15.6 Decrease in Reactor Coolant Inventory Events

Several anticipated operational occurrences (AOO) and postulated accident (PA) events cause a decrease in reactor coolant inventory. Detailed analyses of these reactor coolant inventory events are described in this section, including the following:

- Section 15.6.1 - Inadvertent opening of a pressurizer (PZR) safety relief valve (IOPSRV).
- Section 15.6.2 - Radiological consequences of the failure of small lines carrying primary coolant outside containment.
- Section 15.6.3 - Radiological consequences of a steam generator (SG) tube failure for a pressurized water reactor (PWR).
- Section 15.6.4 - Radiological consequences of main steam line failure outside containment for a boiling water reactor (BWR), which is not applicable to the U.S. EPR.
- Section 15.6.5 - Loss of coolant accidents (LOCA) resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (RCPB).

15.6.1 Inadvertent Opening of a Pressurizer Safety Relief Valve

15.6.1.1 Identification of Causes and Event Description

The PZR of the U.S. EPR has three relief lines, each with a single PZR safety relief valve (PSRV). In addition, a severe accident depressurization line with two manually operated valves in series is present. The severe accident valves are used only for the mitigation of beyond design basis severe accidents. Because opening of the severe accident valves requires two separate manual operations, inadvertent opening of these valves is precluded.

The IOPSRV event is defined as the spurious opening of a PSRV that is normally closed. During power operation, the opening or closing demand of a PSRV is hydraulic and valve-specific, so that a single failure can affect only one PSRV. Because the PSRVs serve combined functions of relief valves and safety valves, no block valves are present downstream to isolate the relief line. Thus, an IOPSRV is similar to a small-break loss of coolant accident (SBLOCA), described in Section 15.6.5, on the hot side of the reactor coolant system (RCS).

The IOPSRV causes a loss of reactor coolant inventory that cannot be offset by the chemical and volume control system (CVCS). This condition causes primary system depressurization and a decrease in reactor coolant density. In the early phase of the event, the reactor power is determined by reactivity feedback (moderator density) and the reaction by the rod position controller (automatic rod control system).

The reactor is tripped automatically on low PZR pressure by the protection system (PS). The reactor trip (RT) signal automatically trips the turbine and closes the main feedwater (MFW) high-load lines (HL) as described in Section 10.4.7. As secondary pressure increases, the turbine bypass valves open, permitting a steam dump to the main condenser. If the condenser is unavailable, as for a loss of offsite power (LOOP), the main steam relief trains (MSRT) open, permitting steam relief to the atmosphere.

Following RT, the SGs are fed by the MFW system (MFWS) through the low-load lines (LL). If the MFWS is unavailable, the startup and shutdown system automatically starts and feeds the SGs through the LL. If the startup and shutdown system is unavailable (as for LOOP), the emergency feedwater system (EFWS) is actuated on a low-level SG or safety injection (SI) signal in combination with LOOP.

RCS pressure continues to decrease throughout the transient. The PZR level increases initially due to the expansion caused by the depressurization and PSRV outflow. The reactor coolant pumps (RCP) continue to run unless there is a LOOP or until an RCP trip signal is generated on the combination of an SI signal and low-pressure differential across the pumps.

The SI signal is generated on very low PZR pressure and automatically starts the medium-head safety injection (MHSI) and low-head safety injection (LHSI) pumps. This signal also initiates a partial cooldown of the secondary system. The partial cooldown accelerates the depressurization of the primary system. MHSI injection causes recovery of RCS inventory, leading to a controlled state. For analysis purposes, the PSRV is assumed to remain open throughout the event.

At the controlled state, core cooling is provided by the safety injection system (SIS). Heat removal from the RCS is provided by continued leak flow through the open PSRV, the SGs, or both the PSRV and the SGs. The PSRVs are qualified to discharge water as well as steam. Following completion of the SG partial cooldown, the operator can initiate a continued cooldown. This action reduces the primary system temperature to the point at which the operator can depressurize the RCS, transition to long-term cooling with the residual heat removal system, and bring the reactor to a safe shutdown condition.

The IOPSRV event is considered an AOO as described in Table 15.0-1. The acceptance criteria for these events are described more fully in Section 15.0.0.2:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
- Fuel-cladding integrity is maintained if the minimum departure from nucleate boiling ratio (MDNBR) remains above the 95/95 DNBR limit.

- An AOO should not develop into a more serious plant condition without other faults occurring independently.

The focus for this event is meeting the specified acceptable fuel design limits (SAFDL).

15.6.1.2 Method of Analysis and Assumptions

The methodology used for this event analysis is described in the Codes and Methods Applicability Report for the U.S. EPR (Reference 1). It uses the S-RELAP5 computer code (described in Section 15.0.2) to calculate the transient thermal and hydraulic response of the primary and secondary systems. The code simulates the necessary components and has the properties necessary to model an IOPSRV event. The calculated transient boundary conditions for the reactor core from the S-RELAP5 analysis are used as input to the thermal margin calculations. The low DNB channel algorithm and the high LPD channel algorithm are simulated to predict RT and the adequacy of the dynamic compensation of the algorithm consistent with the Incore Trip Setpoint and Transient Methodology for U.S. EPR (Reference 2).

The primary criterion for the IOPSRV event is to maintain the fuel-cladding integrity by satisfying the SAFDL. Overpressurization is not an issue for this transient as this is a loss-of-coolant event.

The IOPSRV event analysis considers several cases examining the range of conditions specified in Table 15.0-5 to identify the limiting case. The range of conditions important for the IOPSRV event include time in life (beginning-of-cycle (BOC) versus end-of-cycle (EOC) fuel conditions), LOOP assumption, rod control configuration (manual or automatic), and SG tube plugging level (0 versus 5 percent). Additionally, uncertainties in the pressurizer safety relief valve (PSRV) flow rate and decay heat are considered in the analysis of the IOPSRV event.

The limiting event and initial conditions for the IOPSRV cases are summarized in Table 15.0-62 and Table 15.0-63, respectively. The limiting event is identified by performing a spectrum of calculations that consider various operating parameters and applying single failure and preventive maintenance assumptions that would make the transient worse.

In identifying the limiting scenario, the loss of primary mass through the PSRV is maximized by considering the following conditions:

- The assumption of LOOP on RT causes the RCPs to coastdown. The subsequent degradation of primary-to-secondary heat transfer results in a PZR insurge, which contributes to the mass expelled out the PSRV.

- The single failure requirement for this analysis is satisfied by assuming the failure of one emergency diesel generator, which conservatively removes one train of pumped SIS (including one EFW pump).

- The preventive maintenance assumption removes another train of pumped SIS (including another EFW pump).
- The assumed BOC fuel conditions for the limiting case provides the most positive reactivity feedback during the initial stage of the transient, thus challenging the DNB aspect of the event.

There is no single failure or preventive maintenance assumption other than the above described loss of diesel generator that would have a worse impact on the IOPSRV event.

The limiting case is identified from a set of calculations that include BOC and EOC fuel conditions (with and without automatic rod control) and the availability of the CVCS. Sensitivity calculations are performed to bound the uncertainties in the PSRV flow rate and decay heat.

Description and results of the limiting case are presented in Section 15.6.1.3

Table 15.6-1—IOPSRV Event - Key Input Parameters presents the initial conditions for the limiting case. Table 15.6-2—IOPSRV Event - Key Equipment Status presents the status of mitigating equipment and components. The analysis begins at full power, under normal operating conditions. To minimize the heat removal by the secondary system, the maximum number of plugged SG tubes (five percent) is assumed.

The most reactive control rod is assumed not to insert at RT. LOOP is assumed to occur with RT. Subsequent to an RT, the limiting single failure is taken as the failure of one emergency diesel generator (EDG), resulting in the unavailability of one train of pumped SIS (MHSI, LHSI, and EFWS). A second EDG is assumed to be under maintenance and therefore unavailable, causing a second train of pumped SIS to be unavailable.

Degraded conditions are assumed for the MHSI pump startup and flow rates to produce the most conservative emergency core cooling system (ECCS) response. Degraded containment conditions are also assumed so that the actuation setpoints of mitigating systems use the largest instrument uncertainties.

Operator actions are credited at 30 minutes into the event to align EFWS flow from the two operational trains of EFWS to the four SGs. Later, operator actions are necessary to transition the plant from a controlled state to a safe shutdown condition.

The limiting case uses BOC fuel conditions and assumes the rod position controller is in manual mode. At the BOC, the boron concentration is at its highest. A decrease in density following the IOPSRV results in a decrease in boron concentration. The resulting positive reactivity feedback causes a power increase in the early phase of the event.

End-of-cycle (EOC) fuel conditions are considered in a sensitivity calculation with the assumption the rod position controller is in automatic mode. At EOC, a decrease in density causes negative reactivity feedback because the boron concentration is lower. The rod position controller responds to the core average temperature and turbine generator demand. These parameters do not change rapidly. The net effect is that a decrease in reactor power occurs prior to reaching the RT signal, and this case is less limiting compared to the base BOC case.

Sensitivity studies were also conducted to bound uncertainties in PSRV flow rate (at 20 percent) and core decay heat (at 20 percent). These uncertainties are taken into account in the limiting case used for the thermal-hydraulic DNB analysis. Both uncertainties are included in the limiting case presented.

15.6.1.3 Results

Table 15.6-3—IOPSRV Event - Sequence of Events presents the sequence of events for this case. Figure 15.6-7—IOPSRV Event - Pressurizer Level presents the PZR level after the PSRV opens. After the PSRV opens, reactor power increases slightly prior to RT at 39 seconds (Figure 15.6-1—IOPSRV Event - Transient Reactor Power). The increase in reactor power causes a small increase in core average heat flux (Figure 15.6-6—IOPSRV Event - Core Average Heat Flux). The primary pressure decreases throughout most of the event (Figure 15.6-2—IOPSRV Event - PZR Pressure). The core inlet temperature is stable prior to the RT (Figure 15.6-8—IOPSRV Event - Core Inlet Temperature). Figure 15.6-93—IOPSRV Event-Representative Plot of Normalized Minimum DNBR and Maximum LPD Normalized to the SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

The DNB reactor trip (RT) and high LPD RT setpoints, as well as the dynamic compensation built into the low DNBR channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, the DNB limiting condition for operation (LCO) and the LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and the peak centerline temperatures remain below the fuel centerline melt limit. Figure 15.6-8 through Figure 15.6-10 show the effect of the partial cooldown initiated by the low PZR pressure SI signal. The controlled decrease in SG pressure causes a corresponding cooldown of the core inlet temperature and RCS average temperature.

MHSI injection rate offsets the PSRV discharge rate at about 1100 seconds (Figure 15.6-3—IOPSRV Event - MHSI and PSRV Flow Rates). The reactor vessel fluid mass inventory is shown in Figure 15.6-4—IOPSRV Event - Reactor Vessel Fluid Mass. The core exit void fraction does not exceed approximately 40 percent, indicating that the core remains adequately cooled throughout the transient (Figure 15.6-5—IOPSRV Event - Core Exit Void Fraction).

15.6.1.4 Radiological Consequences

Fuel or cladding damage is not predicted for an IOPSRV event; therefore, radiological consequences are not calculated for this event.

15.6.1.5 Conclusions

The results of the analysis show that the PS provides an early RT to preclude fuel or cladding damage. In the later phase, two MHSI pumps are able to offset the loss of primary inventory through the stuck-open PSRV. The core remains adequately cooled throughout the transient.

For an IOPSRV event, the primary acceptance criterion is that the fuel cladding integrity is maintained. In the early phase of the event, prior to RT, this criterion is met by maintaining the MDNBR above the acceptable fuel design limit. After RT, the criterion is met by recovery of vessel inventory prior to significant voiding in the core. This analysis demonstrates that the criterion is satisfied both before and after RT.

15.6.1.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.6.1 events included in NUREG-0800, Section 15.6.1, (Reference 4), and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

   – Response: The primary pressure decreases throughout most of the event (see Figure 15.6-2) remaining well below the design value. Secondary pressure is also well controlled and remains below the design value (Figure 15.6-10).

2. Fuel cladding integrity is maintained by keeping the MDNBR above the 95/95 DNBR limit.

   – Response: As noted in Section 15.6.1.3, the actions of the low DNB RT and DNB LCO prevent violation of the MDNBR SAFDL prior to RT. After RT, maintenance of fuel cladding integrity is accomplished by recovery of vessel inventory prior to significant voiding in the core.

3. An AOO should not develop into a more serious plant condition without other faults occurring independently.

   – Response: As noted in Section 15.6.1.5, the results of the analysis show that the RPS provides an early RT to preclude fuel or cladding damage. In the later phase, two MHSI pumps are able to compensate for the loss of primary inventory through the stuck-open PSRV. The core remains adequately cooled throughout the transient. Thus, the event does not evolve into a more serious plant condition without other faults occurring independently.

15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

The postulated failures of small lines carrying primary coolant outside containment are analyzed as nonmechanistically initiated events that are evaluated for radiological consequences. The evaluation considers the rupture of small lines in the nuclear sampling system (NSS) and in the CVCS. The U.S. EPR design has no instrument lines, that carry primary coolant outside of containment. Small breaks of other sizes evaluated either lead to an automatic isolation or the release of a smaller RCS inventory. Because no operator action is credited initially, reactor coolant is assumed to discharge outside containment for 30 minutes. The radiological consequences of these events are addressed in Section 15.0.3.5.

15.6.3 Steam Generator Tube Failure (PWR)

15.6.3.1 Identification of Causes and Accident Description

The SG tube rupture (SGTR) event is defined as the double-ended rupture of a single SG tube and is a PA (see Section 15.0 for event categorization). The main acceptance criterion for this event is to maintain the radiological releases below acceptable limits. A secondary criterion is to prevent overfill of the SG secondary to prevent water entering the steam lines.

The tube rupture is postulated to occur in the shortest SG tube, near the tube sheet location, to maximize break flow. Primary coolant from the RCS begins to enter the secondary system, driven by the pressure differential between the RCS and the secondary side of the SG. The inventory, pressure, and activity in the affected SG increase.

The break flow begins to depressurize the RCS and decrease the PZR level. The CVCS charging pumps inject water into the cold legs to maintain PZR level. On the secondary side, the MFW flow to the affected SG reduces in response to the SG level increase.

Radiation monitors located in the steam lines and blowdown lines detect increased activity soon after the break occurrence and identify the affected SG. Although high activity in a steam line (or high SG level) in combination with the initiation of partial cooldown isolates the affected SG, this function is not credited in the SGTR analysis. Other indications to the operator include the mismatch between feed flow and steam flow and the increased activity in the blowdown line of the affected SG.

If one charging pump cannot keep up with the break flow and the PZR level continues to decrease, a second charging pump (normally on standby) is automatically started on low-PZR level. The letdown flow is automatically reduced to its minimum value in response to the decreasing level. The charging pumps take suction from the volume control tank. The pumps are automatically switched to the in-containment refueling water storage tank (IRWST) on low level in the volume control tank. The combined charging pumps are able to offset the coolant loss through a single tube rupture. The operator trips the reactor before the RCS pressure decreases sufficiently to trigger an automatic RT.

If the charging pumps are not available, an automatic RT on low PZR pressure, high SG pressure or high-PZR pressure occurs, depending on the effect of break flow on ruptured SG pressure and reactivity feedback. The PZR heaters are de-energized as PZR level continues to decrease.

The following section describes the analysis of two event scenarios: charging pumps not operating and charging pumps operating.

15.6.3.1.1 Scenario 1 - Charging Pumps Are Not Operating

Without the charging pumps to offset break flow, the reactor trips automatically on low PZR pressure, high SG pressure, or high PZR pressure. This, in turn, trips the turbine and switches MFW flow to the LL. If offsite power is available, the turbine bypass system valves open to dump steam to the condenser. In this case, the radiological pathway is via the condenser as described in Section 15.0.3.

If there is LOOP, which is assumed coincident with turbine trip, the turbine bypass system is blocked automatically to protect the condenser. SG pressure increases to open the MSRTs, which discharge steam to the atmosphere. On the primary side, RCPs lose electrical power and begin coasting down. The EDGs are started and loaded on the de-energized buses.

RCS pressure continues to decrease due to the continued leak through the ruptured SG tube. At the low-low PZR setpoint, the SIS is actuated. This automatically initiates the following actions:

- Partial cooldown of the secondary system in the SGs using the MSRTs to depressurize at a rate corresponding to 180°F/h to 870 psia.
- Starting of the MHSI and LHSI pumps.
- Isolation of the RCS pressure boundary by isolating the CVCS charging and letdown lines.

In combination with LOOP, the SIS signal also automatically starts the EFWS, which subsequently automatically isolates the SG blowdown lines. If not already initiated automatically by the combination of high activity or high SG level in combination with a partial cooldown, the operator isolates the affected SG. To isolate the SG, the operator closes its main steam isolation valve (MSIV), resets its MSRT setpoint high, and closes its MFW and EFWS isolation valves. This action terminates the radiological release from the affected SG.

As the RCS pressure continues to decrease, the loss of coolant is terminated as the pressure difference across the ruptured SG tube decreases to zero. MHSI flow starts when the RCS pressure further decreases below the pump shutoff head restoring RCS inventory. This condition leads to a controlled state.

15.6.3.1.2 Scenario 2 - Charging Pumps Are Operating

With the charging pumps available to offset the break flow, the PS does not detect the loss of coolant. In this case, the operator trips the reactor once the event is detected. The RT, in turn, automatically trips the turbine, and switches MFW flow to the LL.

If offsite power is available, the turbine bypass system valves open to dump steam to the condenser. In this case, the radiological pathway is via the condenser as described in Section 15.0.3.

If LOOP occurs, which is assumed coincident with turbine trip, the turbine bypass system is blocked automatically to protect the condenser. SG pressure increases to open the MSRTs, which discharge steam to the atmosphere. On the primary side, RCPs lose electrical power and begin coasting down. The EDGs are started and loaded on the de-energized buses. The charging pumps are loaded on the SBO diesel generators, but not restarted automatically.

The operator institutes the following SGTR mitigation procedure:

- Close the MSIV in the affected SG to isolate the affected SG.
- Reset the MSRT setpoint high in the affected SG.
- Close the MFW and EFWS isolation valves in the affected SG.
- Start EFWS pumps.
- Initiate partial cooldown in the unaffected SGs, in which the MSRTs depressurize at a rate corresponding to 180°F/h to 870 psia.
- Close the CVCS isolation valves to isolate the charging and letdown lines.
- Start the MHSI pumps.

These actions effectively isolate the affected SG, terminating any radiological release. As the RCS pressure continues to decrease, the loss of coolant is terminated as the pressure difference across the ruptured SG tube decreases to zero. MHSI flow starts when the RCS pressure falls below the pump shutoff head restoring RCS inventory. This condition leads to a controlled state.

Regardless of initiating scenario, continued mitigation of this event is accomplished by managing the pressure difference across the ruptured SG tube, so that radiological releases are maintained below acceptable limits, and the affected SG does not overfill. EBS is initiated to provide adequate boration to prevent recriticality. The cooldown and depressurization of the RCS leads to the entry conditions for the residual heat removal (RHR) system to be put into operation. RHR operation takes the plant to shutdown conditions.

15.6.3.2 Method of Analysis and Assumptions

The methodology used to analyze this event is described in Codes and Methods Topical Report (Reference 1), and uses the S-RELAP5 computer code (described in Section 15.0.2.5) to calculate the transient thermal and hydraulic response of the primary and secondary systems. The S-RELAP5 system model includes the necessary components and contains the features necessary to simulate this event.

The primary concern for the steam generator tube rupture (SGTR) event is to maintain the radiological releases below the acceptable limits. The secondary criterion of avoiding overfill of the affected SG secondary (i.e., to prevent water from entering the steam lines) is also evaluated.

The SGTR analysis presented considers several cases examining the range of conditions specified in Table 15.0-5 to identify the limiting event scenario. The range of conditions important for the SGTR event include time in life (BOC versus EOC fuel conditions), LOOP assumption, SG tube plugging level (0 versus 5 percent), availability of the CVCS system (charging pumps), and the assumed initial coolant temperature (the nominal 594°F versus 584°F corresponding to coastdown at EOC conditions). Operator intervention is a key factor in mitigating the SGTR event. Thus, various combinations of timings and sequence of operator actions are considered in the analysis of the event.

The limiting event and initial conditions for these cases are summarized in Table 15.0-62 and Table 15.0-63, respectively. The limiting radiological and overfill cases are identified by performing a spectrum of calculations that consider the biasing of various operating parameters and applying single failure and preventive maintenance assumptions that would make the transient worse.

The radiological consequences of the event are maximized by imposing the following conditions for the limiting dose case:

- The assumption of LOOP, coincident with turbine trip, renders the turbine bypass system unavailable; thus, forcing the activity in the affected SG to be released to the atmosphere through the MSRT.
- Single failure of the affected SG main steam relief control valve (MSRCV) to stick fully open maximizes the dose consequence of the event (the main steam relief isolation valve (MSRIV) closes automatically on low SG pressure).
- The assumption of affected SG EFW pump to be in preventive maintenance maximizes flashing of the break flow which contributes to the severity of the released dose to the atmosphere.
- The lower initial SG secondary pressure associated with the use of the lowest allowed primary coolant average temperature at full power (584°F) results in slightly higher integrated flashed mass.

The limiting case for the SG overfill scenario is characterized by the following initial condition biasing and plant configuration assumptions:

- The assumption of LOOP in combination with the SIS signal automatically starts the EFW pumps. The case with LOOP is determined to be most limiting as this scenario results in the most integrated break flow, maximizing the ruptured SG inventory.
- Single failure assumption of the affected SG EFW control valve failing in the open position and the SG level increasing past the EFW shutoff setpoint maximizes the overfilling of the ruptured SG.
- The EFW pump feeding one of the intact SGs is assumed to be in preventive maintenance, which minimizes heat transfer to the secondary system and serves to maximize primary pressure, subsequently contributing to higher break flow.
- As with the radiological case, the lower initial SG secondary pressure associated with the use of the lowest allowed primary coolant average temperature at full power (584°F) results in slightly higher integrated flashed mass, making the SG overfill transient worse.

A combination of various other single failure and preventive maintenance assumptions are analyzed for this event for the radiological consequences and the SG overfill scenarios. These include unavailability of the feedwater condensate system, failure of one extra borating system (EBS) pump, failure of one EFWS train and another in preventive maintenance, failure of MSRCV in an intact SG in closed position, and failure of the affected SG MSIV to close with affected SG EFWS in preventive maintenance. Additionally, a hot zero power (HZP) case is considered and the overfill rate is determined to be less penalizing compared to the hot full power (HFP) case.

A break on the SG hot-side is found to be most limiting based on a sensitivity calculation that considers a break on the cold-side of the SG.

In general, initial plant conditions and setpoints are biased such that the primary-to-secondary break flow is maximized. The upper bound of the PZR pressure (2300 psi) is used along with the high span of the PZR liquid level (59.3 percent).

Nominal blowdown flow is included for the radiological cases, whereas no SG blowdown flow is assumed for the SG overfill cases. Parameters such as EFW startup delay, EFW flowrates and temperatures, and EFW low-low SG water level signal are biased differently for the radiological analysis than the SG overfill scenarios to maximize the severity of the event.

Description and results of the limiting radiological case are presented in Section 15.6.3.3.

For the thermal-hydraulic analysis, the break is postulated to occur near the tube sheet to maximize the break flow (lowest hydraulic resistance). It is modeled on the hot-leg side of the SG to maximize the flashing fraction for determining the radiological release. Additionally, for the radiological release, the break is assumed to be at the apex of the tubes to minimize iodine scrubbing.

BOC initial conditions are assumed for the fuel and coolant. The automatic rod position controller is assumed to be in manual mode and does not respond to a change in reactor power. This assumption maximizes the reactivity feedback effects in the early period due to the combination of the moderator temperature coefficient (MTC) and the decrease in boron concentration associated with the initial decrease in RCS pressure (decrease in fluid density). The most reactive control rod is assumed not to insert at RT.

The availability of offsite power has a significant impact on the progress of this event. LOOP is more limiting because the turbine bypass system is available otherwise and limit radiological releases. LOOP is assumed with RT.

Because the availability of equipment affects the course of the event, the analysis considers the single failure and maintenance of safety-related equipment as well as the operation of non-safety-related equipment that makes the outcome worse. This equipment includes the CVCS charging pumps, the EBS, MHSI pumps, MFW and EFWS, MSIVs, turbine bypass system, and the MSRT. The MSRT includes the MSRCV and the MSRIV. Different single failures are limiting depending on whether the analysis seeks to maximize radiological release or the potential for SG overfill.

The initial conditions are biased to either maximize the radiological release or potential for overfill of the affected SG. HFP initial conditions are limiting for radiological release. Both HFP and HZP initial conditions are analyzed for overfill of the affected SG. Similarly, uncertainties in PS setpoints are biased depending on the objectives of the analysis. Operator actions are required to mitigate this event. No operator actions are credited in this analysis prior to 30 minutes.

15.6.3.3 Results

The analysis shows that the limiting case for radiological release is one in which the charging pumps are operating, LOOP occurs at RT, and a single failure occurs in the MSRT of the affected SG. The MSRCV is postulated to stick fully open. This action releases steam to the environment until the MSRIV closes automatically on low SG pressure. In addition, it is assumed that one emergency feedwater (EFW) pump is in maintenance.

The SGTR SG overfill analysis is based on plant characteristics and initial conditions that are selected to maximize the potential for overfill of the affected SG. The assumed single failure for this case is that the affected SG (SG-4) EFW control valve (CV) fails in the open position. The EFW pump feeding the intact SG-1 is assumed to be in preventive maintenance, which minimizes heat transfer to the secondary system and serves to maximize primary system pressure. The EFW pump feeding the affected SG continues to inject, and the EFW flow characteristics to the affected SG are biased to maximize the EFW flow into the affected SG. Because the EFW CV on the affected SG fails in the open position, the EFW flow to the affected SG continues even if the level in the SG exceeds the EFW shutoff value. The EFW flow to the affected SG is terminated by the operator closing the EFW isolation valve. These assumptions maximize the potential for overfill of the affected SG.

15.6.3.3.1 Analysis Initial Conditions

Table 15.6-4—SGTR Event - Key Input Parameters presents the initial conditions for the analysis. The break is assumed to occur near the tube sheet because it maximizes the break flow (lower hydraulic resistance), and on the hot side of the tube because it maximizes the fraction of the break flow that flashes. The analysis is initiated from full power conditions. The analysis assumes the maximum number of plugged SG tubes, five percent, to minimize heat removal. This assumption leads to a lower initial SG pressure, which increases break flow and flashing fraction.

The analysis assumes primary coolant average temperature is at the lowest allowed temperature at full power (584°F, corresponding to coastdown at EOC conditions) because it leads to a lower initial secondary pressure and slightly higher integrated flashed mass.

15.6.3.3.2 Equipment Status

Table 15.6-5—SGTR Event - Key Equipment Status lists the assumed status of mitigating equipment and components. Although a non-safety-related system, the PZR heaters are simulated because they have the penalizing effect of delaying depressurization. The charging system, another non-safety-related system, is modeled because it is similarly penalizing by its response to a decrease in PZR level. The analysis conservatively does not model the letdown system.

The standby charging pump is activated when the PZR level drops to its low-level setpoint. The analysis assumes that both charging pumps start injecting at this time. The CVCS charging system functions as designed until RT, which is assumed to cause LOOP. When LOOP occurs, the charging pumps are de-energized. The occurrence of LOOP also de-energizes the RCPs, which coast down and stop the main sprays. The power supplies of the CVCS charging pumps and the auxiliary spray control valve are automatically switched to the SBO diesel generators, but they are not actuated. The operator does not start the charging pump. Thus, auxiliary sprays are also unavailable after RT.

The turbine bypass system, a non-safety system, is assumed unavailable because it has a beneficial effect. Hence, secondary steam relief is always assumed to be via the MSRTs.

15.6.3.3.3 Transient Calculation

Table 15.6-6—SGTR Radiological Case - Sequence of Events presents the sequence of events for the limiting radiological release scenario. The postulated tube rupture is assumed to occur with the plant operating at HFP with both CVCS pumps operating and the letdown isolated. PZR level and pressure do not decrease sufficiently to cause a RT. The operator detects the event through high activity alarms in the affected SG steam line and blowdown line. The operator begins to take action at 30 minutes and completes the initial SGTR mitigation steps within an additional 10 minutes.

Figure 15.6-11—SGTR Radiological Case - Reactor Power shows reactor power. Power decreases initially because of reactivity feedback due to RCS depressurization. The operator trips the reactor at 1800 seconds, which is assumed to cause LOOP with subsequent de-energizing of the RCPs, CVCS, and MFW pumps. Figure 15.6-12—SGTR Radiological Case Pressurizer and Affected SG Dome Pressure shows pressures in the primary system and the affected SG. Primary pressure starts to decrease initially, and then increases because of the injection of two CVCS pumps. It decreases rapidly after RT.

Because LOOP is assumed concurrent with RT, SG pressure increases (Figure 15.6-12). The operator is assumed to complete SGTR mitigation actions at 2400 seconds. These actions include closing the MSIV in the affected SG, resetting its MSRT setpoint high, isolating its EFWS and blowdown lines, starting the EFW pumps, and initiating partial cooldown of the unaffected SGs using their MSRTs. Pressure in the affected SG reaches the MSRT setpoint at 2450 seconds. When the MSRCV opens, it is assumed to fail fully open and cause a rapid decrease in the affected SG pressure. MSRT relief is terminated in the affected SG when the MSRIV closes automatically at the low SG pressure setpoint of 570 psia, at 2570 seconds. Subsequently, the affected SG pressure equalizes with the primary pressure at about 1250 psia, and then begins to decrease slowly as the unaffected SGs remove heat from the RCS.

Figure 15.6-13—SGTR Radiological Case - SG Blowdown Flow Rates shows flow rate in the affected SG blowdown line, indicating isolation at 2400 seconds. At the same time, EFWS flow begins in the unaffected SGs (Figure 15.6-14—SGTR Radiological Case - EFW Flow Rates) in conjunction with the operator-initiated partial cooldown. Since the EFWS line to the affected SG is isolated by the operator at 2400 seconds, there is no injection into the affected SG.

The MHSI flow begins when the RCS pressure falls below the MHSI shutoff head (Figure 15.6-15—SGTR Radiological Case - Total MHSI Flow Rate). Partial cooldown is complete in the unaffected SGs at 3600 seconds as the pressure in the SGs falls to 870 psia. At this time, the operator continues the cooldown at 90°F/hour and starts the EBS to provide sufficient boration (Figure 15.6-16—SGTR Radiological Case - EBS Flow Rate). EBS flow continues until the EBS tanks empty at approximately 14000 seconds. MHSI is terminated by the operator when the core exit subcooling exceeds 50°F, at 5412 seconds (Figure 15.6-19—SGTR Radiological Case - Core Exit Subcooling). The primary system is refilled at this time, as shown by the PZR level (Figure 15.6-20—SGTR Radiological Case - Pressurizer Level).

Primary pressure continues to decrease slowly beyond this time due to the heat removal from the unaffected SGs. During this time, the operator opens the PSRVs occasionally to accelerate the decrease in primary pressure (Figure 15.6-17). This equalizes primary and secondary pressure in the affected SG (Figure 15.6-12), thereby minimizing break flow (Figure 15.6-18).

Inventory in the affected SG stabilizes before reaching an overfilled condition as shown by the SG wide range (WR) level (Figure 15.6-21—SGTR Radiological Case - SG Wide Range Levels) and liquid volume (Figure 15.6-22—SGTR Radiological Case - Affected SG Liquid Volume). This stabilization achieves a controlled state. The analysis is stopped at 10,000 seconds. The operator continues with the cooldown and depressurization process to reach the RHR entry conditions, which takes the plant to cold shutdown.

The radiological analysis is conducted using the results of the thermal-hydraulic analysis. These include the integrated mass of break flow (Figure 15.6-23—SGTR Radiological Case - Integrated Break Mass Flow), the integrated mass of steam release to the environment (Figure 15.6-24—SGTR Radiological Case - Integrated Steam Mass Release), the integrated mass flashed (Figure 15.6-25—SGTR Radiological Case - Integrated Mass Flashed), and the liquid volume fraction in the region around the apex of the tubes in the affected SG (Figure 15.6-26—SGTR Radiological Case - Affected SG

Apex Liquid Fractions). The radiological analysis is presented in Section 15.0.3. This is the limiting SGTR radiological release scenario.

The sequence of events for the SG overfill case is shown in Table 15.6-24—SGTR Overfill Case - Sequence of Events. The early part of the transient is similar to the radiological case. The postulated tube rupture is assumed to occur with the plant operating at HFP with both CVCS pumps operating and the letdown isolated.

The reactor is manually tripped by the operator at 1800 seconds (30 minutes) and LOOP is assumed coincident with RT - causing the charging pumps to shut off and the RCPs to trip. On the secondary side, the MFW pumps trip. Figure 15.6-94—SGTR Overfill Case - Reactor Power depicts the initial decrease in reactor power due to reactivity feedback associated with RCS depressurization and the sudden drop in power at 1800 seconds as a result of manual RT.

The ruptured SG is isolated within 10 minutes of the RT (i.e. the MSIV is closed and the MSRT setpoint is raised to 1405.5 psia) prior to the manual initiation of SI and start of the partial cooldown at 2400 seconds (40 minutes). The pressure response of the RCS and the affected SG, illustrated in Figure 15.6-95—SGTR Overfill Case - Pressurizer and Affected SG Dome Pressure, shows that following RT the primary system pressure initially increases (as a result of turbine trip) and then drops rapidly; and the SG dome pressure increases in response to turbine trip as the turbine bypass system becomes unavailable on LOOP. The SG blowdown would isolate on an SI signal but the blowdown flow is conservatively assumed to terminate earlier at the time of LOOP (Figure 15.6-96—SGTR Overfill Case - SG Blowdown Flow Rates).

It is assumed that the operator identifies the EFW CV failure once SG-4 level exceeds the 82% wide range SG level setpoint (approximately 10 minutes after the failure) and closes the EFW isolation valves to SG-4 (Figure 15.6-97—SGTR Overfill Case - EFW Flow Rates and Figure 15.6-104—SGTR Overfill Case - SG Wide Range Level.

MHSI flow starts at about 3500 seconds when the primary system pressure drops below the MHSI pump shutoff head (Figure 15.6-98—SGTR Overfill Case - Total MHSI Flow Rate). Partial cooldown is complete at 3600 seconds when the secondary system pressure drops to 870 psia. At this time a 90°F/hr cooldown is initiated in the three intact loop SGs using the MSRT.

At about 3660 seconds, the operator manually initiates the EBS pumps to add concentrated boron to the primary system and provide RCS makeup (Figure 15.6-99—SGTR Overfill Case - EBS Flow Rate).

The operator realigns the EFW flow in SG-2 at approximately 4260 seconds to feed both SG-2 and SG-1 (SG-1 did not have EFW flow because of preventive maintenance (Figure 15.6-97)).

The operator terminates MHSI at 5306 seconds when the core exit subcooling exceeds 50°F (Figure 15.6-102—SGTR Overfill Case - Core Exit Subcooling). The pressurizer is refilled at this time (Figure 15.6-103—SGTR Overfill Case - Pressurizer Level).

Subsequent to the end of partial cooldown (3600 seconds), the operator opens the PSRVs several times (Figure 15.6-100—SGTR Overfill Case - PSRV Flow Rate) to help decrease the primary system pressure. The primary and secondary system pressures equalize at about 6000 seconds (Figure 15.6-95), canceling the break flow (Figure 15.6-101—SGTR Overfill Case - Break Flow Rate).

Because of the delayed isolation of the EFW flow in the affected SG, the level in the SG approaches 90 percent WR (Figure 15.6-104—SGTR Overfill Case - SG Wide Range Level) at about 14,000 seconds. This results in a continued increase of liquid volume in the affected SG; however, the SG does not overfill, as illustrated in Figure 15.6-105—SGTR Overfill Case - Affected SG Liquid Volume.

Integrated break mass flow, steam mass release, and mass flashed are presented in Figure 15.6-106—SGTR Overfill Case - Integrated Break Mass Flow, Figure 15.6-107—SGTR Overfill Case - Integrated Steam Mass Release, and Figure 15.6-108—SGTR Overfill Case - Integrated Mass Flashed, respectively. These parameters show that the liquid inventory in the affected SG stabilizes.

Thus, a controlled state is achieved. The analysis is terminated at 28,800 seconds (8 hours). The operator would then continue with the cooldown and depressurization process to reach the RHR entry conditions, which would take the plant to cold shutdown.

15.6.3.4 Radiological Consequences

The results of the radiological analysis are presented in Section 15.0.3.

15.6.3.5 Conclusions

The results of the analysis show that with penalizing assumptions, the SGTR event is controlled by a combination of automatic and operator actions.

- The radiological releases are below 10 CFR 100 regulatory limits (or within limits of 10 CFR 50.67 for Alternate Source Term).
- The liquid inventory in the affected SG does not increase to a point where overfill of the SG is a concern.
- This analysis extends to the time when the leak is terminated by pressure equalization between the RCS and the affected SG. Termination of the leak terminates the potential for additional radiological release.

15.6.3.6 SRP Acceptance Criteria

The acceptance criteria for this event are based on the relevant requirements of 10 CFR 100 as it relates to mitigating the radiological consequences of an accident. The plant site and the dose mitigating ESFs are acceptable with respect to the radiological consequences of a postulated SG tube failure accident at a PWR facility if the calculated whole-body and thyroid doses at the exclusion area and the low population zone outer boundaries do not exceed the exposure guidelines. A summary of the SRP acceptance criteria for Section 15.6.3 events included in NUREG-0800, Section 15.6.3, (Reference 4) and descriptions of how these criteria are met are listed below:

1. For the PA with an assumed pre-accident iodine spike in the reactor coolant and for the PA with the highest worth control rod stuck out of the core, the calculated doses should not exceed the guideline values of 10 CFR 100, Section 11.

   – Response: The results of the radiological analysis are presented in Section 15.0.3.

2. For the PA with the equilibrium iodine concentration for continued full power operation in combination with an assumed accident initiated iodine spike, the calculated doses should not exceed a small fraction of the above guideline values, i.e., 10 percent or 2.5 rem and 30 rem, respectively, for the whole-body and thyroid doses.

   – Response: The results of the radiological analysis are presented in Section 15.0.3.

15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

This event does not apply to the U.S. EPR.

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

A postulated loss of coolant accident (LOCA) is initiated by the assumed instantaneous rupture of an RCS pipe. Those smaller than ten percent of the cross-sectional area of the cold leg piping are classified as small-break LOCAs (SBLOCAs). Those larger are considered large-break LOCAs (LBLOCAs). Different methodologies are approved to analyze these two classifications of LOCA.

The acceptance criteria for LOCA are presented in 10 CFR 50.46 and 10 CFR 100 as follows:

- The calculated maximum fuel element cladding temperature does not exceed 2200°F (10 CFR 50.46).

- The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation (10 CFR 50.46).
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall 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 (10 CFR 50.46).
- Calculated changes in core geometry shall be such that the core remains amenable to cooling (10 CFR 50.46).
- After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core (10 CFR 50.46).
- The radiological consequences are within the limits of 10 CFR 100.

15.6.5.1 Large Break Loss of Coolant Accident

15.6.5.1.1 Identification of Causes and Event Description

A postulated LBLOCA is initiated by an assumed instantaneous rupture of an RCS pipe. A spectrum of break sizes for both double-ended guillotine break (DEGB) and double-ended split break (DESB) types is analyzed. The piping breaks are postulated to occur at various locations and include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant pressure boundary. In concurrence with Regulatory Guide 1.157, both the split and the double-ended breaks range in area from 10 percent of A_{PIPE} to twice the cross-sectional area of the largest pipe. The determination of break configuration, split versus double-ended, is made after the break area is selected based on a uniform probability for each occurrence. For an LBLOCA, the most limiting break occurs in a cold-leg pipe between the RCP discharge and the reactor pressure vessel.

An LBLOCA event is described in three phases: blowdown, refill, and reflood. The blowdown phase is defined as the time from initiation of the break until flow from the accumulators or SIS begins. The refill phase is from the end of blowdown until fluid from the ECCS has filled the downcomer and lower plenum to the bottom of the heated length of the fuel. The reflood phase is from the end of refill and continues until the fuel cladding temperature transient is terminated.

Following the instantaneous pipe break, the blowdown phase is characterized by a sudden depressurization from operating pressure to the saturation pressure of the hot leg fluid. The flow out of the break causes an immediate reversal of flow in the downcomer and stagnation of flow in the core. This condition causes the fuel rods to exceed critical heat flux (CHF). Following the initial rapid depressurization, RCS depressurizes gradually as reactor coolant is expelled out the break as vapor.

An RT signal occurs when the PZR or hot-leg low-pressure trip setpoint is reached. However, RT is conservatively neglected in the analysis. Reactor shutdown is accomplished initially by moderator voiding feedback and maintained by the boron content of the ECCS water. An SIS initiation signal is generated when the PZR low-low pressure setpoint is reached.

When system pressure falls below the accumulator pressure, the accumulators discharge into the cold legs, thereby ending the blowdown phase and initiating the refill phase. SIS flow injects into the RCS when system startup-time delays have elapsed and primary system pressure falls below the respective shutoff heads of the MHSI and LHSI systems. While some of the ECCS flow bypasses the core and goes directly out of the break, the downcomer and lower plenum gradually refill. During this refill phase, heat is primarily transferred from the hotter fuel rods to cooler fuel rods and structures by radiative heat transfer.

When the lower plenum is refilled to the bottom of the fuel rod heated length, the refill phase ends and the reflood phase begins. The ECCS fluid flowing into the downcomer provides the driving head to move coolant through the core. As the mixture level moves up the core, steam is generated and liquid is entrained. As this entrained liquid is carried into the SGs, it vaporizes because of the higher temperature in the SGs. This causes steam binding, which reduces the core reflooding rate. The fuel rods are cooled and quenched by radiation and convective heat transfer as the quench front moves up the core.

15.6.5.1.2 Method of Analysis and Assumptions

The analytical methodology used to analyze this event is described in the U.S. EPR Realistic Large Break Loss of Coolant Accident Topical Report (Reference 5). The methodology is a best-estimate evaluation model (EM) for a realistic large break loss of coolant accident (RLBLOCA) and is based on nonparametric statistics.

The RLBLOCA methodology consists of the following computer codes:

- RODEX3A for computation of the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance.
- S-RELAP5 for system thermal-hydraulic calculations. Containment backpressure calculations are performed by an ICECON module within S-RELAP5.

The RLBLOCA methodology uses a nonparametric statistical approach to calculate the peak cladding temperature (PCT), peak local oxidation, and total oxidation values. The peak local oxidation and total oxidation are reported for the limiting cases. The fraction of total hydrogen generated is not calculated; however, it is conservatively bounded by the calculated total percent oxidation, which is below the one percent limit.

The nonparametric statistical approach requires that multiple sampled cases are created and processed. For each case, key LOCA parameters are randomly sampled over a range established through code uncertainty assessment or expected operating limits. The key parameters related to phenomena are presented in Table 15.6-7—RLBLOCA - Sampled Parameters (Phenomenological, Cycle 1 and Equilibrium Cycle). Those related to plant operation are shown with their sampling ranges in Table 15.6-8—RLBLOCA - Sampled Plant Parameters (Cycle 1 and Equilibrium Cycle). The calculation of each sampled case begins with an established steady-state initial condition for the S-RELAP5 model. Equipment status is presented in Table 15.6-9—RLBLOCA - Key Equipment Status.

For the RLBLOCA analysis, reactor power is assumed to be 4612 MWt, which represents the rated thermal power of 4590 MWt with a maximum power measurement uncertainty of 0.48 percent (22 MWt) added to the rated thermal power. The value of 0.48 percent is based on the use of a Caldon CheckPlus™ ultrasonic flow meter (UFM) to measure main feedwater flow. The Caldon CheckPlus™ UFM is approved as noted in NRC Regulatory Issue Summary 2007-24. The uncertainty was verified by a calculation of core thermal power with a secondary side heat balance. The reactor core power for the U.S. EPR RLBLOCA analysis is not sampled.

GDC 35 states that an emergency core cooling system must function for both onsite power available (offsite power unavailable) and offsite power available (onsite power unavailable) cases. By design, there is no significant difference in results between the loss-of-offsite power (LOOP) and non-LOOP cases for the U.S. EPR. The U.S. EPR is designed with an automatic reactor coolant pump trip on coincident safety injection signal and low RCP differential pressure. This feature causes the reactor coolant pumps to trip in the event of a LOCA even if offsite power is available. Furthermore, the LOOP condition produces more conservative PCT results because the delays for commencing ECCS injection are greater than those in the non-LOOP condition. Therefore, this analysis does not sample the availability of offsite power and assumes only LOOP.

Of the four trains of pumped safety injection, one train is assumed conservatively to be unavailable due to maintenance and another train is subject to single failure. On this basis, two of the four trains start and deliver flow. One of the two trains is assumed conservatively to inject into the RCS cold leg with the break. Because the ECCS connection is near the break, all of the ECCS flow delivered to the broken RCS cold leg spills into the containment

Axial power profiles sampled from the power history data are used in each case for the RLBLOCA uncertainty analyses. Therefore, the axial shapes used in the RLBLOCA analyses are assumed to represent a wide range of conditions, which bound or envelope the plant operating range.

PCT is predicted at higher than 95 percent probability level with 95 percent confidence. The EM in the Realistic Large Break LOCA Topical Report (Reference 5) complies with the requirements of 10 CFR 50.46.

Following the establishment of an acceptable steady-state condition, the transient calculation is initiated by introducing a break in the cold leg of the loop containing the PZR. As part of an EM requirement for containment condensation heat-transfer modeling, the 1.7 multiplier on the Uchida heat transfer coefficient for application to containment heat structures is confirmed.

15.6.5.1.3 Results

For the U.S. EPR, RLBLOCA analyses are performed for both an initial fuel cycle and an equilibrium fuel cycle representative of an 18-month core. The U.S. EPR Realistic Large Break LOCA Topical Report (Reference 5) presents only the analysis for the equilibrium fuel cycle. Table 15.6-10—RLBLOCA - Sequence of Events presents the sequence of events for Cycle 1 and for the limiting equilibrium fuel cycle.

The analysis cases causing the highest PCTs are summarized in Table 15.6-11—RLBLOCA - Summary of Maximum PCT Values. The PCTs for the hot rods for the limiting cases are summarized in Table 15.6-12—RLBLOCA - Summary of PCT Values for All Hot Rods for Top PCT Cases. The PCT values for the median cases, for which half of the PCTs are higher and half are lower, are summarized in Table 15.6-13—RLBLOCA - Summary of 50/50 PCT Cases. The maximum local oxidation values for the top PCT cases are summarized in Table 15.6-25—RLBLOCA - Summary of Maximum Local Oxidation Values for Top PCT Cases, and the maximum total oxidation values are summarized in Table 15.6-26—RLBLOCA - Summary of Maximum Total Oxidation Values for Top PCT Cases.

The RLBLOCA parameters of principal interest are presented in the figures listed below.

Parameter	Cycle 1 Figure No.	Equilibrium Cycle Figure No.
• PCT Independent of Elevation	15.6-27	15.6-112
• PCT Independent of Elevation for Hot Rod	15.6-28	15.6-113
• Primary System Pressure	15.6-29	15.6-114
• Flows Supplied to ECCS	15.6-30	15.6-115
• Flows Delivered by ECCS	15.6-31	15.6-116
• Core Inlet Flow	15.6-32	15.6-117
• Core Outlet Flow	15.6-33	15.6-118
• Break Flow	15.6-34	15.6-119
• Collapsed Liquid Level in Downcomer	15.6-35	15.6-120
• Core Liquid Level	15.6-36	15.6-121
• Reactor Power	15.6-37	15.6-122
• Secondary System Pressure	15.6-38	15.6-123
• Downcomer Mass Flowrate	15.6-39	15.6-124
• Core Inlet Temperature	15.6-40	15.6-125
• Core Inlet Quality	15.6-41	15.6-126
• Core Inlet Quality on Smaller Time Scale	15.6-42	15.6-127
• Core Outlet Temperature	15.6-43	15.6-128
• Core Outlet Quality	15.6-44	15.6-129
• Core Outlet Quality on Smaller Time Scale	15.6-45	15.6-130
• In-Core Temperature	15.6-46	15.6-131
• In-Core Quality	15.6-47	15.6-132
• In-Core Quality on Smaller Time Scale	15.6-48	15.6-133
• Cladding Temperature	15.6-49	15.6-134
• Heat Transfer Coefficient	15.6-50	15.6-135
• Primary to Secondary Heat Transfer Rate	15.6-109	15.6-136
• Pump Speed	15.6-110	15.6-137
• Containment Pressure	15.6-111	15.6-138

15.6.5.1.4 Radiological Consequences

The radiological consequences for the LBLOCA are addressed in Section 15.0.3.11.

15.6.5.1.5 Conclusions

The acceptance criteria for LBLOCA are met as follows:

- The maximum calculated PCT of 1695°F is below the acceptance limit of 2200°F.
- For the limiting case, the total local cladding oxidation is 1.53 percent, which is well below the acceptance criterion of 17 percent.
- The amount of calculated hydrogen generated is conservatively bounded by the calculated total percent oxidation, which is below the one percent limit.
- The RLBLOCA methodology demonstrates that the core retains a coolable geometry (see also Section 15.6.5.3).
- Long-term cooling is addressed in Section 15.6.5.3.

The radiological consequences are within the limits of 10 CFR 100 (see Section 15.0.3).

15.6.5.1.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.6.5.1.6 events included in NUREG-0800, Section 15.6.5, (Reference 4) and descriptions of how these criteria are met are listed below:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an EM that satisfies the requirements of 10 CFR 50.46. RG 1.157 and Section I of Appendix K to 10 CFR 50 provide guidance on acceptable EMs. For the full spectrum of reactor coolant pipe breaks, and taking into consideration requirements for RCP operation during a small break LOCA, the results of the evaluation must show that the specific requirements of the acceptance criteria for ECCS are satisfied. This also includes analyses of a spectrum of large and SBLOCAs to verify that boric acid precipitation is precluded for all break sizes and locations.

   The analyses should be performed in accordance with 10 CFR 50.46, including methods referred to in 10 CFR 50.46(a)(1) or (2). The analyses must demonstrate sufficient redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities such that the safety functions could be accomplished assuming a single failure in conjunction with the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available). Additionally the LOCA methodology used and the LOCA analyses should be shown to apply to the individual plant by satisfying 10 CFR 50.46(c)(2), and the analysis results should meet the performance criteria in 10 CFR 50.46(b).

   – Response: The RLBLOCA methodology used to analyze LBLOCA is a best-estimate EM based on non-parametric statistics, described in the Realistic Large Break Loss of Coolant Accident Topical Report (Reference 5). The completed analysis demonstrates that the ECCS design is adequate to satisfy acceptance criteria, with and without offsite power, and with the most limiting single-failure, which is a train of pumped SIS. This analysis satisfies the preceding requirements.

A. The calculated maximum fuel element cladding temperature does not exceed 2200°F.

- Response: The maximum calculated PCT of 1695°F is below the acceptance limit of 2200°F.

B. The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident.

- Response: For the limiting case, the total local cladding oxidation of 1.53 percent is well below the acceptance criterion of 17 percent.

C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed one percent of 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.

- Response: The amount of calculated hydrogen generated is conservatively bounded by the calculated total percent oxidation, which is below the one percent limit.

D. Calculated changes in core geometry are such that the core remains amenable to cooling.

- Response: The RLBLOCA methodology demonstrates that the core retains a coolable geometry. See also Section 15.6.5.3.

E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

- Response: Post-LOCA long term cooling is addressed in Section 15.6.5.4.

2. The radiological consequences of the most severe LOCA are within the guidelines of and 10 CFR 100. For applications under 10 CFR 52, reviewers should use SRP Section 15.0.3, "Radiological Consequences of Design Basis Accidents - for ESP, DC and COL Applications."

– Response: The RLBLOCA radiological consequences are addressed in Section 15.0.3.11.3.

3. The TMI Action Plan requirements of II.E.2.3, II.K.2.8, II.K.3.5, II.K.3.25, II.K.3.30, II.K.3.31, II.K.3.40 have been met.

A. Item II.E.2.3, Uncertainty in Performance Predictions.

- Response: Not applicable to LBLOCA events.

B. Item II.K.2(8): Continued Upgrading of EFW System.

- Response: U.S. EPR is provided with automatic EFW actuation that is initiated automatically on a combination of LOOP and SI as well as on low SG wide range level. The SGs for the U.S. EPR are large, providing a significant supply of inventory for decay heat removal. EFW is not modeled in RLBLOCA analyses.

C. Item II.K.3.5: Automatic Trip of Reactor Coolant Pumps.

- Response: The U.S. EPR is provided with an automatic RCP trip on a combination of SI signal and 80 percent ΔP across the pumps.

D. Item II.K.3.25.

- Response: Not applicable to U.S. EPR (BWR only).

E. Item II.K.3.30 - Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K.

- Response: Not applicable to LBLOCA events.

F. Item II.K.3.31 - Plant-Specific Calculations to Show Compliance with 10 CFR 50.46.

- Response: U.S. EPR plant specific analyses were performed and are presented in Section 15.6.5.1.

G. Item II.K.3.40 - Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA.

- Response: Not applicable to LBLOCA events.

15.6.5.2 Small Break Loss of Coolant Accident

15.6.5.2.1 Identification of Causes and Event Description

A SBLOCA is a PA in accordance with the classification of events presented in Section 15.0.0.1. The postulated SBLOCA is defined as a break in the RCPB that has an area of 0.5 square-feet or less (approximately ten percent of the cold leg pipe cross-sectional area). This range of break areas encompasses the small lines that penetrate the RCPB. Small breaks could involve relief and safety valves, charging and letdown lines, drain lines, and instrumentation lines. The most limiting break location is in the cold leg pipe at the discharge side of the RCP. This break location results in the largest amount of inventory loss and the largest fraction of ECCS fluid being ejected outward through the break. This break produces the greatest degree of core uncovery and the longest fuel rod heatup time; consequently, it poses the greatest challenge to meeting the 10 CFR 50.46 acceptance criteria.

The SBLOCA cases are analyzed until the top of the active fuel is recovered with a two-phase mixture and the cladding temperatures are reduced to temperatures near the saturation temperature. The SBLOCA is a loss of reactor coolant inventory that cannot be offset by the CVCS. Because the CVCS is not a safety-related system, it is assumed unavailable for mitigating an SBLOCA.

The loss of primary coolant causes a decrease in primary system pressure and PZR level. An RT occurs on low PZR pressure or low hot-leg pressure. The RT signal automatically trips the turbine and closes the MFW HL. For LOOP cases, it is assumed that a LOOP occurs with RT. This also terminates MFW. The secondary side pressure increases and, because of the unavailability of the steam dump to the main condenser, the MSRTs open to relieve steam to the atmosphere. The SGs are fed by the EFWS, which is actuated on a combination of SI signal and LOOP.

An SI signal is actuated on low-low PZR pressure. The SI signal automatically starts the MHSI and LHSI pumps and initiates a partial cooldown of the secondary system, which causes the primary system to cool down and decrease in pressure. During the partial cooldown, RCS pressure decreases and MHSI begins. The partial cooldown of the SGs is via MSRT steam relief to the atmosphere. This cooldown automatically decreases the MSRCV setpoints at a rate corresponding to 180°F/h to a value low enough to permit MHSI injection, while still high enough to prevent core re-criticality. For the smallest breaks, the volume of flow through the break is less than the volume addition by MHSI and steam production in the core due to the decay heat. Depressurization of the RCS therefore stops at the end of the partial cooldown.

The RCS inventory continues to decrease as long as MHSI injection is insufficient to compensate for the break flow rate. The break flow rate decreases as the void fraction in the cold legs increases. When the break flow changes to single-phase steam, the ratio between steam production due to core decay heat and steam venting at the break shifts. The break then might become the dominant factor for the subsequent depressurization sequence:

- For the smallest breaks, some condensation in the SG tubes may occur in conjunction with the direct steam venting at the break to remove all steam produced in the core. The RCS pressure remains slightly above the SG pressure.
- Larger breaks vent sufficient steam so that further RCS depressurization occurs without steam condensation in the SG tubes (eventually the heat transfer reverses between primary and secondary side). RCS pressure falls independent of the SG temperature down to the accumulator discharge pressure and, possibly, to the LHSI injection pressure.

The subsequent evolution of the RCS water inventory depends on the balance between ECCS flow rates and break flow rate. The core may uncover before the rate of ECCS water addition exceeds the loss of RCS coolant out the break. If so, the fuel cladding temperature rises above saturation temperature in the uncovered part of the core.

15.6.5.2.2 Method of Analysis and Assumptions

Codes and Methods Used

The SBLOCA analysis is performed using the approved EM documented in Codes and Methods Topical Report (Reference 1). The Small Break LOCA and Non-LOCA Sensitivity Studies and Methodology (Reference 6) describes SG nodalization sensitivity analyses performed to support the SBLOCA methodology of Reference 1. The appropriate conservatisms, prescribed by Appendix K of 10 CFR 50, are incorporated in these analyses.

The computer codes used in this analysis are as follows:

- The RODEX2-2A computer code is used to calculate the burnup dependent initial fuel conditions for each active core region in S-RELAP5.
- The S-RELAP5 computer code (described in Section 15.0.2.4) is used to model the primary system (including the hot rod) and the secondary side of the SGs. The governing conservation equations for mass, energy, and momentum transfer are used along with appropriate correlations consistent with 10 CFR 50.46 and 10 CFR 50, Appendix K.

The RCS is modeled in S-RELAP5 as a network of control volumes interconnected by flow paths. The model includes four accumulators, a PZR, and four SGs in which both the primary and secondary sides are modeled. The four loops are modeled explicitly to provide an accurate representation of the plant. The LHSI are cross connected in pairs, which is modeled explicitly in the calculation. The MHSI injects in the accumulator piping, which also is modeled explicitly.

Decay heat is determined from reactor kinetics equations with actinide and decay heating as prescribed by Appendix K to 10 CFR 50.

