

15.2 Decrease in Heat Removal by the Secondary System

Several anticipated operational occurrences (AOO) and one postulated accident (PA) result in an unplanned decrease in heat removal by the secondary system. These events are described in the following sections:

- Section 15.2.1 - Loss of external load (LOEL).
- Section 15.2.2 - Turbine trip (TT).
- Section 15.2.3 - Loss of condenser vacuum (LOCV).
- Section 15.2.4 - Inadvertent main steam isolation valve closure (MSIVC).
- Section 15.2.5 - Steam pressure regulator failure, boiling water reactors (BWR) only.
- Section 15.2.6 - Loss of non-emergency AC power to the station auxiliaries (LNEP).
- Section 15.2.7 - Loss of normal feedwater flow (LNFF).
- Section 15.2.8 - Feedwater line breaks (FWLB inside and outside containment).

Section 15.0.0.1 describes the classification of these accident and transient events, and Section 15.0.0.2 describes the corresponding acceptance criteria. With the exception of the FWLB, the events listed above are AOOs. The FWLB is classified as a PA.

15.2.1 Loss of External Load

15.2.1.1 Identification of Causes and Event Description

The LOEL event is an AOO initiated by an electrical disturbance that causes the loss of a significant portion of the turbine generator load. In an LOEL, the fast closure of the turbine control valves (TCV) causes a decrease in heat removal by the secondary system. The LOEL may also involve a temporary increase in reactor coolant pump (RCP) speed and flow due to the temporary increase in turbine generator speed following the LOEL.

A TT generates a non-safety-related partial trip (PT) that inserts a limited number of control rods to avoid an unnecessary reactor trip (RT). The non-safety-related turbine bypass system operates to control steam generator (SG) pressure. The non-safety-related pressurizer (PZR) spray operates to control reactor coolant system (RCS) pressure. These non-safety-related functions are not credited in the safety analysis of this event. As a result, the safety-related main steam relief trains (MSRT) and main steam safety valve (MSSV) operate to control SG pressure and the safety-related PZR safety relief valves (PSRVs) open to control RCS pressure.

Following TCV closure, secondary side pressure and temperature increase significantly as thermal energy from the RCS continues to be transferred to the SGs. Increasing secondary system temperature causes a corresponding increase in RCS temperature. The increase in reactor coolant temperature causes an expansion of the reactor coolant inventory into the PZR and an increase in RCS pressure. RT occurs on a high PZR pressure protection system (PS) signal.

If loss of offsite power (LOOP) occurs with RT, all main feedwater (MFW) pumps are de-energized and the RCPs coast down. Emergency feedwater (EFW) is activated on wide range (WR) low SG level following a LOOP to maintain SG levels and the secondary heat sink.

A controlled state is reached following RT, with the plant in a hot shutdown condition with residual heat removed by either forced circulation (no LOOP) or natural circulation (LOOP) of primary coolant from the core to the SGs and secondary side heat removal via the MSRTs and MFW or EFW.

The LOEL event potentially challenges three acceptance criteria:

- The RCS overpressurization limit; requires that the RCS pressure be maintained below 110 percent of the RCS design pressure.
- The secondary side overpressurization limit; requires that the secondary side pressure be maintained below 110 percent of the main steam system (MSS) design pressure.
- Specified acceptable fuel design limits (SAFDL), specifically, departure from nucleate boiling (DNB) and fuel centerline melt (FCM).

15.2.1.2 Method of Analysis and Assumptions

The severity of the LOEL event is determined by the closure time of the TCVs, with a shorter closure time corresponding to more severe conditions. A conservatively small (0.1 s) value is assumed for the closure of the turbine stop valve (TSV) for the TT event (see Section 15.2.2). This assumed value bounds the fast closure time for the TCVs; therefore, the results of the TT event bound those for the LOEL for RCS overpressure.

For secondary overpressure, the LOEL event is bounded by the single MSIVC event discussed in Section 15.2.4. The MSIVC event results in a smaller isolated secondary side volume; therefore, the peak secondary pressure is higher for the single MSIVC event than for LOEL.

15.2.1.3 Conclusions

The consequences of the LOEL event are bounded by the TT event presented in Section 15.2.2 and the single MSIVC event presented in Section 15.2.4.

15.2.2 Turbine Trip

15.2.2.1 Identification of Causes and Event Description

The event description of the TT event is the same as the LOEL event (see Section 15.2.1.1), except that the TT event is initiated by the closure of the TSV. Because of the fast closure of the TSV, there is no increase in RCP speed and flow due to the temporary increase in turbine generator speed following the LOEL.

The TT event potentially challenges three acceptance criteria:

- The RCS overpressurization limit; requires that the RCS pressure be maintained below 110 percent of the RCS design pressure.
- The secondary side overpressurization limit; requires that the secondary side pressure be maintained below 110 percent of the MSS design pressure.
- SAFDLs, specifically, DNB and FCM.

The TT event does not challenge SAFDLs because RCS pressure increases and there is little change in core power. The single MSIVC event (Section 15.2.4) is limiting for secondary system overpressure because of the smaller available steam line volume.

15.2.2.2 Method of Analysis and Assumptions

The S-RELAP5 computer code is used to calculate the transient thermal and hydraulic response of the primary and secondary systems in accordance with the methodology described in the Codes and Methods Applicability Report for the U.S. EPR (Reference 1). The computer code simulates the necessary components and contains the features required to model this event.

TT is the limiting overpressure event for the RCS. Therefore, initial conditions and setpoints for analyzing this event are biased conservatively to maximize peak RCS pressure. The least negative doppler temperature coefficient with an additional 10 percent bias is applied to minimize the negative feedback due to any power increase. The most positive hot full power (HFP) moderator temperature coefficient is applied to minimize the negative feedback due to any RCS temperature increase. PS trip setpoint biases are set to values that provide a maximum delay for RT and safety system initiation. It is assumed conservatively that the malfunction of the turbine or reactor system that initiated the TT does not cause an RT by affecting PS-sensed equipment or instrumentation (e.g., RCP speed or engineered safety features).

The TT event is initiated by rapidly closing the TSVs in 0.1 second. Assumption of LOOP is conservative for RCS overpressurization because it reduces the heat transfer rate from the primary to secondary systems. LOOP is assumed conservatively to occur with RT because it is the earliest time to initiate the coastdown of the RCPs without

causing an earlier RT. The MFW pumps are de-energized at the time of LOOP. To maximize the RCS pressure response, the non-safety-related PZR spray, turbine bypass system, and PT (on TT) are not credited.

Five percent tube plugging is assumed to minimize the heat transfer rate from the RCS to the secondary system.

The limiting single failure for this event is that which would maximize the RCS overpressure response. The peak RCS pressure occurs within 10 seconds of the initiation of the event, which precludes any postulated operator action that would exacerbate the event. The PZR PSRVs are single failure proof; therefore, the limiting equipment single failure is assumed to be the failure of a single train of MSRTs.

Table 15.2-1—Turbine Trip - Key Input Parameters presents a listing of the key initial inputs. Table 15.2-2—Turbine Trip - Key Equipment Status presents a listing of the status of key systems.

15.2.2.3 Results

Table 15.2-3—Turbine Trip-RCS Overpressurization - Sequence of Events presents the sequence of events for the TT analysis. Figure 15.2-1—Turbine Trip RCS Overpressurization - Peak RCS Pressure through Figure 15.2-13—Turbine Trip RCS Overpressurization - MFW Flows show the response of key system parameters.

The event is initiated with the closure of the TSVs in 0.1 second. The resulting pressurization of the secondary side causes a reduction in primary-to-secondary heat transfer and corresponding increase in RCS temperatures. The expansion of the RCS coolant causes an insurge into the PZR. As the steam space is compressed, RCS pressure increases until an RT is generated on high PZR pressure at 6.98 seconds. LOOP is assumed with RT, and the RCPs begin to coast down. The PSRVs open at 9.48 seconds to control pressure. The opening setpoints and capacity of the PSRVs are sufficient to limit the maximum RCS pressure to 2785.2 psia, which is reached at 9.8 seconds. The maximum PZR level peaks at an indicated 72 percent.

The automatic actuation of EFW and SG steam relief via the MSRTs provide adequate cooling after RT to remove decay heat. The PSRVs reseal. As the RCPs coast down, natural circulation conditions are established and a stable controlled state is reached.

15.2.2.4 Radiological Consequences

The radiological consequences of this event are bounded by the inadvertent opening of an MSSV event described in Section 15.1.4.

15.2.2.5 Conclusions

- The analysis demonstrates that the maximum RCS pressure (2785.2 psia) remains below 110 percent of the RCS design pressure (2803.2 psia), thereby satisfying overpressure acceptance criteria for the RCS. This is the limiting overpressure event for peak RCS pressure.
- The single MSIV closure event bounds TT as the limiting overpressure event for the secondary system (see Section 15.2.4).
- The TT event does not challenge SAFDLs because RCS pressure increases and there is little change in core power; therefore, the acceptance criteria are met for TT.

15.2.2.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.2.2 events included in NUREG-0800, Section 15.2.1–15.2.5, (Reference 2) and descriptions of how these criteria are met are listed below:

1. The basic objectives of the review of the initiating events listed in Subsection I of this SRP section are met as follows:
 - A. To identify which moderate-frequency event that results in an unplanned decrease in secondary system heat removal is the most limiting, in particular as to primary pressure, secondary pressure, and long-term decay heat removal.
 - Response: The TT event is identified as the most limiting event with respect to primary pressure. The single MSIVC event (Section 15.2.4) is the most limiting event with respect to secondary pressure and minimum departure from nucleate boiling ratio (MDNBR). The loss of normal feedwater flow event (Section 15.2.7) is the most limiting AOO with regard to long-term decay heat removal.
 - B. To verify whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.
 - Response: The TT event satisfies the acceptance criteria for RCS pressure. The single MSIVC event (Section 15.2.4) is limiting with respect to secondary pressure and MDNBR.
 - C. To verify whether the plant PS setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies as delineated in RG 1.105.
 - Response: Measurement and setpoint uncertainties are conservatively applied in the TT analysis.

D. To verify whether the event evaluation considers single failures, operator errors, and performance of non-safety-related systems consistent with the RG 1.206 regulatory guidelines.

- Response: The limiting single failure of a safety-related system (failure of a single MSRT train to open) is included in the TT analysis. The period of concern for the TT event (time of peak RCS pressure) occurs within the first ten seconds of the event, with the consequences of the event essentially being terminated by the opening of the PZR safety relief valves. Therefore, the operation of non-safety-related systems and operator error does not affect the consequences of the TT event.

2. With the ANS standards as guidance, specific criteria meet the relevant requirements of GDCs 10, 13, 15, 17, and 26 for events of moderate frequency, as follows:

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

- Response: The RCS pressure remains below 110 percent of the design value for the TT event. The single MSIVC event (Section 15.2.4) is bounding for secondary pressure.

B. Fuel-cladding integrity is maintained by keeping the MDNBR above the 95/95 DNBR limit for pressurized water reactors (PWR) and the critical power ratio (CPR) remaining above the minimum CPR safety limit for BWRs based on acceptable correlations (refer to Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design.

- Response: The TT event does not challenge SAFDLs because RCS pressure increases and there is little change in core power; therefore, the acceptance criteria are met for TT.

C. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.

- Response: The TT analysis demonstrates that the operable safety-related systems are capable of mitigating the consequences without generating an aggravated condition.

D. The requirements in RG 1.105, "Instrument Spans and Setpoints," are used for their impact on the plant response to the type of AOOs addressed in this SRP section.

- Response: Reference 1 describes how the methodology biases input values to account for uncertainties in spans and setpoints to achieve a conservative result for the event being analyzed.

- E. The most limiting plant system single failure, as defined in “Definitions and Explanations,” 10 CFR 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.
- Response: The limiting single failure of a safety-related system (failure of a single MSRT train to open) is assumed for the TT event analysis.
- F. Performance of non-safety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of 15.2.1-15.2.5-6 Revision 2 - March 2007, check valves in passive systems) must be evaluated and verified according to the guidance of SECY 77-439 as cited in Reference 2, SECY 94-084 as cited in Reference 2, and RG 1.206.
- Response: Non-safety systems are modeled when they make the consequences of the event more severe. There are no non-safety-related systems whose operation makes the consequences of this event more severe.
3. The applicant should analyze events using an acceptable analytical model. Any other analytical method proposed by the applicant is evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by the appropriate organization for reactor systems. The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable, and are used as described below:
- A. The reactor is initially at 102 percent of the rated (licensed) core thermal power to account for a two percent power measurement uncertainty unless a lower number can be justified through measurement uncertainty methodology and evaluation or unless the uncertainty otherwise is accounted for (refer to Section 4.4), and primary loop flow is at the nominal design flow less the flow measurement uncertainty.
- Response: The TT analysis is performed at rated thermal power plus measurement uncertainty. Thermal design flow is used, which is less than the expected nominal design flow minus measurement uncertainty.
- B. Conservative scram characteristics are assumed (i.e., for a PWR maximum time delay with the most reactive rod held out of the core, for a BWR a 0.8 design conservatism multiplier on the predicted reactivity insertion rate) unless a different conservatism factor can be justified through the uncertainty methodology and evaluation or the uncertainty is otherwise accounted for (refer to Section 4.4).
- Response: A conservative scram curve and scram timing delay are used for the TT analysis. The scram curve assumes the most reactive rod is stuck out of the core.
- C. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

- Response: Conservative values based on an array of proposed fuel cycle operating methods are used for the moderator temperature coefficient and Doppler temperature coefficients.
- D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105.
- Response: Instrument and measurement uncertainties are conservatively applied for the TT analysis.

15.2.3 Loss of Condenser Vacuum

The LOCV is a fault that can cause a TT. The LOCV also isolates the turbine bypass system to prevent overpressurization of the condenser and the MFW pumps trip due to low suction pressure. The consequences of the LOCV event are bounded by those of the TT event. The TT event bounds the LOCV event because the TT event analysis does not credit the non-safety-related turbine bypass system, and a conservatively small TSV closure time is assumed for the TT event. In addition, for TT analysis, the MFW pumps are lost due to the assumption of LOOP.

15.2.3.1 Radiological Consequences

The LOCV event is bounded by the inadvertent opening of an MSSV event (Section 15.1.4) for radiological consequences.

15.2.4 Inadvertent Main Steam Isolation Valve Closure

15.2.4.1 Identification of Causes and Event Description

The MSIVC event is an AOO initiated by steam line or reactor system malfunctions and operator actions. The U.S. EPR instrumentation and control system is designed so that no single random failure, spurious signal, or operator action can cause the inadvertent closure of more than one MSIV. Therefore, only the single MSIVC event is credible and is considered.

The single MSIVC event is characterized by a decrease in heat removal by the secondary system caused by the postulated abrupt closure of one MSIV. The closure of one MSIV isolates steam flow from its associated SG. This condition causes pressure and temperature to increase sharply in the isolated SG, the main steam line, and, consequently, in the affected RCS loop. Pressure in the affected SG and MS line is limited automatically by safety-related steam relief devices (MSRT and MSSVs) located upstream of the MSIV. Steam flow from the remaining SGs increases because of turbine demand, which causes a depressurization of the unaffected SGs. As steam line header pressure begins to drop, the non-safety-related TCV opens in an attempt to maintain turbine pressure.

RT and TT are initiated by a PS signal generated on high SG pressure in the affected SG. A non-safety function trips all but one of the MFW pumps on RT, with one remaining in service to supply MFW through the low-load lines. If the non-safety-related turbine bypass system is available, it controls pressure in the unaffected SGs after RT and TT. Otherwise, the safety-related MSRTs and MSSVs control SG pressure as in the affected SG.

The increase in primary pressure, secondary pressure and primary coolant temperature during the MSIVC event potentially challenges three acceptance criteria:

- The RCS overpressurization limit; requires that the RCS pressure be maintained below 110 percent of the RCS design pressure.
- The secondary side overpressurization limit; requires that the secondary side pressure be maintained below 110 percent of the MSS design pressure.
- The MDNBR SAFDL and FCM.

The TT event is limiting with respect to peak RCS pressure (see Section 15.2.2).

15.2.4.2 Method of Analysis and Assumptions

The methodology for this event uses the S-RELAP5 computer code to simulate the responses to the event of the primary and secondary coolant systems, reactor, protective equipment and systems, and automatic controllers. The transient analysis is performed using the methodology described in Reference 1. Section 15.0.2 provides a description of the S-RELAP5 analysis method.

The algorithm, described in Section 15.0.0.3.9, is simulated to predict RT and adequacy of the dynamic compensation of the incore monitoring system in a manner consistent with Reference 3. Additionally, the core thermal-hydraulic computer code LYNXT is used to deterministically calculate the MDNBR using the RCS response from S-RELAP5 as a boundary condition.

The single MSIVC event is the limiting overpressure event for the secondary system. Therefore, initial conditions and setpoints for analyzing this event are biased conservatively to maximize peak secondary pressure. PS trip setpoint biases are set to values that provide a maximum delay for RT and safety system initiation. Similarly, for the single MSIVC MDNBR case, appropriate setpoint biases are employed to maximize the consequences.

The MSIVC event is initiated by closing one of the MSIVs in a conservatively short time of 0.1 second. The event is bounded by TT for the RCS overpressure criterion because three SGs remain available to remove energy from the primary system. The event is analyzed in separate cases to evaluate overpressure in the affected SG and

compliance with SAFDLs, particularly DNB. To obtain a conservative response for these cases, the analysis does not credit the non-safety-related turbine bypass system.

The limiting single failure for this event is that which would maximize the secondary overpressure and MDNBR responses. The peak secondary pressure occurs at approximately 12 seconds after the initiation of the event, which precludes any postulated operator action that would exacerbate the event. The limiting equipment single failure for the peak secondary pressure and MDNBR cases is assumed to be the failure of the MSRT on the steamline of SG-1. In addition, the lowest setpoint MSSV on this steamline is assumed to be out-of-service.

The peak secondary pressure case is biased to maximize the heat transfer from the RCS to the secondary system. This heat transfer is maximized by assuming zero percent SG tube plugging and by assuming that a LOOP does not occur. In addition, the non-safety-related PZR spray is simulated to reduce the potential for RT on high PZR pressure.

The MDNBR case is biased to minimize the heat transfer from the RCS to the secondary system. This heat transfer is minimized by assuming five percent SG tube plugging and by assuming a LOOP at the time of TT. In addition, the non-safety-related PZR spray is simulated to reduce the increase in RCS pressure, which acts to reduce DNB margin. The analysis conservatively accounts for the effect of asymmetric core inlet coolant temperatures on core reactivity and power distributions.

Table 15.2-4—MSIVC Secondary Overpressurization - Key Input Parameters presents a listing of the key initial inputs used in this analysis. Table 15.2-5—MSIVC Overpressurization - Key Equipment Status presents a listing of the status of key systems.

15.2.4.3 Results

15.2.4.3.1 Peak Secondary Pressure Analysis Results

Table 15.2-6—MSIVC Secondary Overpressurization - Sequence of Events presents the sequence of events for the MSIVC maximum secondary pressure analysis. The MSIV in one loop is assumed to close in 0.1 second. The loss of steam flow from the affected SG causes an increase in demand on the unaffected SGs (Figure 15.2-14—MSIVC Secondary Overpressurization—MSIV Flow Rates). The affected SG pressurizes while the unaffected SGs depressurize (Figure 15.2-23—MSIVC Secondary Overpressurization — Maximum Secondary Pressures).

Pressure in the affected SG reaches the high SG pressure RT setpoint at 6.32 seconds initiating RT. This PS signal also opens the MSRIV in the affected SG, but the main steam relief control valve (MSRCV) is assumed to be failed in the closed position (although normally full open during hot full power operation). Steam flows through

the MSRV just long enough to pressurize the relief train piping ahead of the closed MSRCV. This result is shown in Figure 15.2-15—MSIVC Secondary Overpressurization - Safety Valve Flows for the Affected Loop.

The depressurization of the three unaffected SGs causes an increase in the primary to secondary heat transfer for those loops. This condition lowers the temperature of the fluid returning to the core from the unaffected loops (Figure 15.2-16—MSIVC Secondary Overpressurization—Cold Leg Temperatures). The cooler water returning to the core combined with a negative end of cycle moderator coefficient causes an increase in power (Figure 15.2-19—MSIVC Secondary Overpressurization — Reactor Power).

The pressure in the affected SG continues to increase until the first MSSV opens at 10.77 seconds (Figure 15.2-15). The opening setpoints and capacity of the MSSVs are adequate to limit peak secondary pressure (at the bottom of the SGs) to 1566.7 psia at 10.85 s, which is less than the acceptance criterion of 110 percent of the secondary system design pressure (1593.2 psia).

15.2.4.3.2 DNBR Analysis Results

The S-RELAP5 calculation indicates an increase in core power associated with the reduction in cold leg temperatures in the unaffected loops. This result is an artifact of use of these cold leg temperatures as input to the point kinetics model in S-RELAP5. During the same time period prior to RT, RCS pressure increases slightly. The net result is that this event does not produce core conditions that challenge the MDNBR criterion.

15.2.4.4 Radiological Consequences

There are no radiological consequences for this event because there are no fuel failures. It is bounded radiologically by the inadvertent opening of a MSSV event described in Section 15.1.4.

15.2.4.5 Conclusions

The peak RCS pressure never rises significantly because the three unaffected SGs continue to cool the primary system.

- The peak secondary system pressure (1566.7 psia) is below 110 percent of the secondary system design pressure (1593.2 psia). Therefore, the secondary overpressurization acceptance criterion for the single MSIVC event is met.
- The peak RCS pressure is bounded by TT (see Section 15.2.2).
- The SAFDL limits are not exceeded for the single MSIVC event.

15.2.4.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.2.4 events included in NUREG-0800, Section 15.2.1–15.2.5, (Reference 2) and descriptions of how these criteria are met are listed below:

1. The basic objectives of the review of the initiating events listed in Subsection I of this SRP section are as follows:
 - A. To identify which moderate-frequency event that results in an unplanned decrease in secondary system heat removal is the most limiting, in particular as to primary pressure, secondary pressure, and long-term decay heat removal.
 - Response: The TT event (Section 15.2.2) is identified as the most limiting event with respect to primary pressure. The single MSIVC event (Section 15.2.4) is the most limiting event with respect to secondary pressure and MDNBR. The loss of normal feedwater flow event (Section 15.2.7) is most limiting AOO with regard to long-term decay heat removal.
 - B. To verify whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.
 - Response: The TT event satisfies the acceptance criteria for RCS pressure. The single MSIVC event (Section 15.2.4) is limiting with respect to secondary pressure and MDNBR.
 - C. To verify whether the plant PSs setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies as delineated in RG 1.105.
 - Response: Measurement and setpoint uncertainties are conservatively applied in the single MSIVC analysis.
 - D. To verify whether the event evaluation considers single failures, operator errors, and performance of non-safety-related systems consistent with the RG 1.206 regulatory guidelines.
 - Response: The most severe single failure of a safety system (failure of a single MSRT train to open) is assumed for the analysis of the single MSIVC event. PZR sprays are simulated because they are a non-safety-related system that makes the consequences of the single MSIVC event more severe.
2. With the ANS standards as guidance, specific criteria meet the relevant requirements of GDC 10, GDC 13, GDC 15, GDC 17, and GDC 26 for events of moderate frequency as follows:
 - A. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values; refer to Reference 15 cited in NUREG-

0800 (Reference 2).

- Response: For the single MSIVC event, pressure on the primary side does not approach the design limit. The maximum calculated pressure on the secondary side is lower than the 110 percent of the design value.
- B. Fuel-cladding integrity is maintained by keeping the MDNBR above the 95/95 DNBR limit for PWRs and the CPR remaining above the minimum CPR safety limit for BWRs based on acceptable correlations (refer to Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design.
- Response: The single MSIVC event does not challenge DNBR or linear heat generation limits.
- C. An incident of moderate frequency should not generate an aggravated plant condition without other faults occurring independently.
- Response: The single MSIVC analysis demonstrates that the operable safety-related systems are capable of mitigating the consequences without generating an aggravated condition.
- D. The requirements in RG 1.105, "Instrument Spans and Setpoints," are used for their impact on the plant response to the type of AOOs addressed in this SRP section.
- Response: Reference 1 describes how the methodology biases input values to account for uncertainties in spans and setpoints to achieve a conservative result for the event being analyzed.
- E. The most limiting plant system single failure, as defined in "Definitions and Explanations," 10 CFR 50, Appendix A, must be assumed in the analysis according to the guidance of RG 1.53 and GDC 17.
- Response: The limiting single failure of a safety-related system (failure of a single MSRT train to open) is assumed for the single MSIVC event analysis.
- F. Performance of non-safety-related systems during transients and accidents and single failures of active and passive systems (especially as to the performance of check valves in passive systems) must be evaluated and verified according to the guidance of SECY 77-439 as cited in Reference 2, SECY 94-084 as cited in Reference 2, and RG 1.206.
- Response: Non-safety systems are modeled when they make the consequences of the event more severe. PZR sprays are simulated because they are a non-safety-related system that makes the consequences of the single MSIVC event more severe.
3. The applicant should analyze these events using an acceptable analytical model. Any other analytical method proposed by the applicant is evaluated by the staff for acceptability. For new generic methods, the reviewer requests an evaluation by

the appropriate organization for reactor systems. The values of the parameters in the analytical model should be suitably conservative. The following values are acceptable:

- A. The reactor is initially at 102 percent of the rated (licensed) core thermal power to account for a two percent power measurement uncertainty unless a lower number can be justified through measurement uncertainty methodology and evaluation or unless the uncertainty otherwise is accounted for (refer to Section 4.4), and primary loop flow is at the nominal design flow less the flow measurement uncertainty.
 - Response: The TT analysis is performed at rated thermal power plus measurement uncertainty. Thermal design flow is used, which is less than the expected nominal design flow minus measurement uncertainty.

- B. Conservative scram characteristics are assumed (i.e., for a PWR maximum time delay with the most reactive rod held out of the core, for a BWR a 0.8 design conservatism multiplier on the predicted reactivity insertion rate) unless a different conservatism factor can be justified through the uncertainty methodology and evaluation or the uncertainty is otherwise accounted for (refer to Section 4.4).
 - Response: A conservative scram curve and scram timing delay are used for the TT analysis. The scram curve assumes the most reactive rod is stuck out of the core.

- C. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.
 - Response: Conservative values based on an array of proposed fuel cycle operating methods are used for the moderator temperature coefficient and Doppler temperature coefficients.

- D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105.
 - Response: Instrument and measurement uncertainties are conservatively applied for the single MSIVC analysis.

15.2.5 Steam Pressure Regulator Failure

The steam pressure regulator failure applies to BWR plants and is not applicable to the U.S. EPR.

15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries

The LNEP to the station auxiliaries is initiated by a complete loss of either the external (offsite) grid or the onsite AC distribution system. The complete loss of coolant flow

event described in Section 15.3.2 is analyzed assuming a LOOP or LNEP as the initiating event. Refer to Section 15.3.2 for the short-term response of the LNEP event with respect to SAFDLs. The subsequent evolution of the event after RT is similar to and bounded by the LNFF scenario with a LOOP, which causes a complete loss of RCS flow; see Section 15.2.7 for the long-term plant response to an LNEP event.

The LNEP event potentially challenges three acceptance criteria:

- The RCS overpressurization limit; requires that the RCS pressure be maintained below 110 percent of the RCS design pressure.
- The secondary side overpressurization limit; requires that the secondary side pressure be maintained below 110 percent of the MSS design pressure.
- SAFDL, specifically, DNB and FCM.

The TT event (Section 15.2.2) and MSIVC event (Section 15.2.4) bound the LNEP event for RCS and secondary side overpressurization respectively. The complete loss of coolant flow event (Section 15.3.2) is the same as LNEP with respect to SAFDL.

Following initiation of the LNEP event, the diesel generators are started and provide electric power to vital loads. The sensible and decay heat loads are handled by the steam relief and EFW systems. Different segments of this event have similarities to the complete LNFF and complete loss of RCS flow events.

The loss of power results in immediate RCP coastdown and MFW termination. A RT occurs on low pump speed, if not sooner, as a result of the loss of power. Decaying reactor coolant flow causes an immediate increase in core coolant temperatures. Also, the sudden loss of subcooled MFW flow, the decaying reactor coolant flow, and termination of steam flow to the turbine all cause the SG heat removal rates to decrease. The decrease in SG heat removal rates augments the increase in reactor coolant temperatures.

As reactor coolant temperatures rise, the reactor coolant expands and surges into the PZR, potentially overflowing it. The resulting increase in RCS pressure causes the PZR PSRVs to open prior to termination of the short-term-heatup phase of the event that is terminated by reactor scram.

The termination of steam flow to the turbine and the continuing primary-to-secondary transfer of the decaying core power cause SG pressures to rapidly increase. The SGs are protected from overpressurization by opening of the MSRT and, if necessary, by the MSSVs. The SG pressures stabilize with pressure controlled at the MSRT opening setpoint, with the steam release providing cooling of the RCS.

Liquid levels in the SGs soon decrease to the low-low setpoint, which actuates the EFW system and isolates the blowdown lines. EFW actuation and blowdown isolation

occur for each SG independently. That is, when the low-low setpoint is reached in one SG, the EFW is actuated for only that SG, and only its blowdown line is isolated. When delivery of EFW begins, the rate of level decrease slows in the SGs fed by EFW, and the EFW provides additional cooling of the SGs and RCS. EFW supply to two of the SGs will be lost, one due to single failure and a second due to preventative maintenance.

In the long term, the two SGs that are not fed by EFW will dry out, thereby reducing the energy removal rate from the RCS. At 30 minutes into the transient, operator action can be credited for cross-connection of the EFW so that all four SGs are fed. Eventually, the decay heat level decreases below the level of heat removal via the MSRTs, marking the end of the challenge to the event acceptance criteria.

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 Identification of Causes and Event Description

The LNFF is an AOO in which there is a postulated complete termination of MFW. This condition can be caused by a LOOP or a malfunction in the MFW control system or equipment.

The sudden loss of subcooled MFW flow while the plant continues to operate at power causes SG heat removal rates to decrease, which causes reactor coolant temperatures to increase, thus expanding the RCS fluid. RCS fluid flows into the PZR, thereby increasing pressure, actuating the PZR spray system if available and potentially causing the PSRVs to open to control pressure. SG liquid levels drop steadily following termination of the MFW flow, quickly reaching the low SG narrow range (NR) level RT setpoint. This condition initiates RT and subsequent TT, thereby ending the short-term heatup phase of the event.

If available, the non-safety-related turbine bypass system opens to control secondary side pressure. If not available, SG pressure increases until the safety-related MSRTs open. SG levels continue to drop and soon reach the low SG level EFW setpoint that actuates EFW and isolates the SG blowdown line. When the delivery of EFW begins, the rate of level decrease in the fed SGs slows. For the SGs receiving EFW, liquid levels stabilize and begin to rise. If trains of EFW are unavailable due to maintenance or single failure, the operator can redirect the available EFW to feed the four SGs. The plant transitions to a stable controlled state.

The LNFF event is classified as an AOO. The principal acceptance criteria that apply to this event are listed below:

- DNB SAFDL. The DNB acceptance criterion requires that minimum DNB ratio (MDNBR) is not less than the 95/95 correlation limit.

- The DNB SAFDL is not challenged during the short-term-heatup phase of the LNFF event. That is because the reactor power does not increase, and the RCS pressure increases.
- The DNB SAFDL is not challenged during the long-term-heatup phases of the LNFF event if energy removal by the secondary system through the MSRTs are sufficient to remove decay heat plus RCP heat. Thus, the DNB SAFDL is satisfied for the LNFF event if primary coolant subcooling margin is maintained.
- Fuel-melt SAFDL. The fuel-melt acceptance criterion requires that none of the fuel rods in the core experience centerline melt. The fuel-melt SAFDL is not challenged during the LNFF event because the reactor power does not rise above the initial power level.
- Pressure limit. The pressure acceptance criterion requires that the pressures in the reactor coolant and main steam systems are maintained below 110 percent of their respective system design pressures. RCS and secondary pressure do not rise significantly until TT occurs on RT with delay. Because TT occurs after RT, the capacity of the MSRTs is adequate to prevent opening of the MSSVs. RCS pressure does not increase to the PSRV opening setpoint. Therefore, peak RCS and secondary pressures for the LNFF event are bounded by the TT and MSIVC events, respectively.
- PZR overfill. PZR overfill is evaluated to demonstrate that the loss of normal feedwater event meets the criteria that an AOO does not result in a postulated event without other faults occurring independently. The PSRVs are designed to relieve water. The analysis predicts that there is two-phase release through two openings of the PSRVs at the end of the event. Both releases have a duration of four seconds.
- Decay heat removal. There is sufficient capacity for long-term decay heat removal for the plant to reach a stabilized condition. This case, along with the analysis presented for FWLB, demonstrates the adequacy of the design capacity of the EFW system.

15.2.7.2 Method of Analysis and Assumptions

The methodology for this event uses the S-RELAP5 computer code to simulate the responses to the event of the primary and secondary coolant systems, reactor, protective equipment and systems, and automatic controllers. The transient analysis is performed using the methodology described in Reference 1. Section 15.0.2 provides a more detailed description of the S-RELAP5 analysis method.

To determine the limiting event in regard to each of the criteria above, a spectrum of loss of feedwater events were analyzed. The spectrum considered the range of initial conditions specified in Table 15.0.5, including the time in cycle, with or without LOOP, with or without pressure control, with 0 percent or 5 percent SG tube plugging, and with EOC coastdown.

The complete loss of feedwater flow event is a loss of the secondary heat sink and results in a gradual heatup of the RCS. In regard to all acceptance criteria, full power represents the limiting initial power because it provides the largest mismatch between the power generated and the heat removal capacity. Likewise, the results become more limiting with the lower EFW capacity. For this reason, the limiting single failure for all cases is the failure of an EFW train because it limits the heat removal capacity of the EFW system. In addition, it is assumed that one EFW pump is not available due to maintenance.

Because the reactor power does not increase above the initial power and the RCS pressure gradually increases during this event, the DNB and FCM SAFDLs are not challenged. Sensitivity analyses were performed to demonstrate that the peak RCS and MSS pressure for the loss of normal feedwater event do not challenge limits and are bounded by the TT and MSIV closure events, respectively. Thus, the analysis of the loss of feedwater event focuses on the criteria of PZR overfill and long-term decay heat removal.

For the PZR overfill case, the limiting conditions resulted from a low initial T_{avg} corresponding to the EOC coastdown with LOOP. Challenge to this criteria occurs well after RT. The analysis predicts that there is two-phase release through two openings of the PSRVs at the end of the event. Both releases have a duration of four seconds. For the long-term decay heat removal case, the limiting condition is from normal T_{avg} with offsite power available. This case is challenging to the EFW system and demonstrates that the capacity of the EFW system is adequate for long-term RCP pump heat and decay heat removal. Beginning of cycle reactivity parameters, biased to minimize feedback, are assumed for the long-term heat removal case because it corresponds to the least negative feedback. Since this case is a heatup transient, the least negative feedback maximizes core power. For the PZR overfill case, EOC reactivity parameters are assumed because this case is limiting for the lower initial T_{avg} corresponding to the EOC coastdown. The limiting event and initial conditions for these cases are summarized in Tables 15.0-62 and 15.0-63, respectively. The results for the long-term heat removal case is presented below.

Table 15.2-7—LNFF - Key Input Parameters presents a listing of the key initial inputs used. Table 15.2-8—LNFF - Key Equipment Status presents a listing of the status of key systems.

15.2.7.3 Results

The decay heat removal case is presented below. Table 15.2-9—LNFF - Sequence of Events presents the sequence of events for this event. Figure 15.2-27—Loss of Normal Feedwater - Reactor and Total SG Power through Figure 15.2-48—Loss of Normal Feedwater - Net Heat Addition to RCS present responses of several key parameters.

Upon loss of MFW, the temperature of the primary system increases and liquid expands into the PZR. Steam in the PZR is compressed, and the pressure control system responds by supplying normal spray. Water in the SGs boils off until the low SG NR level setpoint is reached, tripping the reactor and resulting in a TT. Power is maintained to the RCPs so that they continue to run.

Pressure in the SGs, which increases after the TT, reaches the MSRT setpoint. The MSRT is activated by opening the MSRIV so that the MSRCV can control secondary pressure. Because the MSRCVs are fully open during normal full power operation, pressure in the SGs decreases substantially. As pressure decreases below the MSRT opening setpoint, the MSRCVs close. After secondary pressure reaches the MSRT setpoint again, the MSRCVs open to modulate pressure, as shown in Figure 15.2-39—Loss of Normal Feedwater - SG Pressure.

Water in the SGs continues to boil-off until the low SG WR level setpoint in each individual SG is reached, which actuates EFW. EFW flow is supplied to only two SGs due to single failure and maintenance assumptions. The levels begin to rise in the two SGs receiving EFW flow, while the levels in the two SGs that are not fed by EFW continue to fall. Once EFW flow is provided to two SGs, sufficient heat removal capacity exists to remove decay heat and RCP heat, even though the two unfed SGs eventually dry out.

EFW continues to fill two SGs to their initial values. RCS temperatures, RCS pressures, and PZR level are stabilized and controlled. Secondary pressure is controlled by the MSRTs, and the plant reaches a stable controlled condition.

15.2.7.4 Radiological Consequences

This event is bounded by the inadvertent opening of an MSSV event (refer to Section 15.1.4) for radiological consequences.

15.2.7.5 Conclusions

The LNFF analysis demonstrates the following:

- The EFW system provides adequate decay heat removal as the energy removed by the secondary side equals or exceeds the energy added to the primary side and the plant reaches a stable controlled state.
- Pressures in the reactor coolant and main steam systems remain below 110 percent of their design pressures. Peak primary and secondary pressures for the LNFF event are bounded by the TT (Section 15.2.2) and single MSIVC events (Section 15.2.4), respectively.
- SAFDLs are not challenged for this event.

15.2.7.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.2.7 events included in NUREG-0800, Section 15.2.7, (Reference 2) and descriptions of how these criteria are met are listed below:

1. The basic objective in the review of the loss of normal feedwater transient is to confirm that the following criteria are met:
 - A. The plant responds to the loss of feedwater transient in such a way that the criteria regarding fuel damage and system pressure are met.
 - Response: The LNFF event does not significantly challenge the acceptance criteria for fuel damage and system pressures do not approach the design limits.
 - B. There is sufficient capacity for long-term decay heat removal for the plant to reach a stabilized condition.
 - Response: The analysis demonstrates that there is sufficient capacity for long-term decay heat removal and that the plant reaches a stable condition.
 - C. The plant PSs setpoints assumed in the transient analyses are selected with adequate allowance for measurement uncertainties as delineated in RG 1.105.
 - Response: Measurement and setpoint uncertainties are conservatively applied in the analysis.
 - D. The event evaluation takes into consideration single failures, operator errors, and performance of non-safety-related systems that are consistent with regulatory guidelines set forth in RG 1.206.
 - Response: The most severe single failure of a safety system (loss of an EFWS train) is assumed for the analysis.
2. Using the ANS standards as guidance, specific criteria have been developed to meet the relevant requirements of GDC 10, GDC 13, GDC 15, GDC 17, and GDC 26 for events of moderate frequency as follows:
 - A. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
 - Response: The maximum calculated pressures in the reactor coolant and main steam systems are lower than the 110 percent of the design values. RCS pressure does not reach the PSRV setpoint, and main steam system pressure does not reach the MSSV setpoints.
 - B. Fuel-cladding integrity is maintained by keeping the minimum DNBR above the 95/95 DNBR limit for PWRs, and the CPR above the MCPR safety limit for

BWRs based on acceptable correlations (refer to Section 4.4), as well as by satisfaction of any other SAFDL that may be applicable to the particular reactor design.

- Response: The event does not challenge DNBR or linear heat generation limits.
- C. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
- Response: The analysis demonstrates that the plant reaches a stable condition and does not generate a more serious plant condition.
- D. To meet the requirements of GDC 10 and GDC 15, the positions of RG 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of transient addressed in this SRP section.
- Response: Reference 1 describes how the methodology biases input values to account for uncertainties in spans and setpoints to achieve a conservative result for the event being analyzed.
- E. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR 50, shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53 and GDC 17.
- Response: The most severe single failure of a safety system (loss of an EFWS train) is assumed for the analysis.
- F. The guidance provided in SECY 77-439 as cited in Reference 2, SECY 94-084 as cited in Reference 2, and RG 1.206 with respect to the consideration of the performance of non-safety-related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems) must be evaluated and verified.
- Response: Non-safety systems are modeled when they make the consequences of the event more severe. PZR sprays are simulated because they are non-safety-related systems that make the consequences of the event more severe for some cases.
3. The applicant's analysis of the loss of normal feedwater transient should be performed using an acceptable analytical model. If the applicant proposes to use analytical methods that have not been approved, these methods are evaluated by the staff for acceptability. For new generic methods the reviewer requests an evaluation by the appropriate organization for reactor systems. The value of parameters used in the analytical model should be suitably conservative. The following values are considered acceptable for use in the model:
- A. The initial power level is taken as the licensed core thermal power for the number of loops initially assumed to be operating plus an allowance of two

percent to account for power measurement uncertainties, unless a lower power level can be justified by the applicant. The number of loops operating at the initiation of the event should correspond to the operating condition which maximizes the consequences of the event.

- Response: The analysis is performed at rated thermal power plus measurement uncertainty. Thermal design flow is used, which is less than the expected nominal design flow minus measurement uncertainty.
- B. Conservative scram characteristics are assumed (i.e., for a PWR, maximum time delay with the most reactive rod held out of the core and for a BWR, a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate), unless a different conservatism factor can be justified through the uncertainty methodology and evaluation, or the uncertainty has otherwise been accounted for (refer to Section 4.4).
- Response: A conservative scram curve and scram timing delay are used for the analysis. The scram curve assumes the most reactive rod is stuck out of the core.
- C. The core burnup is selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, power profile and radial power distribution.
- Response: Conservative values based on an array of proposed fuel cycle operating methods are used for the moderator temperature coefficient and Doppler temperature coefficients.
- D. Mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with RG 1.105.
- Response: Instrument and measurement uncertainties are conservatively applied for the analysis.

15.2.8 Feedwater Line Breaks Inside and Outside Containment

15.2.8.1 Identification of Causes and Event Description

An FWLB is a PA resulting from a rupture in a feedwater line large enough to prevent the addition of sufficient feedwater to the SGs to maintain secondary side fluid inventory in the SGs. A break upstream of the feedwater line check valve affects the plant only as a loss of feedwater. This scenario is addressed by the evaluation of LNFF (Section 15.2.7) and LNEP (Section 15.2.6). If the break in a feedwater line is between the check valve and the SG, and is large enough, fluid from the SG also could be discharged through the break along with the incoming feedwater.

Depending on the size of the break and the plant operating conditions at the time of the break, the break could cause either an RCS cooldown, due to excessive energy

discharge through the break, or an RCS heatup. The FWLB scenarios that cause a cooldown of the RCS are bounded by the analyses described in Section 15.1.5. Only the FWLB scenarios that cause a heatup of the RCS are analyzed and presented.

The rupture of a feedwater line reduces heat removal from the RCS for the following reasons:

- Feedwater flow to the SGs is diminished, particularly in the SG fed by the affected feedwater line. Because feedwater is subcooled, its loss might cause reactor coolant temperatures to increase prior to RT.
- Inventory in the affected SG might be discharged through the break and be unavailable for decay heat removal after RT.
- When the MFW high-load lines close following RT, the reduced flow from low-load lines can be diverted entirely out the break.
- EFW is supplied to the SGs through a separate line that discharges through a spray ring located outside the steam separator riser tubes. If the break is large, there is the potential for the EFW to the affected SG to be swept out the broken MFW line.

Depending on break size, an RT is generated on low NR level in the affected SG, low SG pressure, SG high-pressure drop, or for the smallest breaks on high PZR pressure. RT causes TT, and if a LOOP occurs, MFW is terminated and RCPs coast down. EFW is actuated in the respective SGs when the PS low WR level setpoint is reached. This signal also isolates the normal blowdown of secondary-side liquid via the SG blowdown system.

Until the MSIVs close, steam from the unaffected SGs can be diverted to the affected SG through the main steam line header and be discharged from the break. For large breaks, MSIV closure occurs automatically on the SG high-pressure drop PS signal. As the break size decreases, the time to automatic closure becomes longer. For the smallest break sizes, MSIV closure may not occur automatically. MSIV closure causes repressurization of the unaffected SGs and a more rapid depressurization of the affected SG.

Subsequent dryout of the affected SG together with repressurization of the unaffected SGs to the MSRT setpoint causes the RCS temperature and pressure to increase. The PSRVs open to prevent overpressurization of the RCS. As the unaffected SGs refill with EFW, their heat transfer capacity improves until they are capable of removing core decay heat and, if the RCPs are in operation, pump heat. This condition establishes the plant in a stable controlled state.

15.2.8.2 Method of Analysis and Assumptions

The S-RELAP5 computer code is used to calculate the transient thermal and hydraulic response of the primary and secondary systems in accordance with the methodology described in Reference 1. The computer code simulates the necessary components and contains the features required to model this event.

The large FWLB event is characterized by an initial primary side cooldown prior to RT with resultant increase in reactor core power. This response is similar to that of a small steam line break (Section 15.1.5). Therefore, the analyses to verify compliance with SAFDLs for the spectrum of steam line break events bound this event. Smaller FWLB events cause a cold leg heatup rather than cooldown prior to RT. This initial heatup is not severe and is accompanied by an increase in RCS pressure. In addition, the core power declines slowly. Therefore, small FWLB transients do not challenge SAFDLs.

The FWLB is evaluated for overpressurization of the RCS and SGs, and radiological consequences. The respective analyses are biased conservatively for the acceptance criterion being evaluated. The severity of the FWLB scenario depends on a number of system parameters, including the break size, initial reactor power, and the functioning of various control and safety-related systems. For this reason, a spectrum of break sizes is investigated that ranges from the minimum break size that can be overfed by the MFW, 0.0184 ft², to the area of the MFW distribution half-ring inlet, 0.922 ft². For conservatism, the analyses apply degraded environment uncertainties to setpoints.

It is assumed that one train of EFW is unavailable due to maintenance. For the cases evaluating RCS overpressurization, the most limiting single failure is the inoperability of another train of EFW. For the cases evaluating SG overpressurization, the most limiting single failure is the inoperability of the MSRT in an unaffected SG. In addition, for the SG overpressurization case, the lowest setpoint MSSV on the steamline that experiences the failed MSRT train is assumed to be out-of-service.

One of the active EFW trains is assumed to feed the affected SG. Because of the possibility that this EFW injection could be entrained and carried out the break before absorbing energy, the EFW is assumed to go directly to the Containment Building. At 30 minutes after RT, the analyses assume the operator redirects the EFW of the affected SG to one of the SGs in which EFW is not operating. This action is adequate to restore the inventories of the SGs experiencing EFW flow and remove decay heat from the RCS.

Cases are analyzed with and without an assumed LOOP on RT. For the non-LOOP cases, operator action is credited to trip two RCPs 30 minutes after RT. The RCPs tripped are in the loop with the affected SG and in one of the loops without EFW.

For evaluating radiological consequences, cases are analyzed that are biased to minimize SG inventory. In these cases, the SG blowdown is not isolated until the low-low SG level signal.

Table 15.2-10—Feedwater Line Break - Key Input Parameters and Table 15.2-11—Feedwater Line Break - Key Equipment Status respectively present key inputs and equipment status for the feedwater line break analyses. Table 15.2-12—Feedwater Line Break - Key Case Specific Assumptions presents case specific assumptions.

The FWLB event is classified as a PA. The acceptance criteria are as follows:

- The EFW system maintains adequate decay heat removal.
- Pressure in the reactor coolant and main steam systems is maintained below 110 percent of design pressures.
- Acceptable minimum DNBR remain above the 95/95 DNBR limit; if the DNBR falls below these values, fuel failure (rod perforation) must be assumed for rods not meeting these criteria.
- Calculated doses are a small fraction of the 10 CFR 100 guidelines (refer to Section 15.0.3).

15.2.8.3 Results

Results are presented for three cases:

- Representative small feedwater line break.
- Maximum RCS pressure.
- Maximum main steam system pressure.

15.2.8.3.1 Representative Small Break Case

The small break case presented has the minimum break area analyzed, 0.0184 ft², and produces the highest PZR level. Table 15.2-13—Feedwater Line Break: Representative Small Break Case - Sequence of Events presents the sequence of events for this case. Figure 15.2-49 through Figure 15.2-68 show the responses of key parameters.

MFW flow is terminated at event initiation. PZR sprays are modeled because of their potential to increase PZR level. The PS initiates RT at 88 seconds on an SG NR low-level signal. LOOP is assumed to occur on the TT following RT. The PZR sprays are lost due to LOOP. RCS temperature and pressure continue to increase as the secondary system pressurizes following closure of the TSVs, and the RCPs coast down. The SG WR low-level setpoint that actuates EFW is reached at 95 seconds in the

unaffected SG that has operable EFW (EFW to the affected SG is assumed to be swept out of the break without absorbing energy and, therefore, is not modeled.

Secondary system temperature and pressure increase in the four SGs reaching the MSRTV opening setpoint. The MSRTs release steam for approximately 17 seconds prior to closing. Liquid is discharged through the PSRVs, which are designed for this function.

At 30 minutes after RT, the EFW train feeding the affected SG is realigned to an unfed SG. After the EFW train is realigned, the liquid inventory increases in the fed SGs, and the energy being removed by the secondary side exceeds the energy being added to the primary side. The plant then reaches a stable, controlled state.

15.2.8.3.2 Maximum Reactor Coolant System Pressure Case

This case has a break area of 0.415 ft² and produces the highest peak pressure in the RCS. This condition occurs at the bottom of the reactor vessel. Table 15.2-14—Feedwater Line Break: Maximum RCS Pressure Case - Sequence of Events presents the sequence of events. Figure 15.2-69—FWLB Maximum RCS Pressure Case - Reactor and Total Steam Generator Power through Figure 15.2-86—FWLB Maximum RCS Pressure Case - Reactivities show the responses of key parameters.

MFW flow is assumed to be terminated at event initiation. The PS initiates RT at 40 seconds on an SG NR low-level signal. LOOP does not occur in this case because sensitivity studies show that it produces the highest peak RCS pressure for this event.

The PS signals the MSIVs to close at 173 seconds when the SG high-pressure drop setpoint is reached. Secondary system temperature and pressure increase in the three unaffected SGs, which forces RCS temperature and pressure to increase. The PSRVs open at 528 seconds and cycle to control pressure. The peak RCS pressure of 2688.3 psia is reached at the bottom of the reactor vessel.

The MSRTVs in the unaffected SGs open at 606 seconds and cycle to control to the high SG pressure setpoint. The PS actuates EFW in an unaffected SG at 747 seconds when the SG WR low-level setpoint is reached. EFW flow to the affected SG is assumed to be swept out of the break without absorbing energy and, therefore, is not modeled.

At 30 minutes, the operator realigns the EFW train feeding the affected SG to an unfed SG. The operator also trips two RCPs to reduce the heat load on the SGs. One RCP is in the loop with the affected SG and the other is in a loop without EFW. As the levels in the two active SGs increase, heat removal capability exceeds decay heat and the plant enters a stable controlled state.

15.2.8.3.3 Maximum Main Steam System Pressure Case

This case has a break area of 0.922 ft² and produces the highest peak pressure in the secondary system. This is located at the bottom of the SG. Table 15.2-15—Feedwater Line Break: Maximum Main Steam System Pressure - Sequence of Events presents the sequence of events. Figure 15.2-87—FWLB Maximum Secondary Pressure Case - Reactor and Total Steam Generator Power through Figure 15.2-104—FWLB Maximum Secondary Pressure Case - Reactivities show the responses of key parameters.

MFW flow is assumed to be terminated at event initiation. The PS initiates RT at 6.38 seconds on an SG high-pressure drop signal, which also closes the MSIVs. LOOP does not occur in this case because sensitivity studies show it produces the highest peak secondary system pressure for this event.

Secondary system temperature and pressure increase in the three unaffected SGs, which forces RCS temperature and pressure to increase. The PSRVs open once at 228 seconds to control the pressure.

The MSRIVs open at 198 seconds in the two unaffected SGs where the MSRIVs are operable. Pressure continues to increase in the unaffected SG with the failed MSRT. At 1694 seconds, the MSSV opens and controls the pressure. The peak secondary system pressure of 1562.2 psia is reached at that time at the bottom of the SG (at the tubesheet).

Three EFW trains are available but only one is effective—the one feeding an unaffected SG with an operable MSRT. This train actuates at 1283 seconds on an SG WR low-level signal. Of the other two operable EFW trains, one feeds the SG with the FWLB and, therefore, is assumed to be swept out of the break without absorbing energy. This train is not modeled. The third operable EFW train feeds the SG with the failed MSRT. This train is not actuated automatically because the level in that SG does not decrease to the PS actuation setpoint.

At 30 minutes after RT, the operator realigns the EFW train feeding the SG with the FWLB to the unfed SG with an operable MSRT. The operator also trips two RCPs to reduce the heat load on the SGs. One is in the loop with the FWLB and the other in the loop with the failed MSRT. As the levels in the two fully functional SGs increase, heat removal capability exceeds decay heat and the plant enters a stable controlled state.

15.2.8.4 Radiological Consequences

The radiological consequences of this event are described in Section 15.0.3.

15.2.8.5 Conclusions

The FWLB analysis demonstrates the following:

- The EFW system provides adequate decay heat removal as the energy removed by the secondary side equals or exceeds the energy added to the primary side and the plant reaches a stable controlled state for all cases.
- Pressure in the reactor coolant and main steam systems remains below 110 percent of design pressure. The peak RCS pressure of 2688.3 psia is below the 110 percent design value of 2803 psia. The main steam system peak pressure of 1562.2 psia is below the 110 percent design value of 1593 psia.
- As described in Section 15.2.8.2, this event is bounded by the MSLB event (Section 15.1.5) with regard to SAFDLs.

