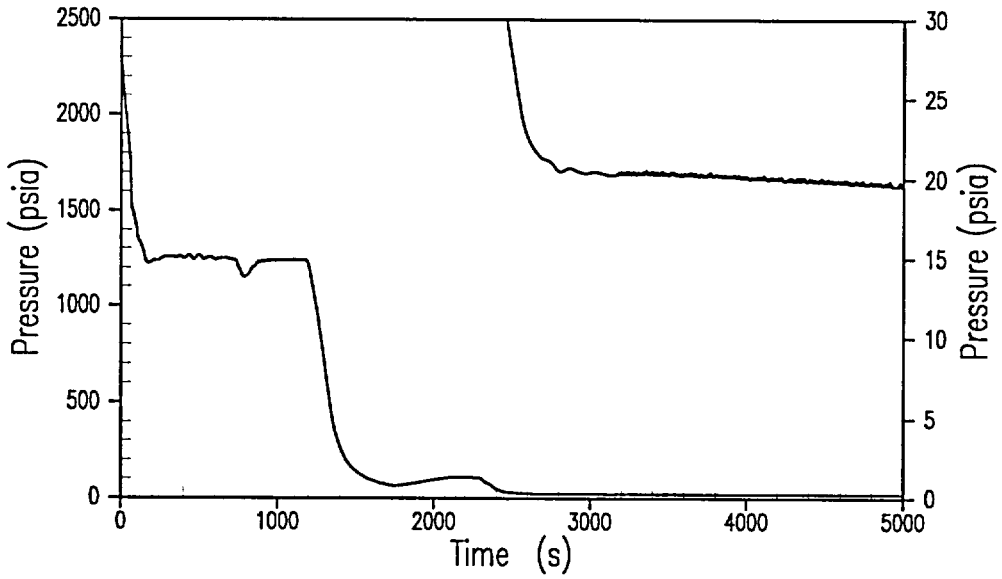



**Figure 440.098-1 2-inch Cold Leg Break in the PRHR Loop - RCS Pressure**

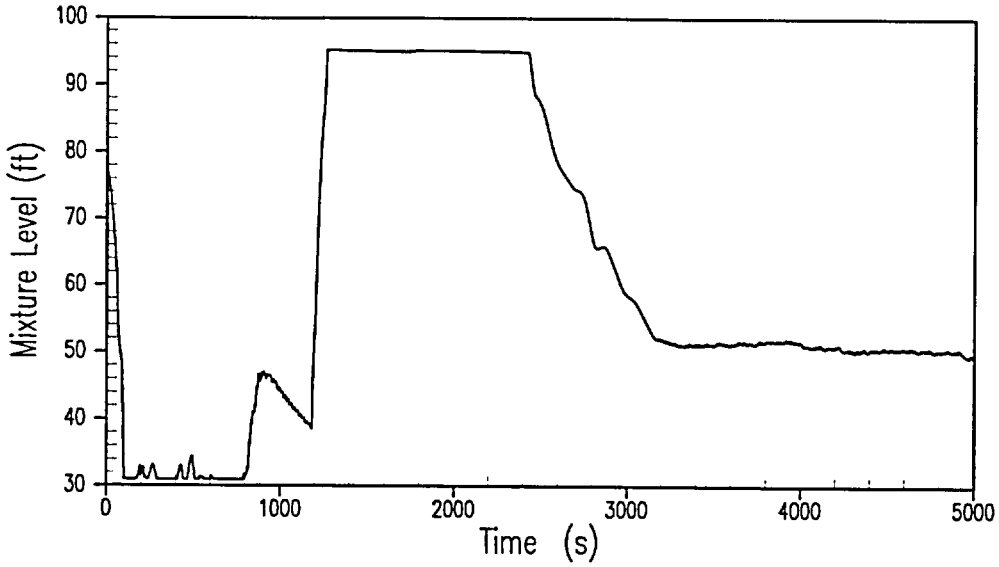



**Figure 440.098-2 2-inch Cold Leg Break in the PRHR Loop - Pressurizer Mixture Level**

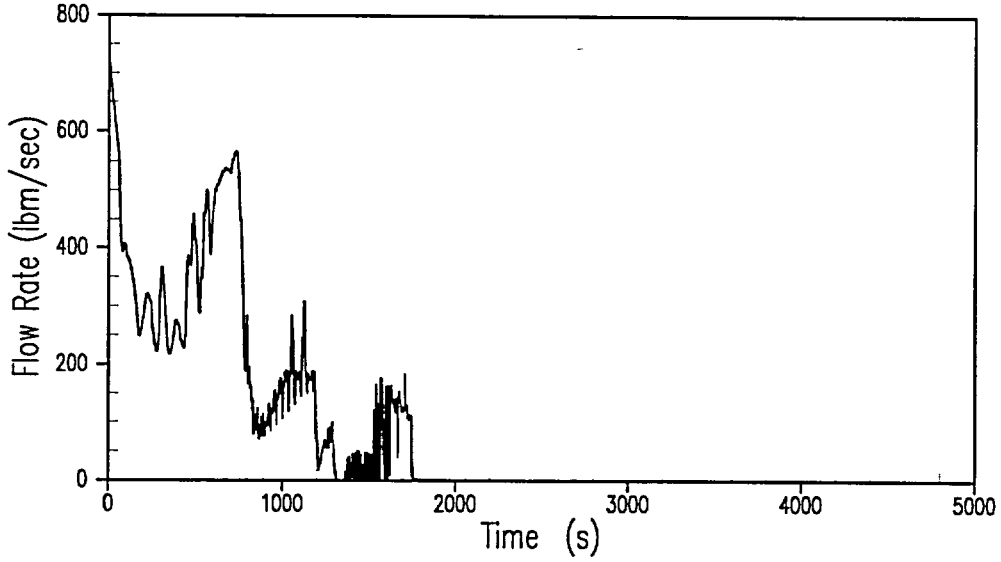



**Figure 440.098-3 2-inch Cold Leg Break in the PRHR Loop - Liquid Break Discharge Rate**

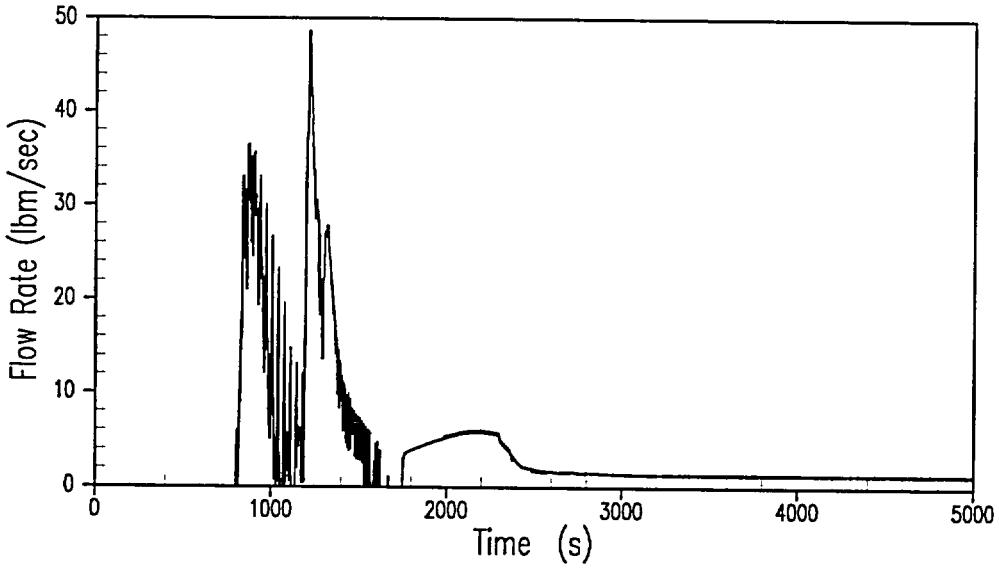



**Figure 440.098-4 2-inch Cold Leg Break in the PRHR Loop - Vapor Break Discharge Rate**

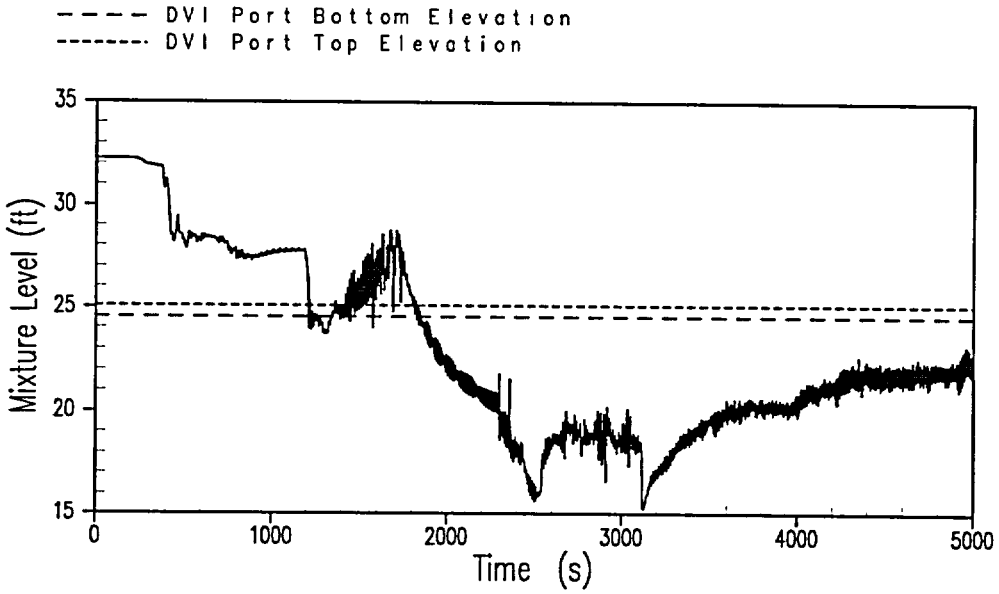



**Figure 440.098-5 2-inch Cold Leg Break in the PRHR Loop - Downcomer Mixture Level**

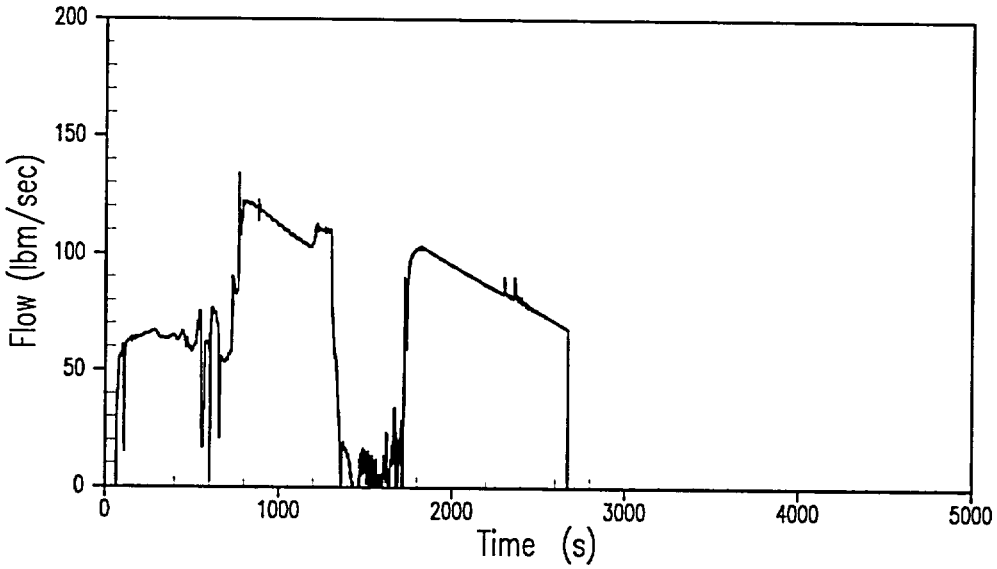



**Figure 440.098-6 2-inch Cold Leg Break in the PRHR Loop - CMT-1 Injection Rate**

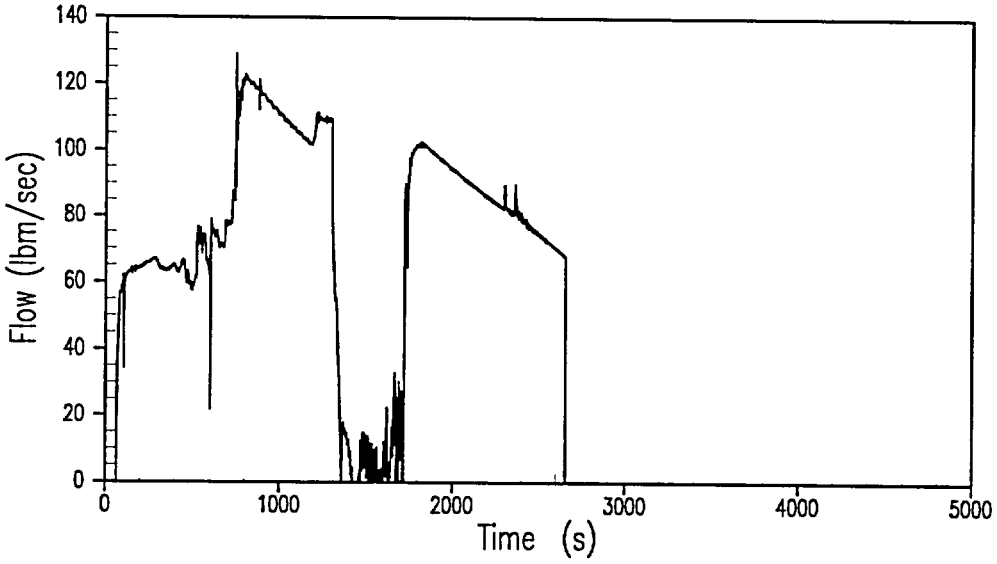



**Figure 440.098-7 2-inch Cold Leg Break in the PRHR Loop - CMT-2 Injection Rate**

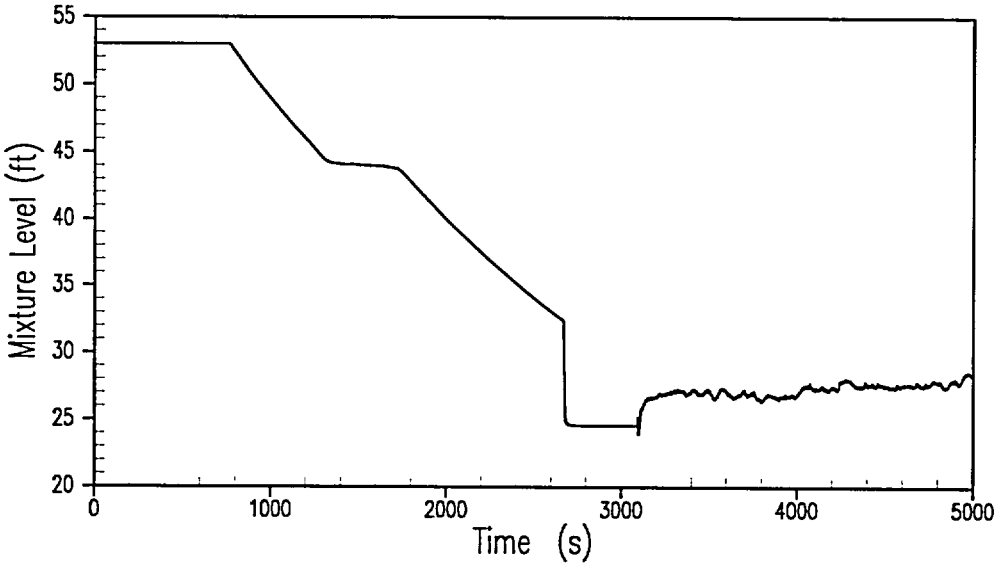



**Figure 440.098-8 2-inch Cold Leg Break in the PRHR Loop - CMT-1 Mixture Level**

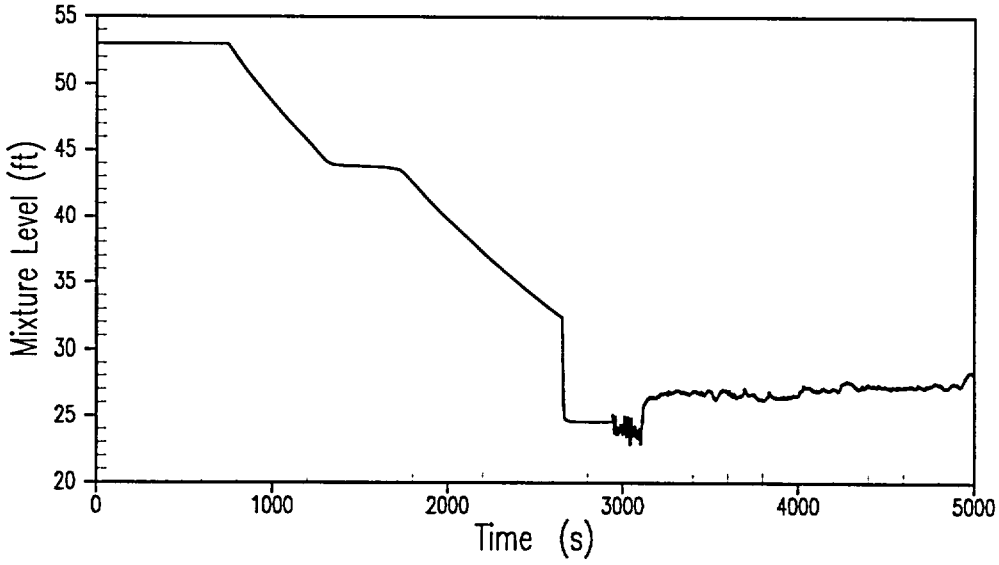



**Figure 440.098-9 2-inch Cold Leg Break in the PRHR Loop - CMT-1 Mixture Level**

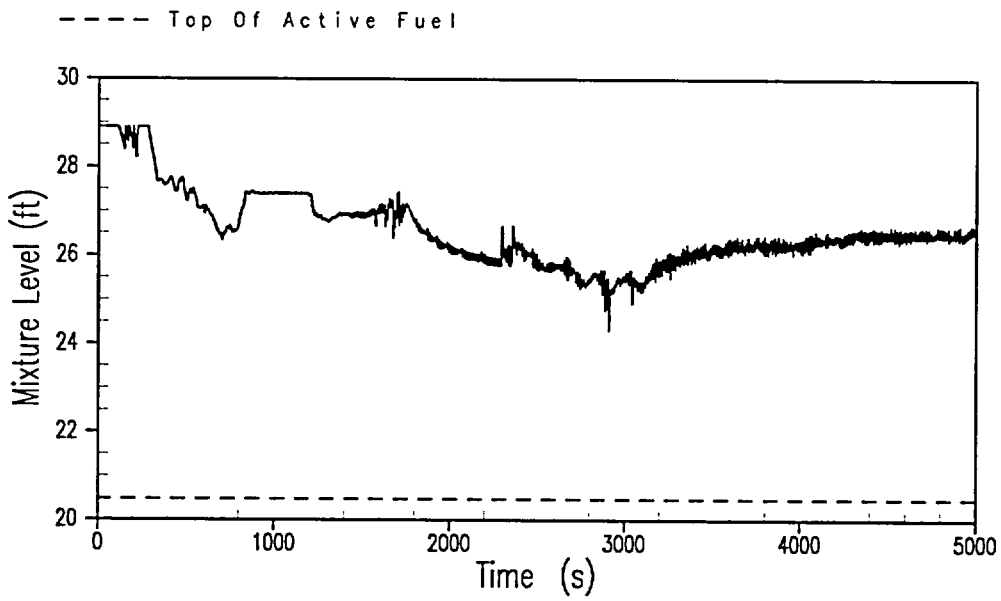



**Figure 440.098-10 2-inch Cold Leg Break in the PRHR Loop - Core/Upper Plenum Mixture Level**

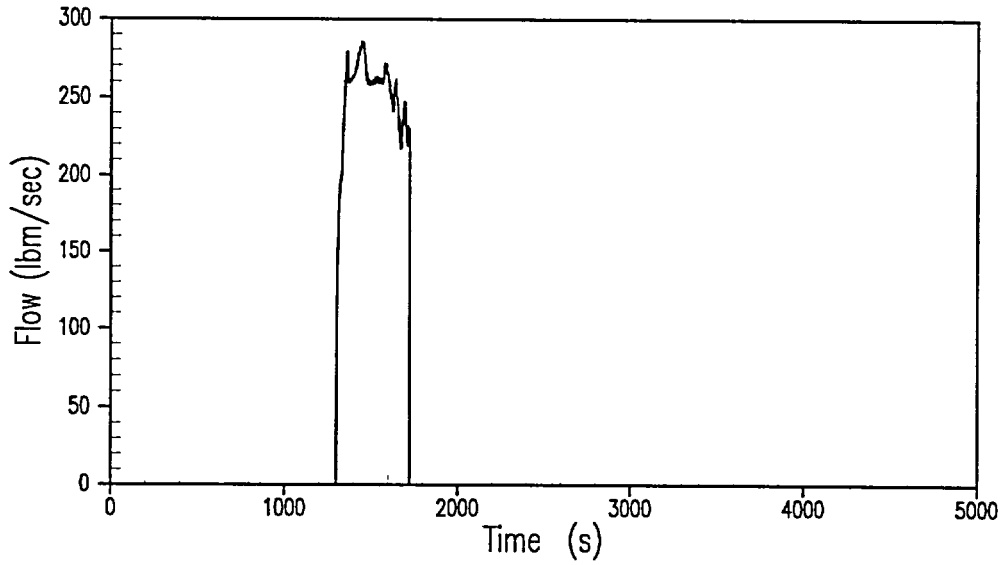



**Figure 440.098-11 2-inch Cold Leg Break in the PRHR Loop - Accumulator-1 Injection Rate**

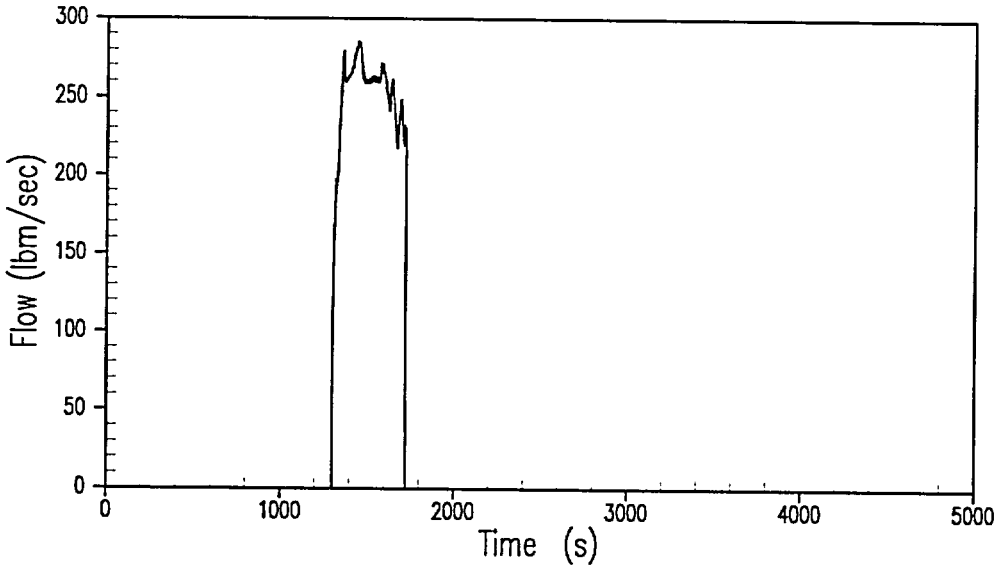



**Figure 440.098-12 2-inch Cold Leg Break in the PRHR Loop - Accumulator-2 Injection Rate**

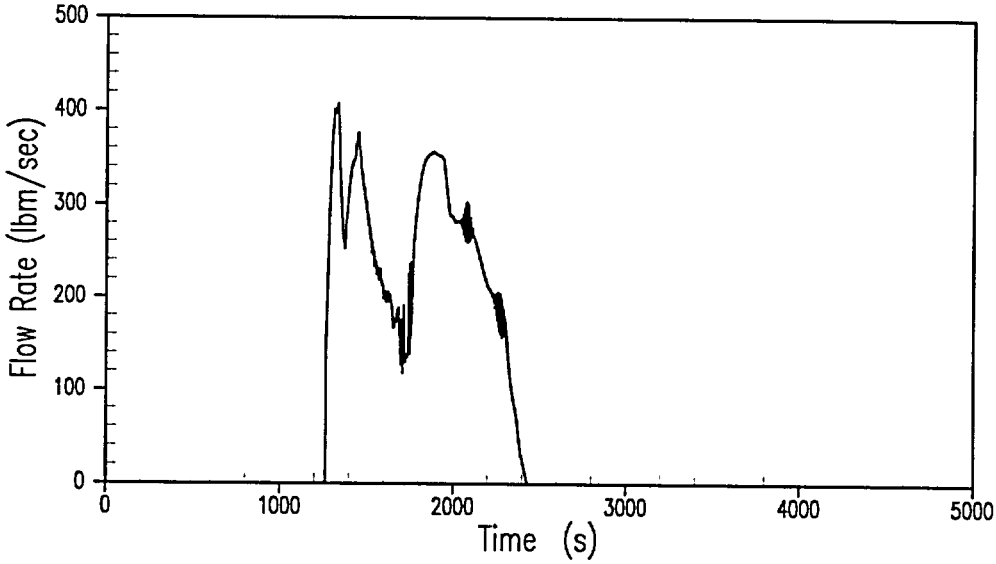



**Figure 440.098-13 2-inch Cold Leg Break in the PRHR Loop - ADS 1-3 Liquid Discharge Rate**

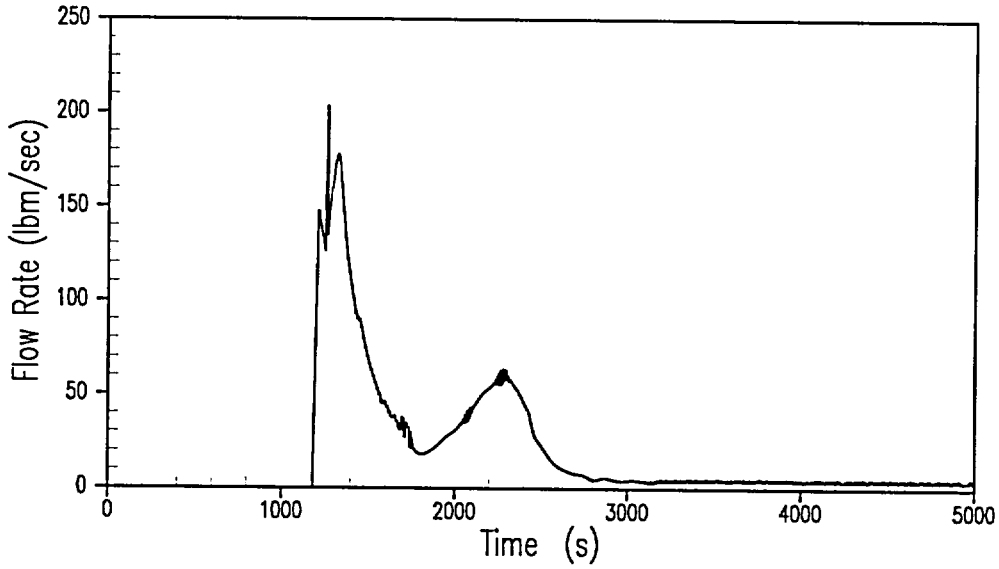



**Figure 440.098-14 2-inch Cold Leg Break in the PRHR Loop - ADS 1-3 Vapor Discharge Rate**

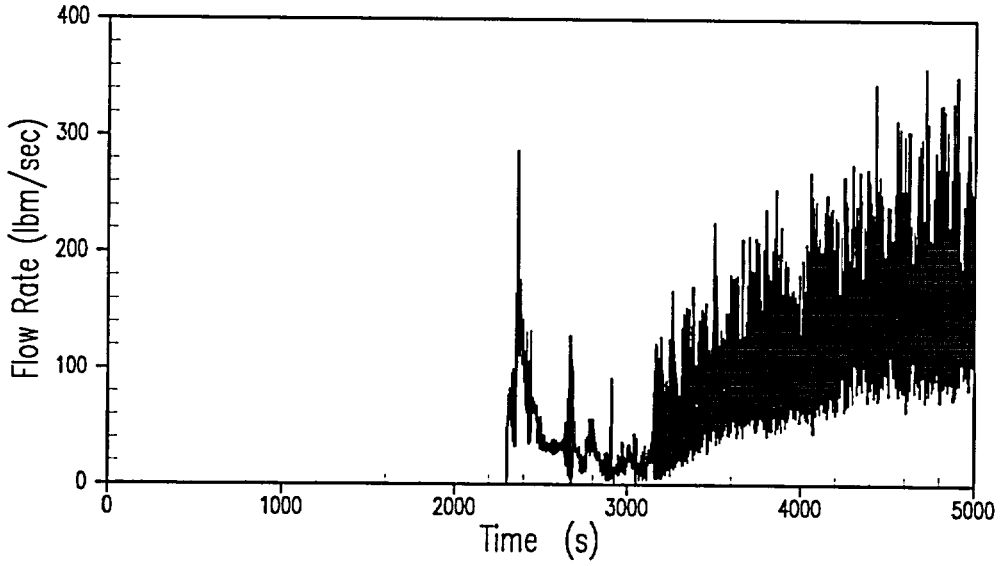



**Figure 440.098-15 2-inch Cold Leg Break in the PRHR Loop - ADS-4 Liquid Discharge Rate**

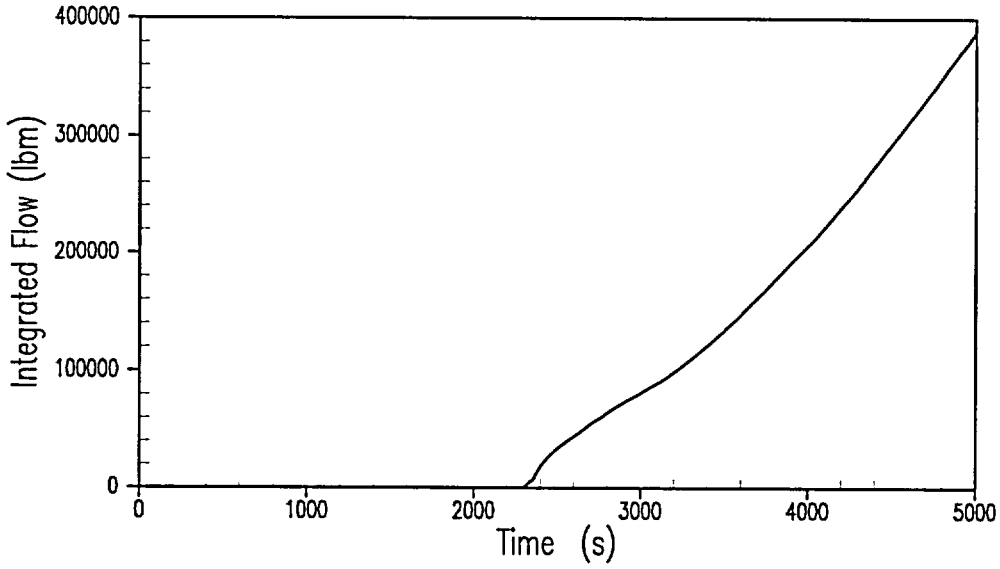



**Figure 440.098-16 2-inch Cold Leg Break in the PRHR Loop - ADS-4 Integrated Discharge Flow**

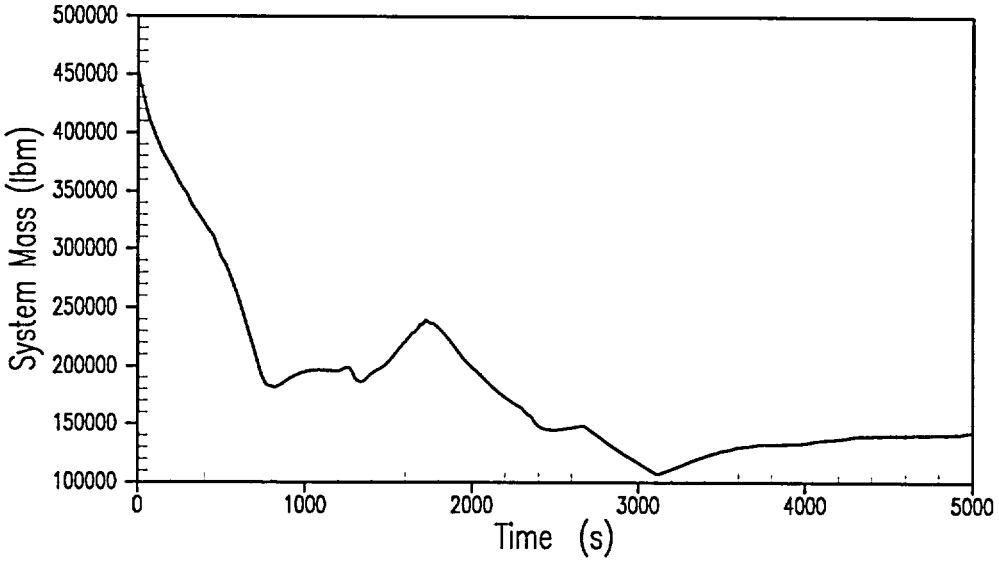



**Figure 440.098-17 2-inch Cold Leg Break in the PRHR Loop - System Mass Inventory**

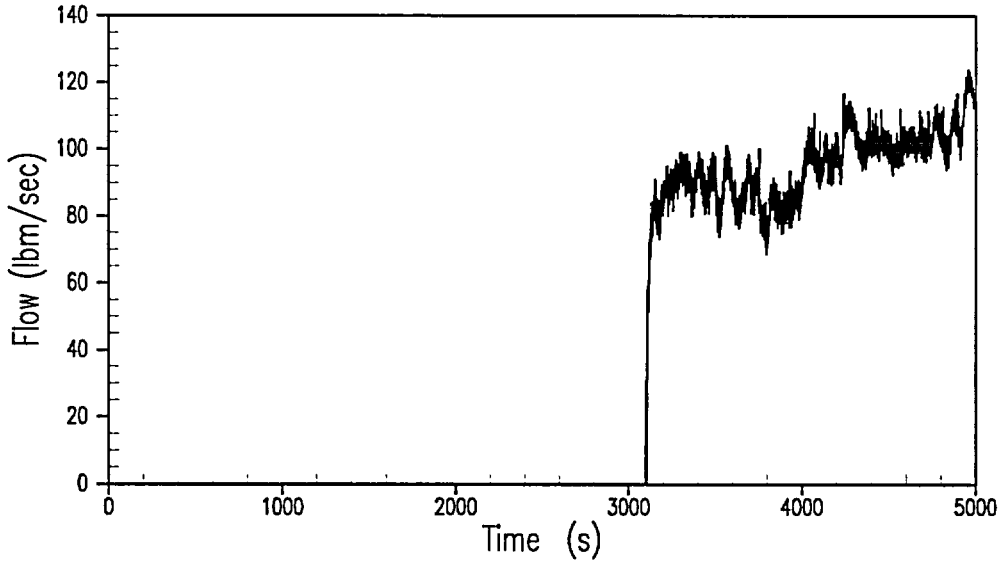



**Figure 440.098-18 2-inch Cold Leg Break in the PRHR Loop - IRWST-1 Injection Rate**

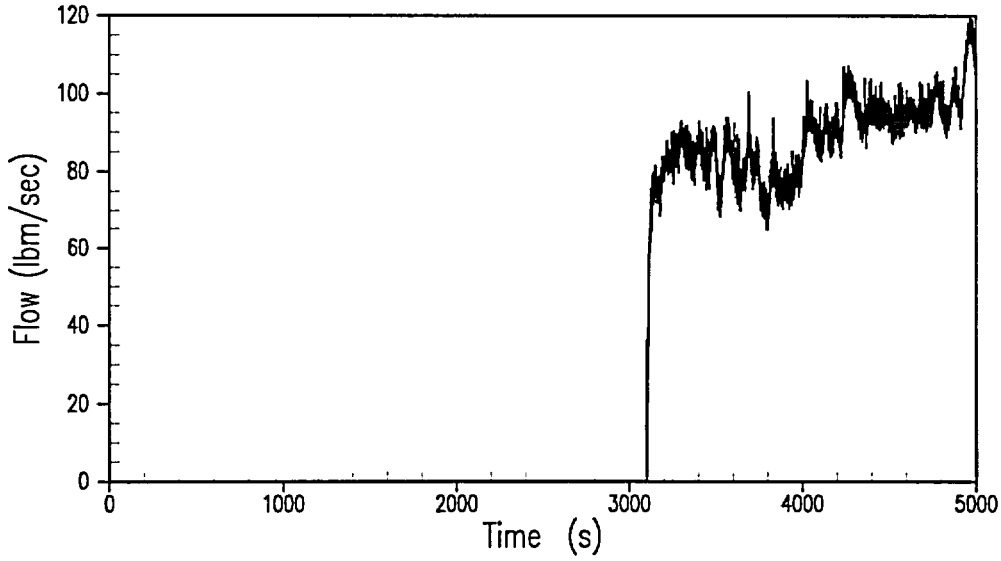



**Figure 440.098-19 2-inch Cold Leg Break in the PRHR Loop- IRWST-2 Injection Rate**

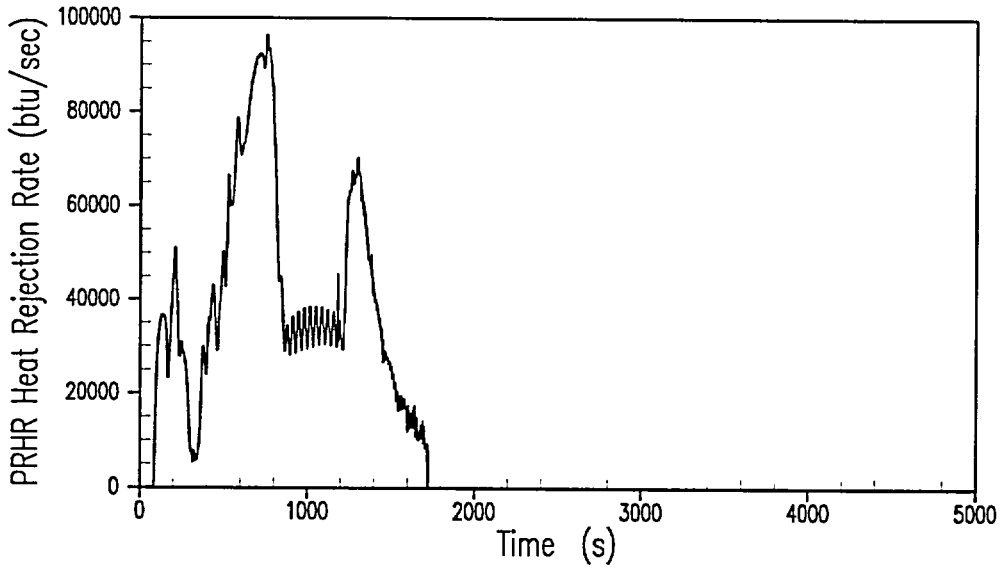



**Figure 440.098-20 2-inch Cold Leg Break in the PRHR Loop - PRHR Heat Removal Rate**

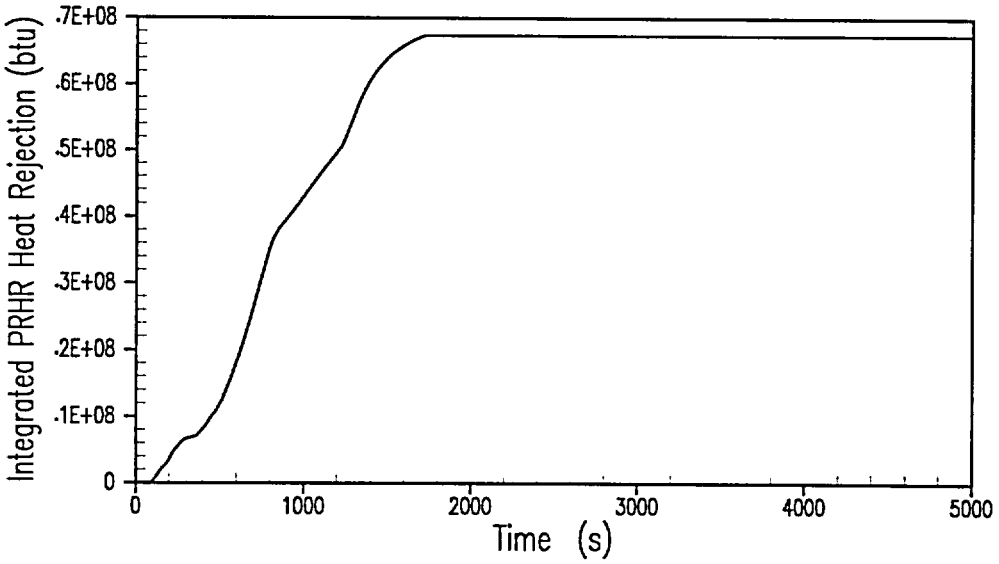



**Figure 440.098-21 2-inch Cold Leg Break in the PRHR Loop - PRHR Integrated Heat Removal**

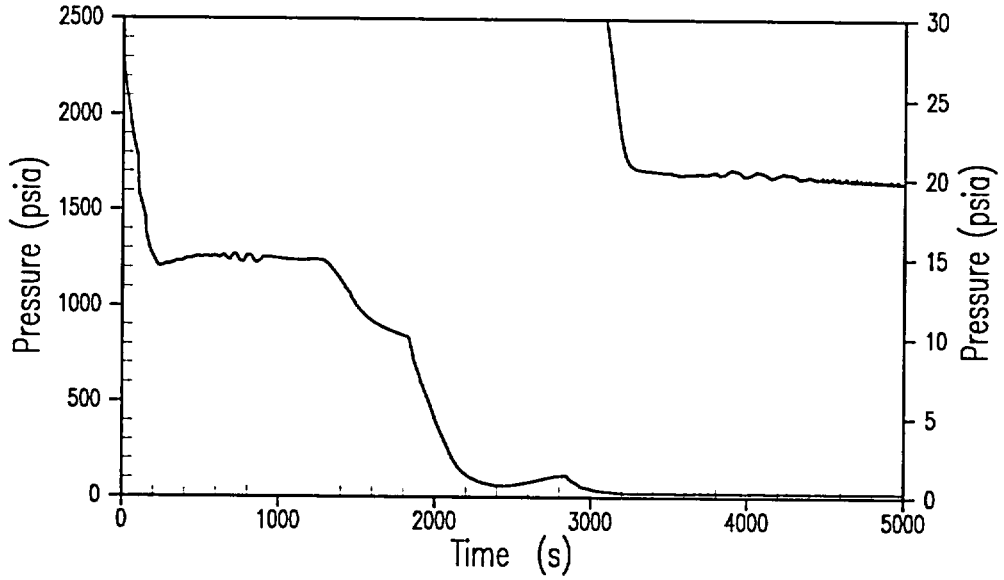



**Figure 440.098-22 2-inch Hot Leg Break in the Loop Without the Pressurizer – RCS Pressure**

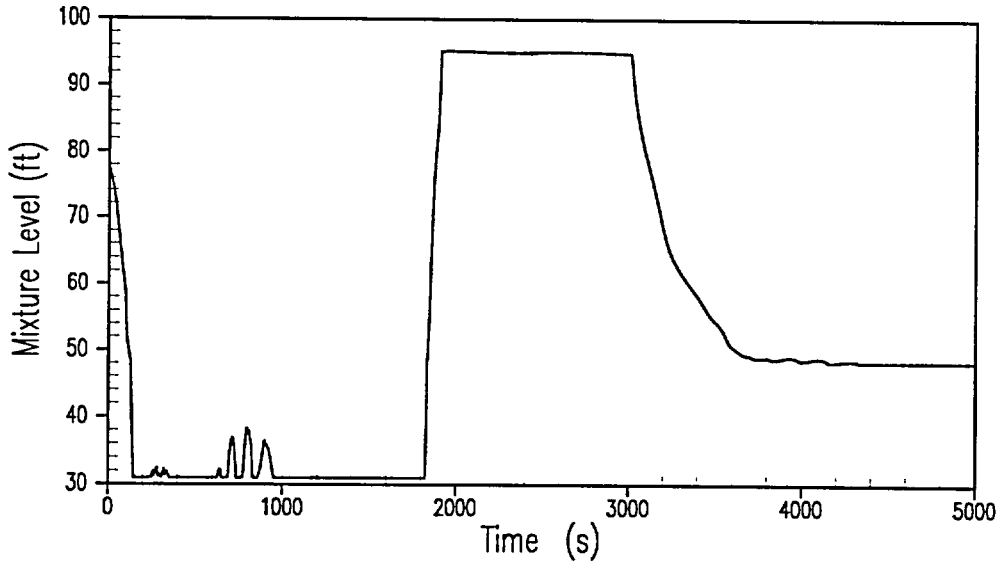



**Figure 440.098-23 2-inch Hot Leg Break in the Loop Without the Pressurizer – Pressurizer Mixture Level**

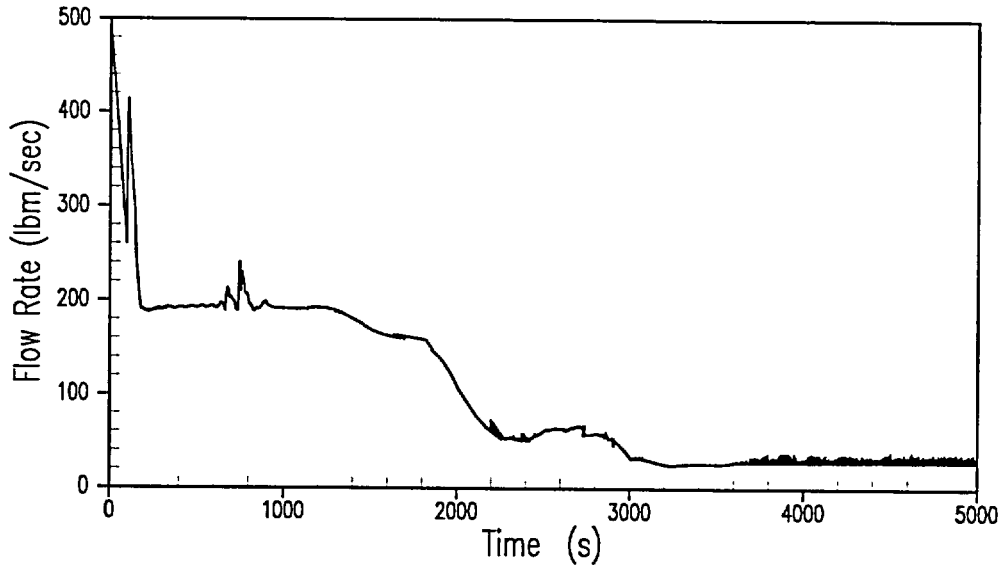



**Figure 440.098-24 2-inch Hot Leg Break in the Loop Without the Pressurizer – Break Liquid Discharge Rate**

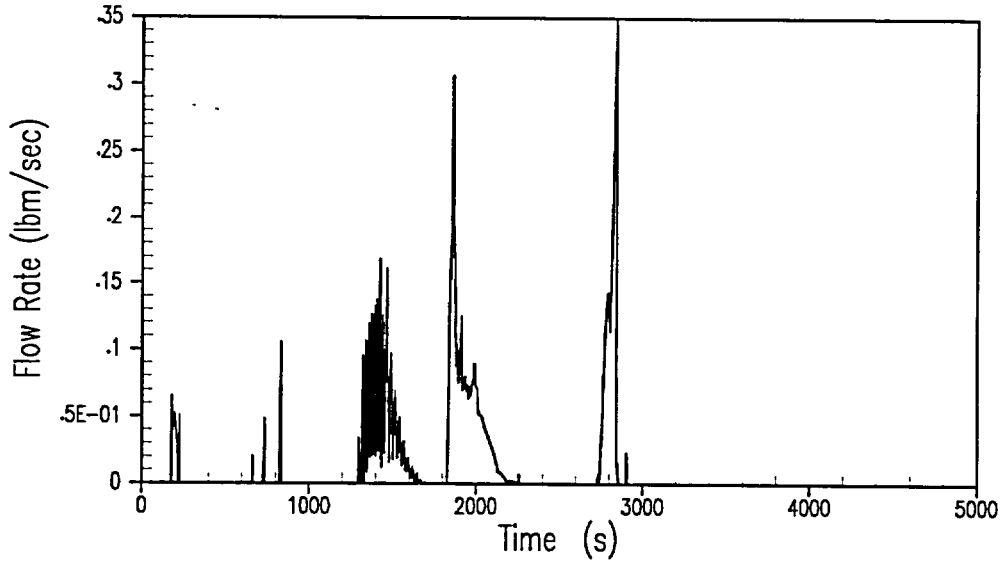



**Figure 440.098-25 2-inch Hot Leg Break in the Loop Without the Pressurizer – Break Vapor Discharge Rate**

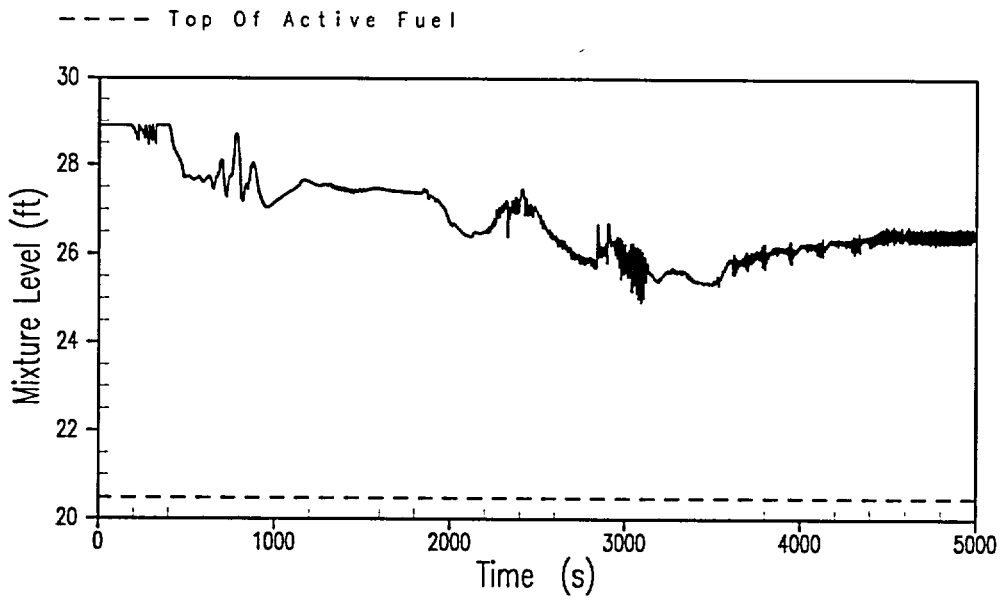



**Figure 440.098-26 2-inch Hot Leg Break in the Loop Without the Pressurizer – Core/Upper Plenum Mixture Level**

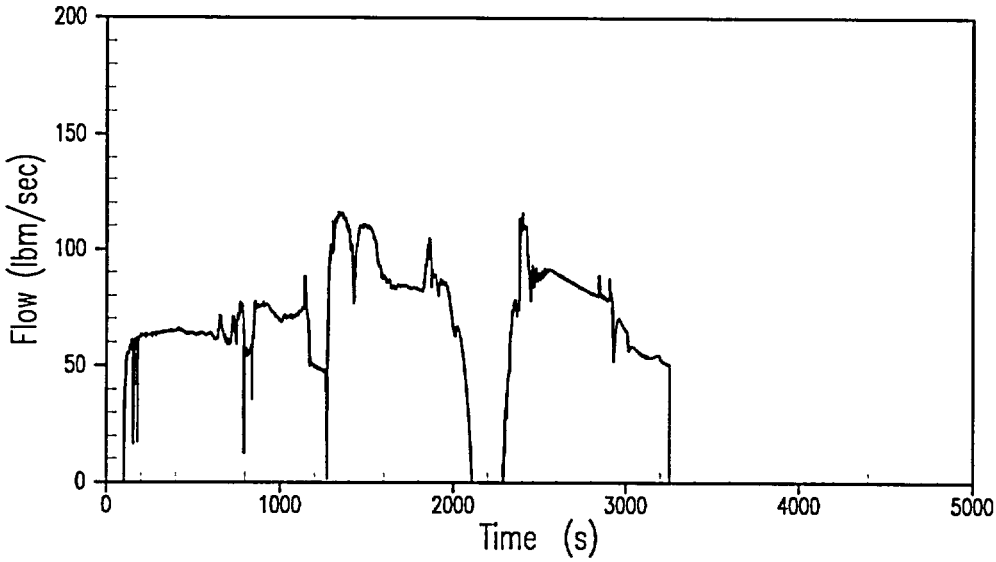



**Figure 440.098-27 2-inch Hot Leg Break in the Loop Without the Pressurizer – CMT-1 Injection Rate**

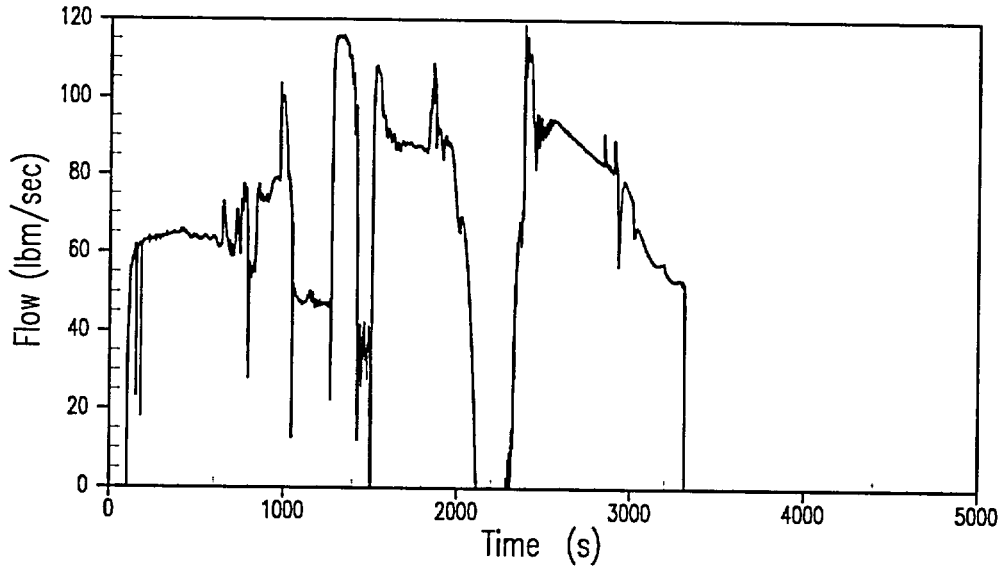



**Figure 440.098-28 2-inch Hot Leg Break in the Loop Without the Pressurizer – CMT-2 Injection Rate**

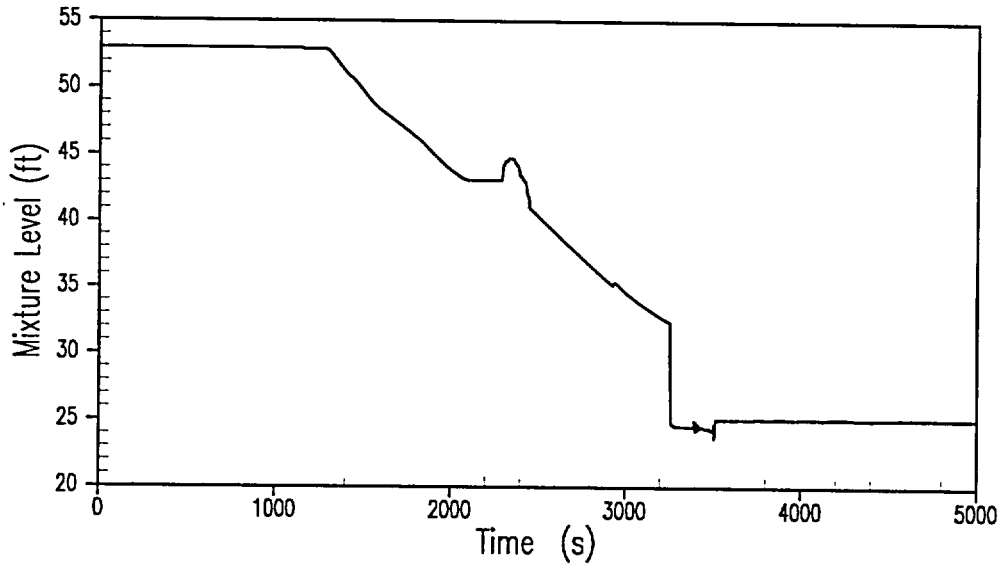



**Figure 440.098-29 2-inch Hot Leg Break in the Loop Without the Pressurizer – CMT-1 Mixture Level**

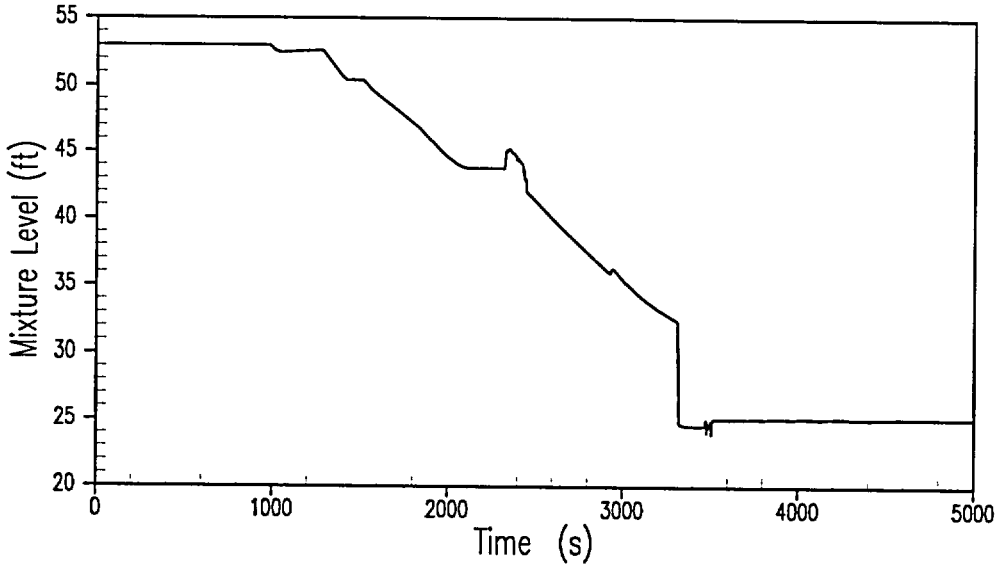



**Figure 440.098-30 2-inch Hot Leg Break in the Loop Without the Pressurizer – CMT-2 Mixture Level**

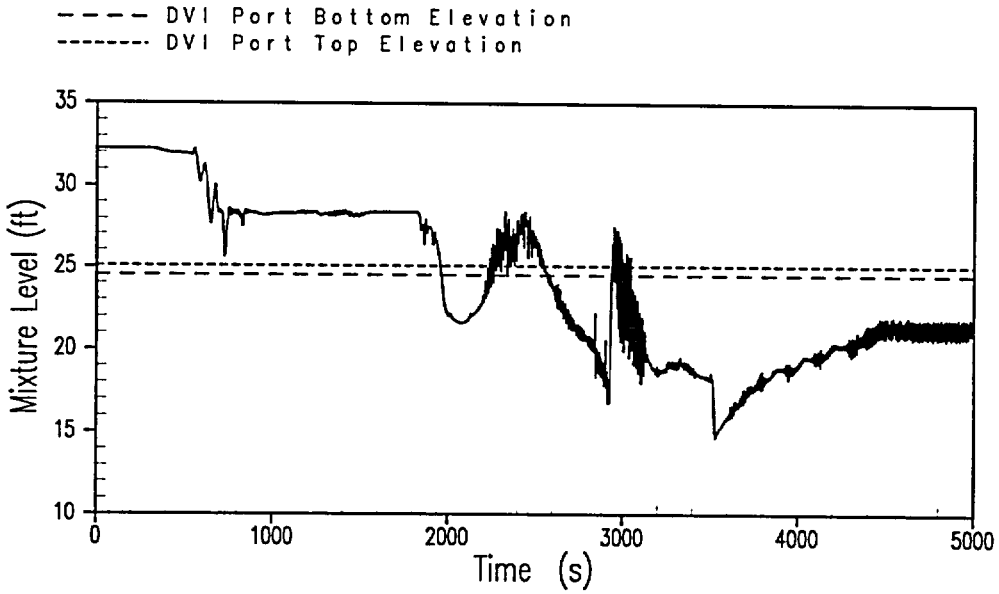



**Figure 440.098-31 2-inch Hot Leg Break in the Loop Without the Pressurizer – Downcomer Mixture Level**

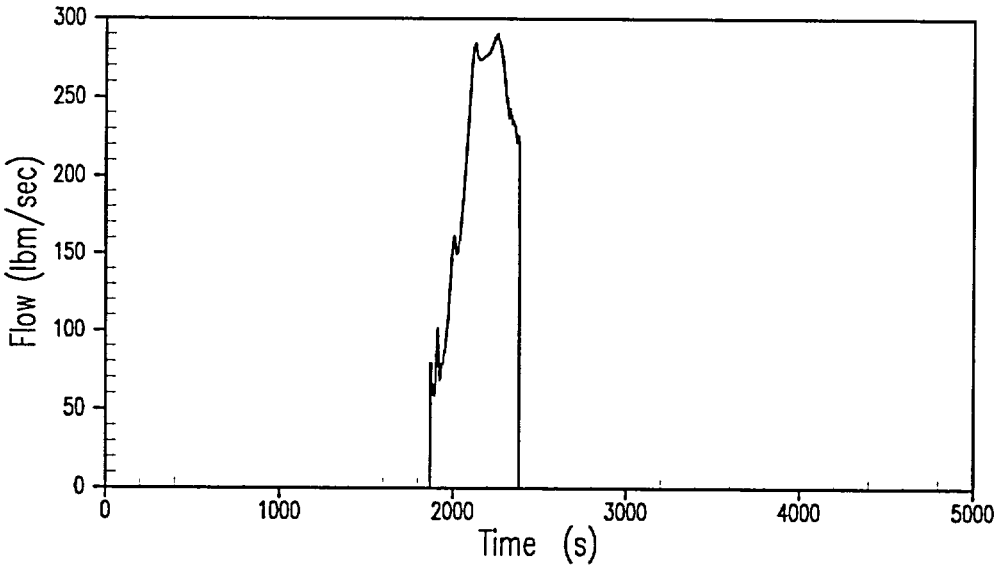



**Figure 440.098-32 2-inch Hot Leg Break in the Loop Without the Pressurizer – Accumulator-1 Injection Rate**

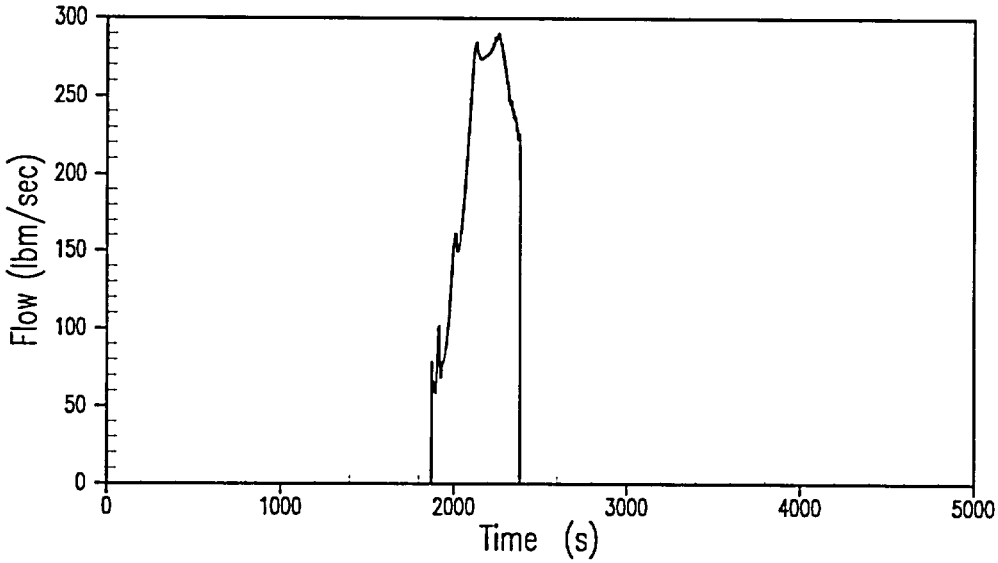



**Figure 440.098-33 2-inch Hot Leg Break in the Loop Without the Pressurizer – Accumulator-2 Injection Rate**

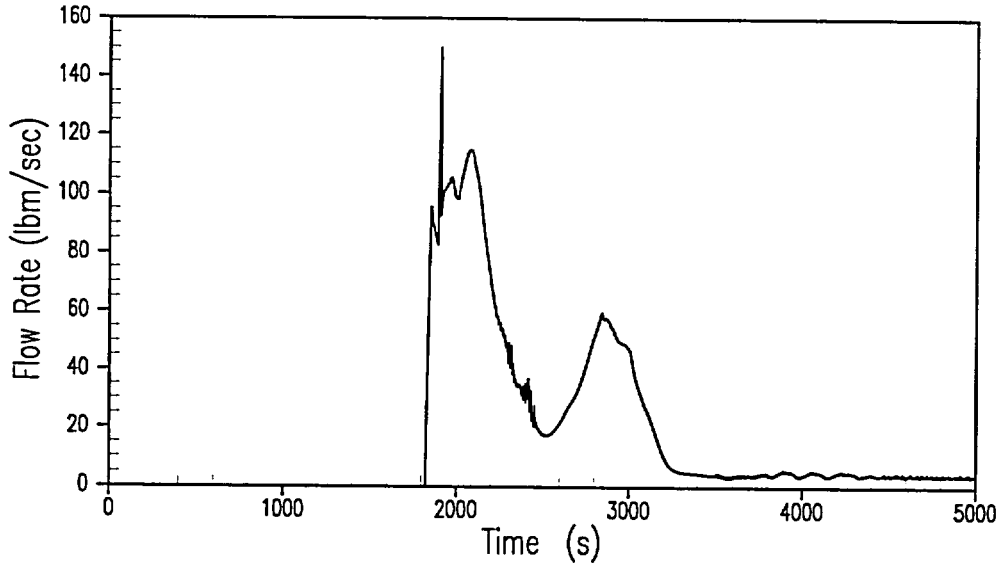



**Figure 440.098-34 2-inch Hot Leg Break in the Loop Without the Pressurizer – ADS 1-3 Liquid Discharge Rate**

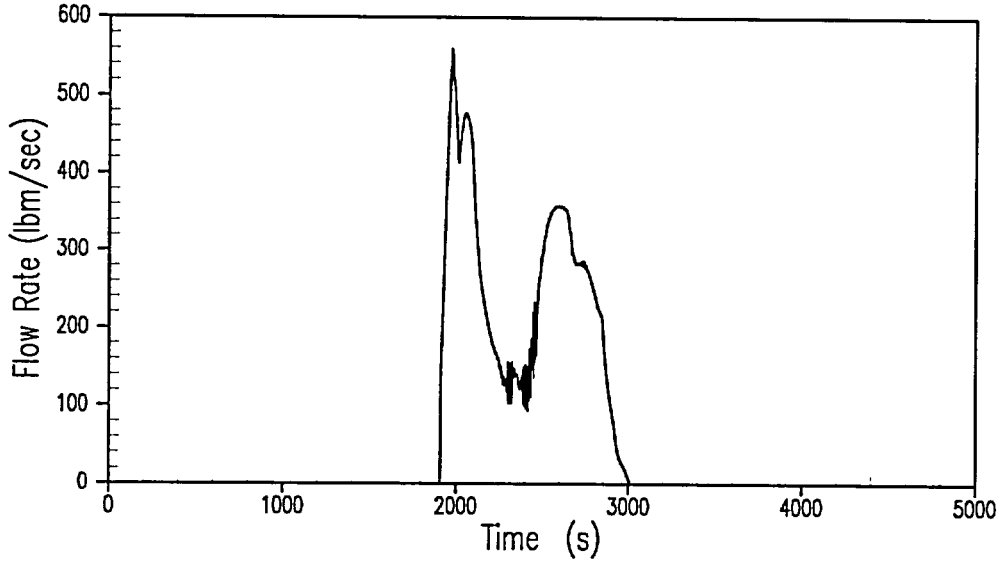



**Figure 440.098-35 2-inch Hot Leg Break in the Loop Without the Pressurizer – ADS 1-3 Vapor Discharge Rate**

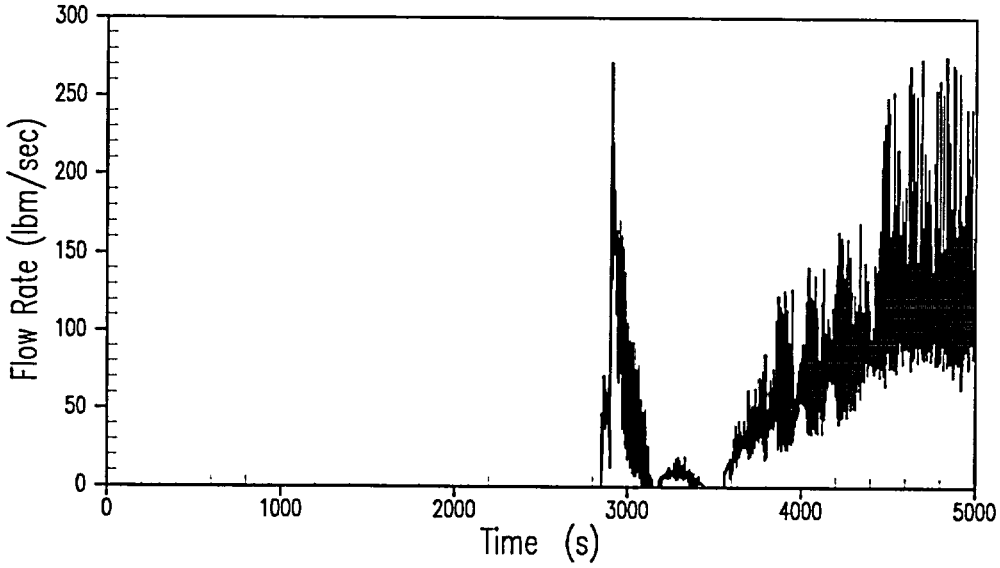



**Figure 440.098-36 2-inch Hot Leg Break in the Loop Without the Pressurizer – ADS-4 Liquid Discharge Rate**

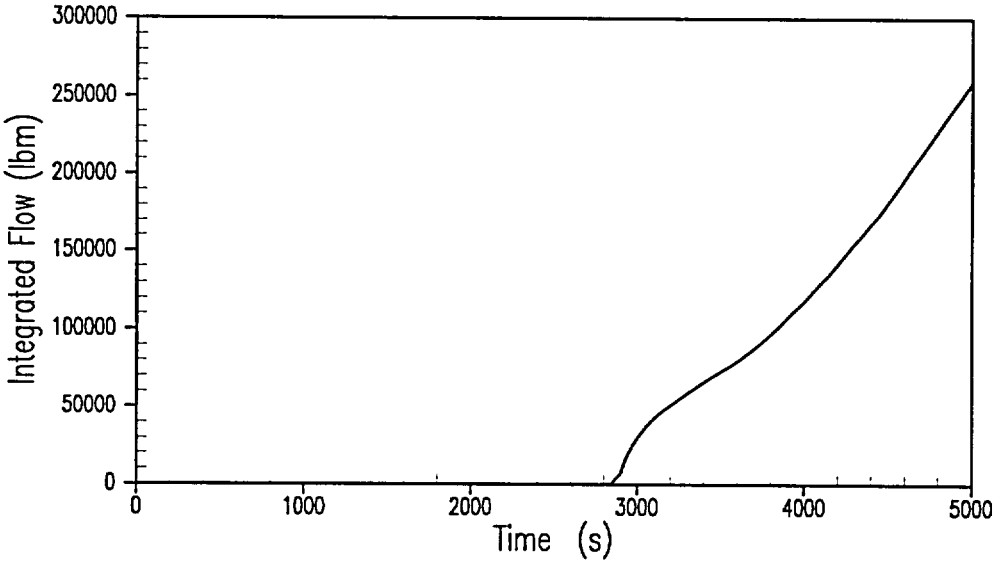



**Figure 440.098-37 2-inch Hot Leg Break in the Loop Without the Pressurizer – ADS-4 Integrated Discharge Flow**

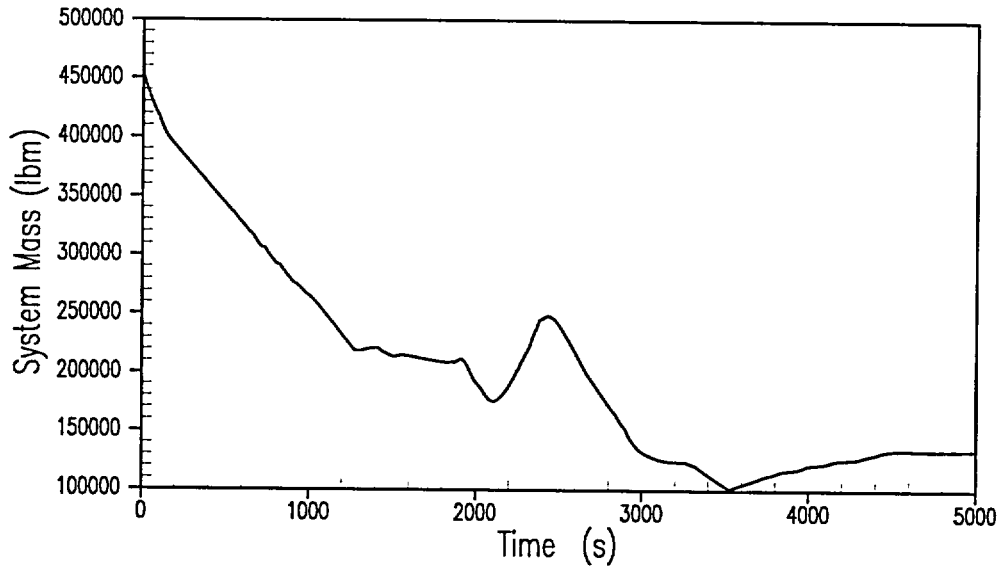



**Figure 440.098-38 2-inch Hot Leg Break in the Loop Without the Pressurizer – RCS System Inventory**

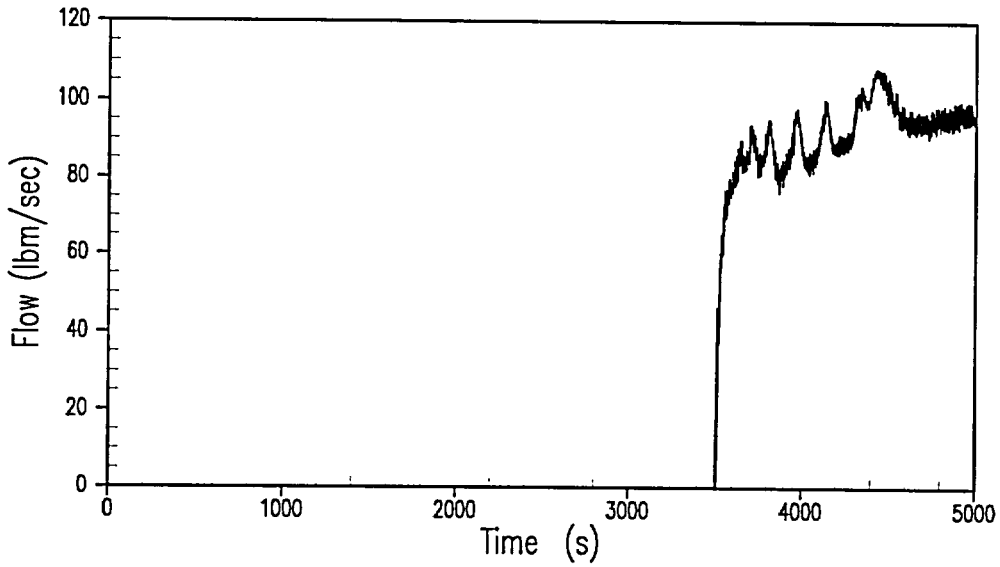



**Figure 440.098-39 2-inch Hot Leg Break in the Loop Without the Pressurizer – IRWST-1 Injection Rate**

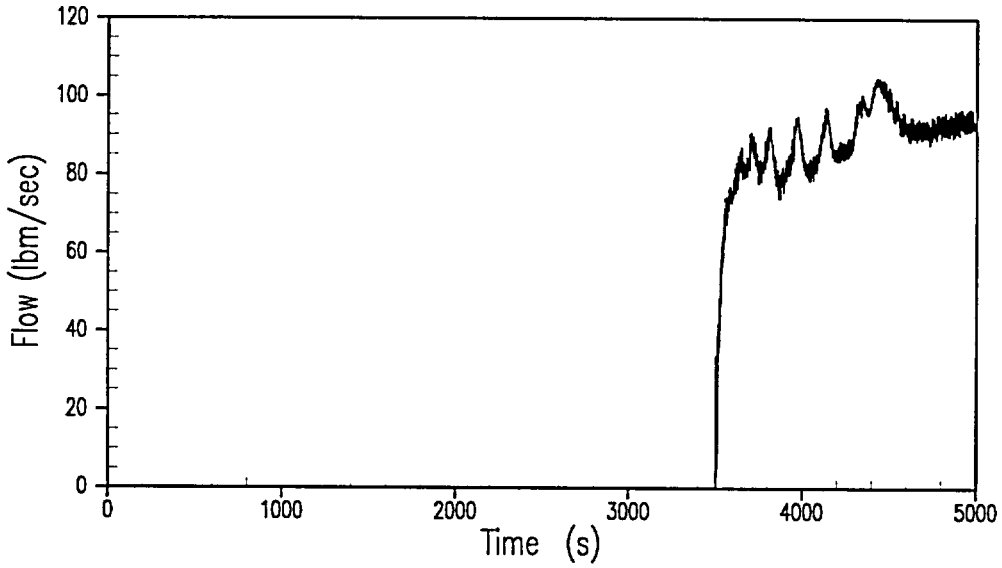



**Figure 440.098-40 2-inch Hot Leg Break in the Loop Without the Pressurizer – IRWST-2 Injection Rate**

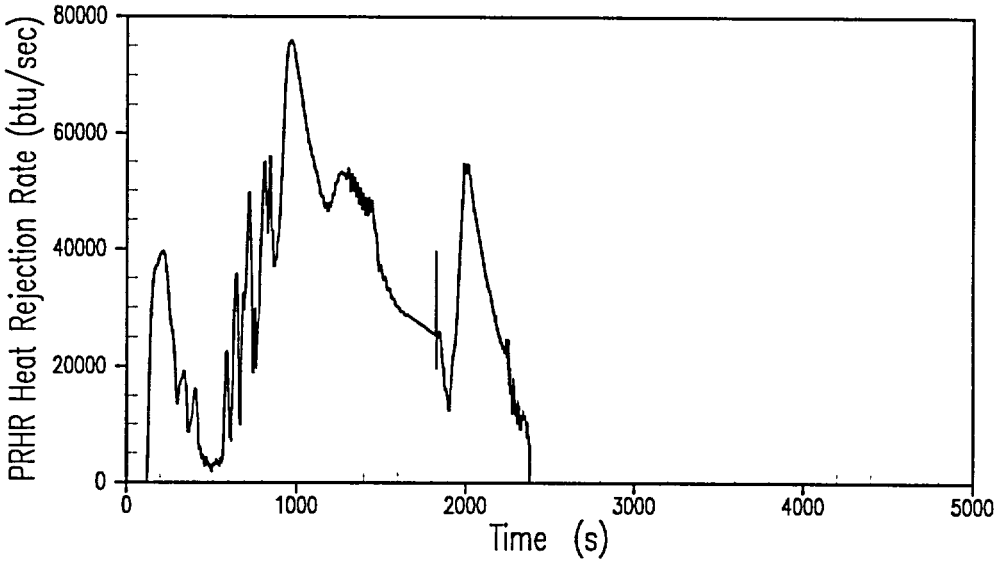



