15.6 Decrease in Reactor Coolant Inventory

This section discusses the following events that result in a decrease in reactor coolant inventory:

- An inadvertent opening of a pressurizer safety valve or inadvertent operation of the automatic depressurization system (ADS)
- A break in an instrument line or other lines from the reactor coolant pressure boundary that penetrate the containment
- A steam generator tube failure
- A loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary

The applicable accidents in this category have been analyzed. It has been determined that the most severe radiological consequences result from the major LOCA described in subsection 15.6.5. The LOCA, chemical and volume control system letdown line break outside the containment and the steam generator tube rupture (SGTR) accident are analyzed for radiological consequences. Other accidents described in this section are bounded by these accidents.

15.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS

15.6.1.1 Identification of Causes and Accident Description

Two types of inadvertent depressurization are discussed in this section. One covers all inadvertent operation of ADS valves. The other covers inadvertent opening of a pressurizer safety valve.

An inadvertent depressurization of the reactor coolant system can occur as a result of an inadvertent opening of a pressurizer safety valve or ADS valves. Initially, the event results in a rapidly decreasing reactor coolant system pressure. The pressure decrease causes a decrease in power via the moderator density feedback. The average coolant temperature decreases slowly, but the pressurizer level increases until reactor trip.

The reactor may be tripped by the following reactor protection system signals:

- Overtemperature ΔT
- Pressurizer low pressure

The ADS is designed such that inadvertent operation of the ADS is classified as a Condition III event, an infrequent fault.

An inadvertent opening of a pressurizer safety valve is a Condition II event, a fault of moderate frequency.

The ADS system consists of four stages of depressurization valves. The ADS stages are interlocked; for example, Stage 1 is initiated first and subsequent stages are not actuated until previous stages have been actuated. Each stage includes two redundant parallel valve paths such that no single failure prevents operation of the ADS stage when it is called upon to actuate and the spurious opening of a single ADS valve does not initiate ADS flow. To actuate the ADS manually from the main control room, the operators actuate two separate controls positioned at some distance apart on the main control board. Therefore, one unintended operator action does not cause ADS actuation.

ADS Stage 1 has a design opening time of 25 seconds and an effective flow area of 7 in^2 (maximum). ADS Stages 2 and 3 have design opening times of 70 seconds and an effective flow area of 26 in^2 (maximum).

In each ADS path are two valves in series such that no mechanical failure could result in an inadvertent operation of an ADS stage. The ADS Stage 4 squib valves cannot be opened while the reactor coolant system is at nominal operating pressure.

For this analysis, multiple failures and or errors are assumed which actuate both Stage 1 ADS paths. Although ADS Stages 2 and 3 have larger depressurization valves, the opening time of the Stage 1 depressurization valves is faster. This results in the most severe reactor coolant system depressurization due to ADS operation with the reactor at power.

Inadvertent opening of a pressurizer safety valve can only be postulated due to a mechanical failure. Although a pressurizer safety valve is smaller than the combined two Stage 1 ADS valves, the pressurizer safety valve is postulated to open in a short time.

Therefore, analyses are presented in this section for the inadvertent opening of a pressurizer safety valve and the inadvertent opening of two paths of Stage 1 of the ADS. These analyses are performed to demonstrate that the departure from nucleate boiling ratio (DNBR) does not decrease below the design limit values (see Section 4.4) while the reactor is at power.

In meeting the requirements of GDC 17 of 10 CFR Part 50, Appendix A, analyses have been performed to evaluate the effects produced by a possible consequential loss of offsite power during inadvertent reactor coolant system depressurization events. As discussed in subsection 15.0.14, the loss of offsite power is considered as a direct consequence of a turbine trip occurring while the plant is operating at power. The primary effect of the loss of offsite power is to cause the reactor coolant pumps to coast down.

15.6.1.2 Analysis of Effects and Consequences

15.6.1.2.1 Method of Analysis

The accidental depressurization transient is analyzed by using the computer code LOFTRAN (References 14 and 15). The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer safety valves, main steam isolation valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperatures, pressures, and power level.

For reactor coolant system depressurization analyses that include a primary coolant flow coastdown caused by a consequential loss of offsite power, a combination of three computer codes is used to perform the DNBR analyses. First the LOFTRAN code is used to perform the plant system transient. The FACTRAN code (Reference 18) is then used to calculate the core heat flux based on nuclear power and reactor coolant flow from LOFTRAN. Finally, the VIPRE-01 code (see Section 4.4) is used to calculate the DNBR using heat flux from FACTRAN and flow from LOFTRAN.

Plant characteristics and initial conditions are discussed in subsection 15.0.3. The following assumptions are made to give conservative results in calculating the DNBR during the transient:

- Initial conditions are discussed in subsection 15.0.3. Uncertainties in initial conditions are included in the DNBR limit as discussed in WCAP-11397-P-A (Reference 16).

- A least negative moderator temperature coefficient is assumed. The spatial effect of voids resulting from local or subcooled boiling is not considered in the analysis with respect to reactivity feedback or core power shape.

- A large (absolute value) Doppler coefficient of reactivity is used such that the resulting amount of positive feedback is conservatively high to retard any power decrease.

Plant systems and equipment necessary to mitigate the effects of reactor coolant system depressurization are discussed in subsection 15.0.8 and are listed in Table 15.0-6.

Normal reactor control systems are not required to function. The rod control system is assumed to be in the automatic mode to maintain the core at full power until the reactor trip protection function is reached. This is a worst case assumption. The reactor protection system functions to trip the reactor on the appropriate signal. No single active failure prevents the reactor protection system from functioning properly.

15.6.1.2.2 Results

The system response to an inadvertent opening of a pressurizer safety valve is shown in Figures 15.6.1-1 through 15.6.1-5. The figures show the results for cases with and without offsite power available. The calculated sequence of events for both inadvertent opening of a pressurizer safety valve scenarios are shown in Table 15.6.1-1.

A pressurizer safety valve is assumed to step open at the start of the event. The reactor coolant system then depressurizes until the overtemperature ΔT reactor trip setpoint is reached. Figure 15.6.1-3 shows the pressurizer pressure transient.

In the case where offsite power is lost, ac power is assumed to be lost 3 seconds after a turbine trip signal occurs. At this time, the reactor coolant pumps are assumed to start coasting down and reactor coolant system flow begins decreasing (Figure 15.6.1-5). The availability of offsite power has minimal impact on the pressure transient during the period of interest.

Prior to tripping of the reactor, the core power remains relatively constant (Figure 15.6.1-1). The minimum DNBR during the event occurs shortly after the rods begin to be inserted into the core (Figure 15.6.1-2). In the case where offsite power is lost, reactor trip has already been initiated and core heat flux has started decreasing when the reactor coolant system flow reduction starts. The DNBR continues to increase when reactor coolant system flow begins to decrease due to the loss of offsite power. Therefore, the minimum DNBR occurs at the same time for cases with and without offsite power available. The DNBR remains above the design limit values as discussed in Section 4.4 throughout the transient.

The system response for inadvertent operation of the ADS is shown in Figures 15.6.1-6 through 15.6.1-10. The figures show the results for cases with and without offsite power available. The sequences of events are provided in Table 15.6.1-1. The responses for inadvertent operation of the ADS are very similar to those obtained for inadvertent opening of a pressurizer safety valve.

15.6.1.3 Conclusion

The results of the analysis show that the overtemperature ΔT reactor protection system signal provides adequate protection against the reactor coolant system depressurization events. The calculated DNBR remains above the design limit defined in Section 4.4. The long-term plant responses due to a stuck-open ADS valve or pressurizer safety valve, which cannot be isolated, is bounded by the small-break LOCA analysis.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

The small lines carrying primary coolant outside containment are the reactor coolant system sample line and the discharge line from the chemical and volume control system to the liquid radwaste system. These lines are used only periodically. No instrument lines carry primary coolant outside the containment.

When excess primary coolant is generated because of boron dilution operations, the chemical and volume control system purification flow is diverted out of containment to the liquid radwaste system. Before passing outside containment, the flow stream passes through the chemical and volume control system heat exchangers and mixed bed demineralizer. The flow leaving the containment is at a temperature of less than 140°F and has been cleaned by the demineralizer. The flow out a postulated break in this line is limited to the chemical and volume control system purification flow rate of 100 gpm. Considering the low temperature of the flow and the reduced iodine activity because of demineralization, this event is not analyzed. The postulated sample line break is more limiting.

The sample line isolation valves inside and outside containment are open only when sampling. The failure of the sample line is postulated to occur between the isolation valve outside the containment and the sample panel. Because the isolation valves are open only when sampling, the loss of sample flow provides indication of the break to plant personnel. In addition, a break in a sample line results in activity release and a resulting actuation of area and air radiation monitors. The loss of coolant reduces the pressurizer level and creates a demand for makeup to the reactor coolant system. Upon indication of a sample line break, the operator would take action to isolate the break.

The sample line includes a flow restrictor at the point of sample to limit the break flow to less than 130 gpm. The liquid sampling lines are 1/4 inch tubing which further restricts the break flow of a sampling line outside containment. Offsite doses are based on a conservative break flow of 130 gpm with isolation after 30 minutes.

15.6.2.1 Source Term

The only significant radionuclide releases are the iodines and the noble gases. The analysis assumes that the reactor coolant iodine is at the maximum Technical Specification level for continuous operation. In addition, it is assumed that an iodine spike occurs at the time of the accident. The reactor coolant noble gas activities are assumed to be those associated with the design basis fuel defect level.

15.6.2.2 Release Pathway

The reactor coolant that is spilled from the break is assumed to be at high temperature and pressure. A large portion of the flow flashes to steam, and the iodine in the flashed liquid is assumed to become airborne.

The iodine and noble gases are assumed to be released directly to the environment with no credit for depletion, although a large fraction of the airborne iodine is expected to deposit on building surfaces. No credit is assumed for radioactive decay after release.

15.6.2.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.6.2.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.6.2-1.

15.6.2.5 Identification of Conservatisms

The assumptions used contain the following significant conservatisms:

- The reactor coolant activities are based on a fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less than this (see Section 11.1).

- It is unlikely that the conservatively selected meteorological conditions would be present at the time of the accident.

15.6.2.6 Doses

Using the assumptions from Table 15.6.2-1, the calculated total effective dose equivalent (TEDE) doses are determined to be < 2.1 rem at the exclusion area boundary and < 1.1 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. The phrase “a small fraction” is taken as being ten percent or less.

At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after 2 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the small line break outside containment, the resulting total dose remains less than the value reported above.

15.6.3 Steam Generator Tube Rupture

15.6.3.1 Identification of Cause and Accident Description

15.6.3.1.1 Introduction

The accident examined is the complete severance of a single steam generator tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods within the allowance of the Technical Specifications. The accident leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the reactor coolant system. In the event of a coincident loss of offsite power, or a failure of the condenser steam dump, discharge of radioactivity to the atmosphere takes place via the steam generator power-operated relief valves or the safety valves.

The assumption of a complete tube severance is conservative because the steam generator tube material (Alloy 690) is a corrosion-resistant and ductile material. The more probable mode of tube failure is one or more smaller leaks of undetermined origin. Activity in the secondary side is subject to continual surveillance, and an accumulation of such leaks, which exceeds the limits established in the Technical Specifications, is not permitted during operation.

The AP1000 design provides automatic protective actions to mitigate the consequences of an SGTR. The automatic actions include reactor trip, actuation of the passive residual heat removal (PRHR) heat exchanger, initiation of core makeup tank flow, termination of pressurizer heater operation, and isolation of chemical and volume control system flow and startup feedwater flow on high steam generator level. These protective actions result in automatic cooldown and depressurization of the reactor coolant system, termination of the break flow and release of steam to the atmosphere, and long-term maintenance of stable conditions in the reactor coolant system. These protection systems serve to prevent steam generator overfill (see discussion in subsections 15.6.3.1.2 and 15.6.3.1.3) and to maintain offsite radiation doses within the allowable guideline values for a design basis SGTR. The operator may take actions that would provide a more rapid mitigation of the consequences of an SGTR.

Because of the series of alarms described next, the operator can readily determine when an SGTR occurs, identify and isolate the ruptured steam generator, and complete the required recovery actions to stabilize the plant and terminate the primary-to-secondary break flow. The recovery procedures are completed on a time scale that terminates break flow to the secondary system before steam generator overfill occurs and limits the offsite doses to acceptable levels without actuation of the ADS. Indications and controls are provided to enable the operator to carry out these functions.

15.6.3.1.2 Sequence of Events for a Steam Generator Tube Rupture

The following sequence of events occur following an SGTR:

- Pressurizer low pressure and low level alarms are actuated and chemical and volume control system makeup flow and pressurizer heater heat addition starts or increases in an attempt to maintain pressurizer level and pressure. On the secondary side, main feedwater flow to the affected steam generator is reduced because the primary-to-secondary break flow increases steam generator level.

- The condenser air removal discharge radiation monitor, steam generator blowdown radiation monitor, and/or main steam line radiation monitor alarm indicate an increase in radioactivity in the secondary system.

- Continued loss of reactor coolant inventory leads to a reactor trip generated by a low pressurizer pressure or over-temperature ΔT signal. Following reactor trip, the SGTR leads to a decrease in reactor coolant pressure and pressurizer level, counteracted by chemical and volume control system flow and pressurizer heater operation. A safeguards (“S”) signal that provides core makeup tank and PRHR heat exchanger actuation is initiated by low pressurizer pressure or low-2 pressurizer level. The “S” signal automatically terminates the normal feedwater supply and trips the reactor coolant pumps. The power to the pressurizer heaters is also terminated. Startup feedwater flow is initiated on a low steam generator narrow range level signal and controls the steam generator levels to the narrow range low-level setpoint.

- The reactor trip automatically trips the turbine, and if offsite power is available, the steam dump valves open permitting steam dump to the condenser. In the event of a loss of offsite power or loss of the condenser, the steam dump valves automatically close to protect the condenser. The steam generator pressure rapidly increases resulting in steam discharge to the atmosphere through the steam generator power-operated relief valves and/or the safety valves.

- Following reactor trip and core makeup tank and PRHR actuation, the PRHR heat exchanger operation – combined with startup feedwater flow, borated core makeup tank flow, and chemical and volume control system flow – provides a heat sink that absorbs the decay heat. This reduces the amount of steam generated in the steam generators and steam bypass to the condenser. In the case of loss of offsite power, this reduces steam relief to the atmosphere.

- Injection of the chemical and volume control system and core makeup tank flow stabilizes reactor coolant system pressure and pressurizer water level, and the reactor coolant system pressure trends toward an equilibrium value, where the total injected flow rate equals the break flow rate.

15.6.3.1.3 Steam Generator Tube Rupture Automatic Recovery Actions

The AP1000 incorporates several protection system and passive design features that automatically terminate a steam generator tube leak and stabilize the reactor coolant system, in the highly unlikely event that the operators do not perform recovery actions. Following an SGTR, the injecting chemical and volume control system flow (and pressurizer heater heat addition if the pressure control system is operating) maintains the primary-to-secondary break flow and the ruptured steam generator secondary level increases as break flow accumulates in the steam generator. Eventually, the ruptured steam generator secondary level reaches the high-2 steam generator narrow range level setpoint, which is near the top of the narrow range level span.

The AP1000 protection system automatically provides several safety-related actions to cool down and depressurize the reactor coolant system, terminate the break flow and steam release to the atmosphere, and stabilize the reactor coolant system in a safe condition. The safety-related actions include initiation of the PRHR system heat exchanger, isolation of the chemical and volume control system pumps and pressurizer heaters, and isolation of the startup feedwater pumps. In addition, the protection and safety monitoring system provides a safety-related signal to trip the redundant, nonsafety related pressurizer heater breakers.

Actuating the PRHR heat exchanger transfers core decay heat to the in-containment reactor water storage tank (IRWST) and initiates a cooldown (and a consequential depressurization) of the reactor coolant system.

Isolation of the chemical and volume control system pumps and pressurizer heaters minimizes the repressurization of the primary system. This allows primary pressure to equilibrate with the secondary pressure, which effectively terminates the primary-to-secondary break flow. Because the core makeup tank continues to inject when needed to provide boration following isolation of the chemical and volume control system pumps, isolating the chemical and volume control system pumps does not present a safety concern.

Isolation of the startup feedwater provides protection against a failure of the startup feedwater control system, which could potentially result in the ruptured steam generator being overfilled.

With decay heat removal by the PRHR heat exchanger, steam generator steaming through the power-operated relief valves ceases and steam generator secondary level is maintained.

15.6.3.1.4 Steam Generator Tube Rupture Assuming Operator Recovery Actions

In the event of an SGTR, the operators can diagnose the accident and perform recovery actions to stabilize the plant, terminate the primary-to-secondary leakage, and proceed with orderly shutdown of the reactor before actuation of the automatic protection systems. The operator actions for SGTR recovery are provided in the plant emergency operating procedures. The major operator actions include the following:

- Identify the ruptured steam generator – The ruptured steam generator can be identified by an unexpected increase in steam generator narrow range level or a high radiation indication from any main steam line monitor, steam generator blowdown line monitor, or steam generator sample.

- Isolate the ruptured steam generator – Once the steam generator with the ruptured tube is identified, recovery actions begin by isolating steam flow from and stopping feedwater flow to the ruptured steam generator.

- Cooldown of the reactor coolant system using the intact steam generator or the PRHR system – After isolation of the ruptured steam generator, the reactor coolant system is cooled as rapidly as possible to less than the saturation temperature corresponding to the ruptured steam generator pressure. This provides adequate subcooling in the reactor coolant system after depressurization of the reactor coolant system to the ruptured steam generator pressure in subsequent actions.

- Depressurize the reactor coolant system to restore reactor coolant inventory – When the cooldown is completed, the chemical and volume control system and core makeup tank injection flow increases the reactor coolant system pressure until break flow matches the total injection flow. Consequently, these flows must be terminated or controlled to stop primary-to-secondary leakage. However, adequate reactor coolant inventory must first be provided. This includes both sufficient reactor coolant subcooling and pressurizer inventory to maintain a reliable pressurizer level indication after the injection flow is stopped.

Because leakage from the primary side continues after the injection flow is stopped, until reactor coolant system and ruptured steam generator pressures equalize, the reactor coolant system is depressurized to provide sufficient inventory to verify that the pressurizer level remains on span after the pressures equalize.

- Termination of the injection flow to stop primary to secondary leakage – The previous actions establish adequate reactor coolant system subcooling, a secondary side heat sink, and sufficient reactor coolant inventory to verify that injection flow is no longer needed. When these actions are completed, core makeup tank and chemical and volume control system flow is stopped to terminate primary-to-secondary leakage. Primary-to-secondary leakage continues after the injection flow is stopped until the reactor coolant system and ruptured steam generator pressures equalize. Chemical and volume control system makeup flow, letdown, pressurizer heaters, and decay heat removal via the intact steam generator or the PRHR heat exchanger are then controlled to prevent repressurization of the reactor coolant system and reinitiation of leakage into the ruptured steam generator.

Following the injection flow termination, the plant conditions stabilize and the primary-to-secondary break flow terminates. At this time, a series of operator actions is performed to prepare the plant for cooldown to cold shutdown conditions. The actions taken depend on the available plant systems and the plan for further plant repair and operation.

15.6.3.2 Analysis of Effects and Consequences

An SGTR results in the leakage of contaminated reactor coolant into the secondary system and subsequent release of a portion of the activity to the atmosphere. An analysis is performed to demonstrate that the offsite radiological consequences resulting from an SGTR are within the allowable guidelines.

One of the concerns for an SGTR is the possibility of steam generator overfill because this can potentially result in a significant increase in the offsite radiological consequences. Automatic protection and passive design features are incorporated into the AP1000 design to automatically terminate the break flow to prevent overfill during an SGTR. These features include actuation of the PRHR system, isolation of chemical and volume control system flow, and isolation of startup feedwater.

An analysis is performed, without modeling expected operator actions to isolate the ruptured steam generator and cool down and depressurize the reactor coolant system, to demonstrate the role that the AP1000 design features have in preventing steam generator overfill. The limiting single failure for the overfill analysis is assumed to be the failure of the startup feedwater control valve to throttle flow when nominal steam generator level is reached. Other conservative assumptions that maximize steam generator secondary volume (such as high initial steam generator level, minimum initial reactor coolant system pressure, loss of offsite power, maximum chemical and volume control system injection flow, maximum pressurizer heater addition, maximum startup feedwater flow, and minimum startup feedwater delay time) are also assumed.

The results of this analysis demonstrate the effectiveness of the AP1000 protection system and passive system design features and support the conclusion that an SGTR event would not result in steam generator overfill.

For determining the offsite radiological consequences, an SGTR analysis is performed assuming the limiting single failure and limiting initial conditions relative to offsite doses. Because steam generator overfill is prevented for the AP1000, the results of this analysis represent the limiting radiological consequences for an SGTR.

A thermal-hydraulic analysis is performed to determine the plant response for a design basis SGTR, the integrated primary-to-secondary break flow, and the mass releases from the ruptured and intact steam generators to the condenser and to the atmosphere. This information is then used to calculate the radioactivity release to the environment and the resulting radiological consequences.

15.6.3.2.1 Method of Analysis

15.6.3.2.1.1 Computer Program

The plant response following an SGTR until the primary-to-secondary break flow is terminated is analyzed with the LOFTTR2 program (Reference 21). The LOFTTR2 program is modified to model the PRHR system, core makeup tanks, and protection system actions appropriate for the AP1000. These modifications to LOFTTR2 are described in WCAP-14234, Revision 1 (Reference 14).

15.6.3.2.1.2 Analysis Assumptions

The accident modeled is a double-ended break of one steam generator tube located at the top of the tube sheet on the outlet (cold leg) side of the steam generator. The location of the break on the cold leg side of the steam generator results in higher initial primary-to-secondary leakage than a break on the hot side of the steam generator.

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

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

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

The valve is subsequently isolated when the associated block valve is automatically closed on a low steam line pressure protection system signal.

No operator actions are modeled in this limiting analysis, and the plant protection system provides the protection for the plant. Not modeling operator actions is conservative because the operators are expected to have sufficient time to recover from the accident and supplement the automatic protection system. In particular, the operator would take action to reduce the primary pressure before the high-2 steam generator level chemical and volume control system shutoff signal is generated. It is also expected that the operator can close the block valve to the ruptured steam generator power-operated relief valve in much shorter time than the automatic protection signal. The operators can quickly diagnose a power-operated relief valve failure based on the rapid depressurization of the steam generator and increase in steam flow. They can then close the block valve from the control panel.

Consistent with the assumed loss of offsite power, the main feedwater pumps coast down and no startup feedwater is assumed to conservatively minimize steam generator secondary inventory and thus maximize secondary activity concentration and steam release.

15.6.3.2.1.3 Results

The sequence of events for this transient is presented in Table 15.6.3-1. The system responses to the SGTR accident are shown in Figures 15.6.3-1 to 15.6.3-10.

Offsite power is lost concurrent with the rupture of the tube. The reactor trips due to the loss of offsite power. The main feedwater pumps are assumed to coast down following reactor trip. The startup feedwater pumps are conservatively assumed not to start. Following the tube rupture, reactor coolant flows from the primary into the secondary side of the faulted steam generator. In response to this loss of reactor coolant, pressurizer level and reactor coolant system pressure decreases as shown in Figures 15.6.3-1 and 15.6.3-2. As a result of the decreasing pressurizer level and pressure, two chemical and volume control system pumps are automatically initiated to provide makeup flow and the pressurizer heaters turn on.

After reactor trip, core power rapidly decreases to decay heat levels and the core inlet to outlet temperature differential decreases. The turbine stop valves close, and steam flow to the turbine is terminated. The steam dump system is conservatively assumed to be inoperable. The secondary side pressure increases rapidly after reactor trip until the steam generator power-operated relief valves (and safety valves, if their setpoints are reached) lift to dissipate the energy, as shown in Figure 15.6.3-3.

Maximum heat addition to the pressurizer from the pressurizer heaters increases the primary pressure.

As the leak flow continues to deplete primary inventory, low pressurizer level "S" and core makeup tank and PRHR actuation signals are reached. Power to the pressurizer heaters is shut off so that they will not provide additional heat to the primary should the pressurizer level return. The ruptured steam generator power-operated relief valve is assumed to fail open at this time.

The failure causes the intact and ruptured steam generators to rapidly depressurize (Figure 15.6.3-3). This results in an initial increase in primary-to-secondary leakage and a decrease in the reactor coolant system temperatures. Both the intact and ruptured steam generators depressurize because the steam generators communicate through the open steam line isolation valves.

The decrease in the reactor coolant system temperature results in a decrease in the pressurizer level and reactor coolant system pressure (Figures 15.6.3-1 and 15.6.3-2). Depressurization of the primary and secondary systems continues until the low steam line pressure setpoint is reached. As a result, the steam line isolation valves and intact and ruptured steam generator power-operated relief block valves are closed.

Following closure of the block valves, the primary and secondary pressures and the ruptured steam generator secondary water volume and mass increase as break flow accumulates. This increase continues until the steam generator secondary level reaches the high-2 narrow range level when the chemical and volume control system pump is isolated.

With continued reactor coolant system cooldown, depressurization provided by the PRHR heat exchanger, and with the chemical and volume control system isolated, primary system pressure eventually falls to match the secondary pressure. The break flow terminates as shown in Figure 15.6.3-5, and the system is stabilized in a safe condition. As shown in Figure 15.6.3-8, steam release through the intact loop, unfaulted power-operated relief valve does not occur following PRHR initiation because the PRHR is capable of removing the core decay heat.

As shown in Figure 15.6.3-9, the core makeup tank flow trends toward zero because the gravity head diminishes as the core makeup tank temperature approaches the reactor coolant system temperature due to the continued balance line flow. The core makeup tank remains full, and ADS actuation does not occur.

The ruptured steam generator water volume is shown in Figure 15.6.3-6. The water volume in the ruptured steam generator when the break flow is terminated is significantly less than the total steam generator volume of 8868 ft^3.

The design basis SGTR event does not result in fuel failures. In the event of an SGTR, the reactor coolant system depressurizes due to the primary-to-secondary leakage through the ruptured steam generator tube. This depressurization reduces the calculated DNBR. The depressurization prior to reactor trip for the SGTR has been compared to the depressurization for the reactor coolant system depressurization accidents analyzed in subsection 15.6.1. The rate of depressurization is much slower for the SGTR than for the reactor coolant system depressurization accidents. Following reactor trip, the DNBR increases rapidly. Thus, the conclusion of subsection 15.6.1, that the calculated DNBR remains above the limit, is extended to the SGTR analysis, justifying the assumption of no failed fuel.

15.6.3.2.1.4 Mass Releases

The mass release of an SGTR event is determined for use in evaluating the exclusion area boundary and low population zone radiation exposure. The steam releases from the ruptured and intact steam generators and the primary-to-secondary leakage into the ruptured steam generator are determined from the LOFTTR2 results for the period from the initiation of the accident until the leakage is terminated.

Following reactor trip, the releases to the atmosphere are through the steam generator power-operated relief valves (and steam generator safety valves for a short period). Steam relief through the power-operated relief valves continues until the PRHR is automatically initiated and core decay heat is transferred to the IRWST. The mass releases for the SGTR event are presented in Table 15.6.3-2.

15.6.3.3 Radiological Consequences

The evaluation of the radiological consequences of the postulated SGTR assumes that the reactor is operating with the design basis fuel defect level (0.25 percent of power produced by fuel rods containing cladding defects) and that leaking steam generator tubes result in a buildup of activity in the secondary coolant.

Following the rupture, any noble gases carried from the primary coolant into the ruptured steam generator via the break flow are released directly to the environment. The iodine and alkali metal activity entering the secondary side is also available for release, with the amount of release dependent on the flashing fraction of the reactor coolant and on the partition coefficient in the steam generator. In addition to the activity released through the ruptured loop, there is also a small amount of activity released through the intact loop.

15.6.3.3.1 Source Term

The significant radionuclide releases from the SGTR are the noble gases, alkali metals and the iodines that become airborne and are released to the environment as a result of the accident.

The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. In one case, the initial iodine concentrations are assumed to be those associated with the equilibrium operating limits for primary coolant iodine activity. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant.

The second case assumes that the iodine spike occurs before the accident and that the maximum reactor coolant iodine concentration exists at the time the accident occurs.

The reactor coolant noble gas and alkali metal concentrations are assumed to be those associated with the design fuel defect level.

The secondary coolant iodine and alkali metal activity is assumed to be 10 percent of the maximum equilibrium primary coolant activity.

15.6.3.3.2 Release Pathways

The noble gas activity contained in the reactor coolant that leaks into the intact steam generator and enters the ruptured steam generator through the break is assumed to be released immediately as long as a pathway to the environment exists. There are three components to the modeling of iodine and alkali metal releases:

- Intact loop steaming, with credit for partitioning of iodines and alkali metals (includes continued primary-to-secondary leakage at the maximum rate allowable by the Technical Specifications)

- Ruptured loop steaming, with credit for partitioning of iodines and alkali metals (includes modeling of increasing activity in the secondary coolant due to the break flow)

- Release of flashed reactor coolant through the ruptured loop, with no credit for scrubbing (this conservatively assumes that break location is at the top of the tube bundle)

Credit is taken for decay of radionuclides until release to the environment. After release to the environment, no consideration is given to radioactive decay or to cloud depletion of iodines by ground deposition during transport offsite.

15.6.3.3.3 Dose Calculation Models

The models used to calculate doses are provided in Appendix 15A.

15.6.3.3.4 Analytical Assumptions and Parameters

The assumptions and parameters used in the analysis are listed in Table 15.6.3-3.

15.6.3.3.5 Identification of Conservatisms

The assumptions used in the analysis contain a number of significant conservatisms, such as:

- The reactor coolant activities are based on a fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less (see Section 11.1).

- It is unlikely that the conservatively selected meteorological conditions are present at the time of the accident.

15.6.3.3.6 Doses

Using the assumptions from Table 15.6.3-3, the calculated TEDE doses for the case in which the iodine spike is assumed to be initiated by the accident are determined to be less than 1.1 rem at the exclusion area boundary for the limiting 2-hour interval (0-2 hours) and less than 0.8 rem at the low population zone outer boundary. These doses are a small fraction of the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34. A "small fraction" is defined, consistent with the Standard Review Plan, as being ten percent or less.

For the case in which the SGTR is assumed to occur coincident with a pre-existing iodine spike, the TEDE doses are determined to be less than 2.2 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and less than 1.3 rem at the low population zone outer boundary. These doses are within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

At the time the accident occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour exclusion area boundary dose because pool boiling would not occur until after 2.0 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the doses calculated for the steam generator tube rupture, the resulting total doses remain as reported above.

15.6.3.4 Conclusions

The results of the SGTR analysis show that the overfill protection logic and the passive system design features provide protection to prevent steam generator overfill. Following an SGTR accident, the operators can identify and isolate the faulted steam generator and complete the required actions to terminate the primary-to-secondary break flow before steam generator overfill or ADS actuation occurs.

Even when no operator actions are assumed, the AP1000 protection system and passive design features initiate automatic actions that can terminate a steam generator tube leak and stabilize the reactor coolant system in a safe condition while preventing steam generator overfill and ADS actuation.

The resulting offsite radiological doses for the limiting case analyzed are within the dose acceptance limits.

15.6.4 Spectrum of Boiling Water Reactor Steam System Piping Failures Outside of Containment

This section is not applicable to the AP1000.

15.6.5 Loss-of-coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

15.6.5.1 Identification of Causes and Frequency Classification

A LOCA is the result of a pipe rupture of the reactor coolant system pressure boundary. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is considered a Condition IV event (a limiting fault) because it is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis (see subsection 15.0.1).

A minor pipe break (small break), as considered in this subsection, is defined as a rupture of the reactor coolant pressure boundary (Section 5.2) with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This is considered a Condition III event because it is an infrequent fault that may occur during the life of the plant.

The acceptance criteria for the LOCA are described in 10 CFR 50.46 (Reference 1) as follows:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- Localized cladding oxidation shall not exceed 17 percent of the total cladding thickness before oxidation.
- The amount of hydrogen generated from fuel element cladding reacting chemically with water or steam shall not exceed 1 percent of the total amount if all metal cladding were to react.
- The core remains amenable to cooling for any calculated change in core geometry.
- The core temperature is maintained at a low value, and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

These criteria are established to provide significant margin in emergency core cooling system performance following a LOCA.

For the AP1000, the small breaks (less than 1.0 ft^2) yield results with more margin than large breaks.

15.6.5.2 Basis and Methodology for LOCA Analyses

Should a major break occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safeguards actuation ("S") signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

- Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. Insertion of control rods to shut down the reactor is neglected in the large-break analysis.

- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

The acceptability of the computer codes approved for AP600 LOCA analyses for the AP1000 application is documented in Reference 24.

15.6.5.2.1 Description of Large-break LOCA Transient

Before the break occurs, the unit is in an equilibrium condition in which the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay stored energy in the fuel, hot internals, and vessel continues to be transferred to the reactor coolant. At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid, which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break, the core heat transfer is based upon local fluid conditions. Transition boiling and dispersed flow film boiling are the major heat transfer mechanisms.

The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending upon the relative temperatures. In the case of continued heat addition to the secondary, secondary system pressure increases and the main steam safety valves may lift to limit the pressure. The safety injection signal actuates a feedwater isolation signal, which isolates normal feedwater flow by closing the main feedwater isolation valves.

The reactor coolant pumps trip automatically during the accident following an "S" signal. The effects of pump coastdown are included in the blowdown. The blowdown phase of the transient ends when the reactor coolant system pressure (initially assumed at 2250 psia) falls to a value approaching that of the containment atmosphere.

When the "S" signal occurs, the core makeup tank valves in the cold leg pressure balance line are opened. The core makeup tank begins to inject subcooled borated water into the reactor vessel through the direct vessel injection lines.

Subsection 15.6.5.4C presents calculations that show the effective post-LOCA long-term cooling of the AP1000 by passive means.

15.6.5.2.2 Description of Small-break LOCA Transient

The AP1000 includes passive safety features to prevent or minimize core uncovery during small-break LOCAs. The passive safety design approach of the AP1000 is to depressurize the reactor coolant system if the break or leak is greater than the makeup capability of the charging system. By depressurizing the reactor system, large volumes of borated water in the accumulators and in the IRWST become available for cooling the core. This analysis demonstrates that, with a single failure, the passive systems are capable of depressurizing the reactor coolant system while maintaining acceptable core conditions and establishing stable delivery of cooling water from the IRWST.

During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurizer low-pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned for delivery following the generation of an “S” signal when the pressurizer low-pressure setpoint is reached. The passive core cooling system includes two core makeup tanks, two accumulators, a large IRWST, and the PRHR heat exchanger.

The core makeup tanks operate at reactor coolant system pressure. They provide high-pressure safety injection in the event of a small-break LOCA. The core makeup tanks share a common discharge line with the accumulators and IRWST; they are filled with borated water to provide core shutdown margin. The injection of the core makeup tanks is provided by gravity head of the colder water in the core makeup tanks. The core makeup tanks are located above the reactor coolant loops, and each is equipped with a pressure balancing line from a cold leg to the top of the tank.

The pressurized accumulators provide additional borated water to the reactor coolant system in the event of a LOCA. Nominally, these 2000-ft^3 tanks are filled with 1700 ft^3 of water and 300 ft^3 of nitrogen at an initial pressure of 700 psig. Once sufficient reactor coolant system depressurization occurs, either as a result of a LOCA or the actuation of the ADS, accumulator injection commences.

The IRWST provides an additional source of water for long-term core cooling. To attain injection from the IRWST, the reactor coolant system pressure must be lowered to approximately 13 psi above containment pressure. For this pressure to be achieved during a small-break LOCA, the ADS system is initiated.

The ADS consists of a series of valves, connected to the pressurizer and hot legs, which provide a phased depressurization of the reactor coolant system. As the reactor system loses inventory through the break, the core makeup tanks provide flow to the reactor vessel. When the level in the core makeup tank drops to the 67.5-percent level, the ADS valves open to accelerate the reactor coolant system depressurization rate. The ADS Stage 1 4-inch valves open at the 67.5-percent level; the 8-inch Stage 2 and the 8-inch Stage 3 valves open in a timed sequence thereafter. The flow from the first three stages of the ADS is discharged into the IRWST through a sparger system. The fourth stages of the ADS are connected to the reactor coolant system hot legs and discharge to containment atmosphere. The ADS Stage 4 valves are activated when the core makeup tank level reaches the 20-percent level.

As the reactor coolant system depressurizes and mass is lost out the break, mass is added to the reactor vessel from the core makeup tanks and the accumulators. When the system is depressurized below the IRWST delivery pressure, flow from the IRWST continues to maintain the core in a coolable state. Calculations described in subsection 15.6.5.4B indicate that acceptable core cooling is provided for the small-break LOCA transients. Subsection 15.6.5.4C calculations show that effective post-LOCA core cooling is provided in the long term by passive means.

15.6.5.3 Radiological Consequences

Although the analysis of the core response during a LOCA (see subsection 15.6.5.4) shows that core integrity is maintained, for the evaluation of the radiological consequences of the accident, it is assumed that major core degradation and melting occur.

The dose calculations take into account the release of activity by way of the containment purge line prior to its isolation near the beginning of the accident and the release of activity resulting from containment leakage. Purge of the containment for hydrogen control is not an intended mode of operation and is not considered in the dose analysis. While the normal residual heat removal system is capable of post-LOCA cooling, it is not a safety-related system and may not be available following the accident. If it is operable, it would be used only if the source term is not far above the normal shutdown primary coolant source term. It is assumed that core cooling is accomplished by the passive core cooling system, which does not pass coolant outside of containment. Thus, there is no recirculation leakage release path to be modeled.

15.6.5.3.1 Source Term

The release of activity to the containment consists of two parts. The initial release is the activity contained in the reactor coolant system. This is followed by the release of core activity.

15.6.5.3.1.1 Primary Coolant Release

The reactor coolant is assumed to have activity levels consistent with operation at the Technical Specification limits of 280 μCi/gm dose equivalent Xe-133 and 1.0 μCi/gm dose equivalent I-131.

Based on NUREG-1465 (Reference 19), for a plant using leak-before-break methodology, the release of coolant into the containment can be assumed to last for 10 minutes. The AP1000 is a leak-before-break plant, and the water in the reactor coolant system is assumed to blow down into the containment over a period of 10 minutes. The flow rate is assumed to be constant over the 10-minute period. As the reactor coolant enters the containment, the noble gases and half of the iodine activity are assumed to be released into the containment atmosphere.

15.6.5.3.1.2 Core Release

The release of activity from the fuel takes place in two stages as summarized in Table 15.6.5-1. First is the gap release which is assumed to occur at the end of the primary coolant release phase (i.e., at ten minutes into the accident) and continue over a period of half an hour. The second stage is that of the in-vessel core melt in which the bulk of the activity releases associated with the accident occur. The source term model is based on NUREG-1465 and Regulatory Guide 1.183 (Reference 20).

The core fission product inventory at the time of the accident is based on operation near the end of a fuel cycle at 102-percent power and is provided in Table 15A-3 of Appendix 15A. Consistent with NUREG-1465, there are three groups of nuclides considered in the gap activity releases: noble gases, iodines, and alkali metals (cesium and rubidium). For the core melt phase, there are five additional nuclide groups for a total of eight. The five additional nuclide groups are the tellurium group, the noble metals group, the cerium group, the lanthanide group, and barium and strontium. The specific nuclides included in the source term are as shown in Table 15A-3.

Gap Activity Release

Consistent with NUREG-1465 guidance for a plant using leak-before-break methodology, the gap release phase begins after the primary coolant release phase ends at ten minutes and has a duration of 0.5 hour.

In-vessel Core Release

After the gap activity release phase, there is an in-vessel release phase which lasts for 1.3 hours and which releases activity to the containment due to core melting. The fractions of the core activity released to the containment atmosphere during this phase are from NUREG-1465:

Noble gases	0.95
Iodines	0.35
Alkali metals	0.25
Tellurium group	0.05
Noble metals	0.0025
Ba and Sr	0.02
Cerium group	0.0005
Lanthanide group	0.0002

Consistent with NUREG-1465, the releases are assumed to occur at a constant rate over the 1.3-hour phase duration.

15.6.5.3.1.3 Iodine Form

The iodine form is consistent with the NUREG-1465 model. The model shows the iodine to be predominantly in the form of nonvolatile cesium iodide with a small fraction existing as elemental iodine. Additionally, the model assumes that a portion of the elemental iodine reacts with organic materials in the containment to form organic iodine compounds. The resulting iodine species split is as follows:

- Particulate 0.95
- Elemental 0.0485
- Organic 0.0015

If the post-LOCA cooling solution has a pH of less than 6.0, part of the cesium iodide may be converted to the elemental iodine form. The passive core cooling system provides sufficient trisodium phosphate to the post-LOCA cooling solution to maintain the solution pH at 7.0 or greater following a LOCA (see subsection 6.3.2.1.4).

15.6.5.3.2 In-containment Activity Removal Processes

The AP1000 does not include active systems for the removal of activity from the containment atmosphere. The containment atmosphere is depleted of elemental iodine and of particulates as a result of natural processes within the containment.

Elemental iodine is removed by deposition onto surfaces. Particulates are removed by sedimentation, diffusiophoresis (deposition driven by steam condensation), and thermophoresis (deposition driven by heat transfer). No removal of organic iodine is assumed. Appendix 15B provides a discussion of the models and assumptions used in calculating the removal coefficients.

15.6.5.3.3 Release Pathways

The release pathways are the containment purge line and containment leakage. The activity releases are assumed to be ground level releases.

During the initial part of the accident, before the containment is isolated, it is assumed that containment purge is in operation and that activity is released through this pathway until the purge valves are closed. No credit is taken for the filters in the purge exhaust line.

The majority of the releases due to the LOCA are the result of containment leakage, with credit for aerosol removal from containment cracks that form the leakage paths, as discussed in Reference 31. The containment is assumed to leak at its design leak rate for the first 24 hours and at half that rate for the remainder of the analysis period.

15.6.5.3.4 Offsite Dose Calculation Models

The offsite dose calculation models are provided in Appendix 15A. The models address the determination of the TEDE doses from the combined acute doses and the committed effective dose equivalent doses.

The exclusion area boundary dose is calculated for the 2-hour period over which the highest doses would be accrued by an individual located at the exclusion area boundary. Because of the delays associated with the core damage for this accident, the first 2 hours of the accident are not the worst 2-hour interval for accumulating a dose.

The low population zone boundary dose is calculated for the nominal 30-day duration of the accident.

For both the exclusion area boundary and low population zone dose determinations, the calculated doses are compared to the dose guideline of 25 rem TEDE from 10 CFR Part 50.34.

15.6.5.3.5 Main Control Room Dose Model

There are two approaches that may be used for modeling the activity entering the main control room. If power is available, the normal heating, ventilation, and air-conditioning (HVAC) system will switch over to a supplemental filtration mode (Section 9.4). The normal HVAC system is not a safety-class system but provides defense in depth.

Alternatively, if the normal HVAC is inoperable or, if operable, the supplemental filtration train does not function properly resulting in increasing levels of airborne iodine in the main control room, the emergency habitability system (Section 6.4) would be actuated when high iodine activity is detected. The emergency habitability system provides passive pressurization of the main control room from a bottled air supply to prevent inleakage of contaminated air to the main control room. There is a 72-hour supply of air in the emergency habitability system. After this time, the main control room is assumed to be opened and unfiltered air is drawn into the main control room by way of an ancillary fan. After 7 days, offsite support is assumed to be available to reestablish operability of the control room habitability system by replenishing the compressed air supply. As a defense-in-depth measure, the nonsafety-related normal control room HVAC would be brought back into operation with the supplemental filtration train if power is available.

The main control room is accessed by a vestibule entrance, which restricts the volume of contaminated air that can enter the main control room from ingress and egress. The design of the emergency habitability system (VES) provides 65 cfm ±5 cfm to the control room and maintains it in a pressurized state. The path for the purge flow out of the main control room is through the vestibule entrance and this results in a dilution of the activity in the vestibule and a reduction in the amount of activity that might enter the main control room. Without this purge through the vestibule, the project unfiltered inleakage into the main control room is 5 cfm. However, the impact of the purge flow is to reduce the effective unfiltered inleakage rate to 2.654 cfm. Additionally, during the first 24 hours following the LOCA, personnel entering the control room will be required to wait inside the vestibule for a short period of time until the activity concentration is reduced by a factor of five or more. This reduces the effective unfiltered inleakage rate to 0.531 cfm. Conservatively, assuming a purge flow of only 55 cfm through the vestibule, the factor of five reduction in activity concentration would be achieved in less than 11 minutes.

Activity entering the main control room is assumed to be uniformly dispersed. No credit is taken for the removal of airborne activity in the main control room although elemental iodine and particulates would be removed by deposition and sedimentation.

The main control room dose calculation models are provided in Appendix 15A for the determination of doses resulting from activity which enters the main control room envelope.

15.6.5.3.6 Analytical Assumptions and Parameters

The analytical assumptions and parameters used in the radiological consequences analysis are listed in Table 15.6.5-2.

15.6.5.3.7 Identification of Conservatisms

The LOCA radiological consequences analysis assumptions include a number of conservatisms. Some of these conservatisms are discussed in the following subsections.

15.6.5.3.7.1 Primary Coolant Source Term

The source term is based on operation with the design fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less.

15.6.5.3.7.2 Core Release Source Term

The assumed core melt is a major conservatism associated with the analysis. In the event of a postulated LOCA, no major core damage is expected. Release of activity from the core is limited to a fraction of the core gap activity.

15.6.5.3.7.3 Atmospheric Dispersion Factors

The atmospheric dispersion factors assumed to be present during the course of the accident are conservatively selected. Actual meteorological conditions are expected to result in significantly higher dispersion of the released activity.

15.6.5.3.8 LOCA Doses

15.6.5.3.8.1 Offsite Doses

The doses calculated for the exclusion area boundary and the low population zone boundary are listed in Table 15.6.5-3. The doses are within the 10 CFR 50.34 dose guideline of 25 rem TEDE.

The reported exclusion area boundary doses are for the time period of 1.2 to 3.2 hours. This is the 2-hour interval that has the highest calculated doses. The dose that would be incurred over the first 2 hours of the accident is well below the reported dose.

At the time the LOCA occurs, there is the potential for a coincident loss of spent fuel pool cooling with the result that the pool could reach boiling and a portion of the radioactive iodine in the spent fuel pool could be released to the environment. The loss of spent fuel pool cooling has been evaluated for a duration of 30 days. There is no contribution to the 2-hour site boundary dose because pool boiling would not occur until after 8 hours. The 30-day contribution to the dose at the low population zone boundary is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

15.6.5.3.8.2 Doses to Operators in the Main Control Room

The doses calculated for the main control room personnel due to airborne activity entering the main control room are listed in Table 15.6.5-3. Also listed on Table 15.6.5-3 are the doses due to direct shine from the activity in the adjacent buildings and sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-scattering. The total of the three dose paths is within the dose criteria of 5 rem TEDE as defined in GDC 19.

As discussed above for the offsite doses, there is the potential for a dose to the operators in the main control room due to iodine releases from postulated spent fuel boiling. The calculated dose from this source is less than 0.01 rem TEDE and, when this is added to the dose calculated for the LOCA, the resulting total dose remains less than that reported in Table 15.6.5-3.

15.6.5.4 Core and System Performance

Subsection 15.6.5.4A describes the large-break LOCA analysis methodology and results. Subsections 15.6.5.4B.1.0 through 15.6.5.4B.4.0 describe the small-break LOCA analysis methodology and results.

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.

The post-LOCA long-term core cooling and core boron concentration analyses discussed in subsection 15.6.5.4C are applicable to the large-break LOCA transient.

15.6.5.4A.1 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 nodalization 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, the 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.

15.6.5.4A.2 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.

15.6.5.4A.3 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.

15.6.5.4A.4 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 backpressure 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.

15.6.5.4A.5 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 indicates 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. Further, local and core-wide cladding oxidation values have been determined using the Reference 10 approved methodology.

Figures 15.6.5.4A-1 through 15.6.5.4A-12 present the parameters of principal interest for the Reference Case DECLG break analysis. Values of the following parameters are presented:

- The highest calculated cladding 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.

15.6.5.4A.6 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 15.6.5.4A-1 and 15.6.5.4A-2) during the flow reversal (Figure 15.6.5.4A-3). In Figure 15.6.5.4A-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 15.6.5.4A-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 that 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 15.6.5.4A-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 15.6.5.4A-5). Accumulator actuation shuts off core makeup tank flow (Figure 15.6.5.4A-6), which has been delivering since the isolation valve opened. The 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 15.6.5.4A-7 through 15.6.5.4A-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 15.6.5.4A-3). The variables plotted are the same as those in Figure 15.6.5.4A-3 for the respective assemblies.

Figures 15.6.5.4A-8 and 15.6.5.4A-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 15.6.5.4A-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 15.6.5.4A-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 15.6.5.4A-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 15.6.5.4A-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 15.6.5.4A-1), although effective cooling is achieved.

Figure 15.6.5.4A-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 cladding 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 15.6.5.4A-10) and vessel (Figure 15.6.5.4A-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 15.6.5.4A-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.

15.6.5.4A.7 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.

15.6.5.4A.8 Large-Break LOCA Conclusions

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 core cooling system provides effective core cooling following a large-break LOCA event, even assuming the limiting single failure of a core makeup tank delivery line isolation valve. The large-break LOCA transient has been extended beyond fuel rod quench until 1800 seconds, a time at which the CMT liquid level has decreased to the low-2 setpoint that actuates the fourth-stage ADS valves and IRWST injection. A significant increase in safety injection flow rate occurs when the IRWST becomes active. The analysis performed demonstrates that CMT injection is sufficient to maintain the mass inventory in the core and downcomer, from the period of fuel rod quench until IRWST injection. The AP1000 passive core cooling system provides effective post-LOCA long-term core cooling.

Table 15.6.5-8 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.

15.6.5.4B Small-break LOCA Analyses

Should a small break LOCA occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal occurs when the pressurizer low-pressure trip setpoint is reached. An "S" signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

- Reactor trip leads to a rapid reduction of power to a residual level corresponding to fission product decay heat by the insertion of control rods to shut down the reactor.

- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

15.6.5.4B.1 Description of Small-break LOCA Transient

The AP1000 plant design includes passive safety features to prevent or minimize core uncovery during small-break LOCAs. The passive safety design approach of the AP1000 is to depressurize the reactor coolant system if the break or leak is greater than the capability of the makeup system or if the nonsafety makeup system fails to perform. By depressurizing the reactor system, large volumes of borated water in the accumulators and in the IRWST become available for cooling the core. This analysis demonstrates that, with a single failure, the passive systems are capable of depressurizing the reactor coolant system while maintaining acceptable core conditions and establishing stable delivery of cooling water from the IRWST.

During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurizer low-pressure setpoint, actuating a reactor trip signal. The passive core cooling system is aligned for delivery following the generation of an "S" signal when the pressurizer low-pressure setpoint is reached. The passive core cooling system includes two core makeup tanks, two accumulators, a large IRWST, and the PRHR heat exchanger.

The core makeup tanks operate at reactor coolant system pressure. They provide high-pressure safety injection in the event of a small-break LOCA. The core makeup tanks share a common discharge line with the accumulators and IRWST; they are filled with borated water to provide core shutdown margin. Gravity head of the colder water in the core makeup tanks provides the injection of the core makeup tanks. The core makeup tanks are located above the reactor coolant loops, and each is equipped with a pressure balancing line from a cold leg to the top of the tank.

The pressurized accumulators provide additional borated water to the reactor coolant system in the event of a LOCA. Nominally, these 2000-ft^3 tanks are filled with 1700 ft^3 of water and 300 ft^3 of nitrogen at an initial pressure of 700 psig. Once sufficient reactor coolant system depressurization occurs, either as a result of a LOCA or the actuation of the ADS, accumulator injection begins.

The IRWST at a minimum provides an additional 73,900 ft^3 of water for long-term core cooling. To attain injection from the IRWST, the reactor coolant system pressure must be lowered to approximately 13 psi above containment pressure. For this pressure to be achieved during a small-break LOCA, the actuation of the ADS valves is required.

The ADS consists of a series of valves, connected to the pressurizer and hot legs, which provide a phased depressurization of the reactor coolant system. As the reactor system loses inventory through the break, the core makeup tanks provide flow to the reactor vessel. When the level in the core makeup tank drops to the 67.5-percent level, the ADS valves open to accelerate the reactor coolant system depressurization rate. The ADS Stage 1 4-inch valves open at the 67.5-percent level; the 8-inch Stage 2 and the 8-inch Stage 3 valves open in a timed sequence thereafter. The flow from the first three stages of the ADS is discharged into the IRWST through a sparger system. The fourth stages of the ADS are connected to the reactor coolant system hot legs and discharge to containment atmosphere. The ADS Stage 4 valves are activated when the core makeup tank level reaches the 20-percent level.

As the reactor system depressurizes and mass is lost out the break, mass is added to the reactor vessel from the core makeup tanks and the accumulators. When the system is depressurized below the IRWST delivery pressure, flow from the IRWST continues to maintain the core in a coolable state. Calculations described in this section indicate that acceptable core cooling is provided for the small-break LOCA transients.

15.6.5.4B.2 Small-break LOCA Analysis Methodology

Small-break LOCA response is evaluated for AP1000 with an evaluation model that conforms to 10 CFR 50 Appendix K. The elements of the AP1000 small-break LOCA evaluation model are the following:

- NOTRUMP computer code
- NOTRUMP homogeneous sensitivity model
- Critical heat flux assessment during accumulator injection

15.6.5.4B.2.1 NOTRUMP Computer Code

The NOTRUMP computer code is used in the analysis of LOCAs due to small-breaks in the reactor coolant system. The NOTRUMP computer code is a one-dimensional, general network code, which includes a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The version of NOTRUMP used in AP1000 small-break LOCA calculations has been validated against applicable passive plant test data (Reference 22). The code has limited capability in modeling upper plenum and hot leg entrainment and did not predict the core collapsed level during the accumulator injection phase adequately. The NOTRUMP homogeneous sensitivity model (discussed in subsection 15.6.5.4B.2.2) and the critical heat flux assessment during the accumulator injection phase (discussed in subsection 15.6.5.4B.2.3) supplement the base NOTRUMP analysis to demonstrate the adequacy of the design.

In NOTRUMP, the reactor coolant system is nodalized into volumes interconnected by flow paths. The transient behavior of the system is determined from the governing conservation equations of mass, energy, and momentum applied throughout the system. A description of NOTRUMP is given in References 12 and 13. The AP600 modeling approach, described in Reference 17, is also used to develop the AP1000 model; NOTRUMP's applicability to AP1000 is documented in Reference 24.

The use of NOTRUMP in the analysis involves the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables an explicit and detailed spatial representation of various system components. Table 15.6.5-9 lists important input parameters and initial conditions of the analysis.

A steady-state input deck for the AP1000 was set up to comply, where appropriate, with the standard small-break LOCA Evaluation Model methodology. Major features of the modeling of the AP1000 follow:

- Accumulators are modeled at an initial pressure of 715 psia.

- The flow through the ADS links is modeled using the Henry-Fauske, the homogeneous equilibrium (HEM), and the Murdock/Baumann critical flow models. The Henry-Fauske correlation is used for low-quality two-phrase flow, and the HEM model, for high-quality flow, with a transition between the two beginning at 10-percent static quality. The Murdock-Bauman model is used if the ADS flow path is venting superheated steam.

- Isolation and check valves used in the passive safety systems are modeled.

- The IRWST is modeled as two connected fluid nodes. The lower node is connected to the direct vessel injection line and is the source of injection water to the DVI lines driven by gravity head. The upper node acts as a sink for the ADS flow from the pressurizer and as a heat sink for the PRHR heat exchanger. These nodes are modeled as having an initial temperature of 120°F, a pressure of 14.7 psia, and the nominal full-power operation level of 28.8 feet. Therefore, the minimum head for IRWST injection is assumed. For the DEDVI simulations, a conservative 20 psia containment pressure was used based on containment pressurization calculations performed with the WGOTHIC containment model.

- The PRHR system is modeled in accordance with the guidance provided in References 22 and 24. The PRHR isolation valve is modeled as opening with the maximum delay after the generation of an "S" signal to conservatively deny the cooling capability of the heat exchanger to the reactor coolant system for an extended period.

- The core power is initially set to 102 percent of the nominal core power. The reactor trip signal occurs when the pressurizer pressure falls below 1800 psia. A conservative delay time is modeled between the reactor trip signal and reactor trip. Decay heat is modeled according to the ANS-1971 (Reference 2) standard, with 20-percent uncertainty added.

- The "S" signal is generated when the pressurizer pressure falls below 1700 psia. The isolation valves on the core makeup tank injection lines begin to open after the signal setpoint is reached; the valves are then assumed to open linearly. The main feedwater isolation valves are ramped closed between 2 and 7 seconds after the "S" signal. The reactor coolant pumps are tripped 6.0 seconds after the "S" signal.

- The ADS actuation signals are generated on low core makeup tank levels and the ADS timer delays. A list of the ADS parameters is given in Table 15.6.5-10 for AP1000. ADS Stages 1, 2, and 3 are modeled as discharging through spargers submerged in the IRWST at the appropriate depth.

- The pressure in the boundary node modeling of the containment is 14.7 psia in all NOTRUMP cases except the DEDVI line break, which used 20.0 psia.

- The steam generator secondary is isolated 6 seconds after the reactor trip signal, due to closure of the turbine stop valves. The main steam safety valves actuate and remove energy from the steam generator secondary when pressure reaches 1235 psia.

Active single failures of the passive safeguards systems are considered. The limiting failure is judged to be one out of four ADS Stage 4 valves failing to open on demand, the failure that most severely impacts depressurization capability. The safety design approach of the AP1000 is to depressurize the reactor coolant system to the containment pressure in an orderly fashion such that the large reservoir of water stored in the IRWST is available for core cooling. The mass inventory plots provided for the breaks show the minimum inventory condition generally occurs at the start of IRWST injection. Penalizing the depressurization is the most conservative approach in postulating the single failure for such breaks.

The small-break LOCA spectrum analyzed for AP1000 includes a break that exhibits a minimum reactor vessel inventory early in the transient, before the accumulators become active: the 10-inch cold leg break. In this transient, the early mass inventory decrease is terminated by injection flow from the intact accumulator, and depressurization through the break enables accumulator injection to begin with no contribution from the actuation of ADS Stages 1, 2, and 3. For consistency, the conservative failure of one of the ADS Stage 4 valves located off the PRHR inlet pipe, which adversely affects the depressurization necessary to achieve IRWST injection in small-break LOCAs, is assumed in all cases. Sensitivity analysis shows that assuming failure of one ADS Stage 4 valve on the non-PRHR loop does not significantly impact core cooling.

15.6.5.4B.2.1.1 AP1000 Model-Detailed Noding

Refer to Reference 17 for details of the AP600 NOTRUMP modeling. The AP1000 model was developed in the same fashion with modifications to the AP600 model introduced as follows. A modification performed for AP1000 was the addition of two core nodes one foot each in length to reflect the added active fuel length of this design. The ADS-4 flow path resistances were increased to accommodate shortcomings in NOTRUMP identified during the integral test facility simulations, namely, the lack of a detailed momentum flux model in the ADS-4 discharge paths. A detailed calculation of the energy and momentum equations is performed for the ADS-4 piping over a range of flow and pressure conditions to provide a benchmark for the NOTRUMP ADS-4 flow path resistance. The methodology used to determine the resistance increase is described in Reference 24. By increasing the ADS-4 resistances, the onset of IRWST injection is more appropriately calculated. This methodology directly addresses the effect of momentum flux in ADS-4. The ADS-4 resistance increase utilized is computed for the NOTRUMP analyses in this section to be a 70 percent ADS-4 flow path resistance increase.

15.6.5.4B.2.1.2 Plant Initial Conditions/Steady-State

A steady-state calculation is performed prior to initiating the transient portion of the calculation.

Table 15.6.5-9 contains the most important initial conditions for the transient calculations. The behaviors of the primary pressure and pressurizer level, steam generator pressures, and the core flow rate are stable at the end of the 100-second steady-state calculation.

15.6.5.4B.2.2 NOTRUMP Homogeneous Sensitivity Model

In order to address the uncertainties associated with entrainment in the upper plenum and hot leg following ADS-4 operation, a sensitivity study is performed with the limiting break with respect to these phenomena, effectively maximizing the amount of entrainment downstream of the core. This methodology is described and the results are presented for the double-ended direct vessel injection (DEDVI) line break in detail in Reference 24.

[*In order to maximize the entrainment downstream of the core for the limiting break with respect to entrainment, NOTRUMP is run with the regions of the upper plenum, hot leg, and ADS-4 lines in a homogeneous fluid condition, with slip = 1, to demonstrate that even with maximum entrainment, the 10 CFR 50.46 criteria are met.*]*

15.6.5.4B.2.3 Critical Heat Flux Assessment During Accumulator Injection

[*An assessment is performed of the peak core heat flux with respect to the critical heat flux during the later ADS depressurization time period for a double-ended rupture of the direct vessel injection line. This time period corresponds to the accumulator injection phase of the transient. The predicted average mass flux at the core inlet and the reactor pressure from the NOTRUMP computer code base model analysis are used as input parameters to critical heat flux correlation as described in Reference 30. The requirements of 10 CFR 50.46 are met provided the maximum heat flux is less than the critical heat flux calculated by the correlation.*]* NOTRUMP has been shown (Reference 24) to adequately predict mass flux and pressure for integral systems tests.

The predicted mass flux at the core inlet is on the average constant and corresponds to 7.2 lbm ft^{-2} s^{-1} (~35 kg m^{-2} s^{-1}). The key thermal-hydraulic parameters at different times during the ADS depressurization time period are summarized in following table.

Time (sec)	UP Pressure (kPa)	UP Pressure (psia)	Mass Flux (kg/m^2s)	Average Heat Flux (kW/m^2)
400	1293	190	35	20.2
500	646	95	35	19.1
570	340	50	35	18.5
600	272	40	35	18.2

* NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

For the critical heat flux assessment, the peak core heat flux is applied to simulate the hot assembly condition in a conservative manner. No credit is taken for increased flow in the hot assembly that is known to occur in rod bundles.

The correlation applied for this assessment is from vertical tube data (Reference 30) and recognizes two regimes depending on the mass flux. The main difference between the two is the mass flux dependence. They are as follows:

$$q^*_{CL} = q^*_{CF} + 0.01351\left(D^*\right)^{-0.473}\left(L/D\right)^{-0.533}\left|G^*\right|^{1.45} \quad \text{for low } G^*$$

and,

$$q^*_{CH} = q^*_{CF} + 0.05664\left(D^*\right)^{-0.247}\left(L/D\right)^{-0.501}\left|G^*\right|^{0.77} \quad \text{for high } G^*$$

The first term of above correlations is,

$$q^*_{CF} = 1.61\left(\frac{A}{Ah}\right)\frac{\left(D^*\right)^{0.5}}{\left[1+\left(\frac{\rho_g}{\rho_l}\right)^{0.25}\right]^2}$$

where A is the flow area and Ah is the heated area.

The dimensionless CHF is calculated as,

$$q^*_{CHF} = \min(q^*_{CL}, q^*_{CH})$$

Dimensionless CHF, G, and D are defined as,

$$q^*_{CHF} = \frac{q''_{CHF}}{h_{fg}\sqrt{\lambda\rho_g g\Delta\rho}}$$

$$G^* = \frac{G}{\sqrt{\lambda\rho_g g\Delta\rho}}$$

$$D^* = \frac{D}{\lambda}$$

where λ is the length scale of the Taylor instability:

$$\lambda = \sqrt{\frac{\sigma}{g\Delta\rho}}$$

Conservative application of this correlation with the AP1000 parameters indicates that the peak AP1000 heat flux during this period is at least 40 percent below the predicted critical heat flux.

This CHF assessment addresses core cooling during a time period where the NOTRUMP computer code may not conservatively predict the core average void fraction. The requirements of 10 CFR 50.46 are met during this period since this CHF assessment indicates peak core heat flux is less than critical heat flux. Cladding temperatures will remain near the coolant saturation temperature, well below the 10 CFR 50.46 peak cladding temperature limit.

15.6.5.4B.3 Small-break LOCA Analysis Results

Several small-break LOCA transients are analyzed using NOTRUMP, and the results of these calculations are presented. The results demonstrate that the minimum reactor coolant system mass inventory condition occurs for the relatively large system pipe breaks. Smaller breaks exhibit a greater margin-to-core uncovery.

15.6.5.4B.3.1 Introduction

The small-break LOCA safety design approach for AP1000 is to provide for a controlled depressurization of the primary system if the break cannot be terminated, or if the nonsafety-related charging system is postulated to be lost or cannot maintain acceptable plant conditions. Nonsafety-related systems are not modeled in this design basis analysis; the testing conducted in the SPES-2 facility has indicated that the mass inventory condition during small LOCAs is significantly improved when these nonsafety-related systems operate. The core makeup tank level activates primary system depressurization. The core makeup tank provides makeup to help compensate for the postulated break in the reactor coolant system. As the core makeup tank level drops, Stages 1 through 4 of the ADS valves are ramped open in sequence. The ADS valve descriptions for the AP1000 plant design are presented in Table 15.6.5-10. The reactor coolant system depressurizes due to the break and the ADS valves, while subcooled water from the core makeup tanks and accumulators enters the reactor vessel downcomer to maintain system inventory and keep the core covered. Design basis maximum values of passive core cooling system resistances are applied to obtain a conservative prediction of system behavior during the small LOCA events.

During controlled depressurization via the ADS, the accumulators and core makeup tanks maintain system inventory for small-break LOCAs. Once the reactor coolant system depressurizes, injection from the IRWST maintains long-term core cooling. For continued injection from the IRWST, the reactor coolant system must remain depressurized. To conservatively model this condition, design maximum resistance values are specified for the IRWST delivery lines.

A series of small-break LOCA calculations are performed to assess the AP1000 passive safety system design performance. In these calculations, the decay heat used is the ANS-1971

(Reference 2) plus 20 percent for uncertainty as specified in 10 CFR 50, Appendix K (Reference 1). This maximizes the core steam generation to be vented. The breaks analyzed in this document include the following:

Inadvertent ADS Actuation

A "no-break" small-break LOCA calculation that uses an inadvertent opening of the 4-inch nominal size ADS Stage 1 valves is a situation that minimizes the venting capability of the reactor coolant system. Only the ADS valve vent area is available; no additional vent area exists due to a break. This case examines whether sufficient vent area is available to completely depressurize the reactor coolant system and achieve injection from the IRWST without core uncovery. The worst single failure for this situation is a failure of one of four ADS Stage 4 valves connected to either of the two hot legs. The ADS Stage 4 valve is the largest ADS valve, and it vents directly to the containment with no additional backpressure from the spargers being submerged in the IRWST.

2-inch Break in a Cold Leg with Core Makeup Tank Balance Line Connections

The small size of the break leads to a long period of recirculatory flow from the cold leg into the core makeup tank. This delays the formation of a vapor space in the core makeup tank and therefore the actuation of the ADS.

Double-ended Rupture of the Direct Vessel Injection Line

The injection line break evaluates the ability of the plant to recover from a moderately sized break with only half of the total emergency core cooling system capacity available. The vessel side of the break of the DEDVI line break is 4 inches in equivalent diameter. The double-ended nature of this break means that there are effectively two breaks modeled:

- Downcomer to containment. The direct vessel injection nozzle includes a venturi, which limits the available break area.

- Direct vessel injection line into containment from the cold leg balance line and the broken loop core makeup tank.

The containment pressure was conservatively assumed to pressurize to 20 psia. This pressure was selected based on iterative execution of the NOTRUMP and WGOTHIC codes. The NOTRUMP code provides the mass and energy releases from the AP1000 DEDVI break to the AP1000 WGOTHIC containment model while the WGOTHIC code calculates the containment pressure response. The containment pressure assumed in the NOTRUMP simulations was conservatively selected from the generated pressure history curves obtained from the WGOTHIC runs.

The peak core heat flux during the accumulator injection period is assessed relative to the predicted critical heat flux as discussed in subsection 15.6.5.4B.2.3.

An additional injection line break case is analyzed assuming containment pressure is at 14.7 psia.

Double-ended Rupture of the Direct Vessel Injection Line Entrainment Sensitivity

The sensitivity case is performed to assess the effect of higher than expected entrainment in the upper plenum and hot legs on the overall system response and core cooling.

10-inch Cold Leg Break

This break models a break size that approaches the upper limit size for small-break LOCAs.

15.6.5.4B.3.2 Transient Results

The transient results are presented in tables and figures for the key AP1000 parameters of interest in the following sections.

15.6.5.4B.3.3 Inadvertent Actuation of Automatic Depressurization System

An inadvertent ADS signal is spuriously generated and the 4-inch ADS valves open. The plant, which is operating at 102-percent power, is depressurized via the ADS alone. Only safety-related systems are assumed to operate in this and other small-break LOCA cases. Additional ADS valves open; after a 70-second delay, the ADS Stage 2 8-inch valves open, and after an additional 120 seconds, the ADS Stage 3 valves open. At the 20-percent core makeup tank level, the ADS Stage 4A valve, which is connected to the hot leg, receives a signal to open. After a 60-second delay, both Stage 4B valves (one connected to the hot leg and the other connected to the PRHR inlet pipe) open. The path that fails to open as the assumed single active failure is the Stage 4A valve off the PRHR inlet pipe. The reactor steady-state initial conditions assumed can be found in Table 15.6.5-9. The sequence of events for the transient is given in Table 15.6.5-11.

The transient is initiated by the opening of the two ADS Stage 1 paths. Reactor trip, reactor coolant pump trip, and safety injection signals are generated via pressurizer low-pressure signals with appropriate delays. After generation of the reactor trip signal, the turbine stop valves begin to close. The main feedwater isolation valves begin to close 2 seconds after the “S” signal pressure setpoint is reached. The opening of the ADS valves and the reduction in core power due to reactor trip causes the primary pressure to fall rapidly (Figure 15.6.5.4B-1). Flow of fluid toward the open ADS paths causes the pressurizer to fill rapidly (Figure 15.6.5.4B-2), and the ADS flow becomes two-phase (Figures 15.6.5.4B-3 and -4). The safety injection signal opens the valves isolating the core makeup tanks and circulation of cold water begins (Figures 15.6.5.4B-5 and -6). The mixture level (Figures 15.6.5.4B-7 and -8) in the core makeup tanks is relatively constant until the accumulators inject (Figures 15.6.5.4B-10 and -11). The reactor coolant pumps begin to coast down due to an automatic trip signal following a 6.0-second delay.

Continued mass flow through the ADS Stage 1, 2, and 3 valves drains the upper parts of the circuit. The steam generator tube cold leg sides start to drain, followed by the drop in mixture levels in the hot leg sides. As the ADS Stage 2 and 3 paths begin to open, increased ADS flow causes the primary pressure to fall rapidly (Figure 15.6.5.4B-1). Following the emptying of the steam generator tube cold leg sides, the cold legs have drained and a mixture level forms in the downcomer (Figure 15.6.5.4B-9).

The primary pressure falls below the pressure in the accumulators thus causing the accumulator check valves to open and accumulator delivery to begin (Figures 15.6.5.4B-10 and -11). The accumulators, and then the core makeup tanks inject until they empty. The ADS flow falls off as the primary pressure decreases. The flow from the accumulators raise the mixture levels in the upper plenum and downcomer (Figures 15.6.5.4B-16 and 15.6.5.4B-9).

As the levels in the core makeup tanks reach the ADS Stage 4 setpoint, one out of two paths are opened from the top of the hot leg (loop 1) and begin discharging fluid. After 30 seconds, the second path in loop one opens, as does a loop 2 Stage 4 path. Activating the Stage 4 paths leads to reduced flow through ADS Stages 1, 2, and 3. The reduced flow allows the pressurizer level to fall, and these stages begin to discharge only steam. Once the core makeup tanks are empty, delivery ceases (Figures 15.6.5.4B-7 and -8). Once the reactor coolant system pressure has fallen sufficiently due to the ADS Stage 4 discharge, (Figure 15.6.5.4B-12) gravity drain from the IRWST begins (Figures 15.6.5.4B-13 and -14). At 5000 seconds, the calculation is considered complete; IRWST delivery exceeds the ADS flows (which are removing the decay heat), and the reactor coolant system inventory is slowly rising (Figure 15.6.5.4B-15). Core uncovery does not occur and the upper plenum mixture level remains well above the core elevation throughout (Figure 15.6.5.4B-16).

The inadvertent opening of the ADS Stage 1 transient confirms the minimum venting area capability to depressurize the reactor coolant system to the IRWST pressure. The analysis indicates that the ADS sizing is sufficient to depressurize the reactor coolant system assuming the worst single failure as the failure of a Stage 4 ADS path to open and decay heat equal to the 10 CFR 50 Appendix K (Reference 1) value of the ANS-1971 Standard (Reference 2) plus 20 percent, which over estimates the core steam generation rate. Even under these limiting conditions, IRWST injection is obtained, and the core remains covered such that no cladding heatup occurs.

15.6.5.4B.3.4 2-inch Cold Leg Break in the Core Makeup Tank Loop

This case models a 2-inch break occurring in the bottom of cold leg connected to the balance line of CMT-1. The reactor steady-state initial conditions assumed for this transient can be found in Table 15.6.5-9. The event times for this transient are given in Table 15.6.5-12.

The break opens at time zero, and the pressurizer pressure begins to fall as shown in Figure 15.6.5.4B-17 as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 15.6.5.4B-18. The break fluid flow is shown in Figures 15.6.5.4B-32 and -33. 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 Figures 15.6.5.4B-22 and -23.

As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 15.6.5.4B-30, and the core remains completely covered. 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 15.6.5.4B-18) 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 15.6.5.4B-17; the accumulator setpoint is reached and the accumulators begin to inject. The injection flow from the core makeup tanks are shown in Figures 15.6.5.4B-22 and -23, and from the accumulators, in Figures 15.6.5.4B-24 and -25.

As Figures 15.6.5.4B-22 and -23 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 2 valves, maintaining the depressurization rate as shown in Figure 15.6.5.4B-17. ADS Stage 3 valves open, thereby increasing the system venting capability. The ADS Stage 4 valves open when the core makeup tank water level is reduced to 20 percent. Figures 15.6.5.4B-28 and -31 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 15.6.5.4B-18. The Figure 15.6.5.4B-29 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. The mixture level in the reactor vessel is approximately at the hot leg elevation as shown in Figure 15.6.5.4B-30 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 break cases exhibit large margin-to-core uncovery.

15.6.5.4B.3.5 Direct Vessel Injection Line Break

This case models the double-ended rupture of the DVI line at the nozzle into the downcomer. The broken loop injection system (consisting of an accumulator, a core makeup tank, and an IRWST delivery line) is modeled to spill completely out the DVI side of the break. The steady-state reactor coolant system conditions for this transient are shown in Table 15.6.5-9. Design maximum resistances are applied to the inlet and outlet lines of that core makeup tank to conservatively minimize intact loop core makeup tank delivery through the time of minimum reactor coolant system mass inventory. Minimum resistances are applied to the broken loop IRWST injection line to maximize the spill to containment, thus minimizing the reactor coolant system mass inventory. This case uses a containment backpressure defined to be a constant 20 psia. While not exactly reflecting the containment pressure history that occurs as a result of the DVI line break, it represents a conservatively low estimate of the expected containment pressure response during a DEDVI transient. The containment pressurizes for a DEDVI break as a result of the break mass and energy releases in addition to the ADS-4 discharge paths that vent directly to the containment atmosphere.

The containment pressurization was calculated using the mass and energy releases from the NOTRUMP small-break LOCA code in the WGOTHIC containment model. Mass and energy releases from both sides of the DVI break (both vessel side and DVI side) and ADS-4 valve discharges were provided in a tabular form to the WGOTHIC AP1000 model used to compute containment pressurization for the long-term cooling analysis.

The event times for this transient are shown in Table 15.6.5-13. The break is assumed to open instantaneously at 0 seconds. The accumulator on the broken loop starts to discharge via the DVI line to the containment. Figure 15.6.5.4B-36 shows the subcooled discharge from the downcomer nozzle, which causes a rapid reactor coolant system (RCS) depressurization (Figure 15.6.5.4B-38). A reactor trip signal is generated, followed by generation of the "S" signal. Following a delay, the isolation valves on the core makeup tank and PRHR delivery lines begin to open. The "S" signal also causes closure of the main feedwater isolation valves after a 2-second delay and trips the reactor coolant pumps after a 6-second delay. The opening of the core makeup tank isolation valves allows the broken loop core makeup tank to discharge directly to the containment (Figure 15.6.5.4B-39), and a small circulatory flow develops through the intact loop core makeup tank (Figure 15.6.5.4B-40).

As the pressure falls, the reactor coolant system fluid saturates, and a mixture level forms in the upper plenum and then falls to the hot leg elevation (Figure 15.6.5.4B-41). The upper parts of the reactor coolant system start to drain, and a mixture level forms in the downcomer (Figure 15.6.5.4B-42) and falls below the elevation of the break. Two-phase discharge, then vapor flow occurs from the downcomer side of the break (Figure 15.6.5.4B-37).

In the core makeup tank connected to the broken loop, a level forms and starts to fall. The ADS Stage 1 setpoint is reached, and the ADS Stage 1 valves open after the signal delay time elapses. The ensuing steam discharge from the top of the pressurizer (Figure 15.6.5.4B-43) increases the reactor coolant system depressurization rate. The depressurization rate is also increased due to the steam discharge from the downcomer to the containment (Figure 15.6.5.4B-37) as the downcomer mixture level falls below the DVI nozzle (Figure 15.6.5.4B-42).

During the initial portion of the DEDVI break, only liquid flows out the top of the core (Figure 15.6.5.4B-45). Soon, steam flows out also (Figure 15.6.5.4B-46) because the void fraction in the core increases (Figure 15.6.5.4B-44). The break in the downcomer draws fluid from the bottom of the core (Figure 15.6.5.4B-47) and insufficient liquid remains in the core and upper plenum to sustain the mixture level. The mixture level therefore starts to decrease (Figure 15.6.5.4B-41). The mixture level falls to a minimum and then starts to recover, as flow re-enters the core from the downcomer (Figure 15.6.5.4B-41 compared to -47).

The ADS Stage 2 valves open after the appropriate time delay between the actuation of the first two stages of the ADS. The intact loop accumulator starts to inject into the downcomer (Figure 15.6.5.4B-50) causing the mixture level in the downcomer to slowly rise (Figure 15.6.5.4B-42). The mixture level also increases within the upper plenum.

The ADS Stage 3 valves open upon completion of the time delay of 120 seconds between the actuation of Stages 2 and 3 of the ADS. The broken loop core makeup tank level reaches the ADS Stage 4 setpoint, but the ADS Stage 4 valves do not open until the minimum time delay between the actuation of ADS Stages 3 and 4 occurs. Two-phase discharge ensues through three of the four

Stage 4 paths (Figures 15.6.5.4B-48 and -49). The broken loop core makeup tank and accumulator empty rapidly.

The fluid level at the top of the intact loop core makeup tank starts to decrease slowly (Figure 15.6.5.4B-52) because injection from the tank has begun (Figure 15.6.5.4B-40). The intact loop accumulator has emptied (Figure 15.6.5.4B-50) and the reduced pressure in the injection line allows the core makeup tank to inject continuously.

During the period of accumulator injection, the downcomer mixture level rises slowly (Figure 15.6.5.4B-42). Figure 15.6.5.4B-53 presents the RCS mass inventory. With only intact loop core makeup tank injection available for a period of time, the downcomer level once again falls and core boil-off increases the rate of reactor coolant system inventory depletion until sufficient CMT/IRWST injection flow can be introduced. However, the level in the upper plenum is maintained near the hot leg elevation (Figure 15.6.5.4B-41) throughout the remainder of the transient.

Once the pressure in the broken DVI line falls below that in the IRWST, the water from the tank is spilled to the containment.

Stable, but decreasing, injection continues from the intact loop core makeup tank as the reactor coolant system pressure declines slowly. The reactor coolant system pressure continues to fall until it drops below that of the IRWST and injection begins (Figure 15.6.5.4B-51). With the reduced initial RCS inventory recovery from the accumulators and only a single intact injection path available for the DEDVI line break, the minimum inventory occurs near the initiation of IRWST injection flow. After injection flow greater than the sum of the break and ADS flows exists, a slow rise in the reactor coolant system inventory (Figure 15.6.5.4B-53) occurs. Since no core uncovery is predicted for this scenario, no cladding heatup occurs.

The critical heat flux assessment described in subsection 15.6.5.4B.2.3 addresses core cooling during a time period where the NOTRUMP computer code may not conservatively predict the core average void fraction. The requirements of 10 CFR 50.46 are met during this period since this CHF assessment indicates peak core heat flux is less than critical heat flux. Cladding temperatures will remain near the coolant saturation temperature, well below the 10 CFR 50.46 peak cladding temperature limit.

Another DEDVI line break analysis is performed that is the same as the case discussed above except that containment pressure is assumed to be at 14.7 psia. Table 15.6.5-13A provides the time sequence of events for this analysis. Figures 15.6.5.4B-36A through -55A provide the transient results for this analysis. The transient is like the case at 20 psia except that IRWST injection occurs somewhat later due to the lower containment pressure.

15.6.5.4B.3.6 10-inch Cold Leg Break

This case models a 10-inch break occurring in the bottom of a cold leg connected to the balance line of CMT-1. The reactor steady-state initial conditions assumed for this transient are found in Table 15.6.5-9. The event times for this transient are given in Table 15.6.5-14.

The break opens at time zero, and the pressurizer pressure begins to fall, as shown in Figure 15.6.5.4B-56, as mass is lost out the break. The pressurizer mixture level initially decreases as given in Figure 15.6.5.4B-57. The break fluid flow is shown in Figures 15.6.5.4B-75 and -76 for the liquid and vapor components respectively. The pressurizer pressure falls below the reactor trip set point. This causes the reactor to trip (after the appropriate time delay) and 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 energy being removed by the steam generator safety valves, the core, and the PRHR heat exchanger. 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 Figures 15.6.5.4B-61 and -62.

As time proceeds, the loops drain to the reactor vessel. The mixture level in the downcomer begins to drop as seen in Figure 15.6.5.4B-60, and the core remains completely covered. Due to the size and location of the break involved, the accumulator setpoint is reached prior to the core makeup tanks transitioning from recirculation to injection mode. The flows from the core makeup tanks are shown in Figures 15.6.5.4B-61 and -62, and from the accumulators, in Figures 15.6.5.4B-63 and -64. The response of core makeup tank 1 is offset compared to that of core makeup tank 2 as a result of the break size/location being modeled. Core makeup tank 2 reaches the 67.5-percent level first, and after an appropriate delay, the ADS Stage 1 valves open. When the ADS is actuated, the mixture level increases in the pressurizer (Figure 15.6.5.4B-57) 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 15.6.5.4B-56.

During the initial portion of the 10-inch break, both liquid and steam flow out the top of the core (Figures 15.6.5.4B-71 and -72) as the void fraction in the core increases (Figure 15.6.5.4B-73). The break in the cold leg draws fluid from the bottom of the core, and insufficient liquid remains in the core and upper plenum to sustain the mixture level. The mixture level, therefore, starts to decrease (Figure 15.6.5.4B-69). The mixture level falls to a minimum and then starts to recover as accumulator flows enter the downcomer (Figures 15.6.5.4B-63 and -64). During this time period (~75-125 seconds), a portion of the core exhibits the potential for core dryout to occur without the prediction of a traditional core uncovery period (for example, core two-phase mixture level dropping into the active fuel region). To conservatively account for this potential core dryout period, a composite core mixture level was created which collapses to the minimum of the actual core/upper plenum two-phase mixture level and the bottom of the lowest core node that exceeds the core dryout onset conditions. A 90-percent quality limit was chosen as the indicator of the onset of core dryout indicative of the critical heat flux (as predicted by Griffith’s modification of the Zuber equation, in References 28 and 29); dryout is assumed at core qualities above this value. The resulting composite core mixture level resulting from this approach can be seen in Figure 15.6.5.4B-70. To conservatively estimate the effects of this dryout period, an adiabatic heat-up calculation was performed, and the resulting peak cladding temperature is determined to be approximately 1370°F. Even under these conservative adiabatic heat-up assumptions, the AP1000 plant design exhibits large margins to the 10 CFR 50.46 Appendix-K limits for the 10-inch break.

As Figures 15.6.5.4B-61 and -62 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 injection lines of the core makeup tanks by the accumulators. The opening of ADS Stage 2 valves maintains the depressurization rate as shown in Figure 15.6.5.4B-56. ADS Stage 3 valves subsequently open. This increases the system venting capability. The ADS Stage 4 valves open when the core makeup tank water level is reduced to 20 percent. Figures 15.6.5.4B-67 and -74 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 15.6.5.4B-57. The Figure 15.6.5.4B-68 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. Once the downcomer pressure drops below the IRWST injection pressure, flow enters the reactor vessel from the IRWST. The mixture level in the reactor vessel is approximately at the hot leg elevation as shown in Figure 15.6.5.4B-69 throughout this transient; the core never uncovers, even though the period of potential core dryout was predicted to occur during the initial blowdown period. Even when the core dryout is conservatively accounted for, large margins to the 10 CFR 50.46 Appendix-K limits of 2200°F exist.