15.2.8.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.2.8 events included in NUREG-0800, Section 15.2.8, (Reference 2) and descriptions of how these criteria are met are listed below:

1. Requirements for maintenance of adequate decay heat removal by the auxiliary feedwater system (AWFS) are in 10 CFR 50.34(f)(1)(ii), (TMI issue II E 1.1) and 10 CFR 50.34(f)(2)(xii), (TMI issue II E 1.2). Requirements for RCP operation are in 10 CFR 50.34(f)(1)(iii), (TMI issue 2 K 2).
 - Response: The EFW system maintains adequate decay heat removal as the energy removed by the secondary side equals or exceeds the energy added to the primary side and the plant reaches a stable controlled state for all cases.
2. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures in the ASME Boiler and Pressure Vessel Code, Section III as cited in Reference 2 for low-probability events and below 120 percent for very low-probability events like double-ended guillotine breaks.
 - Response: Pressure in the reactor coolant and main steam systems is maintained below 110 percent of design pressures for all cases. The RCS peak pressure of 2688.3 psia is below the 110 percent design value of 2803 psia. The main steam system peak pressure of 1562.2 psia is below the 110 percent design value of 1593 psia.
3. The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit for PWRs based on acceptable correlations (refer to Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for rods not meeting these criteria unless, from an acceptable fuel damage model (see SRP Section 4.2) including the potential adverse effects of hydraulic instabilities, fewer failures can be shown to occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core remains in place and intact with no loss of core cooling capability.

- Response: As described in Section 15.2.8.2, this event is bounded by the MSLB event (refer to Section 15.1.5) with regard to SAFDLs.
4. Calculated doses at the site boundary from any activity release must be a small fraction of the 10 CFR 100 guidelines.
 - Response: The dose criteria are met as described in Section 15.0.3.
 5. The integrity of the RCPs should be maintained so loss of alternating current power and containment isolation do not result in seal damage.
 - Response: RCP seal integrity is maintained as discussed in Section 5.4.1.2.1.
 6. The AFWS must be safety grade and automatically initiated when required.
 - Response: The U.S. EPR EFW system is a safety-related system that is actuated automatically by the safety-related PS (refer to Section 10.4.9).
 7. Certain assumptions should be in the analysis of important parameters that describe initial plant conditions and postulated system failures.
 - A. The power level assumed and number of loops operating at the initiation of the transient should correspond to the operating condition which maximizes accident consequences.
 - Response: Initial conditions are established based on the methodology described in Reference 1 and sensitivity studies.
 - B. The assumptions as to whether offsite power is lost and the time of loss should be conservative.
 - Response: Sensitivity studies are performed to evaluate the effect of the availability of offsite power and the timing of a LOOP.
 - C. The effects (such as pipe whip, jet impingement, reaction forces, temperature, and humidity) of the postulated feedwater line breaks on other systems should be considered consistently with the intent of BTP 3-3 and BTP 3-4.
 - Response: The effects of an FWLB on other systems such as plant instrumentation are considered. BTP 3-3 and BTP 3-4 are addressed in Section 3.6.
 - D. The worst single active component failure should be assumed to occur in the systems required to control the transient. For new applications, LOOP should not be considered a single failure; FWLBs should be analyzed with and without LOOP, as in assumption B, in combination with a single, active failure.
 - Response: Sensitivity studies are performed to determine the worst single failure for each acceptance criterion. The effect of LOOP is evaluated in addition to a single failure.

- E. The maximum rod worth should be assumed to be held in the fully withdrawn position per GDC 25. An appropriate rod reactivity worth versus rod position curve should be assumed.
- Response: A conservative reactivity insertion curve and timing delay are used for the FWLB analysis. The analysis assumes the most reactive rod is stuck out of the core.
- F. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.
- Response: The analysis conservatively bounds the preceding core performance parameters for the proposed fuel design as described in Reference 1.
- G. The initial core flow assumed for the analysis of the feedwater line rupture accident should be chosen conservatively.
- Response: The initial cooldown effect of large FWLBs is bounded by the MSLB event described in Section 15.1.5. Initial conditions biased for heatup are established based on the methodology described in Reference 1 and sensitivity studies.
- H. During the initial 10 minutes of the transient, if credit for operator action is required (i.e., RCP trip), an assessment for the limiting consequence must account for operator delay and error.
- Response: Operator actions are not credited before 30 minutes.

15.2.9

References

1. ANP-10263P-A, Revision 0, "Codes and Methods Applicability Report for the U.S. EPR," AREVA NP Inc., November 2007.
2. NUREG-0800, "U.S. NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 2007.
3. ANP-10287P, Revision 0, "Incore Trip Setpoint and Transient Methodology for U.S. EPR Topical Report," AREVA NP Inc., November 2007.
4. NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design: Chapters 10-20," U.S. Nuclear Regulatory Commission, Volume 2, September 2004.

Table 15.2-1—Turbine Trip - Key Input Parameters

Parameter	Analysis Value
Initial reactor power	4612 MW _t
Initial RCS loop flow rate	119,692 gpm/loop
Initial reactor vessel average temperature	594°F
Initial PZR pressure	2250 psia
Initial PZR liquid level	59.3%
Initial SG secondary-side saturation pressure	1103 psia (5% SG tube plugging)
SG tube plugging	5%
Initial SG level	49% NR
MTC	0 pcm/°F
Delayed neutron fraction (β)	0.007358
Doppler reactivity coefficient	-1.17 pcm/°F
MSRT opening pressure	1414.7 psia
MSRT capacity	2,844,146 lb _m /hr at 1384.7 psia per train
PSRV open setpoints	2600.4 psia
PSRV capacity	661,400 lb _m /hr per valve at 2549.7 psia
PSRV opening time	0.9 s
MSSV open setpoints	1518.5 psia 1549.4 psia
MSSV capacities	1,422,073 lb _m /hr at 1518.5 psia 1,422,073 lb _m /hr at 1549.4 psia
TSV stroke time	0.1 s
Time of LOOP	Coincident with RT
High PZR pressure trip setpoint (RT, TT)	2439.9 psia

Table 15.2-2—Turbine Trip - Key Equipment Status

Plant Equipment or System	Status
RPS	Operable
MSRT opening on SG pressure	One fails to open
PSRV opening on PZR pressure	Operable
MSSV opening on SG pressure	Operable
PZR pressure control (PZR spray)	Not operable
MFW	Isolated at RT
RCPs	Operating until LOOP
Turbine bypass system	Not credited
PT	Not credited

Table 15.2-3—Turbine Trip-RCS Overpressurization - Sequence of Events

Event	Time (s)
TT	0.0
TSV fully closed	0.1
RT setpoint reached (high PZR pressure)	6.48
RT signal issued	6.98
Rod movement begins	7.38
LOOP occurs (RCPs and MFW tripped)	7.39
1st PSRV opens	9.48
2nd PSRV opens	9.48
3rd PSRV opens	9.48
Peak RCS pressure	9.80
MSRTs open	11.31
Peak secondary pressure	11.99

Table 15.2-4—MSIVC Secondary Overpressurization - Key Input Parameters

Parameter	Analysis Value
Initial reactor power	4612 MW _t
Initial RCS loop flow rate	119,692 gpm/loop
Initial reactor vessel average temperature	598°F
Initial PZR pressure	2250 psia
Initial PZR liquid level	59.3%
Initial SG secondary-side saturation pressure	1167 psia
SG tube plugging	0%
Initial SG level	49% NR
MTC	-50 pcm/°F
Delayed neutron fraction (β)	0.005151
Doppler reactivity coefficient	-1.47
MSIV stroke time	0.1 seconds
MSRT opening pressure	1414.7 psia
MSRT capacities	2,844,146 lb _m /hr at 1384.7 psia per train
MSSV open setpoints	1518.5 psia 1549.4 psia
MSSV capacities	395.02 lb _m /s at 1518.5 psia and 395.02 lb _m /s at 1549.4 psia
TSV stroke time	0.1 s

Table 15.2-5—MSIVC Overpressurization - Key Equipment Status

Plant Equipment or System	Status
RPS	Operable
MSRT opening on SG pressure	One fails to open
PSRV opening on PZR pressure	Operable
MSSV opening on SG pressure	Operable ¹
PZR pressure control (PZR spray)	Operable
MFW	Isolated at RT
RCPs	Operating
Turbine bypass system	Not credited
PT	Not credited
RT on TT	Not credited

Note:

1. Low setpoint MSSV on the steamline of the SG-1 assumed to be out-of-service.

Table 15.2-6—MSIVC Secondary Overpressurization - Sequence of Events

Event	Time (s)
Loop 1 MSIV fully closed	0.1
RT setpoint reached (high SG pressure)	5.82
RT signal received	6.32
Rod insertion begins	6.72
Peak power (4834 MW _t)	6.73
TT	7.32
Peak RCS pressure (2410 psia)	8.90
MSSV opens	10.77
Peak secondary pressure (1567 psia)	10.85

Table 15.2-7—LNFF - Key Input Parameters
Sheet 1 of 2

Parameter	Analysis Value
Initial reactor power	4612 MW _t
Initial PZR liquid level	59.3%
Low SG level setpoint actuating RT	0% NR span ¹
Low-low SG level setpoint for EFW actuation and normal SG blowdown isolation	38% WR span
EFW pump start time	15 s delay (no LOOP) 60 s delay (LOOP)
EFW flow rate	400 gpm at EFW temperature (122°F) per credited train at pressures up to 1426 psia, decreasing linearly to 124.5 gpm at 1568 psia
MTC	0 pcm /°F
U-238 capture-to-fission ratio	0.85
Single active failure assumption	Loss of EFW train
Maintenance assumption	Loss of EFW train
Initial RCS loop flow rate	119,692 gpm/loop
Initial reactor vessel average temperature	594°F for base cases 584°F for EOC coast-down
Initial PZR pressure	2250 psia
PSRV open setpoint pressures, same setpoint for three valves	2600.4 psia
PSRV blowdown, close setpoint pressures, same value for three valves	2445.3 psia
PSRV capacities	661,400 lb _m /hr per valve at 2549.7 psia
PSRV opening time	0.9 s
MSRT opening setpoint pressure	1414.7 psia
MSRT close setpoint pressure	609.7 psia
MSRT flow rate	2,844,146 lb _m /hr at 1384.7 psia per train
MSRCV normal initial position	Fully open
MSRCV stroke time	40 s
MSRIV opening time	1.8 s
MSSV open setpoint pressures	1518.5 psia 1549.4 psia

**Table 15.2-7—LNFF - Key Input Parameters
Sheet 2 of 2**

Parameter	Analysis Value
MSSV capacities	1,422,073 lb _m /hr at 1518.5 psia 1,422,073 lb _m /hr at 1549.4 psia
MSSV opening time	0.04 s
SG blowdown flow rate	1% of initial feedwater flow rate per SG
SG blowdown isolation time	20 s after signal
Initial SG level	49% NR
EFW temperature	122°F
Delayed neutron fraction (β)	0.007358
Doppler reactivity coefficient	-1.17 pcm/°F

Note:

1. Conservative value, per Table 15.0-7 the setpoint minus normal uncertainty is 15%.

Table 15.2-8—LNFF - Key Equipment Status

Plant Equipment or System	Status
Rod position controller	Manual
PZR heaters	Disabled
PZR spray (normal and auxiliary)	Available
PSRVs	Available
Turbine bypass system	Not available
RCPs	Available
MFW	Auto mode
EFW	Available
CVCS (charging/letdown)	Disabled
PT	Disabled
Low SG level RT	Available
Low-low SG ESFAS trip	Available

Table 15.2-9—LNFF - Sequence of Events

Event	Time (s)
Feedwater flow is terminated	0.0
SG water level reaches low NR level	54
RT	54.5
RT rod insertion begins	54.9
TT	55.5
Maximum PZR level	58
MSRTs open on high SG pressure	≈64
SG water level reaches low WR level (SG 1 / SG 2 / SG 3 / SG 4) Actuate EFW Isolate blowdown	74.9/74.8/73.8/74.9
EFW flow begins in unaffected SG 1 / SG 2	90.4/90.3
Minimum liquid mass in unaffected SG	≈101
All MSRCVs closed	≈104
All MSRCVs open and modulating pressure to setpoint	≈236
Dryout in affected SG 3 / SG 4	≈2000
End of analysis	4000

Table 15.2-10—Feedwater Line Break - Key Input Parameters
Sheet 1 of 2

Parameter	Analysis Value
Initial reactor power	4612 MW _t
Initial RCS total flow rate	478,768 gpm total for the 4 loops
Initial reactor vessel average temperature	594°F (base cases) 584°F (EOC coastdown cases)
Initial PZR pressure	2250 psia
Initial PZR liquid level	59.3%
Initial SG secondary-side level	49% of NR span
Normal SG blowdown flow	1% of total MFW flow per SG
Normal SG blowdown isolation time	20 s after signal
Minimum cross-section within MFW inlet nozzle and distribution half-ring	0.922 ft ²
Low SG level RT setpoint	0% of NR span ¹
MSIV closure setpoints	Low SG pressure: 649.7 psia or 799.7 psia High SG pressure decrease: P _{init} - 177–29 psi/min or P _{init} - 27–29 psi/min
MSIV closure time	5 s after signal
MSRIV opening pressure	1459.7 psia
MSRT flow rate	2,844,146 lb _m /hr at 1384.7 psia per train
Low-low SG level EFW actuation and normal SG blowdown isolation setpoint for unaffected SGs	23.5% of WR span
EFW actuation time	15 s delay for pump start for cases with offsite power 60 s delay for pump start for cases with LOOP
Single failure	All except SG overpressure - SG 3 (unaffected) EFW train fails to operate For SG overpressure - SG 3 MSRT fails to open
Number of EFW trains credited	1 credited for RCS overpressure (1 assumed out of service for maintenance, another assumed to fail, and flow from other operable train assumed delivered to affected SG/ lost out the break) 2 credited for SG overpressure

**Table 15.2-10—Feedwater Line Break - Key Input Parameters
Sheet 2 of 2**

Parameter	Analysis Value
EFW flow rate	400 gpm per credited train at pressures up to 1426 psia, linearly ramping to 125 gpm per credited train at 1568 psia
EFW temperature	86°F
PSRV opening setpoint	2600.4 psia for all cases except representative small break case and 2499.0 psia for representative small break case
PSRV blowdown	6% or 152.1 psi
PSRV reseal setpoint	2445.3 psia for all cases except representative small break case and 2349.9 psia for representative small break case
MTC	0 pcm/°F
Doppler temperature coefficient	-1.17 pcm/°F
Delayed neutron fraction (β) for converting reactivities to units of \$	0.007358
Delayed neutron fraction divided by prompt neutron lifetime (β/λ)	483.30/s
Core average U-238 capture-to-fission ratio	0.85

Note:

1. Conservative value, per Table 15.0-7 the setpoint minus degraded uncertainty is 1%.

Table 15.2-11—Feedwater Line Break - Key Equipment Status

Plant Equipment or System	Status
PZR normal spray	Available for representative small break case until LOOP Disabled for other cases
PZR auxiliary spray	Available for representative small break case until LOOP Disabled for other cases
PZR heaters	Disabled
RCS letdown	Not modeled
RCPs	Operating until LOOP at RT (for LOOP cases) or until operator trips pump in loop with affected SG and pump in loop with another unfed SG at 30 minutes after RT (for other cases)
MSRTs	One fails to open for secondary/SG pressure biased cases (SG-3), all available for other cases
EFW	1 train available, until operator realigns train from affected SG to unfed unaffected SG (30 minutes after RT), which then makes 2 trains available. Exception: Secondary/SG overpressure biased cases where 2 trains available, until operator realigns train from affected SG to unfed unaffected SG (30 minutes after RT), which then makes 3 trains available
MHSI	Available, but not actuated
Normal SG blowdown	Modeled, until isolated, for minimum SG inventory cases Not modeled for other cases
MSSVs	Available ¹

Note:

1. Lowest setpoint MSSV is assumed to be OOS on SG-3.

Table 15.2-12—Feedwater Line Break - Key Case Specific Assumptions

Parameter	Representative Small Break Case	Maximum RCS Pressure Case	Maximum Main Steam System Pressure Case
PZR liquid level	59.3%	59.3%	59.3%
Initial SG NR level	49%	49%	49%
Reactivity parameters	BOC	BOC	BOC
Initial T _{AVG}	584°F	594°F	594°F
LOOP	Yes	No	No
SG tube plugging	5%	0%	0%
MSIV closure setpoint bias	Biased to occur later	Biased to occur later	Biased to occur sooner
PZR sprays	Available until LOOP	Not credited	Not credited
PSRV opening setpoint bias	Low	High	High
Break area	0.01844 ft ²	0.4149 ft ²	0.922 ft ²
MSRT trains available	4	4	3
EFW trains available	1 Before realignment 2 After realignment	1 Before realignment 2 After realignment	2 Before realignment 3 After realignment
SG blowdown system	Not included	Included	Not included

**Table 15.2-13—Feedwater Line Break: Representative Small Break Case -
Sequence of Events**

Event	Time (s)
0.01844 ft ² feedwater line break occurs, MFW flow is terminated	0.00
Normal PZR spray on	14.0
SG 4 NR level reaches low level setpoint with delay RT	87.6
Scram rod insertion begins	88.0
TT	88.6
LOOP assumed on TT Trip RCPs Normal PZR sprays turned off	88.6
PSRVs open, start cycling	90.0
SG water level reaches low WR level in SG 1 Actuate EFW after 0.5 s delay for signal processing	95.1
MSRIVs open on high SG pressure	142
EFW flow begins in SG 1	156
MSRCVs closed	159
Maximum PZR level (112.35% ¹)	1884
EFW from affected SG 4 is cross-connected to SG 2 at 30 minutes after RT (rod motion)	1887.6
End of analysis	7000

Note:

1. The PZR level is extended above the upper water level tap for comparative purposes. The top of the upper dome for the PZR is at an equivalent level of 112.57% in the S-RELAP5 model.

**Table 15.2-14—Feedwater Line Break: Maximum RCS Pressure Case -
Sequence of Events**

Event	Time (s)
0.4149 ft ² feedwater line break occurs, MFW flow is terminated	0.0
SG-4 tube uncover begins	16.0
SG-4 narrow range level reaches low level setpoint with delay, RT	39.3
Scram rod insertion begins	39.7
TT	40.3
SGs 1, 2 and 3 tube uncover begins	48.6
MSIV closure signal on high SG pressure drop	173.1
MSIVs close	178.1
PSRVs open for the first time	528
Peak RCS pressure occurS	530
MSRIVs open on high SG pressure for SGs 1, 2, 3	606
MSRCV closes for SGs 1, 2, 3	624
MSRCV reopens and start modulating to maintain pressure for SGs 1, 2, 3	648
SG water level reaches low WR level in SG-1 Actuate EFW after 0.5 sec delay for signal processing	747.2
EFW flow begins in SG-1	762.7
MSRCVs closes for SGs 2 and 3	1390
MSRCV reopens and start modulating to maintain pressure for SG 3	1400
MSRCVs closes for SG 3	1745
MSRCVs reopens and start modulating to maintain pressure for SGs 2 and 3	1760
EFW from affected SG-4 is cross-connected to SG-2 at 30 minutes after reactor trip (rod motion)	1839.7
RCPs for SG-3 and SG-4 manually tripped at 30 minutes after reactor trip (rod motion)	1839.7
MSRCV closes for SG 2	1845
MSRCV reopens and start modulating to maintain pressure for SG 2	1870
MSRCV closes for SG 3	1905
End of analysis	7000

Table 15.2-15—Feedwater Line Break: Maximum Main Steam System Pressure - Sequence of Events

Event	Time (s)
0.922 ft ² feedwater line break occurs, MFW flow is terminated	0.00
High SG pressure drop with delay, RT and closure of MSIVs	6.38
Scram rod insertion begins	6.78
TT	7.38
SG 4 tube uncover begins	9.66
MSIVs close	11.38
MSRIVs open on high SG pressure for SGs 1 and 2	198
MSRCVs close for SGs 1 and 2	215
MSRCVs open on high SG pressure for SGs 1 and 2 and remain open with flow reducing	228
PSRVs open for the first and only time	228
SGs 1 and 2 tube uncover begins	558
SG water level reaches low WR level in SG 1, Actuate EFW after delay for signal processing	1268
EFW flow begins in SG 1	1283
High setpoint MSSV for SG 3 opens once	1694
Maximum SG bottom pressure (1562.2 psia) is reached	1696
EFW from affected SG 4 is cross-connected to SG 2 at 30 minutes after RT (rod motion)	1807
RCPs for SG 3 and SG 4 manually tripped at 30 minutes after RT (rod motion)	1807
End of analysis	2000

Figure 15.2-1—Turbine Trip RCS Overpressurization - Peak RCS Pressure

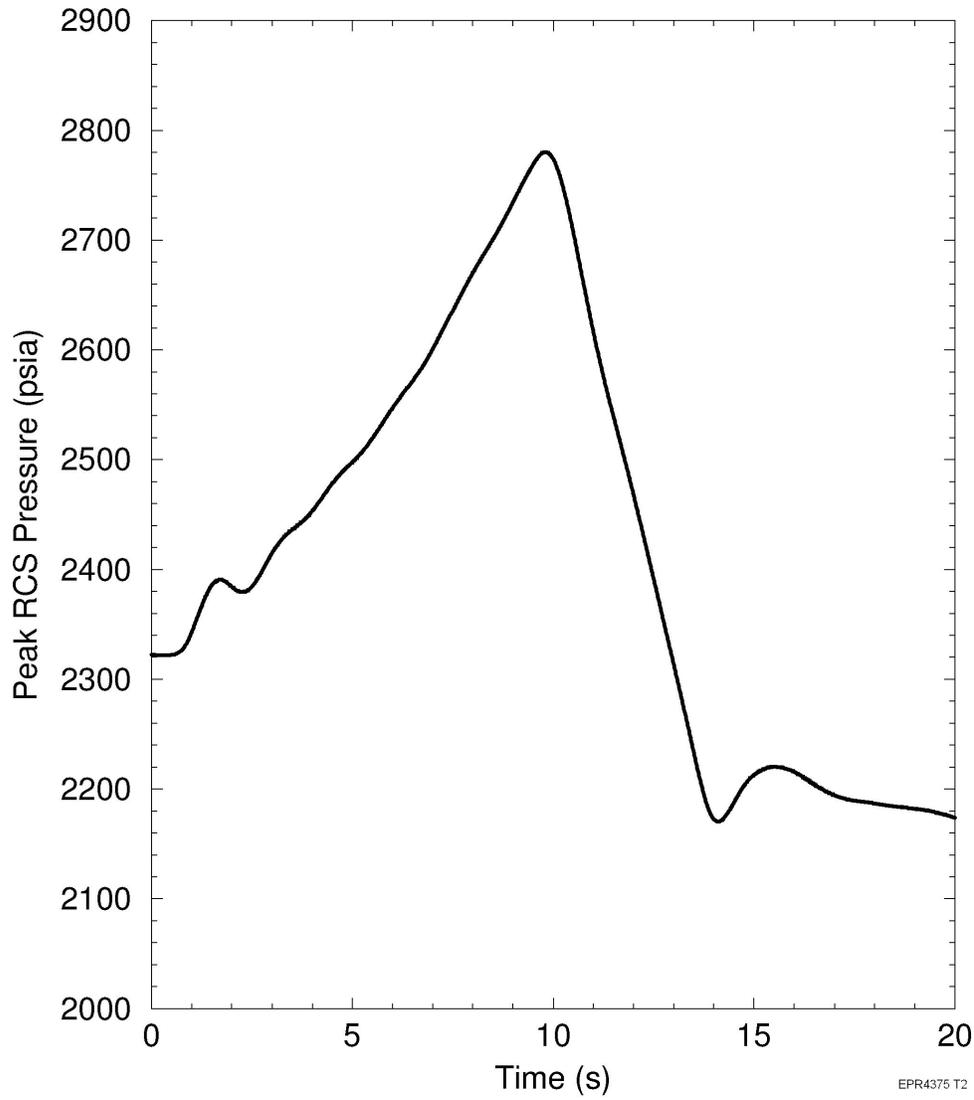


Figure 15.2-2—Turbine Trip RCS Overpressurization - Peak Secondary Pressure

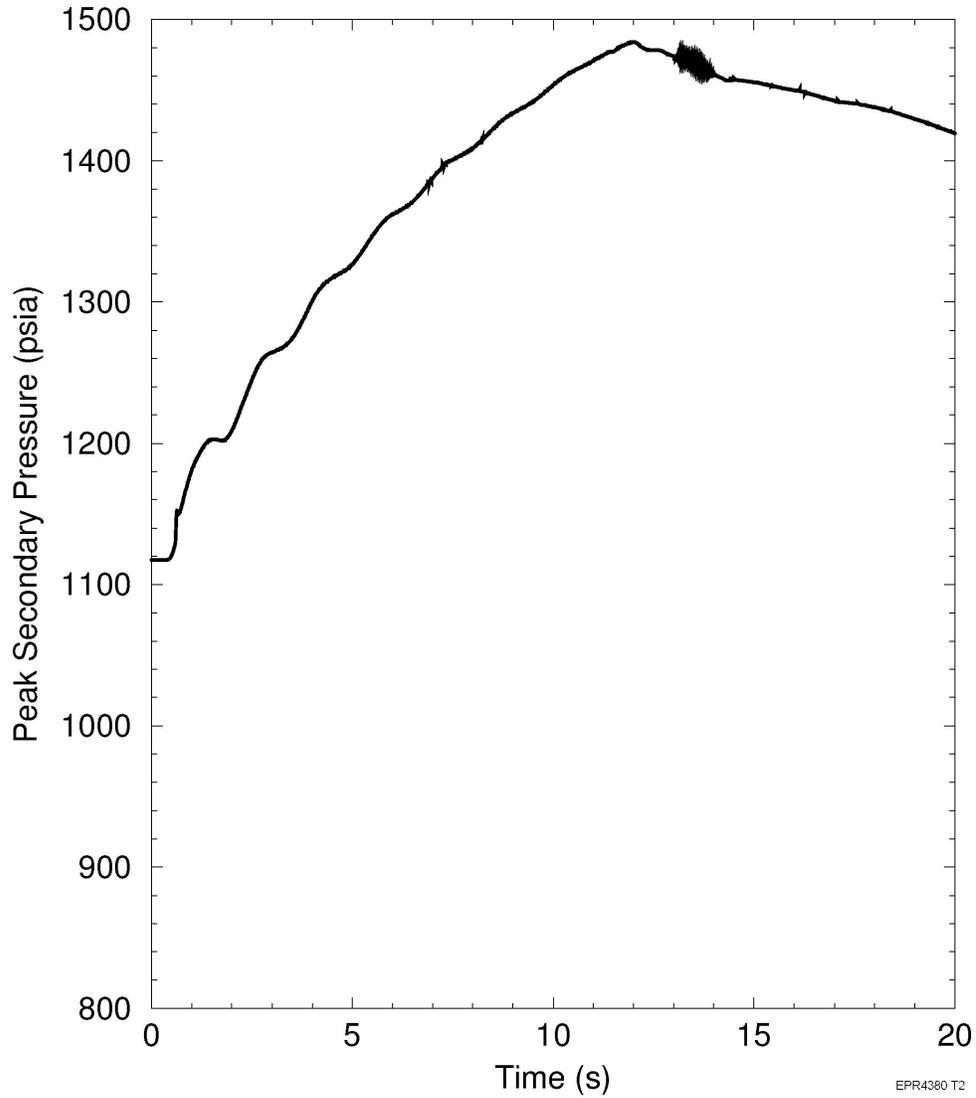
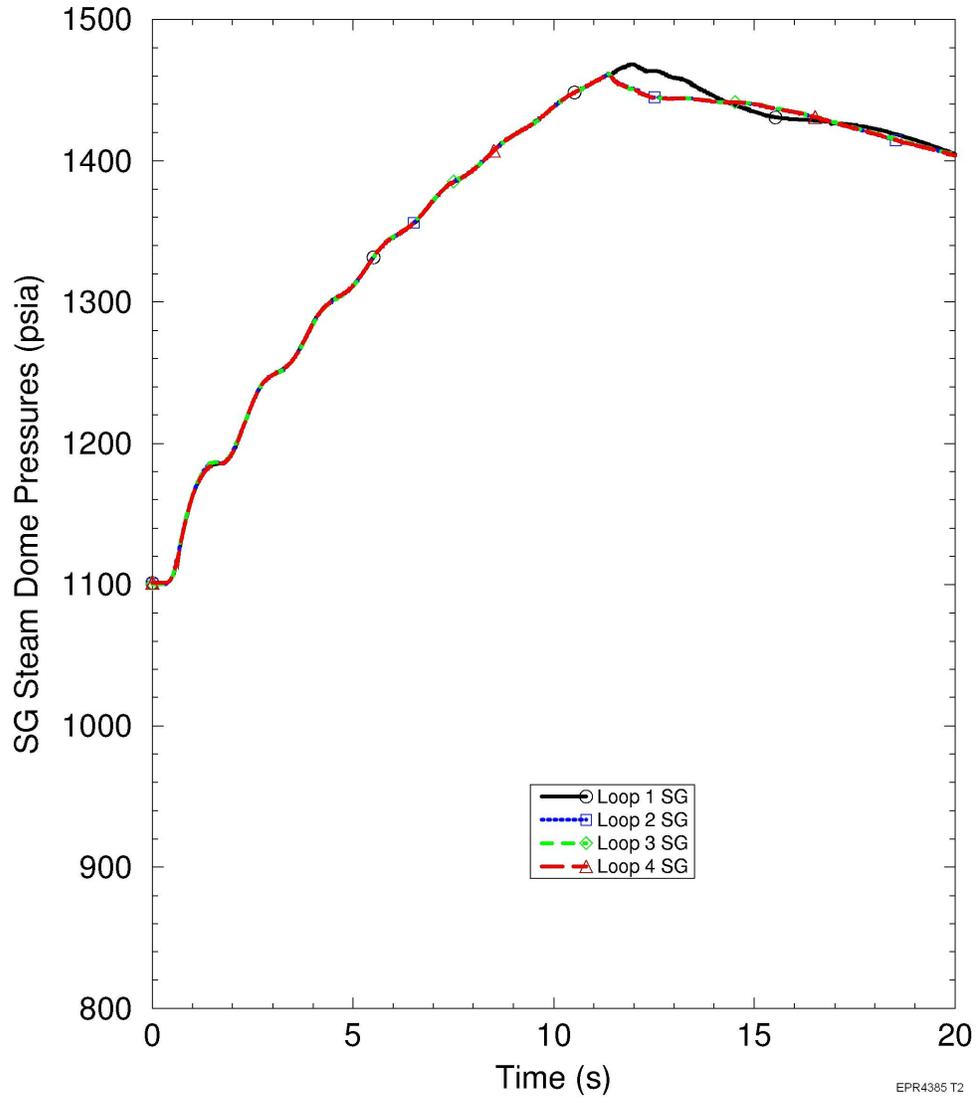


Figure 15.2-3—Turbine Trip RCS Overpressurization - Steam Generator Steam Dome Pressures