To determine the limiting SBLOCA case with respect to the acceptance criteria in 10 CFR 50.46, break spectrum calculations were performed for breaks ranging from 0.4 percent to 10 percent of cold leg area. Two break spectrum calculations were performed, one with the assumption that the LOOP occurs concurrent with reactor scram and the other with the assumption that offsite power is available. The offsite power availability results in changes in equipment availability and actuation times as well as differences in RCPs availability.

The single failure criterion required by Appendix K is satisfied by assuming the failure of one train of pumped SI and EFW. In addition, one train of pumped SI and EFW is assumed unavailable because of maintenance, leaving active only two MHSI pumps, two LHSI pumps and two emergency feedwater pumps. All four accumulators are assumed to inject. This is the most detrimental single failure for an SBLOCA event with respect to the 10 CFR 50.46 criteria since it results in the worst degradation of heat removal capacity by reducing the available SI flow and the EFW flow. A single failure analysis demonstrated that the failure of one MSRT train (with one EDG in preventive maintenance) and the failure of one accumulator (with one EDG in preventive maintenance) are both bounded by the failure of one EDG, with a second EDG in preventive maintenance. Other potential single failures have been evaluated and determined to be less limiting than the loss of one EDG. For example, the failure of one MHSI pump would be less limiting than the loss of a diesel generator since it has no effect on the LHSI flow or the EFW flow.

The EFWS is actuated on the combination of LOOP and SI signal or on SG low wide-range level. The two active trains of MHSI are assumed to inject respectively into Loop 4, the broken loop, and into Loop 1, the intact loop adjacent to the broken loop. The adjacent loop is chosen because it provides the greatest opportunity for injected ECCS to flow directly to the break and bypass the core.

For the scenarios that assume LOOP occurs coincident with RT, LOOP de-energizes the MFW system and RCPs. For the break scenarios without LOOP, the RT signal automatically trips the turbine and closes the MFW high-load lines. The addition of MFW through the LL is conservatively neglected. For the non-LOOP break spectrum, the EFWS is actuated on a low-low SG level signal.

The axial power shape used is a conservatively top-skewed, EOC shape. The power peak occurs at a normalized distance of 0.8542. The power in the hot rod is assumed at the design peaking limit for the U.S. EPR.

The loop seal elevations on the broken loop (Loop 4) and the adjacent intact loop (Loop 1) are biased so that they are 1.0 foot lower than the loop seals in the other two loops. This bias makes the seal in the broken loop less likely to clear before the ones in the intact loops. Sensitivity analyses show that for SBLOCA, higher PCTs result when the loop seal in the broken loop remains plugged longer than in the intact loops. SG tube plugging is set to five percent symmetrically.

Following receipt of an SI signal, the SG MSRT system initiates a partial cooldown, which is a controlled secondary system depressurization from 1414.7 psia to 900 psia at a rate corresponding to 180°F/h.

The core is modeled with a two-dimensional component having 28 axial nodes and three radial nodes. The Baker-Just metal water reaction correlation is used for all fuel rod heat structures. The rupture model is invoked for the hot rod.

The limiting case is identified via a break spectrum analysis.

Cases Analyzed

SBLOCA cases are analyzed over a spectrum of break sizes ranging from 2.0 inches to 8.0 inches in diameter in 0.5-inch increments. The breaks are located in the RCP discharge piping. The break spectrum cases fall into two categories: (a) with LOOP assumed, in which the RCPs trip on RT; and (b) without LOOP, in which the RCPs continue to operate after RT and are tripped on low ΔP across two of the four RCP pumps. Two additional cases are analyzed with LOOP: a double-ended guillotine break of an accumulator line and a 9.71-inch diameter break corresponding to ten percent of the cold leg cross-sectional area. The 6.5-inch break with LOOP produces the limiting PCT.

For the accumulator line break, in addition to the loss of ECCS trains due to single failure and maintenance, one ECCS train (consisting of one MHSI, one LHSI and one accumulator) injects into the broken accumulator line, which spills directly into the containment. Because it is assumed that the remaining operational LHSI is cross connected to the broken ECCS line, it too is discharged to the containment. This leaves only a single MHSI train that is effective delivering pumped injection to the primary system.

Initial Conditions

Table 15.6-14—SBLOCA - U.S. EPR System Analyses Parameters presents the initial conditions used in the analysis. Several initial conditions can have a significant impact on the SBLOCA results. These parameters are biased in the analysis in order to pose a greater challenge to the acceptance criteria in 10 CFR 50.46. A description of these parameters is provided here:

- Axial Power Shape: an EOC top-skewed power shape is used since the high powered zones experience core uncover the longest.
- MHSI/LHSI Fluid Temperature: an upward bias is assumed corresponding to the maximum incontainment refueling water storage tank; this reduces the condensation of the primary coolant coming in contact with the injected fluid and it has an adverse effect on the mixture level rise.
- Accumulator Pressure: the accumulators are activated when the primary system depressurizes to the accumulator pressure; a minimum pressure setpoint is used to delay the initiation of cool water into the core.
- Core Bypass Flow: a conservatively maximum core bypass is assumed to reduce the coolant flow entering the active core region.
- Loop Flow Rate: a bounding low value is assumed to bias low the amount of coolant entering the reactor vessel.

- EFW: minimum EFW flow is initiated consistent with the single failure criterion which allows two trains of EFW to be available. The EFW fluid temperature is selected to be a bounding high value. These assumptions degrade the heat sink capacity of the steam generators and thus the ability to remove the primary energy.
- SG Tube Plugging: a maximum tube plugging is assumed. This assumption degrades the heat transfer from the primary system.

Neutronics Data and Decay Heat

The plant is assumed to be operating at nominal full power plus calorimetric uncertainty until RT. The moderator and Doppler feedbacks are not significant up to RT and are therefore not accounted for in the SBLOCA calculation. For conservatism, it is assumed that the most reactive RCCA does not insert. After RT, the residual fission power is defined by the ANS 5.1-1973 standard (Reference 7) plus 20 percent uncertainty. An EOC top-skewed axial power shape is used in the analysis because it represents a distribution with power concentrated in the upper region of the core. This distribution is limiting because it minimizes coolant level swell, while maximizing vapor superheating and fuel rod heat generation at the uncovered elevation.

Table 15.6-15—SBLOCA - Axial Power Shape presents the axial power shape used in the analysis. A nominal cycle length of 18 months is the basis for all neutronics parameters. However, the top-peaked axial power shape in the 24-month cycle was chosen for the SBLOCA analyses. This power shape bounds the shorter cycles. Table 15.6-14—SBLOCA - U.S. EPR System Analyses Parameters provides additional neutronics data.

Trips & Controls Credited in the SBLOCA Analysis

For SBLOCA events, RT occurs on either low PZR pressure or low hot leg pressure. Table 15.6-16—SBLOCA- Protection System Setpoints presents the safety-related signals credited in SBLOCA analysis. Setpoint uncertainties are for harsh environment conditions. Table 15.6-17—SBLOCA - Equipment Status presents the equipment status for these analyses.

Pumped ECCS Input

Table 15.6-18—SBLOCA - Minimum MHSI Flow and Table 15.6-19—SBLOCA- Minimum LHSI Flow present the minimum, degraded MHSI and LHSI flows, respectively, to each loop (as delivered to the accumulator lines). The coolant temperature for MHSI and LHSI injection is assumed the maximum IRWST temperature (122°F).

Operator Action

The analyses presented in this section do not credit operator action. Throughout the analyses, automatic actions provide the necessary accident mitigation to satisfy applicable acceptance criteria.

NUREG-0800 (Reference 4) action item II.K.3.5 is satisfied in the analysis by conservatively addressing the operation of the RCPs, including requirements for RCP trip during SBLOCAs as presented in NRC Generic Letters 85-12, 86-05, and 86-06. This guidance states that the RCPs should be tripped when necessary during an SBLOCA so that the criteria of 10 CFR 50.46 and 10 CFR 100 are not exceeded by inappropriate RCP operation. The U.S. EPR incorporates an automatic safety-related RCP trip for SBLOCA mitigation when there is an 80 percent ΔP across the pumps in combination with an SIS actuation signal. For analysis, a degraded uncertainty of five percent is applied such that the pumps are tripped at a 75 percent ΔP across the pumps.

15.6.5.2.3 Results - Break Spectrum Results

Table 15.6-20—SBLOCA - Break Spectrum Results with LOOP and Figure 15.6-51—SBLOCA - PCT - Break Spectrum with LOOP present the results of the break spectrum with LOOP at RT. The results identify the limiting break to be a 6.5-inch break in the cold leg with LOOP. Table 15.6-21—SBLOCA - Sequence of Events for 6.5 Inch Break with LOOP presents the sequence of events.

Limiting 6.5-inch Break with LOOP

After the initiation of the break, the primary pressure drops rapidly to the saturation point (Figure 15.6-53—SBLOCA - 6.5 Inch Break - Primary and Secondary System Pressure). RT occurs at 4.49 seconds due to low hot-leg pressure (Figure 15.6-52—SBLOCA - 6.5 Inch Break - Reactor Power). All RCPs are tripped due to LOOP at RT. Depressurization of the RCS plateaus at about 25 seconds as primary system saturates.

Initially, the secondary side pressure increases rapidly due to the closing of the turbine stop valves at the time of RT. This pressure increase is halted by the opening of the MSRIVs at about 114 seconds, which causes a drop in secondary pressure of about 40 psia (Figure 15.6-54—SBLOCA - 6.5 Inch Break - MSRT Flow). SG pressure decreases when the MSRIV first opens, while the MSRCV is 40 percent open. Because SG pressure drops below the target value of the partial cooldown, the MSRCV strokes close at 134 seconds. At about 170 seconds, the MSRCV reopens when the SG pressure intersects the cooldown curve. From 170 seconds to the end of the transient, the MSRCV valve modulates to depressurize the secondary side at a rate corresponding to 180°F/hr. At about 255 seconds, the primary pressure drops below the secondary pressure.

RCS depressurization increases when the break uncovers at about 250 seconds and the break flow transitions from two-phase to steam (Figure 15.6-55—SBLOCA - 6.5 Inch Break - Break Flow). Loops 2 and 3 clear at 234 and 237 seconds, respectively. This condition creates a flow path for the steam to vent out of the break (Figure 15.6-56—SBLOCA - 6.5 Break - Loop Seal Void Fraction). The depressurization continues until the accumulator flow begins at about 346 seconds, at which time the pressure increases slightly (Figure 15.6-57—SBLOCA - 6.5 Break - ECCS Flow). Loops 1 and 4 then clear at 360 and 362 seconds, respectively.

The SG MFW flow is terminated at 4.49 seconds when LOOP occurs on RT (Figure 15.6-58—SBLOCA - 6.5 Break - MFW Flow).

The EFWS pumps in SGs 1 and 4 begin injecting at 76.80 seconds after their actuation on the combination of SI and LOOP signals. In the SGs receiving EFW, injection starts before steam relief through the MSRCV causes a noticeable decrease in SG inventory (Figure 15.6-59—SBLOCA - 6.5 Inch Break - Steam Generator Mass Inventory). In SGs 2 and 3, which do not receive EFW injection, inventory decreases.

RCS inventory and, consequently, collapsed liquid level in the hot assembly, fall rapidly as primary fluid is lost out of the break (Figure 15.6-63—SBLOCA 6.5 Inch Break - Primary System Inventory and Figure 15.6-64—SBLOCA - 6.5 Inch Break - Hot Assembly Collapsed Liquid Level). At about 246 seconds, two MHSI pumps begin to inject into Loops 1 and 4 (Figure 15.6-65—SBLOCA - 6.5 Inch Break - MHSI Flow Rate). However, the inventory lost out the break exceeds that supplied by the MHSI pumps, resulting in RCS net inventory loss and core uncovery. A PCT of 1638°F occurs at 360.26 seconds (Figure 15.6-67—SBLOCA - 6.5 Inch Break - Peak Cladding Temperature and Coolant Temperature).

The large quantity of water supplied by the accumulators terminates the net loss of primary coolant inventory, thereby recovering the core level and ultimately quenching the core. As the RCS depressurizes further, the MHSI and LHSI overcome the break flow and provide adequate long-term RCS coolant inventory.

Additional figures (Figure 15.6-68—SBLOCA - 6.5 inch Break - Hot Assembly Cladding Temperature and Coolant Temperature through Figure 15.6-81—SBLOCA - 6.5 inch Break - RC Speed) of system variables as a function of time are presented. These figures provide additional information for the limiting SBLOCA case. Figure 15.6-68 through Figure 15.6-70 present the peak cladding temperature for the other three regions of the core. The heat transfer coefficient for all regions of the core, at the PCT location is presented in Figure 15.6-71. Figure 15.6-72 through Figure 15.6-75 present the equilibrium quality and fluid temperature at the inlet and outlet of the hot assembly region. Figure 15.6-76 and Figure 15.6-77 present the mass flow rate at inlet and outlet of each of the three core regions, respectively. The other variables presented are the quality at the PCT location (Figure 15.6-78), the downcomer mass flow rate (Figure 15.6-79), the primary to secondary heat transfer rate (Figure 15.6-80) and the RCP speed (Figure 15.6-81).

Non-LOOP Spectrum Analysis

A non-LOOP pump trip break spectrum was evaluated for breaks between 2.0 inches and 8.0 inches. The results of this sensitivity study are presented in Table 15.6-22—SBLOCA - Delayed Pump Trip (Without LOOP) Break Spectrum Results and Figure 15.6-82—SBLOCA - PCT - Delayed Pump Trip (Without LOOP) Break Spectrum. Table 15.6-23—SBLOCA - PCT Comparison between SBLOCA with RCPs Tripped at RT and RCPs Tripped on ΔP and Figure 15.6-83—SBLOCA - Comparison PCT - Break Spectrum With/Without LOOP present comparisons between the PCTs for the SBLOCAs with LOOP, i.e., RCPs are de-energized at the time of LOOP (RT), and the cases without LOOP, where the RCPs are tripped later on low ΔP. The comparisons show that for smaller break sizes, a later pump trip produces higher PCTs. The increase in PCT is due to a longer period of liquid break flow when the RCPs continue to operate. As the break size increases, however, the difference in PCT between the cases with and without LOOP becomes less and the PCT becomes somewhat lower for the non-LOOP cases.

The pump trip occurs well before the minimum RCS inventory. The pump trip study shows that tripping the pumps on ΔP across the pump is adequate to satisfy 10 CFR 50.46. The limiting break size for the non-LOOP spectrum is the 6-inch break, which has a PCT of 1585°F.

15.6.5.2.4 Radiological Consequences of the SBLOCA

The radiological consequences for the SBLOCA are bounded by a LBLOCA (see Section 15.0.3.11).

15.6.5.2.5 Conclusions

The limiting SBLOCA case is the 6.5-inch cold leg break at the RCP discharge piping with LOOP at RT.

The analysis demonstrates that the acceptance criteria are met as follows:

- A PCT of 1638°F was calculated for the limiting case. This is below the 2200°F PCT limit specified in 10 CFR 50.46(b)(1).
- The total cladding oxidation at the PCT location is 0.383 percent for the limiting case. This is below the 17 percent limit specified in 10 CFR 50.46(b)(2).
- The hydrogen generated in the core during the SBLOCA by cladding oxidation, 0.00897 percent, is below the one percent limit specified in 10 CFR 50.46(b)(3).

- The calculation shows that the core retains a coolable geometry (see Section 15.6.5.3). Thus, the coolable geometry criterion in 10 CFR 50.46 (b)(4) is satisfied.

15.6.5.2.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.6.5.2.6 events included in NUREG-0800, Section 15.6.5, (Reference 4) and descriptions of how these criteria are met are listed below:.

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an EM that satisfies the requirements of 10 CFR 50.46. RG 1.157 and Section I of Appendix K to 10 CFR 50 provides guidance on acceptable EMs. For the full spectrum of reactor coolant pipe breaks, and taking into consideration requirements for RCP operation during a small break LOCA, the results of the evaluation must show that the specific requirements of the acceptance criteria for ECCS are satisfied. This also includes analyses of a spectrum of small-break LOCAs to verify that boric acid precipitation is precluded for all break sizes and locations.

   The analyses should be performed in accordance with 10 CFR 50.46, including methods referred to in 10 CFR 50.46(a)(1) or (2). The analyses must demonstrate sufficient redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities such that the safety functions could be accomplished assuming a single failure in conjunction with the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available). Additionally the LOCA methodology used and the LOCA analyses should be shown to apply to the individual plant by satisfying 10 CFR 50.46(c)(2), and the analysis results should meet the performance criteria in 10 CFR 50.46(b).

   – Response: The SBLOCA analyses performed with and without LOOP are performed with an approved EM that complies with the requirements of 10 CFR 50.46. The methodology is described in Codes and Methods Topical Report (Reference 1). The SBLOCA analyses described in Section 15.6.5.2.4 demonstrate that sufficient redundancy is provided by the assumption of the worst single failure in combination with the most limiting maintenance. The performance criteria of 10 CFR 50.46 are thereby satisfied. Boron precipitation is addressed in Section 15.6.5.3.

   A. The calculated maximum fuel element cladding temperature does not exceed 2200°F.

      - Response: A PCT of 1638°F was calculated for the limiting case. This is below the 2200°F PCT limit specified in 10 CFR 50.46(b)(1).

   B. The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident.

- Response: The total cladding oxidation at the PCT location is 0.383 percent for the limiting case. This is below the 17 percent limit specified in 10 CFR 50.46(b)(2).

C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed one percent of 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.

- Response: The hydrogen generated in the core during the SBLOCA by cladding oxidation, 0.00897 percent, is below the one percent limit specified in 10 CFR 50.46(b)(3).

D. Calculated changes in core geometry are such that the core remains amenable to cooling.

- Response: The calculation shows that the core retains a coolable geometry. See Section 15.6.5.3. Thus, the coolable geometry criterion in 10 CFR 50.46 (b)(4) is satisfied.

E. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

- Response: Post-LOCA long-term cooling is addressed in Section 15.6.5.4.

2. The radiological consequences of the most severe LOCA are within the guidelines of 10 CFR 100 or 10 CFR 50.67. For applications under 10 CFR 52, reviewers should use SRP Section 15.0.3, "Radiological Consequences of Design Basis Accidents - for ESP, DC and COL Applications."

– Response: The SBLOCA radiological consequences are addressed in Section 15.0.3.11.3.

3. The TMI Action Plan requirements of II.E.2.3, II.K.2.8, II.K.3.5, II.K.3.25, II.K.3.30, II.K.3.31, and II.K.3.40 have been met.

A. Item II.E.2.3, Uncertainty in Performance Predictions.

- Response: The methodology used was assessed against an array of pertinent experimental data. In addition, use of Appendix K requirements bound possible operational uncertainty. The frequency of a system failure severe enough to approximate the Appendix K single failure assumptions was estimated to be, at most, 0.1/demand. Given a small LOCA, a modeling uncertainty, and something approximating the worst-case single failure, the actual peak cladding temperature could be greater than that calculated by the analyses. However, considerable margin to significant core damage remains for three reasons. First, the small-break analysis for U.S. EPR is not limiting. About a 500°F margin exists between the calculated small-break peak cladding temperature and the 2200°F limit. Second, U.S. EPR operates well within the LOCA limits (i.e., is not “LOCA-limited”). Third, for severe damage to occur, a significant amount of cladding must achieve a temperature significantly higher than 2200°F. The case of the hottest point of the core barely exceeding the temperature limit does not automatically imply severe damage.

B. Item II.K.2(8): Continued Upgrading of EFW System.

- Response: The U.S. EPR is provided with an automatic EFW actuation, which is actuated on LOOP and SI and on low SG wide range level. The SG for the U.S. EPR are large, providing a significant supply of inventory for decay heat removal.

C. Item II.K.3.5: Automatic Trip of Reactor Coolant Pumps.

- Response: The U.S. EPR is provided with an automatic pump trip on 80 percent ΔP across the pumps. For SBLOCA, the pump trip is assumed at 75 percent ΔP across the pumps (with five percent degraded uncertainty)

D. Item II.K.3.25

- Response: Not applicable to the U.S. EPR (BWR only).

E. Item II.K.3.30 - Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K.

- Response: SBLOCA analyses were performed with an approved methodology that complies with 10 CFR 50.46 (see Reference 1).

F. Item II.K.3.31 - Plant-Specific Calculations to Show Compliance with 10 CFR 50.46.

- Response: U.S. EPR plant-specific analyses are presented in Section 15.6.5.2.

G. Item II.K.3.40 - Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA.

4. The RCP shaft seal package shall maintain sealing integrity during an SBLOCA coincident with a LOOP such that the RCP shaft seals in each RCP do not fail and potentially create a coincident LOCA in each loop.

– Response: Shaft seal integrity is provided by maintaining cooling to the RCP shaft seal during an SBLOCA coincident with LOOP (see Section 5.4.1).

15.6.5.3 Coolable Core Geometry

10 CFR 50.46 requires that calculated changes in core geometry following a LOCA shall be such that the core remains amenable to cooling. Several potential conditions

cause geometry degradation. The first is mechanical grid crush and strain caused by the physical movement of the fuel assemblies relative to the heavy reflector and the supports due to a seismic event in conjunction with the hydrodynamic loads generated by the depressurization occurring during a LOCA. Another potential cause of geometry degradation during a LOCA is cladding swelling and rupture associated with elevated cladding temperature in conjunction with increased fuel pin cladding pressure differential as the RCS pressure decreases. The two mechanisms are evaluated separately and then combined to determine the net effect on a coolable core geometry.

Analyses are performed to determine the impact of the combined forces from the safe shutdown earthquake (SSE) event in combination with LOCA forces for the fuel assembly grids for faulted conditions. The maximum grid load calculated by combining the LOCA and SSE impact loads is 16.3 kN. The 95 percent confidence interval allowable for M5 cladding intermediate grid is the elastic limit force of 18.2 kN. Thus the margin of safety is:

$$MS = (18.2/16.3 - 1)*100 = 12\%$$

Because the elastic limit is not exceeded, no permanent grid deformation occurs, and grid crush does not degrade the coolable core geometry.

15.6.5.3.1 Large Break LOCA Clad Swelling

The LBLOCA cases with the highest calculated PCT for the initial cycle and for the equilibrium cycle are evaluated for cladding swelling and rupture. These are Case 38 of the equilibrium cycle, with a PCT of 1625°F, and Case 85 of the initial cycle, with a PCT of 1695°F (Section 15.6.5.1). In both cases, some hot rods fuel pin cladding reach the rupture temperatures (based on the approved rupture model in the Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel (Reference 8)).

The evaluation of cladding pre-rupture and rupture strain due to large break LOCA demonstrates that the maximum local hot fuel assembly flow blockage is 75 percent. These core assembly flow blockage results are conservatively representative of the core-wide average flow blockage.

15.6.5.3.2 Small-break LOCA Clad Swelling

The methodology uses the S-RELAP5 capability to predict cladding rupture. The cladding strain and rupture model is applied to the hot pin. Rupture does not occur for any of the breaks analyzed. Reference 8 provides the cladding strain and assembly blockage as a function of cladding rupture temperature from 1112°F to 2192°F for various cladding temperature ramp rates. It demonstrates that the strain increases as the temperature ramp rate decreases. The low temperature ramp rate of 0°C/s causes the highest strain and blockage. In the alpha phase region, the maximum pre-rupture strain is about 40 percent up to about 1800°F. Because the predicted SBLOCA PCTs are less than 1800°F, the maximum swelling and blockage for the SBLOCA is comparable to the limiting LBLOCA case. This is 75 percent coolant channel blockage or less, depending on the actual cladding temperature and stress time history. Reference 8 demonstrates that the core remains coolable at decay heat levels for up to 90 percent coolant channel blockage.

15.6.5.3.3 Conclusion - Coolable Core Geometry

Based on a conservative swelling and rupture analysis, the evaluation of mechanical degradation of coolable core geometry due to combined seismic and LBLOCA loads demonstrates that the maximum local fuel assembly blockage is 75 percent. Therefore, it is conservatively assumed that the core-wide average blockage is 75 percent. Because this value is less than the 90 percent coolant channel blockage threshold for adequate cooling, the U.S. EPR maintains a coolable core geometry following a LOCA.

15.6.5.4 Long-Term Core Cooling

After the initial mitigation of a LOCA, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for an extended time as required by the long-lived radioactive isotopes remaining in the core. The core remains subcritical.

Several issues are addressed to demonstrate adequate long term cooling following a LOCA:

- Boron precipitation. Boron in the coolant can concentrate and precipitate in the upper region of the core when there is protracted boiling following a LOCA.
- Boron dilution during SBLOCA. GSI-185 raises a concern regarding the potential for recriticality during an SBLOCA if unborated water accumulates in the SGs and cold leg piping due to condensation and moves to the core as a slug.
- Containment debris. GSI-191 raises concerns regarding the potential damage to ECCS equipment and blockage of core channels due to debris in the water re-circulated from the IRWST.

15.6.5.4.1 Prevention of Boric Acid Precipitation

The U.S. EPR provides the operator the capability to redirect an LHSI train so that at least 75 percent of it is injected through the hot leg letdown line of the residual heat removal system (RHRS). Analyses show that switching the LHSI to hot leg injection is effective at limiting the boron concentration in the core region regardless of the break location. When started within 6200 seconds, precipitation is prevented in the core and other regions of the reactor vessel and RCS. The small break analyses show that more water is retained in the core region than for the large break LOCA. Since the core boron concentration, and correspondingly the margin to the precipitation limit, is dependent on the volume of liquid in the core, the large break LOCA bounds small break LOCAs relative to boron precipitation.

The mitigating effect of hot leg injection is confirmed by extending the S-RELAP5 calculations for a representative range of breaks analyzed in Sections 15.6.5.1 and Section 15.6.5.2.

15.6.5.4.1.1 Small-Break LOCA Flow Behavior

For breaks up to approximately 4 inches in diameter, the RCS refills in less than four hours with two trains of MHSI and LHSI and returns to natural circulation.

In the 6.5 inch break, following completion of the automatic partial cooldown, operator action is assumed at 1800 seconds to continue depressurization of the SGs at a rate corresponding to 90°F/h. At the same time, the operator is assumed to realign the two operating trains of LHSI to inject approximately 75 percent of their flow into the respective hot legs. The exact value of the hot leg/cold leg flow split depends on the RCS pressure. Operating procedures control the timing of hot leg injection initiation to within an hour. The S-RELAP5 SBLOCA analysis initiated the hot leg injection at 30 minutes as an example, which is conservative because earlier in time there is a higher system pressure, resulting in less LHSI flow, and a higher decay heat, leading to slightly more steam production and a greater resistance to reverse flow from the hot legs into the core. As seen in Figure 15.6-84—SBLOCA - 6.5 Inch Break - Integral of Upper Plenum Flow to the Hot Legs, the redirected flow reverses the flow in loops 1 and 4 into the upper plenum, making additional coolant available to the core region. These are the loops with the operating SI trains and are the same loops receiving EFW. Figure 15.6-85—SBLOCA - 6.5 Inch Break - Integral of Core Region Exit Flows shows that the hot leg injected flows further reverse the fuel assembly flow in the peripheral region of the core. Some of the downflow continues out through the lower plenum to the lower head region (Figure 15.6-86—SBLOCA - 6.5 Inch Break - Integral of Lower Plenum Flow to Lower Head). This removes the concentrated boron that accumulated prior to the initiation of the hot leg injection. The hot leg injection then maintains the boron concentration below 3000 ppm, which is well below the boron precipitation limit of 38,500 ppm (see Figure 15.6-92). At 7000 seconds into the event, the continued cooldown of the steam generators has not caused the secondary side pressure to reach the point where the steam generators will remove decay heat, as illustrated by the RCS pressure and the steam generator 1 pressure (Figure 15.6-87—SBLOCA - 6.5 Inch Break - Pressurizer and Steam Generator 1 Pressure).

15.6.5.4.1.2 Large-Break LOCA Flow Behavior

A representative LBLOCA case is analyzed to demonstrate the effectiveness of hot leg injection for break sizes too large for the MHSI and LHSI to refill the loops.

In the analysis depicted in the following figures, the operator is assumed to switch to hot leg injection at 1 hour. As seen in Figure 15.6-88—LBLOCA with Hot Leg Injection at 60 Minutes - Integrated Flow from Upper Plenum to Hot Legs, the switch to hot leg injection causes flow to reverse from the hot legs receiving hot leg injection back into the upper plenum. Flow reversal is indicated when the slope becomes negative. The flow proceeds down the peripheral region, the guide tubes, and the heavy reflector into the lower plenum as seen in Figure 15.6-89—LBLOCA with Hot Leg Injection at 60 Minutes - Integrated Flow from Core Regions to Upper Plenum and Figure 15.6-139—LBLOCA with Hot Leg Injection at 60 Minutes - Integrated Flow through Core Bypass Regions. Forward flow continues into and out of the hot assembly, the central core region, and the average core and into the two loops without hot leg injection. Figure 15.6-90—LBLOCA with Hot Leg Injection at 60 Minutes - Integrated Flow from Lower Plenum to Core Regions shows the reverse flows into the lower plenum from the peripheral region and the forward flow to the central regions. Approximately 75 percent of the peripheral region flow mixes with the other core regions, with the remainder penetrating into the lower plenum. The downflow into the lower plenum penetrates further into the lower head, as seen in Figure 15.6-91—LBLOCA with Hot Leg Injection at 60 Minutes - Integrated Flow from Lower Head to Lower Plenum. The flow reverses to the downcomer, increasing the flow out of the vessel side of the break. This removes the concentrated boron that accumulated prior to the initiation of the hot leg injection. The hot leg injection then maintains the concentration below 3000 ppm, which is well below the boron precipitation limit of 38,500 ppm (see Figure 15.6-92—Time Dependent Boron Concentration During the Pool Boiling Period with and without Hot Leg Injection at 60 Minutes).

If the LOCA is a large hot leg break, the ECCS injection into the cold leg exceeds the core boil off rate and the excess ECCS has sufficient flow through the core to prevent the formation of a boron concentration that approaches the precipitation limit even with redirection of approximately 75 percent of the LHSI flow to the hot legs.

15.6.5.4.1.3 Boron Precipitation Assessment

The maximum injection concentration, determined by weighting the flow rates of SI and EBS in the most penalizing injection configuration, is 2051 ppm. This value is used in the calculation of concentration over time using the methodology described in U.S. EPR Boron Precipitation and Boron Dilution (Reference 9).

The calculation conservatively neglects the following mitigating processes:

- Increased boron solubility due to other solutes.

- Increased boiling temperature due to boric acid concentration.
- Carryout of dissolved boric acid by steam generated in the core.
- Carryout of boric acid due to droplet entrainment.
- Addition of nonborated water from sources such as the CVCS.

Figure 15.6-92—Time Dependent Boron Concentration During the Pool Boiling Period with and without Hot Leg Injection at 60 Minutes shows the predicted boron concentration over time for the limiting LBLOCA PCT case. The LBLOCA has a shorter time to precipitation than the SBLOCA, and therefore is the boundary boron precipitation event. The curve demonstrates that boric acid does not concentrate to the degree that boron precipitates out of solution. Moreover, there is adequate time for the operator to initiate hot leg injection to limit the buildup of boron in the core region and prevent precipitation in other regions of the RCS.

15.6.5.4.2 SBLOCA Boron Dilution

GSI-185, "Control of Recriticality Following Small-Break LOCAS in PWRs," raises the concern for SBLOCA events that de-borated water could accumulate in cold leg pump suction piping due to the condensation of steam. When natural circulation is restored, this de-borated water gets flushed as a slug to the RV and core, potentially causing recriticality and fuel damage.

The conditions necessary for this condition to occur develop for a narrow range of break sizes. Breaks smaller than this range do not interrupt natural circulation and therefore do not accumulate de-borated water. Those larger than this range depressurize quickly to low pressure, during which time the secondary sides of the SGs are a heat source to the primary system. Even if heat transfer is re-established to the SGs after they are depressurized, the break is too large for the LHSI to refill the loops. Because natural circulation does not restart, de-borated water is not flushed to the core as a slug.

AREVA performed tests at the PKL integral-loop test facility to investigate boron dilution during SBLOCA, as described in Final Report of the PKL Experimental Program Within the OECD/SETH Project (Reference 10). Some of the tests simulate the controlled cooldown of the SGs representative of the U.S. EPR plant design. The tests demonstrate that natural circulation does not restart abruptly. It is preceded by a period of intermittent circulation. Moreover, the circulation starts first in one active loop and is followed independently by circulation in other active loops. This thermal-hydraulic behavior provides a basis for evaluating boron concentrations in the cold legs and core.

Bounding calculations of boron concentration in the cold legs, RV, and core are performed assuming different natural circulation restart scenarios. These calculations are described in Reference 9. The calculations demonstrate that the concentration in coolant entering the core does not fall to the minimum core average concentration for recriticality of 1005 ppm with equilibrium xenon.

15.6.5.4.3 IRWST Recirculation Cooling

GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance," raises the concern that the high-energy jet from a LOCA may rip away insulation, pulverize concrete, and create other miscellaneous debris particles. Debris generated and transported to the IRWST may potentially penetrate the strainers and screens, degrade the performance of plant mitigating systems, and block coolant channels in the core.

ANP-10293 (Reference 11) describes the design features that address the GSI-191 concerns. The U.S. EPR design reduces the potential for debris generation by using reflective metal insulation to insulate RCS components. This insulation does not produce particulate or fibrous debris that is easily transported to the SIS inlet and ingested. In addition, a defense in depth approach is used to enable heavy materials to settle out in the Containment Building. Multiple levels of filtration prevent debris from reaching the SIS pumps and being transported to the RCS. This system is described in Section 6.3.

15.6.5.4.4 Conclusions

The evaluations described in the preceding sections demonstrate that the U.S. EPR satisfies the requirement that following the initial mitigation of a LOCA, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period required by the long-lived radioactivity.

- Boron precipitation. Calculations demonstrate that for the complete spectrum of breaks, boric acid does not concentrate to the degree that boron precipitates out of solution. Moreover, the operator has adequate time to initiate hot leg injection to limit the buildup of boron in the core region and, if started early, precipitation in other regions of the RCS.

- Boron dilution during SBLOCA. PKL test results and bounding scenario calculations demonstrate that the boron concentration in coolant entering the core during the restart of natural circulation does not fall below the minimum core average concentration for recriticality.

- Containment debris. The use of reflective metal insulation on RCS components to reduce the generation of particulate and fibrous debris and a defense in depth approach to preventing its migration to the ECCS pump inlet effectively mitigates the concern for equipment degradation and blockage due to the ingestion of debris.

15.6.6 References

1. ANP-10263P-A, Revision 0, “Codes and Methods Applicability Report for the U.S. EPR,” AREVA NP Inc., August 2007.

2. ANP-10287P, Revision 0, “Incore Trip Setpoint and Transient Methodology for U.S. EPR,” AREVA NP Inc., November 2007.

3. ANP-10269P, Revision 0, “The ACH-2 CHF Correlation for the U.S. EPR,” AREVA NP Inc., November 2006.

4. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, March 2007.

5. ANP-10278P-A, Revision 1, “U.S. EPR Realistic Large Break Loss of Coolant Accident Topical Report,” AREVA NP Inc., November 2011.

6. ANP-10291P, Revision 0, “Small Break LOCA and Non-LOCA Sensitivity Studies and Methodology,” October, 2007.

7. ANS-5.1-1973, “Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors,” American Nuclear Society, October 1971, revised October 1973.

8. BAW-10227P-A, Revision 1, “Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel,” Framatome Cogema Fuels, February 2000.

9. ANP-10288P, Revision 1, “U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report,” AREVA NP Inc., January 2010.

10. FANP NGTT1/04/en/04, Revision A, “Final Report of the PKL Experimental Program Within the OECD/SETH Project,” Framatome ANP, December 2004.

11. ANP-10293P, Revision 4, “U.S. EPR Design Features to Address GSI-191 Technical Report,” AREVA NP Inc., November 2011.

Table 15.6-1—IOPSRV Event - Key Input Parameters

Parameter	Analysis Value
Initial reactor power	4612 MW
Initial RCS loop flow rate	119,688 gpm/loop
Initial RCS average temperature	594°F
Upper head temperature	594°F
Initial PZR pressure	2250 psia
Initial PZR liquid level	54.3%
Initial main steam pressure	1090 psia
Feedwater flow rate	1442 lb_m/s
Feedwater temperature	446.0°F
SG level	49% NR
SG tube plugging	5%
Moderator temperature coefficient	0 pcm/°F
Doppler reactivity feedback	-1.17 pcm/°F

Table 15.6-2—IOPSRV Event - Key Equipment Status

Plant Equipment or System	Status
Rod Position Controller	Available (if limiting)[1]
PZR Heaters	Off
PZR Sprays	Off
RCPs	Operating until LOOP
Charging Pumps	Off
Letdown Flow	Off
MHSI Pumps	Two pumps available
LHSI Pumps	Two pumps available
EFW Pumps	Two pumps available feeding two SGs until 1800 s; flow from two pumps distributed to the four SGs by operator action at 1800 s
MSSVs and MSRT	Available

Note:

1. The equipment is modeled in the analysis if its availability results in a more limiting transient.

Table 15.6-3—IOPSRV Event - Sequence of Events

Event	Time (s)
Inadvertent opening of the PSRV.	0
RT on low PZR pressure, TT, and LOOP.	39
Two SIS trains start.	54
EFW actuation on SI signal.	116
Two MHSI pumps inject into cold legs 2 and 4.	423
MHSI inflow begins to exceed PSRV outflow.	1100
End of partial cooldown; primary side pressure stays at level of secondary side pressure; natural circulation occurs in primary loops.	1300
EFW manual actuation and beginning of SG refilling in loops 1 and 3.	1800
Primary side pressure decouples from the secondary pressure and continues dropping because of RCS subcooling.	2200
End of calculation.	4000

Table 15.6-4—SGTR Event - Key Input Parameters
Sheet 1 of 2

Parameter	Analysis Value	
Initial reactor power	4612 MW	
Initial RCS loop flow rate	119,692 gpm/loop	
Initial reactor vessel average temperature	584°F	
Initial reactor vessel upper head temperature	T_{hot}	
Initial PZR pressure	2300 psia	
Initial PZR liquid level	59.3% of span	
Initial main steam pressure	984 psia	
Break type/location	Double-ended guillotine break in a single U-tube at the tubesheet on the hot side of SG 4	
Break choked flow model	Moody critical flow	
EFW flow rate	400 gpm per SG (radiological) / 490 gpm per SG (SG overfill)	
EFW temperature	122°F (radiological) / 50°F (SG overfill)	
Moderator reactivity feedback	(lbm/ft^3) 42.270 43.671 44.929 47.554 49.765	($) 0.09 0.06 0.00 -0.22 -0.50
Doppler reactivity feedback	(°F) 100.0 200.0 300.0 400.0 600.0 800.0 1000.0 1008.0 1400.0 1600.0 2000.0	($) 1.74 1.52 1.31 1.11 0.73 0.36 0.01 0.00 -0.64 -0.95 -1.53
Core average U-238 capture-to-fission ratio	0.85	
Charging flow	176 gpm flow per pump	
Charging flow temperature	122°F	

Table 15.6-4—SGTR Event - Key Input Parameters
Sheet 2 of 2

Parameter	Analysis Value
MSRT opening pressure	Intact SGs: Initially 1414.7 psia, then reduced at 180°F/hr rate to 900 psia to implement a Partial Cooldown
	Ruptured SG: Initially 1354.7 psia, then stepped to 1405.5 psia
MSRT flow rate	Intact SGs: Minimum 790 lb_m/s at 1230 psia when MSRCV fully open
	Ruptured SG: Maximum 869 lb_m/s at 1230 psia when MSRCV fully open
SGTP level	5%
Initial SG level	49% NR
Low SG pressure MSIV setpoint	694.7 psia
Low-low SG water level EFW signal setpoint	38% WR (radiological) 42% WR (SG overfill)
MSRCV initial position	Fully open
MSRCV stroke time	40 s
Low-low PZR pressure setpoint actuating MHSI and partial depressurization of SGs	1692.9 psia
MHSI pump shutoff head	1407 psia
MHSI flow	Maximum 1-pump curve for each of 4 credited trains
MHSI temperature	122°F

Table 15.6-5—SGTR Event - Key Equipment Status

Plant Equipment or System	Status
Rod Position Controller	Manual mode, assumed to not respond
RCCAs	Most reactive RCCA stuck in fully withdrawn position following RT
PZR Heaters	Available, ON until LOOP
PZR Sprays	Available until LOOP, not credited
PSRVs	Available
RCPs	Operating until LOOP
Charging Pumps	Both pumps available until LOOP
Letdown Flow	Available until LOOP, not modeled (penalizing assumption)
MHSI Pumps	4 pumps available
LHSI Pumps	4 pumps available
MFW Pumps	Operating until LOOP
EFW Pumps	3 pumps available, One EFW pump feeding unaffected SG is assumed to be in maintenance
MSIVs	Available
Turbine bypass system	Available until LOOP, not modeled
MSRTs	Available initially, MSRCV in affected SG fails in fully open position when activated - single failure
MSSVs	Available
SG Blowdown	Available until LOOP

Table 15.6-6—SGTR Radiological Case - Sequence of Events

Event	Time (s)
DEG rupture of a single U-tube on the hot side of the tubesheet	0
CVCS charging pumps start	204
Manual RT with LOOP MFW pumps and RCPs lose power CVCS charging pumps lose power	1800
Initiate closure of affected SG MSIV Reset affected SG MSRT setpoint to 1405.5 psia, affected MSRT closes SG blowdown isolates CVCS isolates Start of Partial Cooldown in unaffected SGs Isolation of EFW in affected SG Start EFW pumps, EFW pump in affected SG assumed unavailable (maintenance) Start MHSI pumps	2400
Affected SG pressure increases to MSRT setpoint, MSRCV fails open in fully open position (single failure)	2450
Affected SG MSRIV closure initiated on low SG pressure	2570
Partial cooldown ends in unaffected SGs Initiate 90°F/h cooldown in unaffected SG using MSRTs Manual Initiation of EBS pumps to add concentrated boron and provide RCS makeup	3600
Terminate MHSI flow, subcooling > 50°F	5412
Operator cycles PSRV to maintain RCS pressure approximately equal to affected SG pressure	> 3600
End of calculation EBS running, EBS tanks estimated to empty at 14,131 seconds	10,000

Table 15.6-7—RLBLOCA - Sampled Parameters (Phenomenological, Cycle 1 and Equilibrium Cycle)

Time in cycle (axial shape, rod properties, and burnup)
Peaking factors
Break type (guillotine versus split)
Break size
Critical flow discharge coefficients (break)
Offsite power availability (not sampled; LOOP assumed)
Decay heat
Critical flow discharge coefficients (surge line)
Initial upper head temperature
Film boiling heat transfer
Dispersed film boiling heat transfer
CHF
Tmin (intersection of film and transition boiling)
Initial stored energy (only for hot assembly, hot rods, and surrounding assembly)[1]
Downcomer hot wall effects
SG interfacial drag
Condensation interphase heat transfer
Metal-water reaction

Note:

1. For the central and peripheral regions, the fuel centerline is set at maximum densification.

Table 15.6-8—RLBLOCA - Sampled Plant Parameters (Cycle 1 and Equilibrium Cycle)

Parameter	Min	Max
Core power (not sampled)	4612 MW	
Initial RCS flow rate	176.44 Mlb_m/hr	198.00 Mlb_m/hr
Initial operating temperature	590°F	598°F
PZR pressure	2214 psia	2286 psia
PZR level	49.3%	59.3%
Containment volume	2,888,000 ft^3	3,934,000 ft^3
IRWST temperature	100°F	131°F
Accumulator pressure	652.7 psia	710.7 psia
Accumulator liquid volume	1236 ft^3	1412.6 ft^3
Intact cold leg with operational MHSI and LHSI	Loop 1, 2, or 4	

Table 15.6-9—RLBLOCA - Key Equipment Status

Plant Equipment or System	Status
SIS Actuation	SIS actuation is on the very low PZR pressure setpoint, 1667.9 psia (with an uncertainty of ± 25 psia for normal conditions and ± 55 psia for degraded conditions).
MHSI and LHSI	• One train out of service for preventive maintenance. • One train out of service due to single failure. • One MHSI pump to the broken cold leg. • One LHSI pump to the broken cold leg and one intact leg. through a cross-connection. • One MHSI pump to one of the intact cold legs (sampled). • One LHSI pump to one of the intact cold legs (sampled – same cold leg receiving MHSI) and to another cold leg through a cross-connection.
Accumulators	All four accumulators are available.
Control Rod Scram	Rod insertion is not credited.
RCPs	The RCPs trip on LOOP or “on low ΔP over RCP and SIS signal,” where the minimum ΔP over the RCP setpoint is defined as 75% of the nominal ΔP.
Partial Cooldown	Per the RLBLOCA EM, SG isolation occurs at break initiation; hence partial cooldown is not simulated. The S-RELAP5 model for the RLBLOCA analysis does not incorporate the partial cooldown feature. Neglecting the MSRT cooldown feature reduces the energy being removed from the primary system and, therefore, is conservative.
SG Main Steam and Feedwater	Per the RLBLOCA EM, SG isolation occurs at break initiation.

Table 15.6-10—RLBLOCA - Sequence of Events

Event	Time (sec)	
	Cycle 1	Equilibrium Cycle
Begin analysis	0	0
Break opened	0	0
RCP tripped	0	0
SI Actuation Signal (SIAS) issued	10.0	10.3
Start of broken loop accumulator injection	8.2	11.6
Start of intact loop accumulator injection	13.9	13.2
Start of MHSI	50.0	50.3
Beginning of core recovery (beginning of reflood)	27.8	28.4
LHSI available	50.0	50.3
PCT occurred	134.1 (1695°F)	8.5 (1625°F)
Broken loop LHSI delivery began	50.0	50.3
Intact loop 4 LHSI delivery began	50.0	50.3
Broken loop MHSI delivery began	50.0	50.3
Intact loop 4 MHSI delivery began	50.0	50.3
Broken loop accumulator emptied	46.9	59.1
Intact loop accumulator emptied (Loop 1, 2, and 4, respectively)	49.7, 46.4, and 49.7	55.2, 55.5, and 56.2
Transient calculation terminated	688.4	738.8

Table 15.6-11—RLBLOCA - Summary of Maximum PCT Values

Fuel Cycle	Case Number	Break Type	PCT (°F)	Hot Rod	Total Oxidation (%)	Maximum Local Oxidation (%)	PCT Time (sec)	PCT Elevation (ft)	End Time (s)
Equilibrium	38	DESB	1625	8% Gd Rod	0.0206	0.9242	8.5	11.642	738.8
Cycle 1	85	DEGB	1695	UO_2 Rod	0.0224	1.4193	134.1	11.899	688.4

Table 15.6-12—RLBLOCA - Summary of PCT Values for All Hot Rods for Top PCT Cases

Fuel Cycle	Case No.	UO_2 Rod (°F)	Break Type	PCT Elev. (ft)	Fresh 2.0% Gd Rod (°F)	PCT Elev. (ft)	Fresh 4.0% Gd Rod (°F)	PCT Elev. (ft)	Fresh 6.0% Gd Rod (°F)	PCT Elev. (ft)	Fresh 8.0% Gd Rod (°F)	PCT Elev. (ft)
Equilibrium	38	1614	DESB	11.3851	1603	11.3851	1625	11.6421	N/A	N/A	1625	11.6421
Cycle 1	85	1695	DEGB	11.899	1682	11.899	1679	12.4129	1658	12.4129	1642	12.4129

Table 15.6-13—RLBLOCA - Summary of 50/50 PCT Cases

Fuel Cycle	Case Number	Break Type	PCT (°F)	Hot Rod	Total Oxidation (%)	Maximum Local Oxidation (%)	PCT Time(s)	PCT Elevation (ft)	End Time (s)
Equilibrium	13	DEGB	1243	UO_2 Rod	< 0.01	0.2054	152.9	11.899	406.6
Cycle 1	107	DESB	1244	4% Gd Rod	< 0.01	0.1591	3.8	2.200	413.0

Table 15.6-14—SBLOCA - U.S. EPR System Analyses Parameters
Sheet 1 of 2

Parameter	Analysis Value
Core Power	4612 MW (100.5%)
Axial Power Shape and Power Peaking	EOC top skewed
Fq	2.6
FΔH	1.7
Scram Worth	6161 pcm
Core Average Capture to Fission	0.85
Delayed Neutron Fraction (β)	0.00515
Delayed neutron fraction to prompt neutron lifetime ratio (β/l)	214.083 s^{-1}
Gamma Smearing Factor	0.98
Fraction of energy deposited in the fuel when the fuel is fully moderated	0.974
Loop Flow Rate/per loop	119,692 gpm
RCS Average Temperature	594.0°F
Primary System Pressure (PZR Pressure)	2250 psia
Initial PZR Liquid Level	54.3%
Total Bypass	5.5% split: 1.24%-max heavy reflector bypass + max baffle bypass 3.93%-total thimble flow 0.33%--min downcomer to upper head bypass 0%-downcomer to hot legs bypass
Secondary System Pressure	1103.2 psia (consistent with the SG tubes plugging level)
Initial SG Level	49% NR
SG Secondary Side Inventory	181,480 lbm/SG
MFW Flow	5.245E6 lb/hr
MFW Temperature	446.0°F (at 100.5% power)
Accumulator Pressure	652.7 psia
Accumulator Volume	1942.3 ft^3
Accumulator Temperature	90.5°F
MHSI Fluid Temperature	122°F
SG Tube Plugging	5%
EFW Flow Rate	400 gpm to each SG

Table 15.6-14—SBLOCA - U.S. EPR System Analyses Parameters
Sheet 2 of 2

Parameter	Analysis Value
EFW Temperature	122°F
EFW-Start Time	15 s after signal (no LOOP) 60 s after signal (LOOP)
Single Failure Assumption	Loss of 1 EDG (1 EDG in maintenance)

Table 15.6-15—SBLOCA - Axial Power Shape

Node	EOC power shape
1	0.356
2	0.671
3	0.754
4	0.783
5	0.814
6	0.860
7	0.917
8	0.977
9	1.033
10	1.082
11	1.119
12	1.145
13	1.161
14	1.169
15	1.172
16	1.172
17	1.174
18	1.182
19	1.198
20	1.222
21	1.243
22	1.220
23	1.020
24	0.556

Table 15.6-16—SBLOCA- Protection System Setpoints

Signal	Analysis Setpoint[1]
RT on low PZR pressure	1950 psia (Degraded conditions)
RT on low hot leg pressure	1930 psia (Degraded conditions)
TT on RT signal	1950 psia in PZR or 1930 psia in hot leg
MFW isolation	Assumed at closure of the turbine valve
EFW initiation on low SG level	23.5% (Degraded conditions)
MRST opening pressure[2]	1414.7 psia (SG), before the beginning of partial cooldown then maintains 180°F/hr partial cooldown to 900 psia.
RCP Trip for LOOP cases	At the time of RT
RCP Trip for non-LOOP cases	SI signal in combination with 75% ΔP across 2 RCPs (Degraded Conditions)
SI, on low-low PZR pressure, partial cooldown, RCPB Isolation	1612.9 psia (Degraded conditions)

Notes:

1. For the signals which occur before degraded conditions, it is appropriate to consider only the normal condition uncertainty. However, the analysis conservatively applies the degraded uncertainty.