15.6.5.4B.3.7 Direct Vessel Injection Line Break (Entrainment Sensitivity)

In order to assess the potential impact of higher than expected entrainment in the upper plenum and hot legs on the overall system response and core cooling, an AP1000 plant sensitivity run was performed. The sensitivity case was performed with the DEDVI line break simulation as described in the following. The simulation is identical to the base DEDVI line simulation presented in subsection 15.6.5.4B.3.5 until ADS-4 actuation, at which time the higher than expected entrainment is included in the analysis by assuming homogenous conditions in the regions downstream of the core. In addition, since homogenous treatment of these regions will eliminate the pressure drop effect of the accumulated mass stored in the upper plenum, the NOTRUMP model was conservatively adjusted to account for this effect following the transition of the ADS-4 flow paths to noncritical conditions.

Figure 15.6.5.4B-79 presents a comparison of the upper downcomer pressure between the base and sensitivity cases. The sensitivity case results in higher upper downcomer pressure and subsequently results in delayed IRWST injection (Figure 15.6.5.4B-80). This can also be observed in the intact DVI line flow, which comprises all intact injection flow components (that is, accumulator, CMT, and IRWST) per Figure 15.6.5.4B-81, and the pressurizer mixture level response (Figure 15.6.5.4B-90), which follows the change in pressure response. As expected, the initial ADS-4 liquid discharge is much higher (Figure 15.6.5.4B-82) until the inventory, which resided in the upper plenum and hot leg regions, depletes (Figure 15.6.5.4B-83). The net effect is a decrease in the ADS-4 vapor discharge rate (Figure 15.6.5.4B-84) and subsequently higher RCS pressures.

Due to the elimination of the inventory stored in the upper plenum, the downcomer mass is also reduced (Figure 15.6.5.4B-85). Since the static head that existed in the upper plenum is eliminated when the model is made homogenous, the downcomer mixture is subsequently driven into the core as the static heads equilibrate. This results in the core region mass increasing initially due to the introduction of cold downcomer fluid to the core region (Figure 15.6.5.4B-86). The net effect of the sensitivity case is that the vessel inventory is substantially decreased over the base model simulation (Figure 15.6.5.4B-87); however, this inventory is sufficient to provide adequate core cooling because the ADS-4 continually draws liquid flow through the core (Figure 15.6.5.4B-82). Even though there is no liquid storage in the upper plenum for the homogenous case (Figure 15.6.5.4B-88), the core collapsed liquid level (Figure 15.6.5.4B-89) is not impacted significantly.

This sensitivity demonstrates that the AP1000 plant response is relatively insensitive to upper plenum and hot leg entrainment. Even with the assumption of homogenous fluid nodes above the core, adequate core cooling is demonstrated. No significant core uncovery/heatup is predicted for this scenario.

15.6.5.4B.4 Conclusions

The small-break LOCA analyses performed show that the performance of the AP1000 plant design to small-break LOCA scenarios is excellent and 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 excessive cladding heatup for small-break LOCAs analyzed, including double-ended ruptures in the passive safeguards system lines. The peak AP1000 heat flux during the accumulator injection period is below the predicted critical heat flux.
- The effect of increasing upper plenum/hot leg entrainment does not significantly affect plant safety margins.

The analyses performed demonstrate that the 10 CFR 50.46 Acceptance Criteria are met by the AP1000. Summarizing the small-break LOCA spectrum:

Break Location/Diameter	AP1000 Minimum RCS Inventory	Peak Cladding Temperature
Inadvertent ADS	105,800	(1)
2-inch cold leg break	106,620	(1)
10-inch cold leg break	78,160	<1370°F
DEDVI	113,710	(1)
DEDVI (Entrainment Study)	~81,000	(1)

The 10-inch cold leg break exhibits the limiting minimum inventory condition that occurs during the initial blowdown period and is terminated by accumulator injection. The AP1000 design is such that the minimum inventory occurs just prior to IRWST injection for all breaks except the

(1) There is no core heatup as a result of this transient. PCT occurs at transient initiation.

10-inch cold leg break. All breaks simulated in the break spectrum produce results that demonstrate significant margin to peak cladding temperature regulatory limits.

15.6.5.4C Post-LOCA Long-Term Cooling

15.6.5.4C.1 Long-Term Cooling Analysis Methodology

The AP1000 safety-related systems are designed to provide adequate cooling of the reactor indefinitely. Initially, this is achieved by discharging water from the IRWST into the vessel. When the low-3 level setpoint is reached in the IRWST, the containment recirculation subsystem isolation valves open and water from the containment reactor coolant system (RCS) compartment can flow into the vessel through the PXS piping. The water in containment rises in temperature toward the saturation temperature. Long-term heat removal from the reactor and containment is by heat transfer through the containment shell to atmosphere.

The purpose of the long-term cooling analysis is to demonstrate that the passive systems provide adequate emergency core cooling system performance during the IRWST injection/containment recirculation time scale. The long-term cooling analysis is performed using the WCOBRA/TRAC computer code to verify that the passive injection system is providing sufficient flow to the reactor vessel to cool the core and to preclude boron precipitation.

The AP1000 long-term cooling analysis is supported by the series of tests at the Oregon State University AP600 APEX Test Facility. This test facility is designed to represent the AP600 reactor safety-related systems and nonsafety-related systems at quarter-scale during long-term cooling. The data obtained during testing at this facility has been shown to apply to the AP1000 (Reference 25). These tests were modeled using WCOBRA/TRAC with an equivalent noding scheme to that used for AP600 (Reference 17) in order to validate the code for long-term cooling analysis.

Reference 24 provides details of the AP1000 WCOBRA/TRAC modeling. The coarse reactor vessel modeling used for AP600 has been replaced with a detailed noding like that applied in the large-break LOCA analyses described in subsection 15.6.5.4A. The reactor vessel noding used in the AP1000 long-term cooling analyses in core and upper plenum regions is equivalent to that used in full-scale test simulations (see Reference 24).

A DEDVI line break is analyzed because it is the most limiting long-term cooling case in the relationship between decay power and available liquid driving head. Because the IRWST spills directly onto the containment floor in a DEDVI break, this event has the highest core decay power when the transfer to sump injection is initiated. In postulated DEDVI break cases, the compartment water level exceeds the elevation at which the DVI line enters the reactor vessel, so water can flow from the containment into the reactor vessel through the broken DVI line; this in-flow of water through the broken DVI line assists in the heat removal from the core. The steam produced by boiling in the core vents to the containment through the ADS valves and condenses on the inner surface of the steel containment vessel. The condensate is collected and drains to the IRWST to become available for injection into the reactor coolant system. The WCOBRA/TRAC analysis presented analyzes the DEDVI small-break LOCA event from a time (3000 seconds) at which IRWST injection is fully established to beyond the time of containment recirculation.

During this time, the head of water to drive the flow into the vessel for IRWST injection decreases from the initial level to its lowest value at the containment recirculation switchover time. PXS Room B is the location of the break in the DVI line. At this break location, liquid level in containment at the time of recirculation is a minimum.

A continuous analysis of the post-LOCA long term cooling is provided from the time of stable IRWST injection through the time of sump recirculation for the DEDVI break. Maximum design resistances are applied in WCOBRA/TRAC for both the ADS Stage 4 flow paths and the IRWST injection and containment recirculation flow paths.

The break modeled is a double-ended guillotine rupture of one of the direct vessel injection lines. The long-term cooling phase begins after the simultaneous opening of the isolation valves in the IRWST DVI lines and the opening of ADS Stage 4 squib valves, when flow injection from the IRWST has been fully established. Initial conditions are taken from the NOTRUMP DEDVI case at 20 psia containment pressure reported in subsection 15.6.5.4B.

15.6.5.4C.2 DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case

This subsection presents the results of a DEDVI line break analysis during IRWST injection phase continuing into sump recirculation. Initial conditions at the start of the case are prescribed based on the NOTRUMP DEDVI break results to allow a calculation to begin shortly after IRWST injection begins in the small break long-term cooling transient. The WCOBRA/TRAC calculation is then allowed to proceed until a quasi-steady-state is achieved. At this time, the predicted results are independent of the assumed initial conditions. This calculation uses boundary conditions taken from a WGOTHIC analysis of this event. During the calculation, which is carried out for 10,000 seconds until a quasi-steady-state sump recirculation condition has been established, the IRWST water level is decreased continuously until the sump recirculation setpoint is reached.

In the analysis, one of the two ADS Stage 4 valves in the PRHR loop is assumed to have failed. The initial reactor coolant system liquid inventory and temperatures are determined from the NOTRUMP calculation. The core makeup tanks do not contribute to the DVI injection during this phase of the transient. Steam generator secondary side conditions are taken from the NOTRUMP calculation (at the beginning of long-term cooling). The reactor coolant pumps are tripped and not rotating.

The levels and temperatures of the liquid in the containment sump and the containment pressure are based on WGOTHIC calculations of the conservative minimum pressure during this long-term cooling transient, including operation of the containment fan coolers. Small changes in the RCS compartment level do not have a major effect on the predicted core collapsed liquid level or on the predicted flow rate through the core. The minimum compartment floodup level for this break scenario is 107.8 feet or greater.

In this transient, the IRWST provides a hydraulic head sufficient to drive water into the downcomer through the intact DVI nozzle. Also, water flows into the downcomer from the broken DVI line once the liquid level in the compartment with the broken line is adequate to support flow. The water flows down the downcomer and up through the core, into the upper plenum.

Steam produced in the core and liquid flow out of the reactor coolant system via the ADS Stage 4 valves. There is little flow out of ADS Stages 1, 2, and 3 even when the IRWST liquid level falls below the sparger elevation, so they are not modeled in this calculation. The venting provided by the ADS-4 paths enables the liquid flow through the core to maintain core cooling.

Approximately 500 seconds of WCOBRA/TRAC calculation are required to establish the quasi-steady-state condition associated with IRWST injection at the start of long-term cooling and so are ignored in the following discussion. The hot leg levels are such that during the IRWST injection phase the quality of the ADS Stage 4 mass flows varies as water is carried out of the hot legs. This periodically increases the pressure drop across the ADS Stage 4 valves and the upper plenum pressure. The higher pressure in the upper plenum reduces the injection flow. This cycle of pressure variations due to changing void fractions in the flow through ADS Stage 4 is consistent with test observations and is expected to recur often during long-term cooling.

The head of water in the IRWST causes a flow of subcooled water into the downcomer at an approximate rate of 170 lbm/s through the intact DVI nozzle at the start of long-term cooling. The downcomer level at the end of the code initiation (the start of long-term cooling) is about 18.0 feet (Figure 15.6.5.4C-1). Note that the time scale of this and other figures in subsection 15.6.5.4C.2 is offset by 2500 seconds; that is, a time of 500 seconds on the Figure 15.6.5.4C-1 axis equals 3000 seconds transient time for the DEDVI break. All of the injection water flows down the downcomer and up through the core. The accumulators have been fully discharged before the start of the time window and do not contribute to the DVI flow.

Boiling in the core produces steam and a two-phase mixture, which flows into the upper plenum. The core is 14 feet high, and the core average collapsed liquid level (Figure 15.6.5.4C-2) is shown from the start of long-term cooling. The boiling process causes a variable rate of steam production and resulting pressure changes, which in turn causes oscillations in the liquid flow rate at the bottom of the core and also variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC noding, the core is divided both axially and radially as described in Reference 24. The void fractions in the top two cells of the hot assembly are shown as Figures 15.6.5.4C-3 and -4. The average void fraction of these upper core cells is about 0.8 during long-term cooling, during IRWST injection, and into the containment recirculation period. There is a continuous flow of two-phase fluid into the hot legs, and mainly vapor flow toward the ADS Stage 4 valve occurs at the top of the pipe. The collapsed liquid level in the hot leg varies between 0.8 feet to 1.6 feet (Figure 15.6.5.4C-5). The hot legs on average are more than 50-percent full. Vapor and liquid flows at the top of the core are shown in Figures 15.6.5.4C-6 and 15.6.5.4C-7, the upper plenum collapsed liquid level in Figure 15.6.5.4C-8. Figures 15.6.5.4C-9 and 15.6.5.4C-10 are ADS stage 4 mass flowrates.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-11. The upper plenum pressure fluctuation that occurs is due to the ADS Stage 4 water discharge. The PCT of the hot rod follows saturation temperature (Figure 15.6.5.4C-12), which demonstrates that no uncovery and no cladding temperature excursion occurs. A small pressure drop is calculated across the reactor vessel, and injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-13 and -14. Figure 15.6.5.4C-14 shows the flow is outward through the broken DVI line at the start of the long-term cooling period, and it increases to a maximum average value of about 52 lbm/s after the compartment water level has increased above the nozzle elevation to permit liquid injection into the reactor vessel. In contrast, the intact DVI line flow falls from 170 lbm/s with a full IRWST to about 65 lbm/s flow from the containment at the end of the calculation. The recirculation core liquid throughput is more than adequate to preclude any boron buildup on the fuel.

Figures 15.6.5.4C-1A through -14A present the sensitivity of long-term cooling performance to a bounding containment pressure of 14.7 psia. The DEDVI break in the PXS "B" Room case is restarted at 6500 seconds to assess in a window mode calculation the effect of this reduced containment pressure at the most limiting time in the transient, the switchover to containment recirculation. The initial 700 seconds of the window establish the reactor vessel pressure condition that is consistent with the 14.7 psia containment pressure. After 7200 seconds, the WCOBRA/TRAC calculation provides the transient behavior of the AP1000 at the reduced containment pressure.

15.6.5.4C.3 DEDVI Break and Wall-to-Wall Floodup; Containment Recirculation

This subsection presents a DEDVI line break analysis with wall-to-wall flooding due to leakage between compartments, using the window mode methodology. All containment free volume beneath the level of the liquid is assumed filled in this calculation to generate the minimum water level condition during containment recirculation. The time identified for this calculation is 14 days into the event, and the core power is calculated accordingly. The initial conditions at the start of the window are consistent with the analysis described in subsection 15.6.5.4C.2. Containment recirculation is simulated during the time window. The calculation is carried out over a time period long enough to establish a quasi-steady-state solution; after 400 seconds of problem time, the flow dynamics are quasi-steady-state and the predicted results are independent of the assumed initial conditions. The liquid level is simulated constant at 28.2 feet above the bottom inside surface of the reactor vessel (refer to Figure 15.0.3-2 for AP1000 reference plant elevations) during the time window, and the liquid temperatures in the containment sump and the PXS "B" room are 196°F and 182°F, respectively. The containment pressure is conservatively assumed to be 14.7 psia. The single failure of an ADS Stage 4 flow path is assumed as in the subsection 15.6.5.4C.2 case.

Focusing on the post 400-second time interval of this case, the containment liquid provides a hydraulic head sufficient to drive water into the downcomer through the DVI nozzles. The water introduced into the downcomer flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core entrains liquid and flows out of the reactor coolant system via the ADS Stage 4 valves. The DVI flow and the venting provided by the ADS paths provide a liquid flow through the core that enables the core to remain cool.

The downcomer collapsed liquid level (Figure 15.6.5.4C-15) varies between 23 and 25 feet during the analysis. Pressure spikes produced by boiling in the core can cause the mass flow of the DVI flow rates shown in Figures 15.6.5.4C-27 and -28 into the vessel to fluctuate upward and downward.

Boiling in the core produces steam and a two-phase mixture, which flows out of the core into the upper plenum. The core is 14 feet high, and the core collapsed liquid level (Figure 15.6.5.4C-16) maintains a mean level close to the top of the core. The boiling process causes pressure variations, which in turn, cause variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. In the WCOBRA/TRAC analysis, the core is nodalized as described in Reference 24. The void fraction in the top cell is shown in Figure 15.6.5.4C-17 for the core hot assembly, and Figure 15.6.5.4C-18 shows the void fraction that exists one cell further down in the hot assembly. The PCT does not rise appreciably above the saturation temperature (Figure 15.6.5.4C.3-26). The flow through the core and out of the reactor coolant system is more than sufficient to provide adequate flushing to preclude concentration of the boric acid solution. Liquid collects above the upper core plate in the upper plenum, where the average collapsed liquid level is about 3.6 feet (Figure 15.6.5.4C-22). There is no significant flow through the cold legs into either the broken or the intact loops, and there is no significant quantity of liquid residing in any of the cold legs.

The pressure in the upper plenum is shown in Figure 15.6.5.4C-25. The upper plenum pressurization, which occurs periodically, is due to the ADS Stage 4 water discharge. The collapsed liquid level in the hot leg of the pressurizer loop varies between 1.0 feet and 2.1 feet, as shown in Figure 15.6.5.4C-19. Injection rates through the DVI lines into the vessel are presented in Figures 15.6.5.4C-27 and -28.

15.6.5.4C.4 Post Accident Core Boron Concentration

An evaluation has been performed of the potential for the boron concentration to build up in the core following a cold leg LOCA. The evaluation methodology, simplified calculations, and their results are discussed in Reference 24. This evaluation considers both short-term operations, before ADS is actuated, and long-term operations, after ADS is actuated. These evaluations and their results are discussed in the follow paragraphs.

Short-Term – Prior to ADS actuation, it is not likely for boron to build up significantly in the core. Normally, water circulation mixes boron in the RCS and prevents buildup in the core. In order for boron to start to build up in the core region, water circulation through the steam generators and PRHR HX has to stop. In addition, significant injection of borated water is needed from the CMTs and the CVS. For this situation to happen, the hot legs need to void sufficiently to allow the steam generator tubes to drain. Once the steam generator tubes void, the cold legs will also void since they are located higher than the hot legs. When the top of the cold legs void, the CMTs will begin to drain. When the CMTs drain to the ADS stage 1 setpoint, ADS is actuated.

Short-Term Results – As shown in subsection 15.6.5.4B.3.4, a 2-inch LOCA requires less than 16 minutes from the time that the hot legs void significantly until ADS is actuated. For larger LOCAs, this time difference is shorter, as seen for the 10-inch cold leg LOCA (subsection 15.6.5.4B.3.6). The core boron concentration will not build up significantly in this short time. If the break is smaller than 2 inches, voiding of the hot legs will occur at a later time. With maximum operation of CVS makeup, it takes more than 3 hours for the core boron concentration to build up significantly. In addition, the volume of the boric acid tank limits the maximum buildup of boron in the core.

Following a small LOCA where ADS is not actuated, the operators are guided to sample the RCS boron concentration and to initiate a post-LOCA cooldown and depressurization. The cooldown and depressurization of the RCS reduces the leak rate and facilitates recovery of the pressurizer level. Recovery of the pressurizer level allows for re-establishment of water flow through the RCS loops, which mixes the boron. The operators are guided to take an RCS boron sample within

3 hours of the accident and several more during the plant cooldown. The purpose of the boron samples is to assess that there is adequate shutdown margin and that the RCS boron concentration has not built up to excessive levels. The maximum calculated core boron concentration 3 hours after a LOCA without ADS actuation is less than 16,000 ppm. Operator action within 3 hours maintains the maximum core boron concentration well below the boron solubility limit for the core inlet temperatures during the cooldown.

Long-Term – Once ADS is actuated, water carryover out the ADS Stage 4 lines limits the potential core boron concentration buildup following a cold leg LOCA. The design of the AP1000 facilitates water discharge from the hot legs as follows:

- PXS recirculation flow capability tends to fill the hot legs and bring the water level up to the ADS Stage 4 inlet.

- ADS Stage 4 lines discharge at an elevation 3 to 4 feet above the containment water level.

With water carried out ADS Stage 4, the core boron concentration increases until the boron added to the core in the safety injection flow equals the boron removed in the water leaving the RCS through the ADS Stage 4 flow. The lower the ADS Stage 4 vent quality, the lower the core boron concentration buildup.

Long-Term Results – Analyses have been performed (Reference 24) to bound the maximum core boron concentration buildup. These analyses demonstrate that highest ADS Stage 4 vent qualities result from the following:

- Highest decay heat levels

- Lowest PXS injection/ADS 4 vent flows, including high line resistances and low containment water levels

The long-term cooling analysis discussed in subsection 15.6.5.4C.2 is consistent with these assumptions. The ADS Stage 4 vent quality resulting from this analysis is less than 40 percent at the beginning of IRWST injection and reaches a maximum of less than 50 percent around the initiation of recirculation. It decreases after this peak, dropping to a value less than 8 percent at 14 days.

With the maximum ADS Stage 4 vent qualities, the maximum core boron concentration peaks at a value of about 7400 ppm at the time of recirculation initiation. After this time, the core boron concentration decreases as the ADS Stage 4 vent quality decreases, reaching 5000 ppm about 9 hours after the accident. The core boron solubility temperature reaches a maximum of 58°F (at 7400 ppm) and quickly drops to 40°F (at 5000 ppm). With these low core boron solubility temperatures, there is no concern with cold PXS injection water causing boron precipitation in the core. With the IRWST located inside containment, its water temperature is normally expected to be above these solubility temperatures. Even considering the minimum IRWST temperature permitted by the Technical Specifications (50°F), the minimum core inlet temperature is greater than the solubility temperature considering heatup of the injection by steam condensation in the downcomer and pickup of sensible heat from the reactor vessel, core barrel, and lower support plate.

The boron concentration water in the containment is initially about 2980 ppm. As the core boron concentration increases, the containment concentration decreases slightly. The minimum boron concentration in containment is greater than 2950 ppm. The solubility temperature of the containment water at its maximum boron concentration is 32°F.

With high decay heat values, the ADS Stage 4 vent flows and velocities are high. These high vent velocities result in flow regimes that are annular for more than 30 days. The annular flow regime moves water up and out the ADS Stage 4 lines. This flow regime is based on the Taitel-Dukler vertical flow regime map. Lower decay heat levels can be postulated later in time or just after a refueling outage. Significantly lower decay heat levels result in lower ADS Stage 4 vent qualities. They also result in ADS Stage 4 vent flows/velocities that are lower. Even with low ADS Stage 4 vent flow velocities, the AP1000 plant will move water out the ADS Stage 4 operating as a manometer. Small amounts of steam generated in the core reduce the density of the steam/water mixture in the core, upper plenum, and ADS Stage 4 line as it bubbles up through the water. As a result, the injection head is sufficient to push the less dense, bubbly steam/water mix out the ADS Stage 4 line.

At the time recirculation begins, the containment level will be about 109.3 feet (for a non-DVI LOCA) and will be about 108.0 feet (for a DVI LOCA). Over a period of weeks after a LOCA, water may slowly leak from the flooded areas in containment to other areas inside containment that did not initially flood. As a result, the minimum containment water could decrease to 103.5 feet. During recirculation operation following a LOCA and ADS actuation, the operators are guided to maintain the containment water level above the 107-foot elevation by adding borated water to the containment. In addition, if the plant continues to operate in the recirculation mode, the operators are guided to increase the level to 109 feet within 30 days of the accident. These actions provide additional margin in water flow through the ADS Stage 4 line. The operators are also guided to sample the hot leg boron concentration prior to initiating recovery actions that might introduce low temperature water to the reactor.

15.6.5.4C.5 Conclusions

Calculations of AP1000 long-term cooling performance have been performed using the WCOBRA/TRAC model developed for AP1000 and described in Reference 24. The DEDVI case was chosen because it reaches sump recirculation at the earliest time (and highest decay heat). A window mode case at the minimum containment water level postulated to occur 2 weeks into long-term cooling was also performed.

The DEDVI small-break LOCA exhibits no core uncovery due to its adequate reactor coolant system mass inventory condition during the long-term cooling phase from initiation into containment recirculation. Adequate flow through the core is provided to maintain a low cladding temperature and to prevent any buildup of boric acid on the fuel rods. The wall-to-wall floodup case using the window mode technique demonstrates that effective core cooling is also provided at the minimum containment water level. The results of these cases demonstrate the capability of the AP1000 passive systems to provide long-term cooling for a limiting LOCA event.

15.6.6 References

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Table 15.6.1-1

TIME SEQUENCE OF EVENTS FOR INCIDENTS THAT CAUSE A DECREASE IN REACTOR COOLANT INVENTORY

Accident	Event	Time (seconds)
Inadvertent opening of a pressurizer safety valve with offsite power available	Pressurizer safety valve opens fully	0.0
	Overtemperature ΔT reactor trip setpoint reached	18.55
	Rods begin to drop	20.55
	Minimum DNBR occurs	21.3
Inadvertent opening of a pressurizer safety valve without offsite power available	Pressurizer safety valve opens fully	0.0
	Overtemperature ΔT reactor trip setpoint reached	18.55
	Turbine trip signal	20.23
	Rods begin to drop	20.55
	Minimum DNBR occurs	21.3
	ac power lost, reactor coolant pumps begin coasting down	23.23
Inadvertent opening of two ADS Stage 1 trains with offsite power available	ADS valves begin to open	0.0
	Overtemperature ΔT reactor trip setpoint reached	18.40
	Rods begin to drop	20.40
	Minimum DNBR occurs	21.30
	ADS valves fully open	25.0
Inadvertent opening of two ADS Stage 1 trains without offsite power available	ADS valves begin to open	0.0
	Low pressurizer pressure reactor trip setpoint reached	18.40
	Turbine trip signal	20.1
	Rods begin to drop	20.40
	Minimum DNBR occurs	21.3
	ac power lost, reactor coolant pumps begin coasting down	23.1
	ADS valves fully open	25.0

Table 15.6.2-1

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A SMALL LINE BREAK OUTSIDE CONTAINMENT

Reactor coolant iodine activity	Initial activity equal to the design basis reactor coolant activity of 1.0 μCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 500 (see Table 15A-2 in Appendix 15A)[(a)]
Reactor coolant noble gas activity	280 μCi/g dose equivalent Xe-133
Break flow rate (gpm)	130[(b)]
Fraction of reactor coolant flashing	0.41
Duration of accident (hr)	0.5
Atmospheric dispersion (χ/Q) factors	See Table 15A-5
Nuclide data	See Table 15A-4

Notes:

a. Use of accident-initiated iodine spike is consistent with the guidance in the Standard Review Plan.

b. At density of 62.4 lb/ft^3.

Table 15.6.3-1

STEAM GENERATOR TUBE RUPTURE SEQUENCE OF EVENTS

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

Table 15.6.3-2

STEAM GENERATOR TUBE RUPTURE MASS RELEASE RESULTS

Total Mass Flow from Initiation of Event to Cooldown to RHR[1] Conditions

	Start of Event to Break Flow Termination (Pounds Mass)	Break Flow Termination to Cut-in of RHR (Pounds Mass)
Faulted steam generator		
– Atmosphere	238,600	93,200
Intact steam generator		
– Atmosphere	183,400	1,234,900
Break flow	385,000	0

Note:
1. RHR = residual heat removal

Table 15.6.3-3

PARAMETERS USED IN EVALUATING THE RADIOLOGICAL CONSEQUENCES OF A STEAM GENERATOR TUBE RUPTURE

Reactor coolant iodine activity	
– Accident initiated spike	Initial activity equal to the equilibrium operating limit for reactor coolant activity of 1.0 µCi/g dose equivalent I-131 with an assumed iodine spike that increases the rate of iodine release from fuel into the coolant by a factor of 335 (see Appendix 15A). Duration of spike is 5.3 hours.
– Preaccident spike	An assumed iodine spike that results in an increase in the reactor coolant activity to 60 µCi/g of dose equivalent I-131 (see Appendix 15A)
Reactor coolant noble gas activity	280 µCi/g dose equivalent Xe-133
Reactor coolant alkali metal activity	Design basis activity (see Table 11.1-2)
Secondary coolant initial iodine and alkali metal	10% of reactor coolant concentrations at maximum equilibrium conditions
Reactor coolant mass (lb)	3.84 E+05
Offsite power	Lost on reactor trip
Condenser	Lost on reactor trip
Time of reactor trip	Beginning of the accident
Duration of steam releases (hr)	13.19
Atmospheric dispersion factors	See Appendix 15A
Nuclide data	See Appendix 15A
Steam generator in ruptured loop	
– Initial secondary coolant mass (lb)	1.66 E+05
– Primary-to-secondary break flow	See Figure 15.6.3-5
– Flashing fraction for break flow	See Figure 15.6.3-10
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02[a]
– Alkali metals partition coefficient	1.0 E-03[a]
Steam generator in intact loop	
– Initial secondary coolant mass (lb)	2.00 E+05
– Primary-to-secondary leak rate (lb/hr)	52.14[b]
– Steam released (lb)	See Table 15.6.3-2
– Iodine partition coefficient	1.0 E-02
– Alkali metals partition coefficient	1.0 E-03

Notes:

a. Iodine partition coefficient does not apply to flashed break flow.

b. Equivalent to 150 gpd at psia cooled liquid at 62.4 lb/ft^3.

Table 15.6.5-1

CORE ACTIVITY RELEASES TO THE CONTAINMENT ATMOSPHERE

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

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

Table 15.6.5-2 (Sheet 1 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Primary coolant source data	
– Noble gas concentration	280 µCi/g dose equivalent Xe-133
– Iodine concentration	1.0 µCi/g dose equivalent I-131
– Primary coolant mass (lb)	3.72 E+05
Containment purge release data	
– Containment purge flow rate (cfm)	8800
– Time to isolate purge line (seconds)	30
– Time to blow down the primary coolant system (minutes)	10
– Fraction of primary coolant iodine that becomes airborne	0.5
Core source data	
– Core activity at shutdown	See Table 15A-3
– Release of core activity to containment atmosphere (timing and fractions)	See Table 15.6.5-1
– Iodine species distribution (%)	
• Elemental	4.85
• Organic	0.15
• Particulate	95
Containment leakage release data	
– Containment volume (ft^3)	2.06 E+06
– Containment leak rate, 0-24 hr (% per day)	0.10
– Containment leak rate, > 24 hr (% per day)	0.05
– Aerosol removal efficiency due to impaction in the containment leakage path(s)/percent	80
– Elemental iodine deposition removal coefficient (hr^{-1})	1.7
– Decontamination factor limit for elemental iodine removal	200
– Removal coefficient for particulates (hr^{-1})	See Appendix 15B
Main control room model	
– Main control room volume (ft^3)	35,700
– Volume of HVAC, including main control room and control support area (ft^3)	105,500
– Normal HVAC operation (prior to switchover to an emergency mode)	
• Air intake flow (cfm)	1925
• Filter efficiency	Not applicable

Table 15.6.5-2 (Sheet 2 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Main control room model (cont.)	
– Atmospheric dispersion factors (sec/m^3)	See Table 15A-6
– Occupancy	
• 0 - 24 hr	1.0
• 24 - 96 hr	0.6
• 96 - 720 hr	0.4
– Breathing rate (m^3/sec)	3.5 E-04
Control room with emergency habitability system credited	
– Main control room activity level at which the emergency habitability system is actuated (Ci/m^3 of dose equivalent I-131)	2.0 E-6
– Time to switch from normal HVAC to emergency habitability system operation after actuation signal generation (sec)	30
– Interval with operation of the emergency habitability system	
• Flow from compressed air bottles of the emergency habitability system (cfm)	60
• Effective unfiltered inleakage via ingress/egress (cfm)	5.0
• 0 - 24 hr	0.531
• > 24 hr	2.654
• Recirculation flow (cfm)	Not applicable
– Time at which the compressed air supply of the emergency habitability system is depleted (hr)	72
– After depletion of emergency habitability system bottled air supply (>72 hr)	
• Air intake flow (cfm)	1700
• Intake flow filter efficiency (%)	Not applicable
• Recirculation flow (cfm)	Not applicable
– Time at which the compressed air supply is restored and emergency habitability system returns to operation (hr)	168
Control room with credit for continued operation of HVAC	
– Time delay to switch from normal operation to the supplemental air filtration mode (sec)	30
– Filtered air intake flow (cfm)	860
– Filtered air recirculation flow (cfm)	2740
– Filter efficiency (%)	
• Elemental iodine	90
• Organic iodine	90
• Particulates	99

Table 15.6.5-2 (Sheet 3 of 3)

ASSUMPTIONS AND PARAMETERS USED IN CALCULATING RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT

Miscellaneous assumptions and parameters	
– Unfiltered air inleakage (cfm)	90
– Offsite power	Not applicable
– Atmospheric dispersion factors (offsite)	See Table 15A-5
– Nuclide dose conversion factors	See Table 15A-4
– Nuclide decay constants	See Table 15A-4
– Offsite breathing rate (m^3/sec)	
0 - 8 hr	3.5 E-04
8 - 24 hr	1.8 E-04
24 - 720 hr	2.3 E-04

Table 15.6.5-3

RADIOLOGICAL CONSEQUENCES OF A LOSS-OF-COOLANT ACCIDENT WITH CORE MELT

	TEDE Dose (rem)
Exclusion zone boundary dose (1.2 - 3.2 hr)(1)	11.5
Low population zone boundary dose (0 - 30 days)	14.2
Main control room dose (emergency habitability system in operation)	
– Airborne activity entering the main control room	2.8 rem
– Direct radiation from adjacent structures	0.15 rem
– Sky-shine	0.01 rem
– Total	2.96 rem
Main control room dose (normal HVAC operating in the supplemental filtration mode)(2)	
– Airborne activity entering the main control room	2.91 rem
– Direct radiation from adjacent structures	0.15 rem
– Sky-shine	0.01 rem
– Total	4.07 rem

Note:
1. This is the 2-hour period having the highest dose.
2. Although the dose is reported for the case in which the normal HVAC operates in the supplemental filtration mode, for the design basis accident it is projected that the system would not maintain the activity in the main control room air low enough to avoid actuation of the emergency habitability system. A “realistic” event could be addressed without actuation of the emergency habitability system.

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.

Table 15.6.5-5

AP1000 LOCA CHRONOLOGY

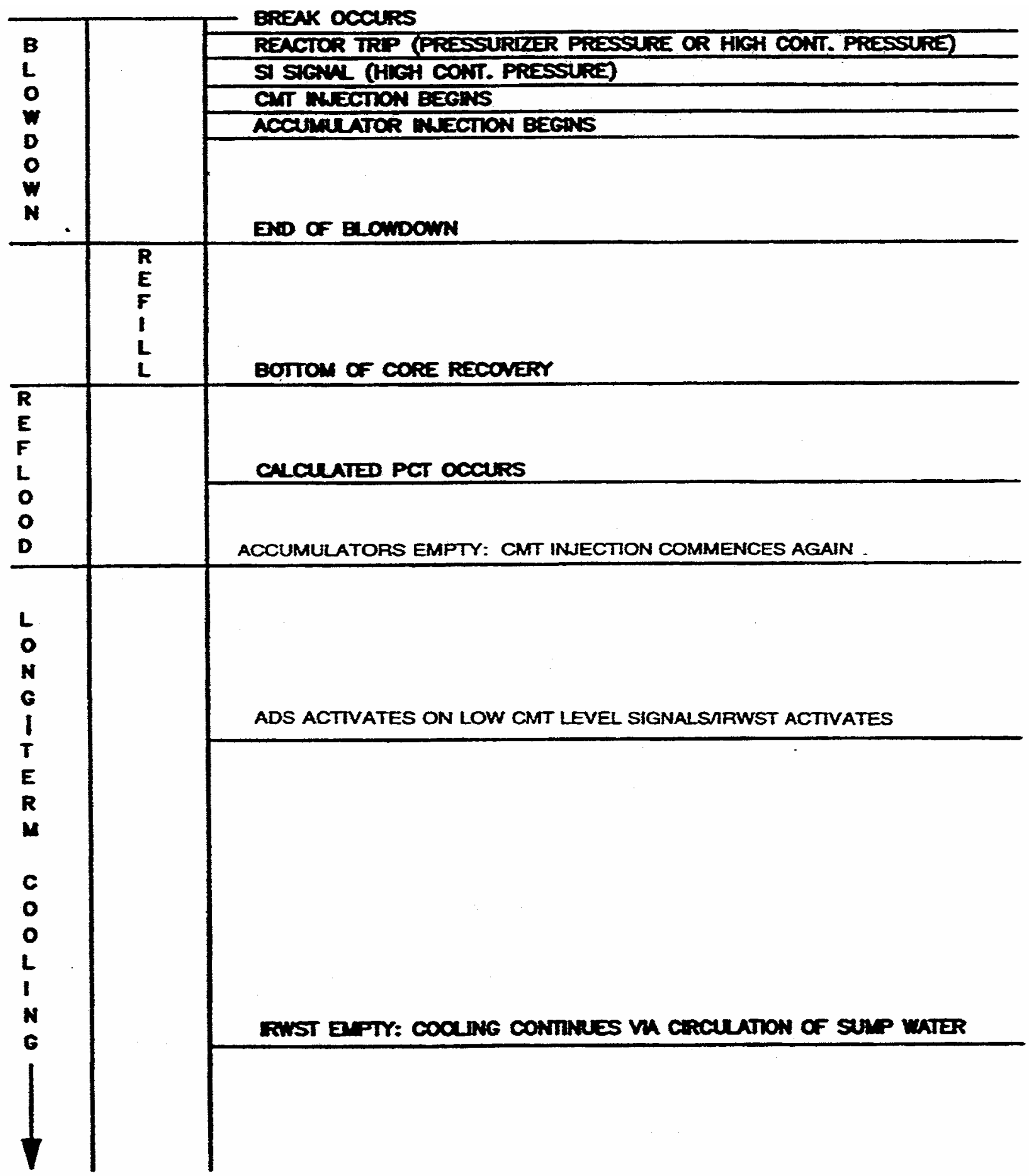

Table 15.6.5-6

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-2 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 15.6.5-8

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 local cladding oxidation (%)	< 12.9	17
Maximum core-wide cladding oxidation (%)	0.73	1
Coolable geometry	Core remains coolable	Core remains coolable
Long-term cooling	Core remains cool in long term	Core remains cool in long term

Table 15.6.5-9

INITIAL CONDITIONS FOR AP1000 SMALL-BREAK LOCA ANALYSIS

Condition	Calculation	Nominal Steady-state
Pressurizer pressure (psia)	2303.1	2300
Vessel inlet temperature (°F)	534.03	534.3
Vessel outlet temperature (°F)	612.83	612.9
Vessel flow rate (lb/sec)	31086	31089
Steam generator pressure (psia)	806.5	788.5

Table 15.6.5-10

AP1000 ADS PARAMETERS

Actuation Signal (percentage of core makeup tank level)		**Actuation Time (seconds)**	**Minimum Valve Flow Area (for each path, in^2)**	**Number of Paths**	**Valve Opening Time (seconds)**
Stage 1 – Control Low 1	67.5	20 after CMT-Low 1	4.6	2 out of 2	≤ 30
Stage 2 – Control		70 after Stage 1	21	2 out of 2	≤ 80
Stage 3 – Control		120 after Stage 2	21	2 out of 2	≤ 80
Stage 4A	20	120 after Stage 3	67	1 out of 2	≤ 2
Stage 4B		60 after Stage 4A	67	2 out of 2	≤ 2

Table 15.6.5-11

INADVERTENT ADS DEPRESSURIZATION SEQUENCE OF EVENTS

Event	AP1000 Time (seconds)
Inadvertent opening of ADS valves	0.0
Reactor trip signal	37.8
Steam turbine stop valves close	43.8
"S" signal	44.1
Main feed isolation valves begin to close	49.1
Reactor coolant pumps start to coast down	50.1
ADS Stage 2	70.0
ADS Stage 3	190.0
Accumulator injection starts	268
Accumulator empties	693
ADS Stage 4	1746
Core makeup tank empty	2112
IRWST injection starts	2663

Table 15.6.5-12

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

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

Table 15.6.5-13

DOUBLE-ENDED INJECTION LINE BREAK SEQUENCE OF EVENTS – 20 psi

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	13.1
Steam turbine stop valves close	19.1
“S” signal	18.6
Main feed isolation valves begin to close	20.6
Reactor coolant pumps start to coast down	24.6
ADS Stage 1	182.5
ADS Stage 2	252.5
Intact accumulator injection starts	254
ADS Stage 3	372.5
ADS Stage 4	492.5
Intact accumulator empties	600.0
Intact loop IRWST injection starts*	1470
Intact loop core makeup tank empties	2123

Note:
*Continuous injection period

Table 15.6.5-13A

DOUBLE-ENDED INJECTION LINE BREAK SEQUENCE OF EVENTS – 14.7 psi

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

Note:
*Continuous injection period

Table 15.6.5-14

10-INCH COLD LEG BREAK IN SEQUENCE OF EVENTS

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

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

Table 15.6.5-15

DOUBLE-ENDED INJECTION LINE BREAK SEQUENCE OF EVENTS (ENTRAINMENT SENSITIVITY)

Event	AP1000 Time (seconds)
Break opens	0.0
Reactor trip signal	13.1
Steam turbine stop valves close	19.1
“S” signal	18.6
Main feed isolation valves begin to close	20.6
Reactor coolant pumps start to coast down	24.6
ADS Stage 1	182.4
ADS Stage 2	252.4
Intact accumulator injection starts	255
ADS Stage 3	372.4
ADS Stage 4	492.4
Intact accumulator empties	608.9
Intact loop IRWST injection starts*	1718
Intact loop core makeup tank empties	2106

Note:
*Continuous injection period

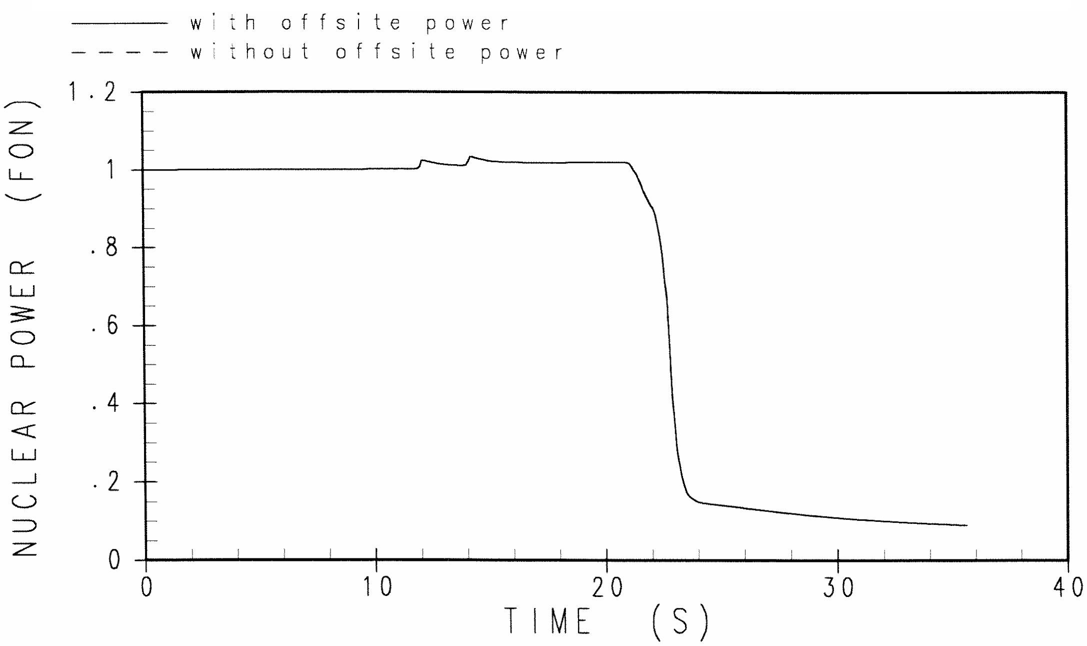

Figure 15.6.1-1

Nuclear Power Transient
Inadvertent Opening of a Pressurizer Safety Valve

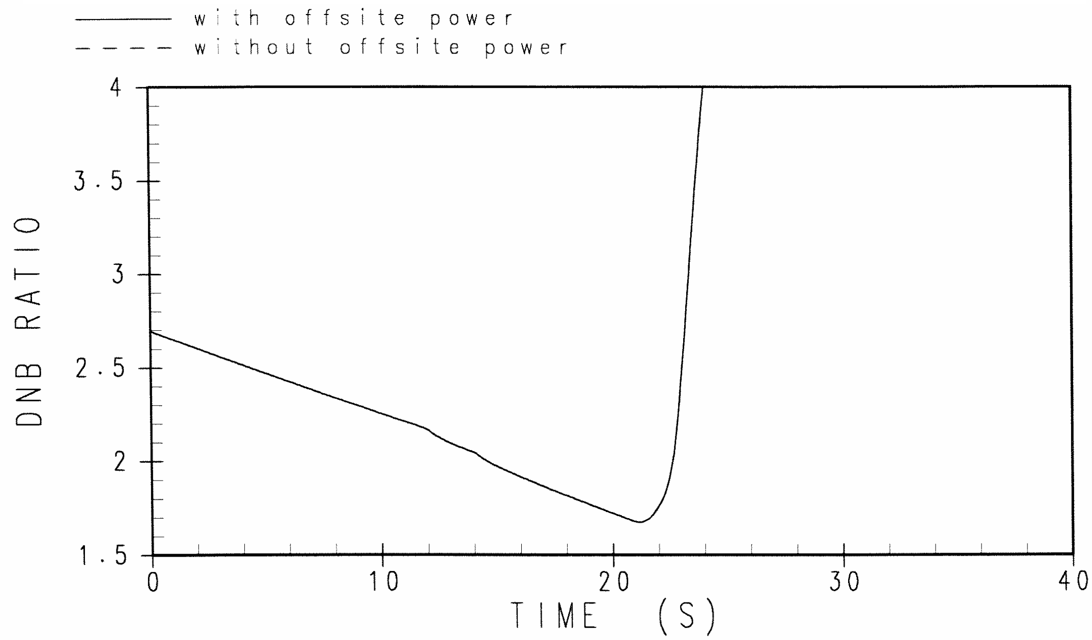

Figure 15.6.1-2

DNBR Transient
Inadvertent Opening of a Pressurizer Safety Valve

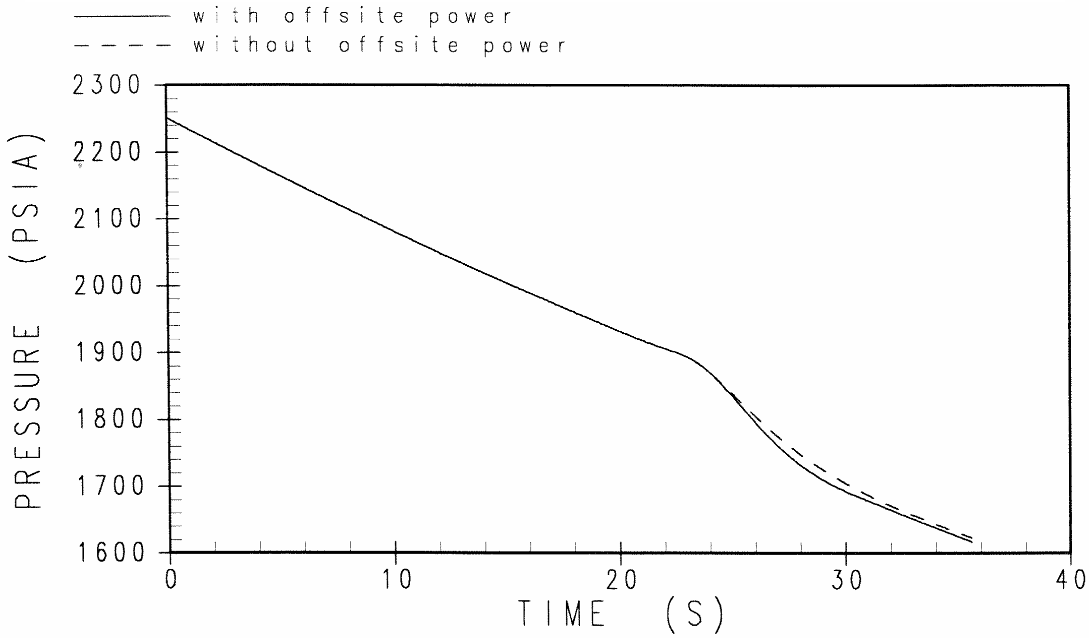

Figure 15.6.1-3

Pressurizer Pressure Transient
Inadvertent Opening of a Pressurizer Safety Valve

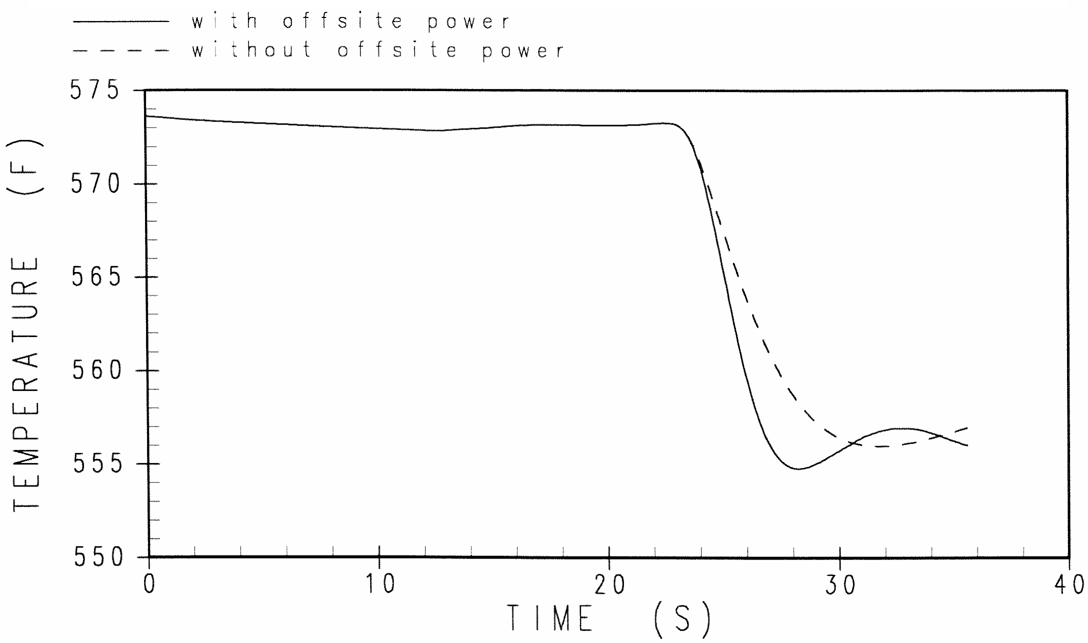

Figure 15.6.1-4

Vessel Average Temperature
Inadvertent Opening of a Pressurizer Safety Valve

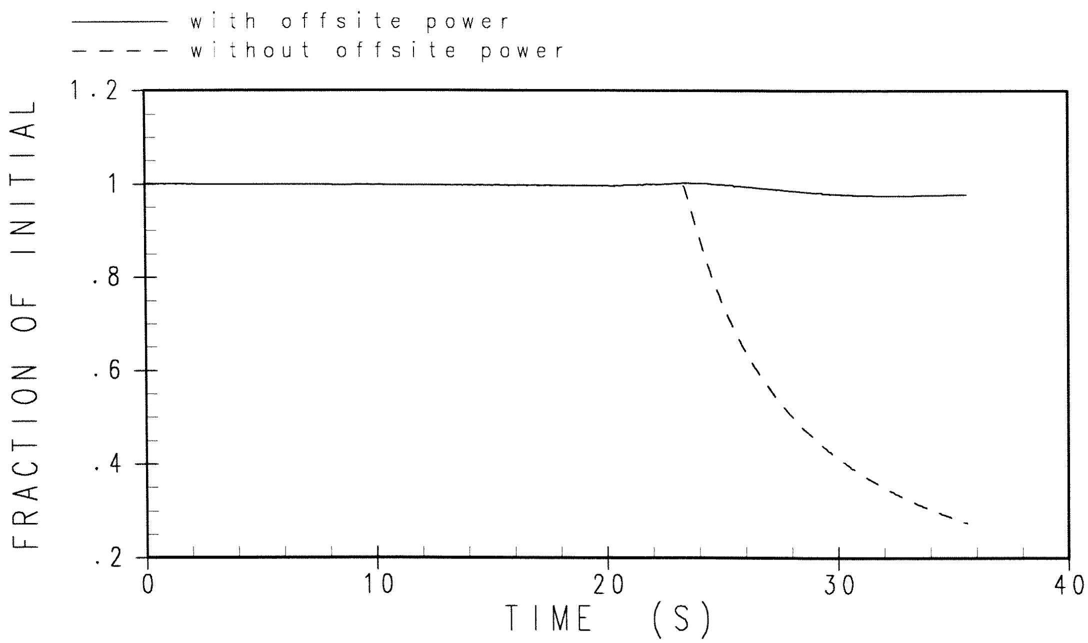

Figure 15.6.1-5

Core Mass Flow Rate
Inadvertent Opening of a Pressurizer Safety Valve

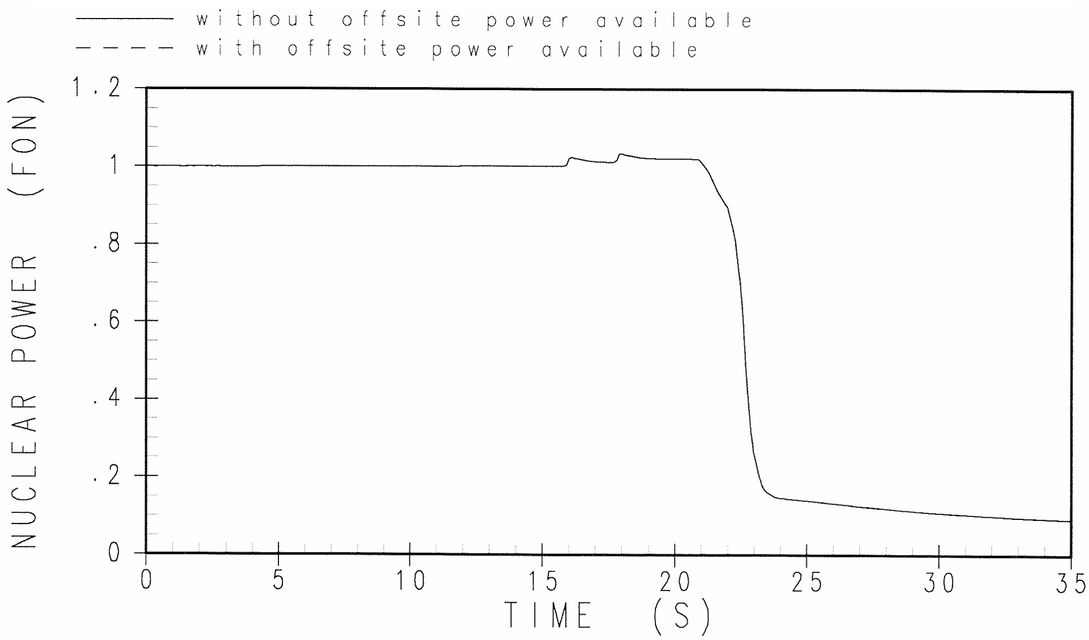

Figure 15.6.1-6

Nuclear Power Transient
Inadvertent Opening of Two ADS Stage 1 Trains

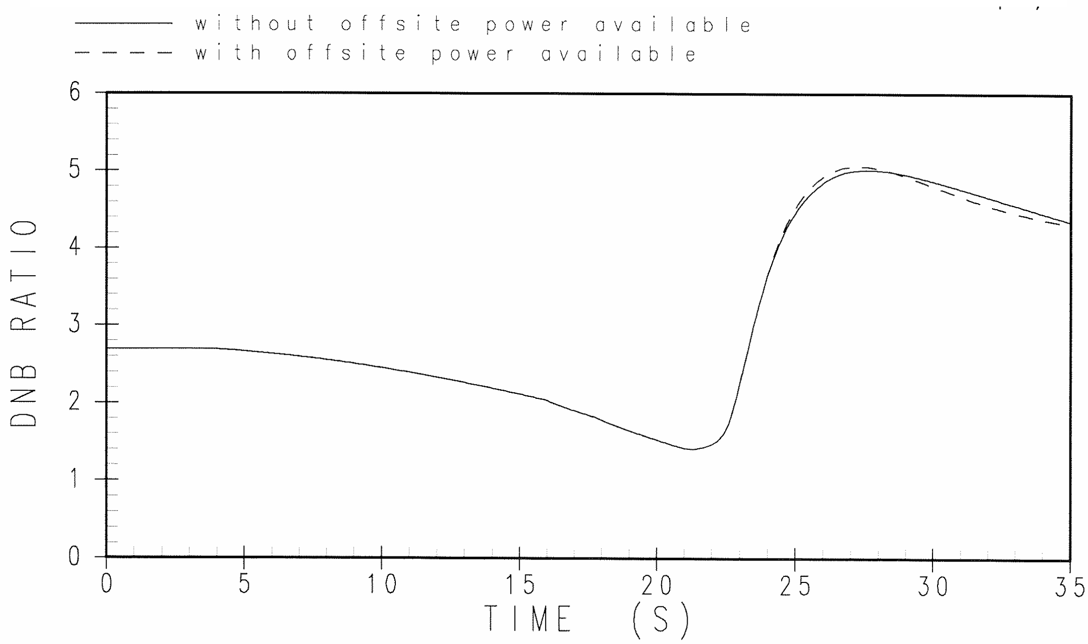

Figure 15.6.1-7

DNBR Transient
Inadvertent Opening of Two ADS Stage 1 Trains

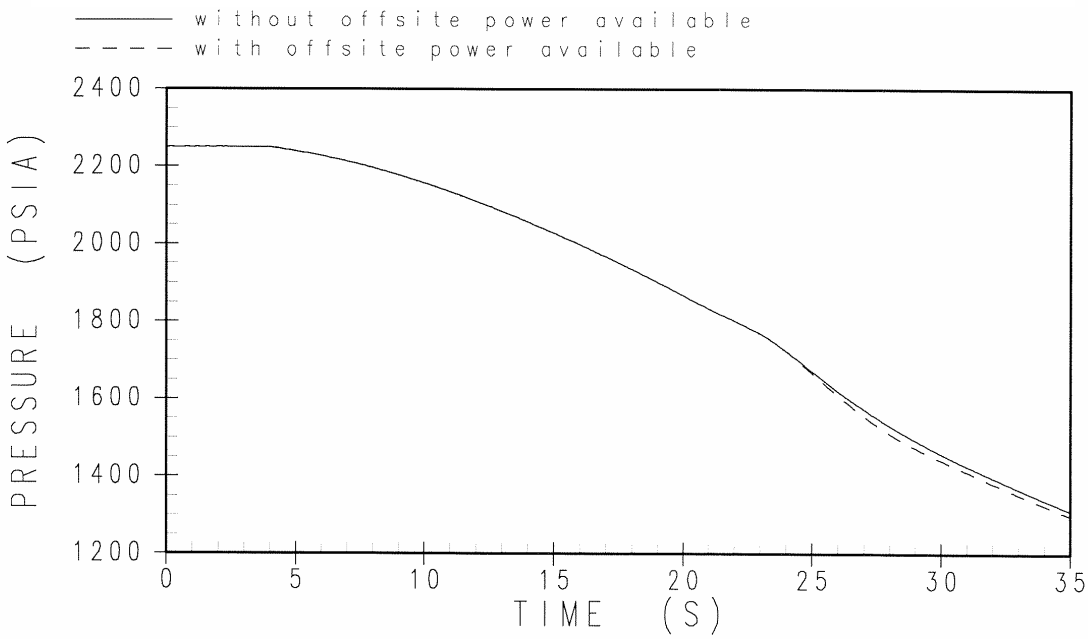

Figure 15.6.1-8

Nuclear Power Transient
Inadvertent Opening of Two ADS Stage 1 Trains

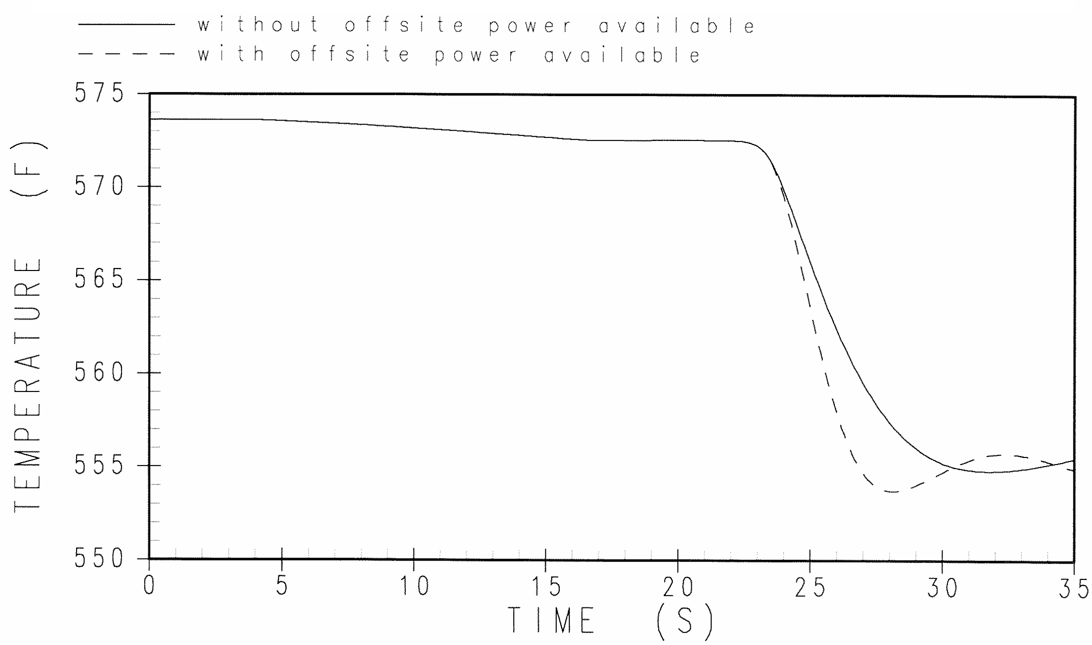

Figure 15.6.1-9

Nuclear Power Transient
Inadvertent Opening of Two ADS Stage 1 Trains

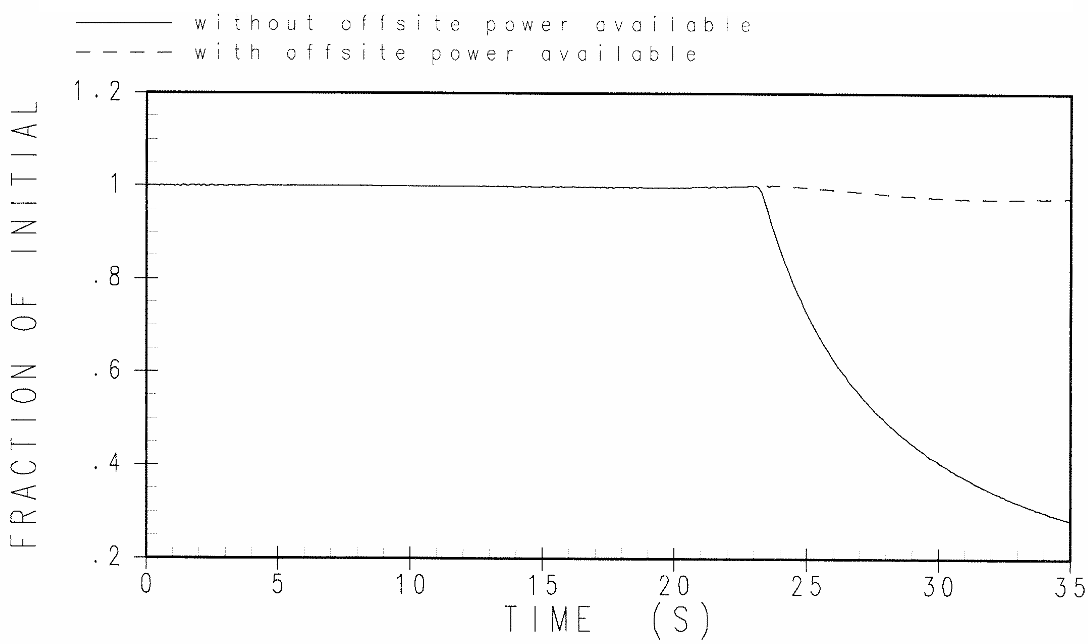

Figure 15.6.1-10

Core Mass Flow Rate
Inadvertent Opening of Two ADS Stage 1 Trains

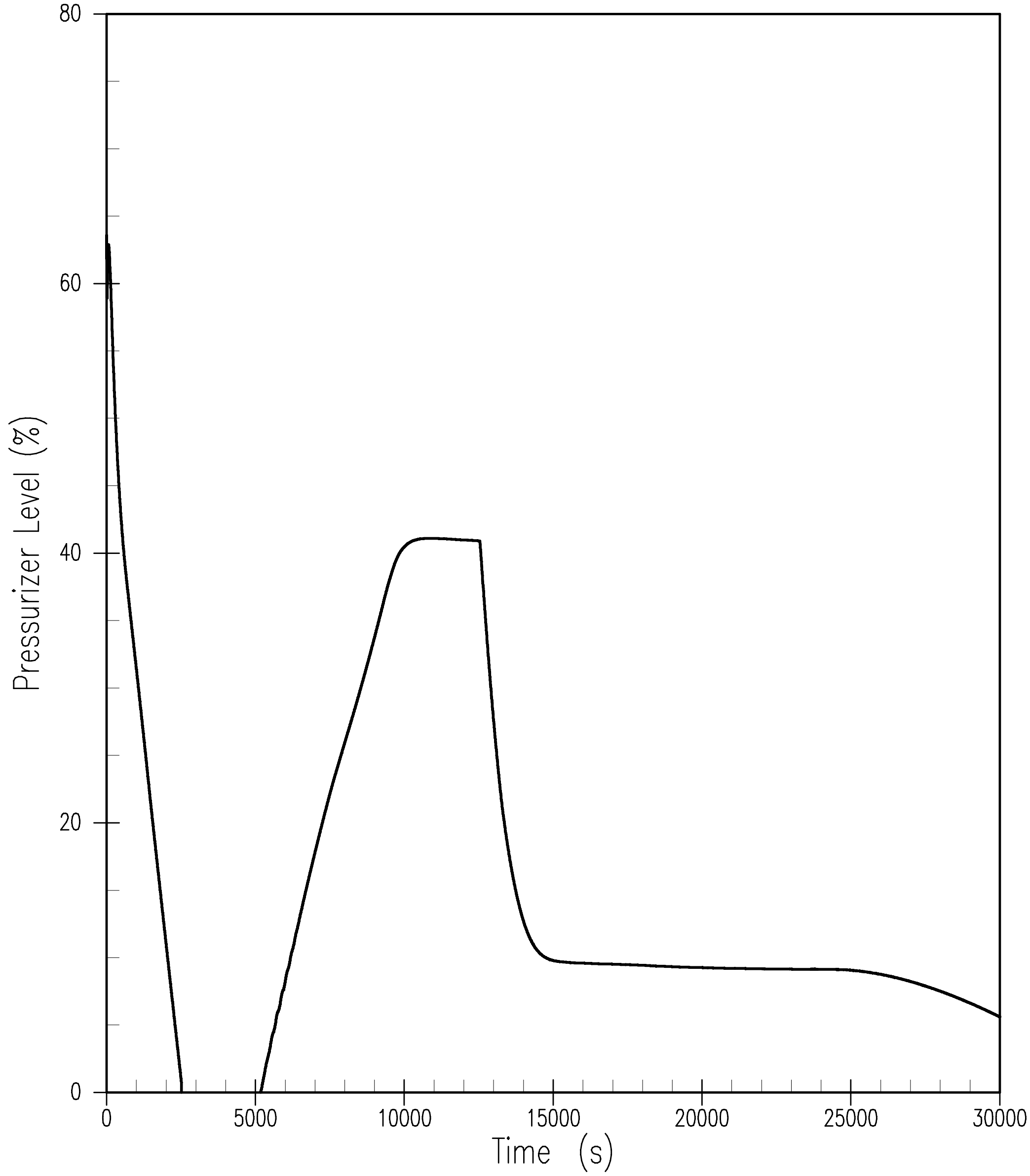

Figure 15.6.3-1

Pressurizer Level for SGTR

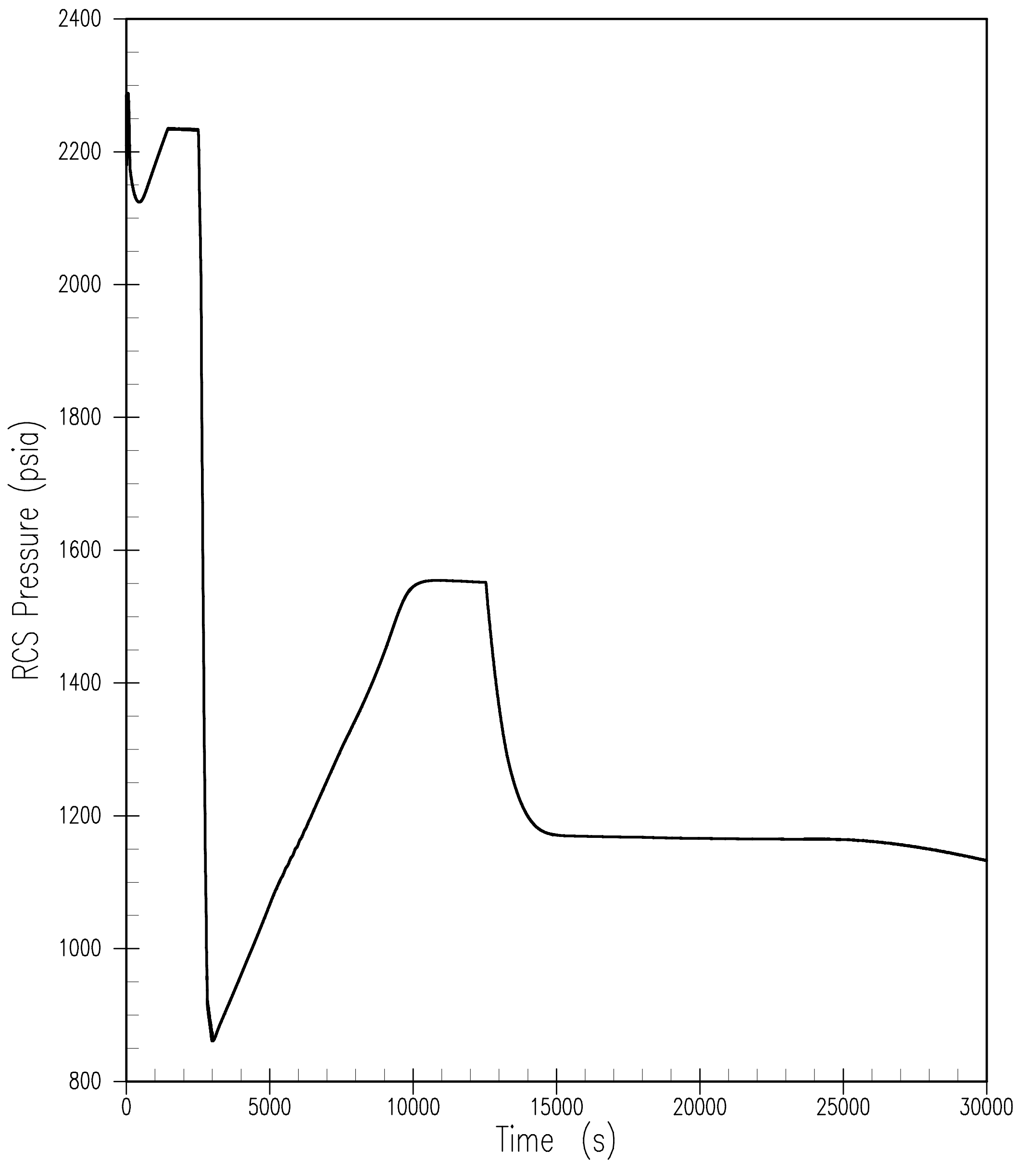

Figure 15.6.3-2

Reactor Coolant System Pressure for SGTR

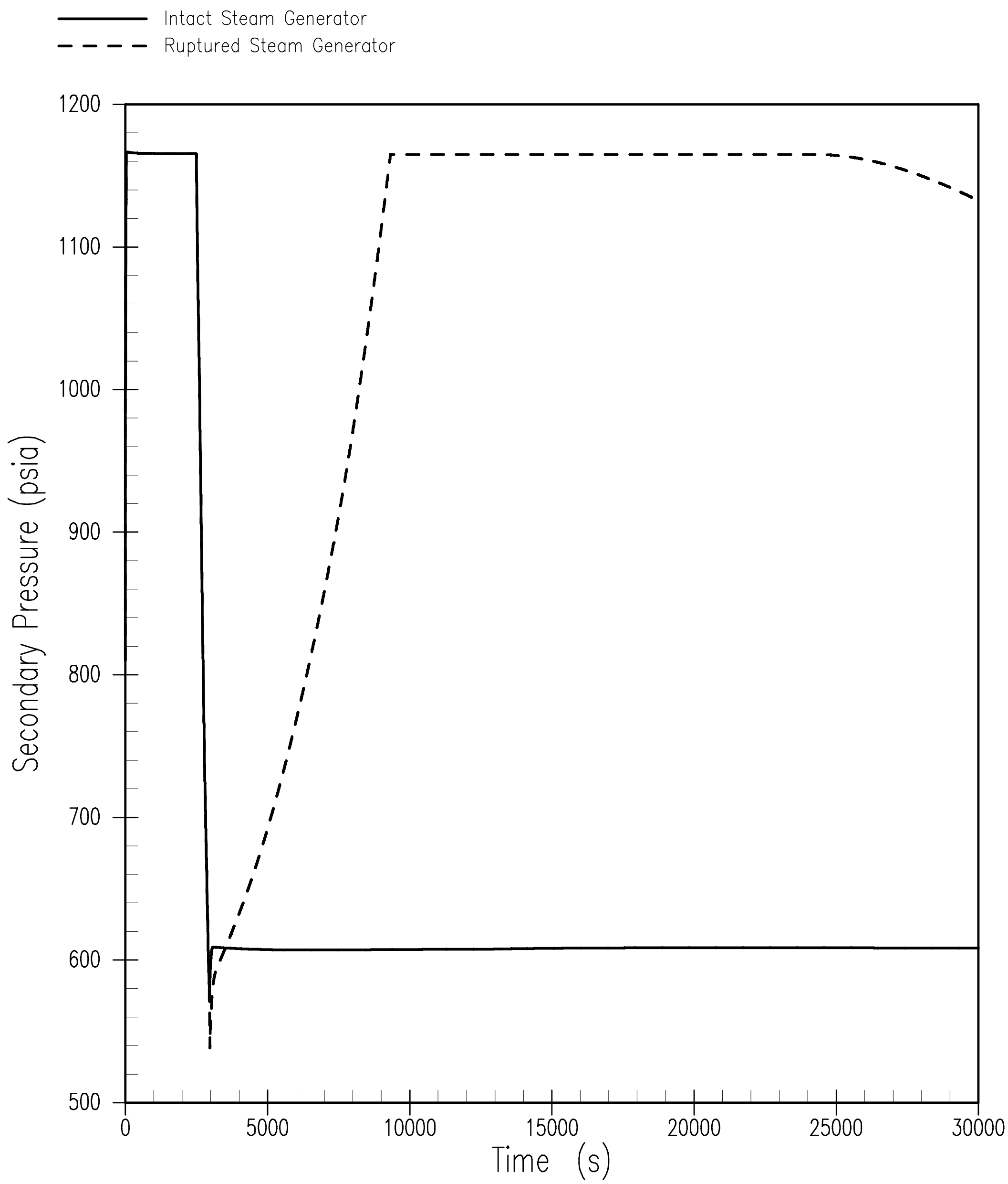

Figure 15.6.3-3

Secondary Pressure for SGTR

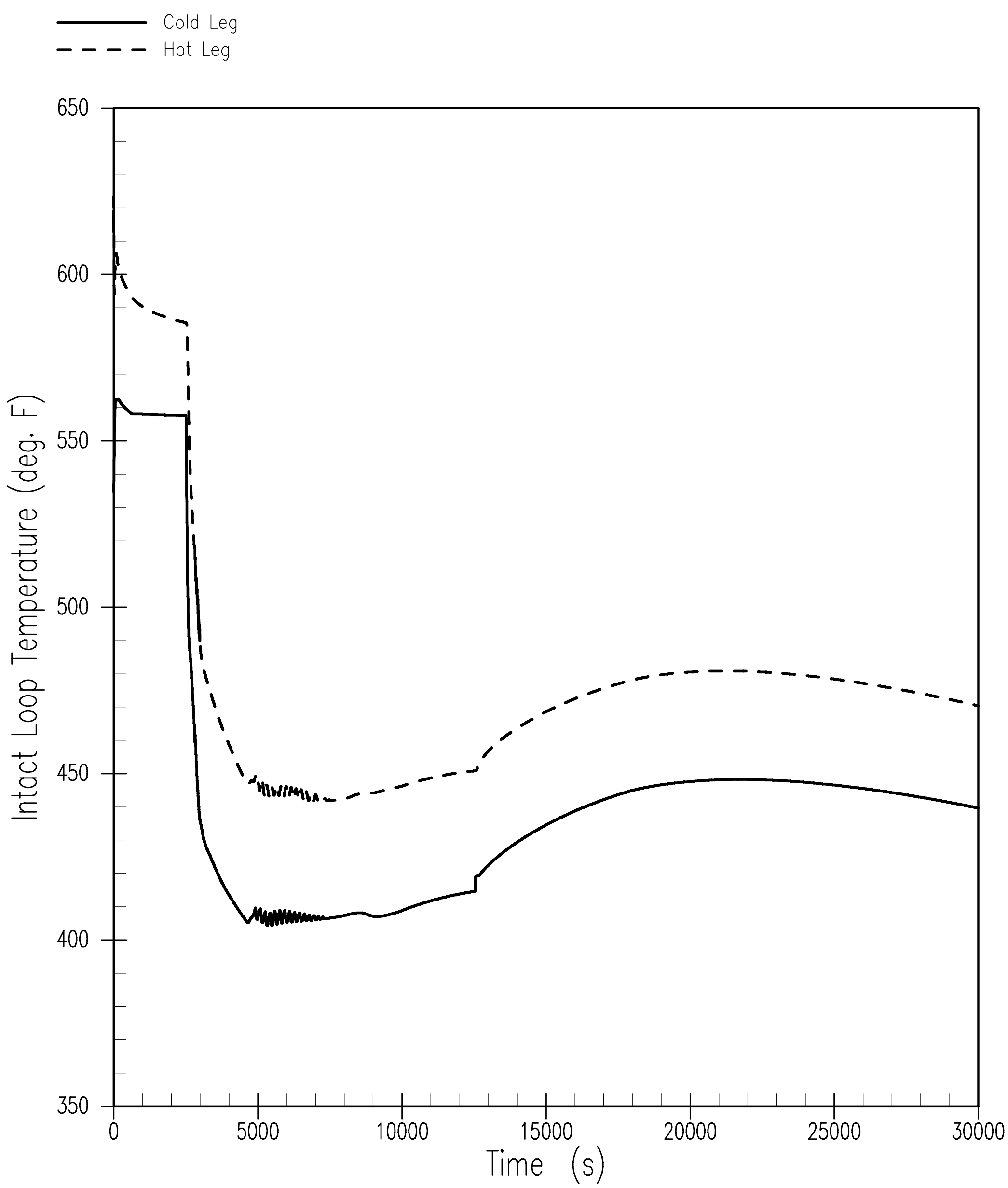

Figure 15.6.3-4

Intact Loop Hot and Cold Leg
Reactor Coolant System Temperature for SGTR

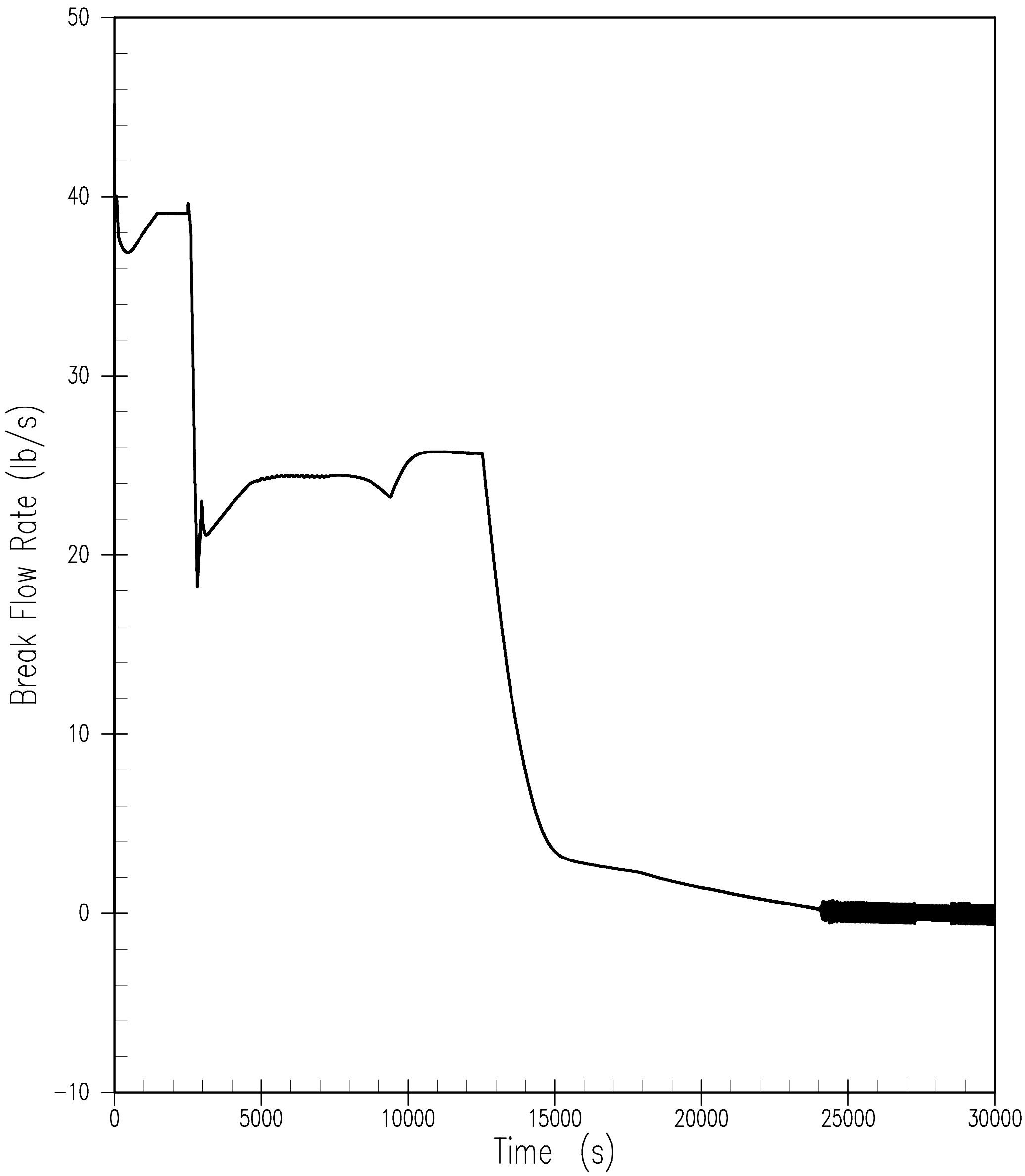

Figure 15.6.3-5

Primary-to-Secondary Break Flow Rate for SGTR

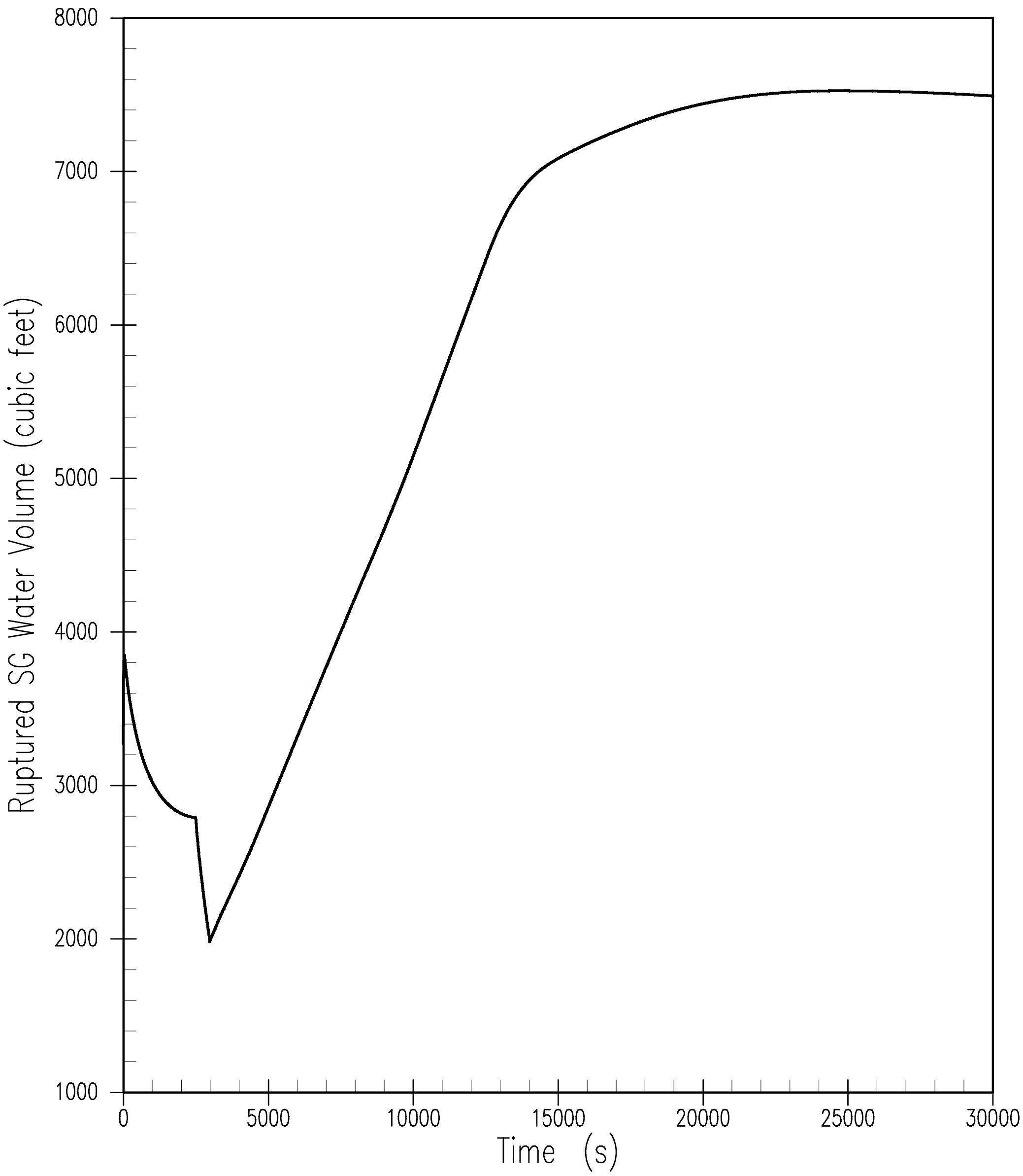

Figure 15.6.3-6

Faulted Steam Generator Water Volume for SGTR

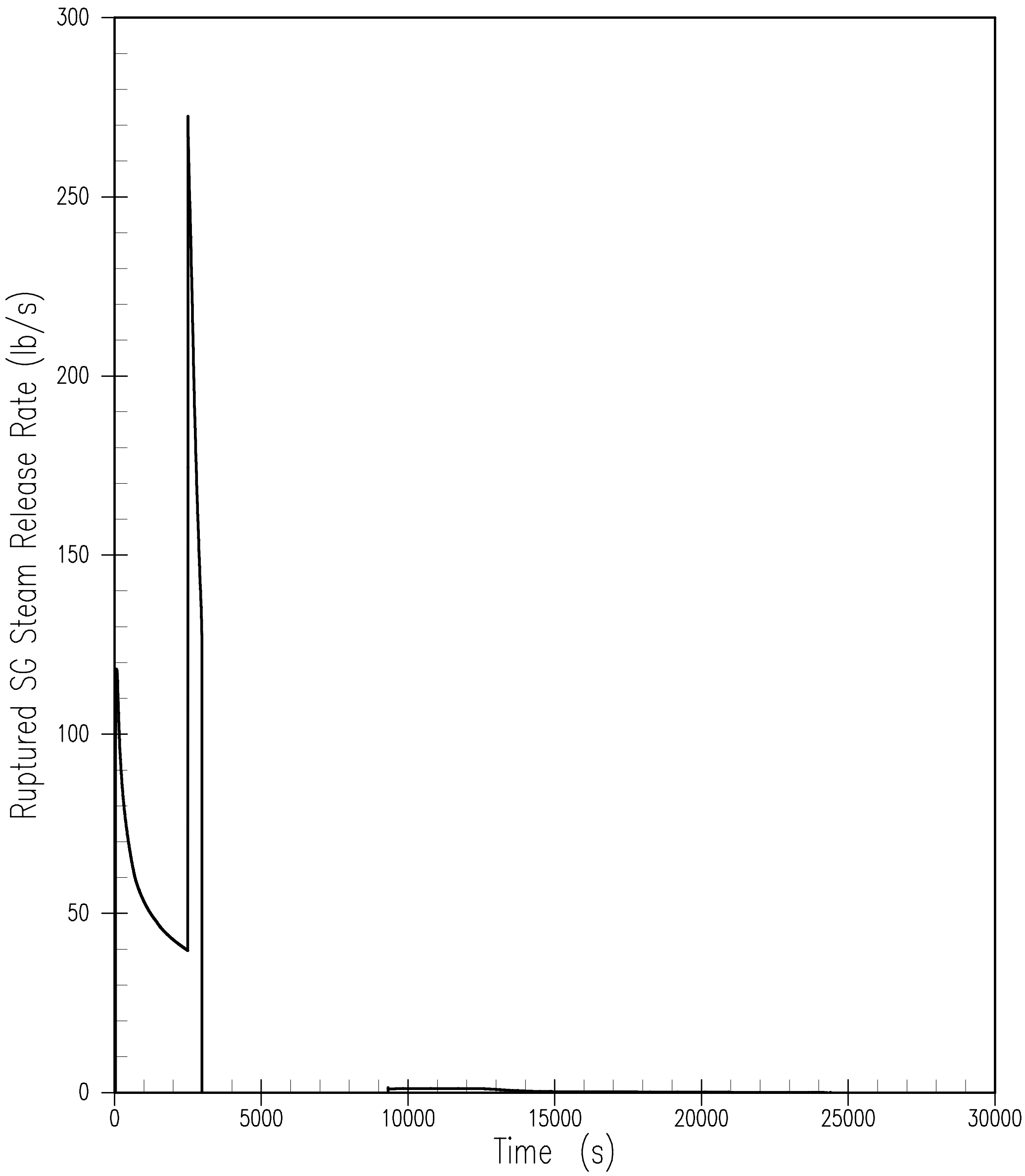

Figure 15.6.3-7

Faulted Steam Generator Mass Release Rate to the Atmosphere for SGTR

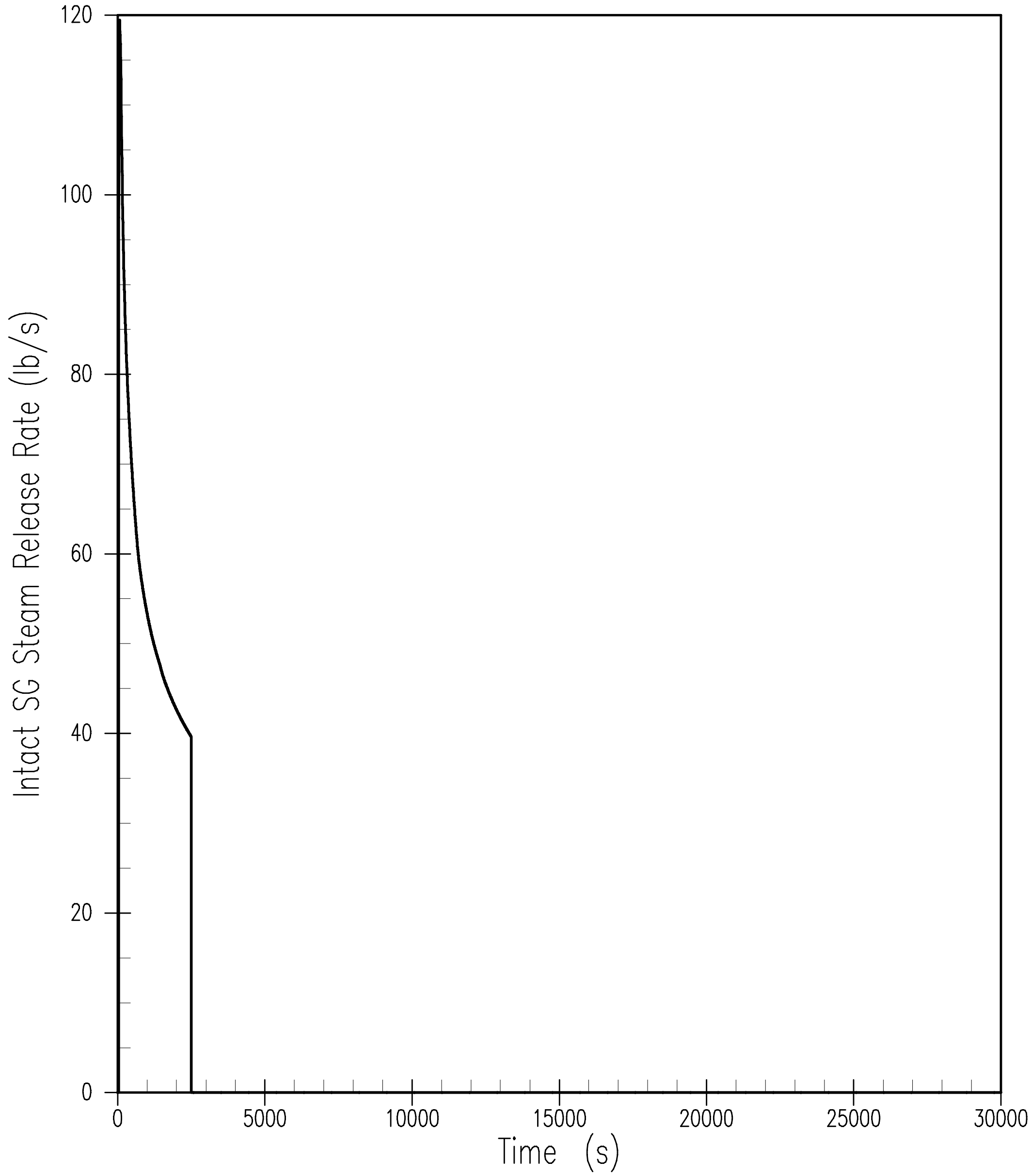

Figure 15.6.3-8

Intact Steam Generator Mass Release Rate to the Atmosphere for SGTR

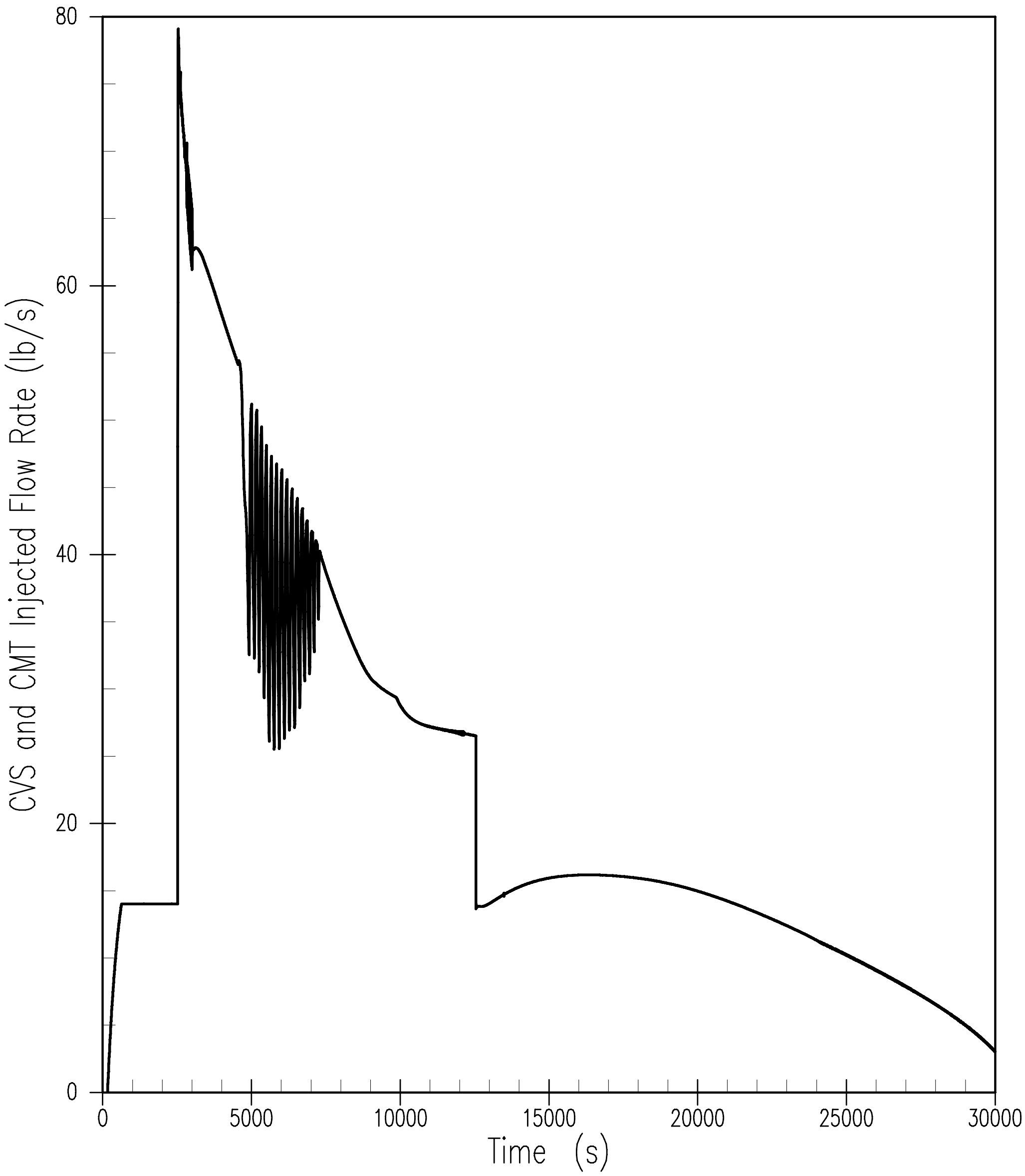

Figure 15.6.3-9

Faulted Loop Chemical and Volume Control System and Core Makeup Tank Injection Flow for SGTR