EPR4385 T2

Figure 15.2-4—Turbine Trip RCS Overpressurization - % Reactor Power

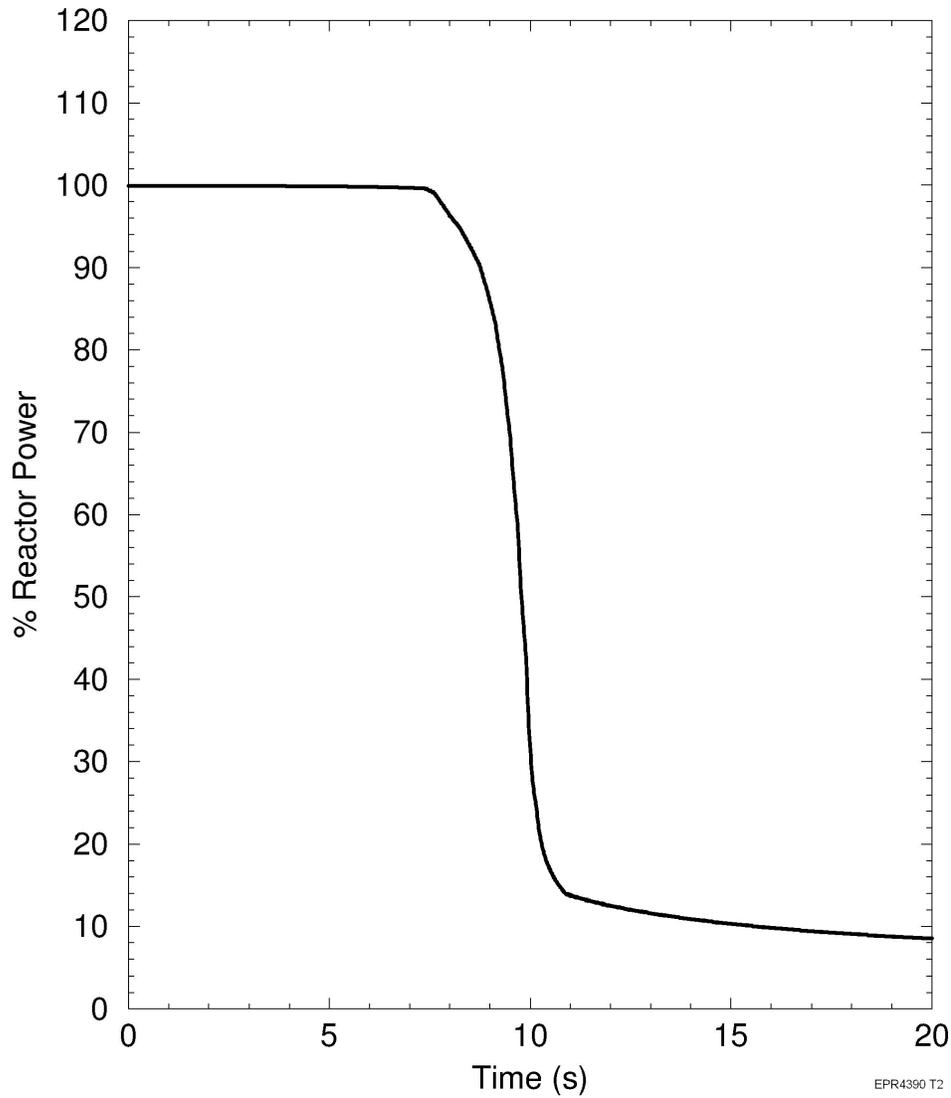


Figure 15.2-5—Turbine Trip RCS Overpressurization - Total Reactivity

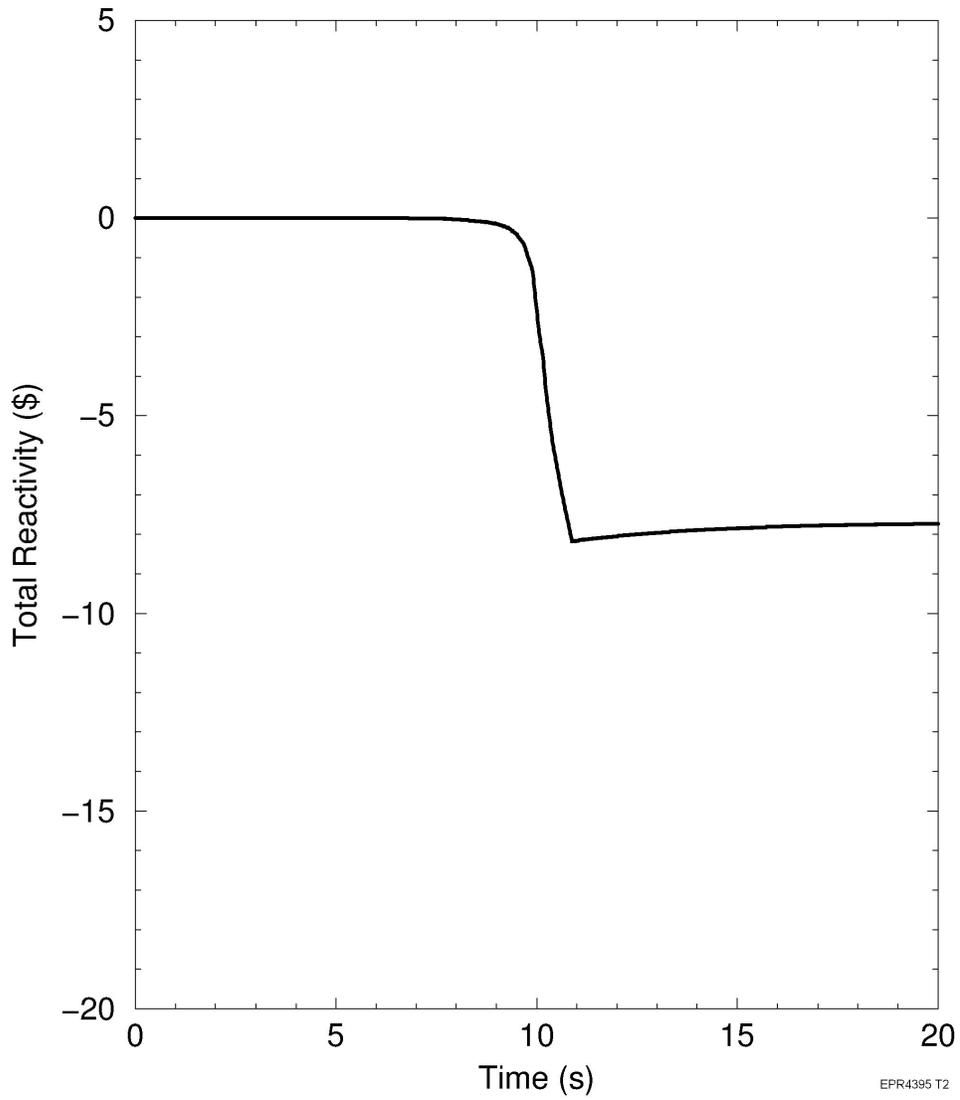


Figure 15.2-6—Turbine Trip RCS Overpressurization - Total RCS Flow

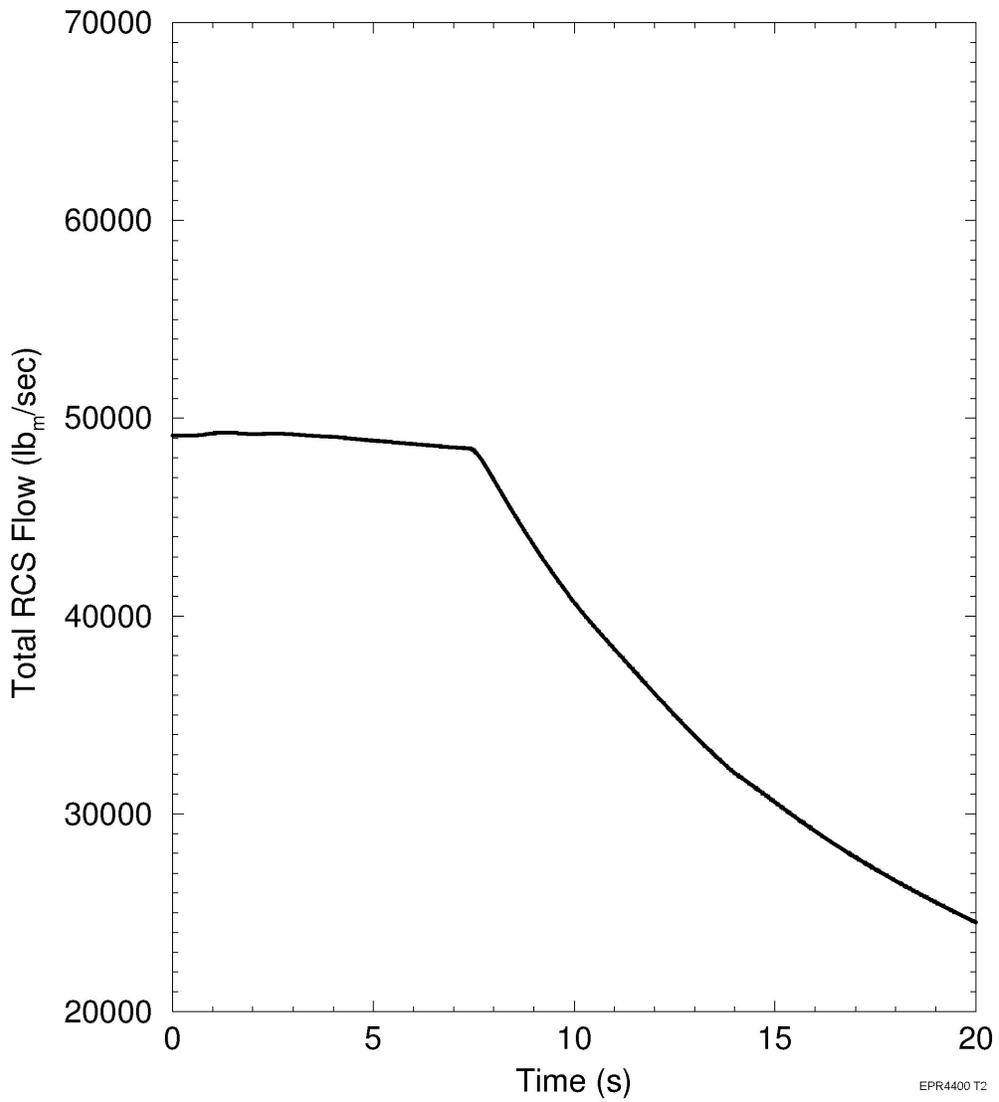


Figure 15.2-7—Turbine Trip RCS Overpressurization - Core Fluid Temperatures

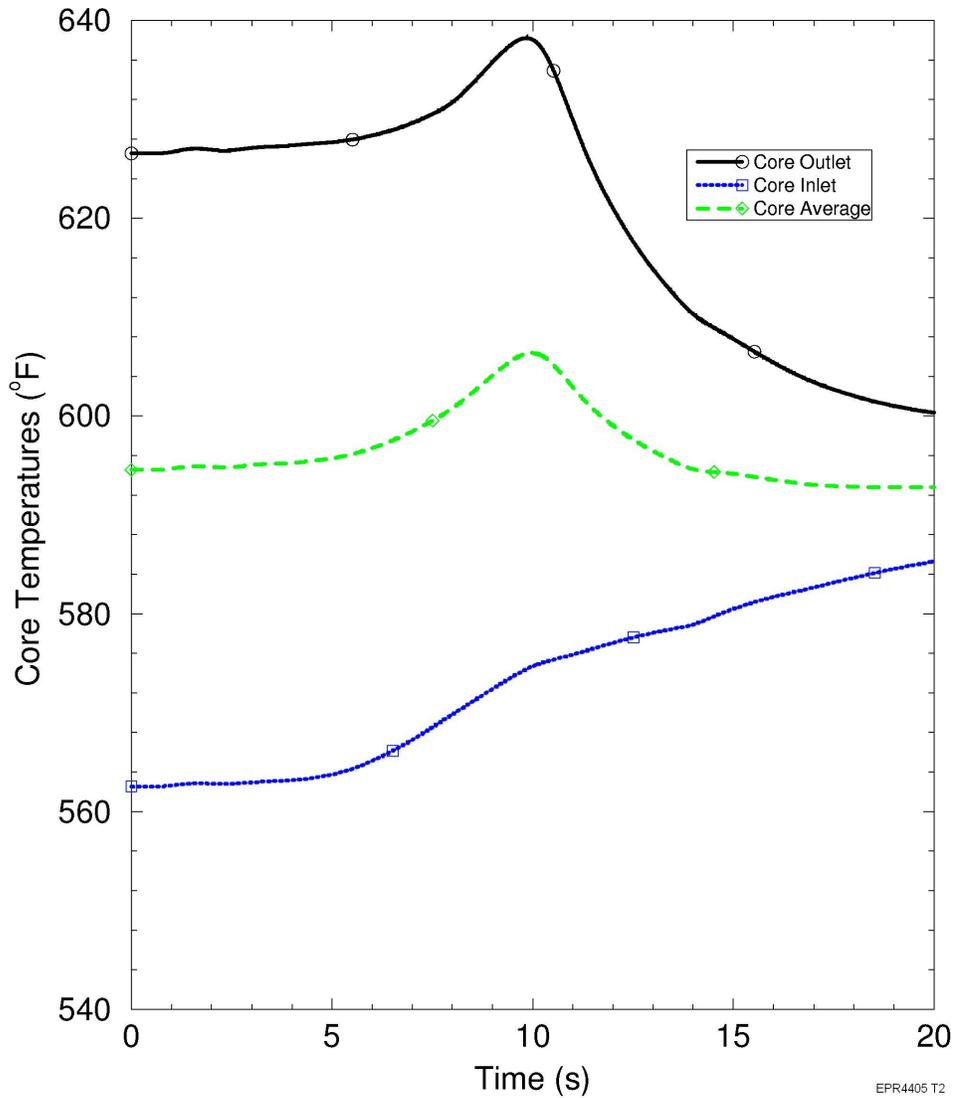


Figure 15.2-8—Turbine Trip RCS Overpressurization - RCS Loop Temperature

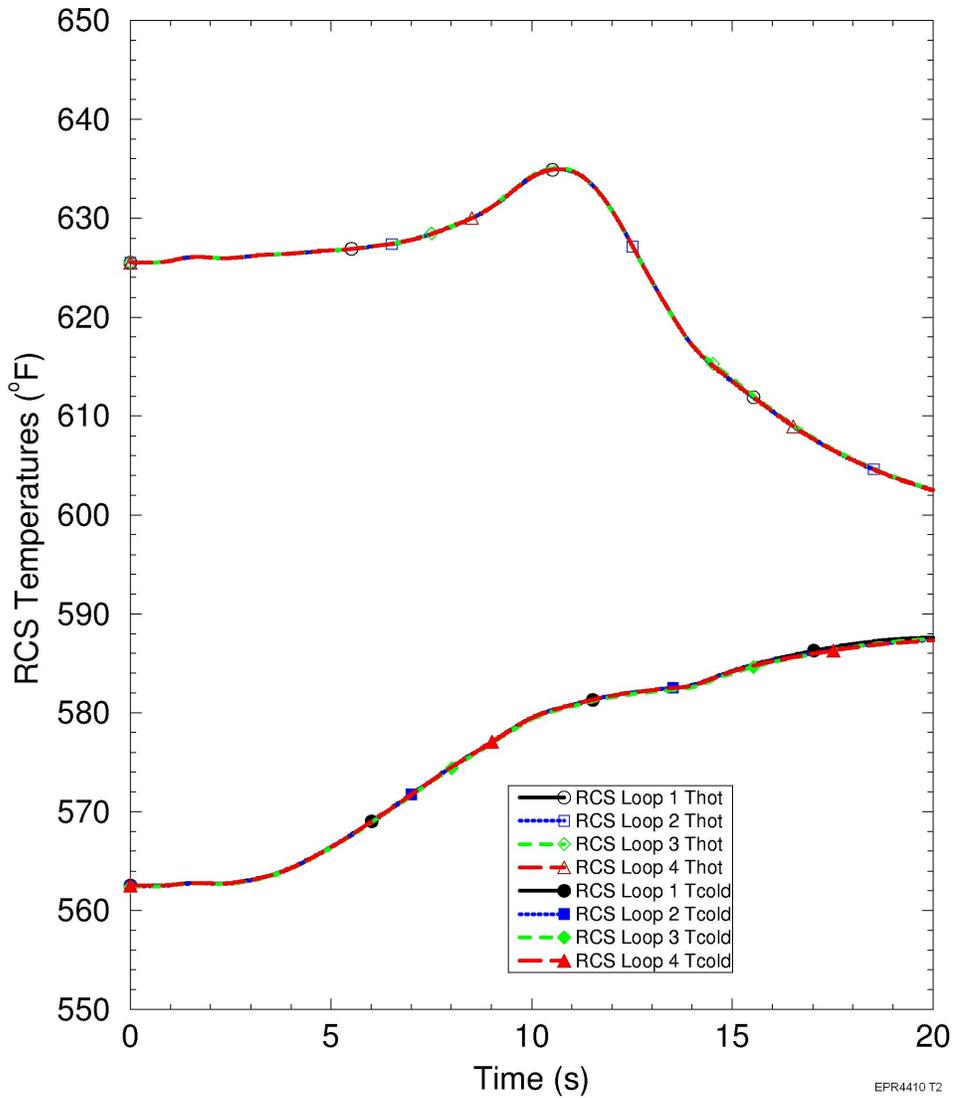


Figure 15.2-9—Turbine Trip RCS Overpressurization - PZR Pressure

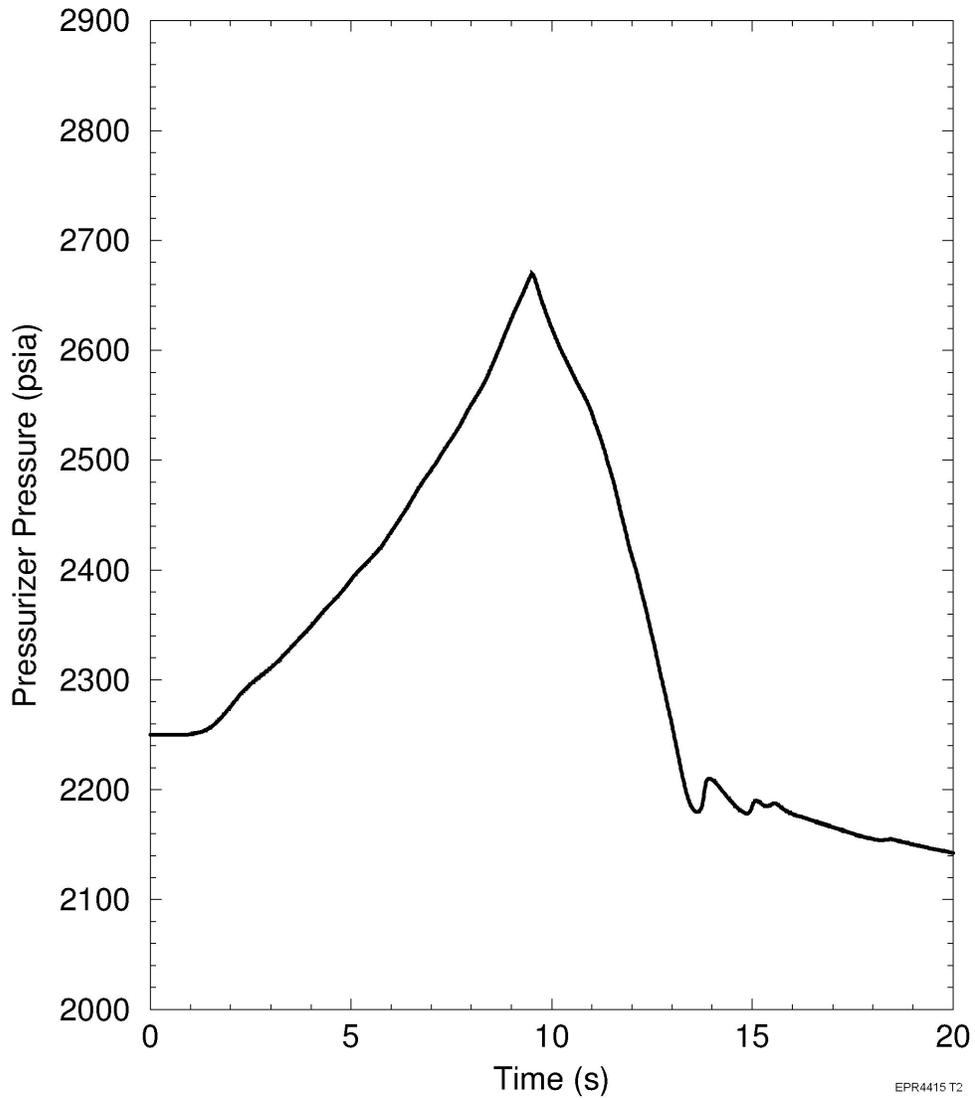


Figure 15.2-10—Turbine Trip RCS Overpressurization - PZR Level

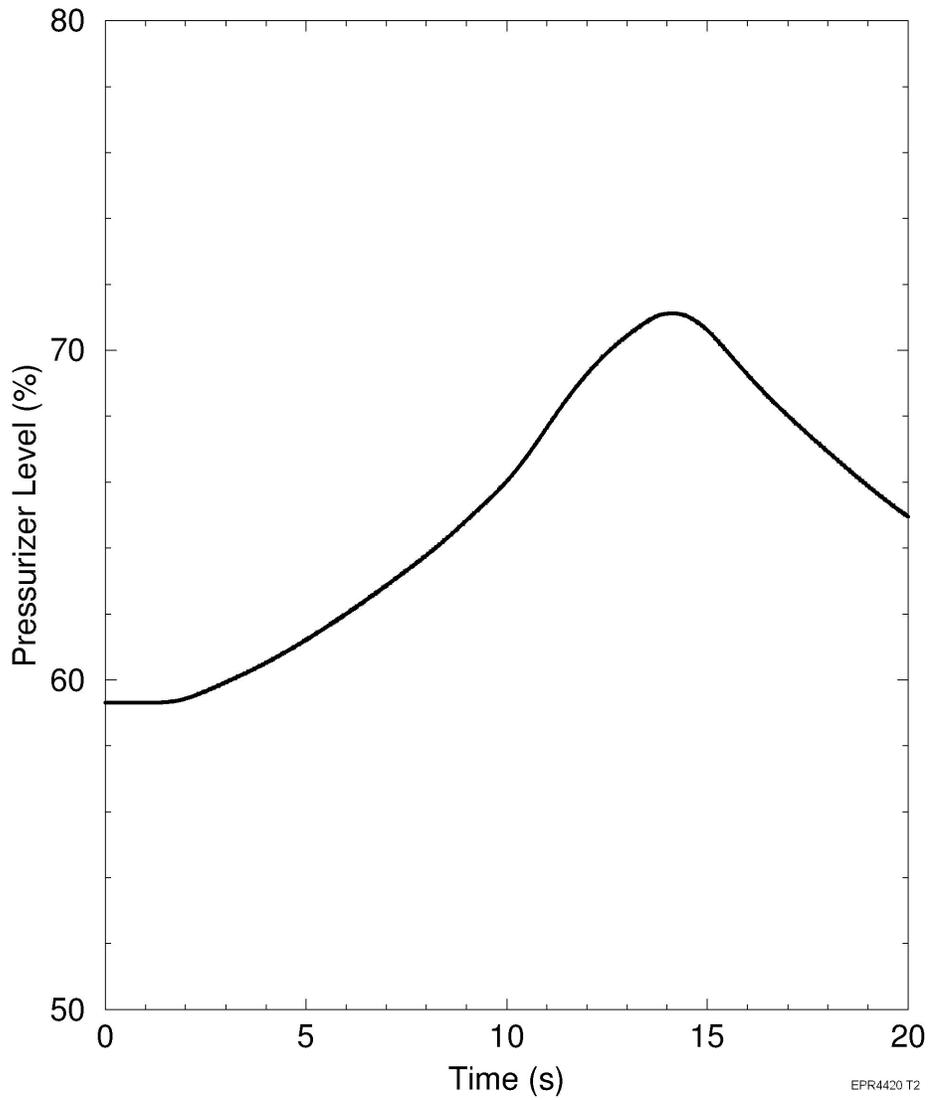


Figure 15.2-11—Turbine Trip RCS Overpressurization - PSRV Flow Rates

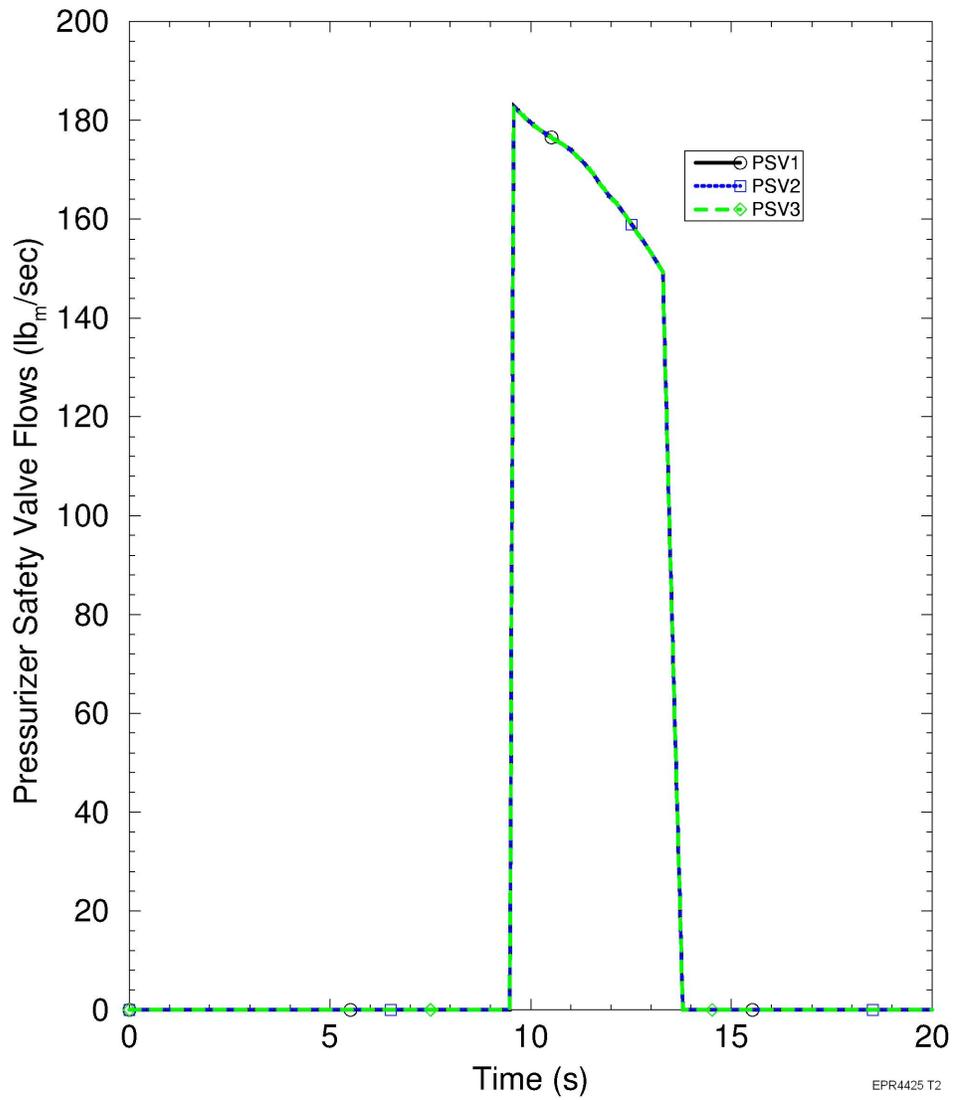


Figure 15.2-12—Turbine Trip RCS Overpressurization - MSRT Flow Rates

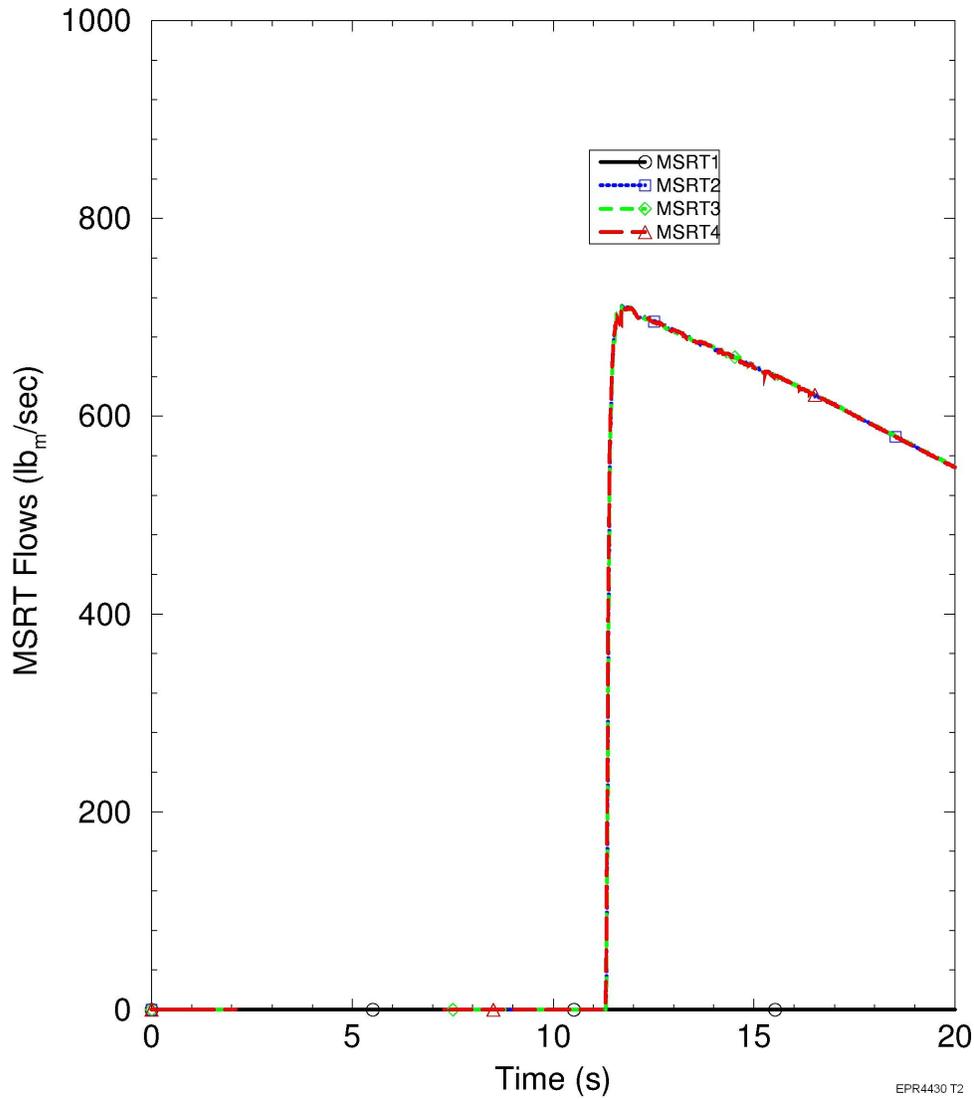


Figure 15.2-13—Turbine Trip RCS Overpressurization - MFW Flows

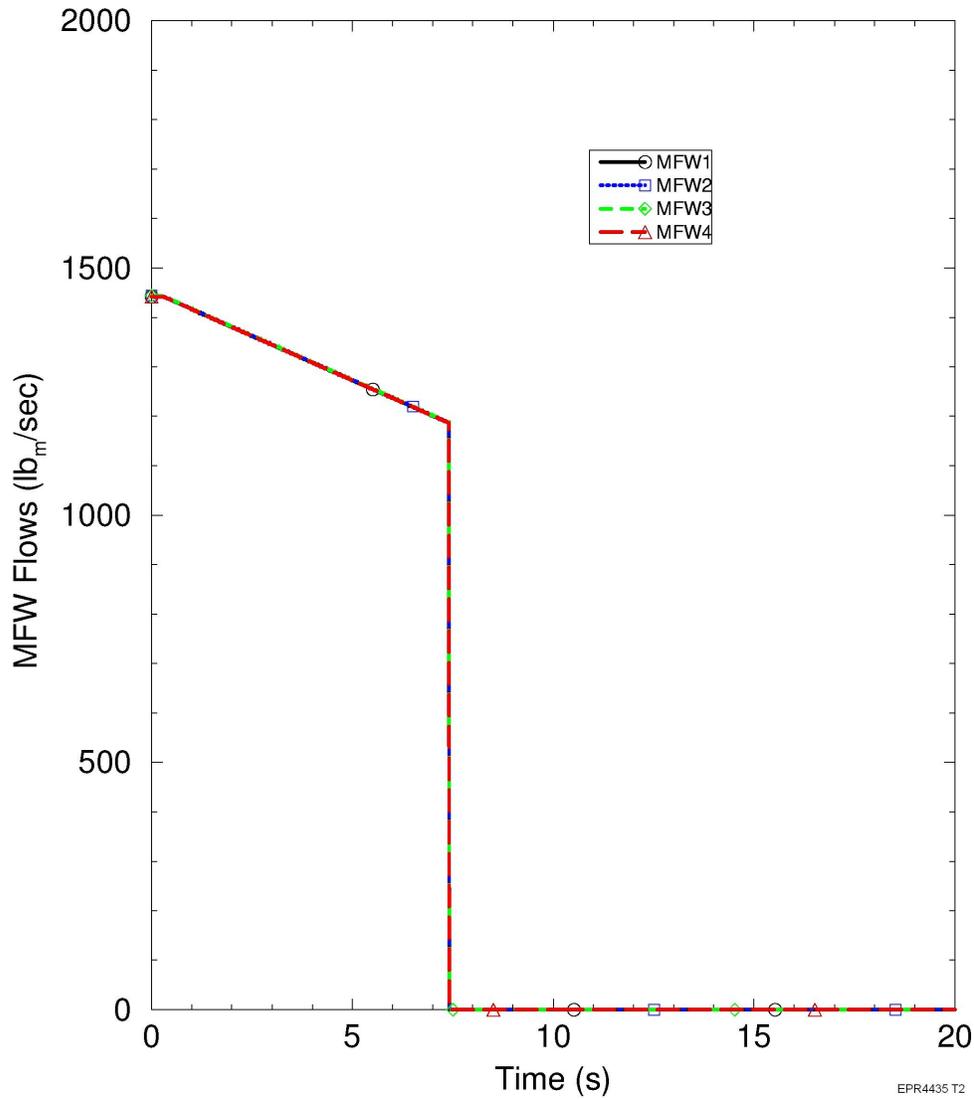


Figure 15.2-14—MSIVC Secondary Overpressurization—MSIV Flow Rates

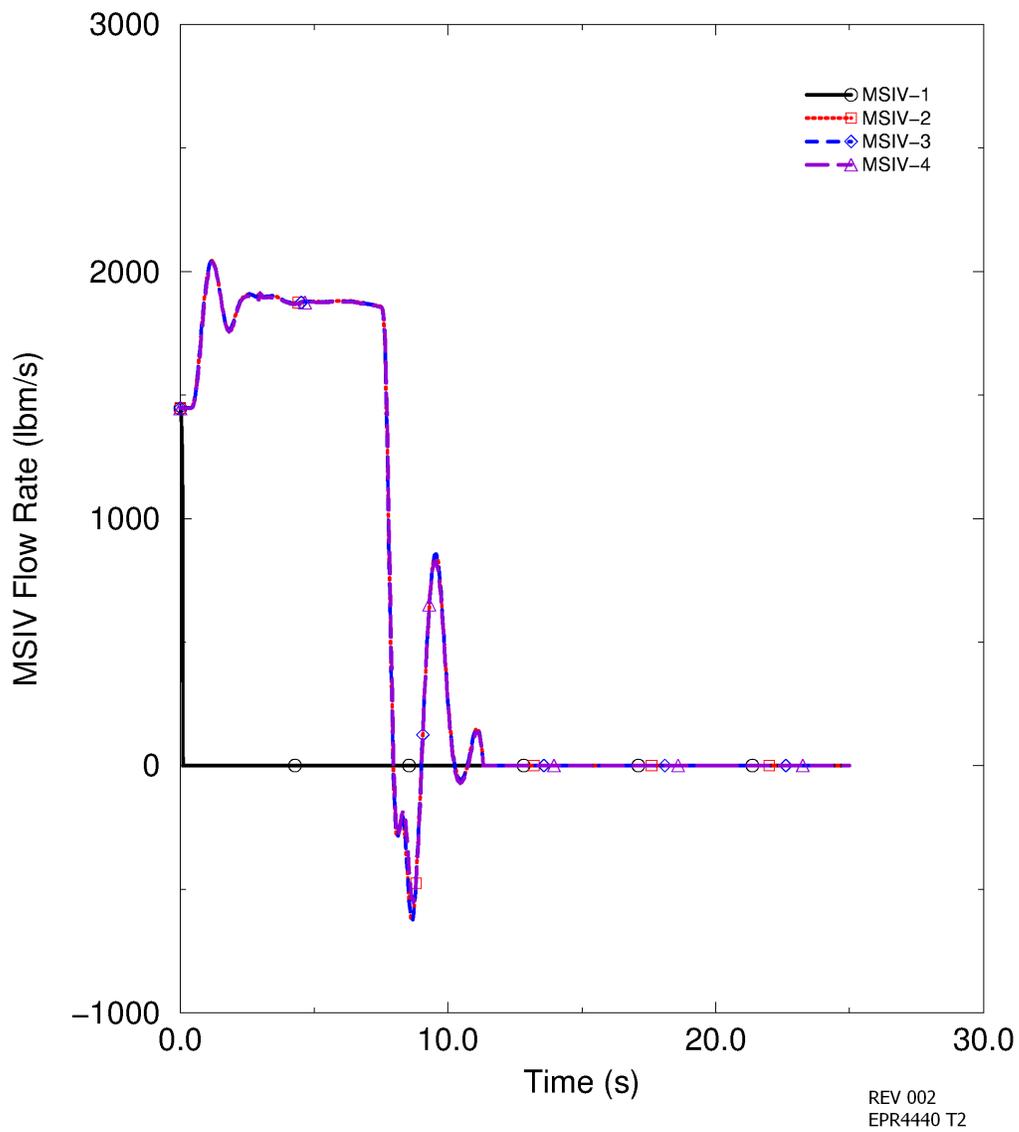
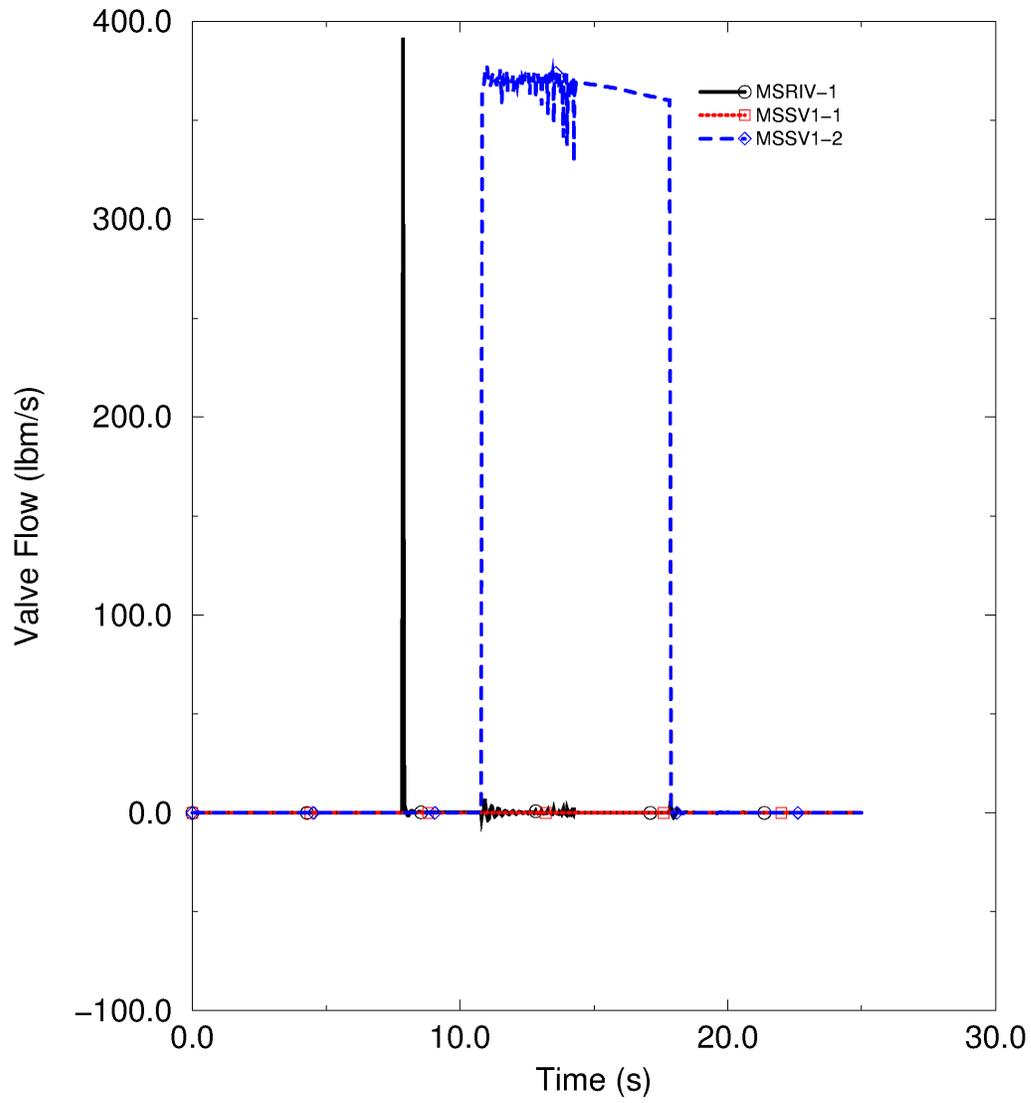
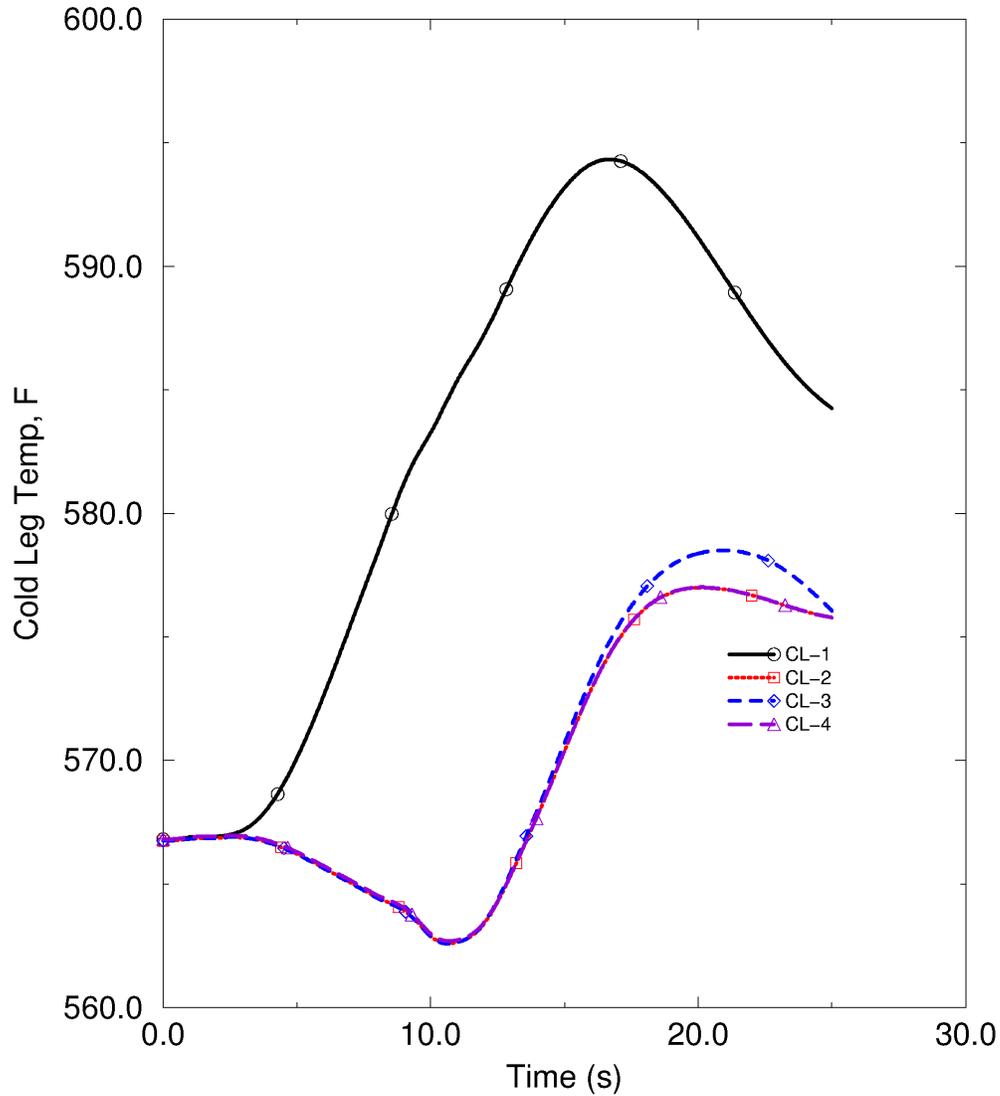


Figure 15.2-15—MSIVC Secondary Overpressurization - Safety Valve Flows for the Affected Loop



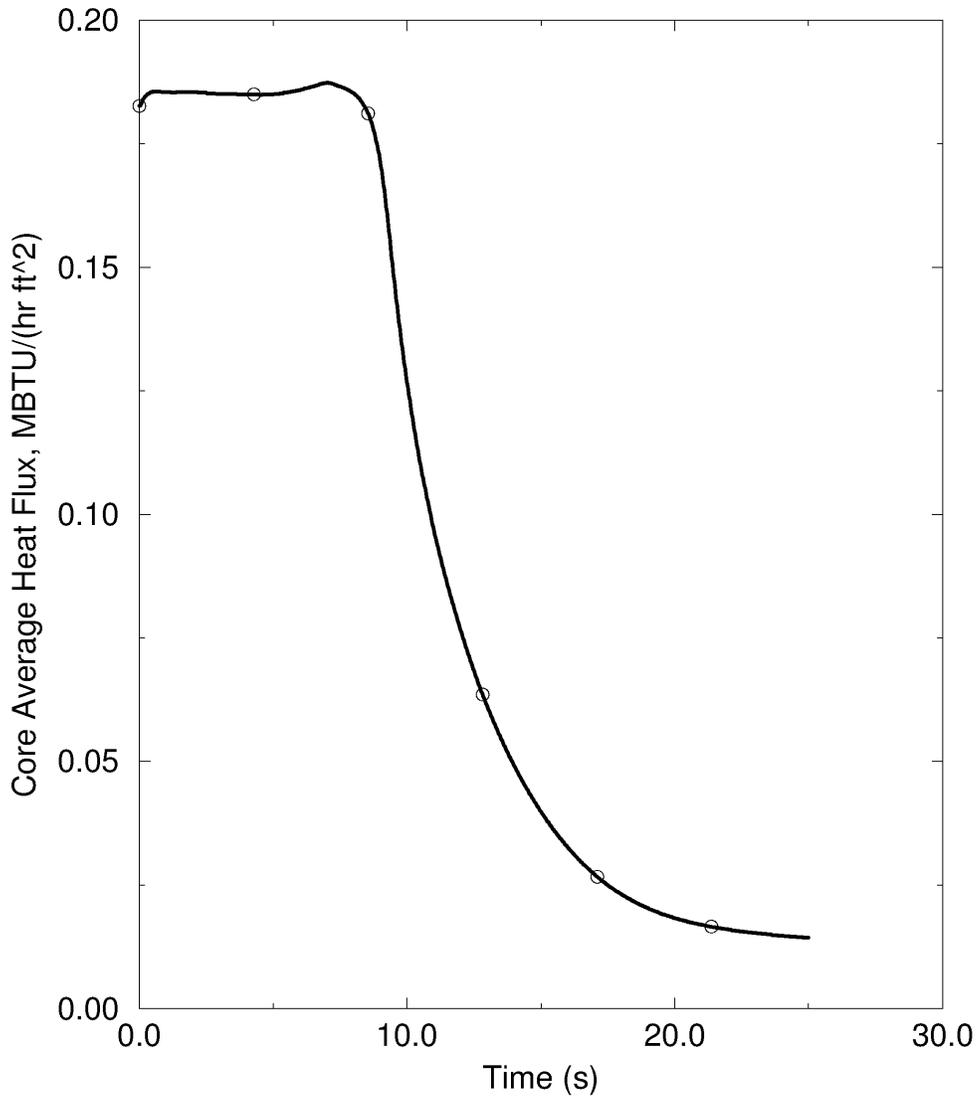
REV 002
EPR4445 T2

Figure 15.2-16—MSIVC Secondary Overpressurization—Cold Leg Temperatures



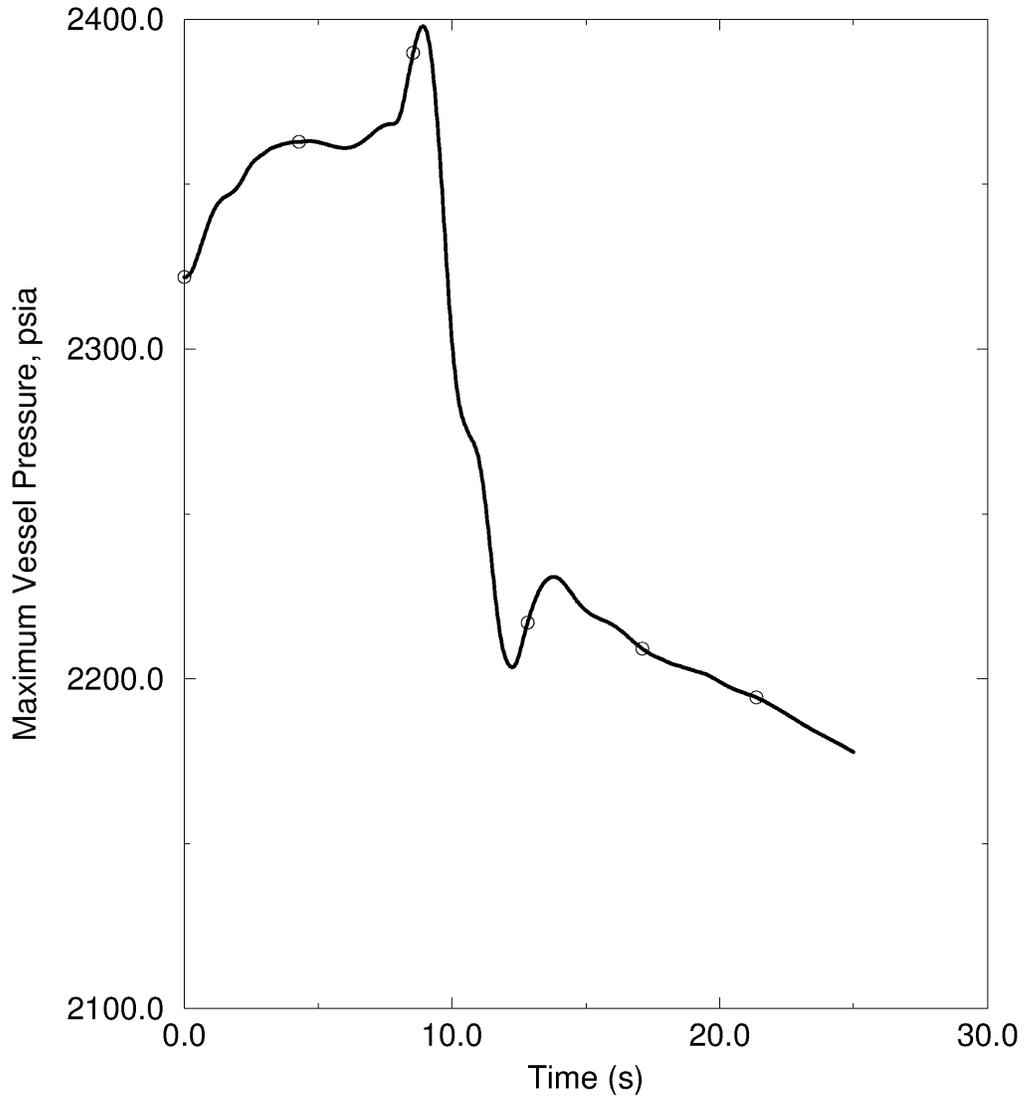
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EPR4450 T2

Figure 15.2-17—MSIVC Secondary Overpressurization — Average Core Heat Flux



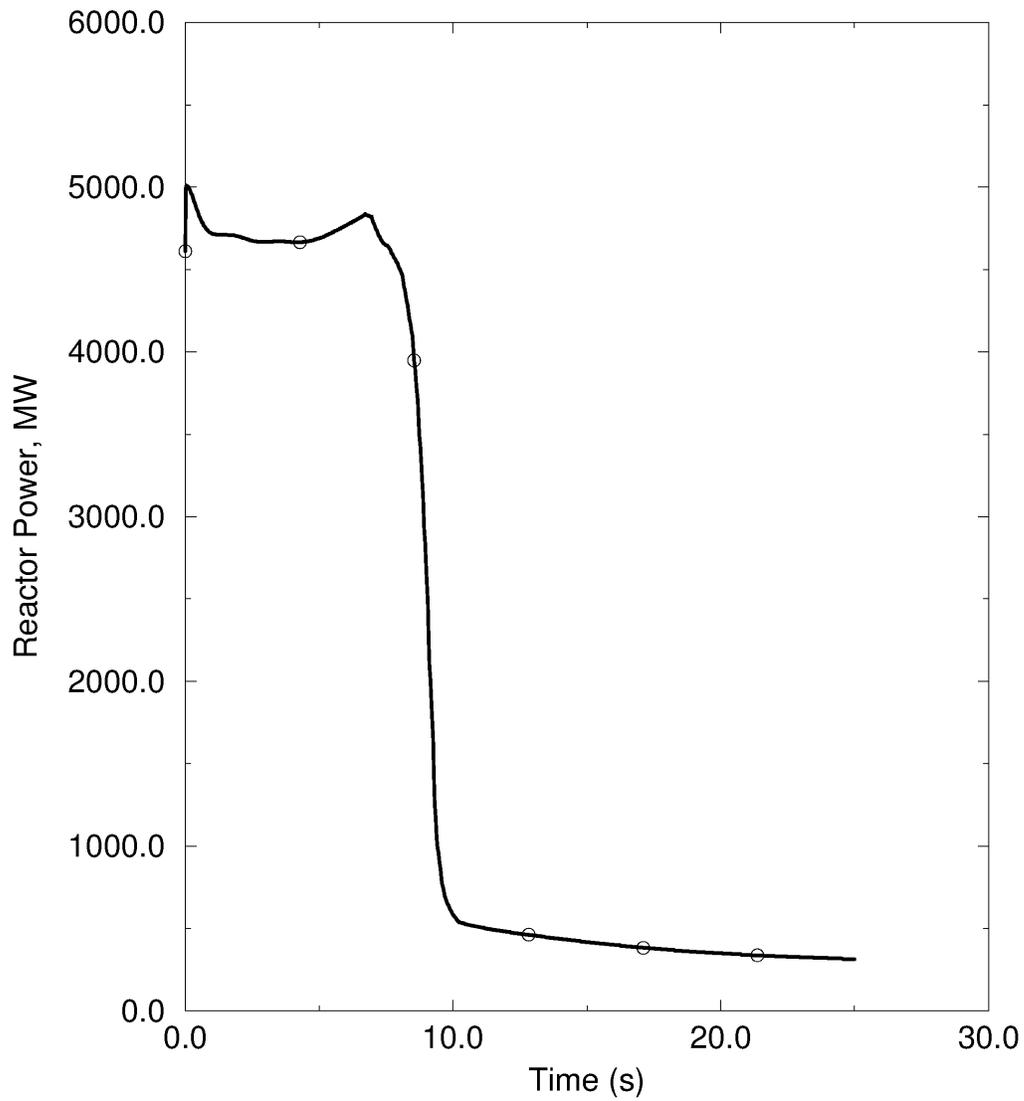
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EPR4455 T2

Figure 15.2-18—MSIVC Secondary Overpressurization — Maximum Primary Pressure



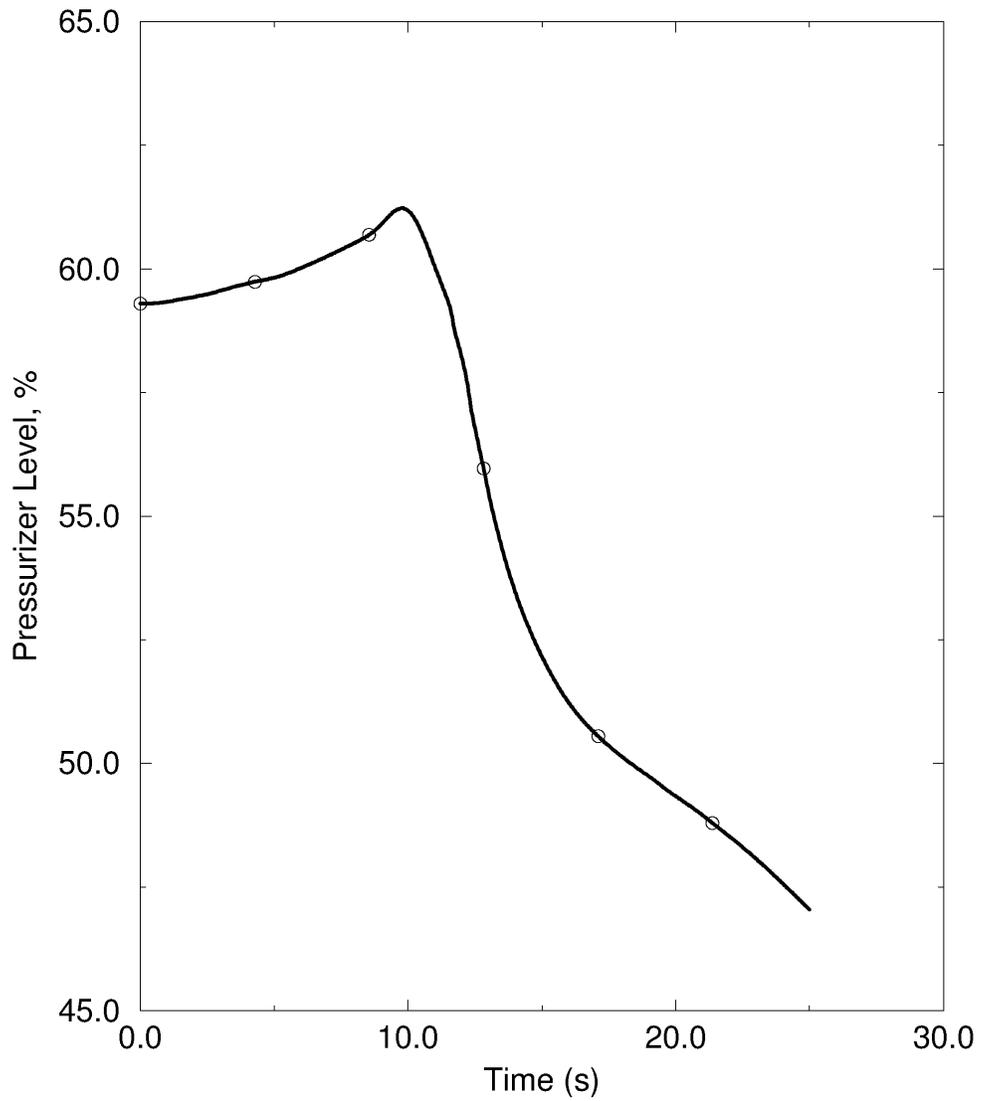
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EPR4460 T2

Figure 15.2-19—MSIVC Secondary Overpressurization — Reactor Power



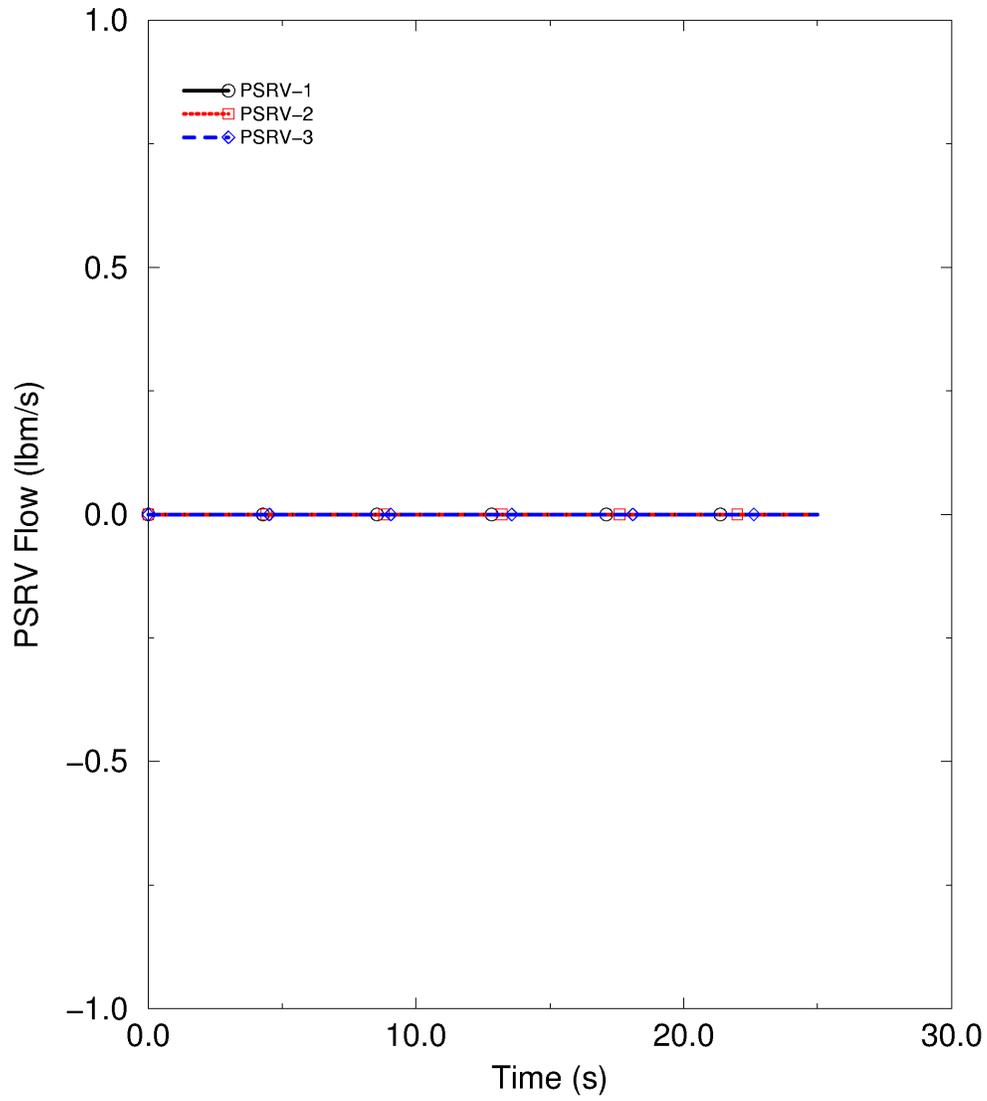
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Figure 15.2-20—MSIVC Secondary Overpressurization — Pressurizer Level



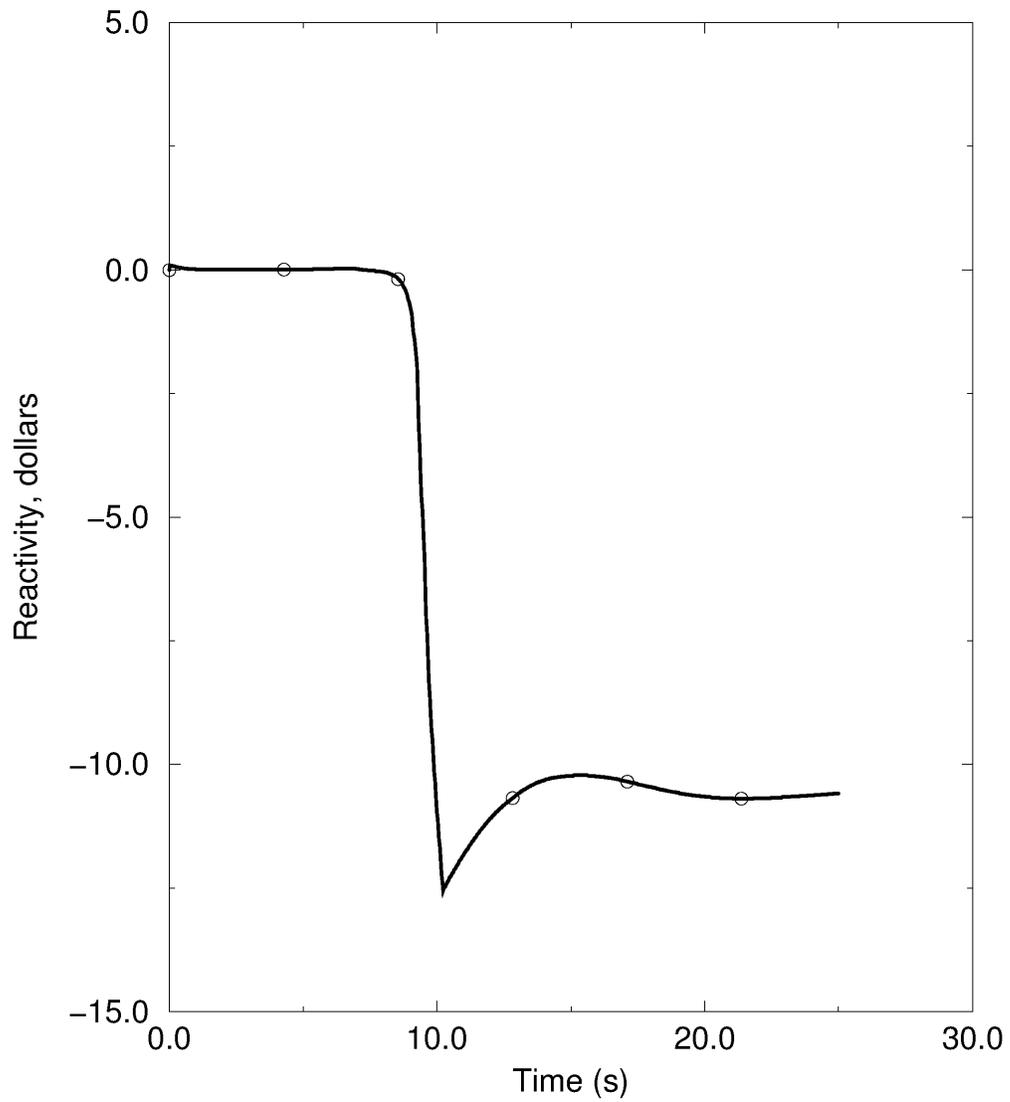
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Figure 15.2-21—MSIVC Secondary Overpressurization — Pressurizer Relief Valve Flows



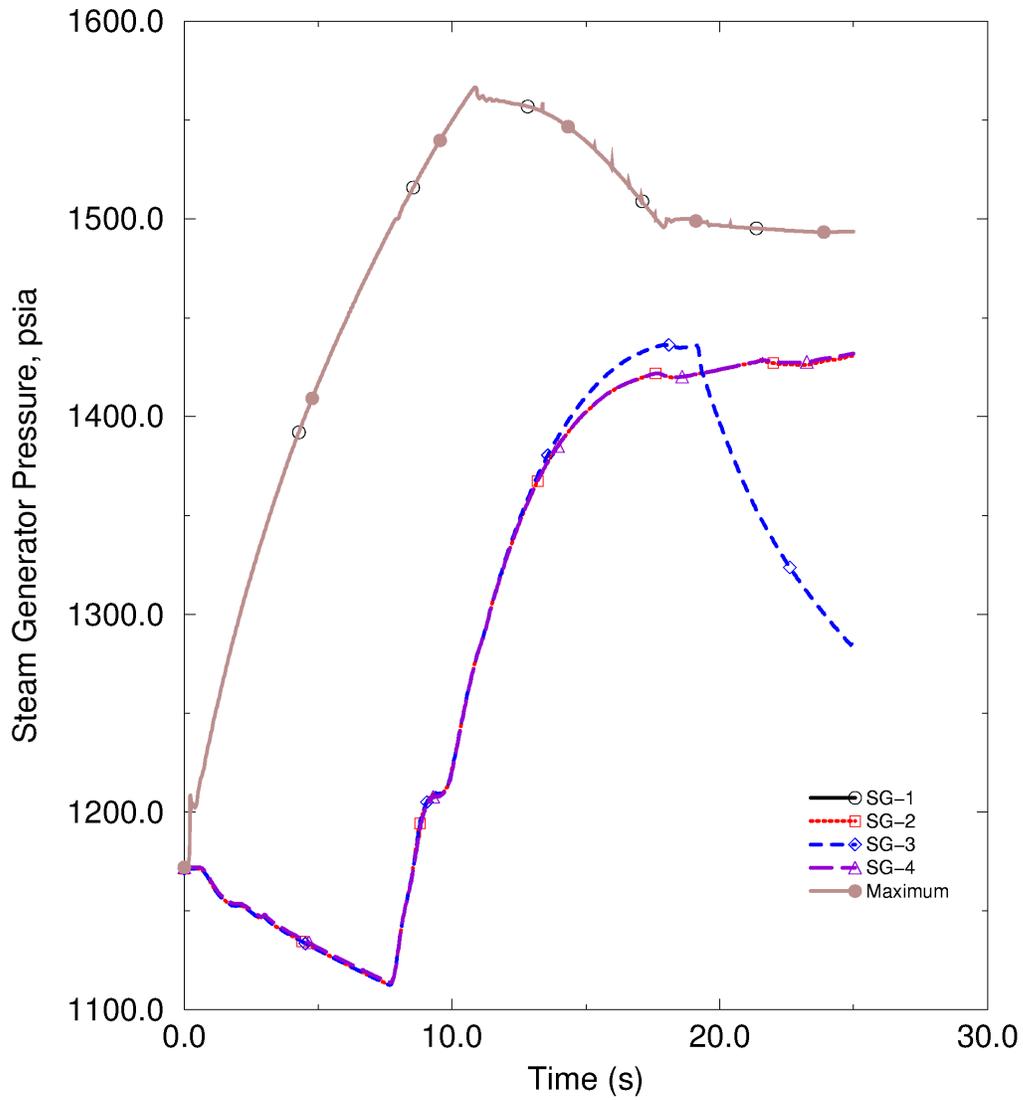
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Figure 15.2-22—MSIVC Secondary Overpressurization—Total Reactivity



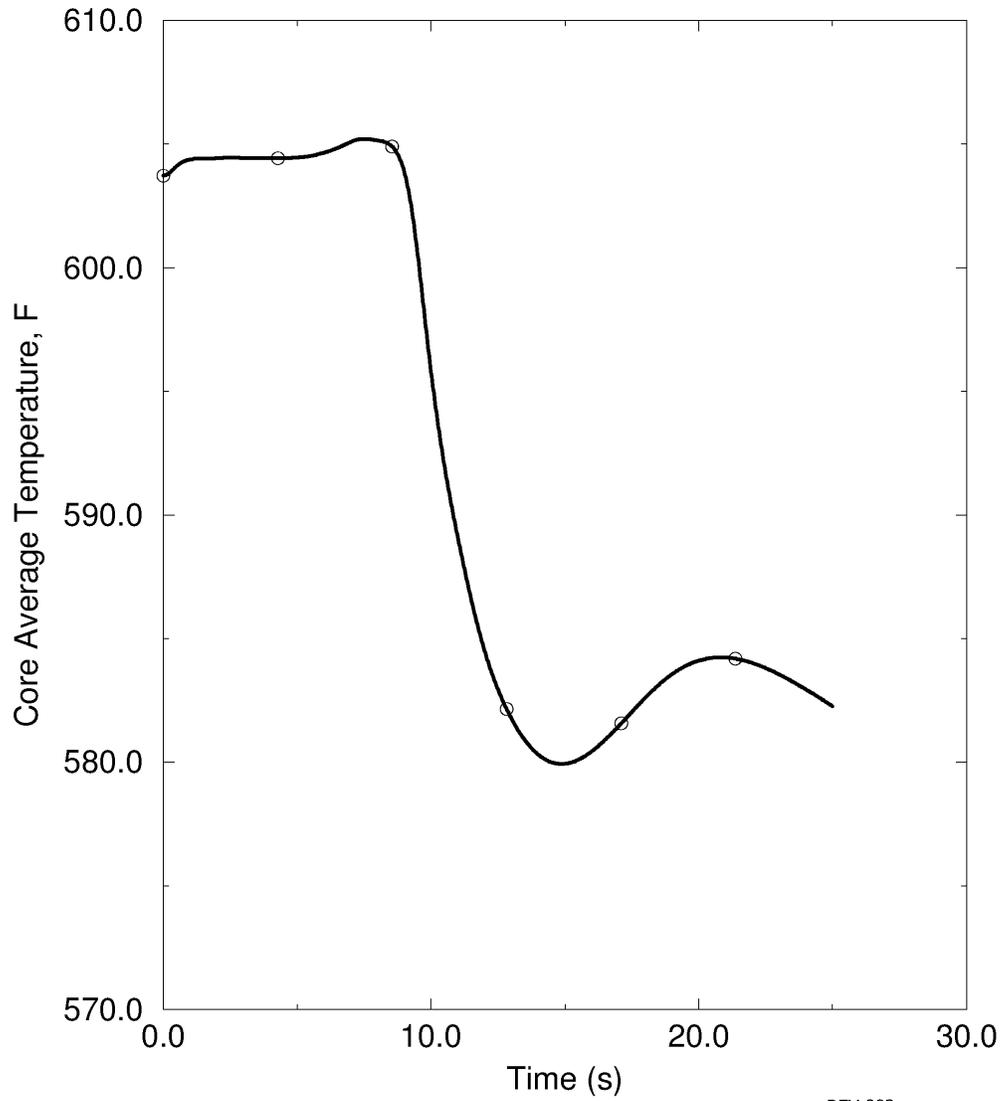
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EPR4480 T2

Figure 15.2-23—MSIVC Secondary Overpressurization — Maximum Secondary Pressures



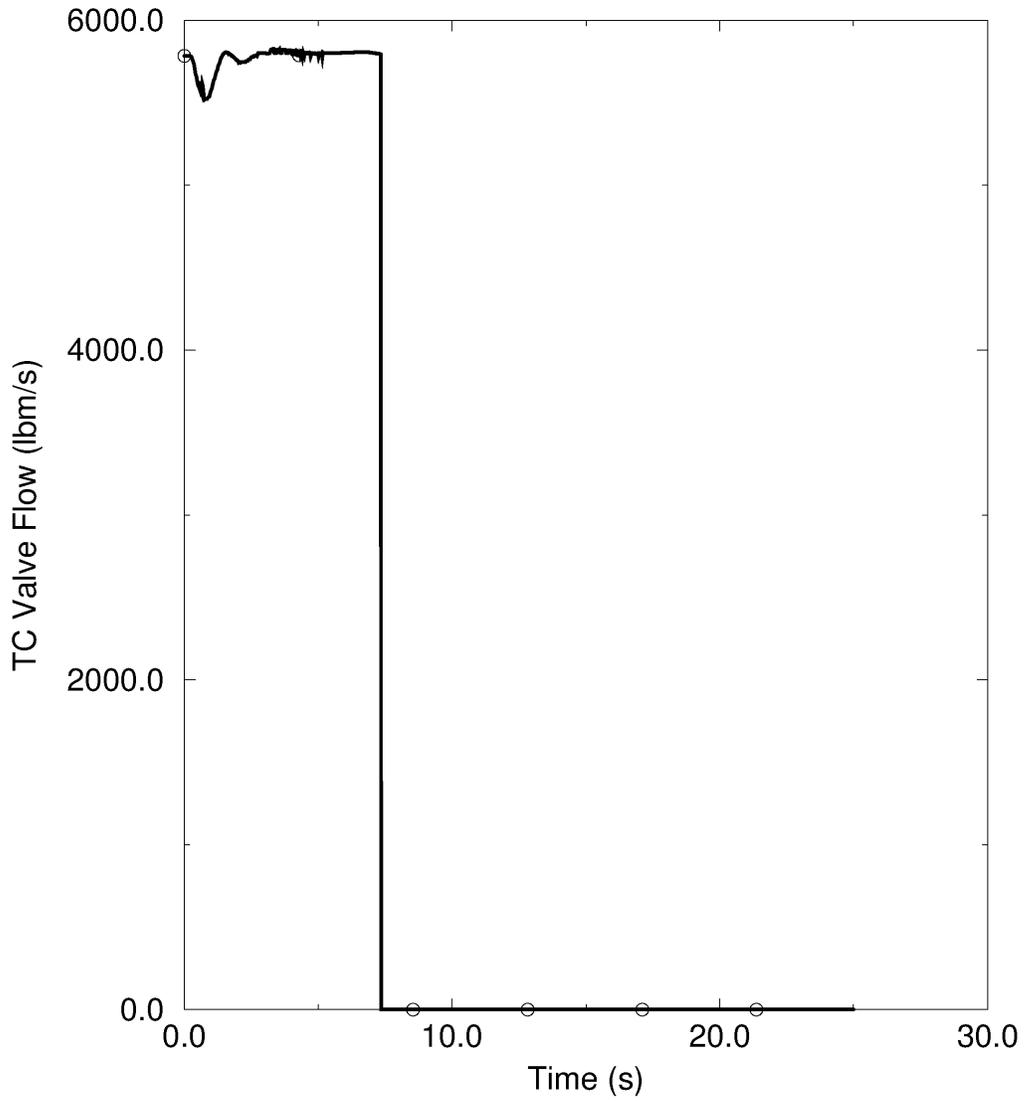
REV 002
EPR4485 T2

Figure 15.2-24—MSIVC Secondary Overpressurization — Average Core Fluid Temperature