2. The non-degraded uncertainty is used because the instrumentation is located outside containment.

Table 15.6-17—SBLOCA - Equipment Status

Plant Equipment or System	Status
PZR Heaters	Available
PZR Spray	Not modeled
RCPs	Operating, until RT, coast down after LOOP
MHSI Pumps	Available, consistent with single failure assumption
LHSI Pumps	Available, consistent with single failure assumption
Turbine bypass system	Not available, LOOP
MSRT	Available
MFW	Available (until RT)
Emergency Feedwater	Available consistent with single failure assumptions

Table 15.6-18—SBLOCA - Minimum MHSI Flow

$P_{injection}$ (psia) Degraded[1]	Flow per Train (lbm/s) Degraded
21.2	130.1
151.6	117.3
296.6	105.5
441.5	94.0
586.5	82.3
731.5	70.0
876.4	56.7
1021.0	41.8
1166.0	24.1
1239.0	13.0
1245.0	0.0

Note:

1. Pressure at injection location.

Table 15.6-19—SBLOCA- Minimum LHSI Flow

$P_{injection}$ (psia) Degraded[1]	Flow per Train (lbm/s) Degraded
36.4	312.2
76.8	273.2
104.0	248.4
131.4	223.4
159.0	197.6
186.6	170.7
214.4	141.9
242.4	110.5
270.5	75.1
298.9	32.3
325.0	0.0

Note:

1. Pressure at injection location.

Table 15.6-20—SBLOCA - Break Spectrum Results with LOOP

Break Diameter (in)	Break Area (ft^2)	PCT (°F)	Time of PCT(s)	Metal Water Reaction: Local Maximum (%)	Metal Water Reaction: Core Wide (%)
2.0	0.0218	No Heatup	N/A	N/A	N/A
2.5	0.0341	1042	5000.2	2.59E-2	5.006E-4
3.0	0.0491	917	2986.4	5.286E-3	1.806E-4
3.5	0.0668	949	1837.9	5.458E-3	2.106E-4
4.0	0.0873	1088	1222.3	1.551E-2	3.193E-4
4.5	0.1104	1223	908.13	4.855E-2	8.999E-4
5.0	0.1364	1085	679.28	1.176E-2	3.543E-4
5.5	0.165	1199	548.34	3.064E-02	6.715E-4
6.0	0.1963	1125	459.28	1.758E-2	5.462E-4
6.5	0.2304	1638	360.26	0.383	8.974E-3
7.0	0.2673	1587	305.39	0.305	7.619E-3
7.5	0.3068	1464	267.45	0.158	3.135E-3
8.0	0.3491	1470	234.85	0.152	2.496E-3
ECCS Line Break DEG: 8.5 (RCS side) 10.126 (ECCS side)	DEG: 0.3941 (RCS side) 0.5592 (ECCS side)	1531	265.16	0.217	5.076E-3
Max Break 9.71	0.5143	1435	165.82	0.108	1.447E-3

Table 15.6-21—SBLOCA - Sequence of Events for 6.5 Inch Break with LOOP

Event	Time(s)
Begin analysis	0
Break opened	0
RT	4.493
RCPs tripped	4.494
SIS signal	16.807
EFW initiated (Loop 1 and 4)	76.807
MSRIV opens	114
MSRCV closes (faster SG depressurization)	134
MSRCV reopens to control SG depressurization at a rate of 180°F/hr	170
Loop seal clearing - Loop 2	234
Loop seal clearing - Loop 3	237
Broken loop 4 MHSI delivery began	246
Intact loop 1 MHSI delivery began	246
Break uncovery	250
Accumulator injection (Loop 1, 2, and 3 and 4 respectively)	346
Loop seal clearing - Loop 1	360
PCT occurred (1638, node #31)	360.3
Loop seal clearing - Loop 4	362
Broken loop 4 LHSI delivery began	380
Intact loop 1 LHSI delivery began	380
Transient calculation terminated	1000

Table 15.6-22—SBLOCA - Delayed Pump Trip (Without LOOP) Break Spectrum Results

Break Diameter (in)	Break Area (ft^2)	PCT (°F)	Time of PCT(s)	Metal Water Reaction	
				Local Maximum (%)	Core Wide (%)
2.0	0.0218	No Heatup	NA	NA	NA
2.5	0.0341	1034	4777.1	2.863E-2	5.720E-4
3.0	0.0491	1276	2604.0	0.1118	2.530E-3
3.5	0.0668	1504	1469.9	0.3175	8.866e-3
4.0	0.0873	1505	1084.6	0.3382	9.111E-3
4.5	0.1104	1399	824.06	0.1478	3.318E-3
5.0	0.1364	1405	658.61	0.1344	3.074E-3
5.5	0.1650	1572	521.25	0.3851	1.139E-2
6.0	0.1963	1585	428.02	0.3530	1.053E-2
6.5	0.2304	1577	358.17	0.2856	5.220E-3
7.0	0.2673	1524	305.71	0.2184	4.86E-3
8.0	0.3491	1429	238.01	0.1285	1.883E-3

Table 15.6-23—SBLOCA - PCT Comparison between SBLOCA with RCPs Tripped at RT and RCPs Tripped on ΔP

Break Diameter (in)	PCT for RCPs Tripped at Scram (°F)	PCT for RCPs Tripped on ΔP (°F)
2.0	No Heatup	No Heatup
2.5	1042	1034
3.0	917	1276
3.5	949	1504
4.0	1088	1505
4.5	1223	1399
5.0	1085	1405
5.5	1199	1572
6.0	1125	1585
6.5	1638	1577
7.0	1587	1524
8.0	1470	1429

Table 15.6-24—SGTR Overfill Case - Sequence of Events

Time (seconds)	Event
0	DEG rupture of a single U-tube on the hot side of the tubesheet
204	Start 2 CVCS charging pumps
1800	Manual Reactor Trip with LOOP
1801	MFW pumps and RCPs lose power CVCS charging pumps lose power
2400	Manual SI start
2400	Start of Partial Cooldown
2401	Initiation of EFW (SI + LOOP), EFW pump 1 PM SG blowdown isolates, EFW 4 CV fails fully open
2401	Initiate closure of affected SG MSIV Reset affected SG MSRT setpoint to 1405.5 psia, affected MSRT closes
3061	EFW flow to SG 4 isolated
3600	End of Partial Cooldown, Initiate 90°F/hr SG cooldown in 3 intact SGs using MSRTs
3660	Manual Initiation of EBS pumps to add concentrated boron and provide RCS makeup
4261	EFW 2 re-aligned to feed SG 1 & SG 2
5306	Terminate MHSI flow, subcooling > 50°F
14191	EBS tanks empty, EBS pumps stop
3600 - 28800	Operator cycles PSRV to maintain RCS pressure approximately equal to affected SG pressure
28800	End of Analysis

Table 15.6-25—RLBLOCA - Summary of Maximum Local Oxidation Values for Top PCT Cases

Fuel Cycle	Case Number	Break Type	PCT (°F)	Hot Rod	Total Oxidation (%)	Maximum Local Oxidation (%)	PCT Time (sec)	PCT Elevation (ft)	End Time (sec)
Equilibrium	38	DESB	1625	8% Gd Rod	0.0206	0.9242	8.5	11.642	738.8
Cycle 1	70	DEGB	1684	UO_2 Rod	0.0105	1.5287	142.1	11.385	664.3

Table 15.6-26—RLBLOCA - Summary of Maximum Total Oxidation Values for Top PCT Cases

Fuel Cycle	Case Number	Break Type	PCT (°F)	Hot Rod	Total Oxidation (%)	Maximum Local Oxidation (%)	PCT Time (sec)	PCT Elevation (ft)	End Time (sec)
Equilibrium	2	DESB	1573	8% Gd Rod	0.0230	0.8125	7.6	11.642	612.7
Cycle 1	63	DEGB	1656	UO_2 Rod	0.0271	1.0287	109.0	10.871	621.4