**Figure 440.098-41 2-inch Hot Leg Break in the Loop Without the Pressurizer – PRHR Heat Removal Rate**

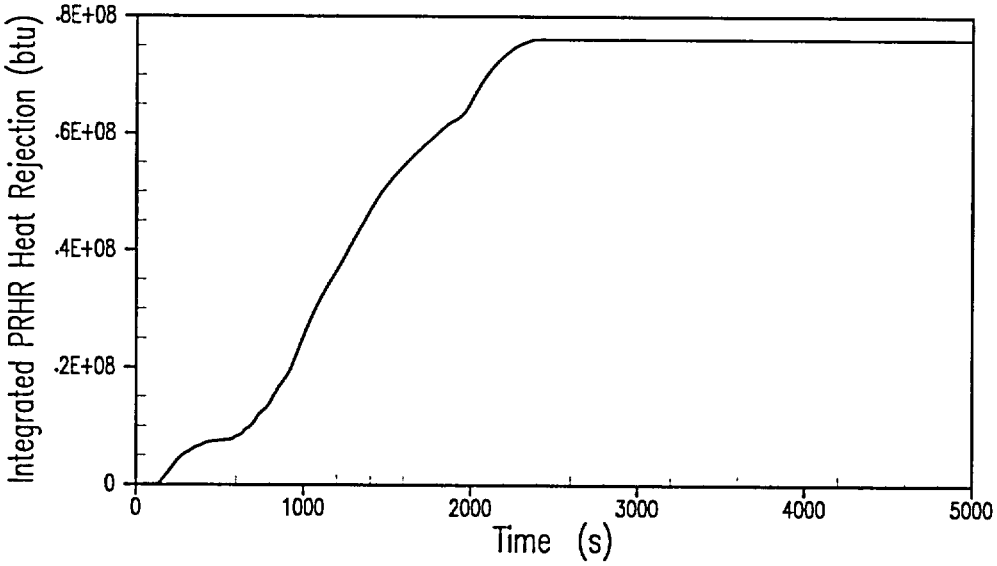



**Figure 440.098-42 2-inch Hot Leg Break in the Loop Without the Pressurizer – Integrated PRHR Heat Removal**

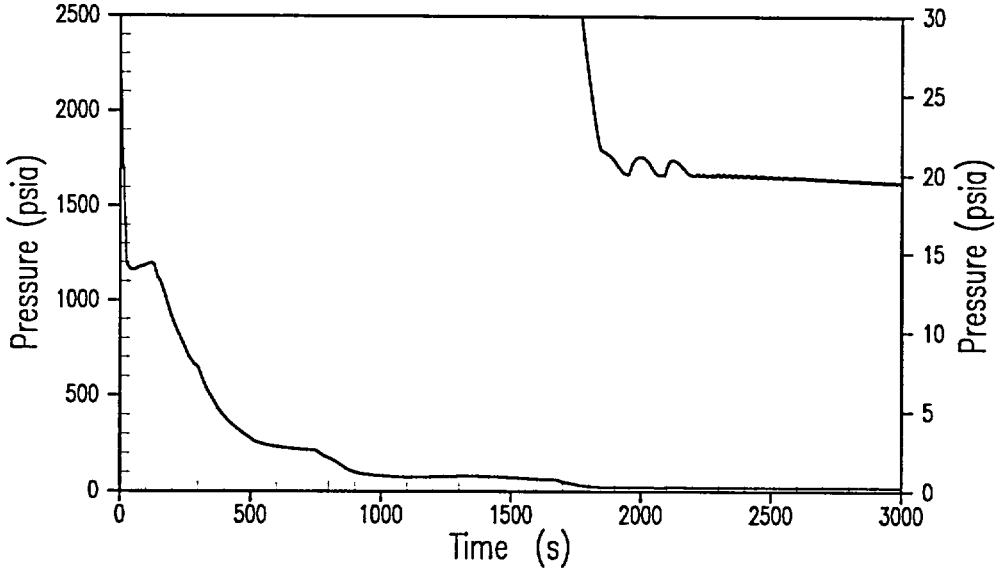



**Figure 440.098-43 Double-Ended Cold Leg Balance Line Break – RCS Pressure**

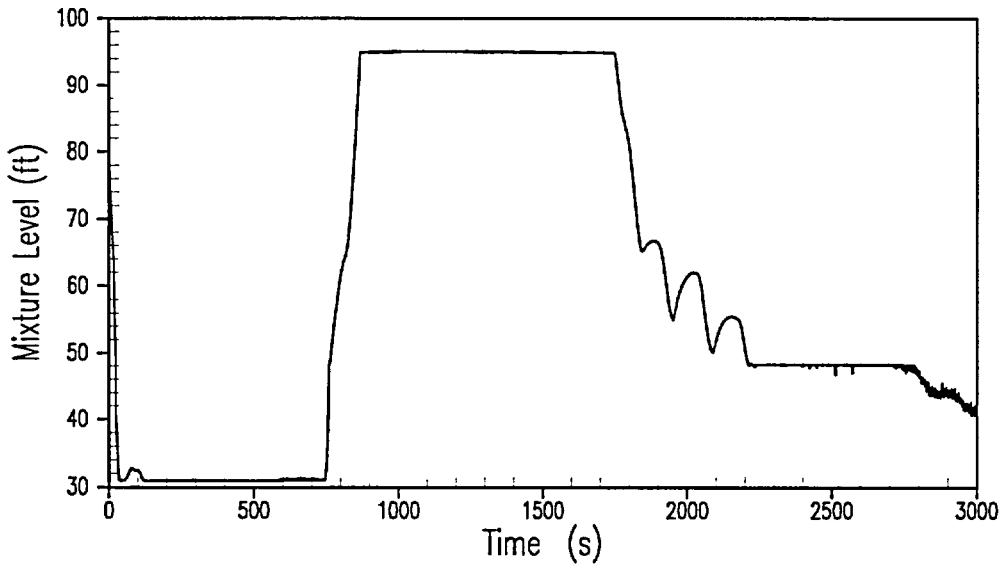



**Figure 440.098-44 Double-Ended Cold Leg Balance Line Break – Pressurizer Mixture Level**

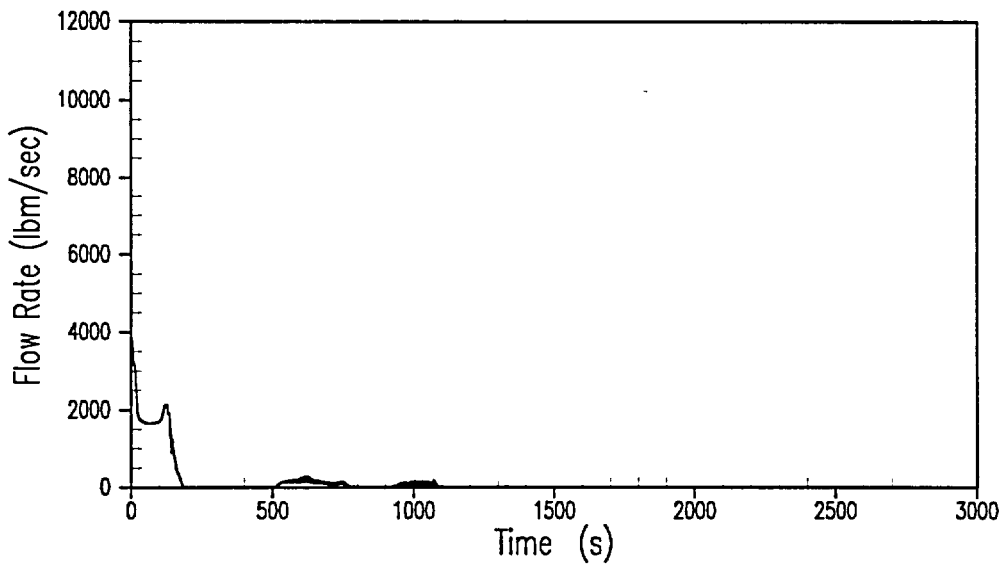



**Figure 440.098-45 Double-Ended Cold Leg Balance Line Break – Break Vapor Discharge Rate**

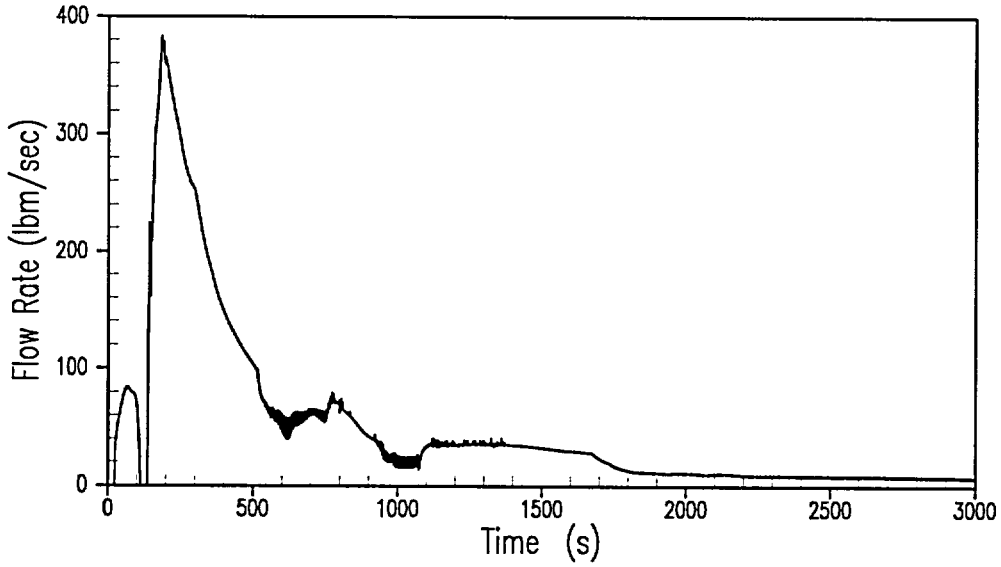



**Figure 440.098-46 Double-Ended Cold Leg Balance Line Break – Break Vapor Discharge Rate**

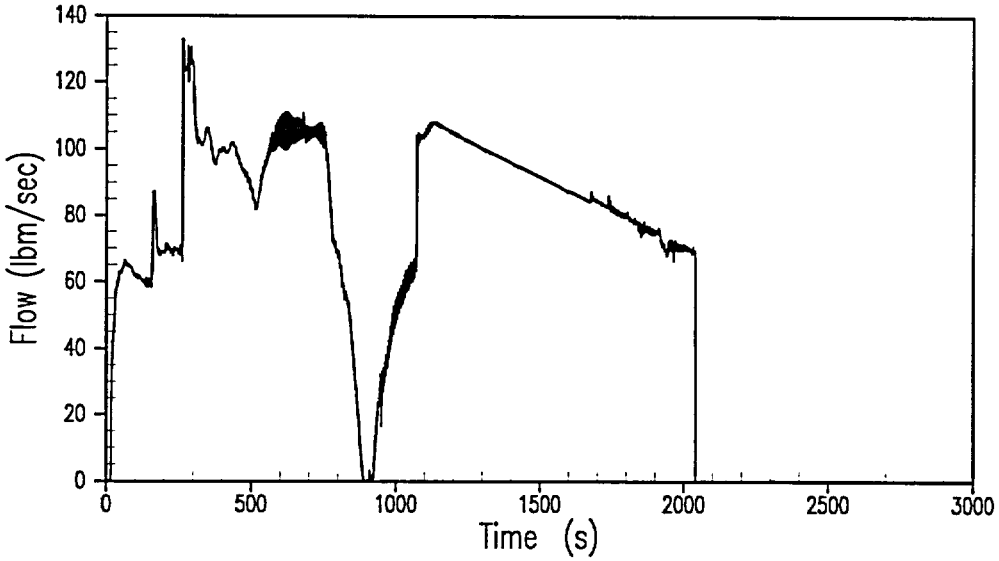



**Figure 440.098-47 Double-Ended Cold Leg Balance Line Break – CMT-2 Injection Rate**

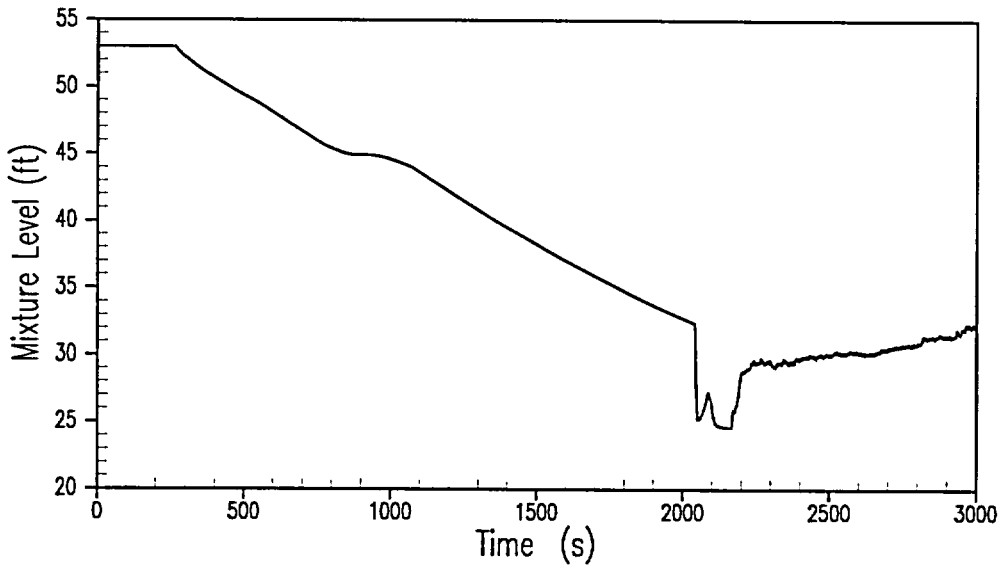



**Figure 440.098-48 Double-Ended Cold Leg Balance Line Break – CMT-2 Mixture Level**

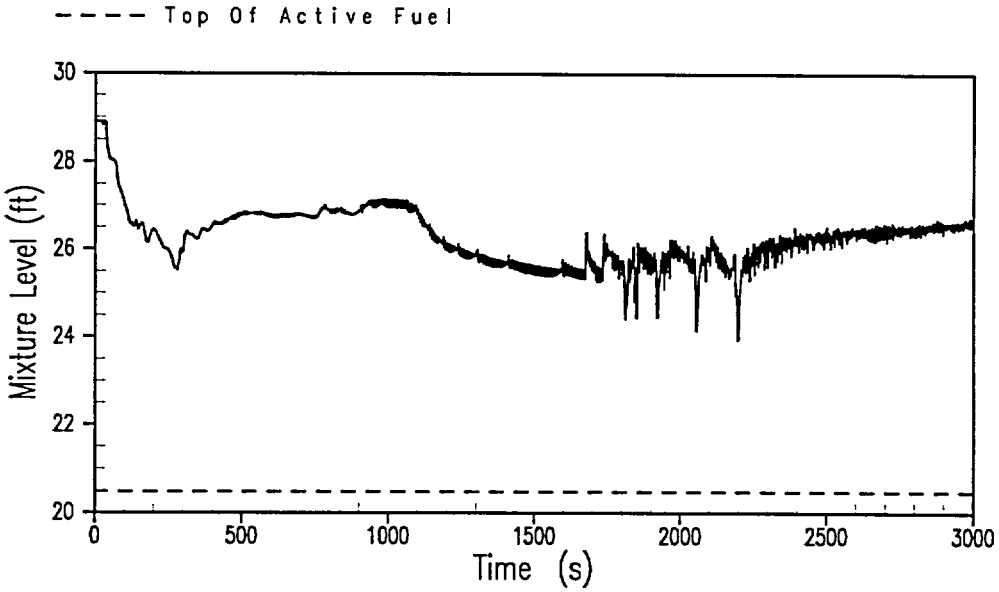



**Figure 440.098-49 Double-Ended Cold Leg Balance Line Break – Core/Upper Plenum Mixture Level**

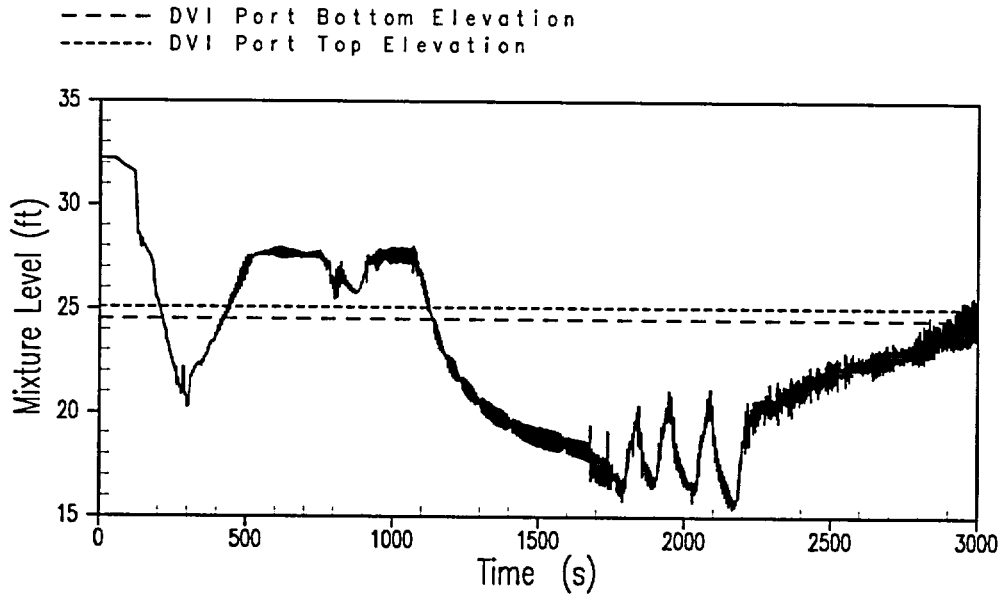



**Figure 440.098-50 Double-Ended Cold Leg Balance Line Break – Downcomer Mixture Level**

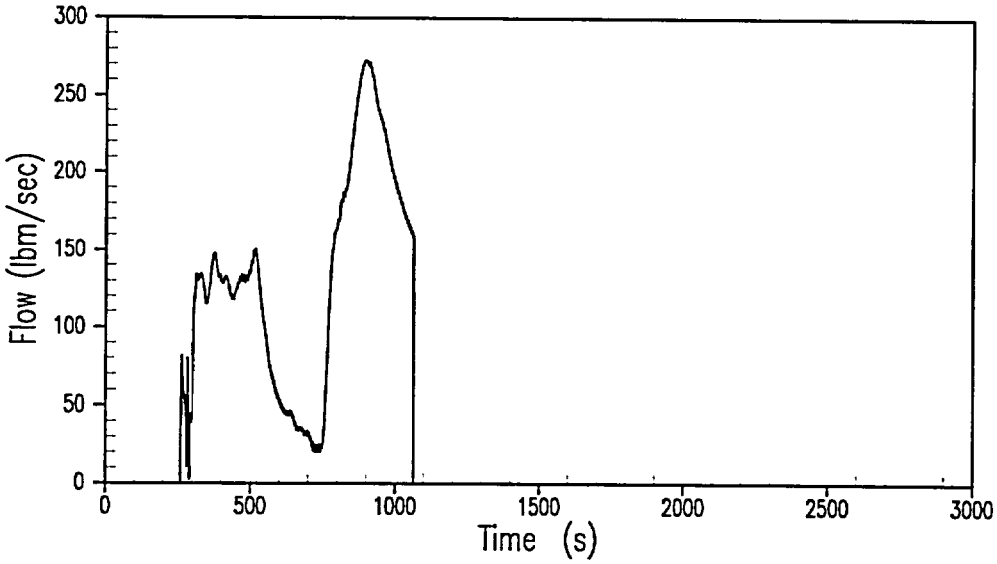



**Figure 440.098-51 Double-Ended Cold Leg Balance Line Break – Accumulator-1 Injection Rate**

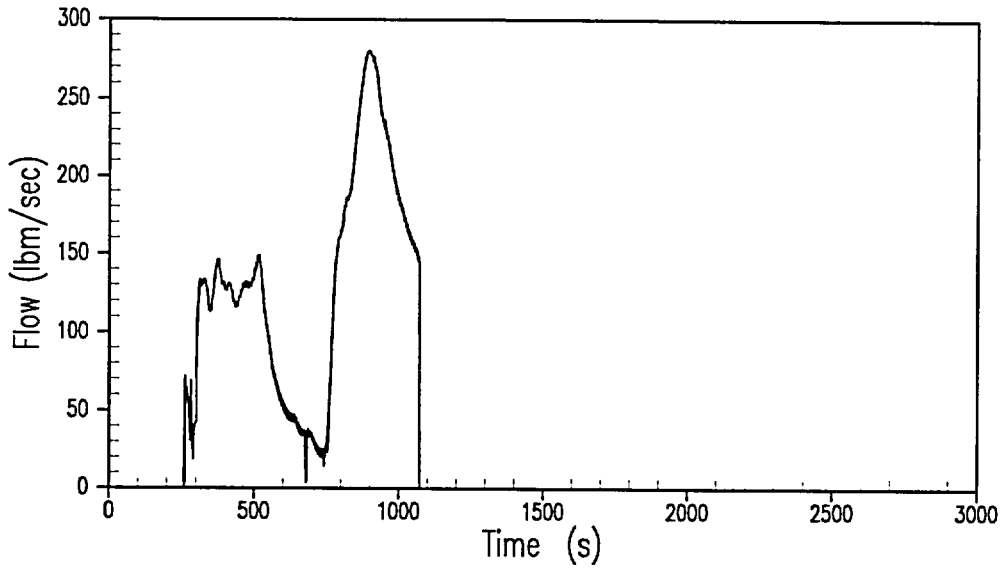



**Figure 440.098-52 Double-Ended Cold Leg Balance Line Break – Accumulator-2 Injection Rate**

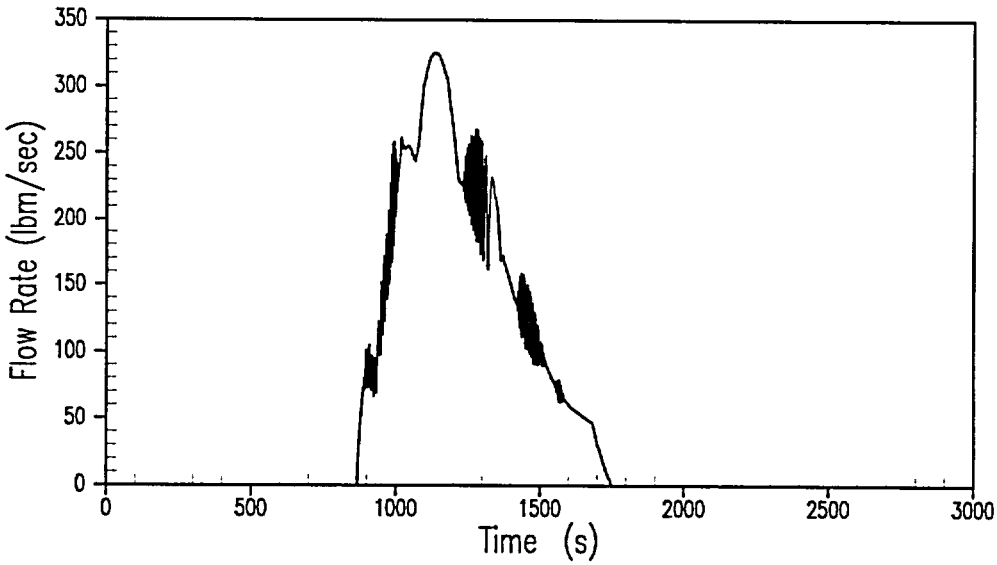



**Figure 440.098-53 Double-Ended Cold Leg Balance Line Break – ADS 1-3 Liquid Discharge Rate**

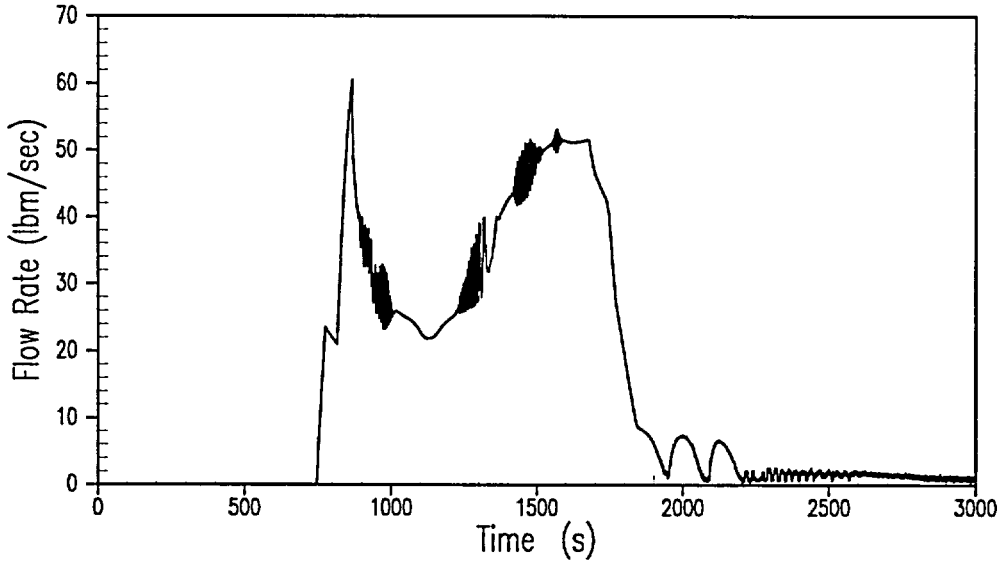



**Figure 440.098-54 Double-Ended Cold Leg Balance Line Break – ADS 1-3 Vapor Discharge Rate**

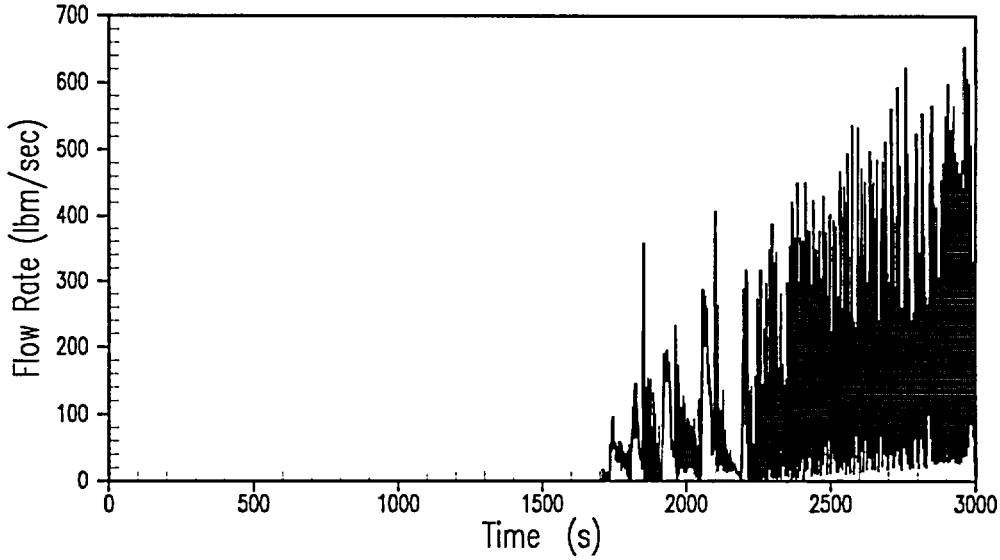



**Figure 440.098-55 Double-Ended Cold Leg Balance Line Break – ADS-4 Liquid Discharge Rate**

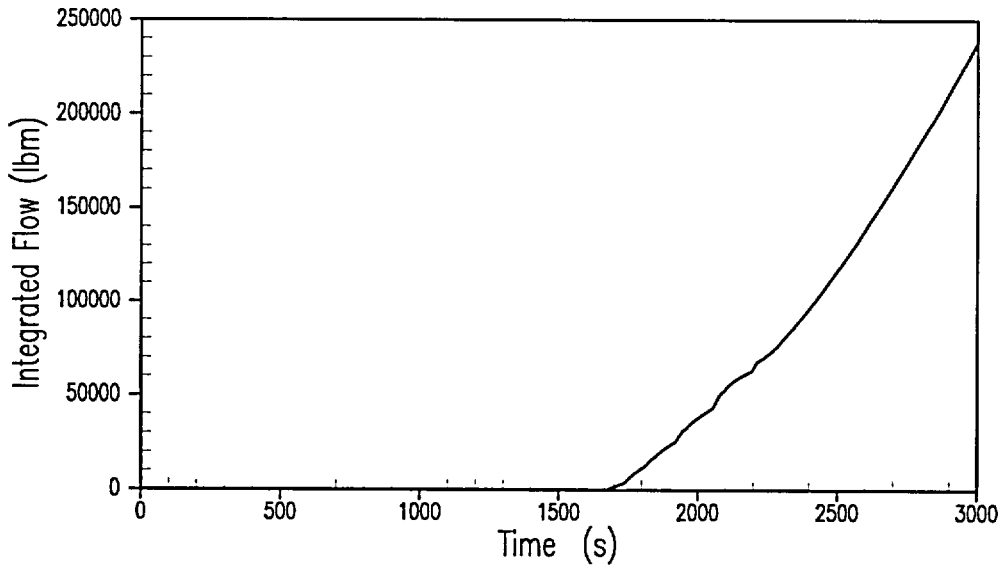



**Figure 440.098-56 Double-Ended Cold Leg Balance Line Break – ADS-4 Integrated Discharge Flow**

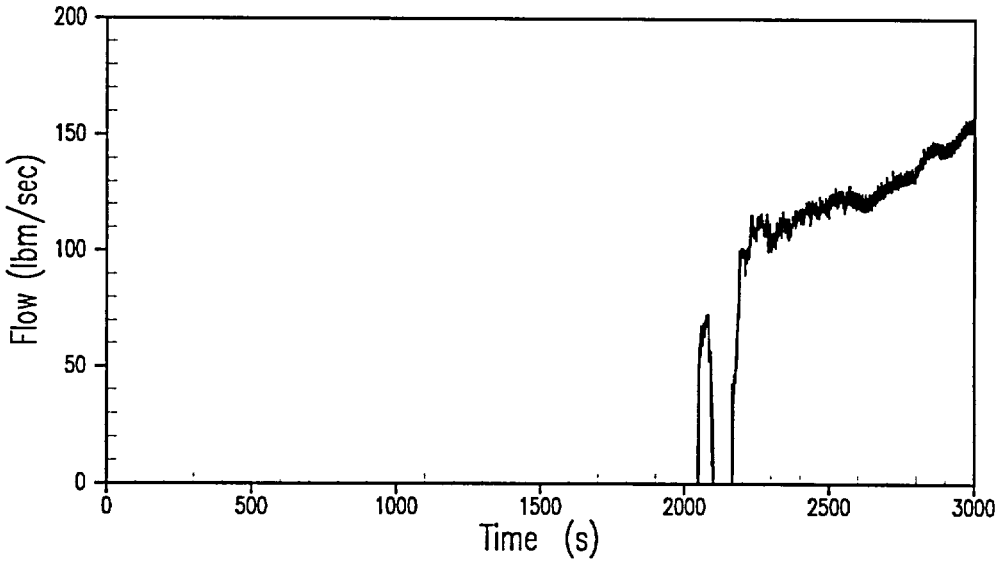



**Figure 440.098-57 Double-Ended Cold Leg Balance Line Break – IRWST-1 Injection Rate**

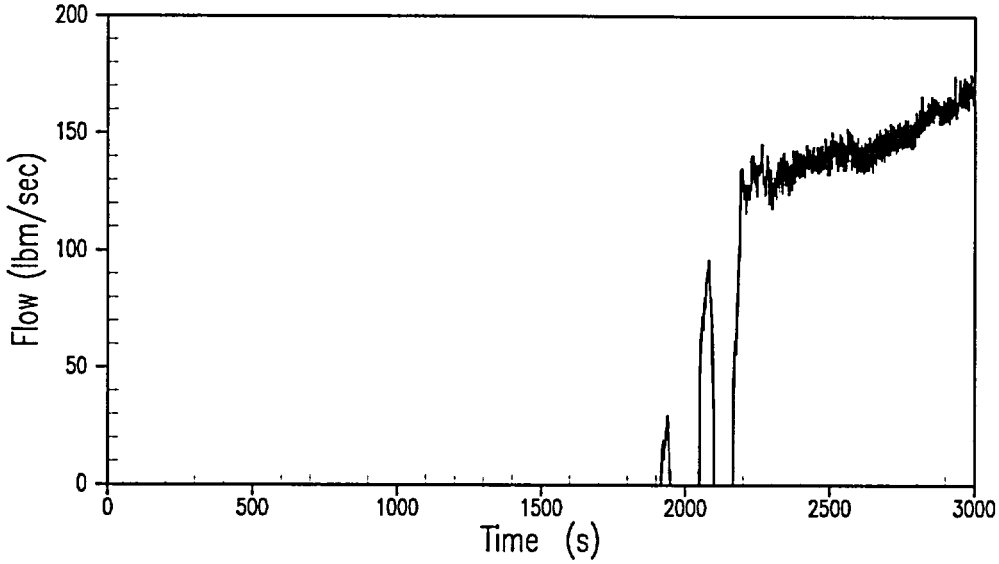



**Figure 440.098-58 Double-Ended Cold Leg Balance Line Break – IRWST-2 Injection Rate**

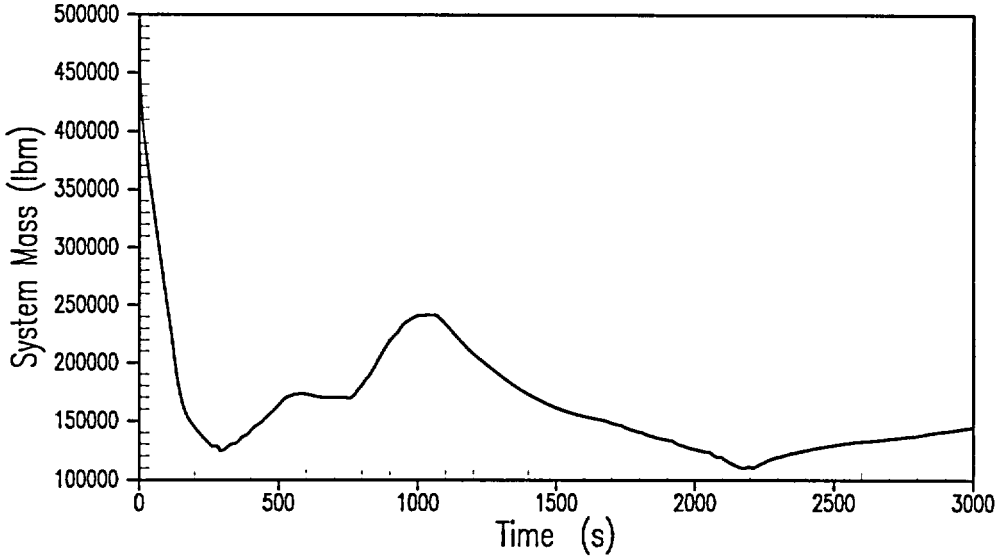



**Figure 440.098-59 Double-Ended Cold Leg Balance Line Break – RCS System Inventory**

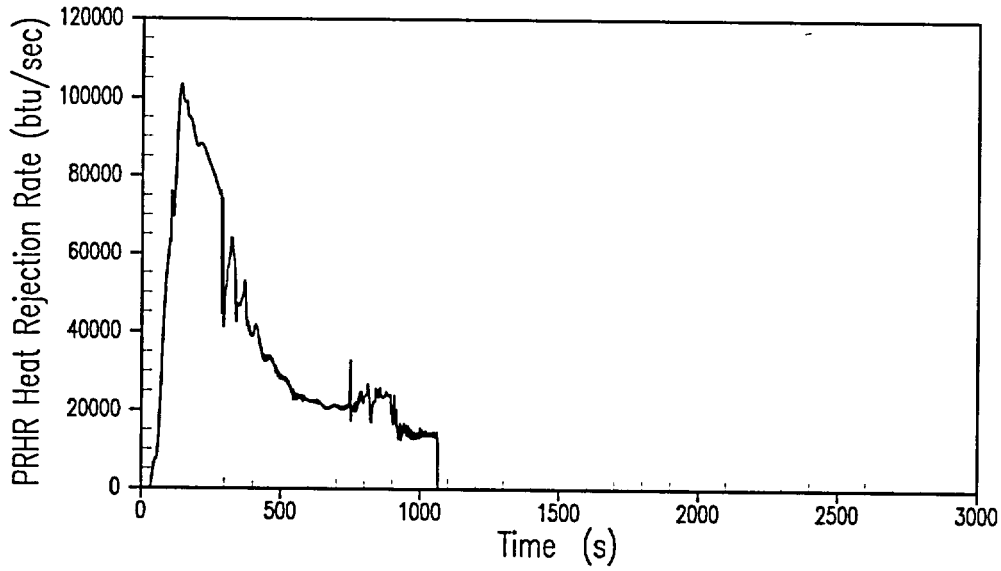



**Figure 440.098-60 Double-Ended Cold Leg Balance Line Break – PRHR Heat Removal Rate**

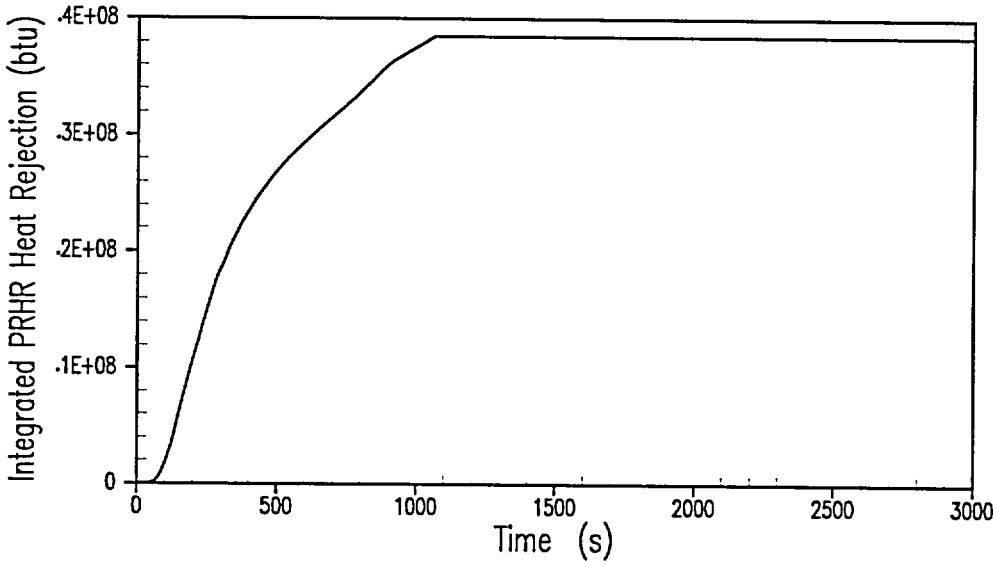



**Figure 440.098-61 Double-Ended Cold Leg Balance Line Break – Integrated PRHR Heat Removal**

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number. 440.129

***Question:***

In WCAP-15833, Section 2.2.2, "Separate Effects Verification," describes assessment of the horizontal stratified flow model in WCOBRA/TRAC using data from the Lim test facility. Upper plenum/hot leg entrainment during the post-ADS period is given a high importance in the phenomena identification and ranking table (PIRT). The maximum steam velocity in the Lim tests is stated to be 18 meters-per-second (m/s) and the maximum water velocity is stated to be 41 centimeters-per-second (cm/s). NRC staff calculations for AP1000 using RELAP5 predict hot leg velocities for both steam and water that are in excess of those in the Lim tests for the period following ADS-4 actuation during a SBLOCA. Please justify that the steam and water velocities in the hot legs of AP1000 following SBLOCA will be within the experimental range of the Lim tests or provide other justification for the horizontal stratified flow model in WCOBRA/TRAC.

**Westinghouse Response:**

The primary use of the Lim tests was for condensation of steam in cocurrent, horizontal flow as described in Section 2.2.2 of WCAP-15833. Condensation heat transfer in the AP1000 hot leg is a minor effect during the ADS-4 to IRWST transition phase as saturated or near-saturated conditions exist in the hot leg during this phase of a SBLOCA transient. Therefore, the WCOBRA/TRAC condensation model for horizontal stratified flow is acceptable for this application to the hot legs in AP1000.

The horizontal stratified flow model is described in Section 2.2.1 of WCAP-15833. This model was used to simulate the OSU test facility that has been shown in WCAP-15613 to be adequately scaled to AP1000. The results of the test simulation (test SB18) for the ADS-4 to IRWST transition phase was judged to be in reasonable agreement with the SB18 OSU test data.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

**WCAP Revision:**

None

RAI Number: 440.151

***Question:***

Section 2.2.1.5 states that the quality ($x$) entering the automatic depressurization system stage 4 (ADS-4) branch line from the hot leg is calculated using,

$$x = R^{3.25(1-R)^2} \tag{1}$$

where, $R = (h/h_b)$. Here, h is the distance between the top of the main pipe and the liquid surface, and $h_b$ is the critical distance for entrainment onset. The onset of entrainment is obtained from an expression based on the gas phase Froude number ($Fr$) at the branch line inlet,

$$Fr_g = \frac{U_g}{\sqrt{\frac{gd\Delta\rho}{\rho_g}}} = C_1\left(\frac{h_b}{d}\right)^{C_2} \tag{2}$$

Various values of $C_1$ and $C_2$ have been proposed by different investigators, several of which are listed in Table 1.

Table 1

| Reference | $C_1$ | $C_2$ |
|---|---|---|
| Anderson and Benedetti [1] | 0.35 | 2.5 |
| Rouse [2] | 5.67 | 2.0 |
| Schrock, et al. [3] | 0.395 | 2.5 |
| Smoglie [4] | 0.355 | 2.5 |
| Maciaszek and Micaelli [5] | 1.75 | 1.5 |