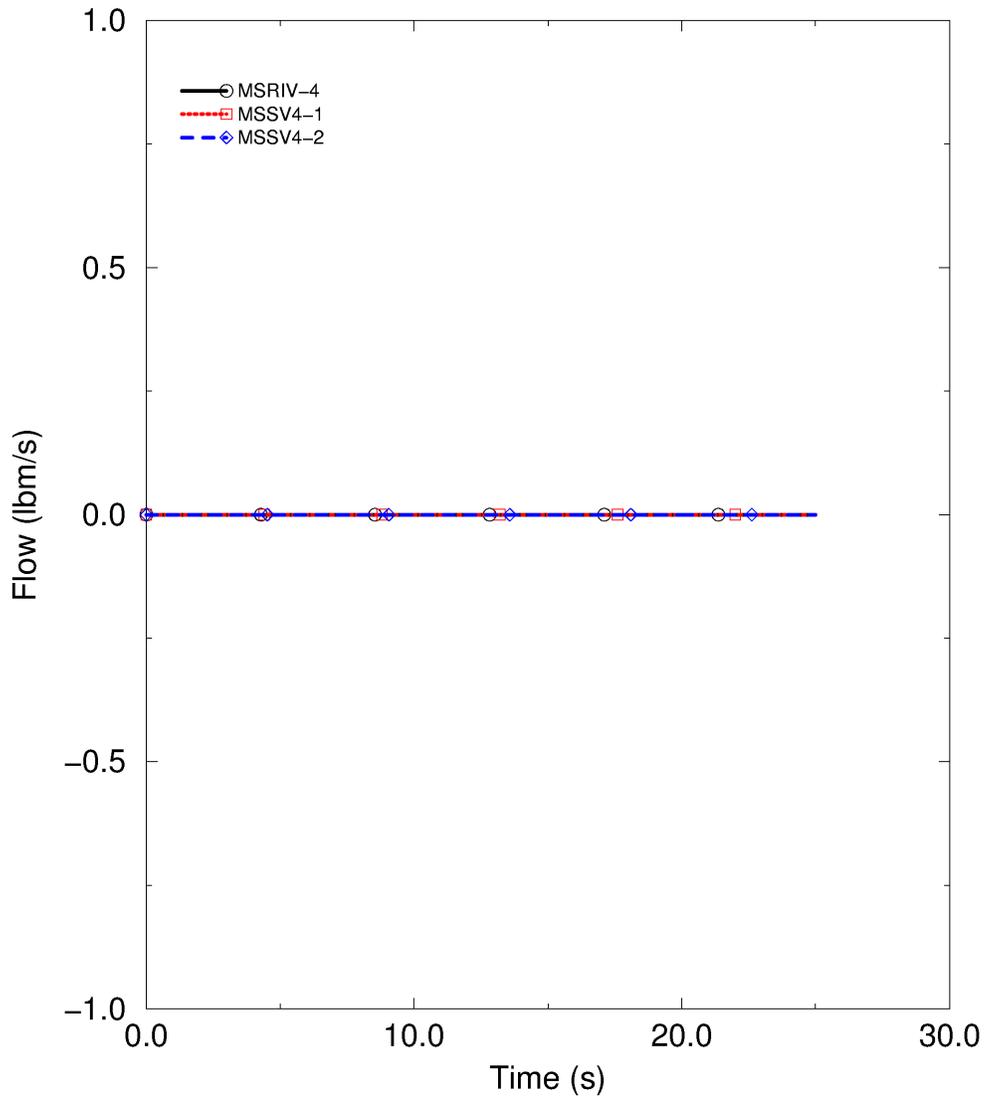
REV 002
EPR4490 T2

Figure 15.2-25—MSIVC Secondary Overpressurization — Turbine Control Valve Flow



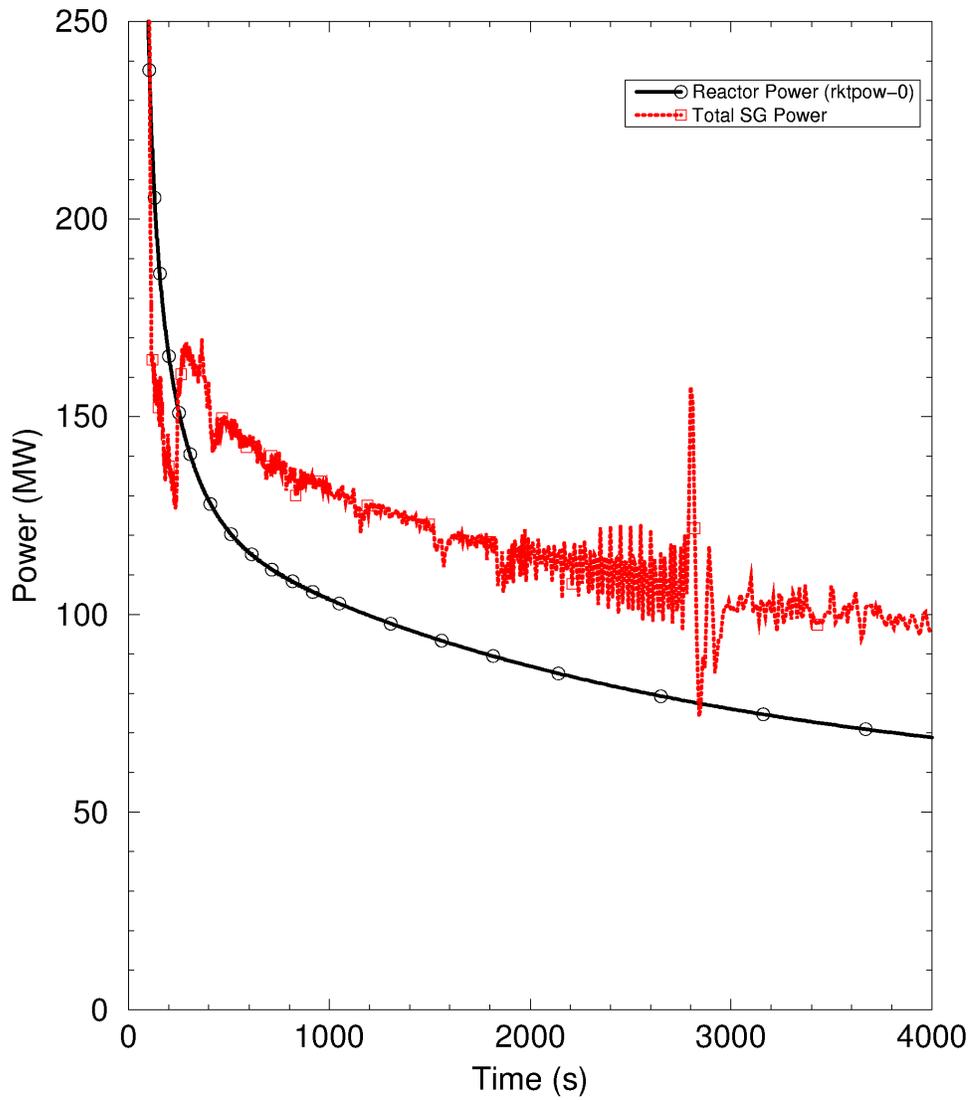
REV 002
EPR4495 T2

Figure 15.2-26—MSIVC Secondary Overpressurization — Safety Valve Flows for an Unaffected Loop



REV 002
EPR 4500 T2

Figure 15.2-27—Loss of Normal Feedwater - Reactor and Total SG Power



REV 002
EPR4505

Figure 15.2-28—Loss of Normal Feedwater - SG Power

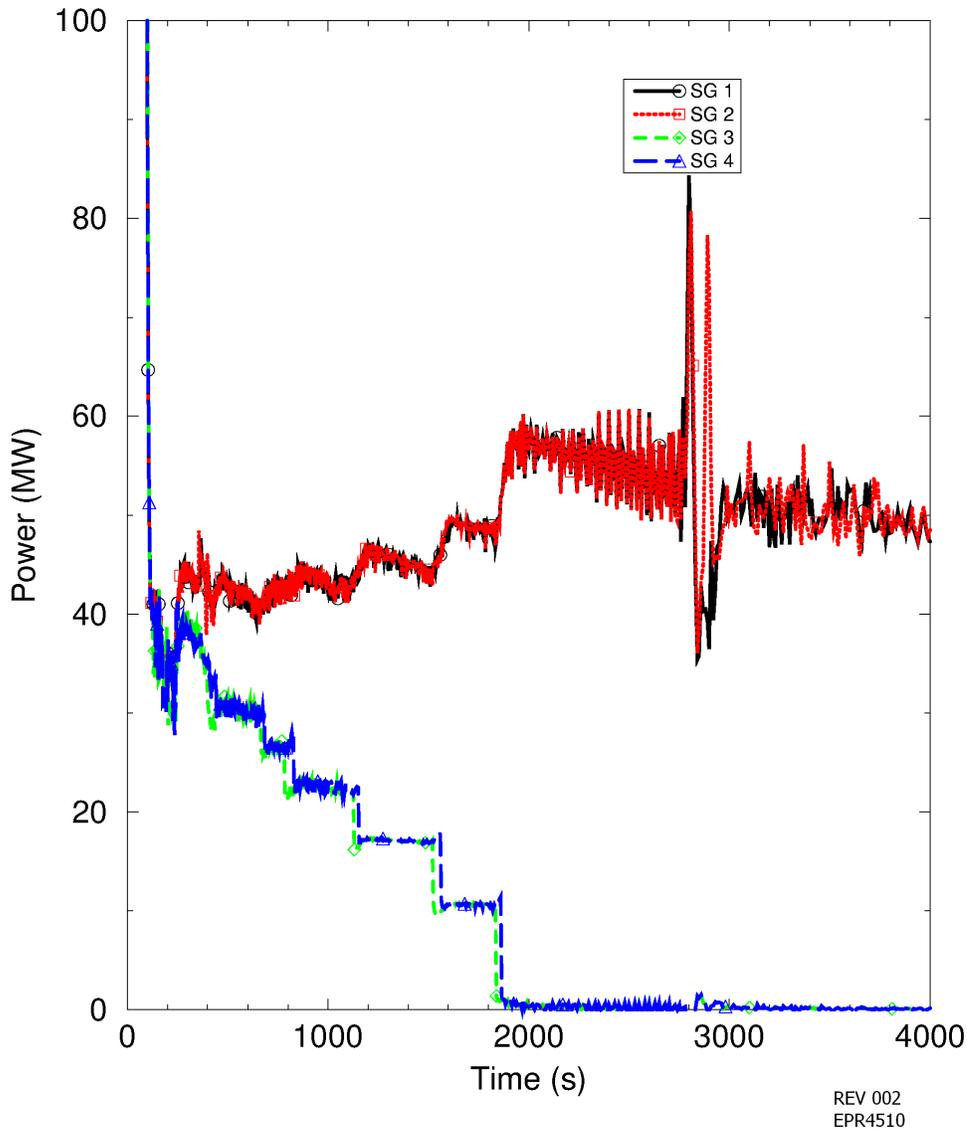
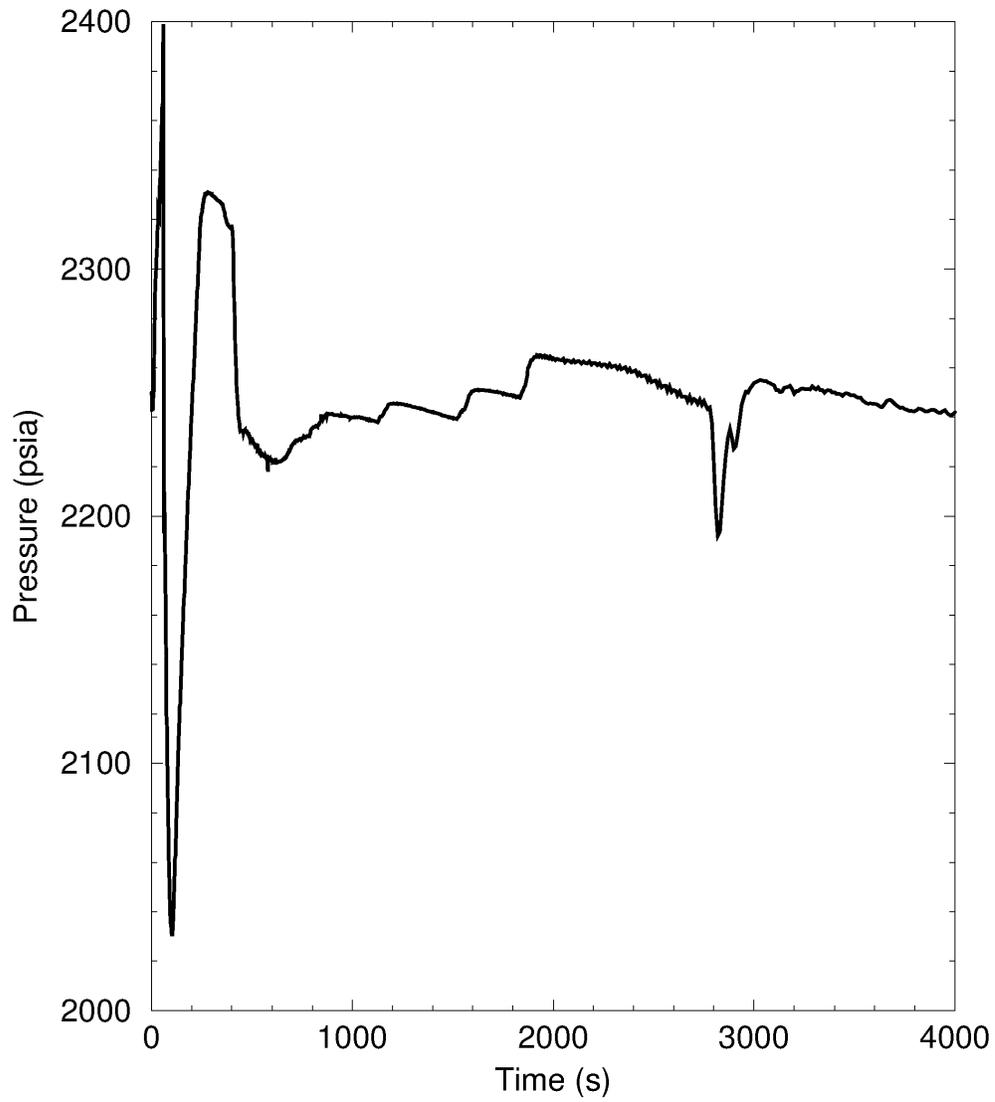
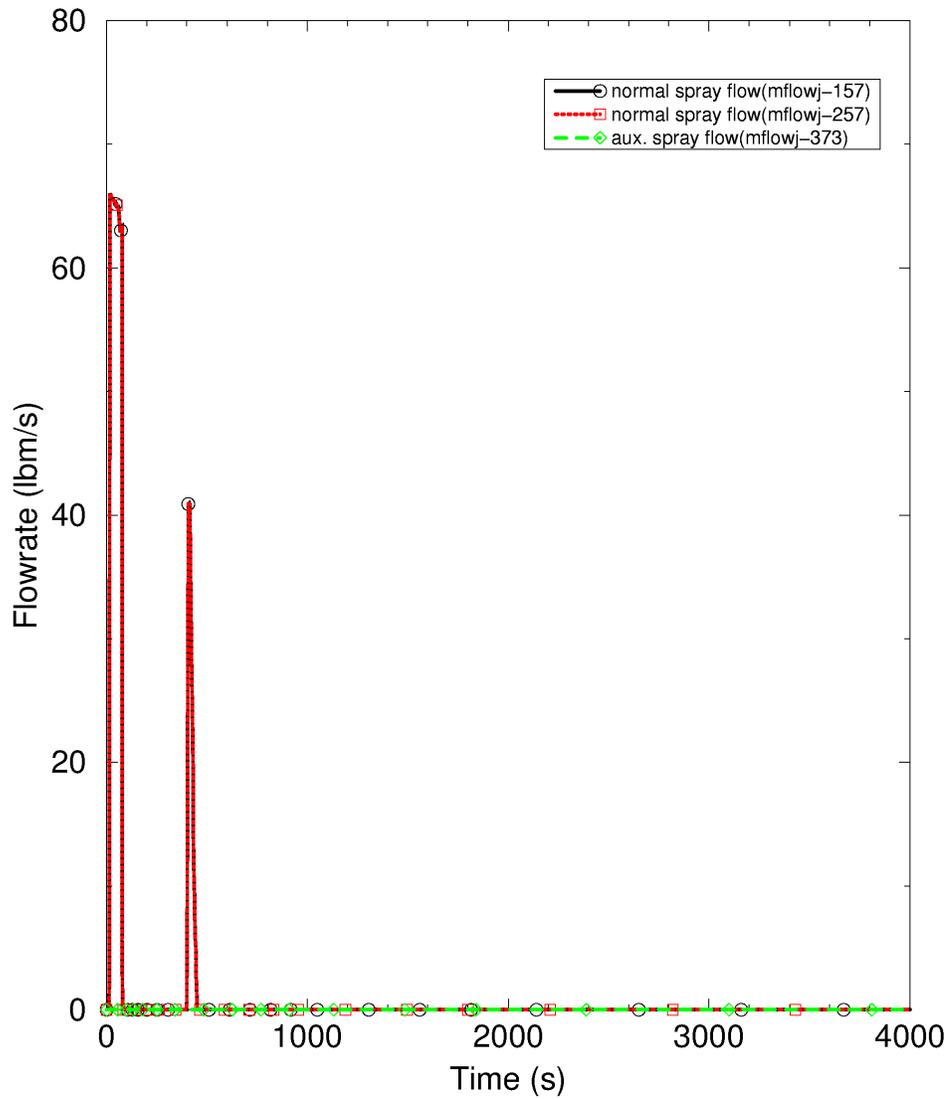


Figure 15.2-29—Loss of Normal Feedwater - PZR Pressure



REV 002
EPR4515

Figure 15.2-30—Loss of Normal Feedwater - PZR Spray Flow



REV 002
EPR4520

Figure 15.2-31—Loss of Normal Feedwater - PSRV Flow

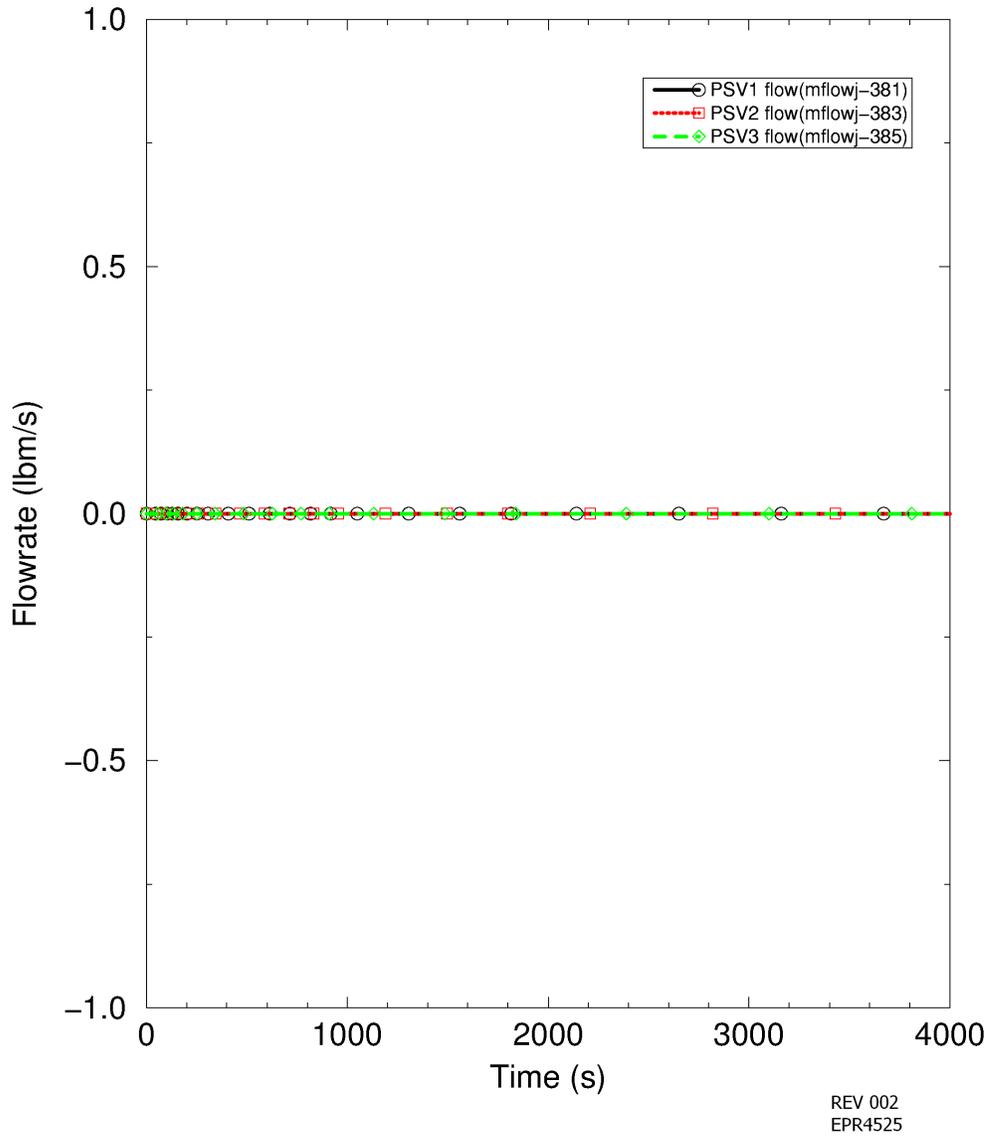
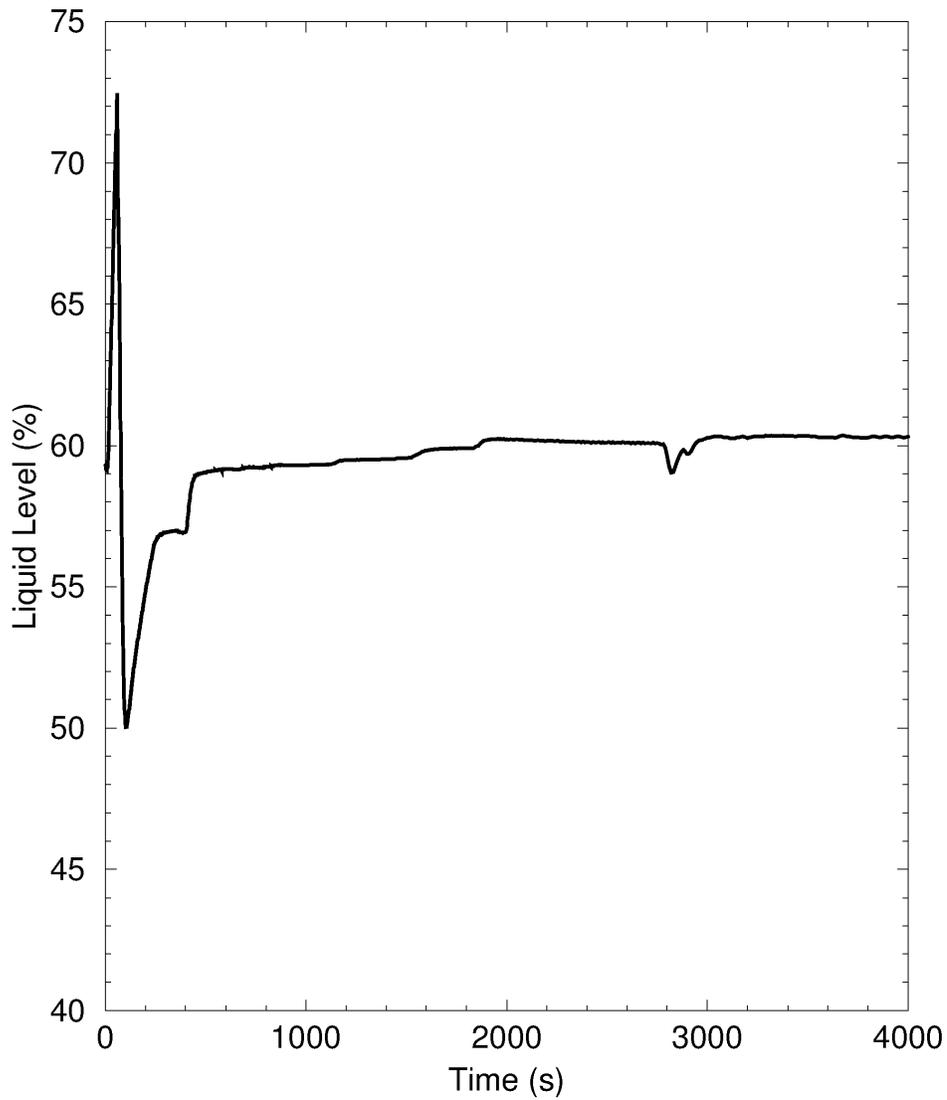
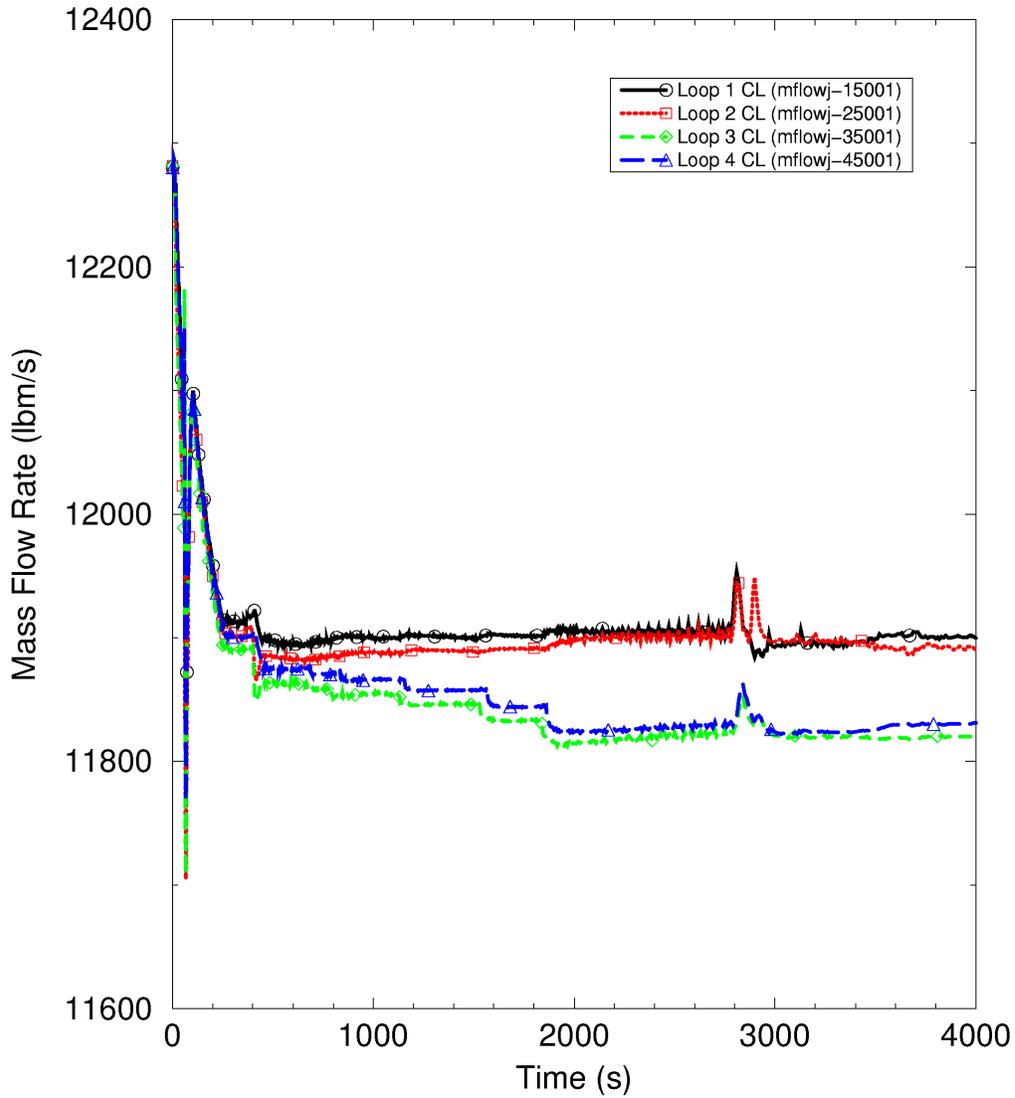


Figure 15.2-32—Loss of Normal Feedwater - PZR Liquid Level



REV 002
EPR4530

Figure 15.2-33—Loss of Normal Feedwater - RCS Loop Mass Flow



REV 002
EPR4535

Figure 15.2-34—Loss of Normal Feedwater - RCS Temperatures

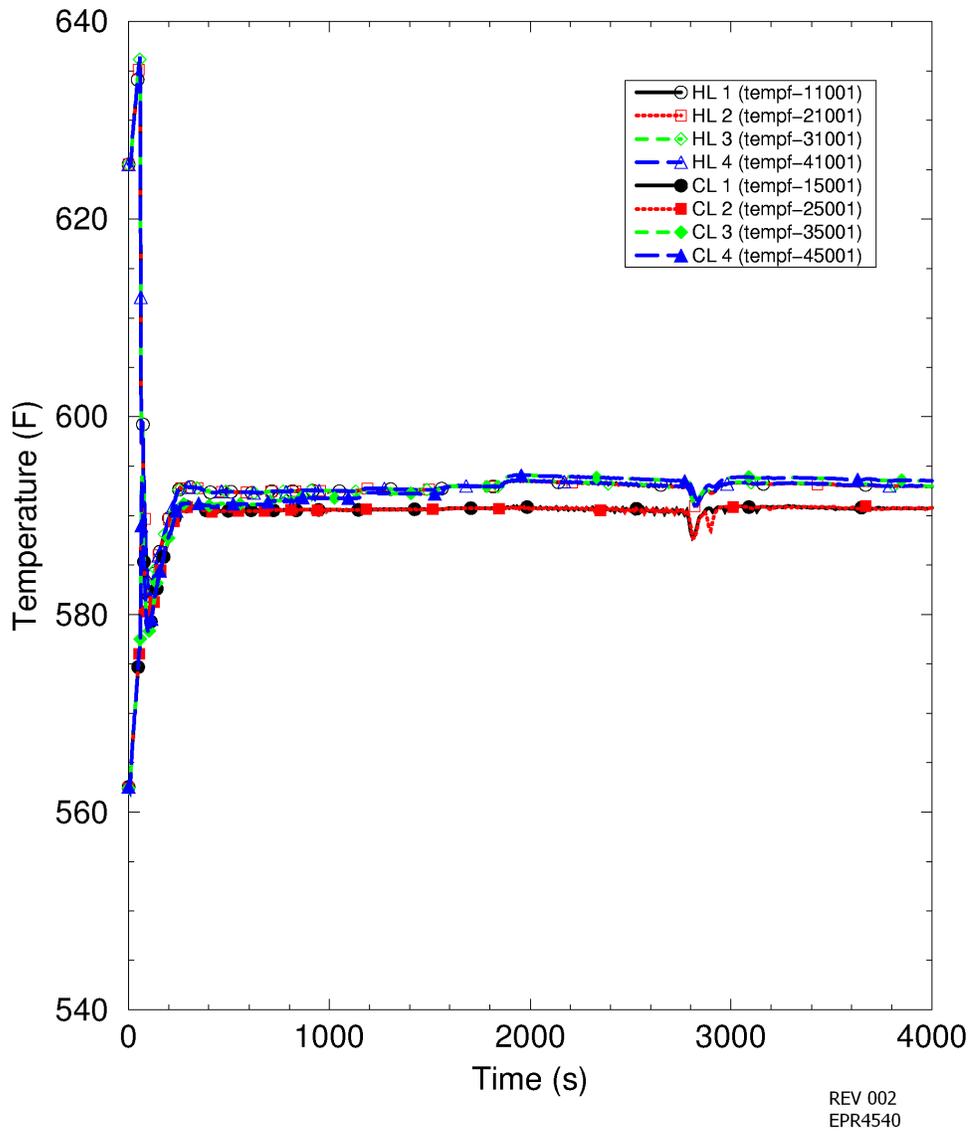
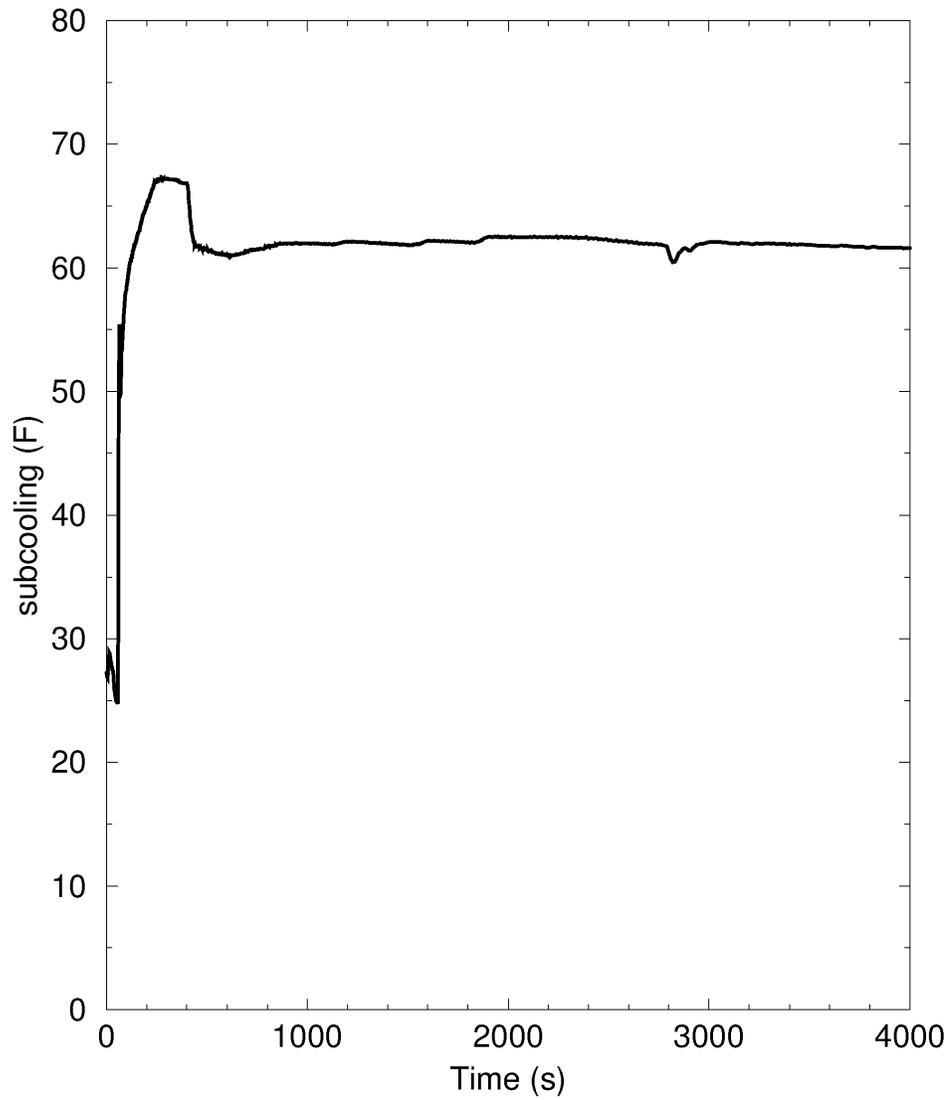


Figure 15.2-35—Loss of Normal Feedwater - Core Exit Subcooling



REV 002
EPR4545

Figure 15.2-36—Loss of Normal Feedwater - Core Flow Rate

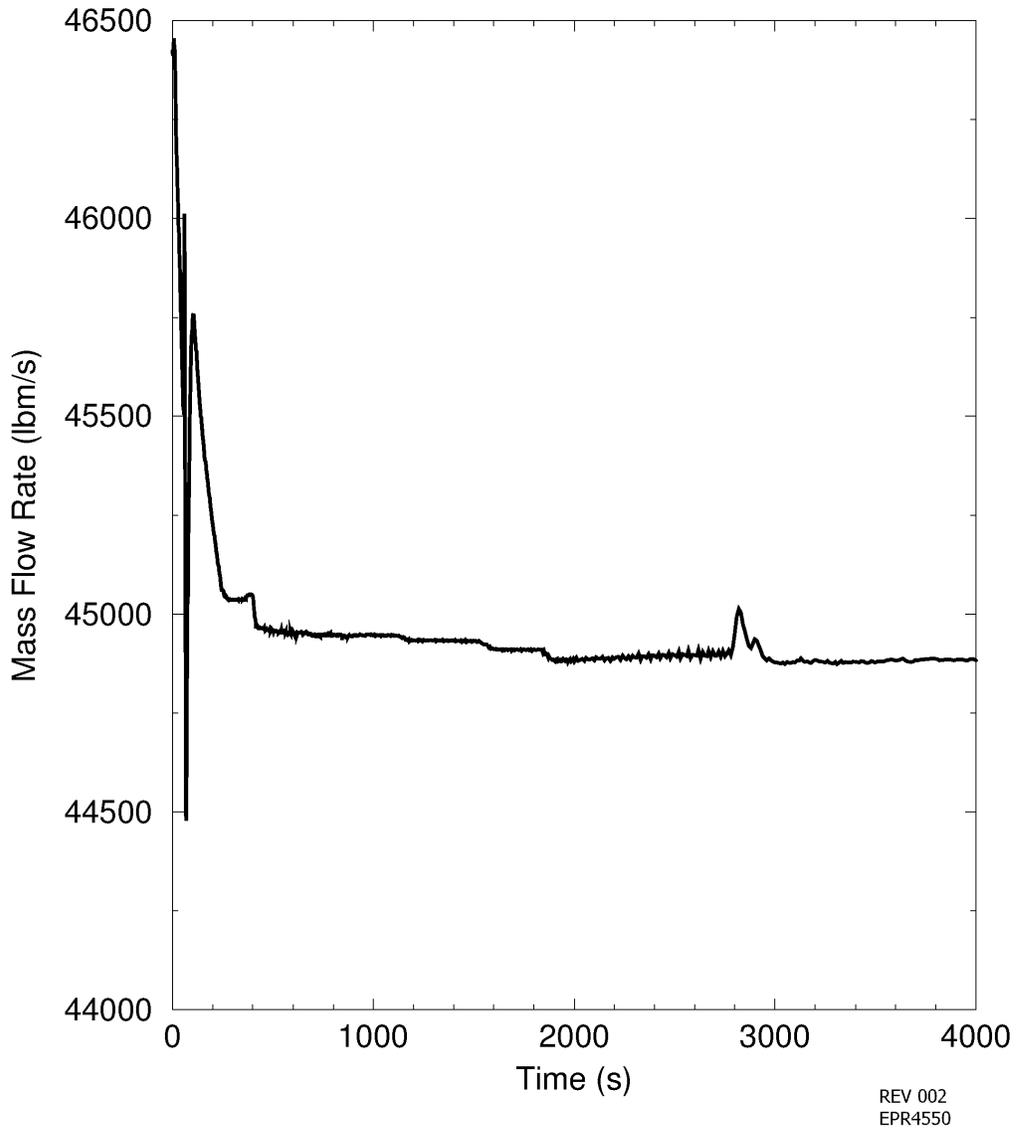
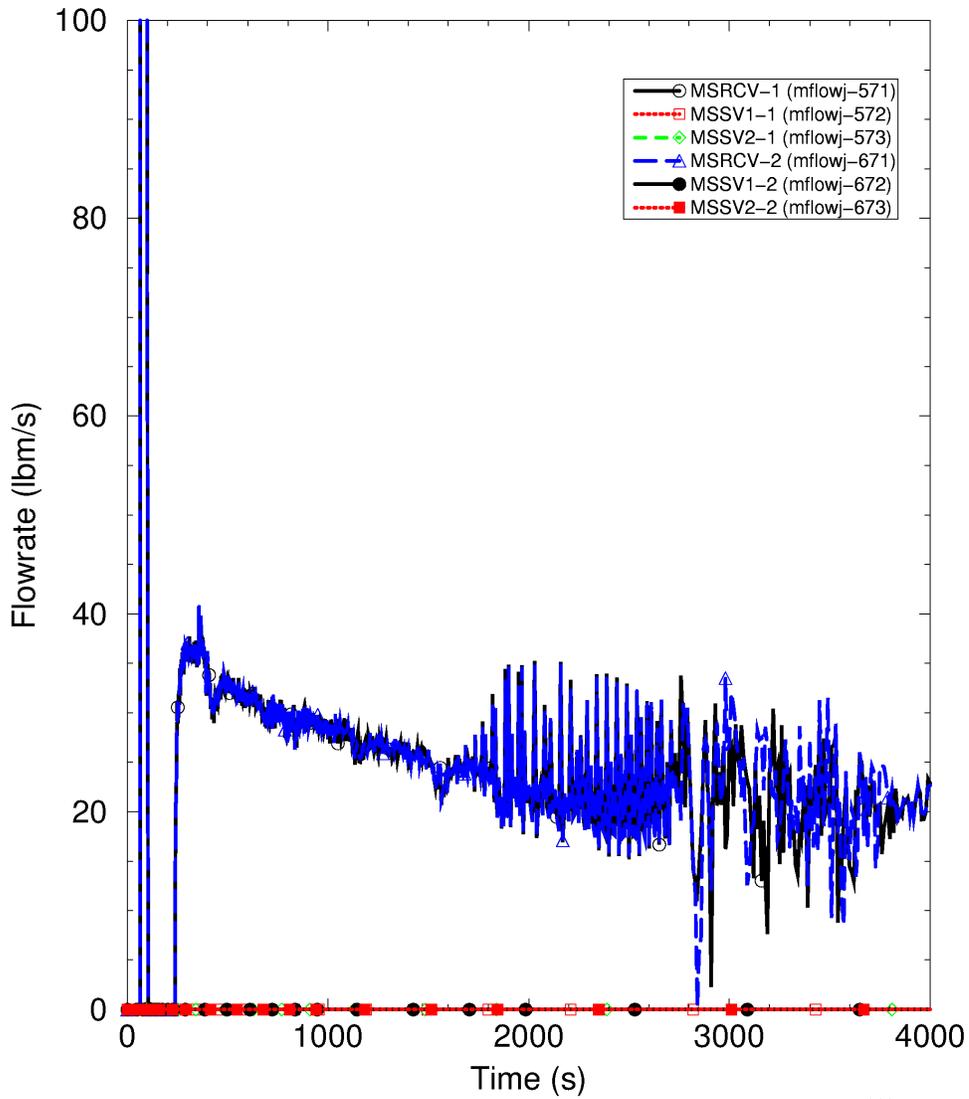
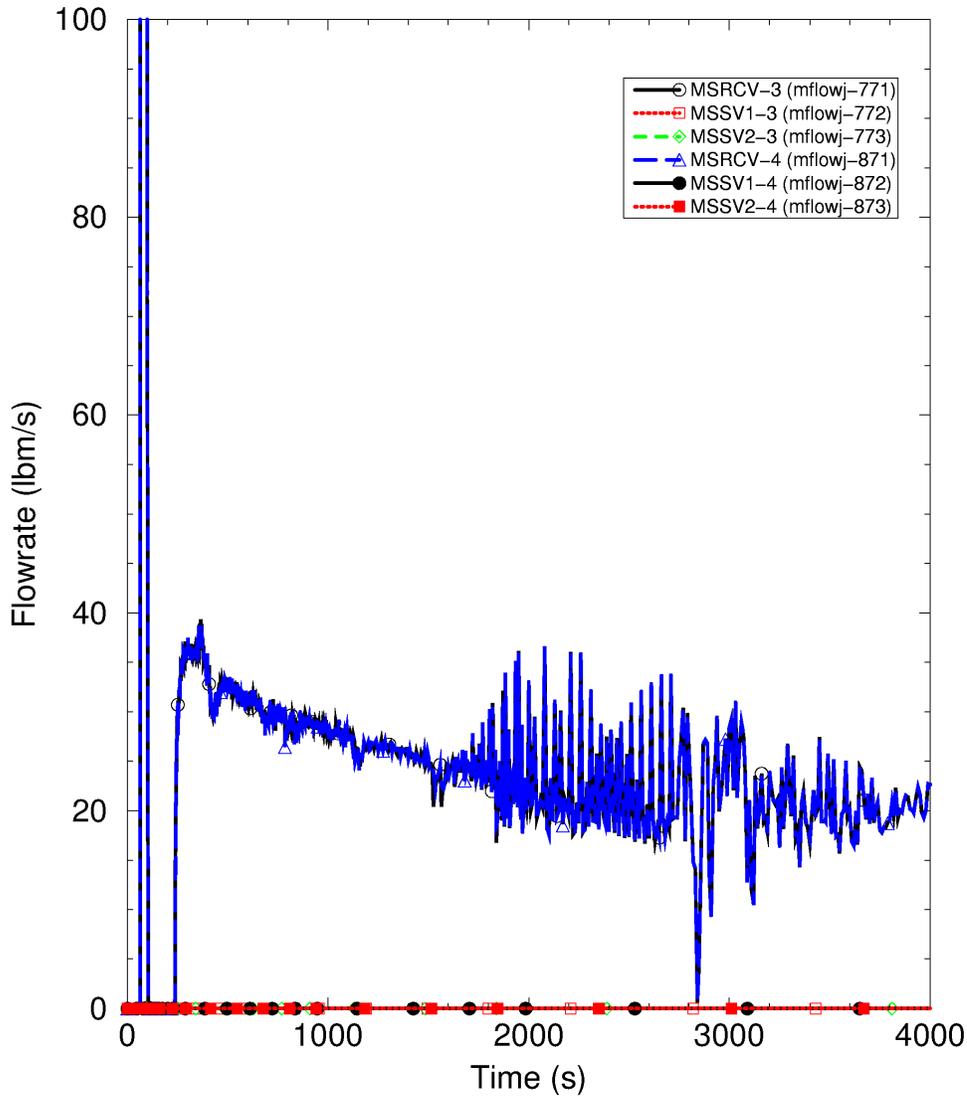


Figure 15.2-37—Loss of Normal Feedwater - Flow Rates through Main Steam Relief Train, Unaffected SGs



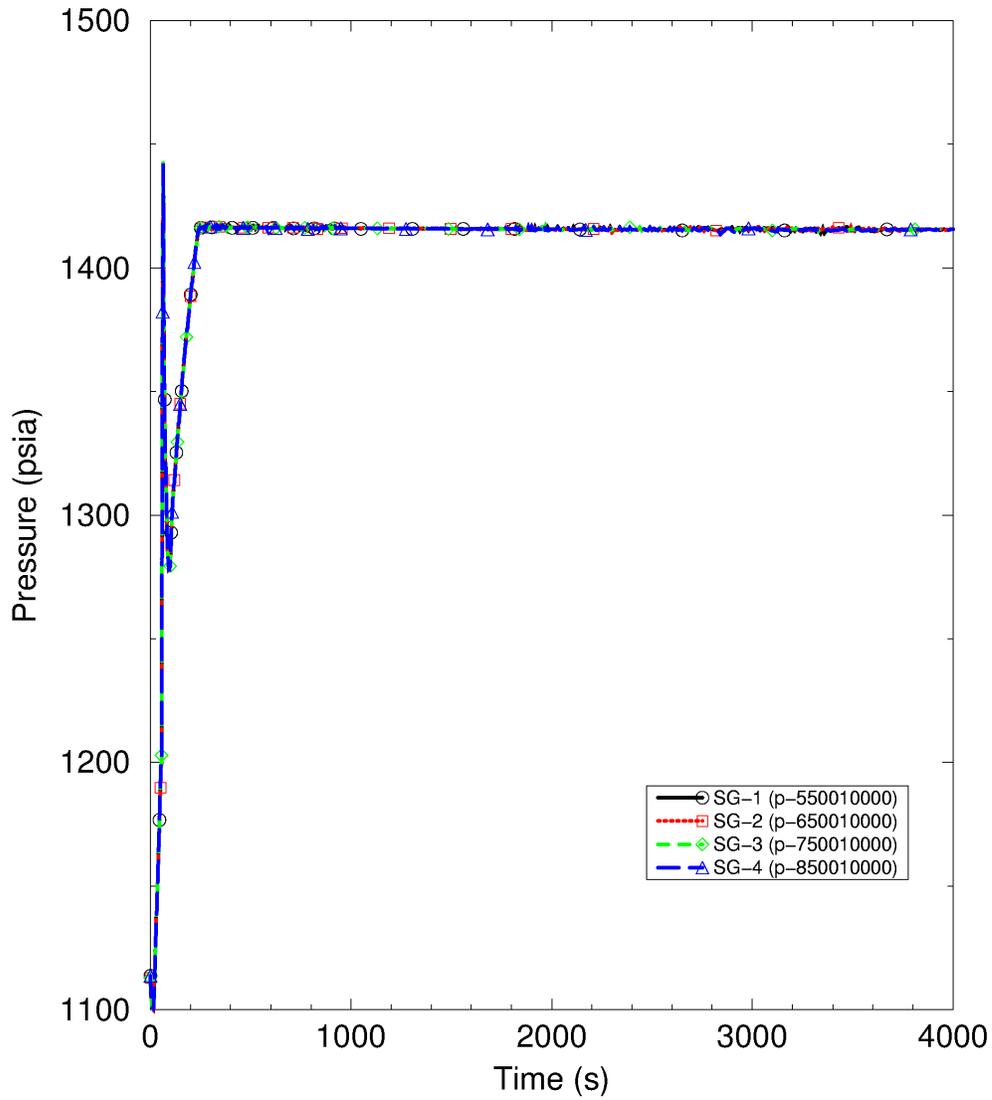
REV 002
EPR4555

Figure 15.2-38—Loss of Normal Feedwater - Flow Rates through Main Steam Relief Train, Affected SGs