Figure 15.6-1—IOPSRV Event - Transient Reactor Power

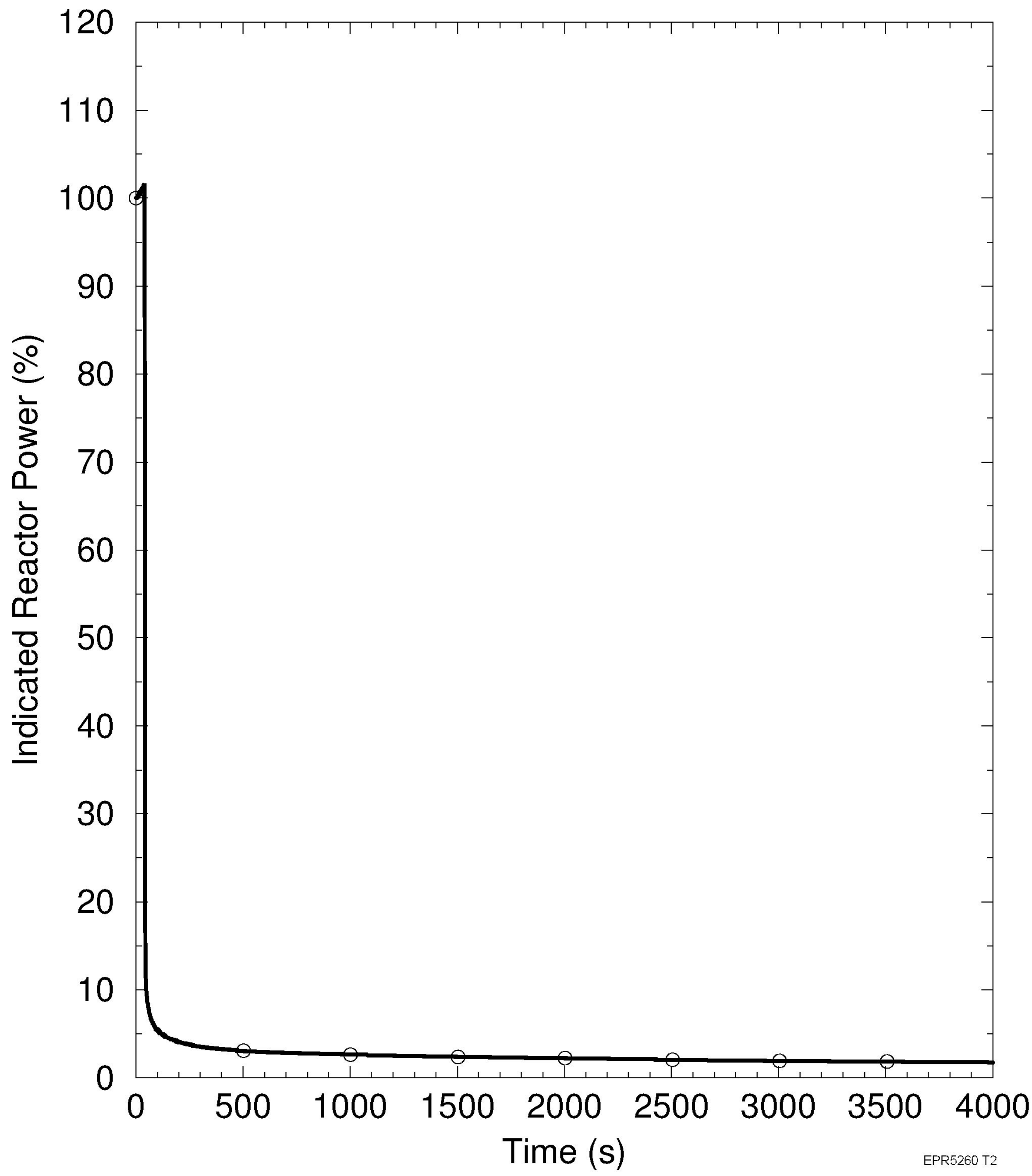

Figure 15.6-2—IOPSRV Event - PZR Pressure

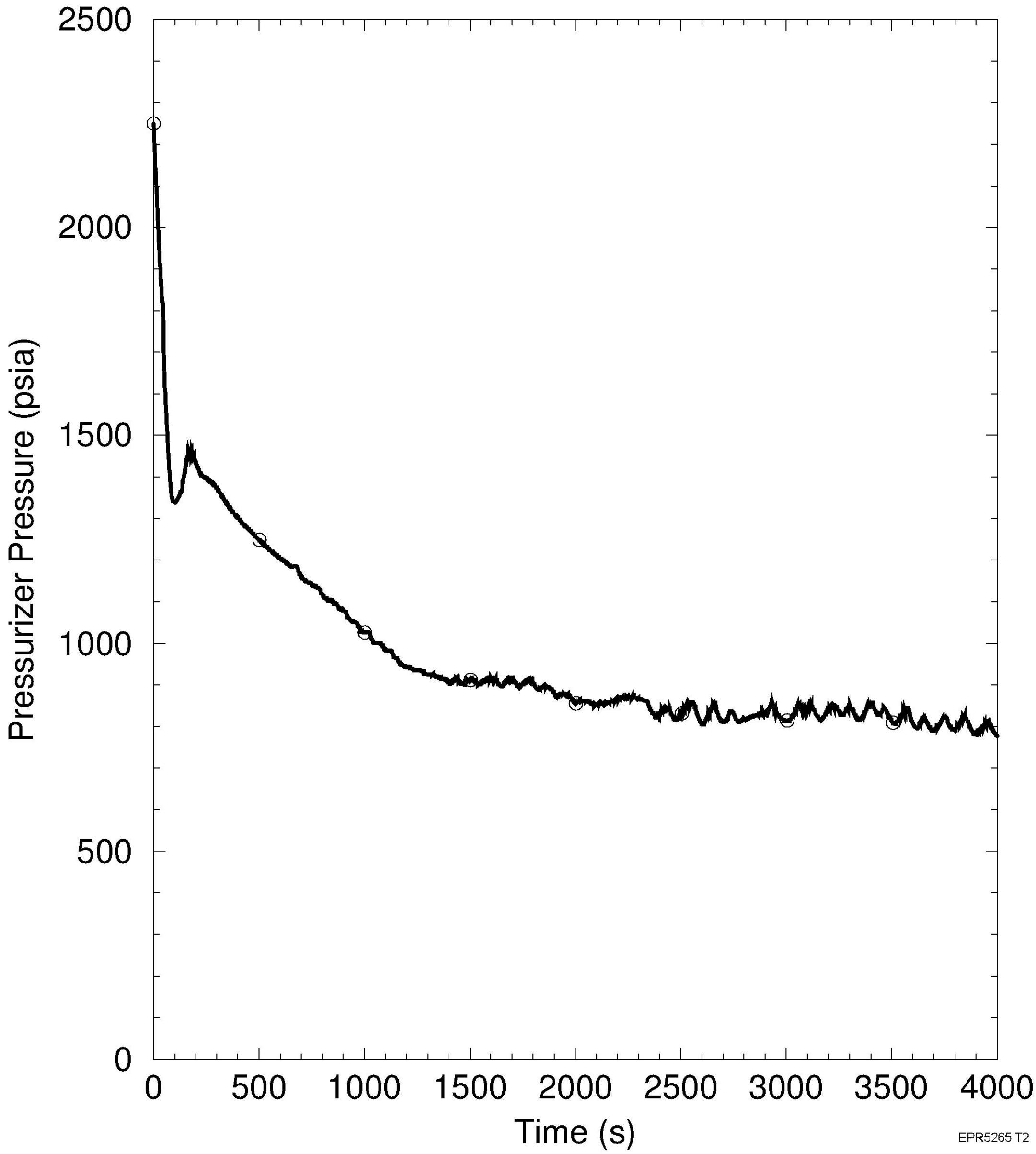

Figure 15.6-3—IOPSRV Event - MHSI and PSRV Flow Rates

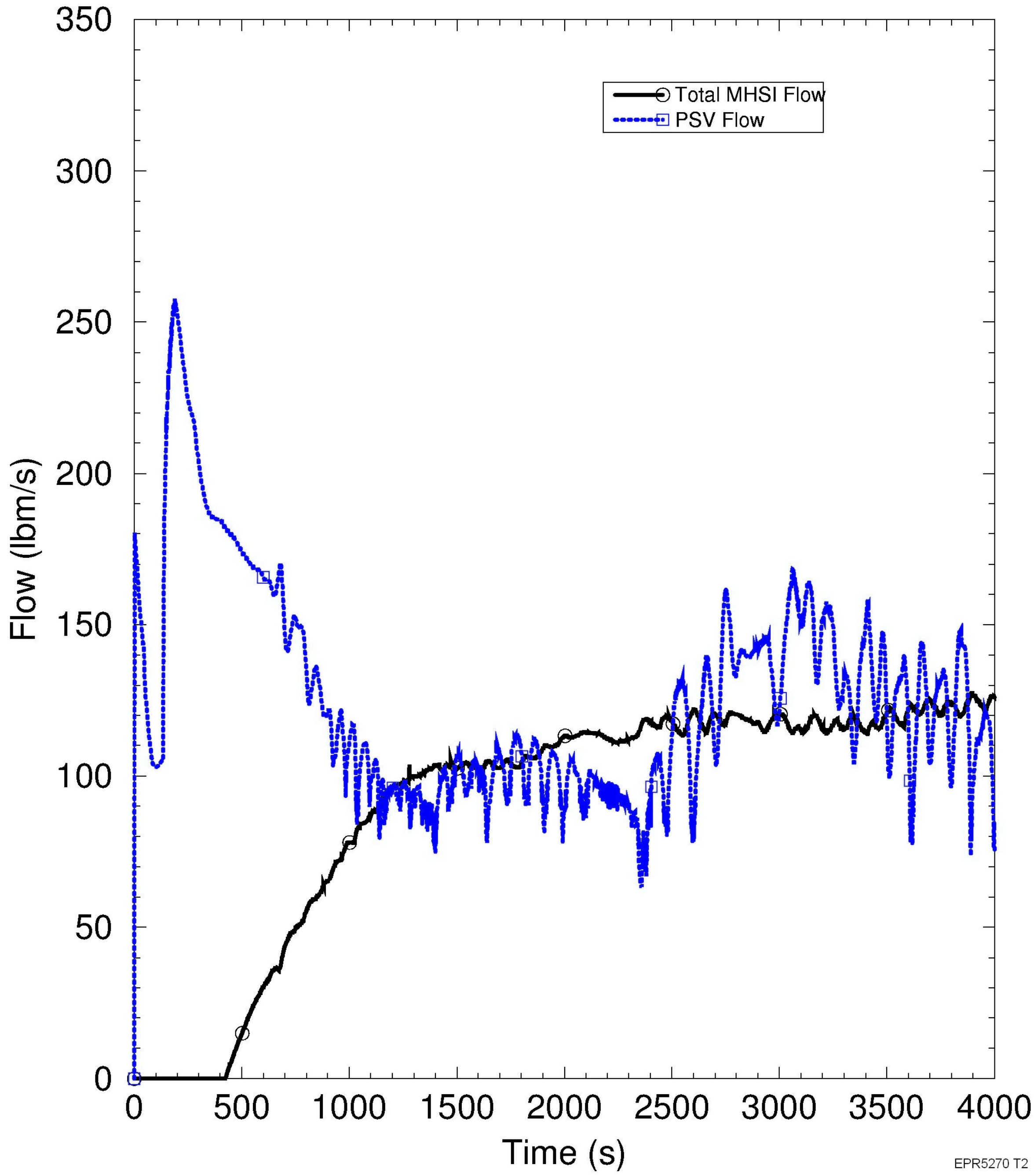

Figure 15.6-4—IOPSRV Event - Reactor Vessel Fluid Mass

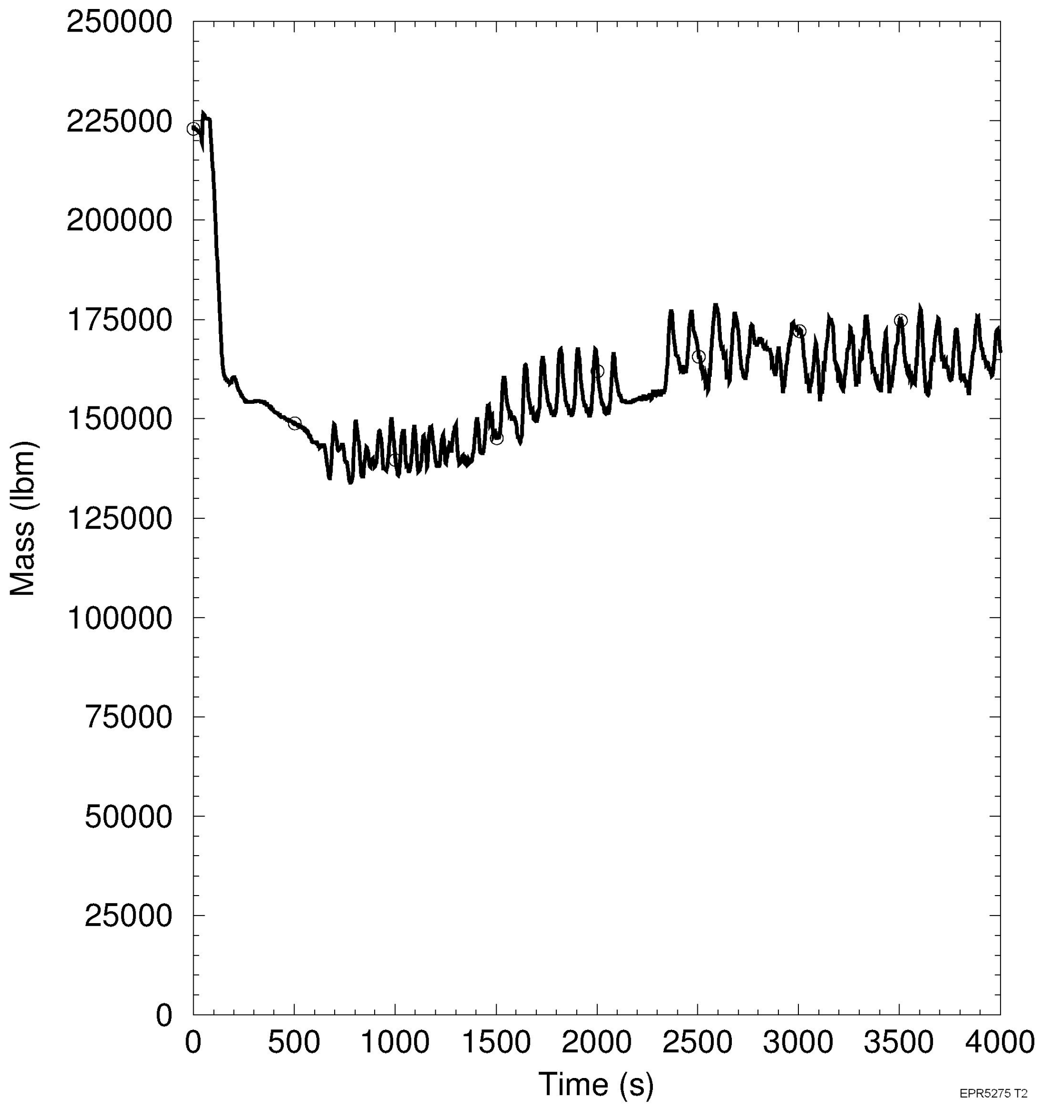

Figure 15.6-5—IOPSRV Event - Core Exit Void Fraction

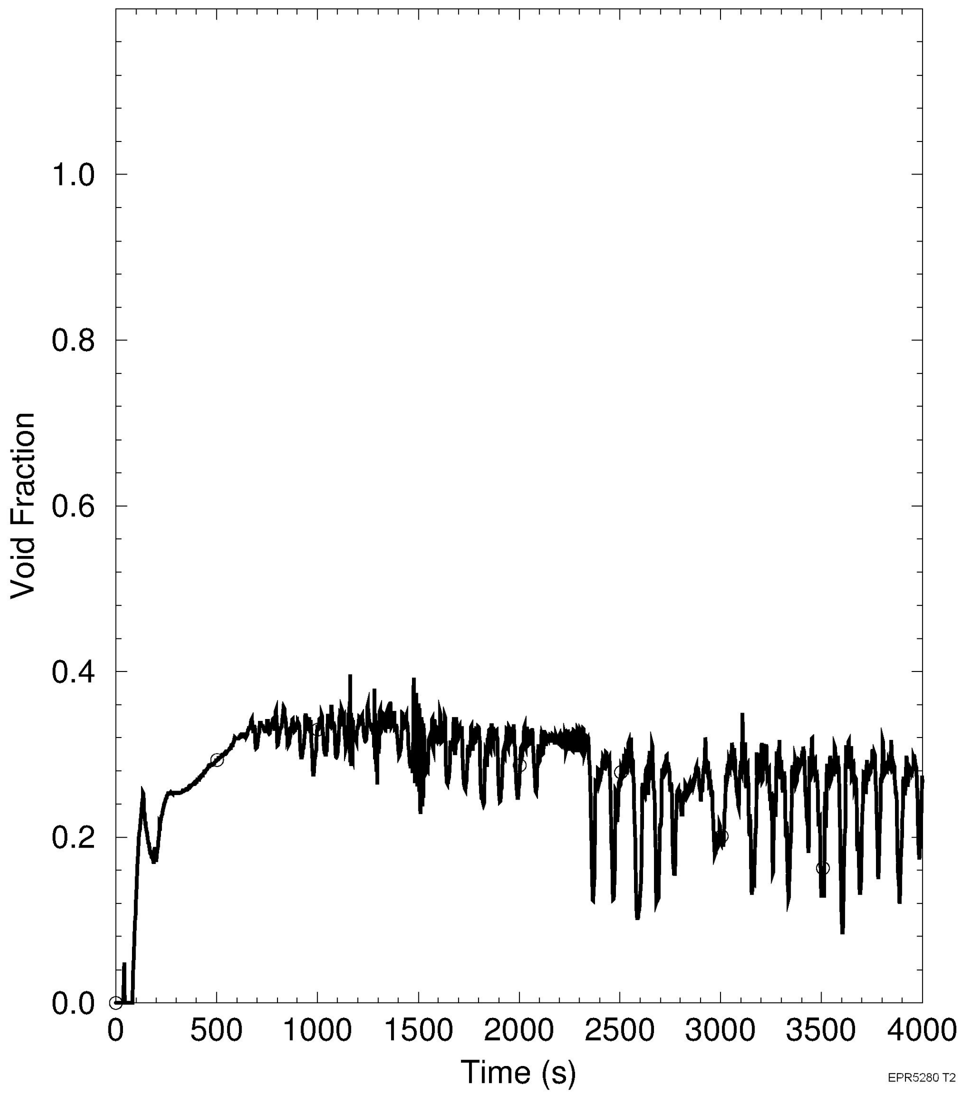

Figure 15.6-6—IOPSRV Event - Core Average Heat Flux

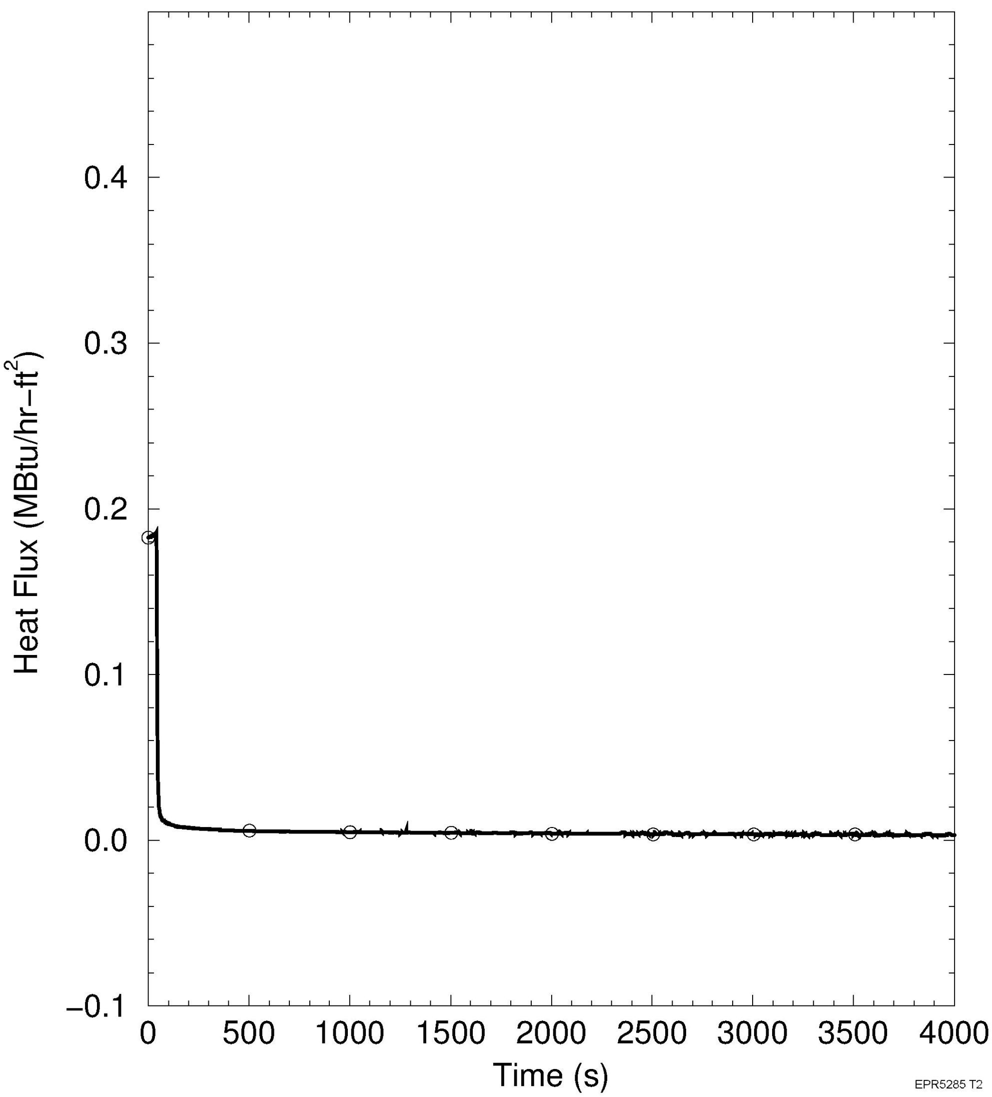

Figure 15.6-7—IOPSRV Event - Pressurizer Level

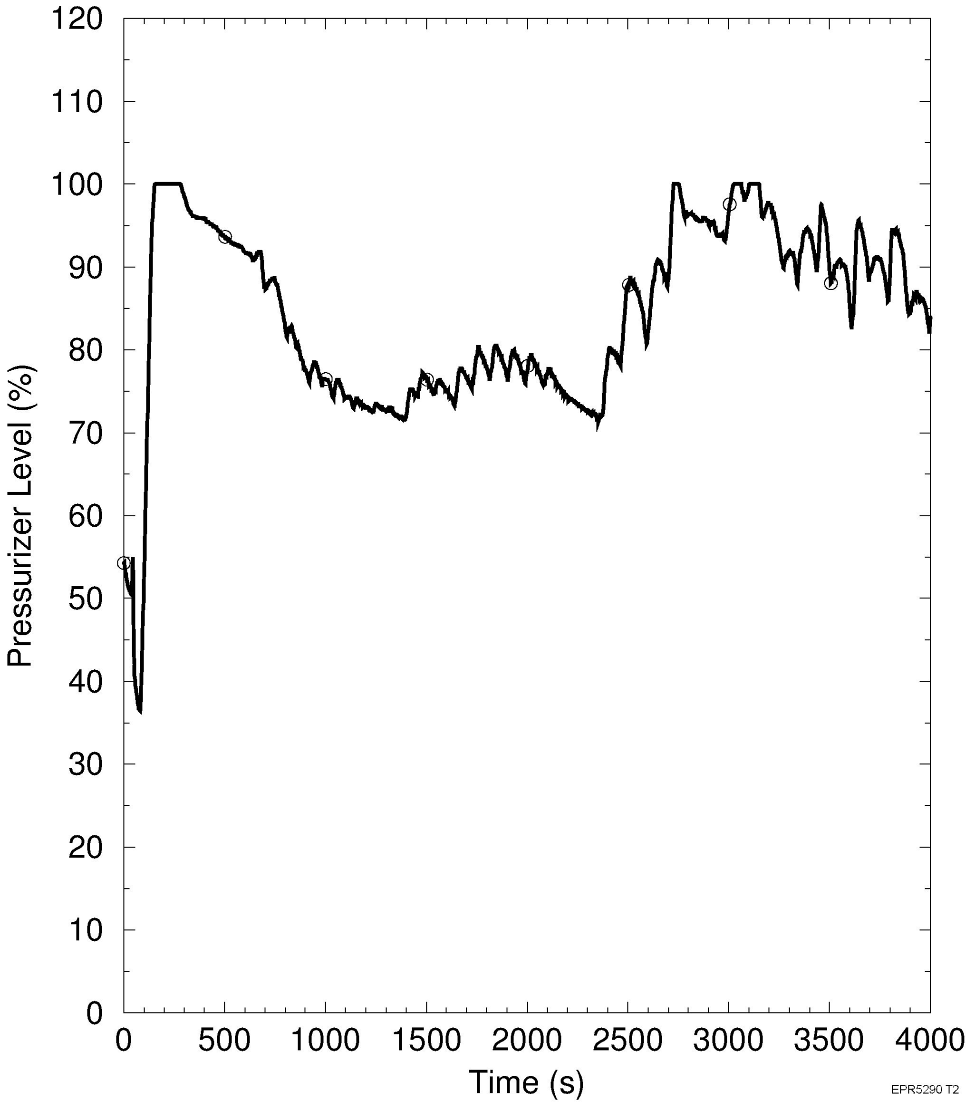

Figure 15.6-8—IOPSRV Event - Core Inlet Temperature

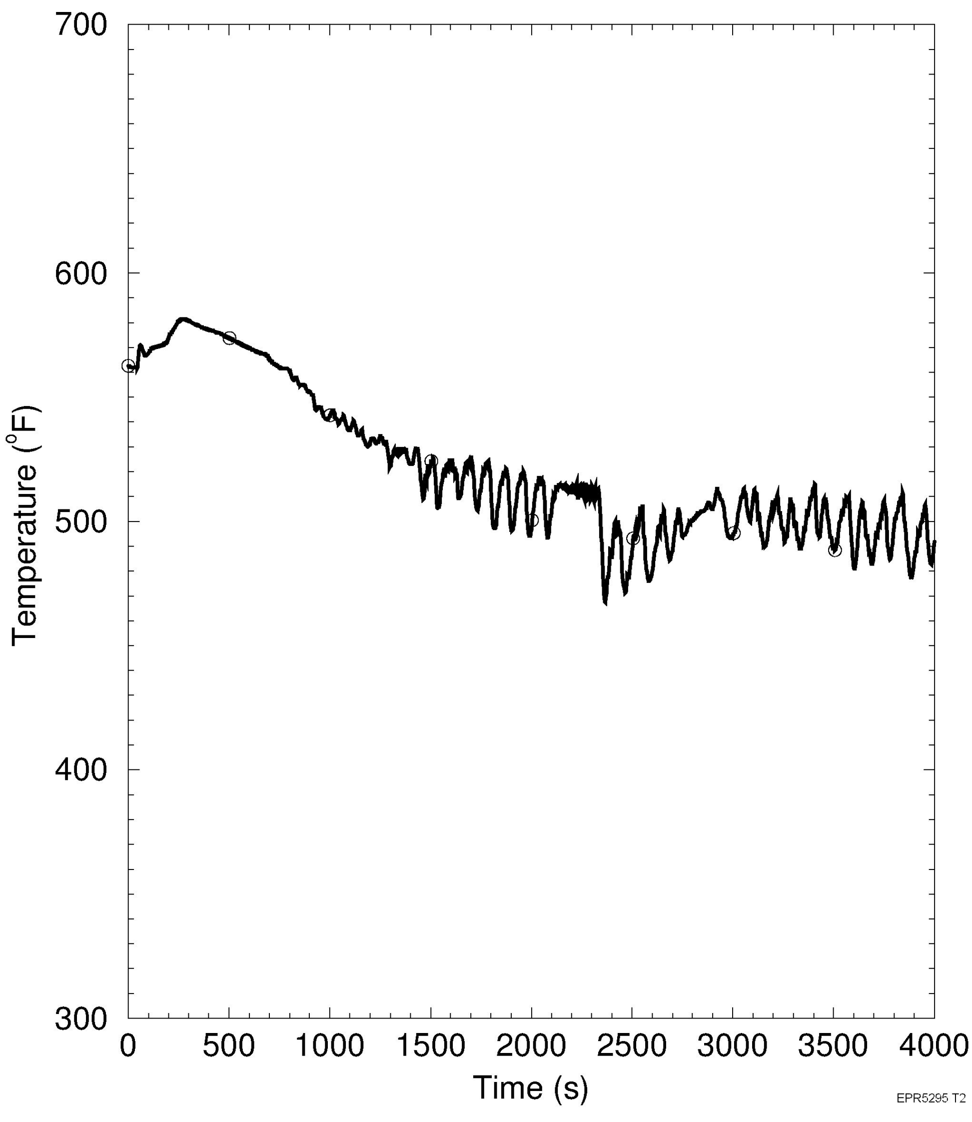

Figure 15.6-9—IOPSRV Event - RCS Average Temperature

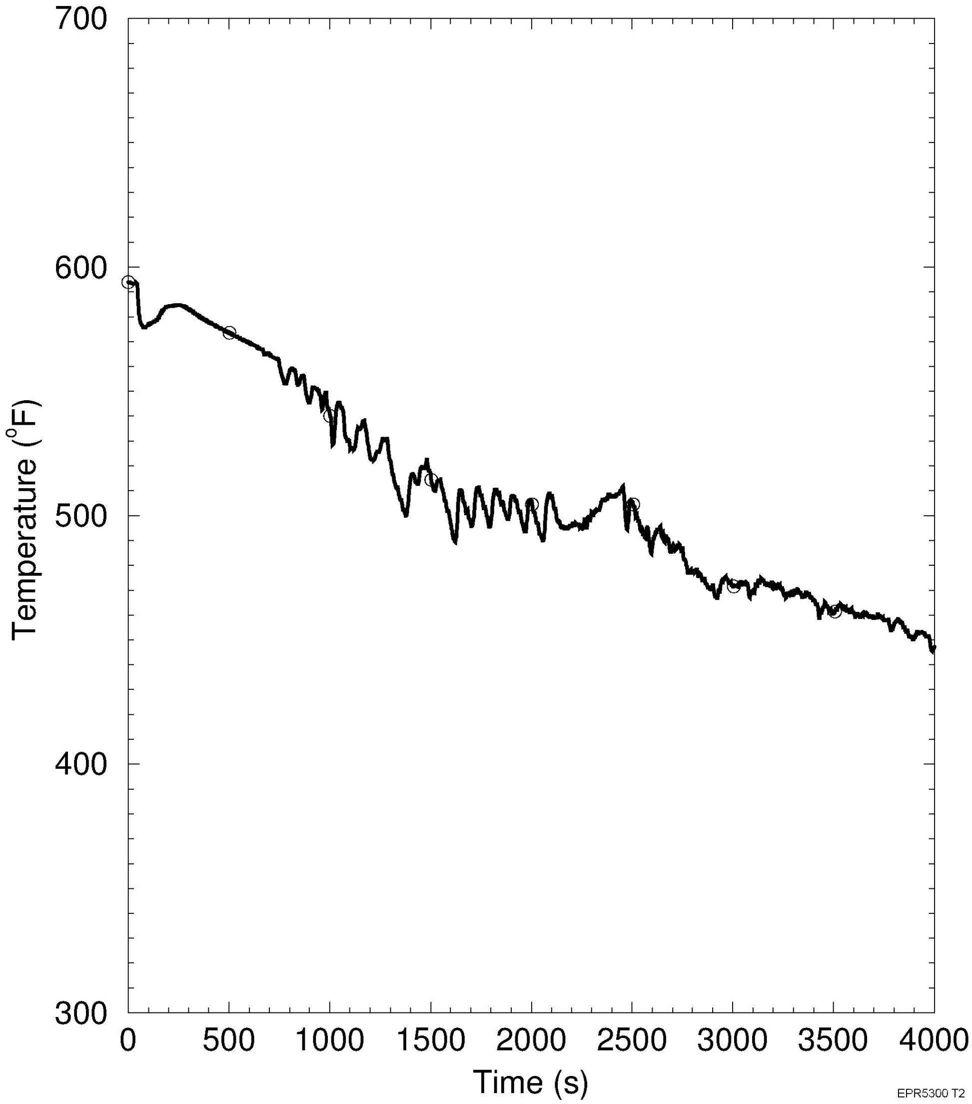

Figure 15.6-10—IOPSRV Event - SG Pressure

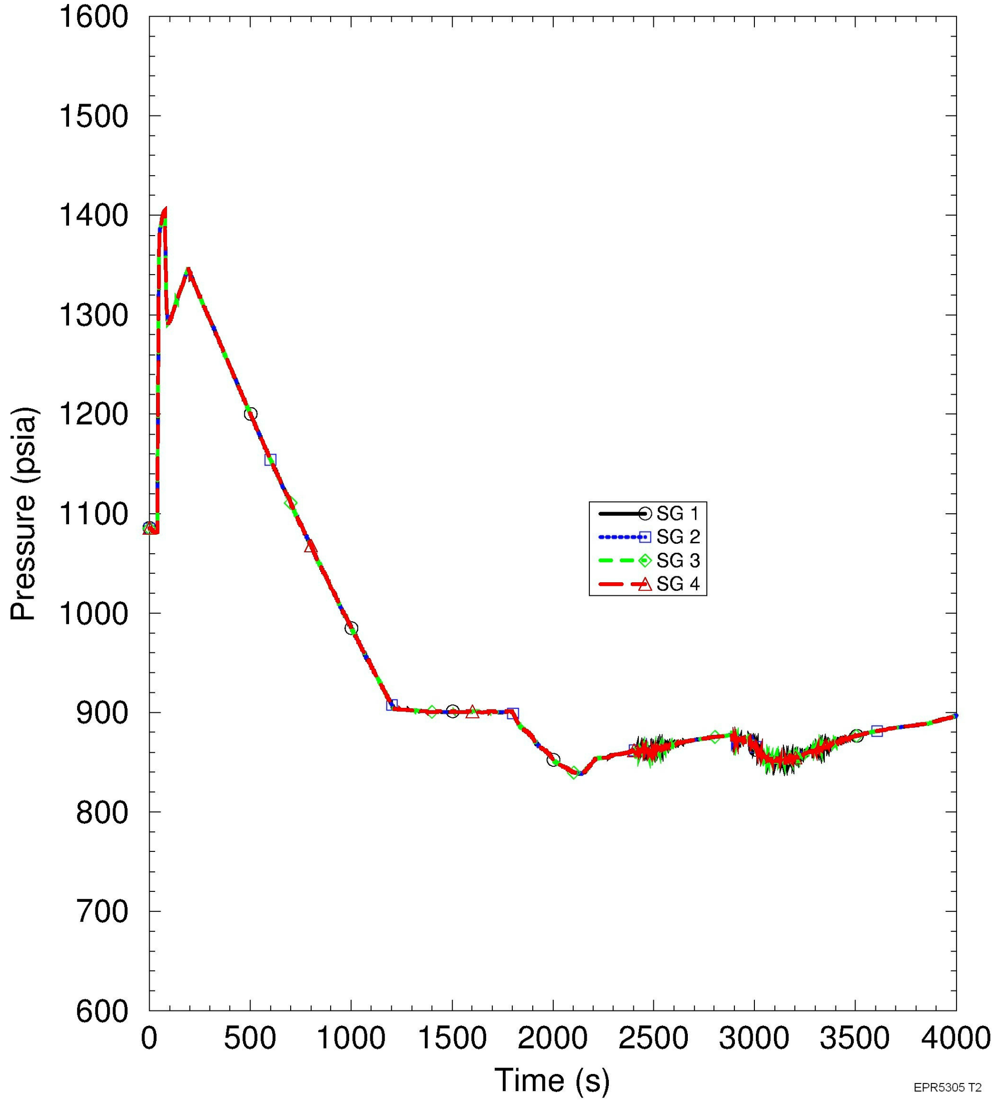

Figure 15.6-11—SGTR Radiological Case - Reactor Power

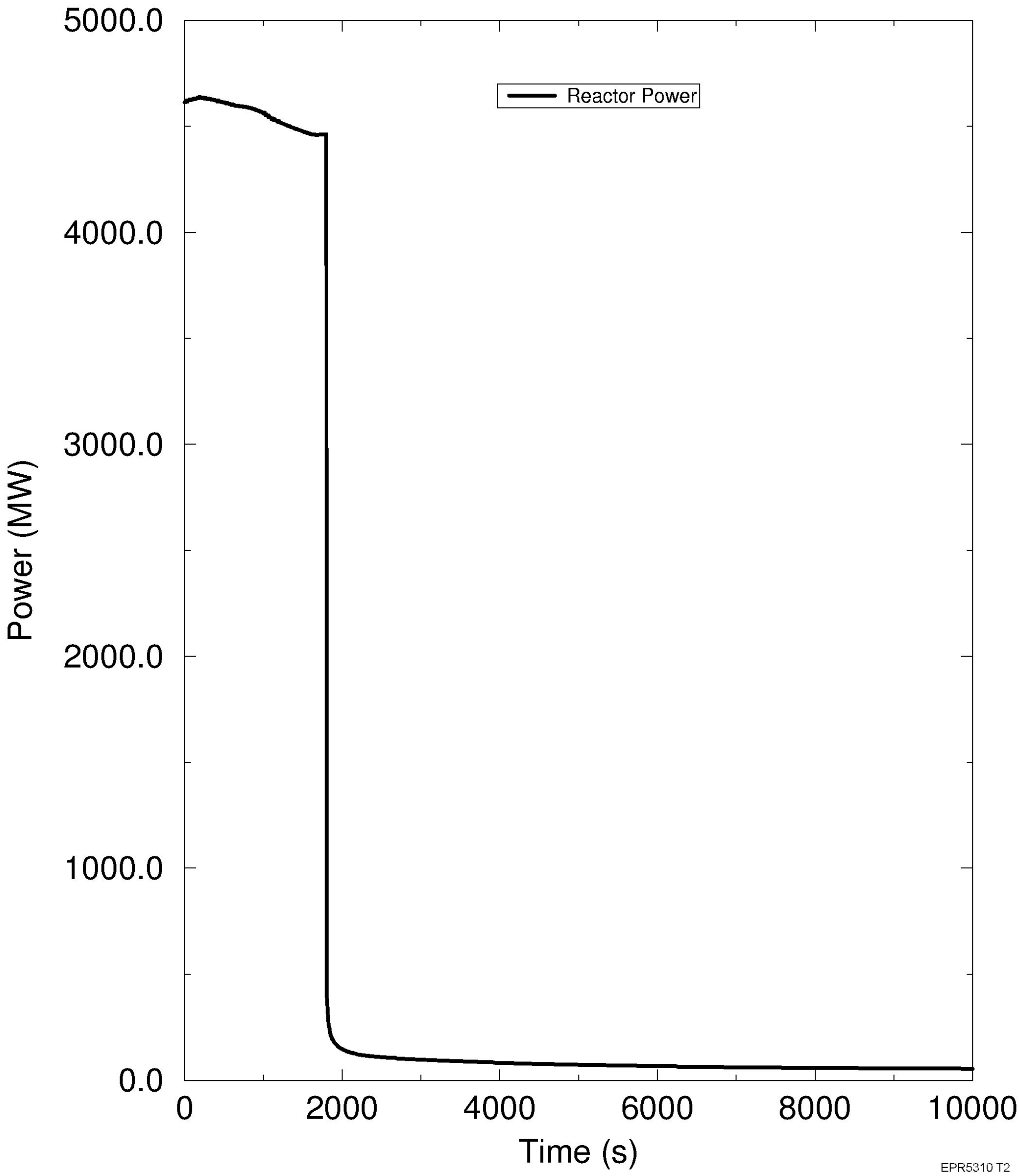

Figure 15.6-12—SGTR Radiological Case Pressurizer and Affected SG Dome Pressure

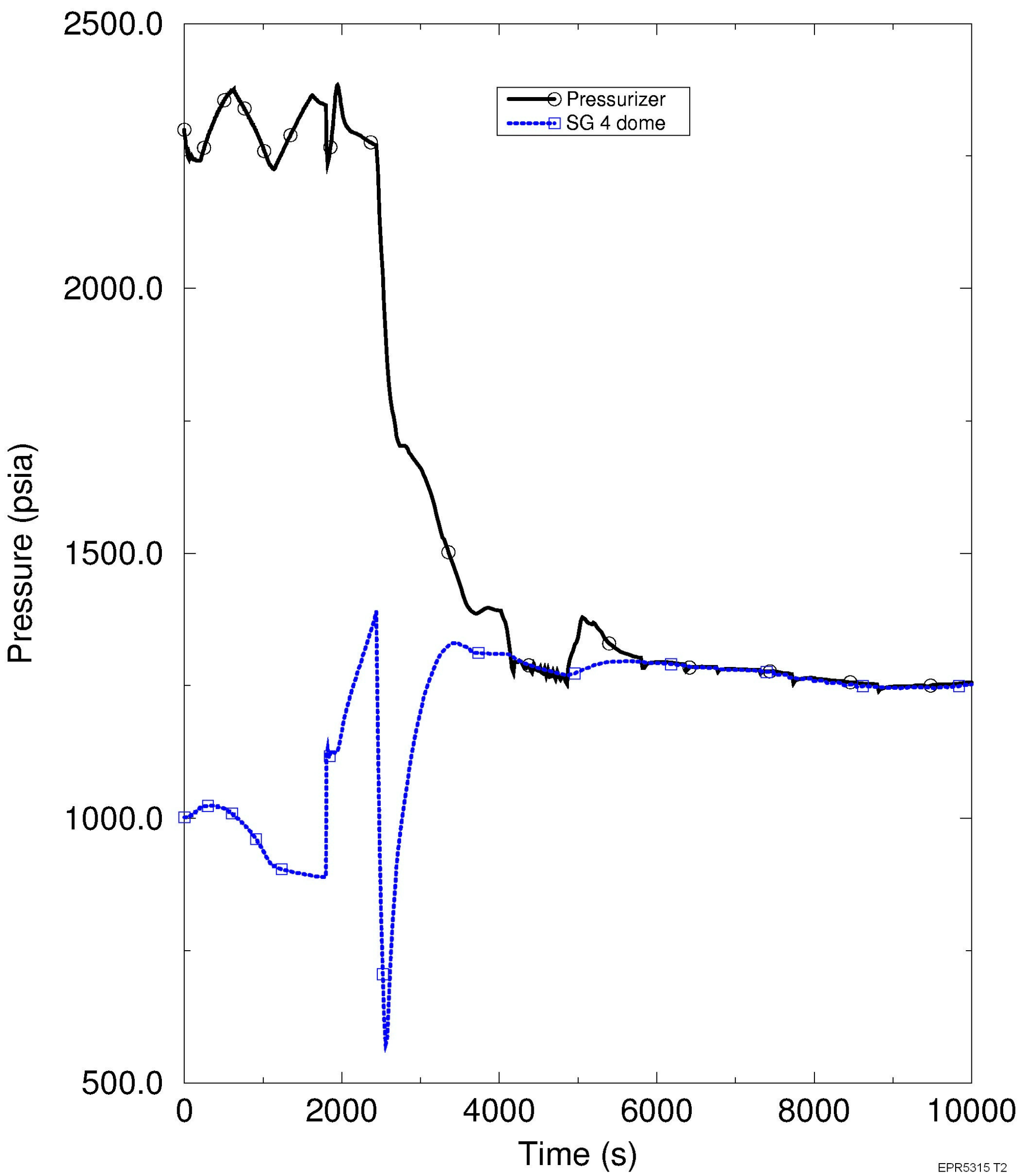

Figure 15.6-13—SGTR Radiological Case - SG Blowdown Flow Rates

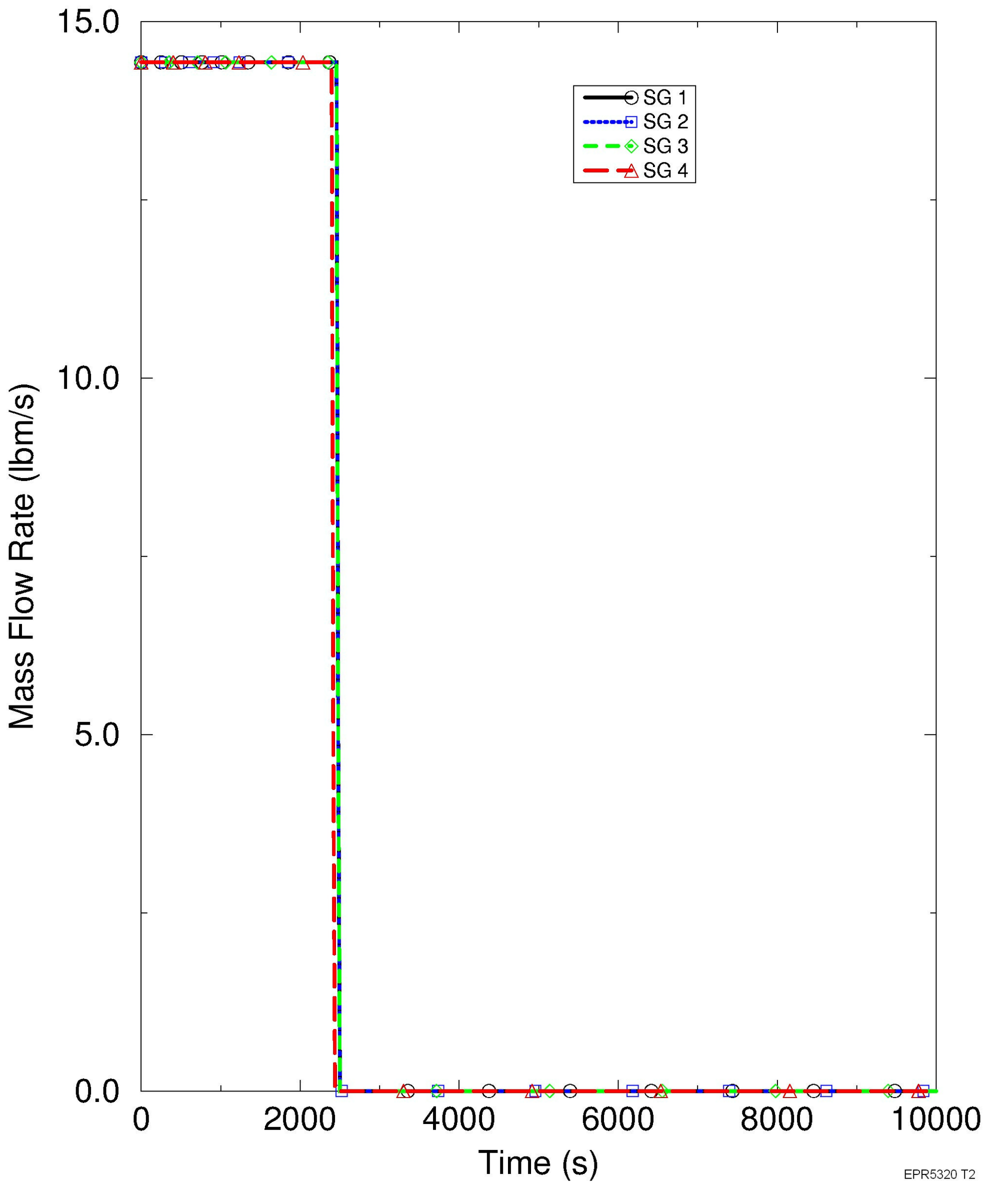

Figure 15.6-14—SGTR Radiological Case - EFW Flow Rates

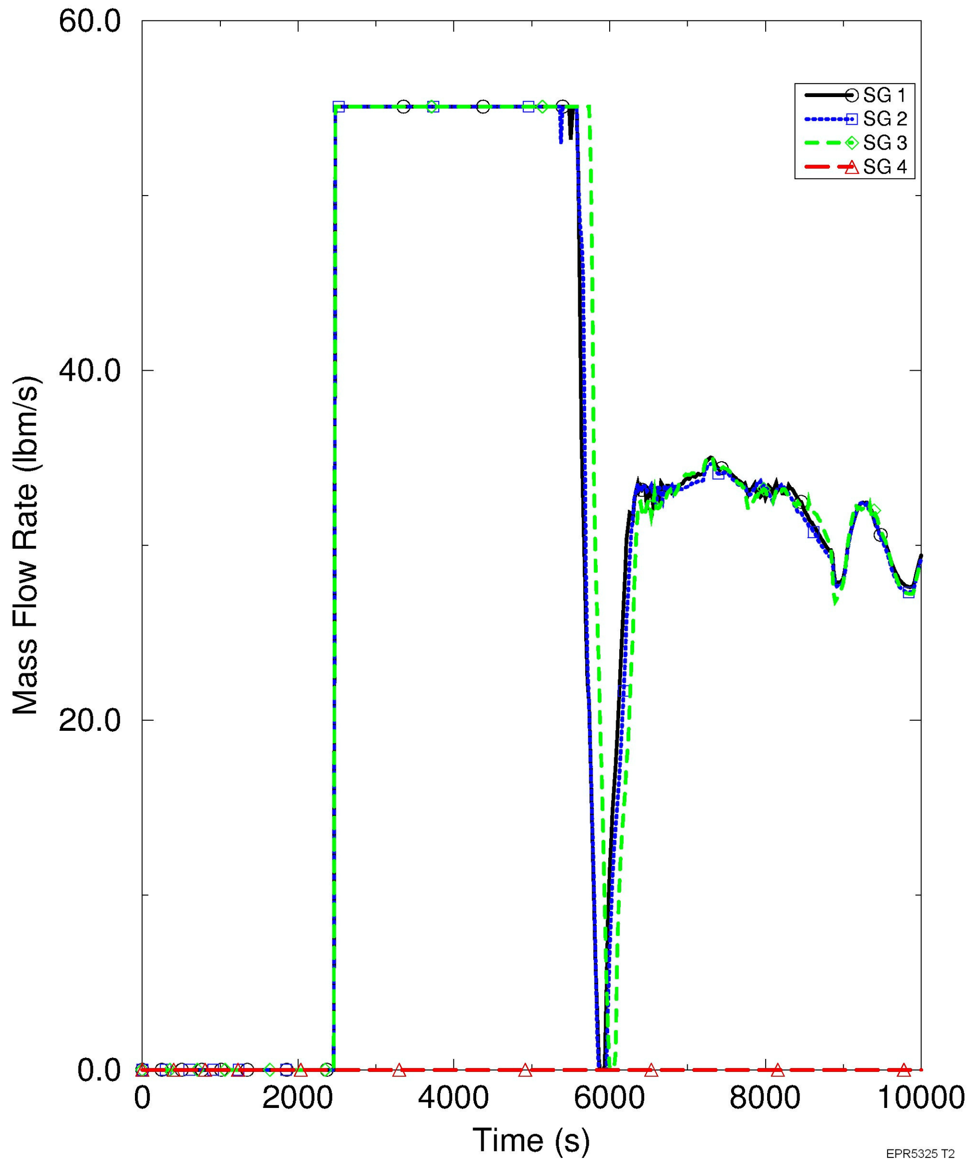

Figure 15.6-15—SGTR Radiological Case - Total MHSI Flow Rate

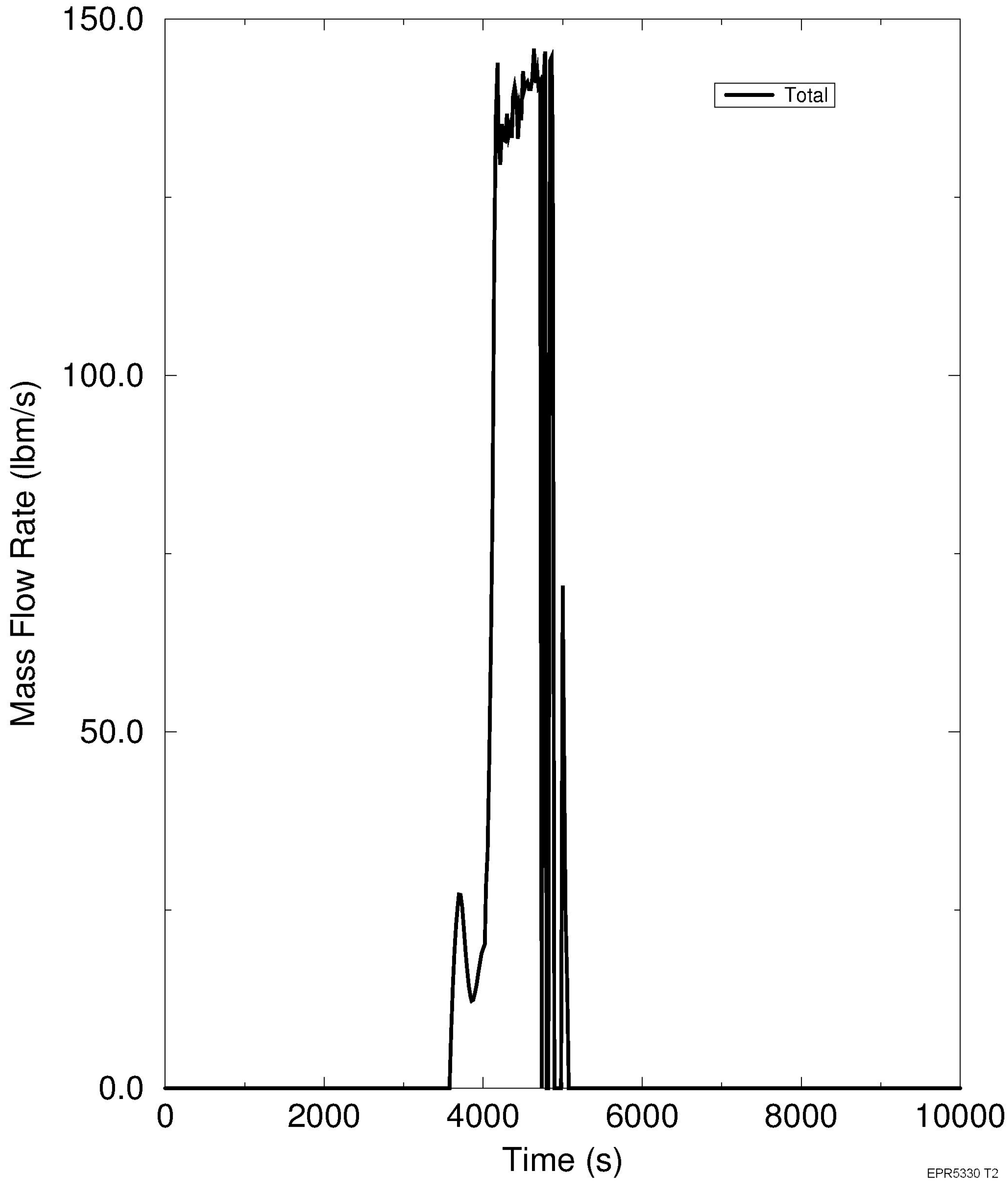

Figure 15.6-16—SGTR Radiological Case - EBS Flow Rate

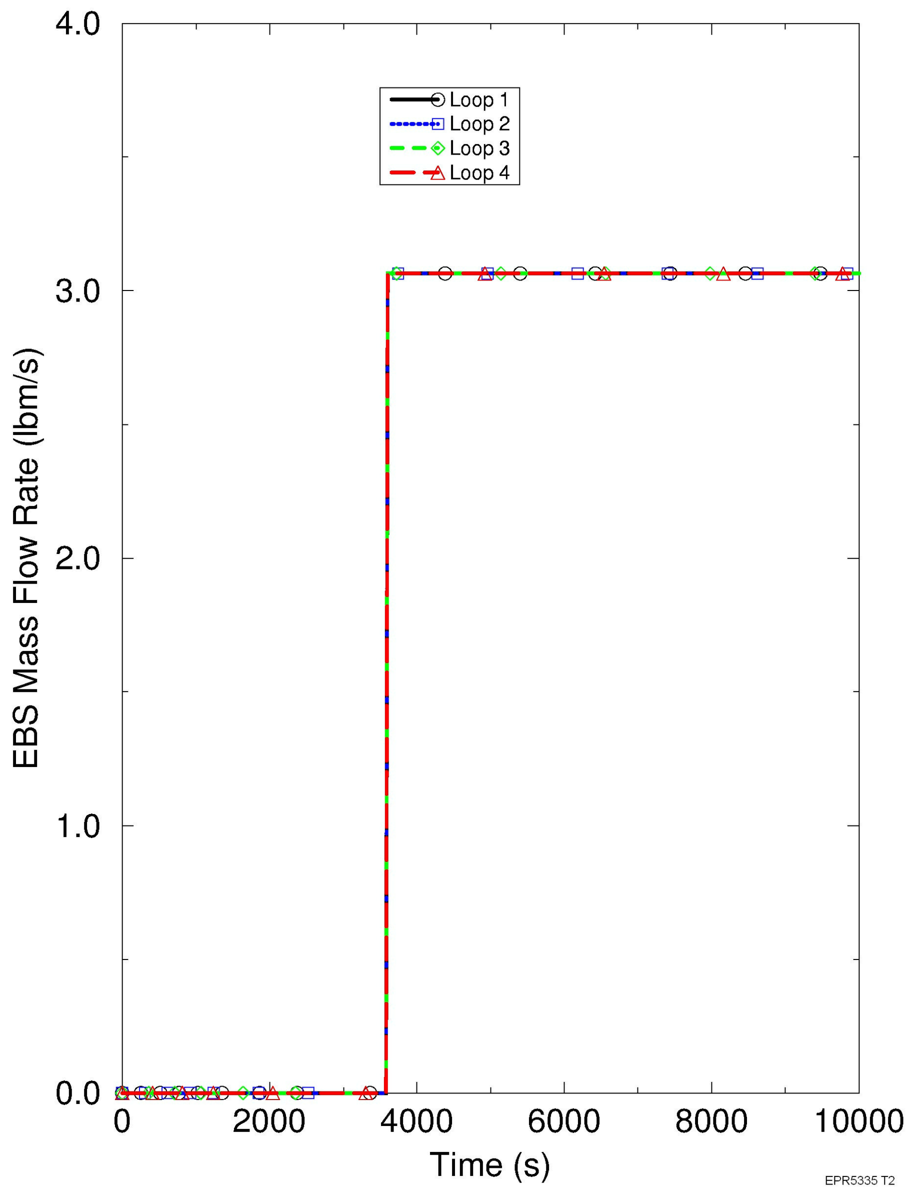

Figure 15.6-17—SGTR Radiological Case - PSRV Flow Rate

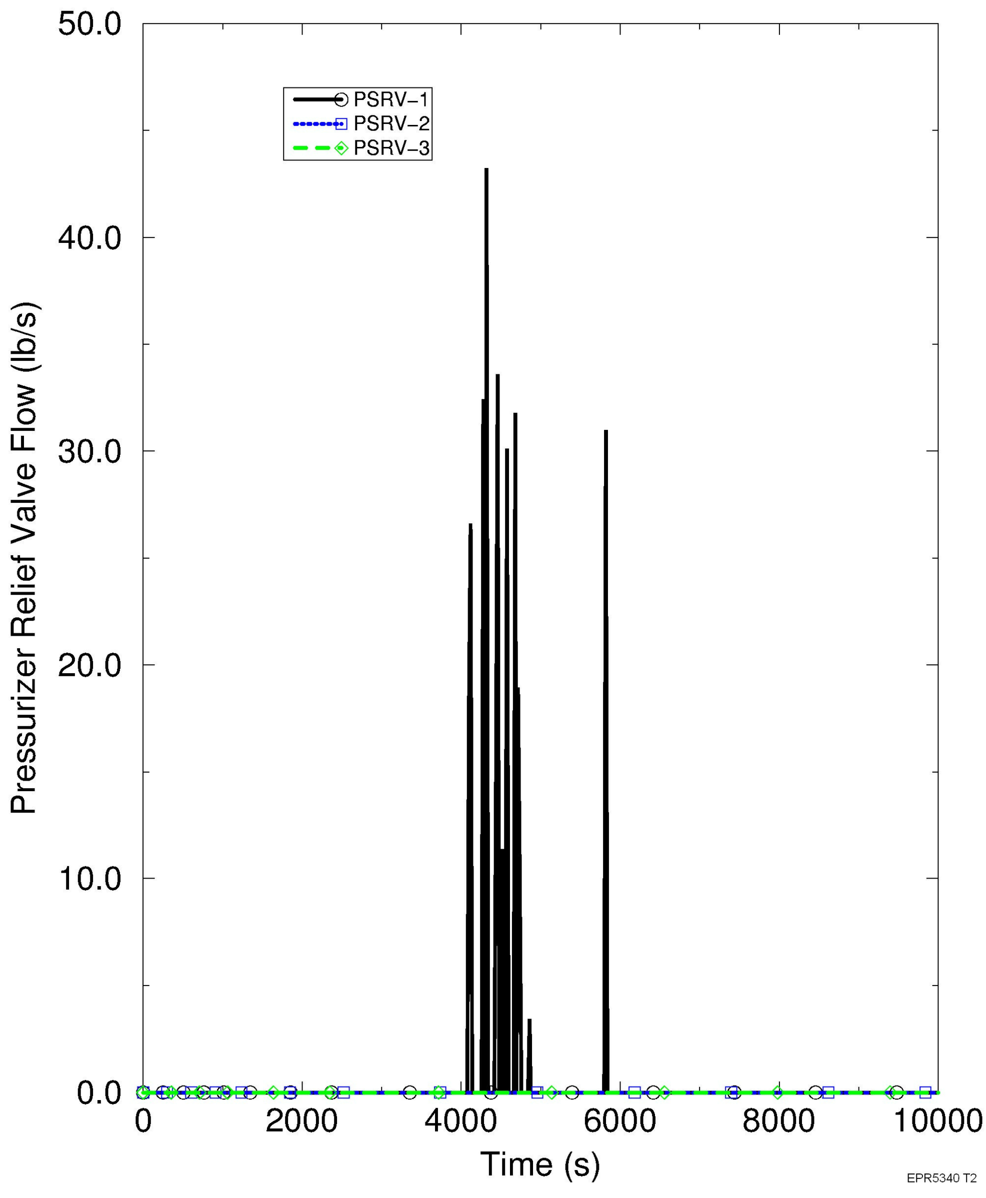

Figure 15.6-18—SGTR Radiological Case - Break Flow Rate

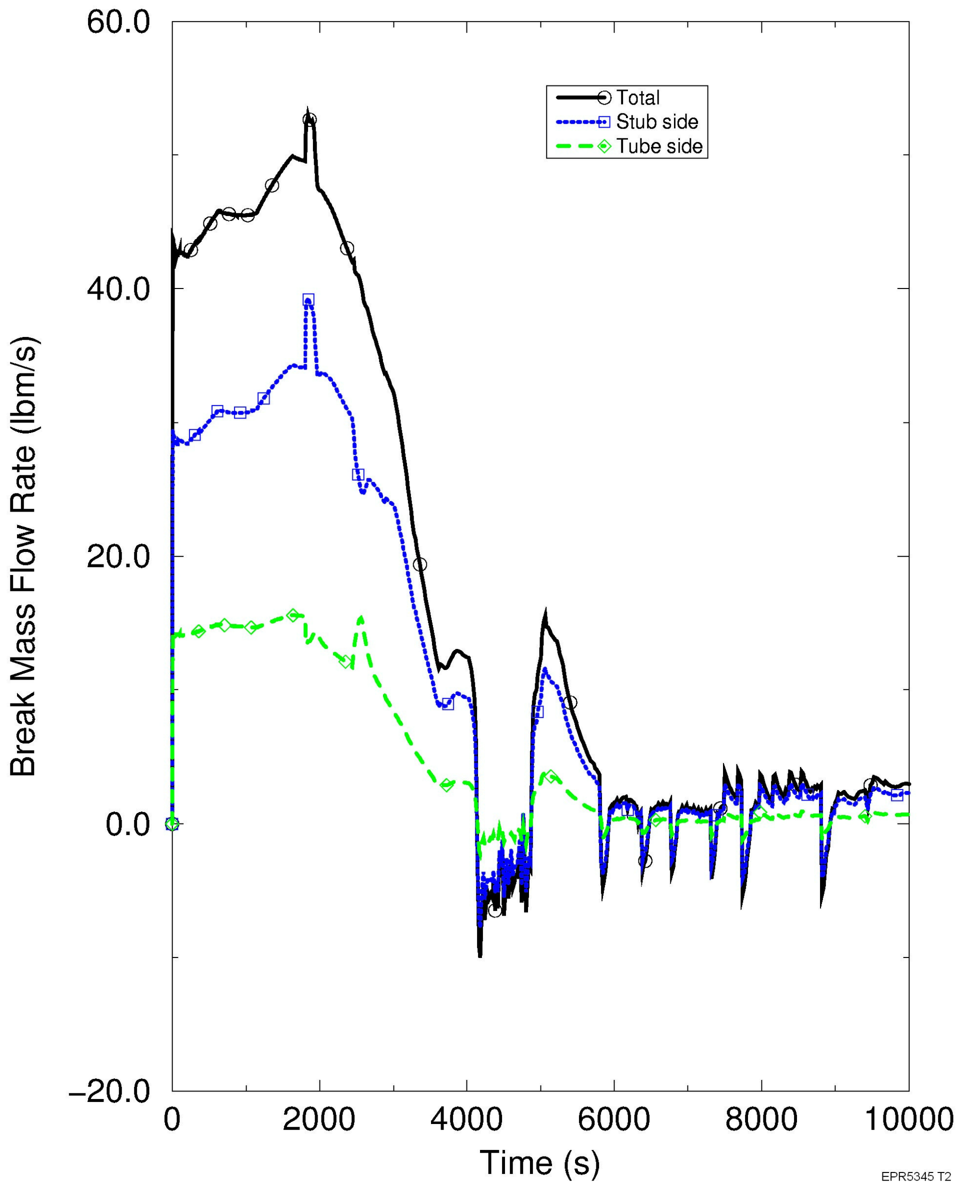

Figure 15.6-19—SGTR Radiological Case - Core Exit Subcooling

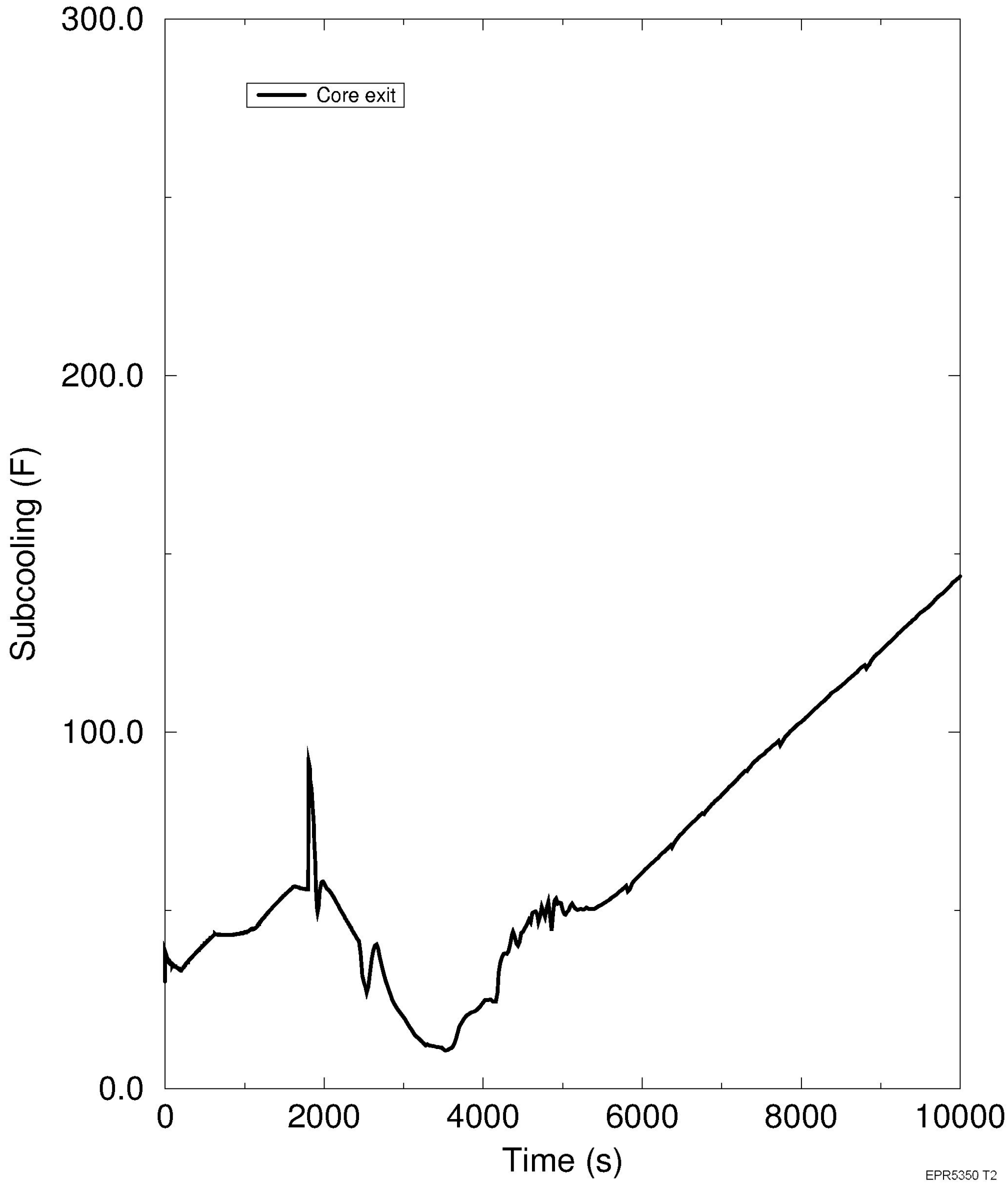

Figure 15.6-20—SGTR Radiological Case - Pressurizer Level

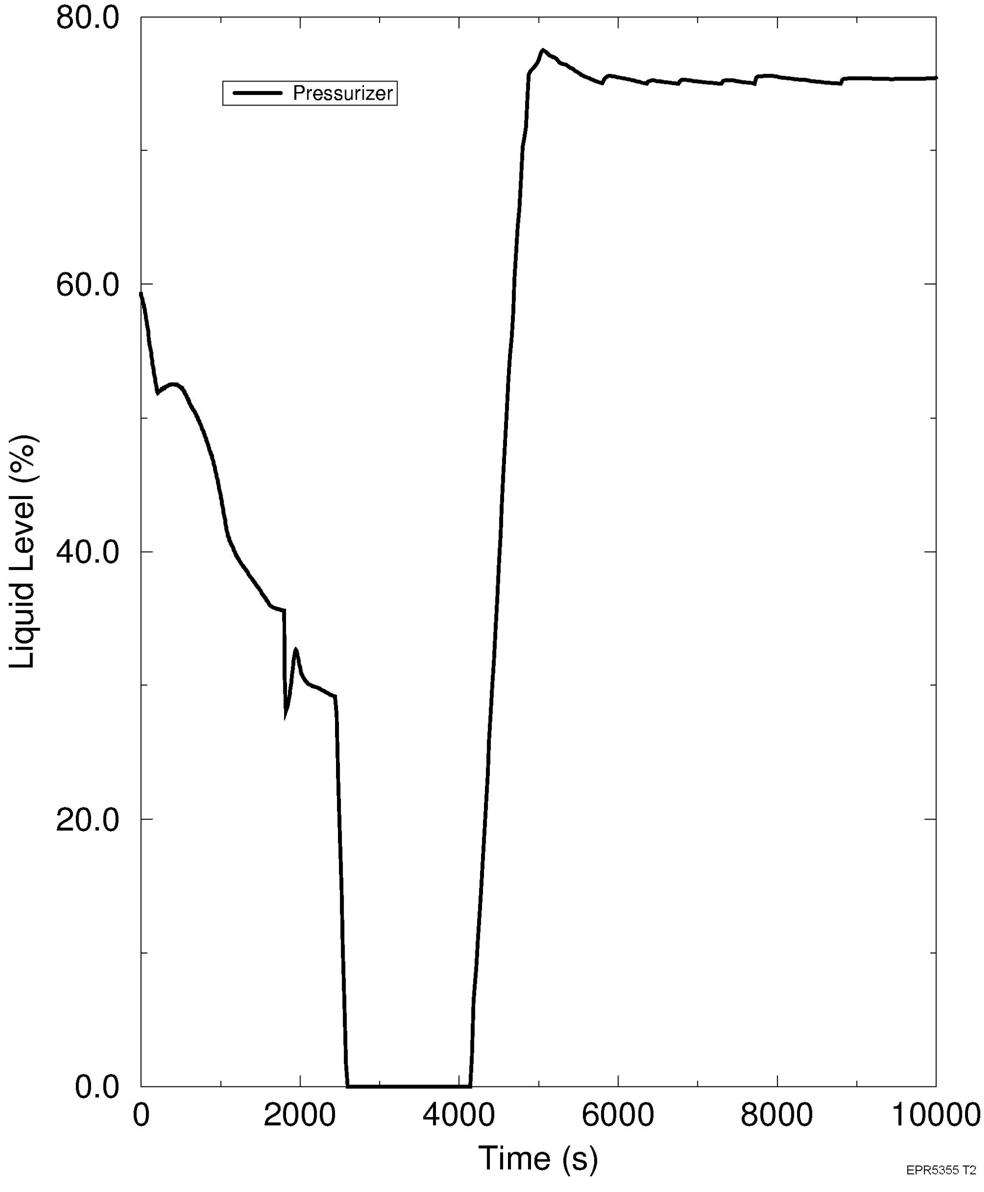

Figure 15.6-21—SGTR Radiological Case - SG Wide Range Levels

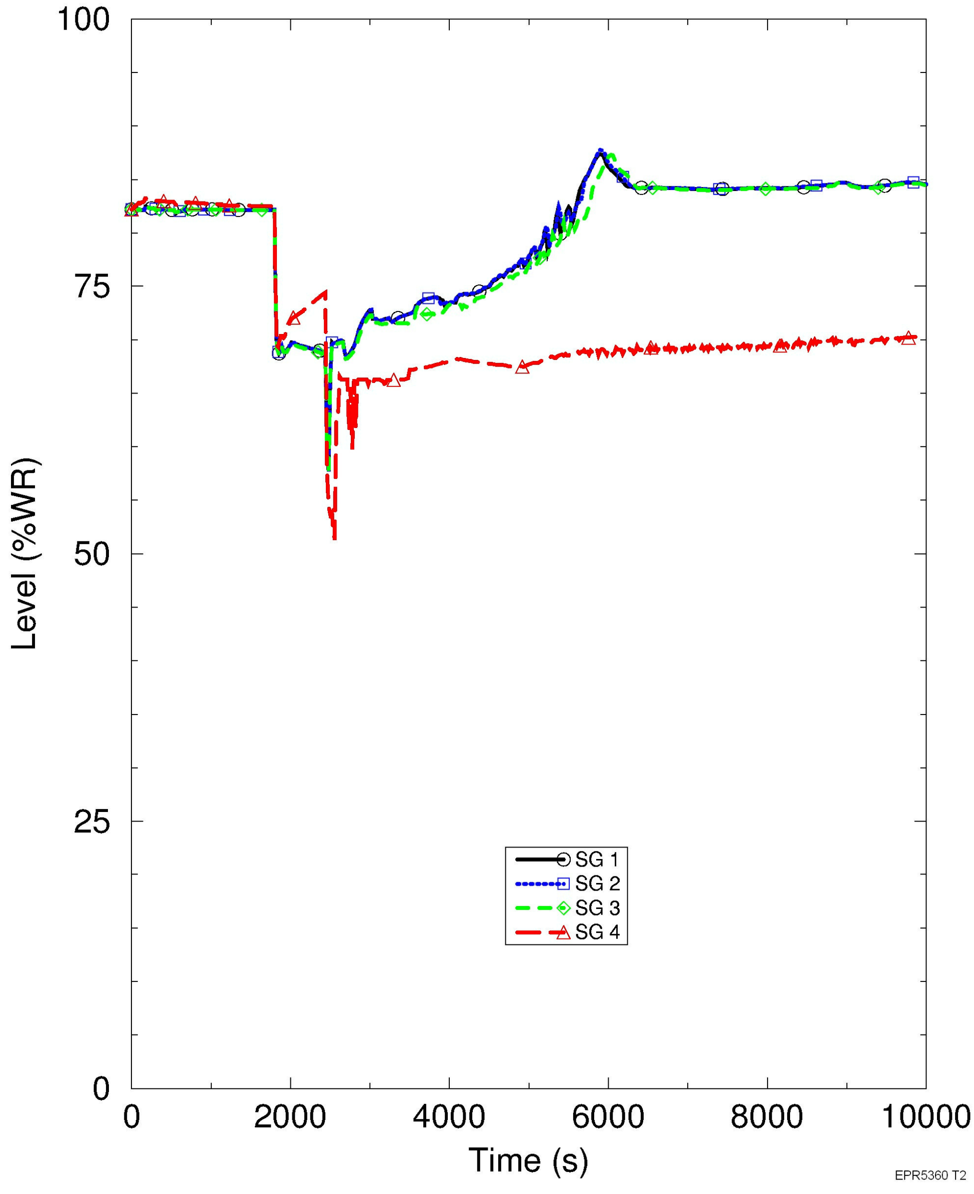

Figure 15.6-22—SGTR Radiological Case - Affected SG Liquid Volume

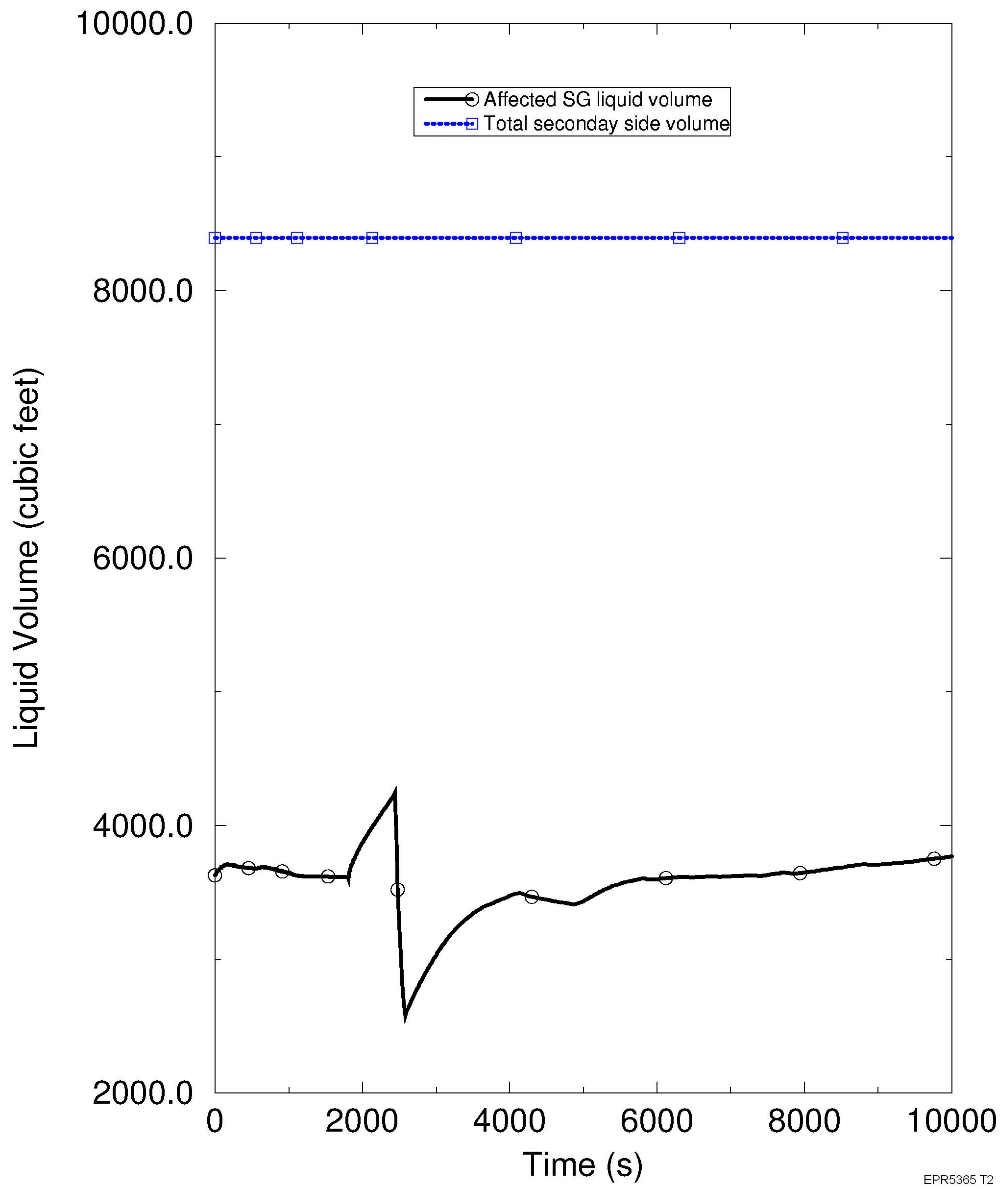

Figure 15.6-23—SGTR Radiological Case - Integrated Break Mass Flow

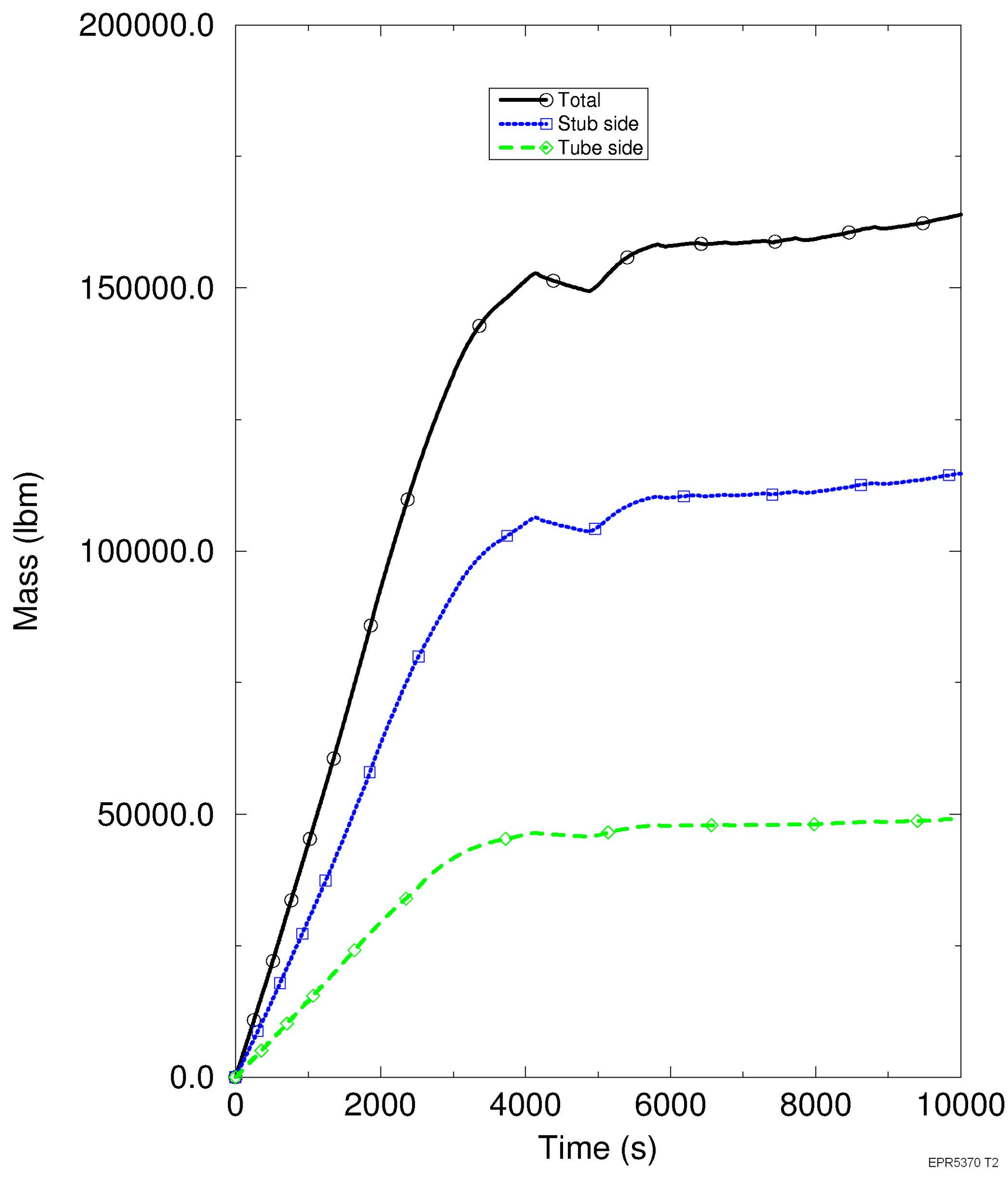

Figure 15.6-24—SGTR Radiological Case - Integrated Steam Mass Release

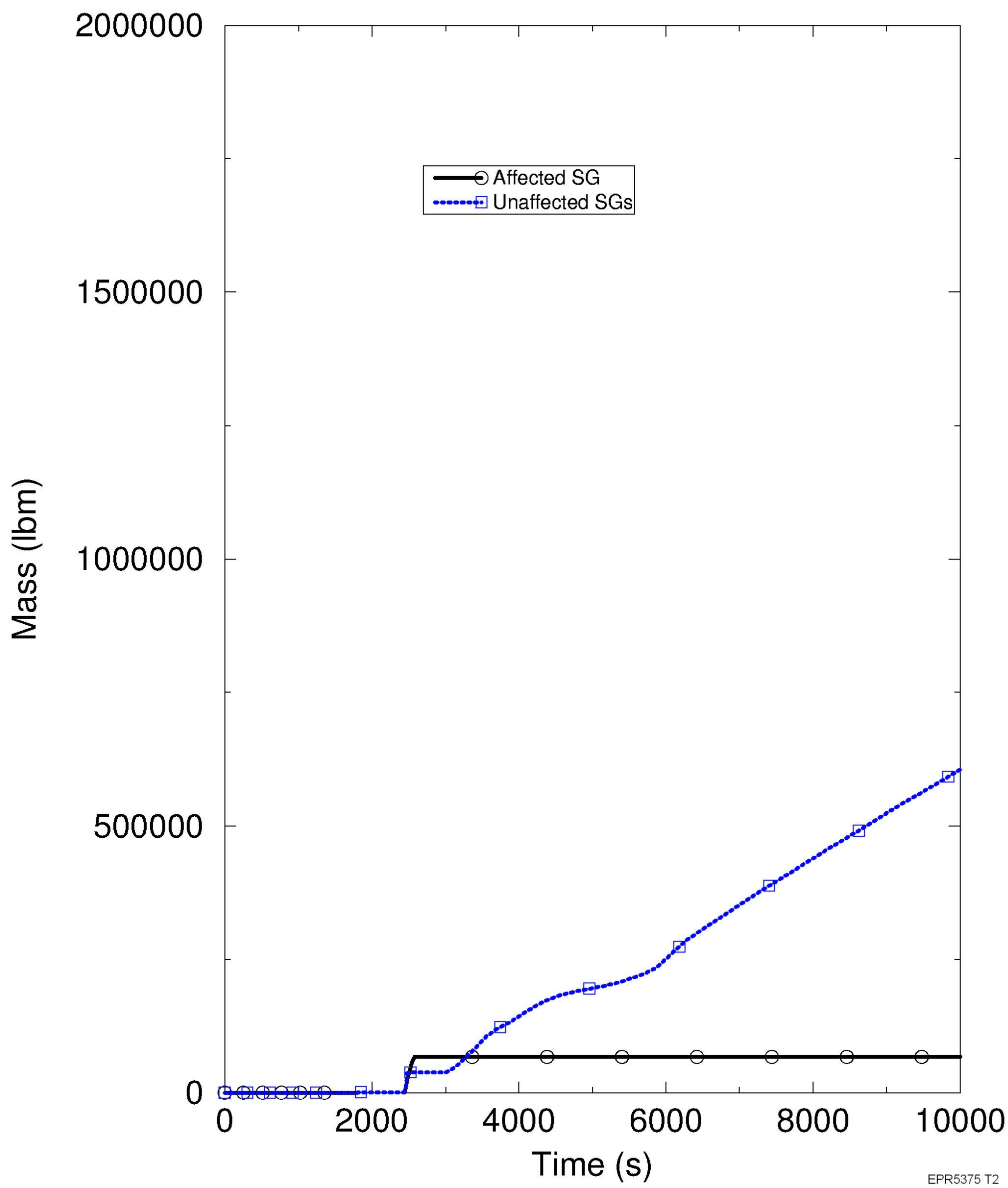

Figure 15.6-25—SGTR Radiological Case - Integrated Mass Flashed

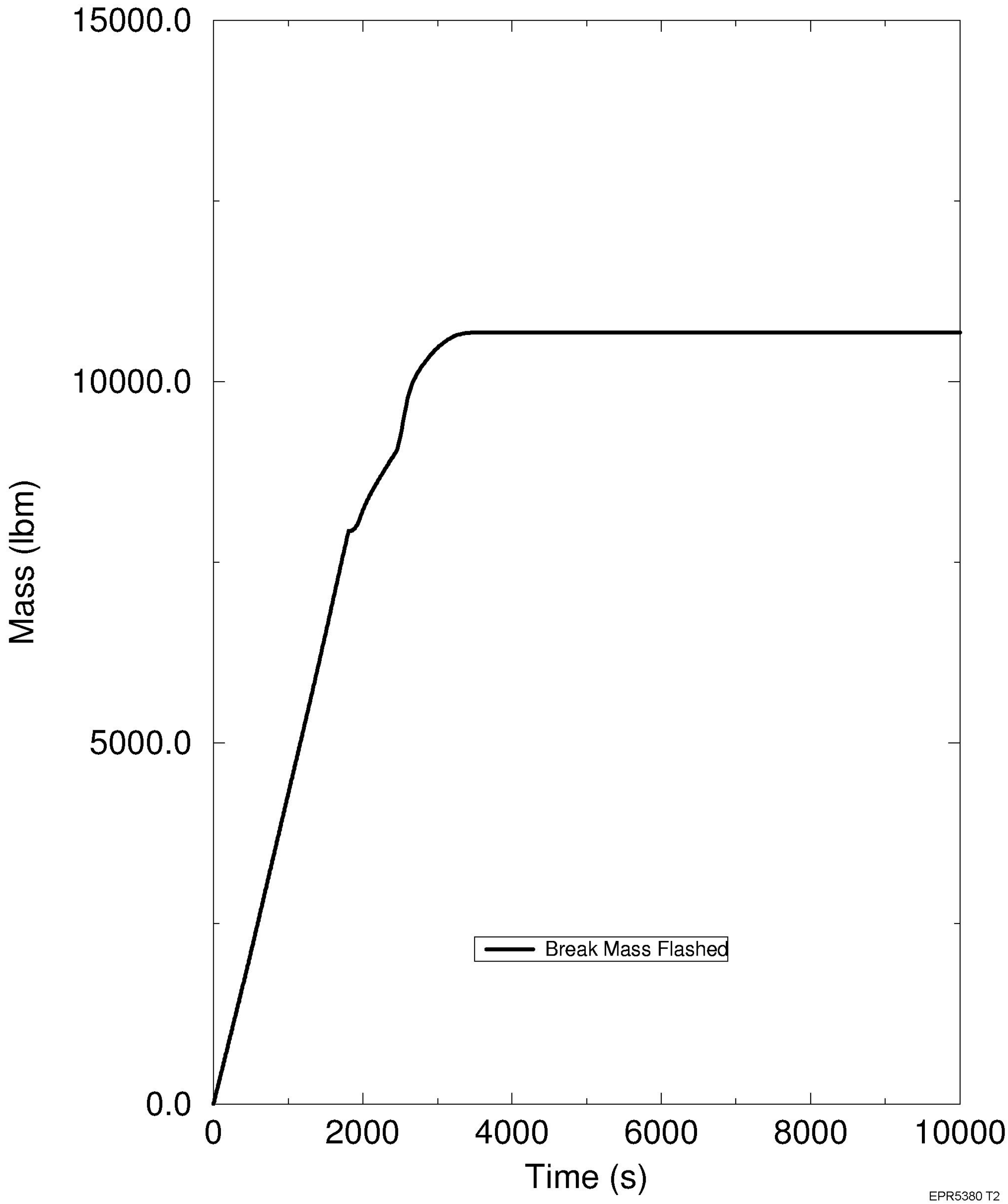

Figure 15.6-26—SGTR Radiological Case - Affected SG Apex Liquid Fractions

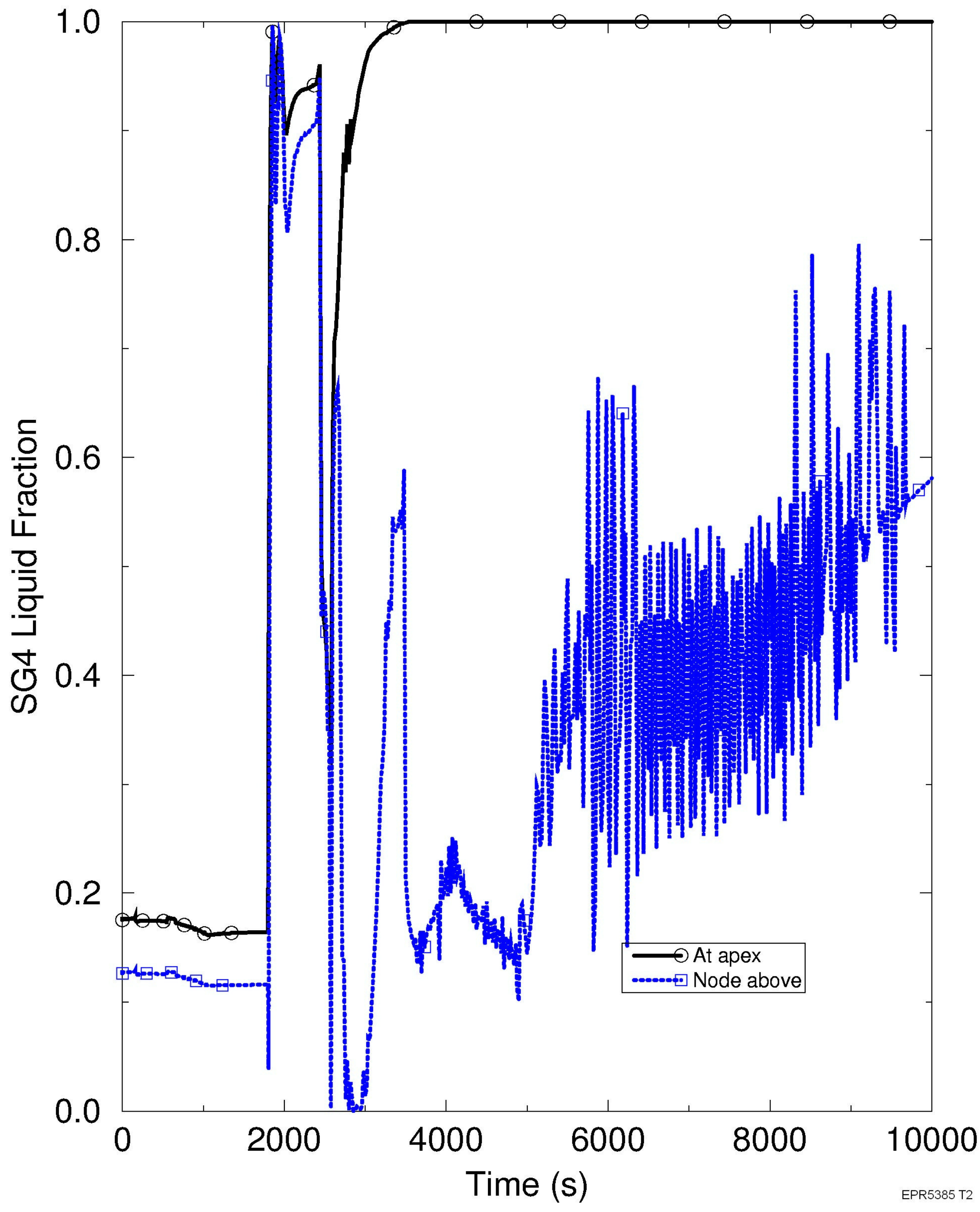

Figure 15.6-27—RLBLOCA - PCT Independent of Elevation for the Limiting PCT Case (Cycle 1)

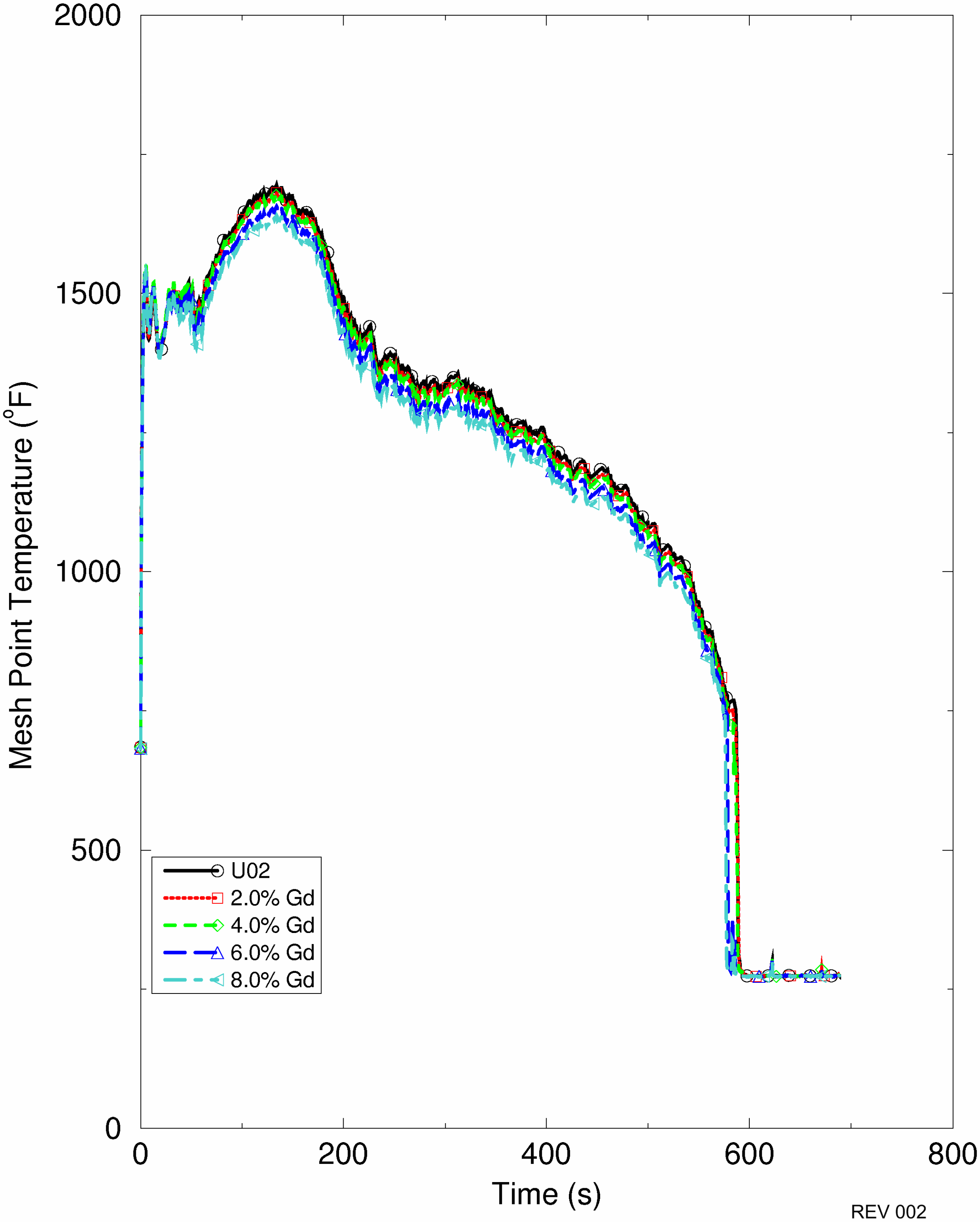

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Figure 15.6-28—RLBLOCA - PCT Independent Elevation for the Limiting PCT Case Hot Rod (Cycle 1)

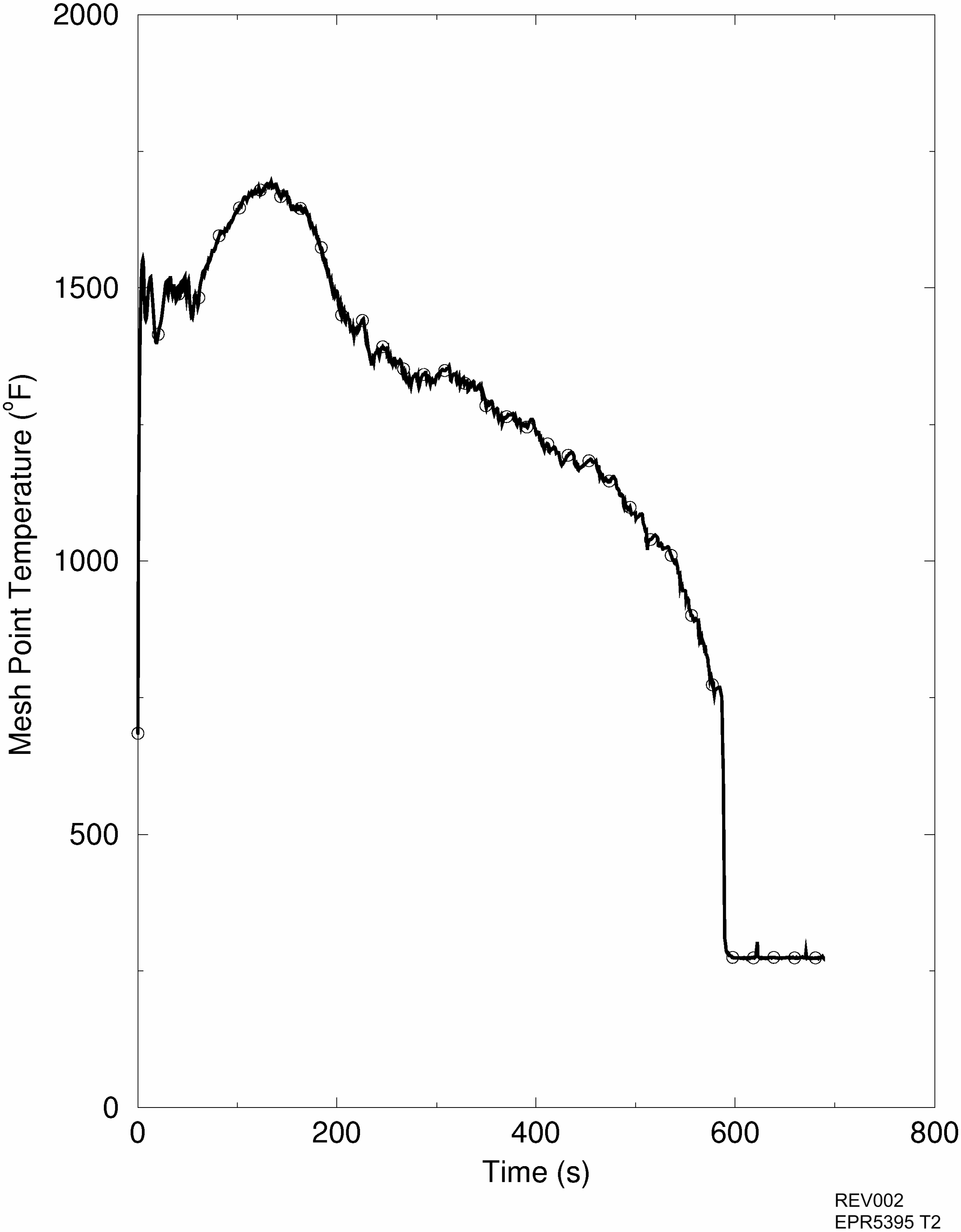


Figure 15.6-29—RLBLOCA - Primary System Pressure for the Limiting PCT Case (Cycle 1)

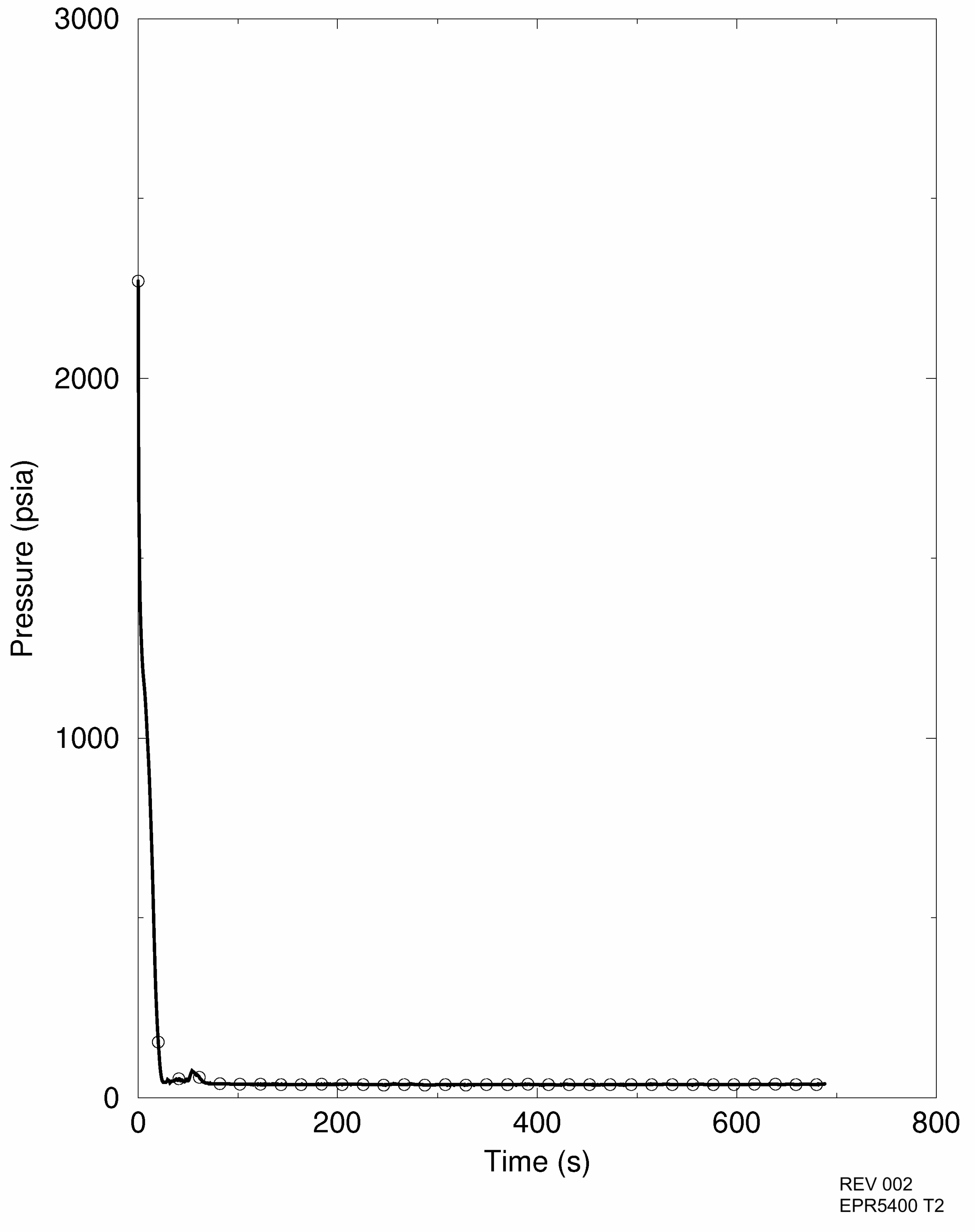


Figure 15.6-30—RLBLOCA - Flows Supplied to ECCS (includes Accumulator, MHSI and LHSI) for the Limiting PCT Case (Cycle 1)

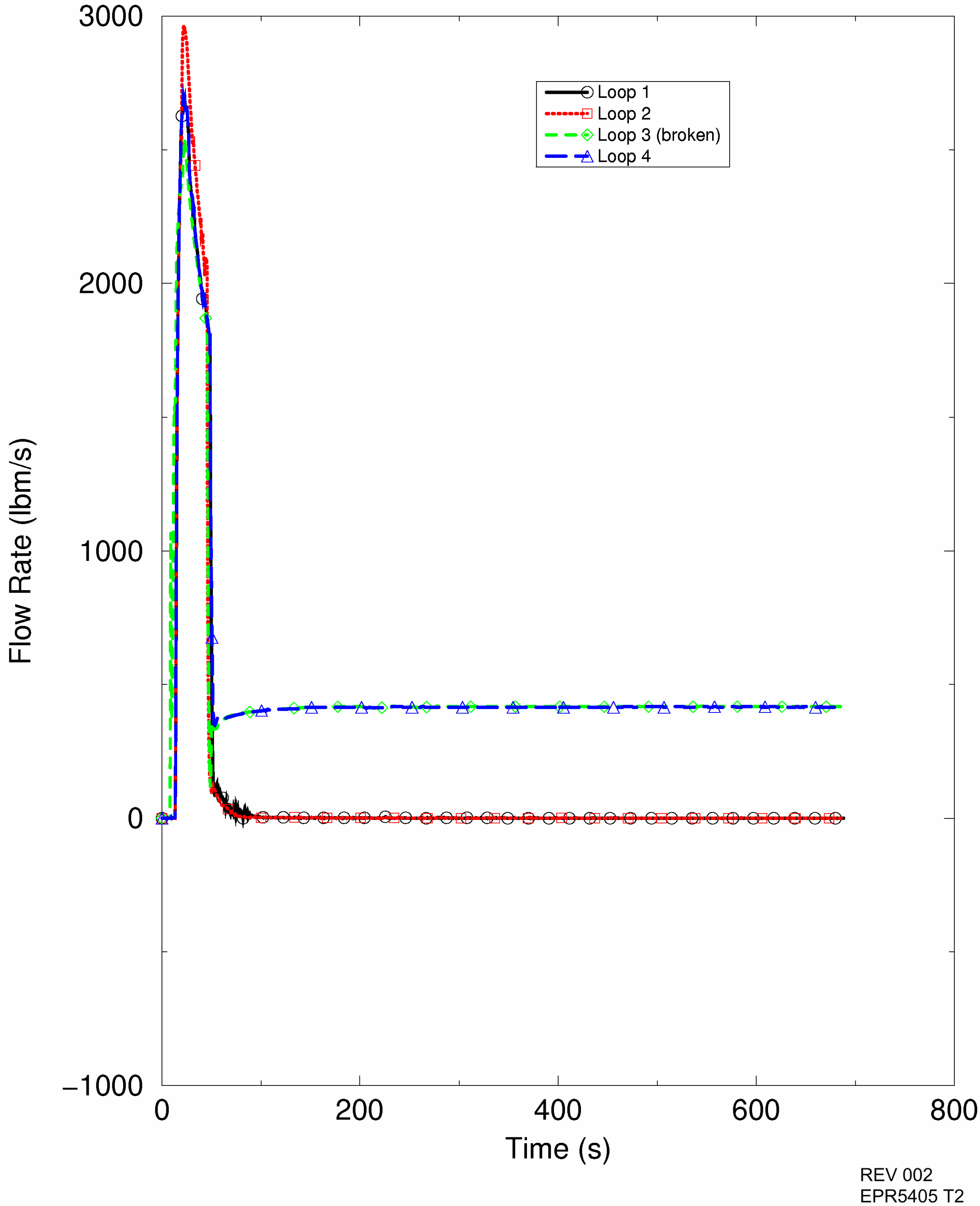


Figure 15.6-31—RLBLOCA - Flows Delivered by ECCS for the Limiting PCT Case (Cycle 1)