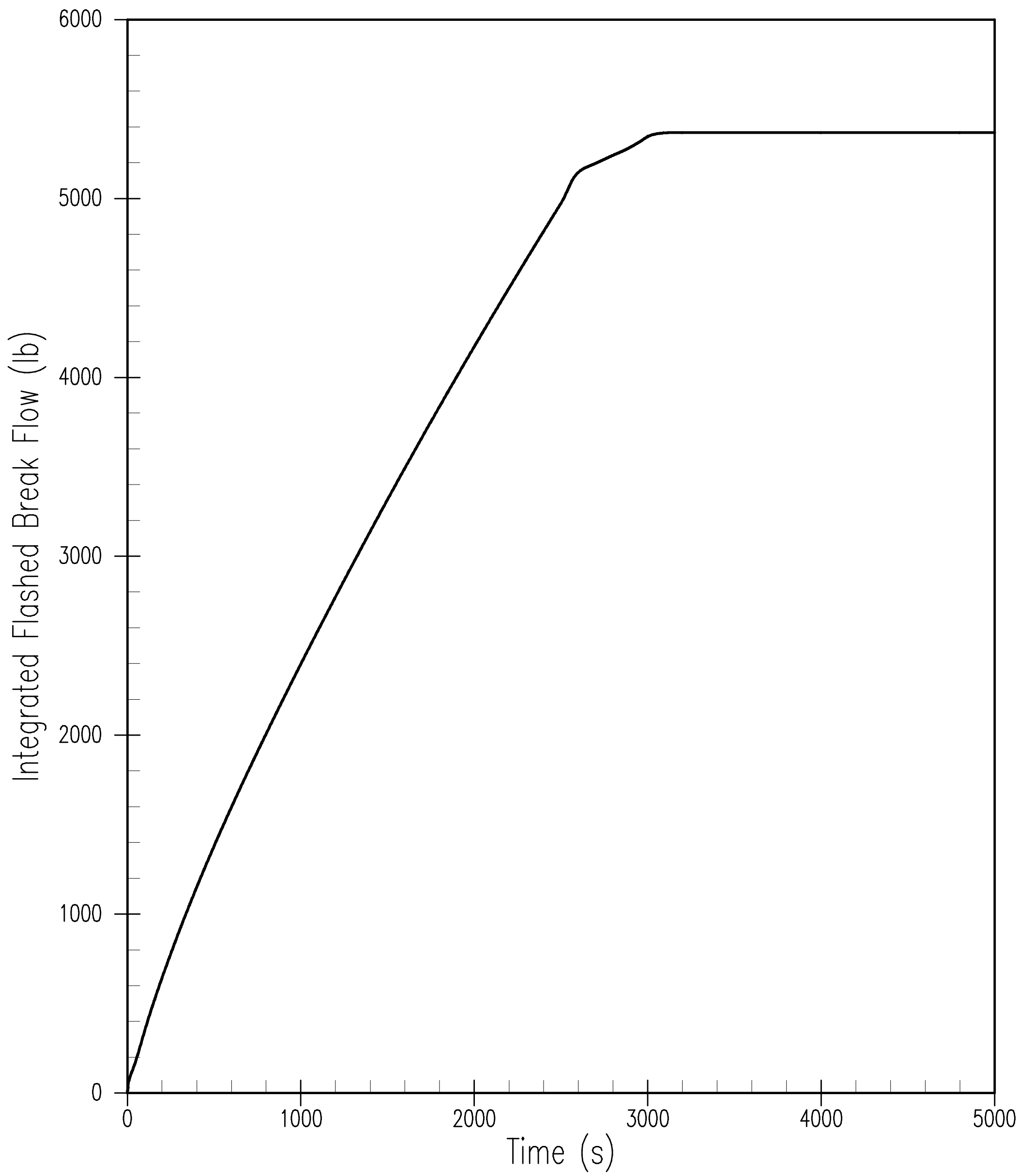

Figure 15.6.3-10

Integrated Flashed Break Flow for SGTR

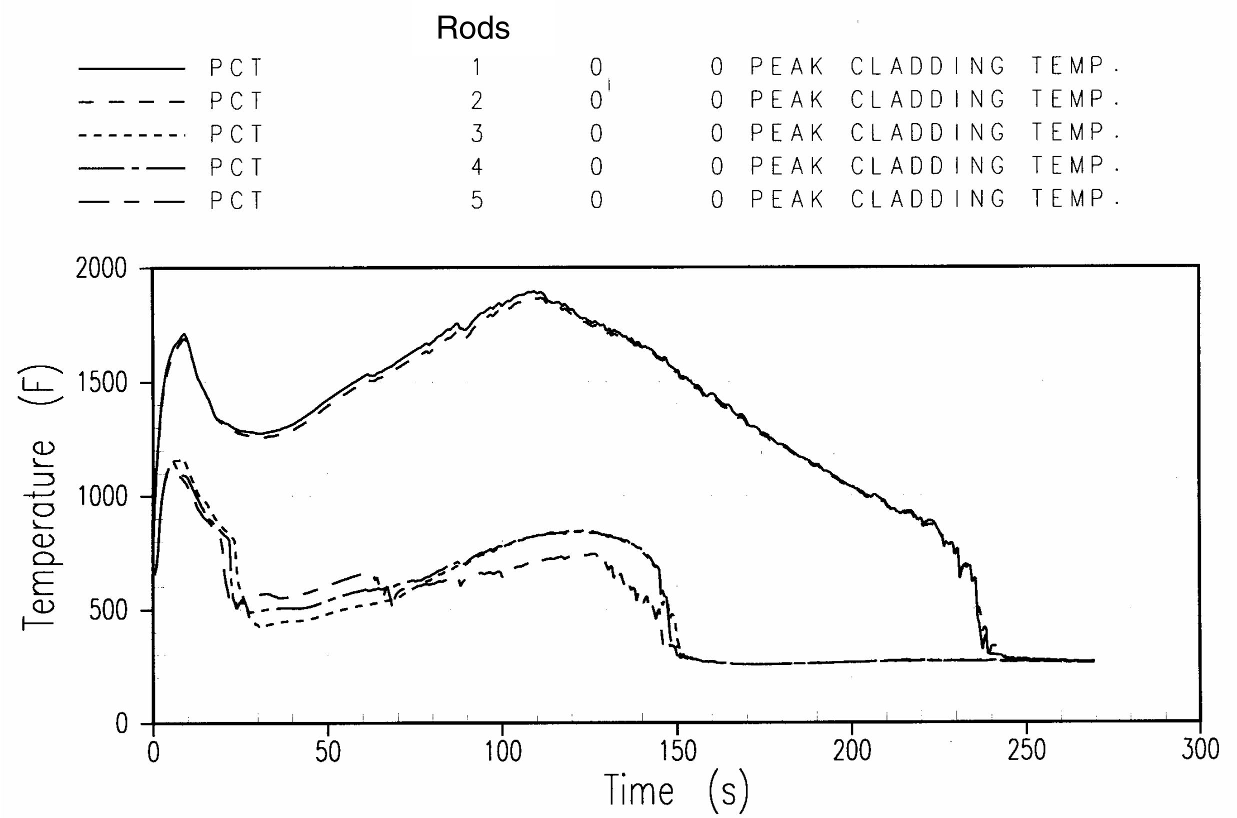

Figure 15.6.5.4A-1

PCT Among All Elevations for Each Fuel Rod

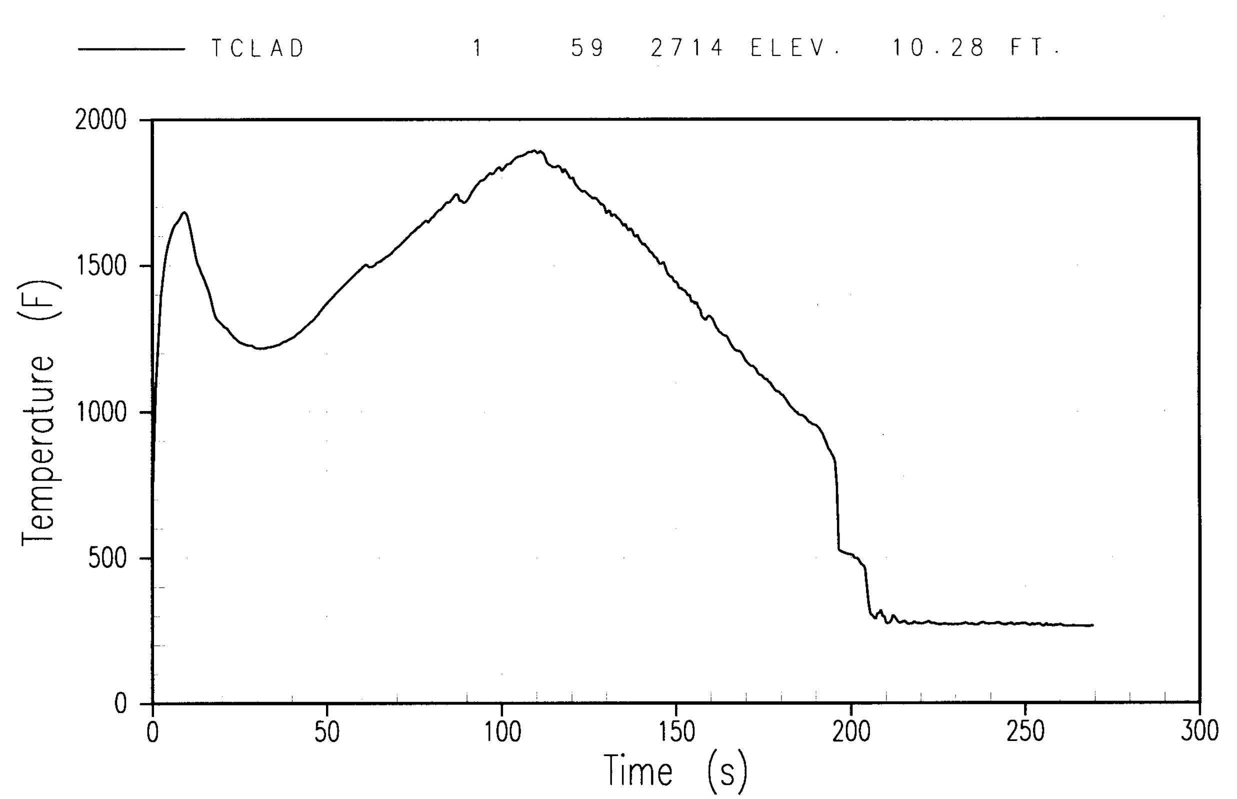

Figure 15.6.5.4A-2

Hot Rod Cladding Temperature Transient, PCT Elevation

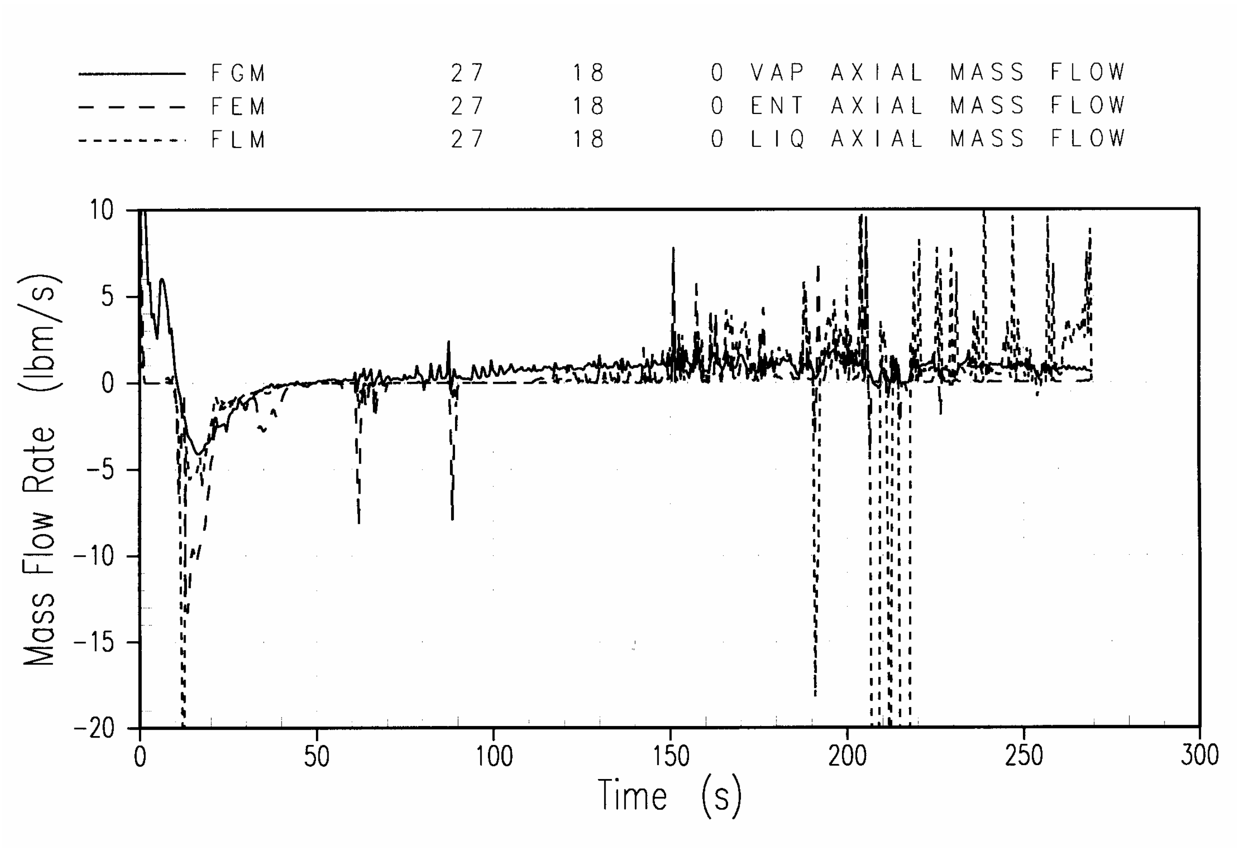

Figure 15.6.5.4A-3

Hot Assembly Exit Vapor, Entrained Drop, Liquid Flow Rates

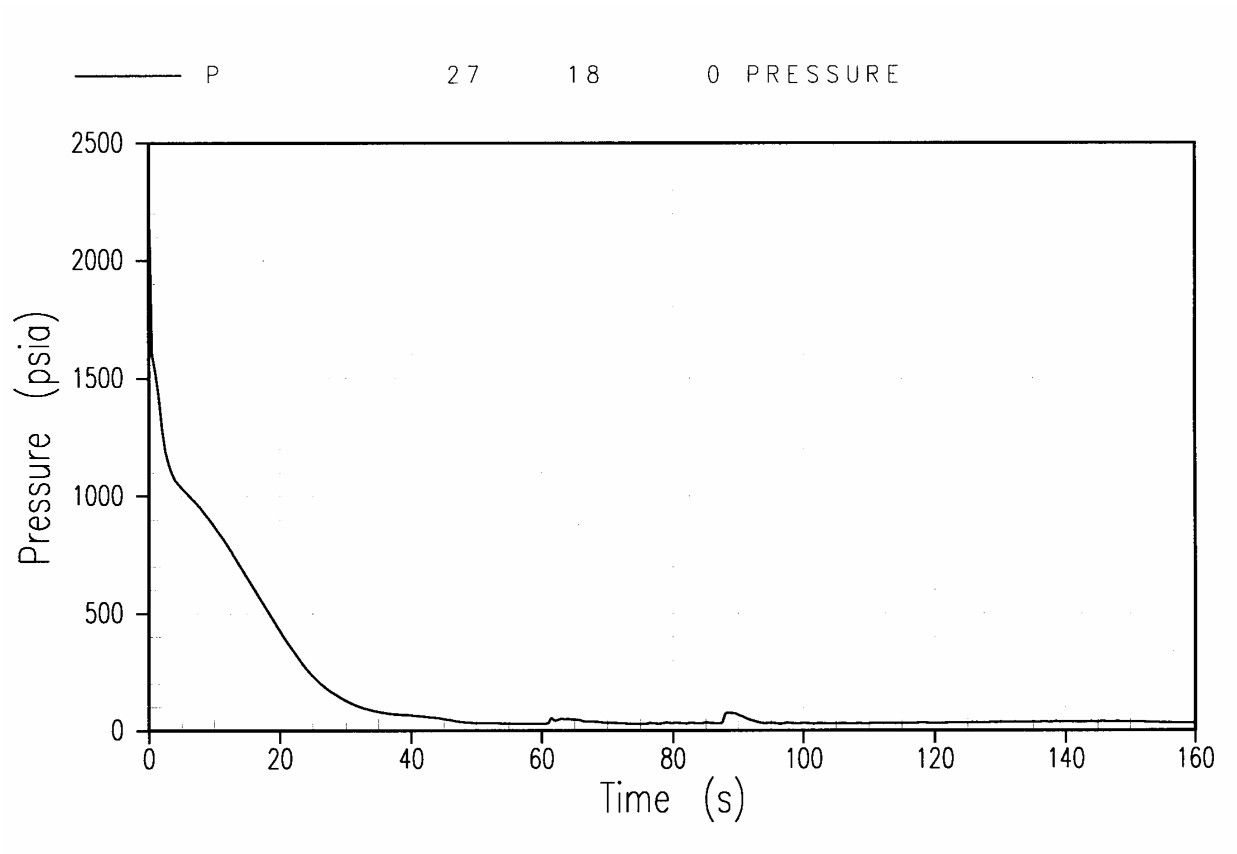

Figure 15.6.5.4A-4

Core Pressure

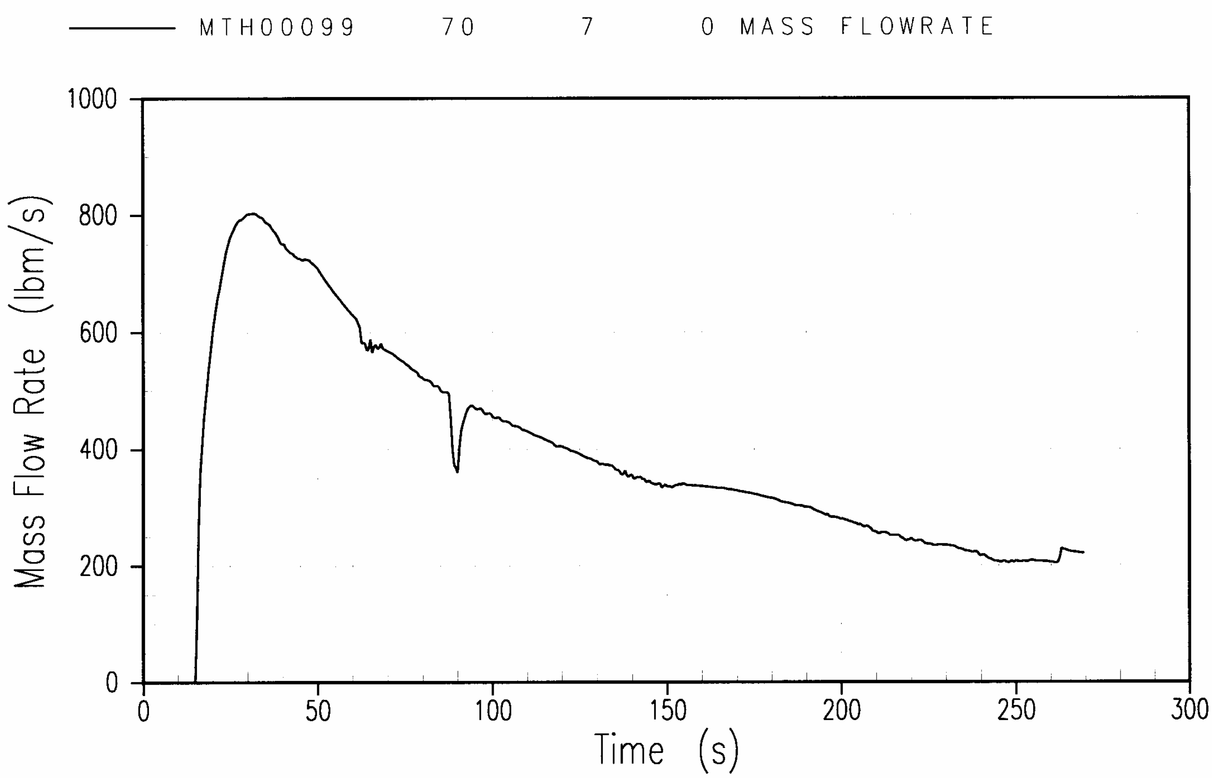

Figure 15.6.5.4A-5

Accumulator Flow Rate

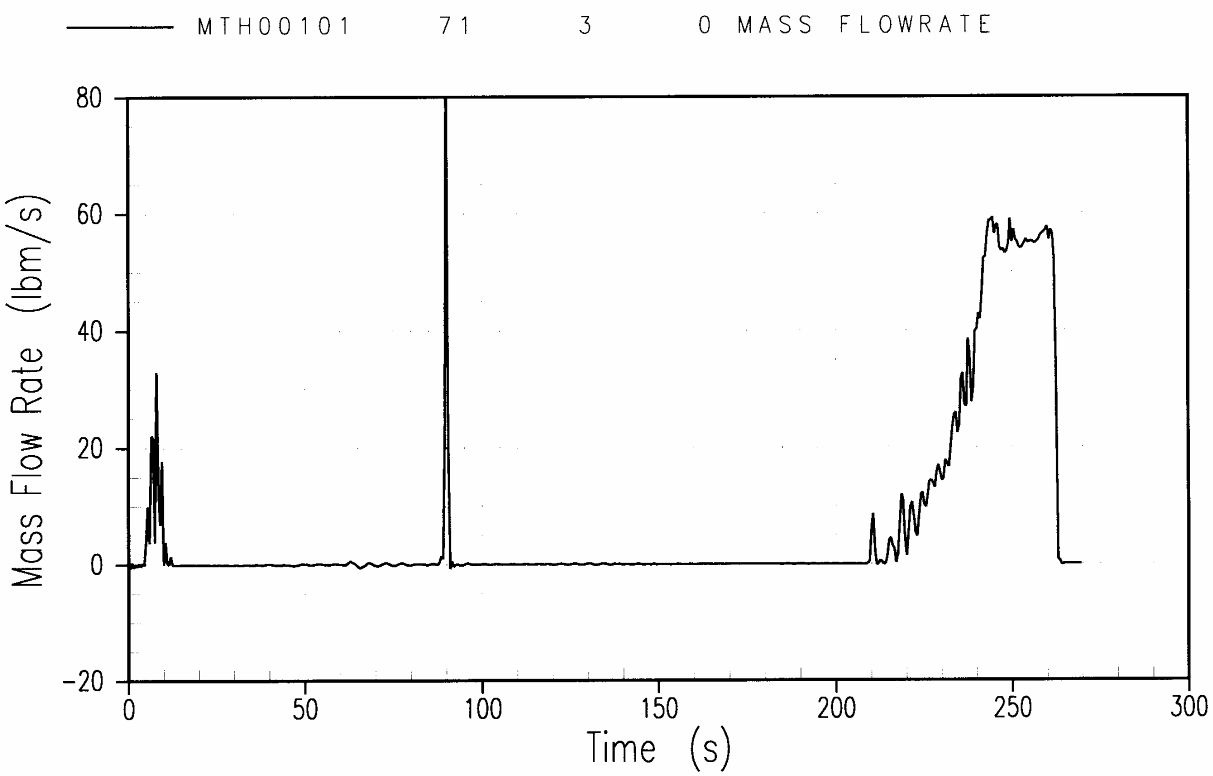

Figure 15.6.5.4A-6

Intact Loop Core Makeup Tank Flow Rate

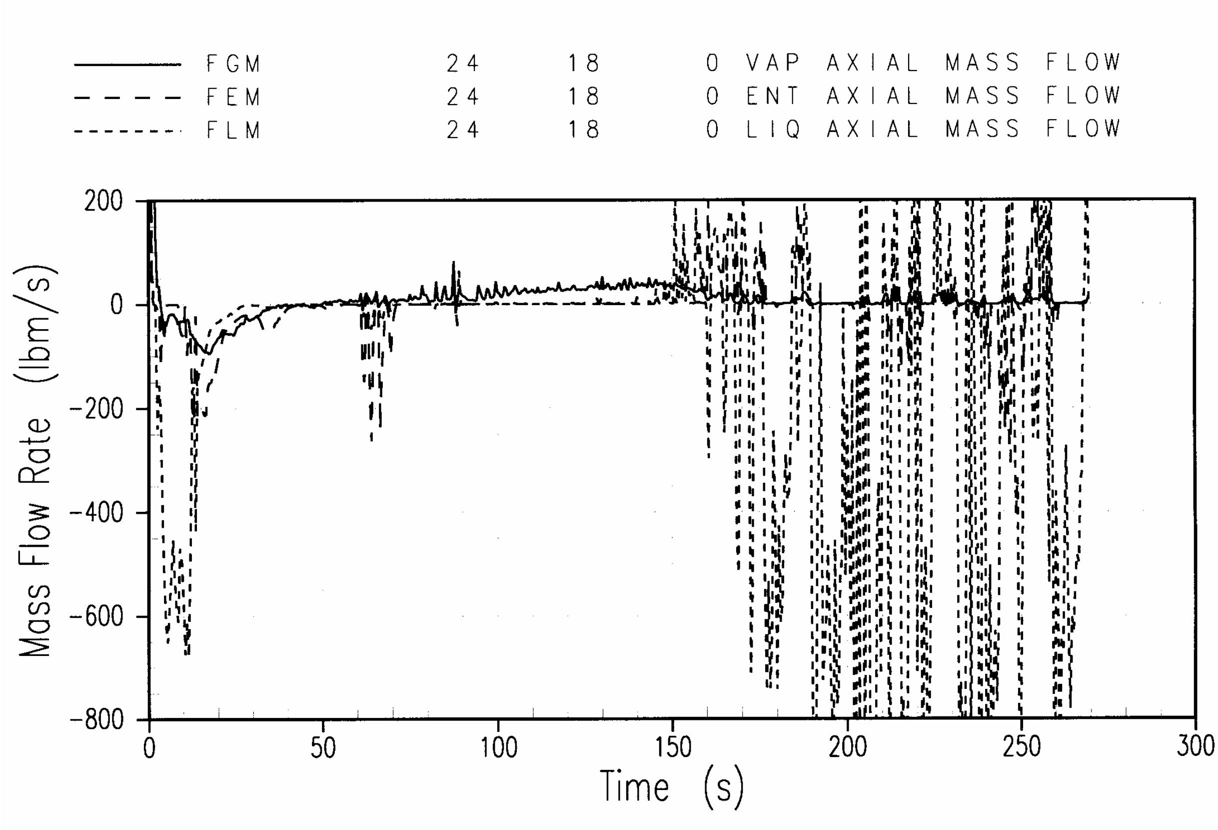

Figure 15.6.5.4A-7

Peripheral Assemblies Exit Vapor, Entrained Drop, Liquid Flow Rates

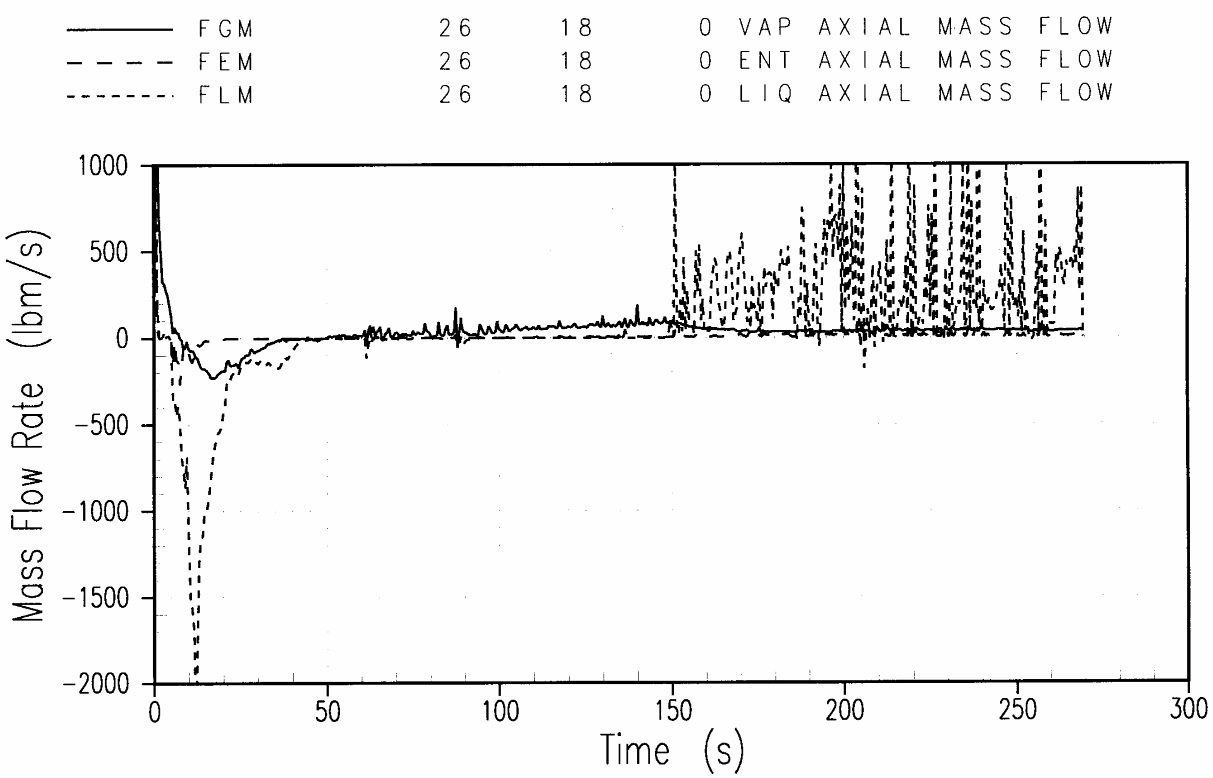

Figure 15.6.5.4A-8

Guide Tube Assemblies Exit Vapor, Entrained Drop, Liquid Flow Rates

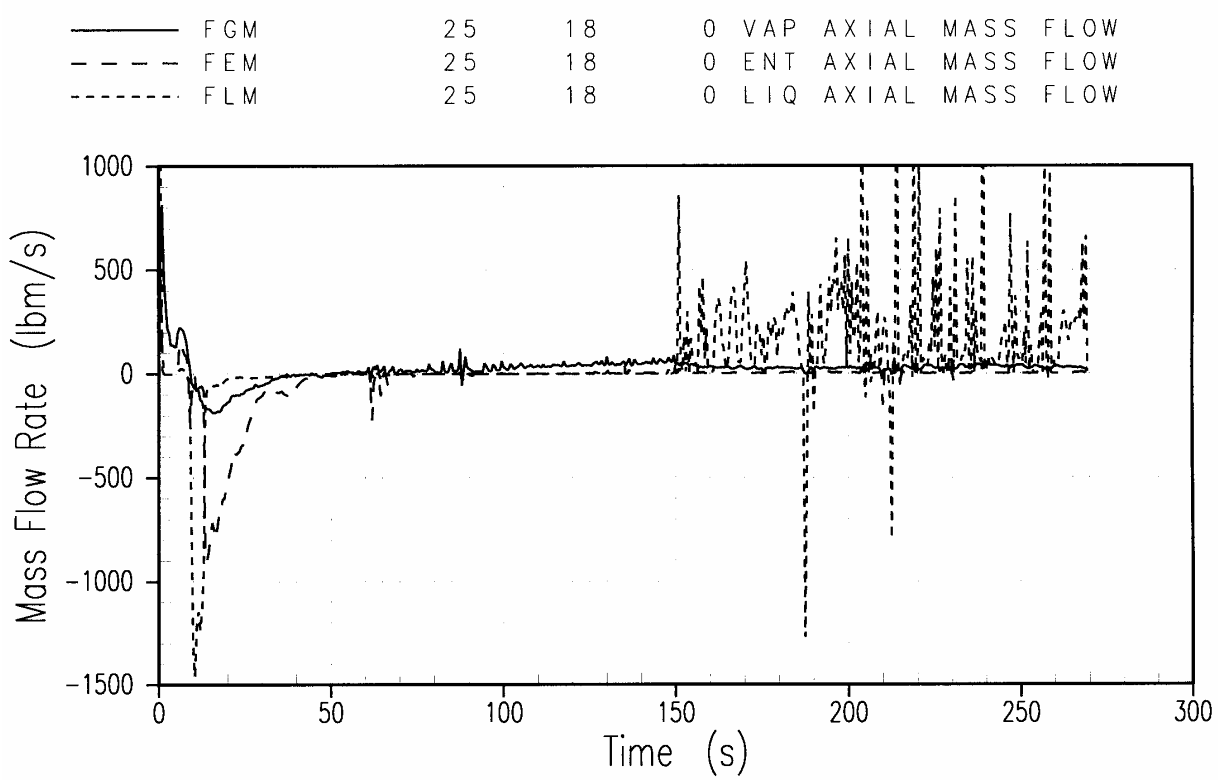

Figure 15.6.5.4A-9

Open Hole Assemblies Exit Vapor, Entrained Drop, Liquid Flow Rates

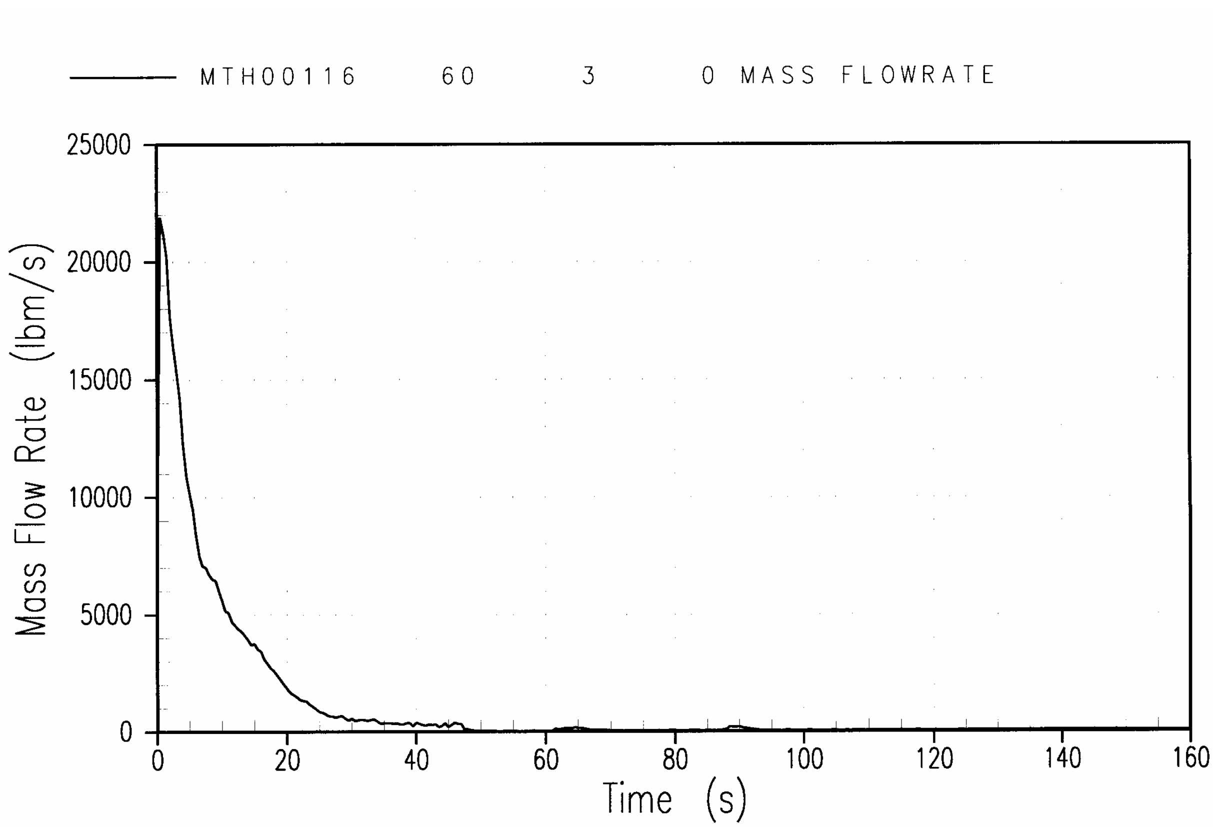

Figure 15.6.5.4A-10

Steam Generator Side DECLG Break Flow Rate

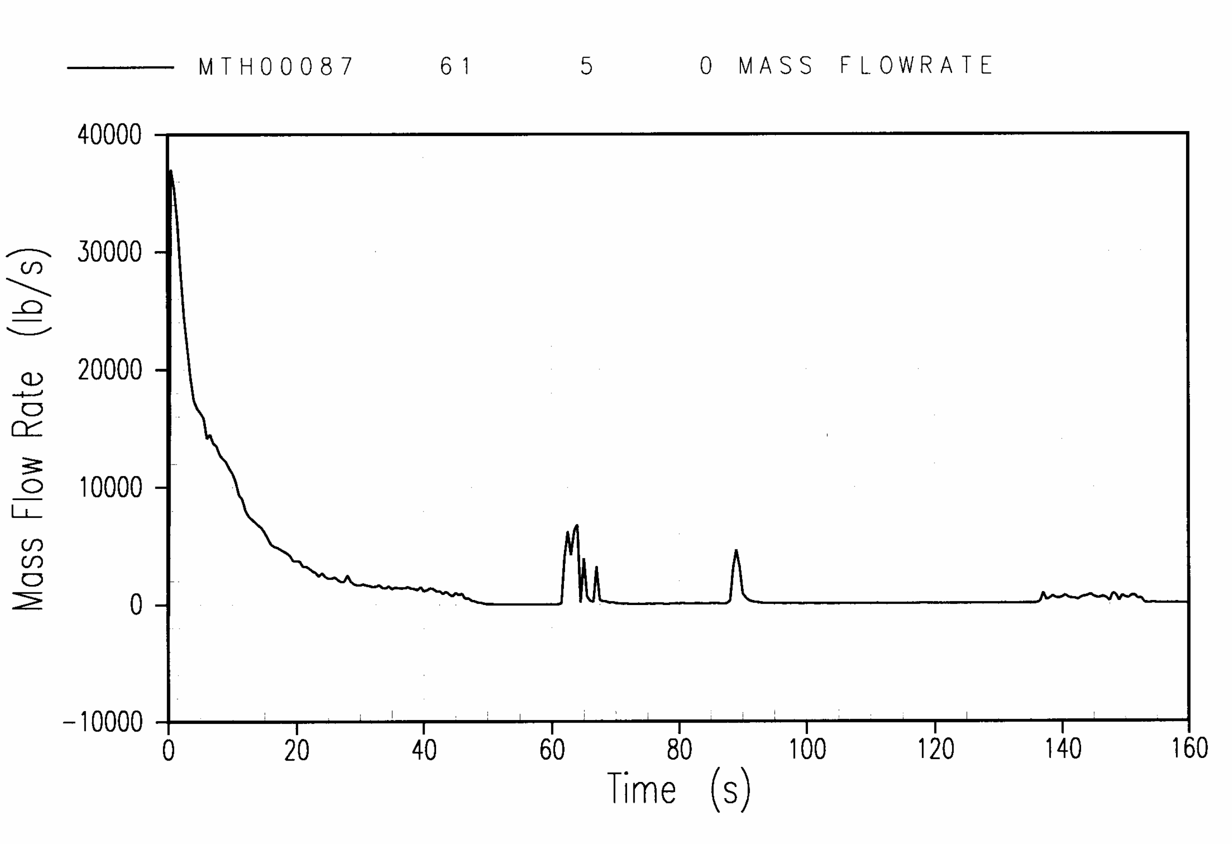

Figure 15.6.5.4A-11

Vessel Side DECLG Break Flow Rate

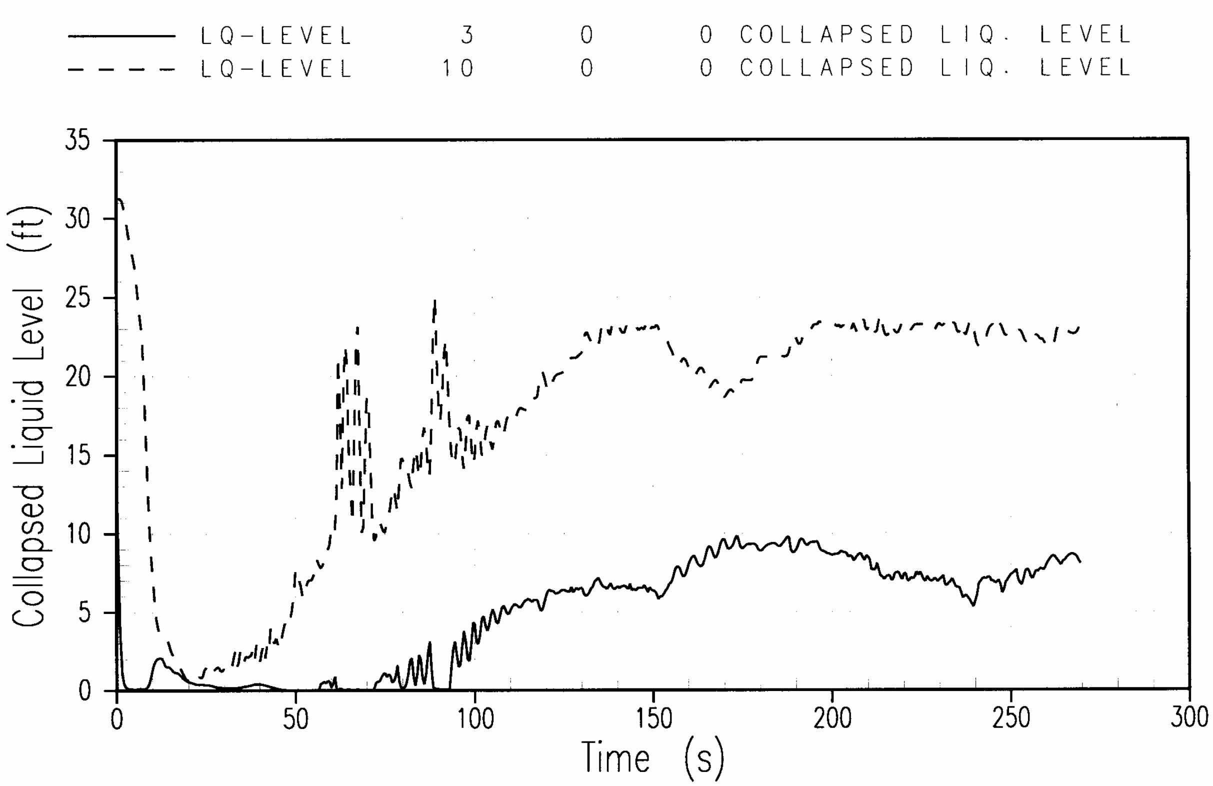

Figure 15.6.5.4A-12

Core and Downcomer Collapsed Liquid Levels

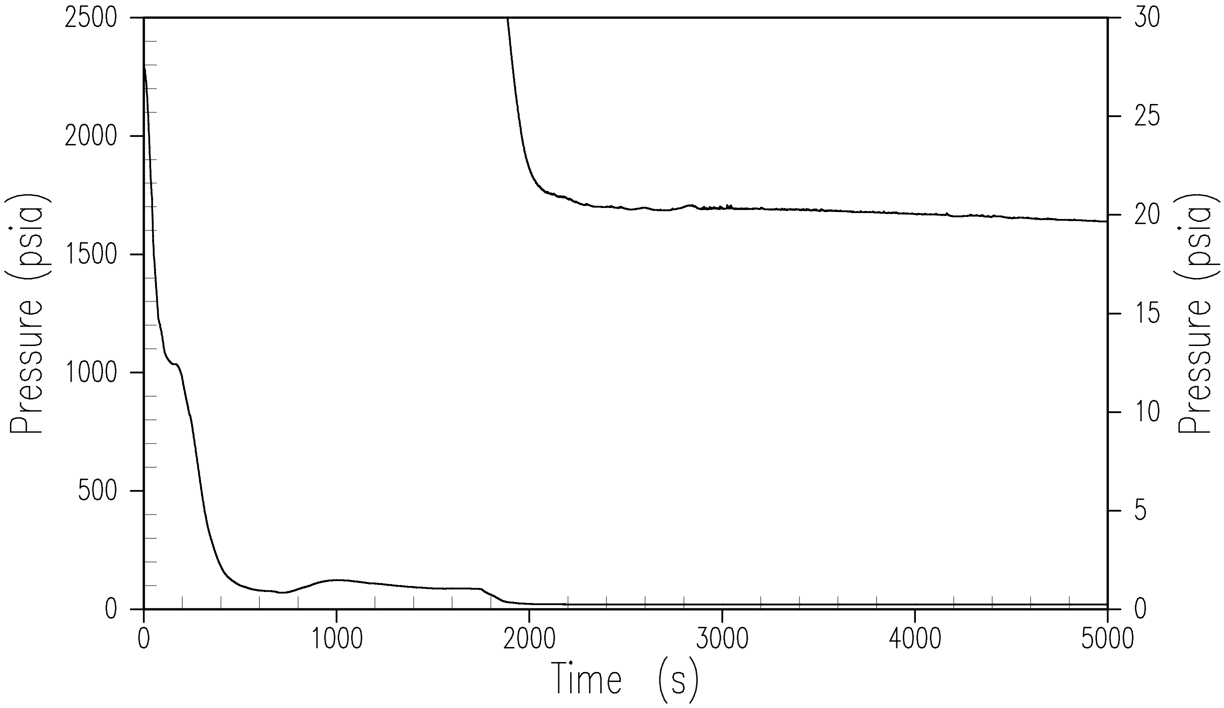

Figure 15.6.5.4B-1

Inadvertent ADS – RCS Pressure

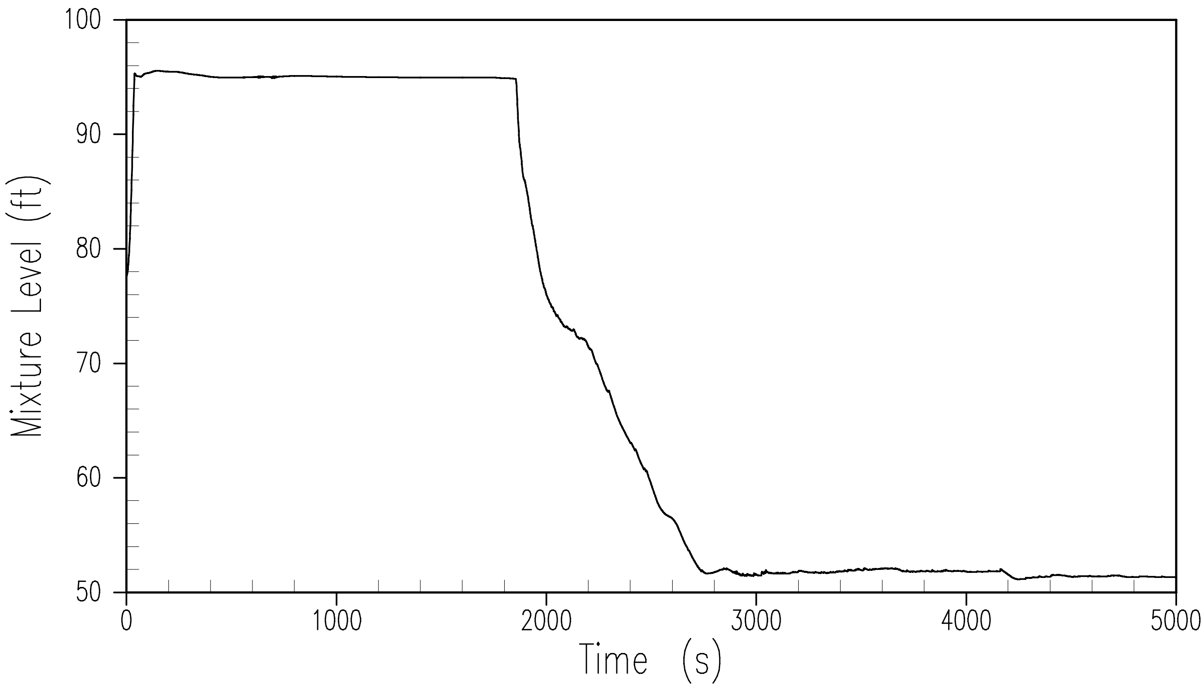

Figure 15.6.5.4B-2

Inadvertent ADS – Pressurizer Mixture Level

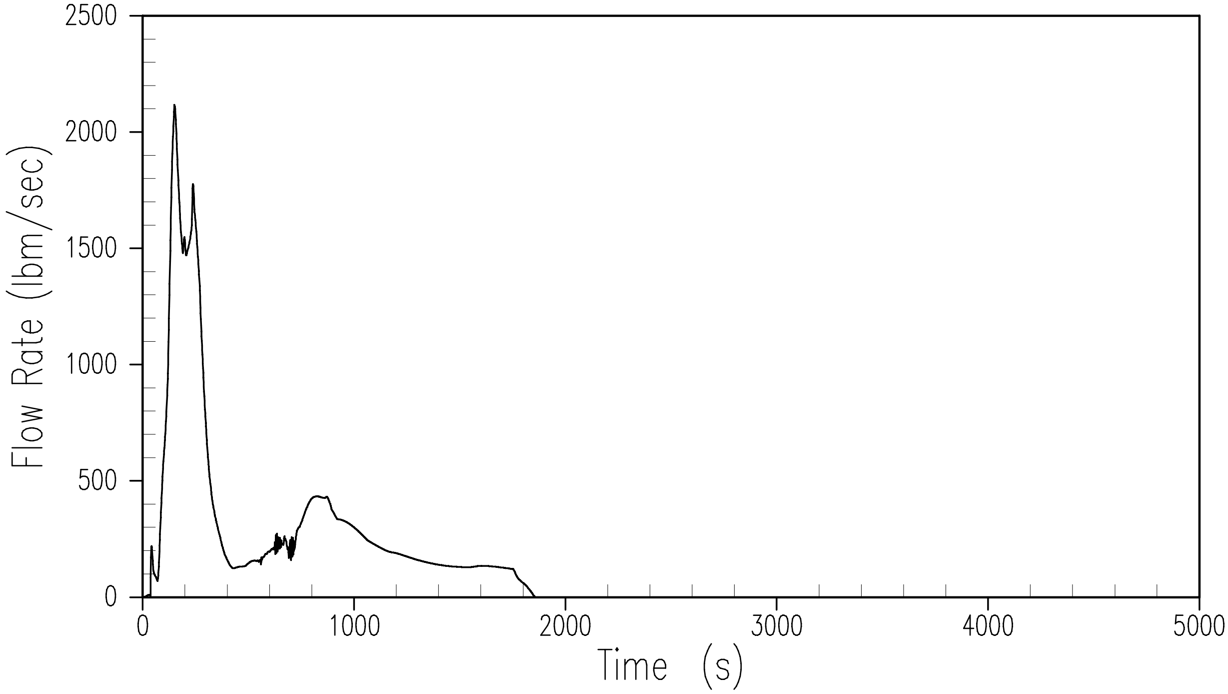

Figure 15.6.5.4B-3

Inadvertent ADS – ADS 1-3 Liquid Discharge

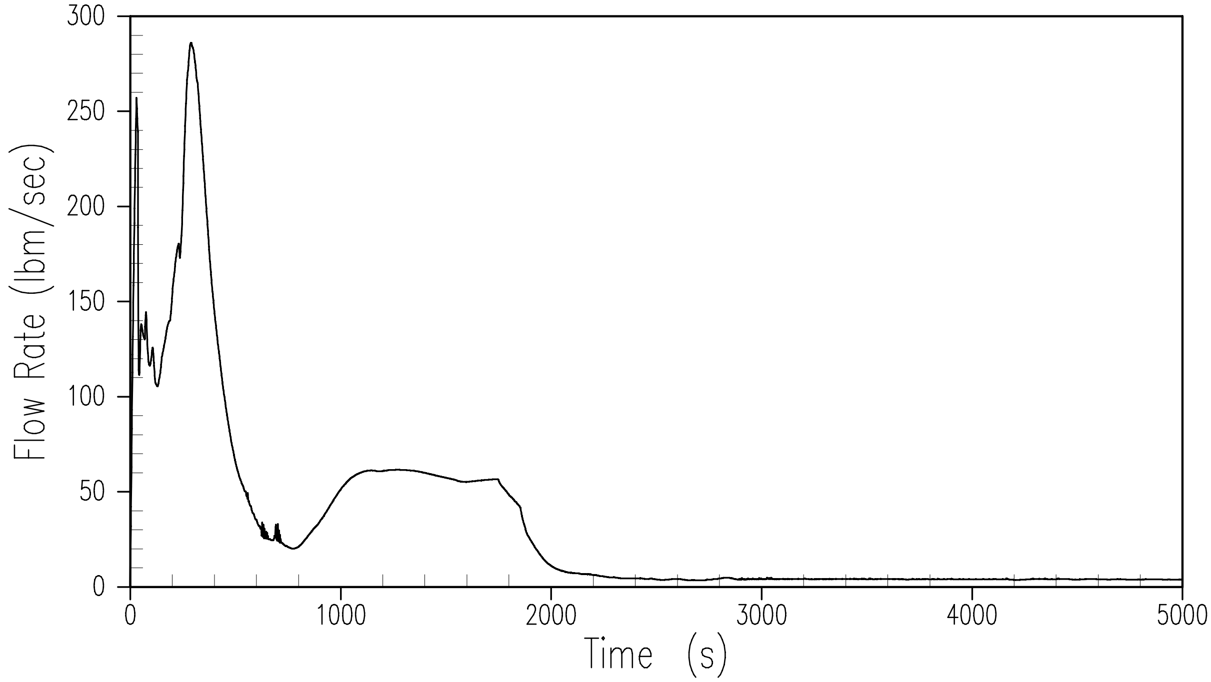

Figure 15.6.5.4B-4

Inadvertent ADS – ADS 1-3 Vapor Discharge

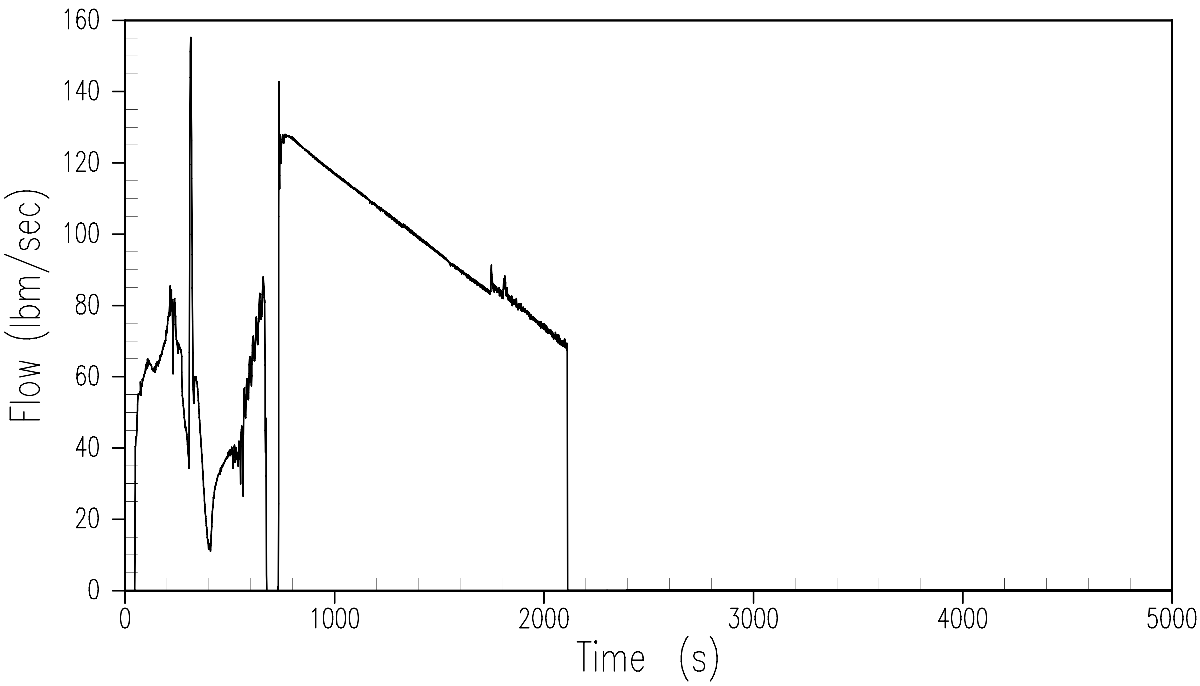

Figure 15.6.5.4B-5

Inadvertent ADS – CMT-1 Injection Rate

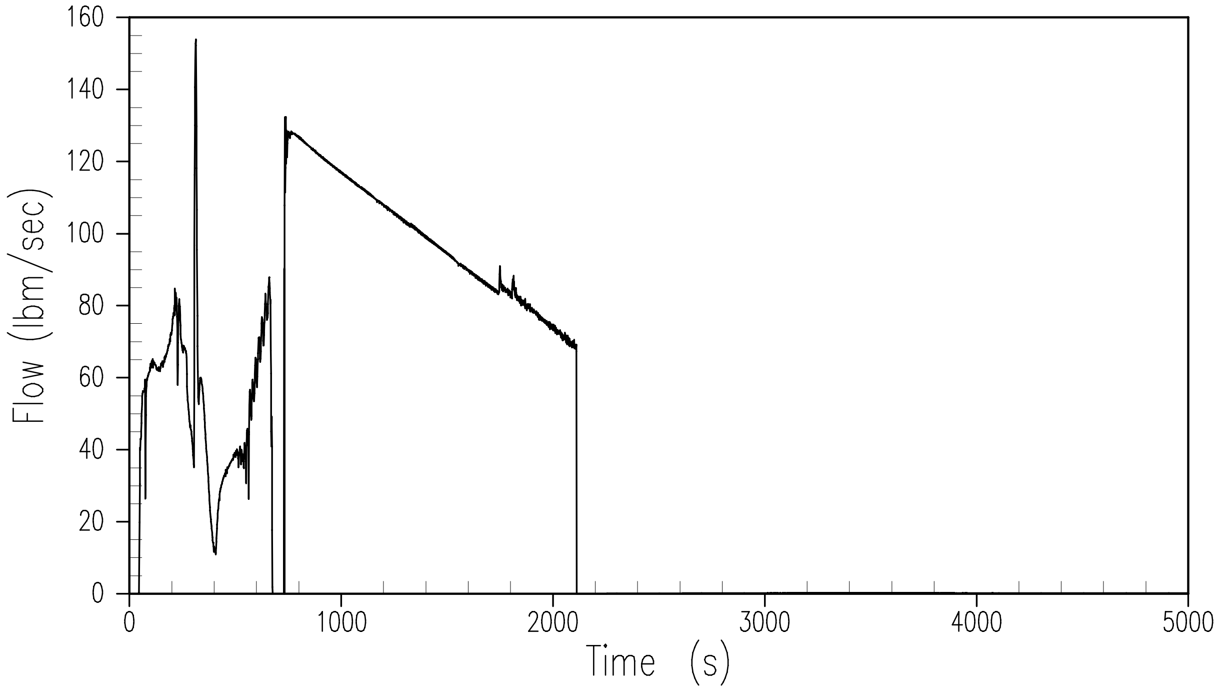

Figure 15.6.5.4B-6

Inadvertent ADS – CMT-2 Injection Rate

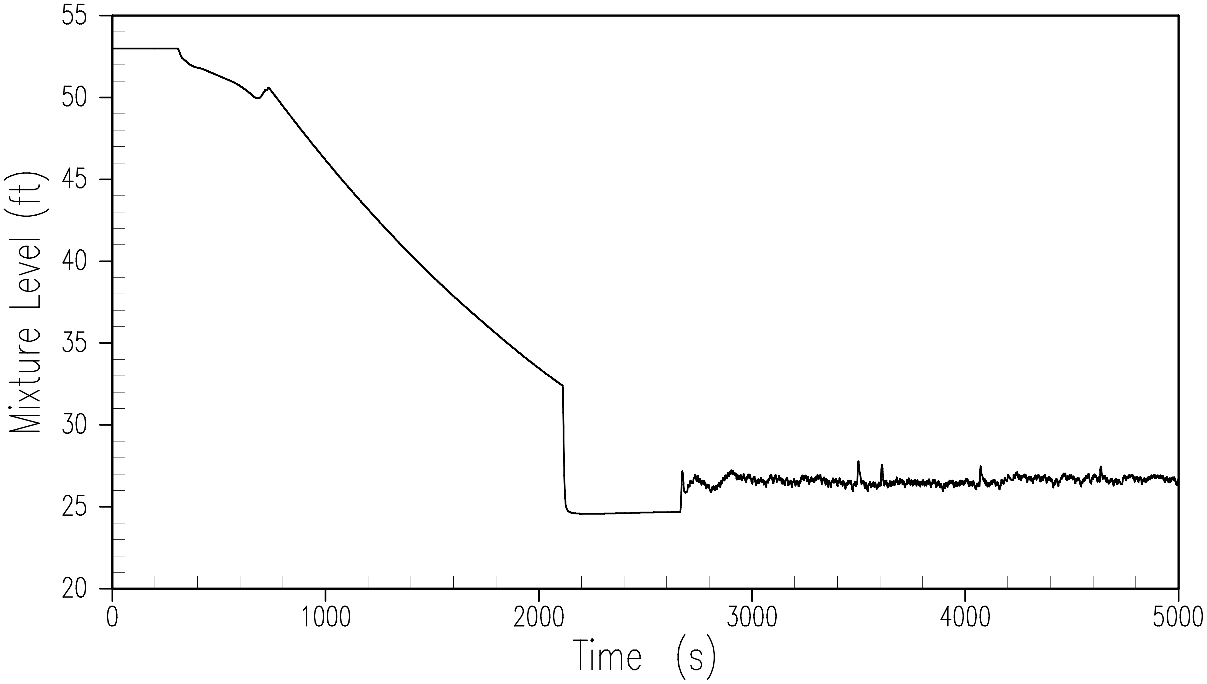

Figure 15.6.5.4B-7

Inadvertent ADS – CMT-1 Mixture Level

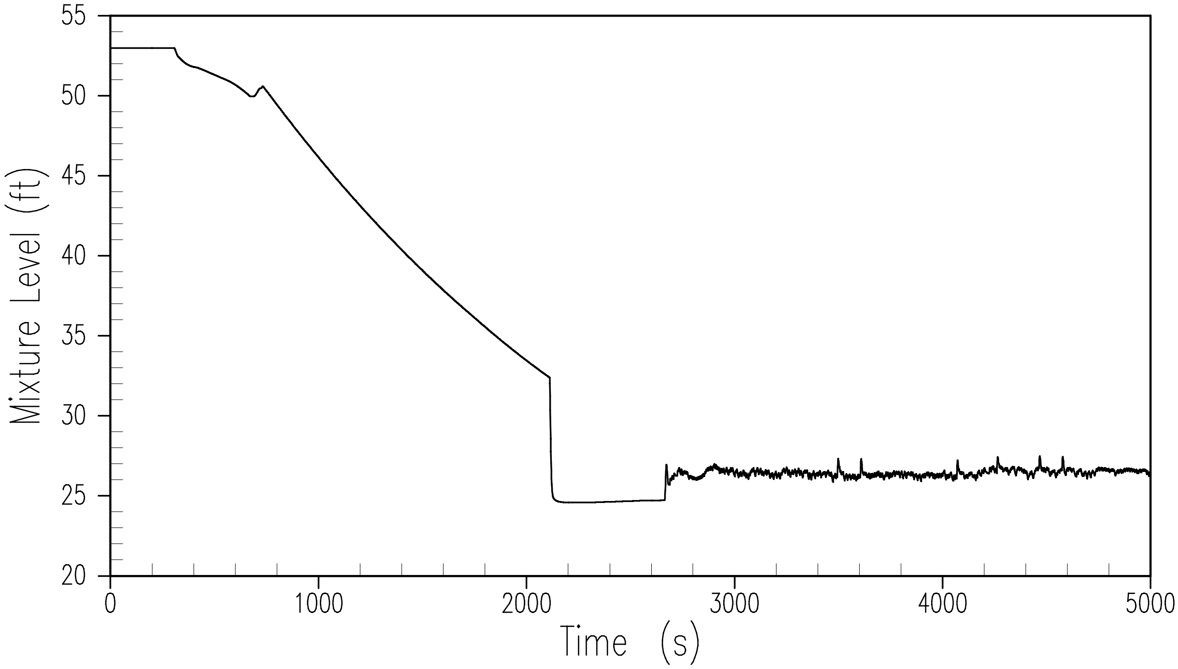

Figure 15.6.5.4B-8

Inadvertent ADS – CMT-2 Mixture Level

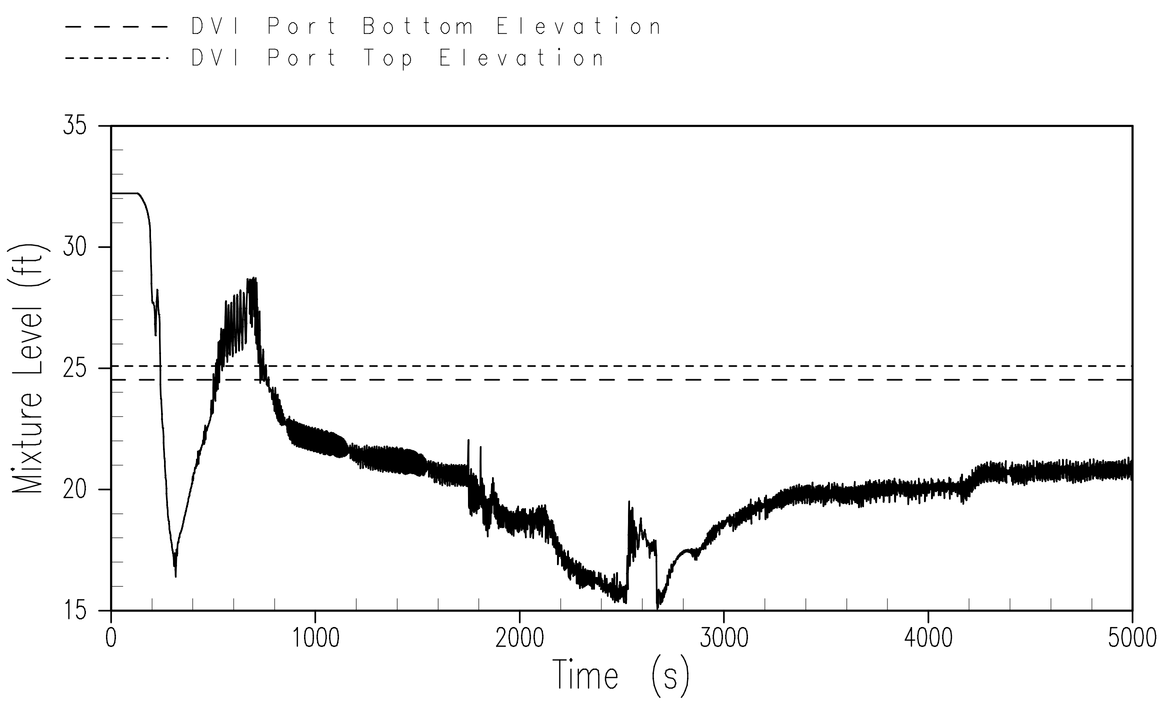

Figure 15.6.5.4B-9

Inadvertent ADS – Downcomer Mixture Level

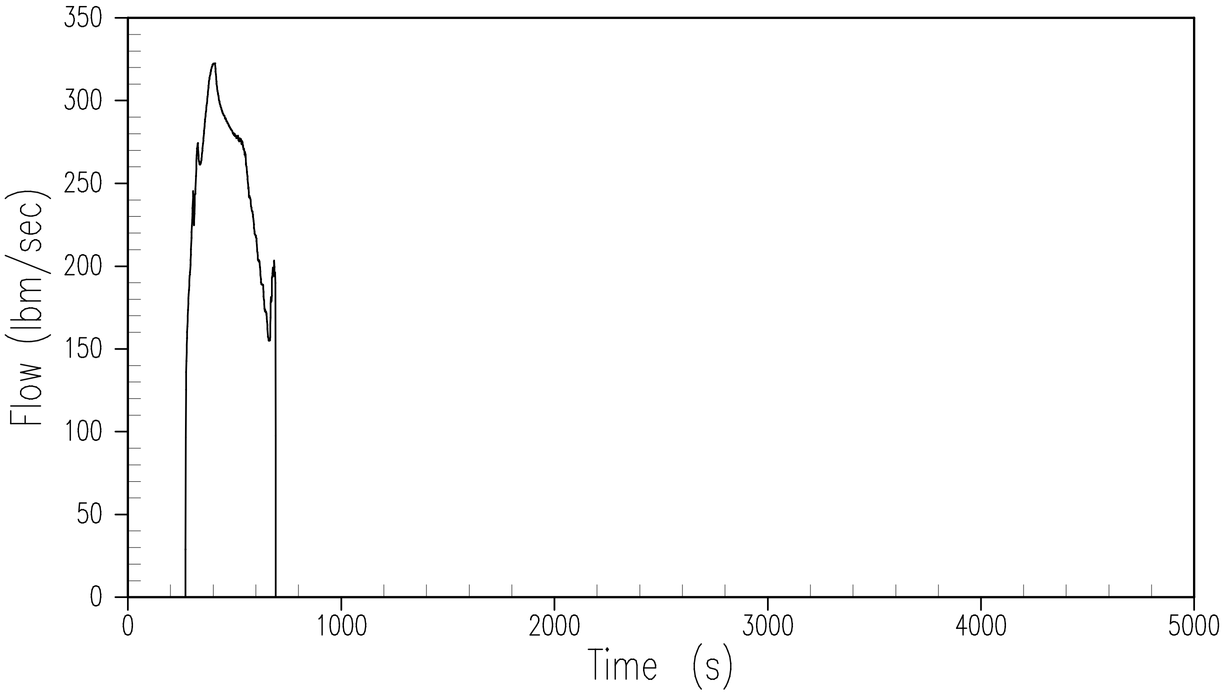

Figure 15.6.5.4B-10

Inadvertent ADS – Accumulator-1 Injection Rate

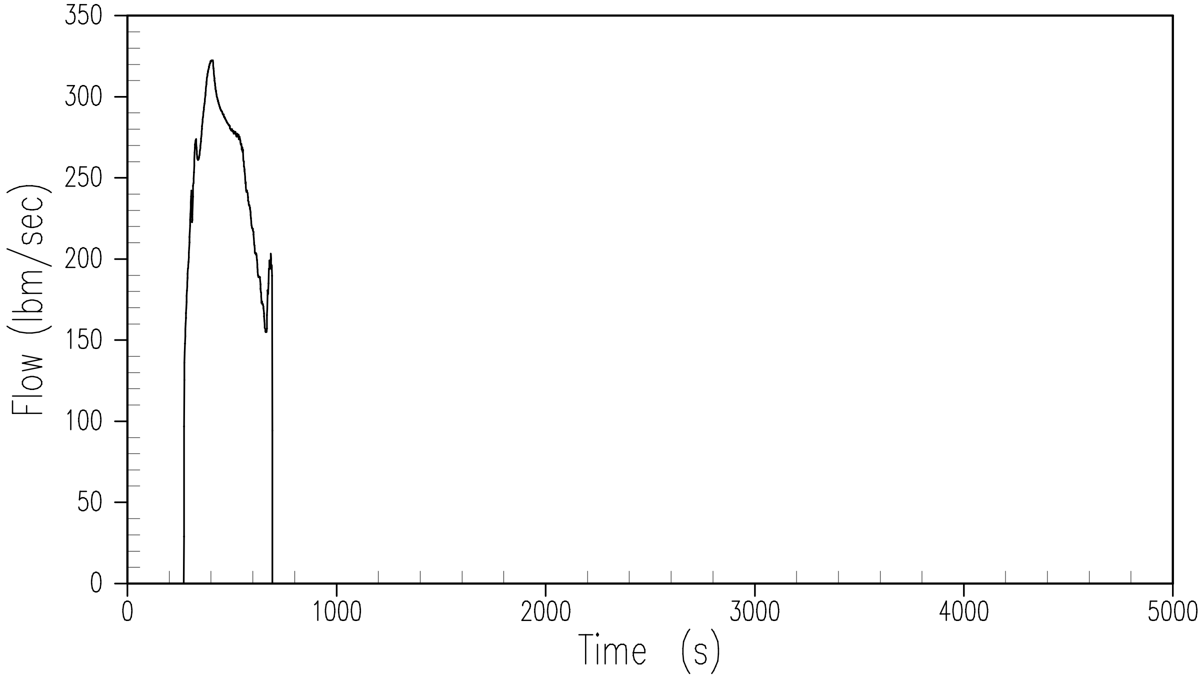

Figure 15.6.5.4B-11

Inadvertent ADS – Accumulator-2 Injection Rate

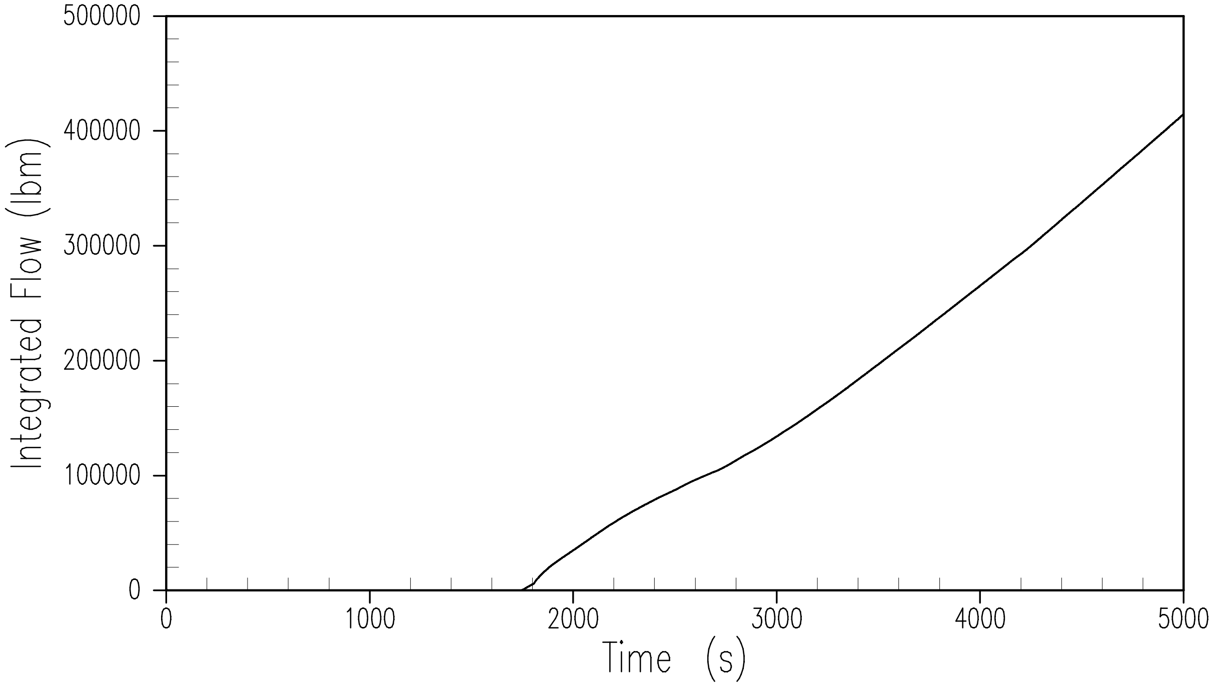

Figure 15.6.5.4B-12

Inadvertent ADS – ADS-4 Integrated Discharge

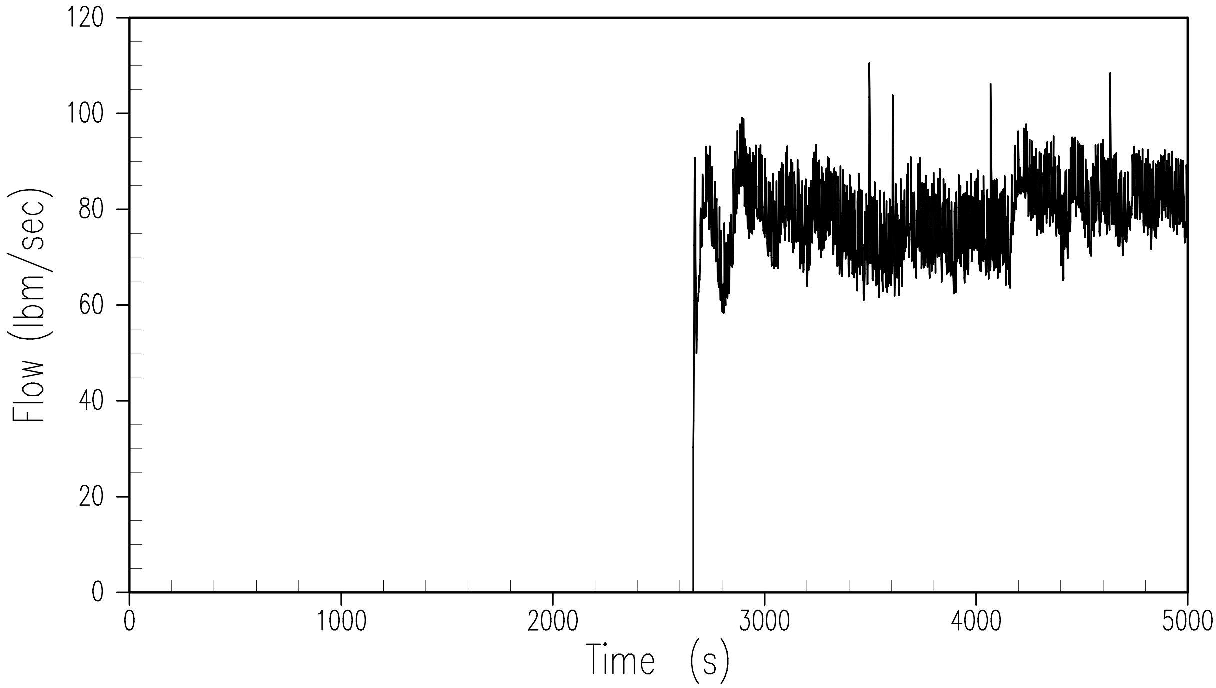

Figure 15.6.5.4B-13

Inadvertent ADS – IRWST-1 Injection Rate

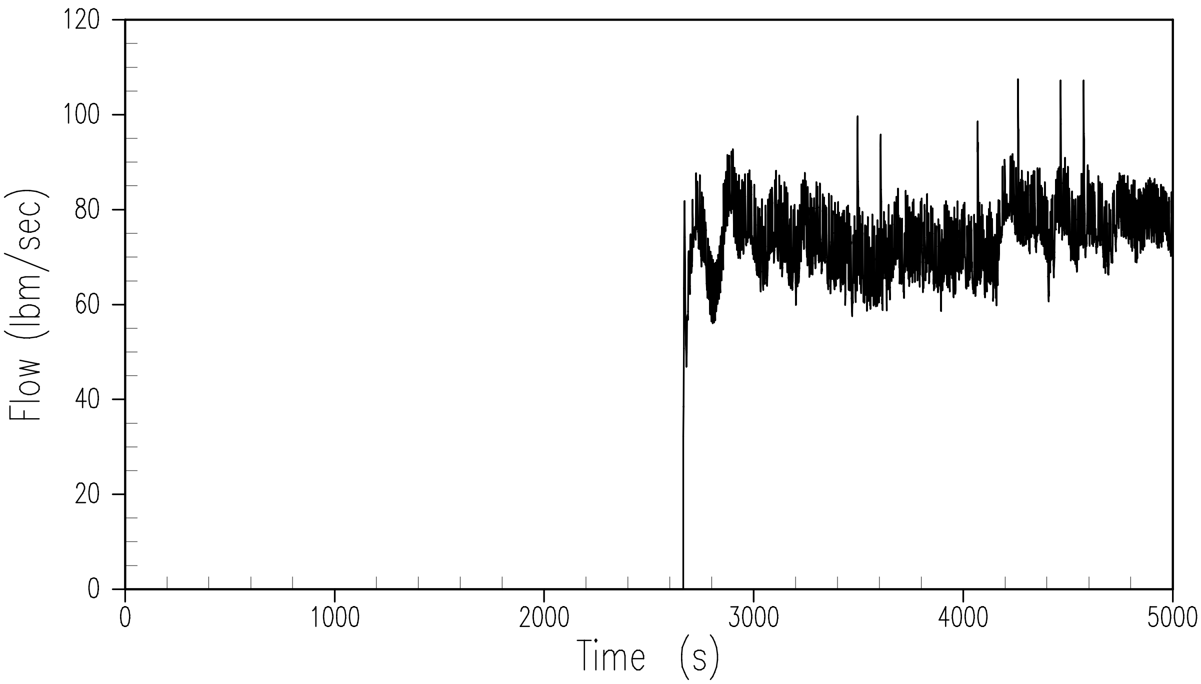

Figure 15.6.5.4B-14

Inadvertent ADS – IRWST-2 Injection Rate

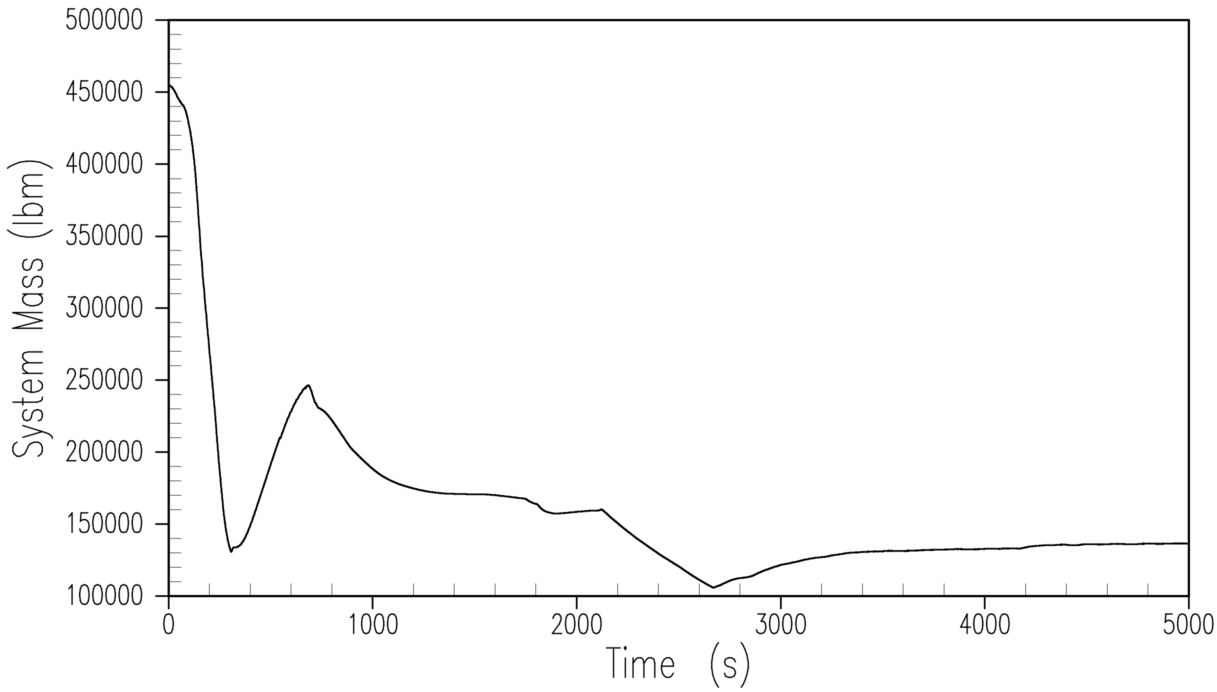

Figure 15.6.5.4B-15

Inadvertent ADS – RCS System Inventory

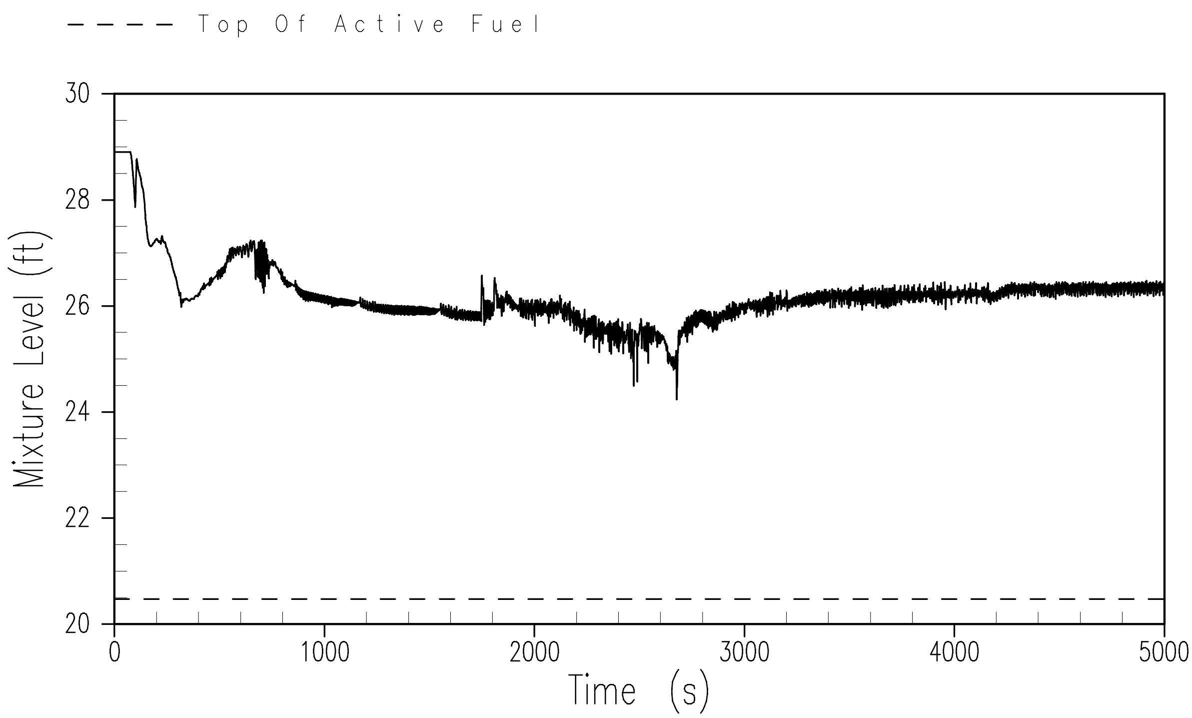

Figure 15.6.5.4B-16

Inadvertent ADS – Core/Upper Plenum Mixture Level