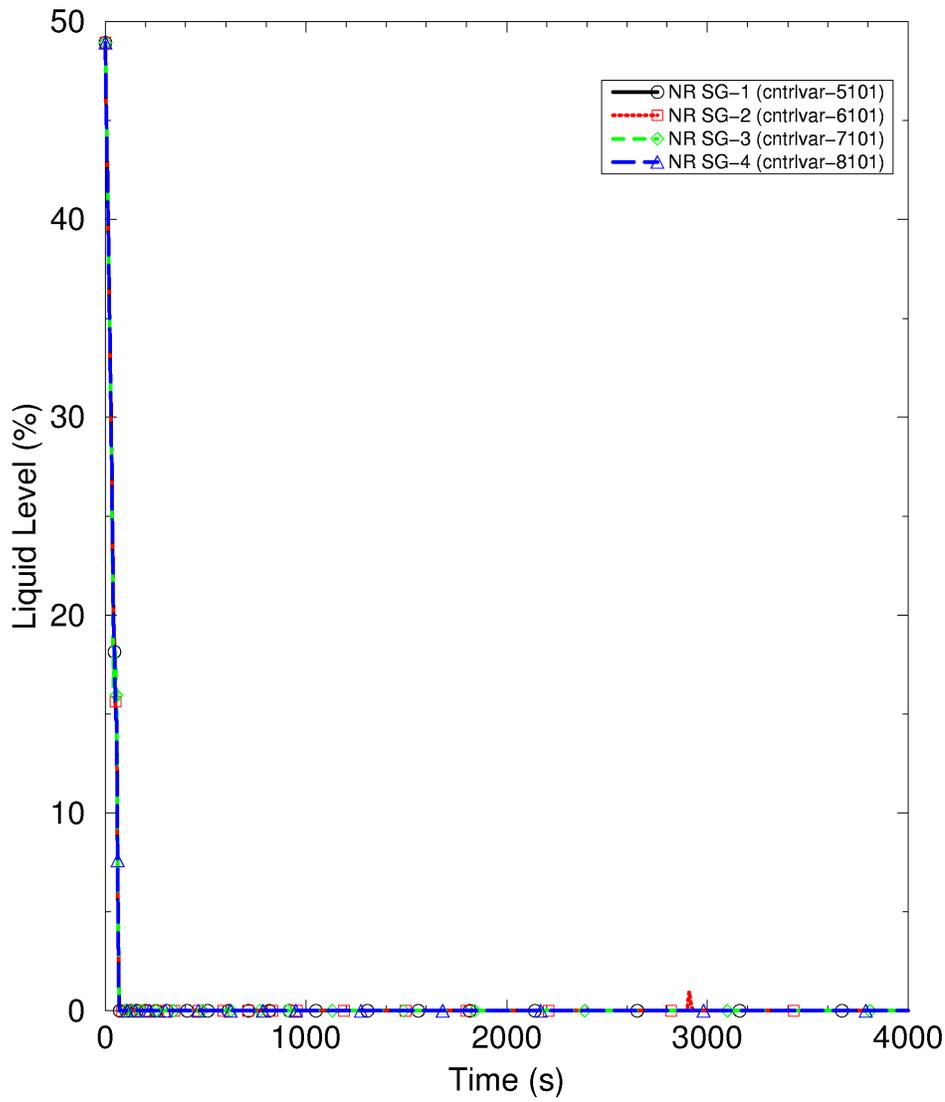
REV 002
EPR4560

Figure 15.2-39—Loss of Normal Feedwater - SG Pressure



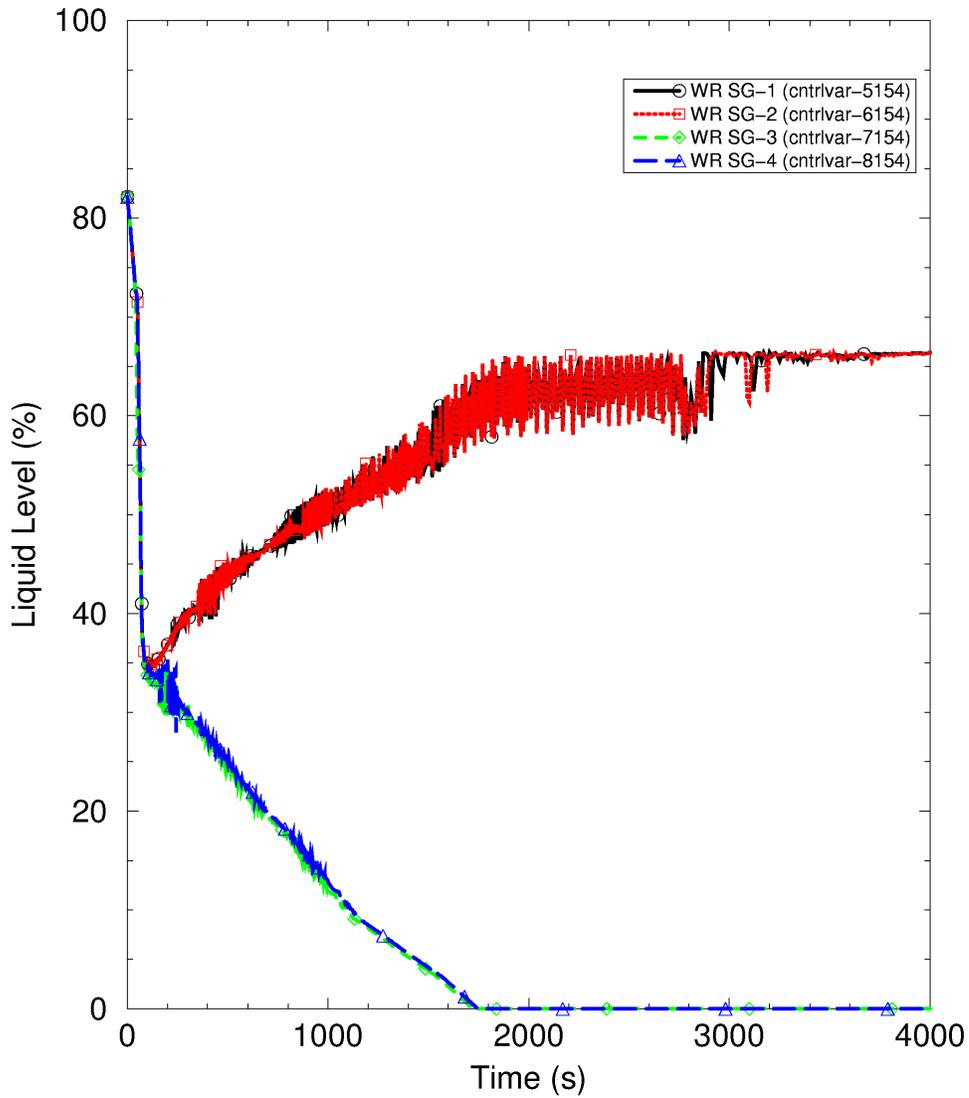
REV 002
EPR4565

Figure 15.2-40—Loss of Normal Feedwater - SG NR Liquid Level



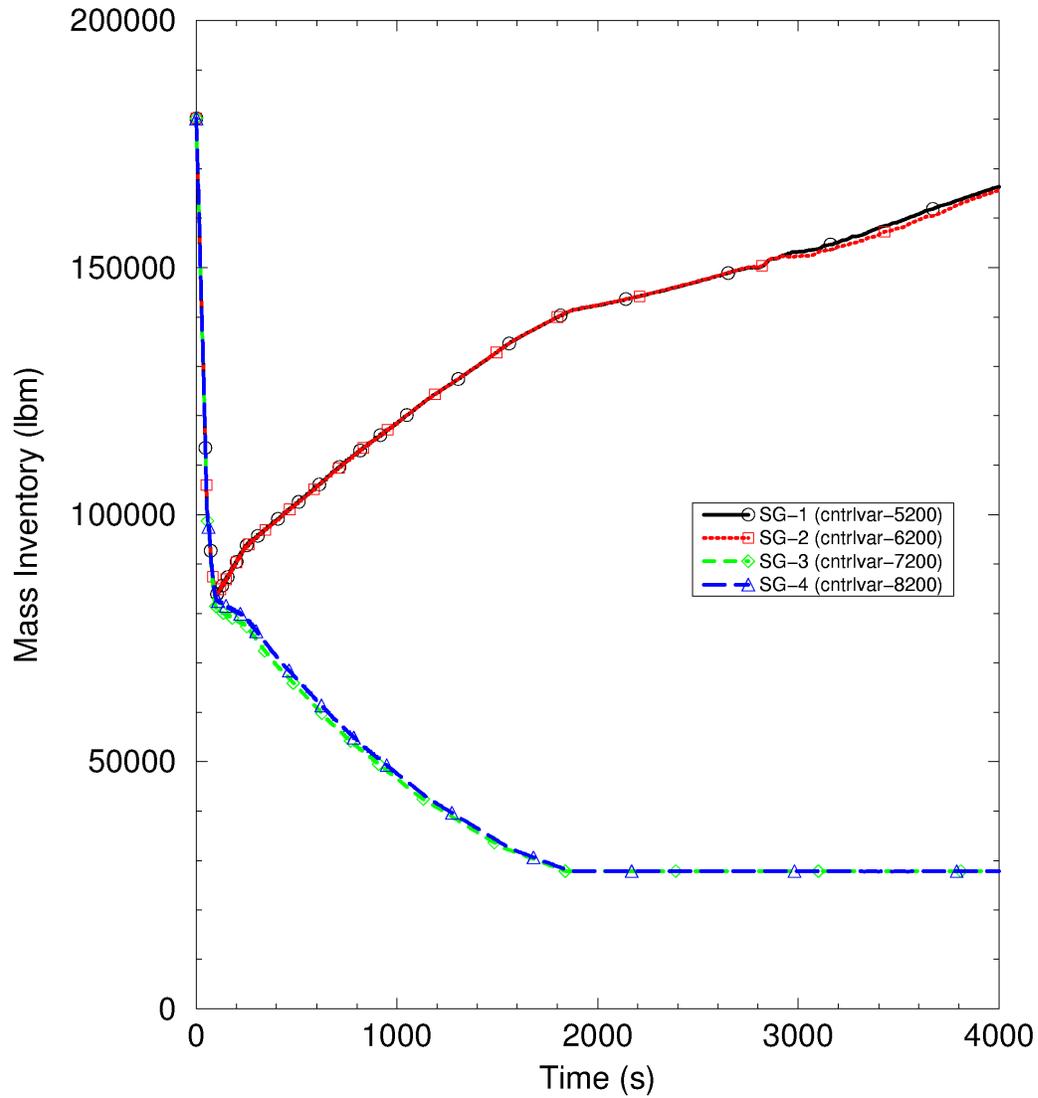
REV 002
EPR4570

Figure 15.2-41—Loss of Normal Feedwater - SG WR Liquid Level



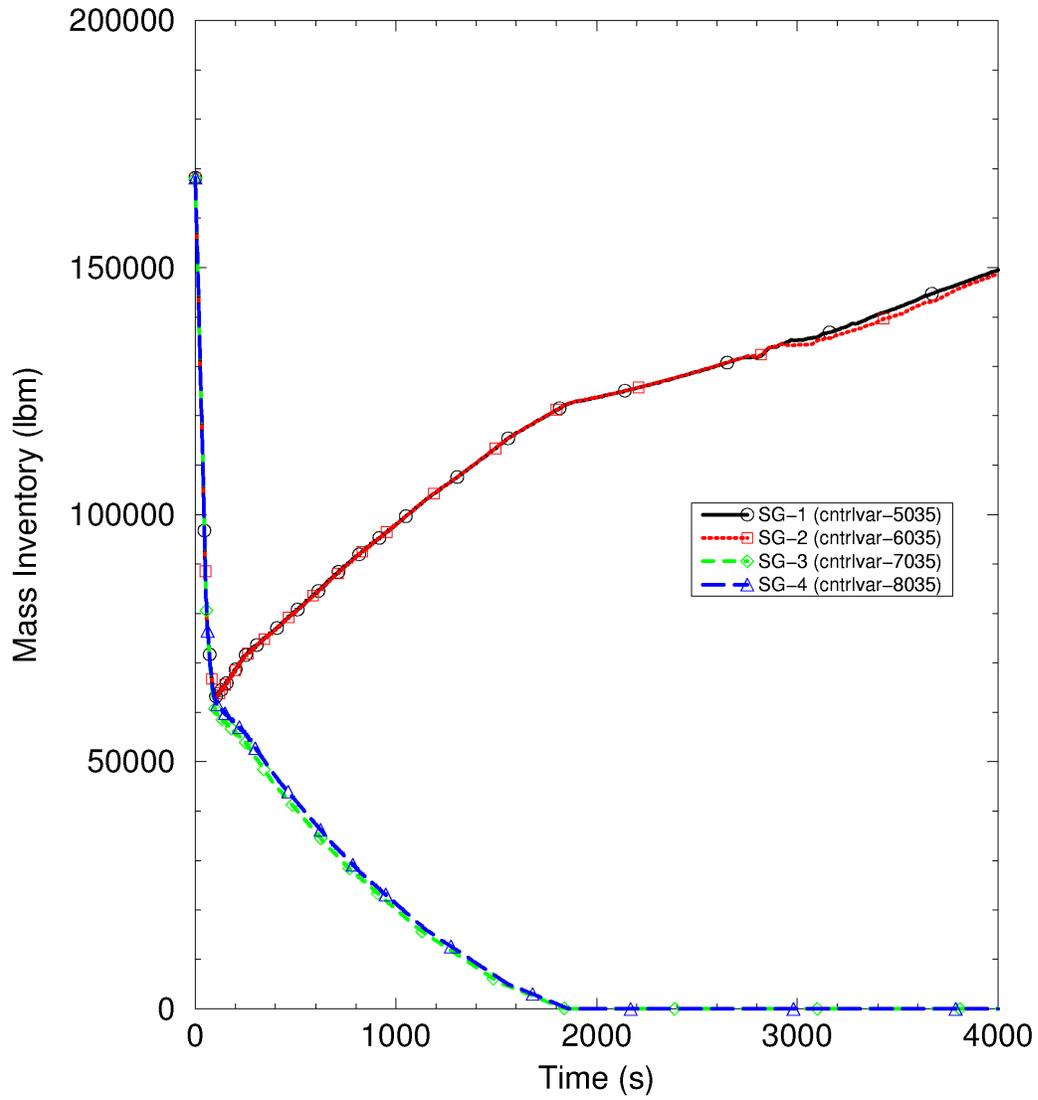
REV 002
EPR4575

Figure 15.2-42—Loss of Normal Feedwater - SG Total Mass Inventory



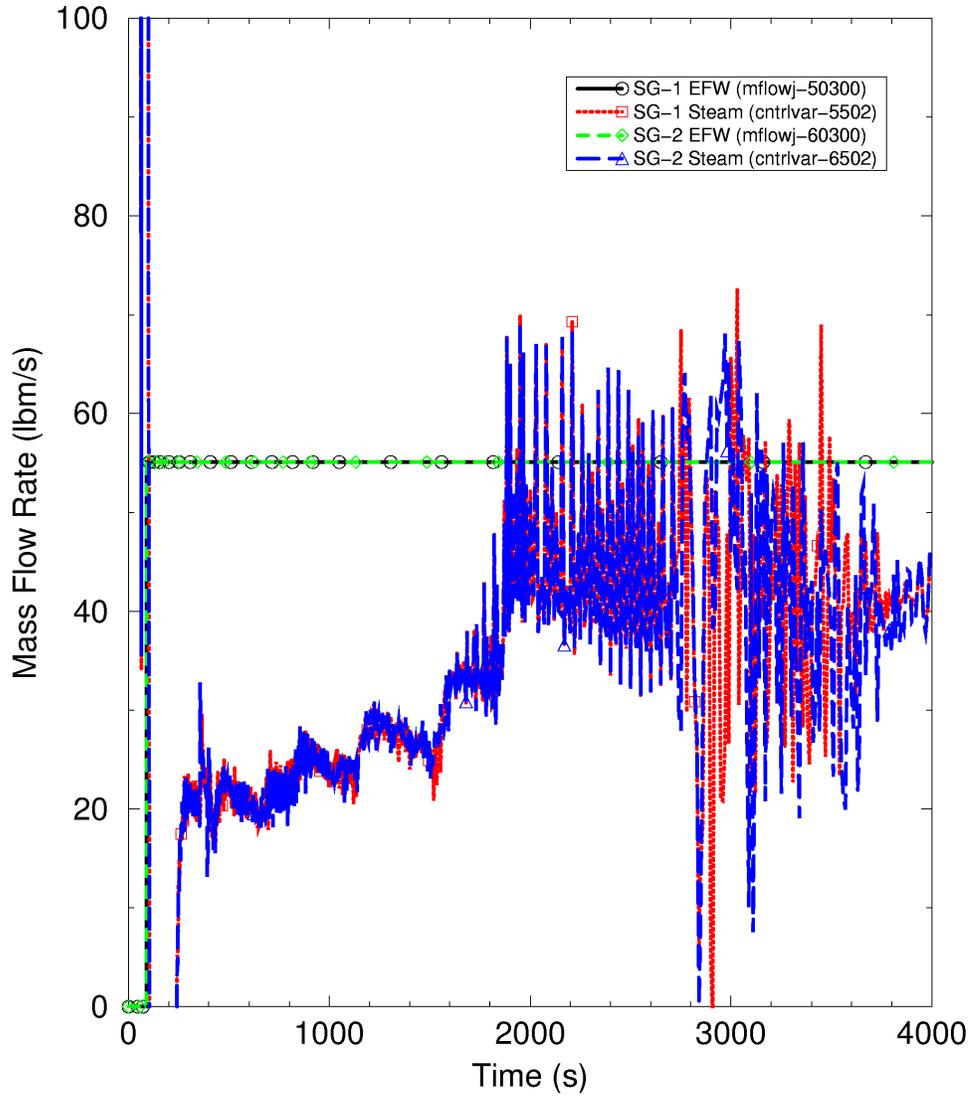
REV 002
EPR4580

Figure 15.2-43—Loss of Normal Feedwater - SG Liquid Mass Inventory



REV 002
EPR4585

Figure 15.2-44—Loss of Normal Feedwater - Steam and EFW Flow, Unaffected



REV 002
EPR4590

Figure 15.2-45—Loss of Normal Feedwater - Steam and EFW Flow, Affected

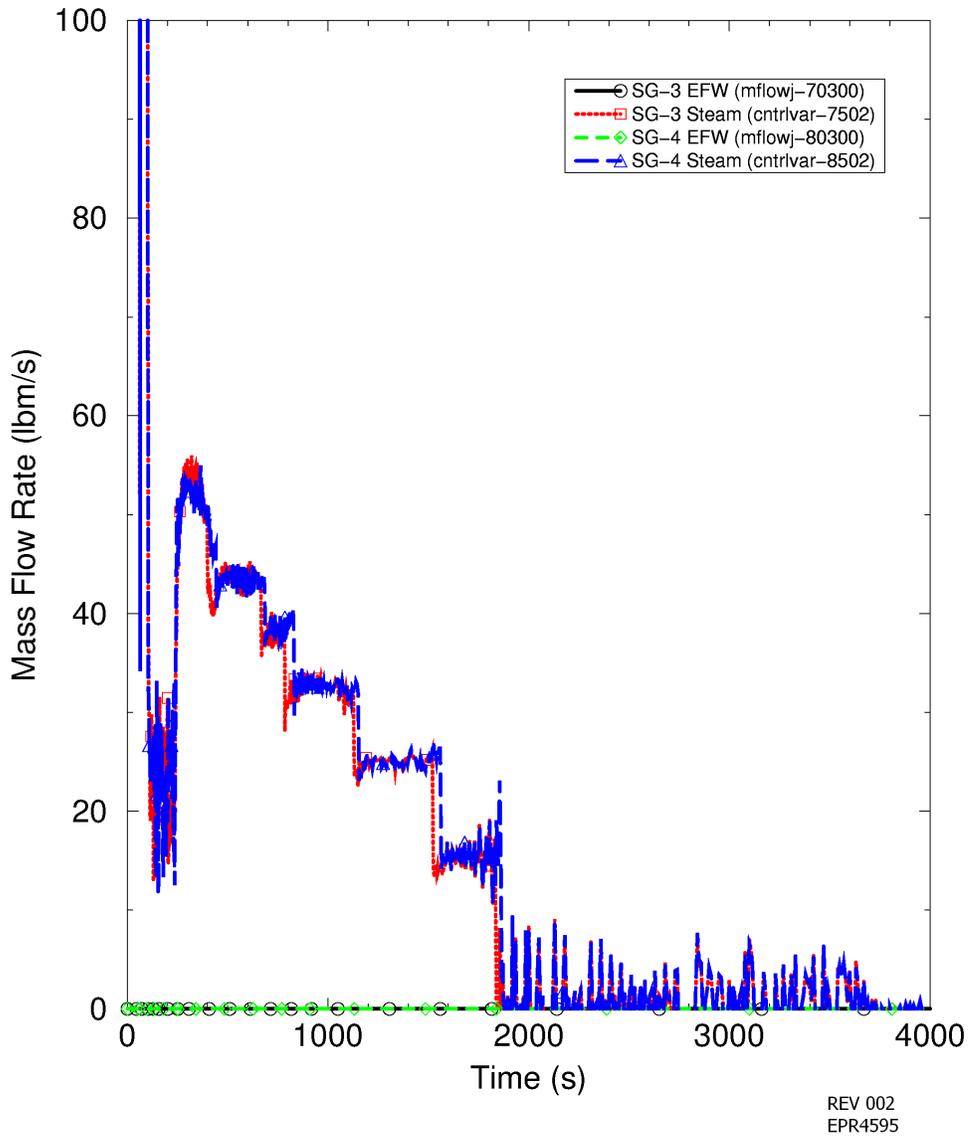
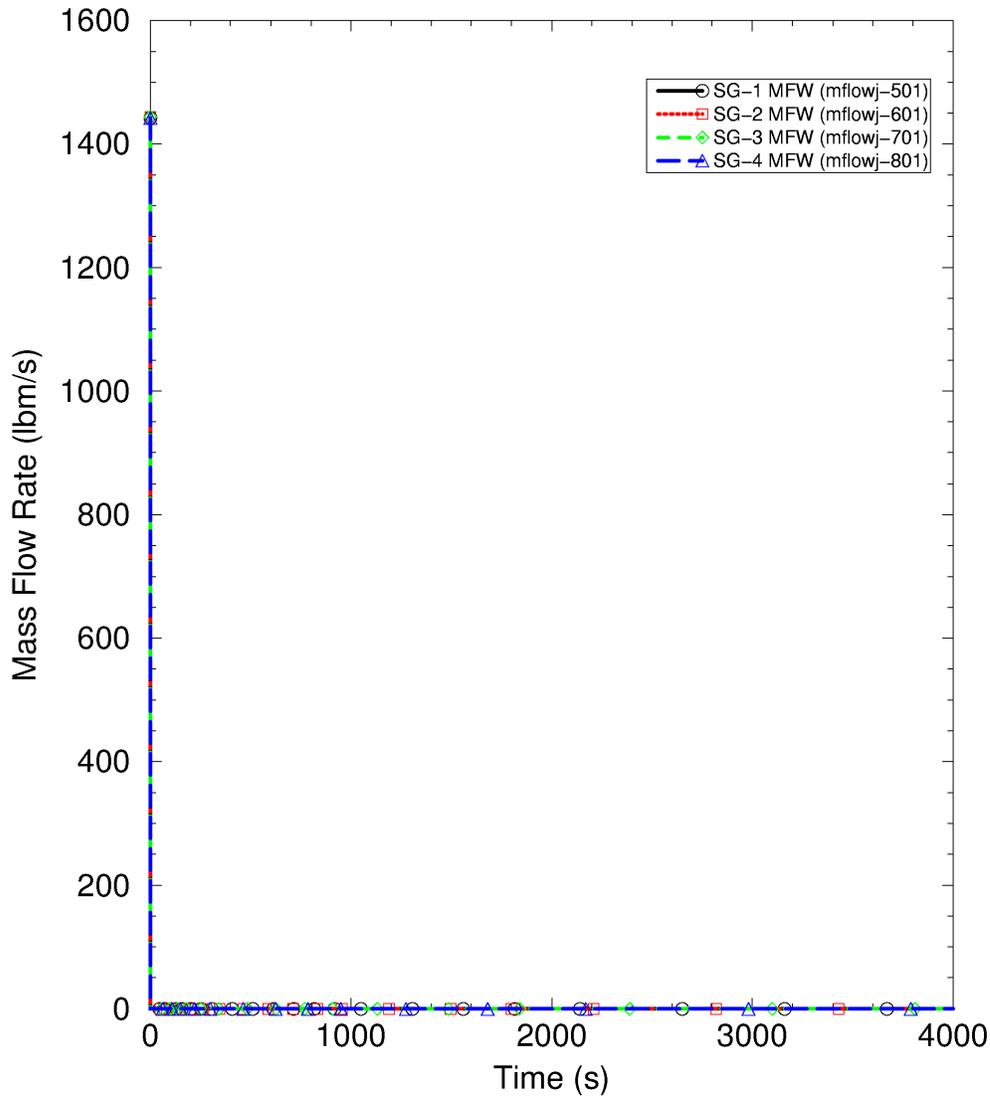
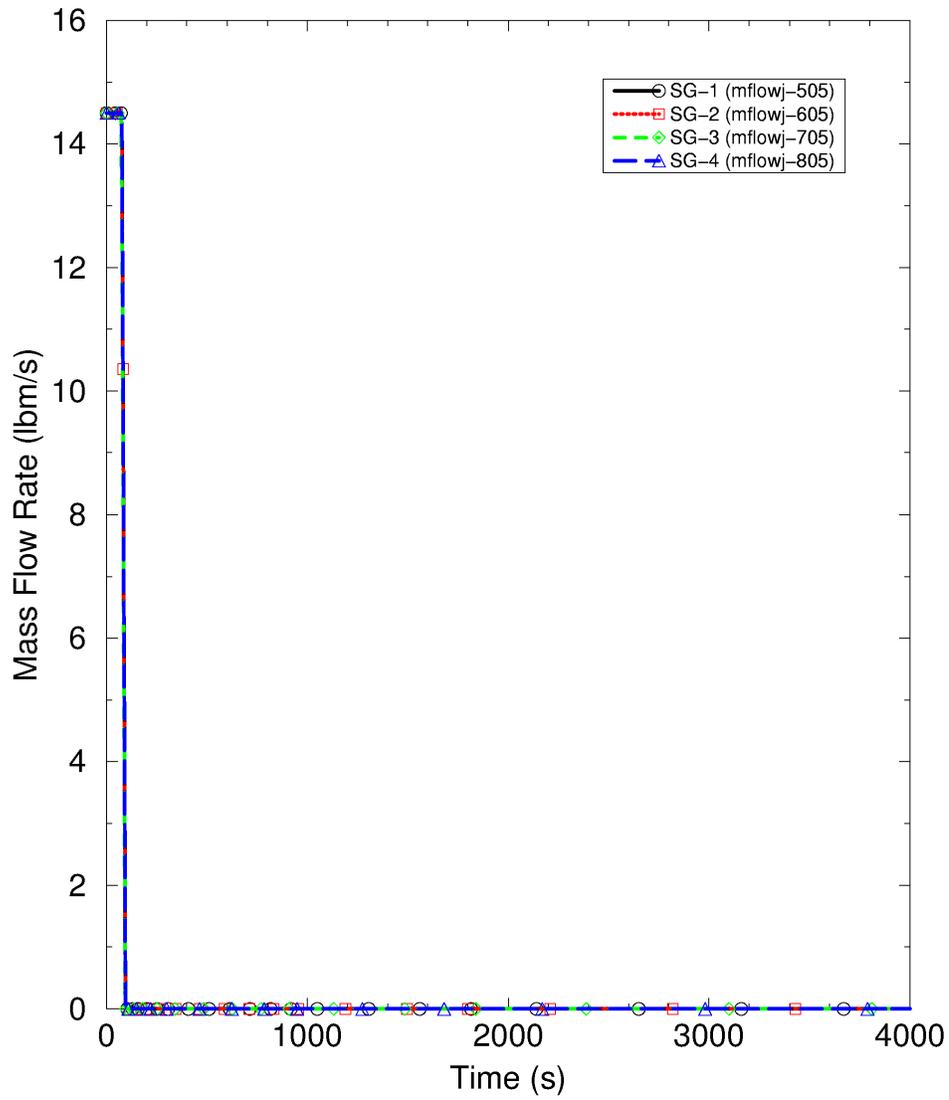


Figure 15.2-46—Loss of Normal Feedwater - MFW Flow



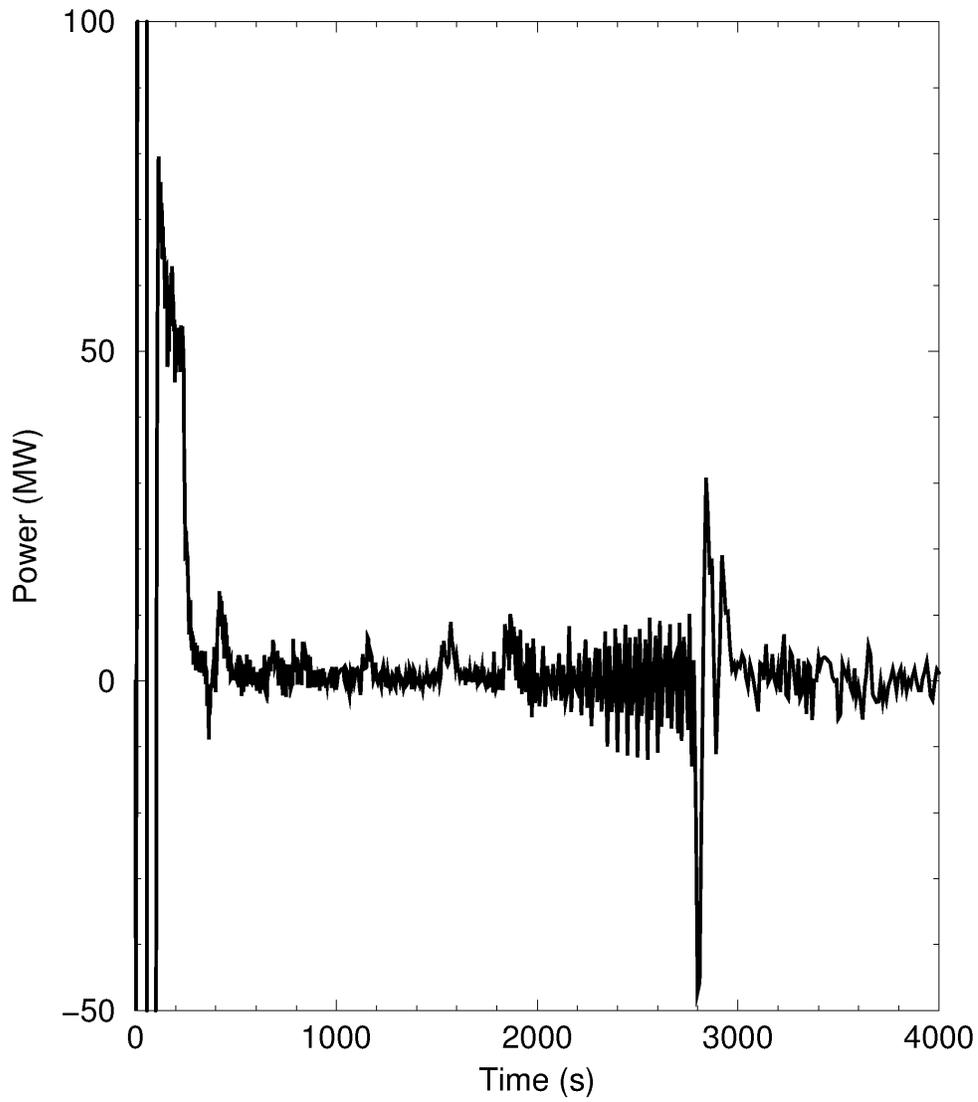
REV 002
EPR4600

Figure 15.2-47—Loss of Normal Feedwater - Blowdown Flow



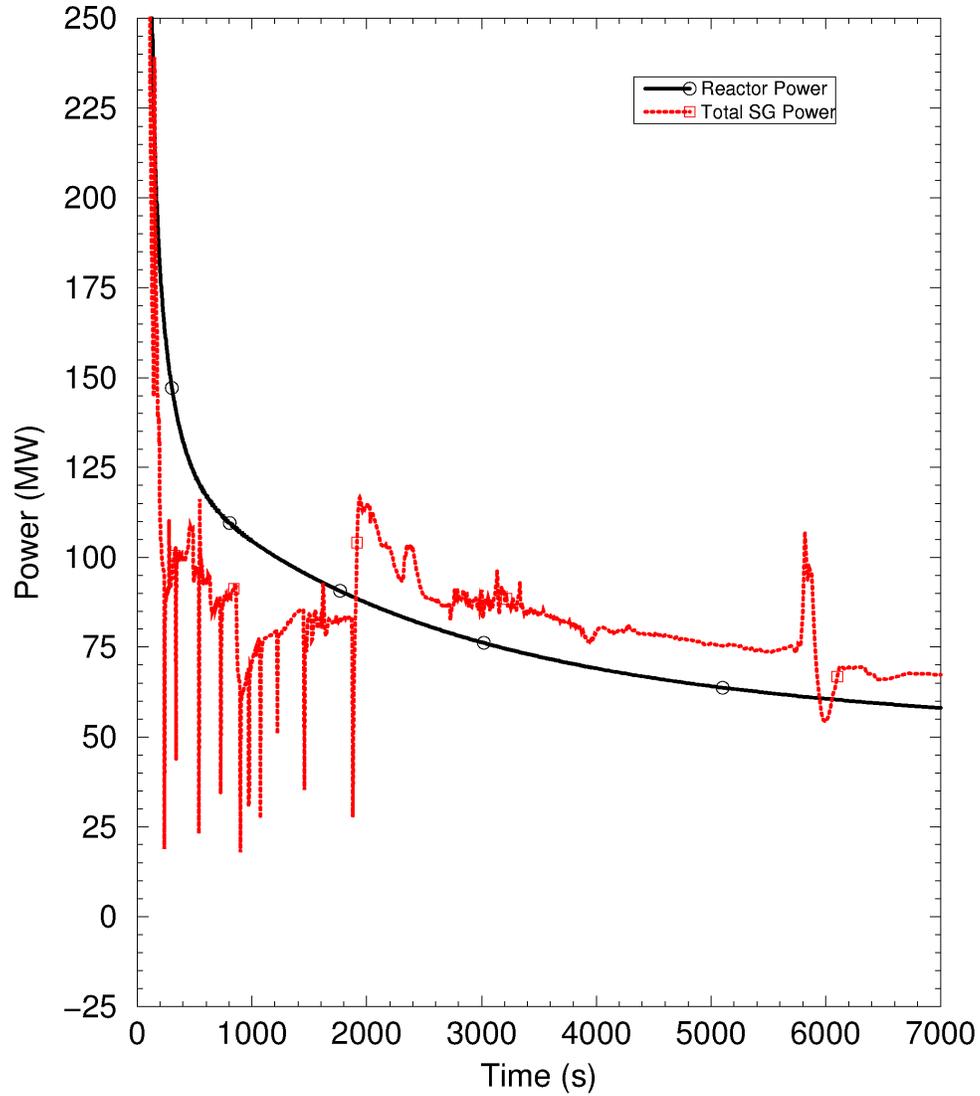
REV 002
EPR4605

Figure 15.2-48—Loss of Normal Feedwater - Net Heat Addition to RCS



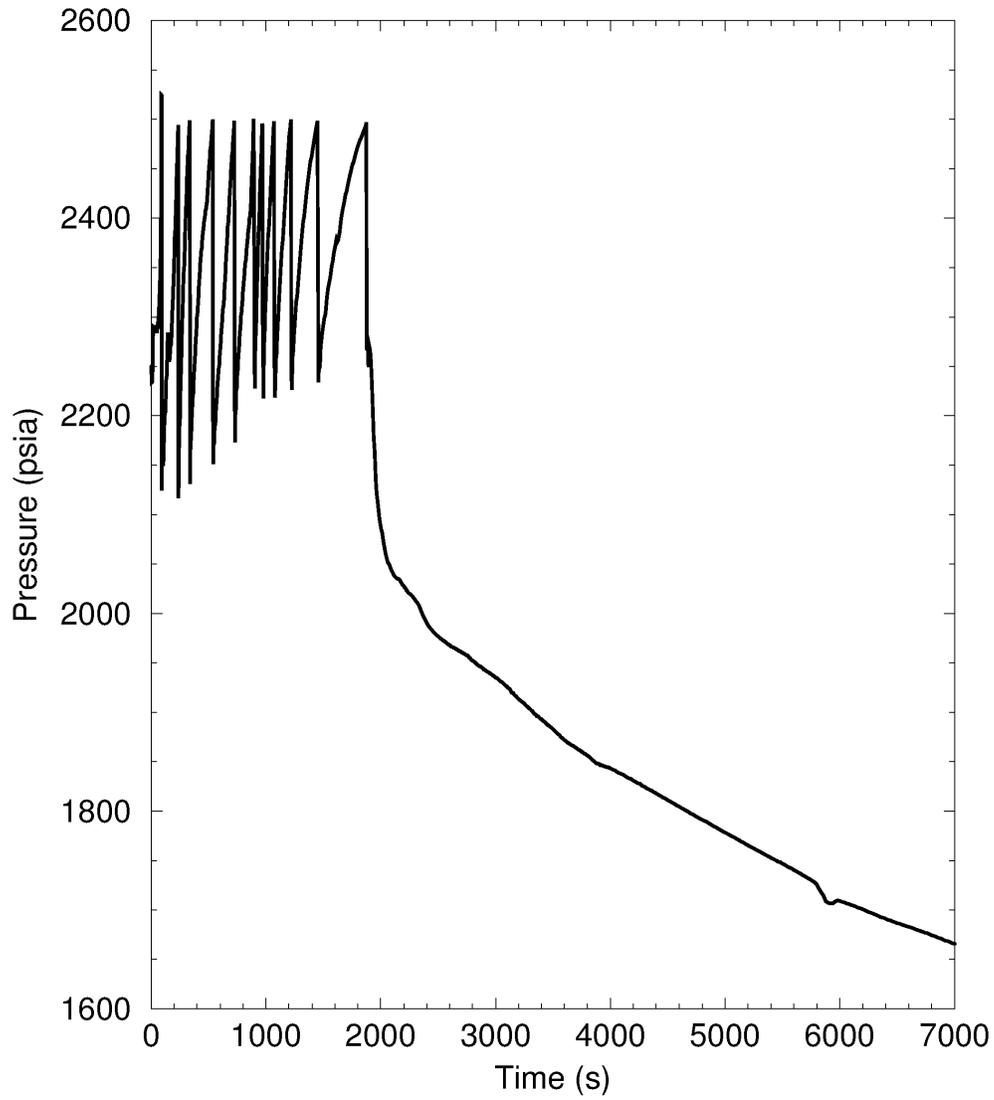
REV 002
EPR4610

Figure 15.2-49—FWLB Representative Small Break - Reactor and Total Steam Generator Power



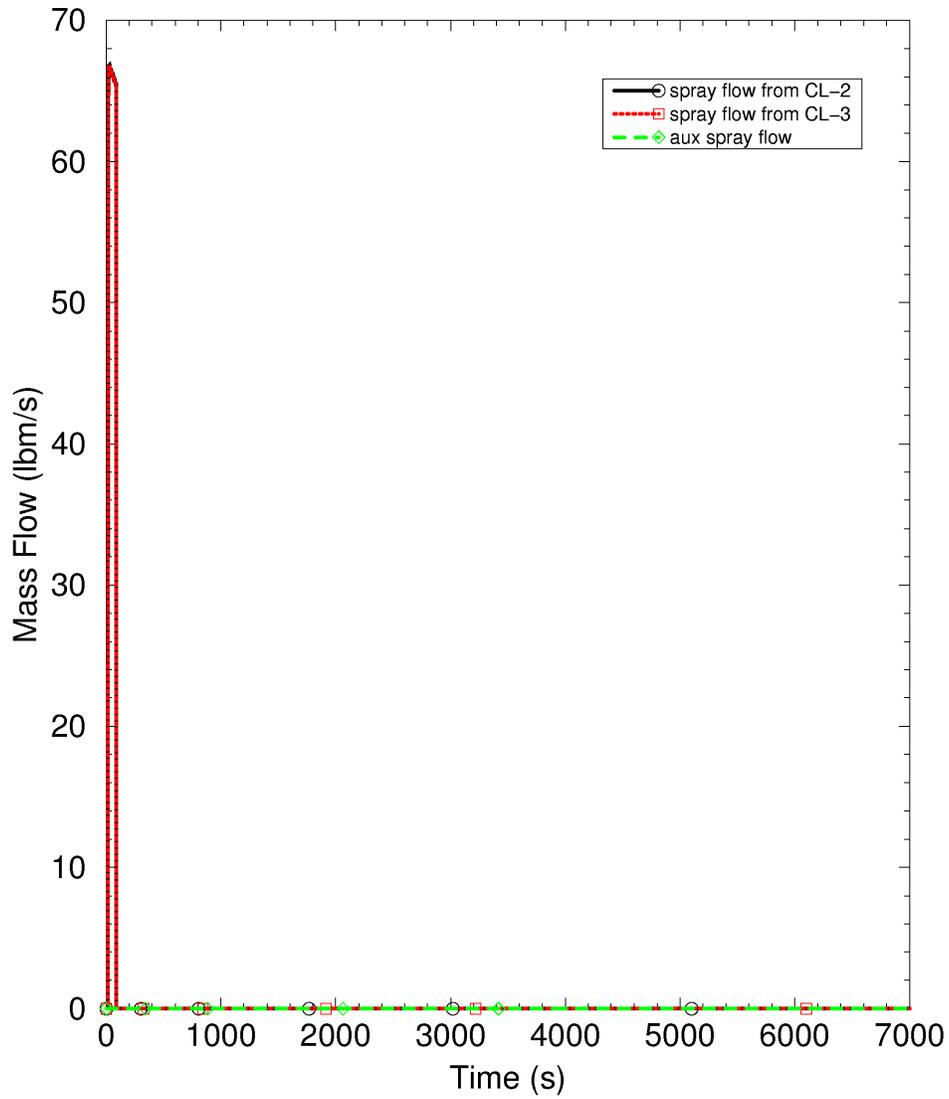
REV 002
EPR4615 T2

Figure 15.2-50—FWLB Representative Small Break - Pressurizer Pressure



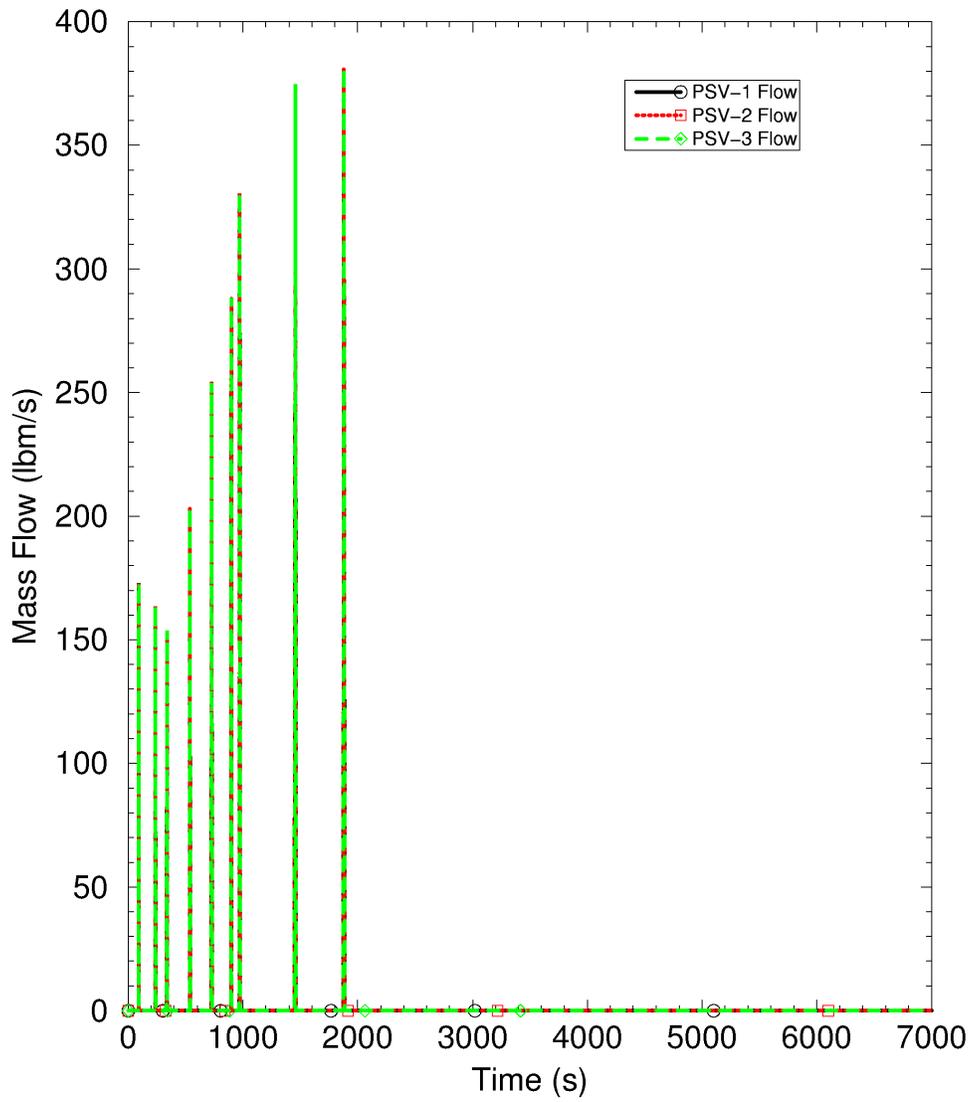
REV 002
EPR4620 T2

Figure 15.2-51—FWLB Representative Small Break - Pressurizer Spray Flow



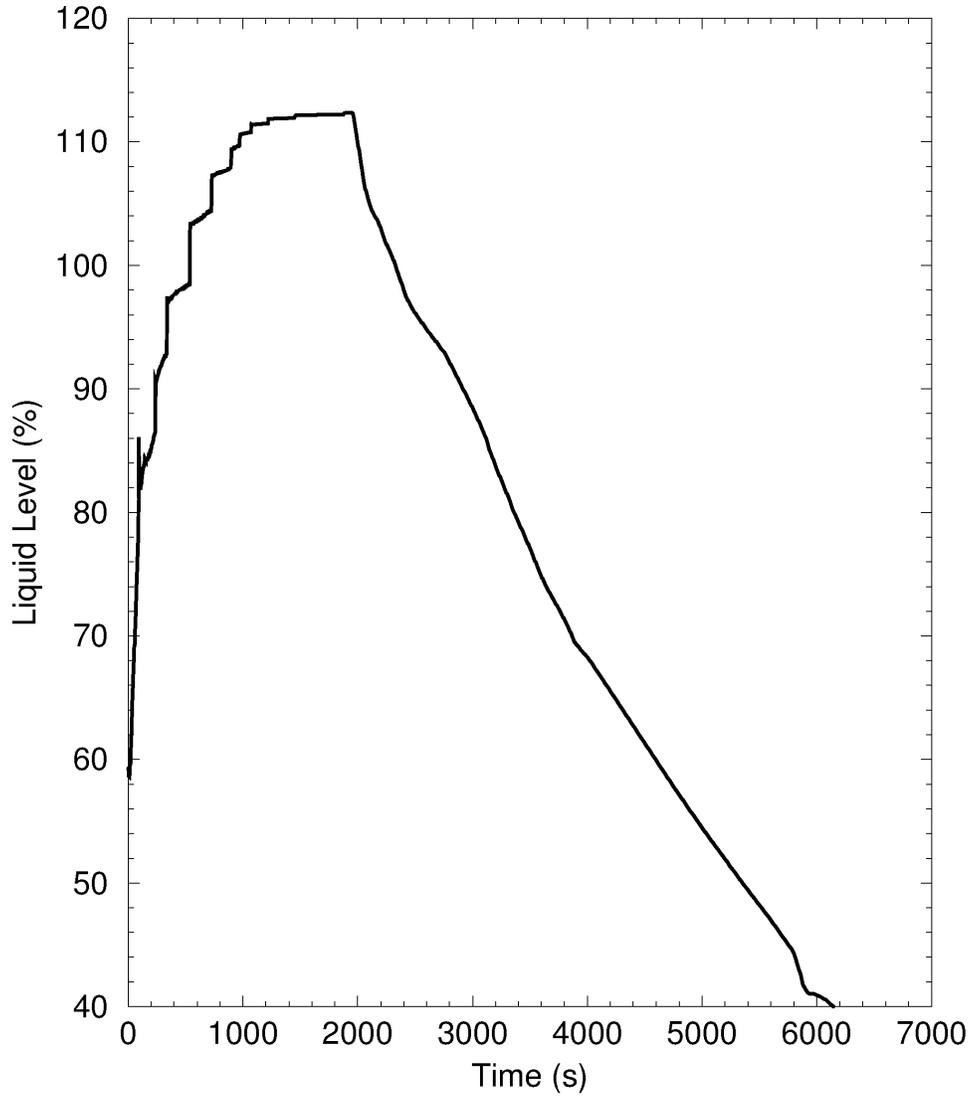
REV 002
EPR4625 T2

Figure 15.2-52—FWLB Representative Small Break - PSRV Flow



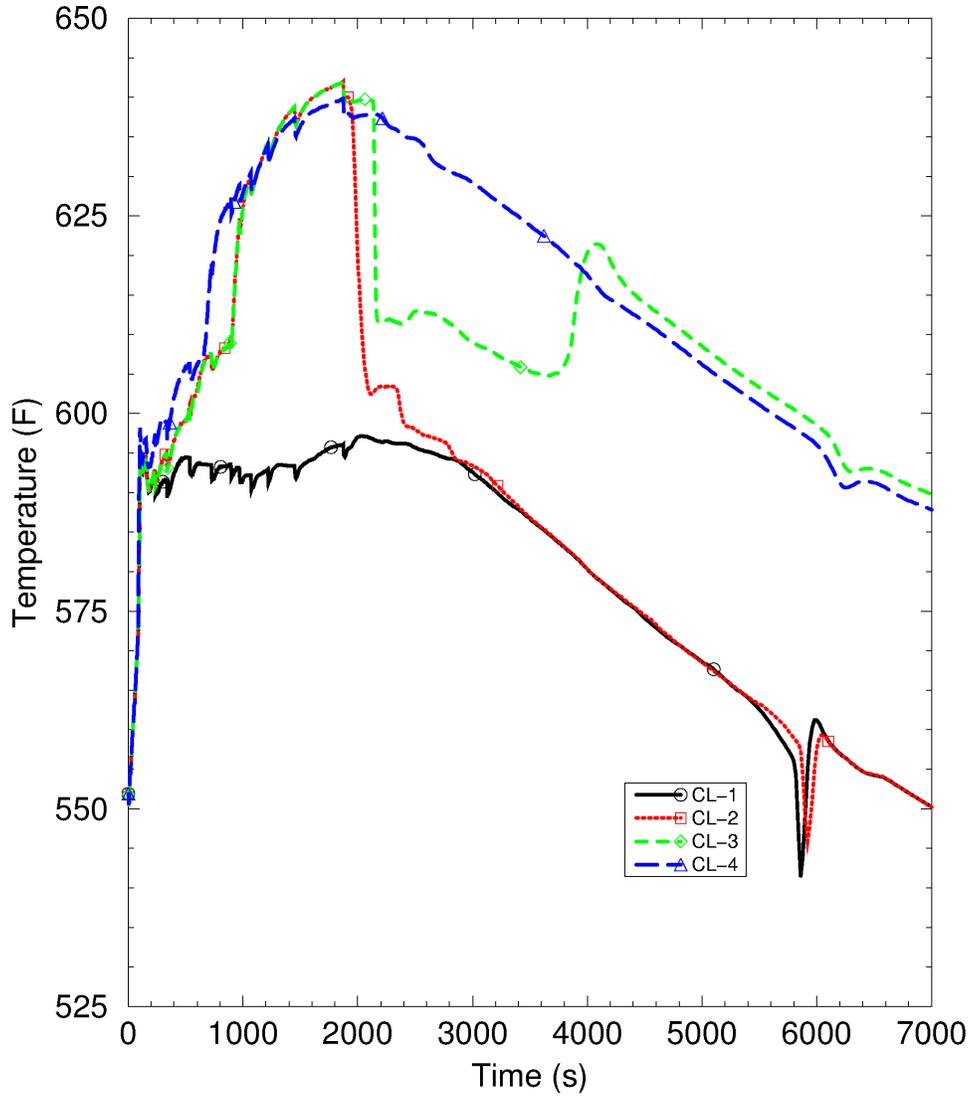
REV 002
EPR4630 T2

Figure 15.2-53—FWLB Representative Small Break - Pressurizer Liquid Level



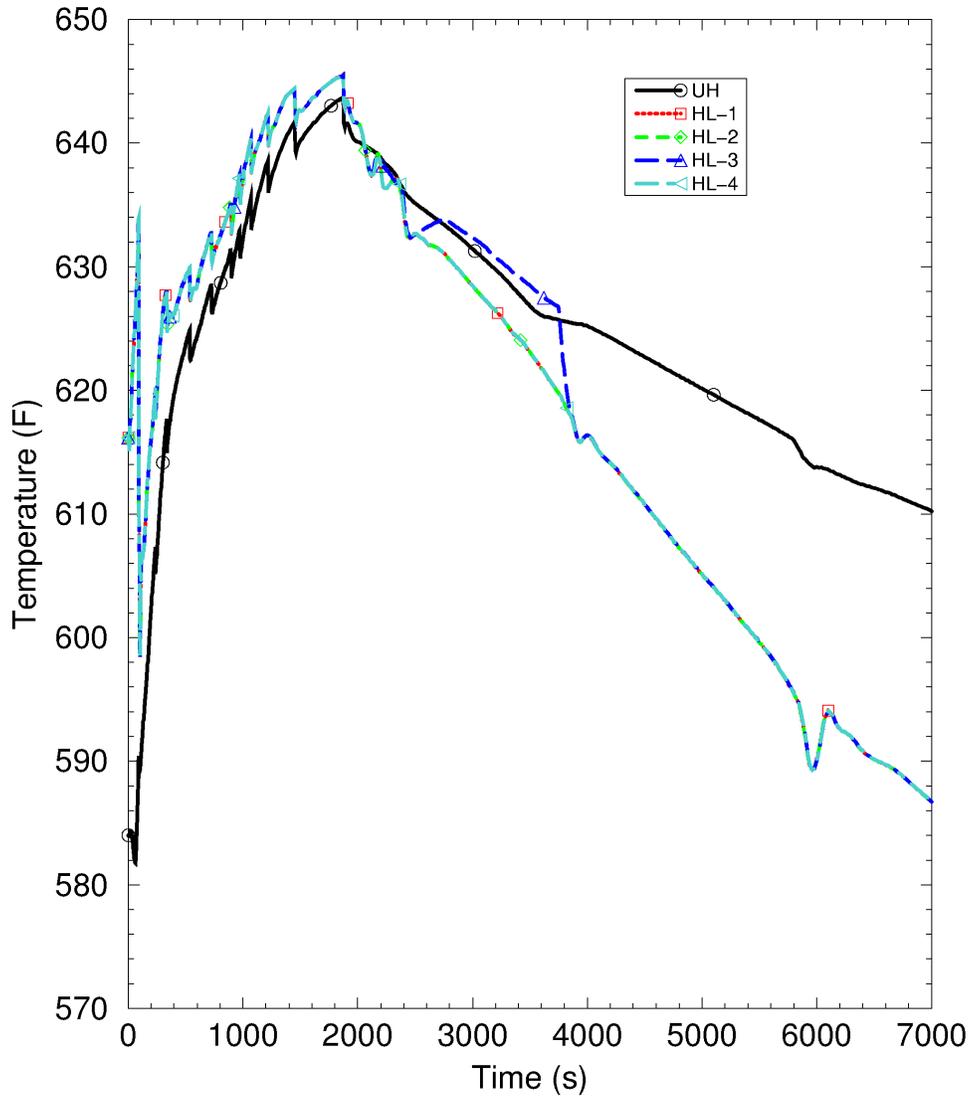
REV 002
EPR4635 T2

Figure 15.2-54—FWLB Representative Small Break - RCS Cold Leg Temperatures



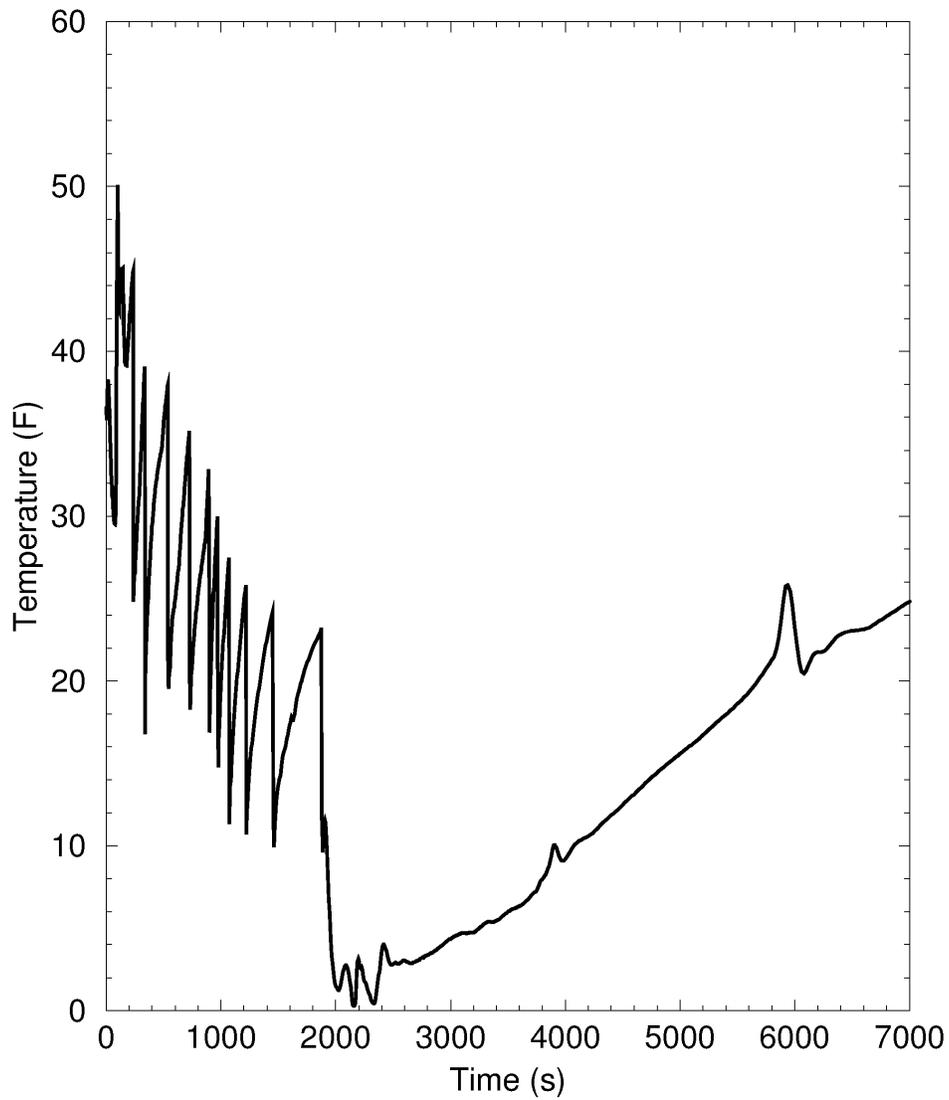
REV 002
EPR4640 T2

Figure 15.2-55—FWLB Representative Small Break - RCS Hot Leg Temperatures



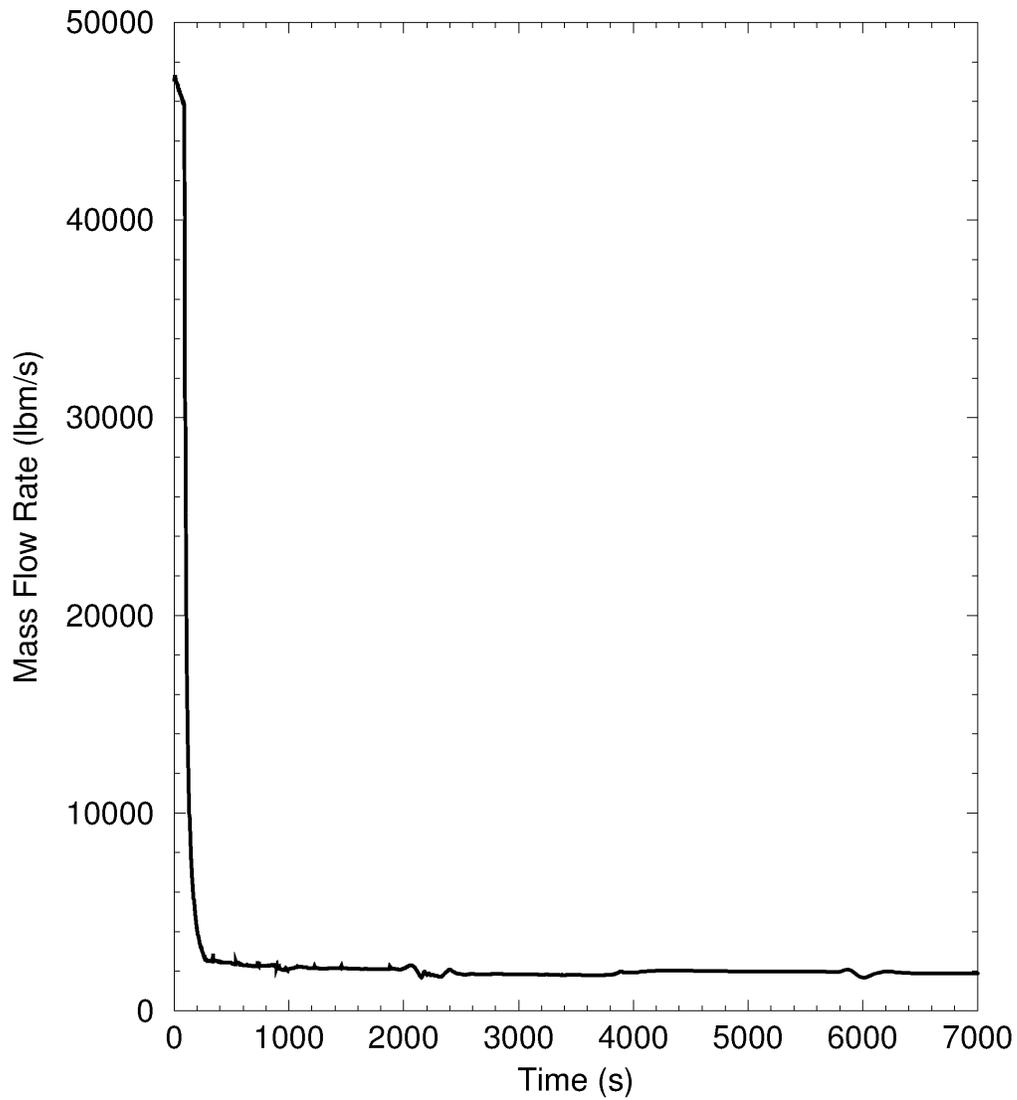
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EPR4645 T2