Equations (1) and (2) are based on experiments in which the branch line diameter is small in comparison to the diameter of the horizontal pipe. In general, the ratio of the branch line to the main pipe was less than 0.1 in development of these correlations.

The coefficients used in WCOBRA/TRAC-AP are those of Anderson and Benedetti. The TRAC-M code also uses Equations (1) and (2) but with the coefficients by Smoglie. However, the coefficients by Smoglie and by Anderson and Benedetti are nearly identical, and the WCOBRA/TRAC-AP and TRAC-M models generate the same results. Figure 1 shows the predicted variation of branch line quality as a function of the branch line Froude number assuming the liquid level in the main pipe is at the midplane (h/D = 0.5) for two cases in which the ratio of the branch line to main pipe diameter is large compared to the database used to determine the coefficients in Table 1. ATLATS is a separate effects test facility with dimensions scaled to those of AP600 that was used to investigate phase separation at the junction between a small branch line and a main pipe. The ratio of the branch line to the main pipe in ATLATS is d/D = 0.33, which is less than the AP1000 ratio of d/D = 0.47. Calculations for both the ATLATS test facility geometry and the AP1000 hot leg and ADS-4 junction are also shown. As the ratio (d/D) increases, the model predicts lower branch line qualities. However, in neither case are the predicted branch line qualities in reasonable agreement with the experimental data shown in Figure 1. These data are from ATLATS tests, with the equilibrium level at approximately h/D = 0.5, consistent with the calculations.

The comparison suggests that the model and coefficients of Equations (1) and (2) and Table 1, grossly overpredict the ADS-4 quality for conditions expected in AP1000. Please provide suitable justification for Equations (1) and (2) and their coefficients for the large d/D ratios in the AP1000 design. Provide justification that the phase separation equations used in the WCOBRA/TRAC-AP code are appropriate for AP1000 ADS-4 analysis in light of these data.

References

[1] Anderson, J. L., and Benedetti, R. L., "Critical Flow Through Small Pipe Breaks," EPRI/NP-4532, 1986.

[2] Rouse, H., "Seven Exploratory Studies in Hydraulics," J. Hydr. Div. Proc. ASCE, HY4, pp (1038) 1-35, August, 1956.

[3] Schrock, V. E., Revankar, S. T., Mannheimer, R., and Wang, C-H., "Small Break Critical Discharge -- The Roles of Vapor and Liquid Entrainment in a Stratified Two-Phase Region Upstream of the Break," NUREG/CR-4791, 1986.

[4] Smoglie, C., "Two-Phase Flow Through Small Branches in a Horizontal Pipe with Stratified Flow, Ph. D. Dissertation, Univ. of Karlsruhe, 1984.

[5] Maciaszek, T., and Micaelli, J. C., "The CATHARE Phase Separation Model in Tee Junctions," SETh/LEML-EM/89-159, 1986.

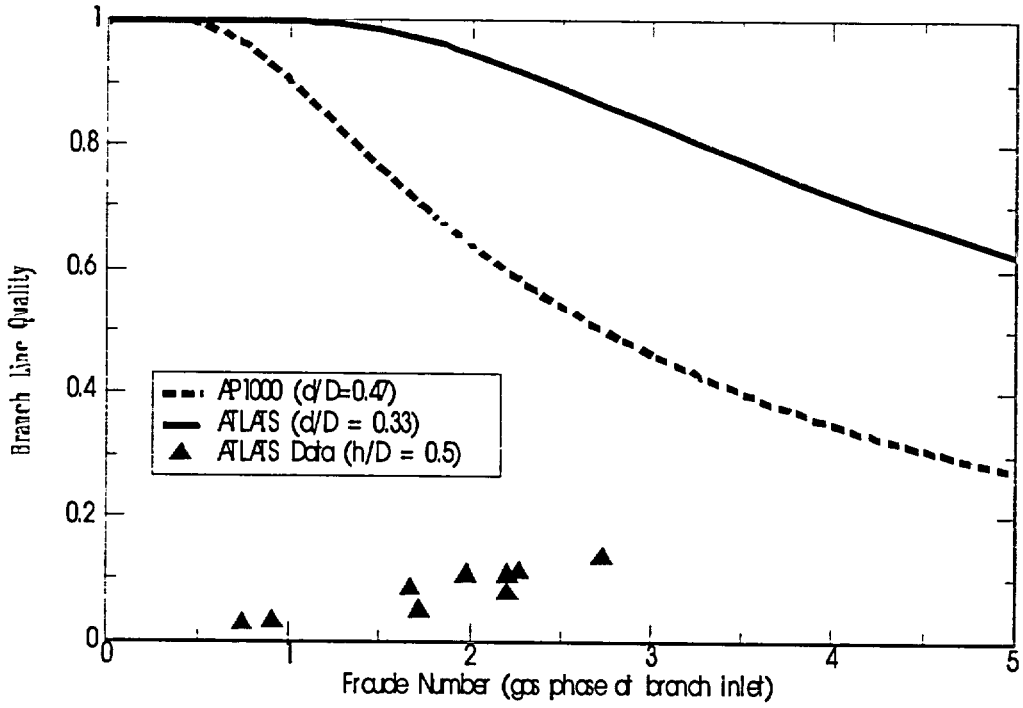



Figure 1. Branch line quality for liquid level at main pipe midplane for ATLATS and AP1000.

**Westinghouse Response:**

Entrainment of liquid from a stratified surface in a horizontal pipe to a vertical off-take has been studied and found by several investigators (including those cited in references 1 through 4 above) to correlate with Froude number and a geometric ratio of entrainment onset height to off-take diameter (Z/d). In section 4 of WCAP-15613 (AP1000 PIRT and Scaling Assessment) it was demonstrated that AP600 test facilities such as OSU adequately scale (relative to AP1000) Froude number in the hot legs and the resulting liquid entrainment inception correlation for stratified flow conditions. The entrainment correlations described in the references listed above use the same basic form consisting of Froude number and a geometric ratio of entrainment onset height to off-take diameter (Z/d). As the staff points out, there is some notable variation among the different experiments as different geometric ratios of off-take diameter to main pipe diameter (i.e. d/D) were tested and various coefficients (C1) or exponents (C2) were proposed to correlate the data as shown in Table 1 above. These investigations covered a range of d/D ratios from about 0.03 to 0.15. The d/D ratio for AP1000 is 0.47. Throughout the tested range,

however, the investigations demonstrated that the basic form of the entrainment correlation remains valid as it appears to do a reasonable job of representing the experimental data. Therefore, although the AP1000 d/D ratio is outside the tested range, based upon the trend of these investigations, it is reasonable to expect the same form of the correlation to remain valid at larger d/D with only variation in the coefficient (C1) or exponent (C2) with scale. To address this, sensitivity calculations of entrainment from the hot leg to the ADS-4 offtake in AP1000 were performed with the WCOBRA/TRAC-AP models where the coefficient (C1) and exponent (C2) associated with entrainment inception correlation were varied as described in Appendix A.4 of WCAP-15833. The calculations indicated that the flow regime for AP1000 during the ADS-IRWST transition phase of a SBLOCA is predominantly stratified in the hot leg upstream of the ADS-4 off-take. Hence, the correlations are applicable. There is little sensitivity to variation in the coefficient (C1) or exponent (C2) associated with the entrainment inception correlation. Based upon the investigations at different geometric scales and the sensitivity study performed with WCOBRA/TRAC-AP it seems reasonable to apply these entrainment correlations to AP1000.