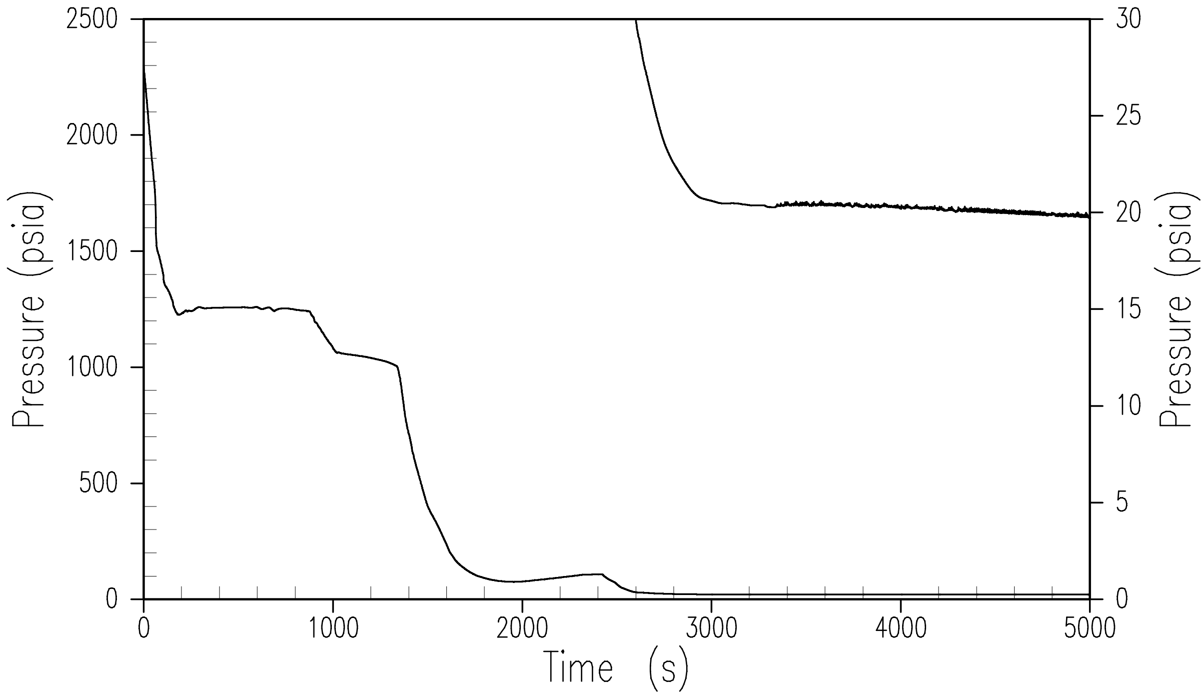

Figure 15.6.5.4B-17

2-Inch Cold Leg Break – RCS Pressure

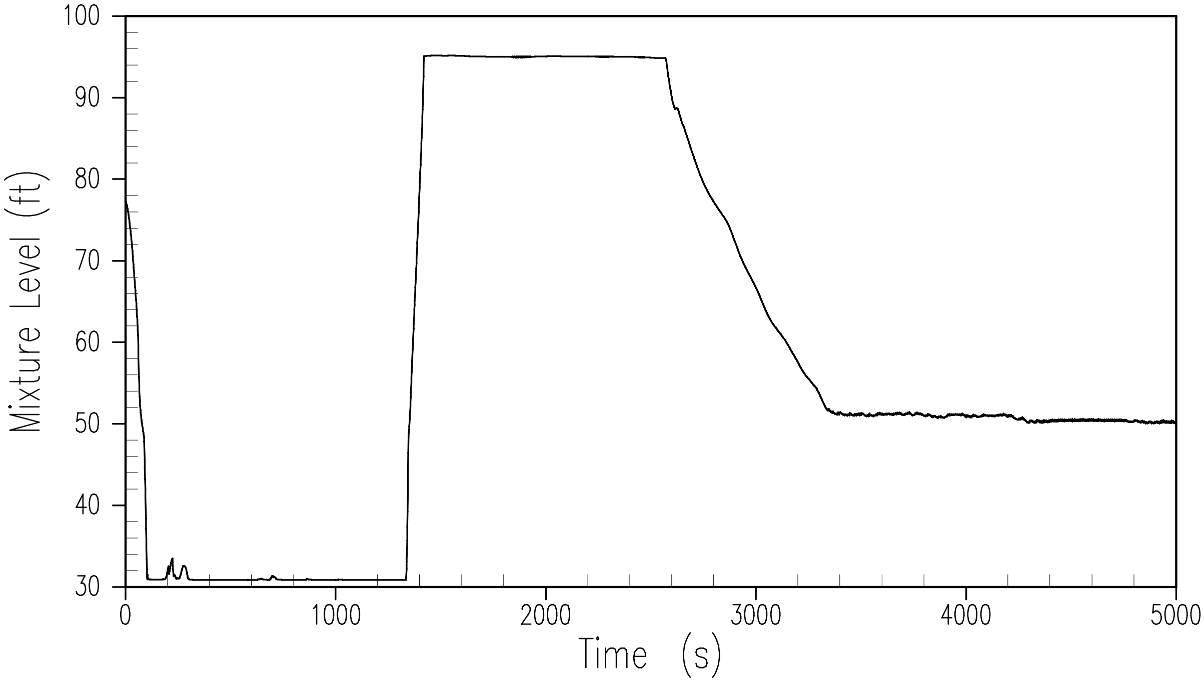

Figure 15.6.5.4B-18

2-Inch Cold Leg Break – Pressurizer Mixture Level

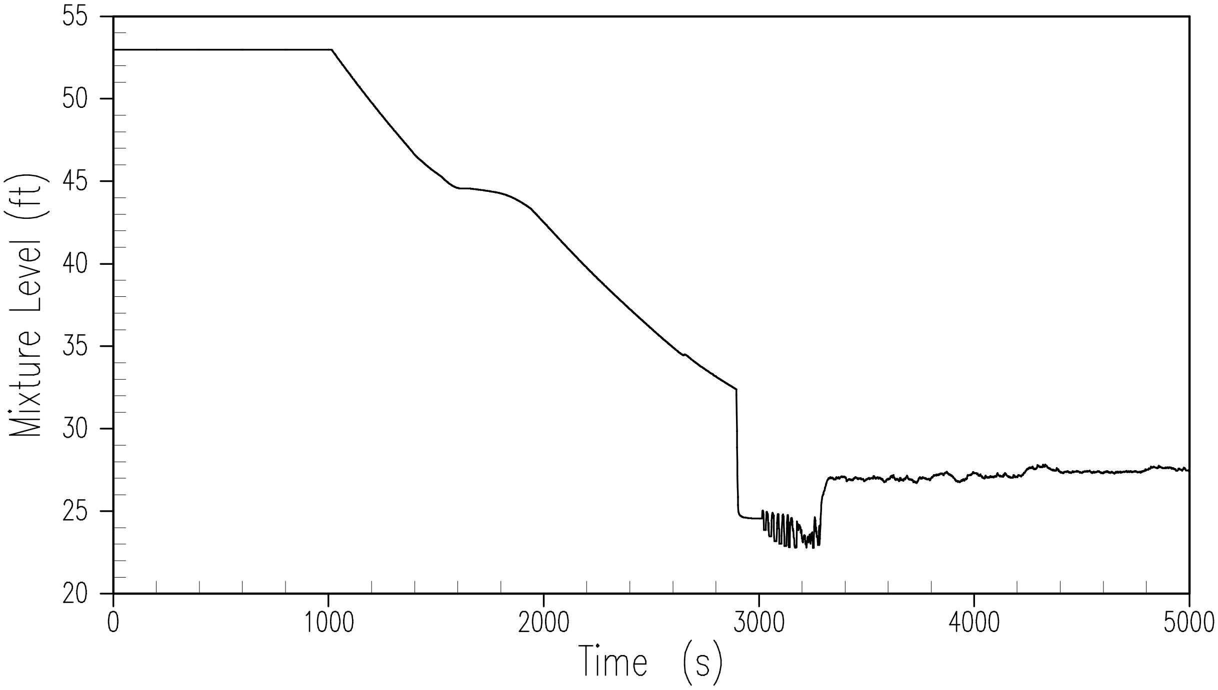

Figure 15.6.5.4B-19

2-Inch Cold Leg Break – CMT-1 Mixture Level

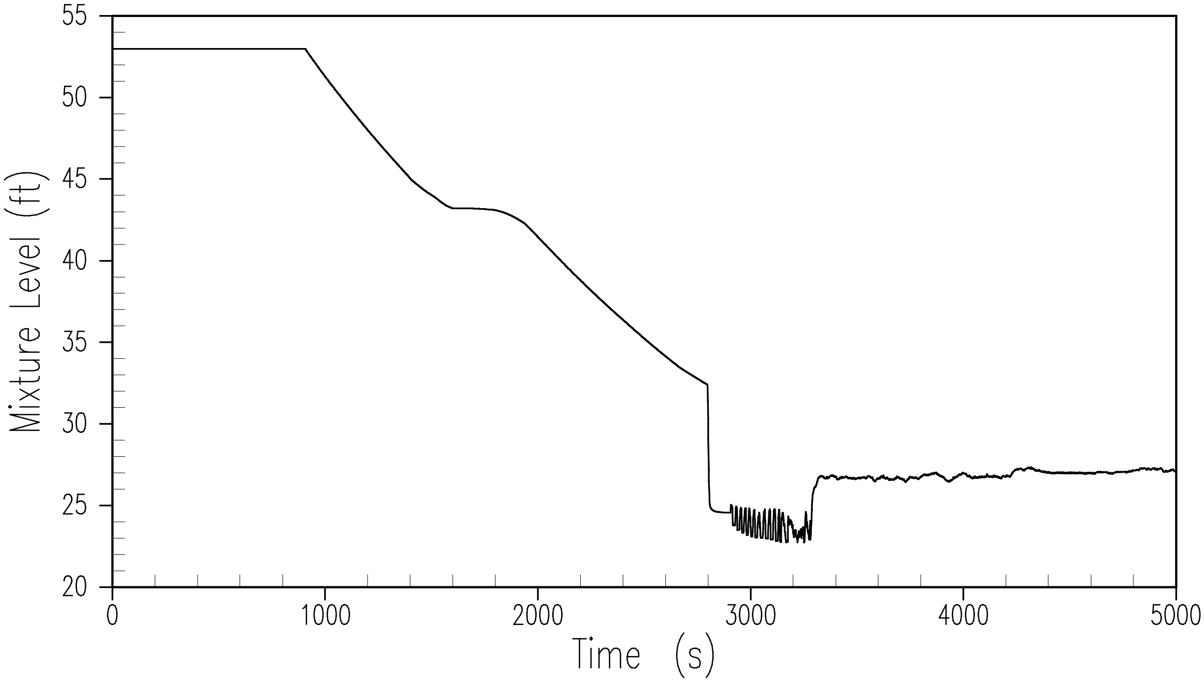

Figure 15.6.5.4B-20

2-Inch Cold Leg Break – CMT-2 Mixture Level

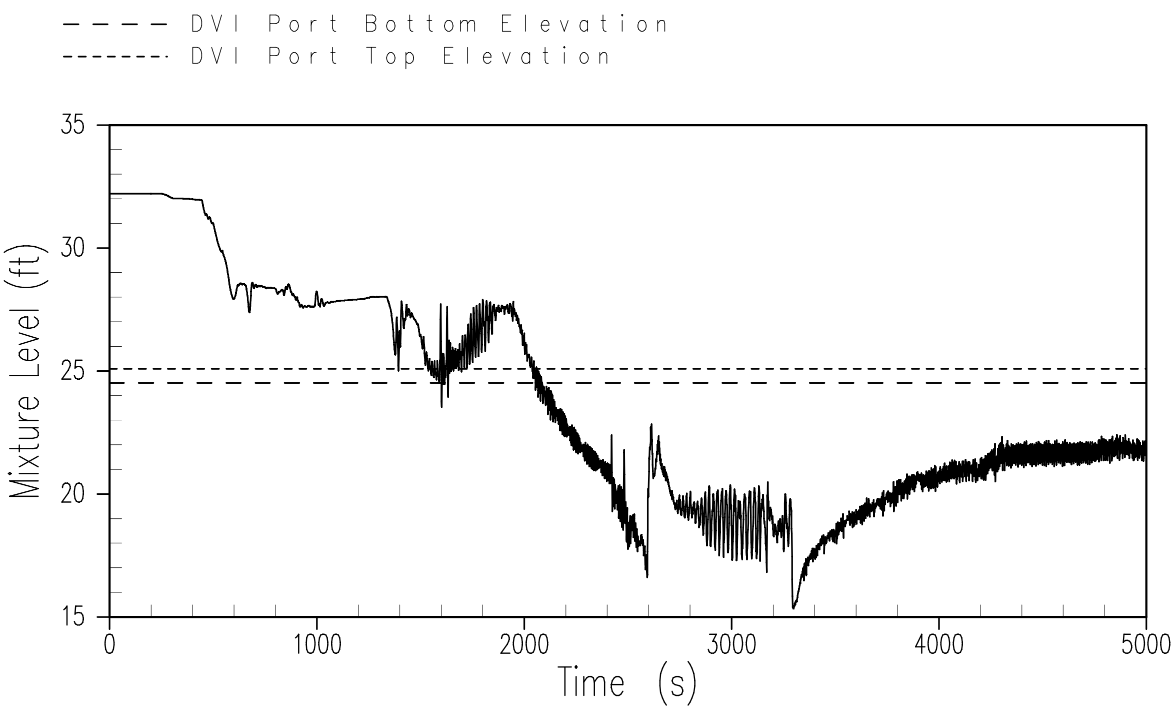

Figure 15.6.5.4B-21

2-Inch Cold Leg Break – Downcomer Mixture Level

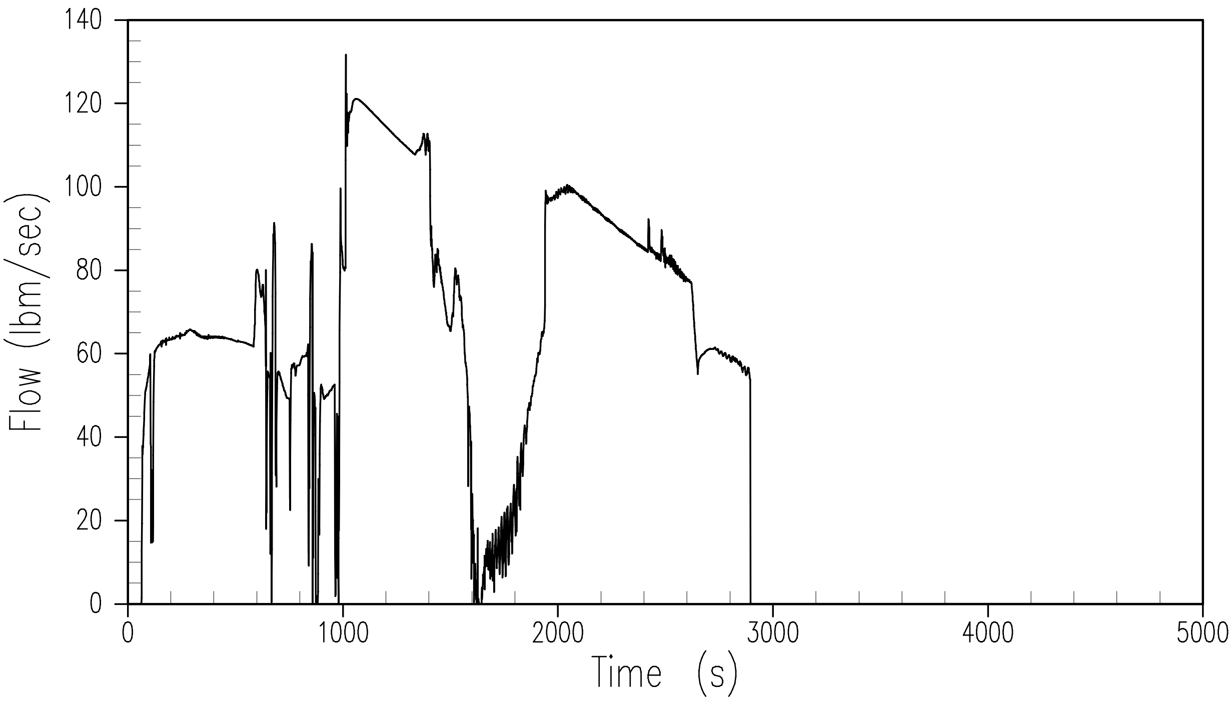

Figure 15.6.5.4B-22

2-Inch Cold Leg Break – CMT-1 Injection Rate

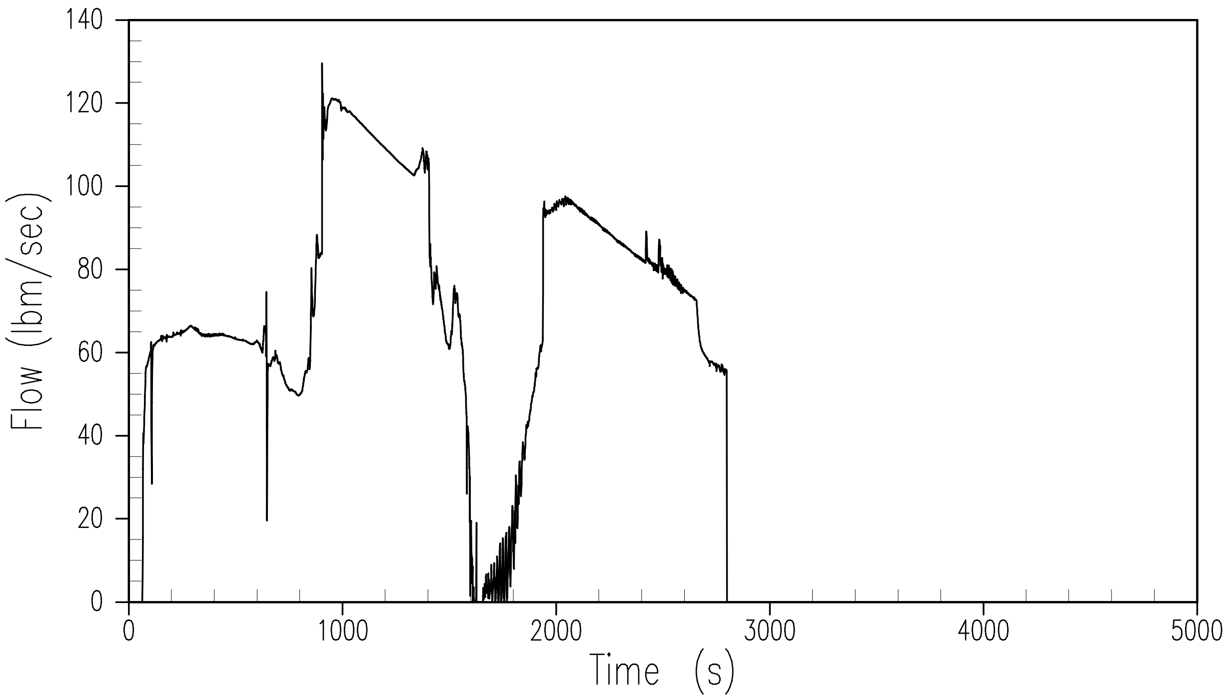

Figure 15.6.5.4B-23

2-Inch Cold Leg Break – CMT-2 Injection Rate

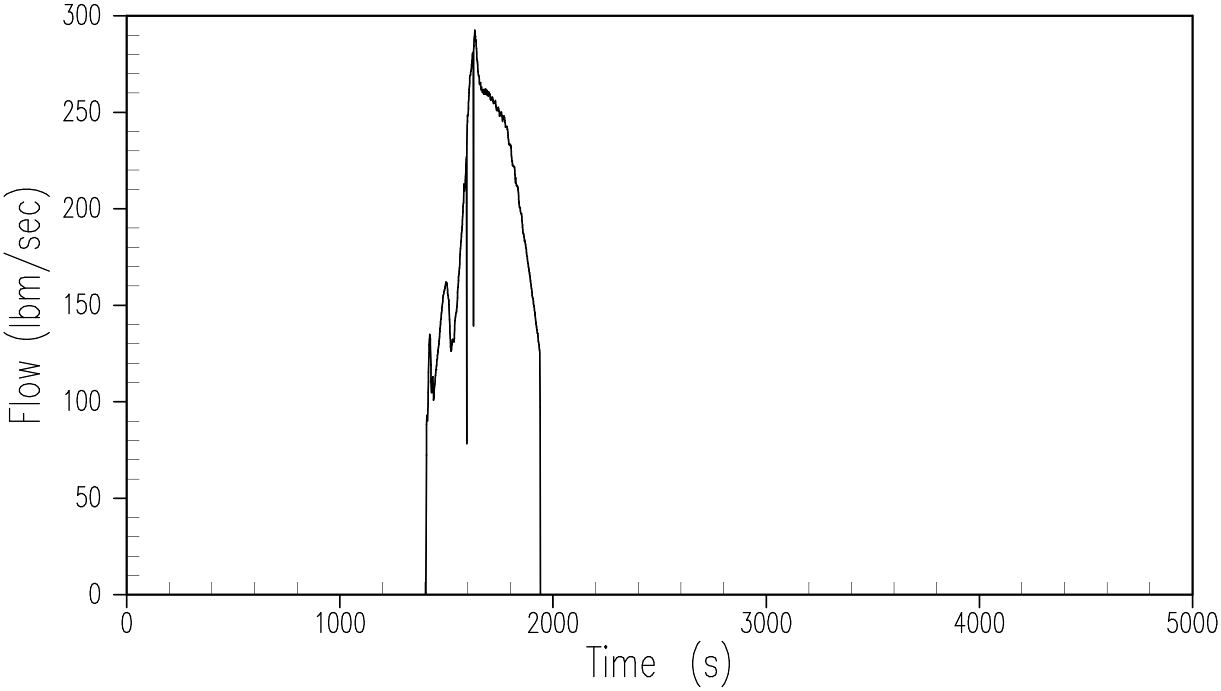

Figure 15.6.5.4B-24

2-Inch Cold Leg Break – Accumulator-1 Injection Rate

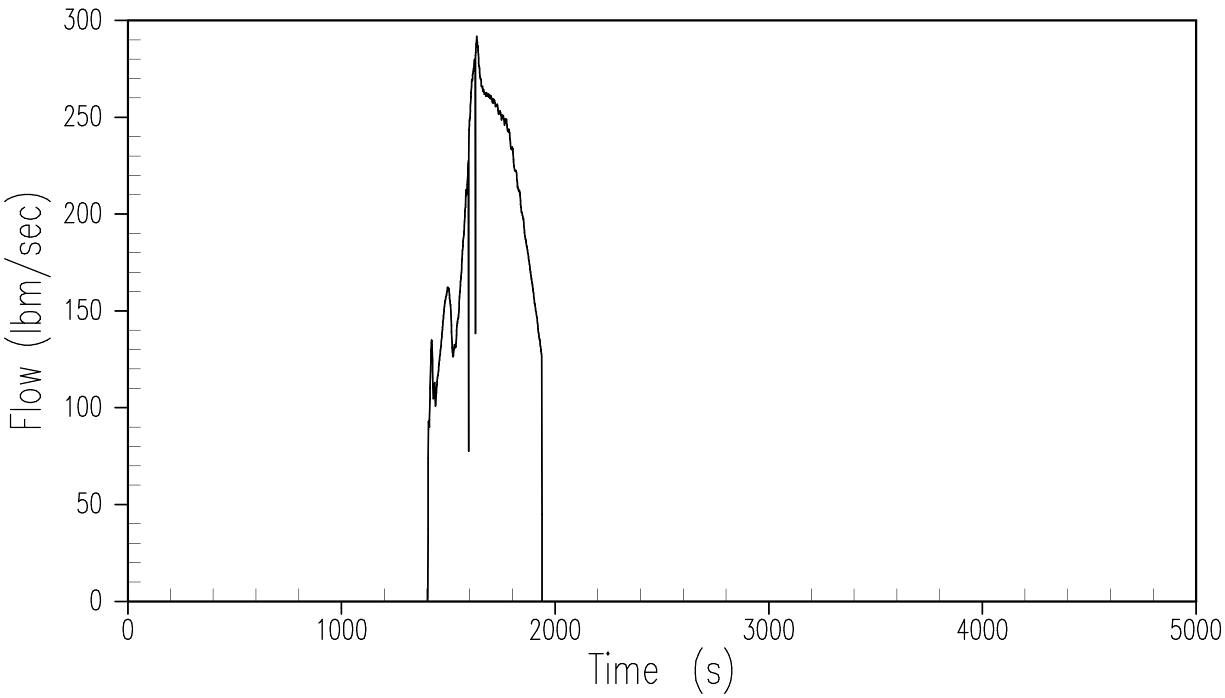

Figure 15.6.5.4B-25

2-Inch Cold Leg Break – Accumulator-2 Injection Rate

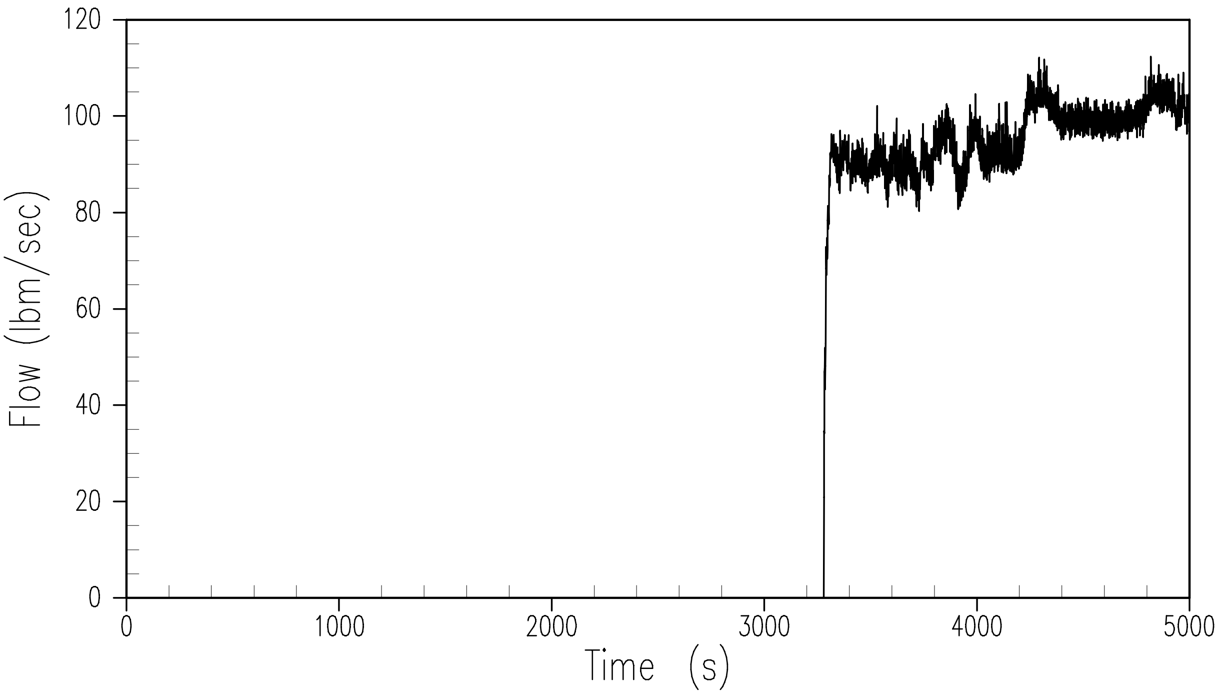

Figure 15.6.5.4B-26

2-Inch Cold Leg Break – IRWST-1 Injection Rate

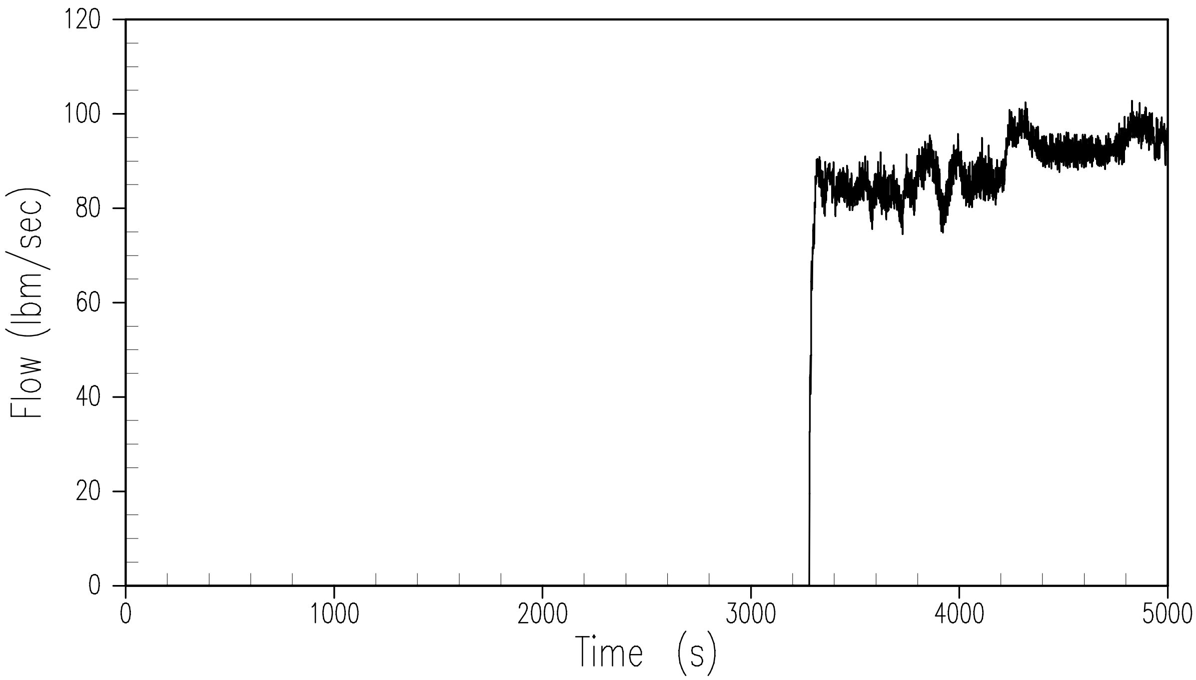

Figure 15.6.5.4B-27

2-Inch Cold Leg Break – IRWST-2 Injection Rate

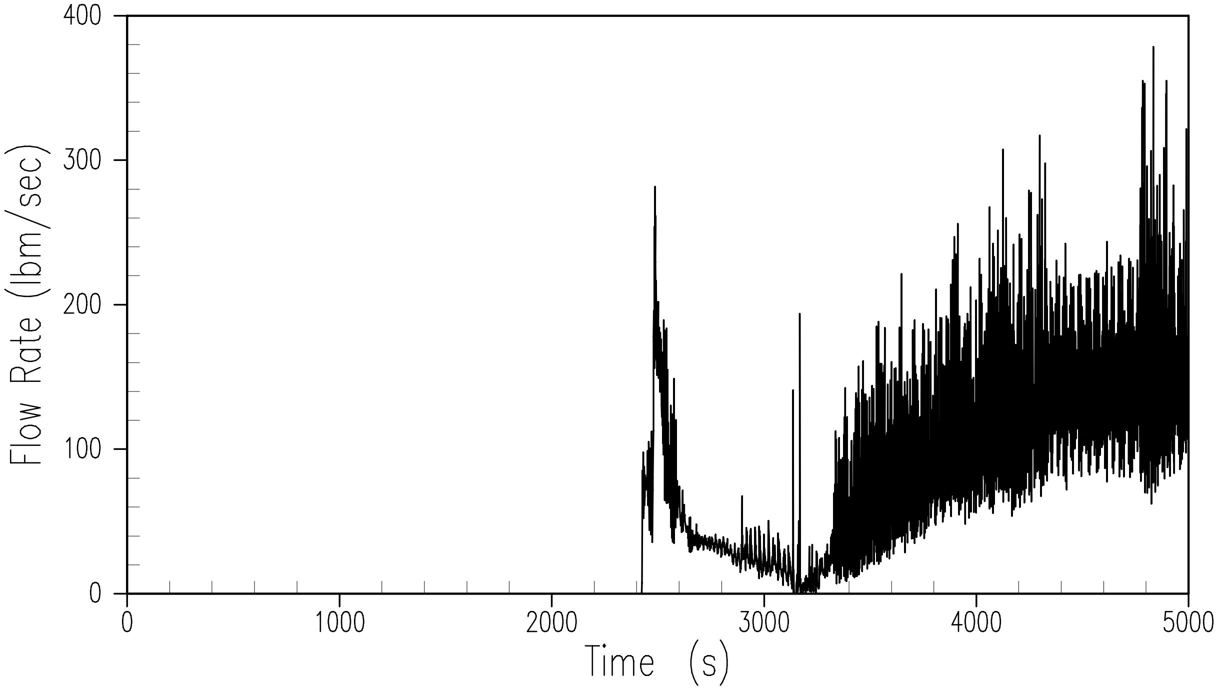

Figure 15.6.5.4B-28

2-Inch Cold Leg Break – ADS-4 Liquid Discharge

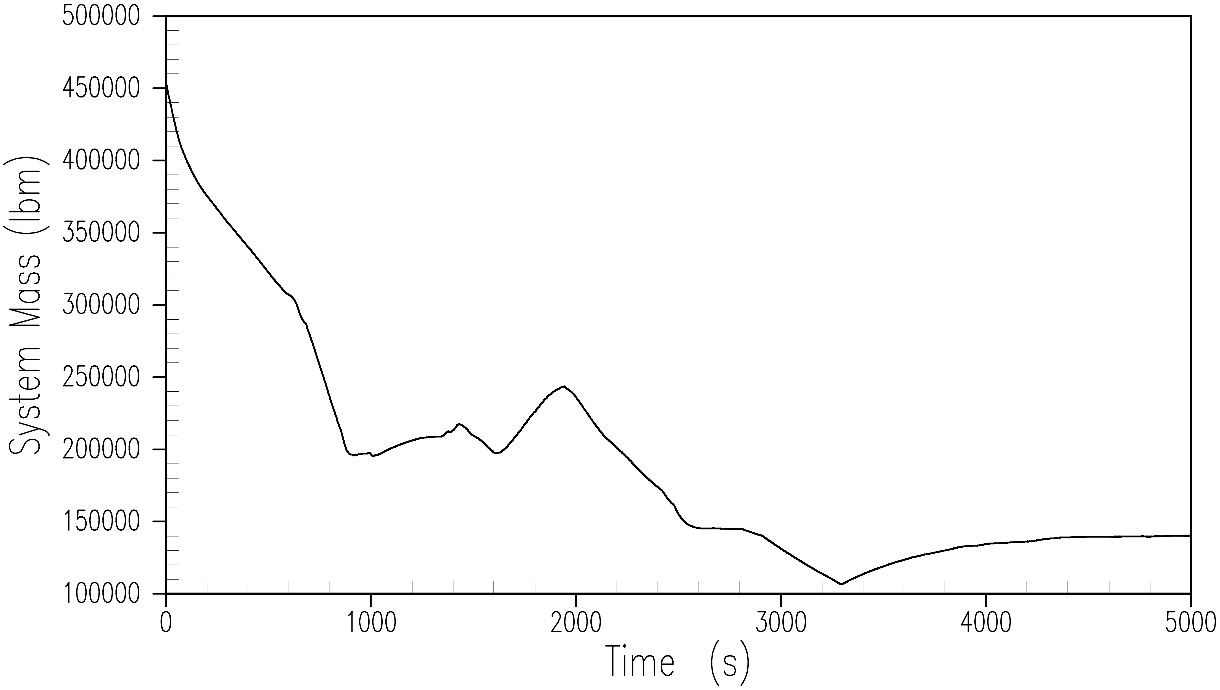

Figure 15.6.5.4B-29

2-Inch Cold Leg Break – RCS System Inventory

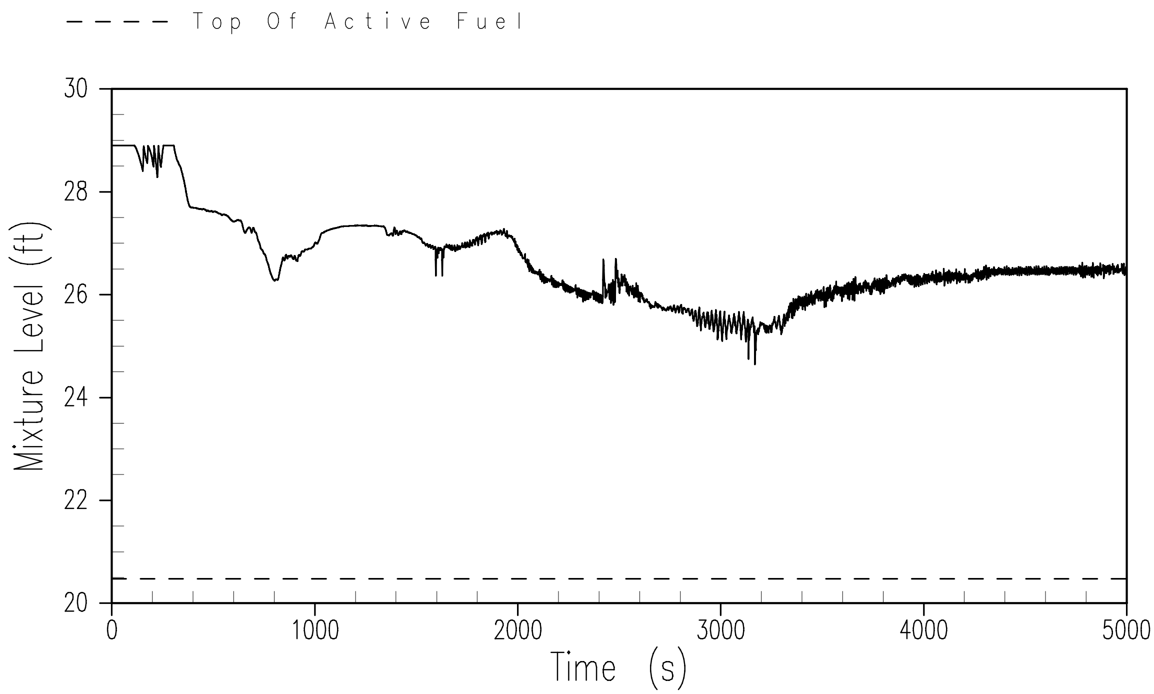

Figure 15.6.5.4B-30

2-Inch Cold Leg Break – Core/Upper Plenum Mixture Level

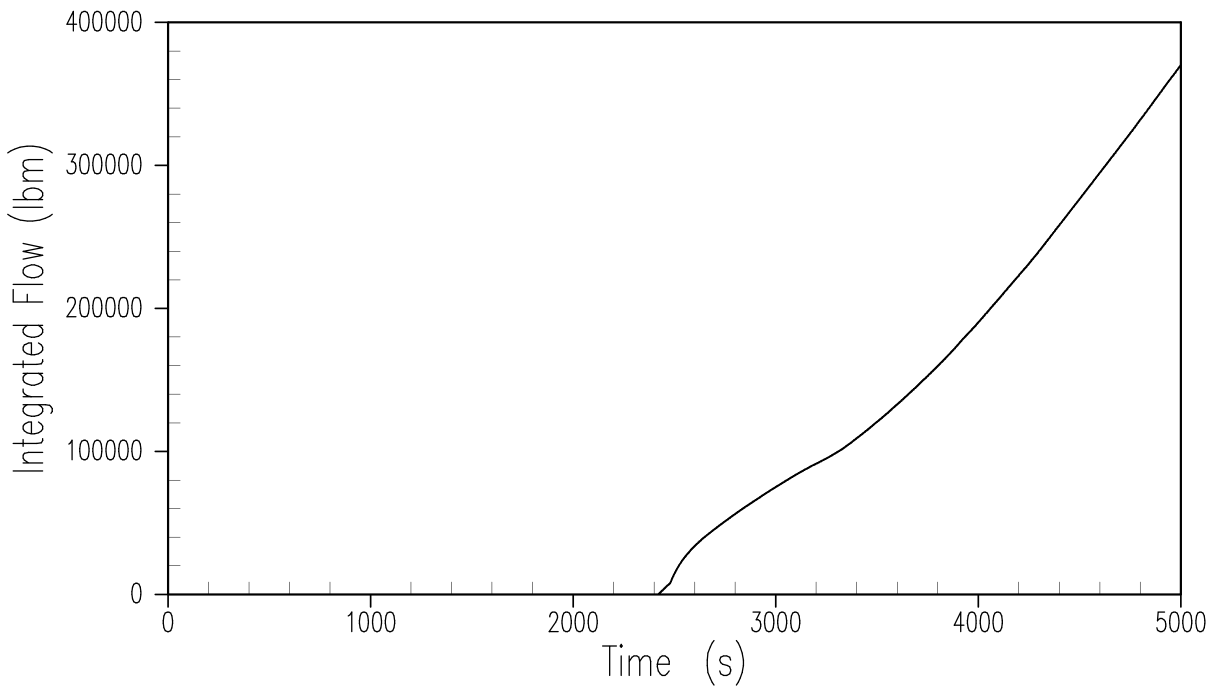

Figure 15.6.5.4B-31

2-Inch Cold Leg Break – ADS-4 Integrated Discharge

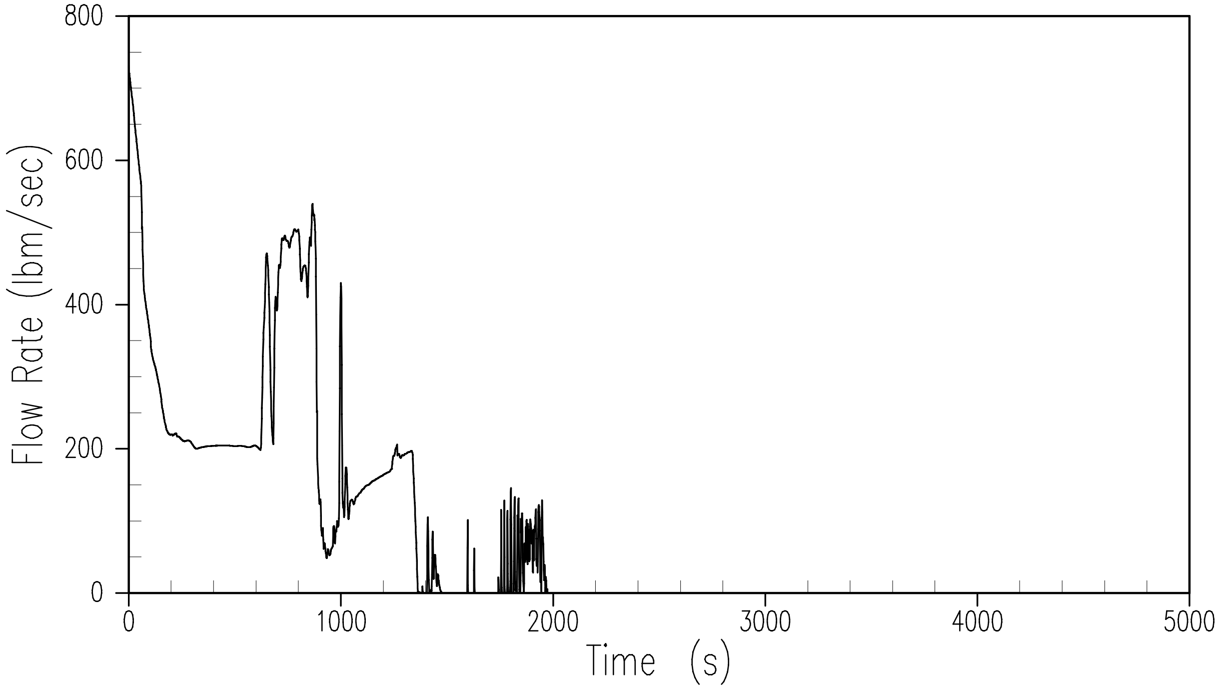

Figure 15.6.5.4B-32

2-Inch Cold Leg Break – Liquid Break Discharge

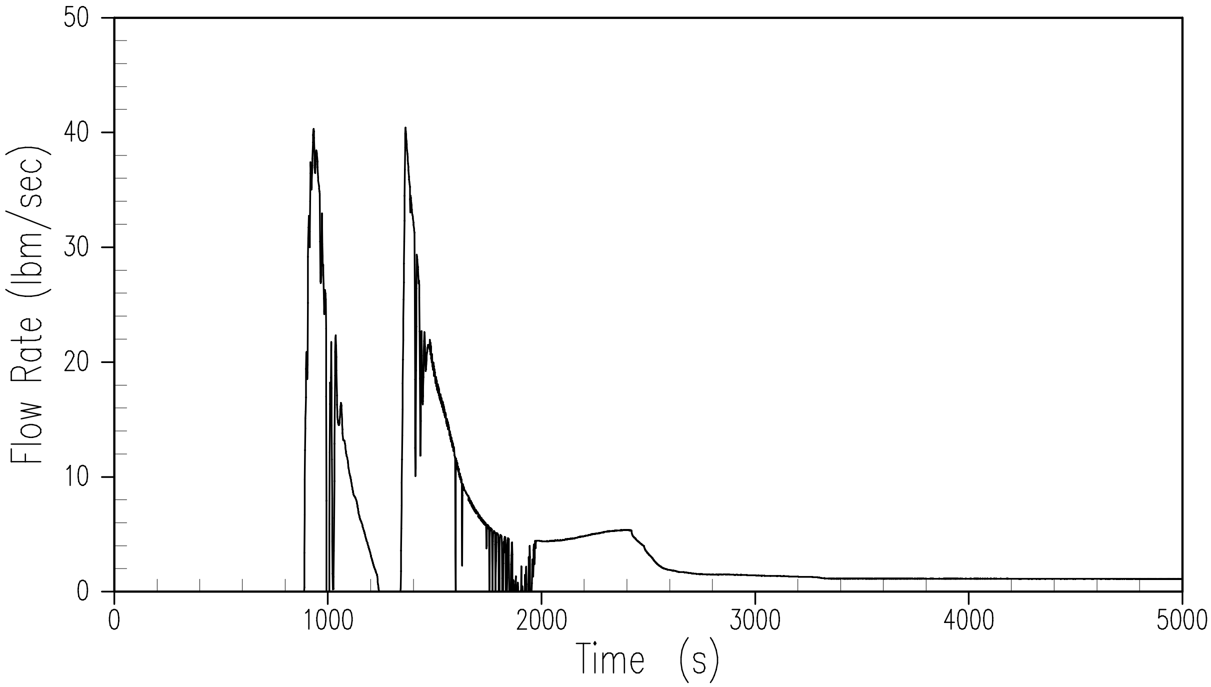

Figure 15.6.5.4B-33

2-Inch Cold Leg Break – Vapor Break Discharge

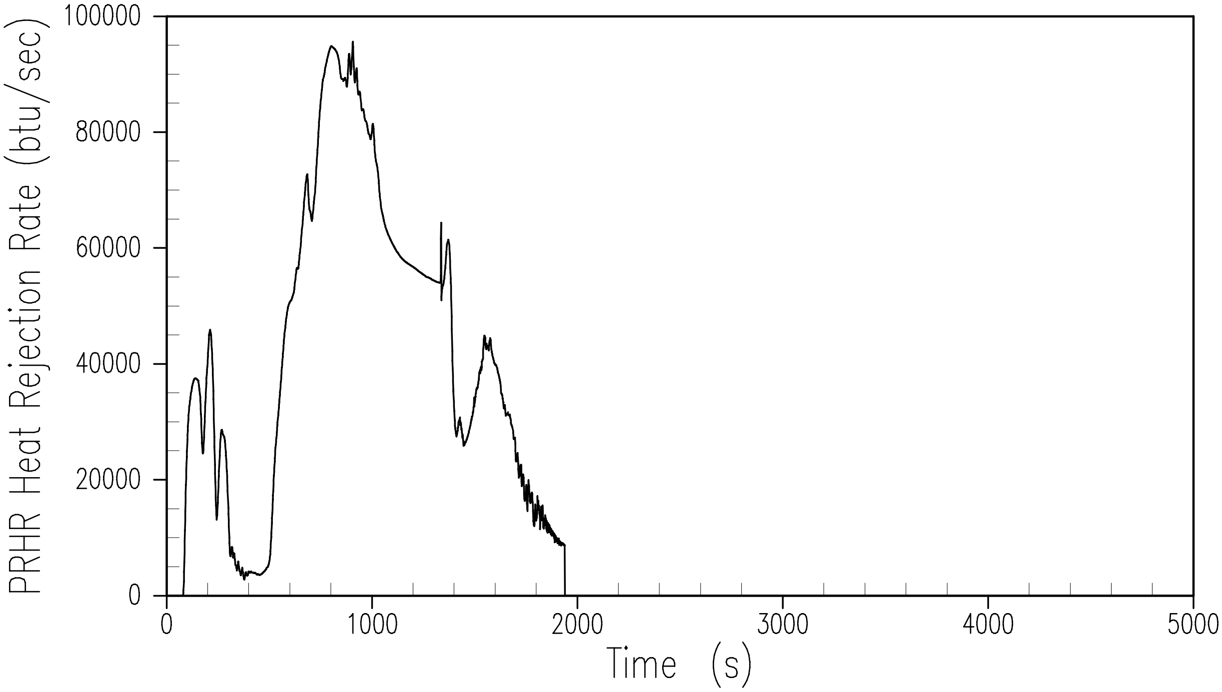

Figure 15.6.5.4B-34

2-Inch Cold Leg Break – PRHR Heat Removal Rate

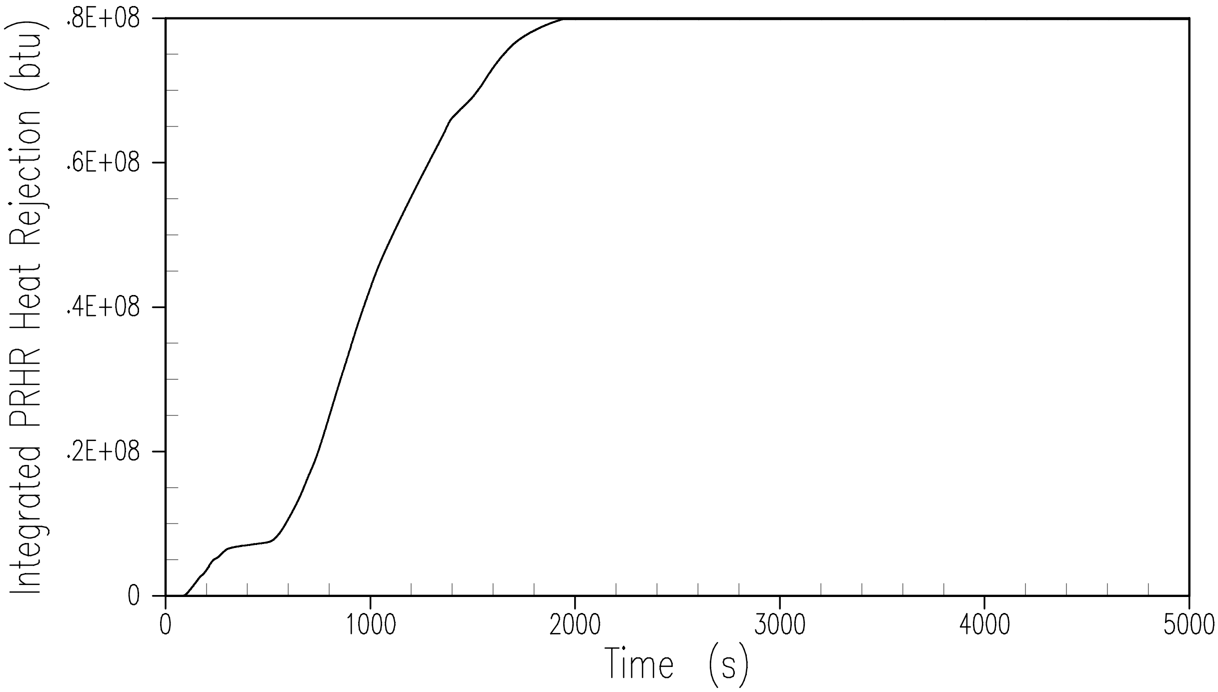

Figure 15.6.5.4B-35

2-Inch Cold Leg Break – Integrated PRHR Heat Removal

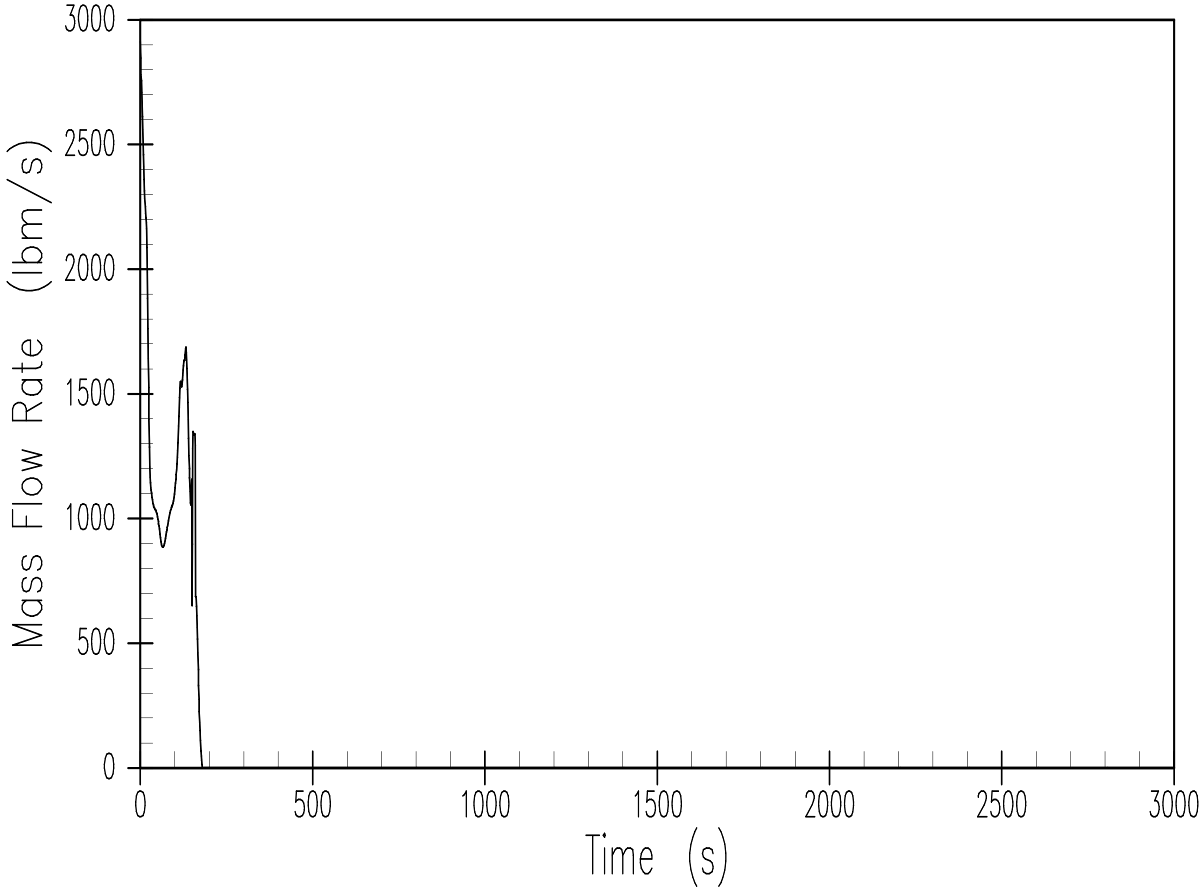

Figure 15.6.5.4B-36

DEDVI – Vessel Side Liquid Break Discharge – 20 psi

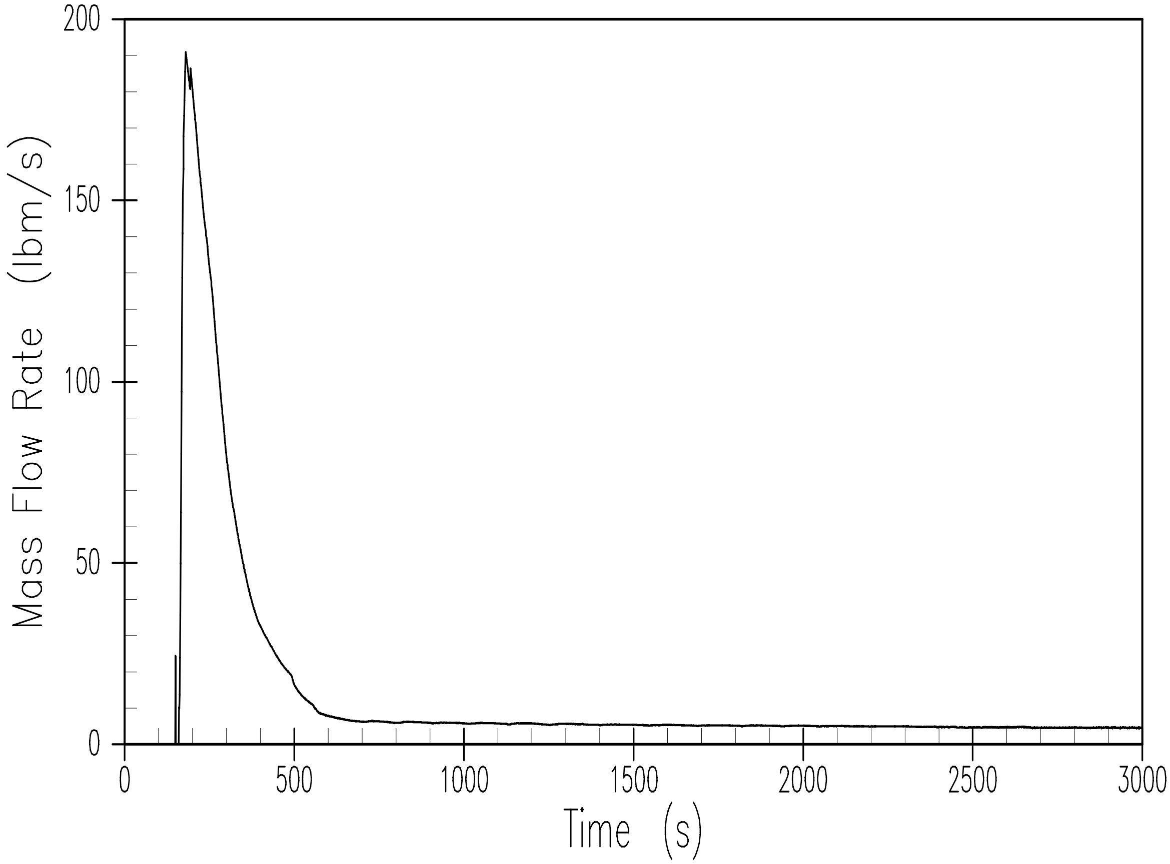

Figure 15.6.5.4B-37

DEDVI – Vessel Side Vapor Break Discharge – 20 psi

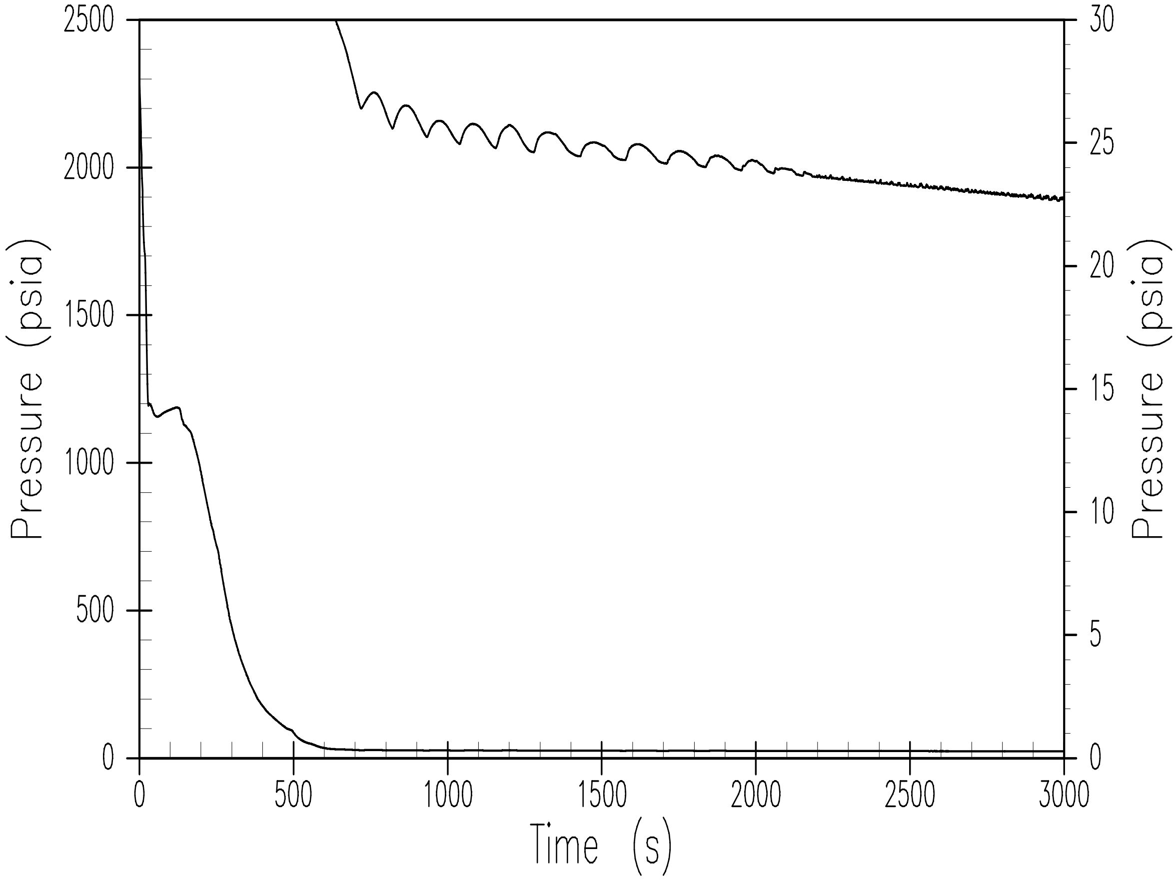

Figure 15.6.5.4B-38

DEDVI – RCS Pressure – 20 psi

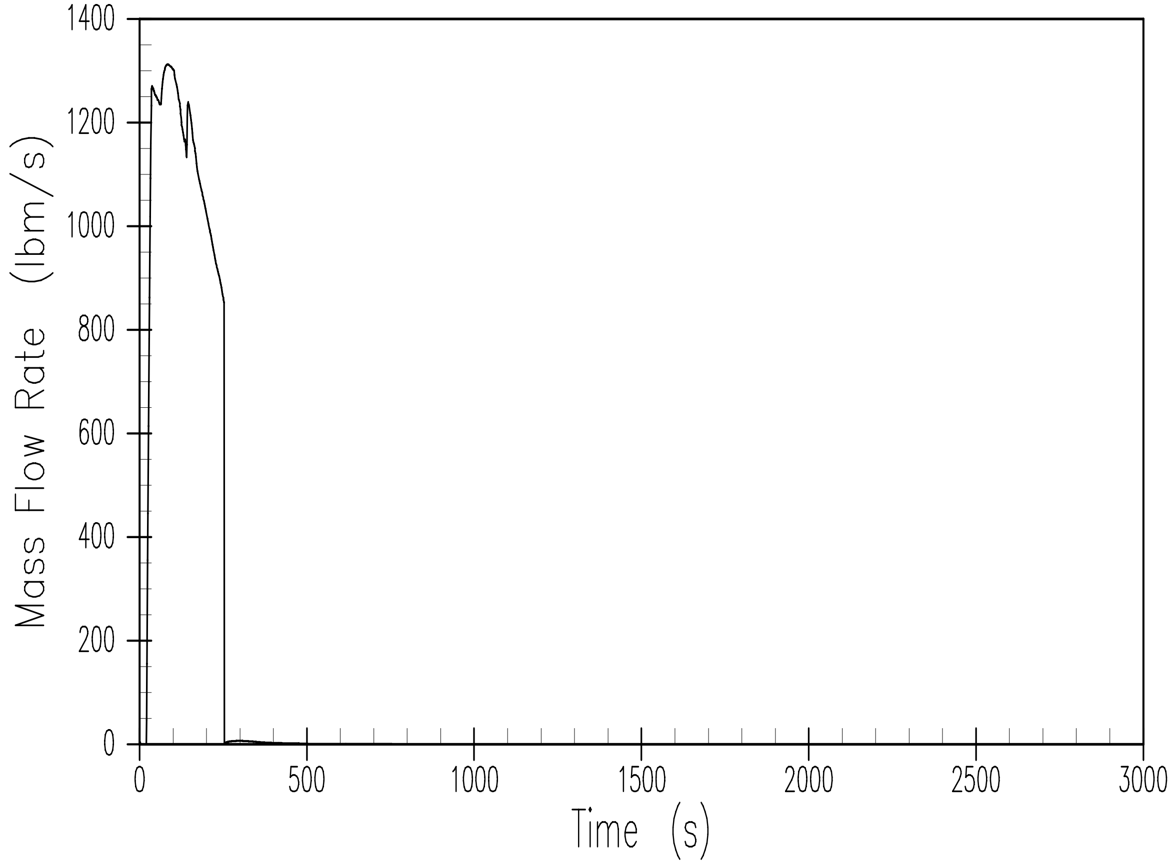

Figure 15.6.5.4B-39

DEDVI – Broken CMT Injection Rate – 20 psi

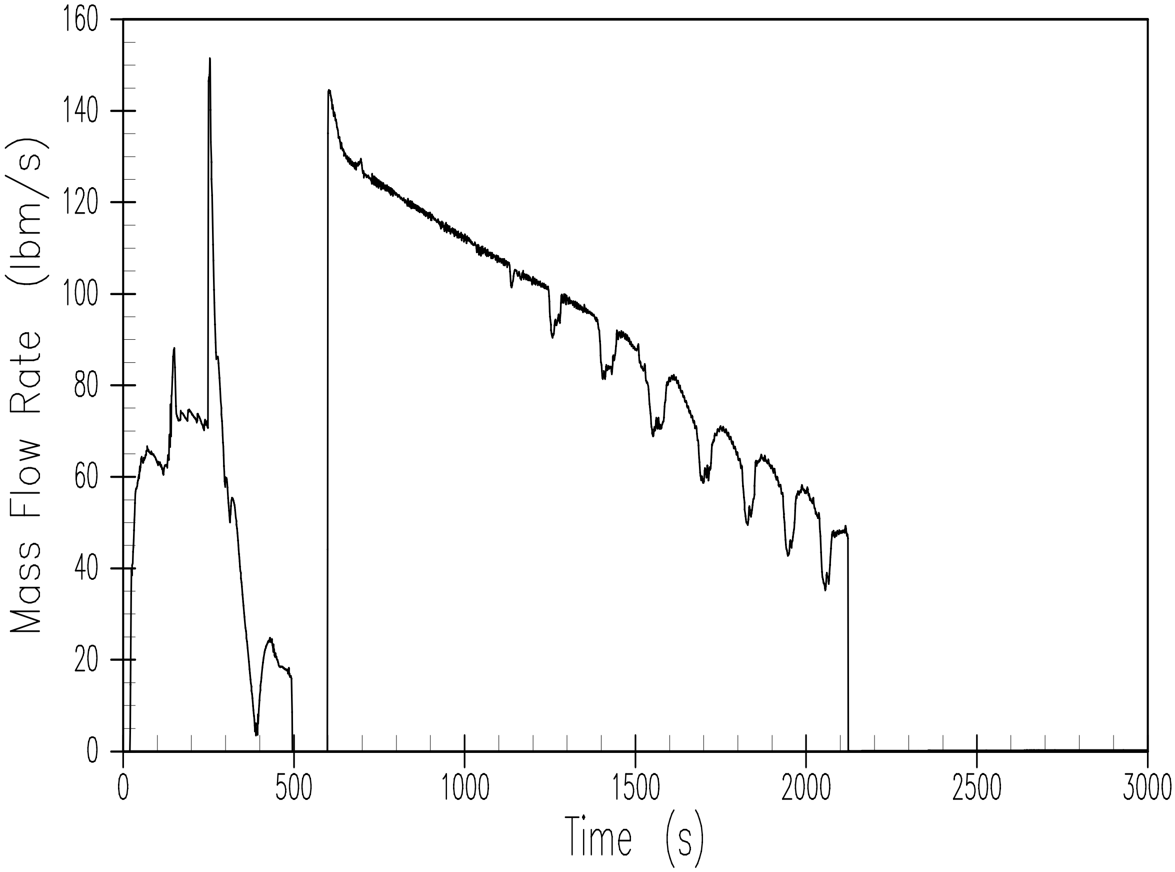

Figure 15.6.5.4B-40

DEDVI – Intact CMT Injection Rate – 20 psi

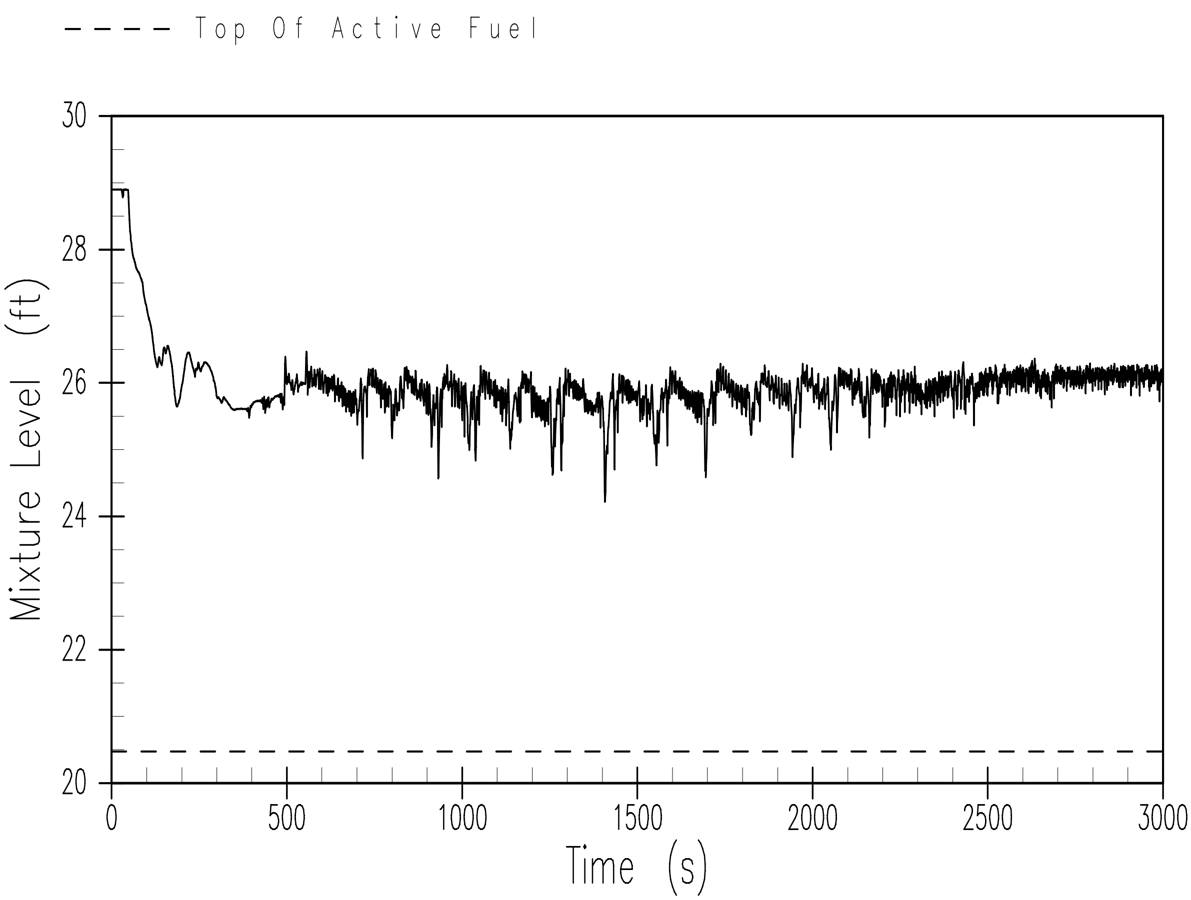

Figure 15.6.5.4B-41

DEDVI – Core/Upper Plenum Mixture Level – 20 psi

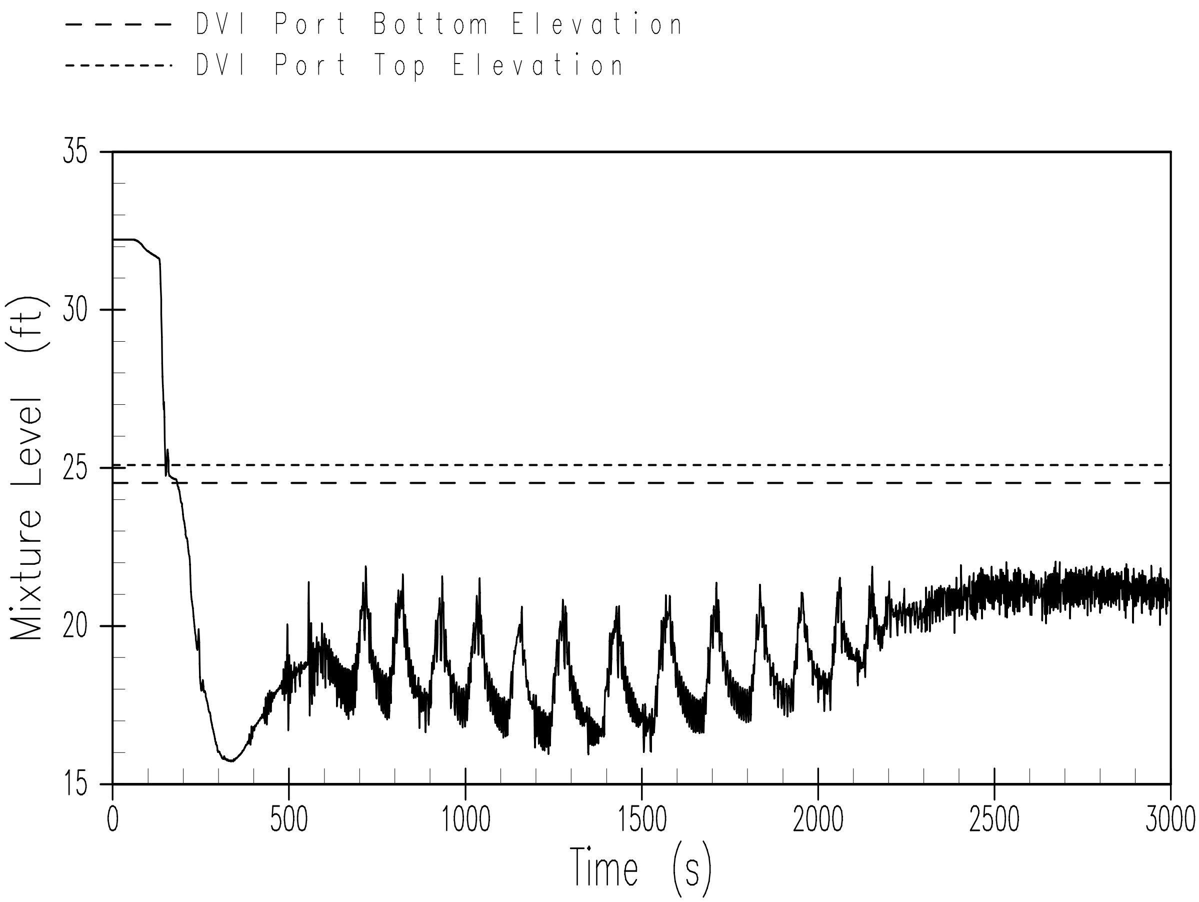

Figure 15.6.5.4B-42

DEDVI – Downcomer Mixture Level – 20 psi

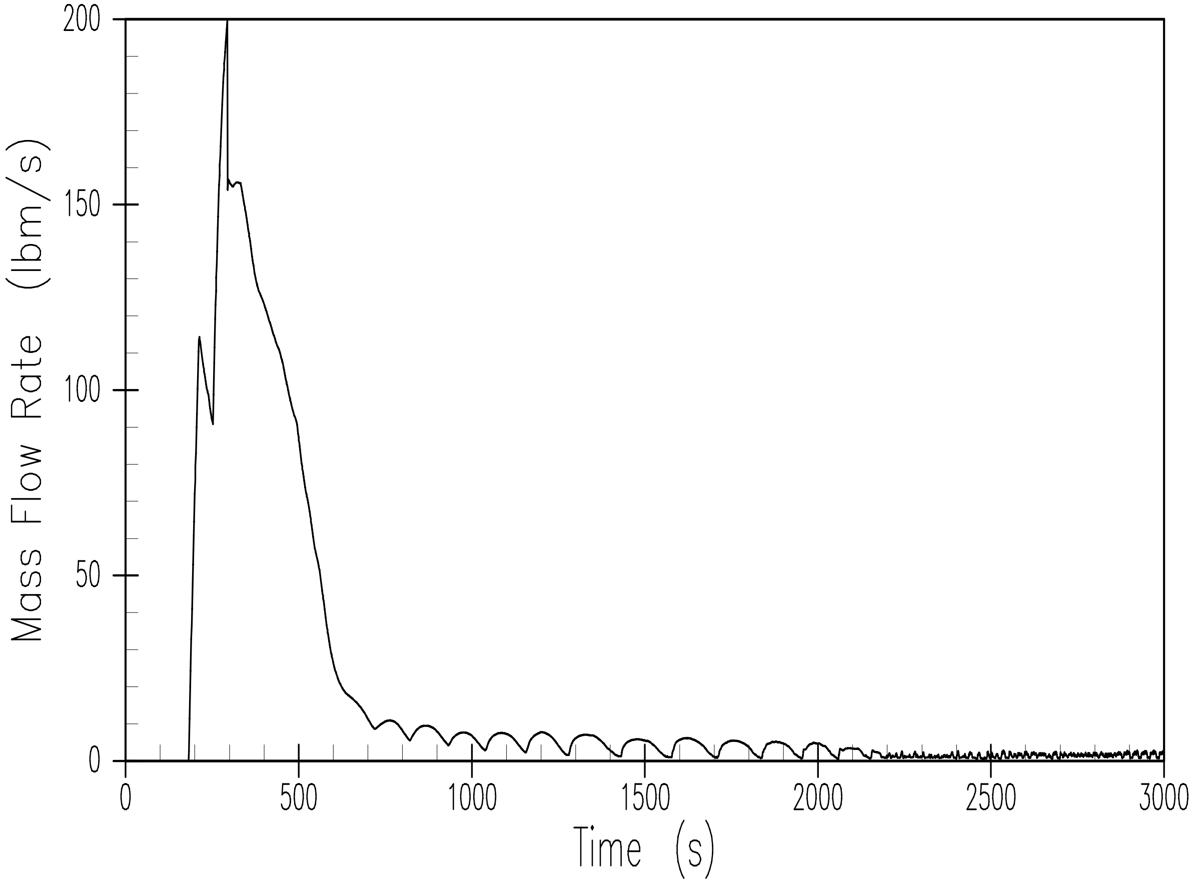

Figure 15.6.5.4B-43

DEDVI – ADS 1-3 Vapor Discharge – 20 psi

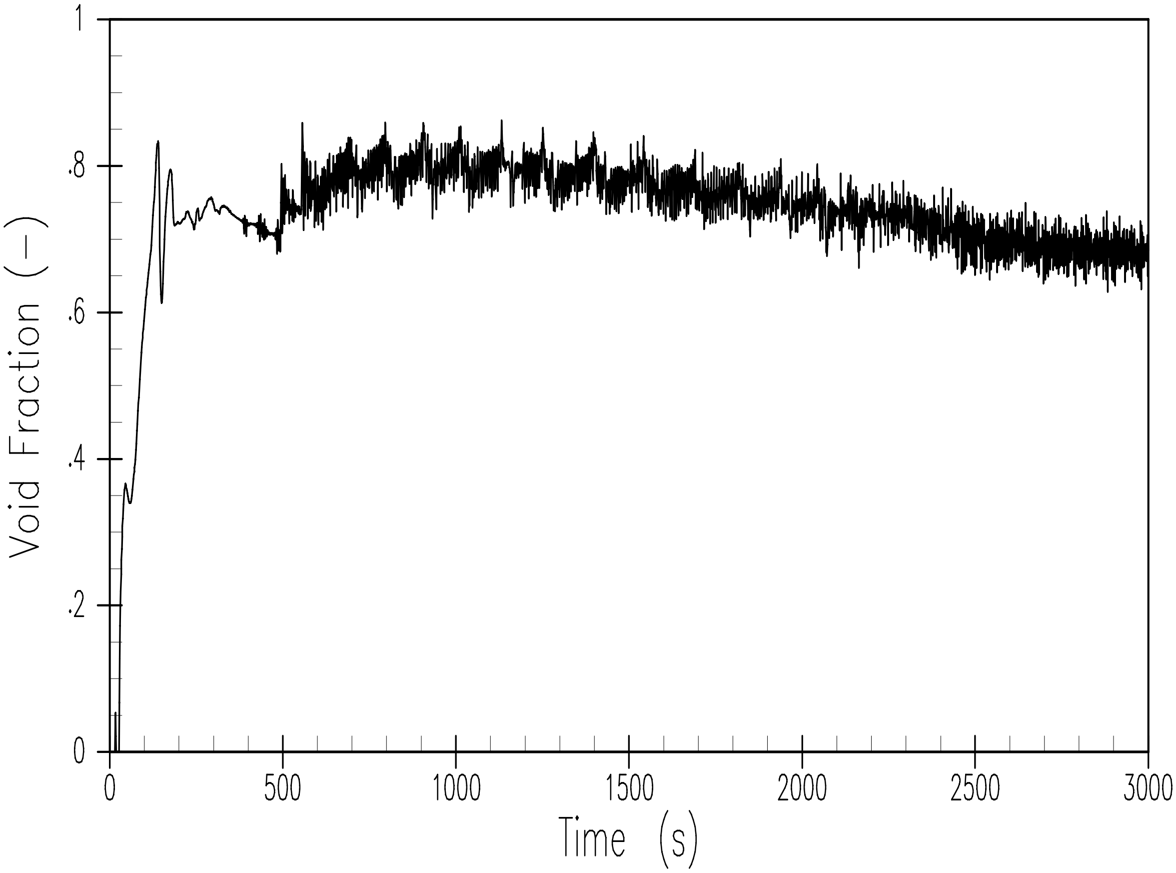

Figure 15.6.5.4B-44