Figure 15.2-56—FWLB Representative Small Break - Core Exit Subcooling



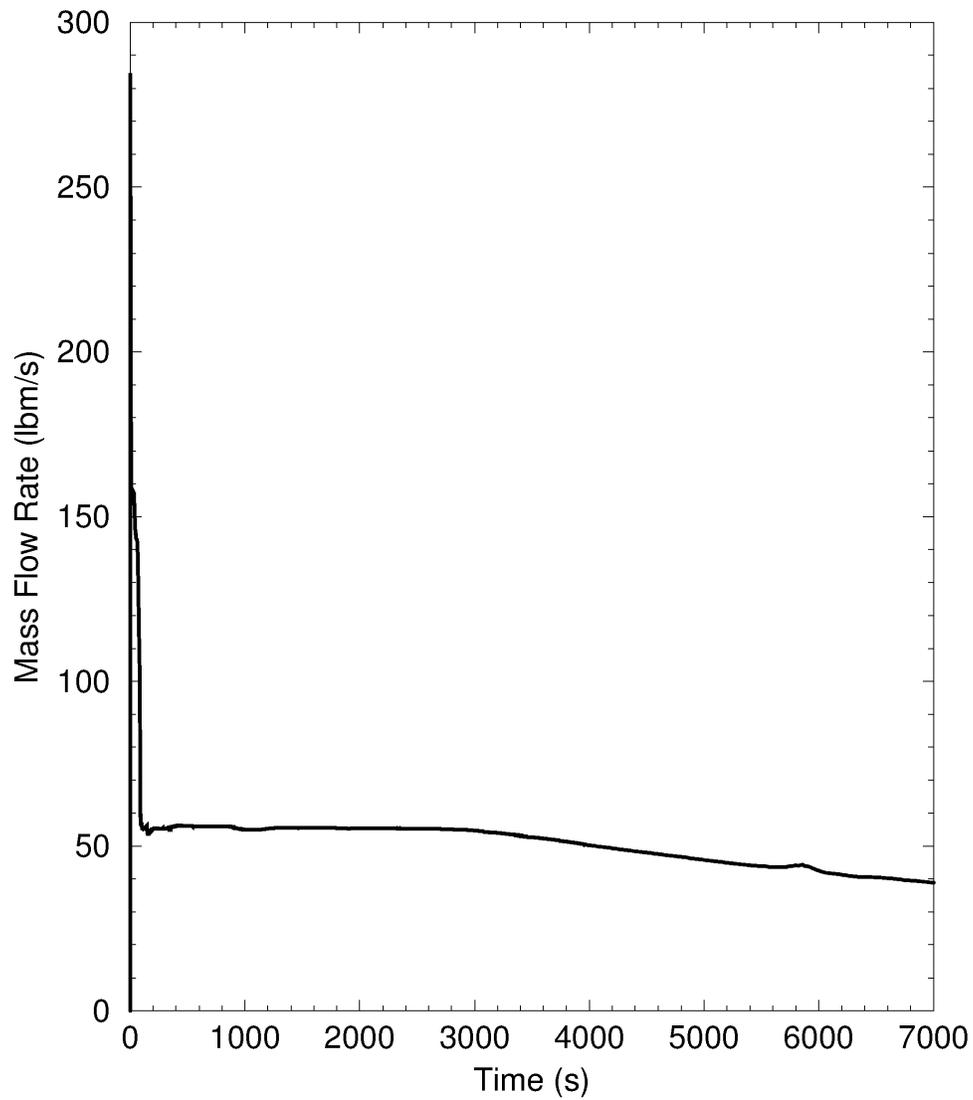
REV 002
EPR4650 T2

Figure 15.2-57—FWLB Representative Small Break - Core Flow



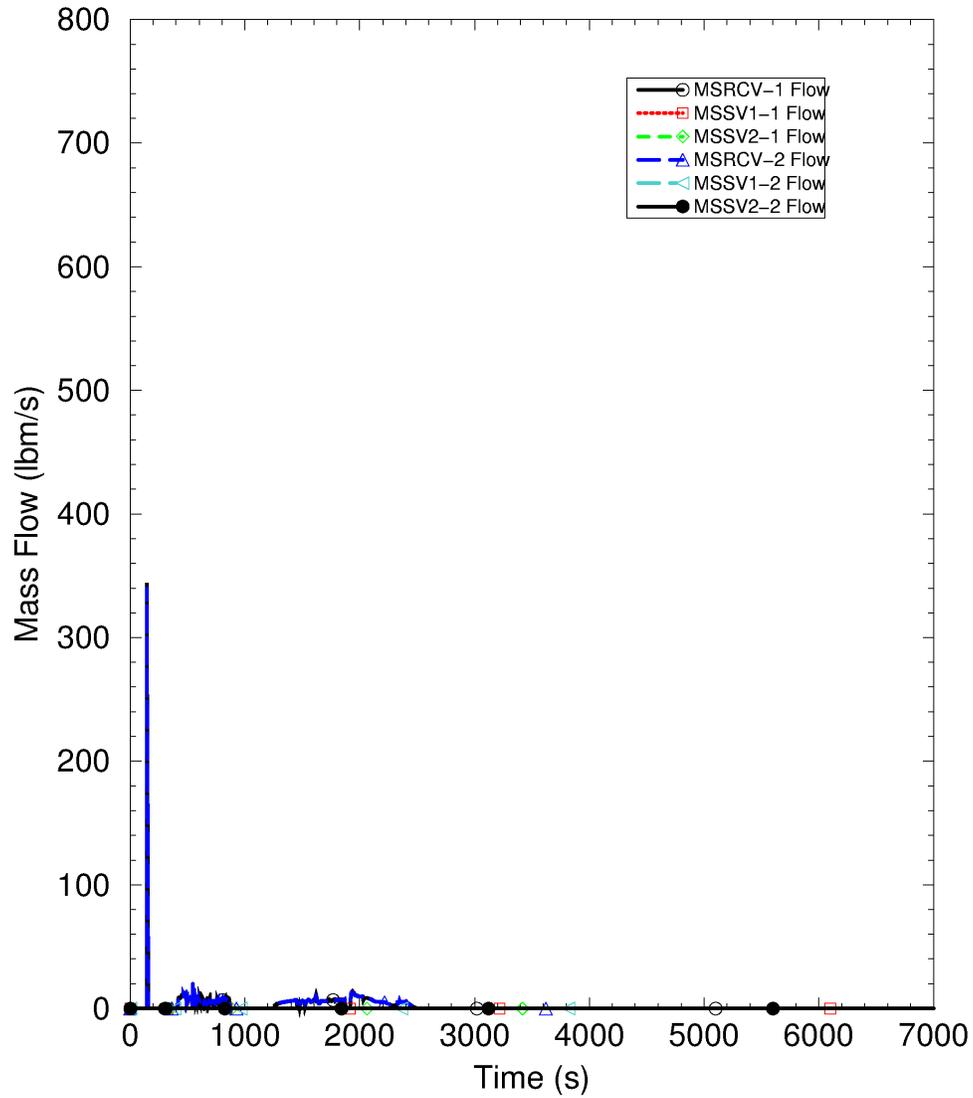
REV 002
EPR4655 T2

Figure 15.2-58—FWLB Representative Small Break - Break Flow



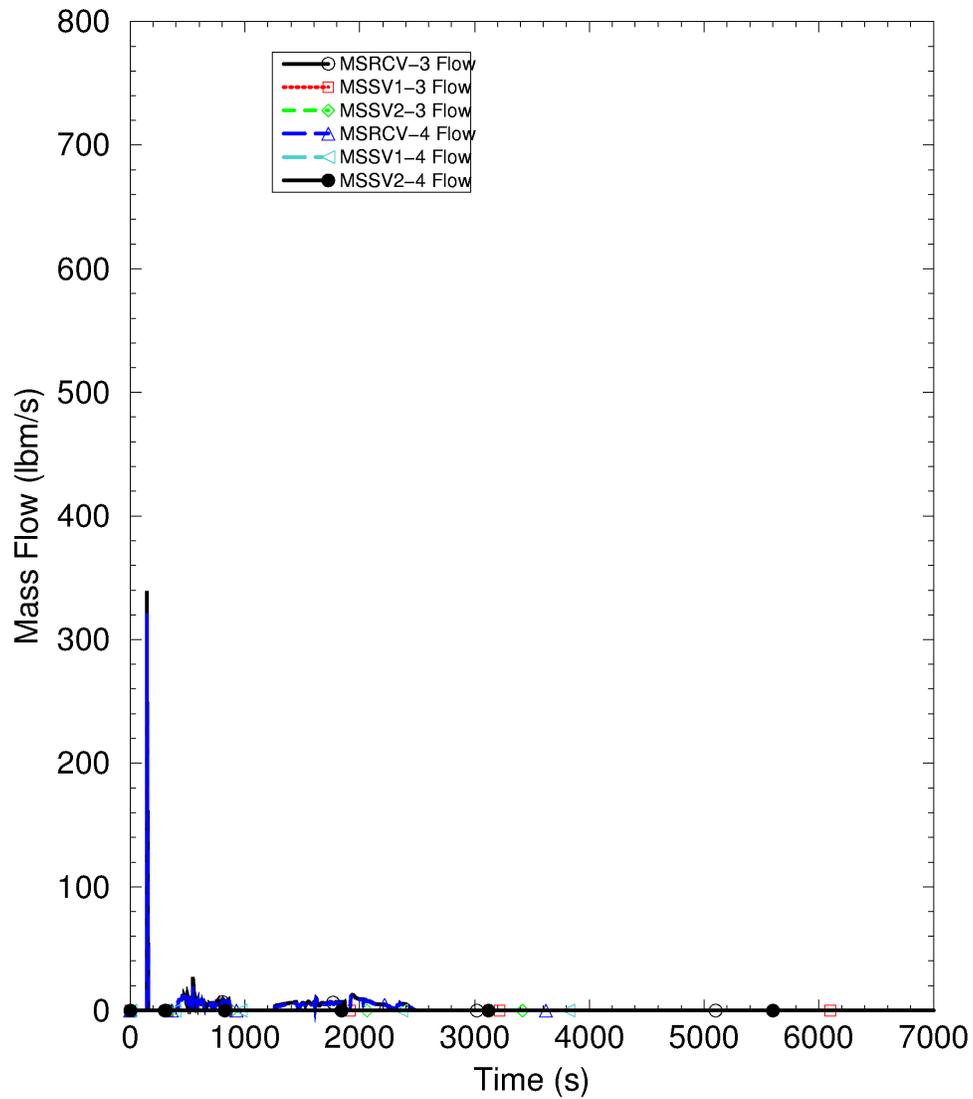
REV 002
EPR4660 T2

Figure 15.2-59—FWLB Representative Small Break - Main Steam Relief Loops 1 and 2



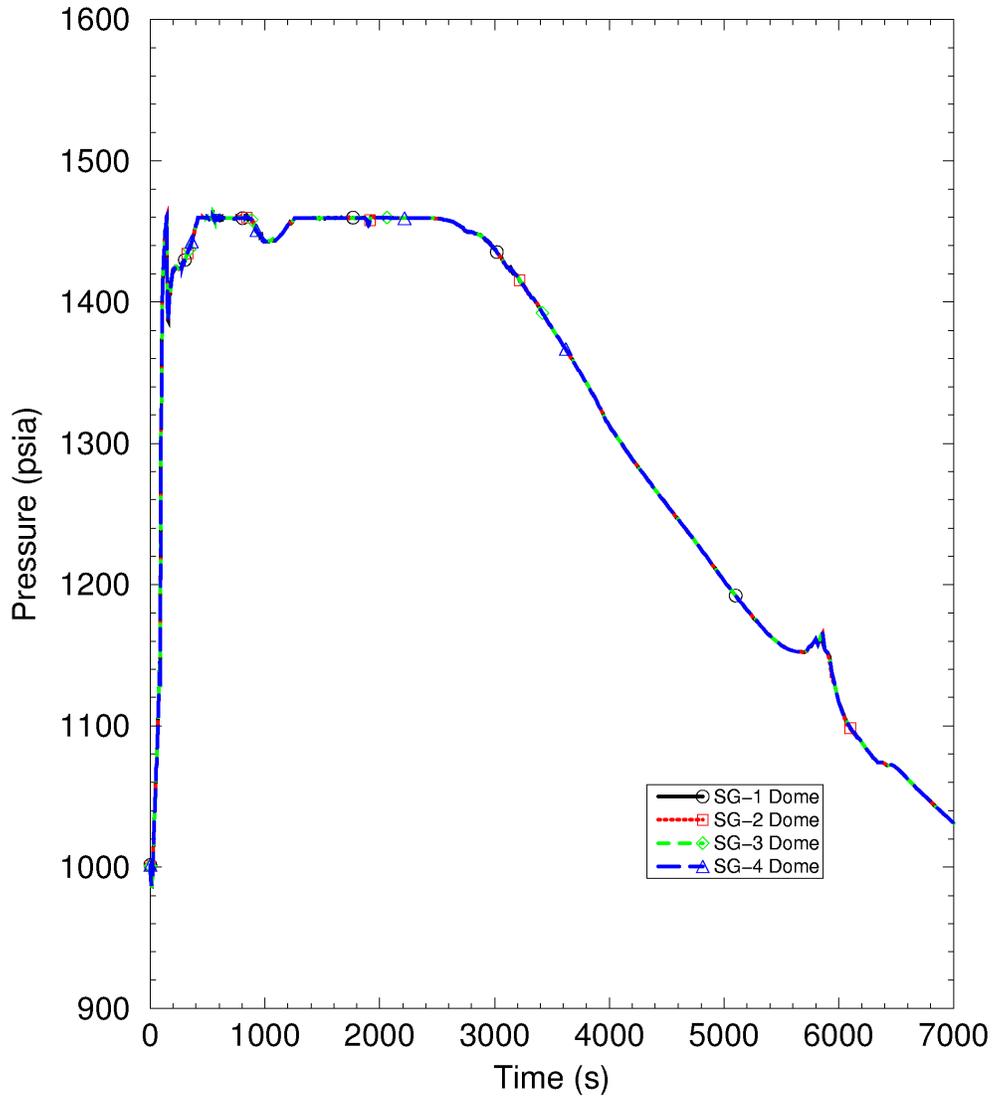
REV 002
EPR4665 T2

Figure 15.2-60—FWLB Representative Small Break - Main Steam Relief
Loops 3 and 4



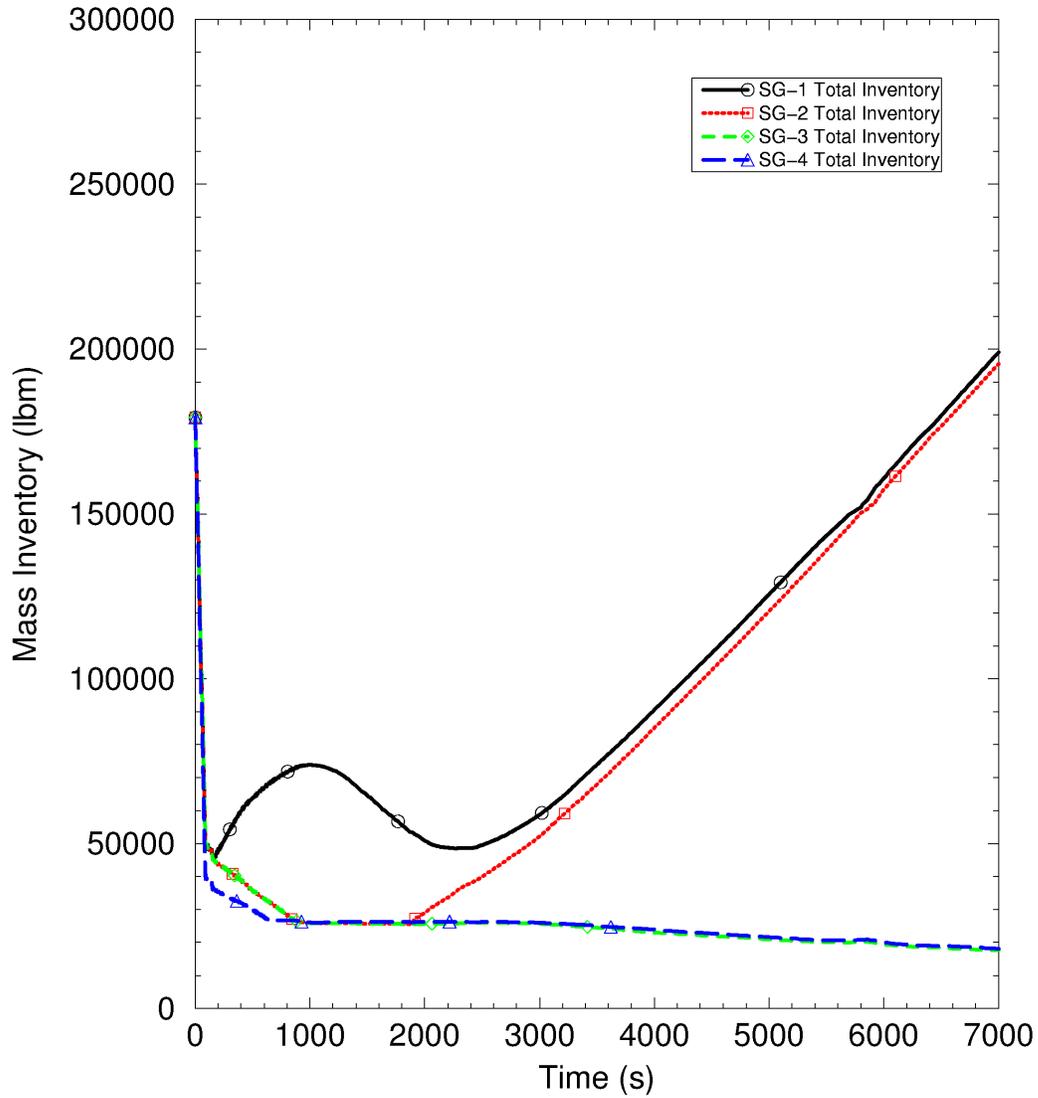
REV 002
EPR4670 T2

Figure 15.2-61—FWLB Representative Small Break - Steam Generator Dome Pressure



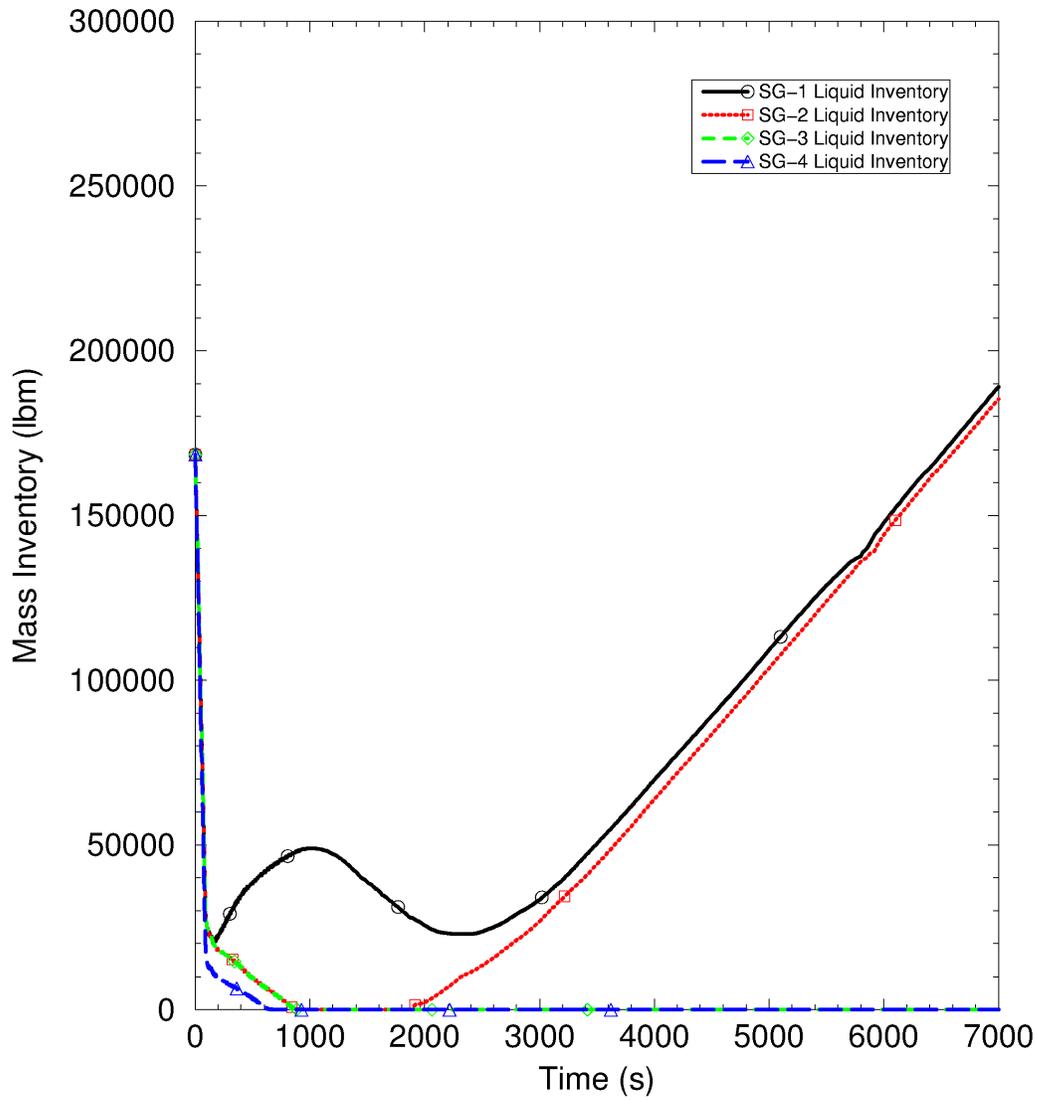
REV 002
EPR4675 T2

Figure 15.2-62—FWLB Representative Small Break - Steam Generator Total Mass



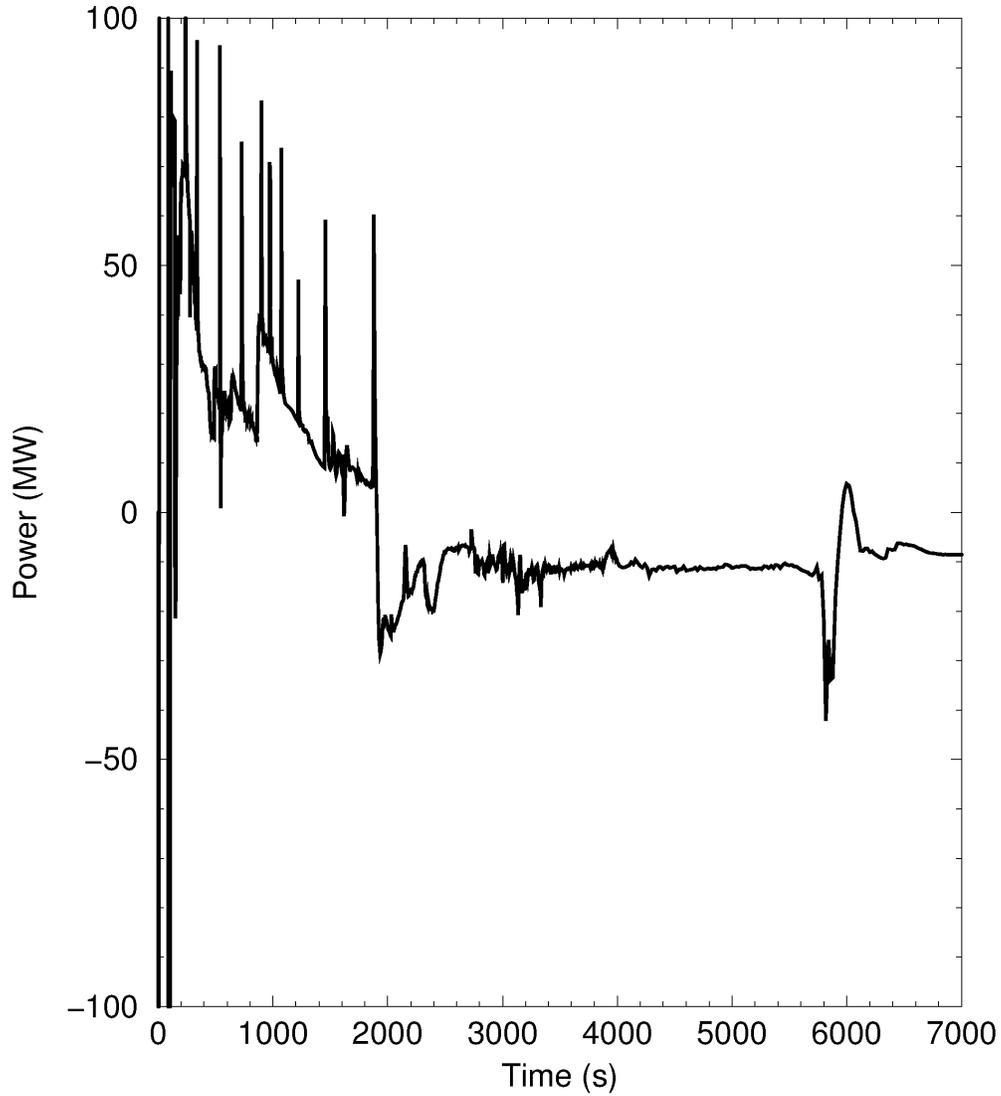
REV 002
EPR4680 T2

Figure 15.2-63—FWLB Representative Small Break - Steam Generator Liquid Mass



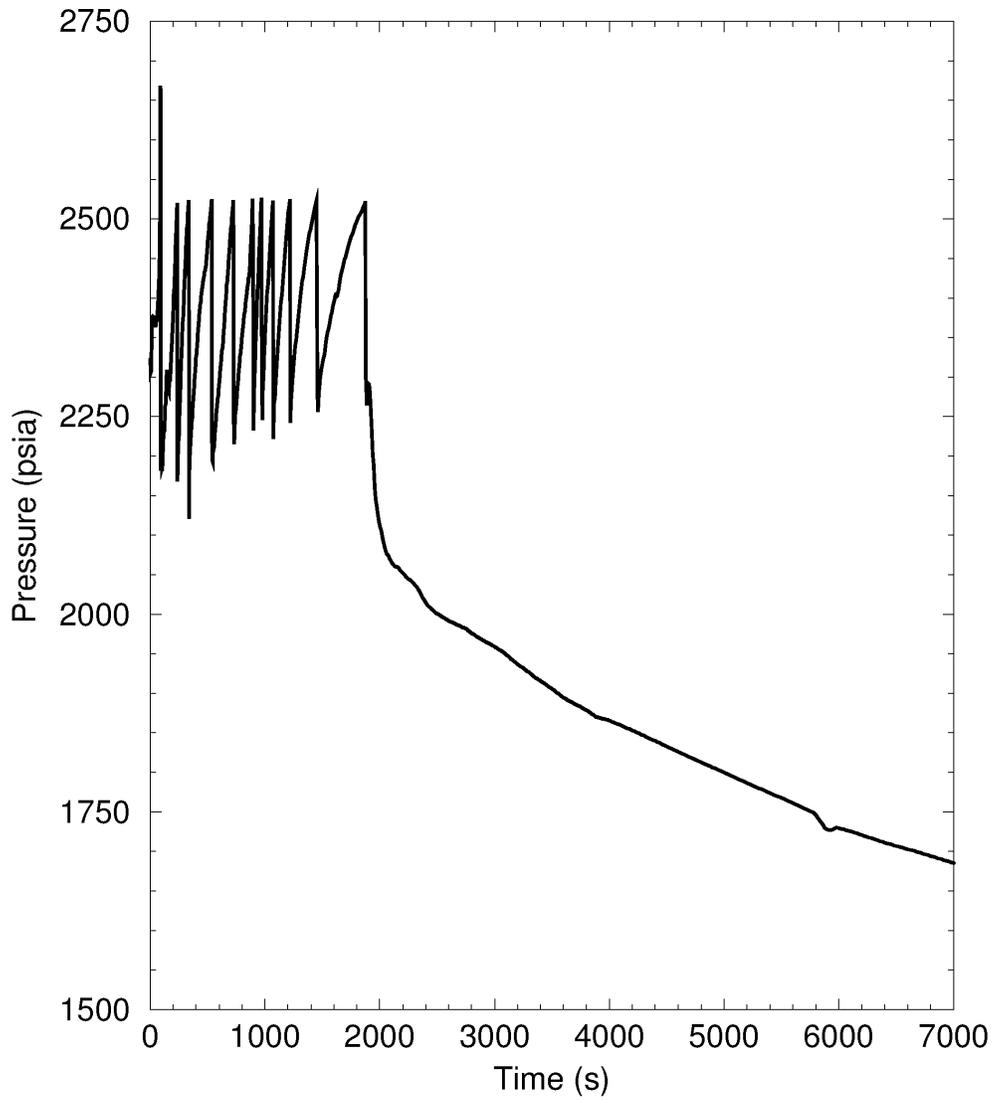
REV 002
EPR4685 T2

Figure 15.2-64—FWLB Representative Small Break - Net Heat Addition to RCS



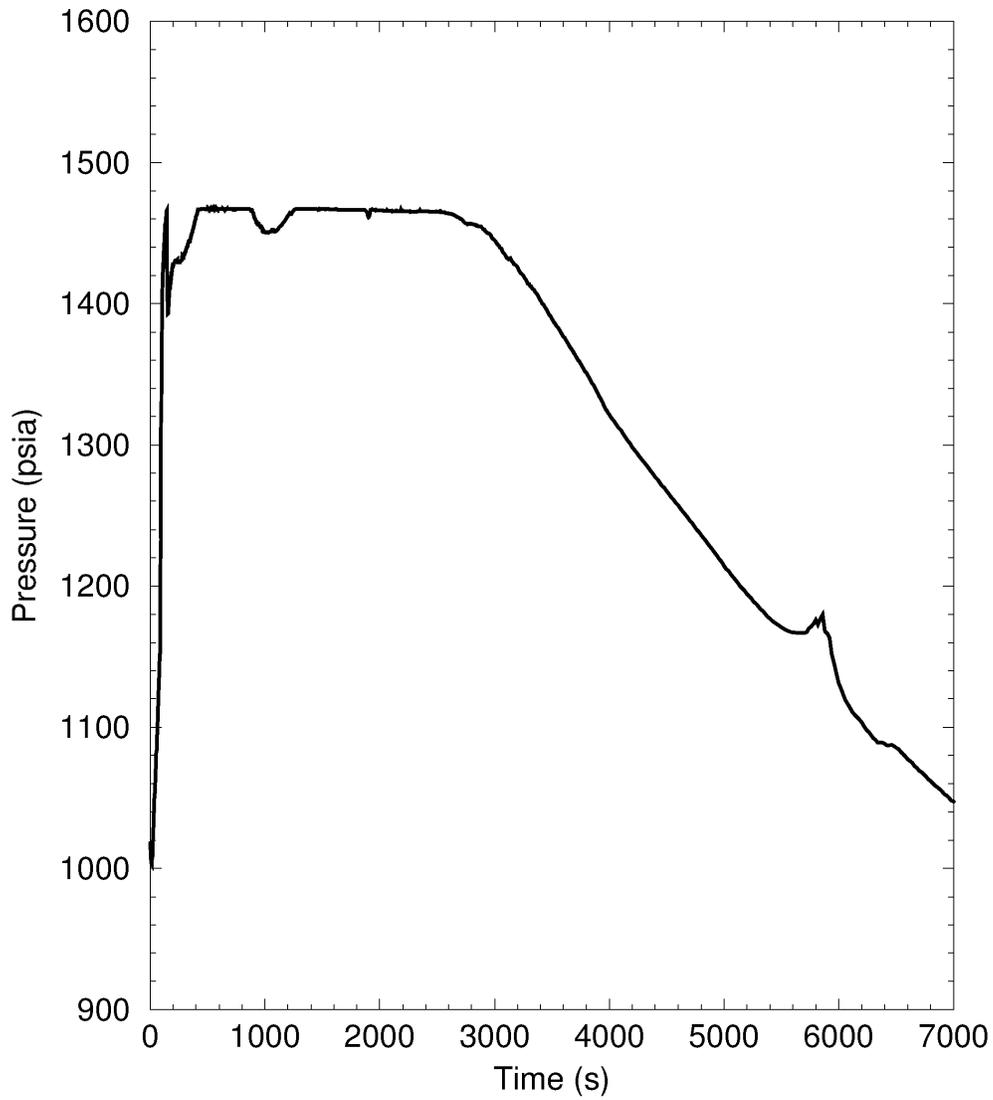
REV 002
EPR4690 T2

Figure 15.2-65—FWLB Representative Small Break - RCS Maximum Pressure



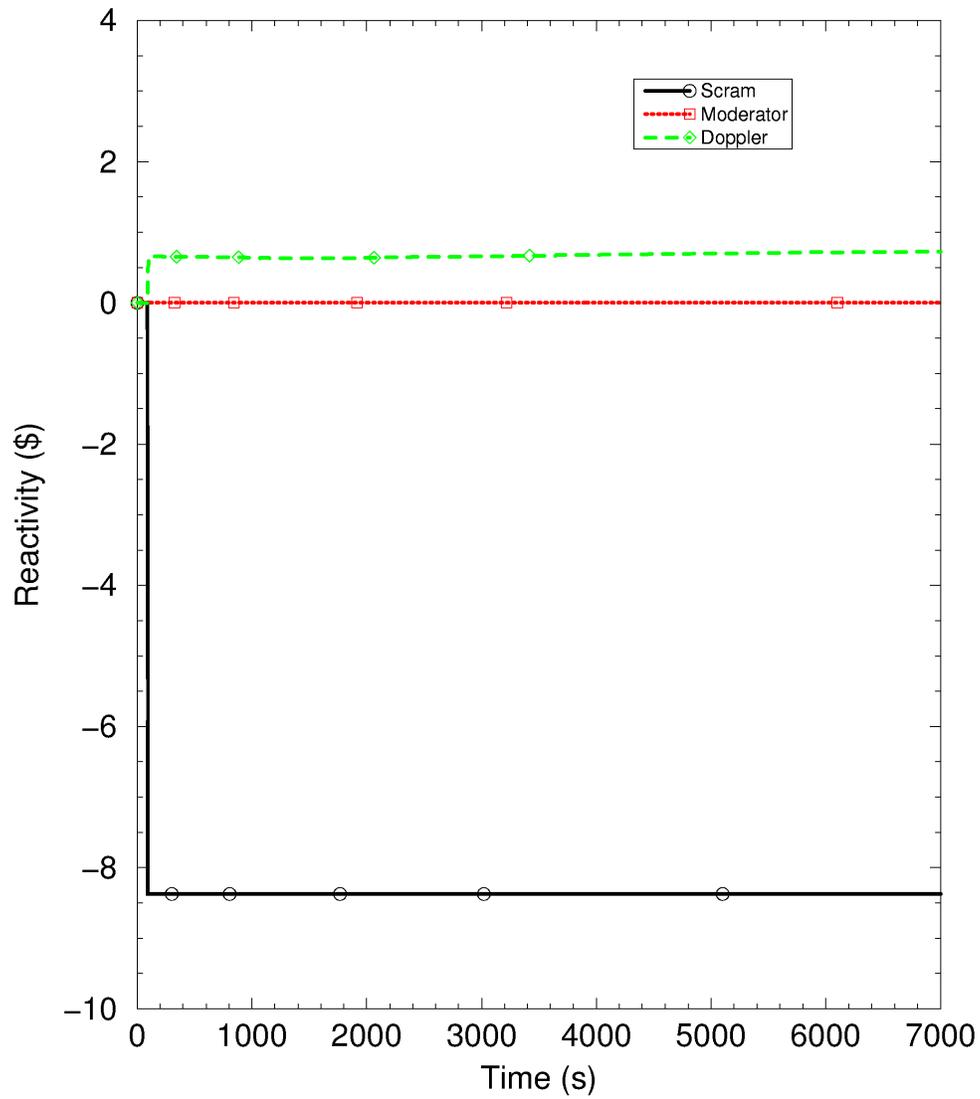
REV 002
EPR4700 T2

Figure 15.2-66—FWLB Representative Small Break - Steam Generator Maximum Pressure



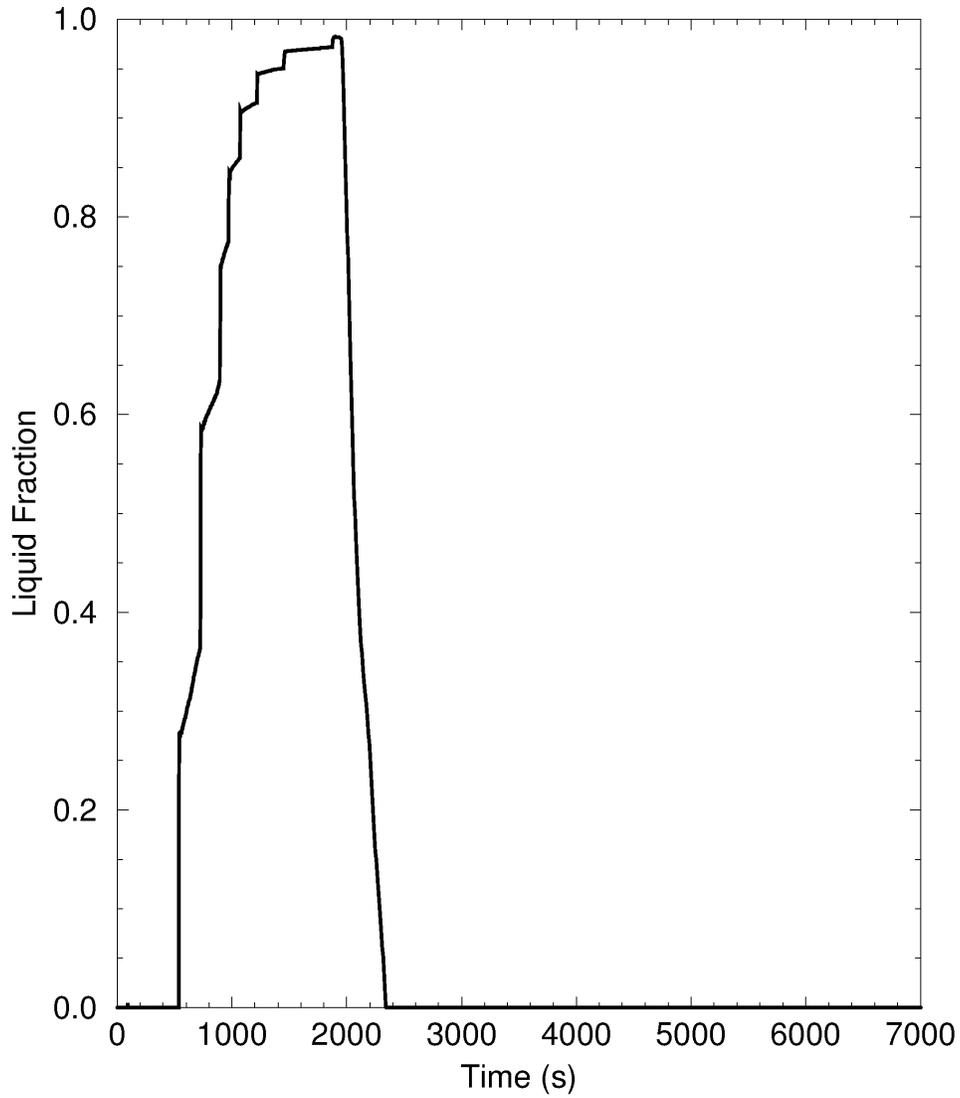
REV 002
EPR4705 T2

Figure 15.2-67—FWLB Representative Small Break - Reactivities



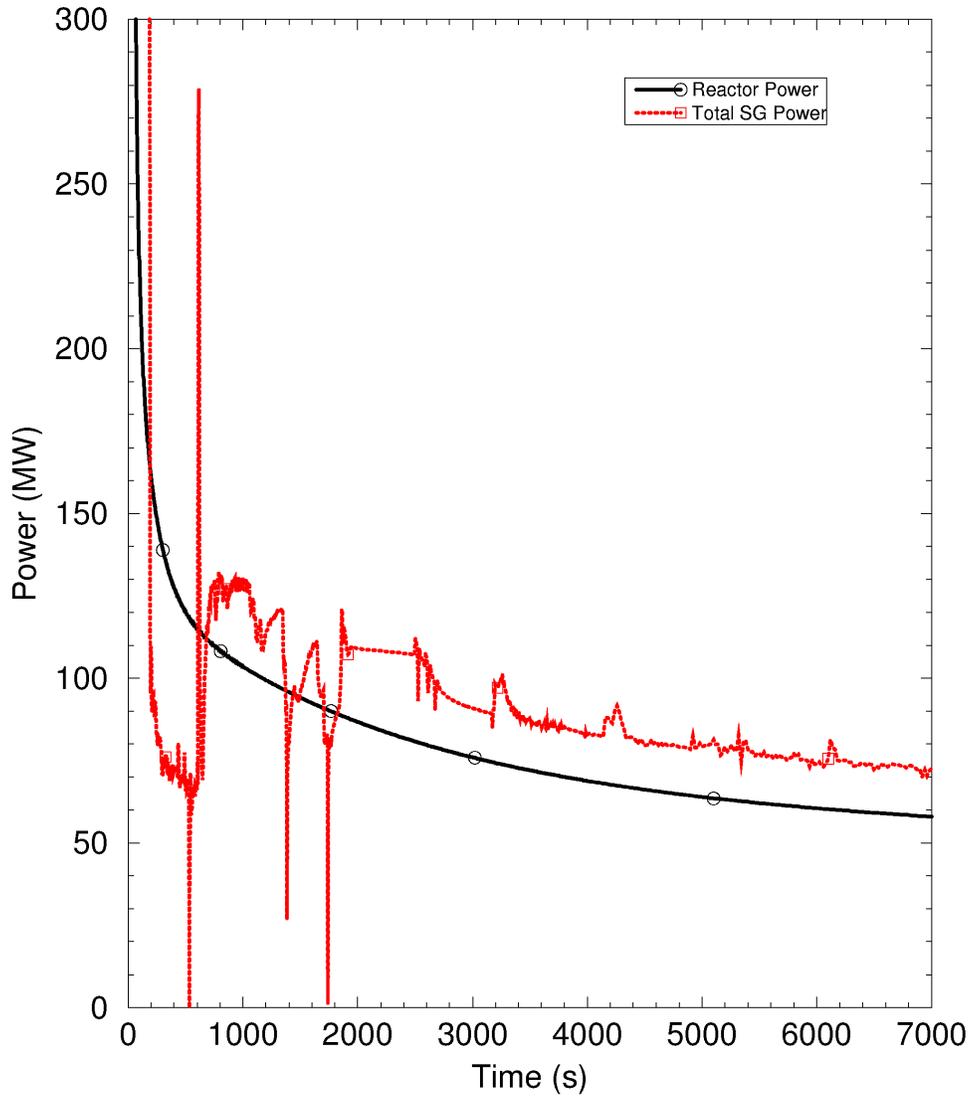
REV 002
EPR4710 T2

Figure 15.2-68—FWLB Representative Small Break - Liquid Volume Fraction in Pressurizer Dome



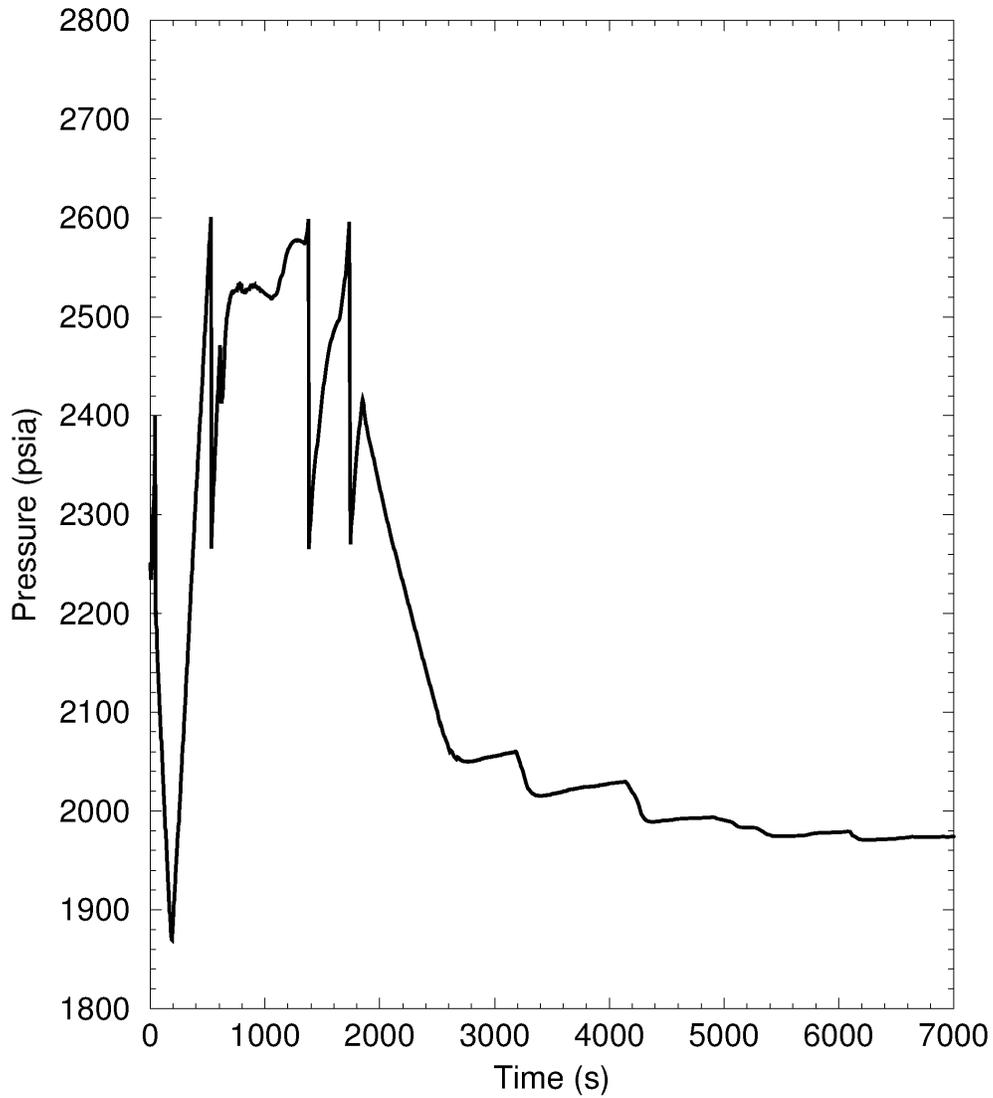
REV 002
EPR4715 T2

Figure 15.2-69—FWLB Maximum RCS Pressure Case - Reactor and Total Steam Generator Power