With respect to addressing the data associated with the ATLATS separate effects test facility, Westinghouse does not have detailed information regarding these tests (i.e facility layout, boundary conditions, test procedures, scaling analysis, etc.) and therefore these tests have not been analyzed by Westinghouse. Based on our understanding of the ATLATS test program, a possible explanation for the different behavior displayed in ATLATS (relative to the stratified flow type entrainment behavior expected in AP1000 and as seen in other test facilities) is that the ATLATS test facility is producing a different flow regime that could be attributed to its non-prototypic or incomplete simulation of the actual AP1000 configuration. The amount of effort necessary for Westinghouse to perform a detailed assessment of the ATLATS tests program, including the detailed evaluation of its design, scaling analysis, boundary conditions, test procedures, and test results that would otherwise be necessary for Westinghouse to consider the results of this test program are not warranted. Westinghouse believes that the staff can make the necessary safety determination for the AP1000 without a detailed assessment by Westinghouse of the ATLATS facility test results. We believe that WCAP-15833 Revision 1 provides the necessary information for the staff to determine the importance of upper plenum and hot leg entrainment and their effect on AP1000 plant safety.

**Design Control Document (DCD) Revision:** None

**PRA Revision:** None

**WCAP Revision:** None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 440.158

***Question:***

WCAP-15833

Specify the steam flow rate for the liquid flows shown in Figure 2-7 and provide a key for the curves in this figure. (For example, there are two curves with open triangles and solid lines, and there are two curves with "X" and a solid line.)

**Westinghouse Response:**

Each curve on the figure indicates the test water thickness versus axial position for a specific liquid ($W_l$) and steam ($W_g$) flow rate. A revised Figure 2-7 of WCAP-15833 is attached that provides the requested clarification.

**Design Control Document (DCD) Revision:**

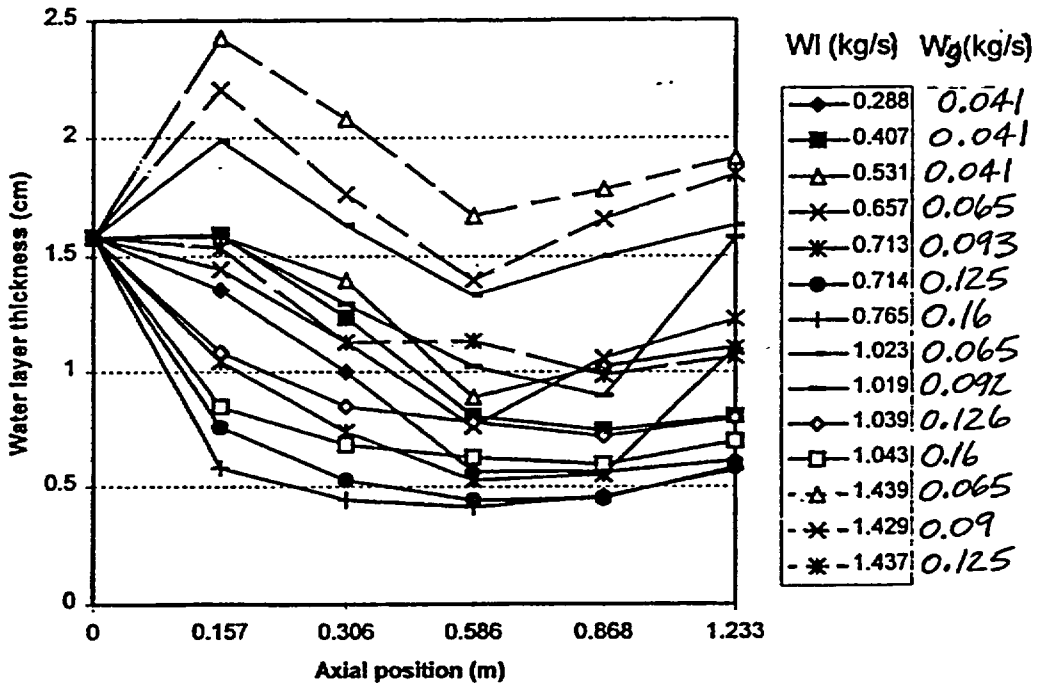
None

**PRA Revision:**

None

**WCAP-15833 Revision:**

Replace Figure 2-7 with the attached figure.




**Measured Water Thickness Versus Axial Position for Various Liquid (WI) Flowrates and Inlet Water Layer Thickness of 1.583 cm**

**Replacement WCAP-15833 Figure 2-7**

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number 440.159

***Question:***

WCAP-15833 Revision 1

What is Figure 2-8 attempting to show? Are the points in Figure 2-8 the liquid level at the 1.233 meter (m) axial position in Figure 2-7?

**Westinghouse Response:**

The liquid level measured at the first axial data location, which is 0.157 meters from the test section inlet, is plotted in Figure 2-8 in a three-dimensional figure against test steam and water flow rates; the tests considered are those presented in Figure 2-7. Figure 2-8 shows a good correlation among these parameters, leading to a conclusion that the test section liquid level is determined primarily by the steam and water flow rates.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

**WCAP Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 440.160

***Question:***

WCAP-15833

Please provide Figures 2-9, 2-11, and 2-13 that include a key, or clearly indicate what the individual curves represents. In addition, explain why there are periodic oscillations in the steam flow rate (see dashed curves in Figure 2-13) although the pressure differentials along the channel remain relatively constant (see Figure 2-11).

**Westinghouse Response:**

Figure 2-9 of WCAP-15833 presents the calculated liquid levels for the individual gaps in the WCOBRA/TRAC nodal network. Gaps connect adjacent channels within a section of a WCOBRA/TRAC VESSEL component model at each vertical cell; Figure 2-6 of WCAP-15833 indicates the gap numbering scheme for the Lim test facility as **G2**, **G4**, etc. An expanded version of the WCAP-15833 plot with each curve identified is attached as Figure 440.160-1. Figure 2-11 presents the calculated steam pressure of [ ]$^{a,c}$ individual channels in the WCOBRA/TRAC nodal network. An expanded version of this WCAP-15833 plot is provided as Figure 440.160-2, wherein each curve is identified individually. Figure 2-13 presents the calculated steam flow rate [

]$^{a,c}$

Figure 440.160-3 exhibits periodic oscillations in the steam flow rate [

]$^{a,c}$ the results are very stable at all gap locations, as shown in the attached Figure 440.160-4. The variation in the [

]$^{a,c}$

**Design Control Document (DCD) Revision:** None

**PRA Revision:** None

**WCAP Revision:** None

a,c

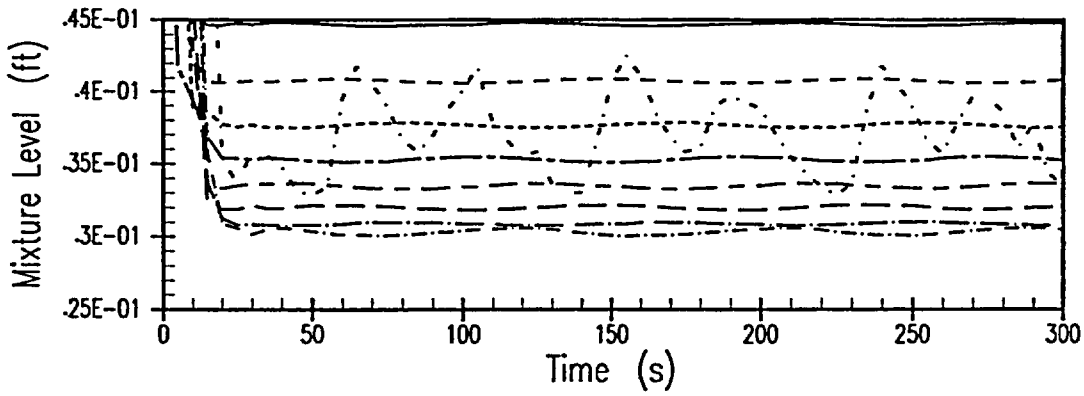



FIGURE 440.160-1: CALCULATED LIQUID LEVEL

[ ] a,c

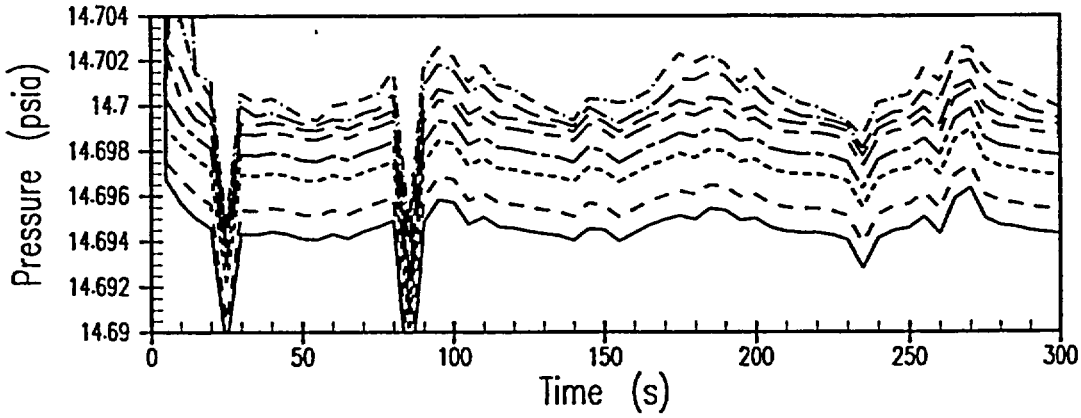



FIGURE 440.160-2: CALCULATED PRESSURE

a,c

[ ]

FIGURE 440.160-3:

Calculated Steam Flowrate (Run 275)

a,c

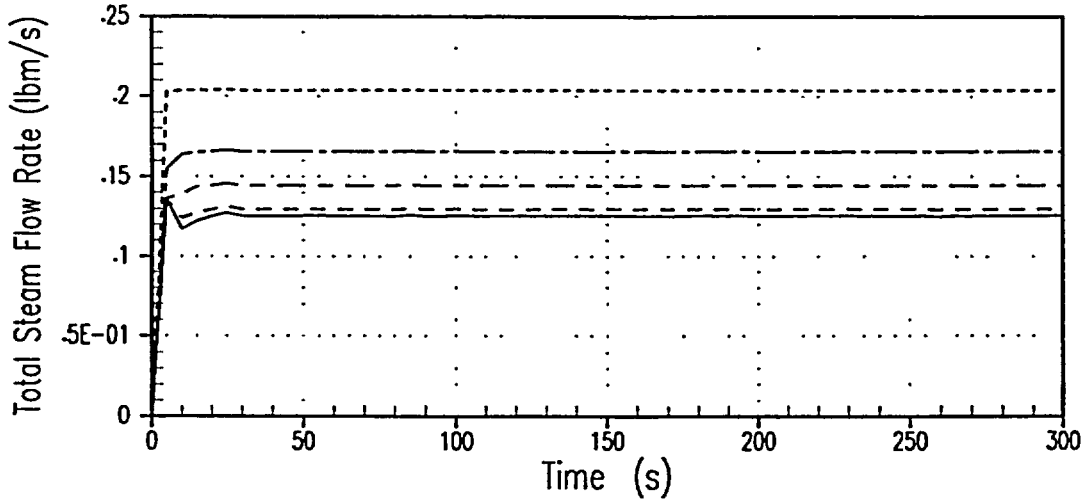



FIGURE 440.160-4: TOTAL GAP STEAM FLOW RATE

RAI Number. 440.164

***Question:***

Figure 3-16 shows the vessel mass inventory for the DEDVI Break, and Figure 3-25 shows the vessel mass inventory for the Inadvertent ADS Actuation scenario as predicted by WCOBRA/TRAC. Since there is less mass in the vessel, there is the possibility that there is less mass in the core and void fractions at the top of the active core may be high. In previous NOTRUMP calculations, the void fraction at the top core was found to exceed 0.90. Therefore, please provide figures showing the axial void gradient in the AP1000 preditions at the time of minimum vessel inventory. Also provide void fraction variation with time for several hydraulic cells in Channel 10, including the top cell.