DEDVI – Core Exit Void Fraction – 20 psi

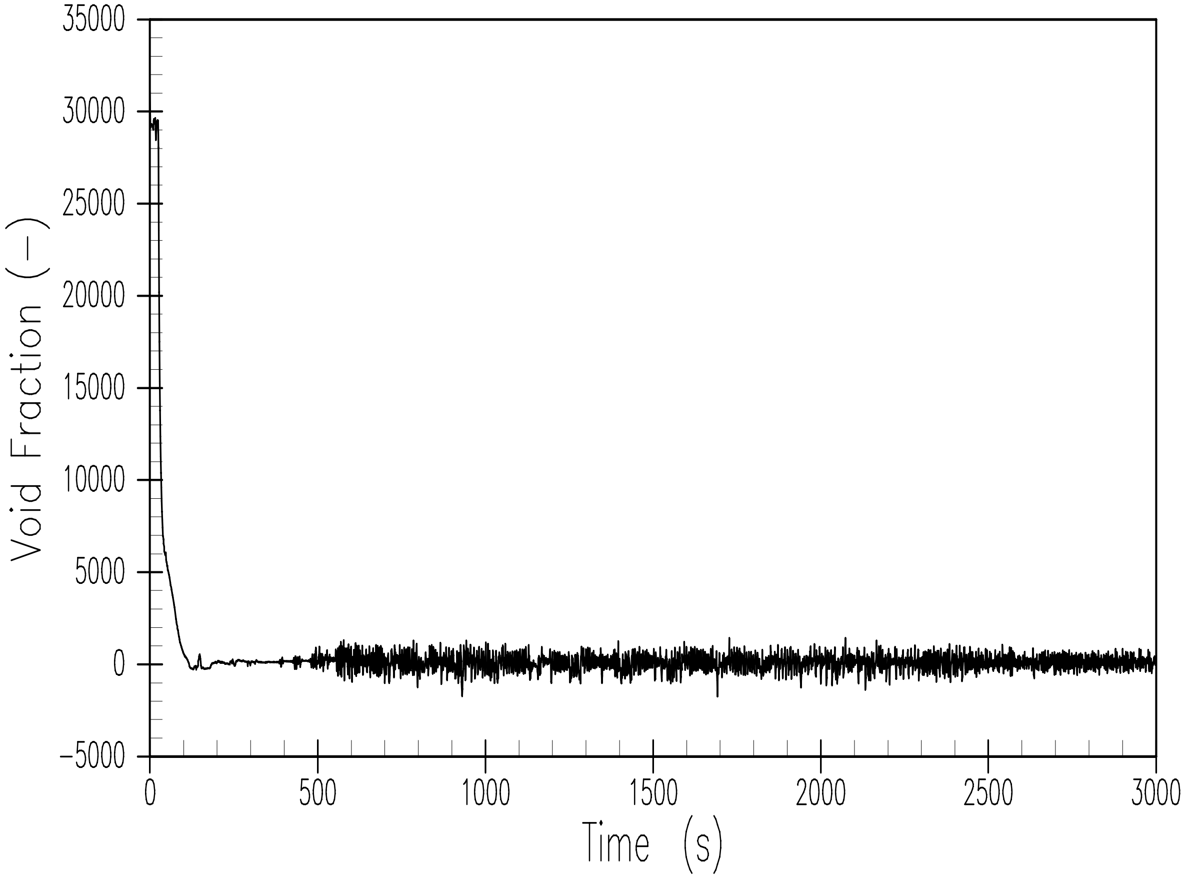

Figure 15.6.5.4B-45

DEDVI – Core Exit Liquid Flow Rate – 20 psi

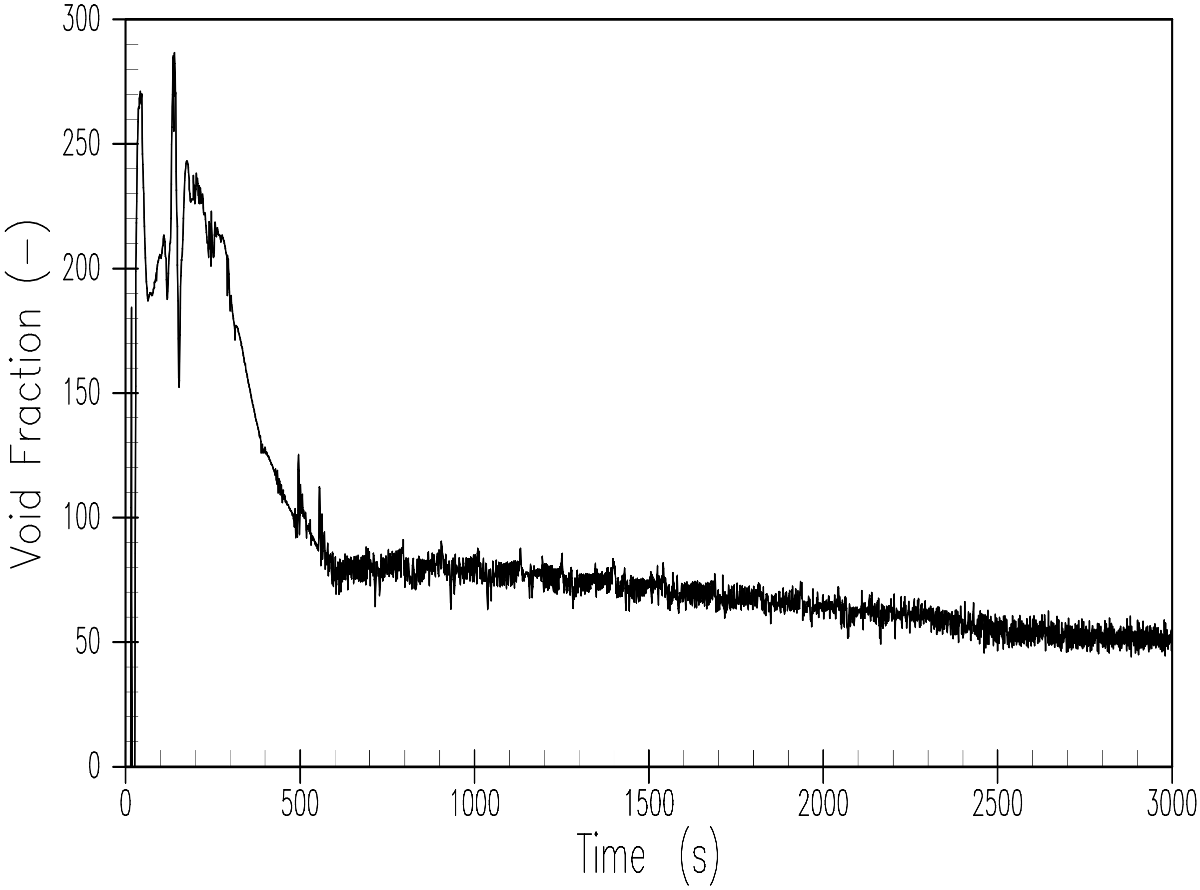

Figure 15.6.5.4B-46

DEDVI – Core Exit Vapor Flow Rate – 20 psi

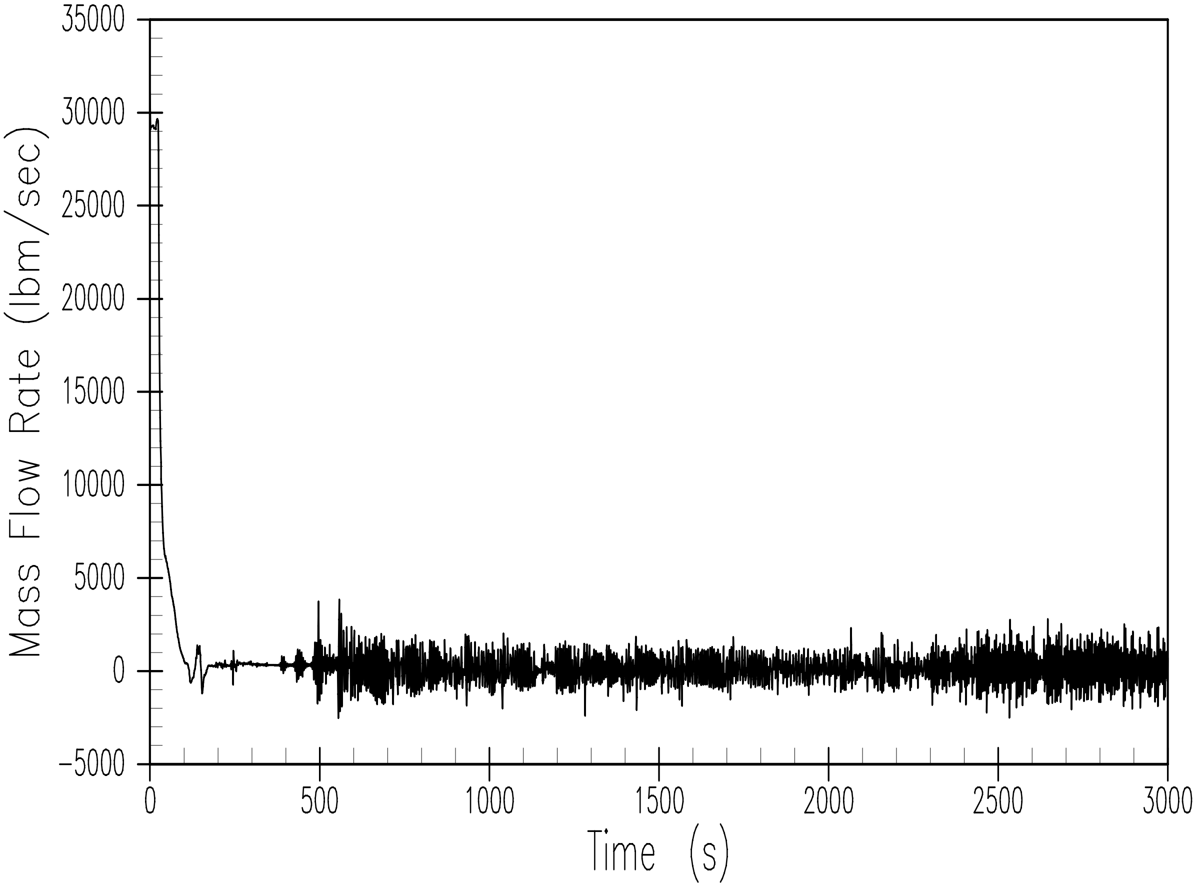

Figure 15.6.5.4B-47

DEDVI – Lower Plenum to Core Flow Rate – 20 psi

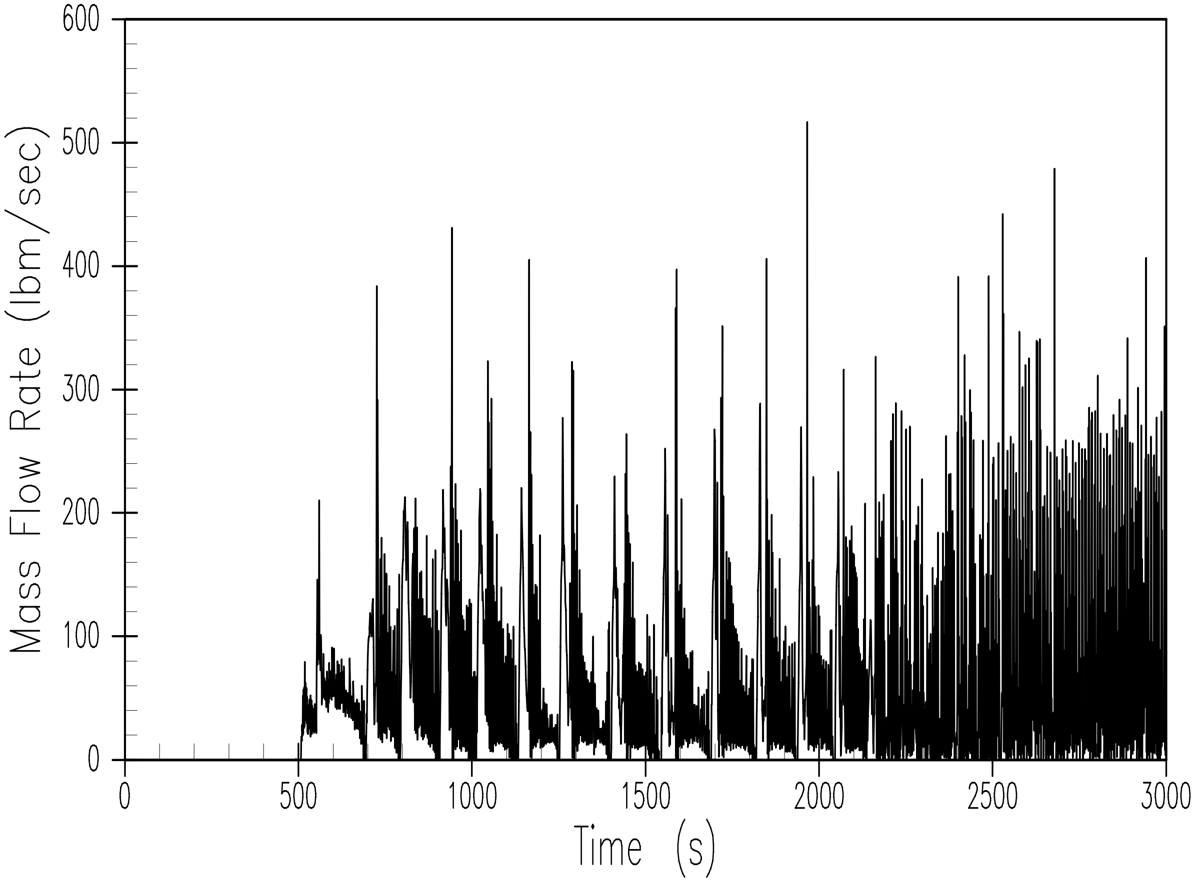

Figure 15.6.5.4B-48

DEDVI – ADS-4 Liquid Discharge – 20 psi

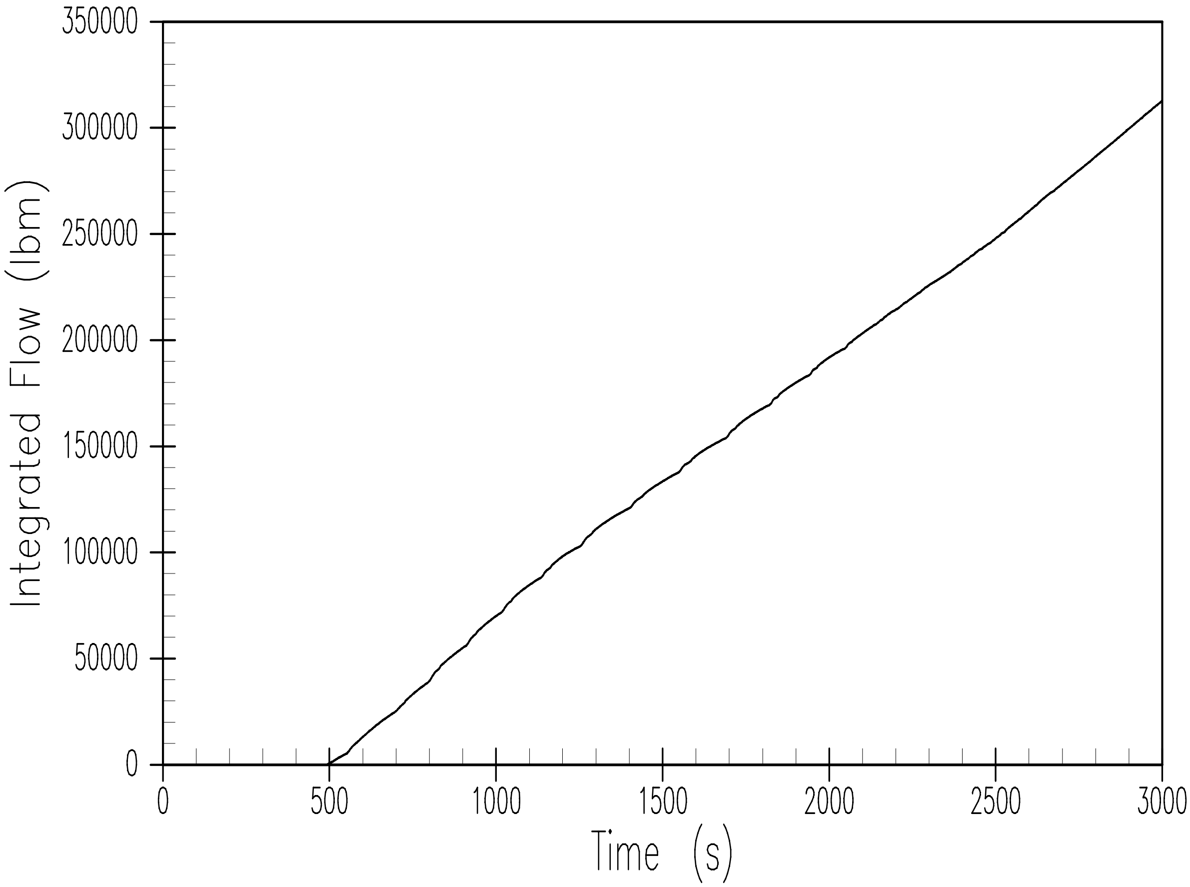

Figure 15.6.5.4B-49

DEDVI – ADS-4 Integrated Discharge – 20 psi

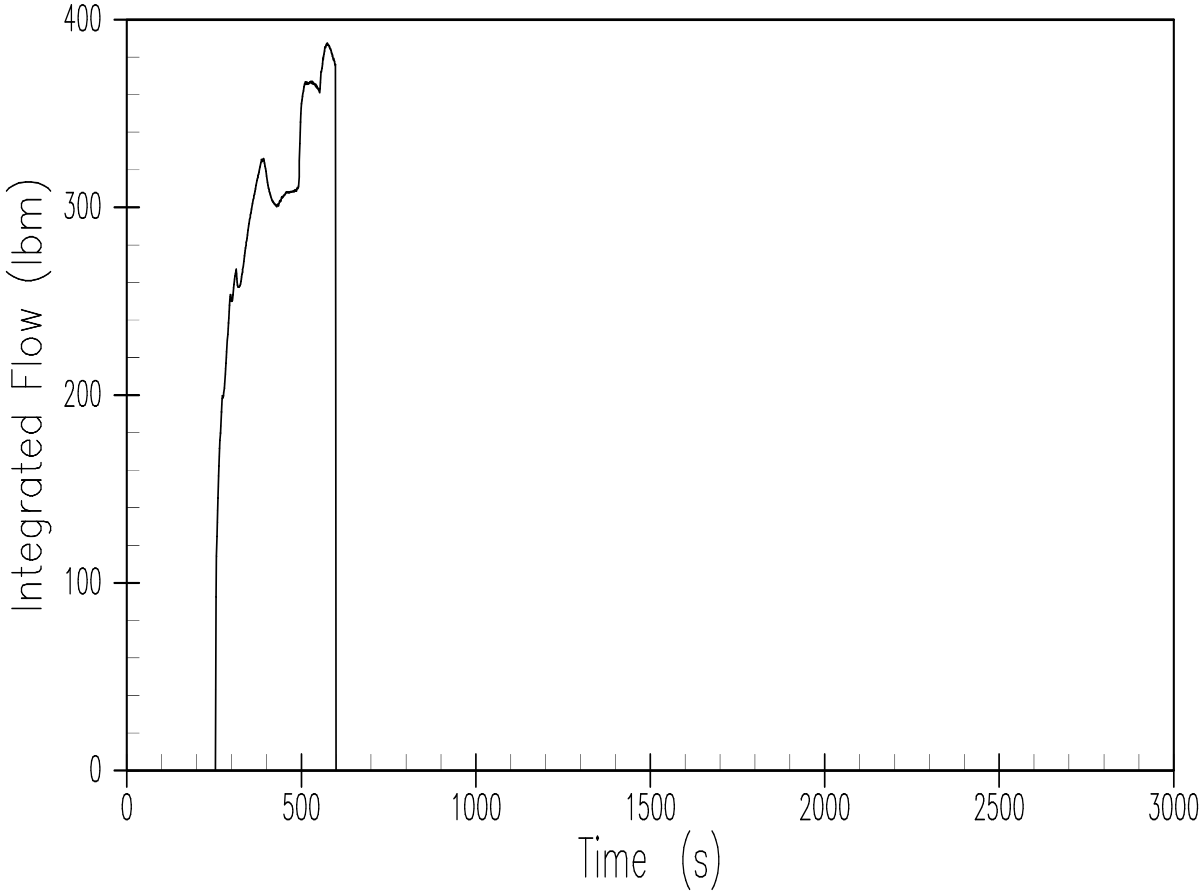

Figure 15.6.5.4B-50

DEDVI – Intact Accumulator Flow Rate – 20 psi

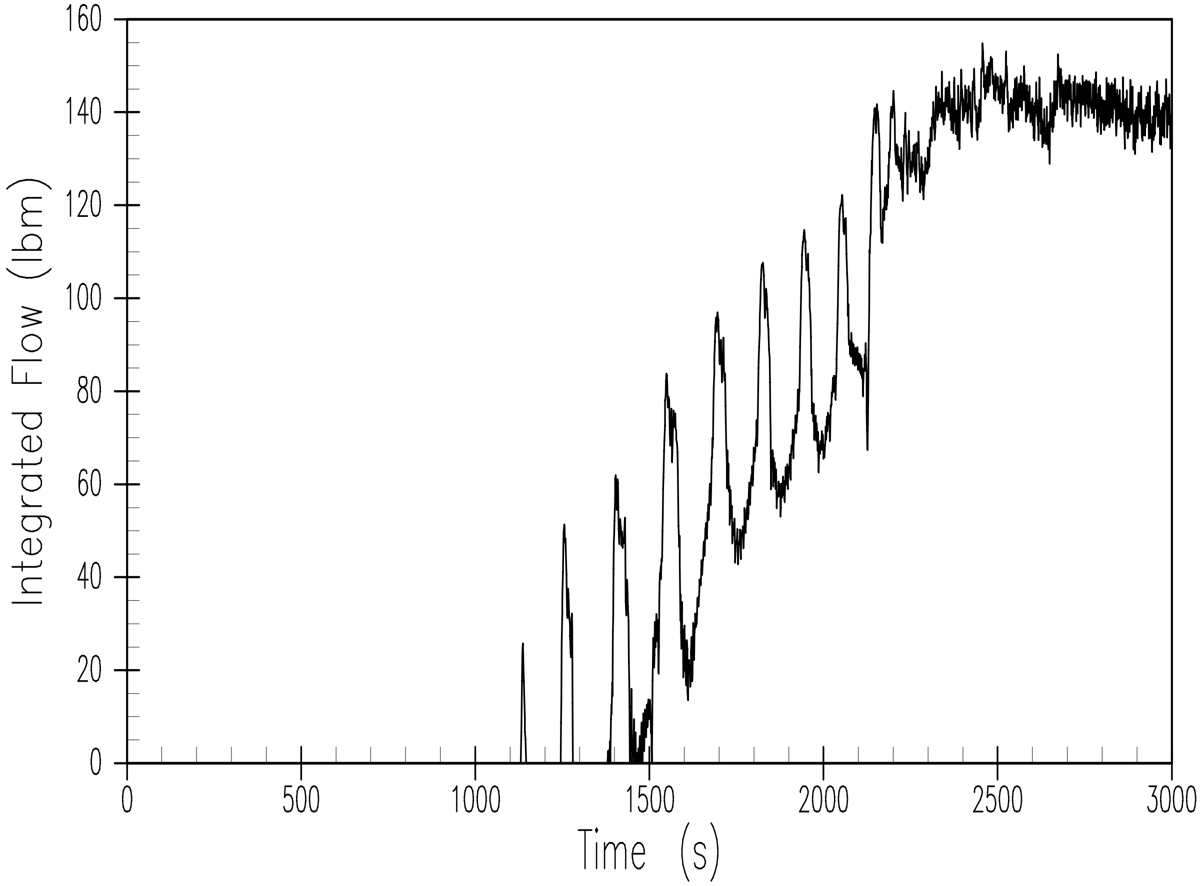

Figure 15.6.5.4B-51

DEDVI – Intact IRWST Injection Rate – 20 psi

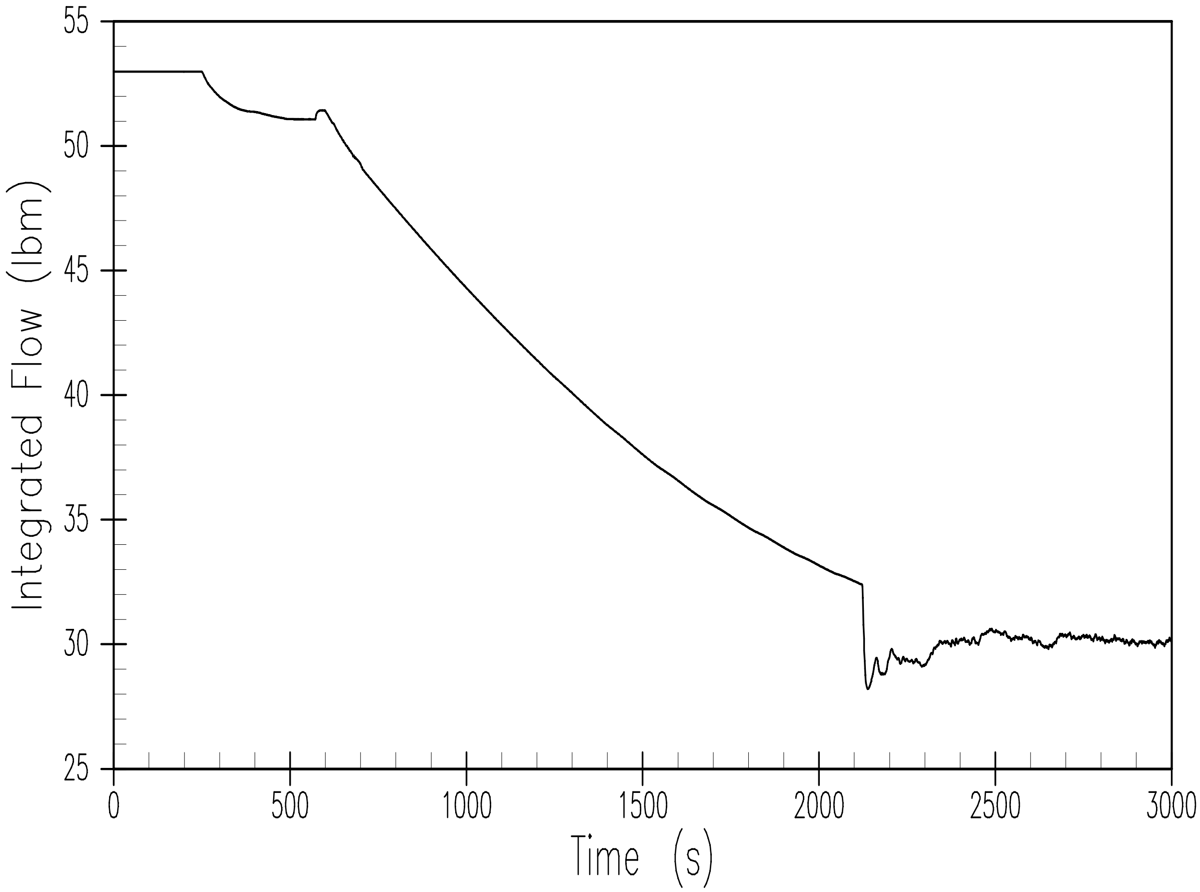

Figure 15.6.5.4B-52

DEDVI – Intact CMT Mixture Level – 20 psi

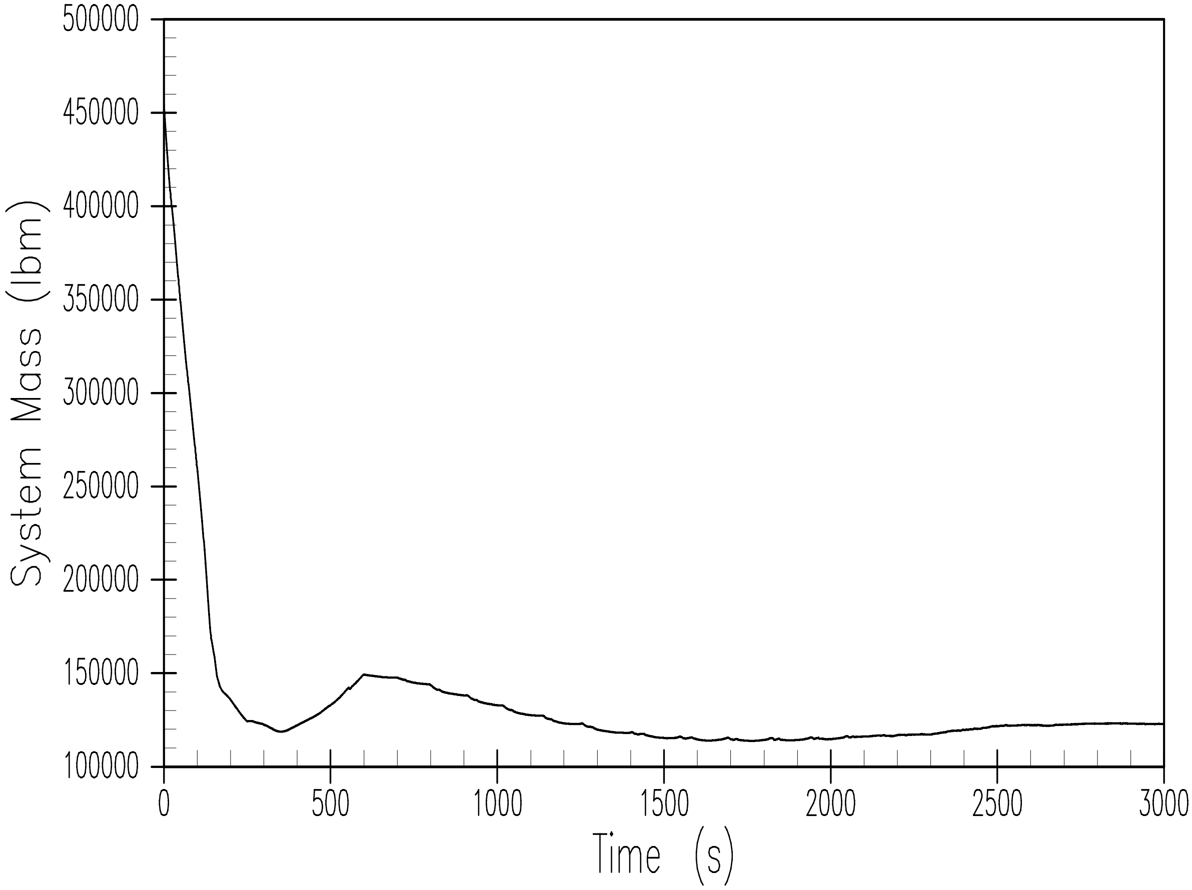

Figure 15.6.5.4B-53

DEDVI – RCS System Inventory – 20 psi

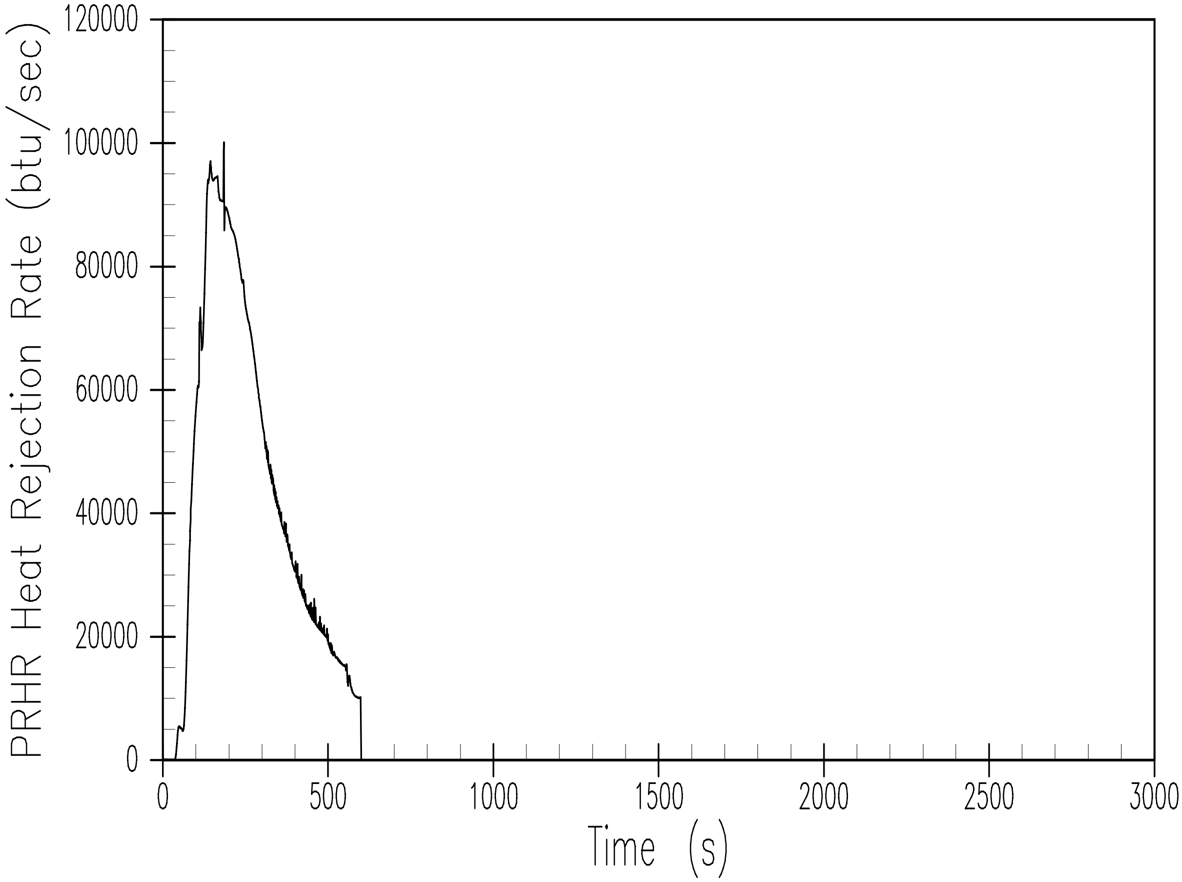

Figure 15.6.5.4B-54

DEDVI – PRHR Heat Removal Rate – 20 psi

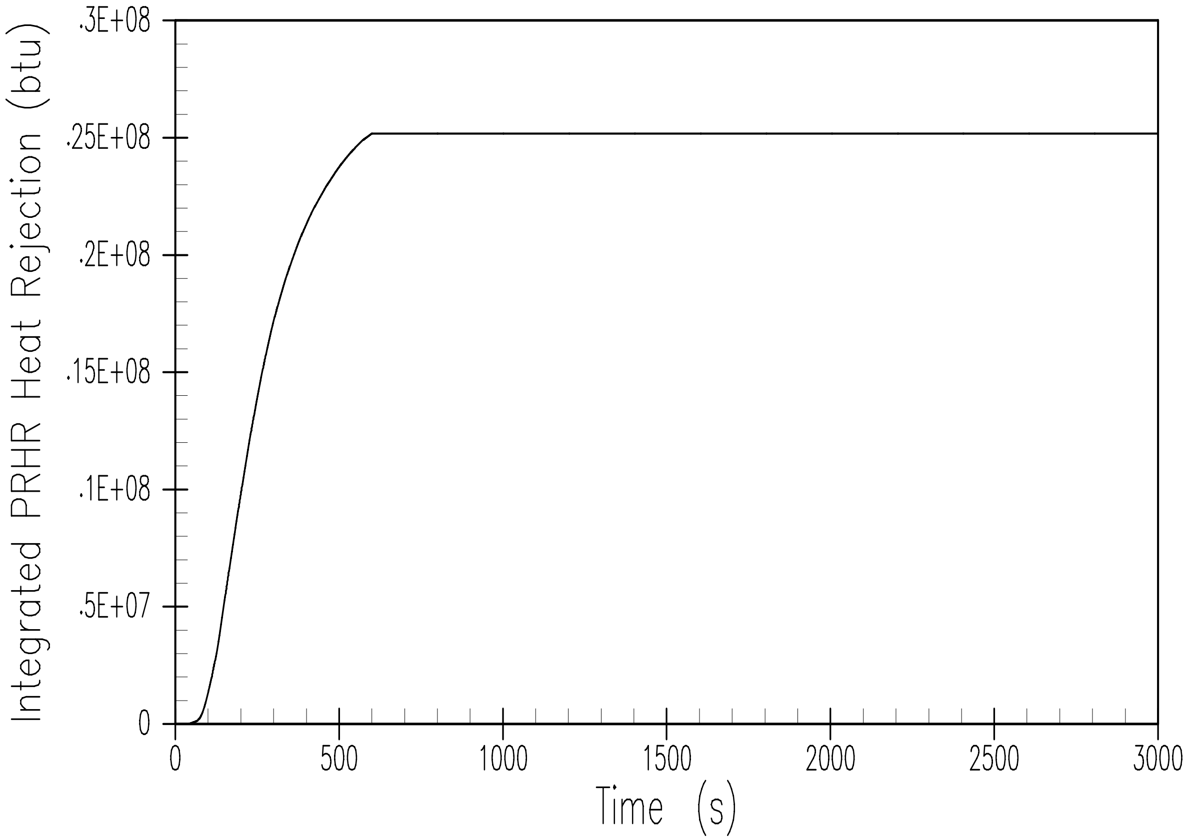

Figure 15.6.5.4B-55

DEDVI – Integrated PRHR Heat Removal – 20 psi

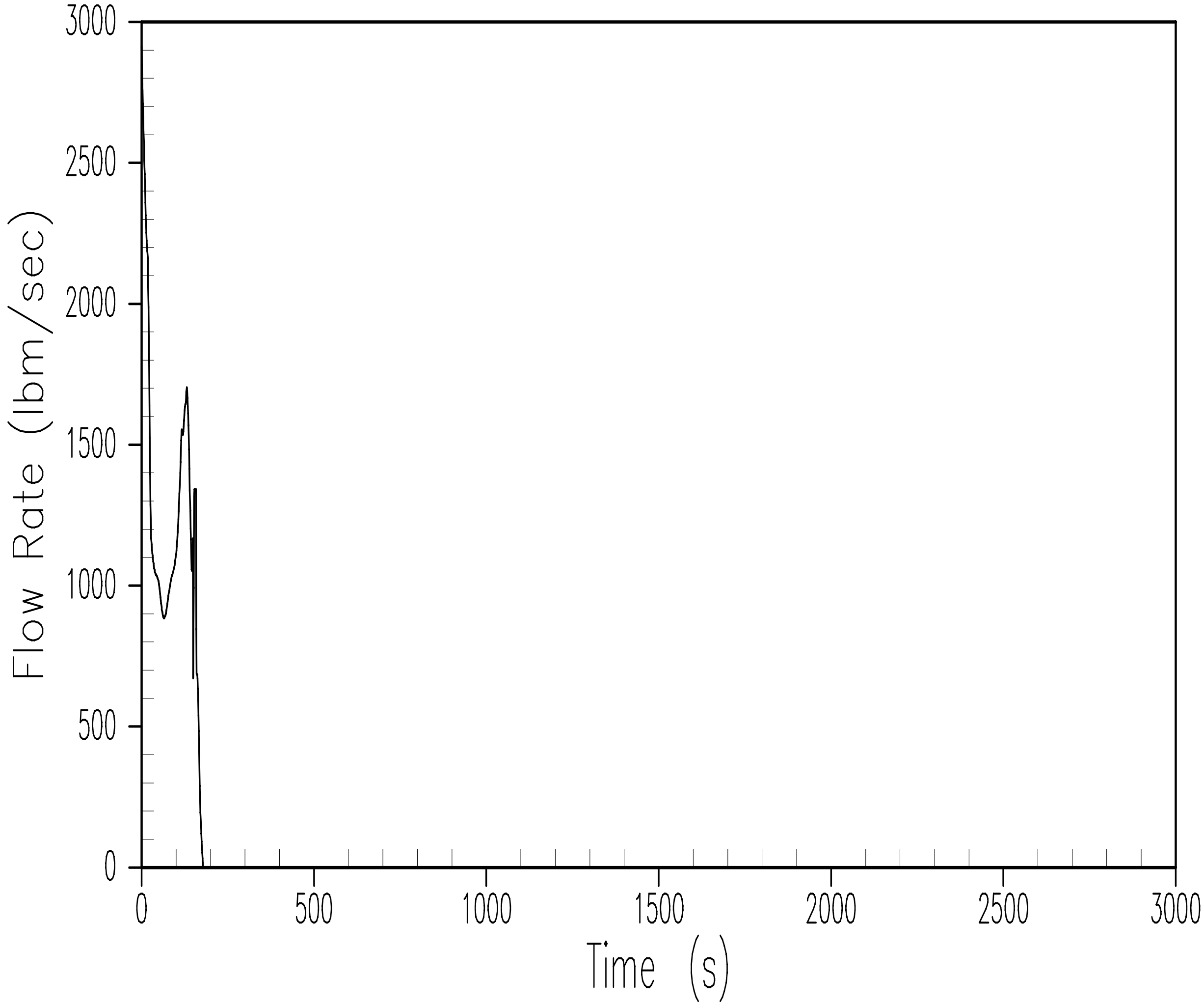

Figure 15.6.5.4B-36A

DEDVI – Vessel Side Liquid Break Discharge – 14.7 psi

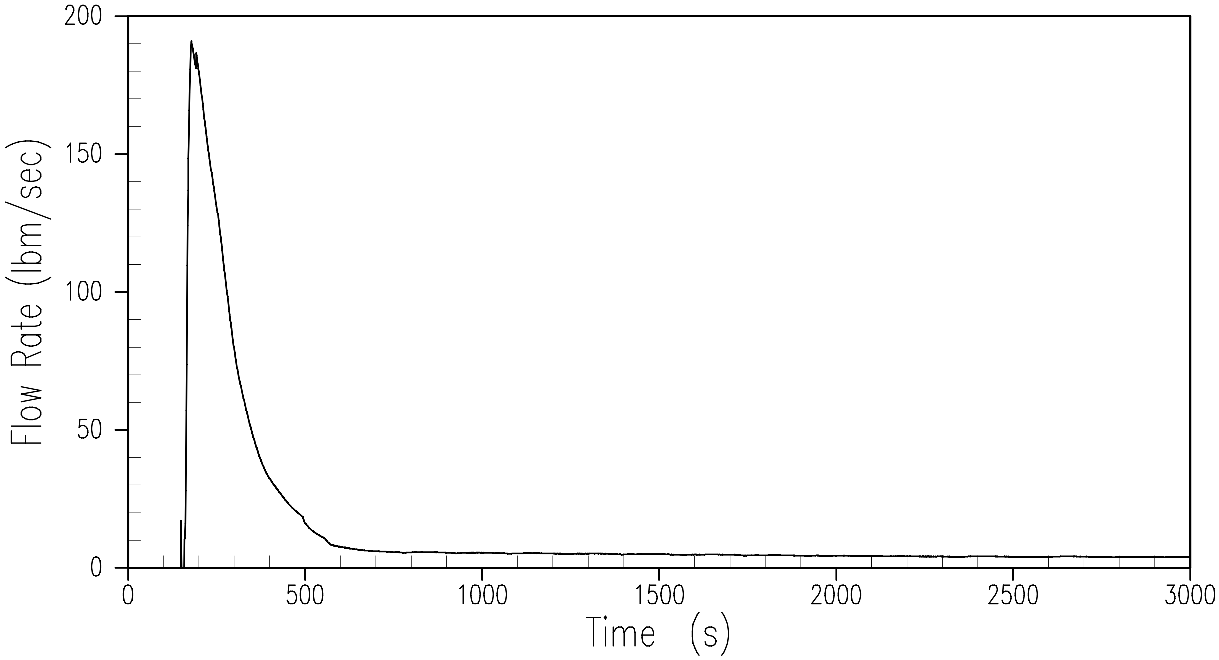

Figure 15.6.5.4B-37A

DEDVI – Vessel Side Vapor Break Discharge – 14.7 psi

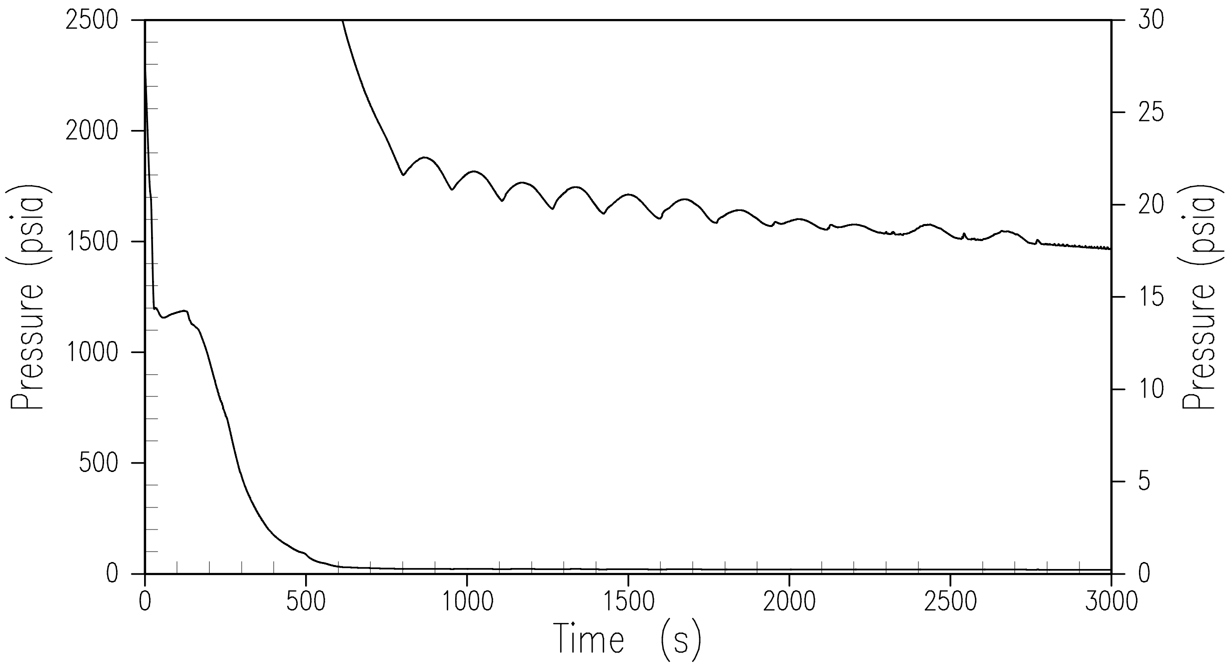

Figure 15.6.5.4B-38A

DEDVI – RCS Pressure – 14.7 psi

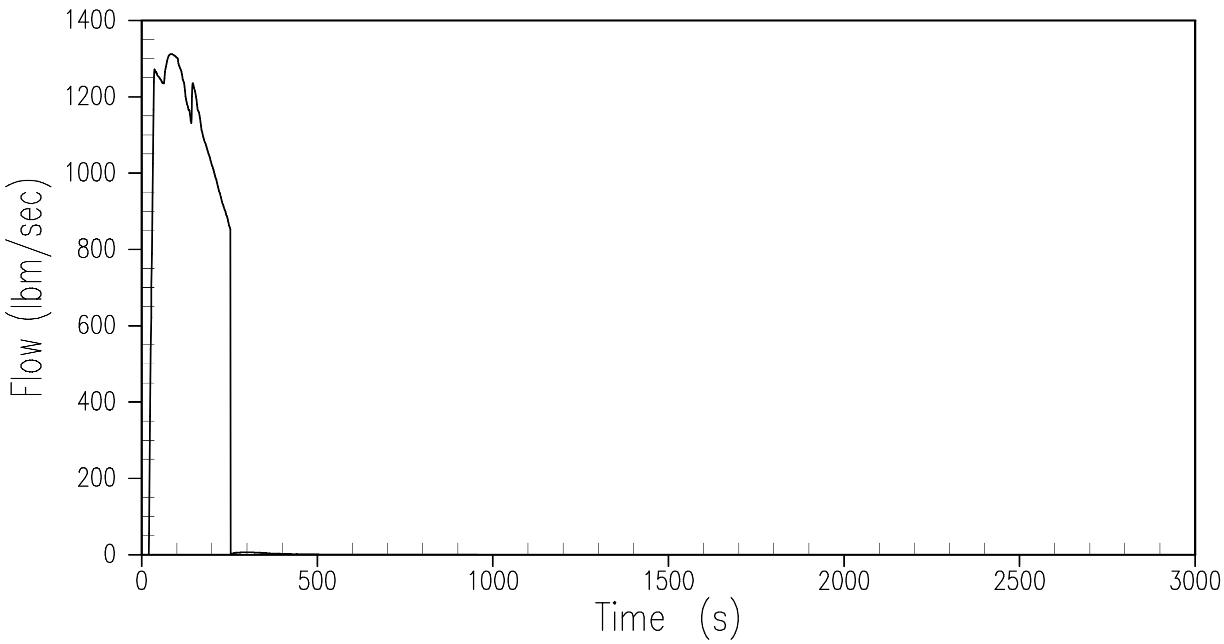

Figure 15.6.5.4B-39A

DEDVI – Broken CMT Injection Rate – 14.7 psi

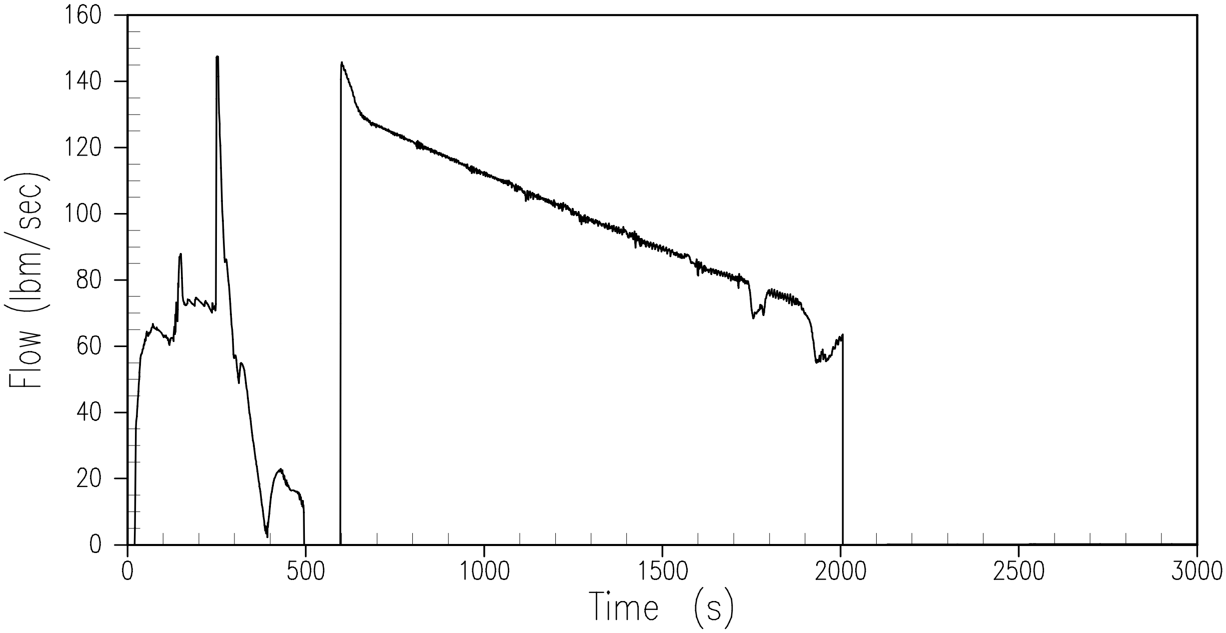

Figure 15.6.5.4B-40A

DEDVI – Intact CMT Injection Rate – 14.7 psi

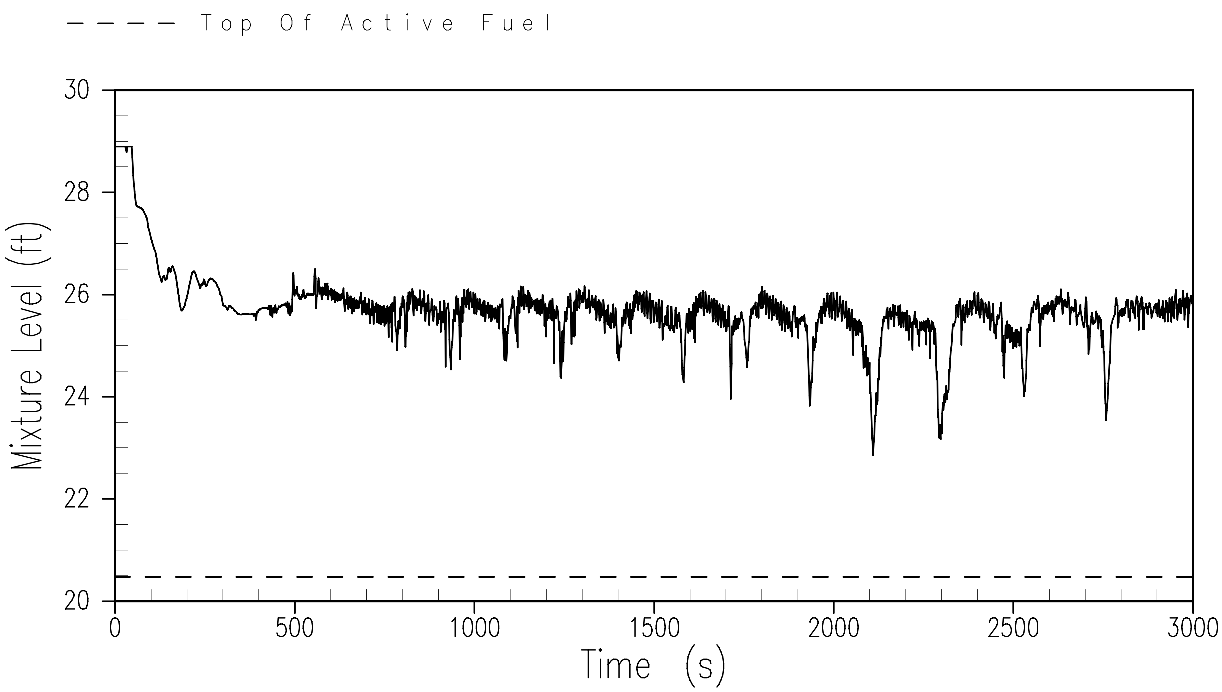

Figure 15.6.5.4B-41A

DEDVI – Core/Upper Plenum Mixture Level – 14.7 psi

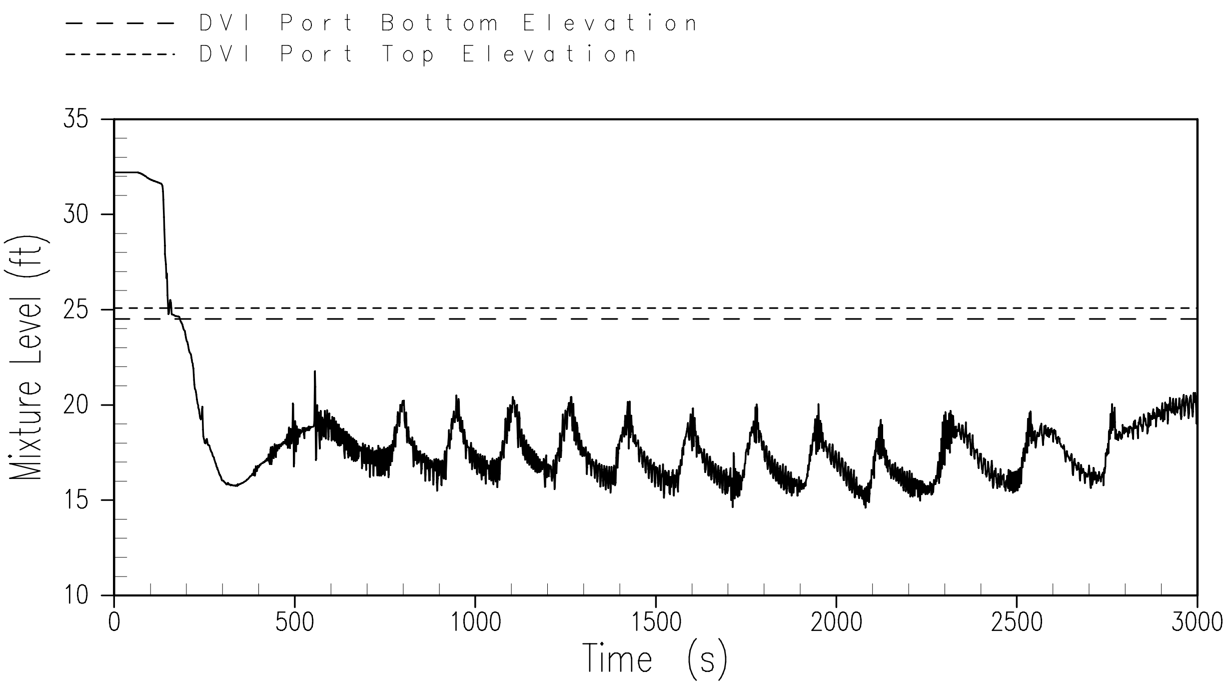

Figure 15.6.5.4B-42A

DEDVI – Downcomer Mixture Level – 14.7 psi

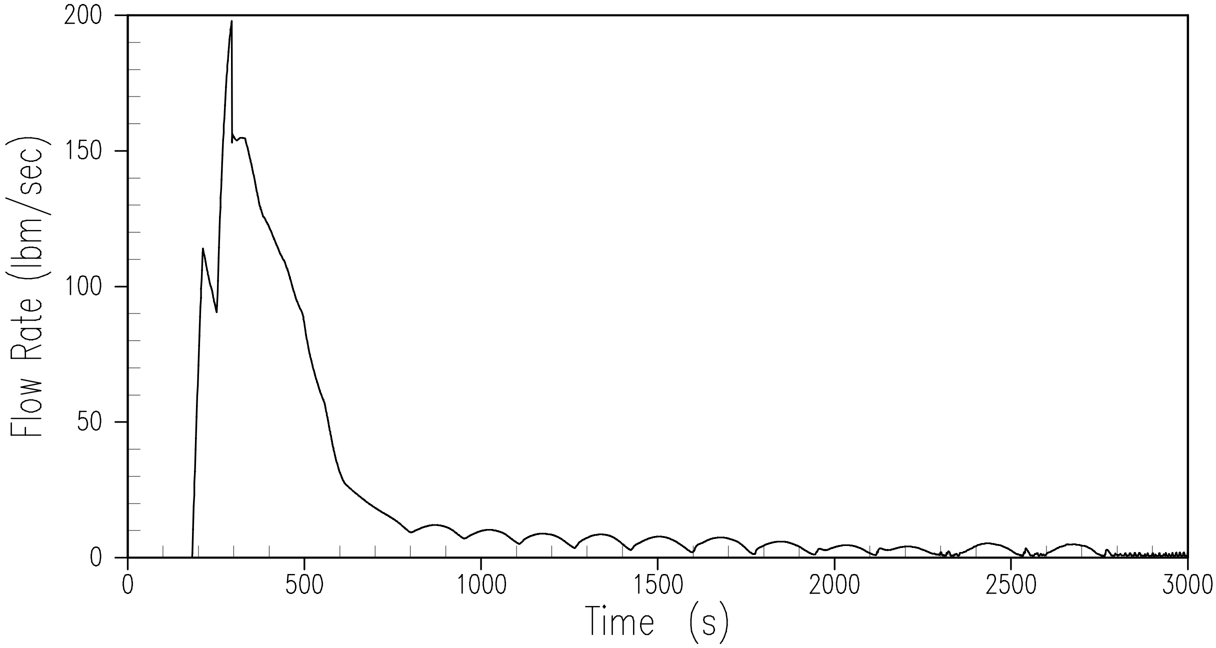

Figure 15.6.5.4B-43A

DEDVI – ADS 1-3 Vapor Discharge – 14.7 psi

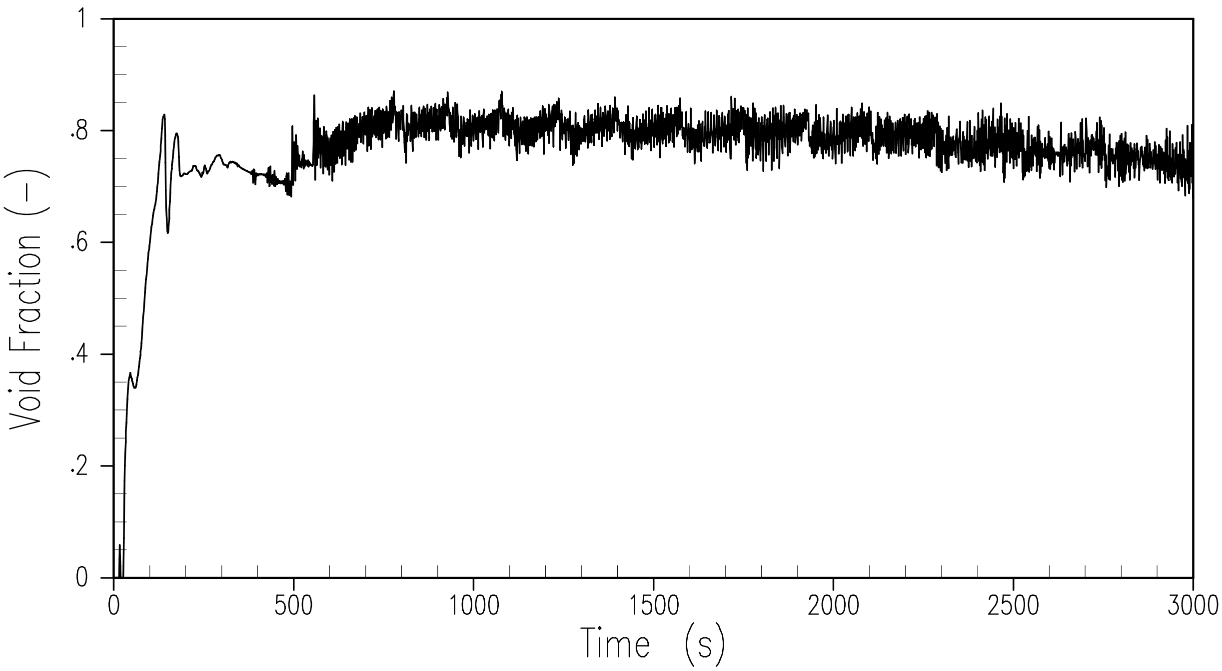

Figure 15.6.5.4B-44A

DEDVI – Core Exit Void Fraction – 14.7 psi

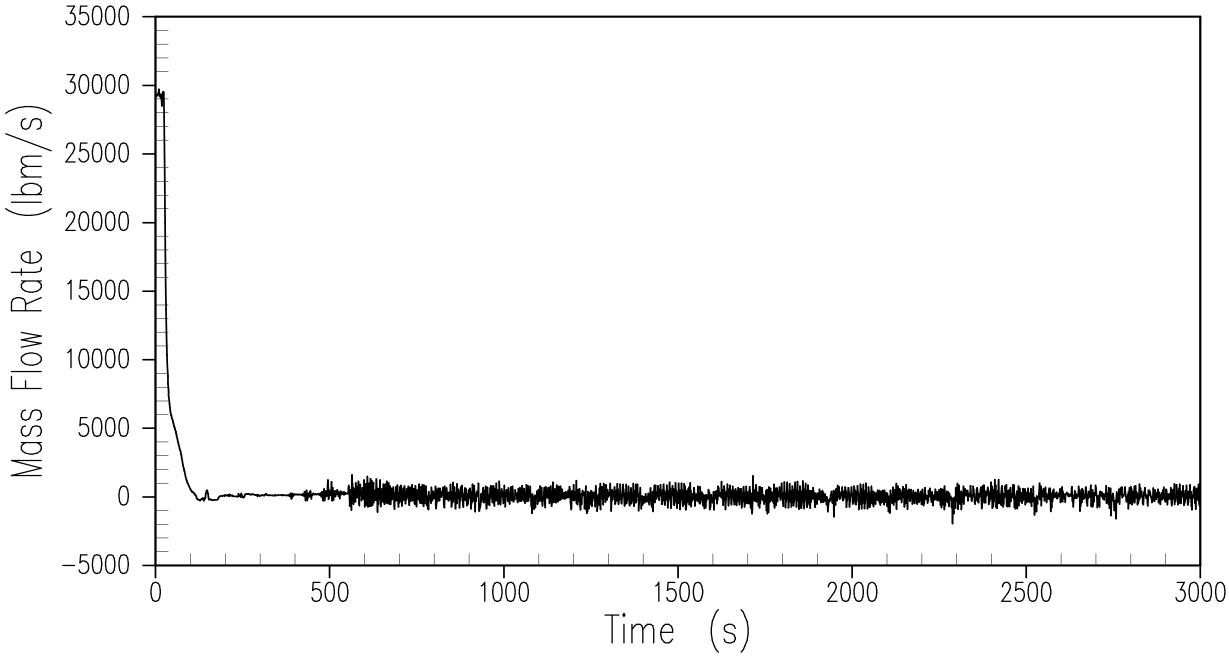

Figure 15.6.5.4B-45A

DEDVI – Core Exit Liquid Flow Rate – 14.7 psi

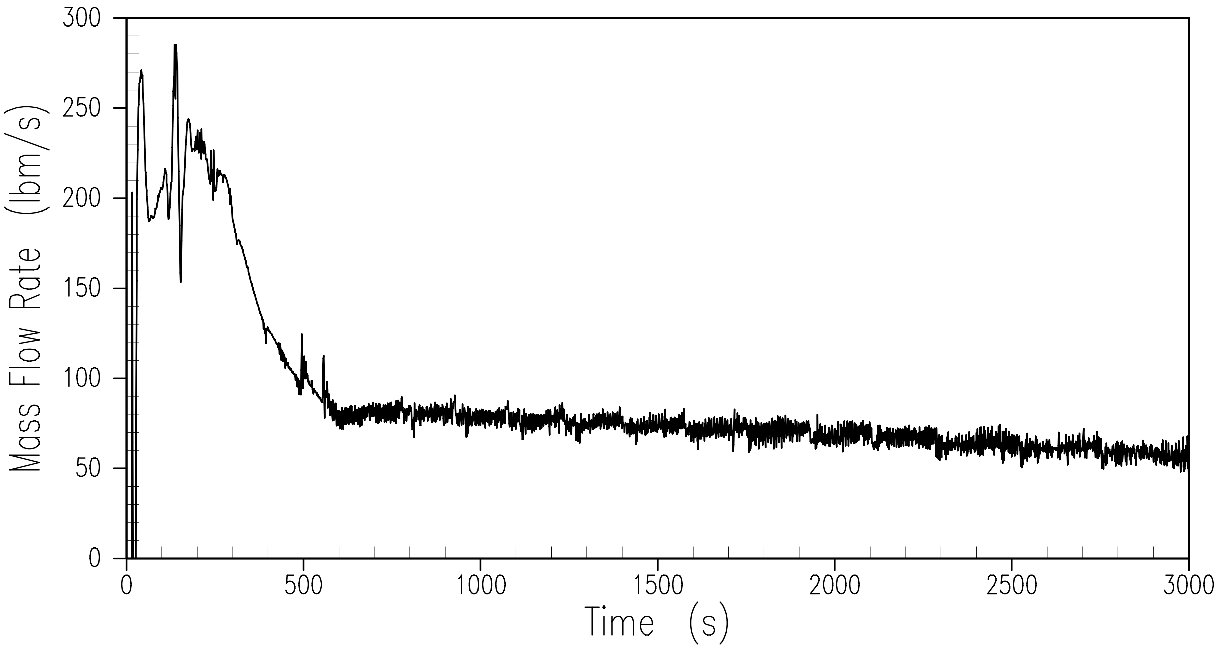

Figure 15.6.5.4B-46A

DEDVI – Core Exit Vapor Flow Rate – 14.7 psi

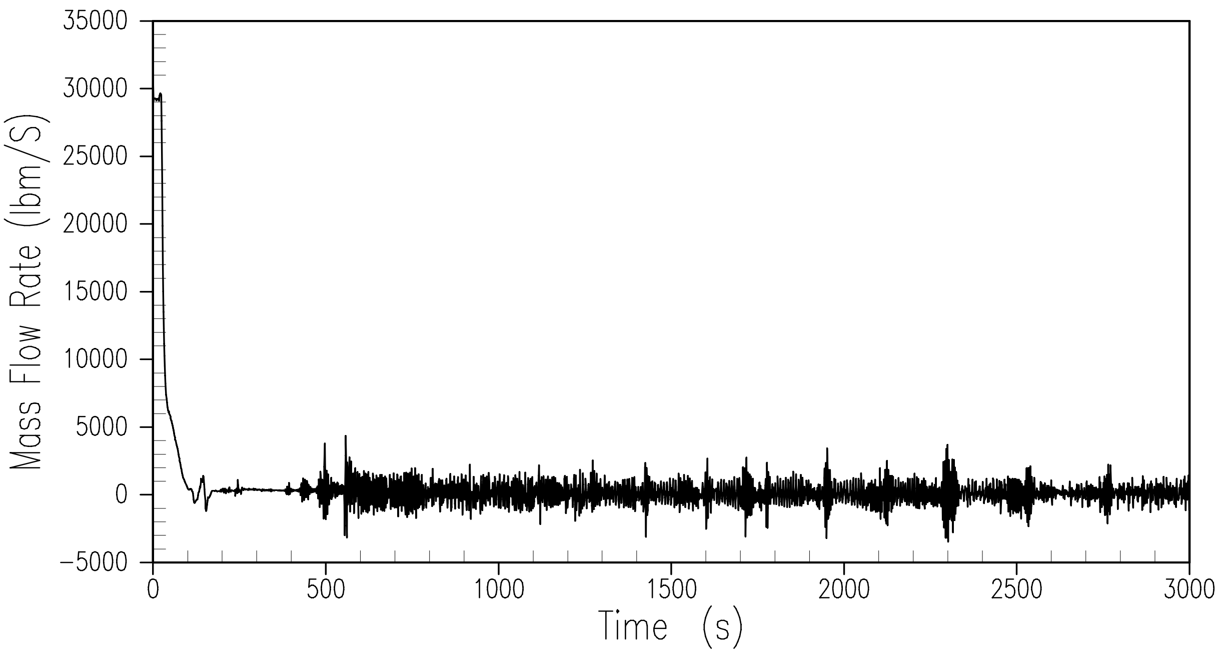

Figure 15.6.5.4B-47A

DEDVI – Lower Plenum to Core Flow Rate – 14.7 psi

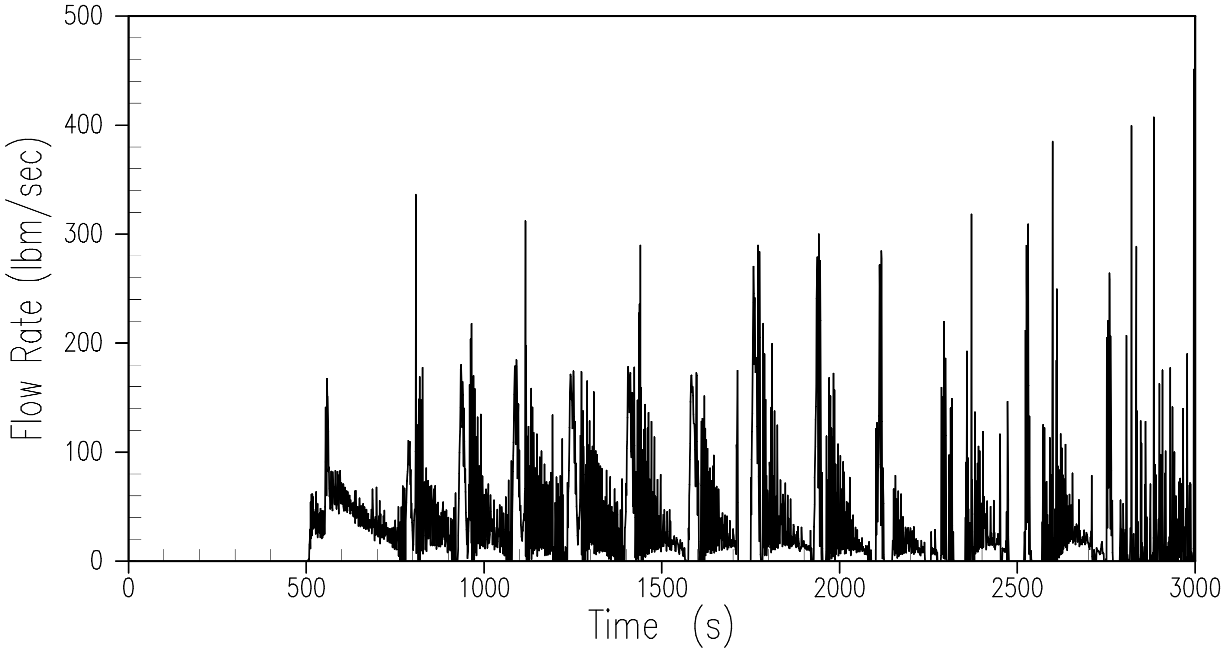

Figure 15.6.5.4B-48A

DEDVI – ADS-4 Liquid Discharge – 14.7 psi

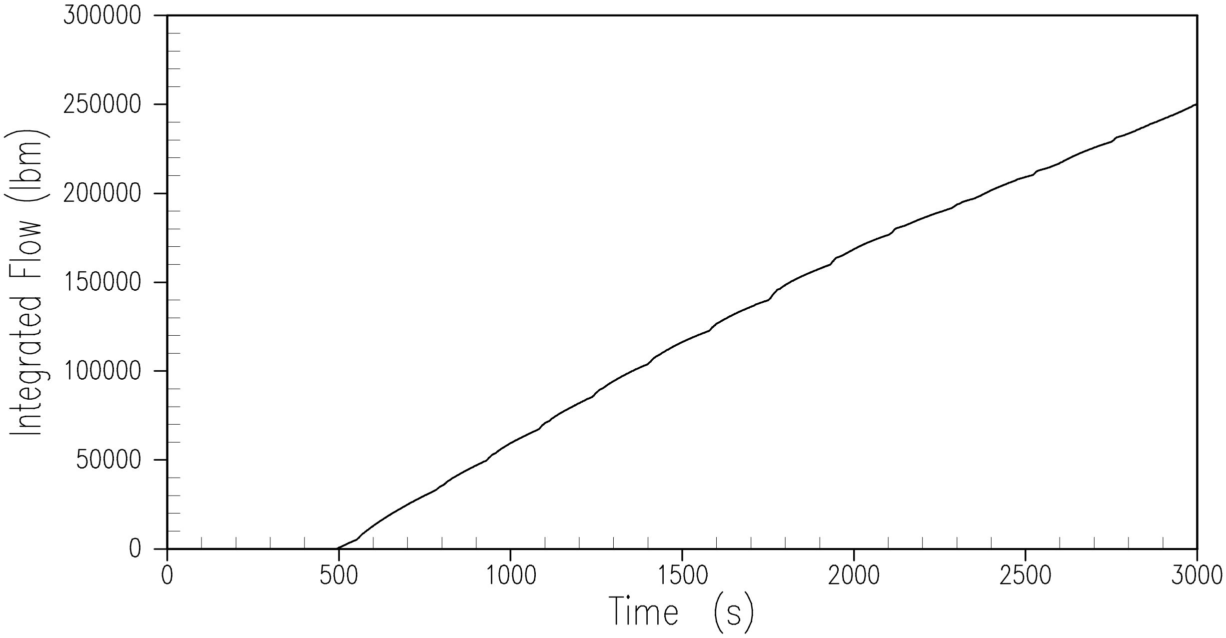

Figure 15.6.5.4B-49A

DEDVI – ADS-4 Integrated Discharge – 14.7 psi

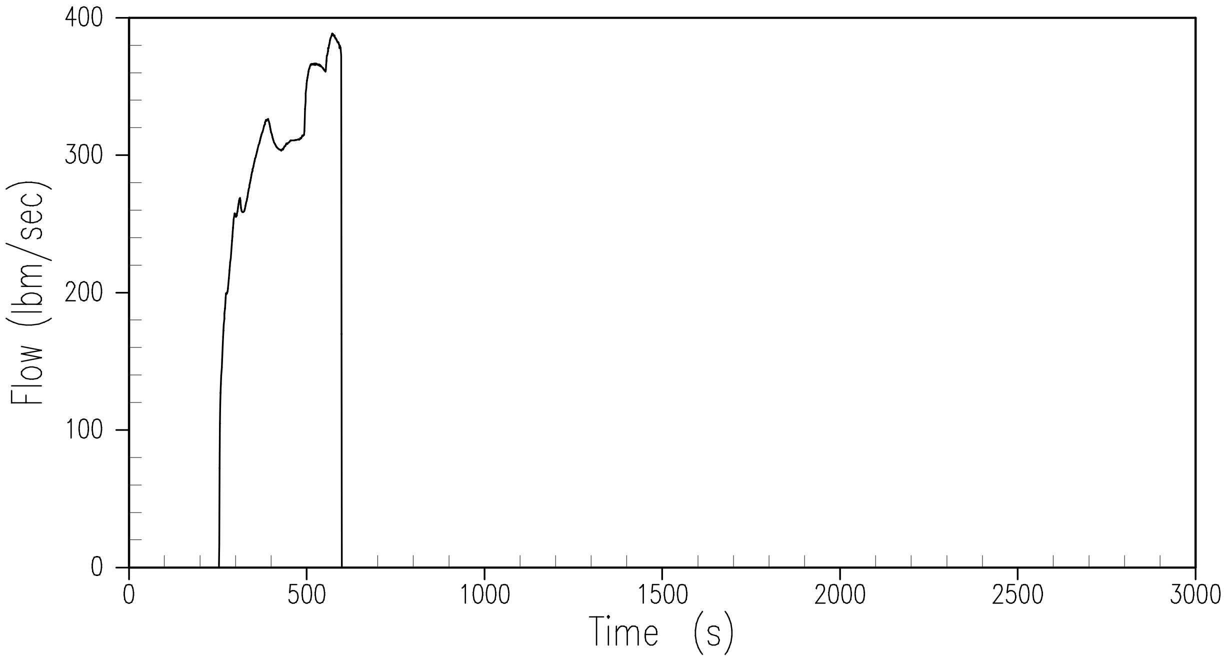

Figure 15.6.5.4B-50A

DEDVI – Intact Accumulator Flow Rate – 14.7 psi

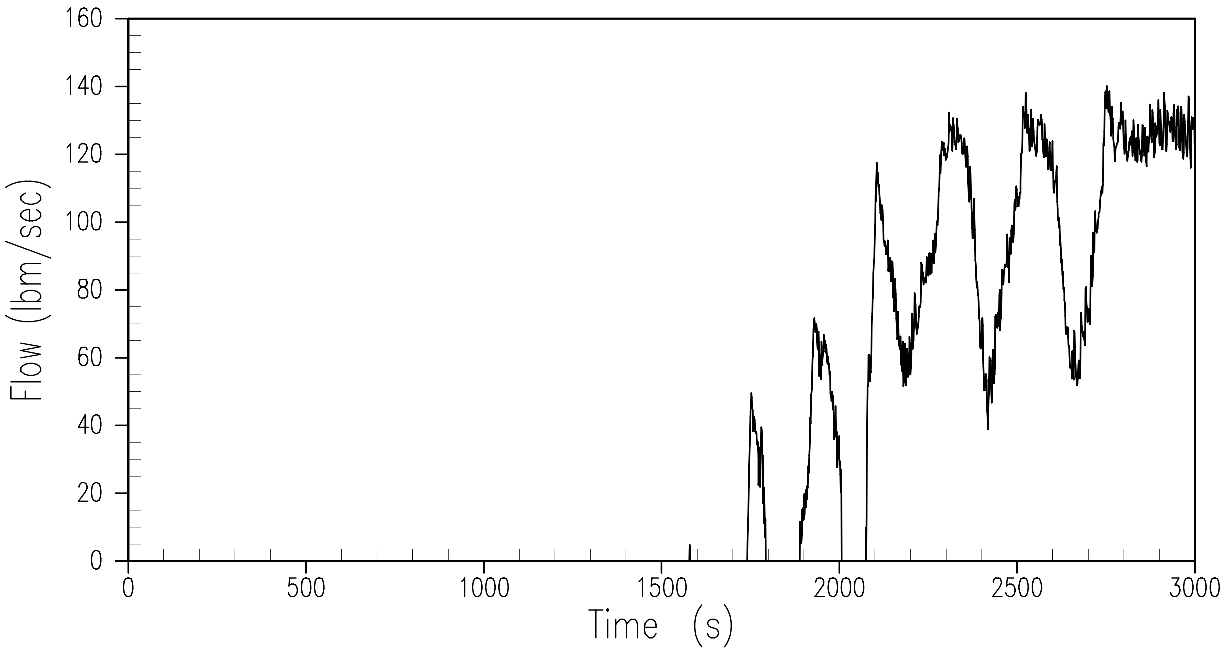

Figure 15.6.5.4B-51A

DEDVI – Intact IRWST Injection Rate – 14.7 psi

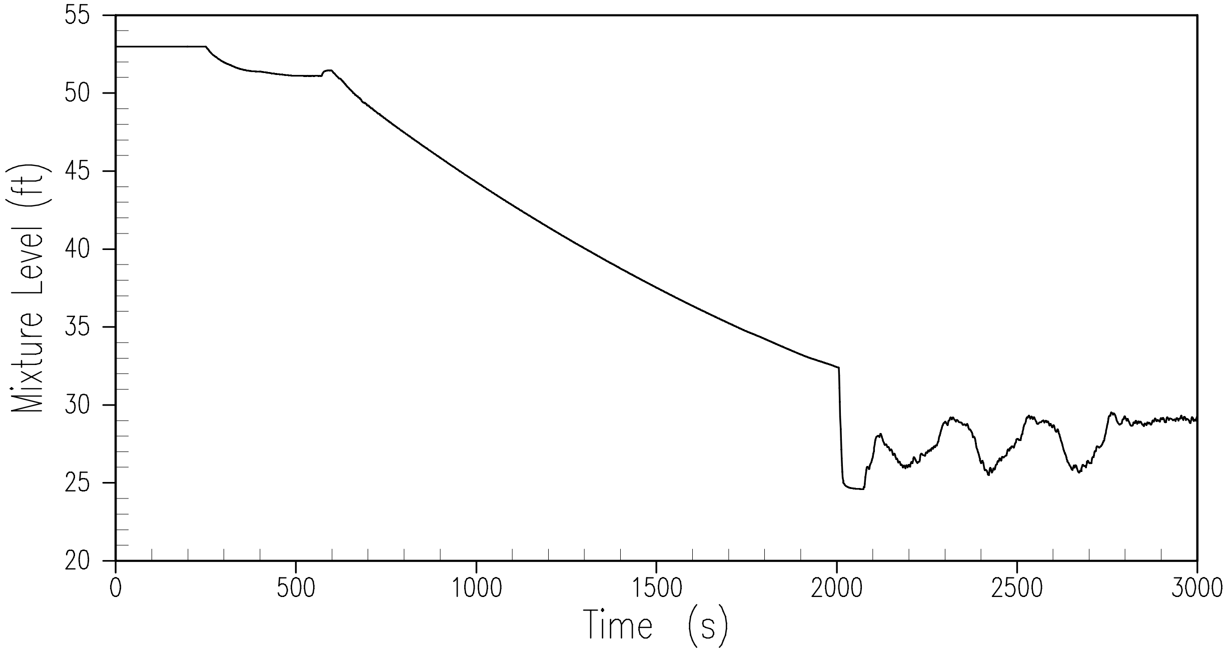

Figure 15.6.5.4B-52A