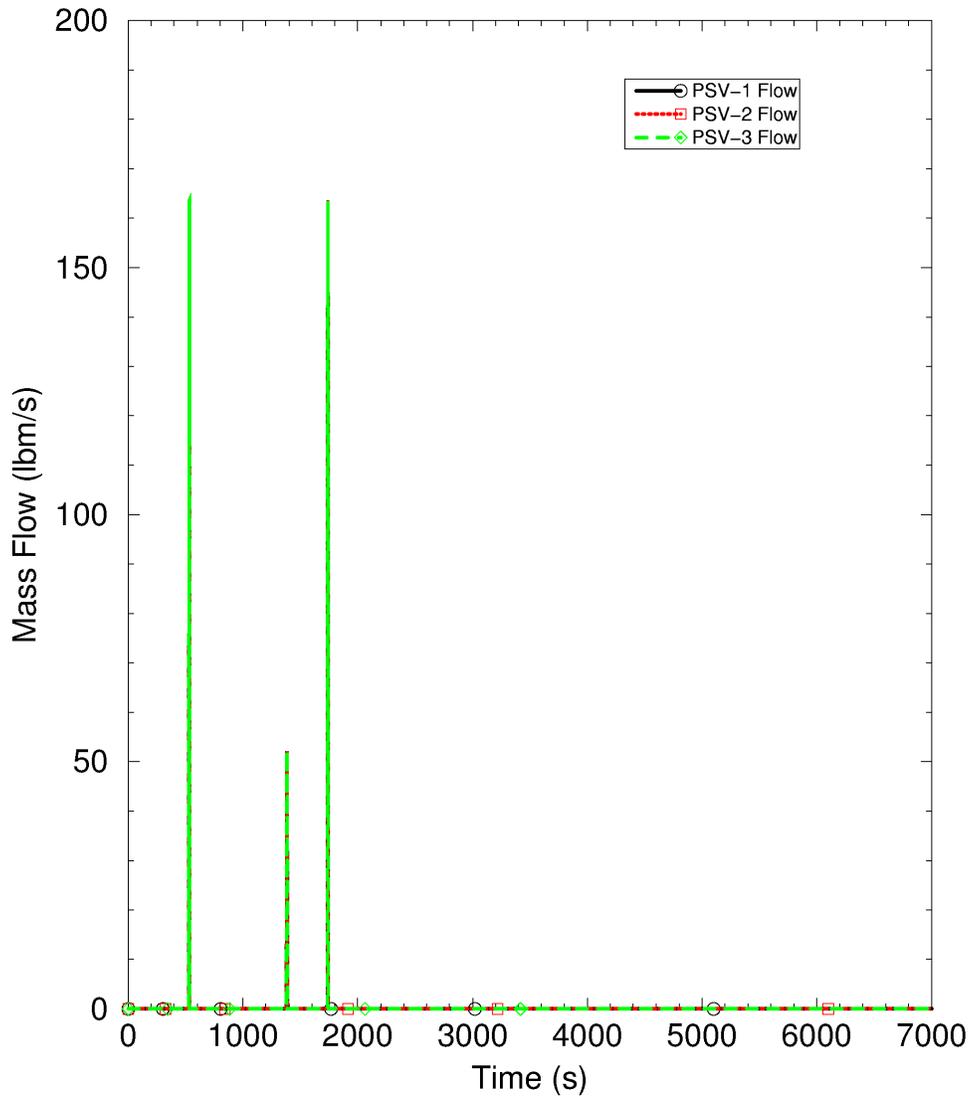
REV 002
EPR4720 T2

Figure 15.2-70—FWLB Maximum RCS Pressure Case - Pressurizer Pressure



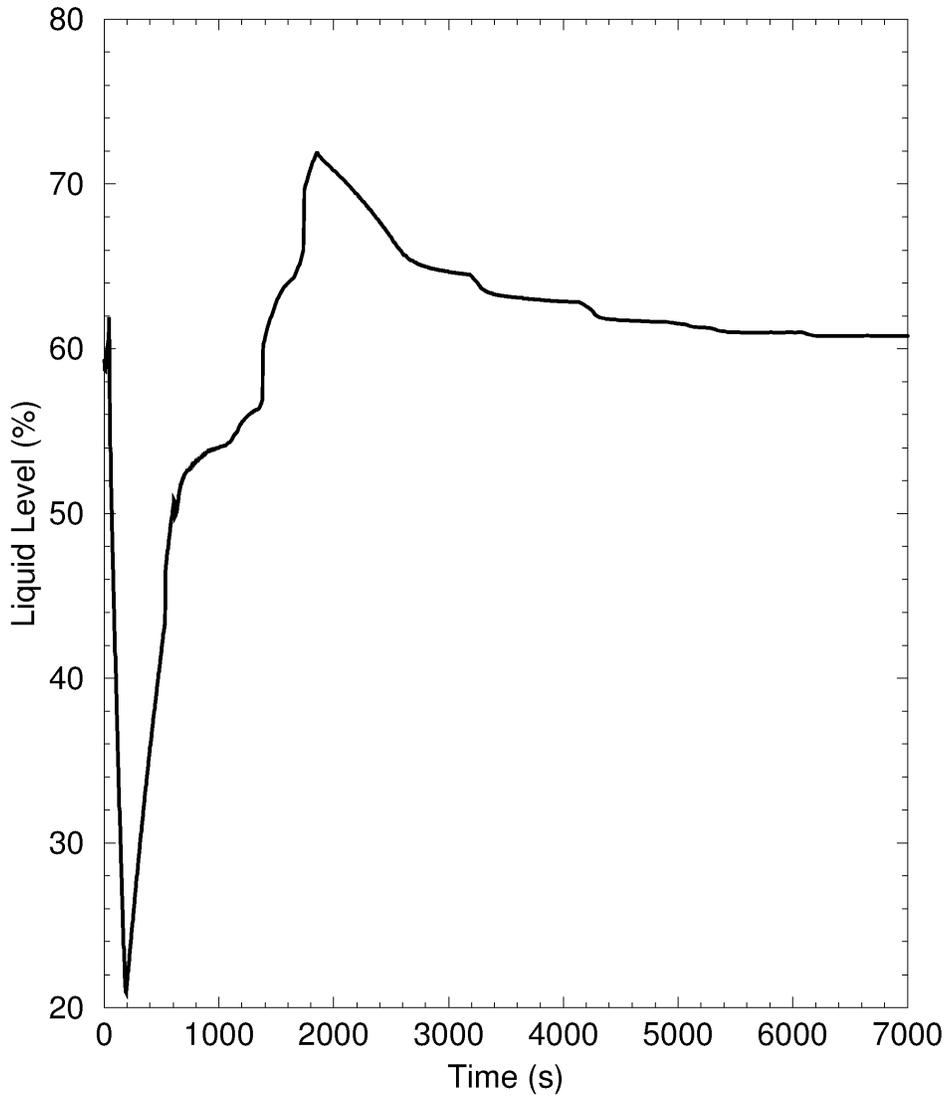
REV 002
EPR4723 T2

Figure 15.2-71—FWLB Maximum RCS Pressure Case - PSRV Flow



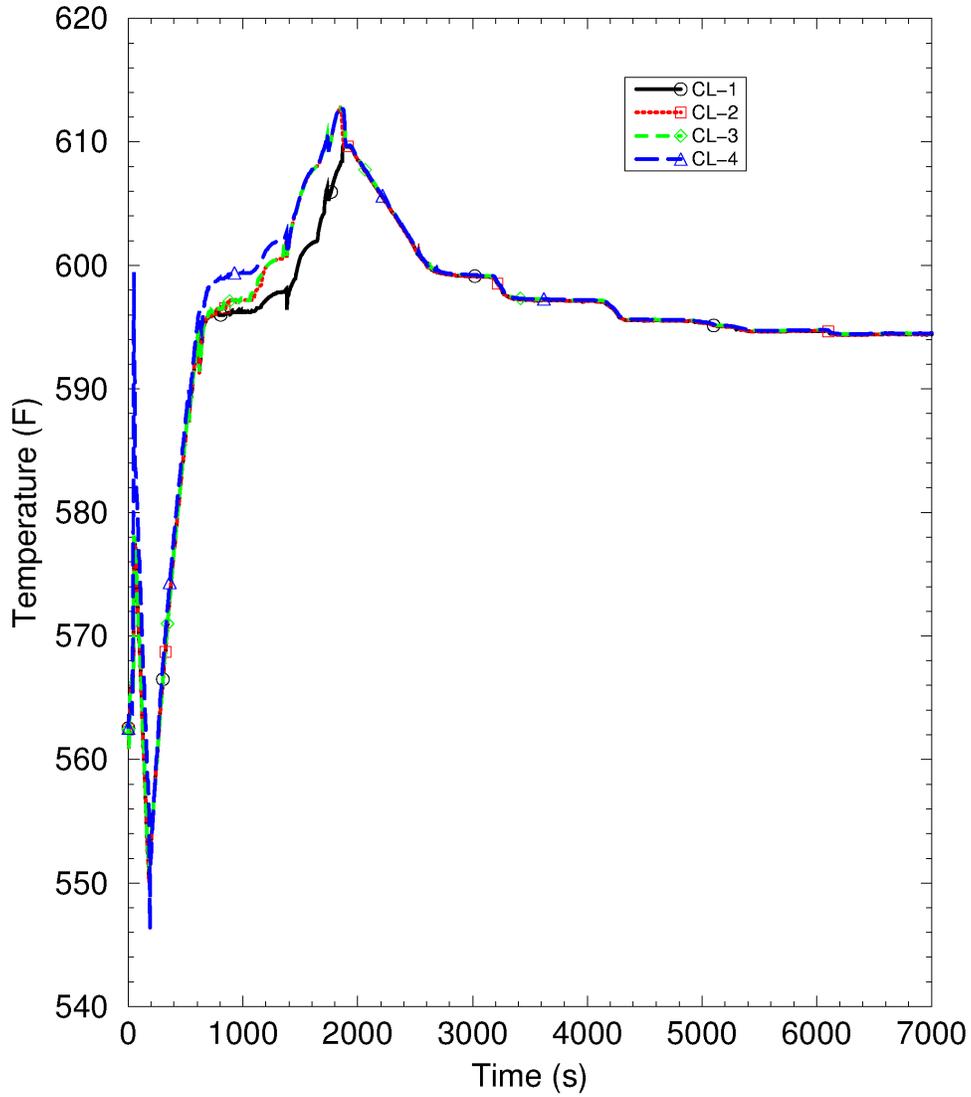
REV 002
EPR4725 T2

Figure 15.2-72—FWLB Maximum RCS Pressure Case - Pressurizer Liquid Level



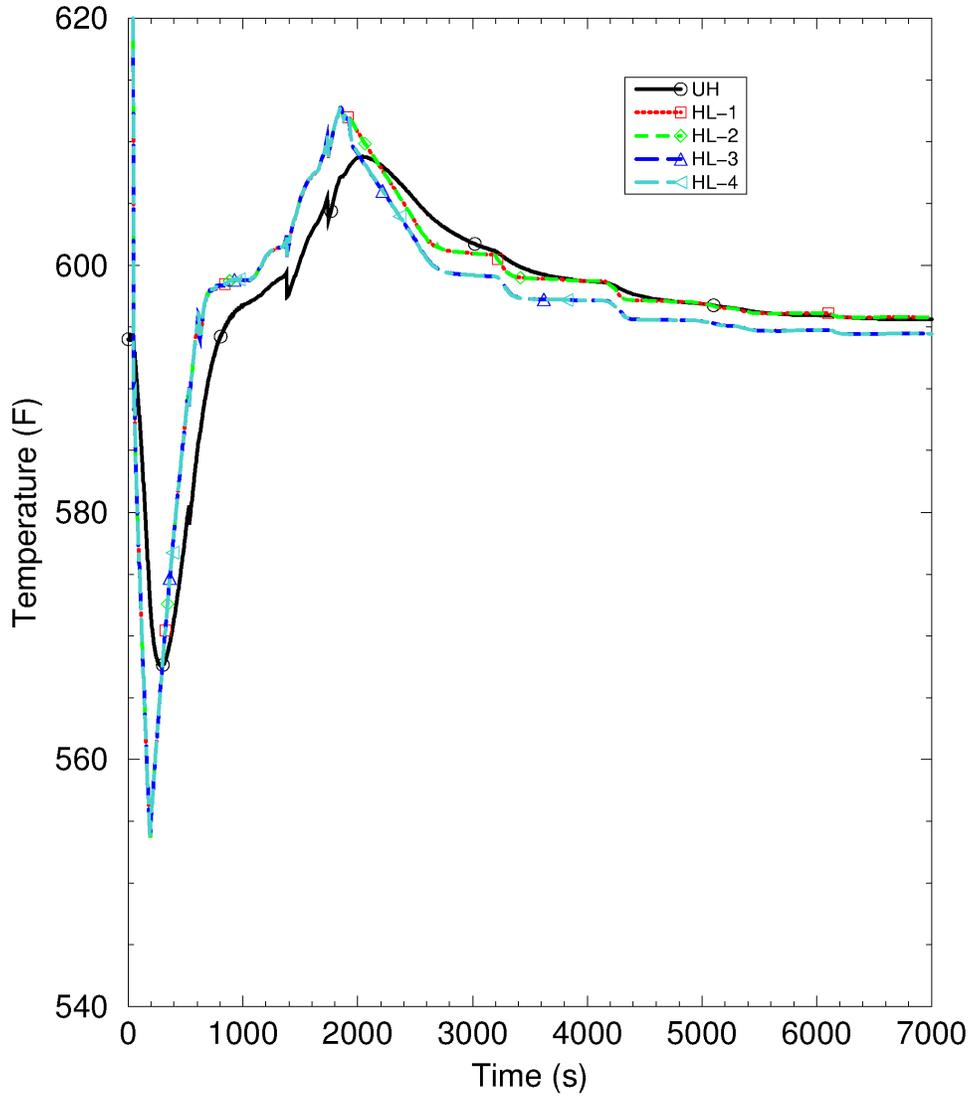
REV 002
EPR4730 T2

Figure 15.2-73—FWLB Maximum RCS Pressure Case - RCS Cold Leg Temperatures



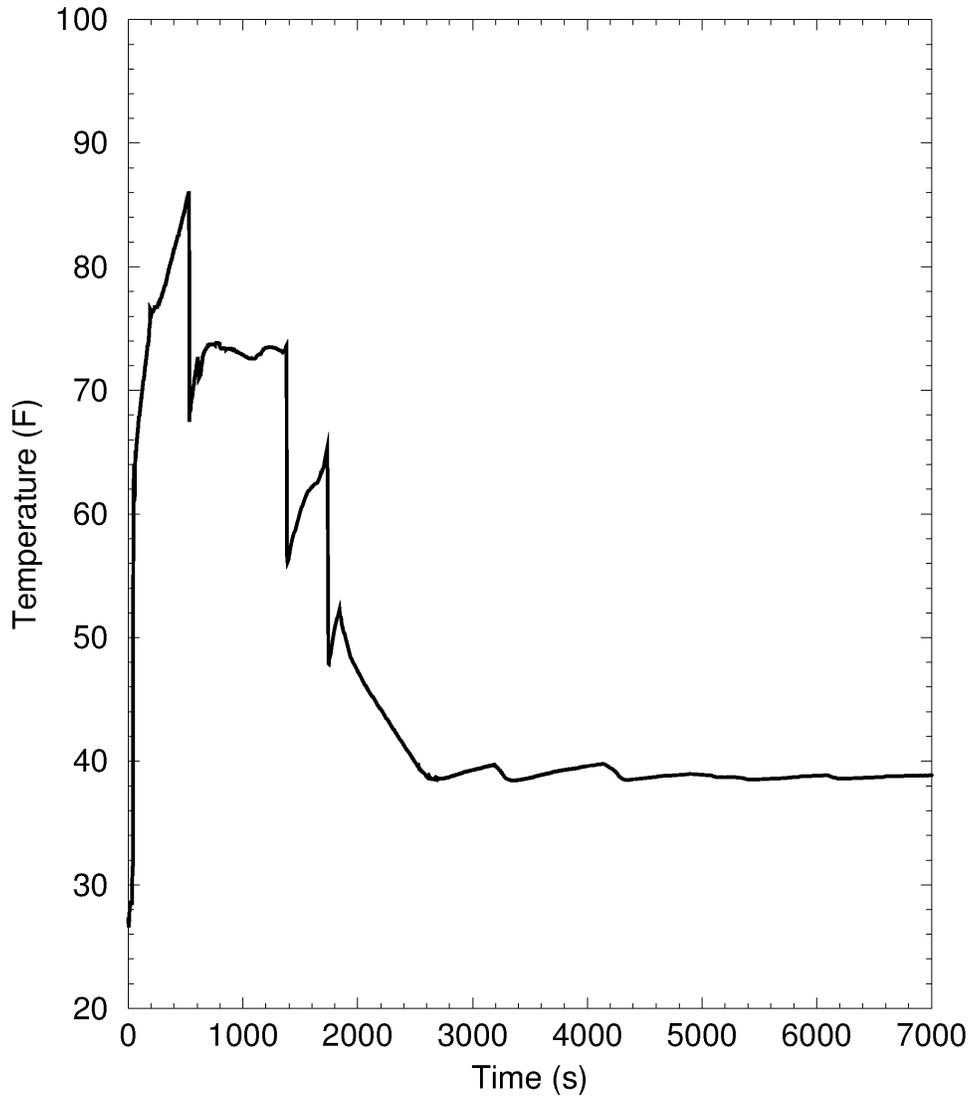
REV 002
EPR4735 T2

Figure 15.2-74—FWLB Maximum RCS Pressure Case - RCS Hot Leg and Upper Head Temperatures