**Westinghouse Response:**

Figure 440.164-1 presents the void fraction variation of the top three cells of Channel 10, cells 3, 4, and 5, at the [ ]$^{a,c}$ during the DEDVI break ADS-4 IRWST initiation phase transient. The time of IRWST injection is 595 seconds. Figure 440.164-2 presents the void fraction variation of the top three cells of Channel 10, cells 3, 4, and 5, for the [ ]$^{a,c}$ during the inadvertent ADS actuation scenario ADS-4 IRWST initiation phase transient. The time of IRWST injection is 1890 seconds.

Figure 440.164-3 provides the axial void profile in the [ ]$^{a,c}$ at the time of the DEDVI break minimum vessel inventory in the WCOBRA/TRAC simulation. Figure 440.164-4 provides the axial void profile in the [ ]$^{a,c}$ at the time of the inadvertent ADS actuation scenario minimum vessel inventory in the WCOBRA/TRAC simulation.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

Figure 440.164-1: Core Void Profile, DEDVI Break Case

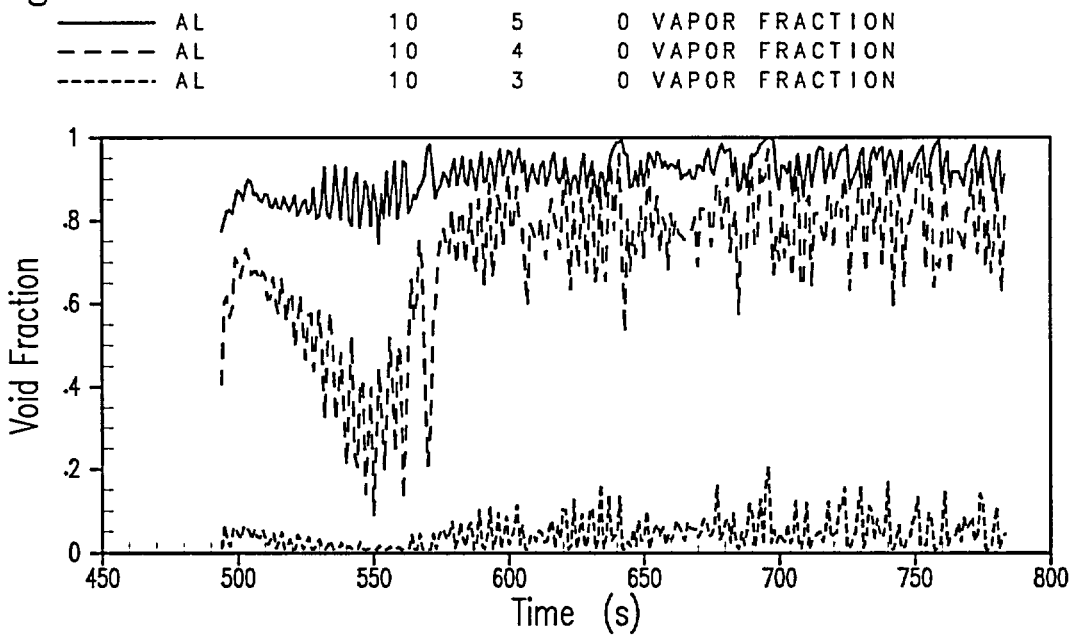
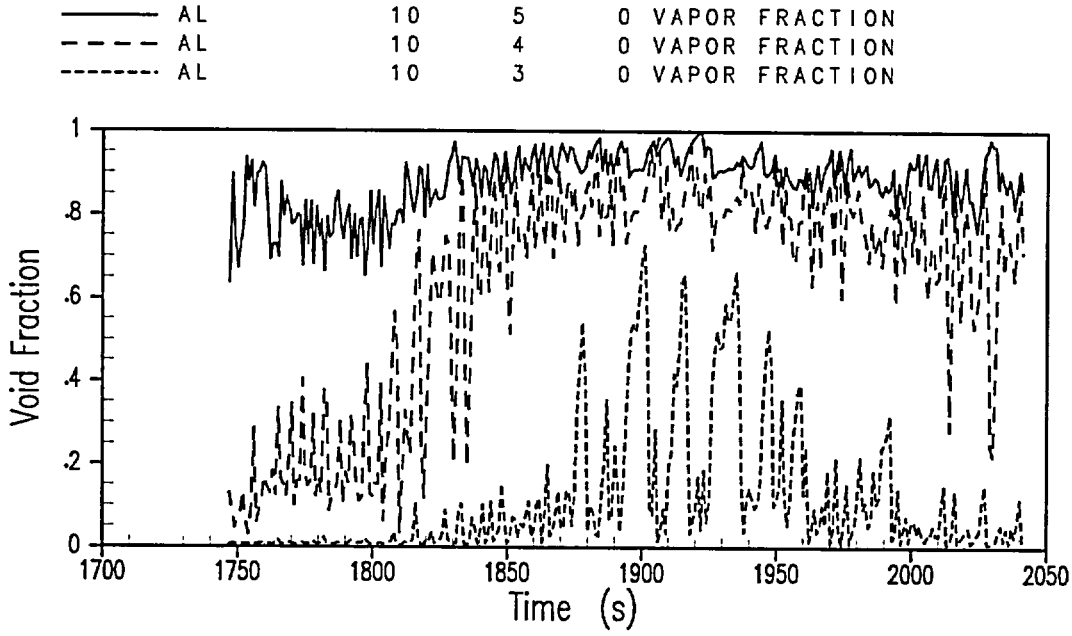


Figure 440.164–2: Core Void Profile, Inadvertent ADS Case



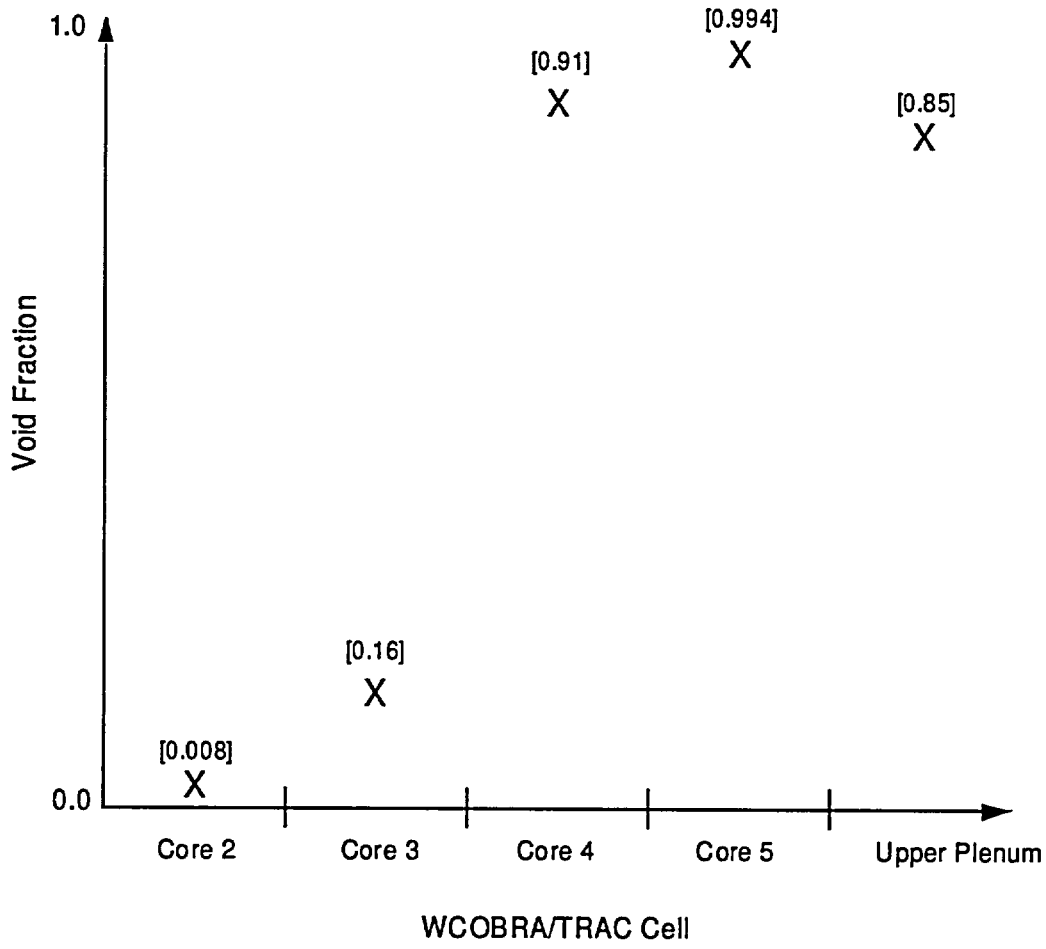



Figure 440.164-3: Axial Void Fraction Gradient at time of Minimum Reactor Vessel Inventory, DEVI Break

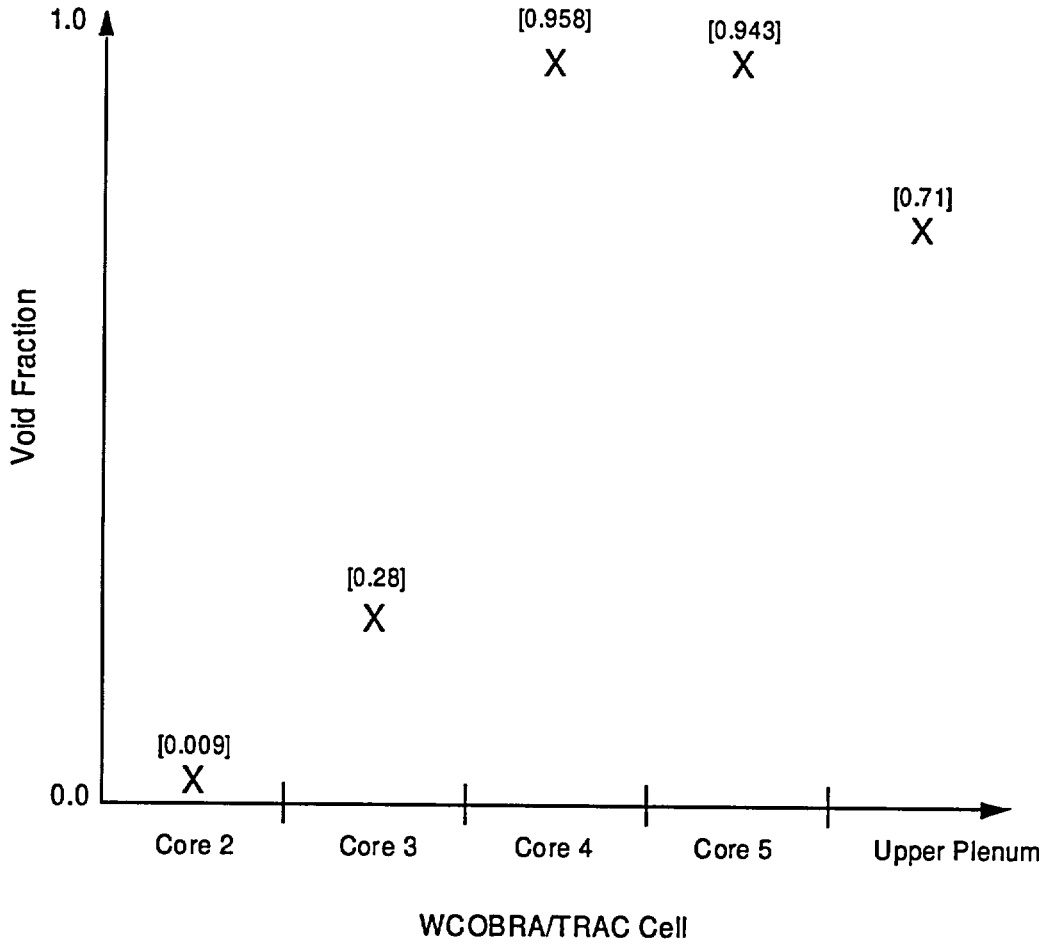



Figure 440.164-4: Axial Void Fraction Gradient at time of Minimum Reactor Vessel Inventory, Inadvertent ADS Case

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 440.169

***Question:***

Section A.3 describes the Kataoka-Ishii model for pool entrainment, which made use of data from several small-scale facilities. Provide justification that the Kataoka-Ishii model is appropriate for full-scale reactor upper plenum geometries.

**Westinghouse Response:**

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

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

**WCAP Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 440.170

***Question:***

Section A.3.2 discusses calculation of upper plenum pool entrainment and presents the results for two WCOBRA/TRAC calculations. Please provide the WCOBRA/TRAC model (nodalization and input, or appropriate description). Indicate if the AP1000 upper plenum (including guide tubes, lateral connections, heat slabs, etc.) was used, or if a simple stand alone model was used. Do the entrainment rate results represent an axial flow to the hot leg level, or do they represent the net flow from the upper plenum to the hot legs?

**Westinghouse Response:**

The WCOBRA/TRAC VESSEL component nodalization presented in Figure 3-1 of WCAP-15833 was used in the calculations reported in Section A.3.2, together with the one-dimensional component network of Figure 3-8. Within the upper plenum, [

] [a,c].

The AP1000 upper plenum modeling / nodalization described in the preceding paragraph was used for the calculation reported in Section A.3.2. The entrainment rate results shown in Section A.3.2 are the sum total of the entrainment generation in [ ][a,c] Channel 15 and in [ ][a,c] Channel 20 within the WCOBRA/TRAC upper plenum nodalization.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number 440.171

***Question:***

Figures A.3-4 and A.3-5 show a prediction of WCOBRA/TRAC upper plenum entrainment versus time for conditions simulating a DVI line break. Please provide the WCOBRA/TRAC collapsed liquid level for this simulation.

**Westinghouse Response:**

The collapsed liquid level in the upper plenum from the WCOBRA/TRAC DEDVI line break simulation presented in Appendix A.3 is shown in Figure 440.171-1. The level shown is relative to the top elevation of the active fuel length.

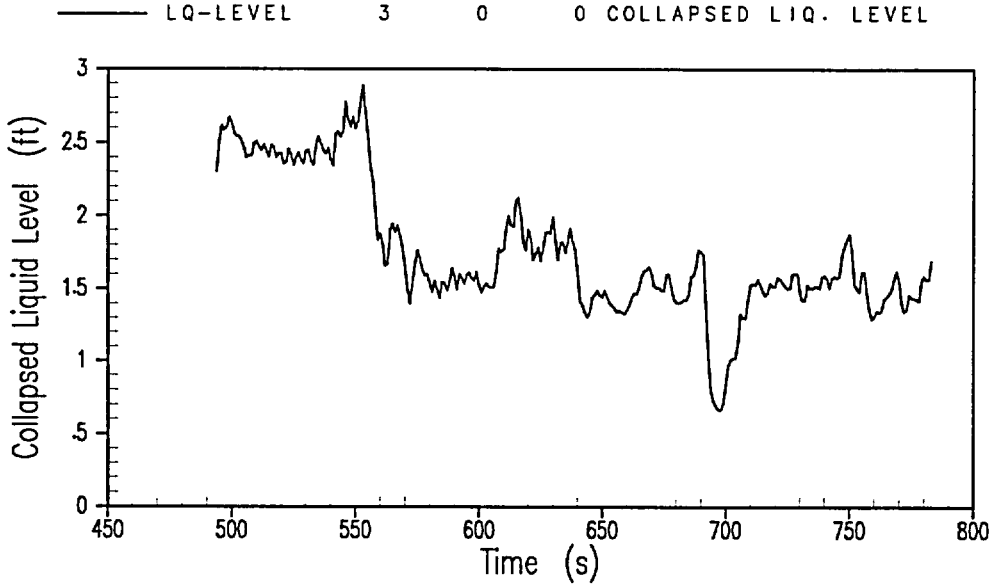
**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

Figure 440.171–1: Upper Plenum Level, DEDVI Break Case



# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number 440.172

***Question:***

Figures A.3-9 and A.3-10 show a prediction of WCOBRA/TRAC upper plenum entrainment versus time for conditions simulating an Inadvertent ADS Actuation. Please provide the WCOBRA/TRAC collapsed liquid level for this simulation. In addition, please provide an explanation for the "spikes" that reduce the entrainment rate to near K-I Region 3 rates even though the steam velocity remains greater than 10 feet-per-second (ft/sec).

**Westinghouse Response:**

The collapsed liquid level in the upper plenum from the WCOBRA/TRAC Inadvertent ADS actuation scenario presented in Appendix A.3 is shown in Figure 440.172-1. The level shown is relative to the top elevation of the active fuel length.

As stated, there are moments during this transient when the predicted entrainment rate decreases to a small value. The predicted entrainment rates shown in Section A.3.2 are the sum total of the entrainment generation in [the single cell of][a,c] Channel 15 and in [the bottom cell of][a,c] Channel 20 within the WCOBRA/TRAC model of the AP1000 upper plenum. The momentary low entrainment rate prediction at a significant steam velocity is the result of two different effects that occur in the two channels. In Channel 20, the cell void fraction at that moment is very close to 1.00, so that there is almost no liquid present to be entrained. Within Channel 15, the net liquid flow at the core / upper plenum boundary becomes downward, the liquid flowing down into the core instead of upward. Although both of these conditions are transitory, each causes the entrainment prediction in the respective channel cells to be minimal at a particular point in time. When the two effects occur coincidentally, the predicted entrainment rate is reduced to near K-I Region 3 rates.

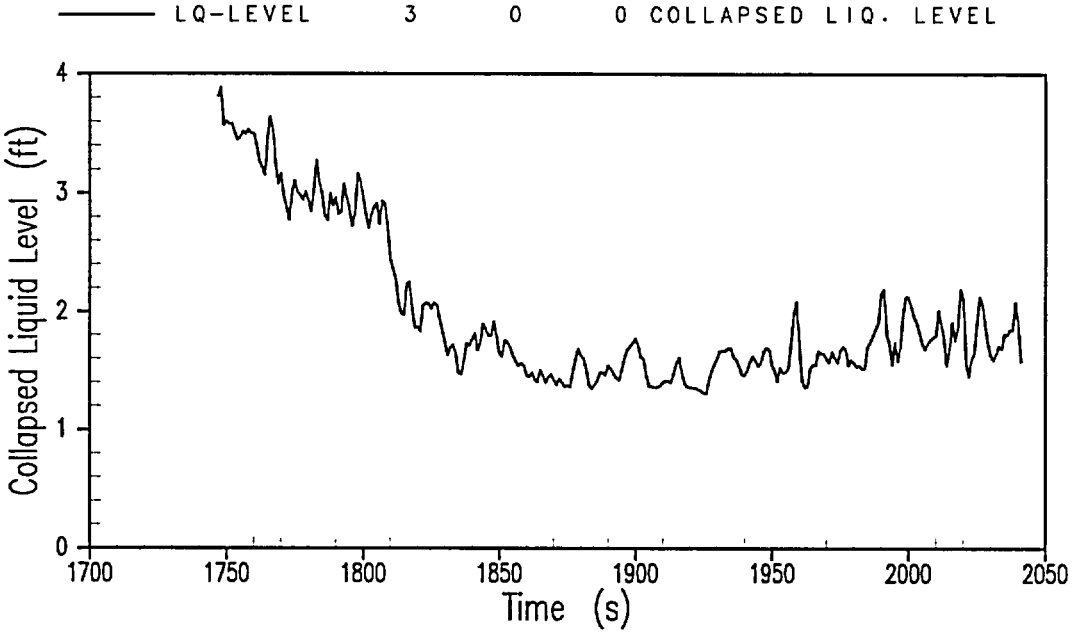
**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

Figure 440.172–1: Upper Plenum Level, Inadvertent ADS Case



# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 480.002

***Question:***

(DCD Section 6.2.3, Containment Isolation System") This section makes no mention of RG 1.141, "Containment Isolation Provisions For Fluid Systems," dated April 1978, which endorses, with exceptions, ANSI [American National Standard Institute] N271-1976/ANS-56.2, "Containment Isolation Provisions For Fluid Systems," dated June 28, 1976.

Provide a discussion of the AP1000 design's conformance to the provisions of this RG.

**Westinghouse Response:**

DCD Appendix 1A Conformance with Regulatory Guides provides information regarding the conformance of the AP1000 design to Reg. Guide 1.141. This conformance assessment can be found on page 1A-51.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 480.003

***Question:***

DCD Section 6.2.1.1.3, "Design Evaluation," states that the passive internal heat sink data used in the WGOTHIC analyses is presented in DCD Reference 20, Section 4. Section 4 of DCD Reference 20 describes the AP600. Section 13 of DCD Reference 20 describes the AP1000. Please revise the DCD to identify the correct section for the description of the AP1000 WGOTHIC model.

**Westinghouse Response:**

The AP1000 DCD will be revised as recommended.

**Design Control Document (DCD) Revision:**

DCD Section 6.2.1.1.3 (9th paragraph) will be revised as shown:

The passive internal containment heat sink data used in the WGOTHIC analyses is presented in Reference 20, Section 4-13. Data for both metallic and concrete heat sinks are presented. The containment pressure and temperature responses to a double-ended cold leg guillotine are presented in Figures 6.2.1.1-5 and 6.2.1.1-6 for the 24 hour portion of the transient and Figures 6.2.1.1-7 and 6.2.1.1-8 for the 72 hour transient. The containment pressure and temperature response to a double-ended hot leg guillotine break are presented in Figures 6.2.1.1-9 and 6.2.1.1-10. The physical properties of the materials corresponding to the heat sink information is presented in Table 6.2.1.1-8.

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 480.005

***Question:***

Section 6.2.1.2.3, "Design Evaluation," indicates that DCD Reference 26 contains the details for the subcompartment pipe break analyses. DCD Reference 26 does not have a document identifier or a document date. By its name, it implies that no changes to the subcompartment models were made to evaluate the AP1000. Please provide a copy of the reference for staff review.

**Westinghouse Response:**

DCD Section 6.2 Reference 26 will be revised to reflect the AP1000-specific reference containing the AP1000 subcompartment models. The subcompartment models were revised to reflect the AP1000 design.

**Design Control Document (DCD) Revision:**

DCD Section 6.2.7 will be revised as follows.

**6.2.7 References**

26. ~~Applicability of AP600 Subcompartment Evaluation Models to AP1000~~**WCAP-15965-P (Proprietary) and WCAP-15965-NP (Non-Proprietary), "AP1000 Subcompartment Models," November 2002.**

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 480.006

***Question:***

Please identify the break size (and location) for DCD Table 6.2.1.3-4. Is this the 1.0 square foot ($ft^2$) break identified in DCD Section 6.2.1.2.3.2, "Pipe Breaks"? Please revise the DCD to include this information in the table. DCD Table 6.2.1.3-4 also is not referenced in DCD Table 6.2.1.2-1. Please revise the DCD appropriately.

**Westinghouse Response:**

The mass and energy release information provided in Table 6.2.1.3-4 is for 1-square foot break of the main steam line. The table heading will be revised to clarify this.

This information is not referenced in the subcompartment pressurization analyses presented in Table 6.2.1.2-1 because it does not represent the limiting mass and energy release for a break inside containment. This mass and energy release information is used in the analysis of subcompartments outside containment where the break of a main steam line is limiting.

See the response to RAI 220.015 for a related discussion of the subcompartment pressurization analysis of rooms outside containment.

**Design Control Document (DCD) Revision:**

Table 6.2.1.3-4

MAIN STEAM LINE BREAK MASS AND ENERGY (1 $FT^2$ BREAK)

| Time (sec) | Mass (lbm/sec) | Energy (Btu/sec) |
|---|---|---|

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 480.007

***Question:***

Please provide the data presented in DCD Tables 6.2.1.3-9 and 6.2.1.4-2 in an electronic format for use by the staff to perform independent confirmatory analyses. Also, please provide the data presented in Figure 7-2, "Minimum Delivered PCS Water Flow Rate," of WCAP-15846, "WGOTHIC Application to AP1000," for the AP1000 in an electronic format for use by the staff to perform independent confirmatory analyses.

**Westinghouse Response:**

The AP1000 DCD was submitted to the NRC in electronic format. The following table contains the information that is plotted in Figure 7-2 of WCAP-15846.

Minimum Delivered PCS Water Flow Rates
(Data used in Figure 7-2 of WCAP-15846)

| AP1000 | | AP600 | |
|---|---|---|---|
| Time (sec) | Flow (GPM) | Time (sec) | Flow (GPM) |
| 0 | 0 | 0 | 0 |
| 35 | 0 | 35 | 0 |
| 36 | 469.1 | 36 | 442.9 |
| 13670 | 451.7 | 10800 | 425.9 |
| 13680 | 226.6 | 10810 | 123.1 |
| 39590 | 218.4 | 108000 | 110.2 |
| 39600 | 176.3 | 108010 | 72.1 |
| 84590 | 166.7 | 259200 | 62.8 |
| 84600 | 144.2 | | |
| 284390 | 100.7 | | |

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 480.009

***Question:***

There are several discrepancies (as cited below) in the DCD regarding containment purge isolation valve closure time.

- DCD Tier 2, Table 14.3-7 (Sheet 1 of 3), "Radiological Analysis," indicates that the maximum closure time for remotely-operated containment purge valves is ≤ 5 seconds.

- DCD Tier 2, Table 6.2.3-1 (Sheet 3 of 4), "Containment Mechanical Penetrations and Isolation Valves," indicates that the closure time for the containment purge valves (VFS-PL-V003, -V004, -V009, and -V010) is 20 seconds.

- DCD Tier 1, Table 2.2.1-3, "Inspections, Tests, Analyses, and Acceptance Criteria," Item 7 indicates that the containment purge isolation valves (VFS-PL-V003, -V004, -V009, and -V010) close within 10 seconds.

- DCD Tier 2, Section 6.2.1.5, "Minimum Containment Pressure Analysis for Performance Capability Studies of Emergency Core Cooling System" (PWR), sub-section 6.2.1.5., indicates that, to help minimize containment pressure, the containment purge supply and exhaust lines were modeled to close 12 seconds after the 8 psig closure setpoint was reached (apparently based on a 10 sec closure time, increased by a factor of 1.2).

Please clarify this issue and correct the apparent discrepancies in the DCD.

**Westinghouse Response:**

The containment purge isolation valve closure time is ≤ 10 seconds. DCD Tier 2 Tables 14.3-7 and 6.2.3-1 will be revised accordingly.

### Design Control Document (DCD) Revision:

DCD Table 6.2.3-1 will be revised as follows (note only selected columns from Table 6.2.3-1 are shown in this response):

| System | Line | Valve/Hatch Identification | Closure Time |
|---|---|---|---|
| VFS | Cont. air filter supply | VFS-PL-V003<br>VFS-PL-V004 | ~~20~~ 10 sec<br>~~20~~ 10 sec |
| | Cont. air filter exhaust | VFS-PL-V010<br>VFS-PL-V009<br>VFS-PL-V008 | ~~20~~ 10 sec<br>~~20~~ 10 sec<br>N/A |

DCD Table 14.3-7 will be revised as follows:

Table 14.3-7 (Sheet 1 of 3)

**RADIOLOGICAL ANALYSIS**

| Reference | | Design Feature | Value |
|---|---|---|---|
| Table | 6. 2. 3-1 | Maximum closure time for remotely operated containment purge valves (seconds) | ≤ ~~5~~ 10 |

### PRA Revision:

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 640.001

***Question:***

Please reconcile the apparent inconsistency between (1) the minimum value of 755,100 gallons stated in DCD Table 6.2.2-1, for PCCWST useable capacity for the PCS (gal) and (2) the acceptance criteria of "greater than 755,000" gallons stated in DCD ITAAC Table 2.2.2-3, Item 7.(f)ii.

**Westinghouse Response:**

The DCD Tier 2 material will be revised as shown.

**Design Control Document (DCD) Revision:**

Table 6.2.2-1

PASSIVE CONTAINMENT COOLING SYSTEM PERFORMANCE PARAMETERS

| | |
|---|---|
| PCCWST[1] initial inventory duration - Minimum | 72 hours |
| PCCWST useable capacity for PCS (gal) - Minimum | 755,~~1~~000 |

**PRA Revision:**

None

RAI Number· 720.006

*Question:*

The MAAP4 analyses to justify the success paths described in Sections A.3.2.1 and A.3.3.1 are for hot leg break locations. This is because analyses in WCAP-14869 indicate that hot leg breaks are more limiting than cold leg breaks of the same break size. The hot legs are at a lower elevation relative to the reactor core than the cold legs and would be expected to cause more water to be lost through a postulated break. Analyses in Chapter 15 of the Design Certification Document (DCD) indicate that cold leg break sizes are more limiting. This is because the cold leg breaks analyzed in the DCD are located close to the pressure balance line of core makeup tank (CMT) causing delayed CMT injection. Please provide analyses demonstrating that the success paths of the PRA are still valid for the break locations analyzed in Chapter 15 of the DCD.

**Westinghouse Response:**

As stated for AP600 in WCAP-14869 (MAAP4/NOTRUMP Benchmarking): "The hot leg break results in a lower vessel inventory when ADS is actuated, causing the system response to be slightly more limiting than a cold leg break. The effect of the break location is an interesting thermal-hydraulic issue, but does not impact the PRA success criteria definitions. The plant response to different break locations is similar enough to define the same set of equipment needed to achieve successful core cooling."

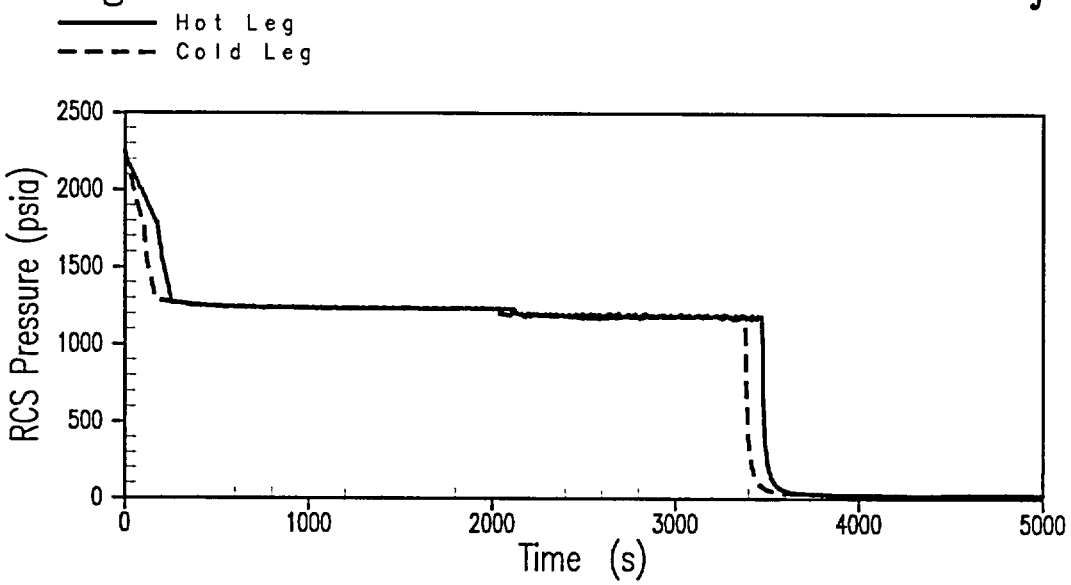
This statement remains true for AP1000. As an example, a 2-inch cold leg break was analyzed with MAAP4 for comparison to the 2-inch hot leg break case presented in Figures A.3.2-8 to A.3.2-13 of the AP1000 PRA. The figures on the following pages compare the cold leg break response to the hot leg break response for the same system parameters shown in the PRA: RCS pressure, CMT water mass, integrated break flow, integrated stage 4 ADS vapor release, integrated IRWST water injection, and core mixture level. The water in the cold leg is more dense than the water in the hot leg, and therefore there is initially more water mass lost from the cold leg break when compared to the hot leg break. However, the higher elevation of the cold leg causes the break location to uncover sooner and transition to vapor-only break flow, resulting in less integrated coolant mass through the break. With more coolant remaining in the RCS, the core uncovery is slightly less limiting for the cold leg break than the hot leg break. This is the reason that the hot leg break has been used for analyses of limiting PRA multiple-failure accident sequences, although the difference is not large.