DEDVI – Intact CMT Mixture Level – 14.7 psi

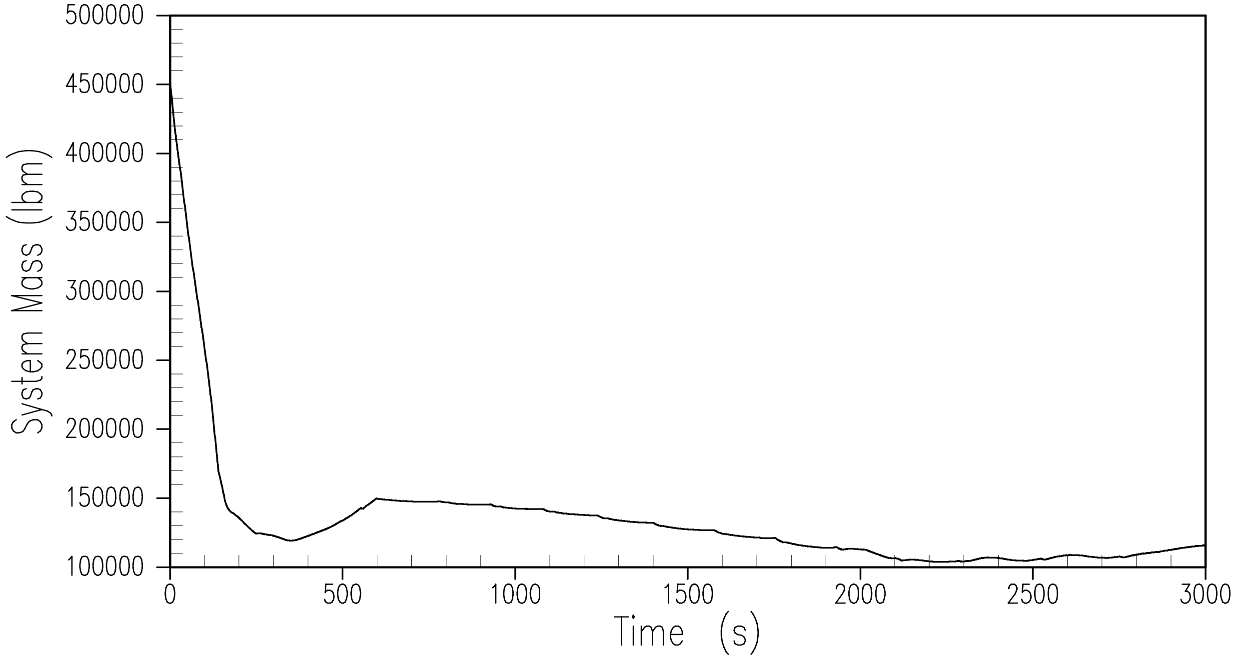

Figure 15.6.5.4B-53A

DEDVI – RCS System Inventory – 14.7 psi

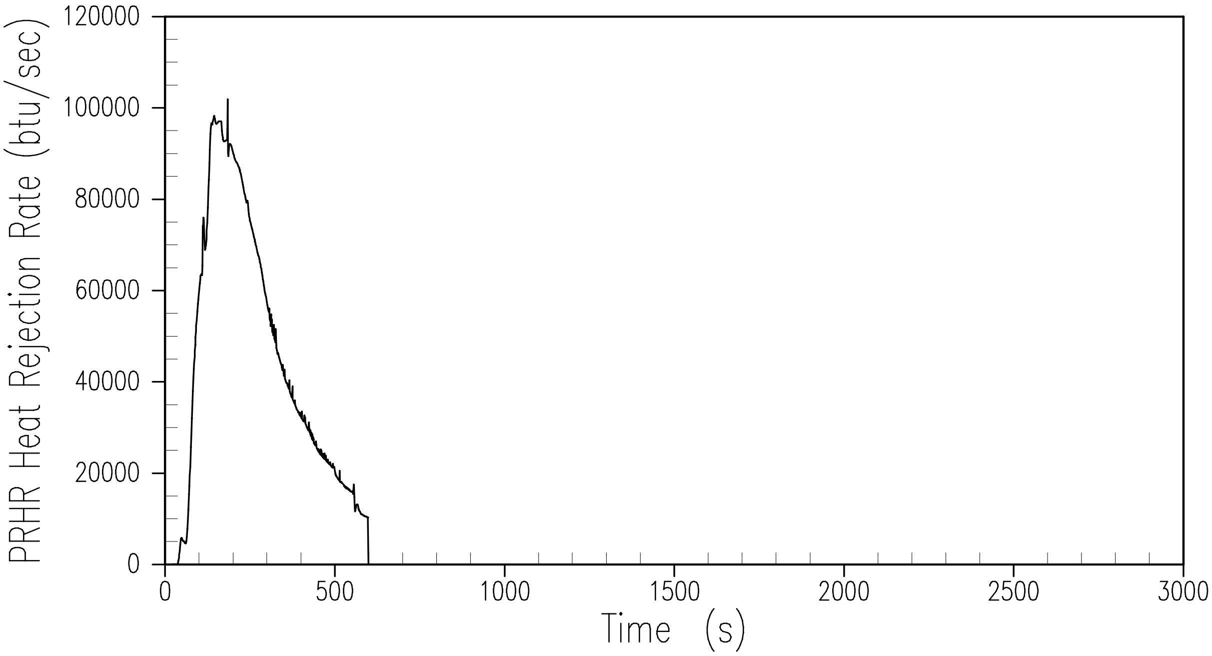

Figure 15.6.5.4B-54A

DEDVI – PRHR Heat Removal Rate – 14.7 psi

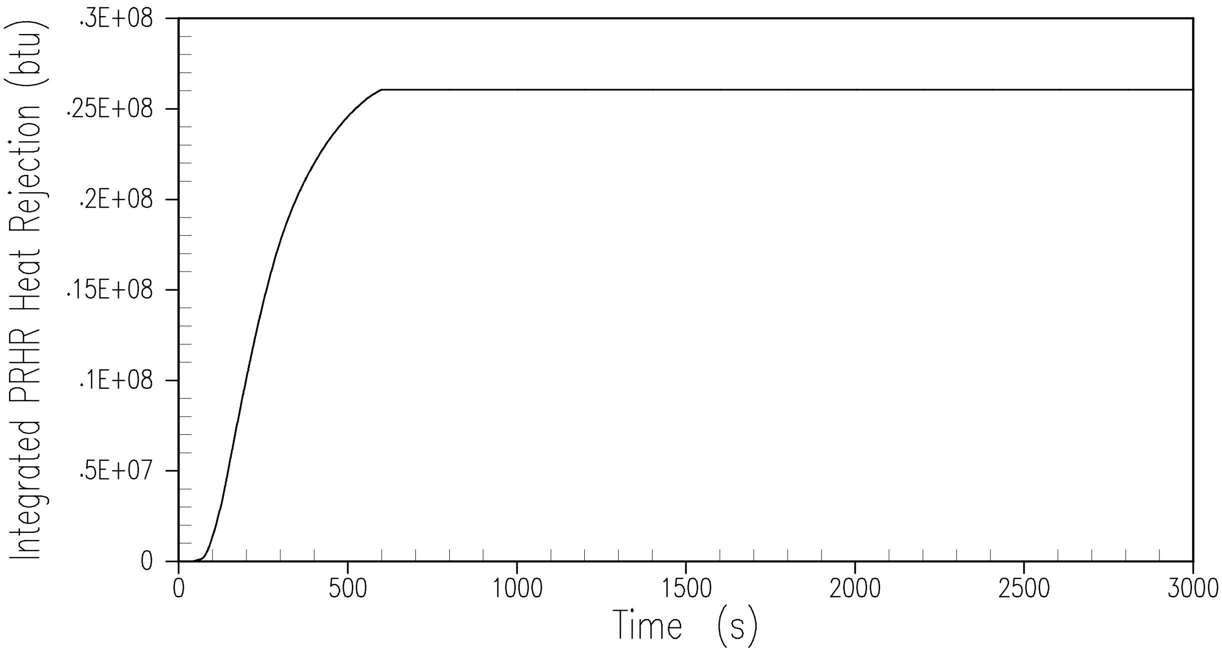

Figure 15.6.5.4B-55A

DEDVI – Integrated PRHR Heat Removal – 14.7 psi

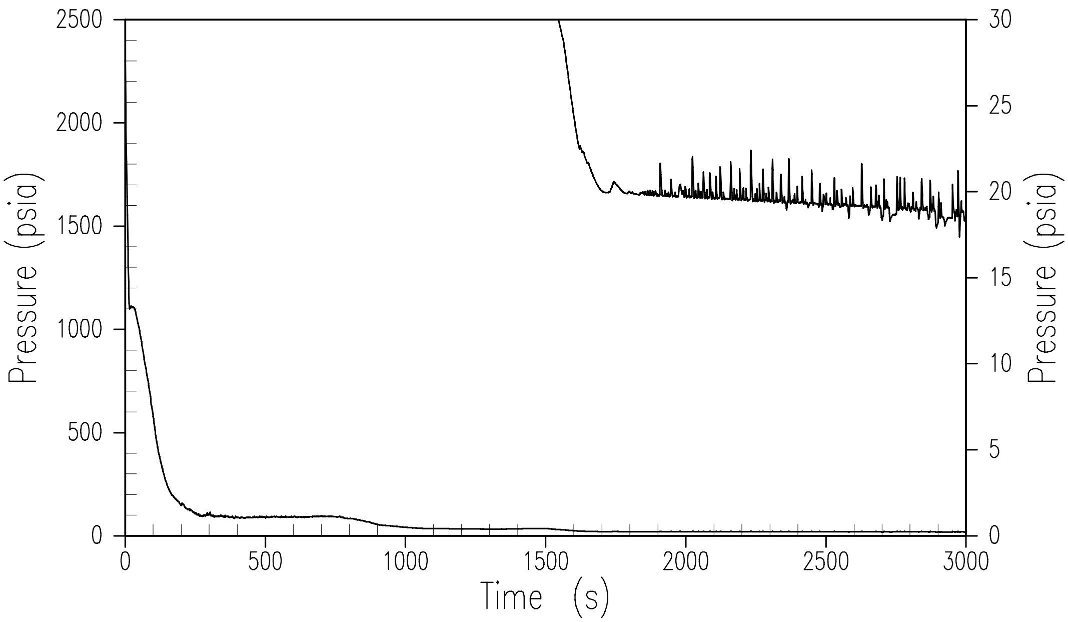

Figure 15.6.5.4B-56

10-Inch Cold Leg Break – RCS Pressure

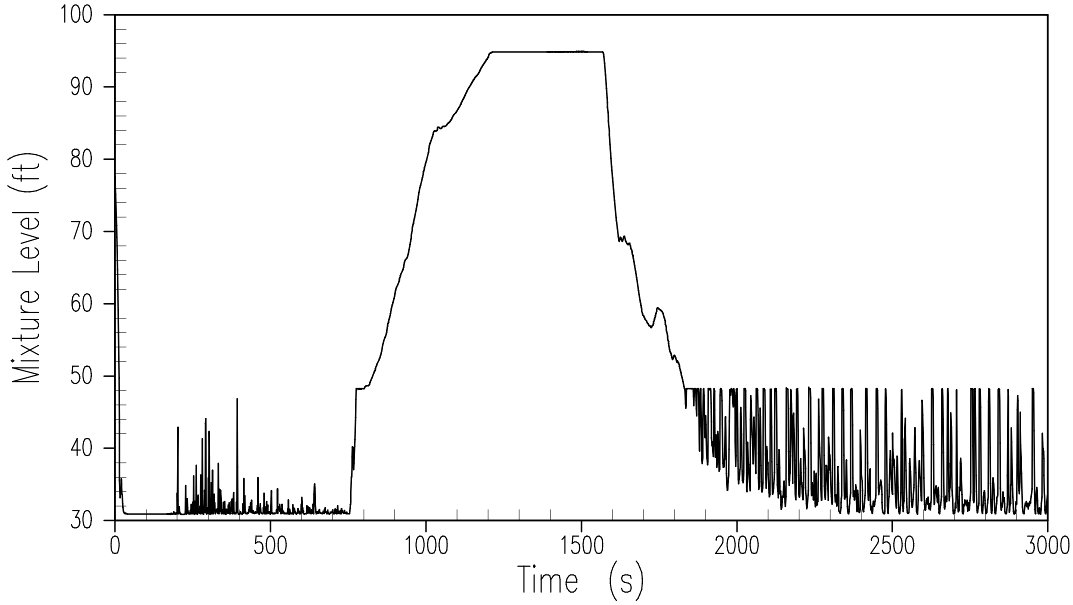

Figure 15.6.5.4B-57

10-Inch Cold Leg Break – Pressurizer Mixture Level

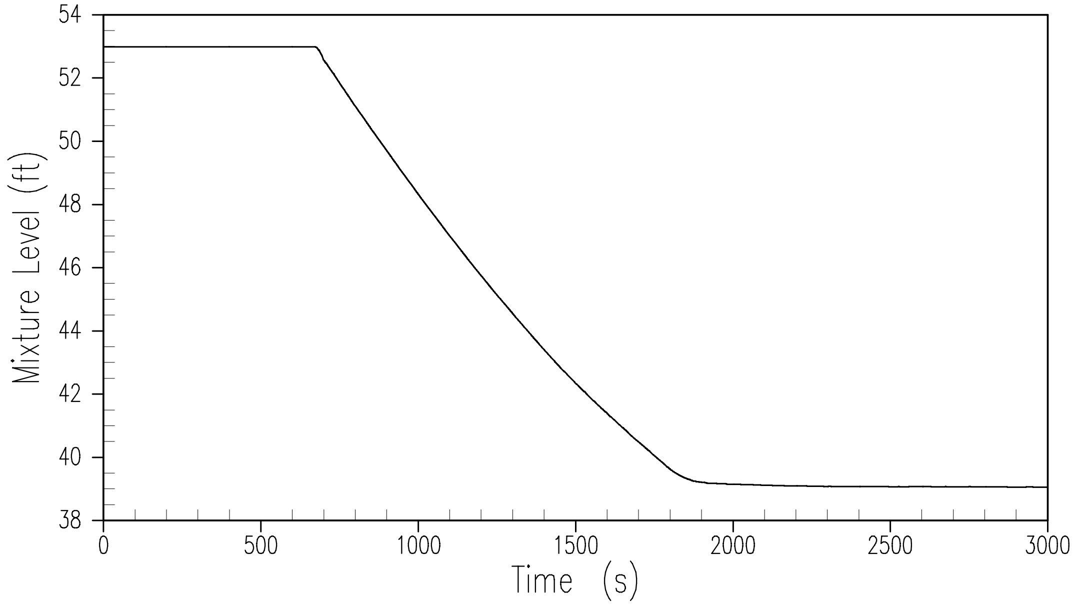

Figure 15.6.5.4B-58

10-Inch Cold Leg Break – CMT-1 Mixture Level

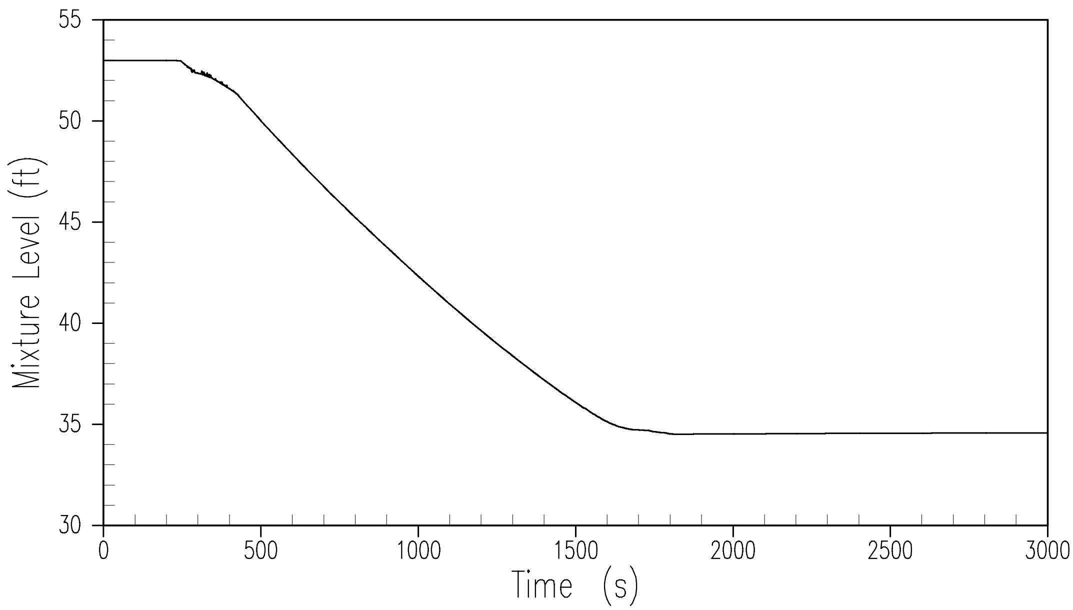

Figure 15.6.5.4B-59

10-Inch Cold Leg Break – CMT-2 Mixture Level

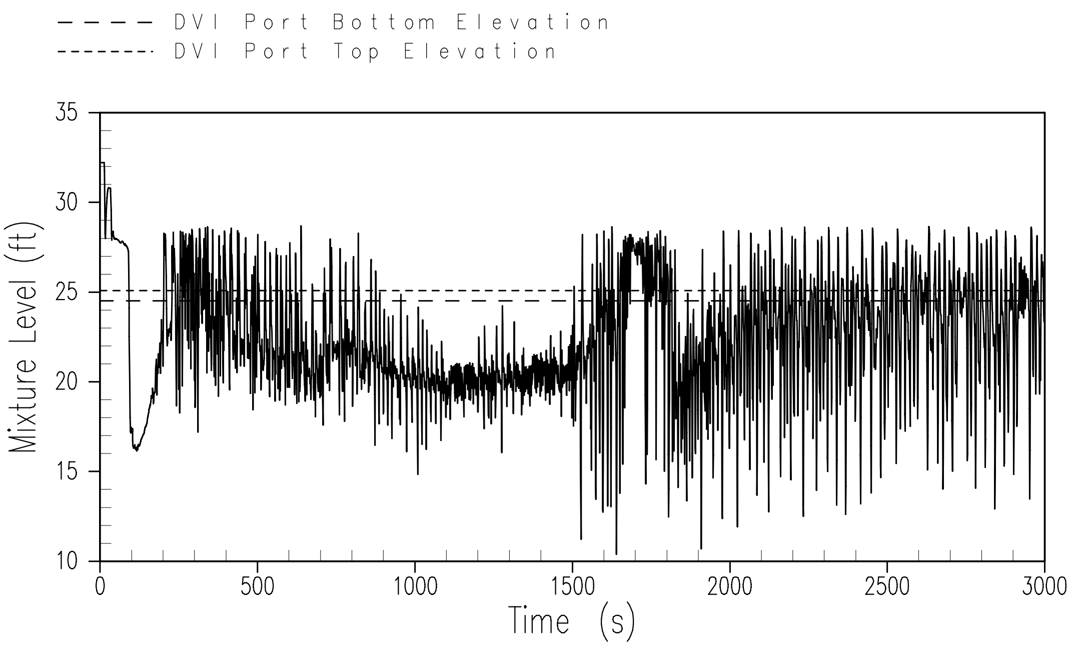

Figure 15.6.5.4B-60

10-Inch Cold Leg Break – Downcomer Mixture Level

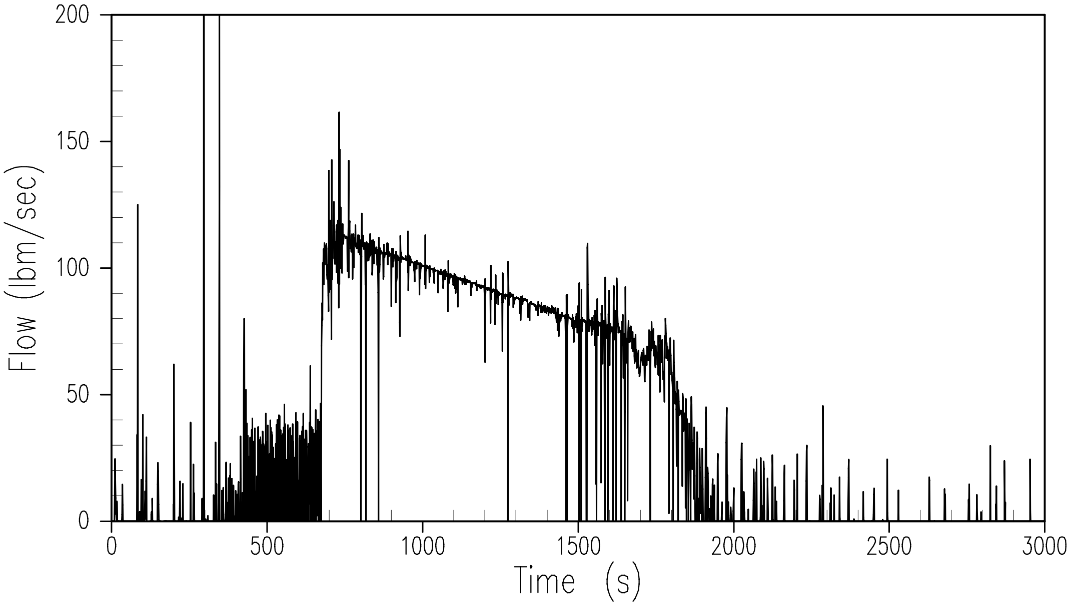

Figure 15.6.5.4B-61

10-Inch Cold Leg Break – CMT-1 Injection Rate

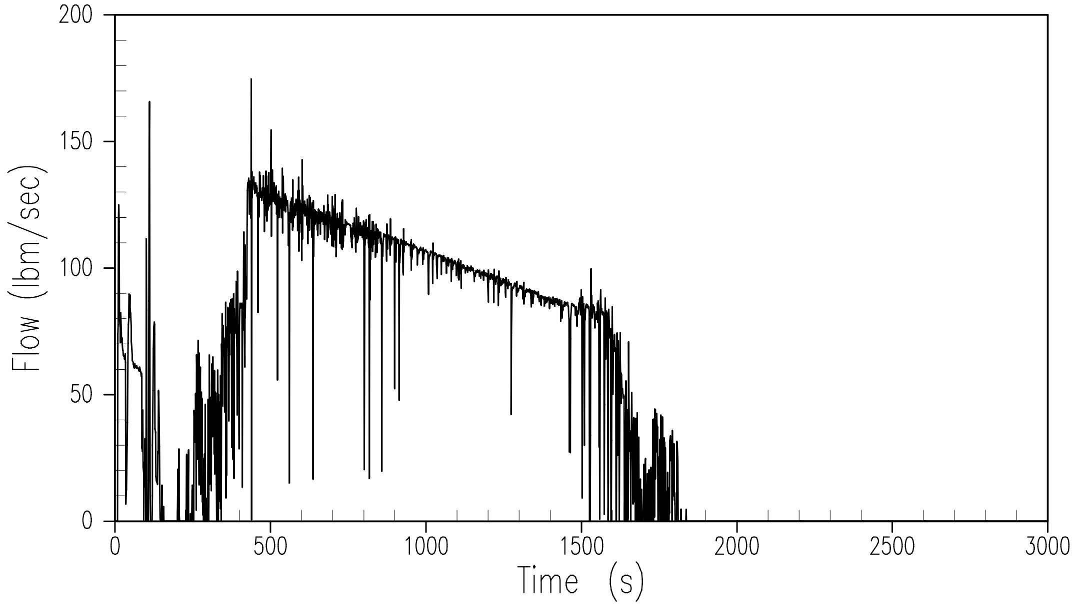

Figure 15.6.5.4B-62

10-Inch Cold Leg Break – CMT-2 Injection Rate

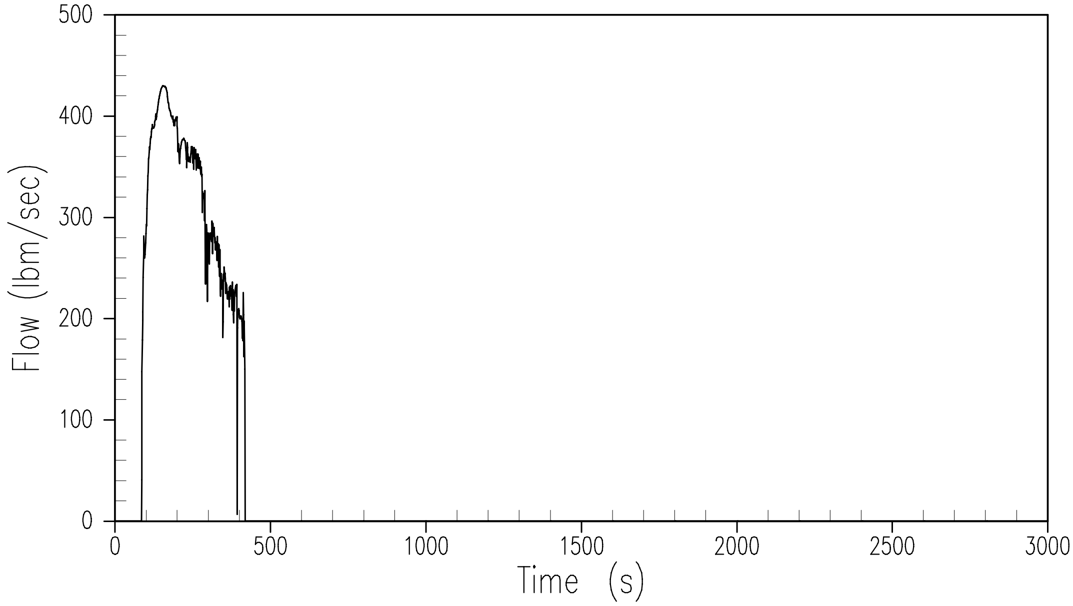

Figure 15.6.5.4B-63

10-Inch Cold Leg Break – Accumulator-1 Injection Rate

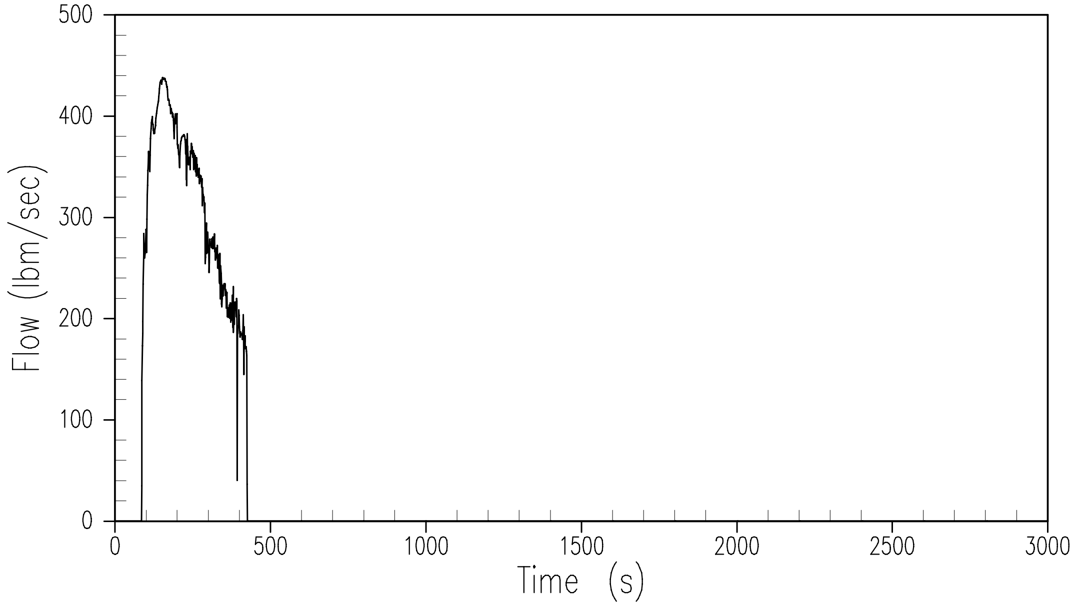

Figure 15.6.5.4B-64

10-Inch Cold Leg Break – Accumulator-2 Injection Rate

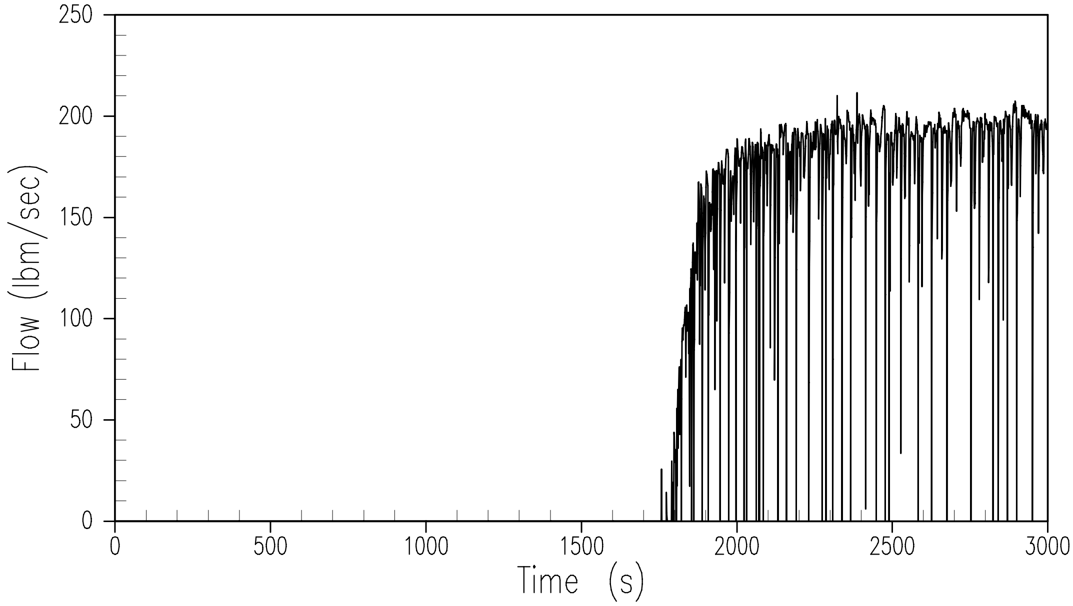

Figure 15.6.5.4B-65

10-Inch Cold Leg Break – IRWST-1 Injection Rate

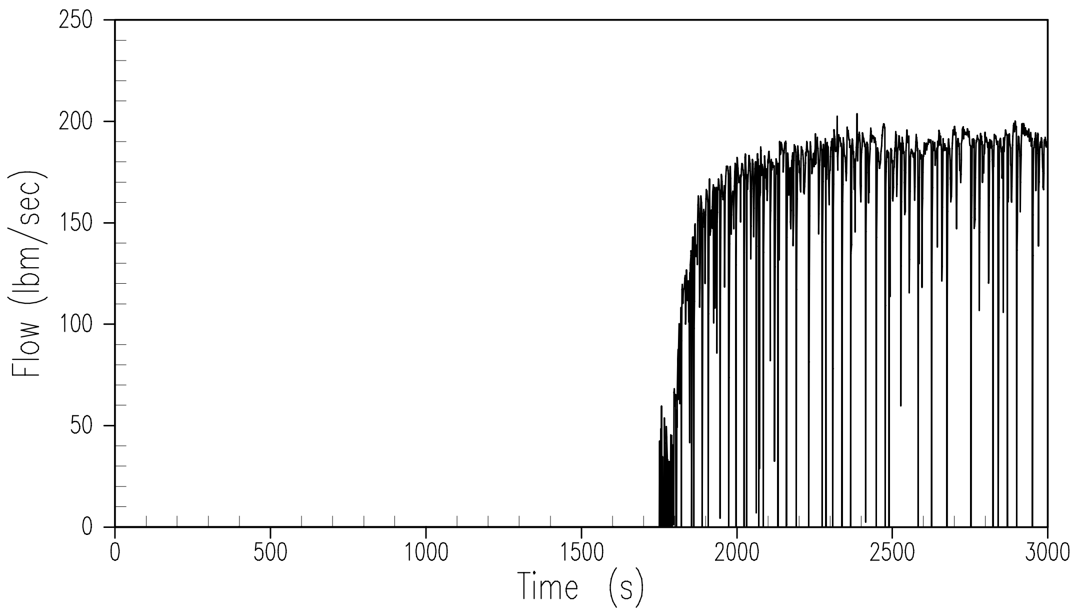

Figure 15.6.5.4B-66

10-Inch Cold Leg Break – IRWST-2 Injection Rate

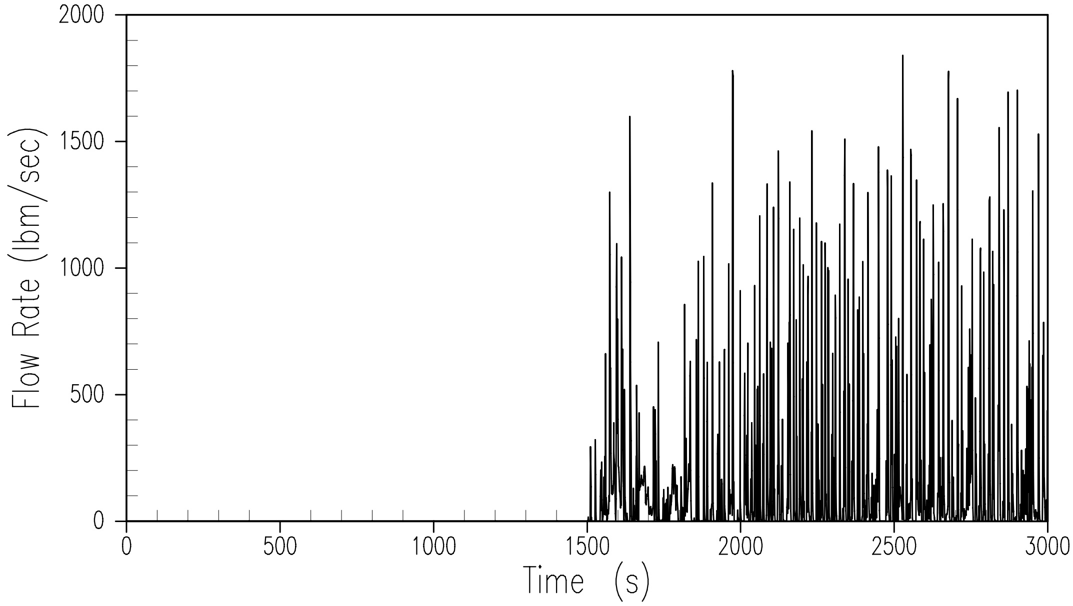

Figure 15.6.5.4B-67

10-Inch Cold Leg Break – ADS-4 Liquid Discharge

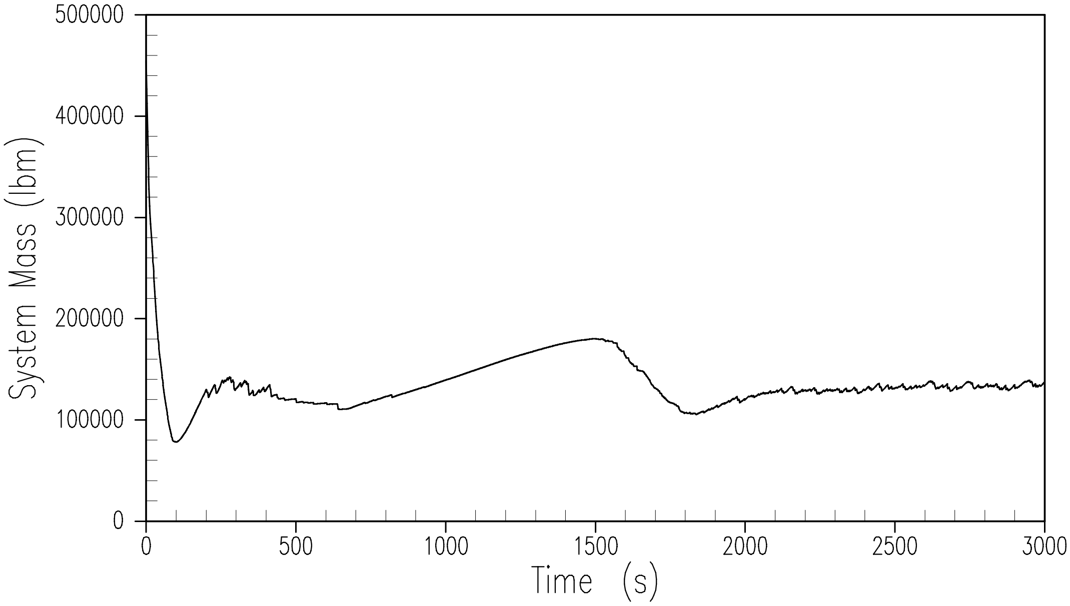

Figure 15.6.5.4B-68

10-Inch Cold Leg Break – RCS System Inventory

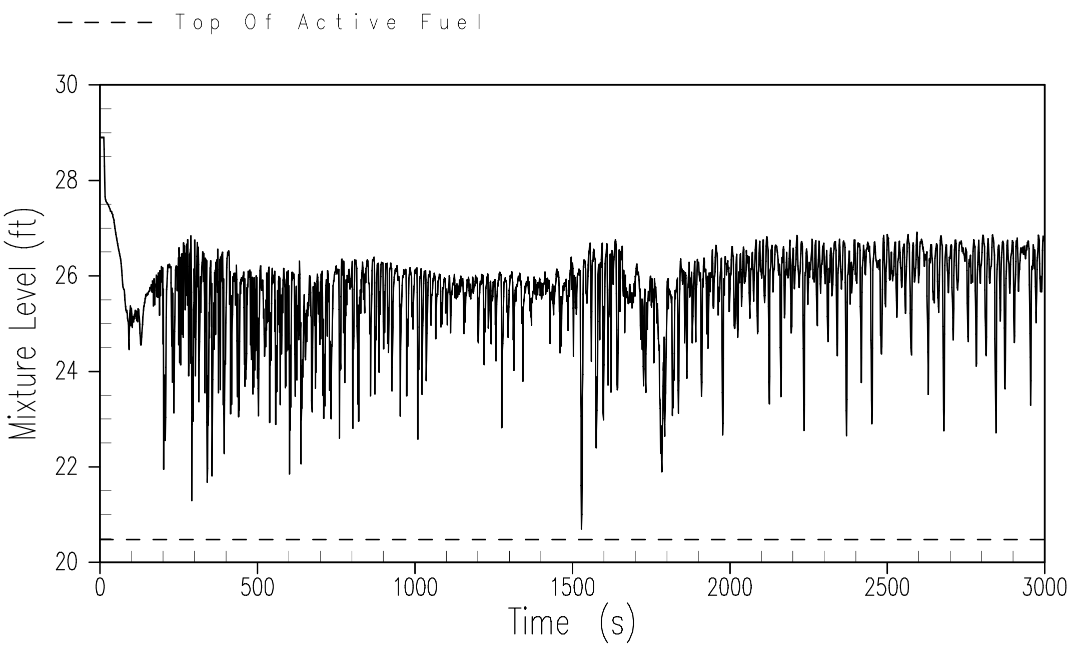

Figure 15.6.5.4B-69

10-Inch Cold Leg Break – Core/Upper Plenum Mixture Level

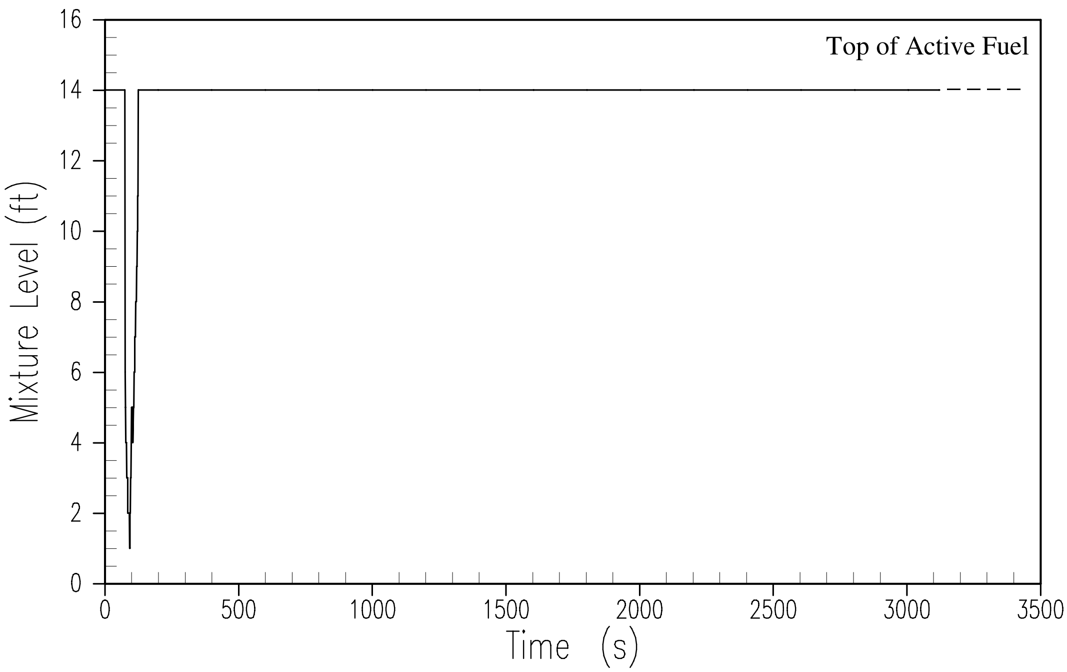

Figure 15.6.5.4B-70

10-Inch Cold Leg Break – Composite Core Mixture Level

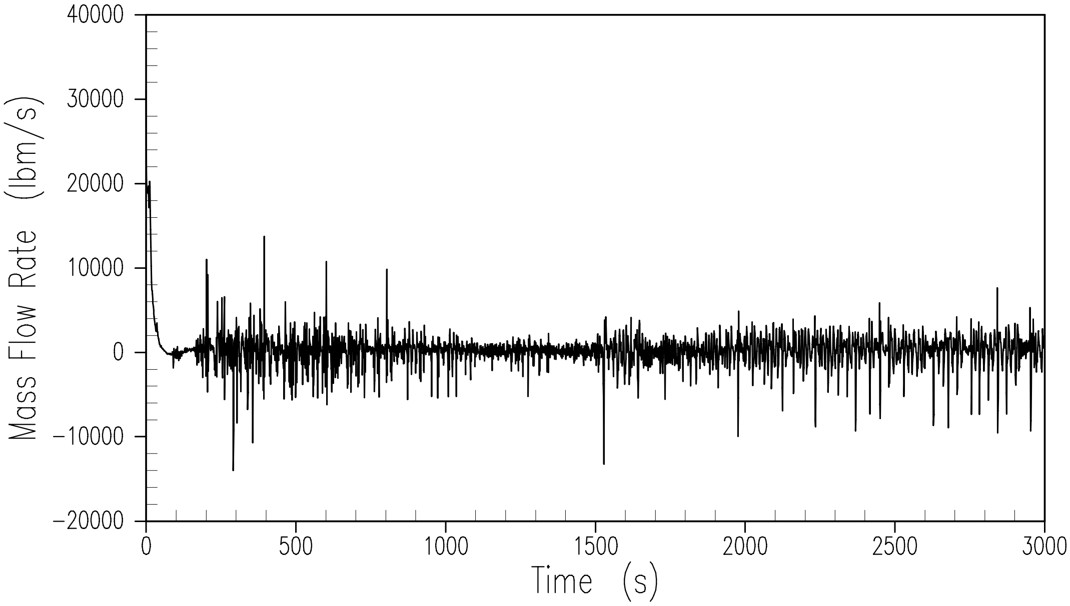

Figure 15.6.5.4B-71

10-Inch Cold Leg Break – Core Exit Liquid Flow

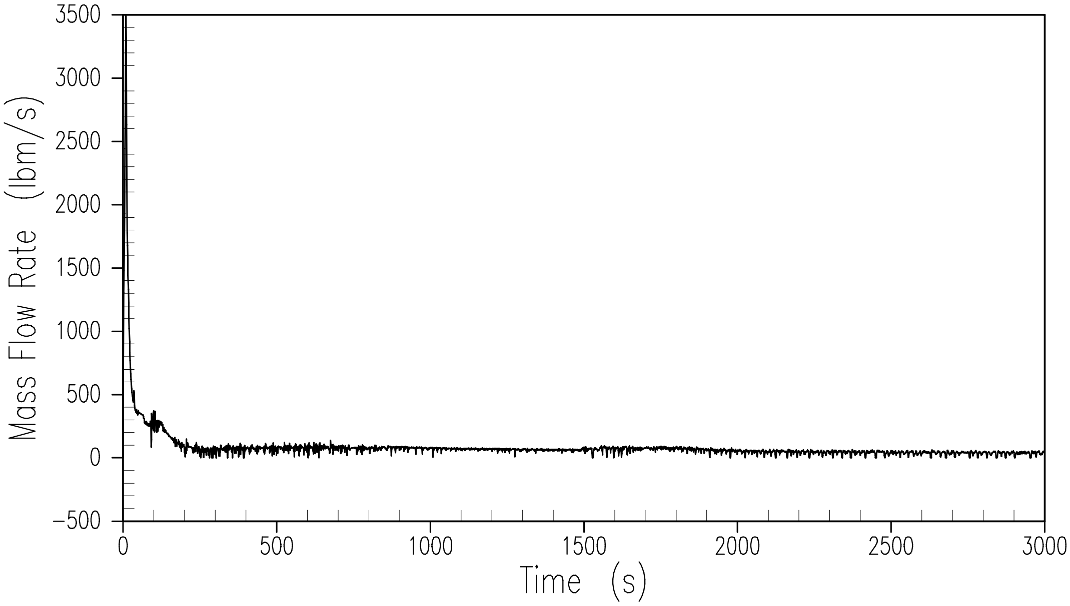

Figure 15.6.5.4B-72

10-Inch Cold Leg Break – Core Exit Vapor Flow

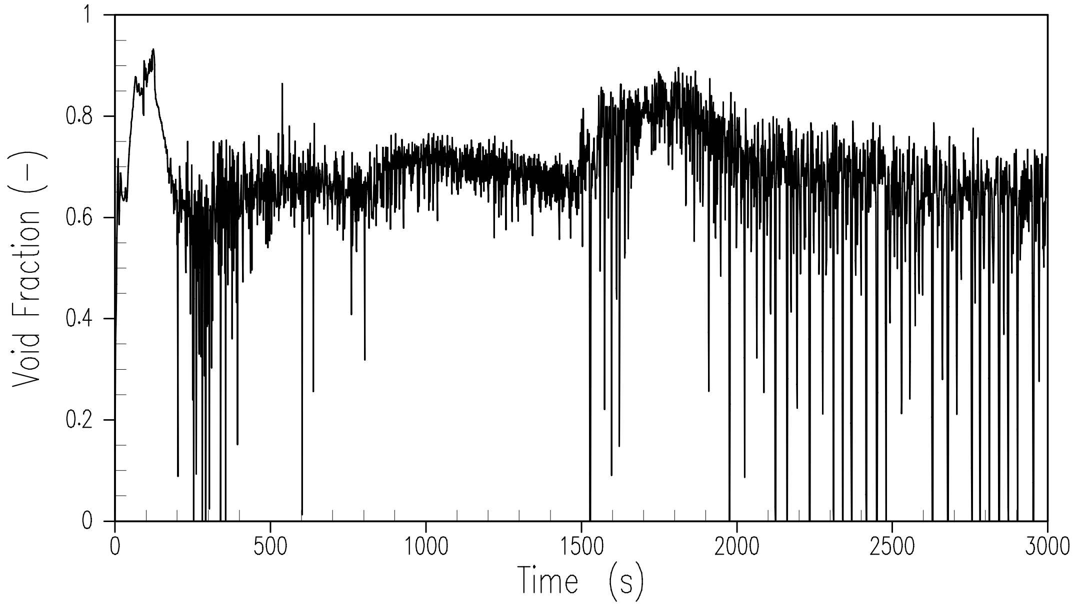

Figure 15.6.5.4B-73

10-Inch Cold Leg Break – Core Exit Void Fraction

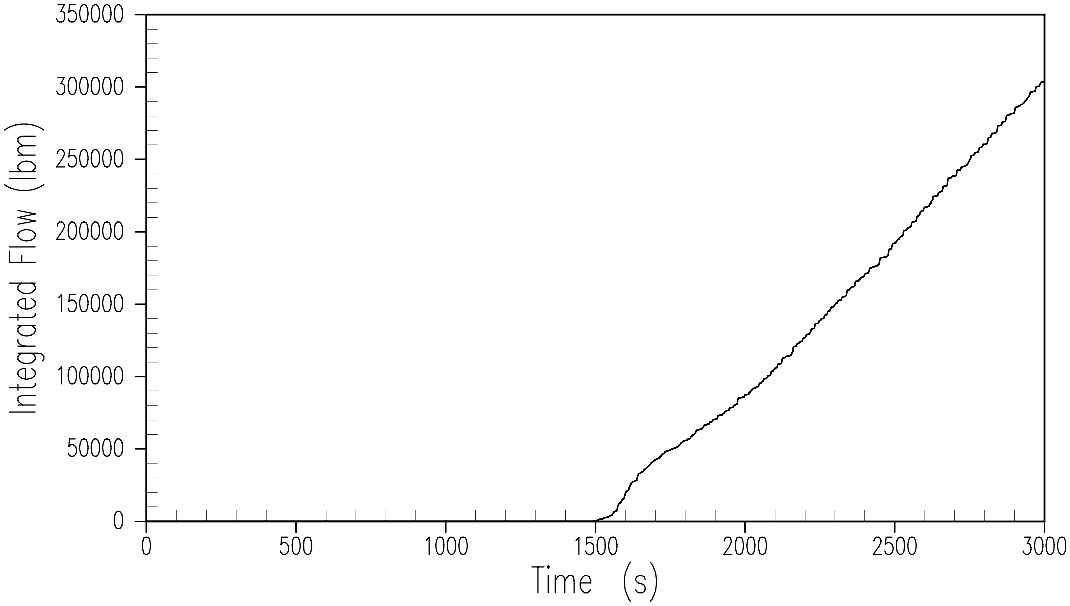

Figure 15.6.5.4B-74

10-Inch Cold Leg Break – ADS-4 Integrated Discharge

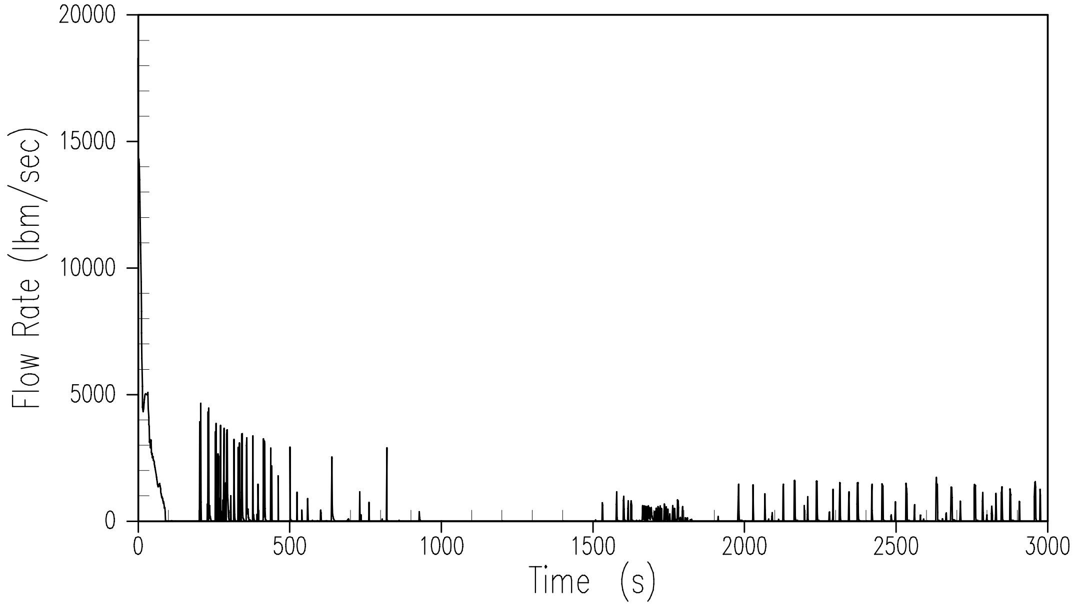

Figure 15.6.5.4B-75

10-Inch Cold Leg Break – Liquid Break Discharge

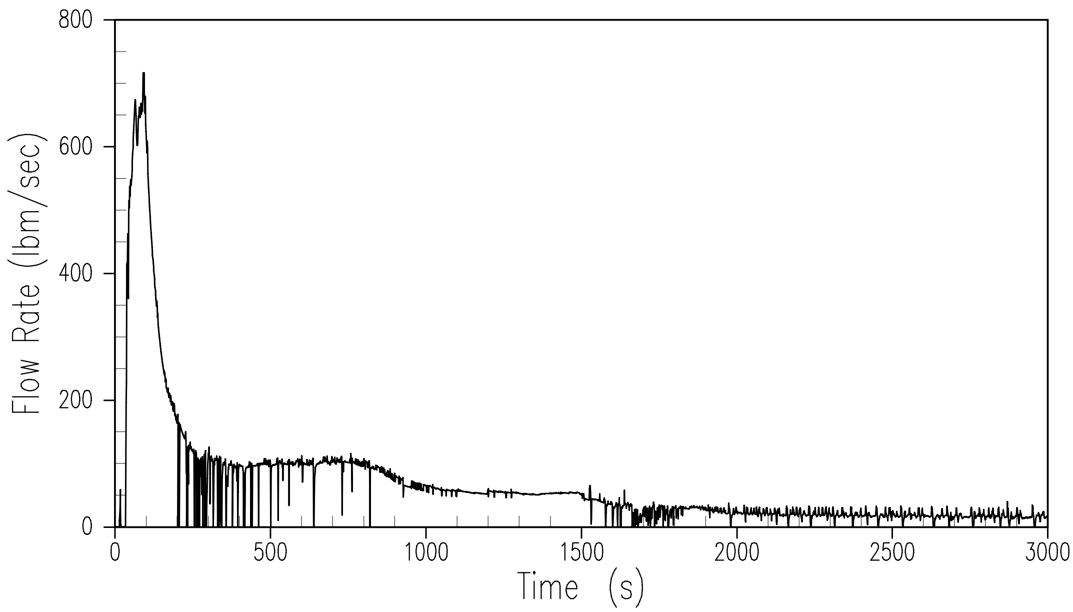

Figure 15.6.5.4B-76

10-Inch Cold Leg Break – Vapor Break Discharge

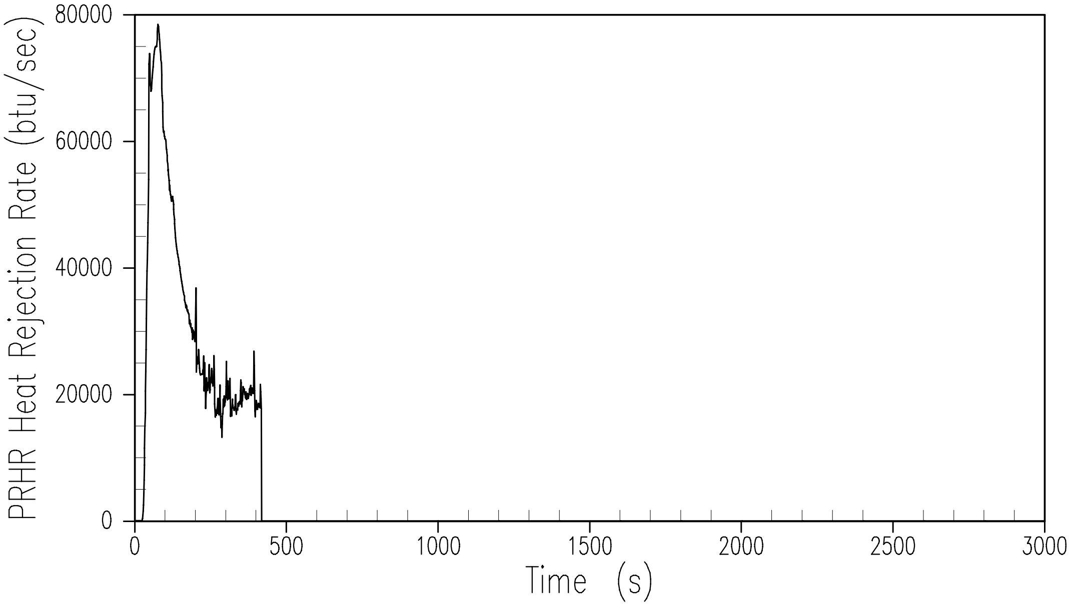

Figure 15.6.5.4B-77

10-Inch Cold Leg Break – PRHR Heat Removal Rate

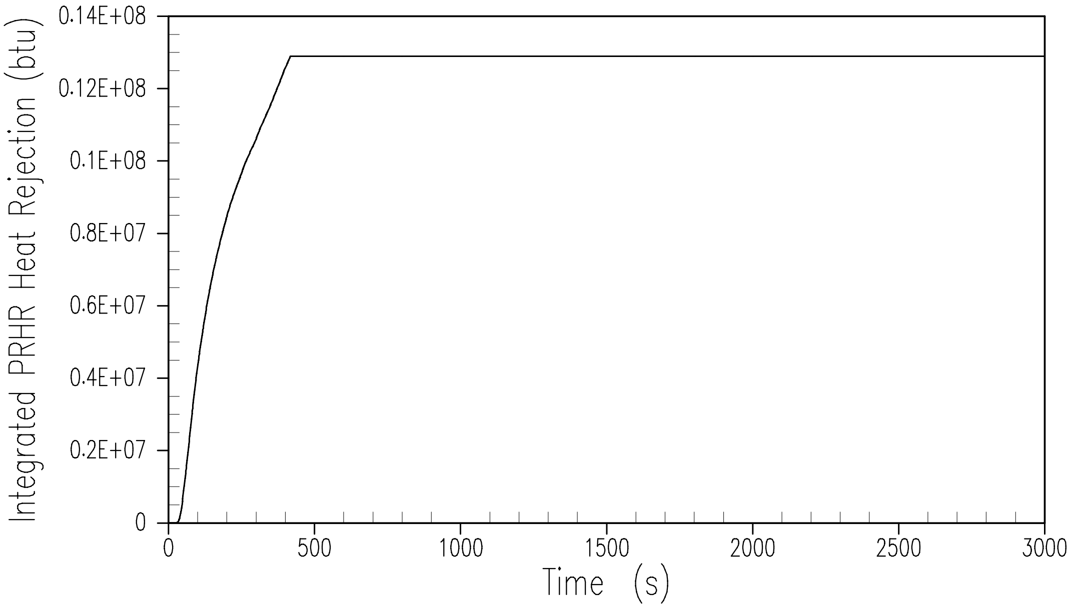

Figure 15.6.5.4B-78

10-Inch Cold Leg Break – Integrated PRHR Heat Removal

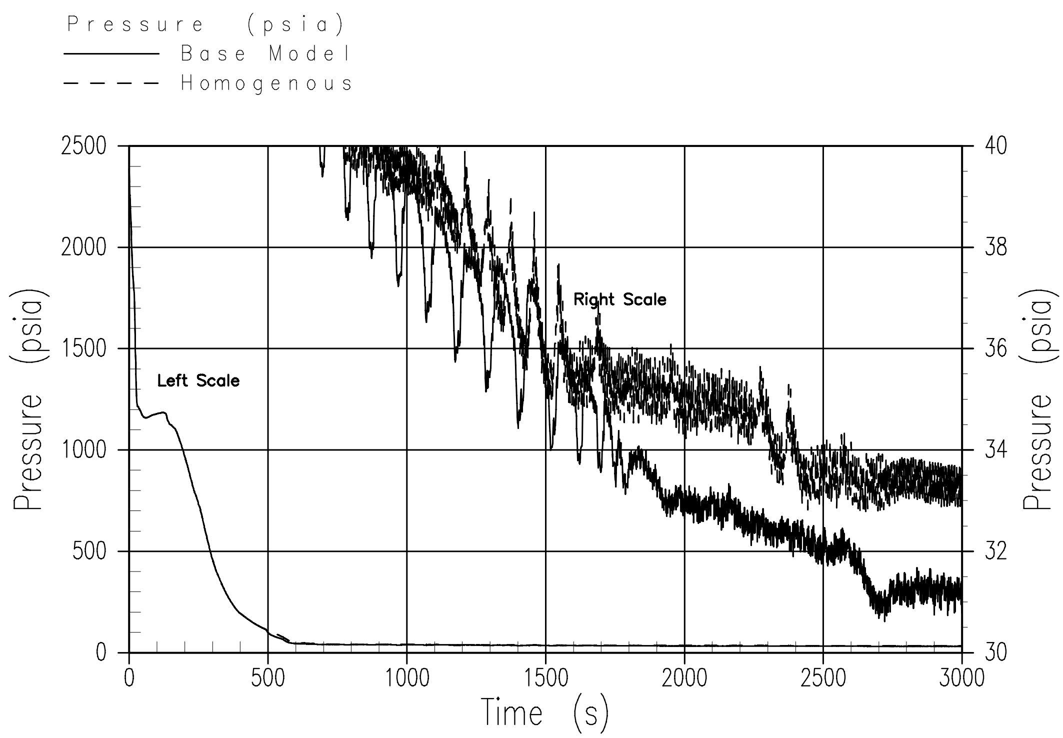

Figure 15.6.5.4B-79

DEDVI – Downcomer Pressure Comparison