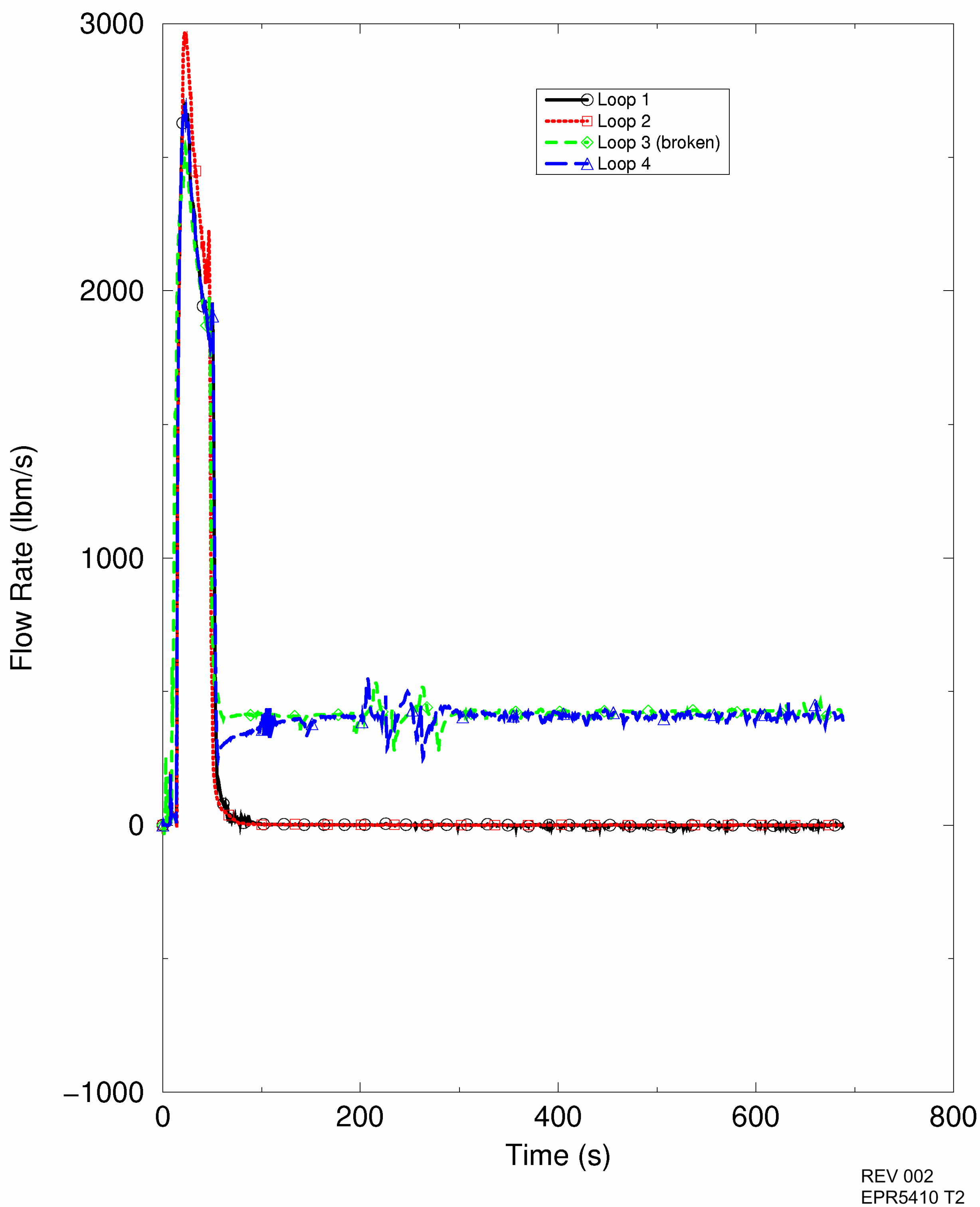
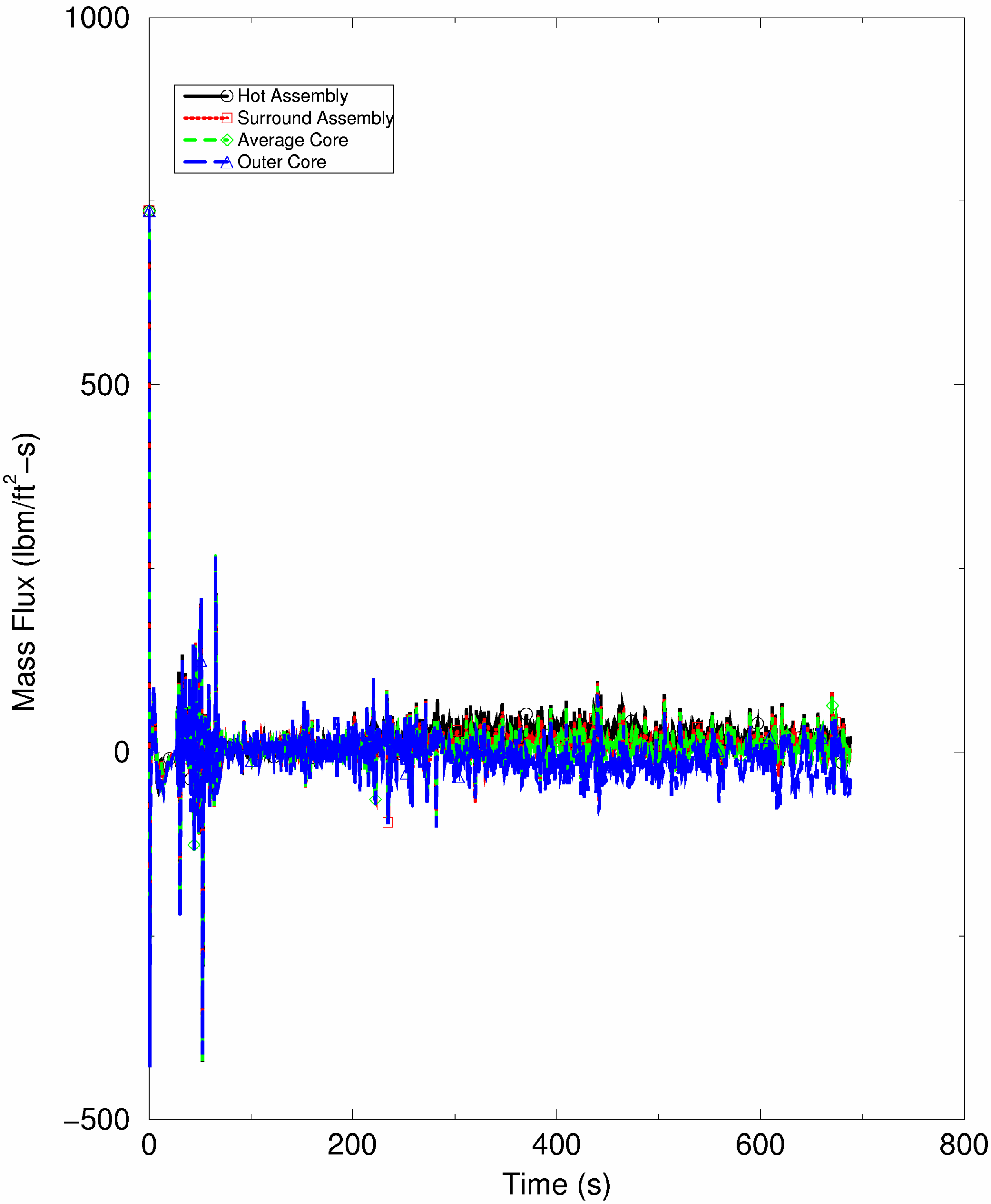


Figure 15.6-32—RLBLOCA - Core Inlet Flow for the Limiting PCT Case (Cycle 1)



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EPR5415 T2

Figure 15.6-33—RLBLOCA - Core Outlet Flow for the Limiting PCT Case (Cycle 1)

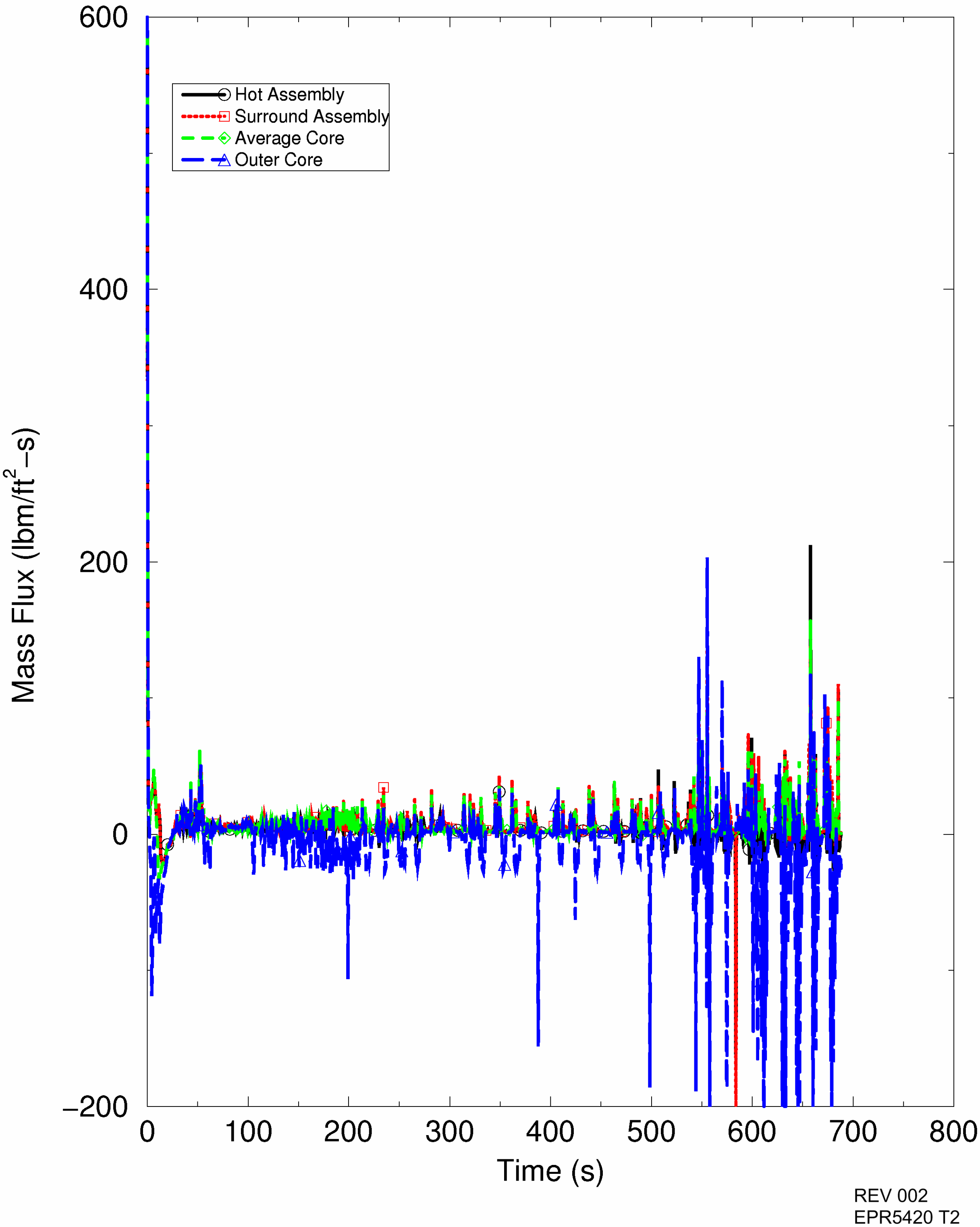


Figure 15.6-34—RLBLOCA - Break Flow for the Limiting PCT Case (Cycle 1)

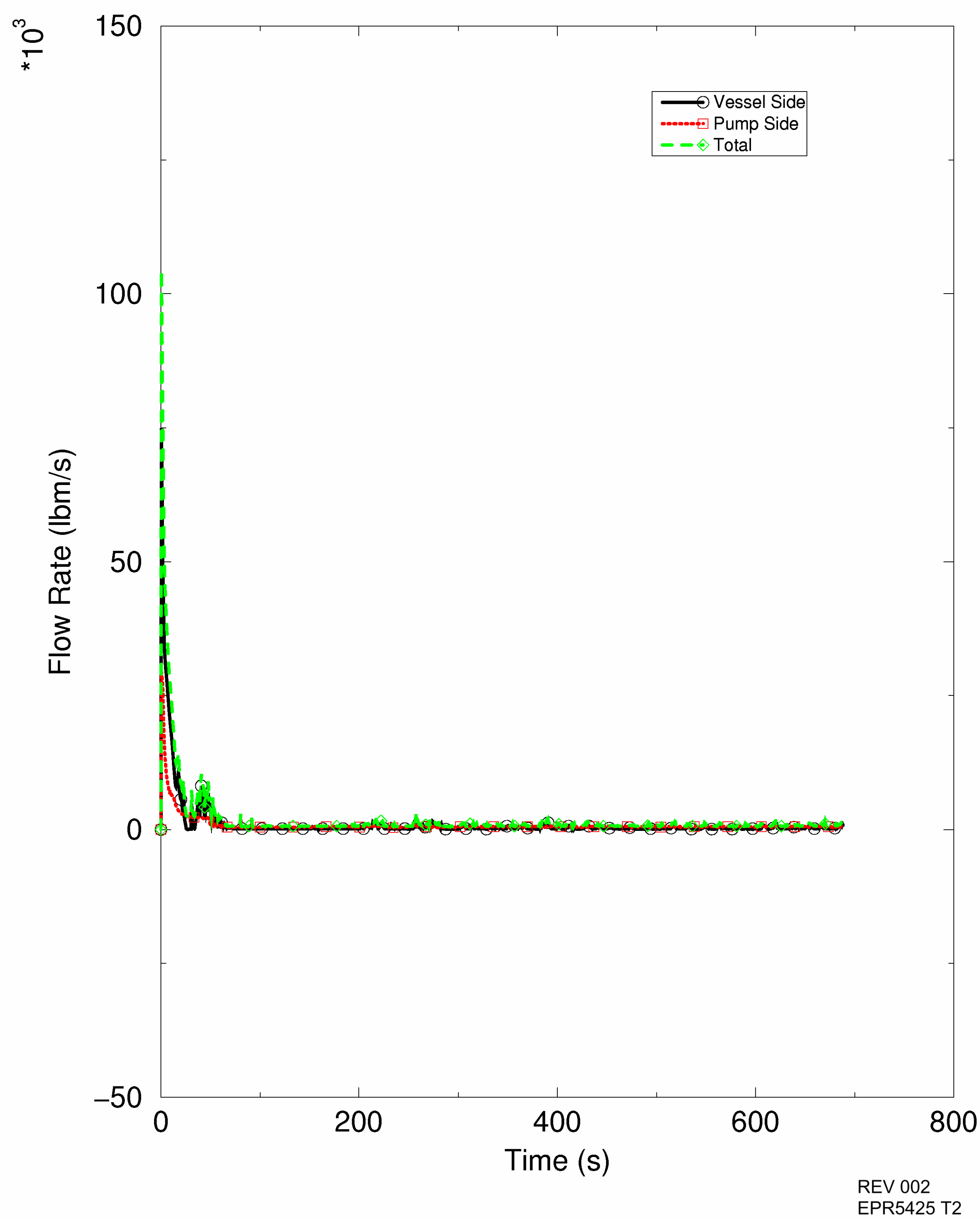


Figure 15.6-35—RLBLOCA - Collapsed Liquid Level in Downcomer for the Limiting PCT Case (Cycle 1)

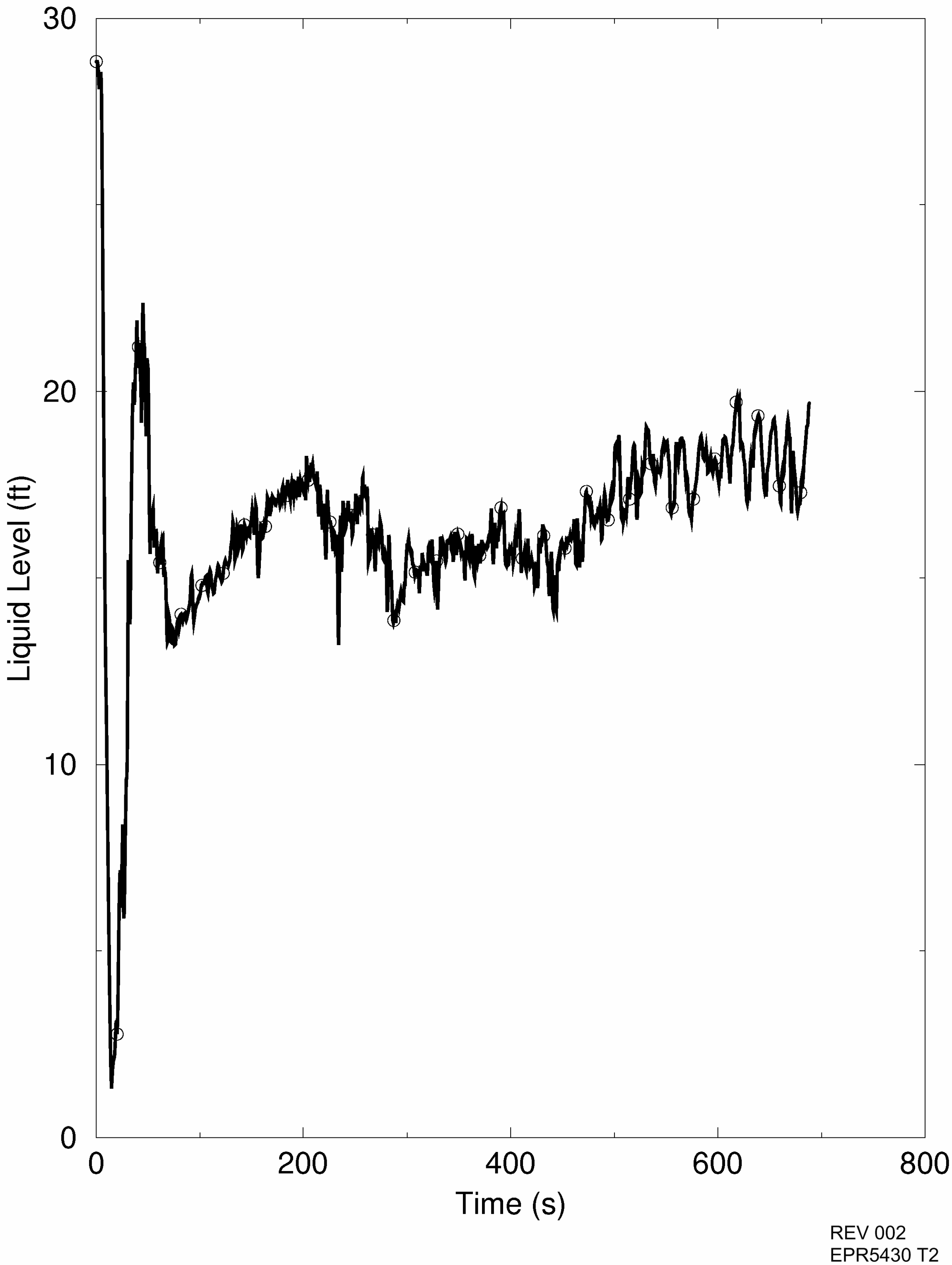
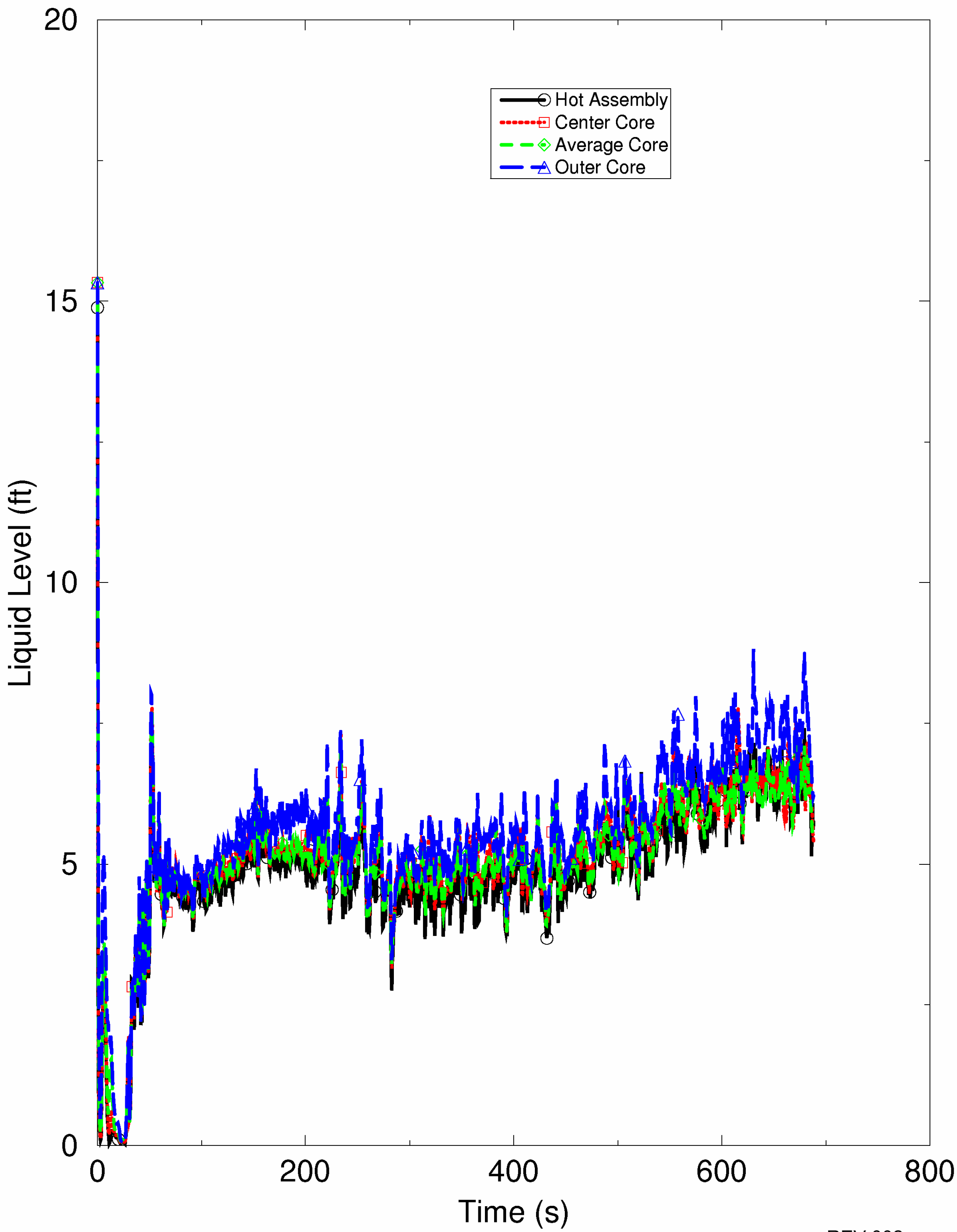


Figure 15.6-36—RLBLOCA - Core Liquid Level for the Limiting PCT Case (Cycle 1)



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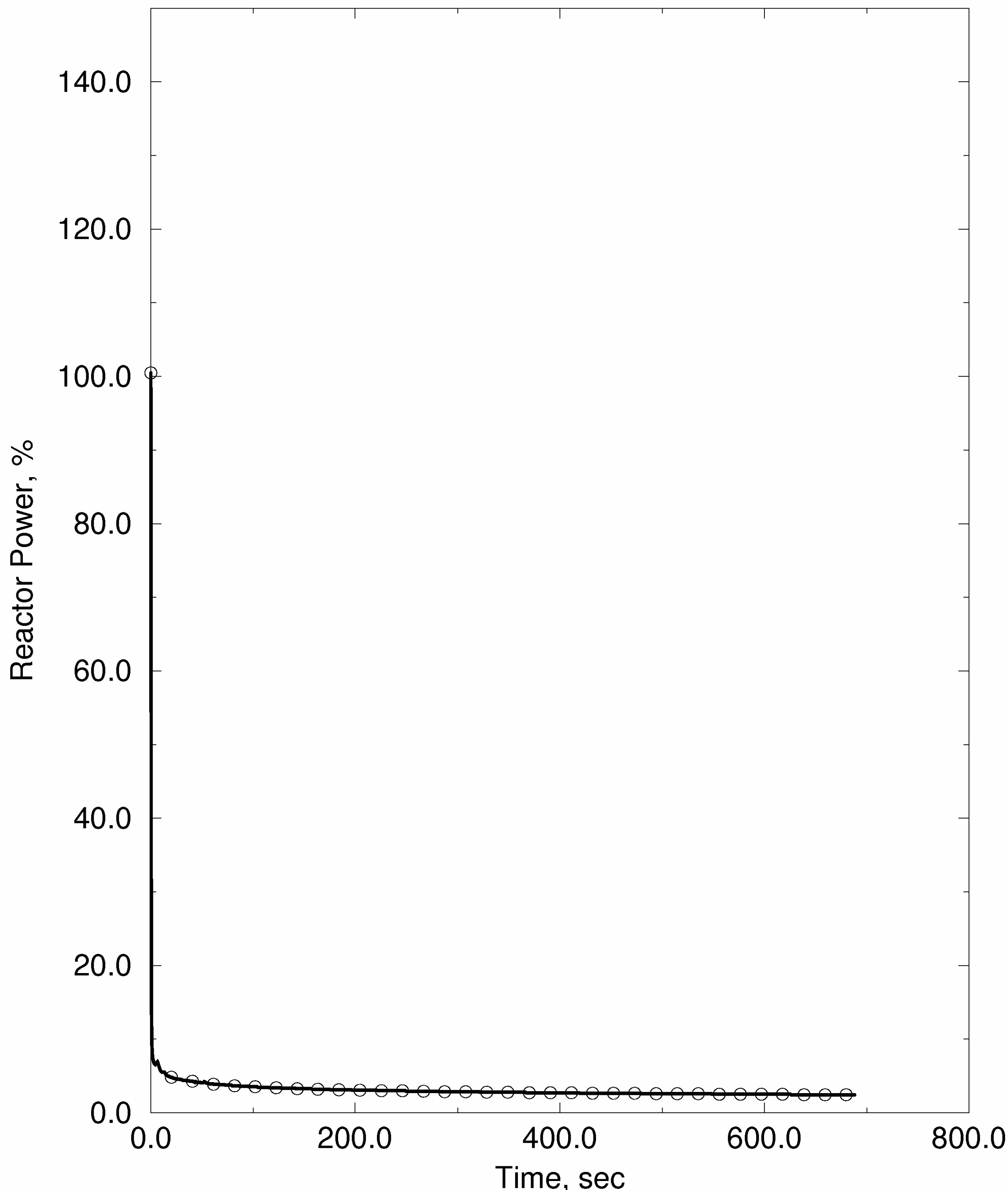
EPR5435 T2

Figure 15.6-37—RLBLOCA - Reactor Power for the Limiting PCT Case (Cycle 1)



REV 002
EPR5440 T2

Figure 15.6-38—RLBLOCA - Secondary Pressure for the Limiting PCT Case (Cycle 1)

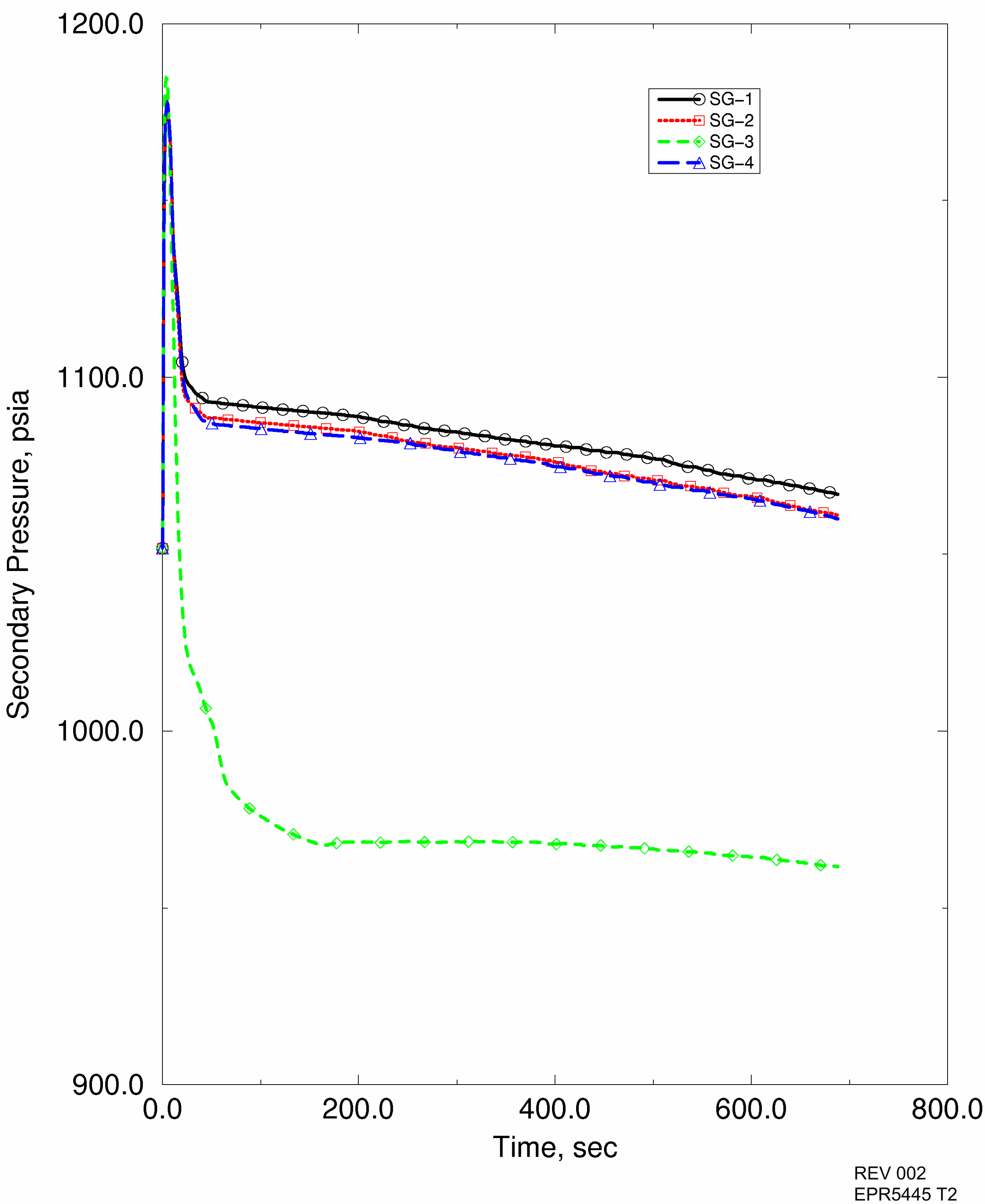


Figure 15.6-39—RLBLOCA - Downcomer Mass Flowrate for the Limiting PCT Case (Cycle 1)

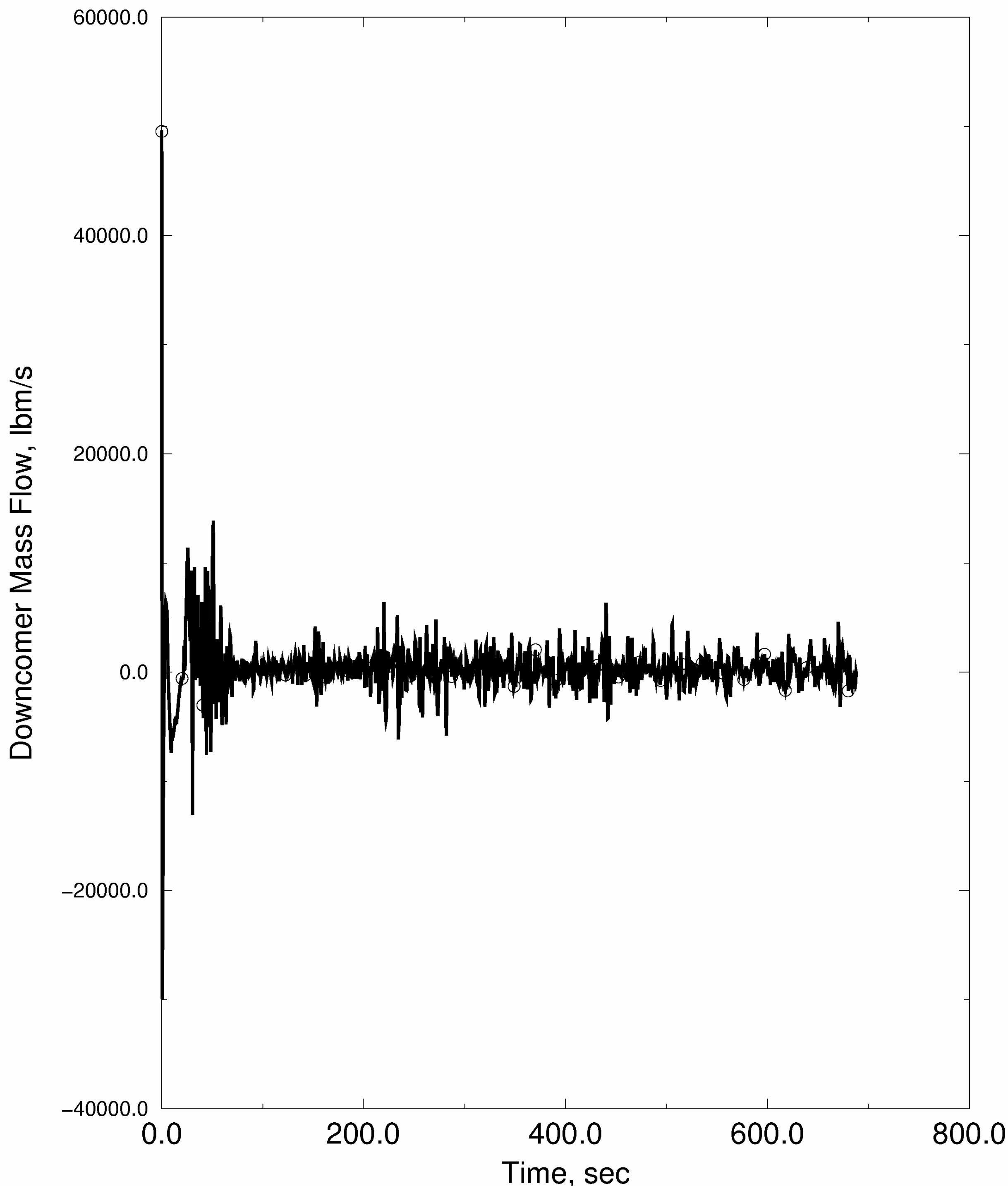


Figure 15.6-40—RLBLOCA - Core Inlet Temperature for the Limiting PCT Case (Cycle 1)

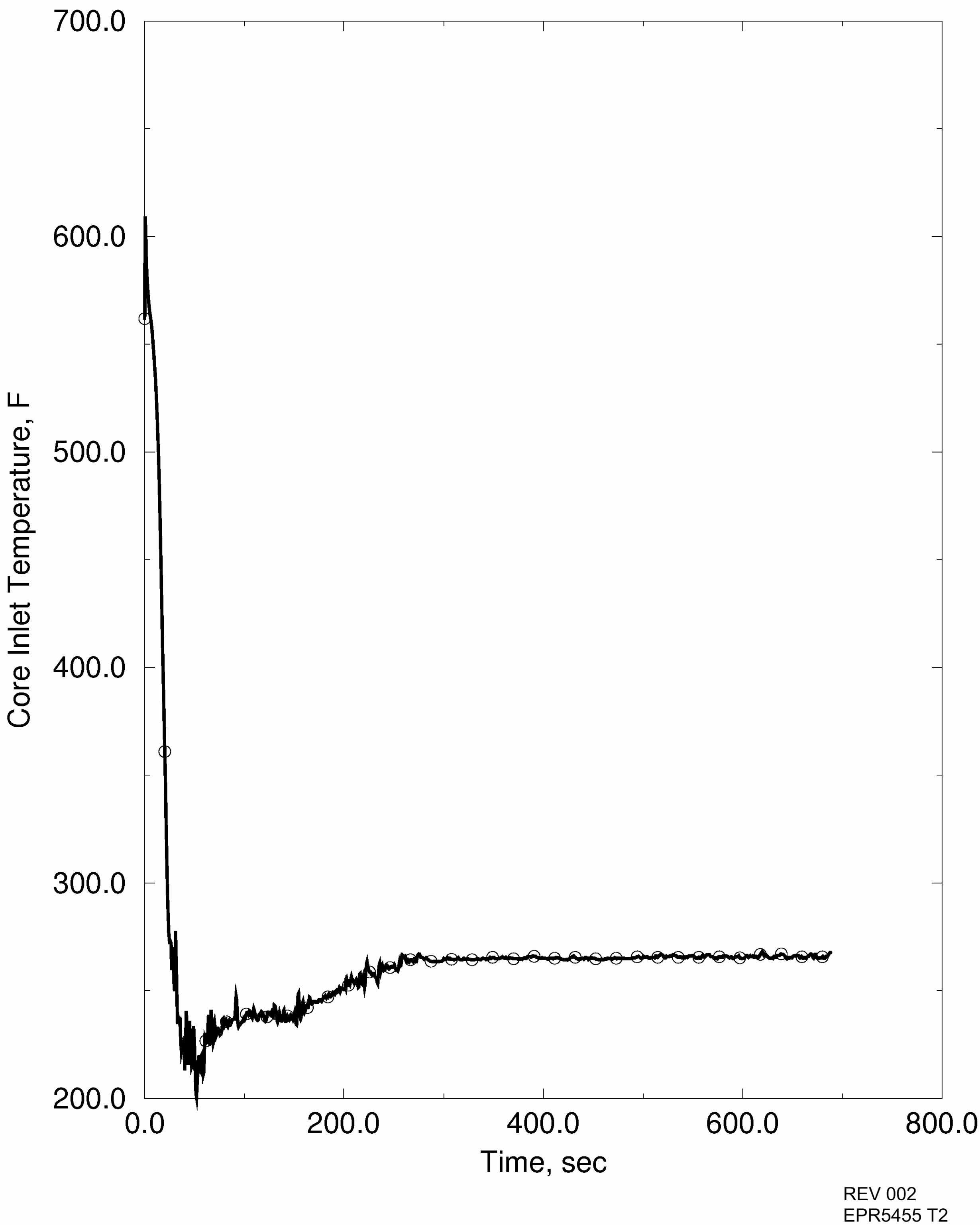


Figure 15.6-41—RLBLOCA - Core Inlet Quality for the Limiting PCT Case (Cycle 1)

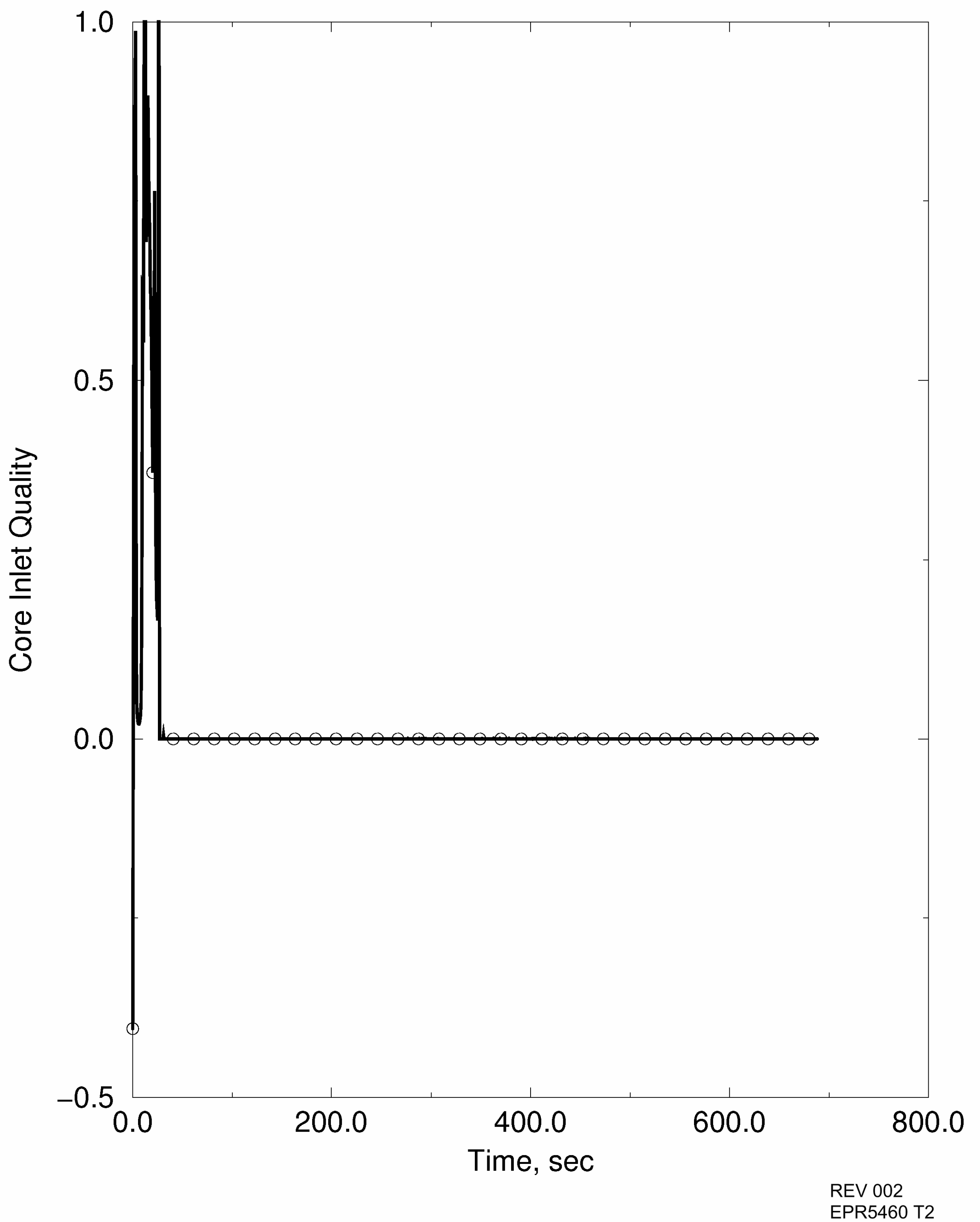


Figure 15.6-42—RLBLOCA - Core Inlet Quality for the Limiting PCT Case on Smaller Time Scale (Cycle 1)

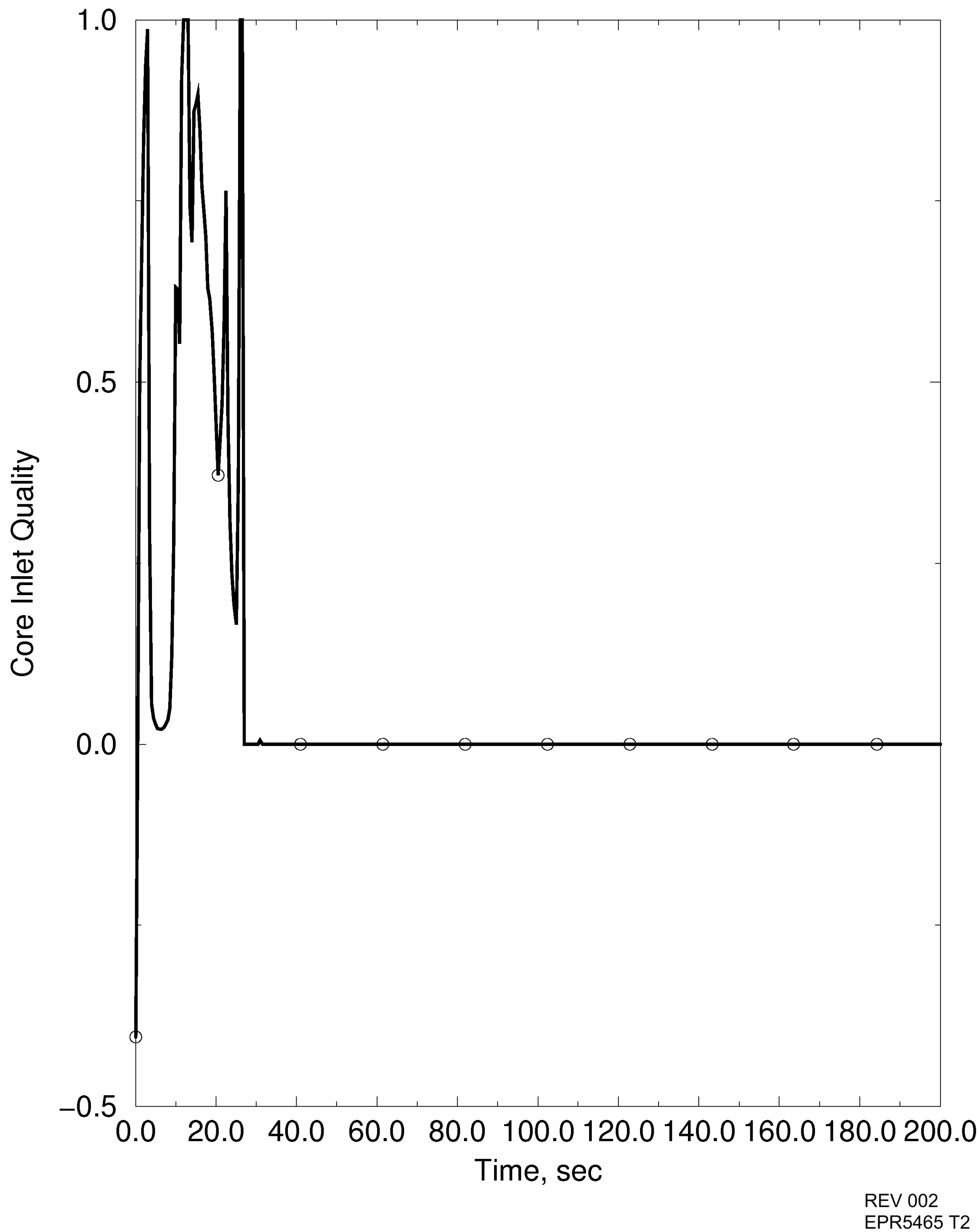


Figure 15.6-43—RLBLOCA - Core Outlet Temperature for the Limiting PCT Case (Cycle 1)

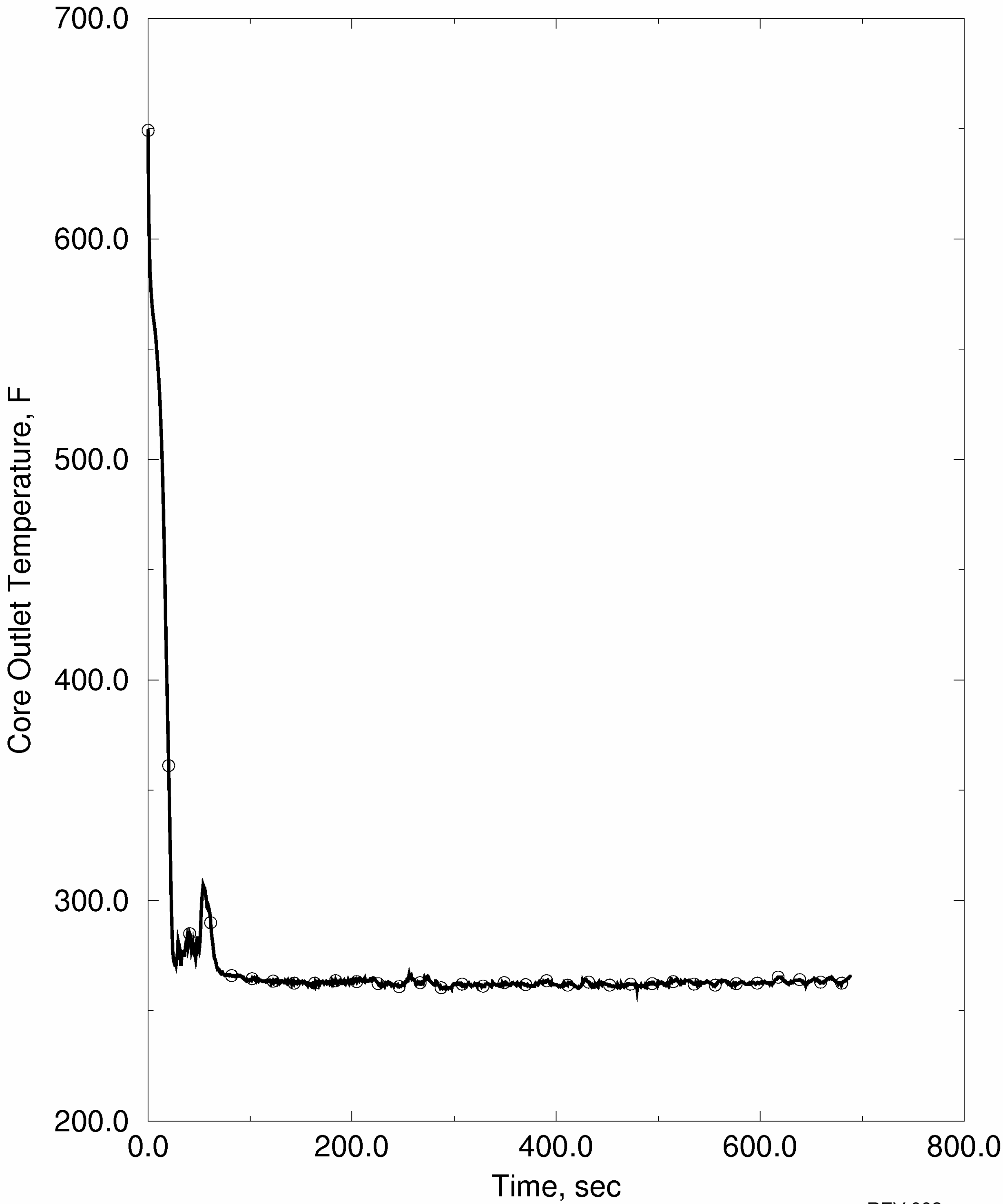


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Figure 15.6-44—RLBLOCA - Core Outlet Quality for the Limiting PCT Case (Cycle 1)

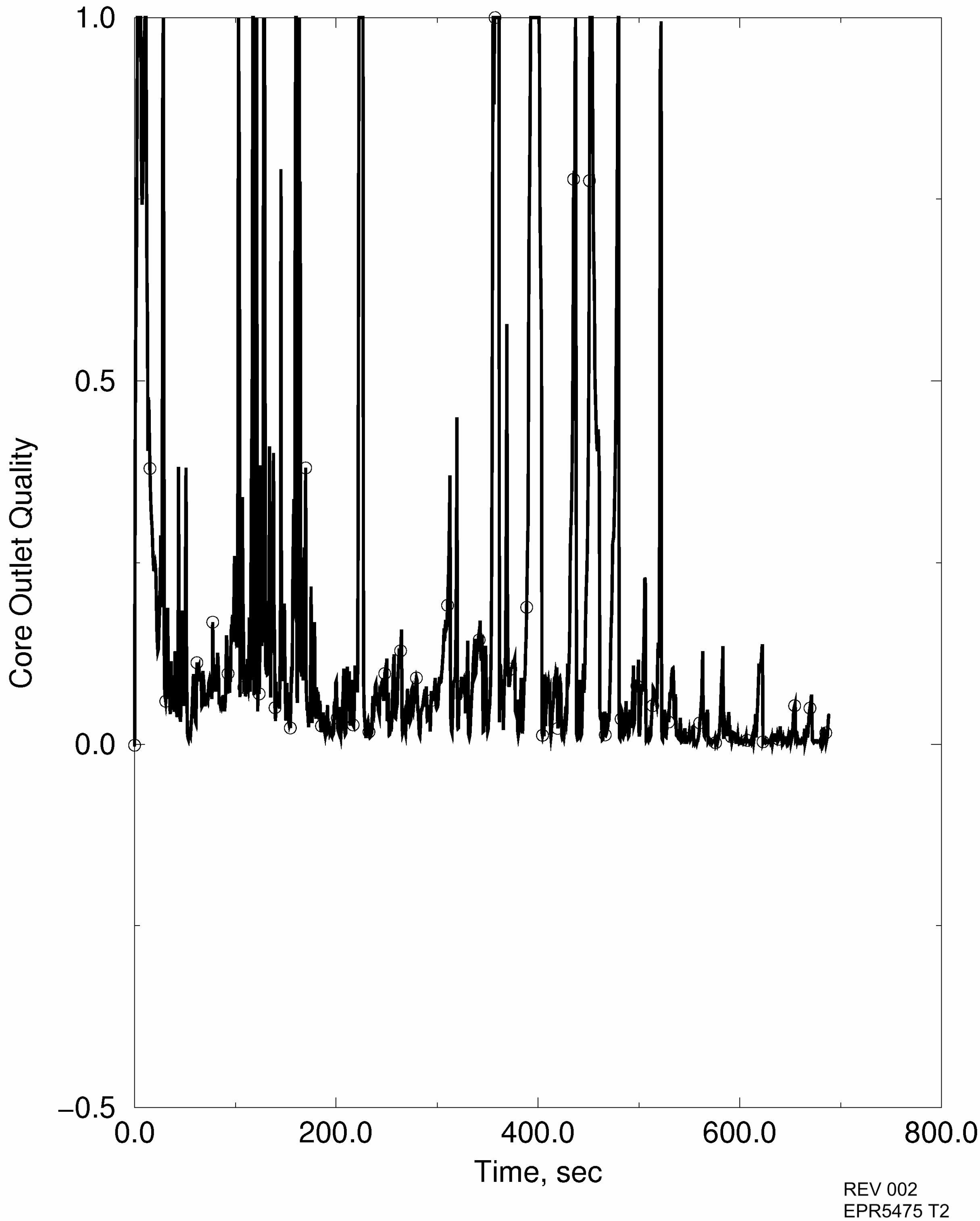


Figure 15.6-45—RLBLOCA - Core Outlet Quality for the Limiting PCT Case on Smaller Time Scale (Cycle 1)

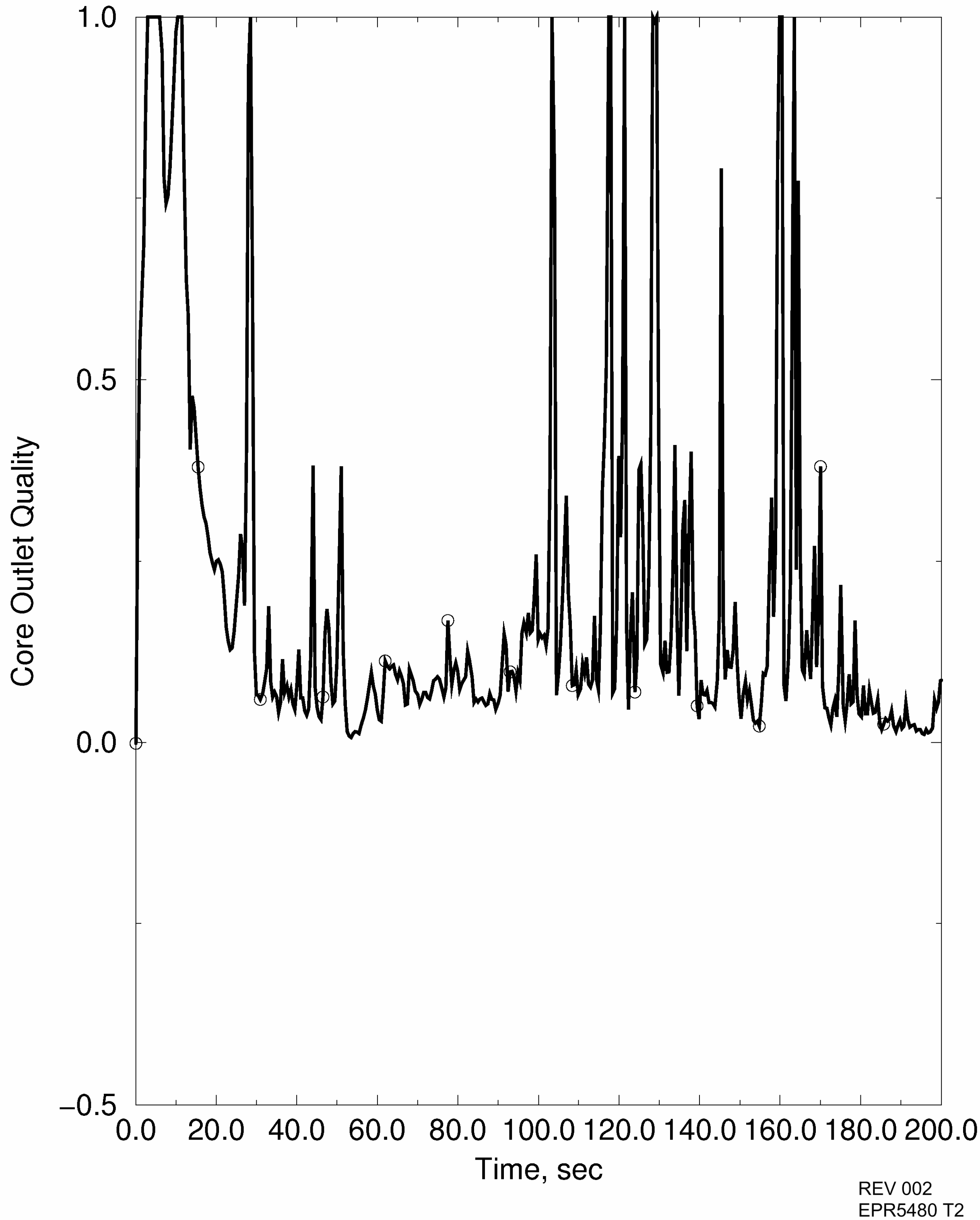


Figure 15.6-46—RLBLOCA - In-Core Temperature for the Limiting PCT Case (Cycle 1)