The response to RAI 440.098 provides results of the spectrum of small break LOCA analysis results including a 2-inch hot leg break. The AP1000 small break LOCA analysis results are acceptable for all breaks analyzed. Comparisons between the analysis of the 2-inch hot leg

break and the 2-inch cold leg break show similar results. The transients are characterized by no core uncovery for either the 2-inch cold leg or 2-inch hot leg break, with no resultant fuel cladding heatup in either case. The minimum RCS inventory for the 2-inch hot leg break is slightly less than for the 2-inch cold leg breaks, however the results are acceptable for both.

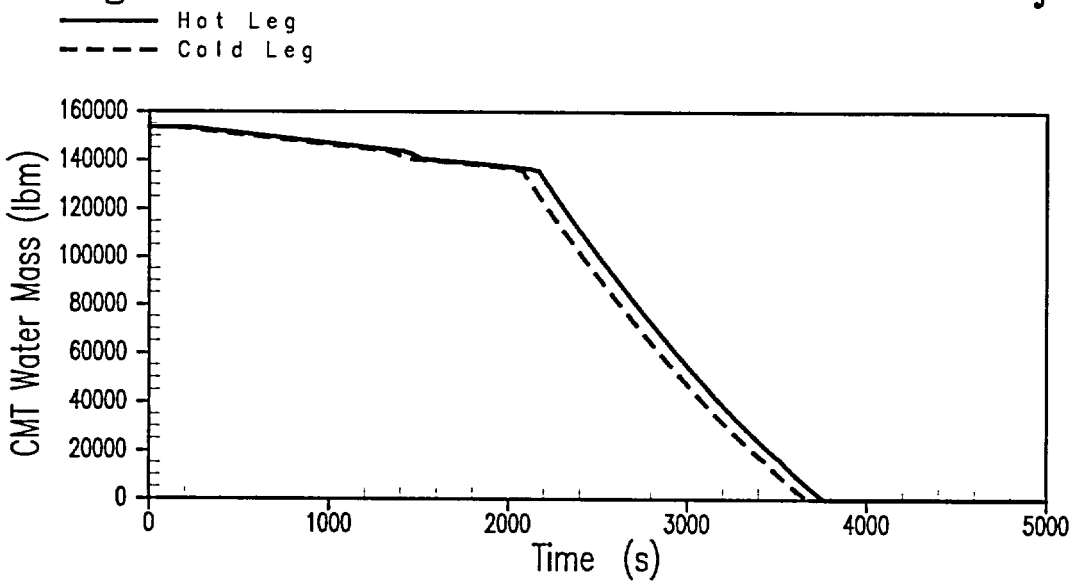
# AP1000 MAAP4 Analysis of 2.0 Inch Break

## 3 stage 4 ADS, 1 CMT, No Accumulators, IRWST Inj



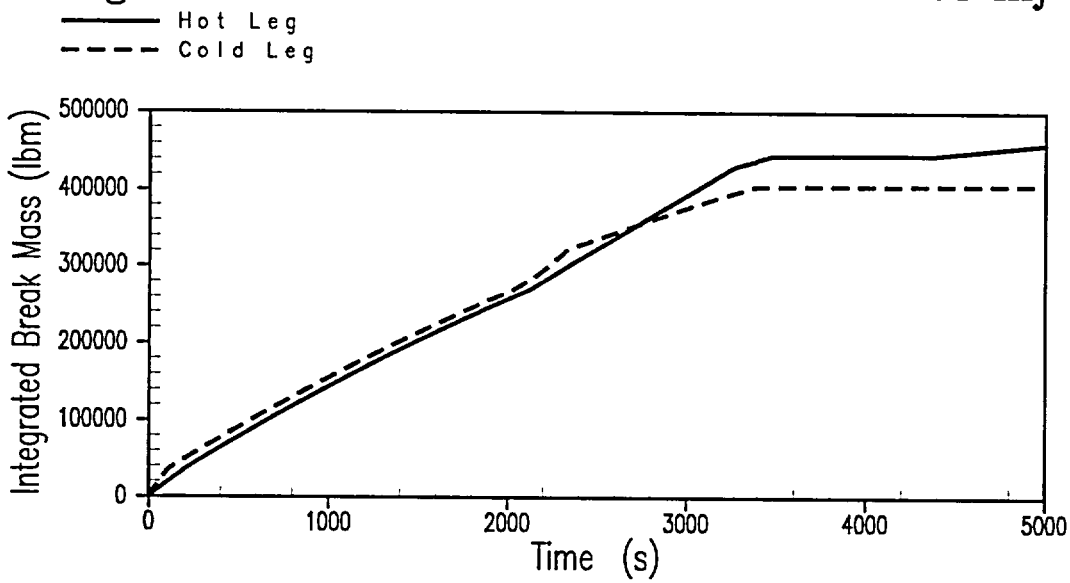
# AP1000 MAAP4 Analysis of 2.0 Inch Break

## 3 stage 4 ADS, 1 CMT, No Accumulators, IRWST Inj



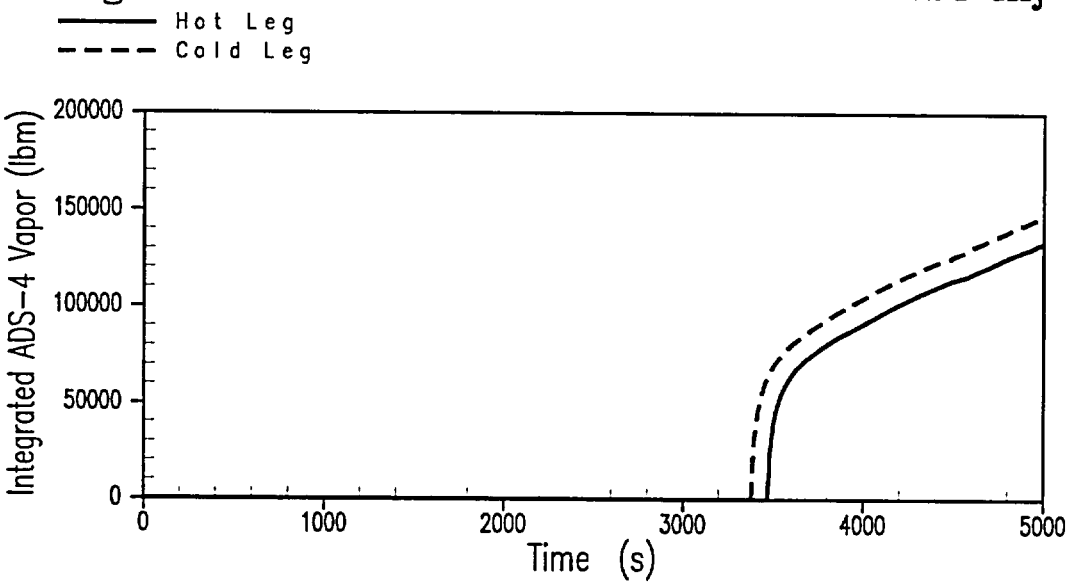
# AP1000 MAAP4 Analysis of 2.0 Inch Break

## 3 stage 4 ADS, 1 CMT, No Accumulators, IRWST Inj



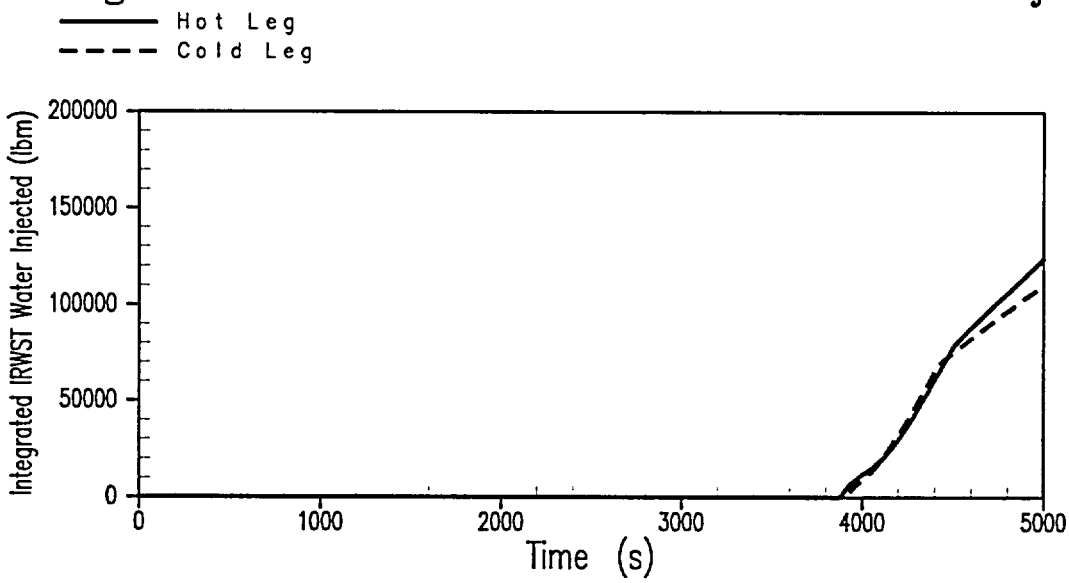
# AP1000 MAAP4 Analysis of 2.0 Inch Break

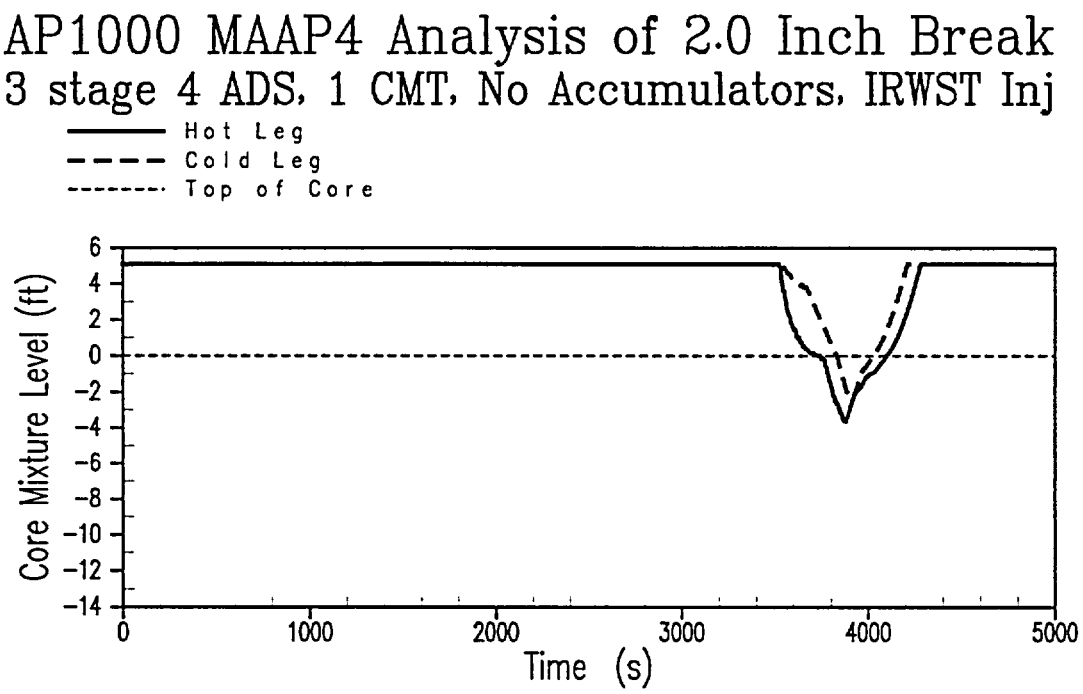
## 3 stage 4 ADS, 1 CMT, No Accumulators, IRWST Inj



# AP1000 MAAP4 Analysis of 2.0 Inch Break

## 3 stage 4 ADS, 1 CMT, No Accumulators, IRWST Inj






**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

RAI Number 720.015

*Question:*

Section A5.2.1.1 states for cases where containment isolation is assumed, a constant containment back pressure of 25 psia is used. This is for the 3.25 inch hot leg break and the double-ended CMT inlet breaks with manual ADS4 actuation.

Provide a containment pressure analyses justifying that this assumption is bounding for the entire duration of the events, in particular for the period between ADS4 actuation and IRWST injection.

**Westinghouse Response:**

The hot leg break case (UC1) and the balance line break case (UC2) assume containment isolation. Containment pressure affects flow out the break and ADS valves, and IRWST injection flow. In general, the lower the assumed back pressure, the lower the IRWST injection flow.

The 25 psia containment back pressure assumed in these analyses is derived from analyzing the containment response to double-ended DVI line break (an equivalent 4 inch pipe break on the cold leg side). The pressure response to this event is shown in Figure 1, and 25 psia is chosen to conservatively bound the range of the transient.

Specific containment analyses were not performed for the two cases (UC1 and UC2) analyzed, and 25 psia was assumed for the containment back pressure for these cases. For UC2 which is a 6.8 inch break on the cold leg side, the 25 psia back pressure is conservatively low. However, for the 3 inch hot leg break (note that a three inch break was found to be more limiting than 3.25 inch), the integrated energy released from the break is expected to be significantly lower than the DEDVI break. This is due to the smaller break size, and the higher upstream enthalpy which results in lower critical mass flux. Thus, it is not conservative to assume 25 psia for the containment back pressure for this case.

To evaluate the effect of containment back pressure on case UC1, a calculation was made with a containment atmosphere pressure of 14.7 psia. Figure 2 shows the IRWST flow for the two cases; with and without containment isolation. For the case with atmosphere containment pressure, the onset of IRWST injection is delayed slightly, and the flow is somewhat lower due to the lower driving head. There is minimal effect on the core mixture level, and the case without containment isolation results in slightly less margin to core uncovery. Thus, the containment back pressure is not an important assumption for this case.

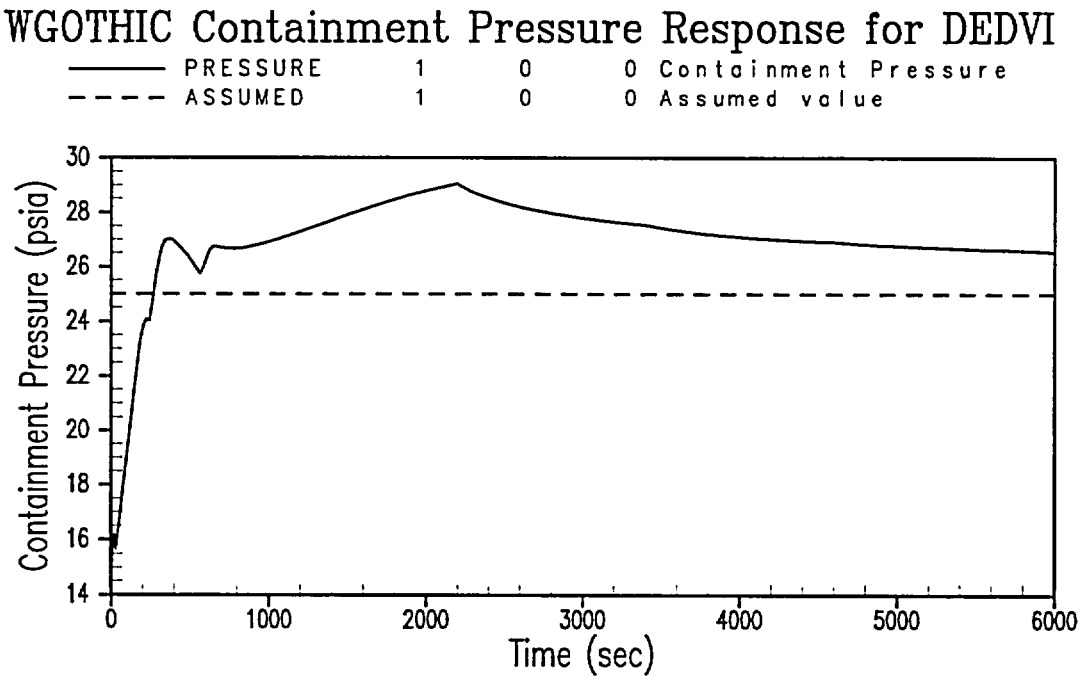



**Figure 1: WGOTHIC Containment Pressure Response for DEDVI Break**

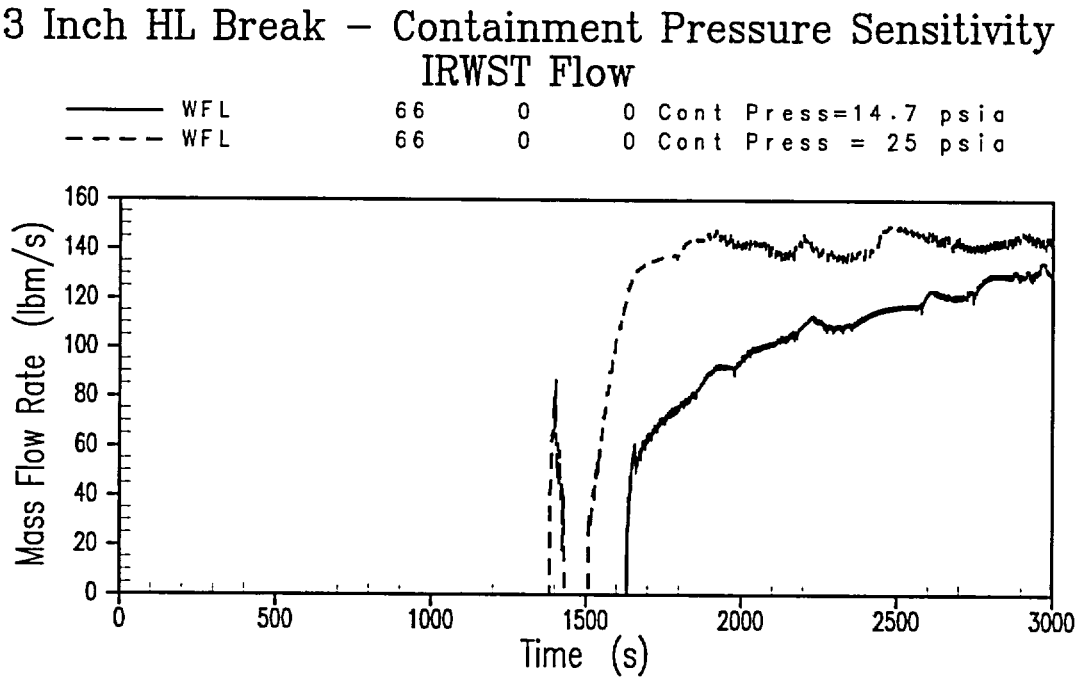



**Figure 2: IRWST Flow vs. Time for 3 Inch Hot Leg Break – Effect of Containment Pressure**

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

The following changes will be included in the next AP1000 PRA Appendix A revision.

### A5.2.1.3.1 Case UC1 Results

Case UC1 is a 3.0~~3.25~~-inch break in the RCS hot leg. **This break size is the maximum that will keep the RCS pressure at or above the accumulator pressure (700 psia) at the time that manual ADS-4 actuation is assumed (1200 seconds after the safety injection signal).** Neither of the 2 CMTs is assumed to operate and, therefore, operator action to actuate the ADS must be assumed. Additional assumptions are:

- Credit for PRHR HX operation
- Credit for only 1 of 2 accumulators
- ADS stages 1, 2, and 3 ADS fail to open
- Credit for 4 out of 4 ADS stage 4 at 20 minutes (1200 seconds) after safety injection (SI) signal
- Only 1 of 2 IRWST lines is assumed available for injection. Further, failure of 1 of the 2 parallel paths in the available IRWST line is assumed
- ~~Credit for containment isolation: e~~ Containment pressure is ~~was originally~~ assumed to be 14.7 ~~25~~ psia~~which was calculated for the DEDVI break~~. **Containment pressures greater than 14.7 psia have been shown to improve the performance of the passive safety systems.** ~~Subsequent evaluations showed that the containment pressure for a 3-inch hot leg break will be significantly lower.~~ **Consequently, the containment pressure for this case is conservatively assumed to be 14.7 psia.**

Figures A5.2-1 through A5.2-13 provide plots of the plant response and Table A5.2-2 provides the sequence of key events. Figures A5.2-3 and A5.2-4 show the liquid and steam break flow rates that lead to depressurization of the RCS, as seen in Figure A5.2-1, and draining of the RCS pressurizer (Figure A5.2-2). The ~~3.25~~3.0-inch break size was selected as the largest break size that together with the PRHR HX would not result in significant accumulator injection before the operator opens the ADS stage 4 valves at 20 minutes. Figure A5.2-1 shows the RCS pressure to be slightly below the accumulator cut-in pressure of 715 psia at 20 minutes. At 20 minutes, the operator opens all 4 ADS stage 4 valves, which results in rapid depressurization down to less than 50 psia. The accumulator injects as a result of the depressurization, refilling the RCS downcomer, and recovering the core. The single accumulator runs dry at about 1430 seconds and the IRWST begins to inject. Core uncovery occurs before operator action to open the ADS stage 4 valves, followed by a rapid recovery of the core due to injection of the single accumulator. A minimum RCS mass of 60,000 lbm occurs shortly after 1200 seconds (Figure A5.2-12), the time of maximum core uncovery.

A cladding heatup calculation for case UC1 (Figure A5.2-13) shows a peak cladding temperature of 719°F at 13.75 feet on the fuel rod occurring at 1238 seconds. These results are well below the 2200°F acceptance criterion.

### A5.2.1.3.2 Case UC2 Results

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

- Credit for PRHR HX operation
- Credit for 2 out of 2 accumulators
- ADS stages 1, 2, and 3 fail to open
- Credit for 4 out of 4 ADS stage 4 at 20 minutes (1200 seconds)
- Only 1 of 2 IRWST lines is assumed to inject. Further, failure of 1 of the 2 parallel paths in the IRWST line to open is assumed
- Credit for containment isolation; containment pressure assumed to be 25 psia which was calculated for the DEDVI break. **Since the DE CMT balance line is a larger break, the 25 psia containment pressure is conservative for this case.**

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

Table A5.2-1

**AP1000 T/H UNCERTAINTY ANALYSIS CASES**

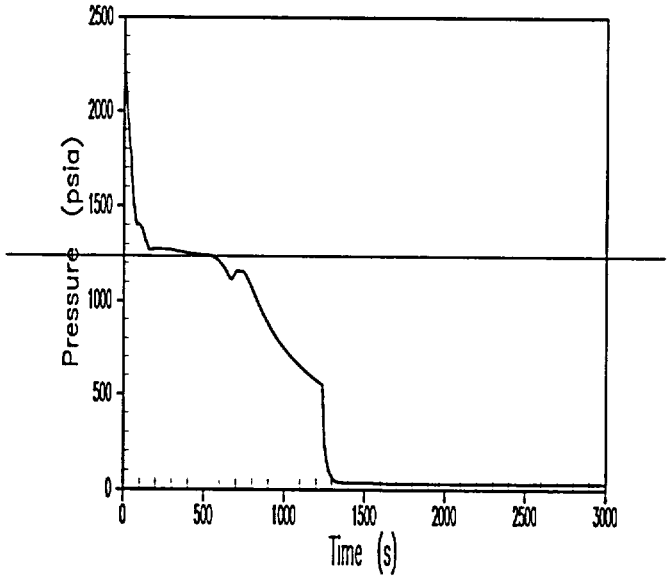
| | Cont Isol | ADS 1/2/3 | ADS 4 | PRHR HX | CMT | Acc | IRWST Val/Path | Recirc Val/Path |
|---|---|---|---|---|---|---|---|---|
| **Small LOCAs** | | | | | | | | |
| 1. ~~3.25~~3.0" HL | yes | 0 | 4 | 1 | 0 | 1 | 1/1 | na |
| 2. DE CMT inlet | yes | 0 | 4 | 1 | 0 | 2 | 1/1 | na |
| 3. DE DVI | no | 0 | 3 | 1? | 1 | 0 | 1/1 | na |
| **Large LOCAs** | | | | | | | | |
| none | | | | | | | | |
| **Long Term Cooling** | | | | | | | | |
| none | | | | | | | | |

Table A5.2-2

**UC1 SEQUENCE OF EVENTS (~~3.25~~3.0-INCH HOT-LEG BREAK)**

| Event | Time (sec) |
|---|---|
| Break Opens | 0.0 |
| Reactor Trip Signal | 39 |
| "S" Signal | 47 |
| Steam Turbine Stop Valves Close | 48 |
| Main Feed Isolation Valves Begin to Close | 48 |
| Reactor Coolant Pumps Start Coastdown | 63 |
| Top of Core Uncovers | 1150 |
| ADS Stage 4 Opens | 1247 |
| Accumulator Injection Starts | 1200 |
| Top of Core Recovers | 1250 |
| IRWST Injection Starts | 1500 |
| Accumulator Empties | 1430 |

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

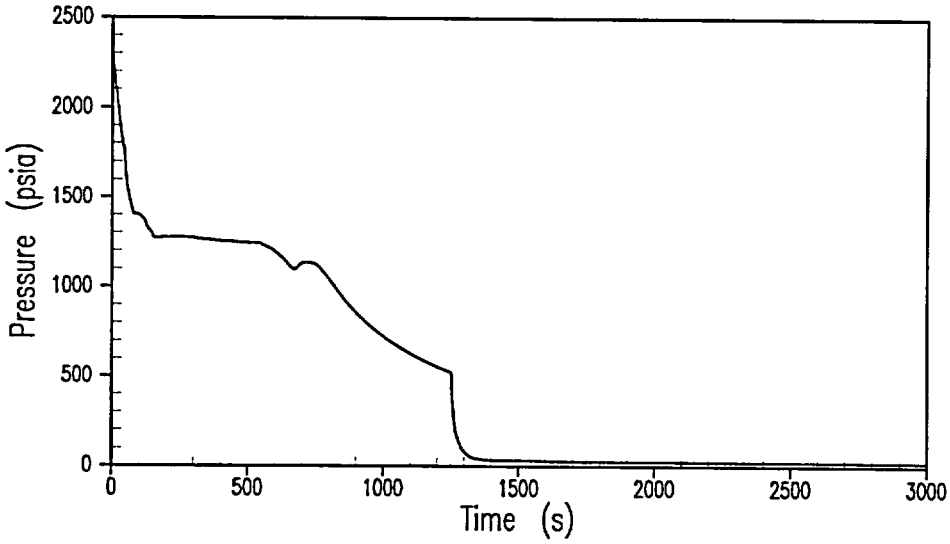
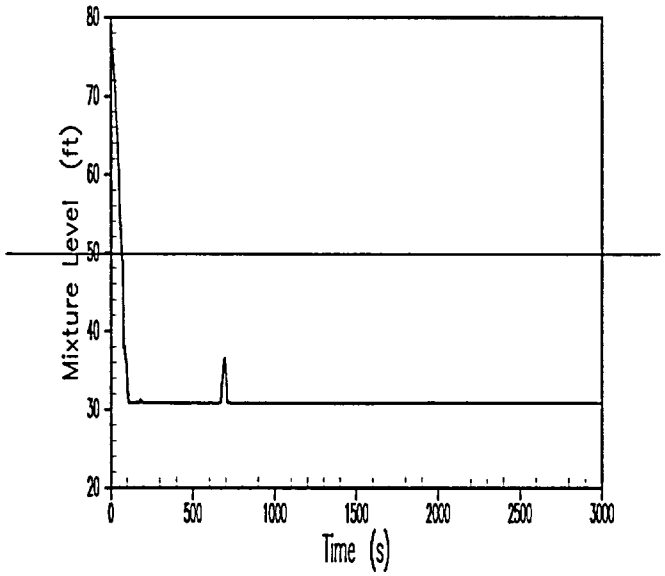



Figure A5.2-1
**Case UC1 Pressurizer Pressure**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

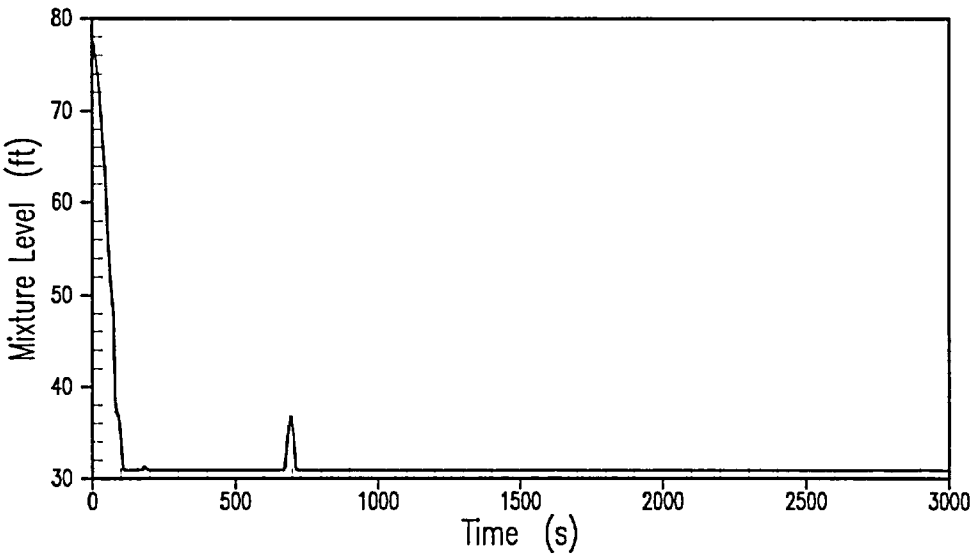
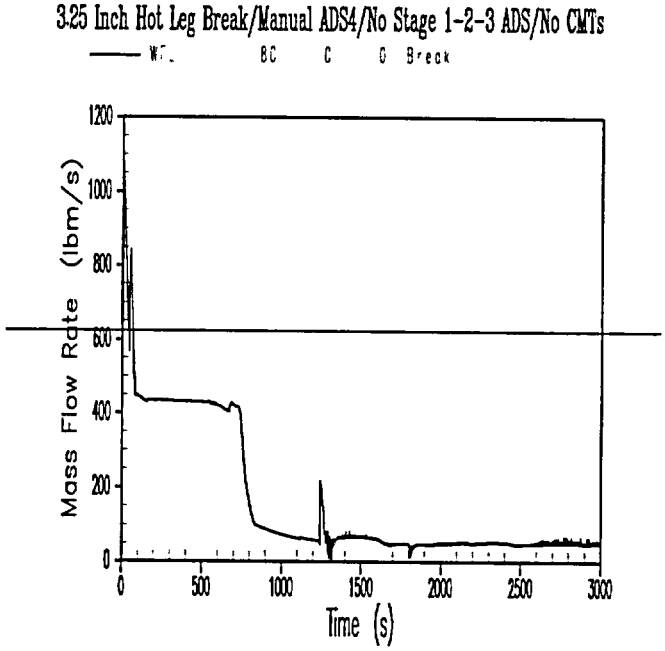



Figure A5.2-2
**Case UC1 Pressurizer Level**




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

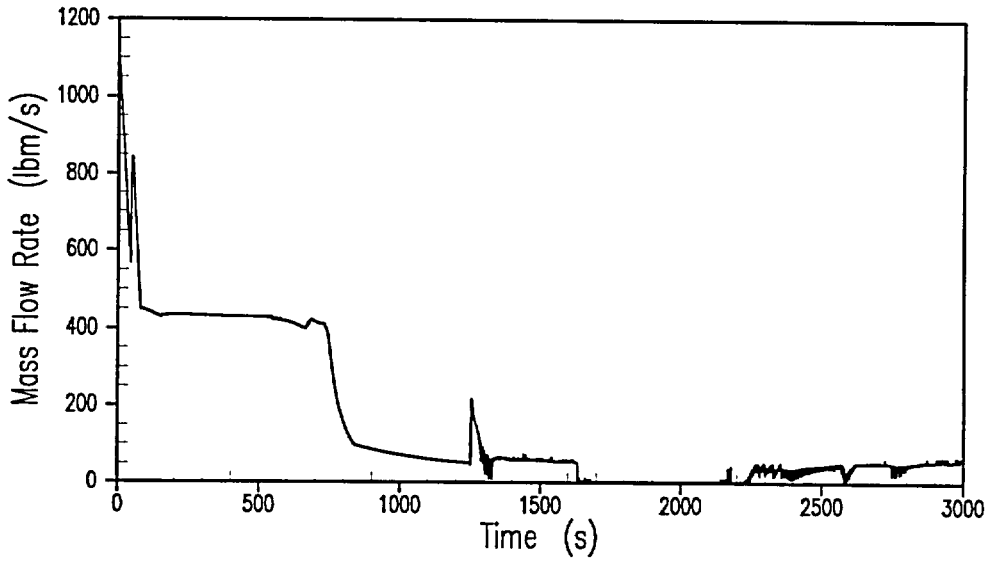
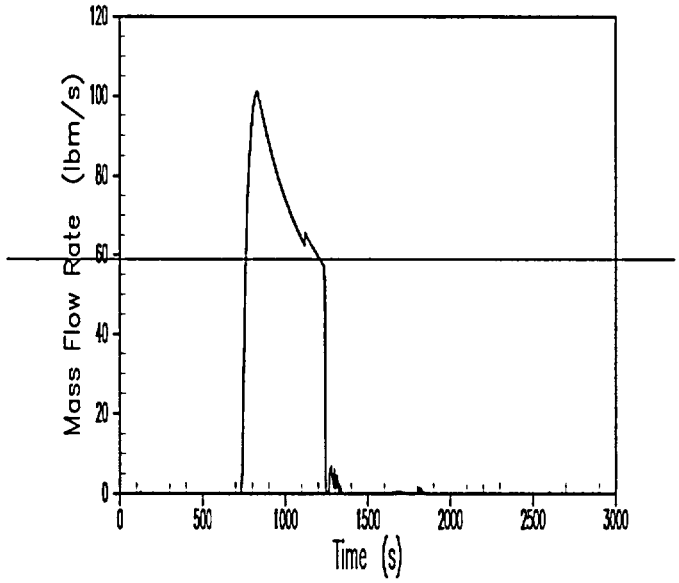