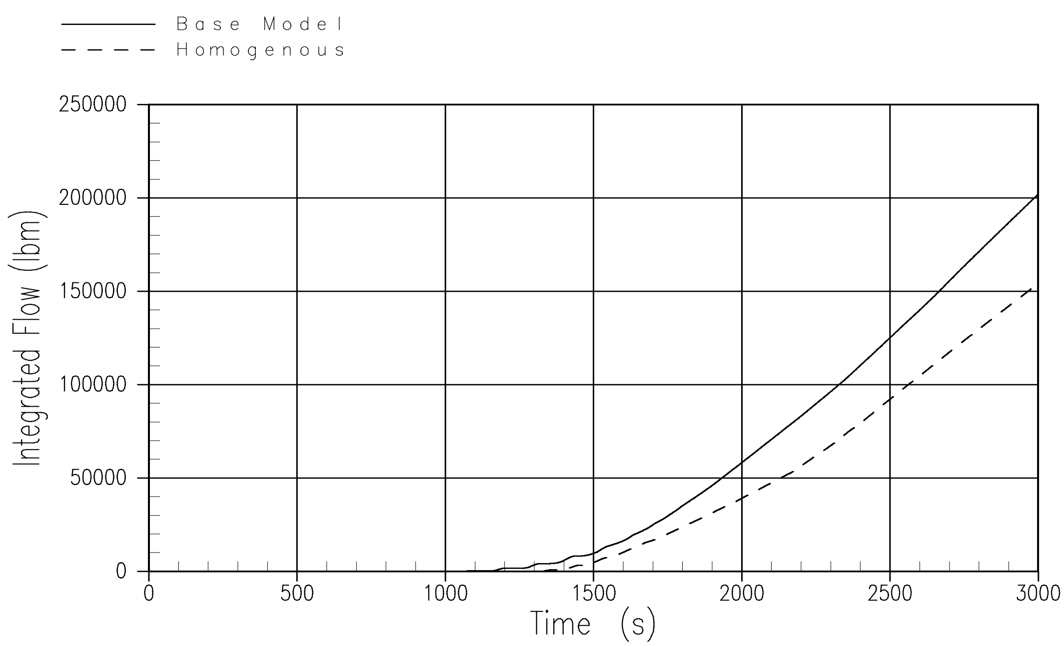

Figure 15.6.5.4B-80

DEDVI – Intact IRWST Injection Flow

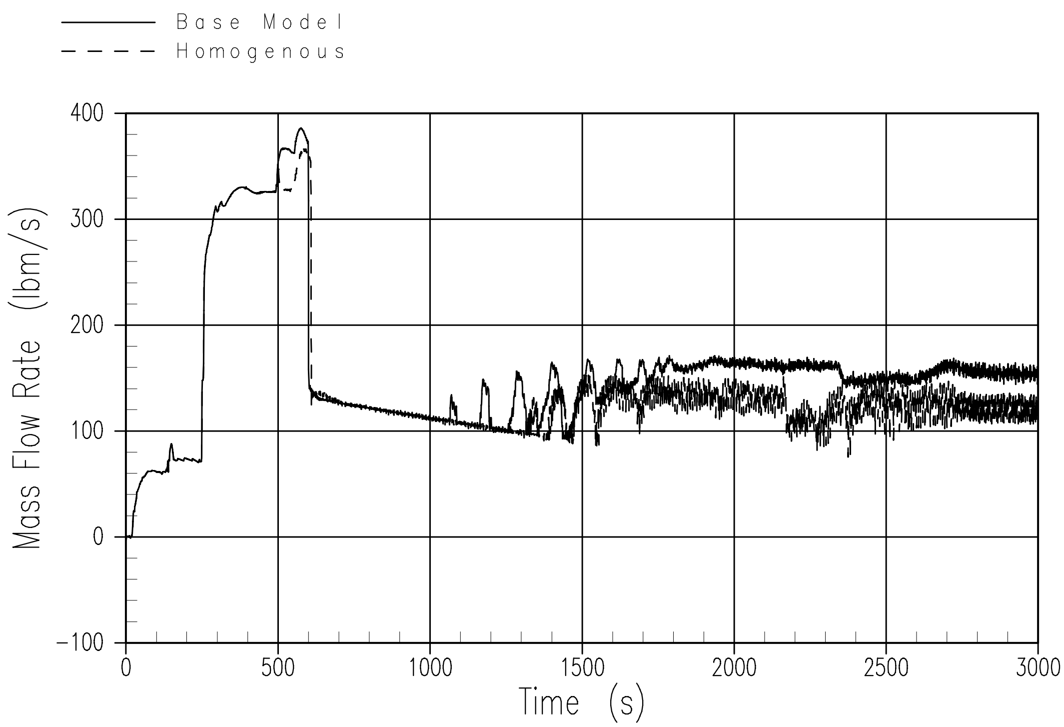

Figure 15.6.5.4B-81

DEDVI – Intact DVI Line Injection Flow

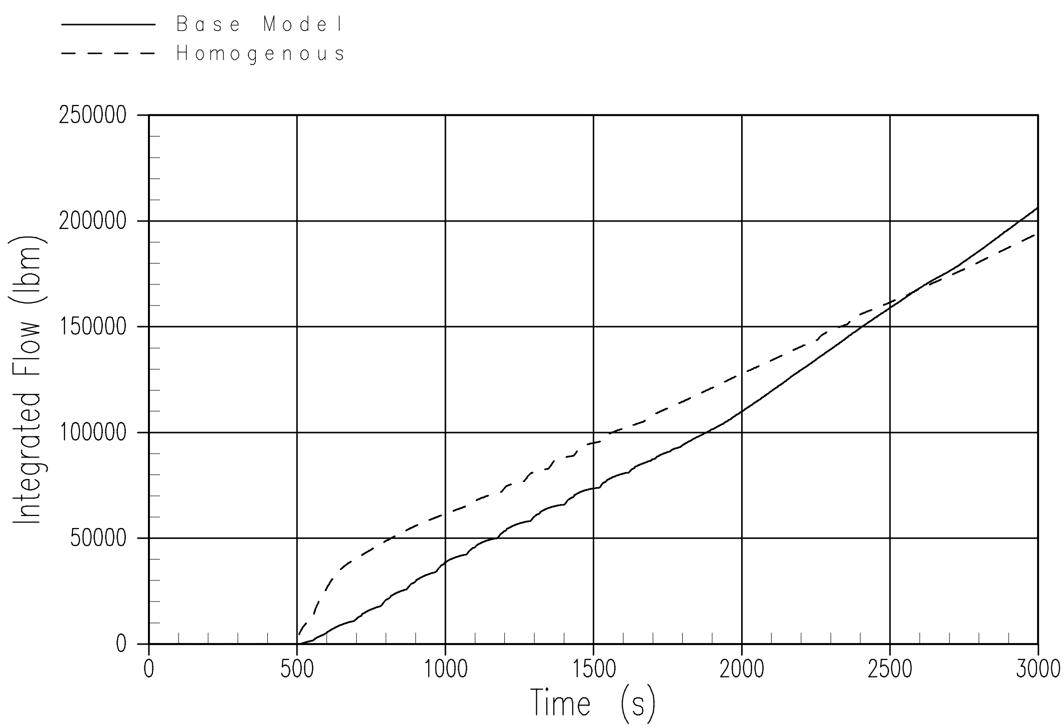

Figure 15.6.5.4B-82

DEDVI – ADS-4 Integrated Liquid Discharge Comparison

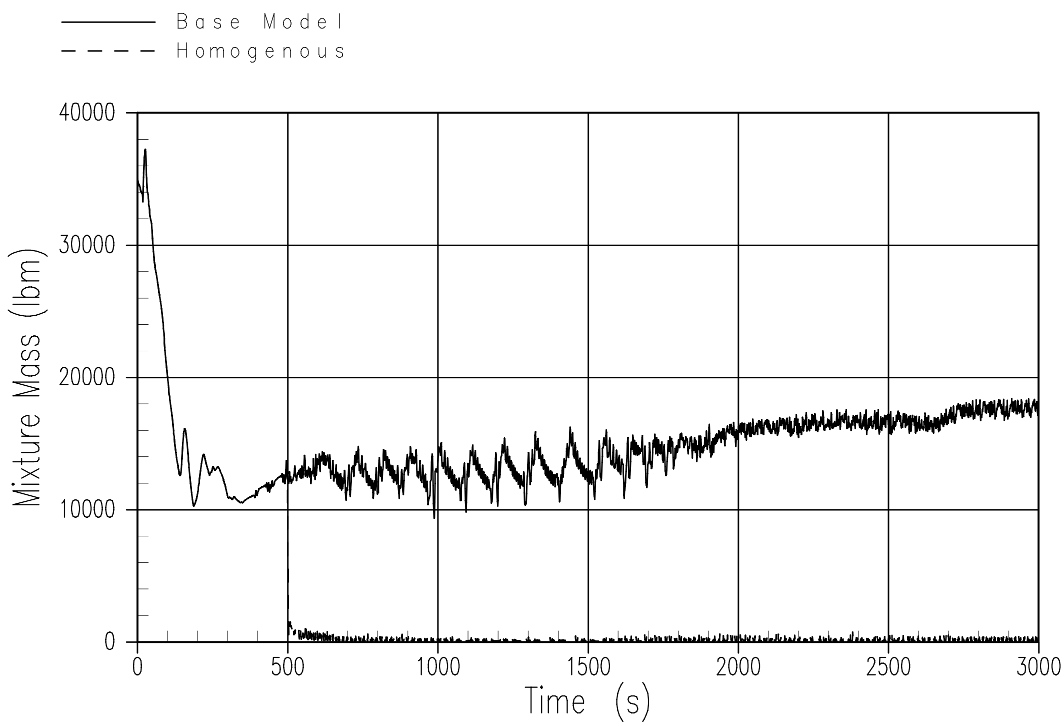

Figure 15.6.5.4B-83

DEDVI – Upper Plenum Mixture Mass Comparison

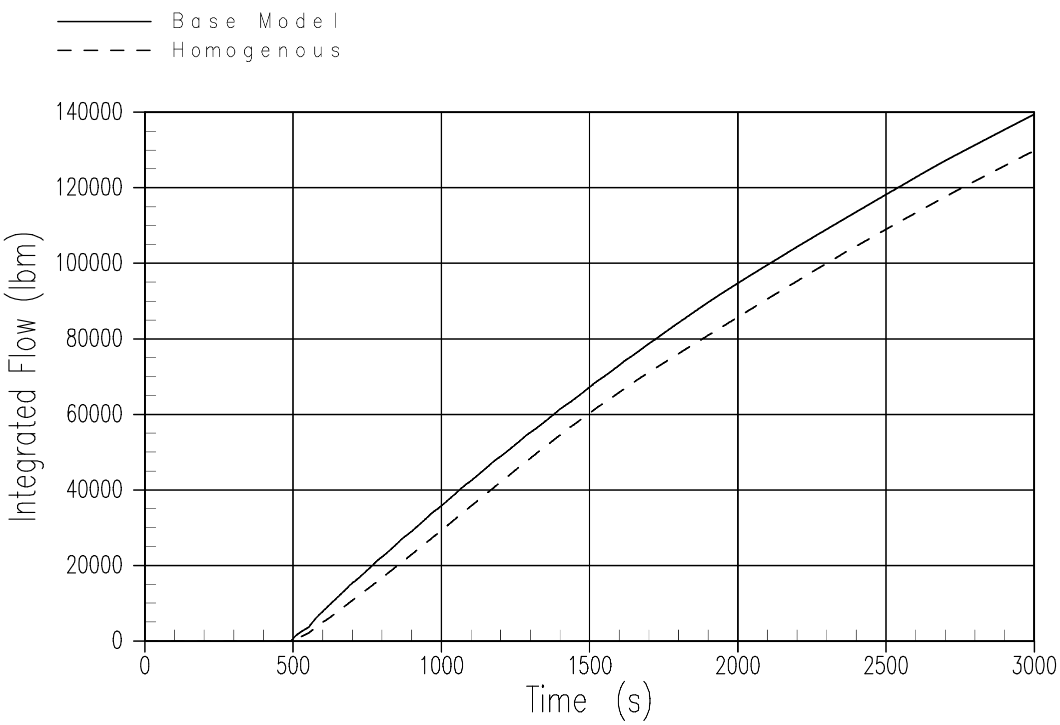

Figure 15.6.5.4B-84

DEDVI – ADS-4 Integrated Vapor Discharge Comparison

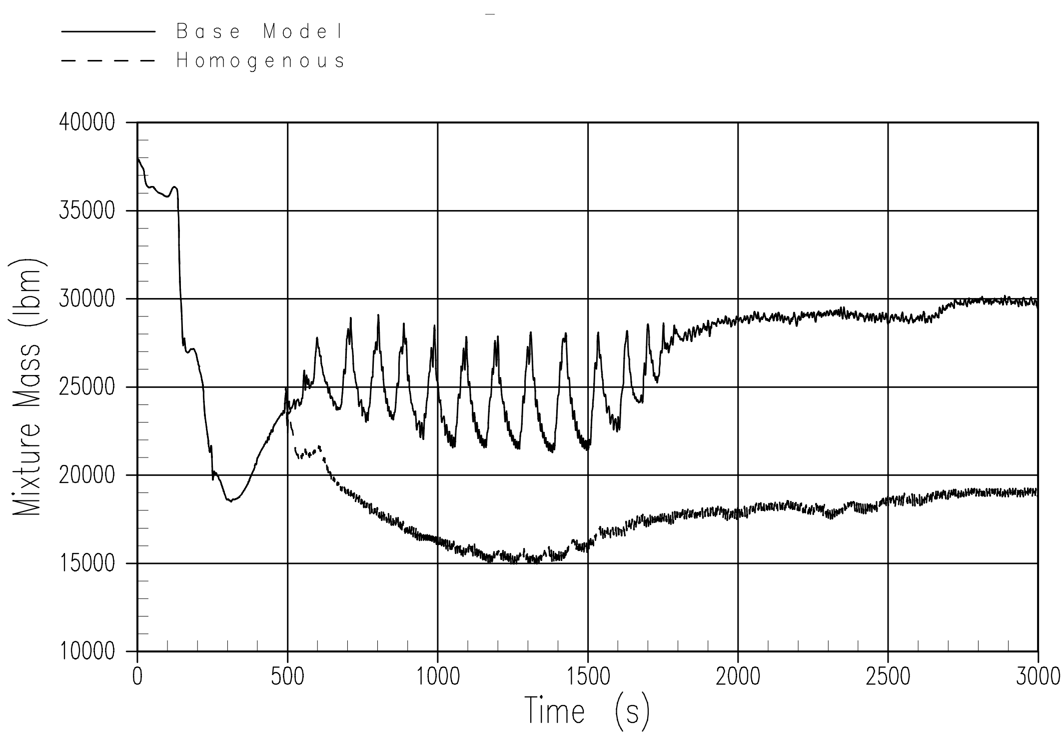

Figure 15.6.5.4B-85

DEDVI – Downcomer Region Mass Comparison

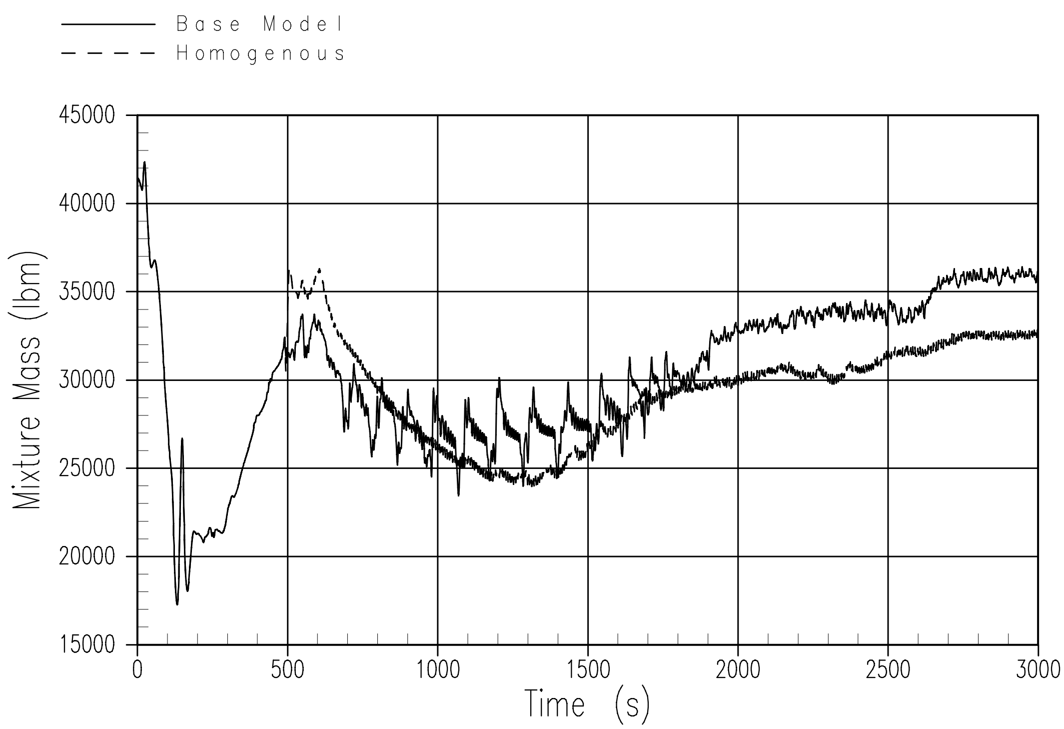

Figure 15.6.5.4B-86

DEDVI – Core Region Mass Comparison

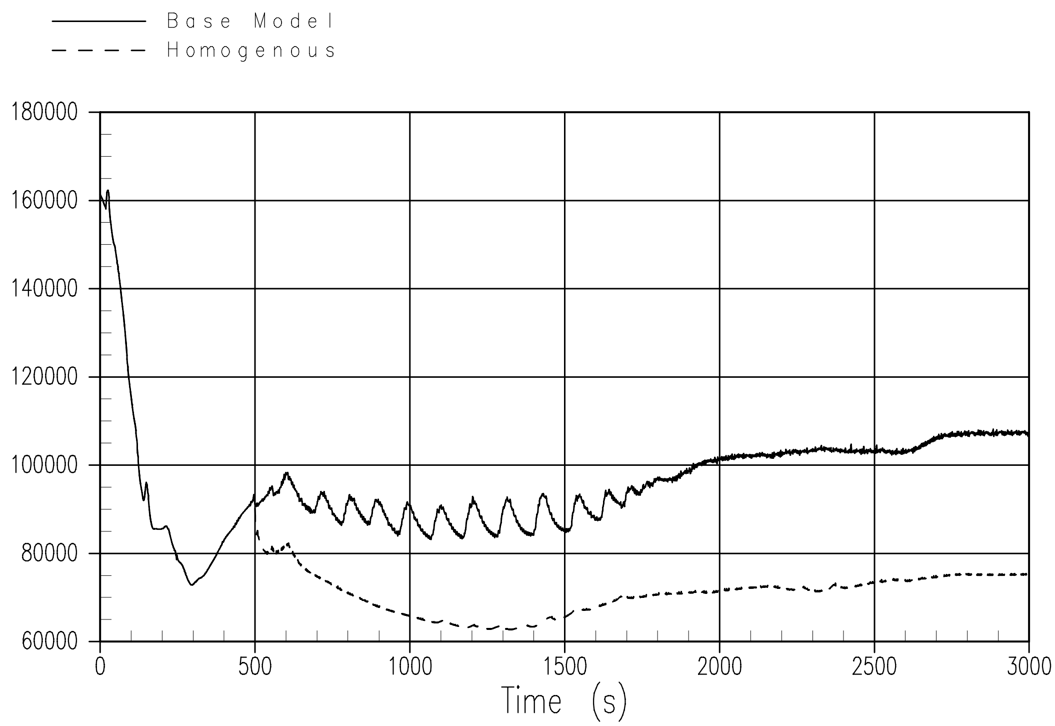

Figure 15.6.5.4B-87

DEDVI – Vessel Mixture Mass Comparison

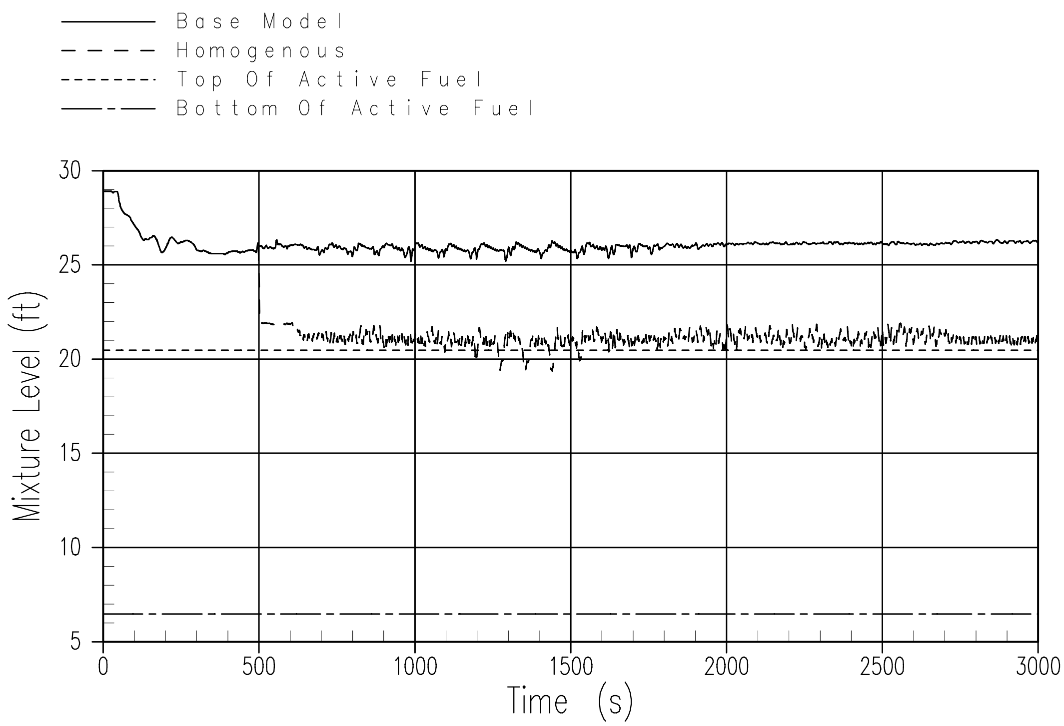

Figure 15.6.5.4B-88

DEDVI – Core/Upper Plenum Mixture Level Comparison

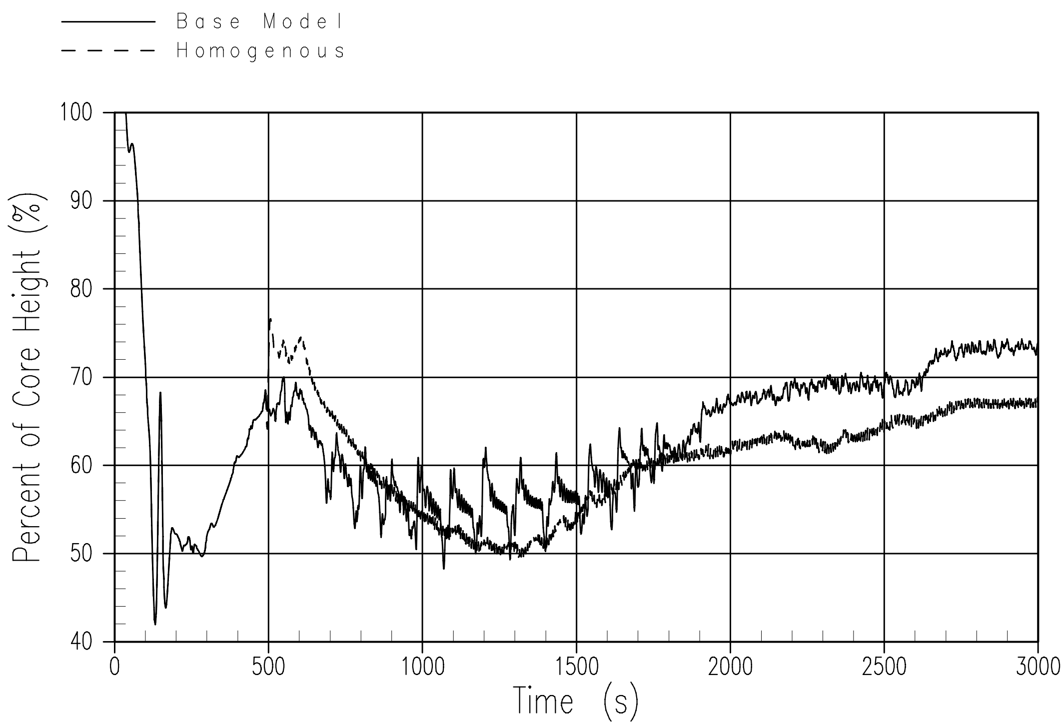

Figure 15.6.5.4B-89

DEDVI – Core Collapsed Liquid Level Comparison

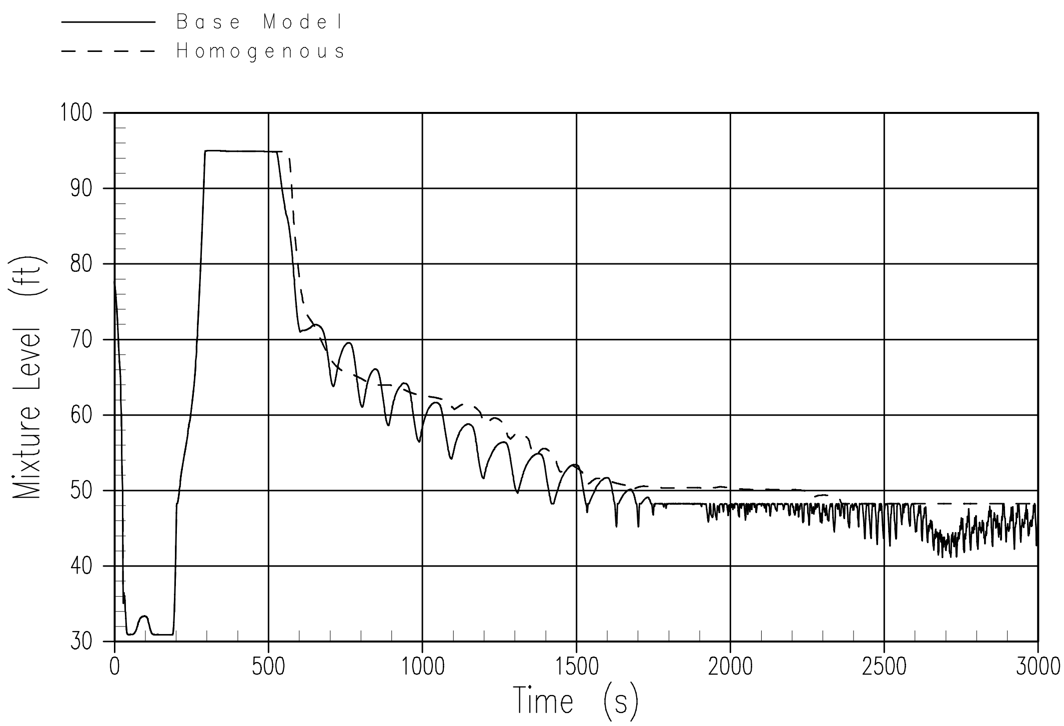

Figure 15.6.5.4B-90

DEDVI – Pressurizer Mixture Level Comparison

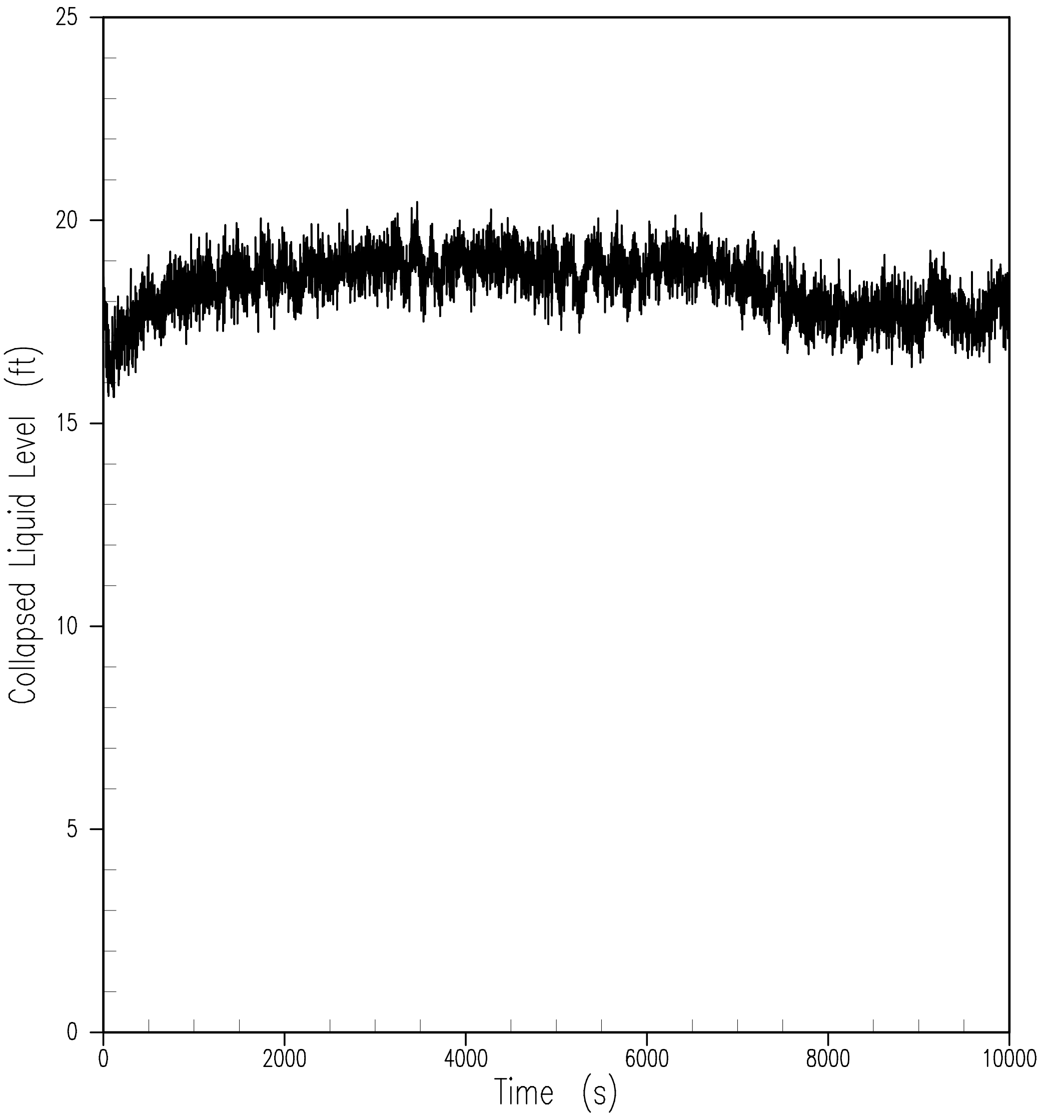

Figure 15.6.5.4C-1

Collapsed Level of Liquid in the Downcomer (DEDVI Case)

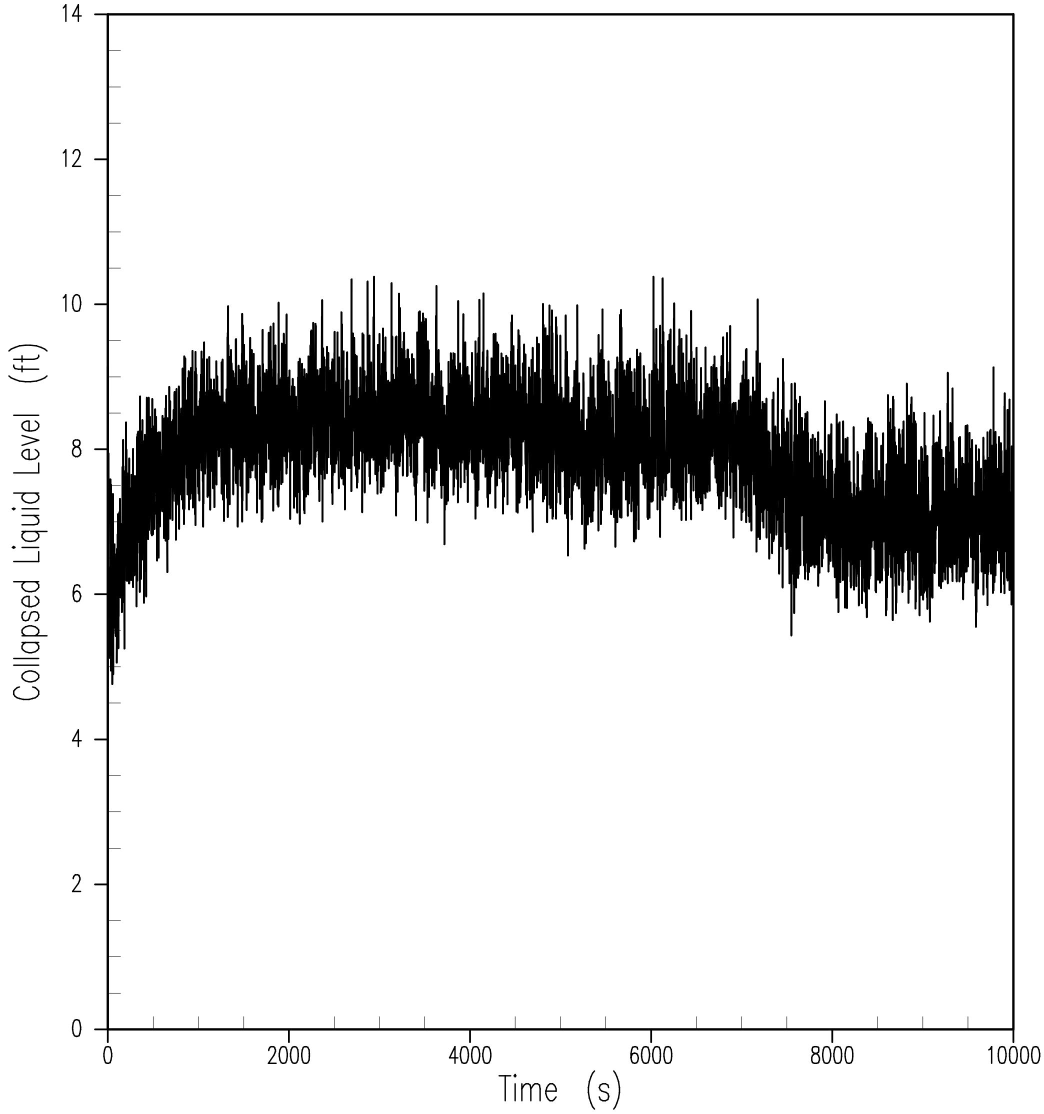

Figure 15.6.5.4C-2

Collapsed Level of Liquid over the Heated Length of the Fuel (DEDVI Case)

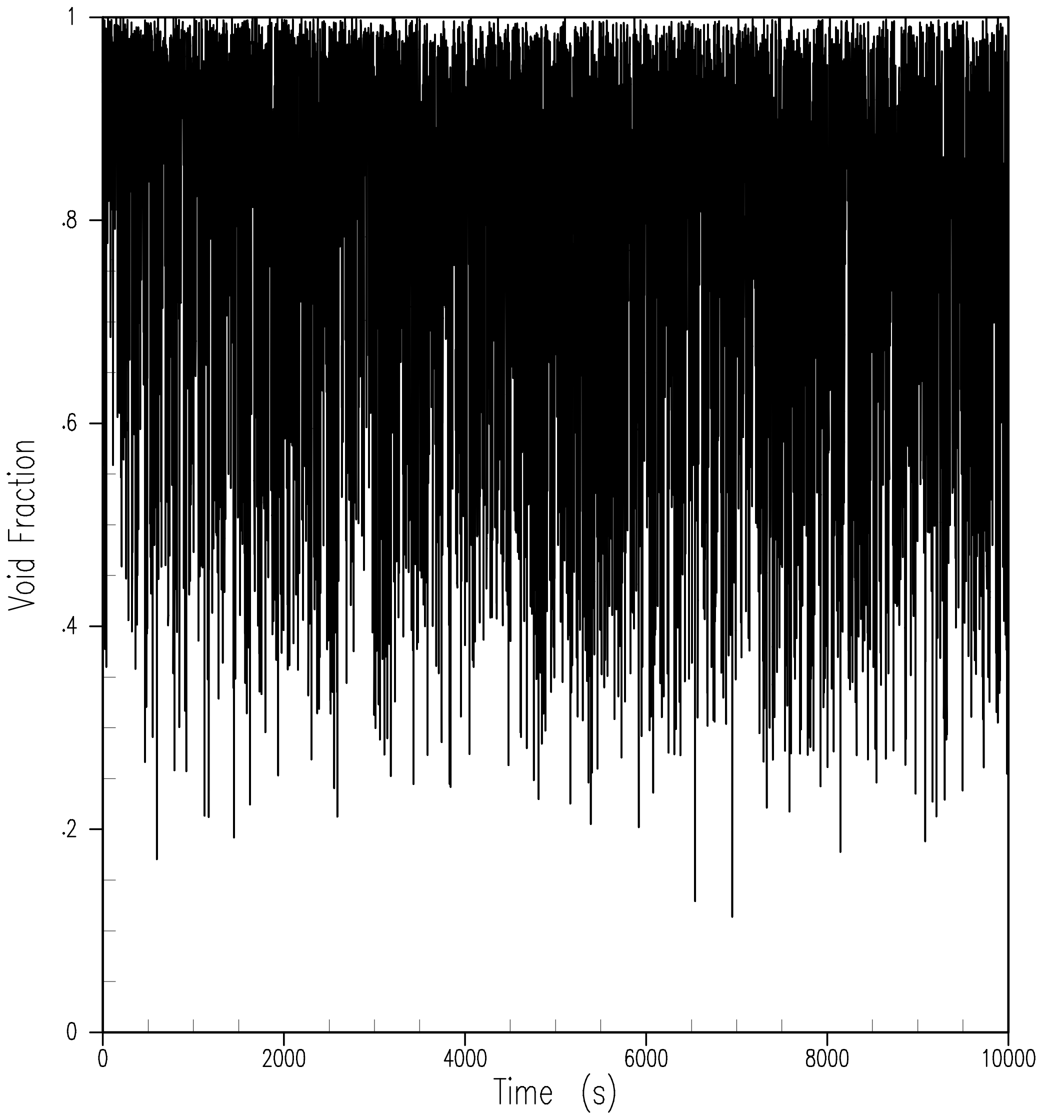

Figure 15.6.5.4C-3

Void Fraction in Core Hot Assembly Top Cell (DEDVI Case)

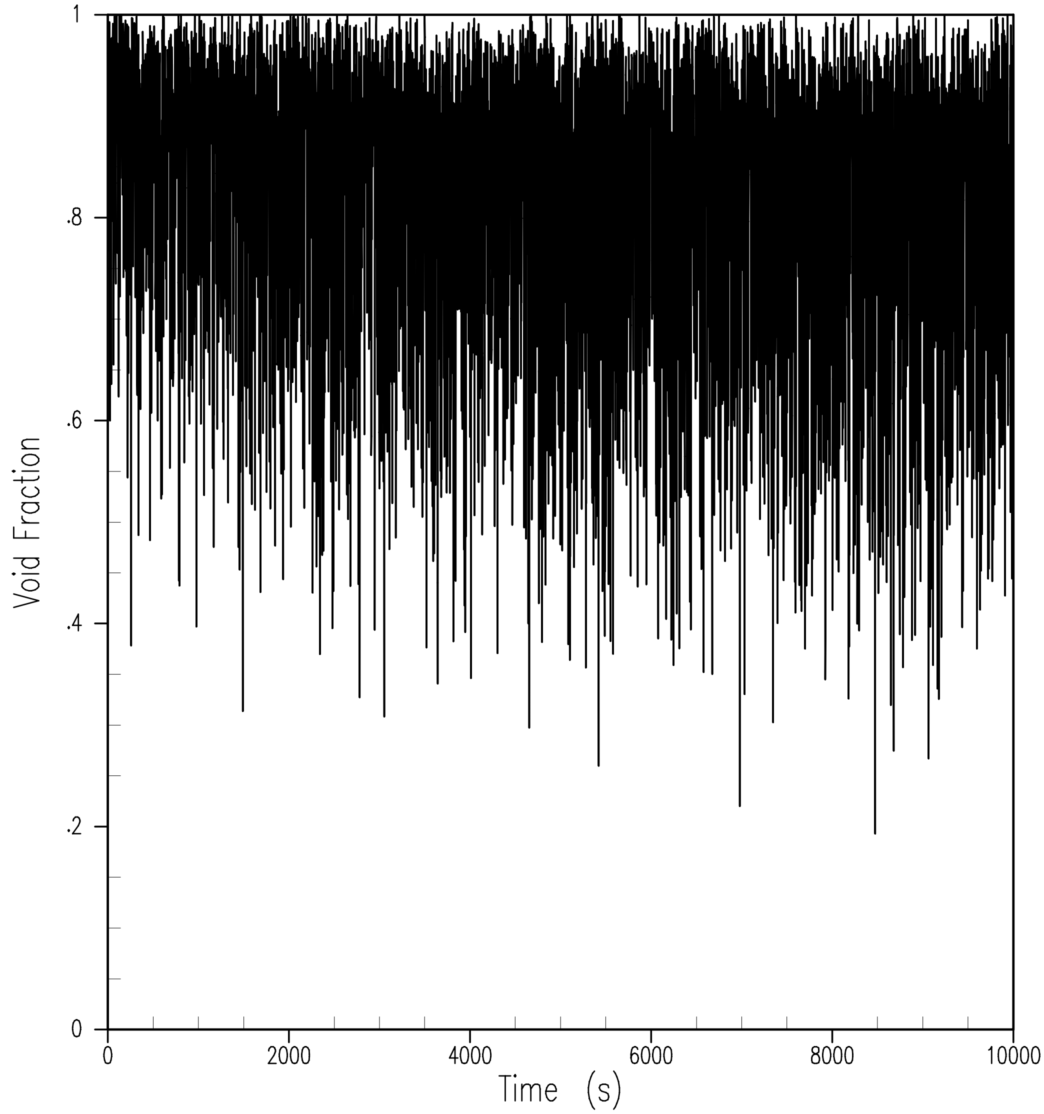

Figure 15.6.5.4C-4

Void Fraction in Core Hot Assembly Second from Top Cell (DEDVI Case)

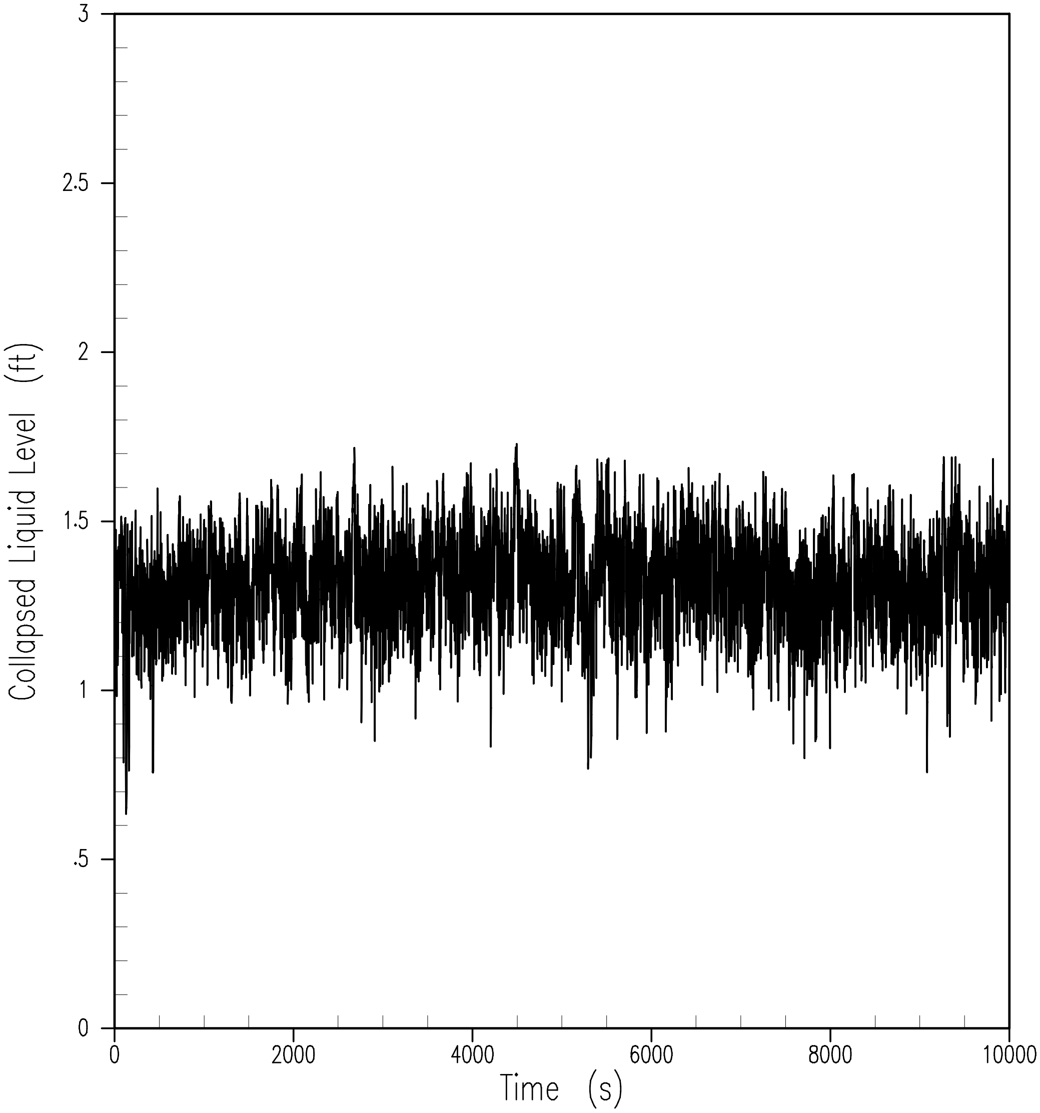

Figure 15.6.5.4C-5

Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (DEDVI Case)

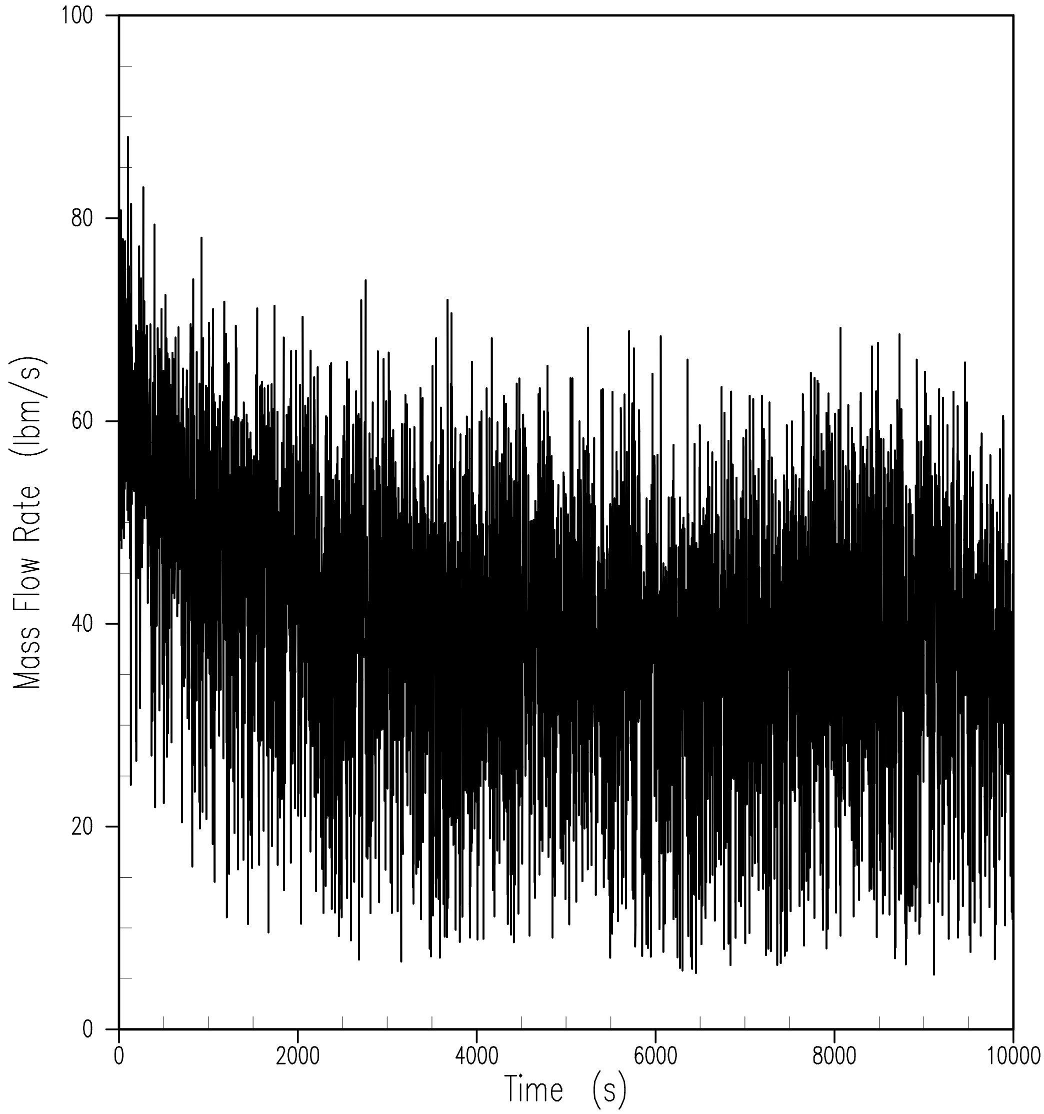

Figure 15.6.5.4C-6

Vapor Rate out of the Core (DEDVI Case)

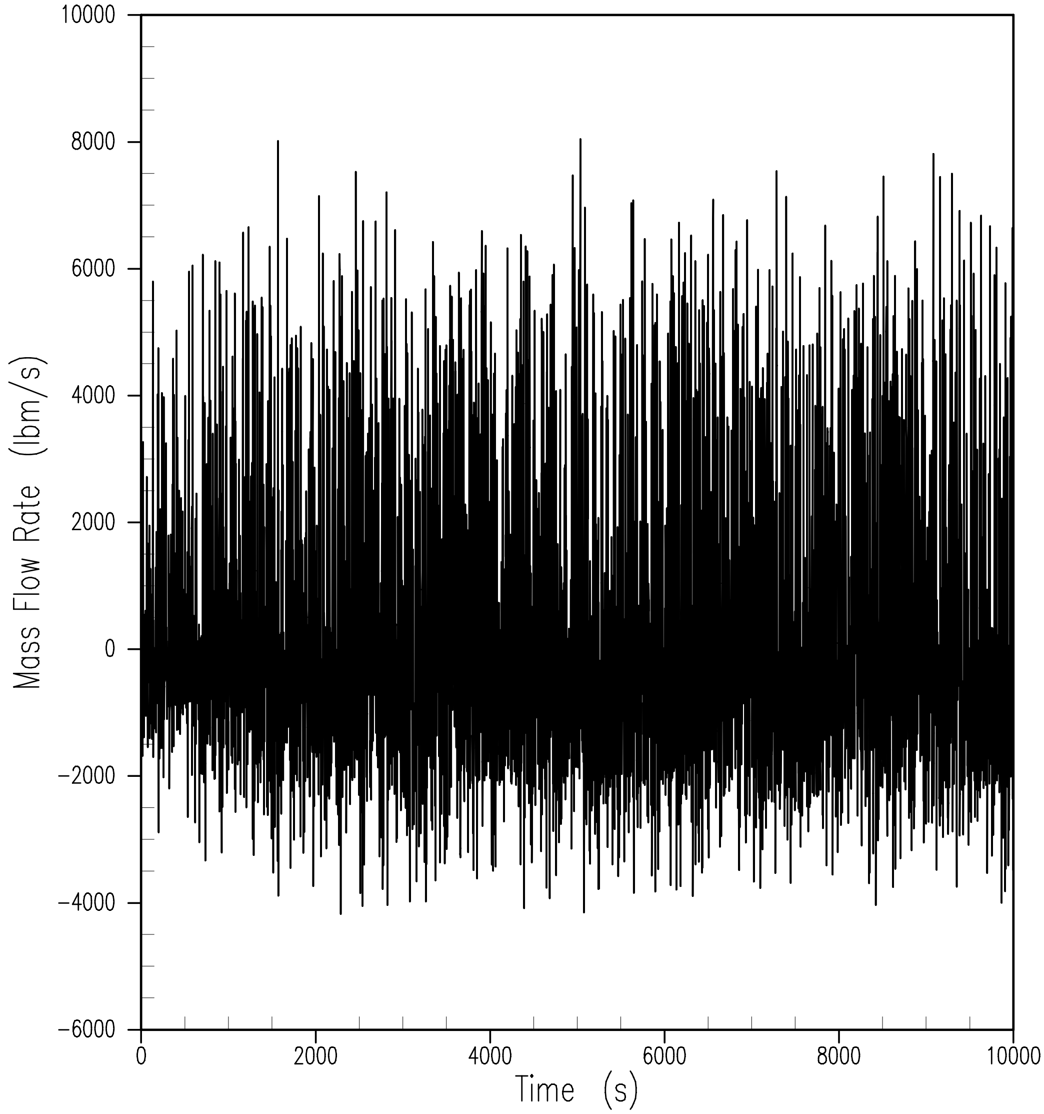

Figure 15.6.5.4C-7

Liquid Flow Rate out of the Core (DEDVI Case)

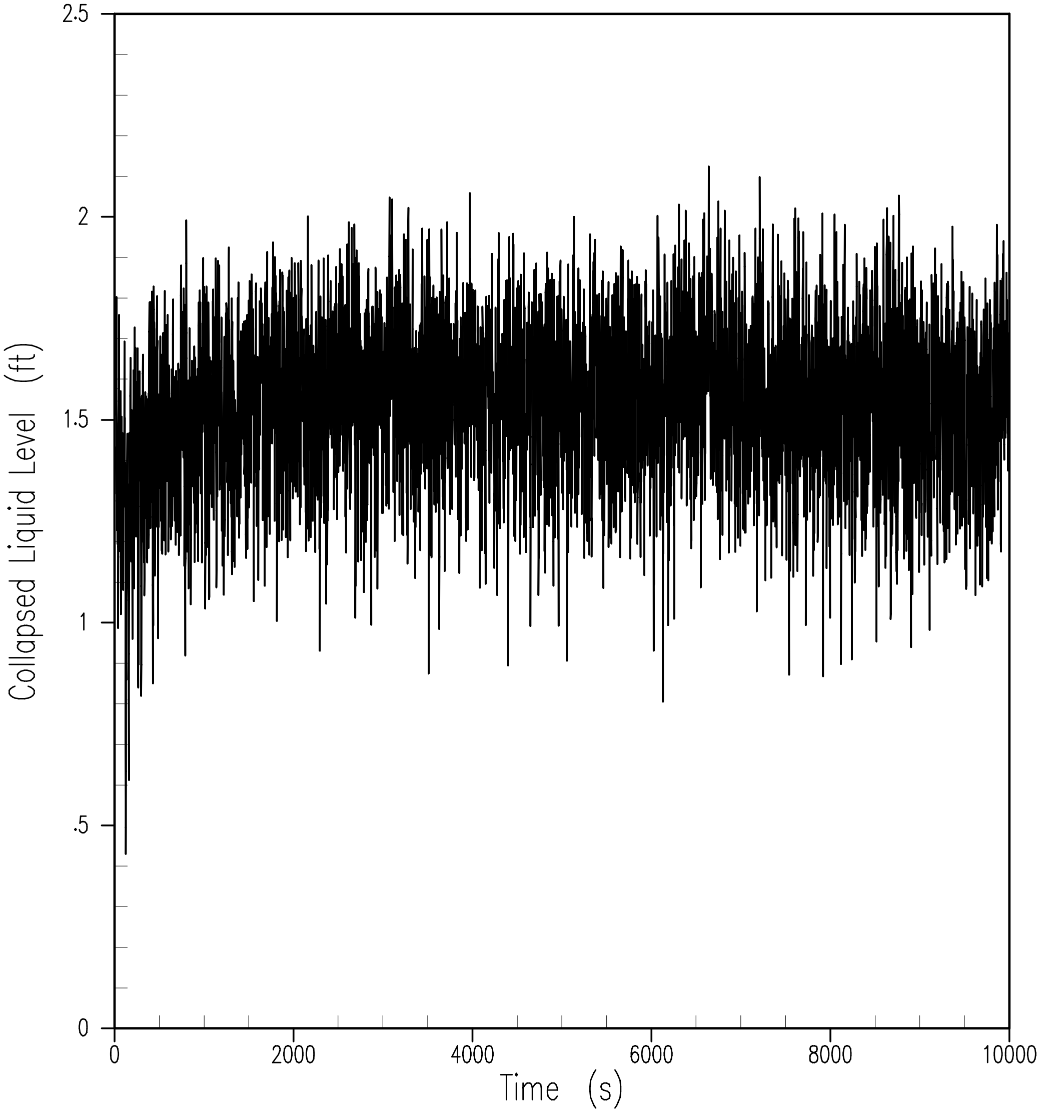

Figure 15.6.5.4C-8

Collapsed Liquid Level in the Upper Plenum (DEDVI Case)

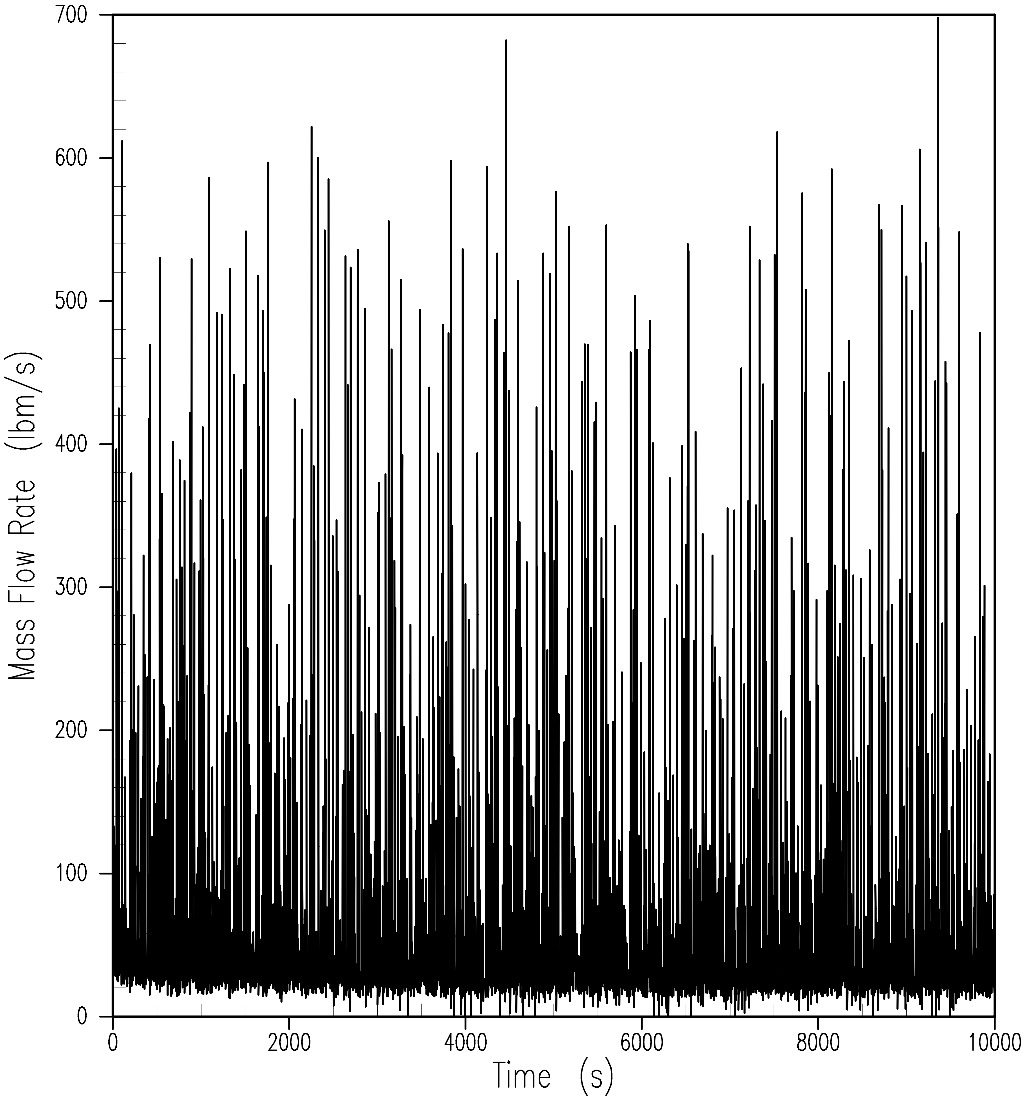

Figure 15.6.5.4C-9

Mixture Flow Rate Through ADS Stage 4A Valves (DEDVI Case)

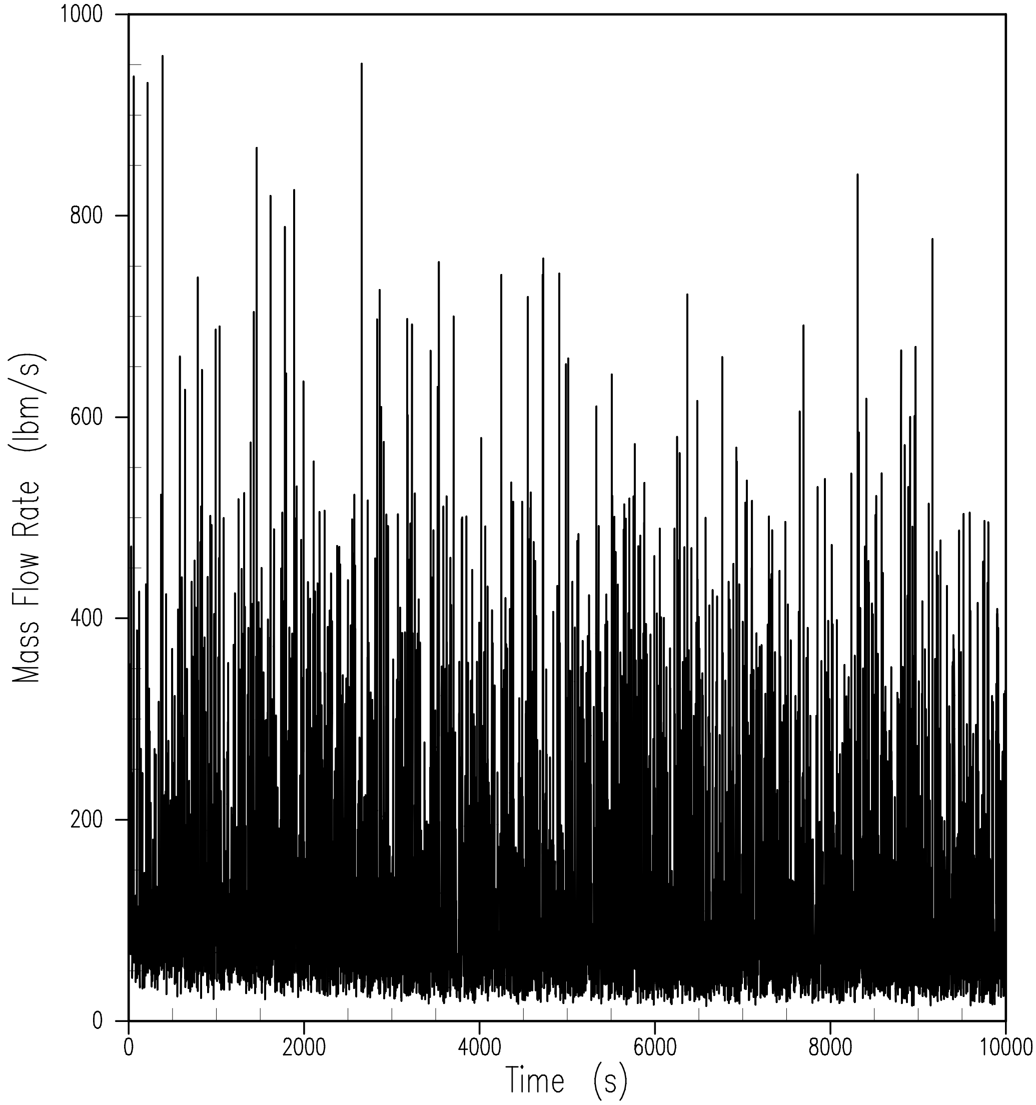

Figure 15.6.5.4C-10

Mixture Flow Rate Through ADS Stage 4B Valves (DEDVI Case)

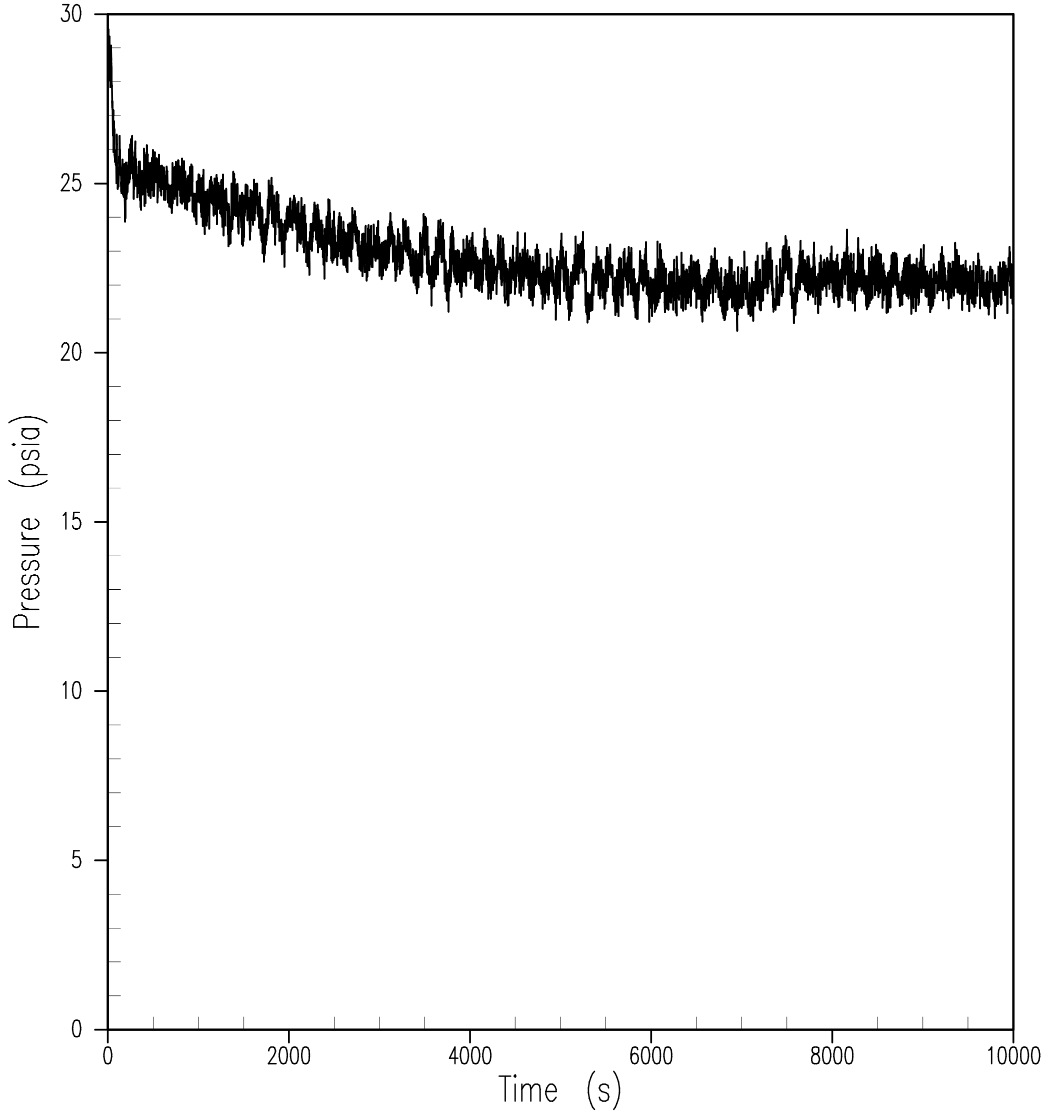

Figure 15.6.5.4C-11

Upper Plenum Pressure (DEDVI Case)

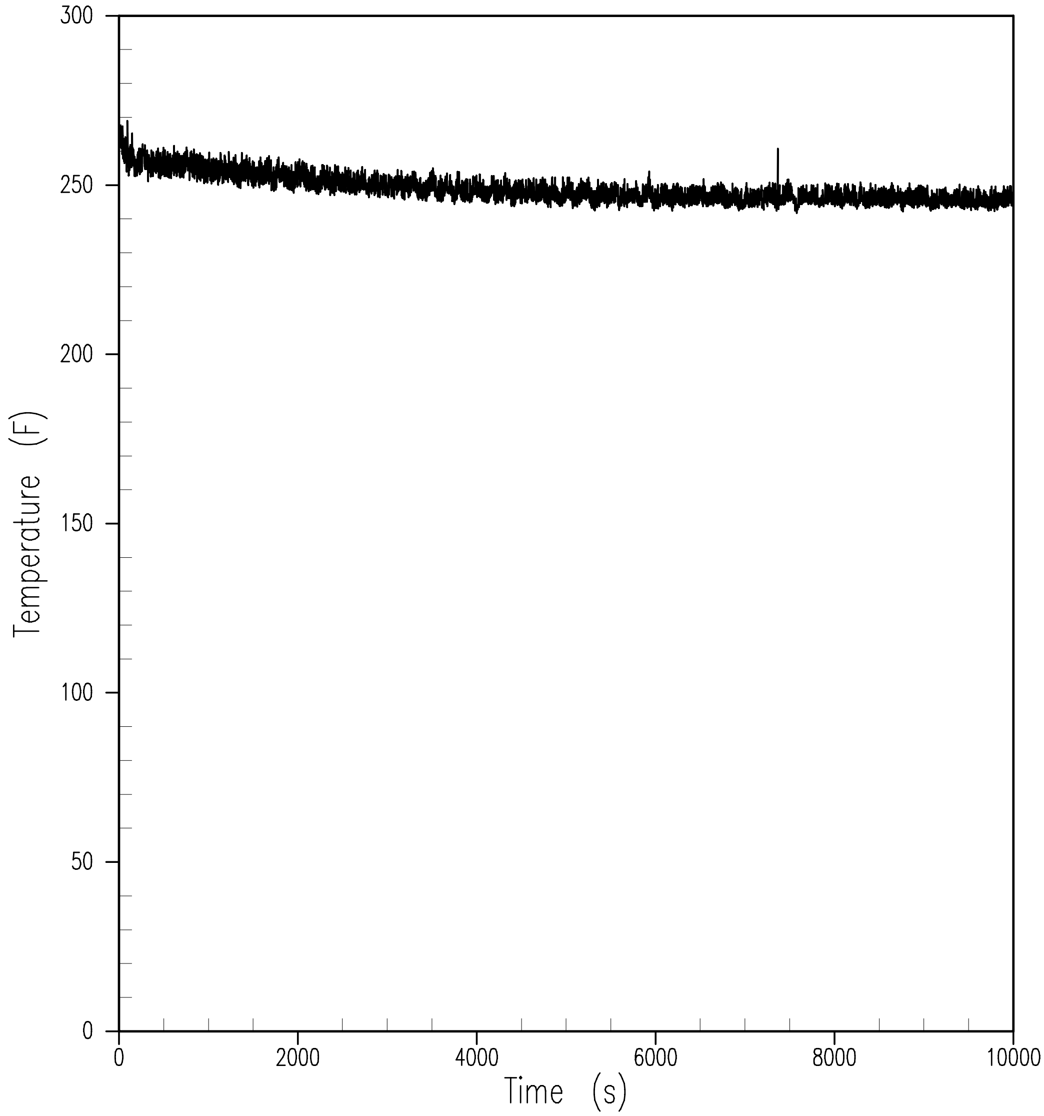

Figure 15.6.5.4C-12

Peak Cladding Temperature (DEDVI Case)

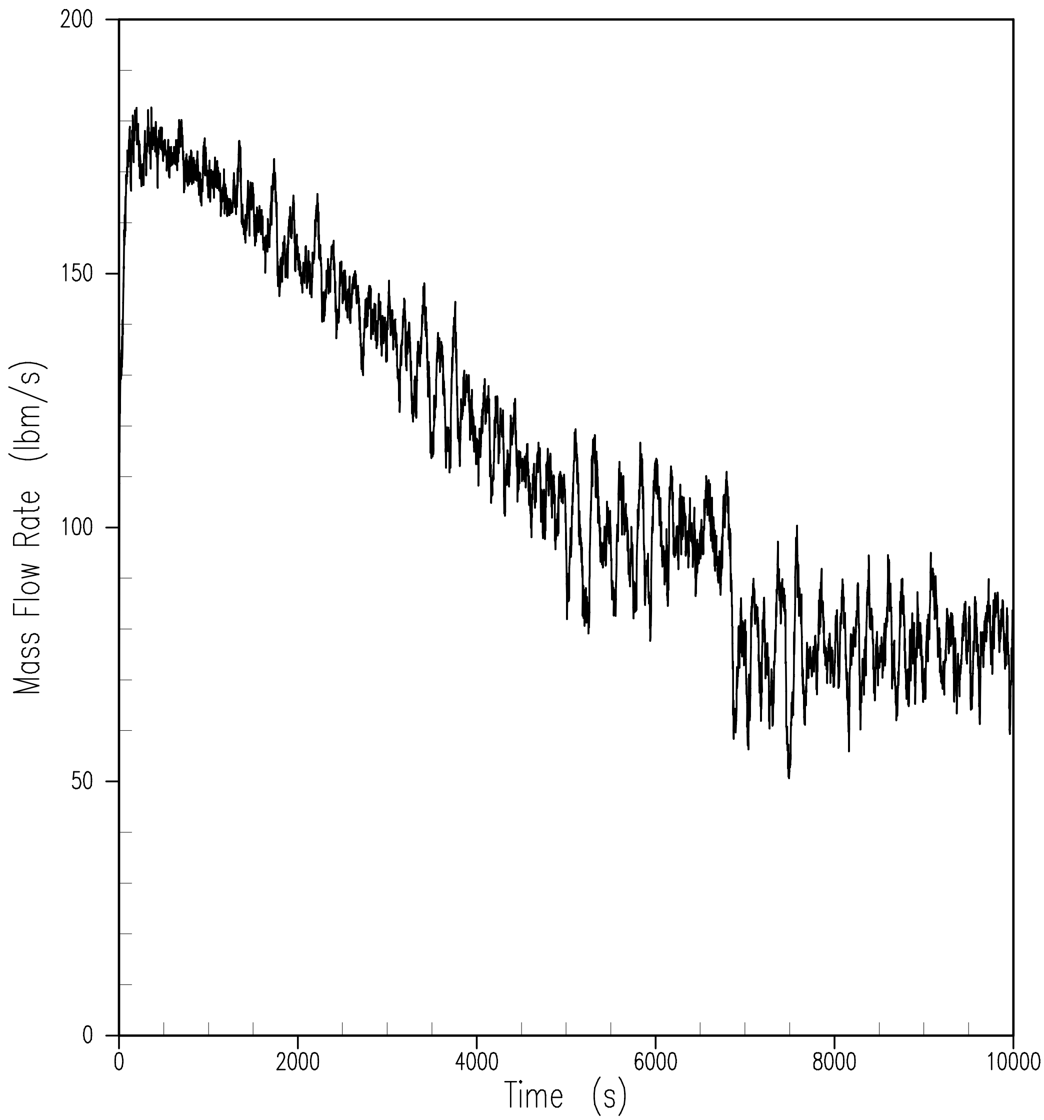

Figure 15.6.5.4C-13

DVI–A Mixture Flow Rate (DEDVI Case)

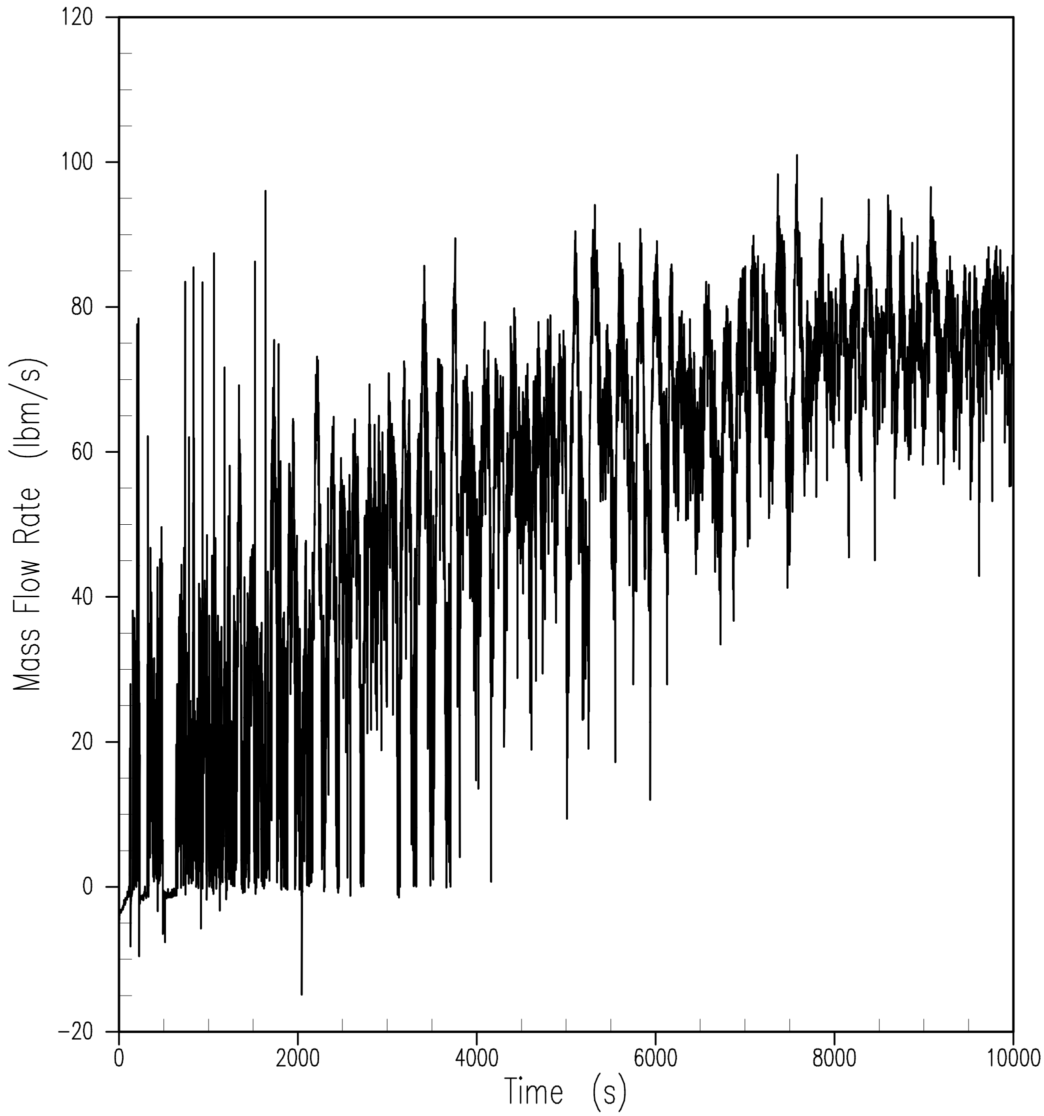

Figure 15.6.5.4C-14

DVI–B Mixture Flow Rate (DEDVI Case)

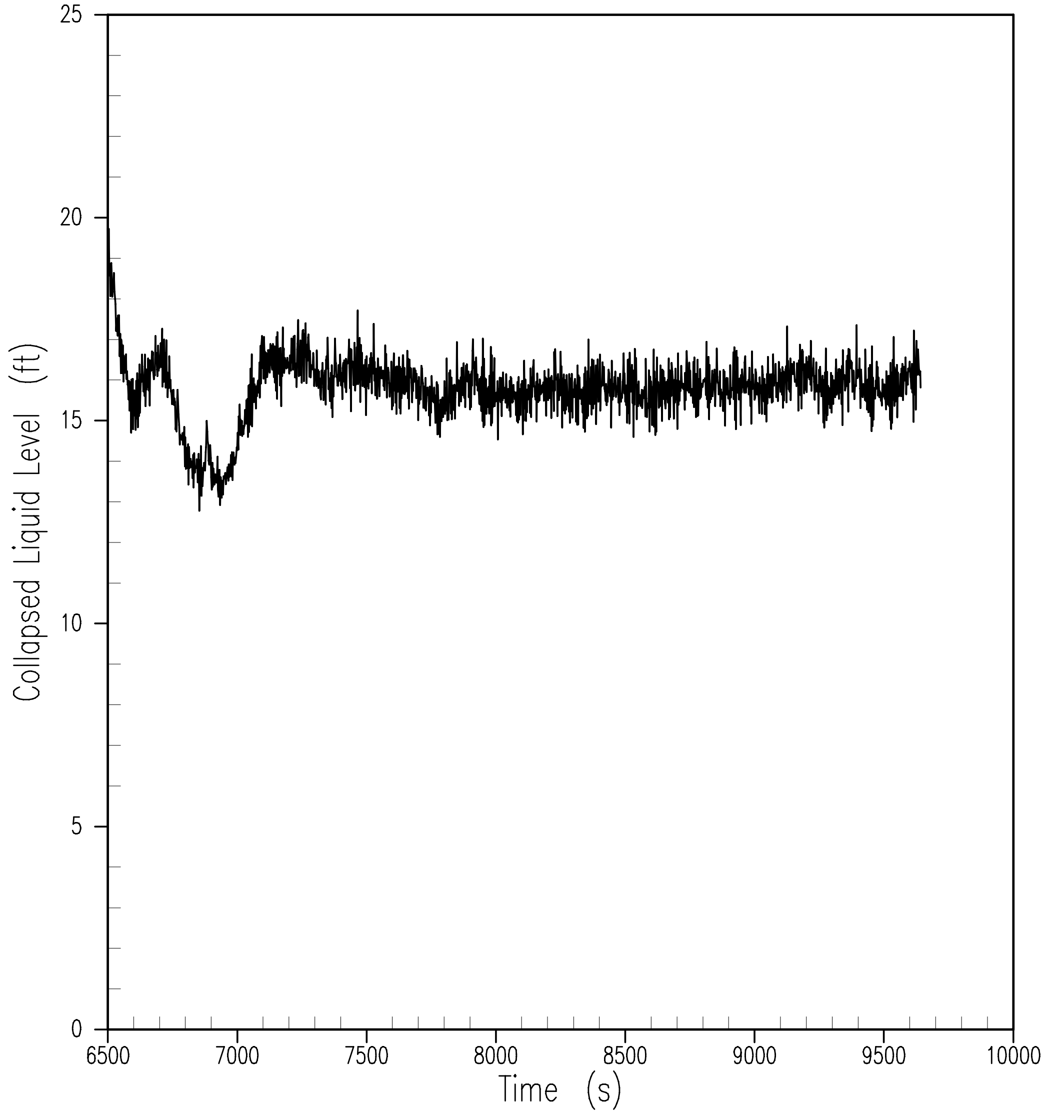

Figure 15.6.5.4C-1A

Collapsed Level of Liquid in the Downcomer (DEDVI Case) – 14.7 psi

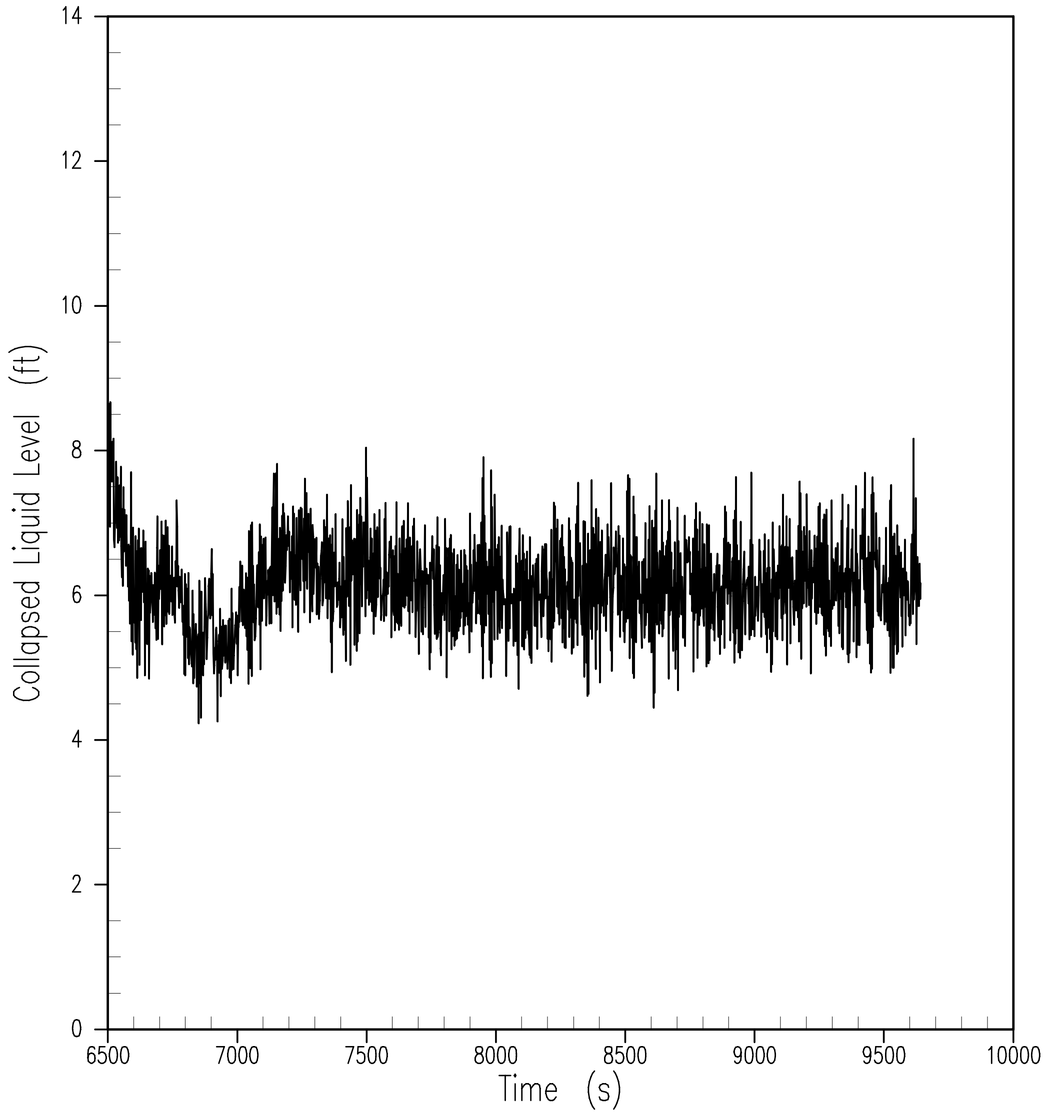

Figure 15.6.5.4C-2A

Collapsed Level of Liquid over the Heated Length of the Fuel (DEDVI Case) – 14.7 psi