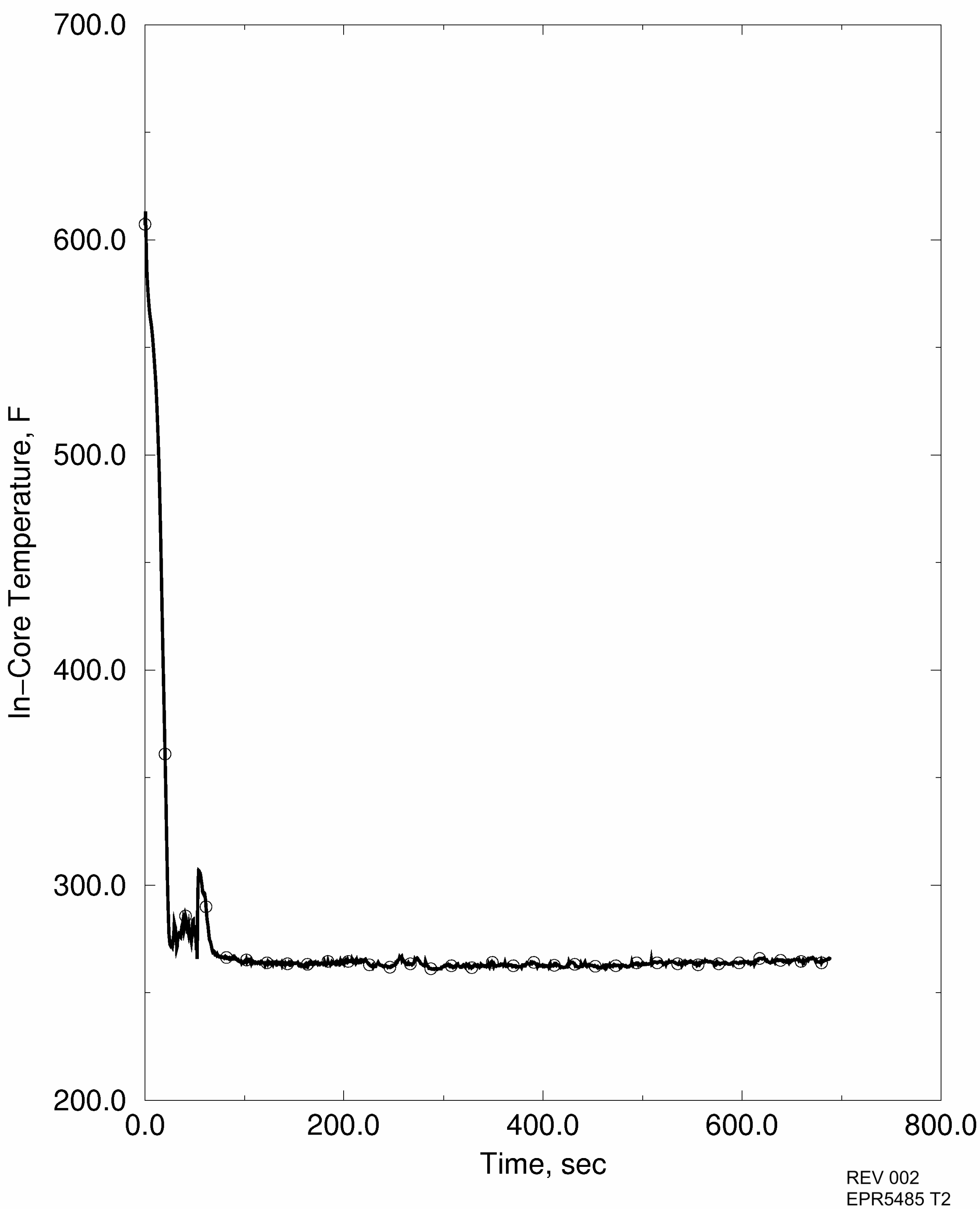
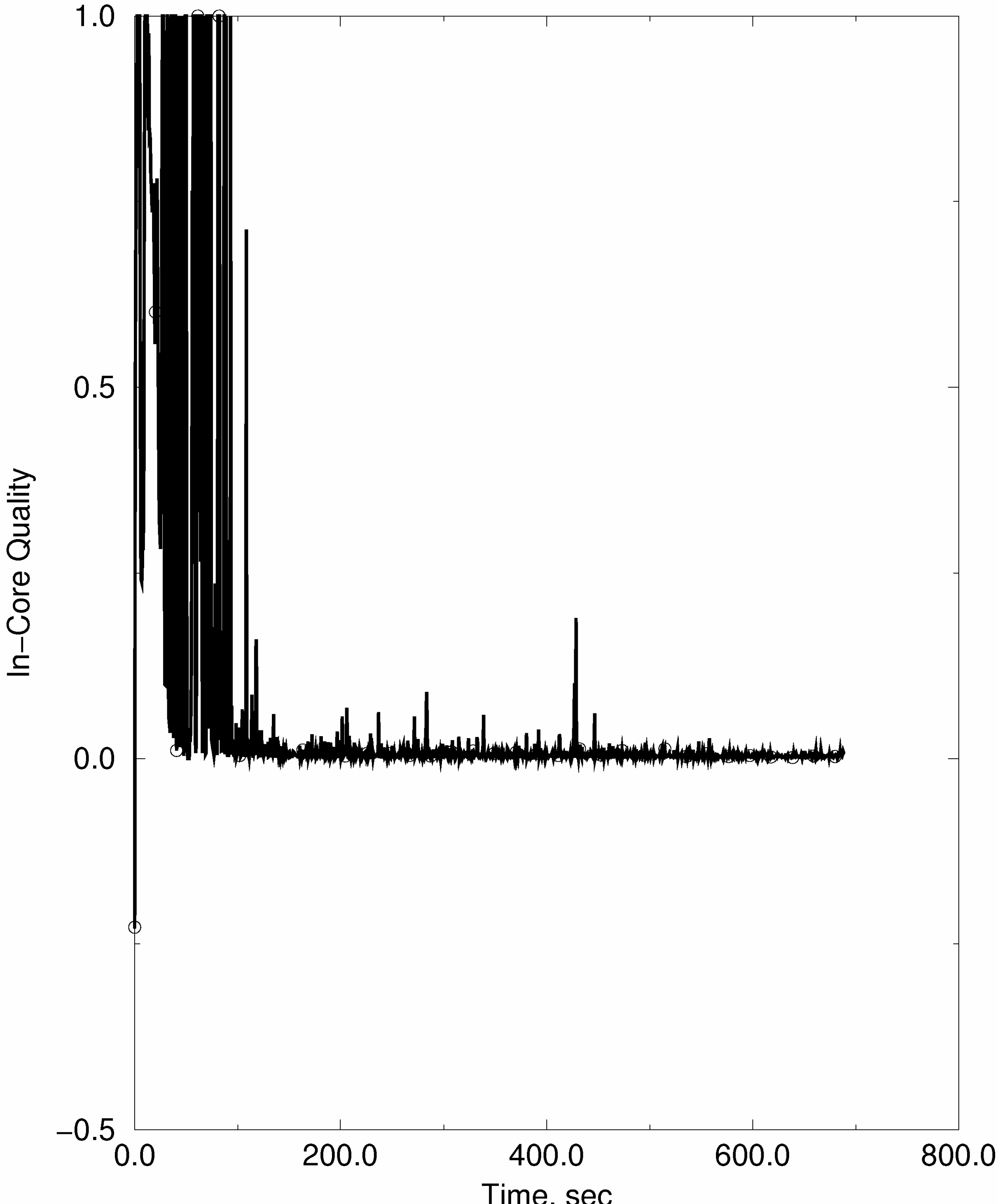


Figure 15.6-47—RLBLOCA - In-Core Quality for the Limiting PCT Case (Cycle 1)



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EPR5490 T2

Figure 15.6-48—RLBLOCA - In-Core Quality for the Limiting PCT Case on Smaller Time Scale (Cycle 1)

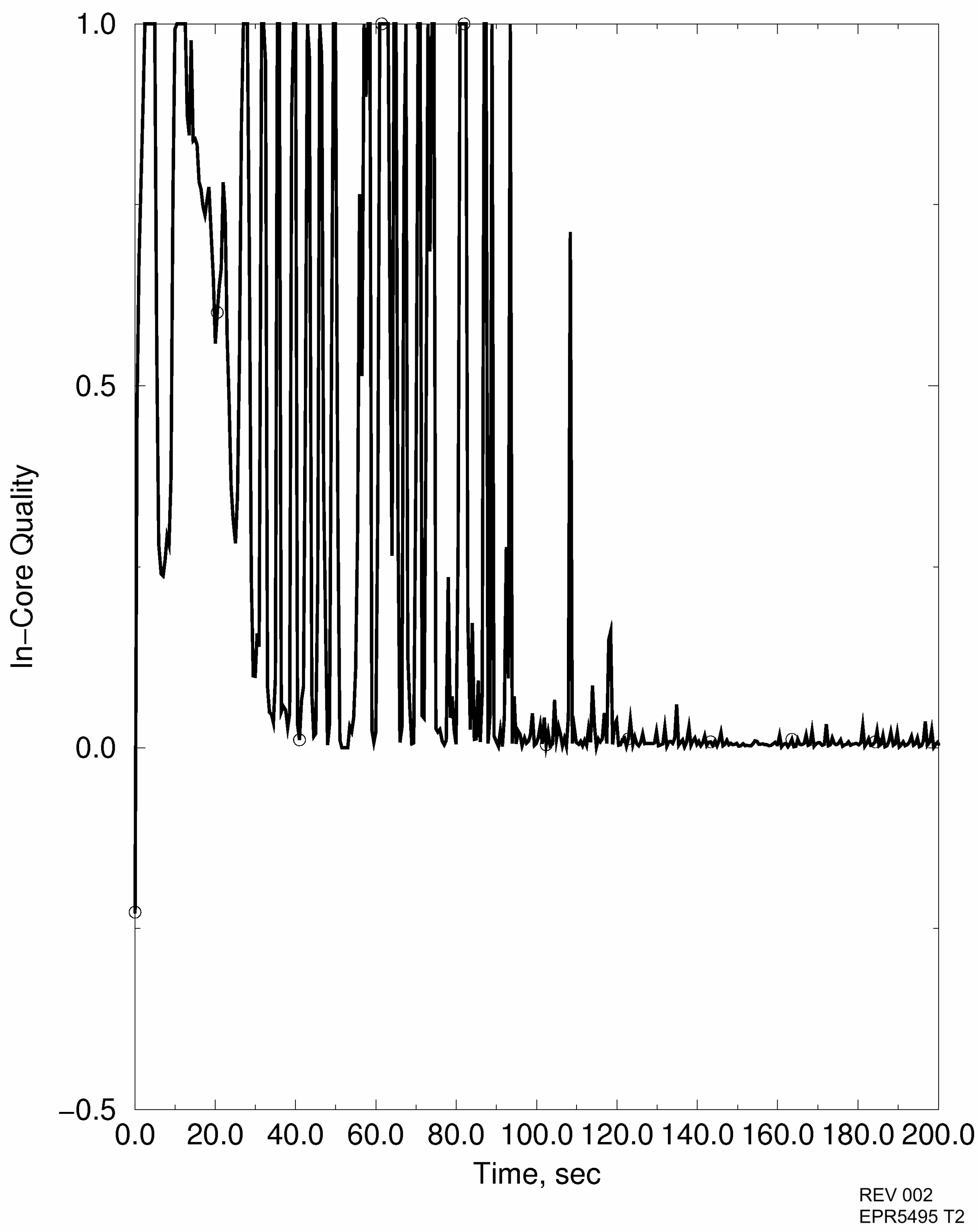

Figure 15.6-49—RLBLOCA - Cladding Temperature for the Limiting PCT Case (Cycle 1)

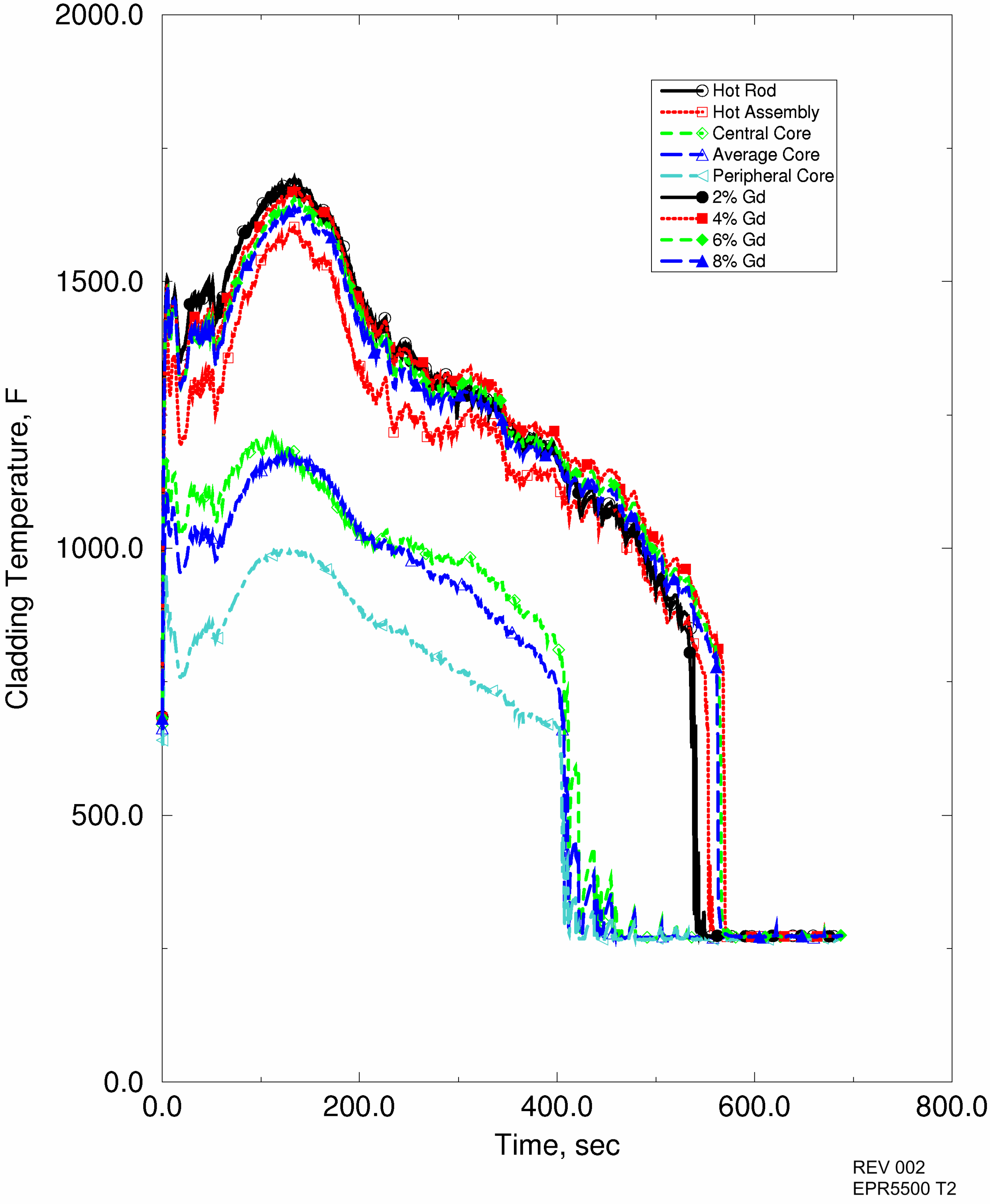

Figure 15.6-50—RLBLOCA - Heat Transfer Coefficient for the Limiting PCT Case (Cycle 1)

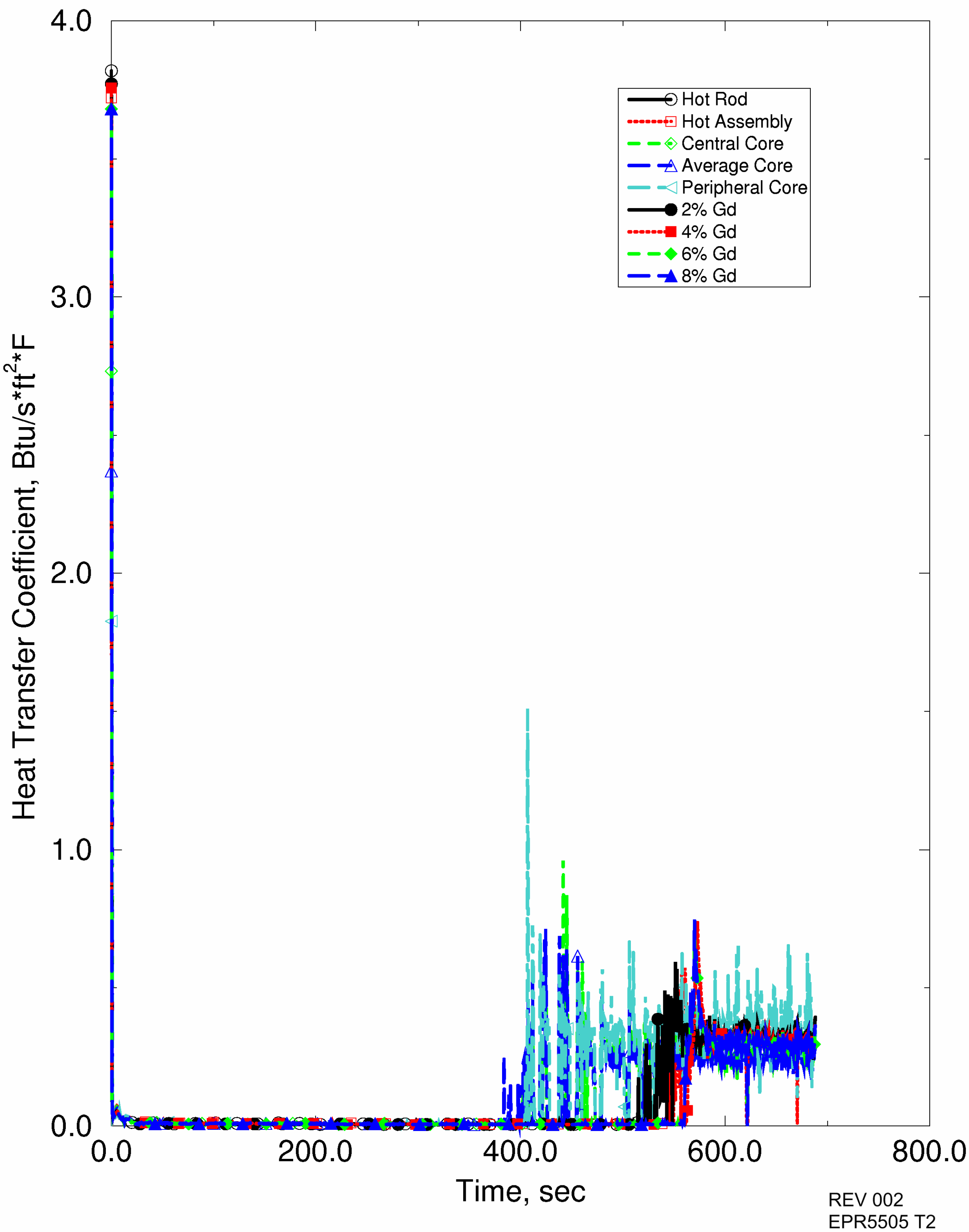

Figure 15.6-51—SBLOCA - PCT - Break Spectrum with LOOP

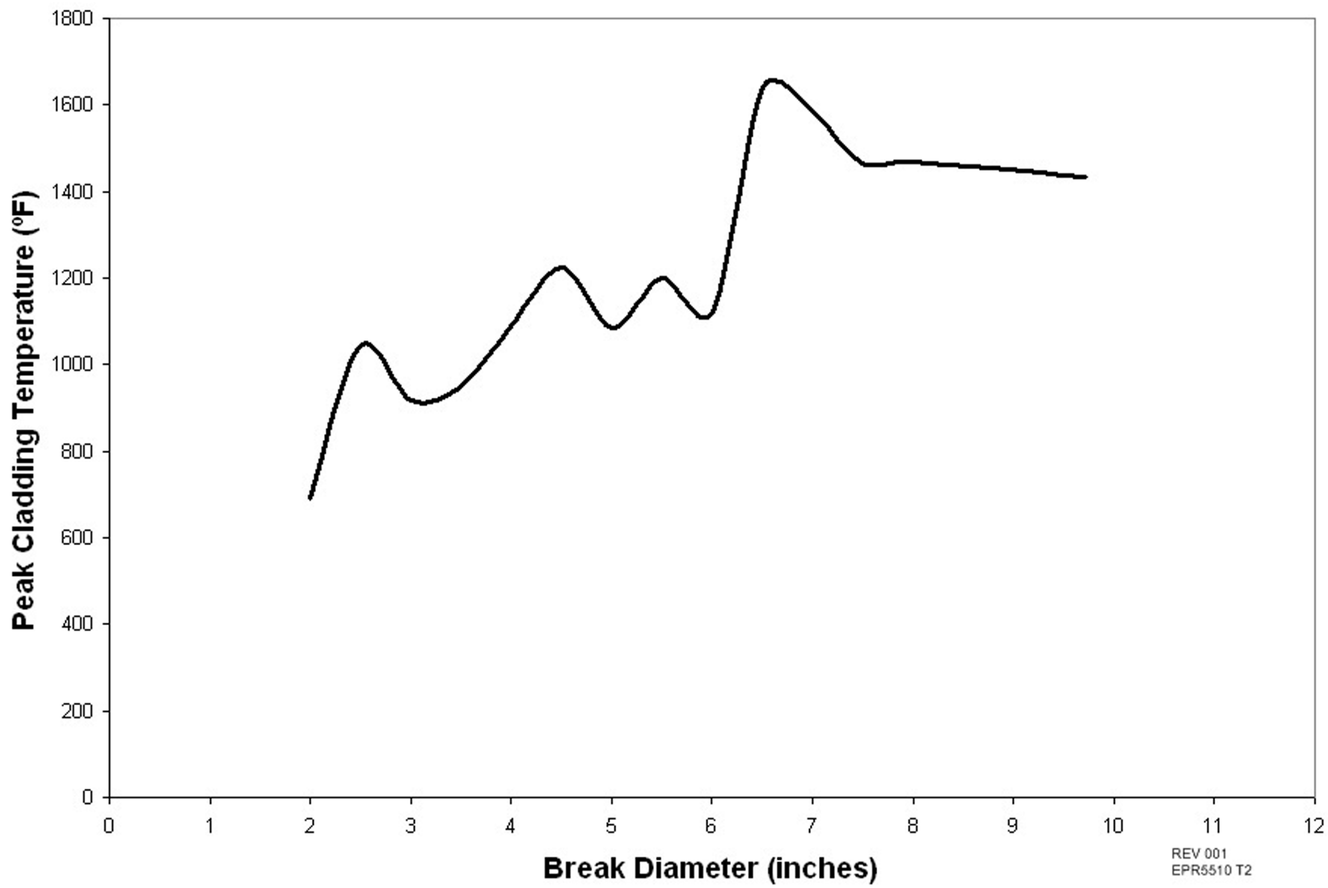

Figure 15.6-52—SBLOCA - 6.5 Inch Break - Reactor Power

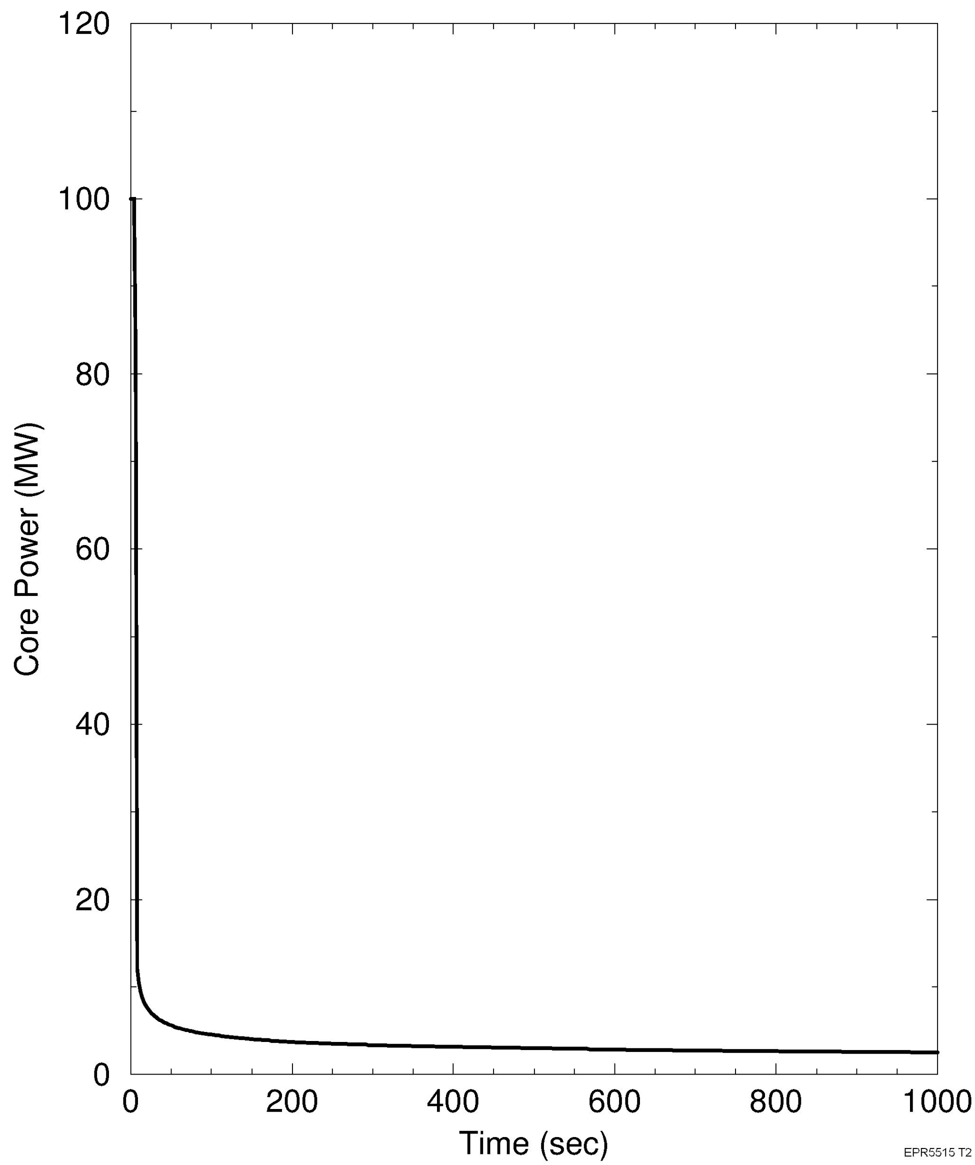

Figure 15.6-53—SBLOCA - 6.5 Inch Break - Primary and Secondary System Pressure

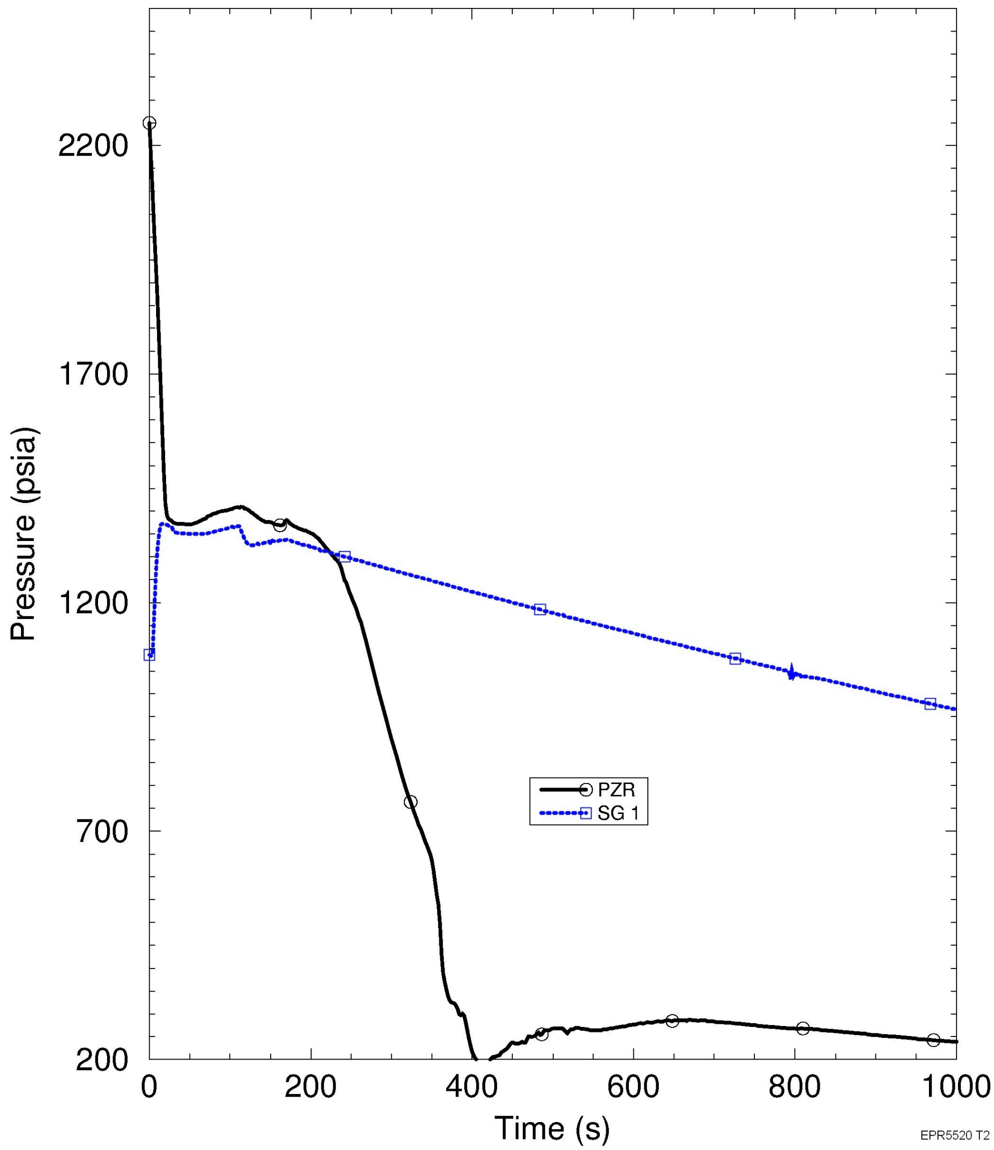

Figure 15.6-54—SBLOCA - 6.5 Inch Break - MSRT Flow

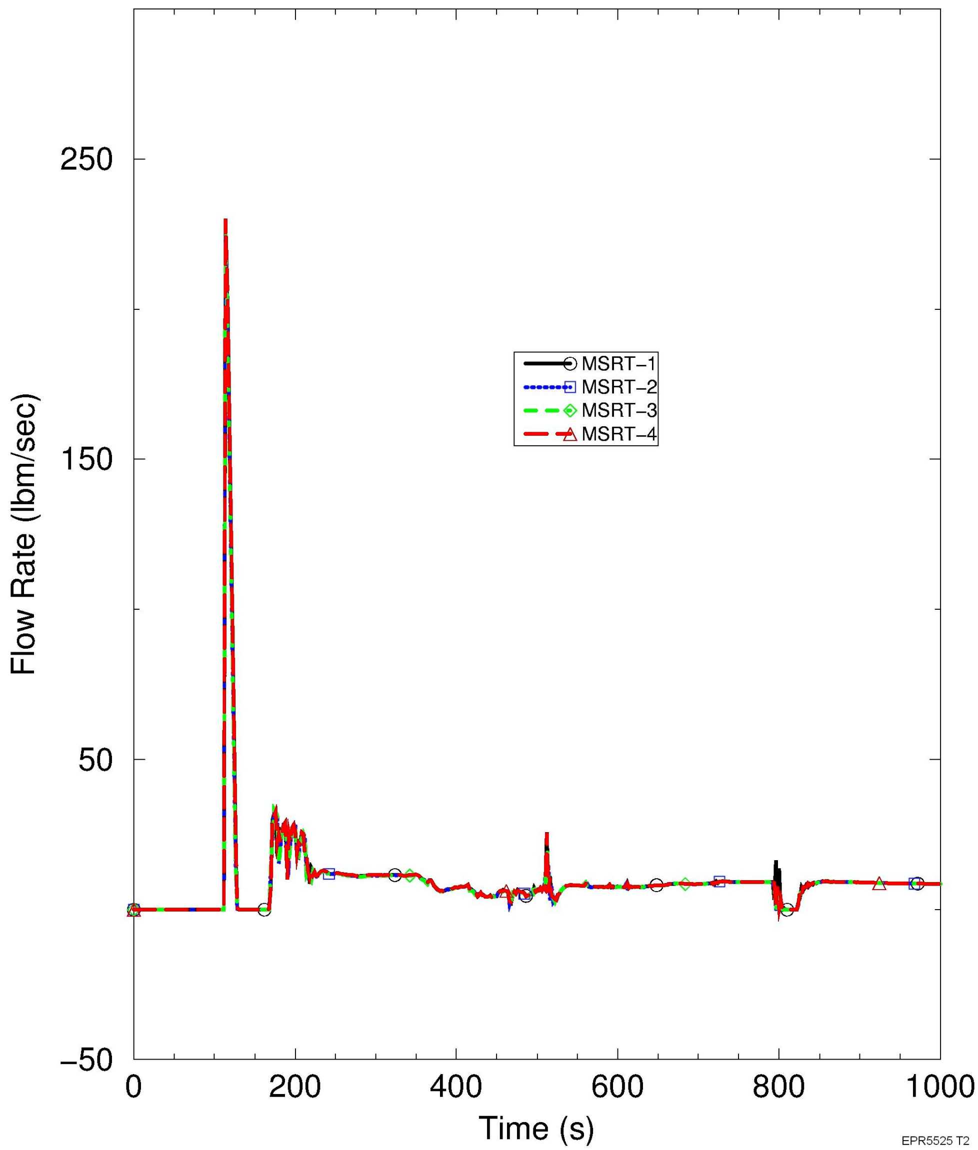

Figure 15.6-55—SBLOCA - 6.5 Inch Break - Break Flow

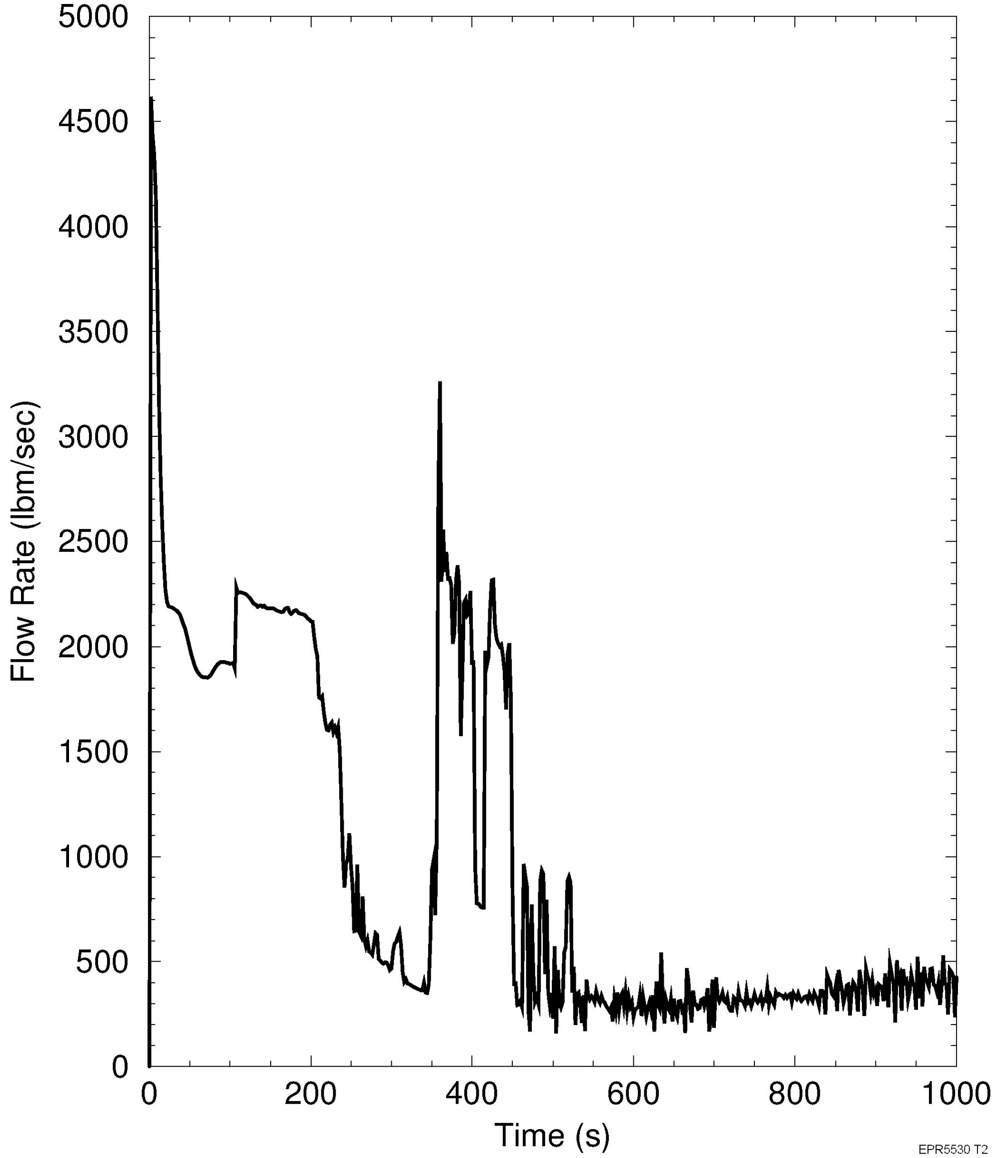

Figure 15.6-56—SBLOCA - 6.5 Break - Loop Seal Void Fraction

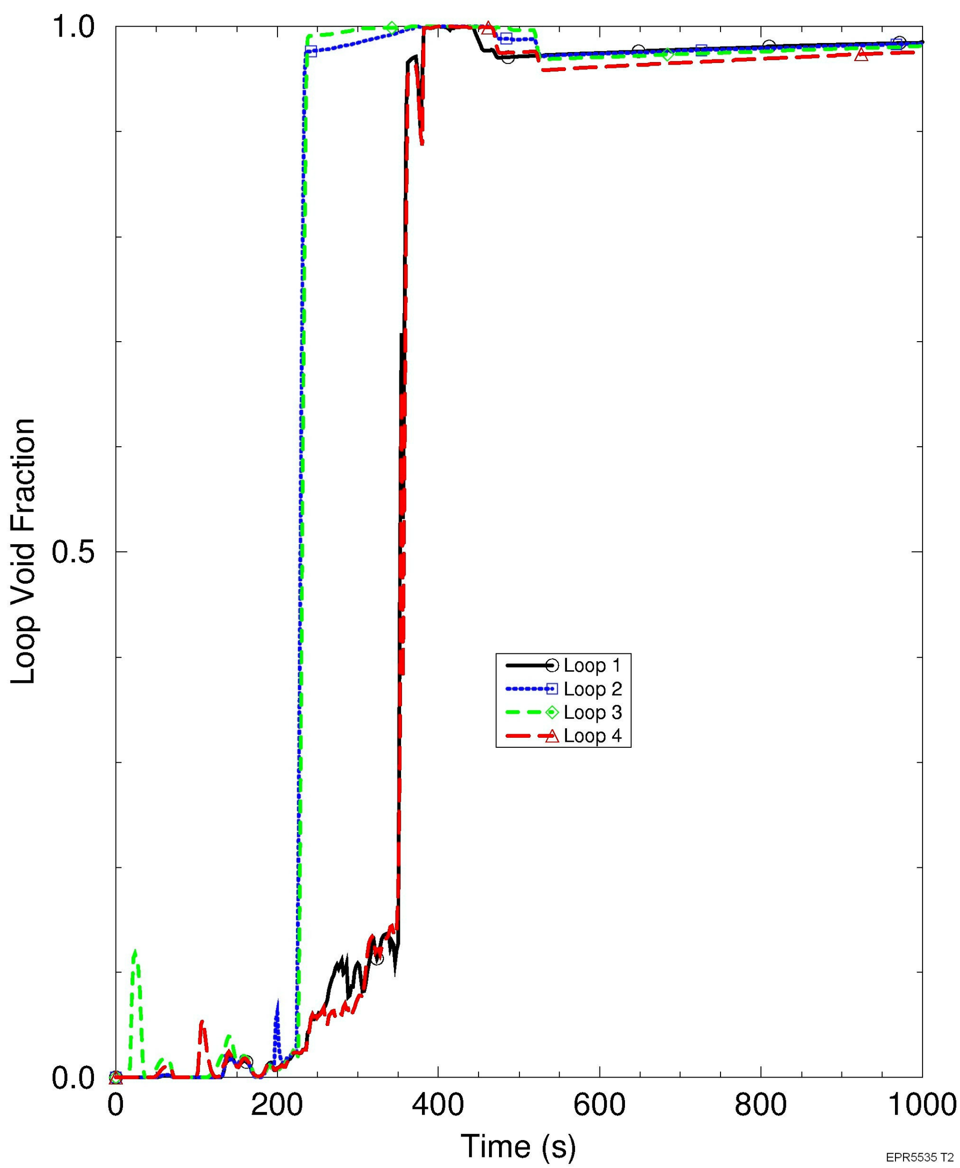

Figure 15.6-57—SBLOCA - 6.5 Break - ECCS Flow

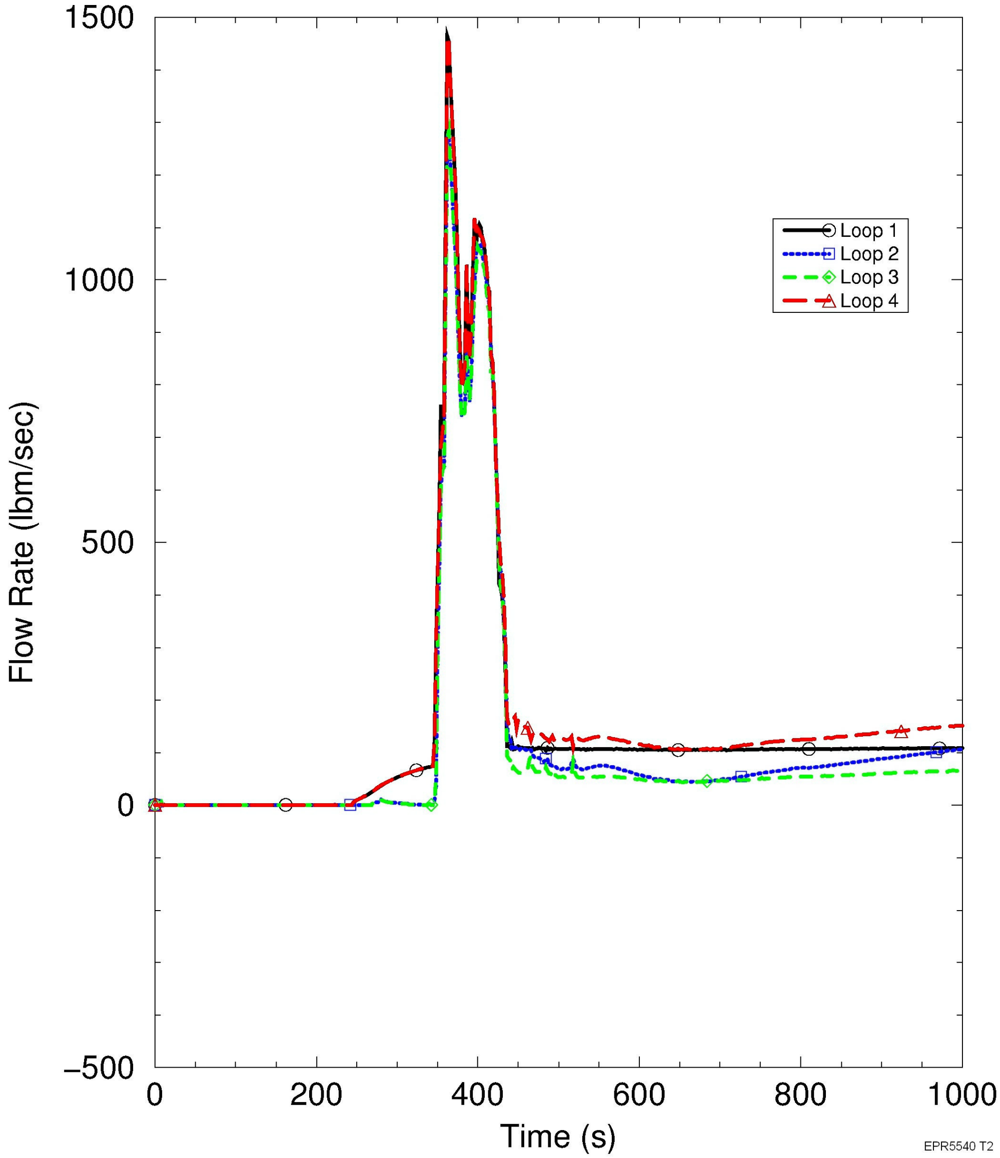

Figure 15.6-58—SBLOCA - 6.5 Break - MFW Flow

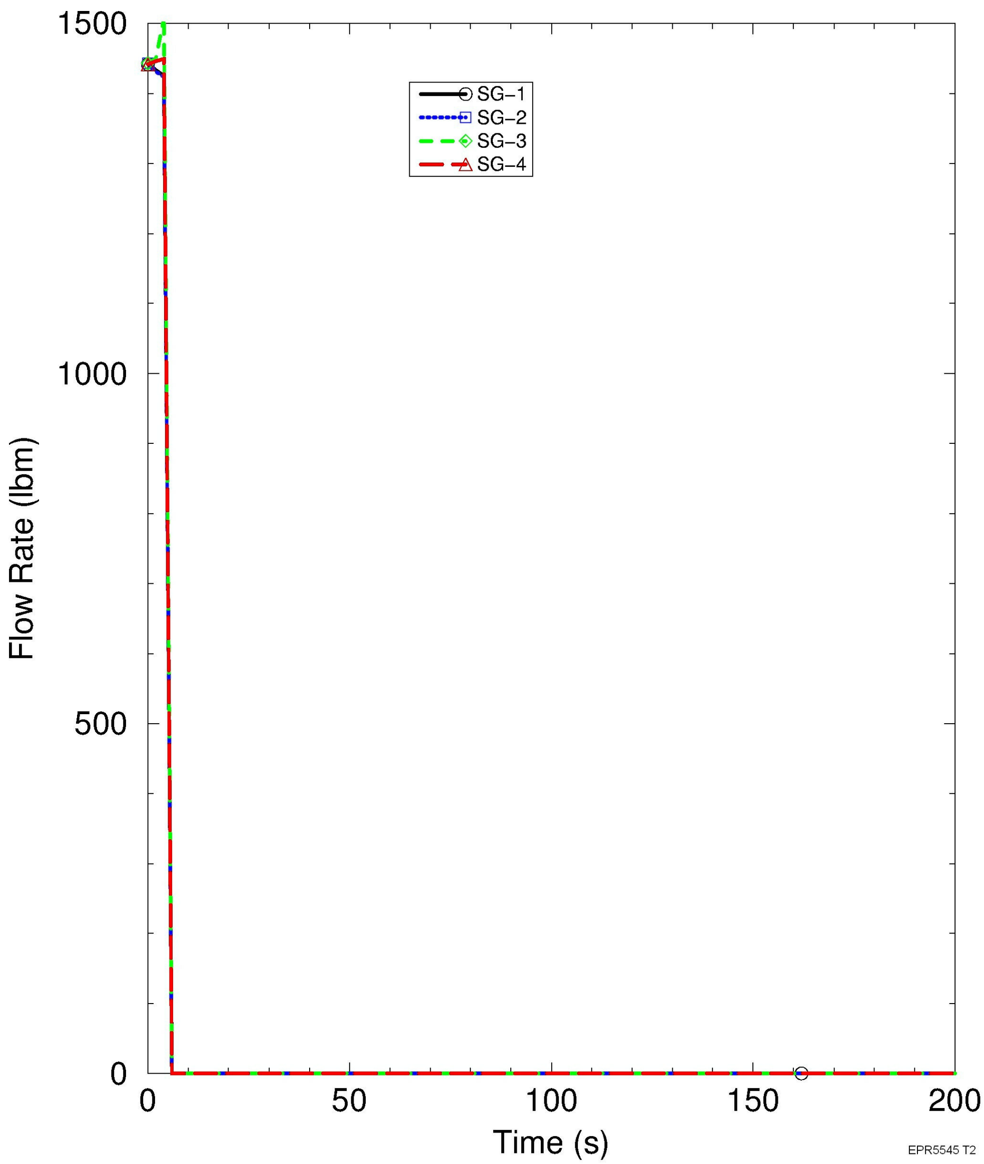

Figure 15.6-59—SBLOCA - 6.5 Inch Break - Steam Generator Mass Inventory

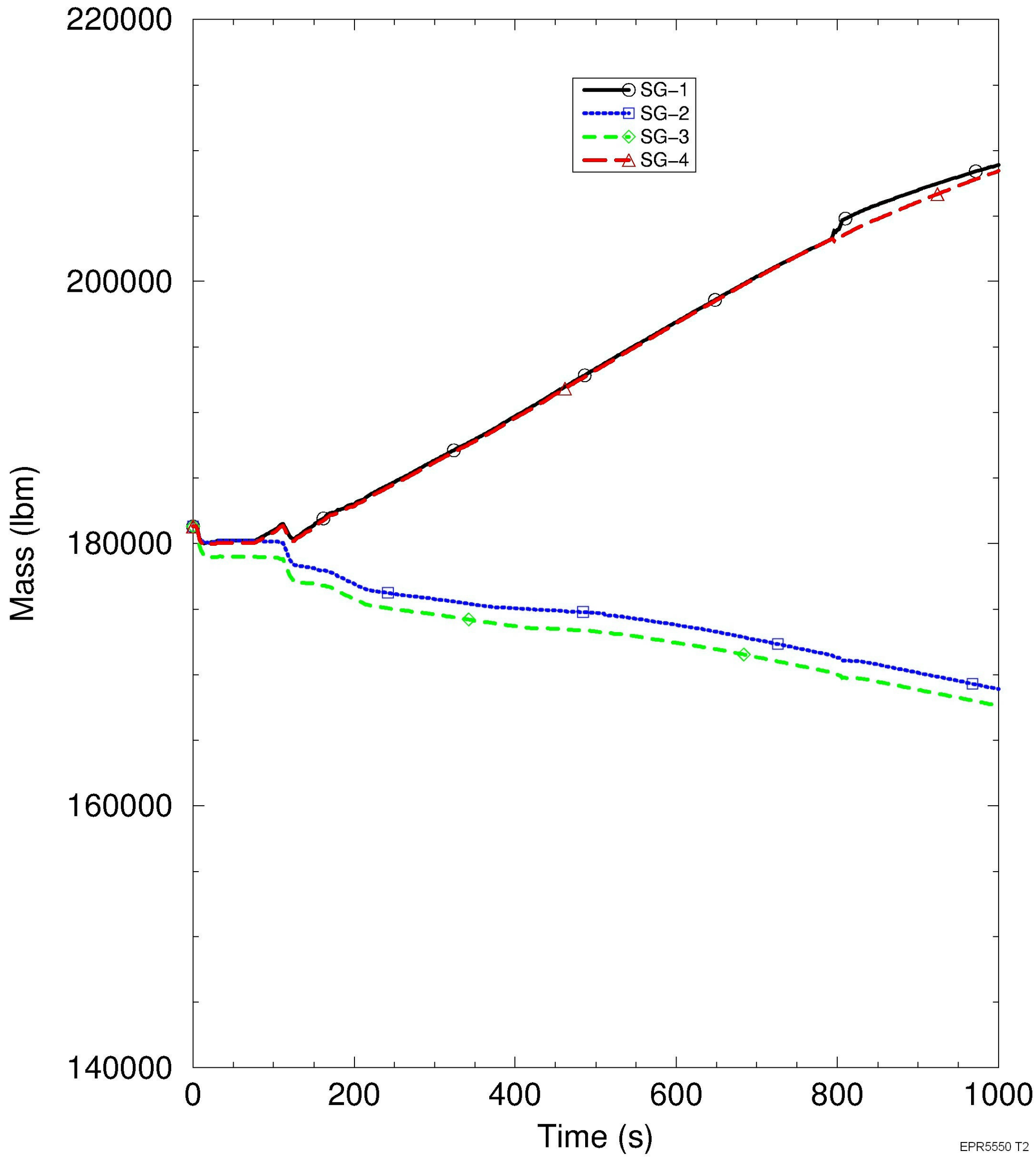

Figure 15.6-60—SBLOCA - 6.5 Inch Break - Steam Generator Wide Range Level

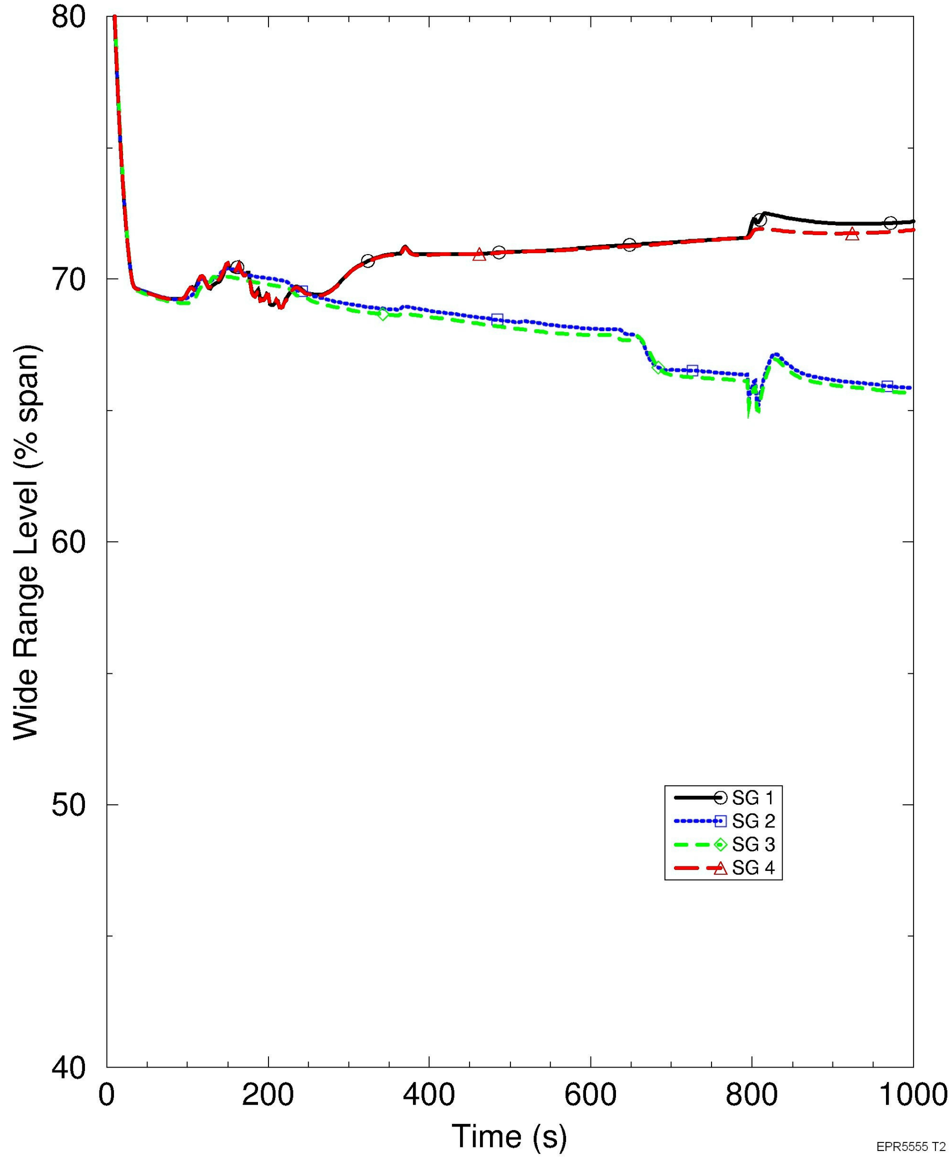

Figure 15.6-61—SBLOCA - 6.5 Inch Break - Steam Generator Steam Line Pressure

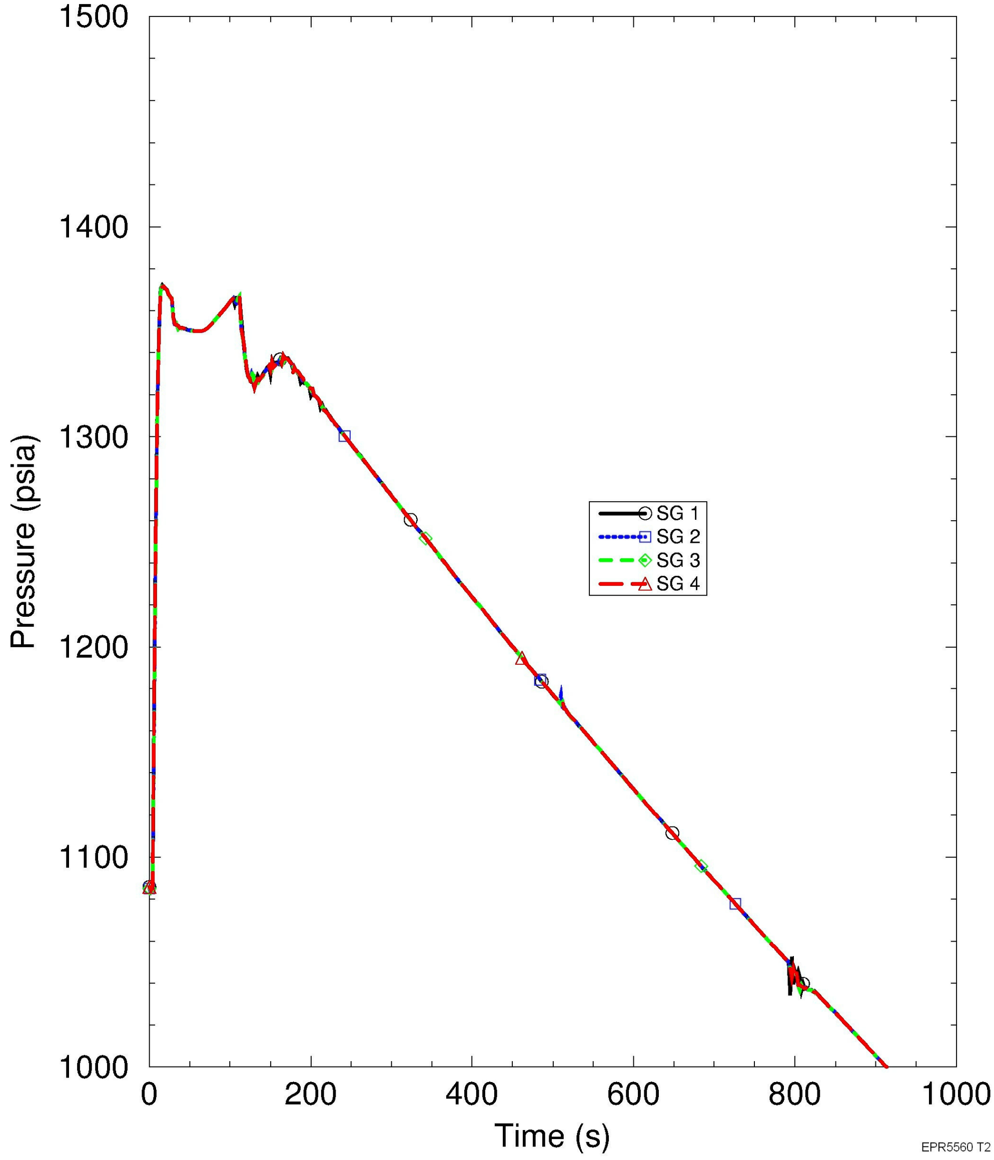

Figure 15.6-62—SBLOCA - 6.5 Inch Break - Steam Generator Steam Line Pressure and MSRT Cooldown Curve