Figure A5.2-3

**Case UC1 Break Liquid Flow**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

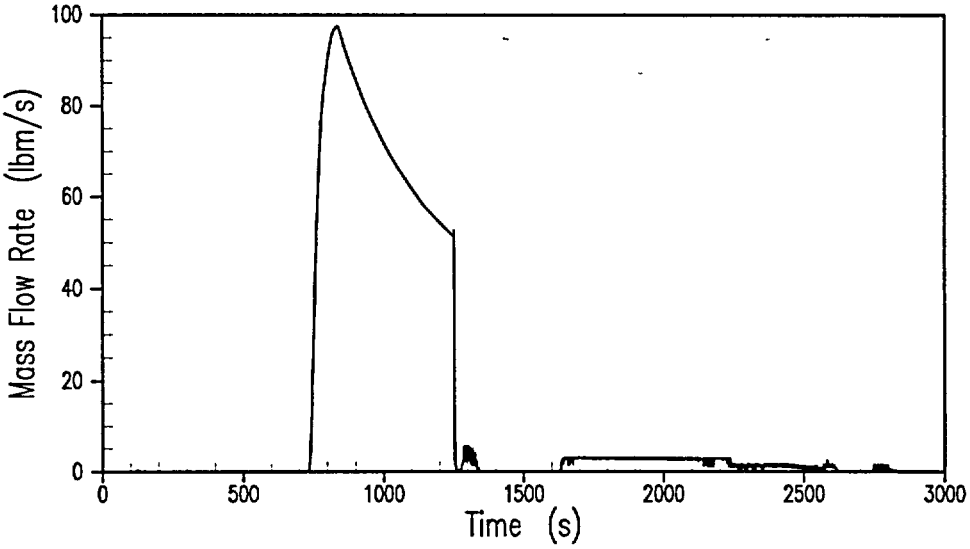
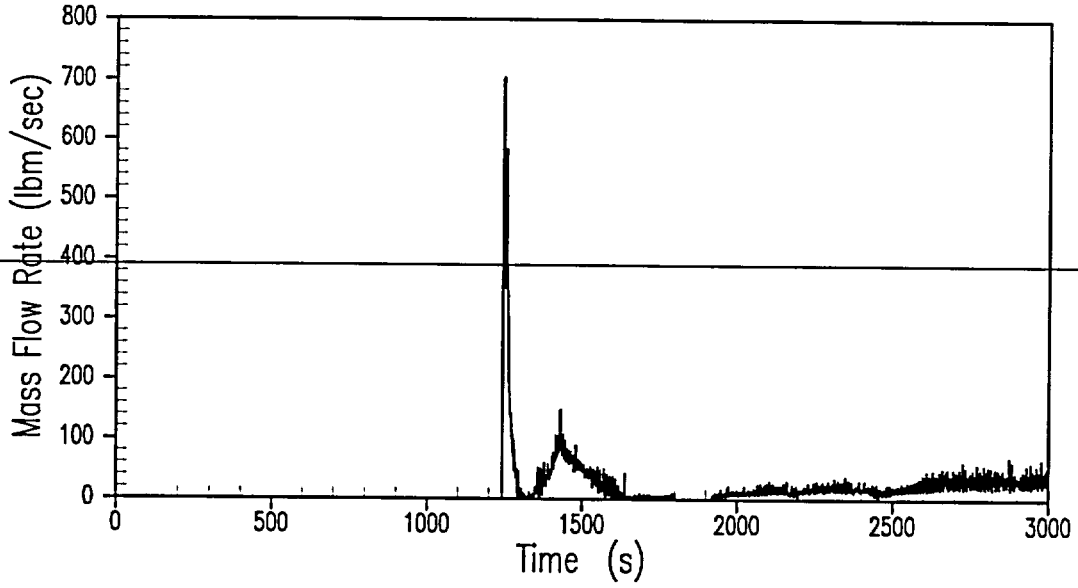



Figure A5.2-4

**Case UC1 Break Vapor Flow**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

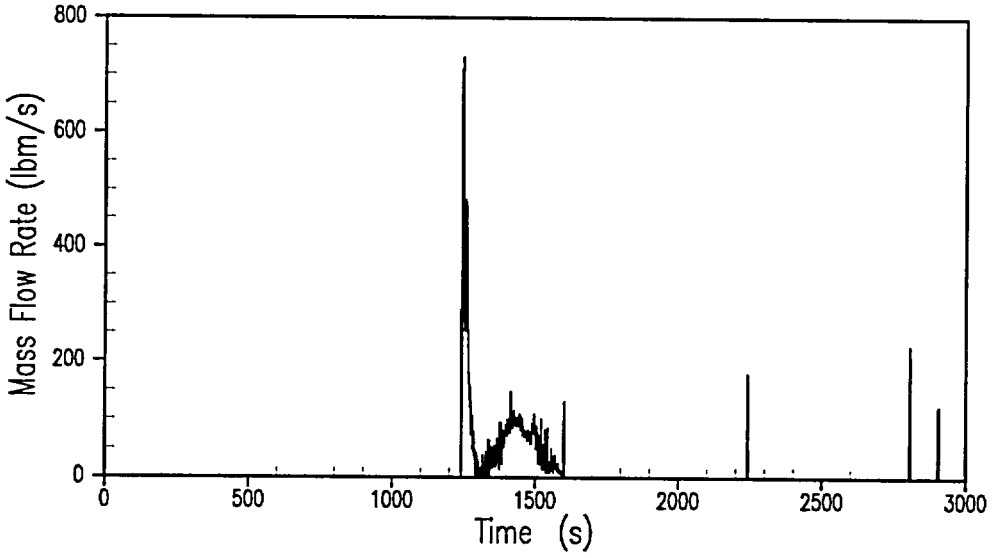
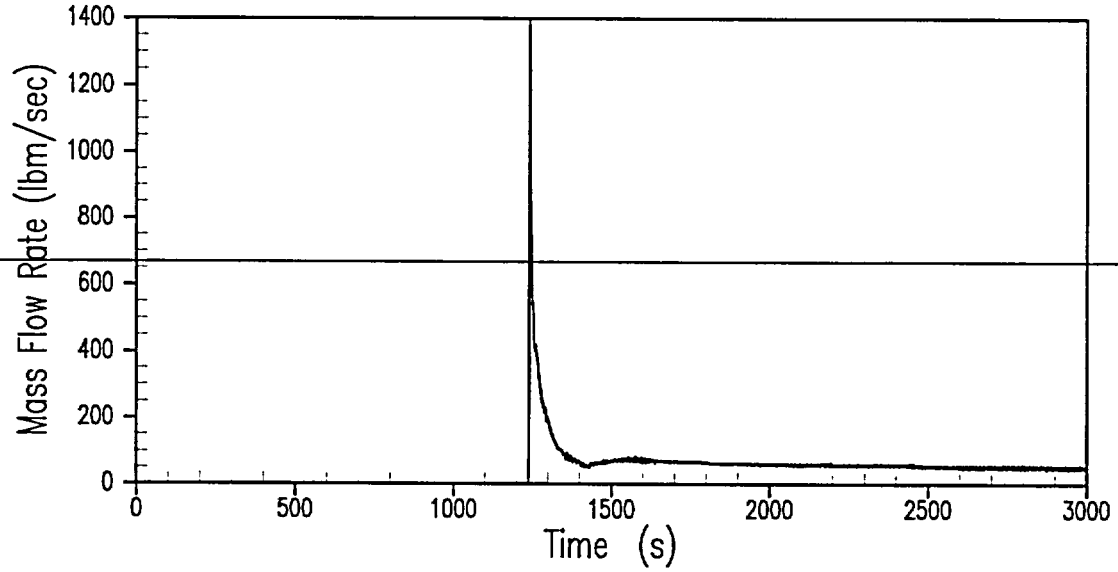



Figure A5.2-5

**Case UC1 4th Stage ADS Liquid Flow Through All Open Paths**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1–2–3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

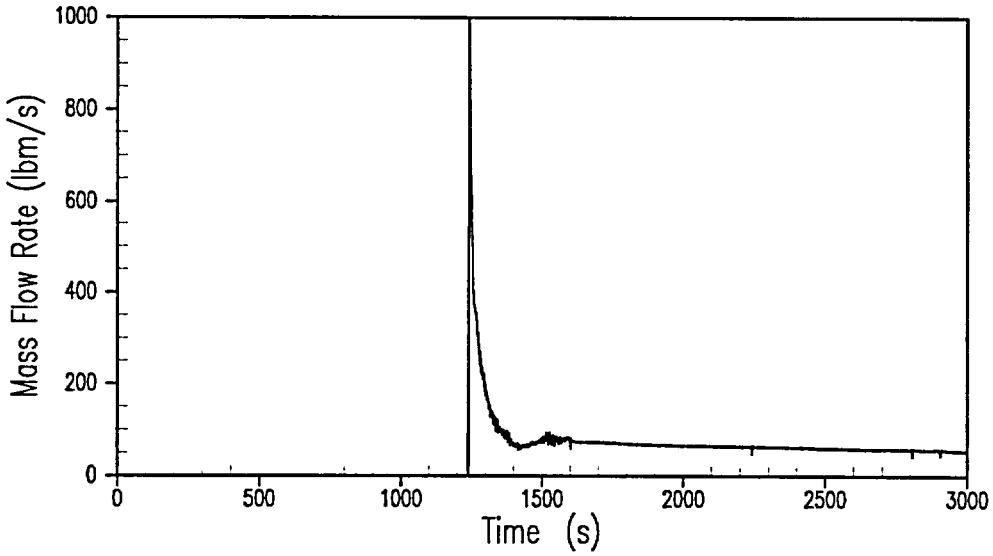
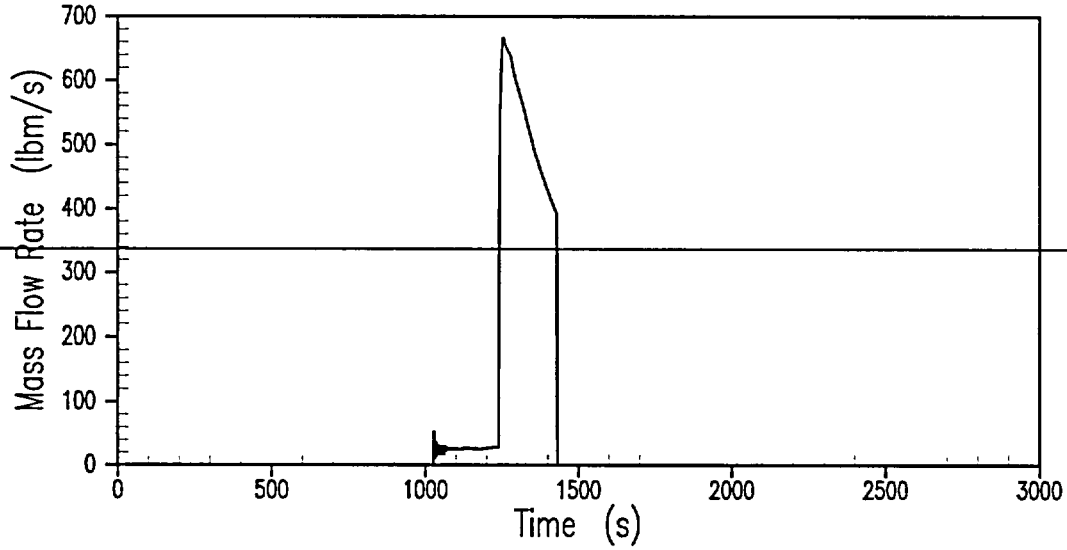



Figure A5.2-6

**Case UC1 4th Stage ADS Vapor Flow**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

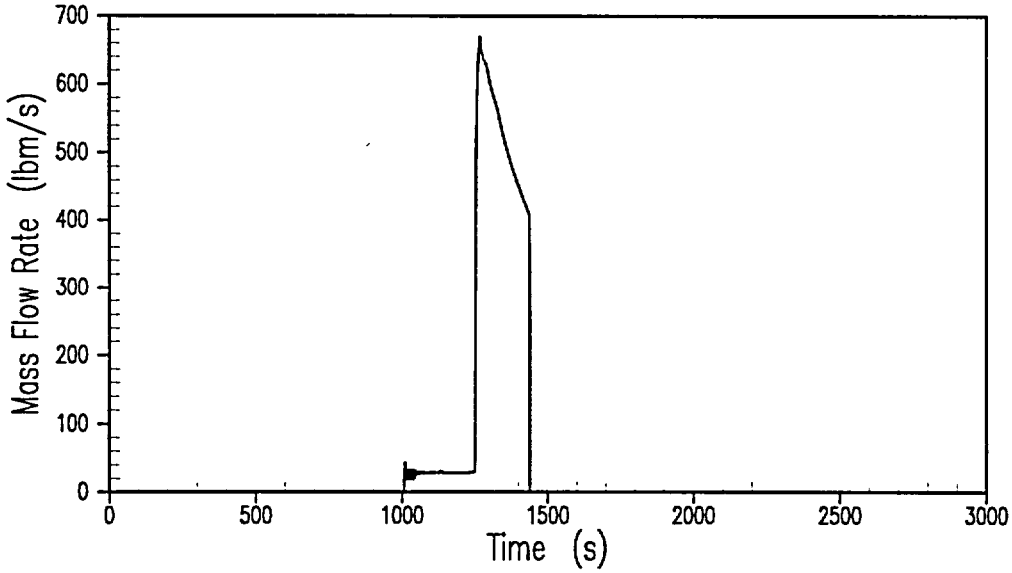
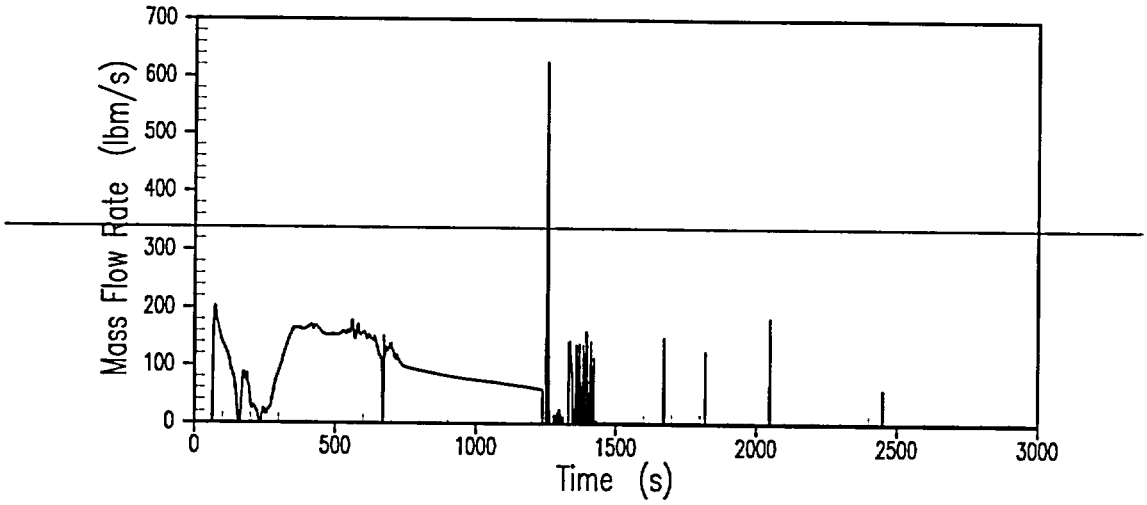



Figure A5.2-7

**Case UC1 Accumulator Injection Flow**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

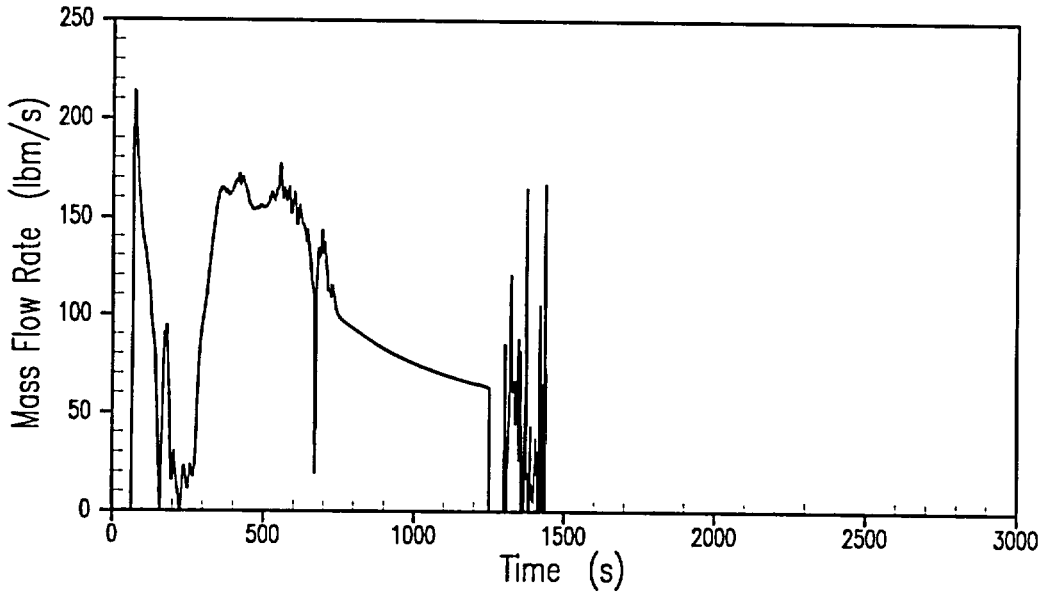
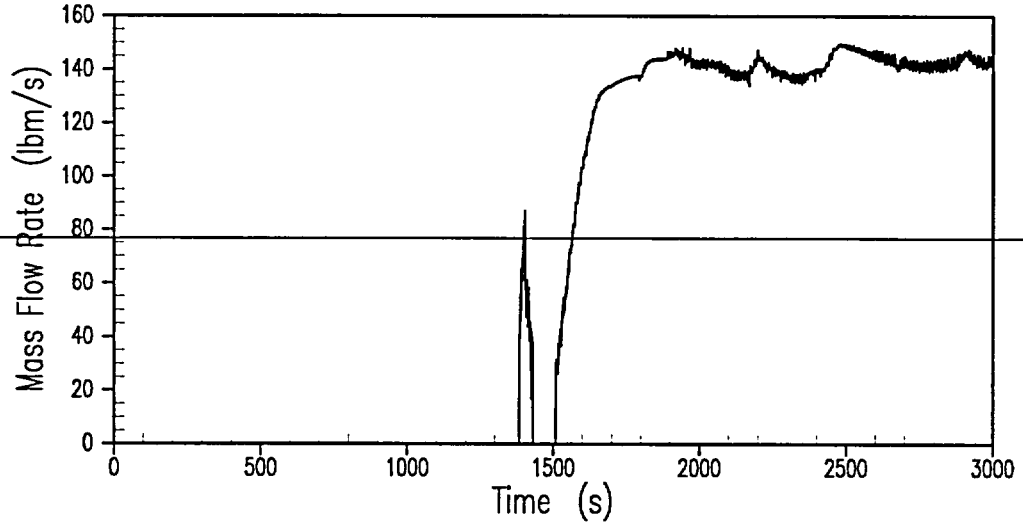



Figure A5.2-8

**Case UC1 PRHR Discharge Flow**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

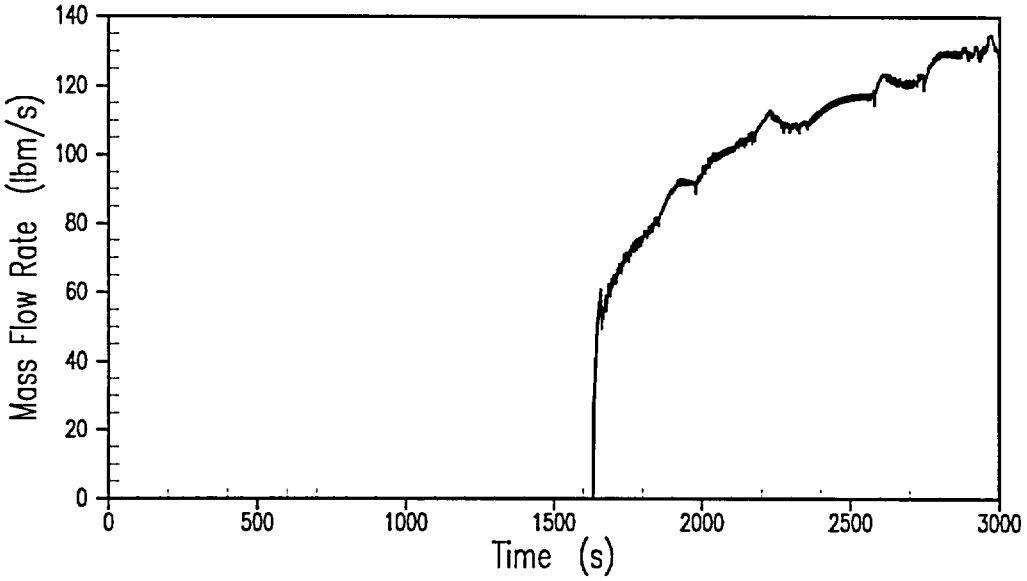
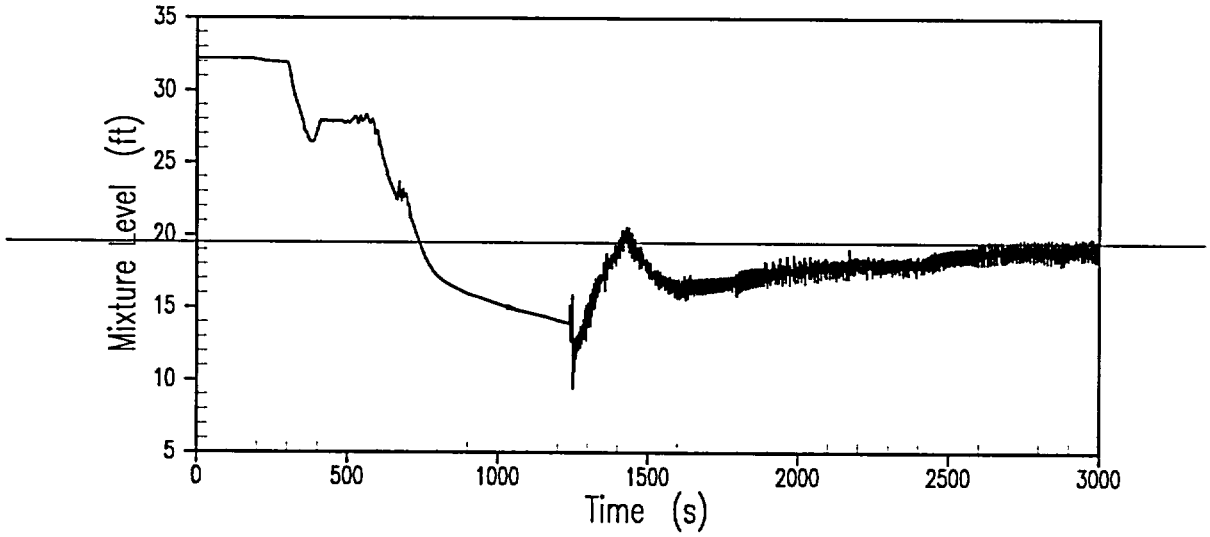



Figure A5.2-9

**Case UC1 IRWST Injection Flow**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

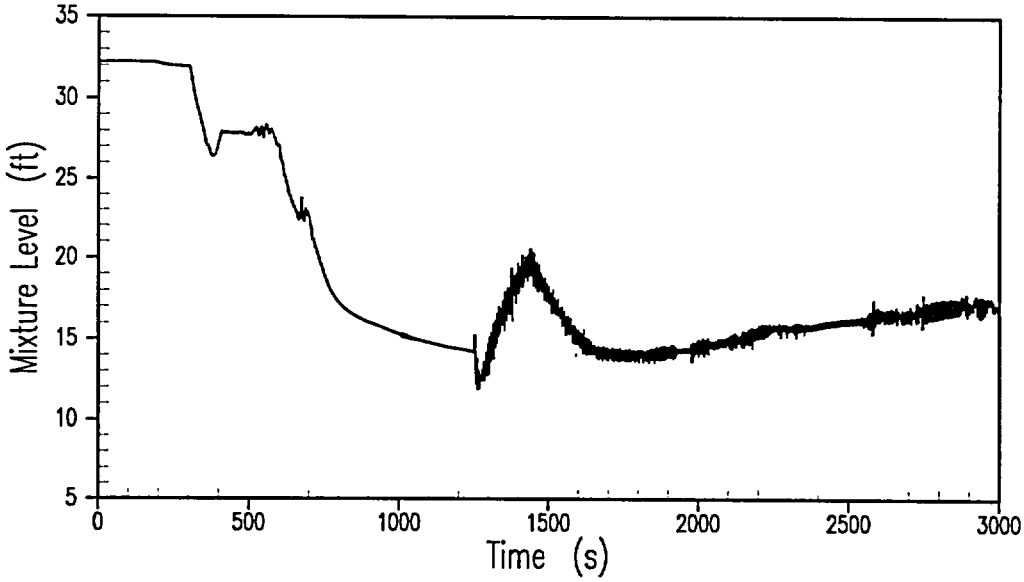
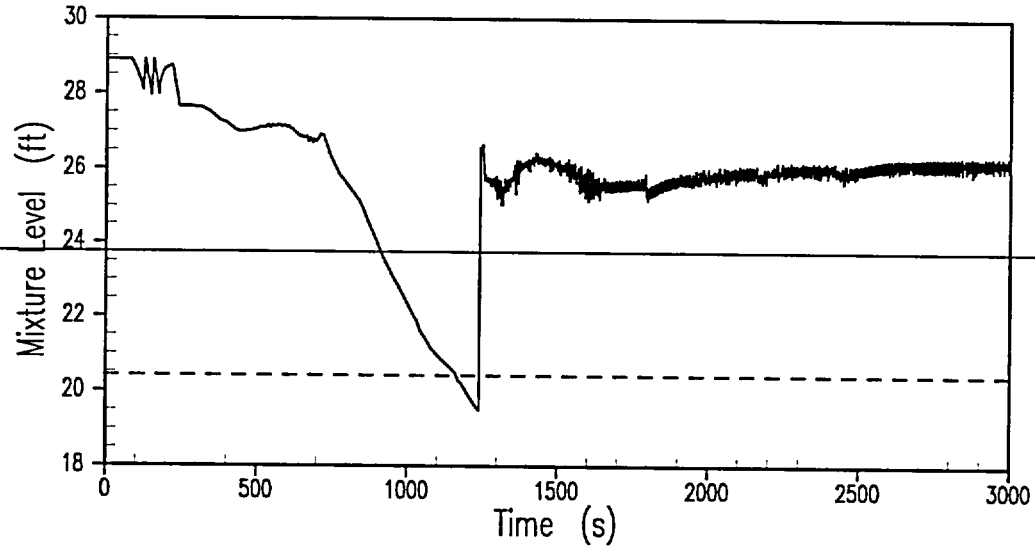



Figure A5.2-10

**Case UC1 Downcomer Mixture Level**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1–2–3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

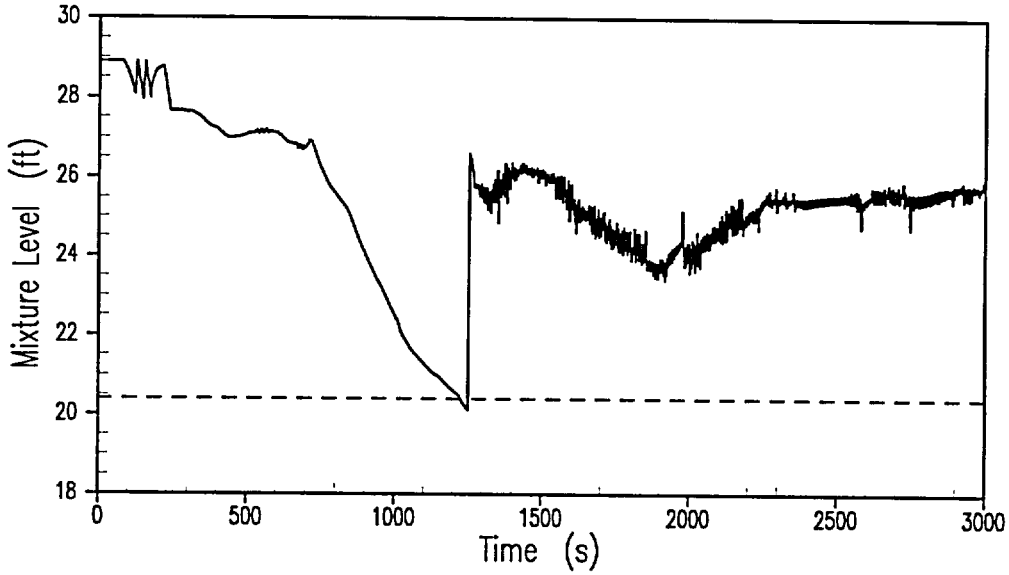
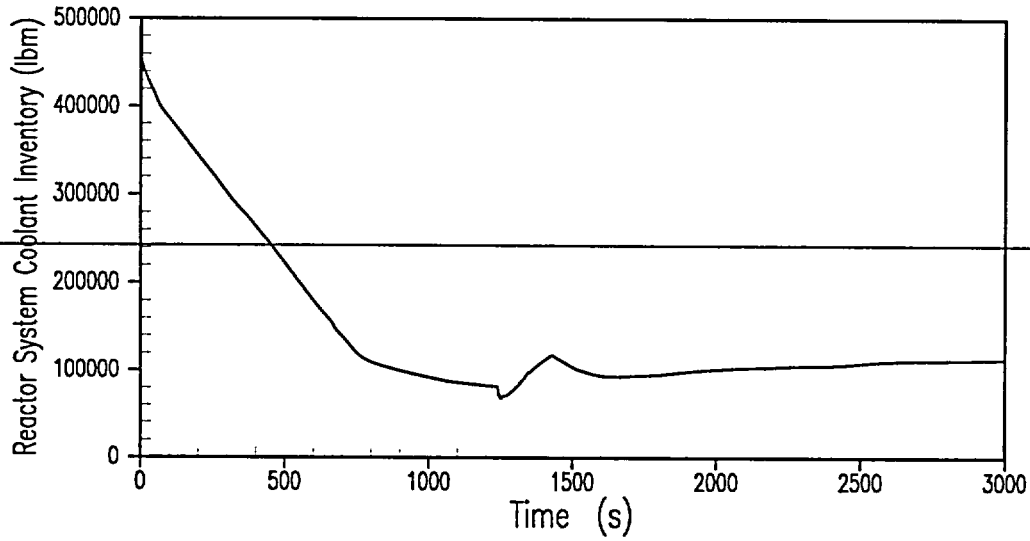



Figure A5.2-11

**Case UC1 Upper Plenum and Core Mixture Level**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1-2-3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 ACC

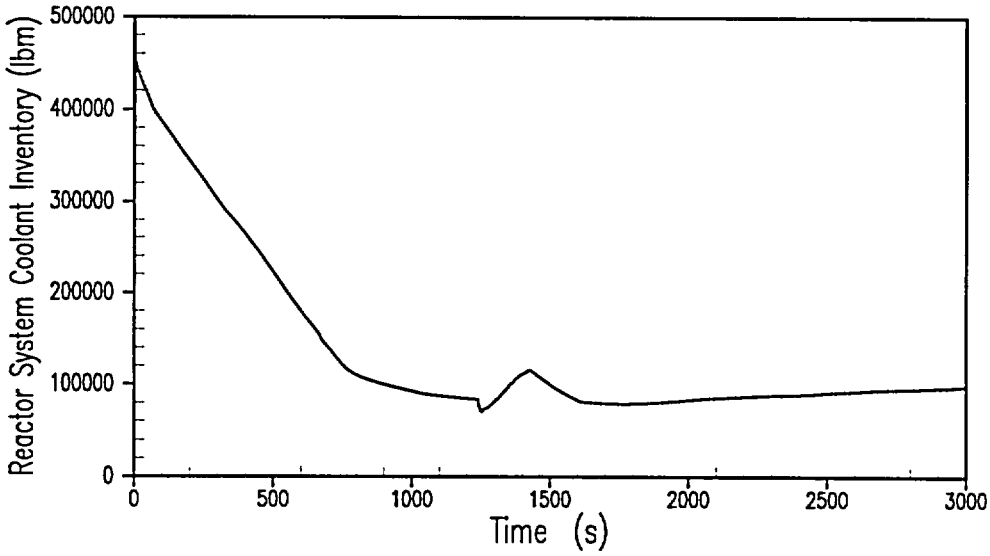
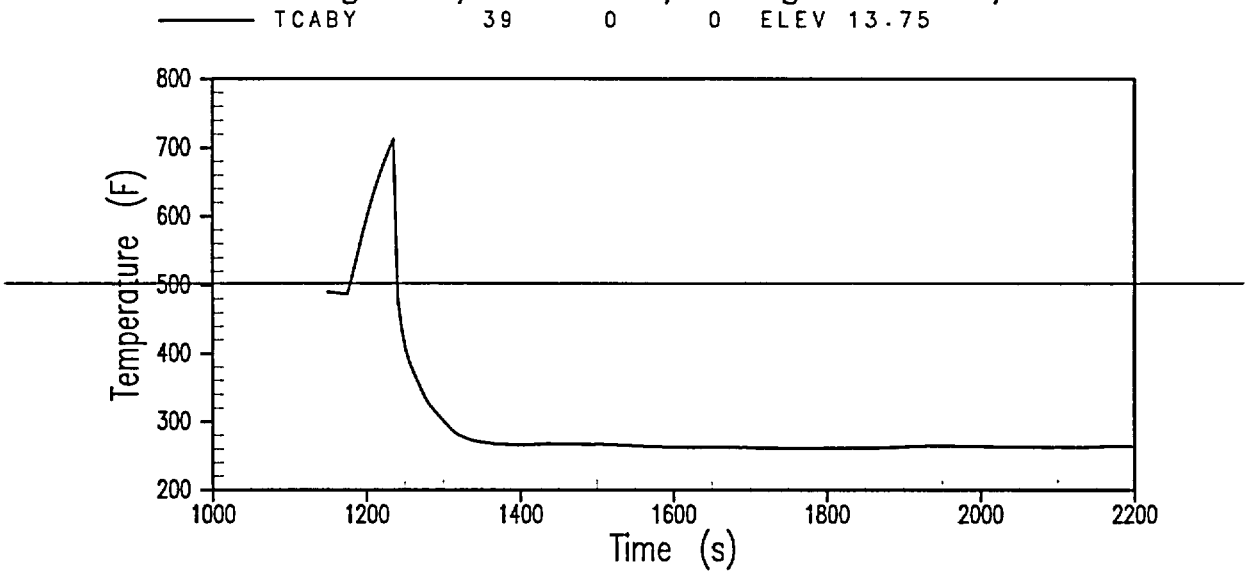



Figure A5.2-12

**Case UC1 Reactor System Coolant Inventory**

3.25 Inch Hot Leg Break/Manual ADS4/No Stage 1–2–3 ADS/No CMTs




3.0 Inch Hot Leg Break/Man ADS4/No ADS123/No CMT/1 Acc

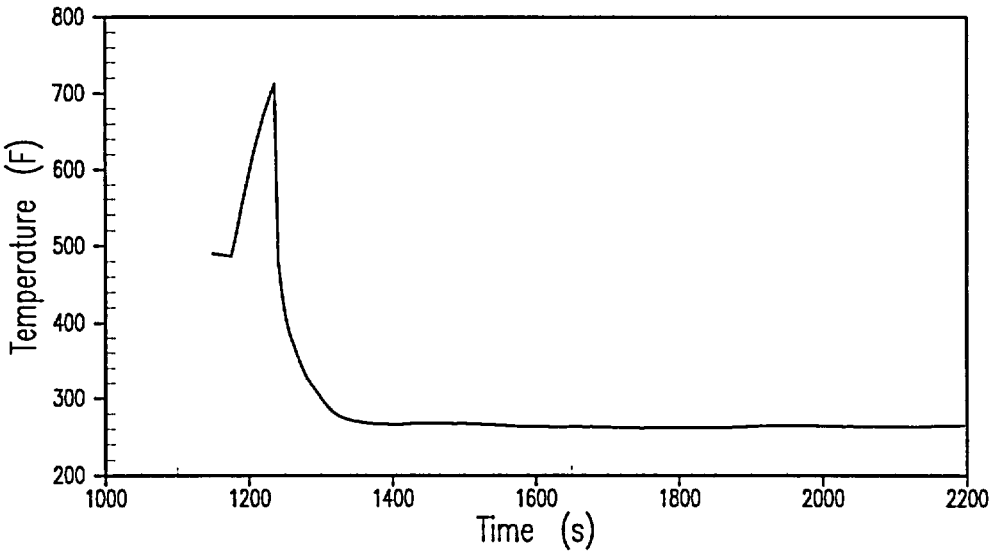



Figure A5.2-13

**Case UC1 Peaking Cladding Temperature**

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number. 720.027

***Question:***

The staff review identified several differences between the AP600 and the AP1000 probabilistic risk assessments (PRAs) in the assumed initiating event frequencies for several accident categories. These differences are related to (a) various loss of coolant accident (LOCA) categories, (b) steam generator tube rupture (SGTR) accidents, and (c) passive residual heat removal (PRHR) heat exchanger (HX) tube rupture accidents.