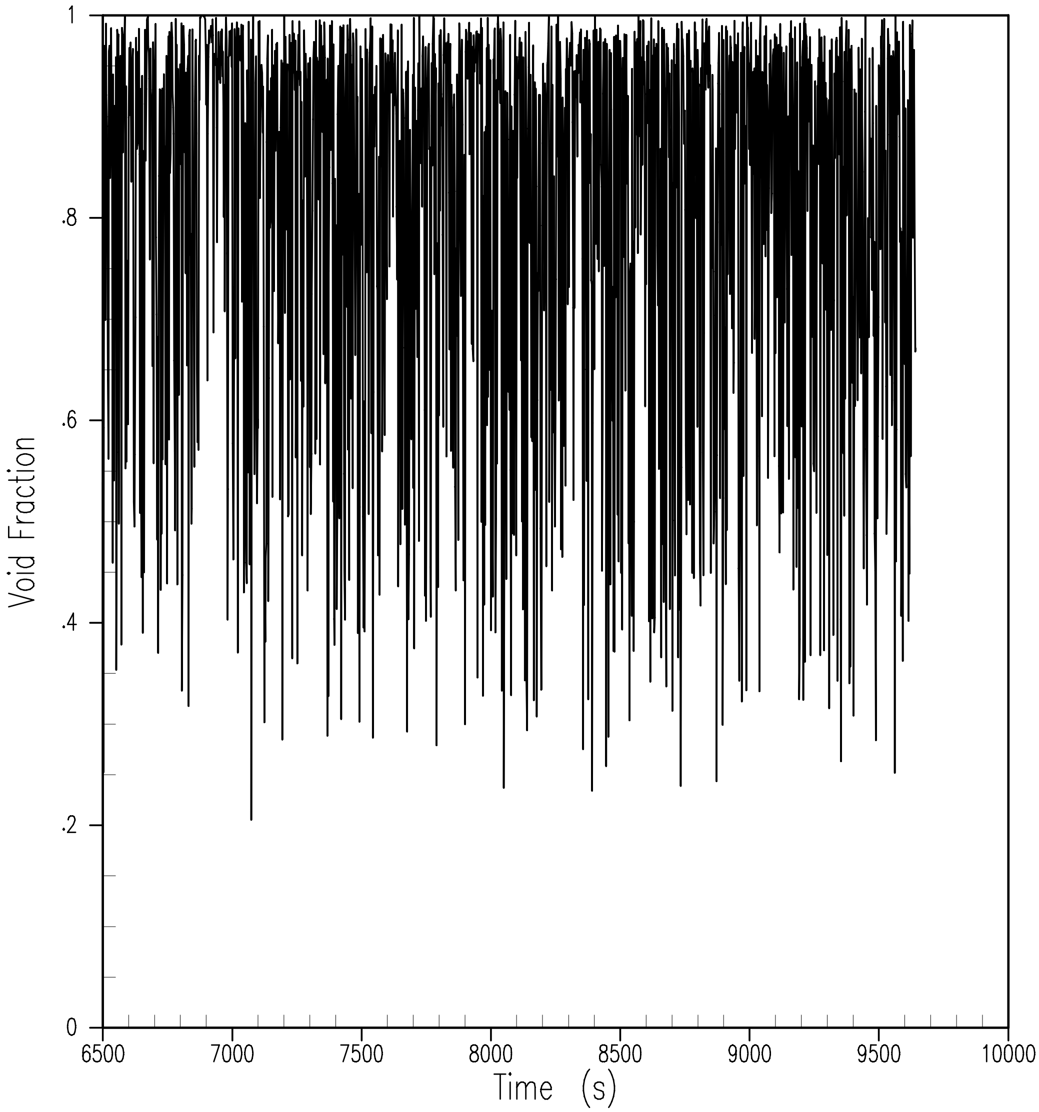

Figure 15.6.5.4C-3A

Void Fraction in Core Hot Assembly Top Cell (DEDVI Case) – 14.7 psi

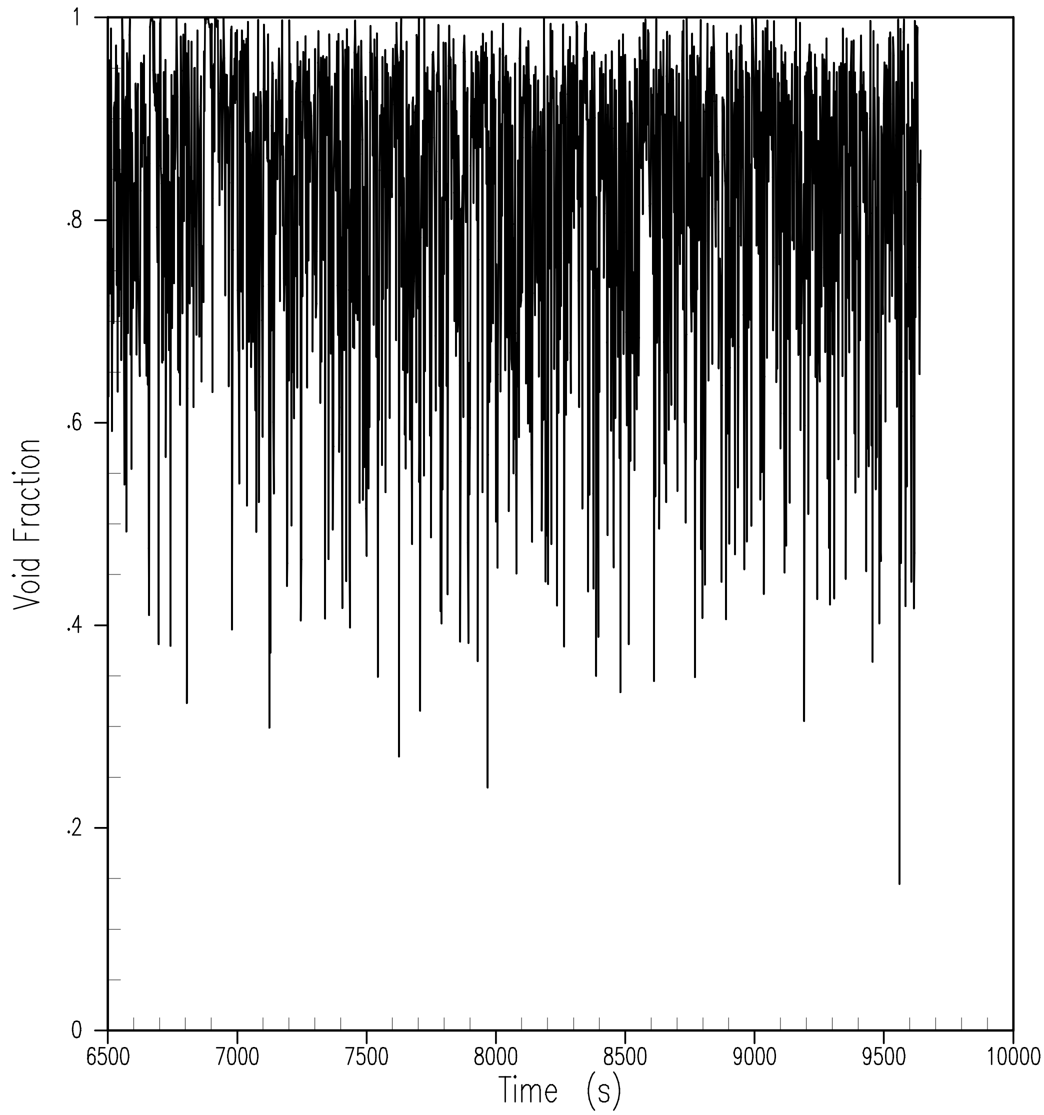

Figure 15.6.5.4C-4A

Void Fraction in Core Hot Assembly Second from Top Cell (DEDVI Case) – 14.7 psi

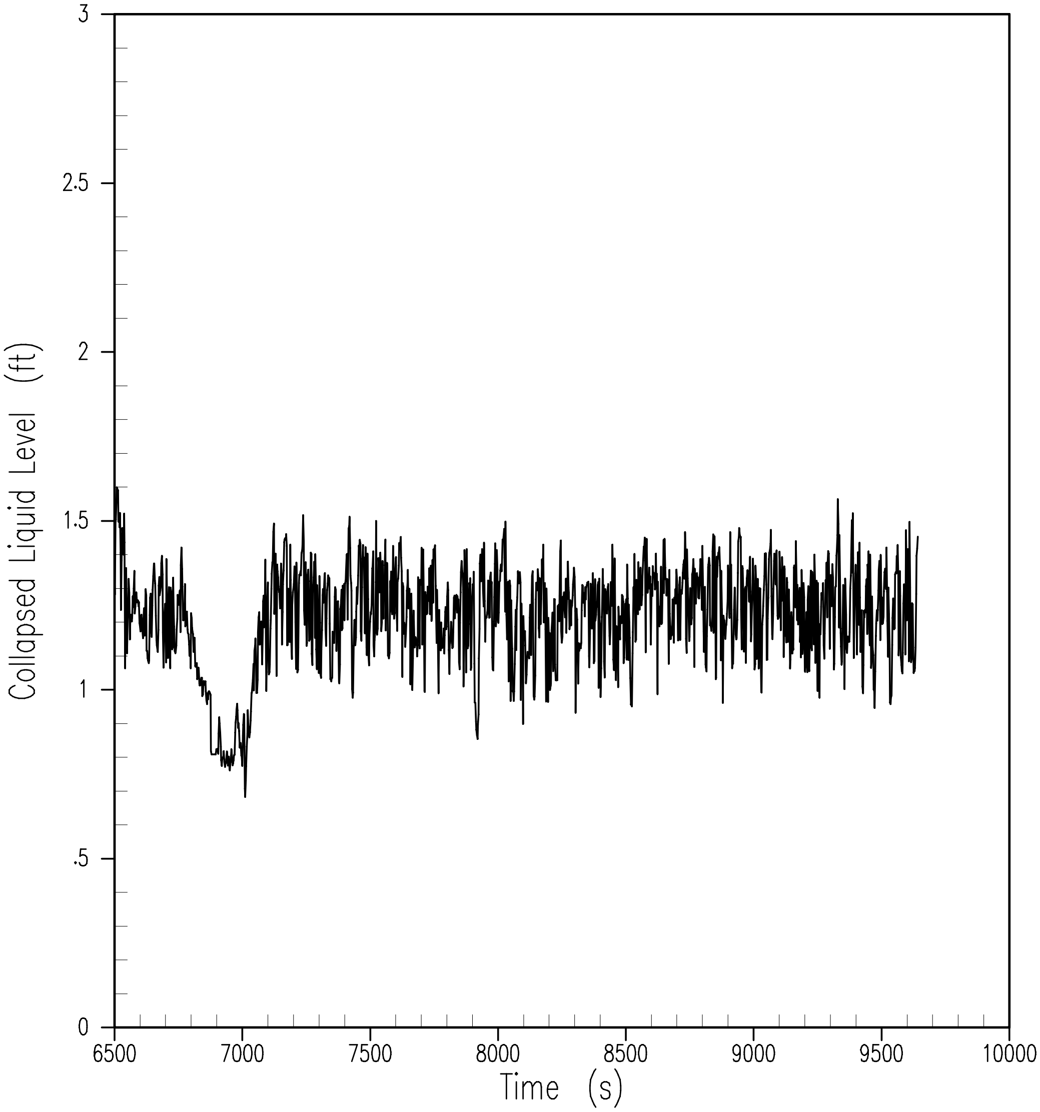

Figure 15.6.5.4C-5A

Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (DEDVI Case) – 14.7 psi

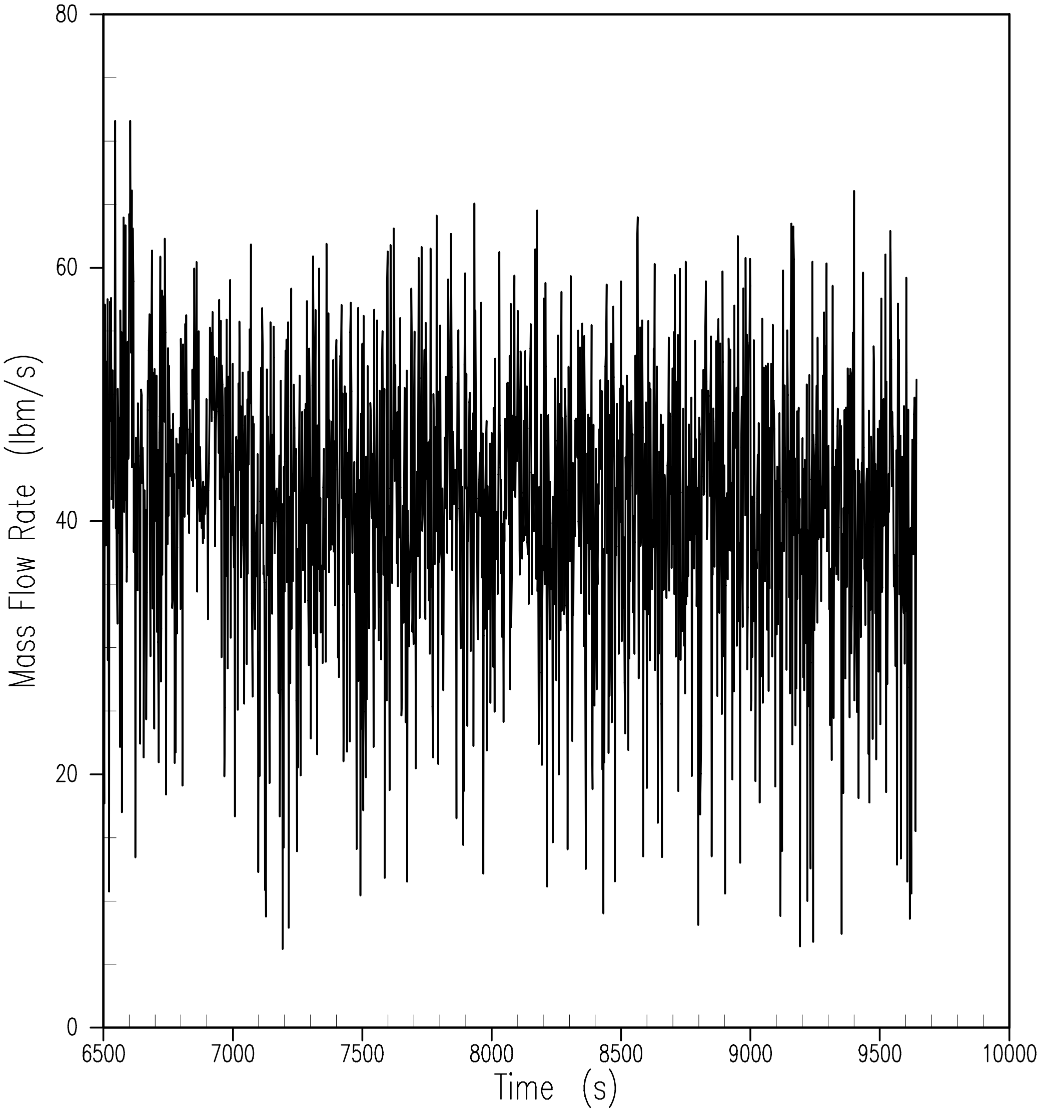

Figure 15.6.5.4C-6A

Vapor Rate out of the Core (DEDVI Case) – 14.7 psi

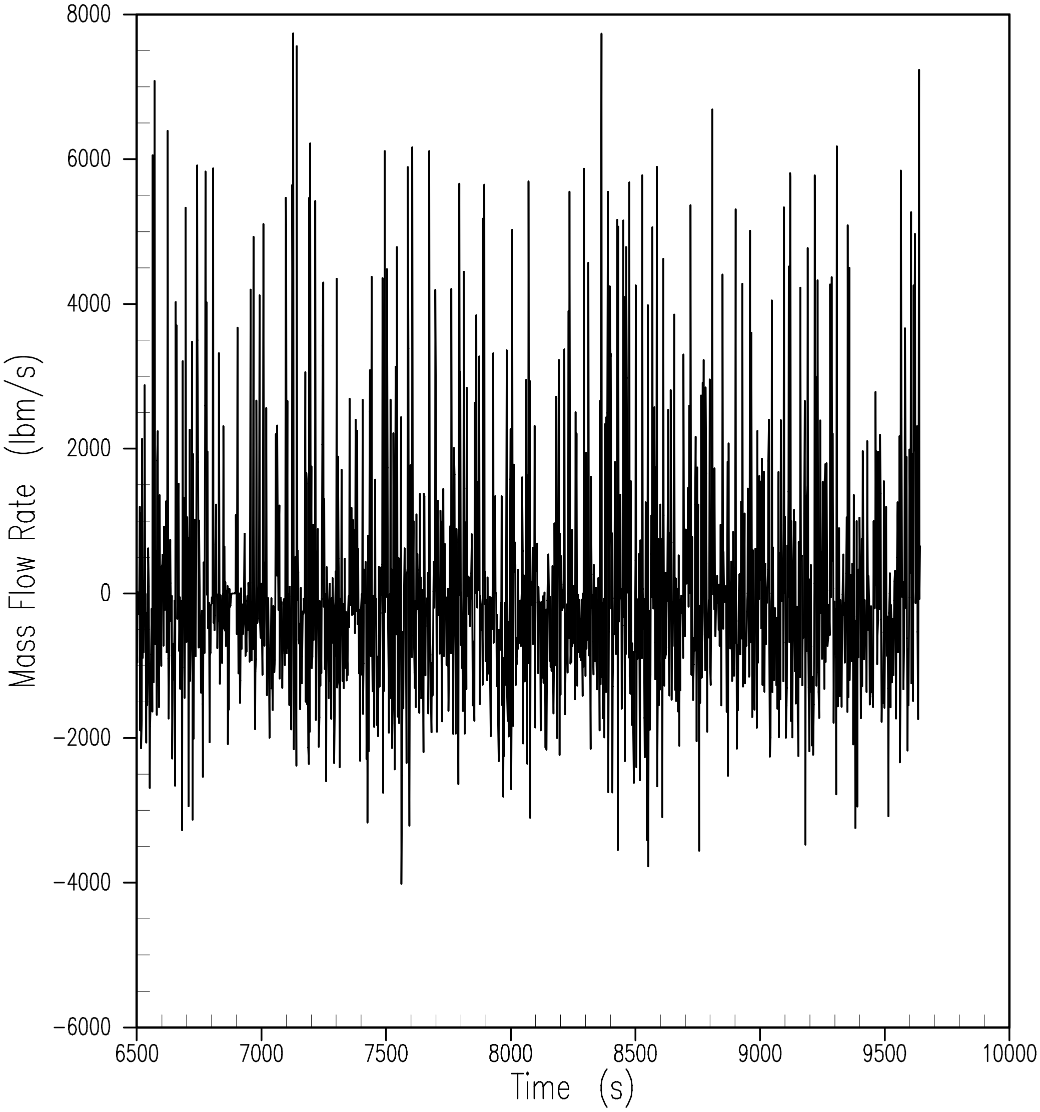

Figure 15.6.5.4C-7A

Liquid Flow Rate out of the Core (DEDVI Case) – 14.7 psi

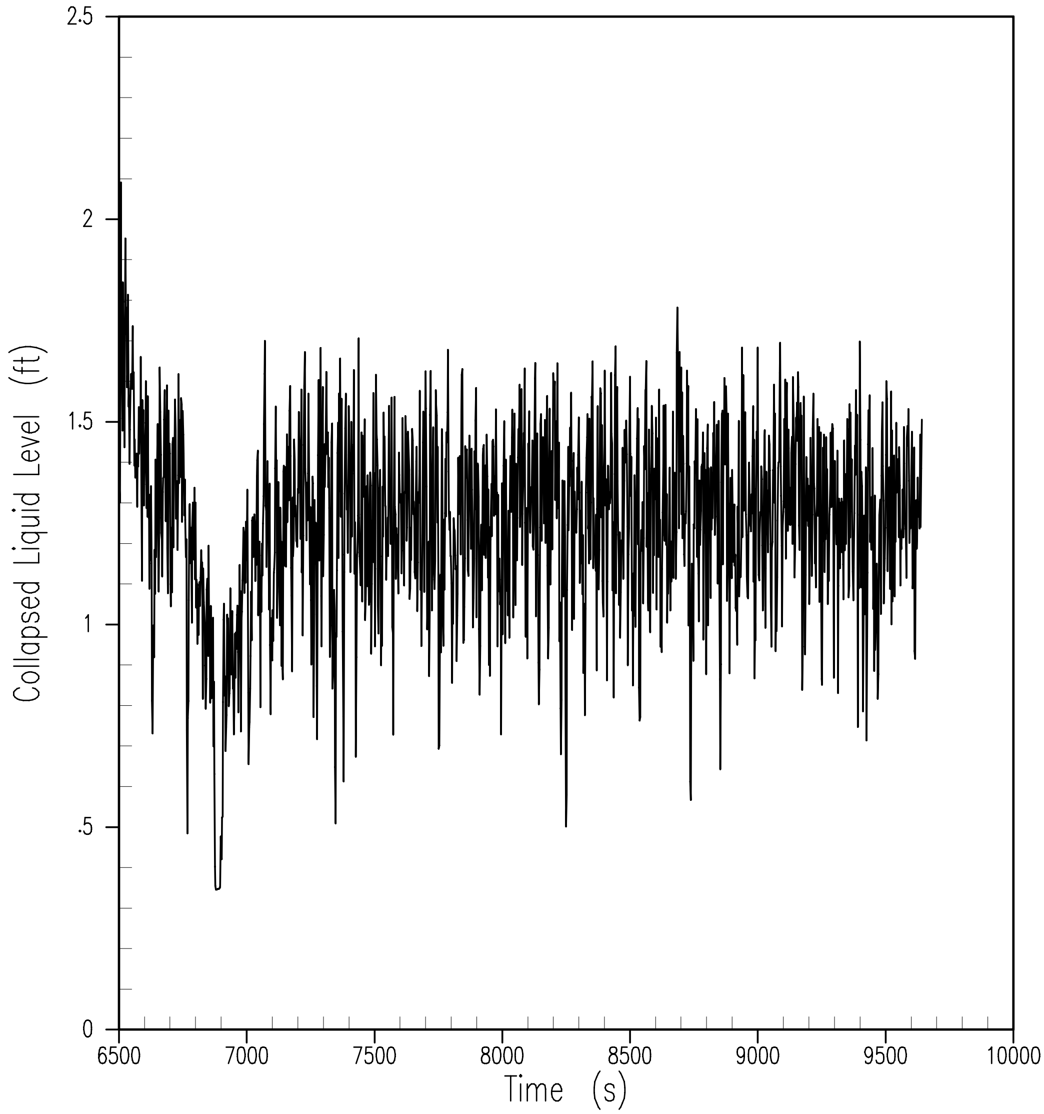

Figure 15.6.5.4C-8A

Collapsed Liquid Level in the Upper Plenum (DEDVI Case) – 14.7 psi

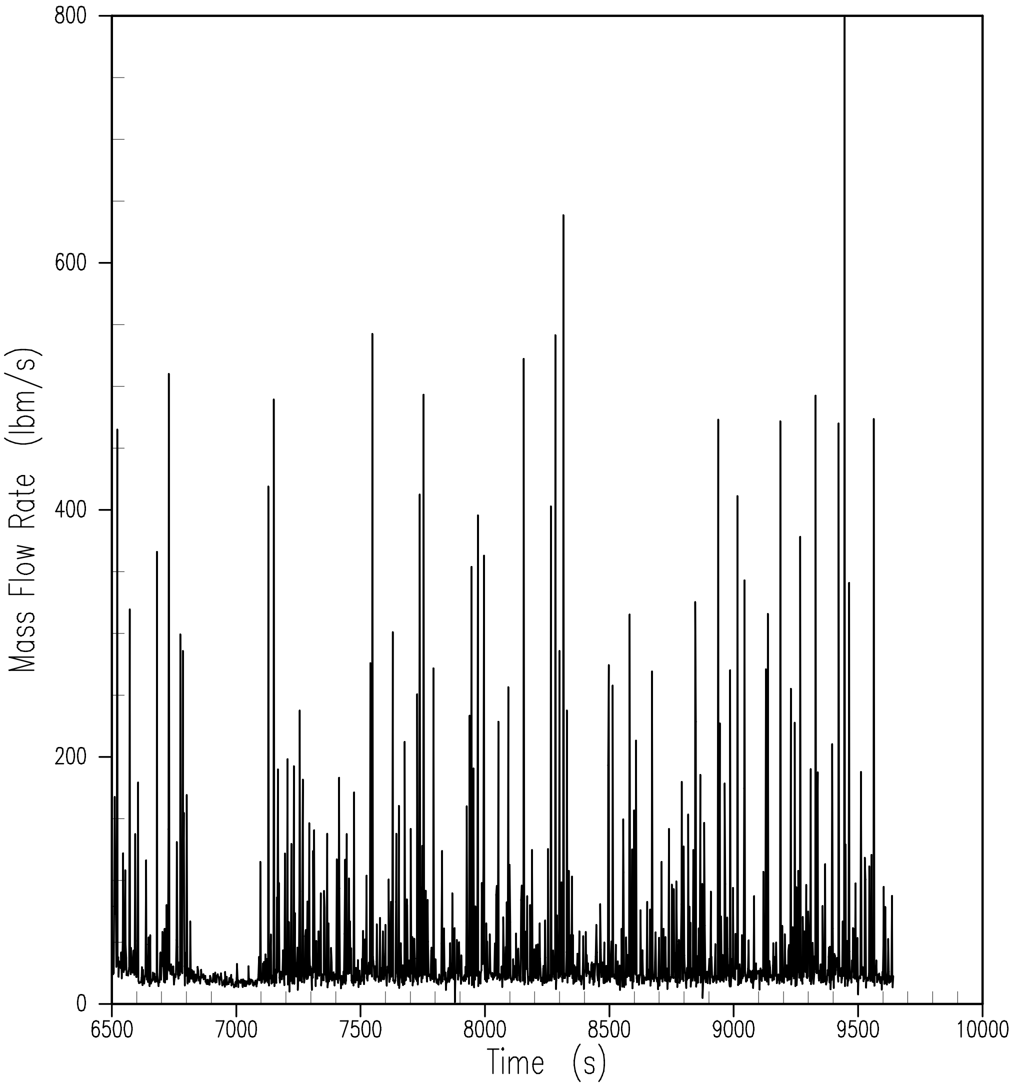

Figure 15.6.5.4C-9A

Mixture Flow Rate Through ADS Stage 4A Valves (DEDVI Case) – 14.7 psi

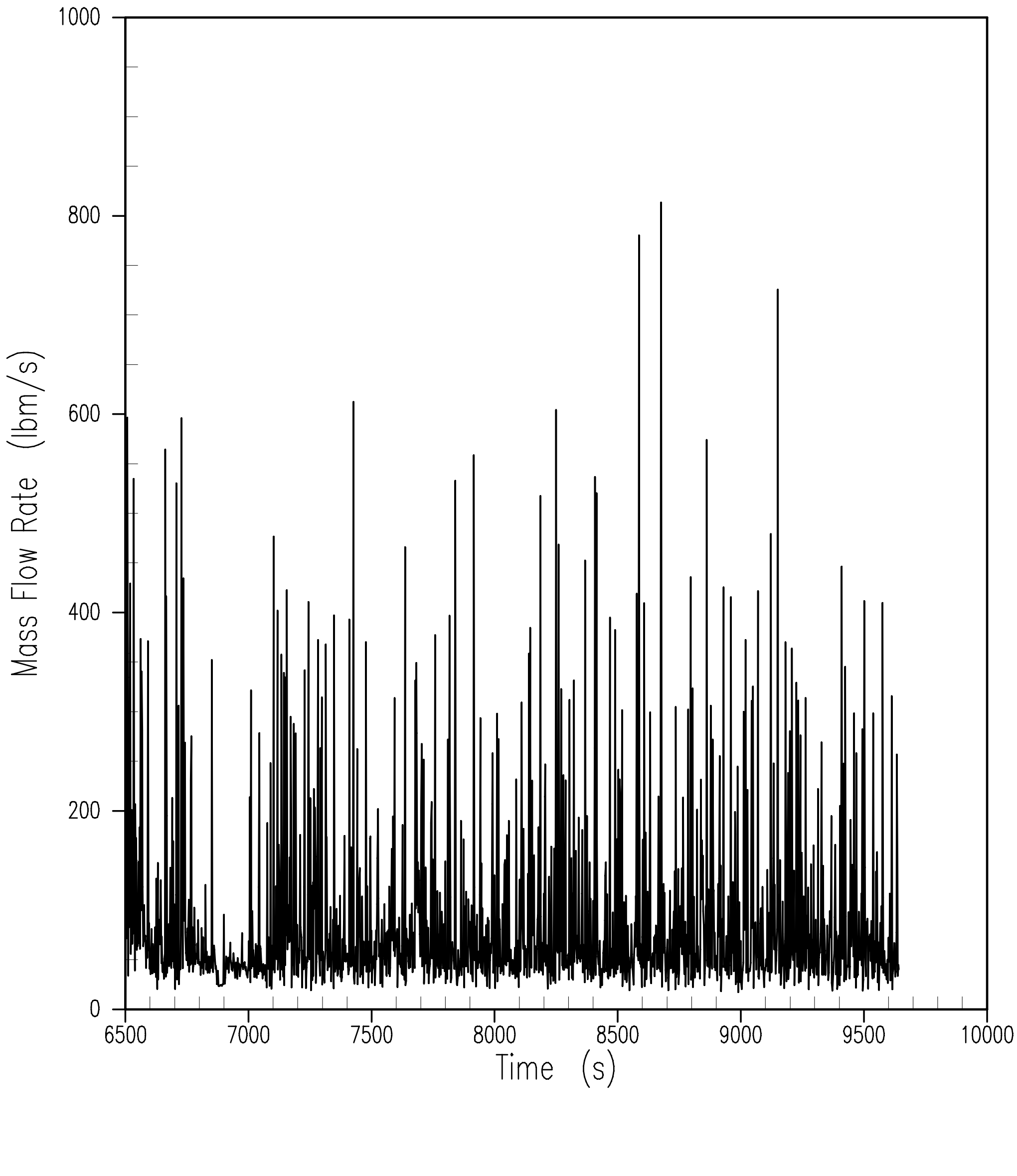

Figure 15.6.5.4C-10A

Mixture Flow Rate Through ADS Stage 4B Valves (DEDVI Case) – 14.7 psi

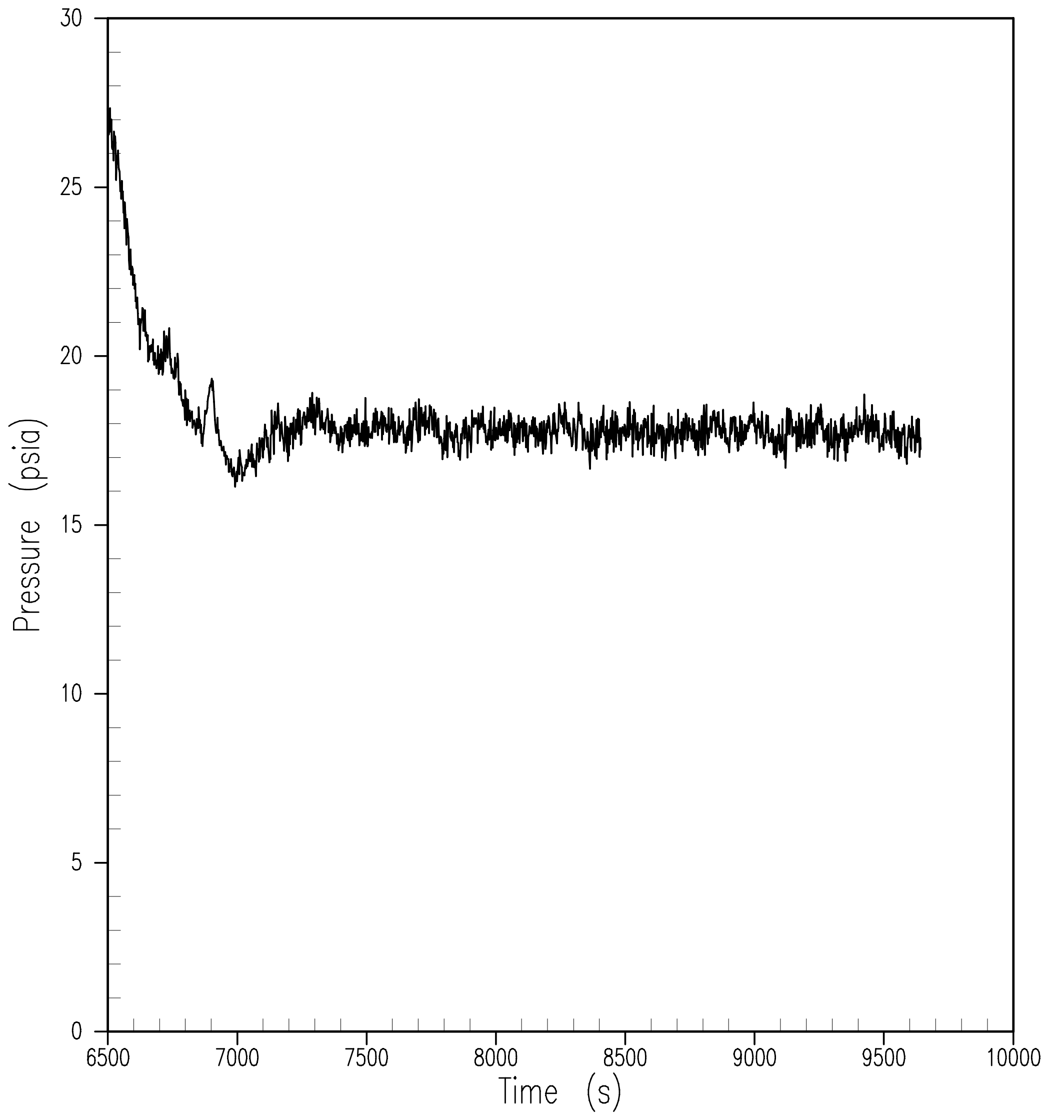

Figure 15.6.5.4C-11A

Upper Plenum Pressure (DEDVI Case) – 14.7 psi

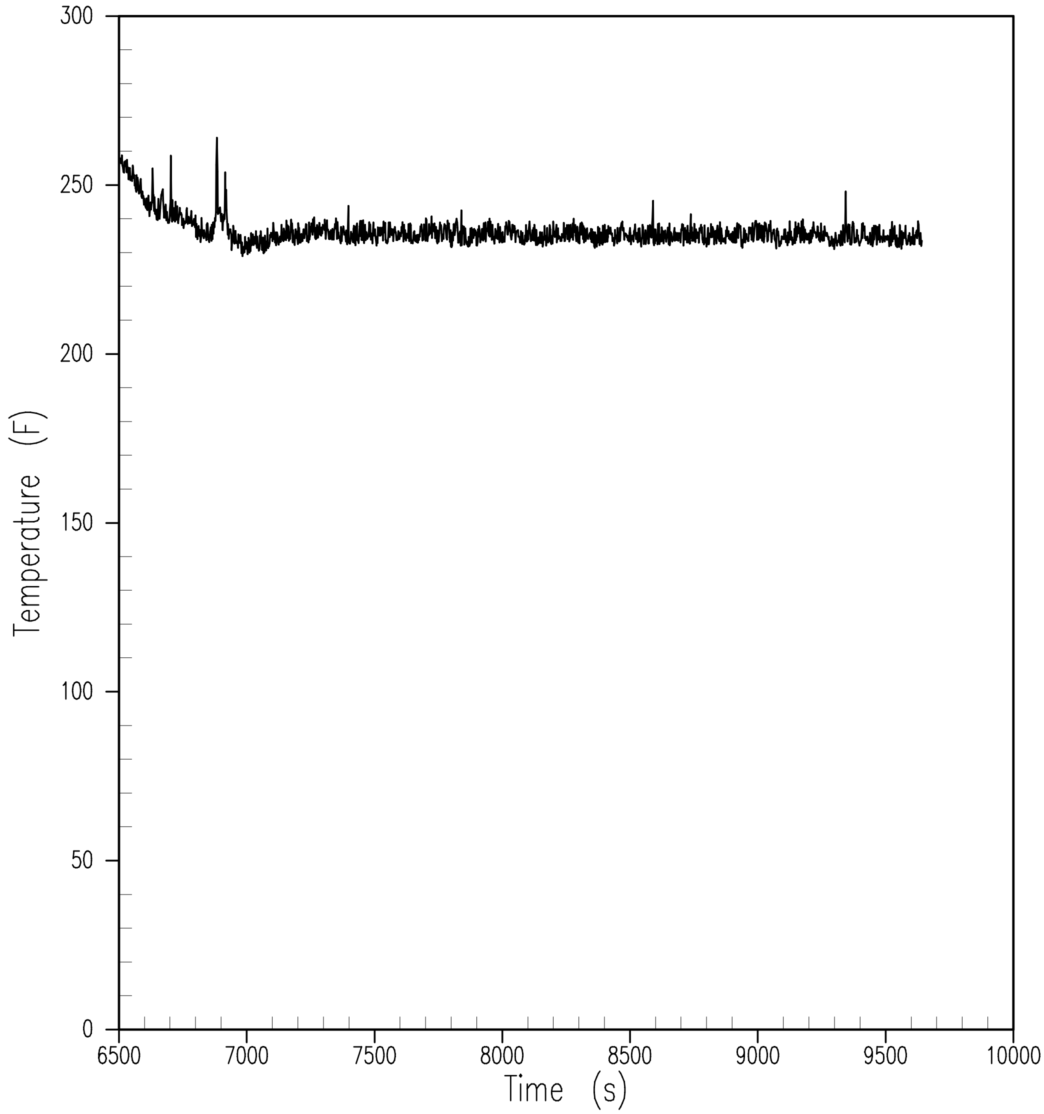

Figure 15.6.5.4C-12A

Peak Cladding Temperature (DEDVI Case) – 14.7 psi

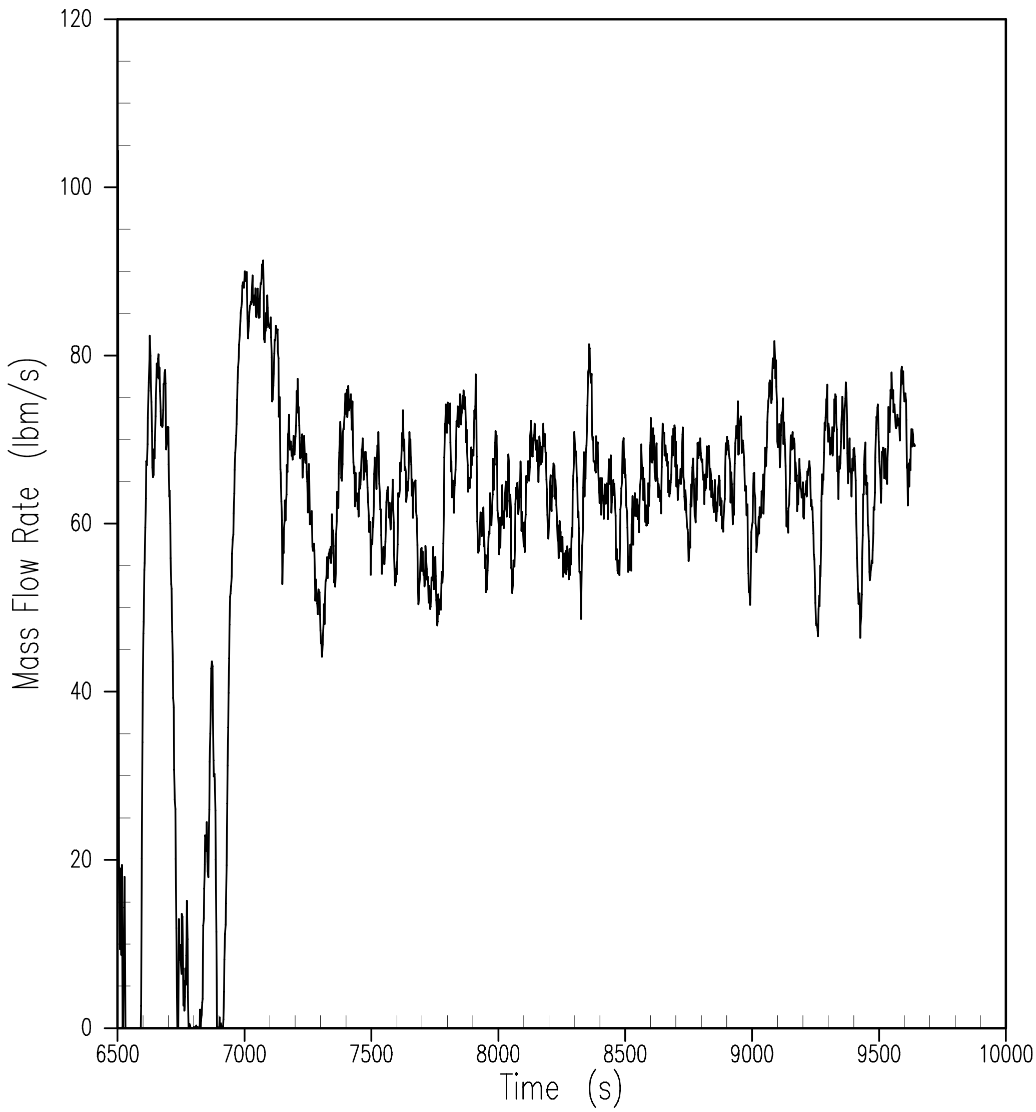

Figure 15.6.5.4C-13A

DVI–A Mixture Flow Rate (DEDVI Case) – 14.7 psi

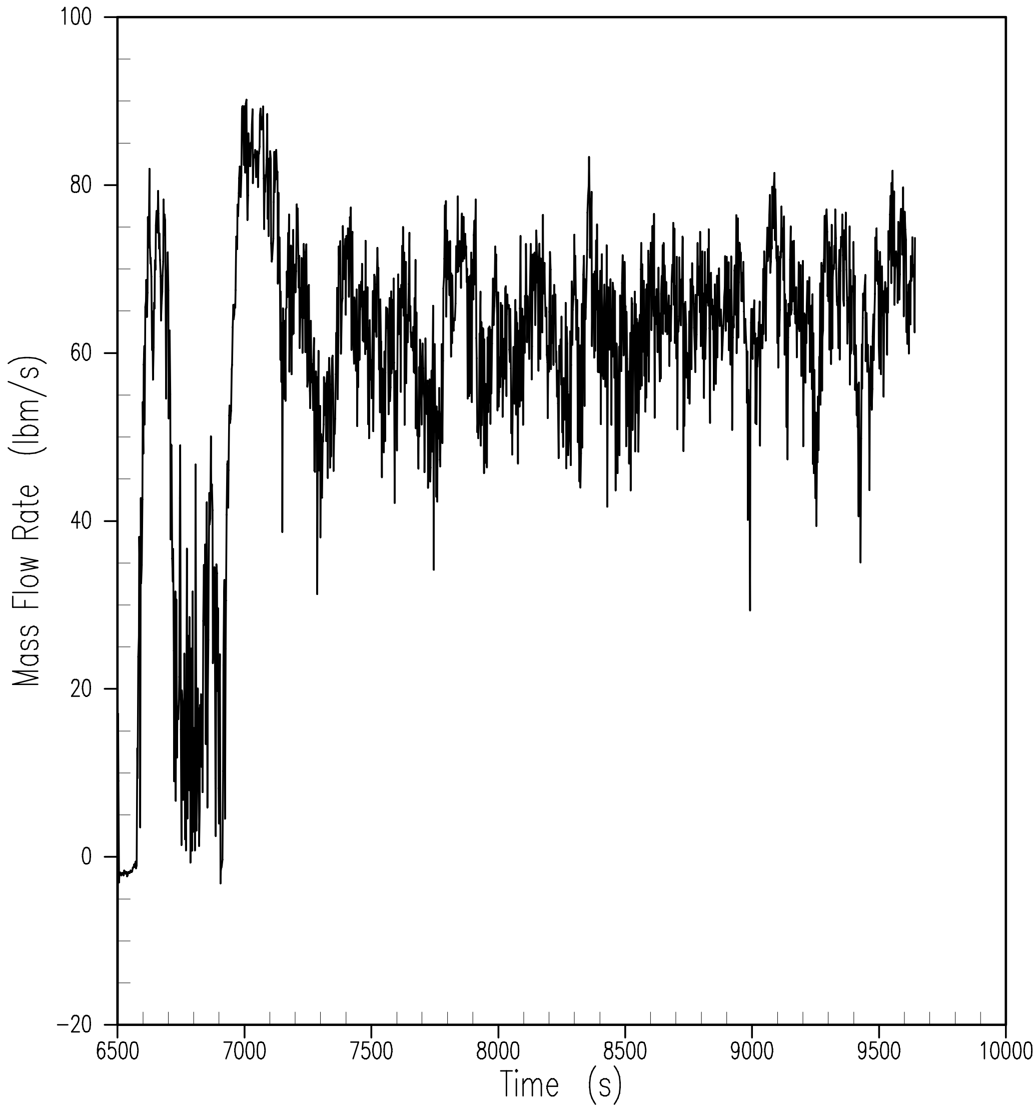

Figure 15.6.5.4C-14A

DVI–B Mixture Flow Rate
(DEDVI Case) – 14.7 psi

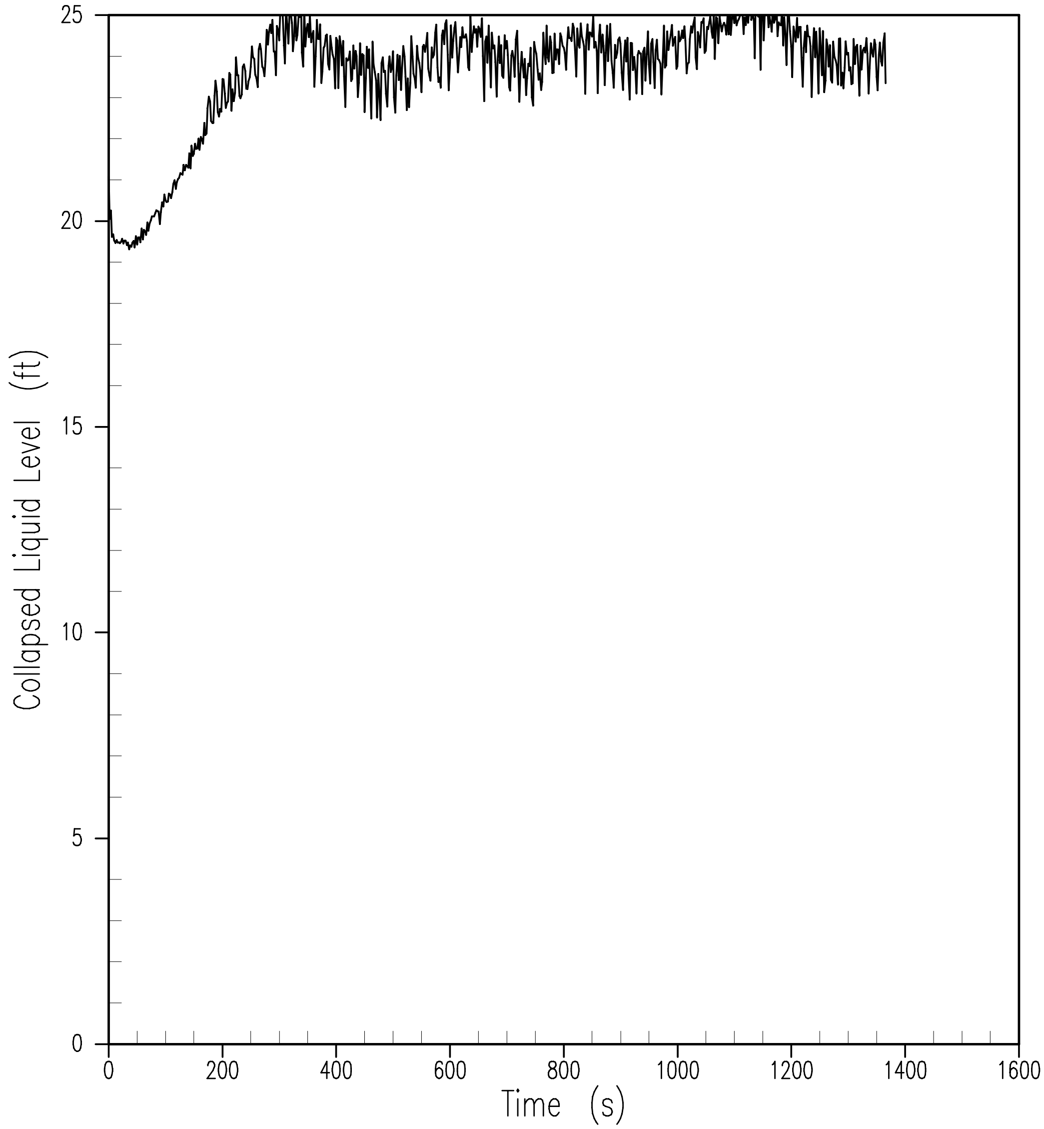

Figure 15.6.5.4C-15

Collapsed Level of Liquid in the Downcomer (Wall-to-Wall Floodup Case) – 14.7 psi

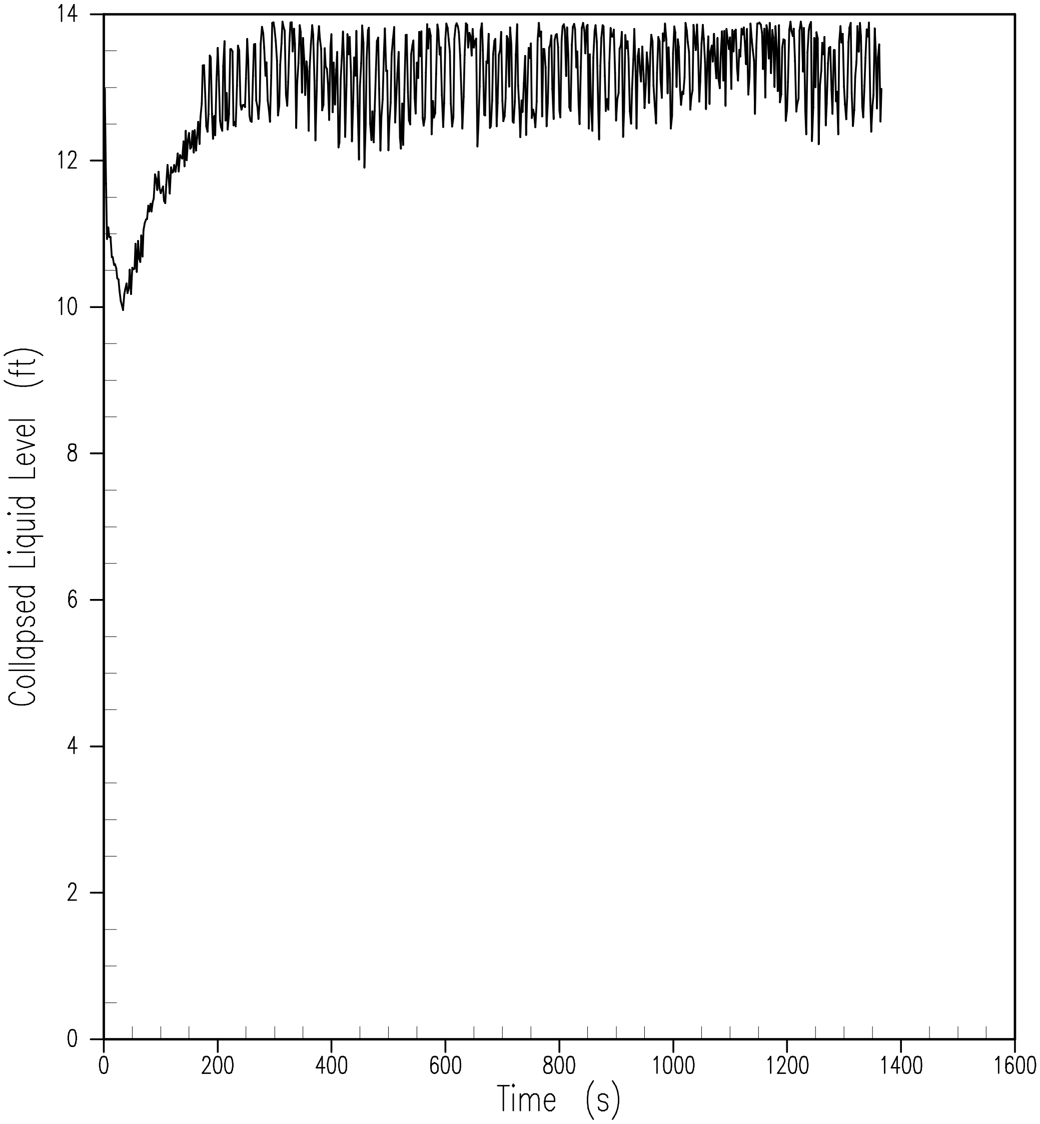

Figure 15.6.5.4C-16

Collapsed Level of Liquid Over the Heated Length of the Fuel (Wall-to-Wall Floodup Case) – 14.7 psi

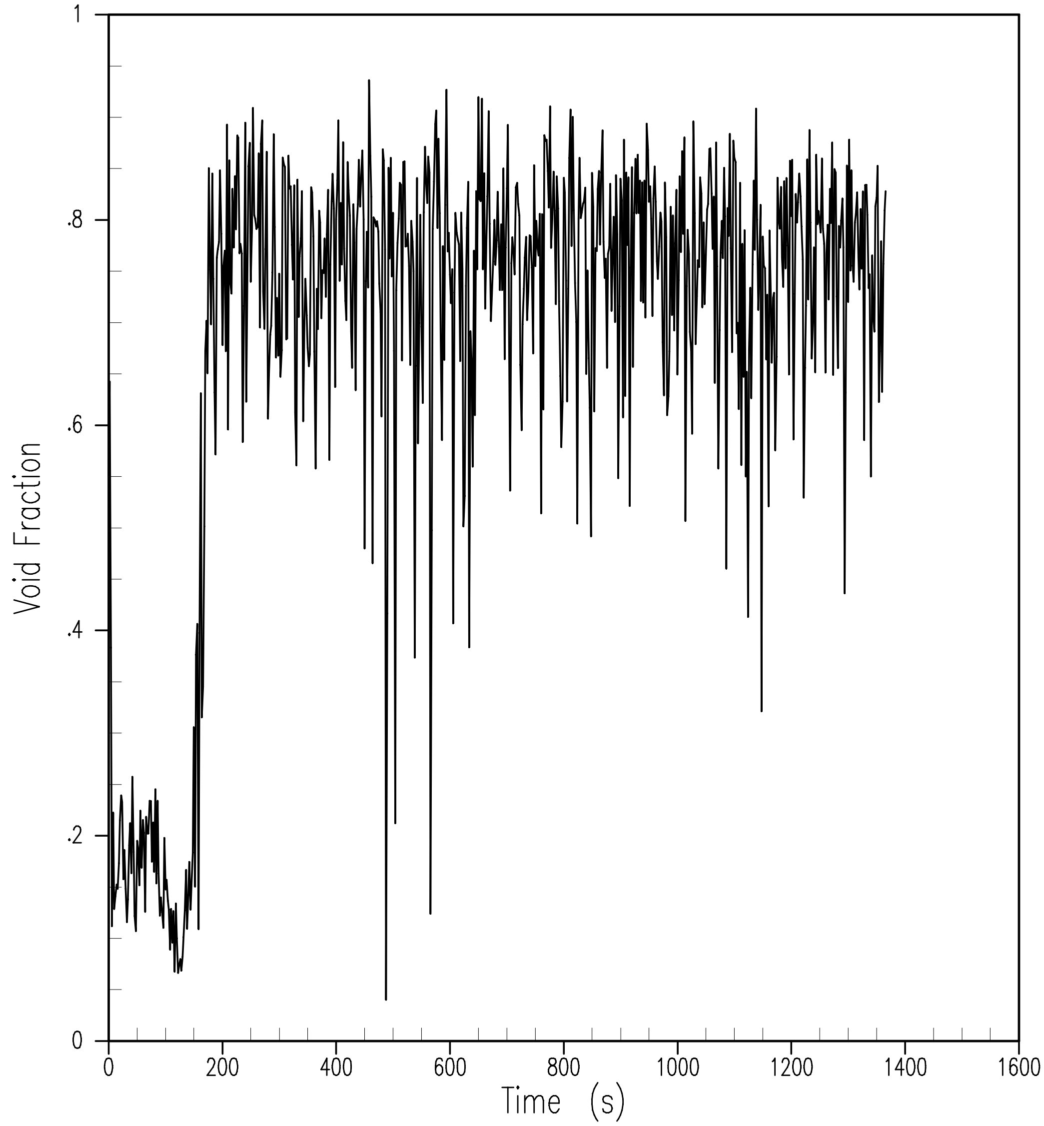

Figure 15.6.5.4C-17

Void Fraction in Core Hot Assembly Top Cell (Wall-to-Wall Floodup Case) – 14.7 psi

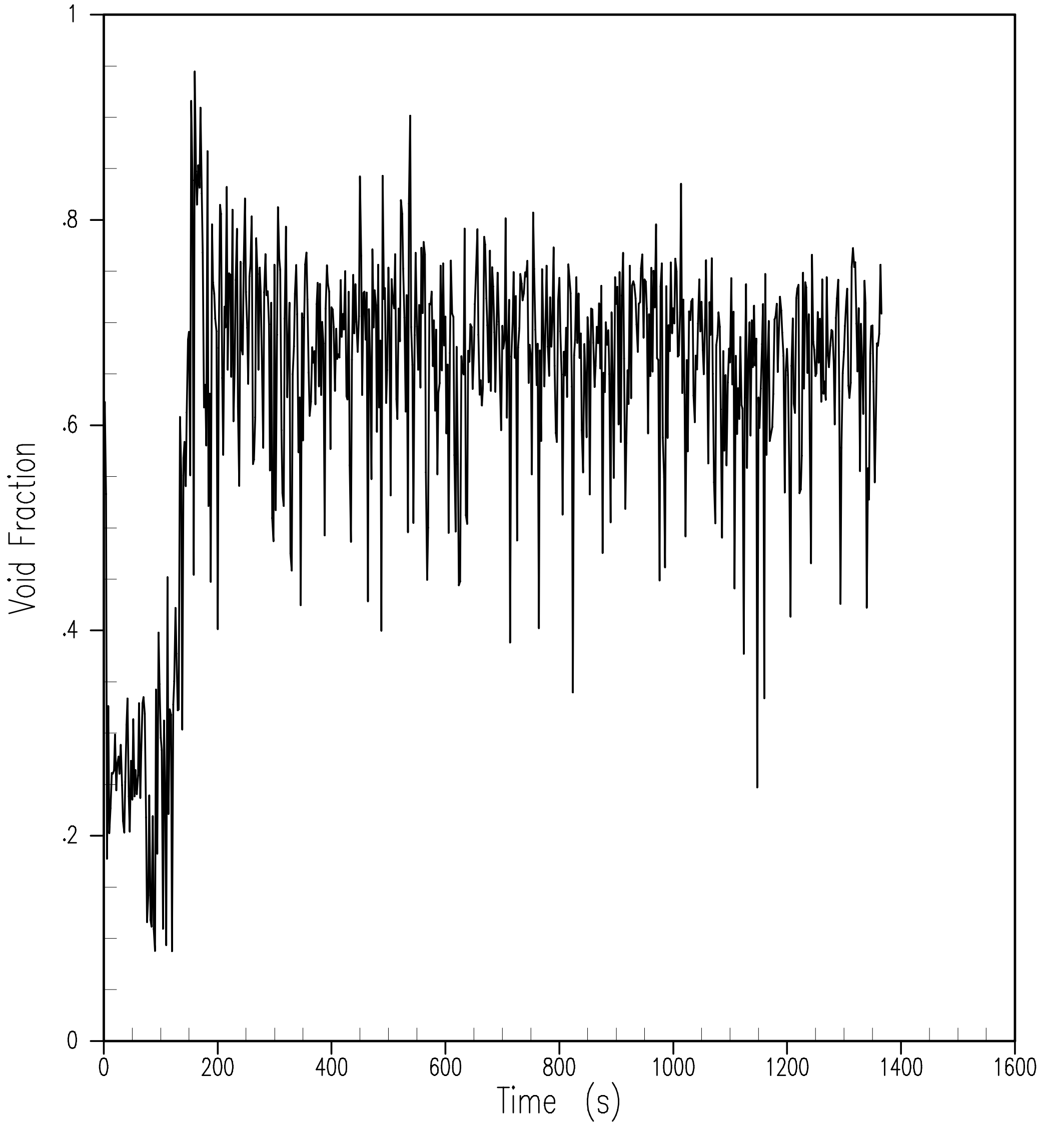

Figure 15.6.5.4C-18

Void Fraction in Core Hot Assembly Second from Top Cell (Wall-to-Wall Floodup Case) – 14.7 psi

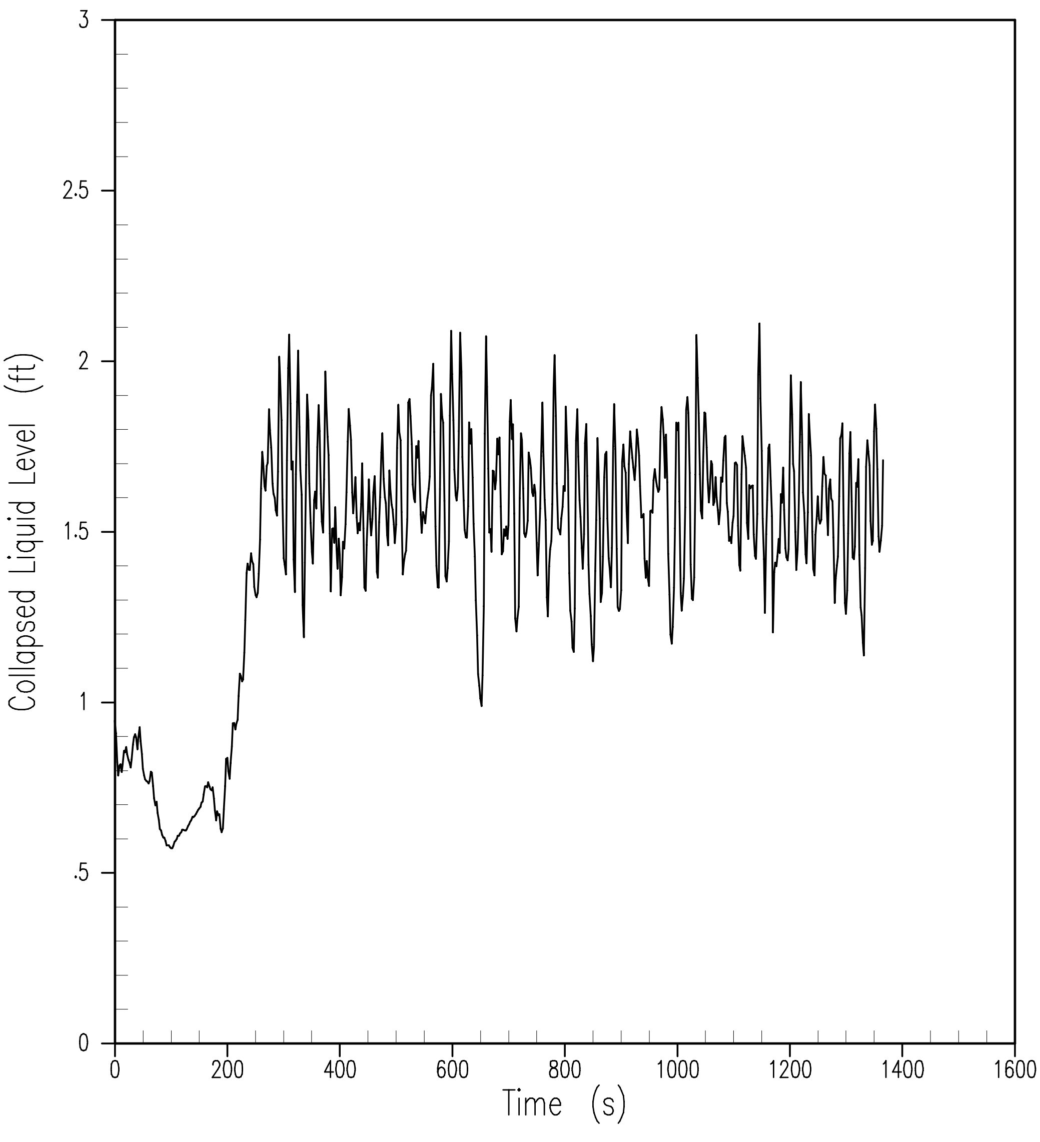

Figure 15.6.5.4C-19

Collapsed Liquid Level in the Hot Leg of Pressurizer Loop (Wall-to-Wall Floodup Case) – 14.7 psi

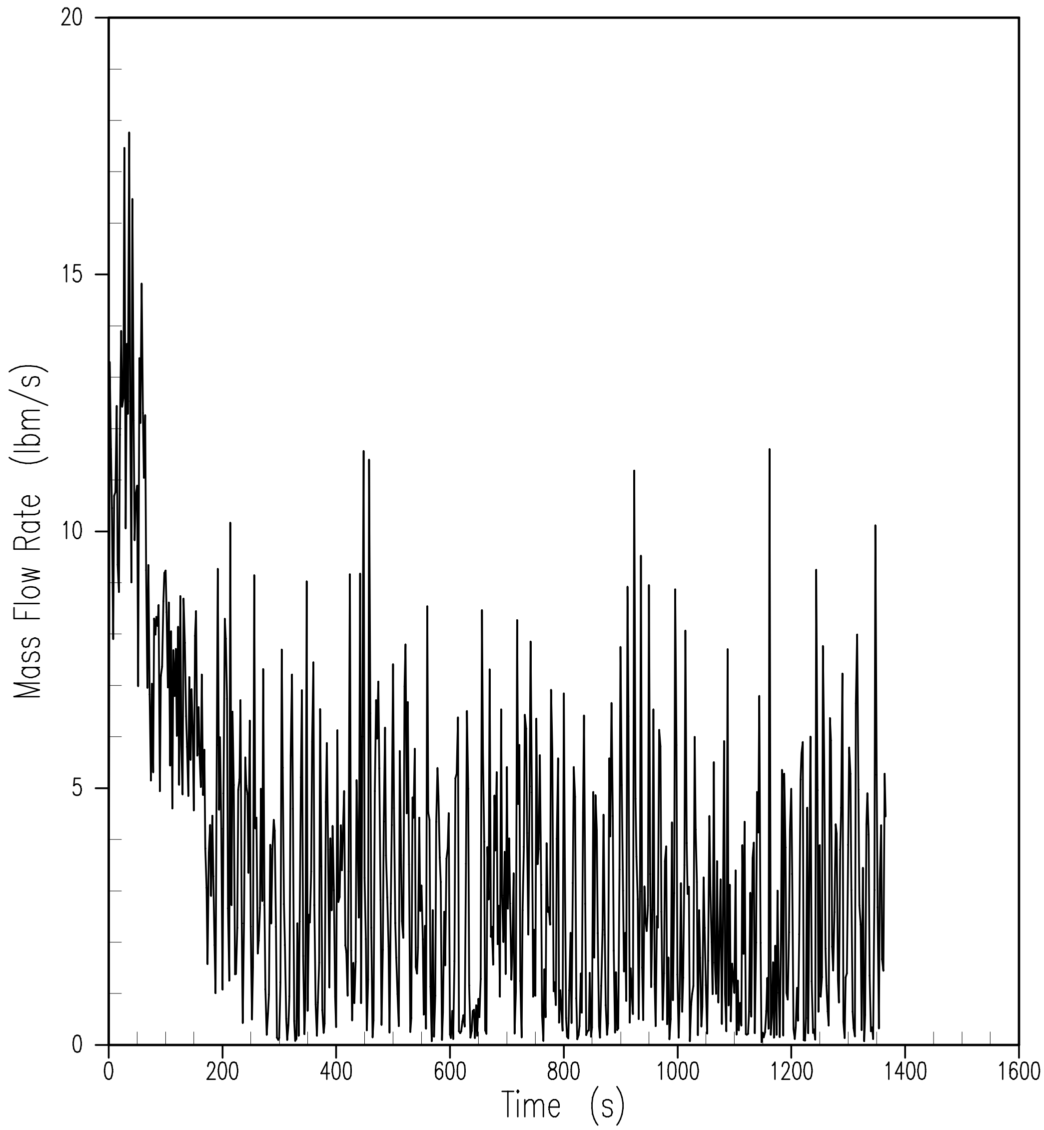

Figure 15.6.5.4C-20

Vapor Rate out of the Core
(Wall-to-Wall Floodup Case) – 14.7 psi

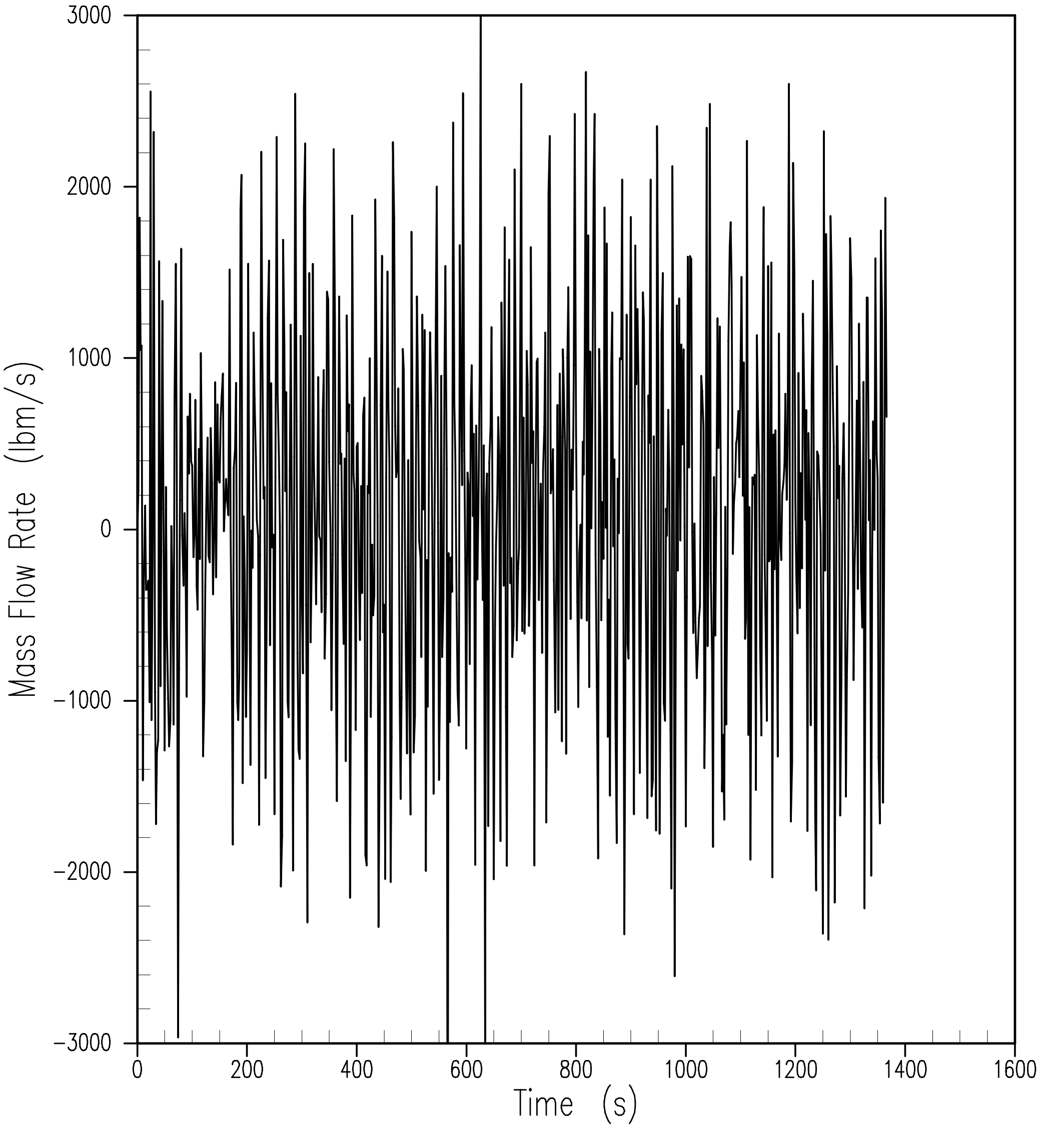

Figure 15.6.5.4C-21

Liquid Flow Rate out of the Core
(Wall-to-Wall Floodup Case) – 14.7 psi

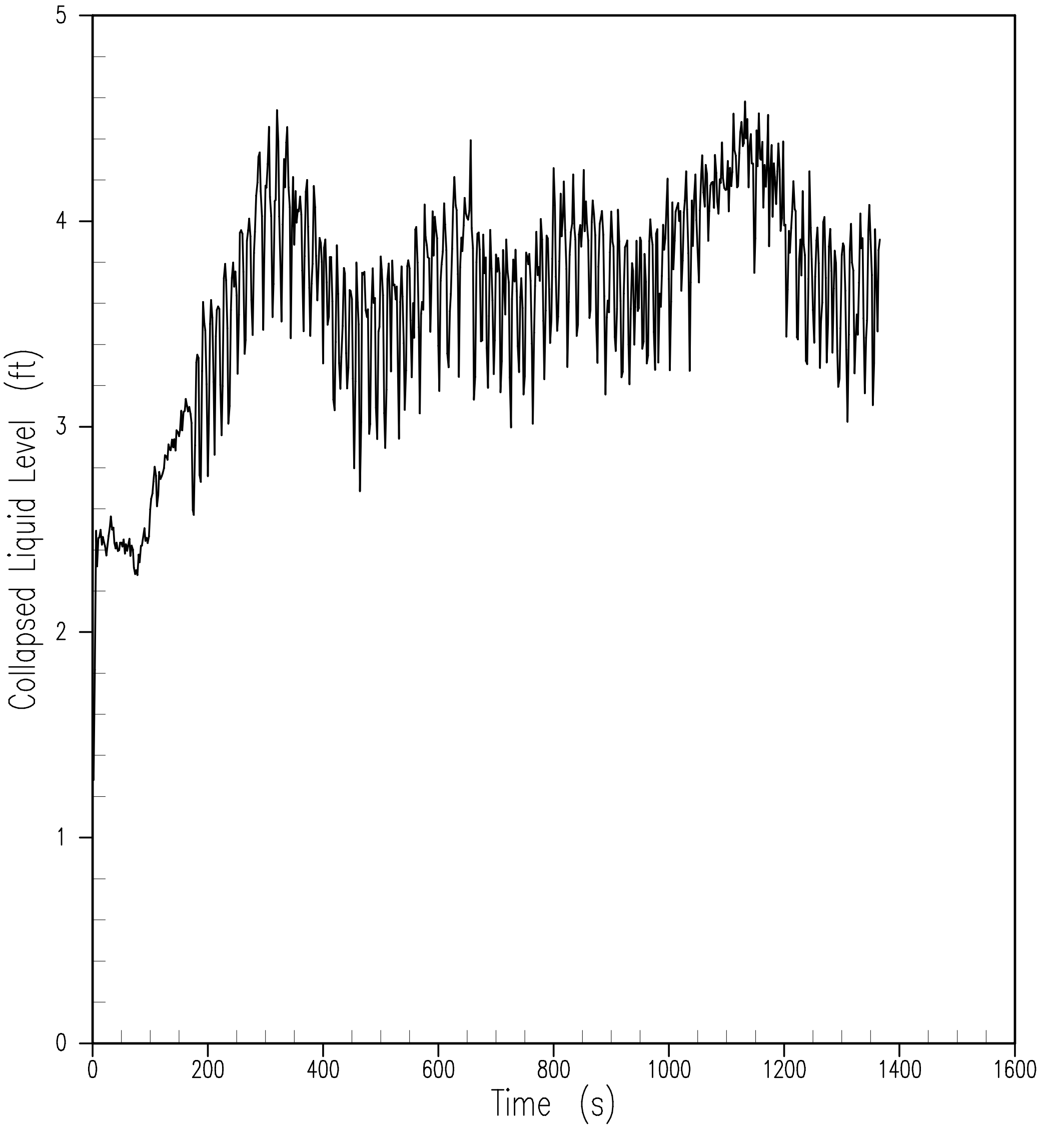

Figure 15.6.5.4C-22

Collapsed Liquid Level in the Upper Plenum (Wall-to-Wall Floodup Case) – 14.7 psi

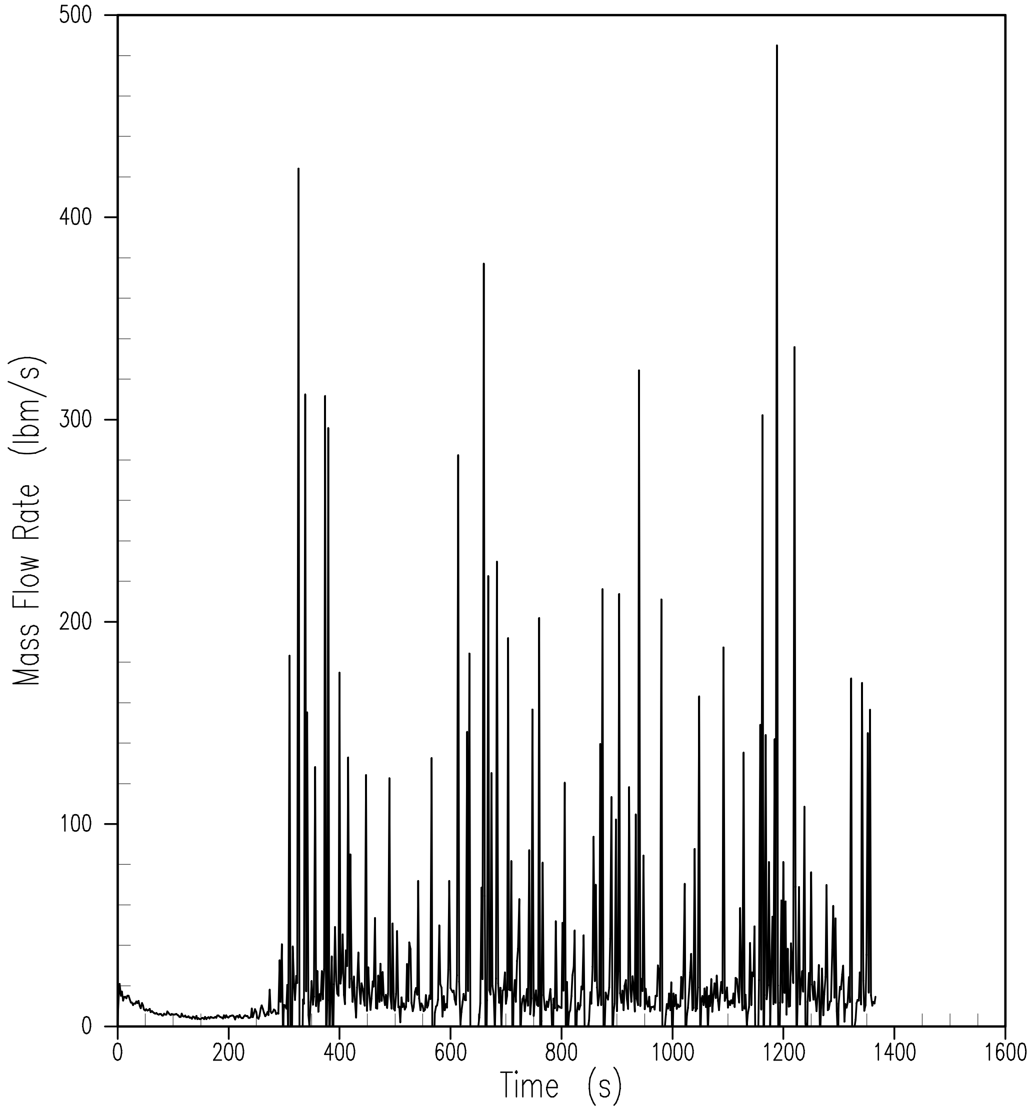

Figure 15.6.5.4C-23

Mixture Flow Rate Through ADS Stage 4A Valves (Wall-to-Wall Floodup Case) – 14.7 psi

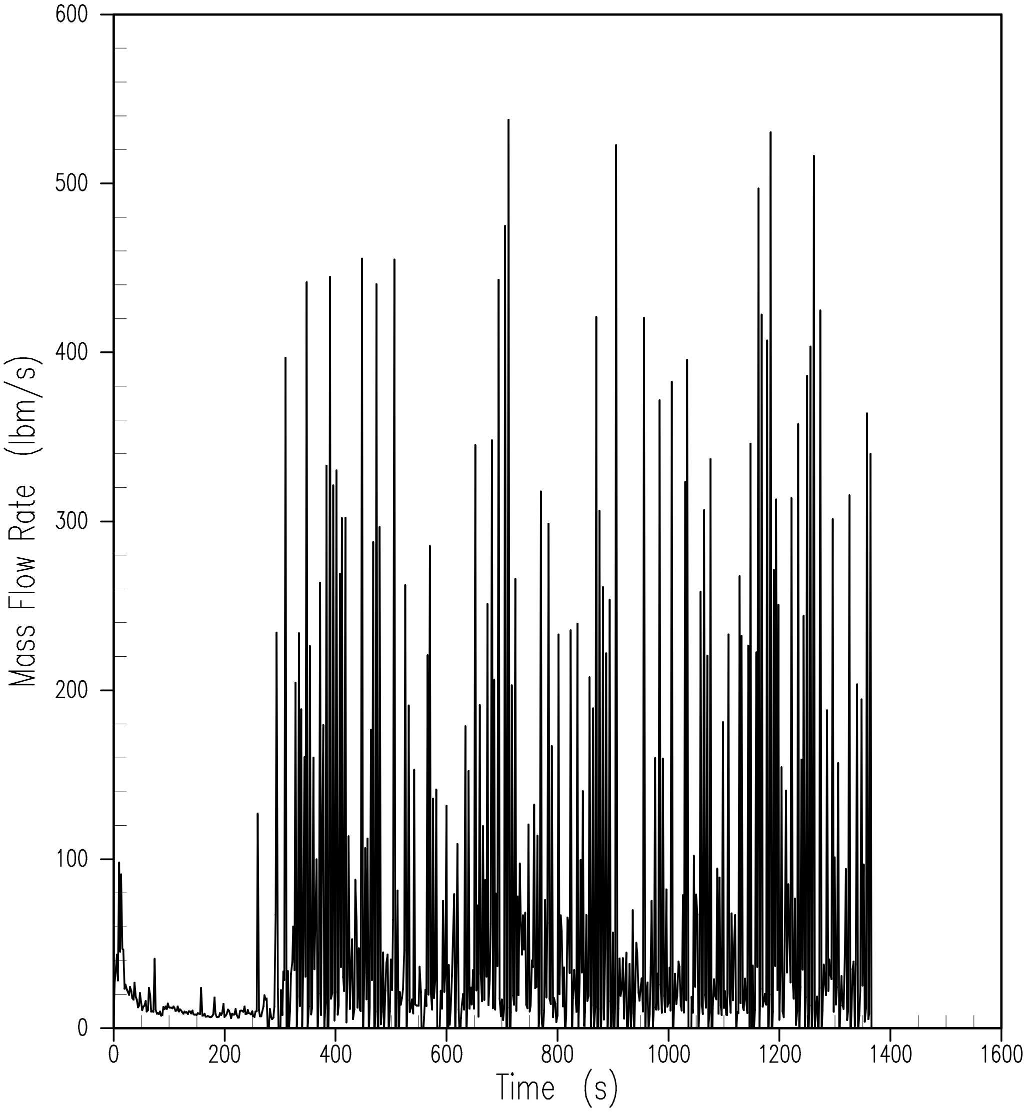

Figure 15.6.5.4C-24

Mixture Flow Rate Through ADS Stage 4B Valves (Wall-to-Wall Floodup Case) – 14.7 psi

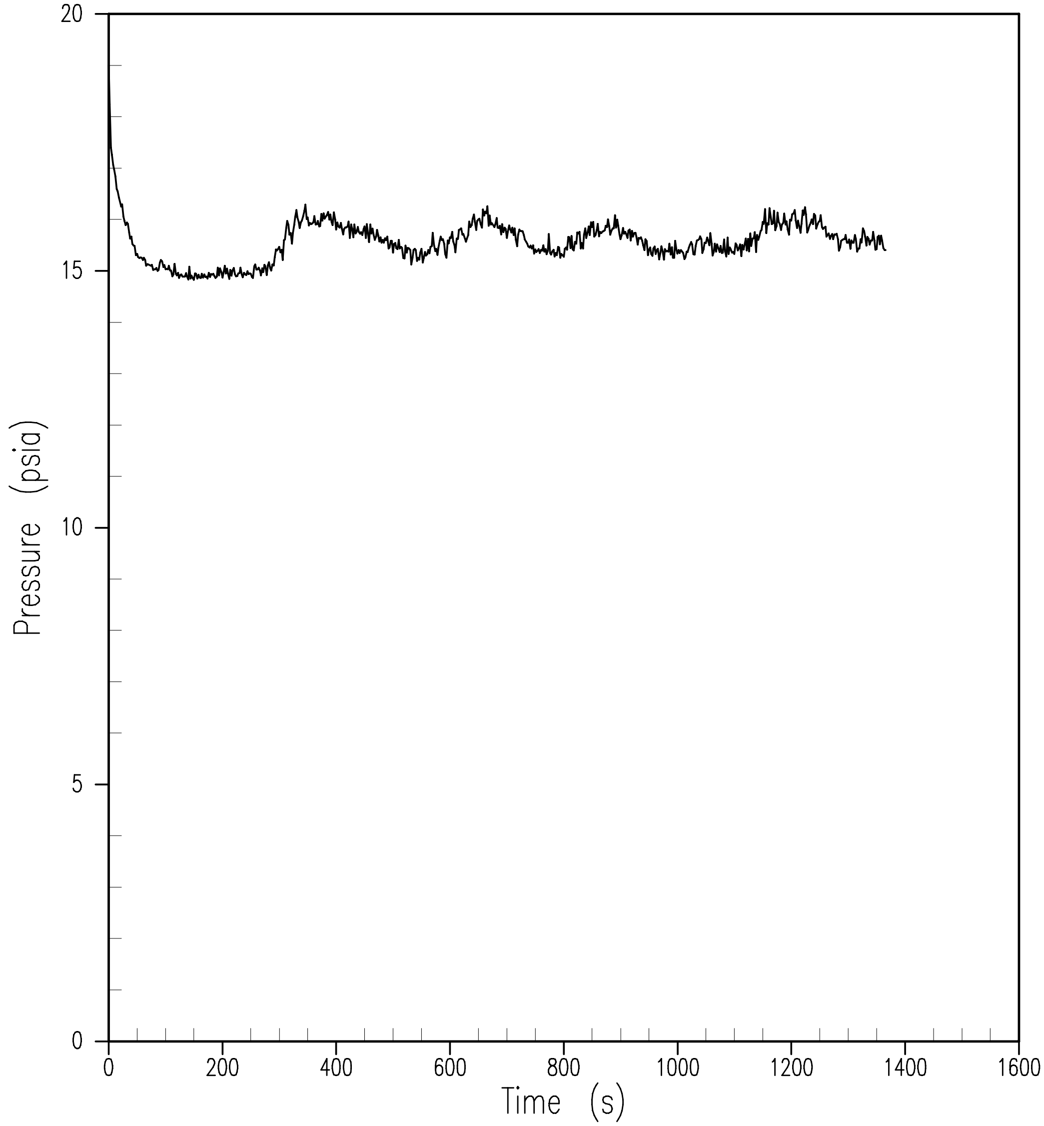

Figure 15.6.5.4C-25

Upper Plenum Pressure
(Wall-to-Wall Floodup Case) – 14.7 psi

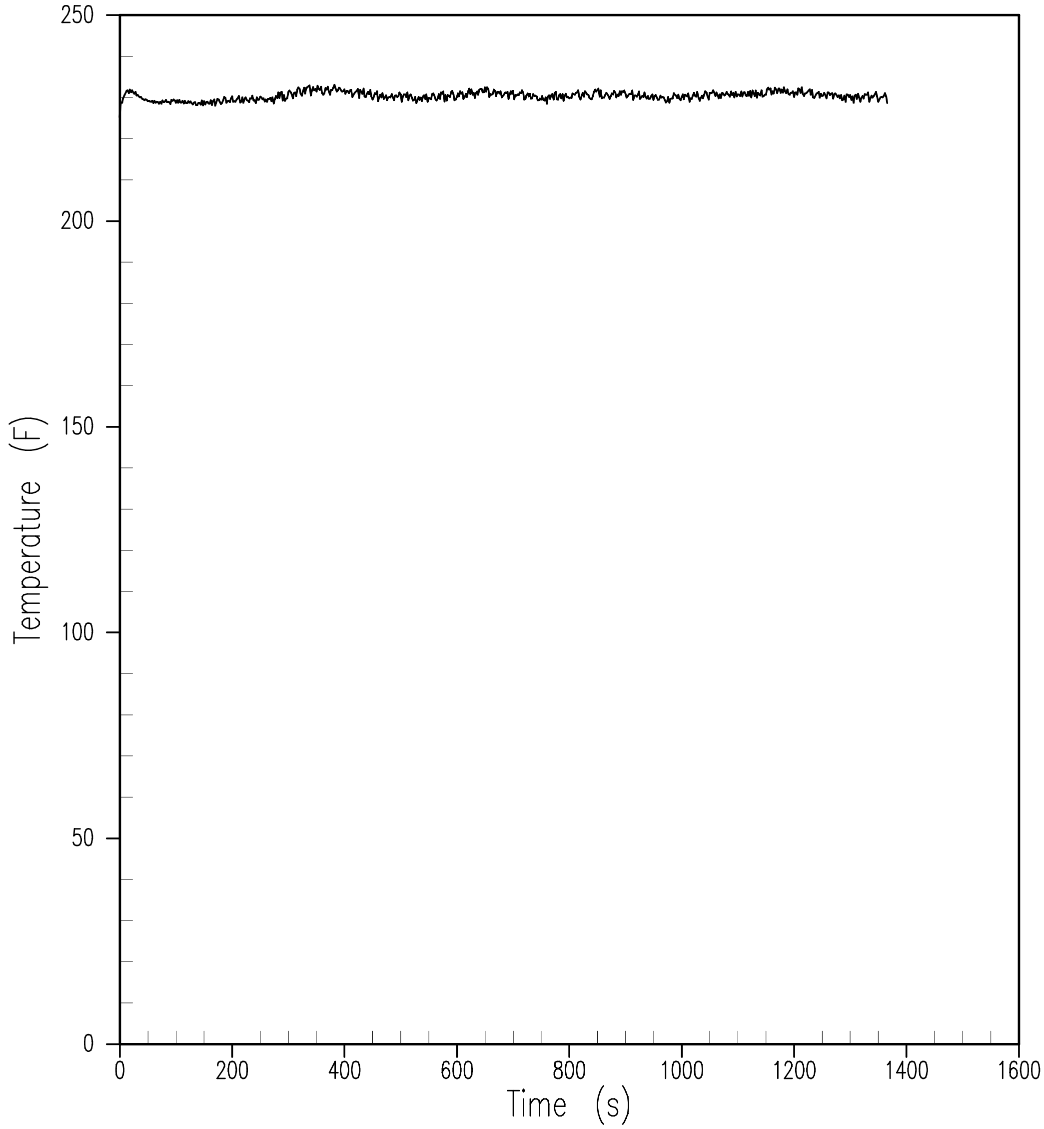

Figure 15.6.5.4C-26

Hot Rod Cladding Temperature Near Top of Core (Wall-to-Wall Floodup Case) – 14.7 psi

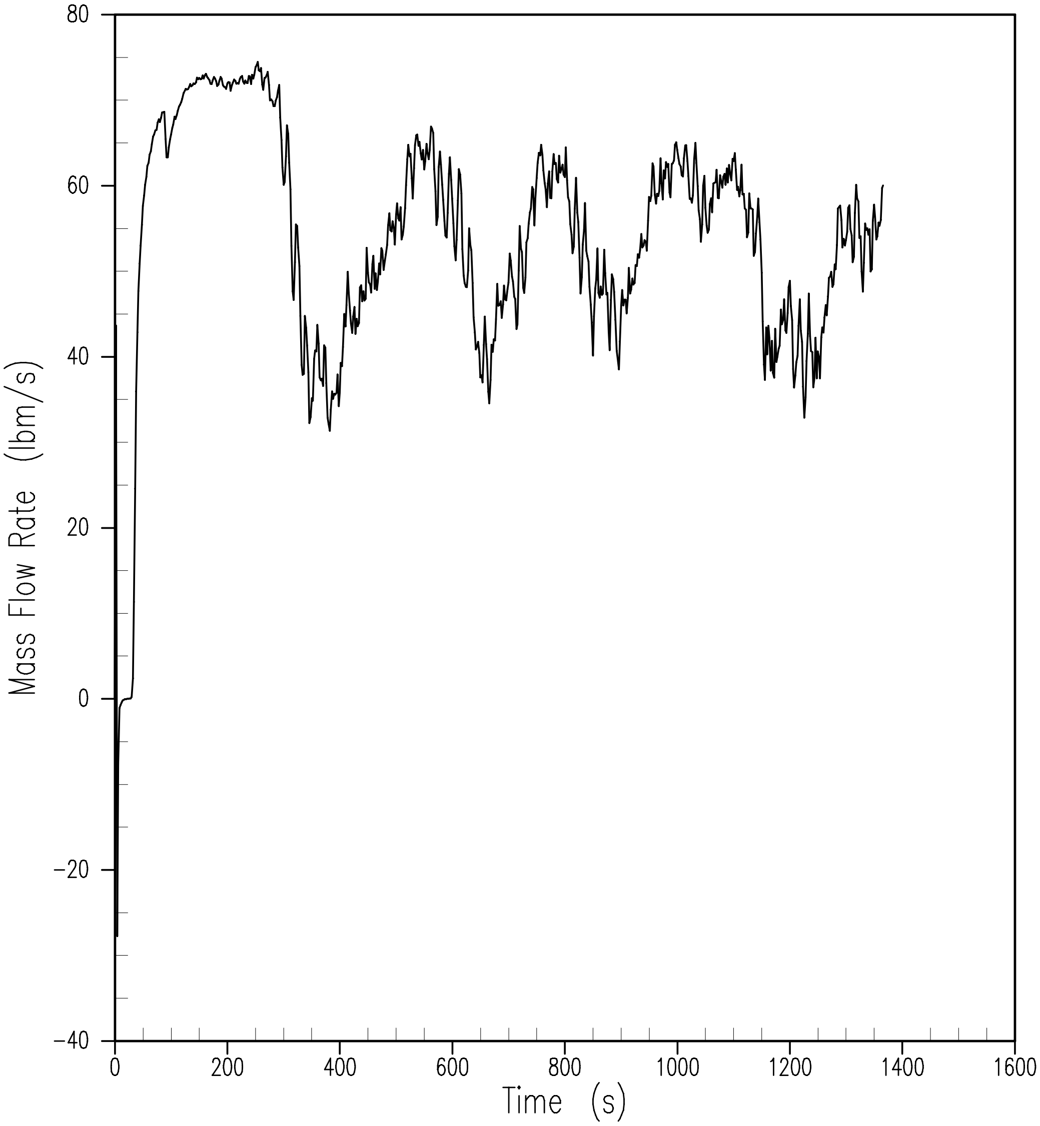

Figure 15.6.5.4C-27

DVI-A Mixture Flow Rate
(Wall-to-Wall Floodup Case) – 14.7 psi

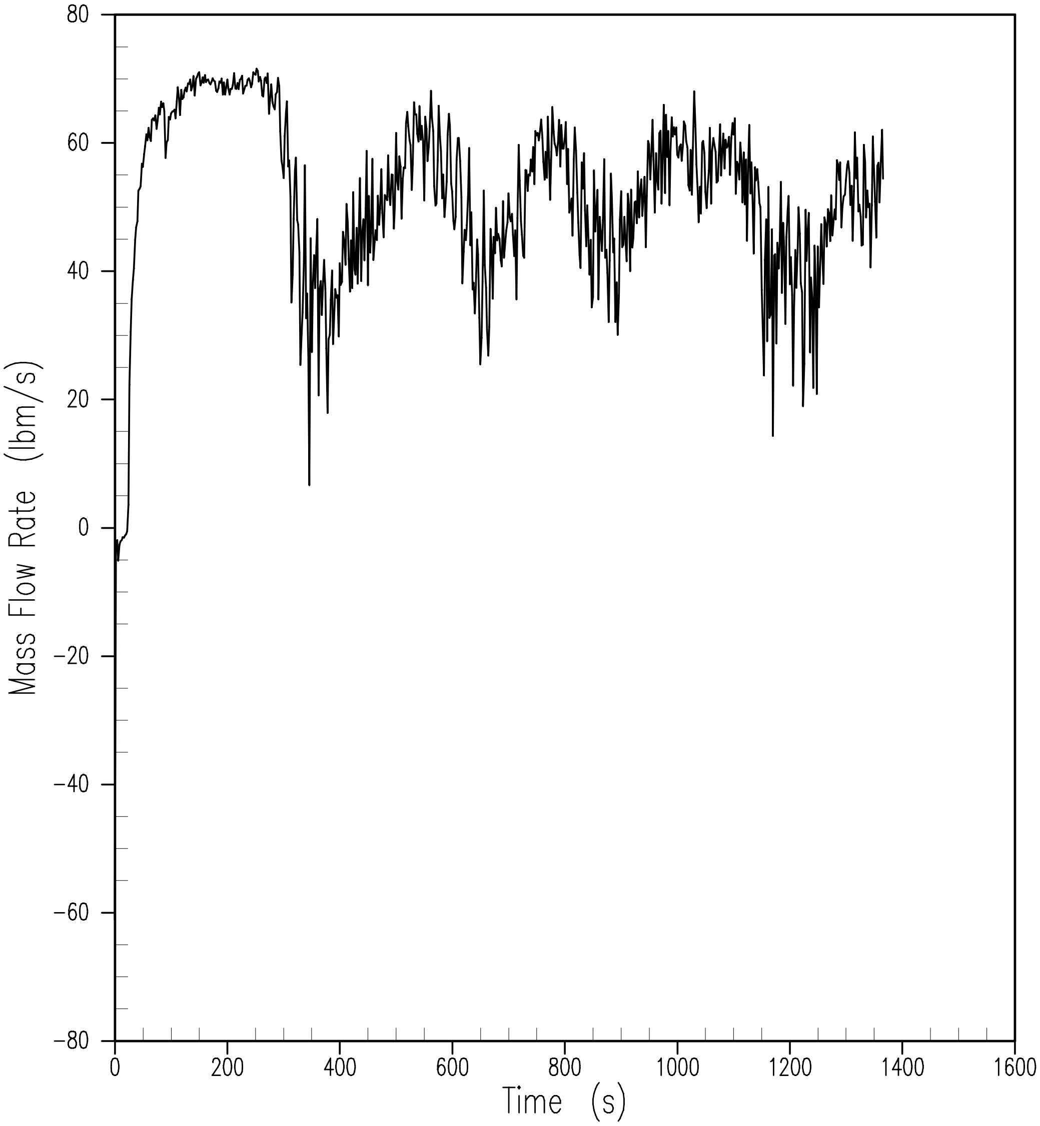

Figure 15.6.5.4C-28

DVI-B Mixture Flow Rate
(Wall-to-Wall Floodup Case) – 14.7 psi