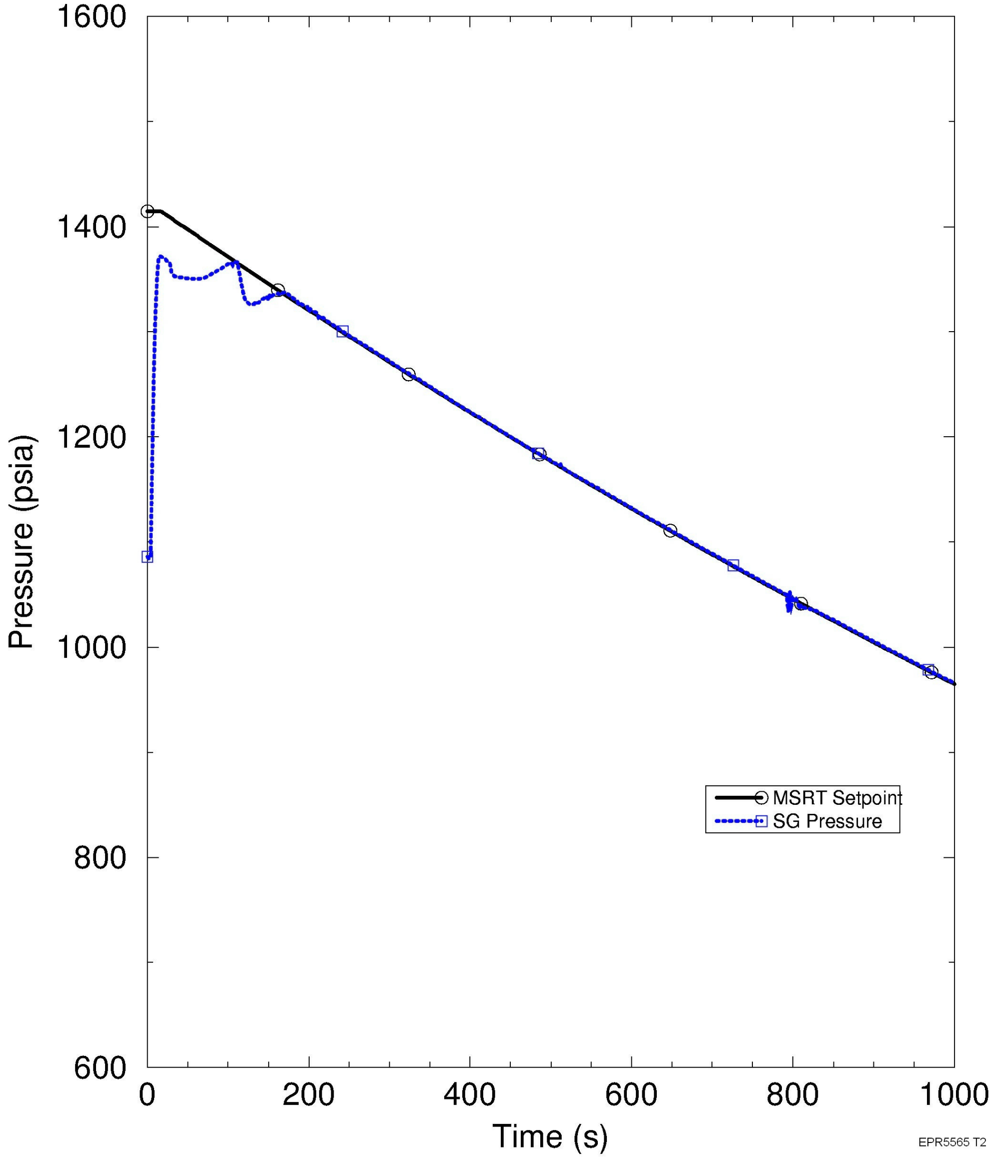

Figure 15.6-63—SBLOCA 6.5 Inch Break - Primary System Inventory

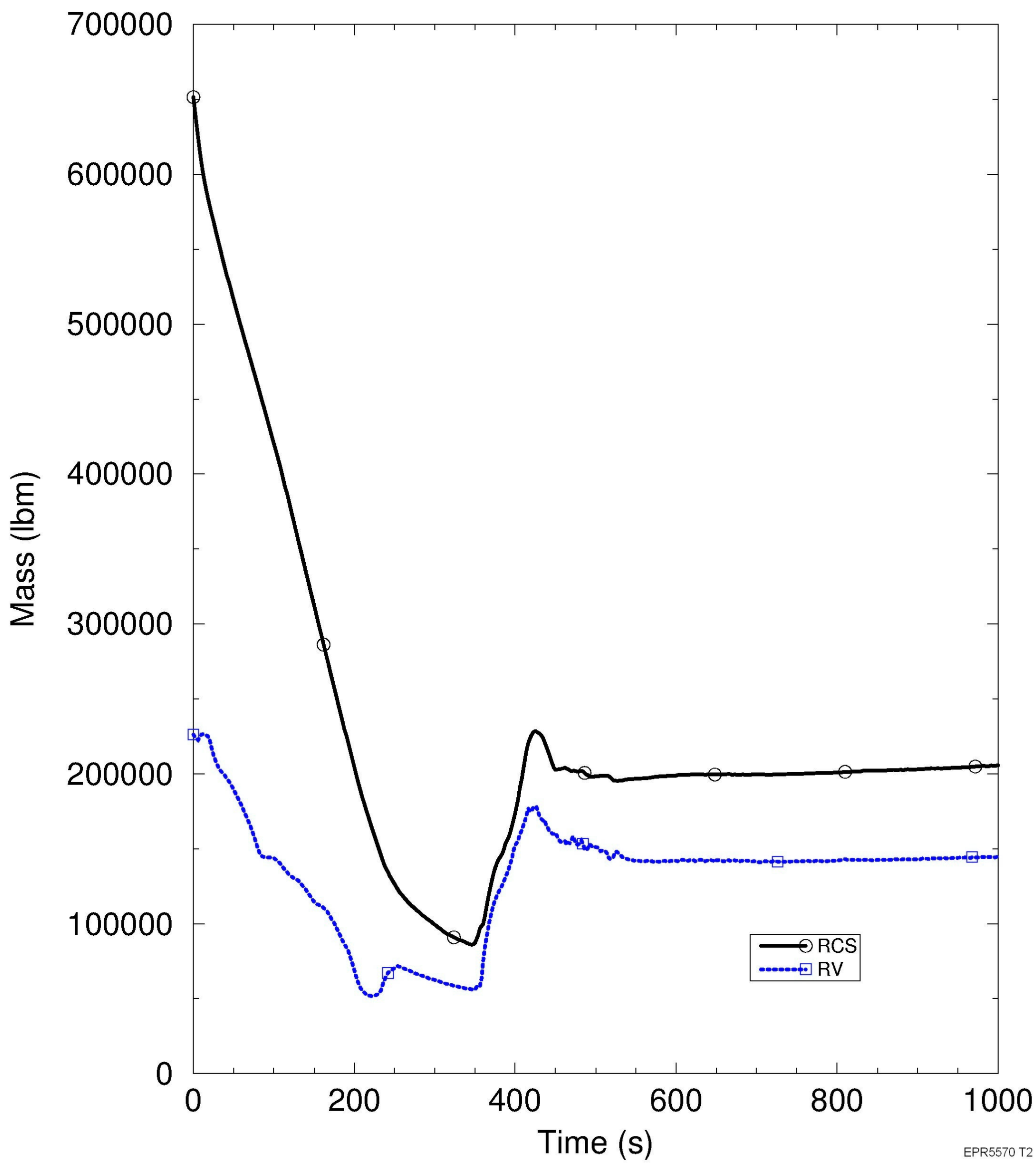

Figure 15.6-64—SBLOCA - 6.5 Inch Break - Hot Assembly Collapsed Liquid Level

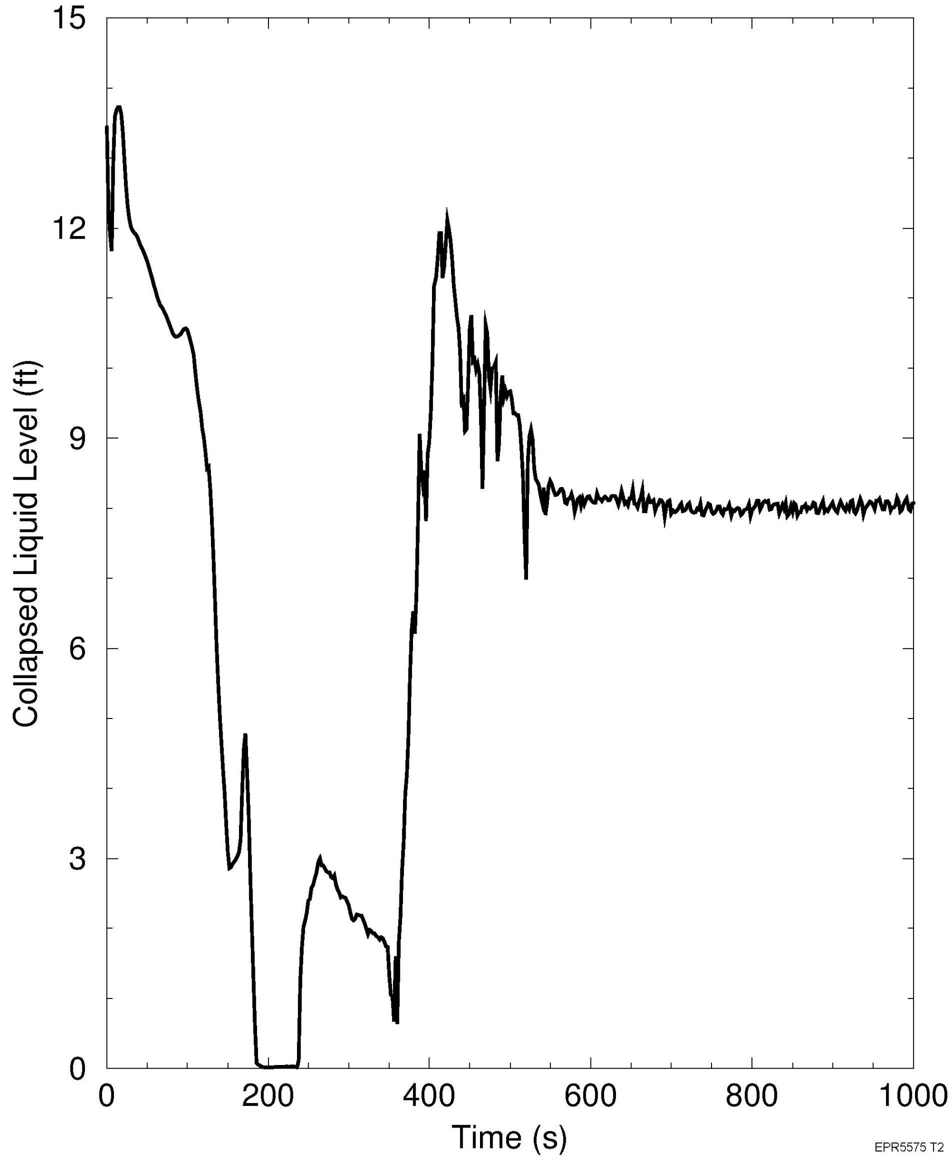

Figure 15.6-65—SBLOCA - 6.5 Inch Break - MHSI Flow Rate

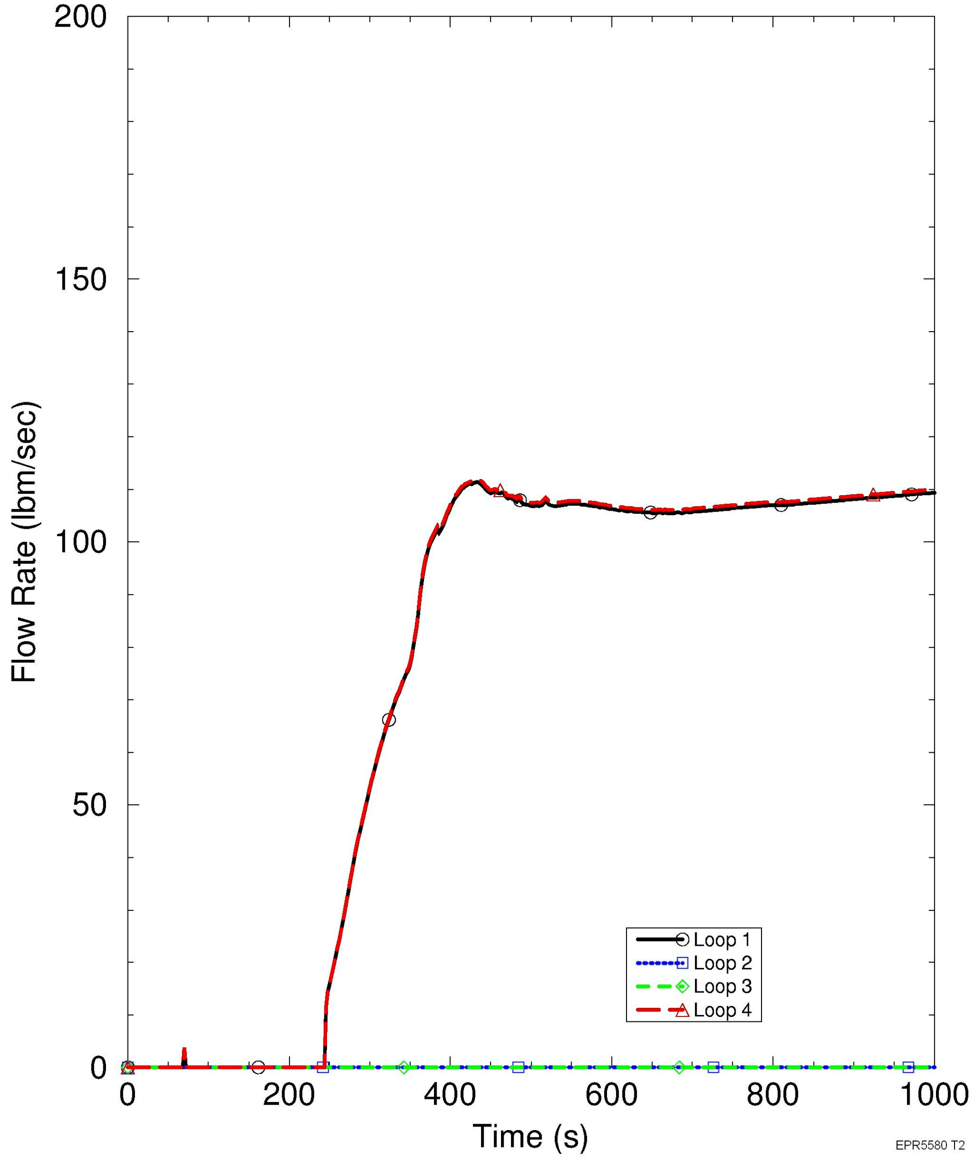

Figure 15.6-66—SBLOCA - 6.5 Inch Break - LHSI Flow

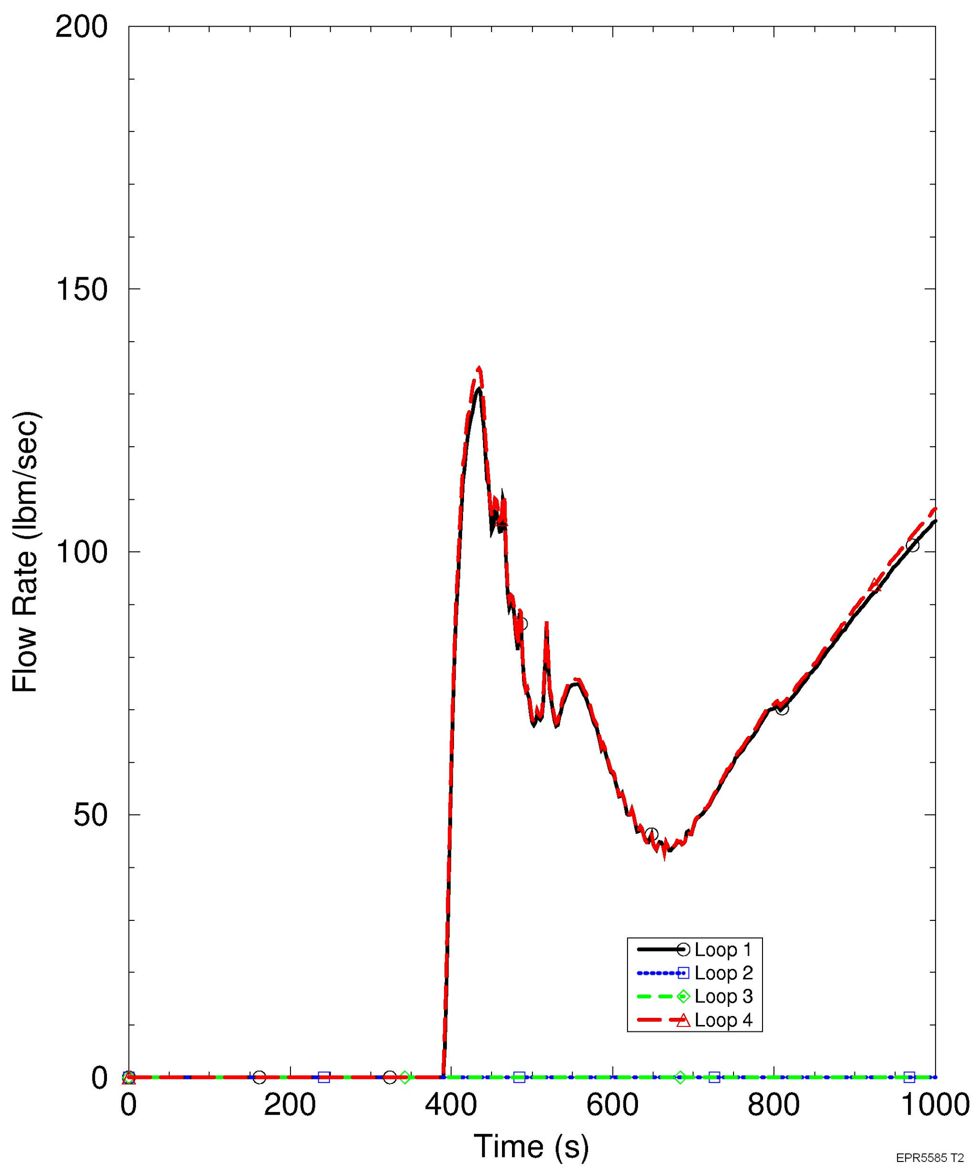

Figure 15.6-67—SBLOCA - 6.5 Inch Break - Peak Cladding Temperature and Coolant Temperature

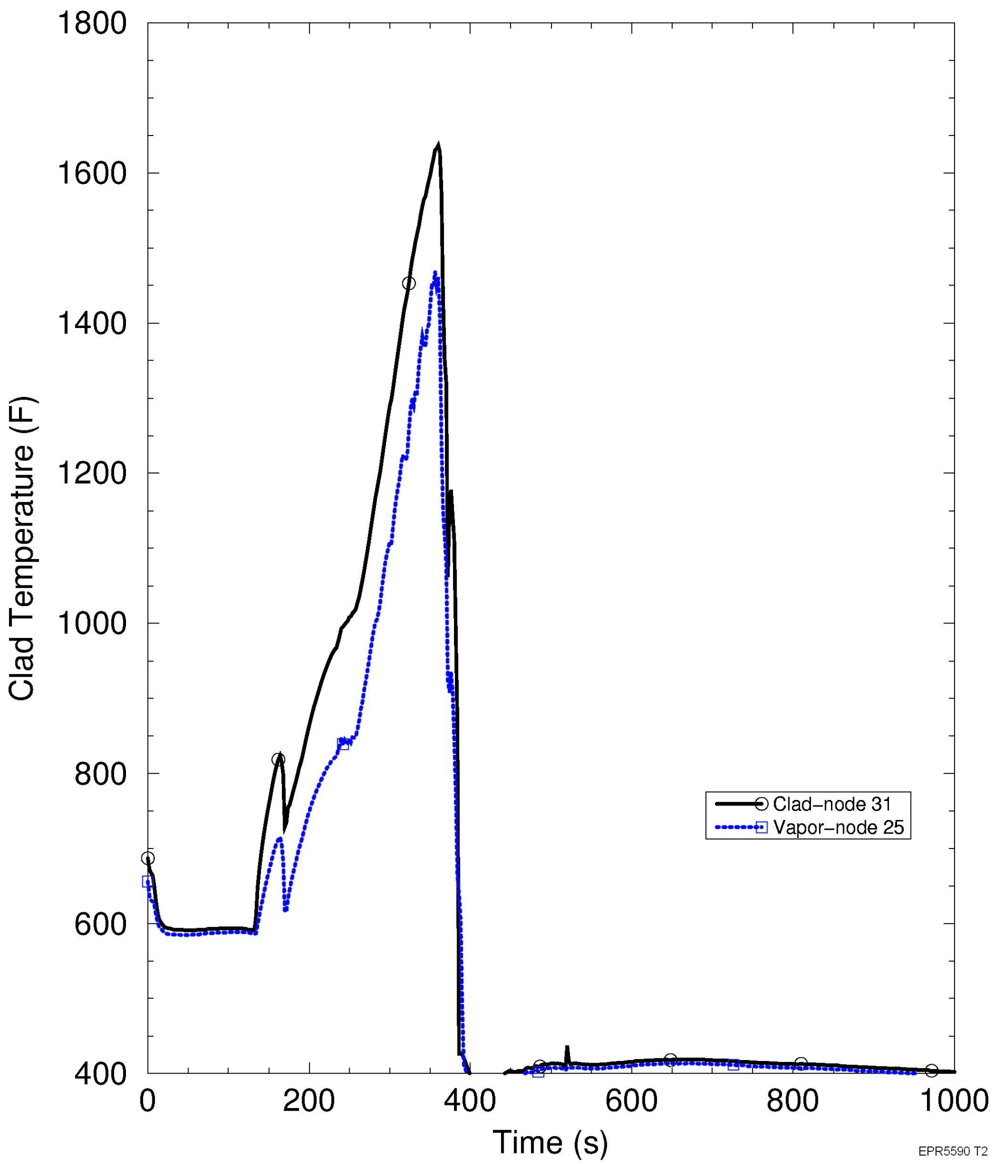

Figure 15.6-68—SBLOCA - 6.5 inch Break - Hot Assembly Cladding Temperature and Coolant Temperature

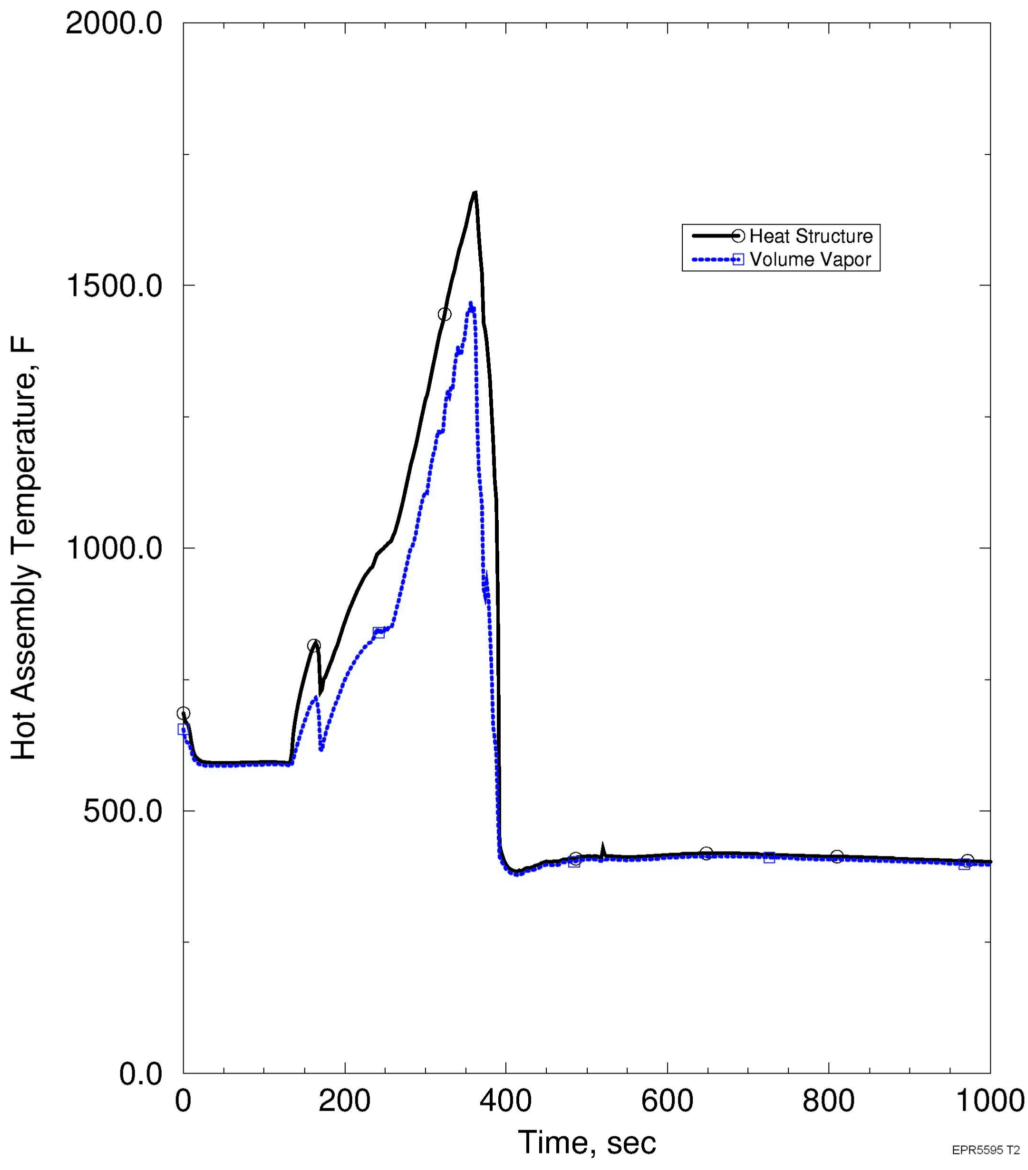

Figure 15.6-69—SBLOCA - 6.5 inch Break - Inner Core Cladding Temperature and Coolant Temperature

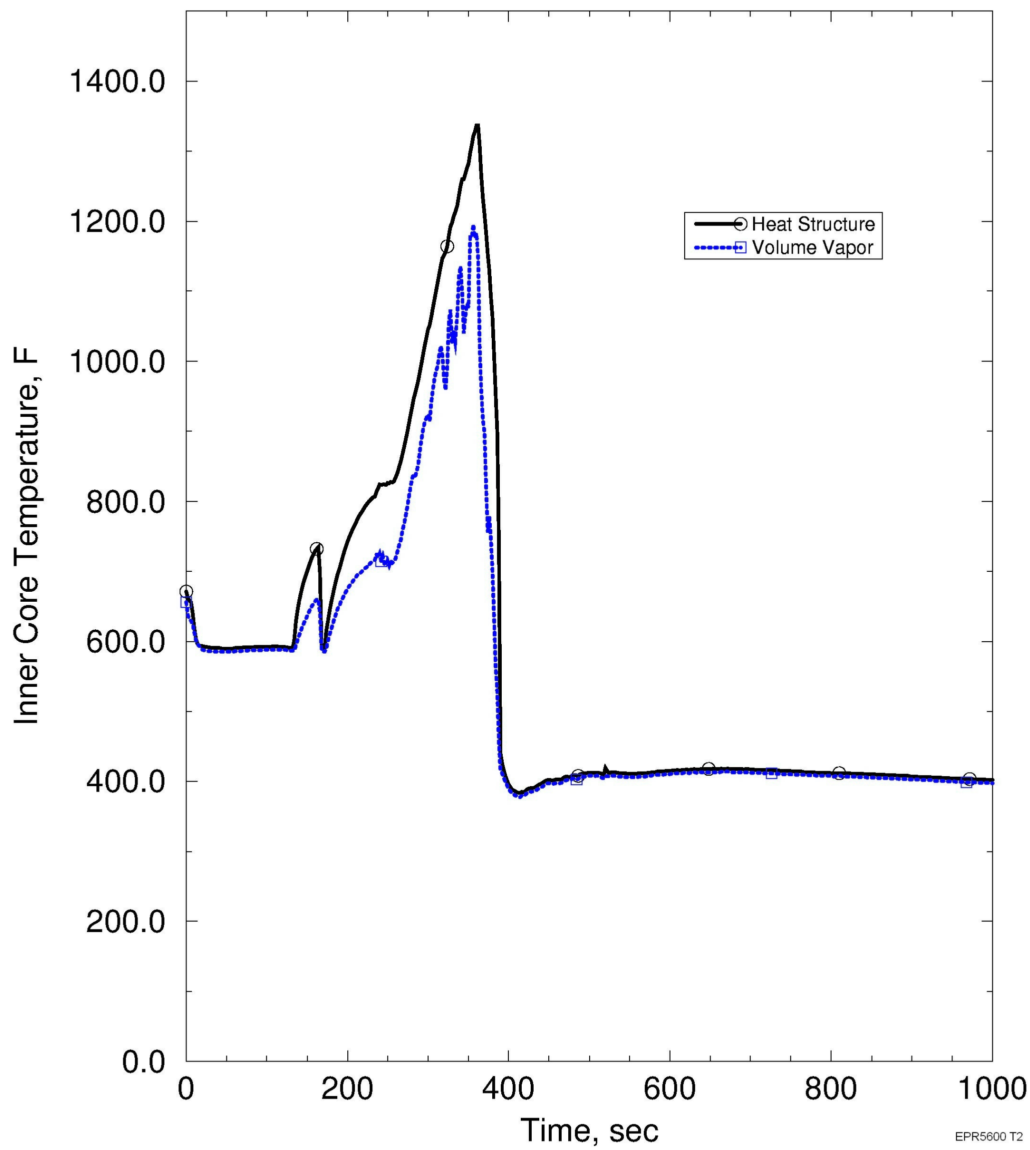

Figure 15.6-70—SBLOCA - 6.5 inch Break - Outer Core Cladding Temperature and Coolant Temperature

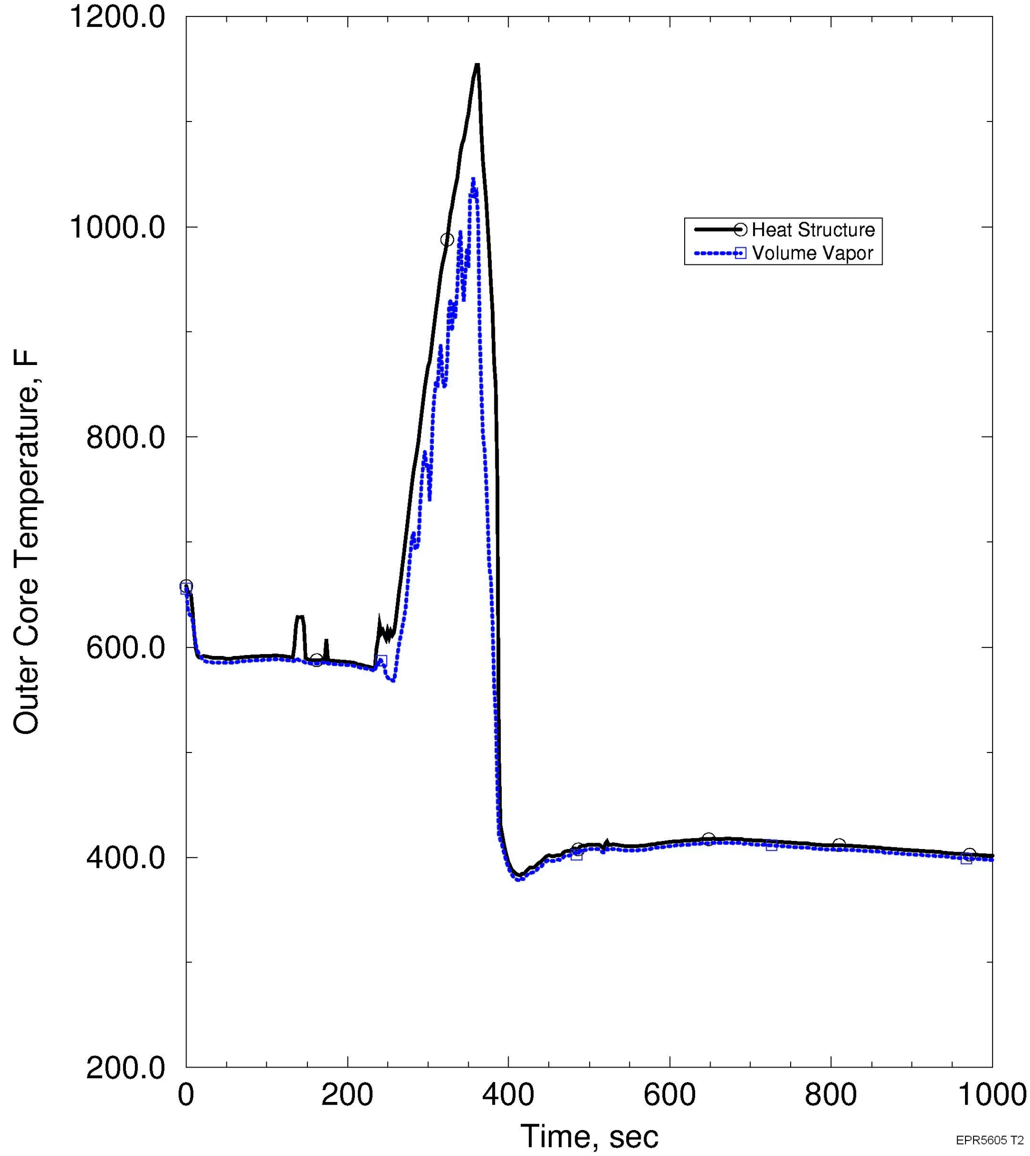

Figure 15.6-71—SBLOCA - 6.5 inch Break - Heat Transfer Coefficients for Hot Rod, Hot Assembly, Inner Core and Outer Core

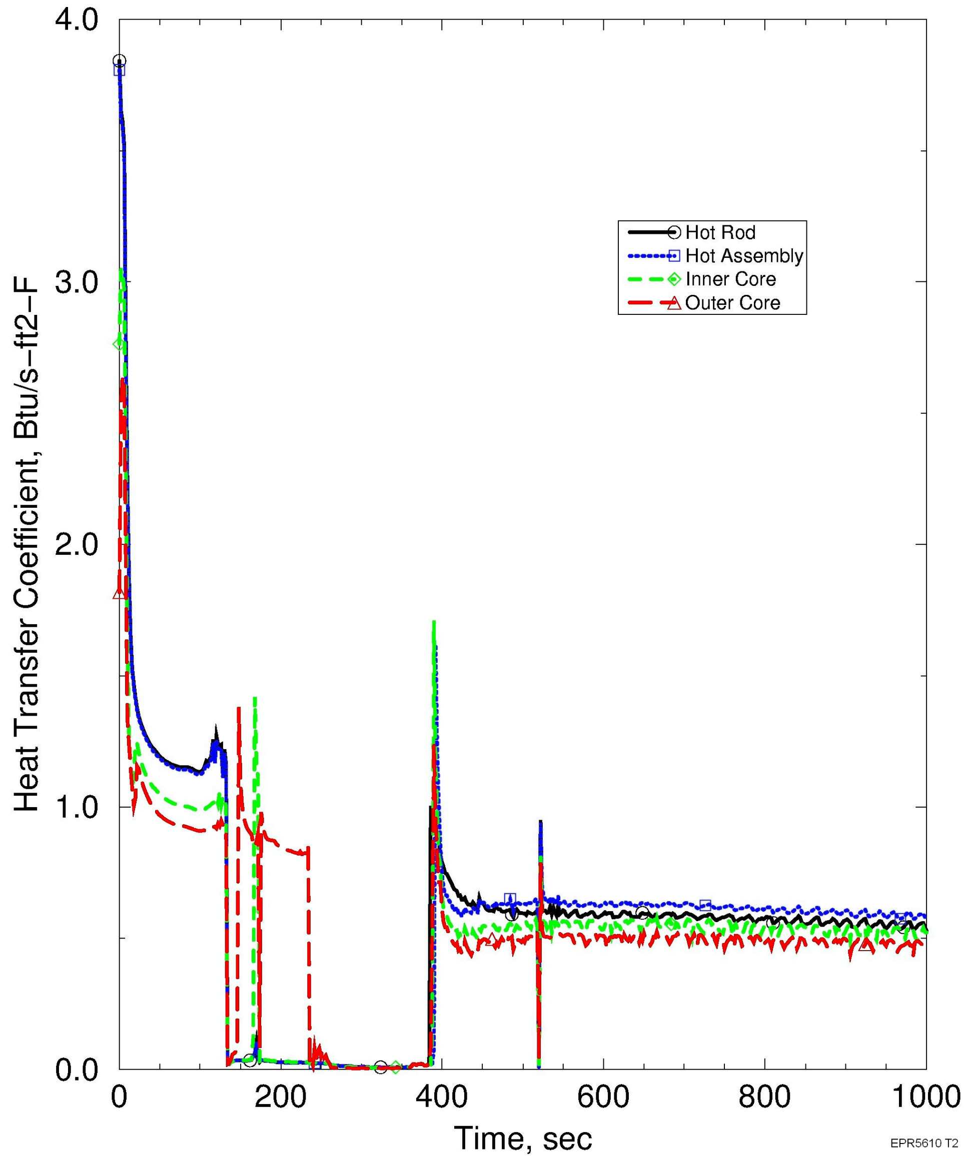

Figure 15.6-72—SBLOCA - 6.5 inch Break - Core Inlet Quality

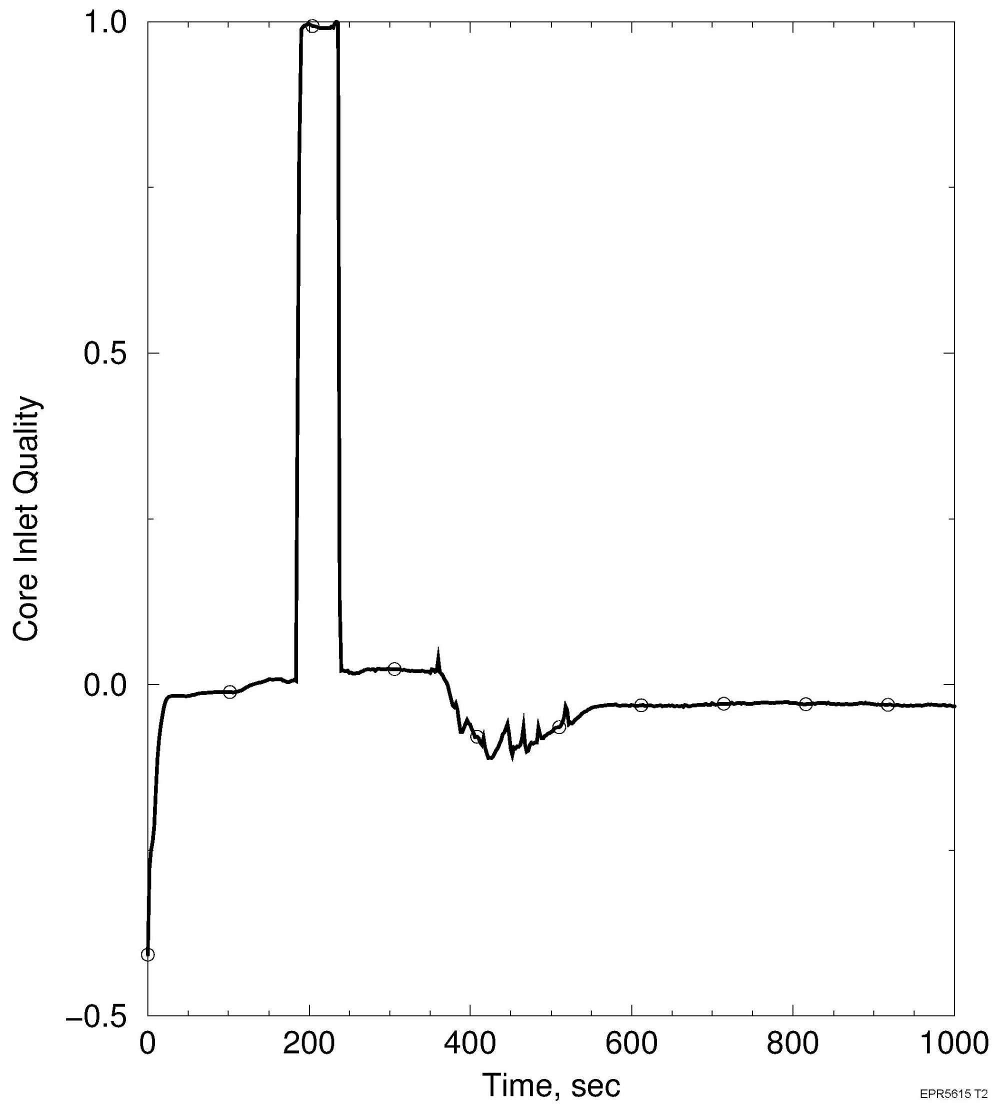

Figure 15.6-73—SBLOCA - 6.5 inch Break - Core Outlet Quality

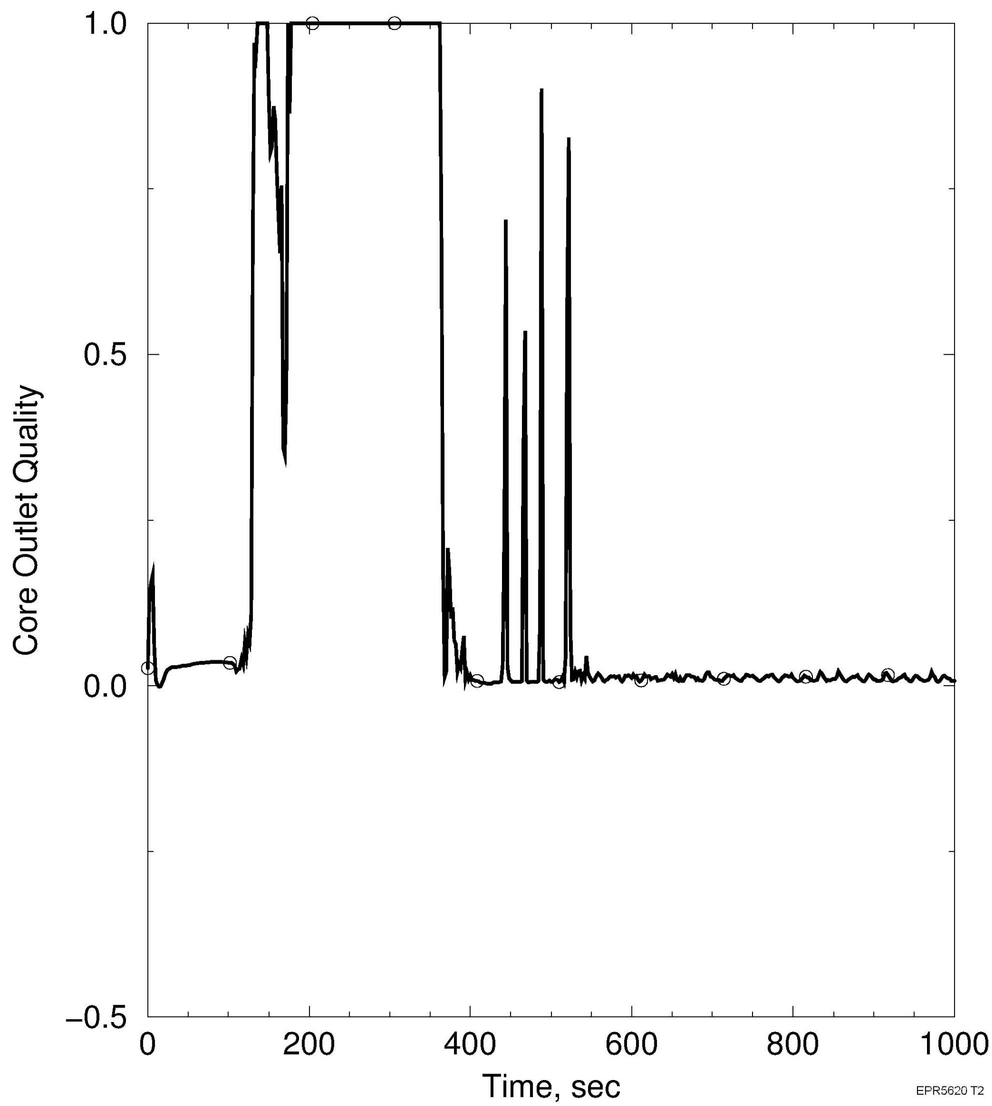

Figure 15.6-74—SBLOCA - 6.5 inch Break - Core Inlet Temperature

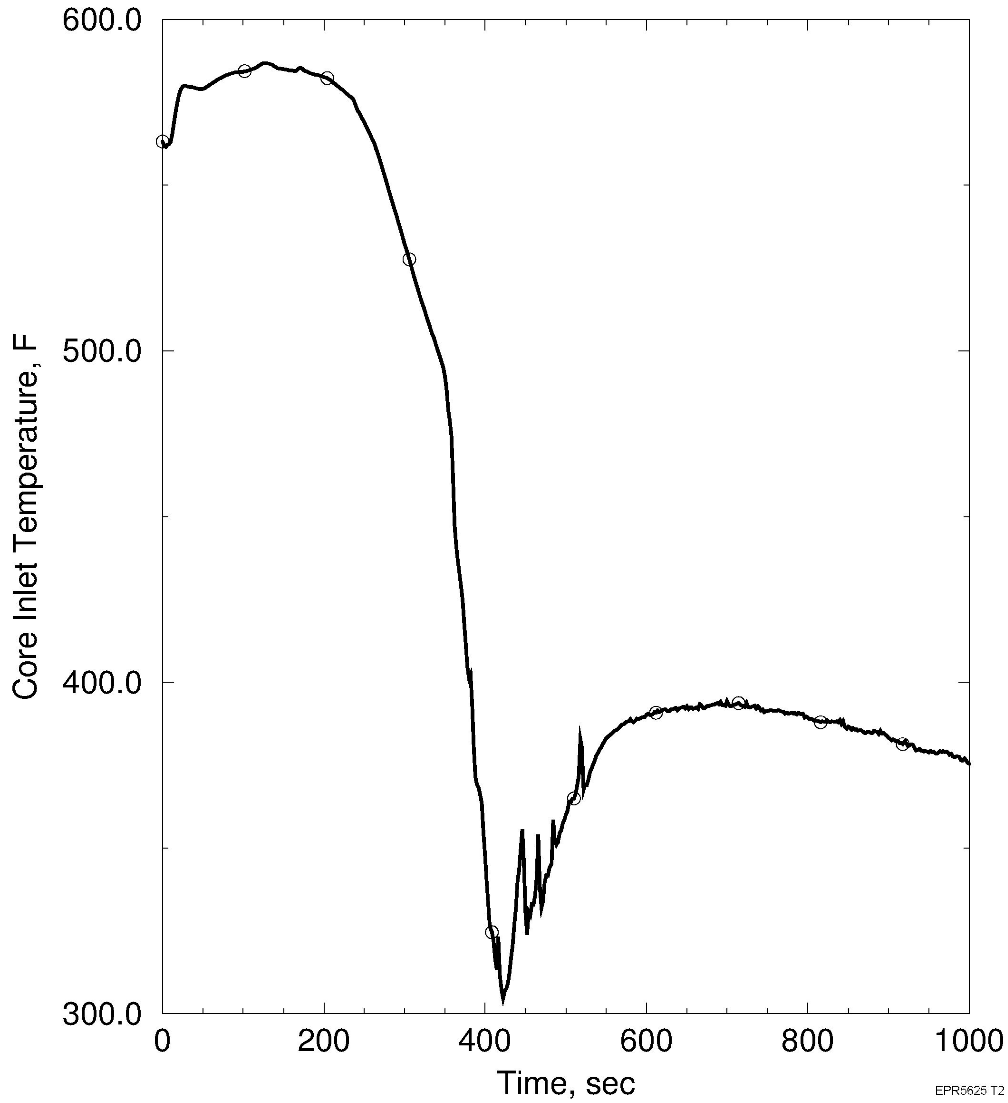

Figure 15.6-75—SBLOCA - 6.5 inch Break - Core Outlet Temperature

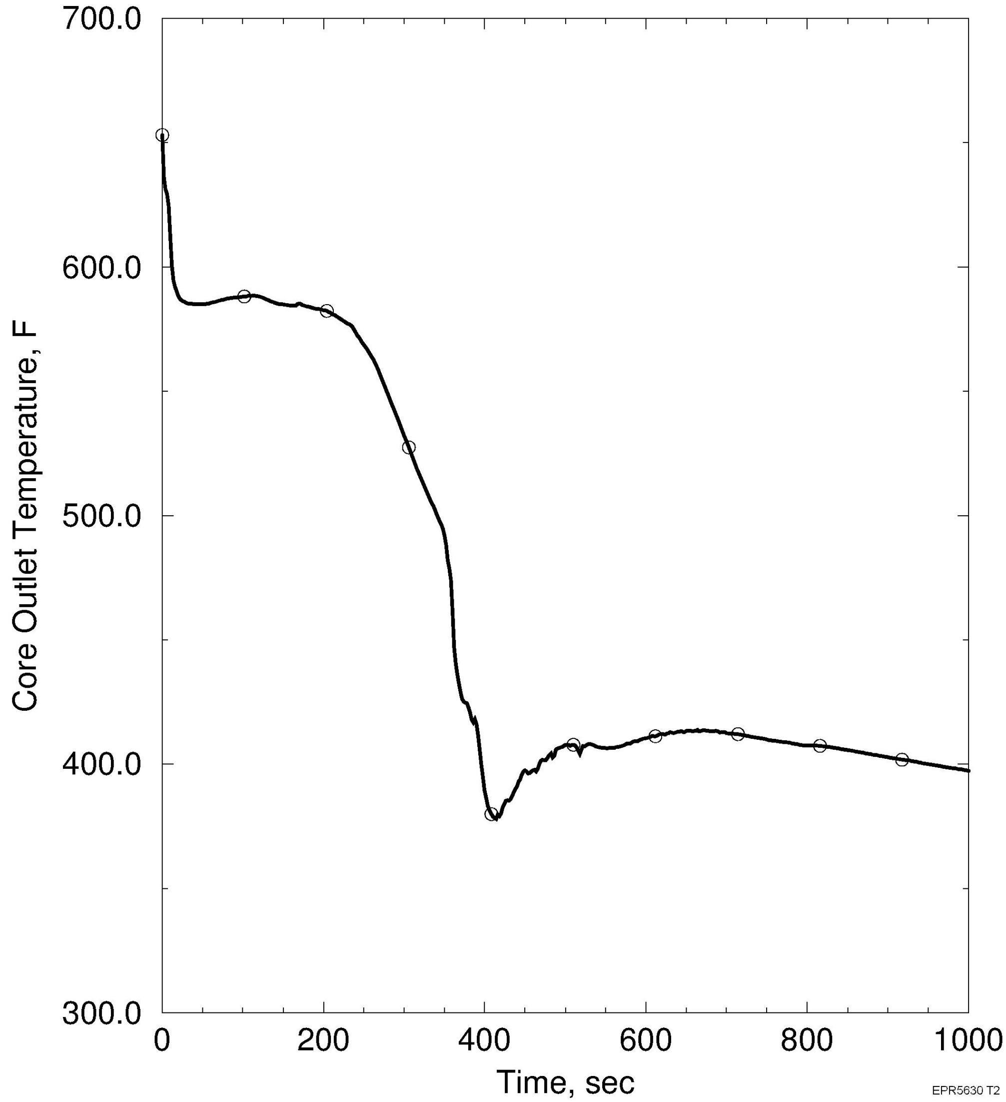

Figure 15.6-76—SBLOCA - 6.5 inch Break - Core Inlet Mass Flow Rate

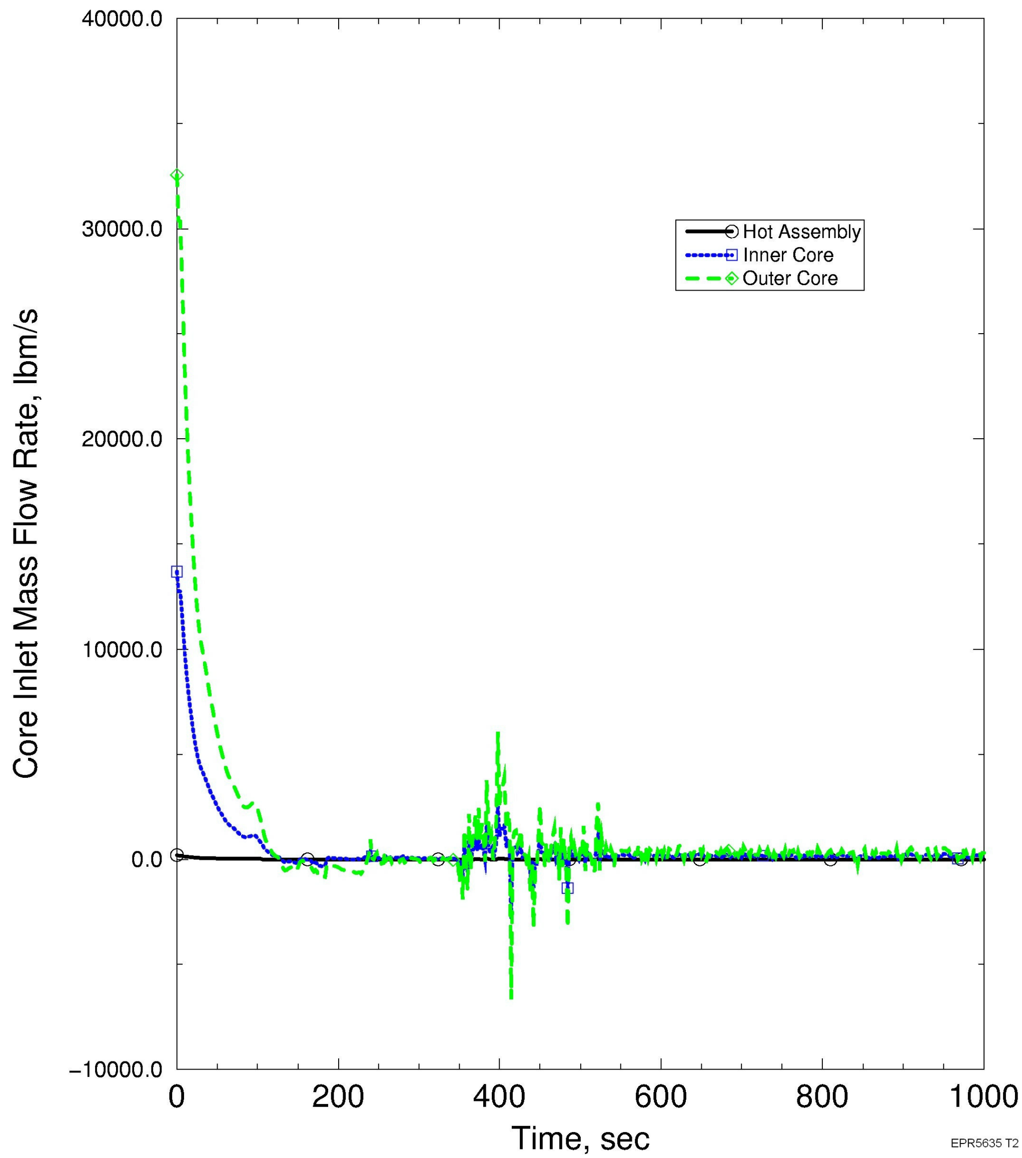

Figure 15.6-77—SBLOCA - 6.5 inch Break - Core Outlet Mass Flow Rate

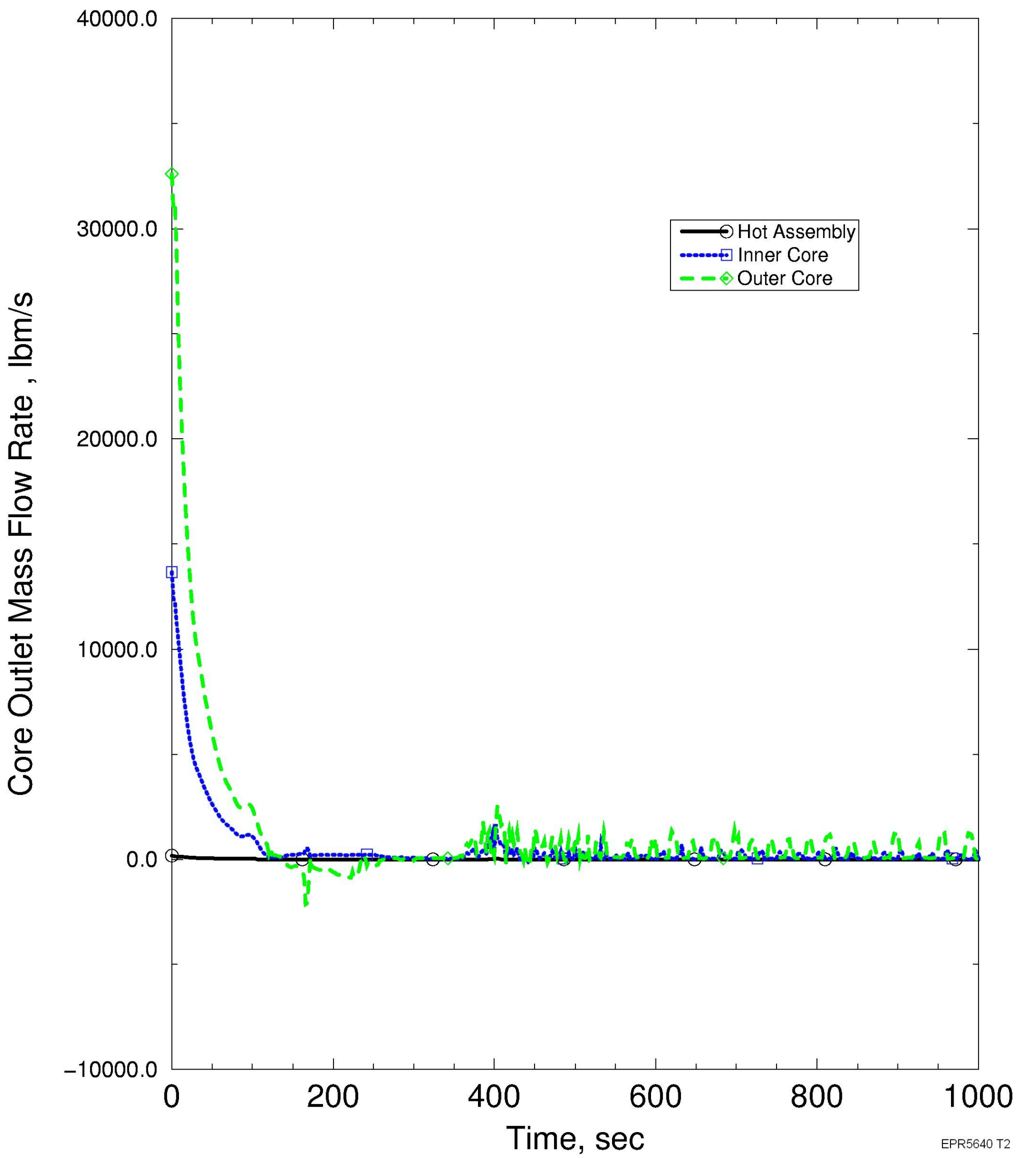

Figure 15.6-78—SBLOCA - 6.5 inch Break - Quality at the PCT Node Location

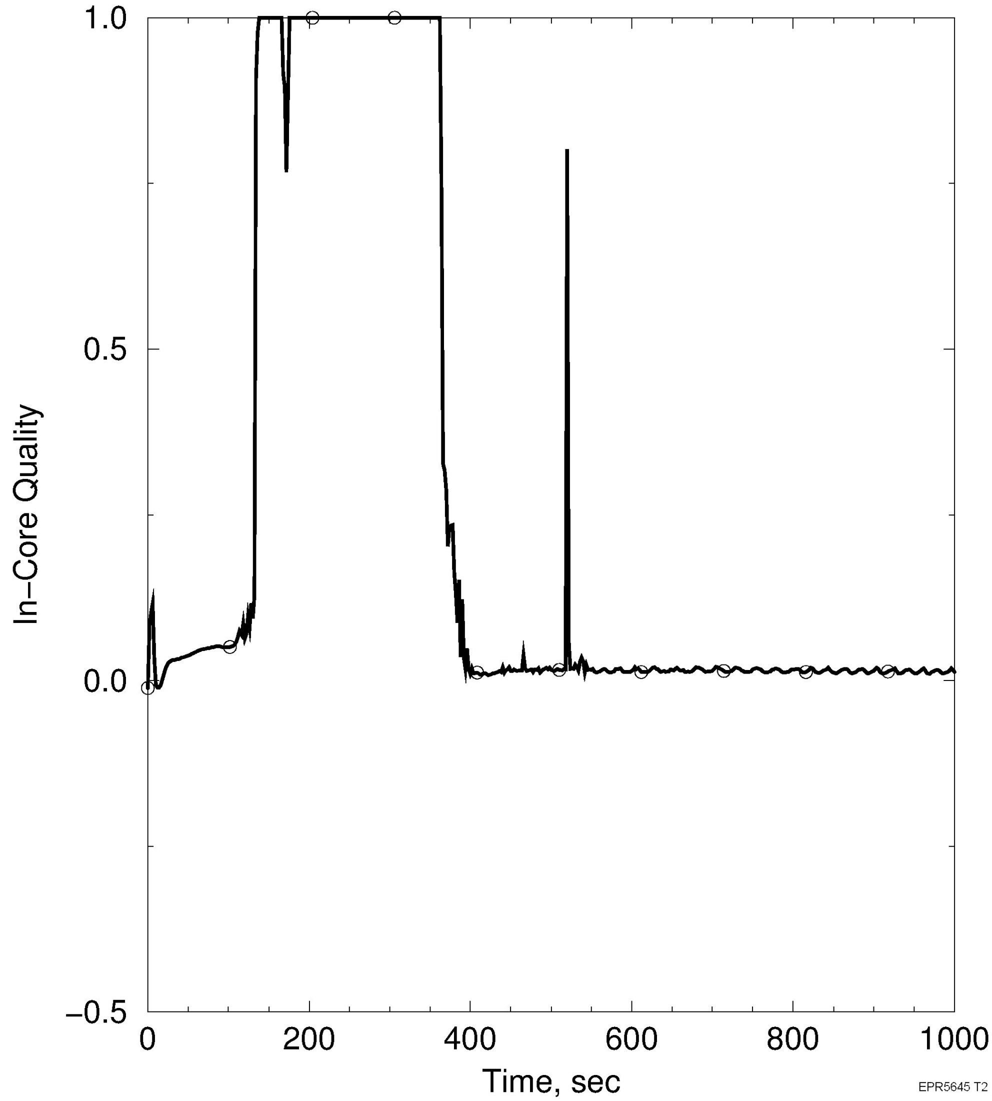

Figure 15.6-79—SBLOCA - 6.5 inch Break - Reactor Vessel Downcomer Mass Flow Rate

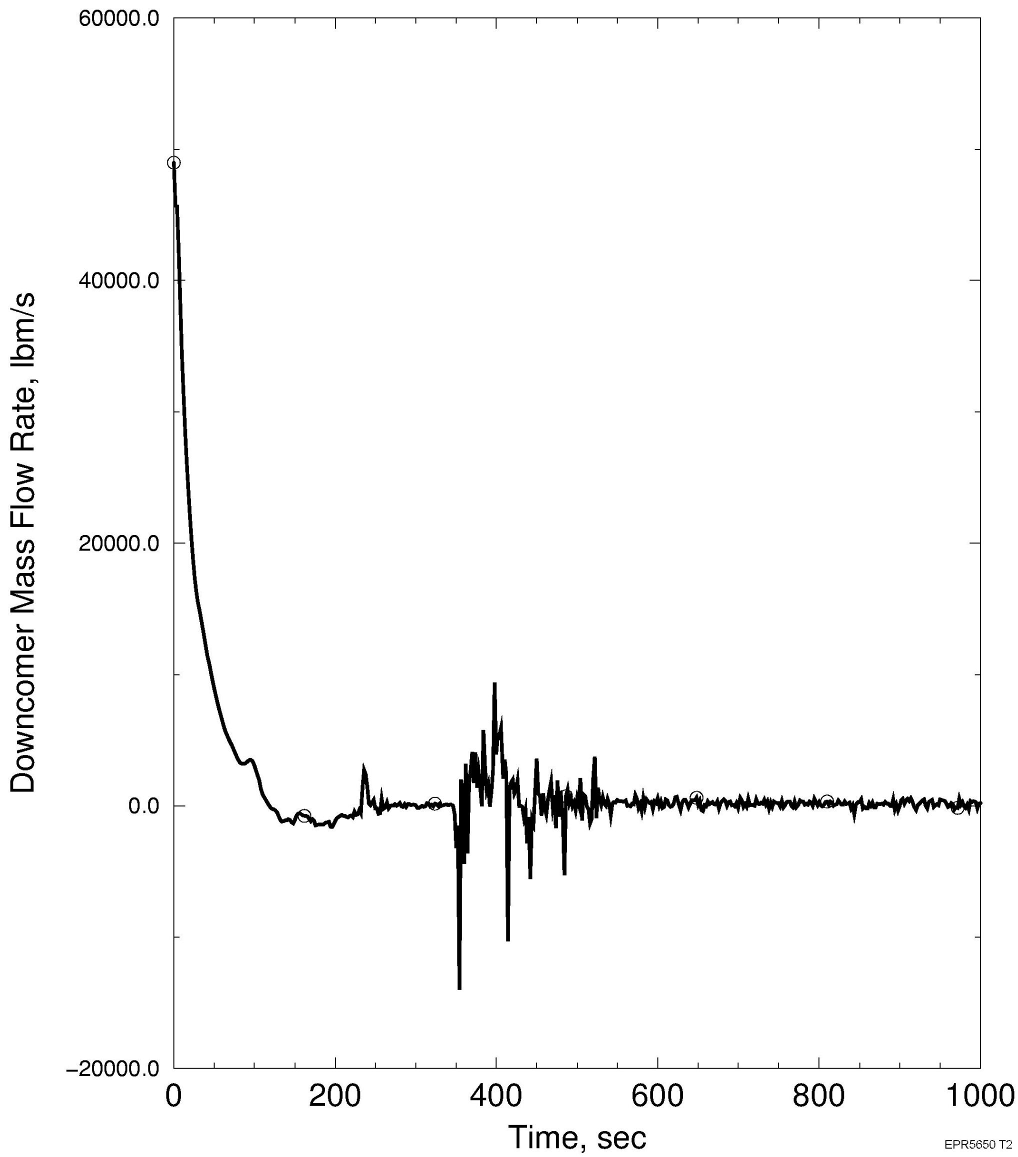

Figure 15.6-80—SBLOCA - 6.5 inch Break - Primary System to Secondary System Heat Transfer Rate