For the LOCA categories, these differences are primarily due to the selective use in the AP1000 PRA of operating experience data reported in NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 - 1995," February 1999, for pipe breaks as opposed to the use of data from pipe break analyses in the AP600 PRA. In the AP1000 PRA, operating experience data are used for some pipe break frequencies (e.g., pipe breaks contributing to medium and large LOCAs) but not for others (e.g., pipe breaks in the direct vessel injection lines). Similarly, the frequencies of contributors to the various LOCA initiating event categories other than pipe breaks, such as stuck-open pressurizer valves, were calculated in the AP1000 PRA while operating experience data are reported in NUREG/CR-5750 for these contributors.

The initiating event frequency for SGTR events is assumed in the AP1000 PRA to be 3.88E-3 events/year as opposed to 5.2E-3 events/year used in the AP600 and 7E-3 events/year reported in NUREG/CR-5750. In Attachment 2A, it is stated that the AP1000 SGTR frequency is based on 1.94E-7 failures per tube-year instead of 1.25E-6 failures per tube year used in the AP600 PRA because the AP1000 steam generator will be manufactured using Alloy 690 which is more resistant to stress corrosion cracking than the Alloy 600 assumed in the AP600 design. It is argued that due to this design improvement the SGTR frequency is smaller for the AP1000 design than the AP600 design even though the AP1000 steam generators have many more tubes (i.e., 20,000 for AP1000 versus 12,614 for AP600). Please explain how the SGTR frequency per tube year for AP1000 was calculated, including data and assumptions.

The initiating event frequency for PRHR HX tube rupture events is assumed in the AP1000 PRA to be 1.34E-4 events/year as opposed to 2.5E-4 events/year used in the AP600 PRA even though there are more and longer tubes in the AP1000 design. Statements made in Section 2.3.1.4 imply that the reliability of the AP1000 design PRHR HX tubes has improved because the reliability of the AP1000 SGTR tubes has improved with respect to the AP600 design. Please explain or re-calculate the PRHR HX tube rupture frequency to address these comments.

The frequency of PRHR HX tube rupture events is shown in the associated event tree (page 4-185) as 2.83E-4 events/year which is different from 1.34E-4 events/year reported in Chapter 2. Please explain this apparent discrepancy and revise appropriately.

The staff believes that a better understanding is needed of the impact of these new frequencies, including uncertainties, on the PRA results and insights which were derived from the AP600 PRA and assumed to be valid also for AP1000. Examples of insights and conclusions that may need to be revised are the identification of low thermal margin but risk significant success sequences for thermal-hydraulic (T-H) uncertainty assessment and the level of availability control for non-safety-related equipment. Please provide information addressing this issue.

**Westinghouse Response:**

Westinghouse responses to the NRC questions asked in this RAI are given below.

1. Table 720.027-1, which is attached, summarizes the revision of the initiating event frequencies from AP600 to AP1000 PRAs for internal events at power. The reasons for changes are summarized in the last column of this table.

When we started AP1000 PRA initiating event analysis, we had to create an unbiased set of decision rules about how, if any, updates to the initiating event frequencies should be handled, before any risk calculations were made. Westinghouse applied the following criteria:

i. Examine each frequency and revise if there are design changes, or corrections;

ii. Update random pipe failure related "generic" frequencies (such as LLOCA, MLOCA, and SLOCA) to match the most recent reference NUREG-5750 that is also being used by the rest of the industry for existing plants (see the need for LLOCA discussed below);

iii. Evaluate recent analyses performed by Westinghouse on other projects and assess whether these insights should be incorporated into the AP1000 PRA.

Westinghouse examined recent industry guidelines and relevant documents to assess the "generic" LOCA frequencies as they are seen now (compared to what they were in early 1990 – for AP600 PRA).

It should also be noted that the AP1000 PRA, like the AP600 PRA, has additional pipe break categories such as CMT and DVI line breaks, that possibly could have been lumped into the generic LLOCA and MLOCA categories. However, Westinghouse retained these separate categories in the AP1000 PRA, and applied the generic initiating event frequency to the LLOCA frequency alone. In addition, "consequential" LOCA (stuck open valves) frequencies were retained from the AP600 PRA since there is no evidence that new information has been developed since the time of the AP600 PRA.

With respect to SGTR frequency, we have applied the results of recent calculations of the initiating event frequencies for SGTR for replacement steam generator projects that take credit for the performance in the field of steam generators with Alloy 690TT tubes during the last 5-10

years. The resulting frequencies are lower, and these frequencies are used to better represent the actual performance of the steam generators that will be also used in the AP1000. This revision also impacted the PRHR initiating event frequency, since the PRHR tube rupture was based on the SGTR frequency.

The total number of transients assumed for AP1000 is at the order of 2.3 events per year. This is larger than the average number of transients that operating plants are experiencing currently. Moreover, since the various transient event categories are not risk dominant as observed in AP600 PRA, and later confirmed in AP1000 PRA , we did not update the transient event frequencies.

Also note that one category has been added and one removed:

i) Spurious ADS category, which was subsumed into LLOCA category in AP600, has been separated out to take credit for the break location and characteristics, compared to other LLOCA events.

ii) Medium and intermediate LOCA ranges are combined into a single LOCA range, MEDIUM LOCA, consistent with common PRA practices. In the AP600 PRA, the intermediate LOCA category was initially in the MLOCA range; it was separated out later (based on AP600 plant-specific MAAP analyses) to take credit for more favorable success criteria. This is not needed (nor applicable) to AP1000 success criteria. The current definition of MLOCA is consistent with the AP1000 plant specific success criteria.

The AP1000 initiating event frequencies used are based on current industry data, and provide a reasonable basis to evaluate the AP1000 plant risk.

2. As mentioned in the response to question 1 above, the AP1000 PRA uses a more recent calculation of the SGTR frequency that was performed in conjunction with a replacement steam generator project. This more recent calculation takes into account field experience of steam generators that utilize Alloy 690 tubes. The AP1000 SGTR initiating event frequency takes the updated IEV from the RSG project calculation, and increases it based on the additional number of tubes included in the AP1000 steam generators. This calculation is proprietary to Westinghouse, and can be made available to the staff in our Westinghouse offices in Rockville.

3. The calculated and used frequency for PRHR HX tube rupture events is based on the SGTR frequency, as discussed in Chapter 2 of the AP1000 PRA. This is believed to be a conservative estimate for the PRHR tube rupture frequency for the following reasons:

   - the PRHR HX tubes are thicker than the SG tubes (0.065 vs. 0.040)

- the SG tubes are in a much harsher operating environment (secondary water chemistry, continuously boiling heat transfer duty during normal operation) than the PRHR HX tubes. The PRHR HX is normally idle, and is only used in case of an accident. The HX tubes sit in a pool of refueling water with strict chemistry controls and at room temperature (50 – 120 F)

Since the calculation of the PRHR HX tube rupture frequency was based on the SGTR frequency, it is reduced with respect to the AP600 estimate. The reasoning is explicitly stated in Chapter 2 of AP1000 PRA and is consistent with the set of initiating event frequencies.

4. The PRHR HX tube rupture initiating event frequency calculated in the AP1000 PRA and used for plant CDF and LERF calculations is 1.34E-4/year, as given in Section 2. The value on the event tree picture in Chapter 4 is a "typo" and is not used anywhere for calculations. Chapter 4 does not contain or affect calculations. The PRA will be revised as shown.

Additional information is provided as follows:

ii) Please see the response to RAI 720.012 for an AP1000 assessment of the T&H uncertainty assessment for the AP1000 PRA.

iii) Please see the responses to RAI 100.003, 261.011 and 440.108 for a discussion of the AP1000 and its compliance to the RTNSSS Policy Statement.

**Table 720.027-1. Comparison of AP1000 and AP600 PRA Initiating Event Frequencies at Power**

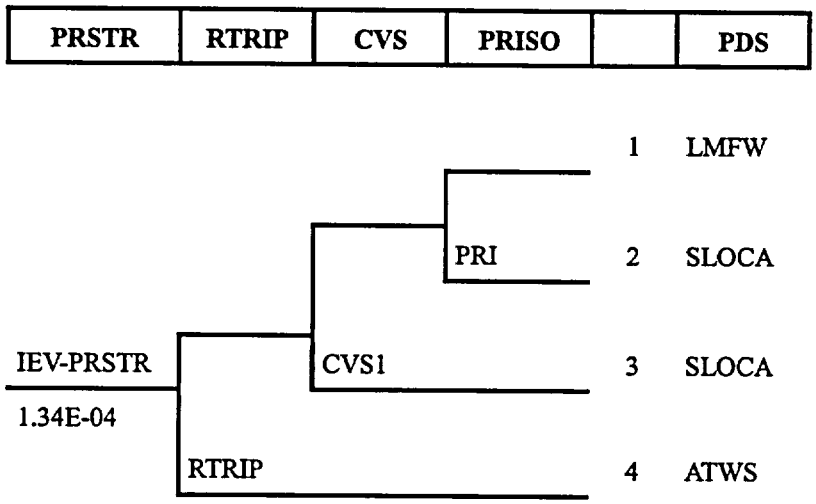
| | Event Category | AP1000 Category Frequency | AP600 Category Frequency | Reason For Change |
|---|---|---|---|---|
| 1 | Large LOCA | 5.04E-06 | 1.05E-04 | use industry data |
| 2 | Large Spurious ADS Actuation | 5.40E-05 | N/A | new category |
| 3 | Medium LOCA | 4.36E-04 | 1.62E-04 | use industry data |
| 4 | Core makeup tank line break | 9.31E-05 | 8.94E-05 | number of pipe segments |
| 5 | Safety injection line break | 2.12E-04 | 1.04E-04 | number of pipe segments |
| | Intermediate LOCA | N/A | 7.70E-04 | included in MLOCA |
| 6 | Small LOCA | 5.00E-04 | 1.01E-04 | use industry data |
| 7 | RCS leakage | 6.20E-03 | 1.20E-02 | use industry data |
| 8 | Passive residual heat removal tube rupture | 1.34E-04 | 2.50E-04 | number of pipe segments; updated database |
| 9 | Steam generator tube rupture | 3.88E-03 | 5.20E-03 | number of pipe segments; updated database |
| 10 | Reactor vessel rupture | 1.00E-08 | 1.00E-08 | |
| 11 | Interfacing system LOCA | 5.00E-11 | 5.00E-11 | |
| 12 | Transient with main FW | 1.40E+00 | 1.4E+00 | |
| 13 | Loss of RCS flow | 1.80E-02 | 1.80E-02 | |
| 14 | Loss of main FW to one steam generator | 1.92E-01 | 1.92E-01 | |
| 15 | Core power excursion | 4.50E-03 | 4.50E-03 | |
| 16 | Loss of component cooling water/service water | 1.44E-01 | 1.44E-01 | |
| 17 | Loss of main FW to both steam generators | 3.35E-01 | 3.35E-01 | |
| 18 | Loss of condenser | 1.12E-01 | 1.12E-01 | |
| 19 | Loss of compressed air | 3.48E-02 | 3.48E-02 | |
| 20 | Loss of offsite power | 1.20E-01 | 1.20E-01 | |
| 21 | Main steam line break downstream of MSIVs | 5.96E-04 | 5.96E-04 | |
| 22 | Main steam line break upstream of MSIVs | 3.72E-04 | 3.72E-04 | |
| 23 | Main steam line stuck-open valve | 2.39E-03 | 1.21E-03 | number of valves increased |
| 24 | ATWS precursor without main feedwater | 4.81E-01 | 4.81E-01 | |
| 25 | ATWS precursor With SI | 1.48E-02 | 2.05E-02 | contributing events changed |
| 26 | ATWS precursor with main feedwater | 1.17E+00 | 1.17E+00 | |
| | | | | |
| | **Total (Excluding ATWS) =** | 2.38E+00 | 2.42E+00 | |

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

The corrected value of the PRHR HX tube rupture event frequency will be included in the next revision of the AP1000 PRA Figure 4B-8




**List of Top Events**

| Event | Description |
|---|---|
| **PRSTR** | PRHR Tube Leak Event Occurs |
| **RTRIP** | Reactor Trip Occurs |
| **CVS** | CVS System is Operable |
| **PRISO** | Operator Isolates the PRS following PRS Tube Leak Event |

**Figure 4B-8**

**AP1000 Passive RHR Tube Rupture Event Tree**

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number. 720.028

***Question:***

The staff review identified two potentially significant differences between the AP600 and the AP1000 PRAs in the categorization of LOCA initiating events. One difference involves combining two AP600 PRA LOCA categories into one AP1000 PRA category. In the AP600 PRA, distinct medium LOCA (MLOCA) and an intermediate LOCA (NLOCA) categories were defined. The MLOCA involved break sizes between six and nine inches equivalent diameter and the NLOCA involved break sizes between two and six inches equivalent diameter. In the AP1000 PRA, however, these two categories were combined into one category (MLOCA which range from two to nine inches equivalent diameter). The other difference involves the splitting of one AP600 PRA LOCA category into two categories in the AP1000 PRA. In the AP600 PRA, the large LOCA category included both pipe breaks and spurious opening of the automatic depressurization system (ADS) valves. In the AP1000, there is one large LOCA category for pipe breaks and another large LOCA category for spurious ADS actuation.

T-H analyses performed for the AP600 design have identified bounding parameters (e.g., break size and location) for each LOCA category. The success criteria for systems and operator actions used in the AP600 PRA were determined by studying the plant response given such bounding parameters. For example, in the AP600 PRA, the success criteria for a MLOCA (i.e., six to nine inches equivalent diameter) are based on a cold leg break while for an NLOCA (i.e., two to six inches equivalent diameter) they are based on a hot leg break. These two AP600 LOCA categories which are associated with different plant responses were combined in the AP1000 PRA. Since the AP1000 PRA success criteria are based to a certain extend on information from analyses performed for the AP600 design, please explain the following:

A. Why these different initiating event categorizations were needed;

B. How information from AP600 design T-H analyses is used to support AP1000 design criteria; and

C. The approach that was followed, including new analyses beyond those performed for AP600, to ensure that the AP1000 success criteria are valid for all break sizes and locations in each category and that the T-H uncertainties for risk-significant sequences are bounded.

Westinghouse Response:

A. The reason for the two different initiating event categorizations is as follows:

Split LLOCA and SPADS:

These two events are split because they have different accumulator success criteria. The AP1000 design-specific accumulator success criteria for events in the spectrum of large loss-of-coolant accident (LLOCA) is 2 of 2 accumulators, which is adequate for low frequency pipe break events. This success criteria is conservative for a specific category of LLOCA events, namely the spurious actuation of ADS stage 4 line(s) (SPADS). Since the frequency assigned to SPADS (the same frequency both in AP600 and AP1000) is higher than that of the pipe break frequency for LLOCA events, we took credit for 1 of 2 accumulators being adequate for success in SPADS. This allows a more realistic and plant-specific estimation of risk from these rare events. That is the reason why the SPADS is separated out from the LLOCA category. In AP600, 1 of 2 accumulators was adequate for all LLOCA events, and a split was not needed.

Combination of MLOCA and NLOCA:

In AP600, the traditional MLOCA range was split into NLOCA (smaller) and MLOCA (larger). The success criteria was the same except that NLOCA required one ADS stage 2/3 valve to open to allow RNS injection and MLOCA did not require any of these valves. In AP1000, the more restrictive NLOCA success criteria was applied to all breaks in the MLOCA range; thus, there is no need to split the range.

B. The information from AP600 design T-H analyses provided experience and insights to the designers and analysts to select the limiting accident conditions. Plant-specific AP1000 analyses were performed on the limiting conditions as documented in Appendix A of the AP1000 PRA.

C. The approach followed to justify the AP1000 PRA success criteria is the same as for AP600. That approach is to use the following justification:

1. DCD analysis where available
2. Perform AP1000 PRA specific analysis
3. Other justifications (other calculations, engineering judgement)

Chapter 6 and Appendix A of the PRA explain which of these approaches was applied to each success criteria and how T-H uncertainty is addressed for AP1000. Additional information is also provided in the response to RAIs (720.010, 720.012 and 720.03).

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 720.031

***Question:***

The success criteria listed in Table 6-2 for the "containment integrity" function appear to apply to the "core cooling" function. Please explain and revise accordingly.

**Westinghouse Response:**

Section 6 and Table 6-2 are intended for description of Level 1 success criteria for core damage event tree models in PRA Chapter 4. The containment integrity is only mentioned for its relation to other core damage mitigating systems in these event trees, specifically for the sump recirculation in the containment. So the presence of "containment integrity" as it affects other mitigation systems for core damage is necessary and appropriate in Table 6-2.

Containment integrity function, as it relates to PRA Level 2 (containment analysis) is discussed and modeled in the Chapter 35 "Containment Event Tree Analysis" of the PRA Report.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

The Table 6-2 (Sheet 9 of 9) will be updated in the next revision of the AP1000 PRA Chapter 6 accordingly (see Attachment).

Table 6-2 (Sheet 9 of 9)

**SUMMARY OF SUCCESS CRITERIA FOR THE MITIGATING SYSTEMS**

| Function/Accident Initiator | Success Sequence Identifiers | Success Criteria | Basis |
|---|---|---|---|
| **Containment Heat Removal** | | | |
| • All initiating events | (all) | 1 out of 3 PCS drain valves<br>**OR**<br>PCA air cooling [17] | Reference 6-4 Chapter 6<br><br>PRA Appendix A |
| • Sequences where RNS is operable | * | 1 out of 2 RNS pumps/HXs and CCW cooling to RNS HXs<br>**OR**<br>1 out of 3 PCS drain valves<br>**OR**<br>PCA air cooling [17] | Design Basis<br><br>Reference 6-4 Chapter 6<br><br>PRA Appendix A |
| **Containment Integrity for Containment Recirculation** | | | |
| • All initiating events which require gravity injection of IRWST | (all) | 1 out of 4 containment recirculation valves are required if containment isolation is successful; with failure of containment isolation, 2 out of 4 containment recirculation valves are required[18] | PRA Appendix A |

*Too many to list

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 720.032

***Question:***

In Section 6.3.1, "General Sequence Success Criteria," it is stated (page 6-3): "In general, if the reactor achieves a stable shutdown condition without core damage, and this condition can be maintained for at least 24 hours following event initiation without further action or system operation, the sequence is categorized as successful. It is not sufficient to avoid core damage during the first 24 hours if conditions have not stabilized and core damage is anticipated shortly following 24 hours. That is, core damage is assumed if the RCS conditions are not stabilized in 24 hours, or if core damage is anticipated following 24 hours without further system or operator action." Please describe the accident sequences where the conditions are not stabilized in 24 hours or core damage is anticipated following 24 hours without further system or operator action.

**Westinghouse Response:**

This issue has been recognized and discussed during AP600 PRA information exchanges between Westinghouse and the NRC, and was resolved satisfactorily, leading to PRA insights. For the AP1000 PRA, the same conditions that avoid the success criteria "cliffs" after the initial 24-hour mission time are kept, and the related insights are maintained. Thus, for the AP1000 PRA sequence success criteria, there are no "cliffs" within a reasonable time period (to 72 hours) beyond 24 hours. Each success sequence leads to and maintains a stable safe shutdown condition, not only within the first 24-hour mission time, but also in a following reasonable time window (to 72 hours).

The following insights, which also apply to AP1000 PRA, are summarized for how the perceived "cliffs" are analyzed and resolved:

**1. Depletion of Passive Containment Cooling Inventory:**

There is sufficient inventory to last 72 hours (additional inventory to achieve a 7 day coping period is available from nonsafety-related sources).

**2. Potential Inventory Loss through PRHR Gutter Valves**

The PRHR gutter valves were changed to a safety-related status in AP600 as the result of PRA insights. The failure of these valves to close is modeled in the AP1000 PRHR system failure models (fault trees). As a result, there is sufficient IRWST inventory to support PRHR HX operation for at least 72 hours.

### 3. Potential Inventory Loss When Containment Isolation Fails

MAAP analyses are made with the containment isolation failure conditions. It has been established that the containment pressure drops to atmospheric pressure and the leak stops. The inventory lost by that time is not consequential, as shown by the analyses.

### 4. ADS Actuation by Timer in 22 Hours After LOSP

In benign transients following a prolonged loss of offsite power, automatic ADS actuation by a timer at 22-hours would transform the event to a LOCA. This design feature was introduced as a defense to battery depletion at 24 hours into a station blackout event (there are 72-hour batteries for monitoring plant parameters). Based on PRA insights, an operator action to disable the timer actuation circuit was introduced in AP600 (and is applicable to AP1000), saving 4 hours of battery capacity, in case later ADS actuation is needed. This allows a benign transient to proceed beyond 24 hours without unnecessary introduction of automatic ADS actuation at 22 hours.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

RAI Number: 720.059

*Question:*

Please provide a description of: (1) the physical characteristics of the door between the reactor cavity compartment and RCDT room, including its approximate size, construction, bouyancy, hinging arrangement (opening direction and jambs), and pressure retaining capability, (2) the expected response of the door during the flood-up period and following a postulated melt-through of the RV, and (3) the potential for the door to break free and block the inlets to the RV insulation system during flood-up, or remain in place and restrict debris spreading within the reactor cavity following a postulated melt-through of the RV. Also explain why these design details should not be included in the system design description in the DCD.

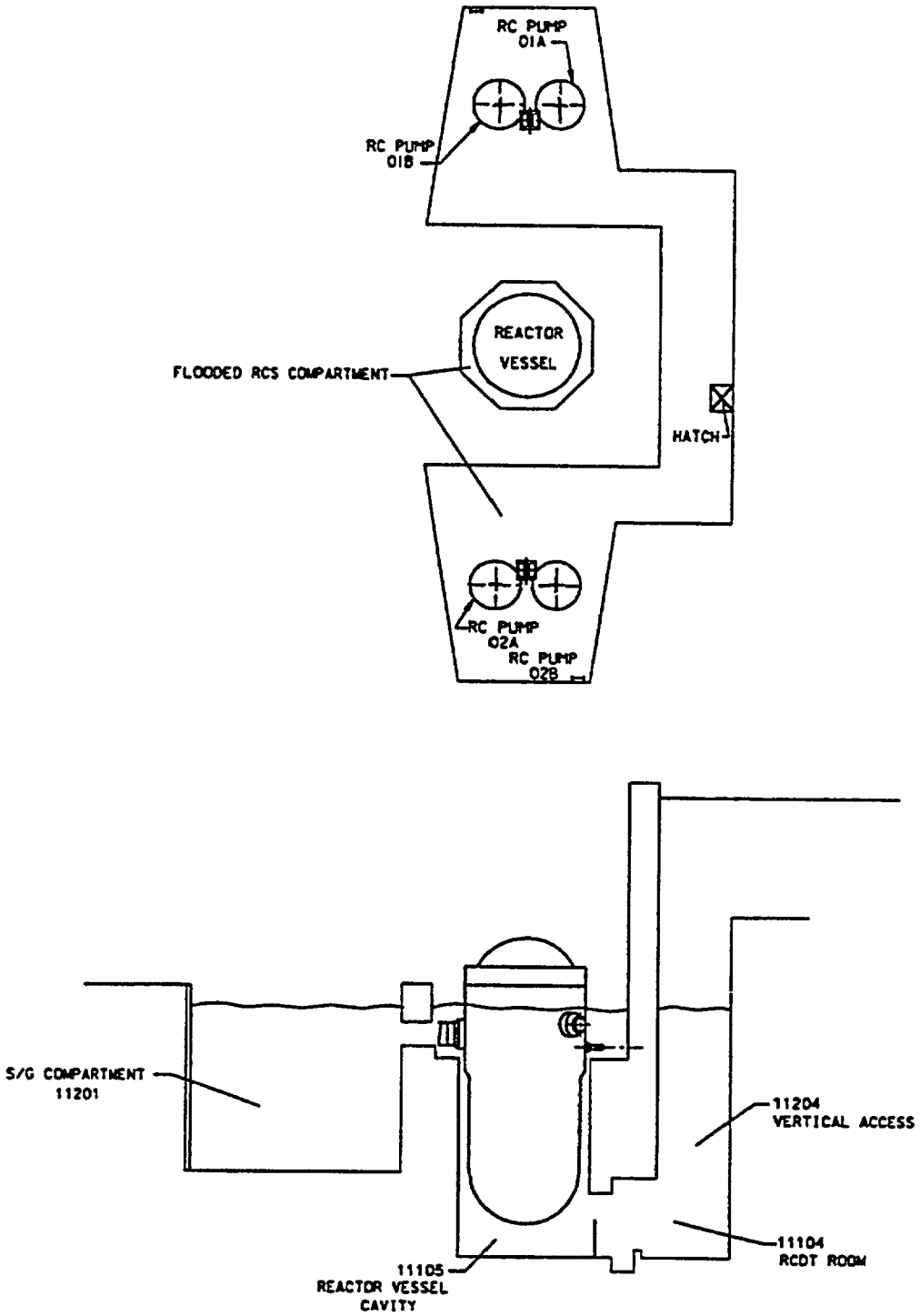
**Westinghouse Response:**

DCD section 5.3.5.3 provides a description of the external vessel flooded compartments including the RCDT room, and the operation of the door and damper during normal operation and post-accident operation. The description provided in the AP1000 DCD includes the same level of detail as provided in the AP600 DCD, and addresses items (1) and (2) from the questions.

The RCDT door is designed to remain intact during flooding of the reactor cavity, the RCDT room, or both. The RCDT door is not designed for the loads associated with ex-vessel severe accidents that can be postulated.

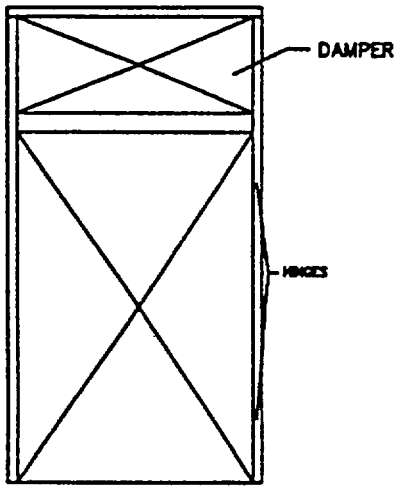
**5.3.5.3 Description of External Vessel Cooling Flooded Compartments**

Ex-vessel cooling during a severe accident is provided by flooding the reactor coolant system loop compartment including the vertical access tunnel, the reactor coolant drain tank room, and the reactor cavity. Water from these compartments replenishes the water that comes in contact with the reactor vessel and is boiled and vented to containment. The opening between the vertical access tunnel and the reactor coolant drain tank room is approximately 100 $ft^2$. Removable steel grating is provided over the inlet to the vertical access tunnel to restrict access to the lower compartments. This grating precludes large debris from being transported into the reactor cavity during ex-vessel cooling scenarios. Figure 5.3-8 depicts the flooded compartments that provide the water for ex-vessel cooling. The doorway between the reactor cavity compartment and the reactor coolant drain tank room consists of a normally closed door and a damper above the door. The door and damper arrangement, shown in Figure 5.3-9, maintains the proper air flow through the reactor cavity during normal operation. The damper prevents air from flowing into the reactor coolant drain tank compartment, but opens to permit flooding of the reactor cavity from the reactor coolant drain tank compartment. The damper opening has a minimum flow area of 8 $ft^2$ and is not susceptible to clogging from debris that can pass through the grating over the inlet to the vertical access tunnel. It is constructed of light-weight material to minimize the force necessary to open the damper and permit flooding and continued water flow through the opening during ex-vessel cooling. The damper provides an acceptable pressure drop through the opening during ex-vessel cooling.

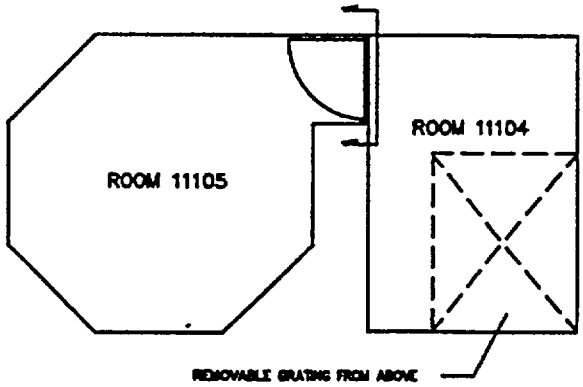



**DCD Figure 5.3-8 RCS Flooded Compartments During Ex-Vessel Cooling**

DOORWAY BETWEEN RCDT ROOM AND REACTOR CAVITY COMPARTMENT




LOOKING SOUTH FROM RCDT ROOM (11104) INTO REACTOR CAVITY (11105)




**DCD Figure 5.3-9 Door Between RCDT Room and Reactor Cavity**

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 720.062

***Question:***

The commitment to cover and lock closed access portals to the Passive Core Cooling System (PXS) that may be near the containment wall is important for limiting diffusion flames. Please discuss how this commitment will be addressed in ITAAC or COL action items.

**Westinghouse Response:**

PRA Chapter 41 discusses the access hatches to the PXS compartments as follows:

"Vents from the PXS and CVS compartments to the CMT room are located well away from the containment shell and containment penetrations. Access hatches to the subcompartments that are near the containment shell are covered and secured closed such that they will not open as a result of a pipe break inside the compartment. Therefore, hydrogen releases to the CMT room from the subcompartments are not considered as a threat to the containment integrity."

A specific design commitment is not included in the ITAAC or COL action items for this level of design detail. DCD section 3.4.1.2.2.1 Containment Flooding Events provides a discussion of the design features of the containment structural compartments including the PXS compartments that are designed to flood following an accident. DCD section 3.4.1.2.2.1 will be revised to incorporate a discussion of the closed access portals provided in these compartments.

**Design Control Document (DCD) Revision:**

DCD Section 3.4.1.2.2.1, 14th paragraph, page 3.4-7 will be revised as follows:

The PXS-A compartment (Room 11206), PXS-B compartment (Room 11207) and the chemical and volume control system compartment (Room 11208) are physically separated and isolated from each other by structural walls and curbs such that flooding in any one of these compartments or in the reactor coolant system compartment cannot cause flooding in any of the other compartments. The access hatch to the PXS-B compartment is located near the containment wall and is normally closed to address severe accident considerations. The access hatch to the PXS-B compartment is accessible from Room 11300 on elevation 107'-2".

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number: 720.077

***Question:***

The AP1000 system equivalent to the AP600 post-accident sampling system is designated as the containment atmosphere sampling system. Are there any differences between the AP600 post-accident sampling system and the AP1000 containment atmosphere sampling system including differences in functions? If so, please describe.

***Westinghouse Response:***

There are no differences between the AP600 and AP1000 containment atmosphere sampling system design features included in the primary sampling system. The AP1000 primary sampling system is described in DCD section 9.3.3. The AP1000 primary sampling system does not include specific post-accident sampling capability. The AP1000 post-accident sampling functional requirements have been eliminated in accordance with the Safety Evaluation by the Office of Nuclear Regulation Related to WCAP-14986, "Westinghouse Owners Group Post Accident Sampling System Requirements," Westinghouse Owners Group Project No. 694, June 14, 2000. (DCD Section 9.3.8 Reference 5). In accordance with this safety evaluation, the AP1000 provides the capability to perform post-accident sampling (on a contingency basis) for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. These samples are for analysis, during the later stages of accident response; reactor coolant for boron, containment atmosphere for hydrogen and fission products, and containment sump water for pH.

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None

# AP1000 DESIGN CERTIFICATION REVIEW

## Response to Request For Additional Information

RAI Number 720.081

***Question:***

The AP600 FSER states that the combined license (COL) applicant referencing the AP600 certified design will perform a thermal lag assessment of the as-built equipment used to mitigate severe accidents to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns. This assessment is COL Action Item 19.2.3.3.7-1 and is contained in Section 19.59.10.5 of the AP600 DCD. Where can this COL Action Item be found in the AP1000 DCD? How can the staff make a reasonable assurance finding, such as in Section 19.2.3.3.7, for the AP1000 without this COL Action Item?

**Westinghouse Response:**

The AP1000 Tier 1 Material contains the commitment to perform a thermal lag assessment of the as-built equipment. This design commitment is item 9.c) in Table 2.2.3-4 as is attached.

| Table 2.2.3-4 (cont.)<br>Inspections, Tests, Analyses, and Acceptance Criteria | | |
|---|---|---|
| **Design Commitment** | **Inspections, Tests, Analyses** | **Acceptance Criteria** |
| 9.c) The equipment listed in Table 2.2.3-6 has sufficient thermal lag to withstand the effects of identified hydrogen burns associated with severe accidents. | Tests, analyses, or a combination of tests and analyses will be performed to determine the thermal lag of this equipment. | A report exists and concludes that the thermal lag of this equipment is greater than the value required. |

| Table 2.2.3-6 | | |
|---|---|---|
| **Equipment** | **Tag No.** | **Function** |
| Containment Air Sample Containment Isolation Valve IRC | PSS-PL-V008 | Transfer open |
| Containment Pressure Sensors | PCS-005, 006, 007, 008 | Sense pressure |
| RCS Wide Range Pressure Sensors | RCS-140A, B, C, D | Sense pressure |
| SG1 Wide Range Level Sensors | SGS-011, 012, 013, 014 | Sense level |
| SG2 Wide Range Level Sensors | SGS-016, 017, 018, 019 | Sense level |
| Hydrogen Ignitors | VLS-EH-01 through 64 | Ignite hydrogen |
| Electrical Penetrations | VUS-JY-E01, 02, 06, 09, 10, 11, 12, 13, 14, 15, 16, 17, 18, 21, 22, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32 | Maintain containment boundary |

**Design Control Document (DCD) Revision:**

None

**PRA Revision:**

None