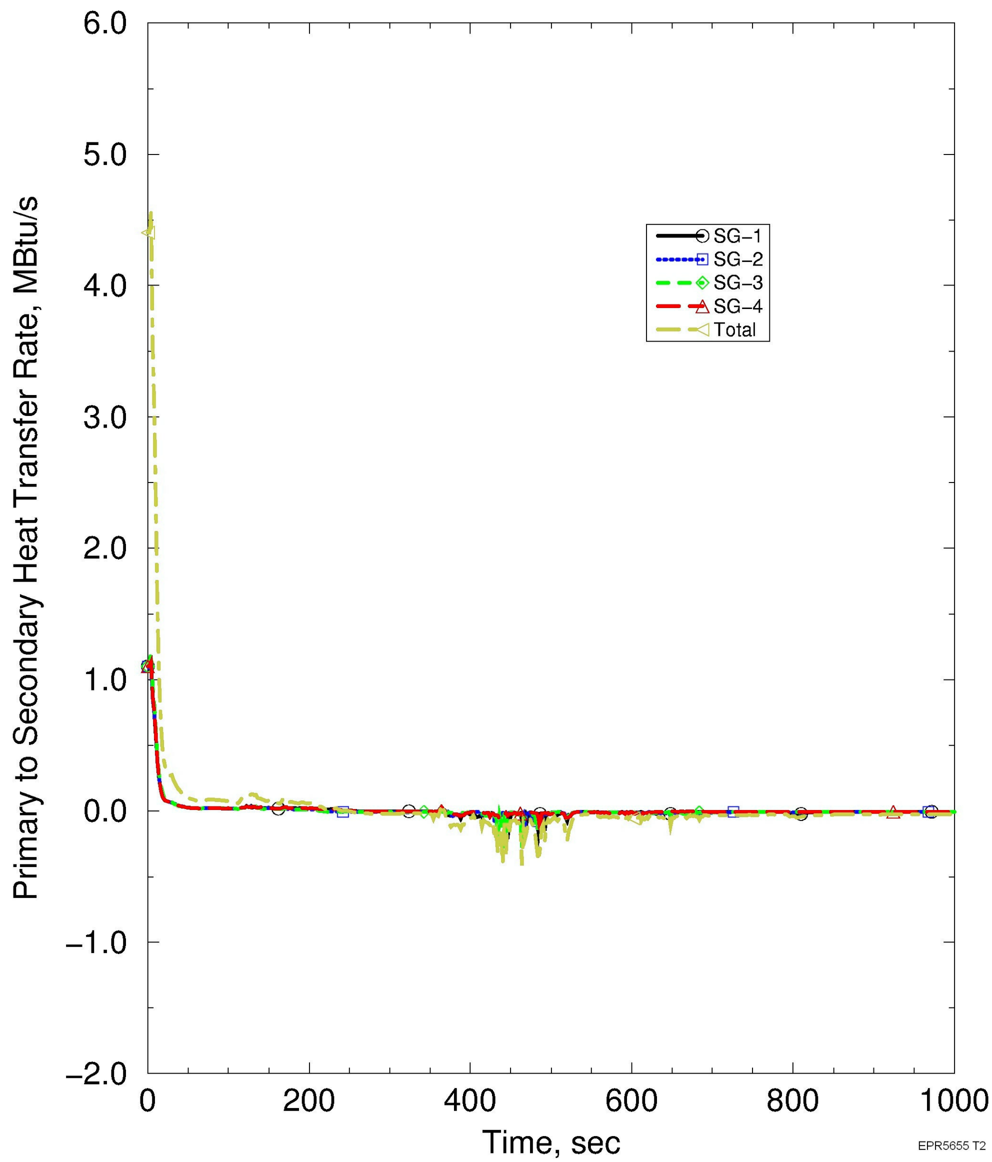

Figure 15.6-81—SBLOCA - 6.5 inch Break - RC Speed

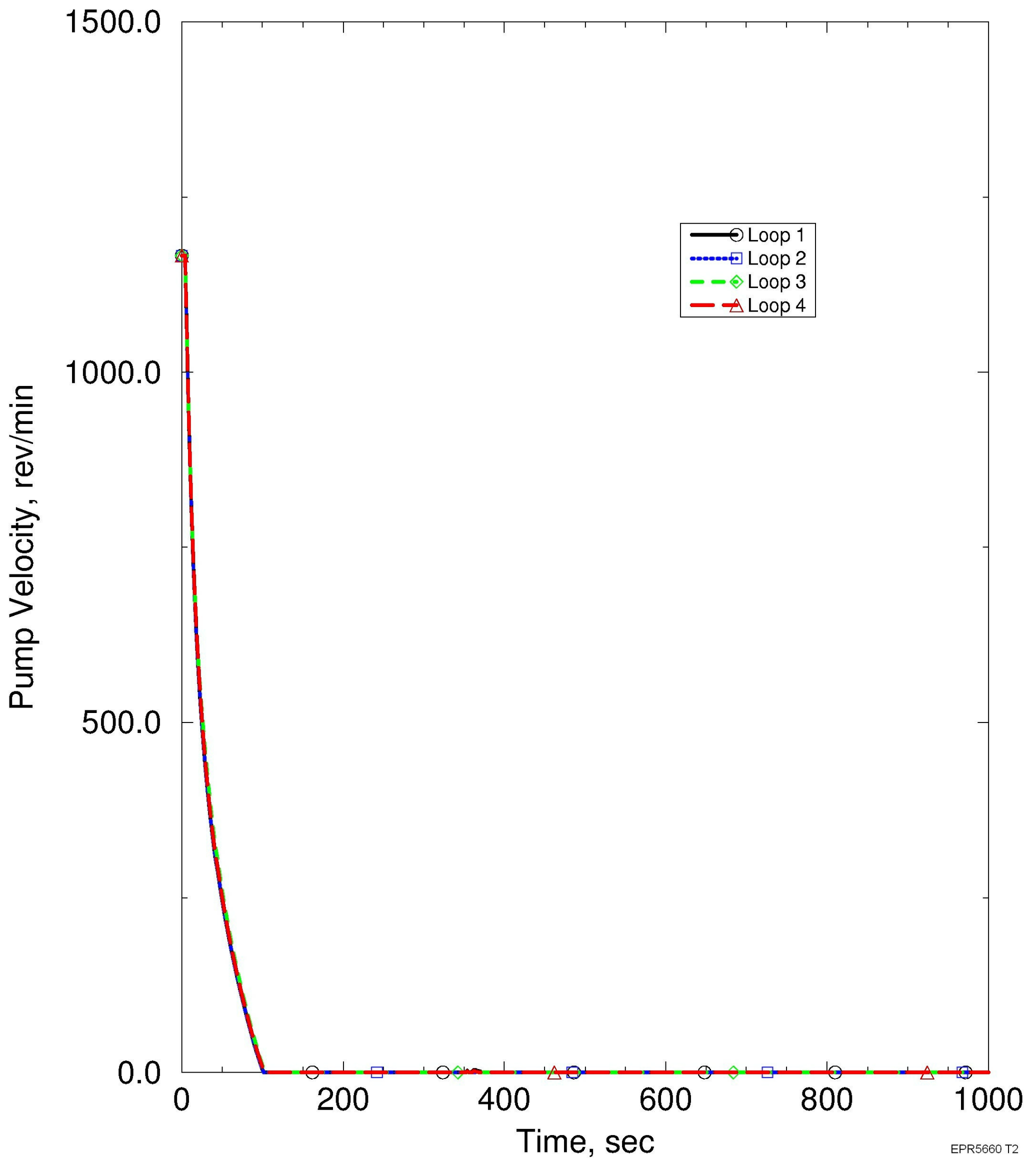

Figure 15.6-82—SBLOCA - PCT - Delayed Pump Trip (Without LOOP) Break Spectrum

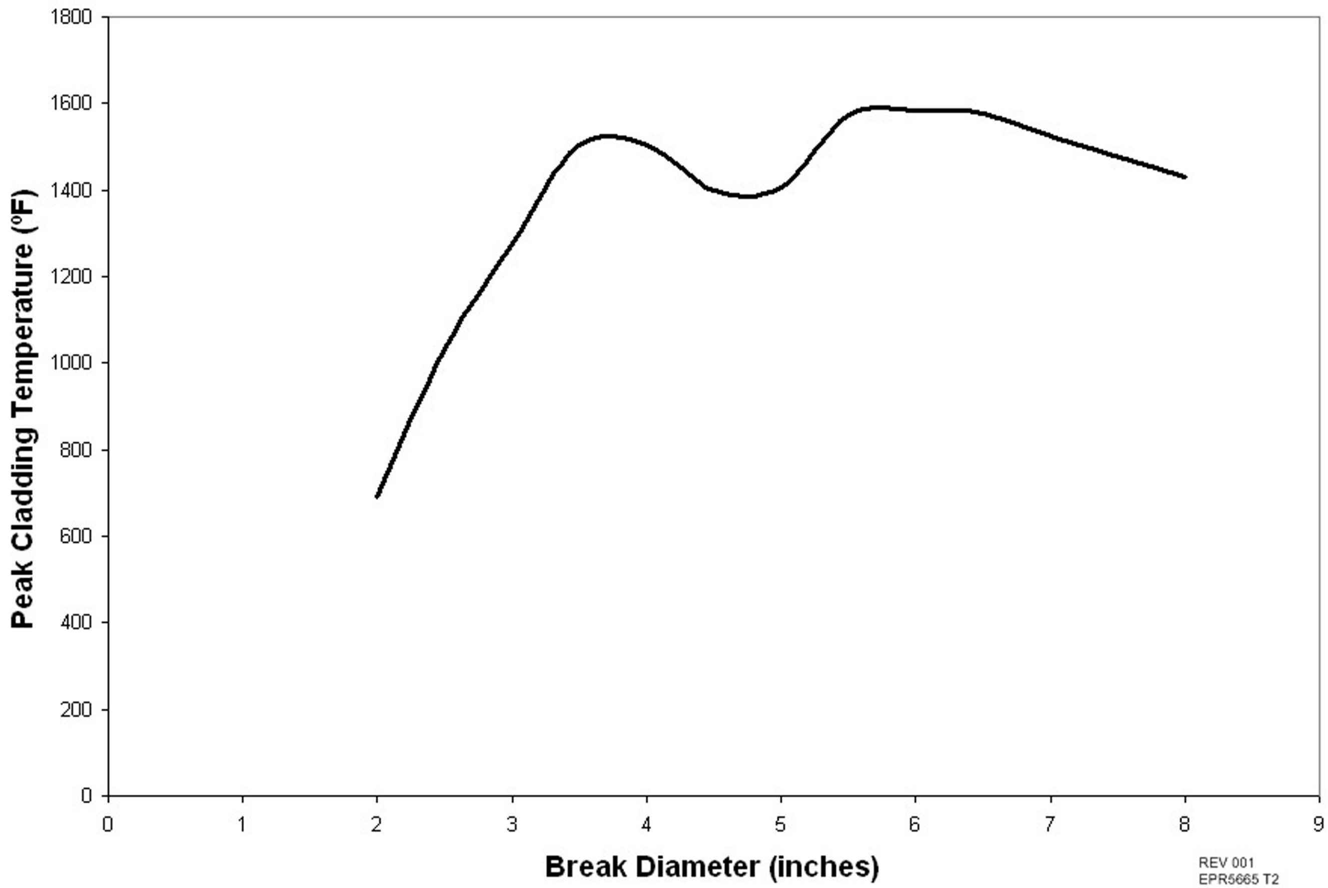

Figure 15.6-83—SBLOCA - Comparison PCT - Break Spectrum With/Without LOOP

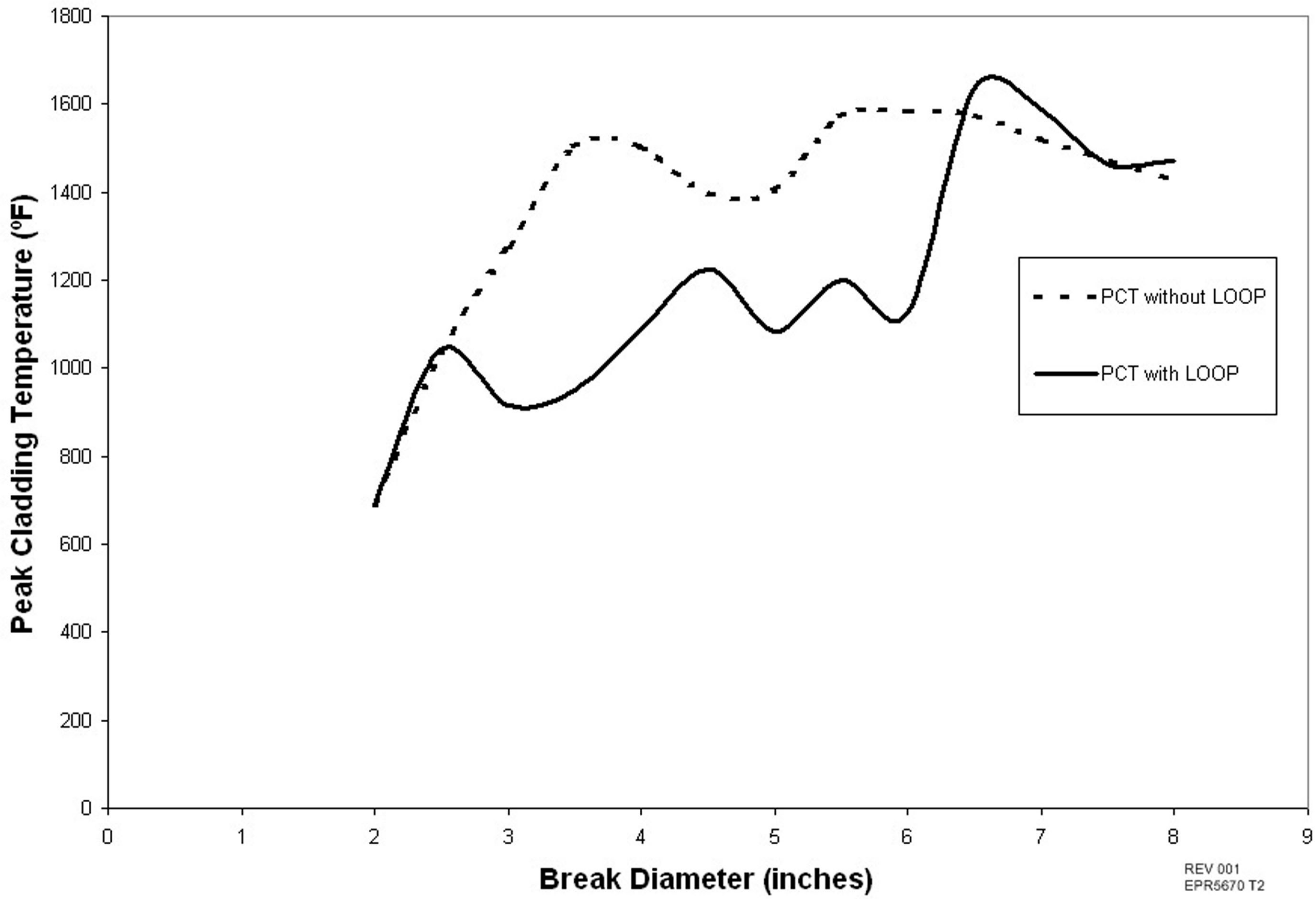

Figure 15.6-84—SBLOCA - 6.5 Inch Break - Integral of Upper Plenum Flow to the Hot Legs

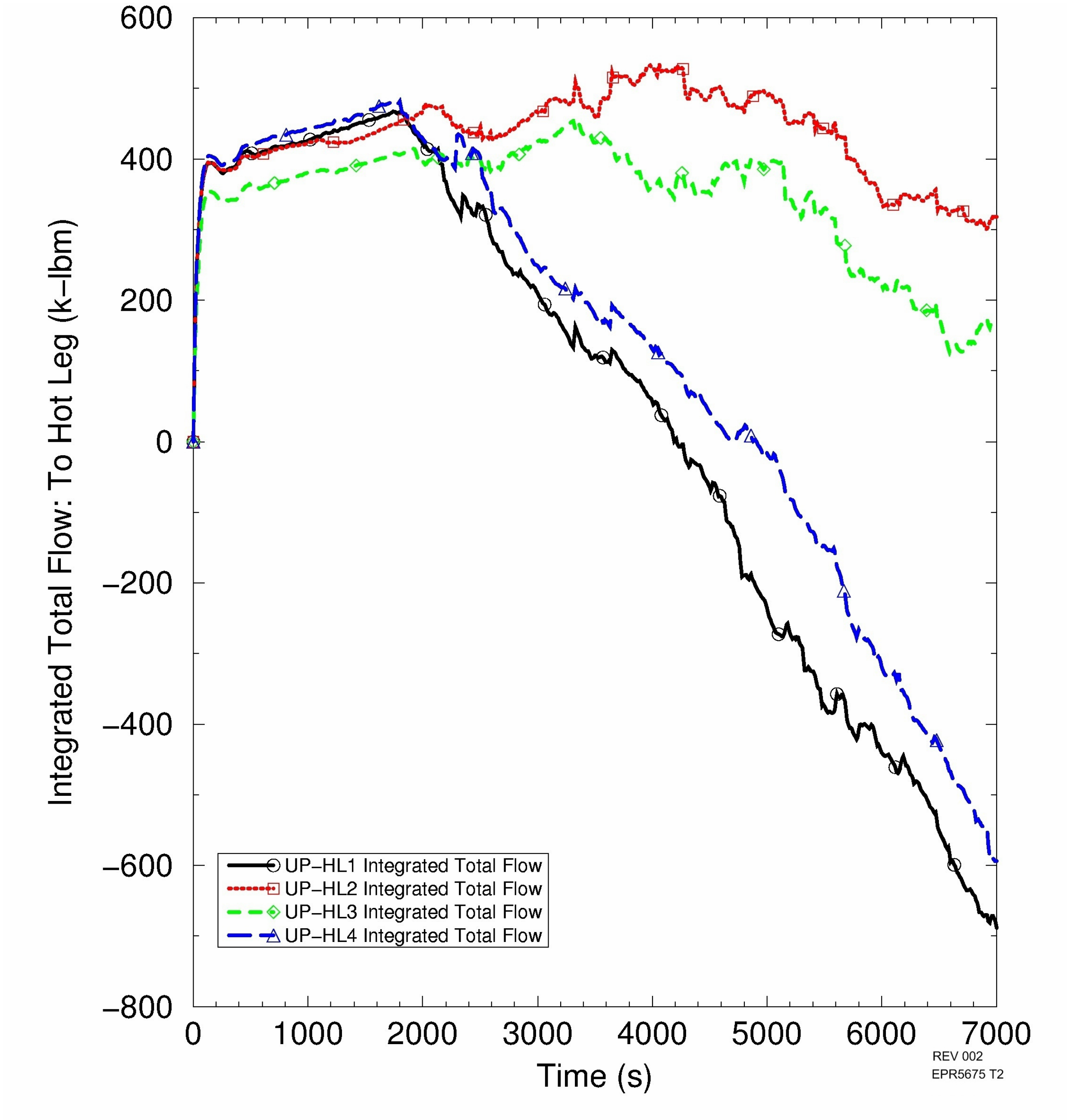

Figure 15.6-85—SBLOCA - 6.5 Inch Break - Integral of Core Region Exit Flows

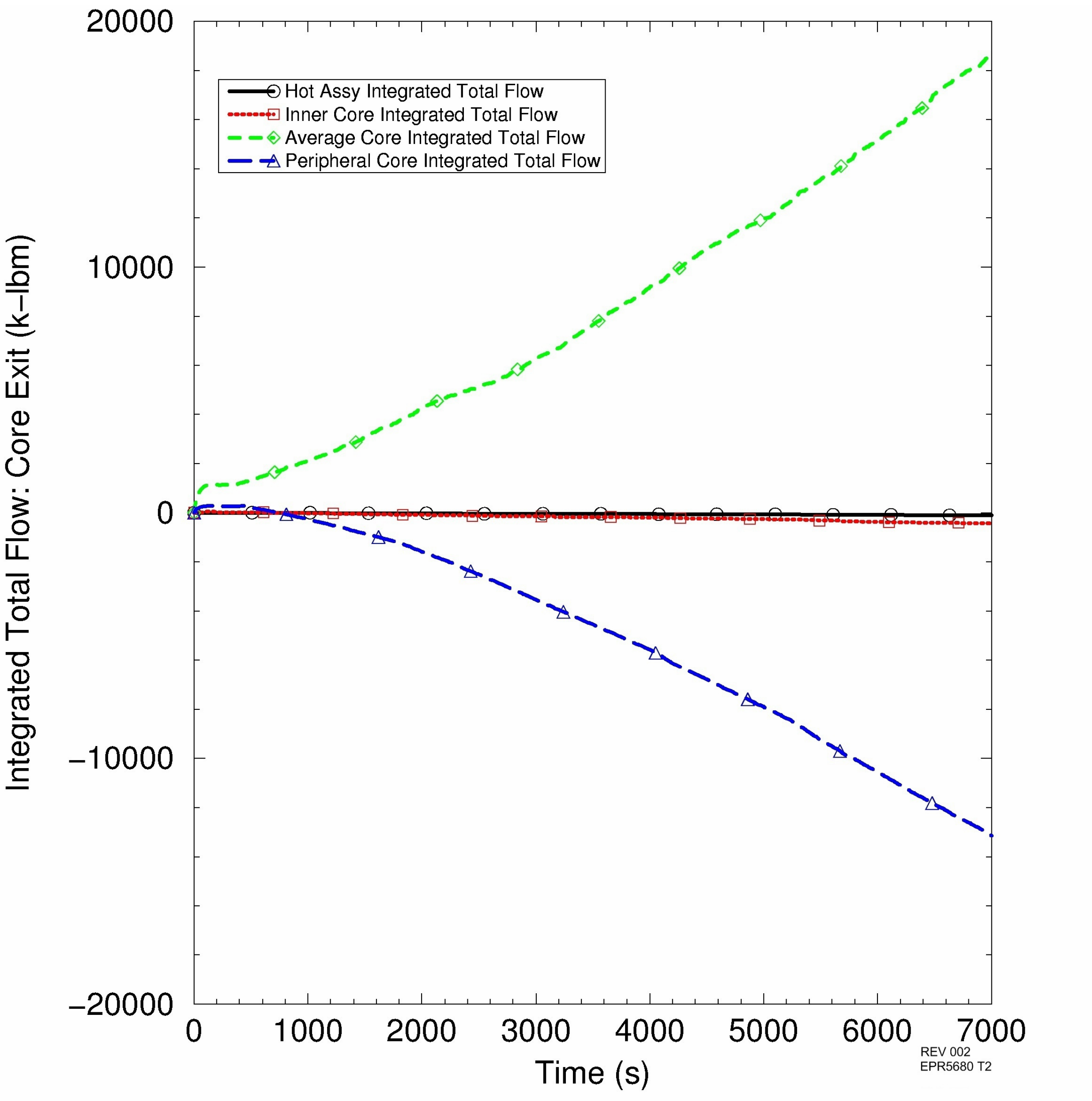

Figure 15.6-86—SBLOCA - 6.5 Inch Break - Integral of Lower Plenum Flow to Lower Head

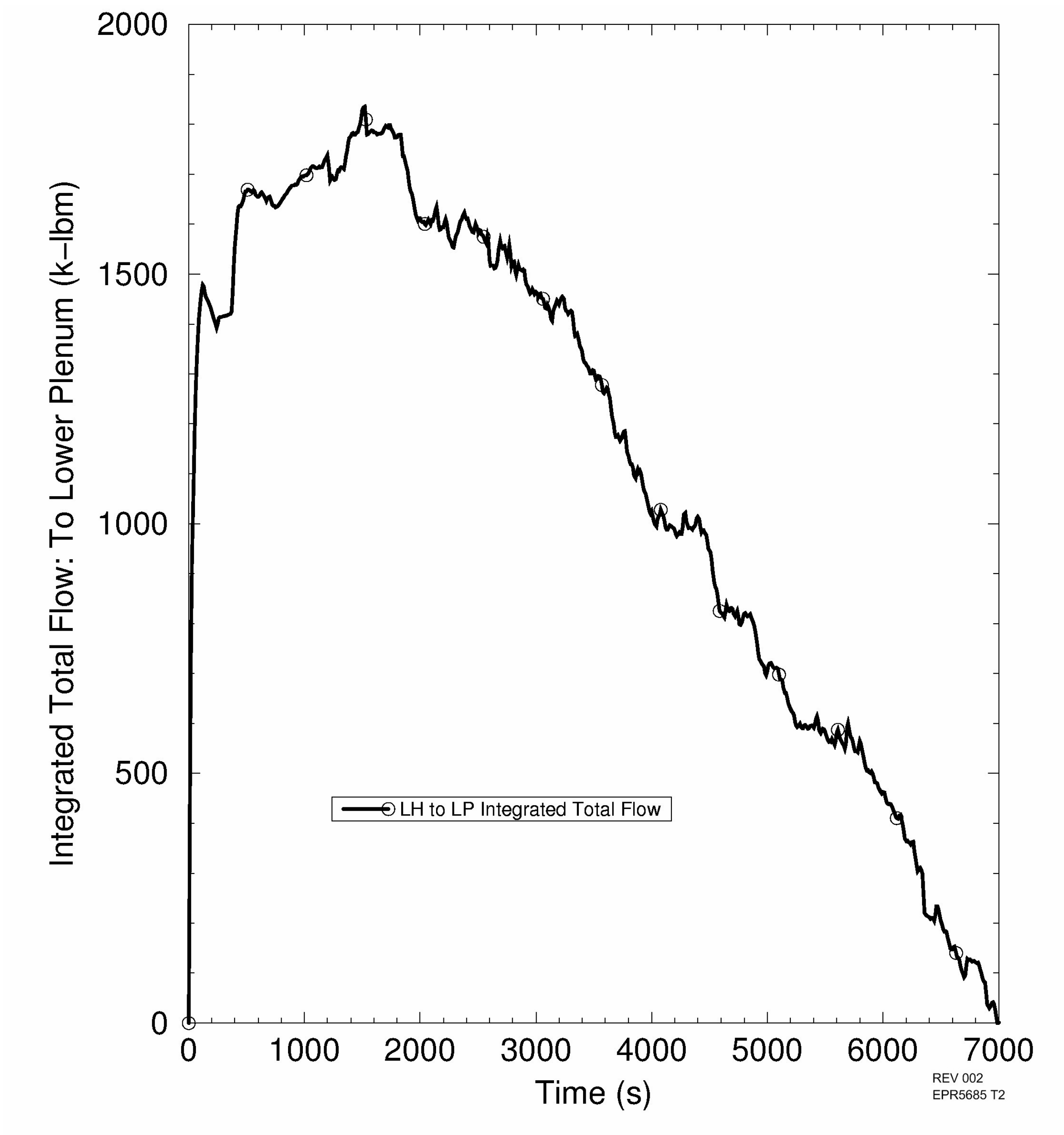

Figure 15.6-87—SBLOCA - 6.5 Inch Break - Pressurizer and Steam Generator 1 Pressure

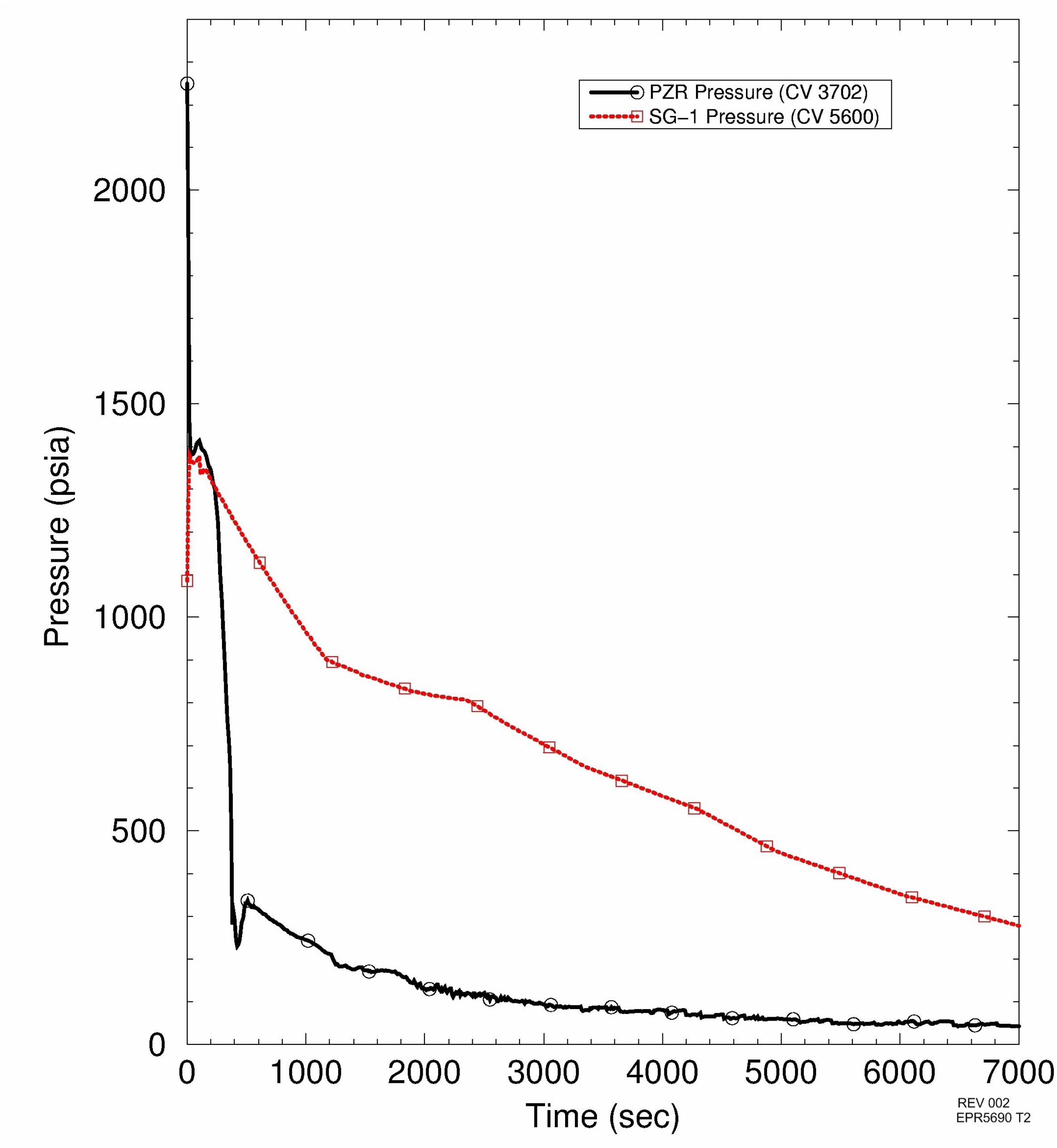

Figure 15.6-88—LBLOCA with Hot Leg Injection at 60 Minutes - Integrated Flow from Upper Plenum to Hot Legs

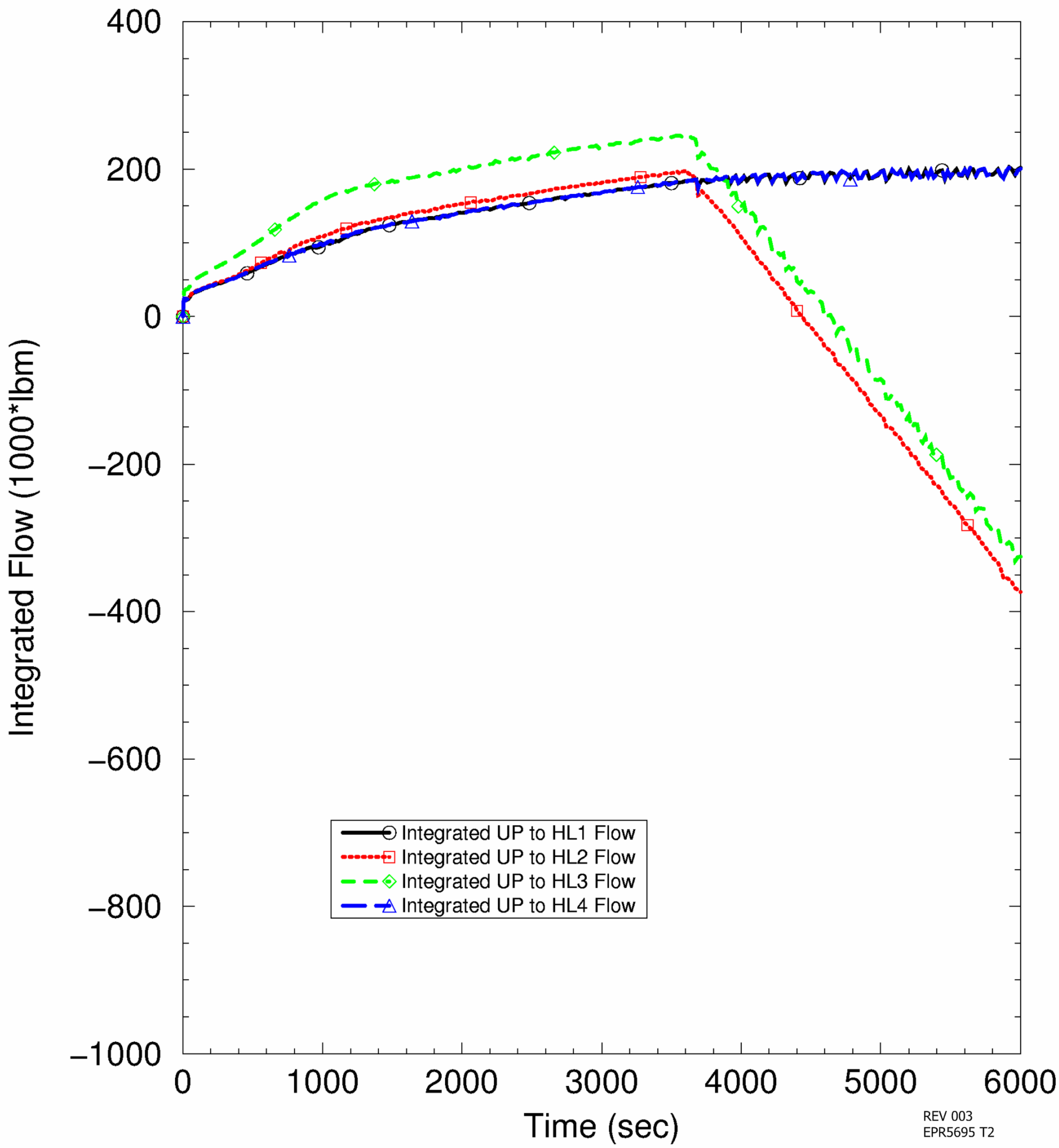

Figure 15.6-89—LBLOCA with Hot Leg Injection at 60 Minutes - Integrated Flow from Core Regions to Upper Plenum

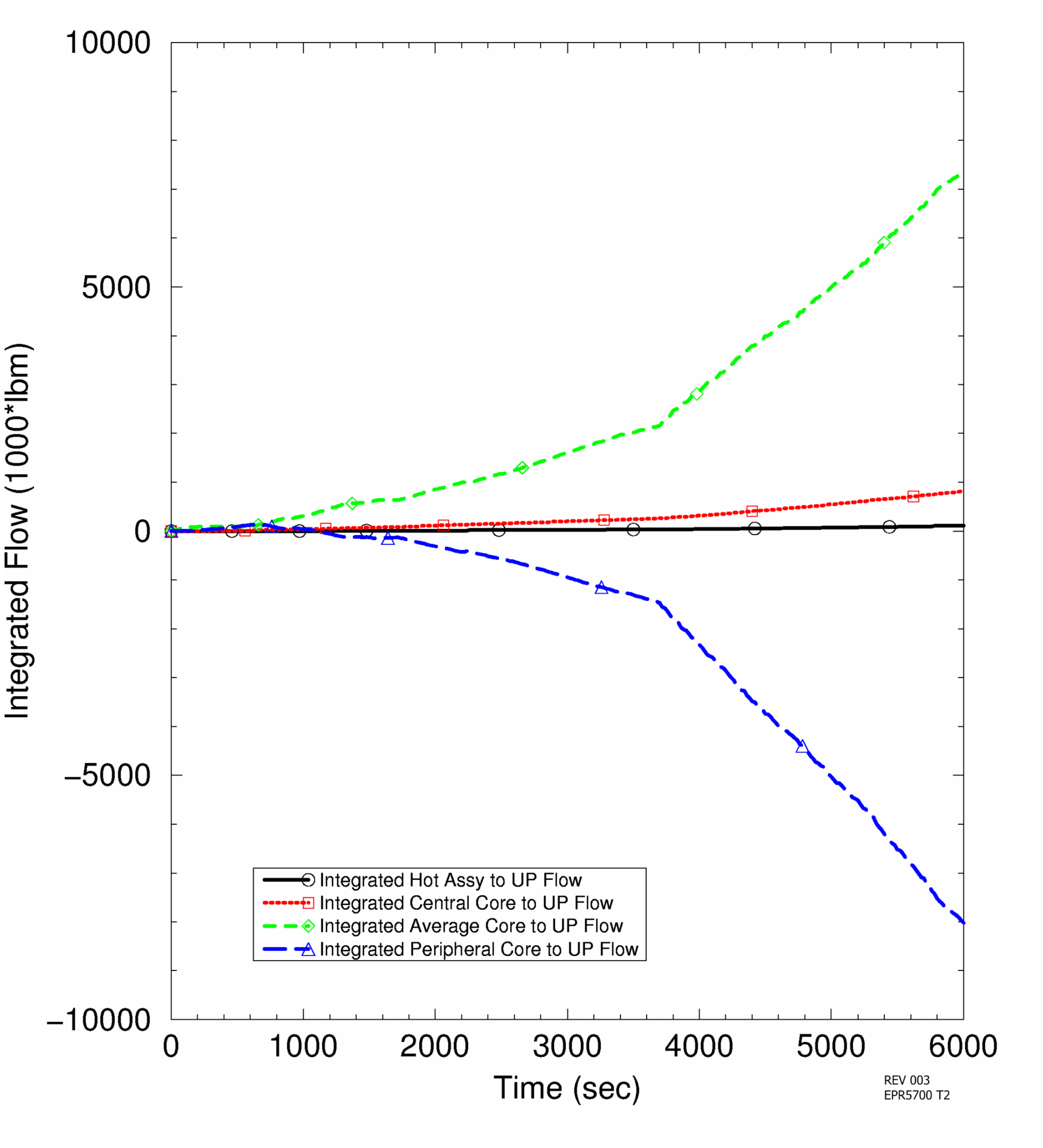

Figure 15.6-90—LBLOCA with Hot Leg Injection at 60 Minutes - Integrated Flow from Lower Plenum to Core Regions

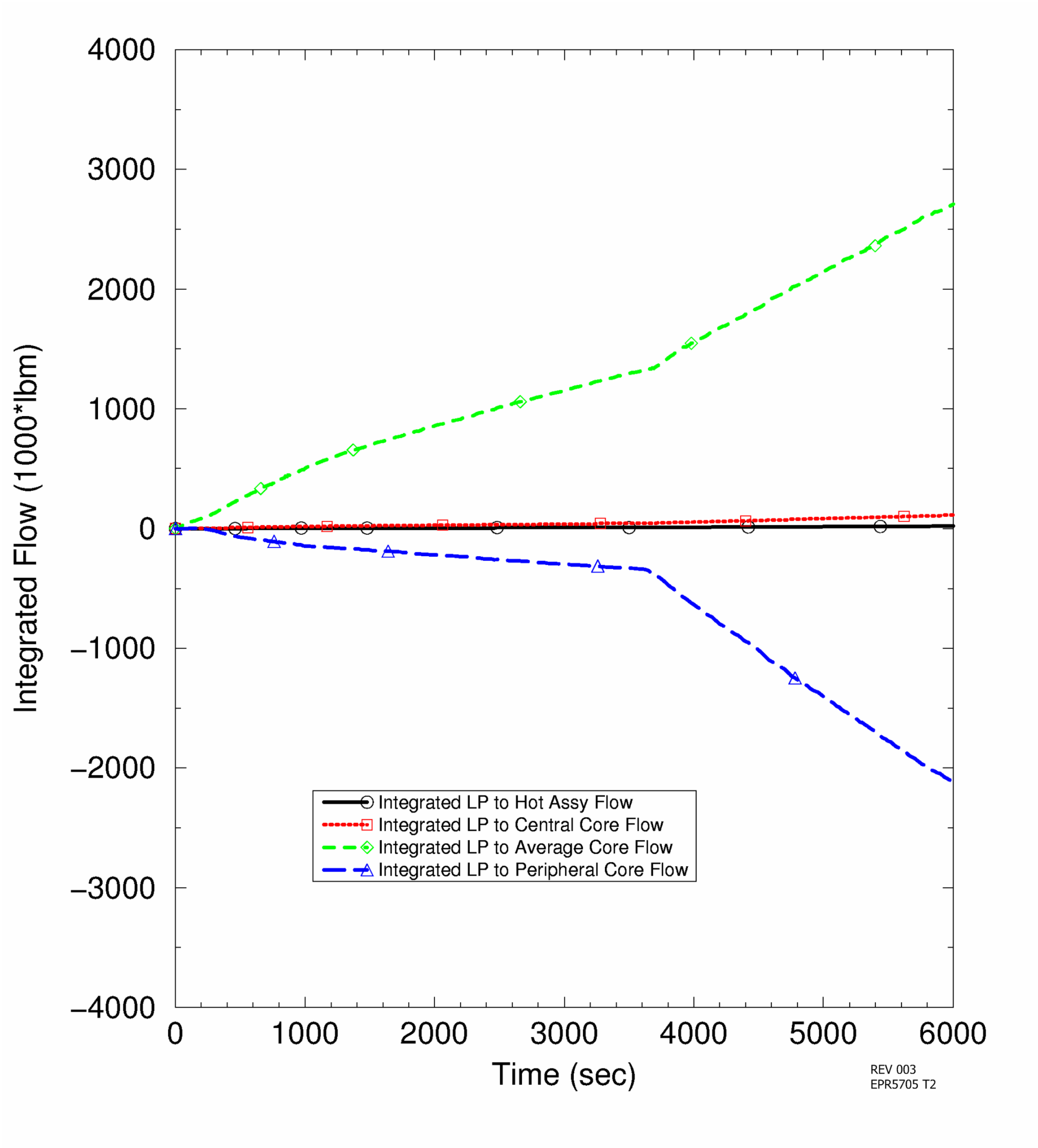

Figure 15.6-91—LBLOCA with Hot Leg Injection at 60 Minutes - Integrated Flow from Lower Head to Lower Plenum

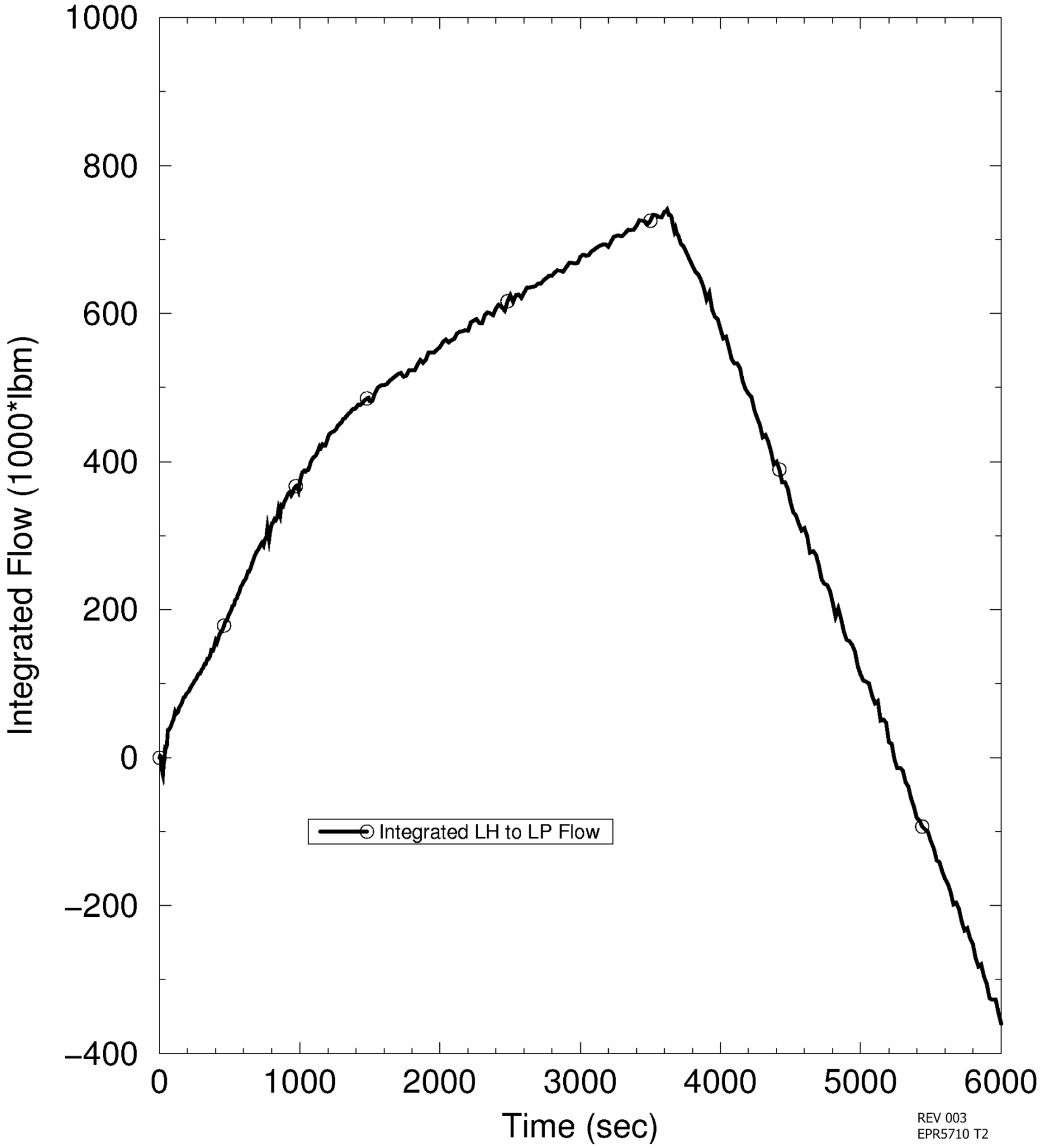

Figure 15.6-92—Time Dependent Boron Concentration During the Pool Boiling Period with and without Hot Leg Injection at 60 Minutes

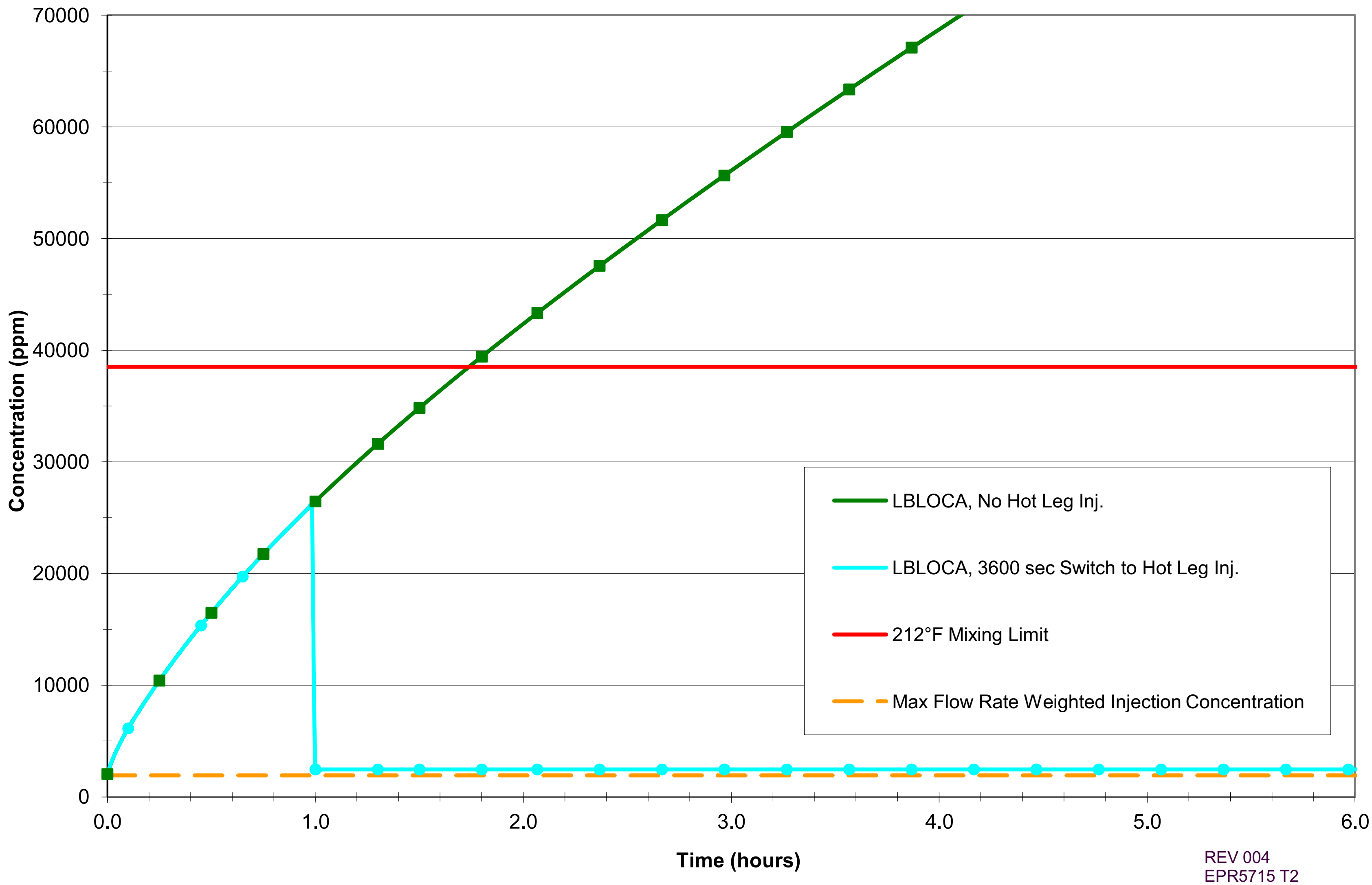

Figure 15.6-93—IOPSRV Event- Representative Plot of Normalized Minimum DNBR and Maximum LPD Normalized to the SAFDL