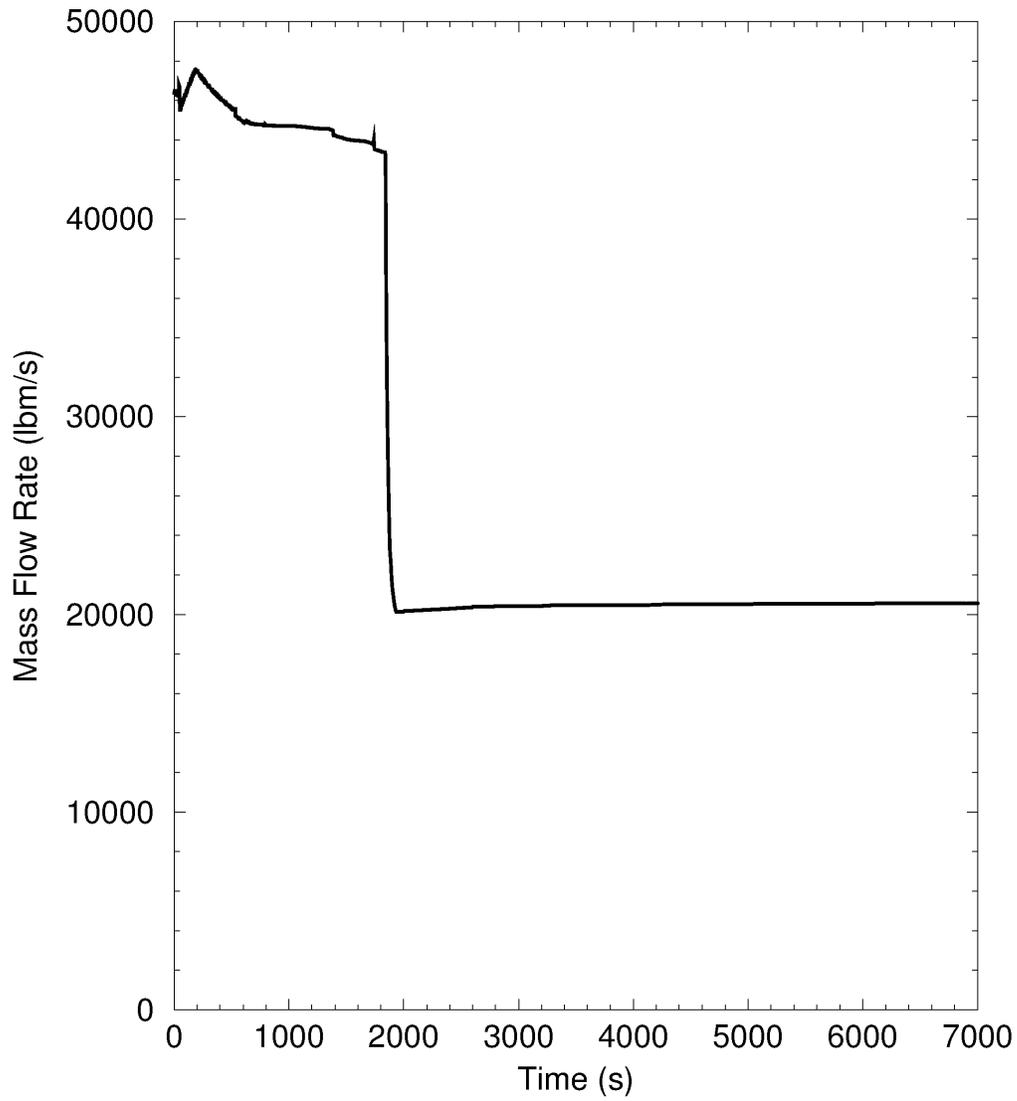
REV 002
EPR4740 T2

Figure 15.2-75—FWLB Maximum RCS Pressure Case - Core Exit Subcooling



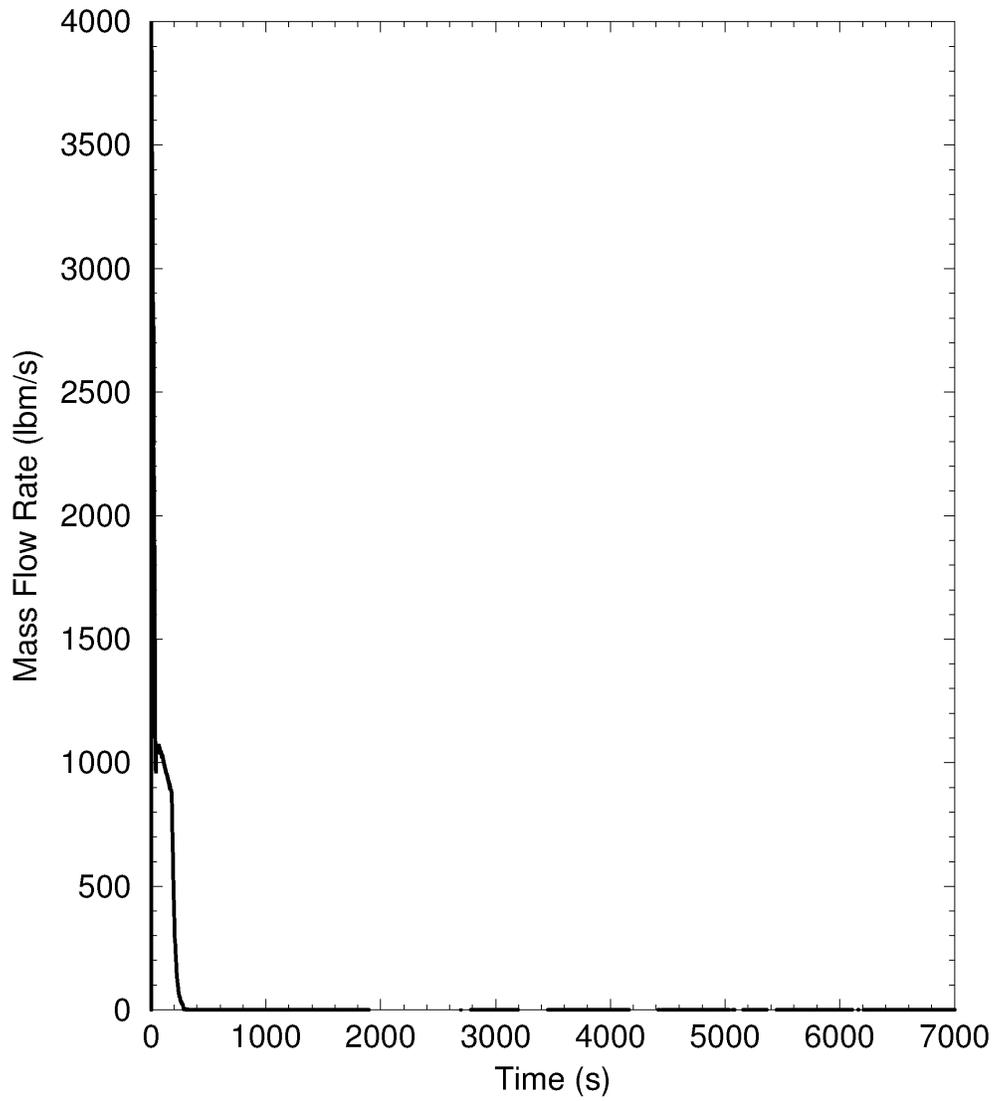
REV 002
EPR4745 T2

Figure 15.2-76—FWLB Maximum RCS Pressure Case - Core Flow



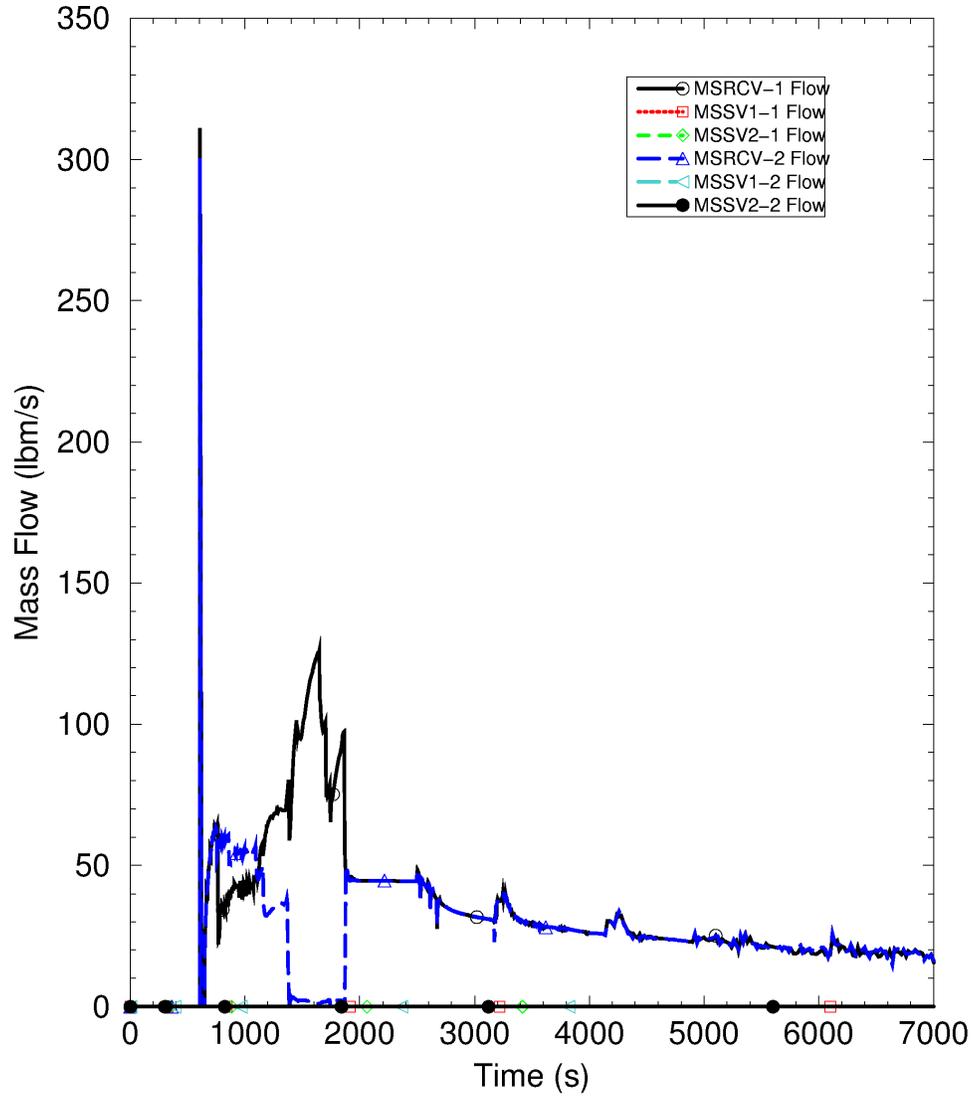
REV 002
EPR4750 T2

Figure 15.2-77—FWLB Maximum RCS Pressure Case - Break Flow



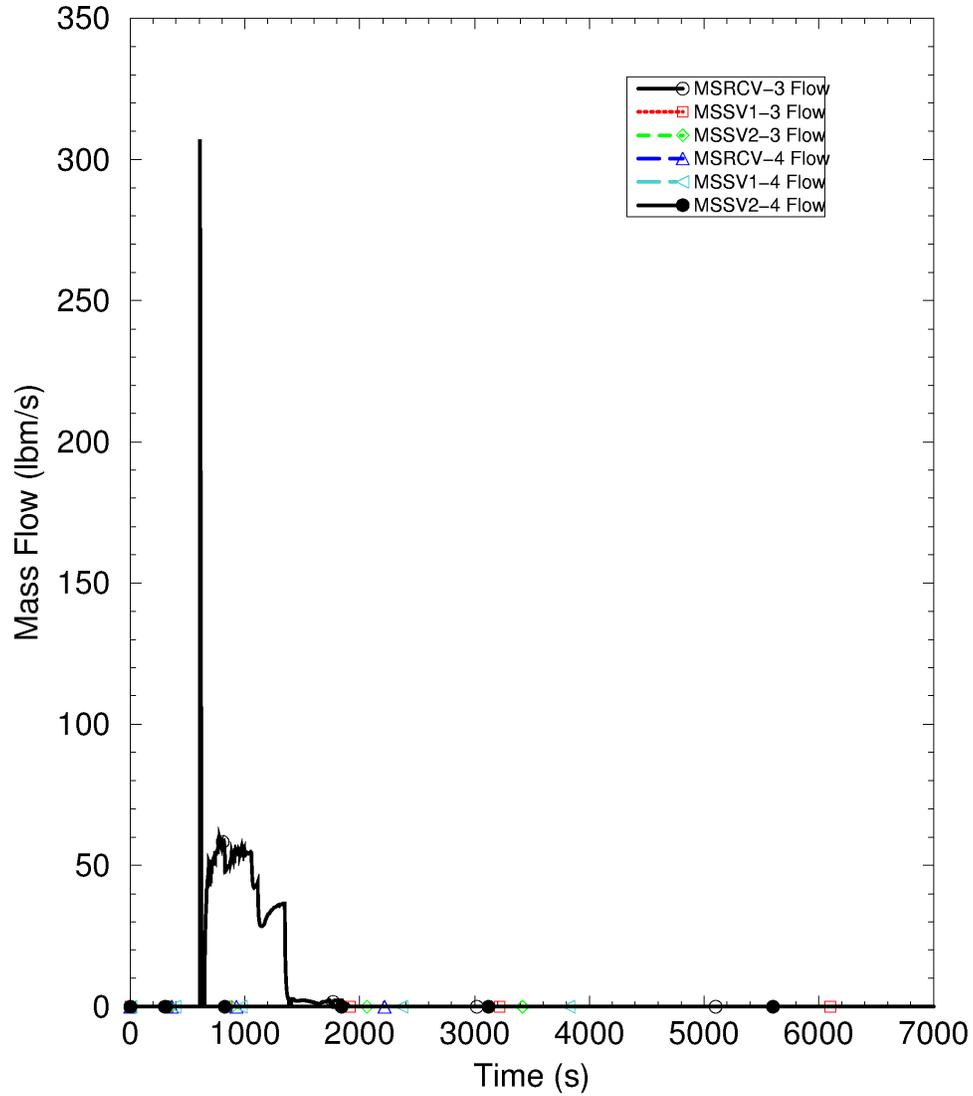
REV 002
EPR4755 T2

Figure 15.2-78—FWLB Maximum RCS Pressure Case - Main Steam Relief
Loops 1 and 2



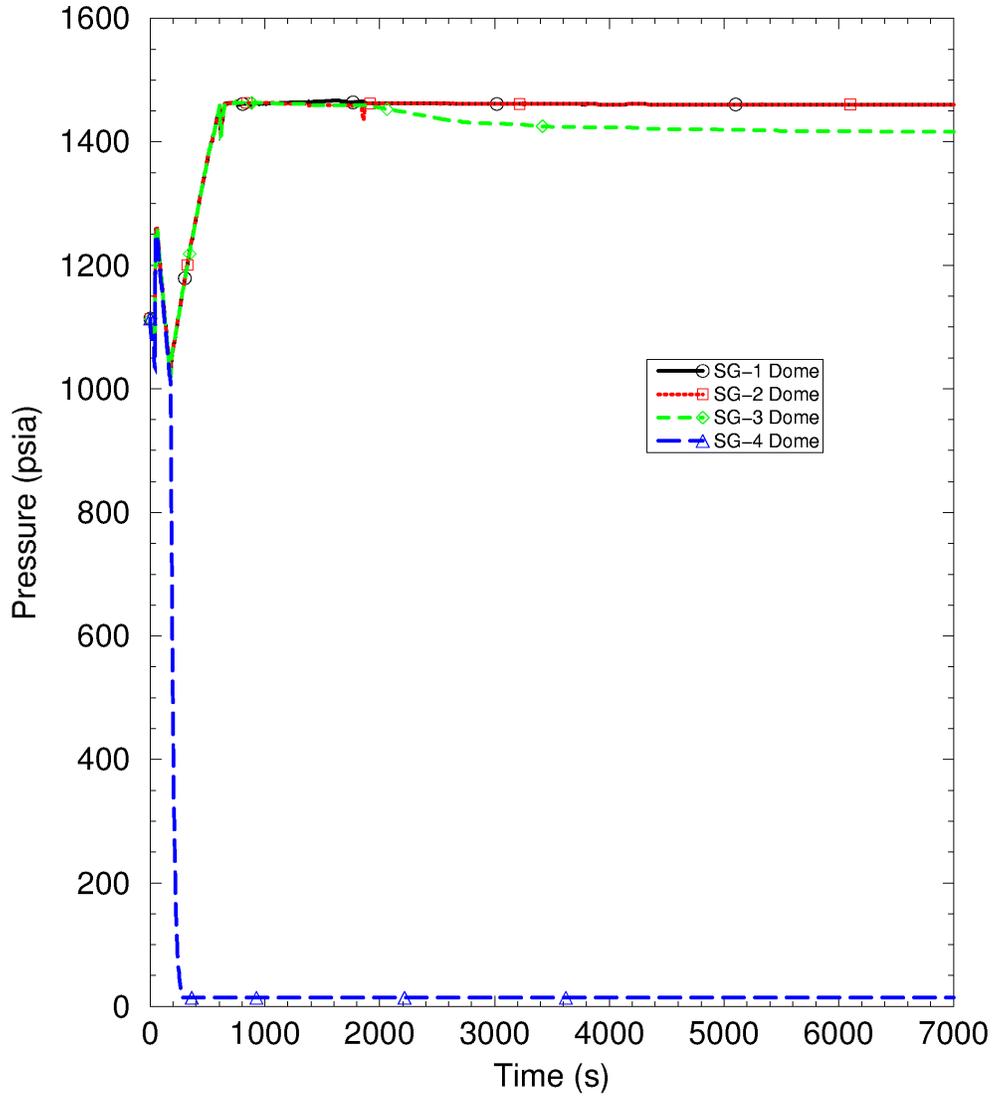
REV 002
EPR4760 T2

Figure 15.2-79—FWLB Maximum RCS Pressure Case - Main Steam Relief Loops 3 and 4



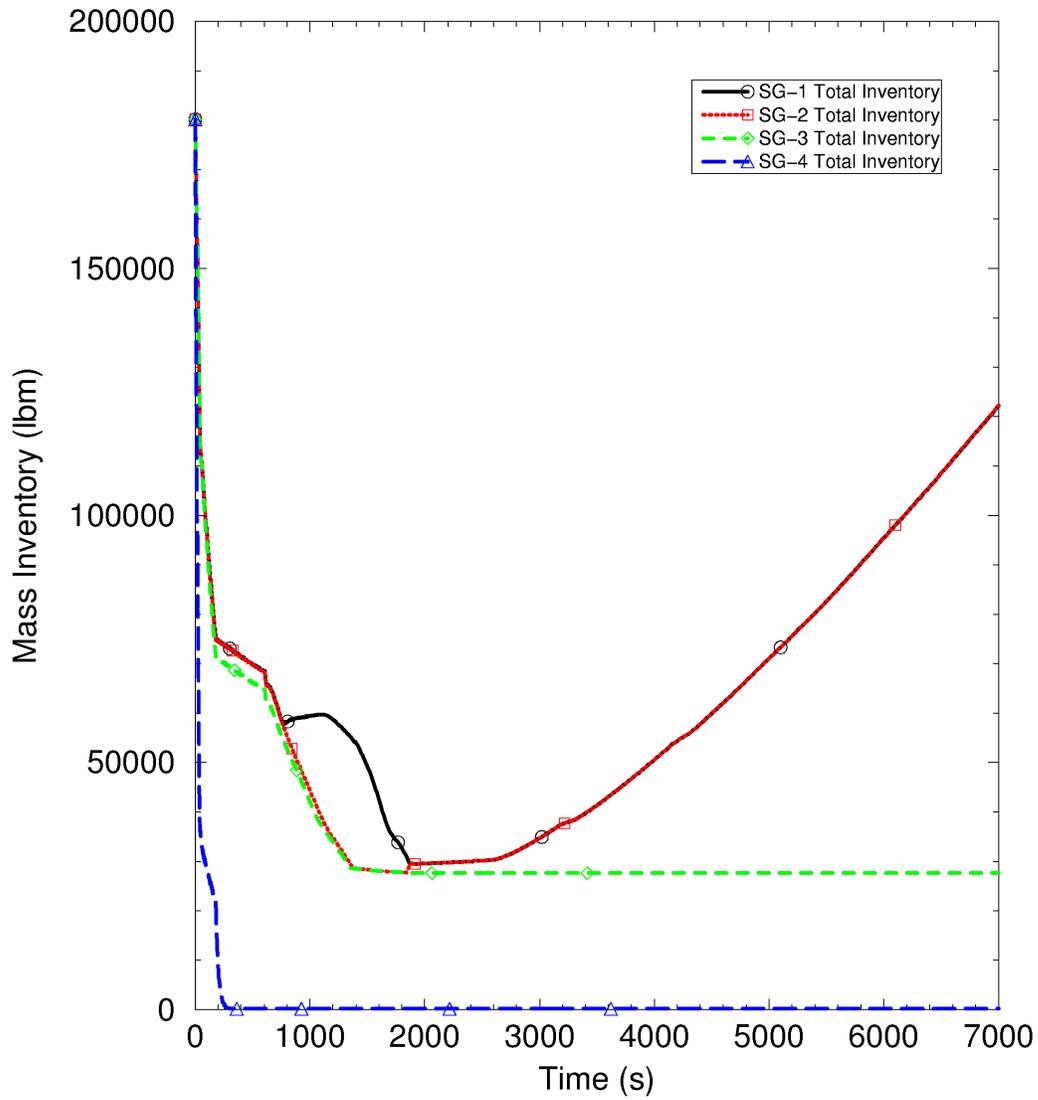
REV 002
EPR4765 T2

Figure 15.2-80—FWLB Maximum RCS Pressure Case - Steam Generator Dome Pressures



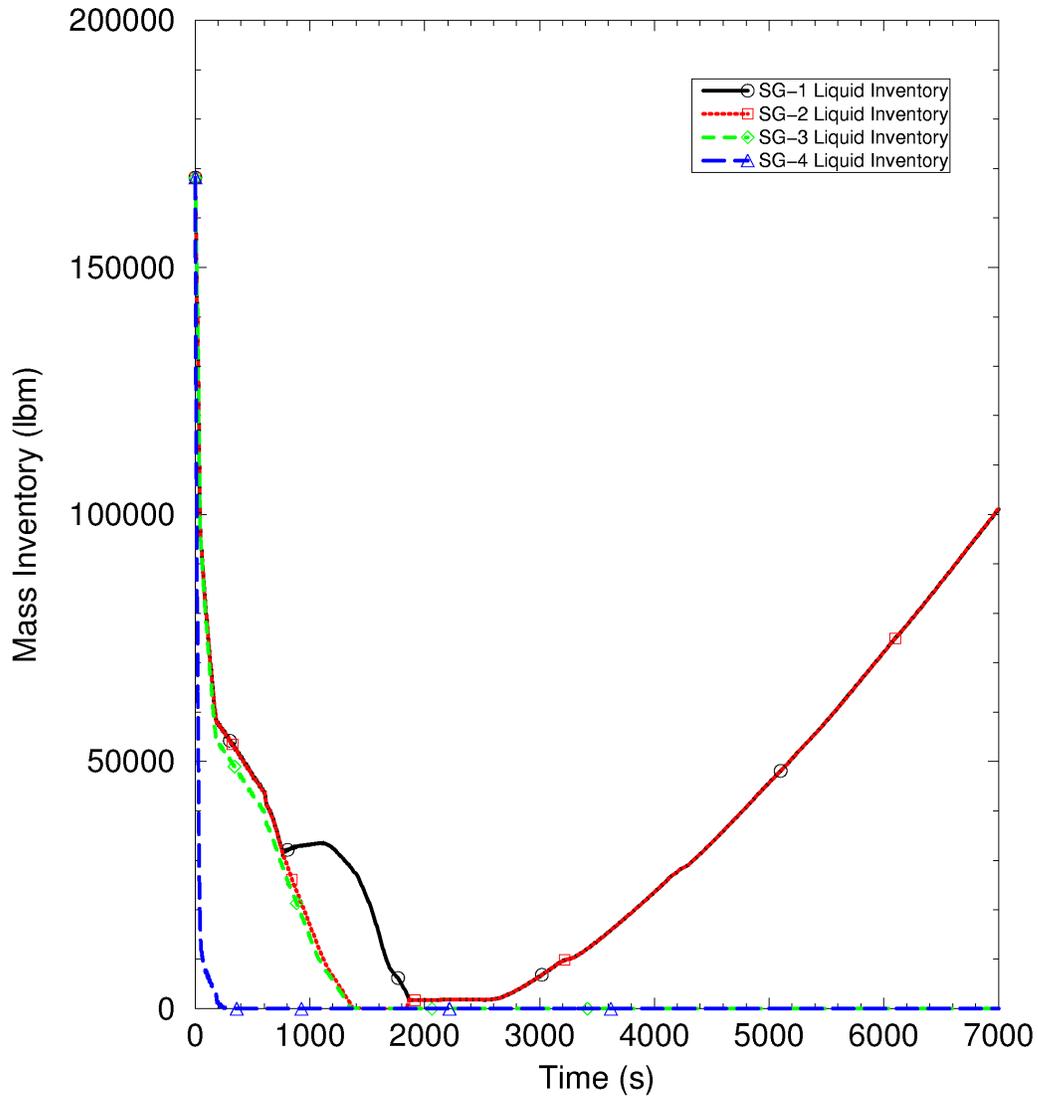
REV 002
EPR4770 T2

Figure 15.2-81—FWLB Maximum RCS Pressure Case - Steam Generator Total Mass



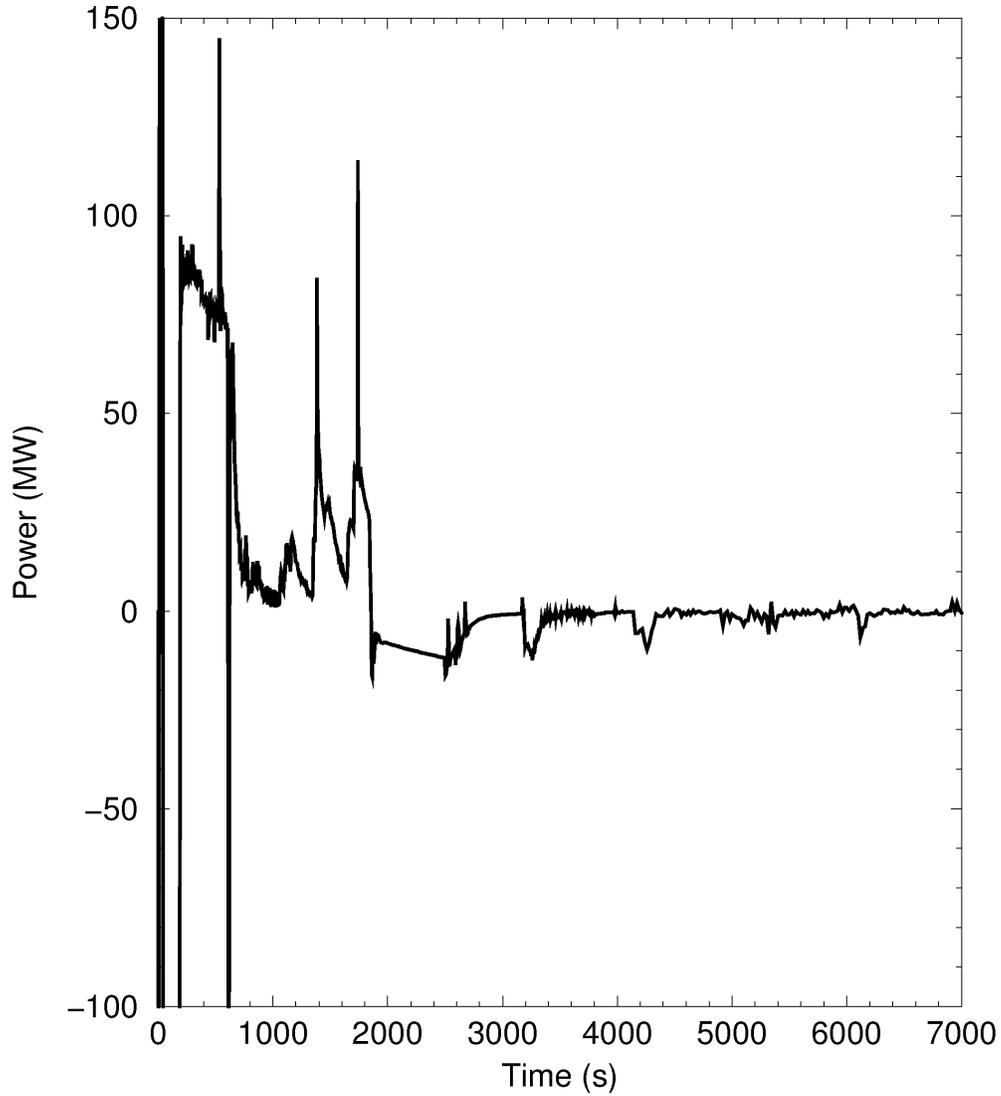
REV 002
EPR4775 T2

Figure 15.2-82—FWLB Maximum RCS Pressure Case - Steam Generator Liquid Mass



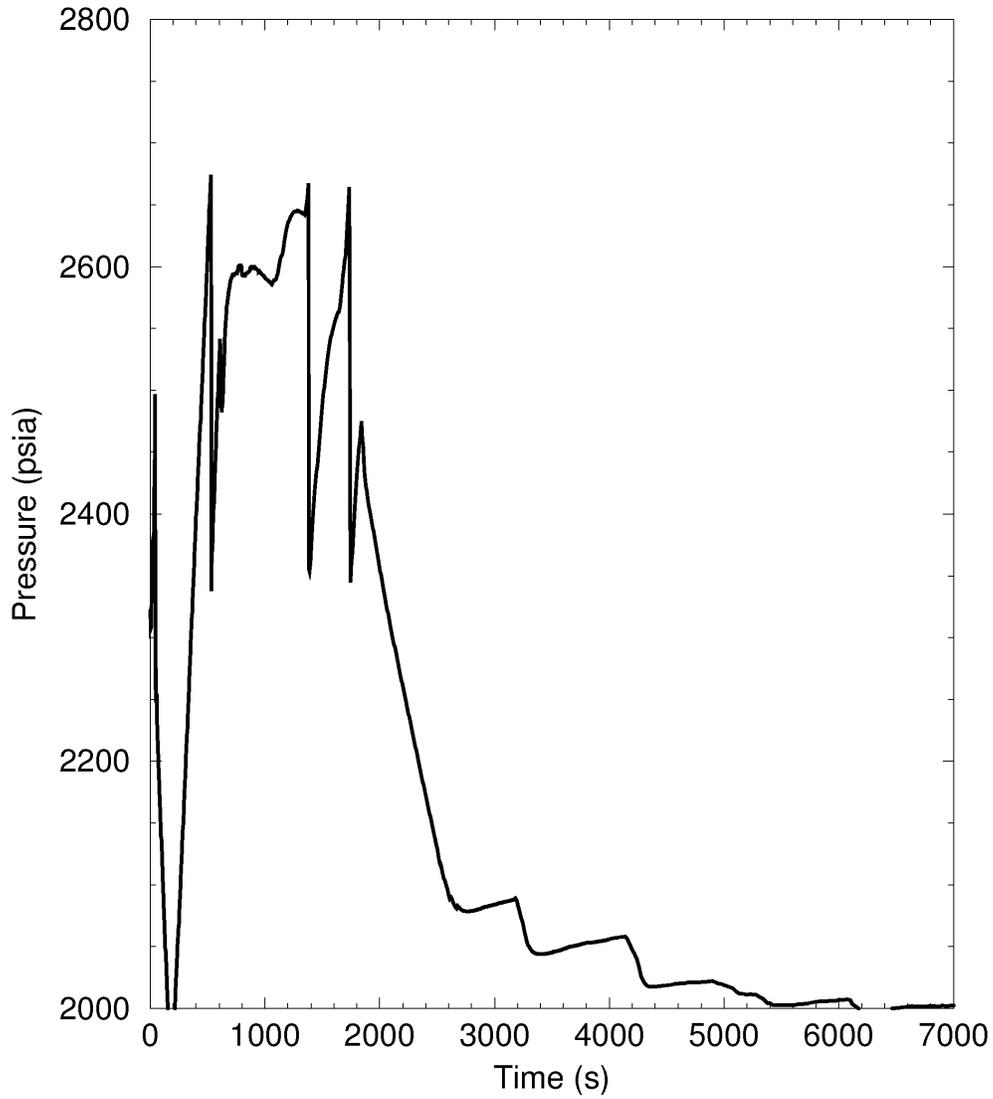
REV 002
EPR4780 T2

Figure 15.2-83—FWLB Maximum RCS Pressure Case - Net Heat Addition to RCS



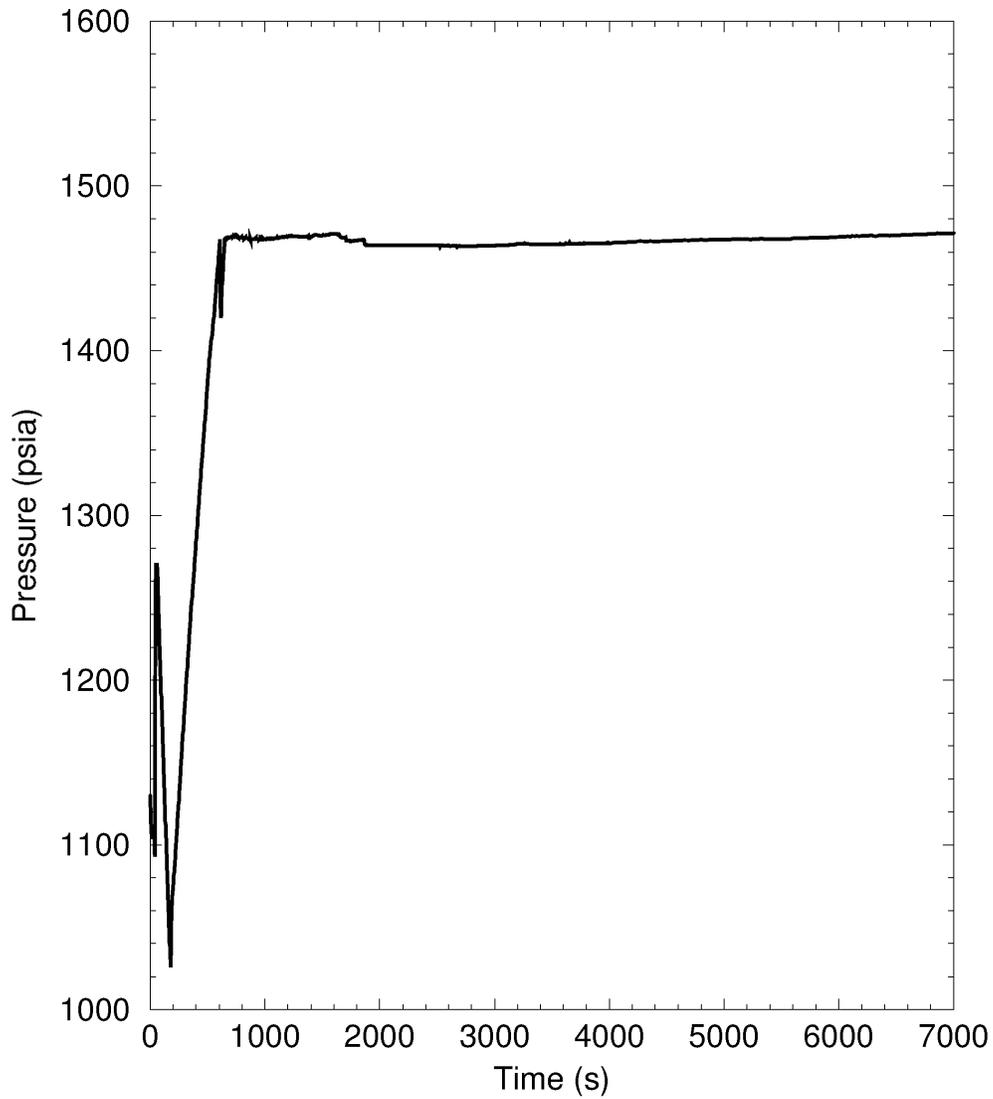
REV 002
EPR4785 T2

Figure 15.2-84—FWLB Maximum RCS Pressure Case - RCS Maximum Pressure



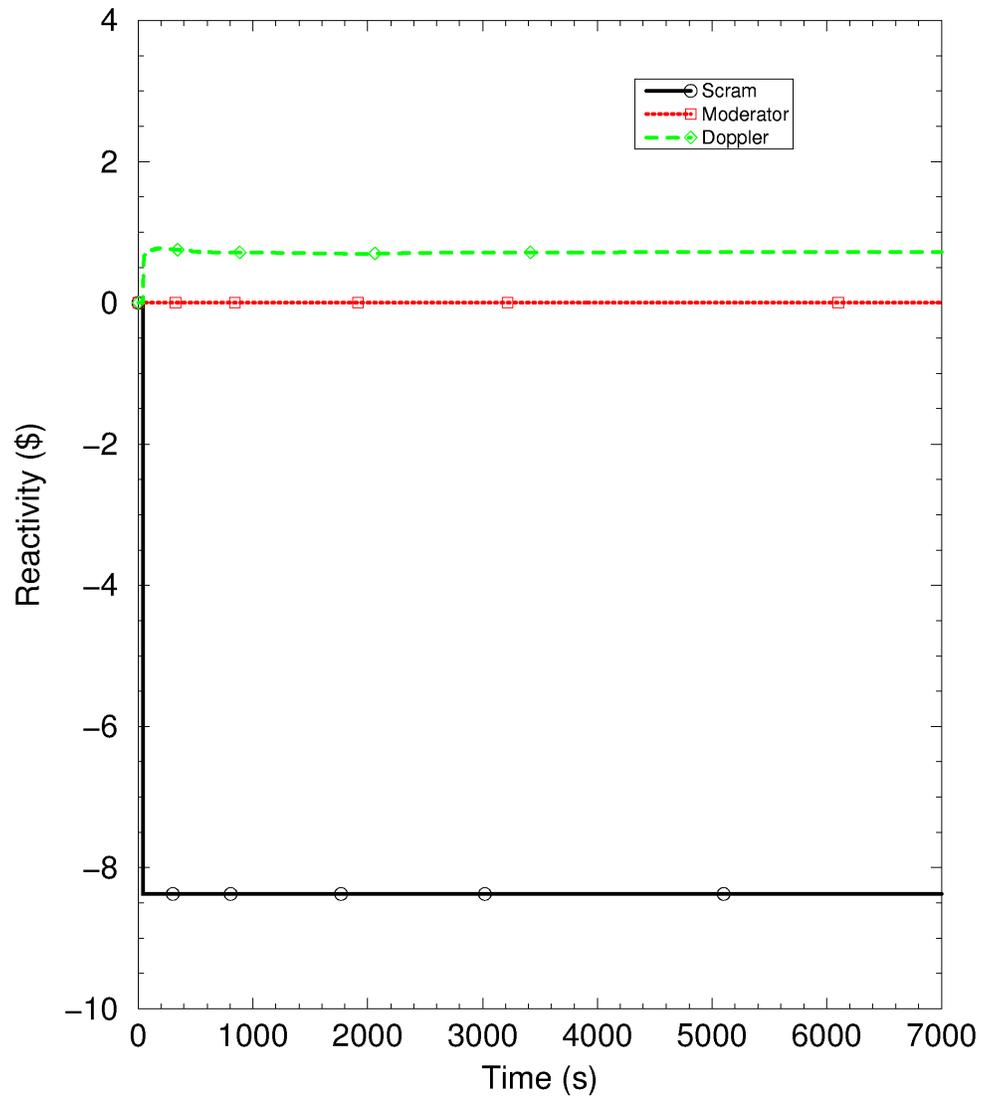
REV 002
EPR4790 T2

Figure 15.2-85—FWLB Maximum RCS Pressure Case - Steam Generator Maximum Pressure



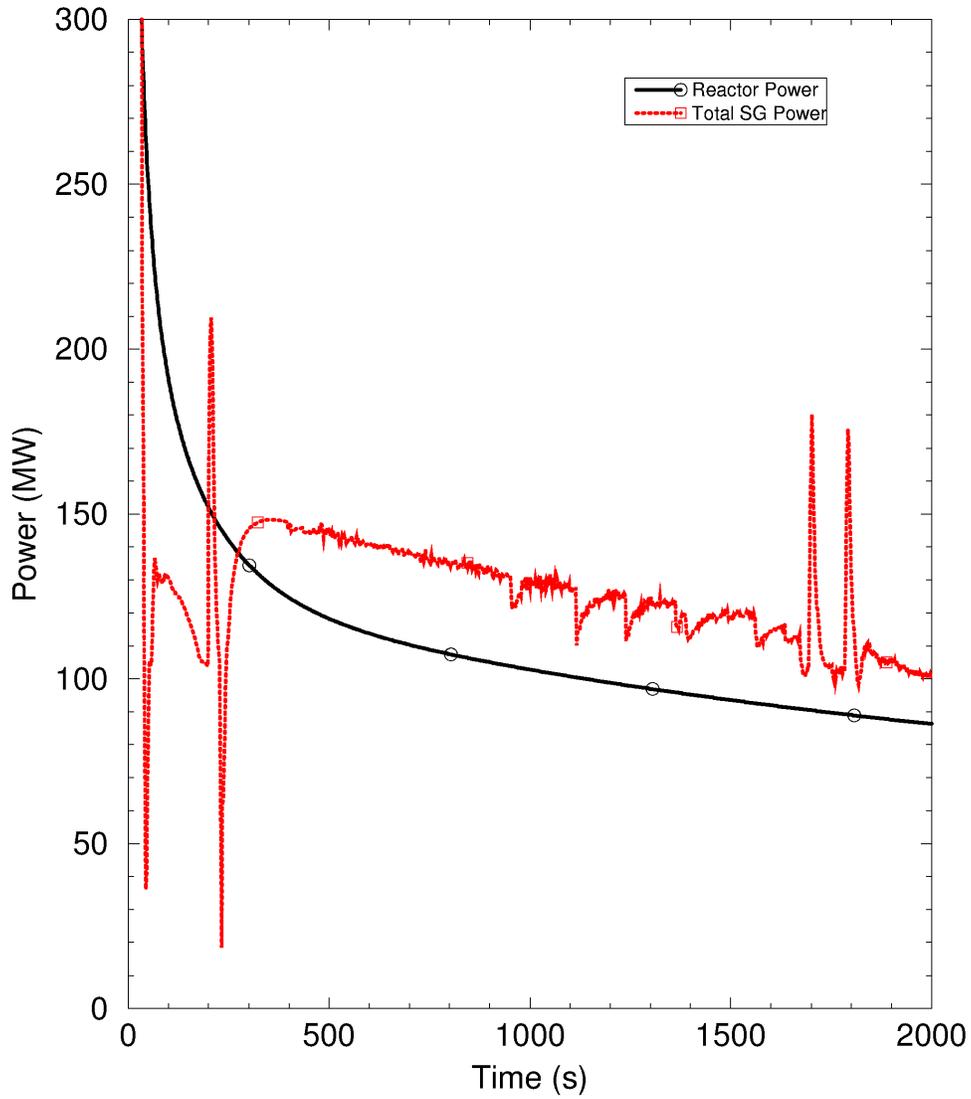
REV 002
EPR4795 T2

Figure 15.2-86—FWLB Maximum RCS Pressure Case - Reactivities



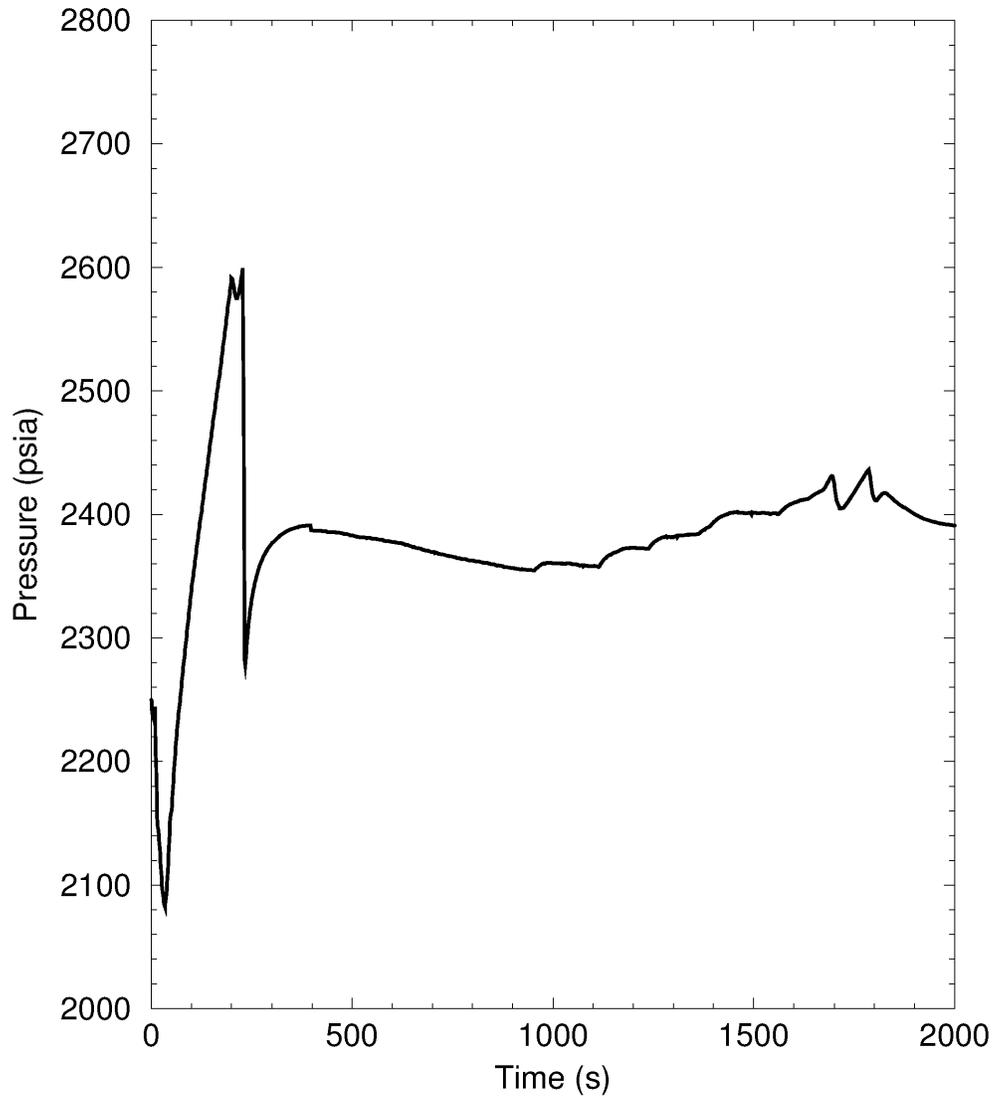
REV 002
EPR4800 T2

Figure 15.2-87—FWLB Maximum Secondary Pressure Case - Reactor and Total Steam Generator Power



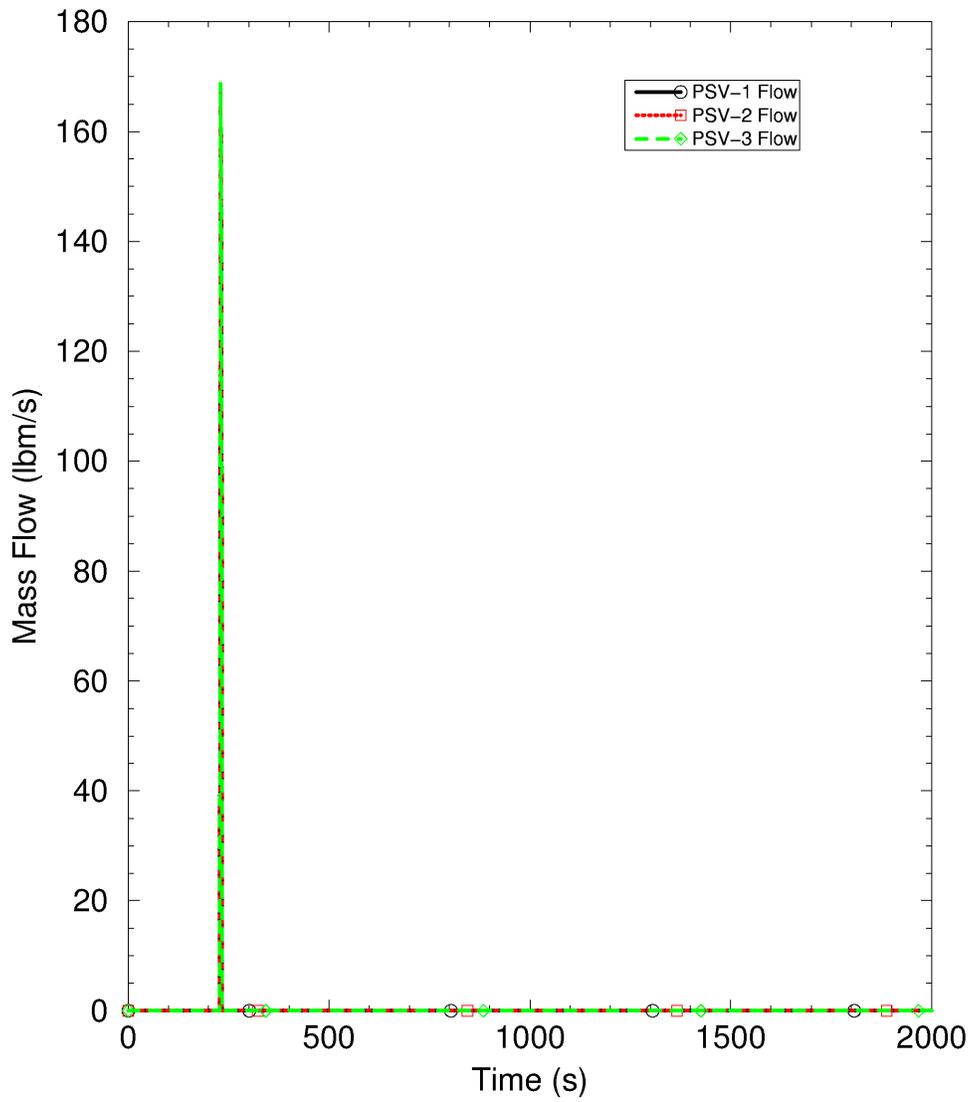
REV 002
EPR4805 T2

Figure 15.2-88—FWLB Maximum Secondary Pressure Case - Pressurizer Pressure



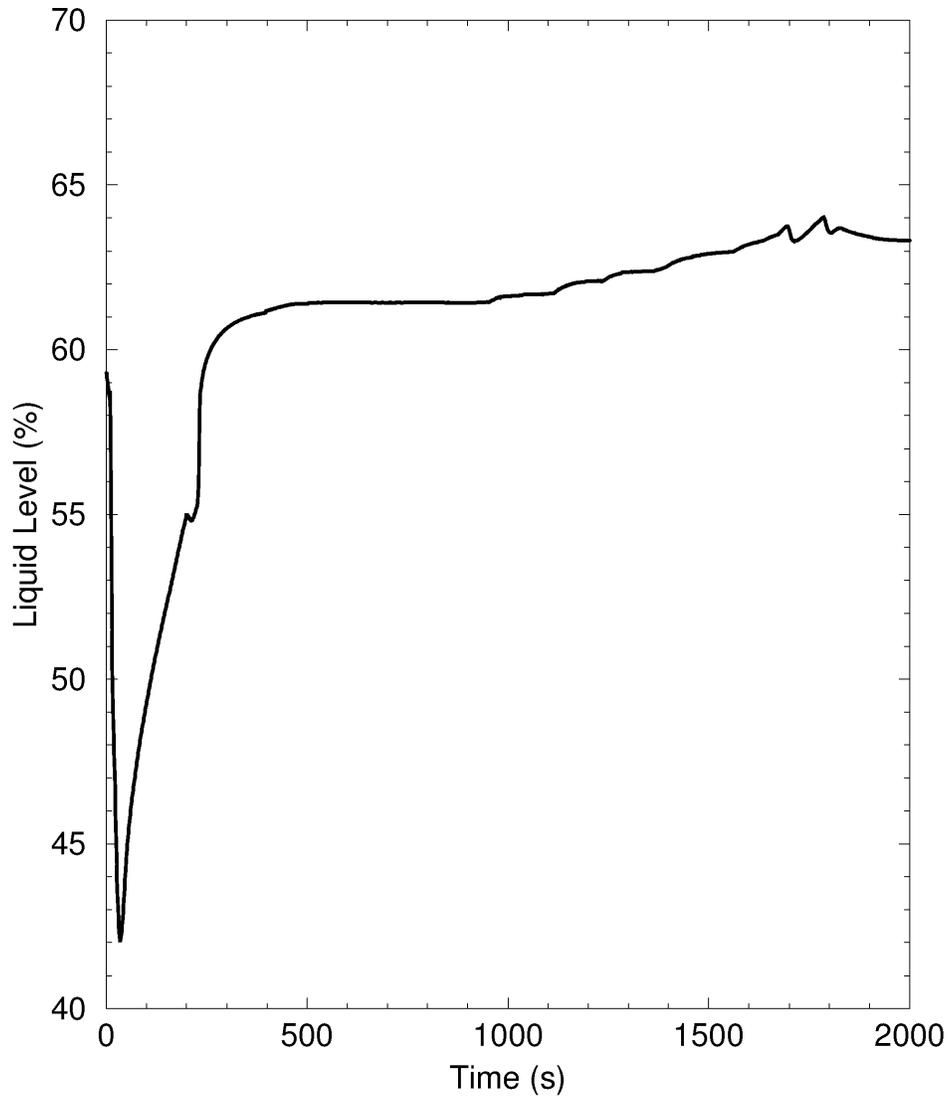
REV 002
EPR4810 T2

Figure 15.2-89—FWLB Maximum Secondary Pressure Case - PSRV Flow



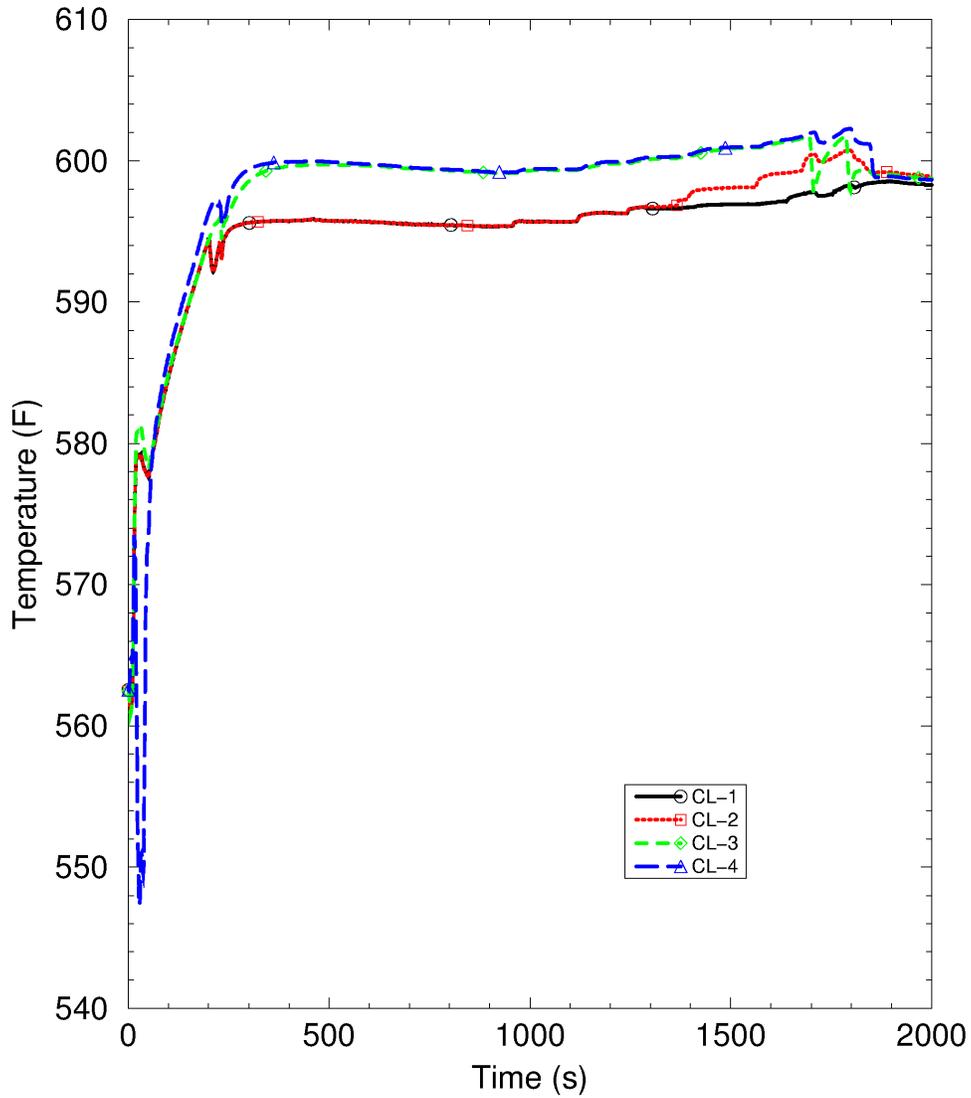
REV 002
EPR4815 T2

Figure 15.2-90—FWLB Maximum Secondary Pressure Case - Pressurizer Liquid Level



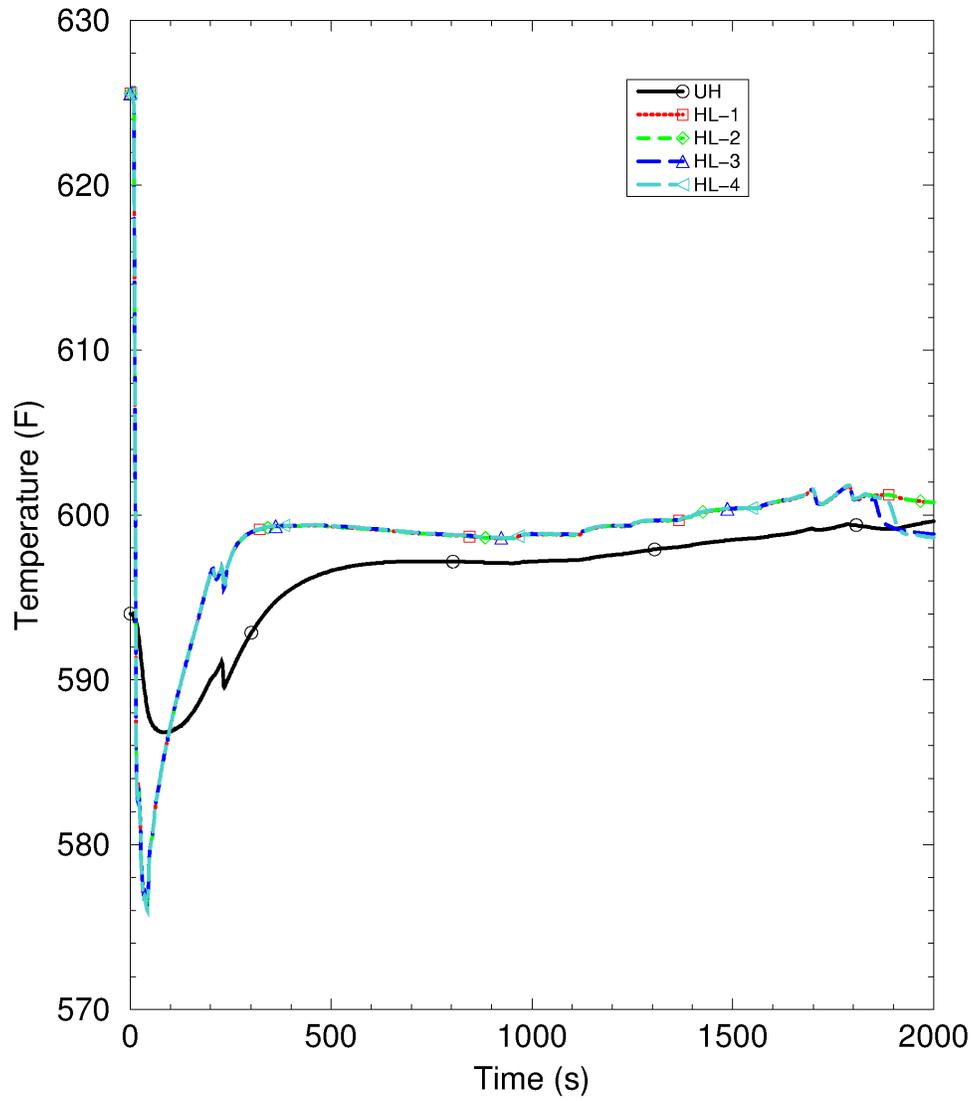
REV 002
EPR4820 T2

Figure 15.2-91—FWLB Maximum Secondary Pressure Case - RCS Cold Leg Temperatures



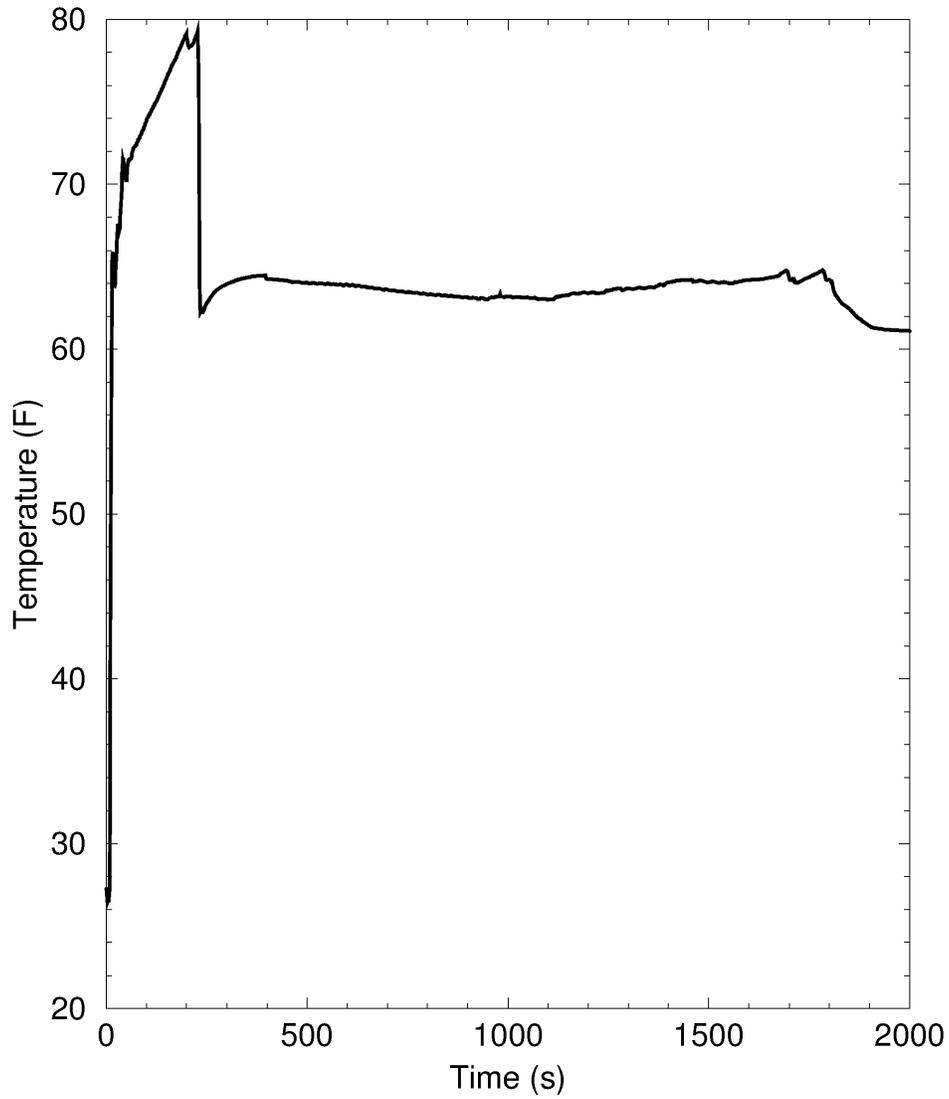
REV 002
EPR4825 T2

Figure 15.2-92—FWLB Maximum Secondary Pressure Case - RCS Hot Leg and Upper Head Temperature



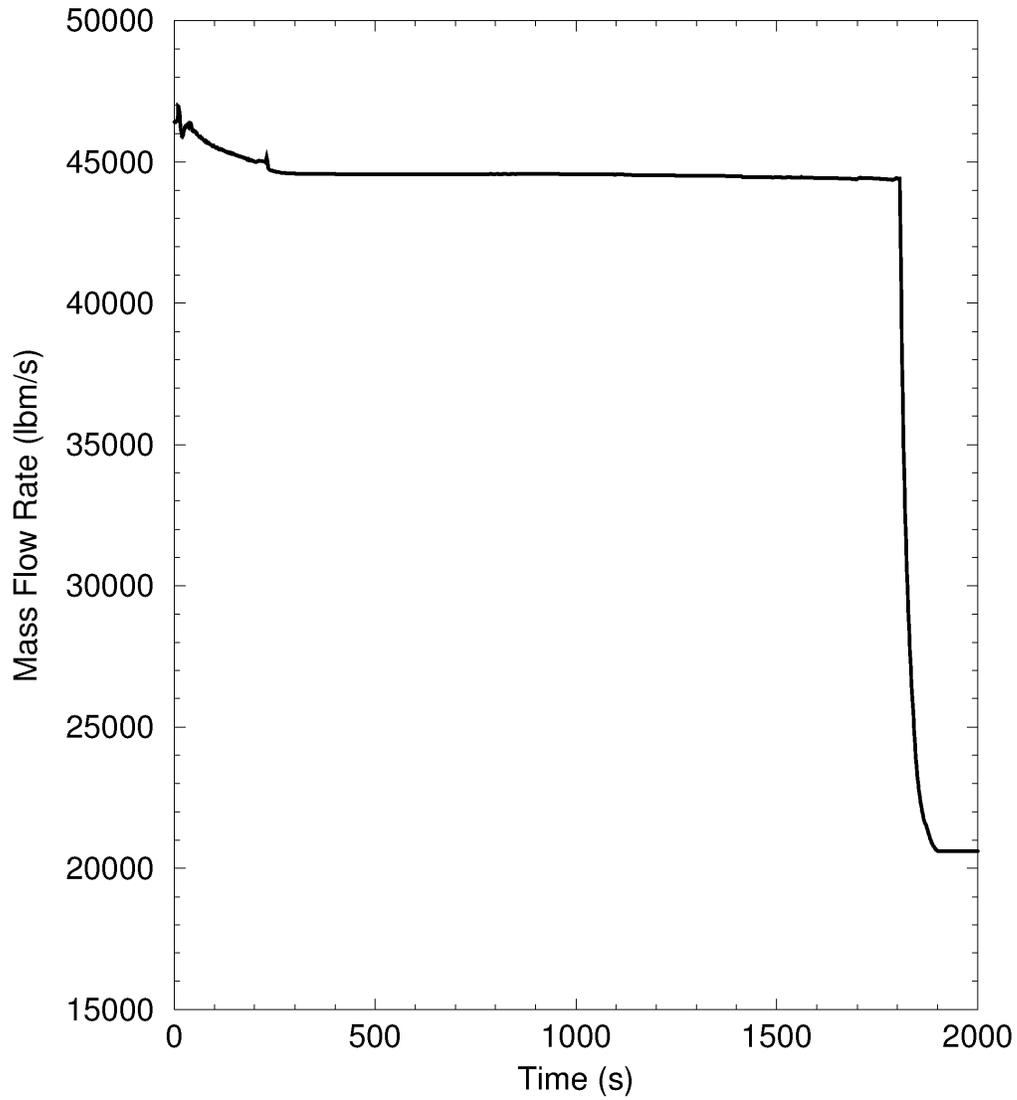
REV 002
EPR4830 T2

Figure 15.2-93—FWLB Maximum Secondary Pressure Case - Core Exit Subcooling



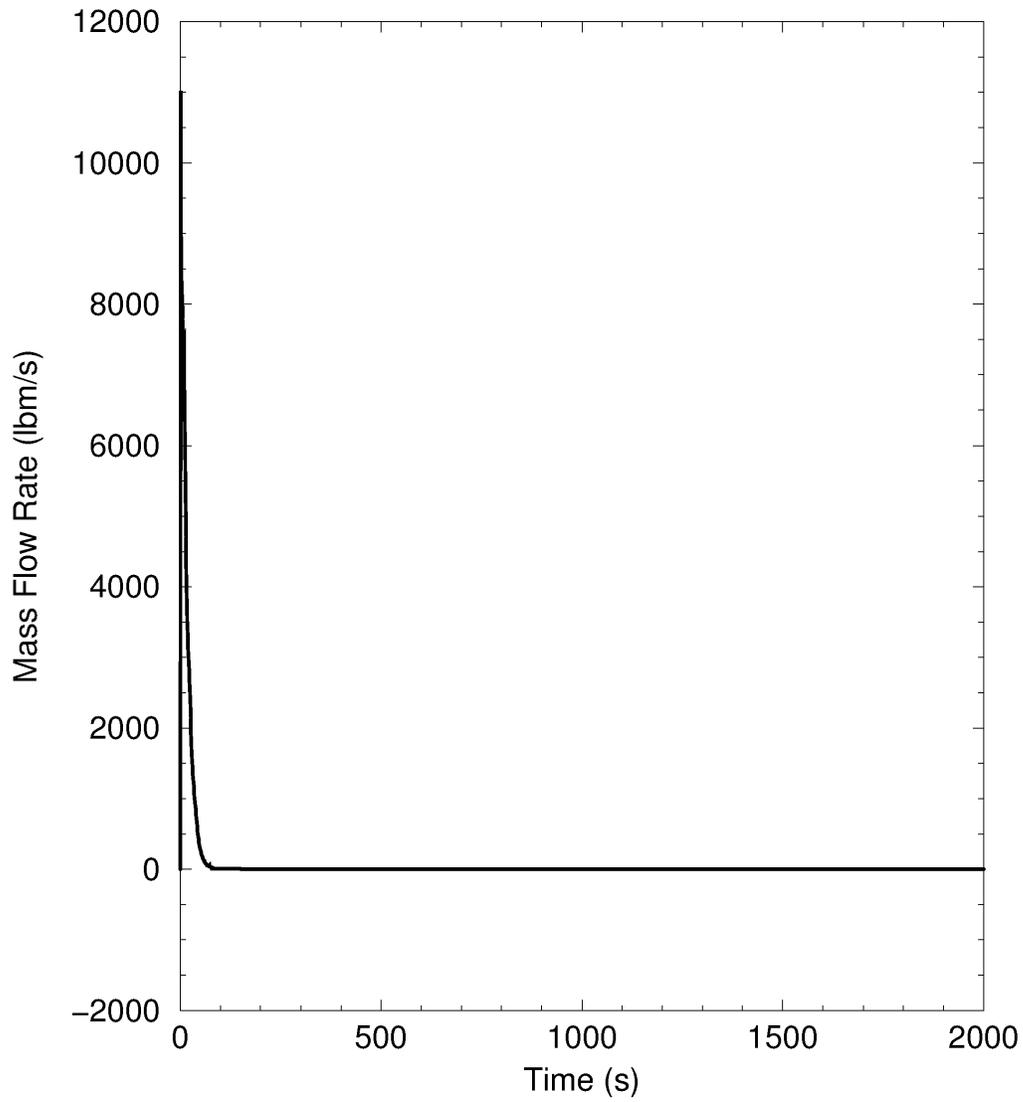
REV 002
EPR4835 T2

Figure 15.2-94—FWLB Maximum Secondary Pressure Case - Core Flow



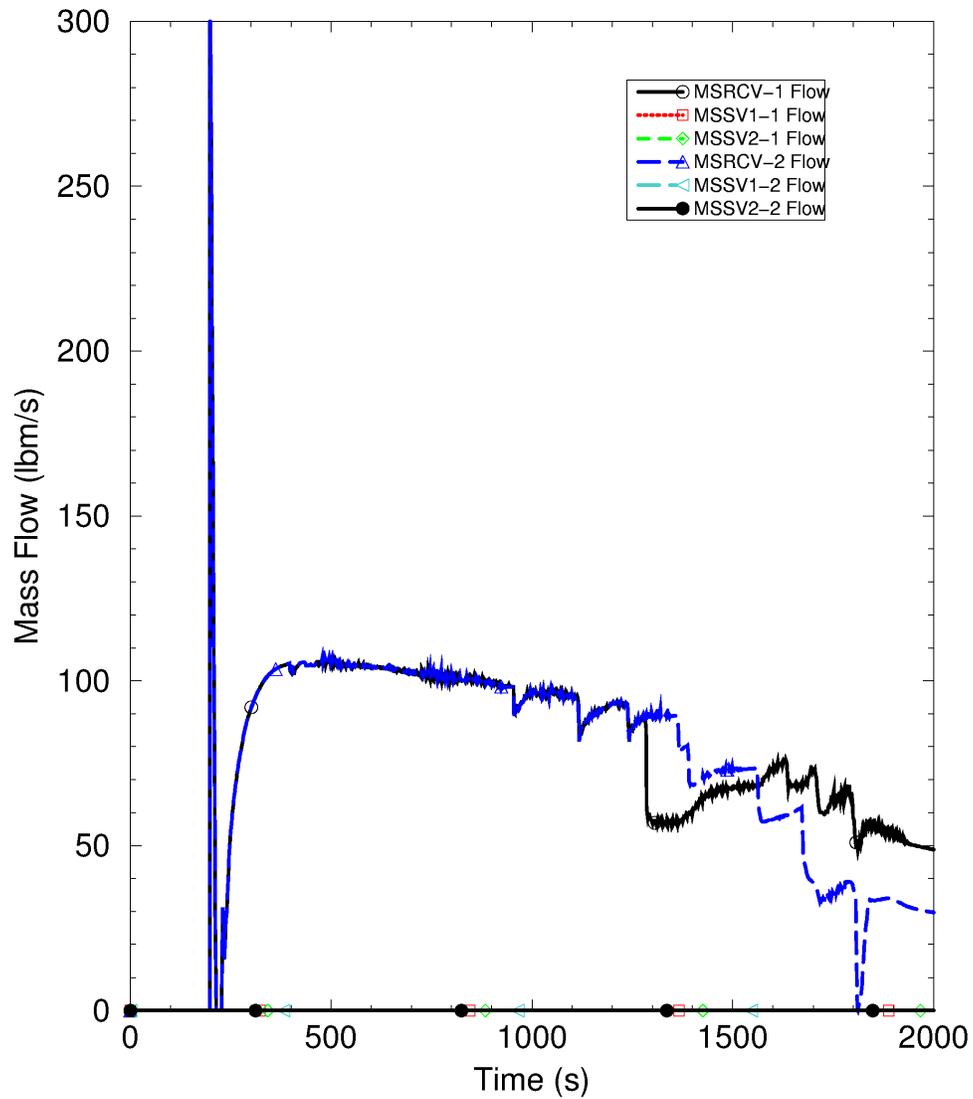
REV 002
EPR4840 T2

Figure 15.2-95—FWLB Maximum Secondary Pressure Case - Break Flow



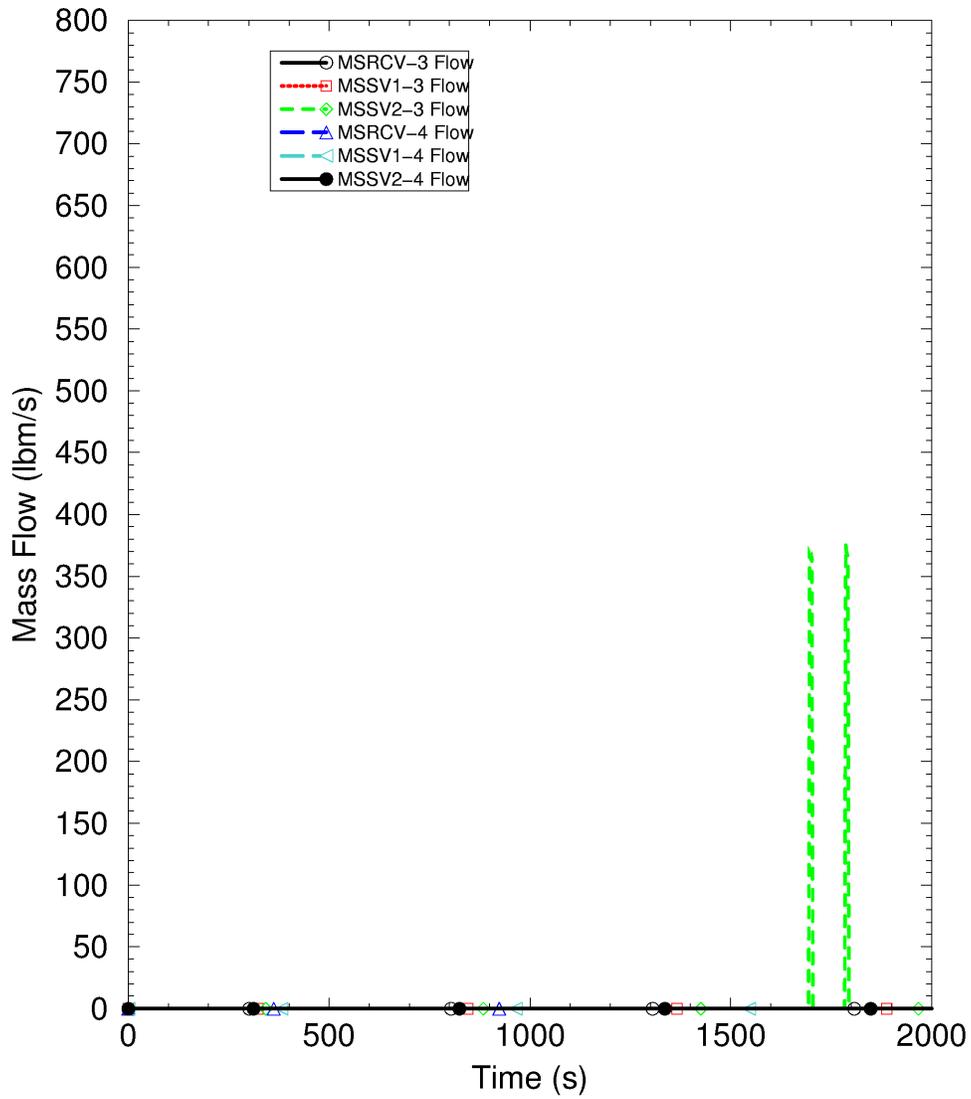
REV 002
EPR4845 T2

Figure 15.2-96—FWLB Maximum Secondary Pressure Case - Main Steam Relief Loops 1 and 2



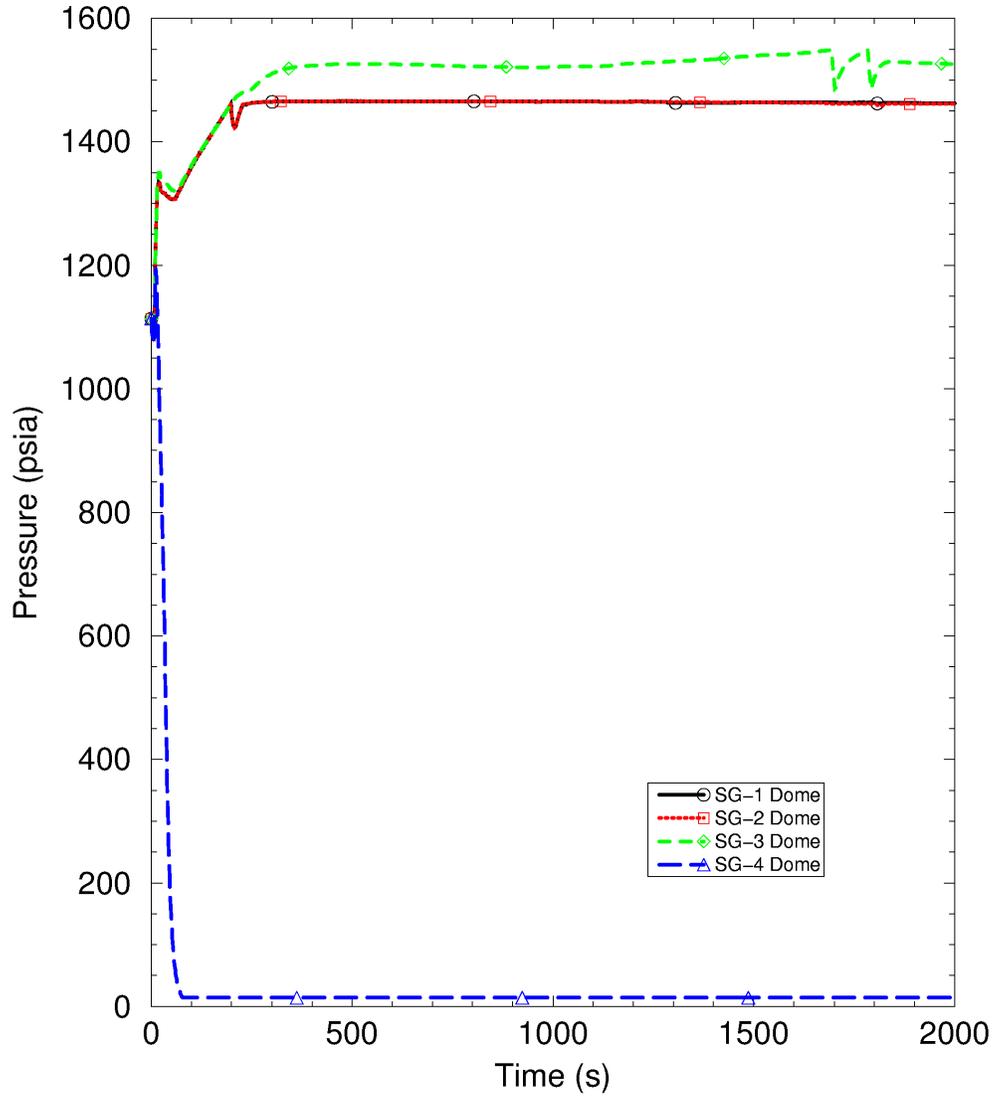
REV 002
EPR4850 T2

Figure 15.2-97—FWLB Maximum Secondary Pressure Case - Main Steam Relief Loops 3 and 4



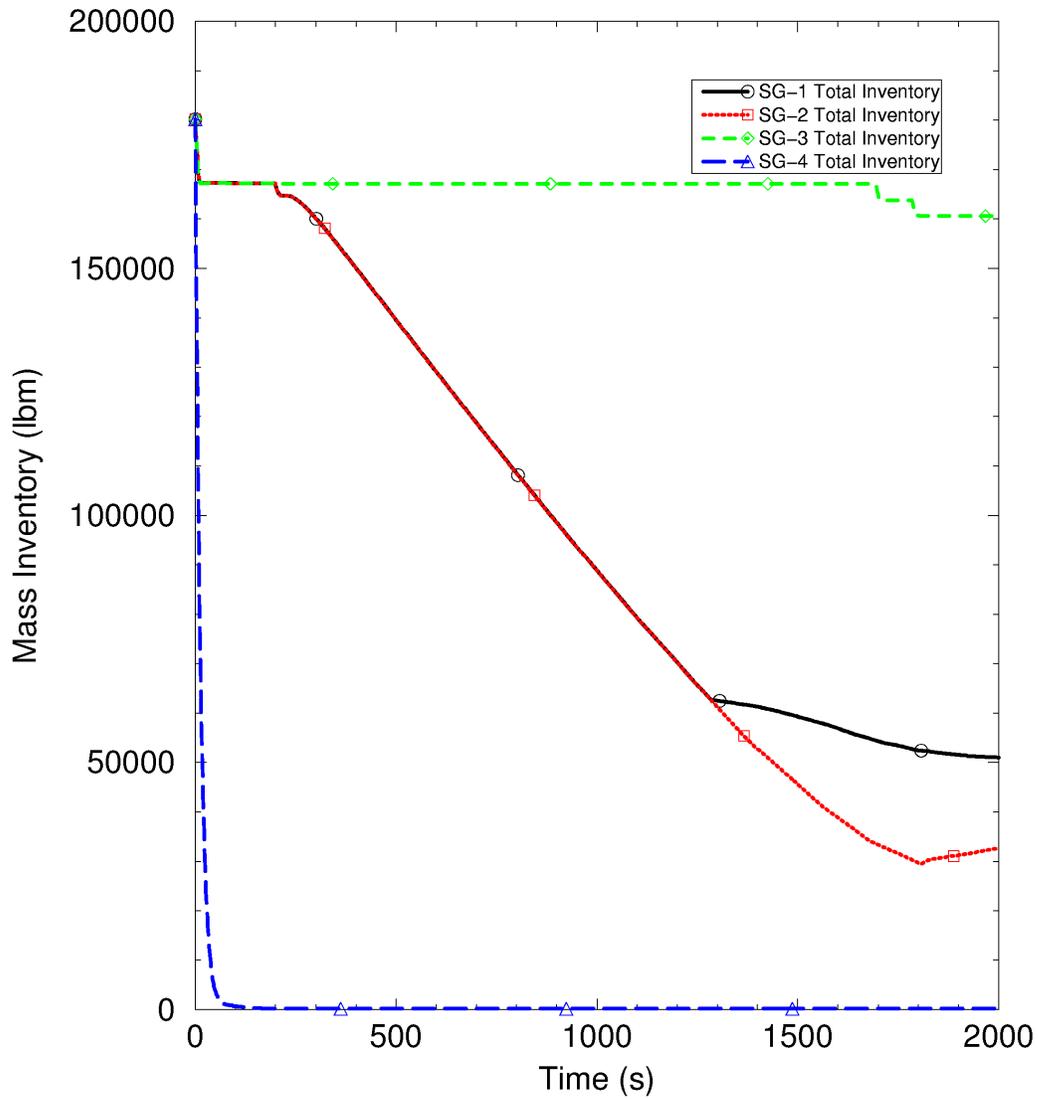
REV 002
EPR4855 T2

Figure 15.2-98—FWLB Maximum Secondary Pressure Case - Steam Generator Dome Pressures



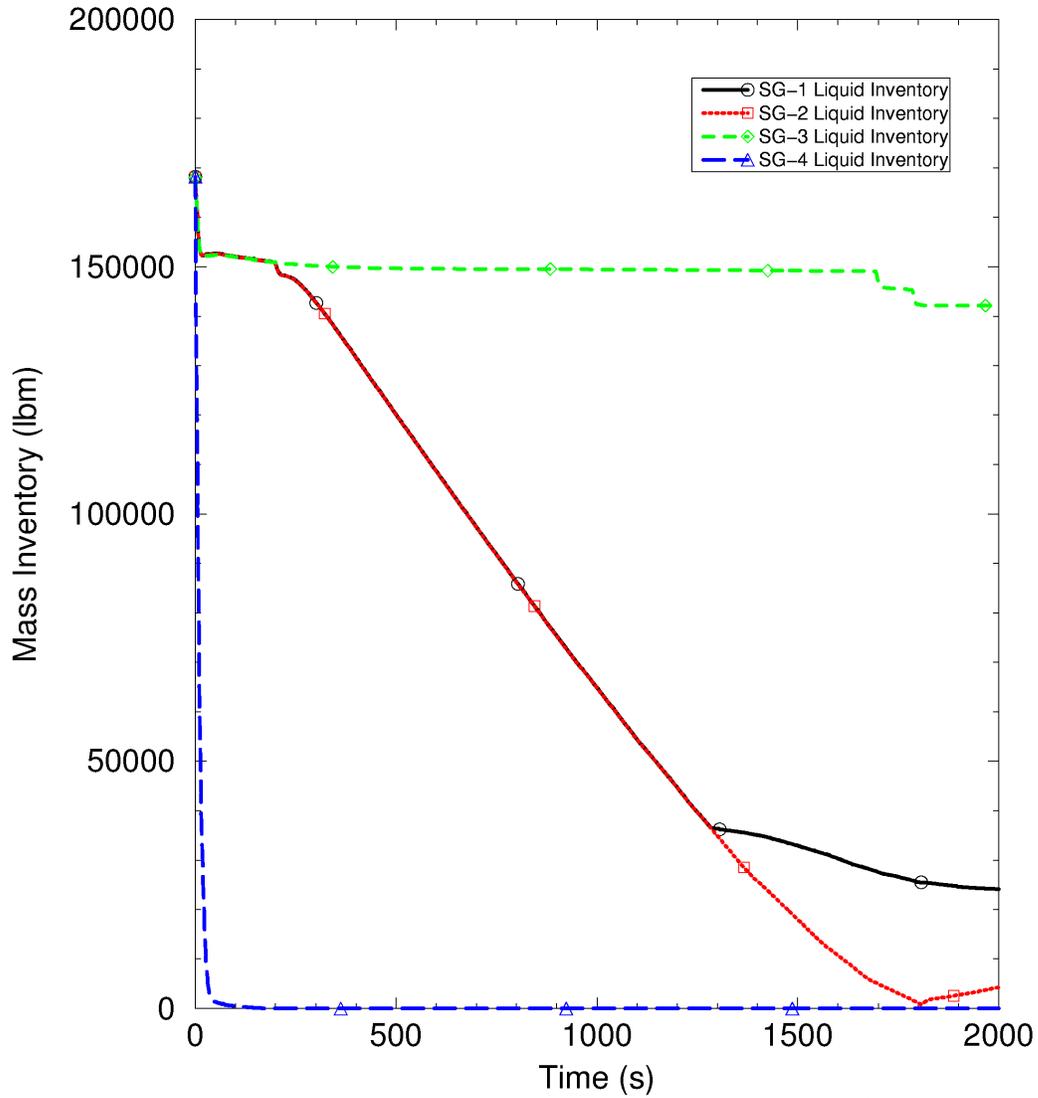
REV 002
EPR4860 T2

Figure 15.2-99—FWLB Maximum Secondary Pressure Case - Steam Generator Total Mass



REV 002
EPR4865 T2

Figure 15.2-100—FWLB Maximum Secondary Pressure Case - Steam Generator Liquid Mass



REV 002
EPR4870 T2

Figure 15.2-101—FWLB Maximum Secondary Pressure Case - Net Heat Addition to RCS

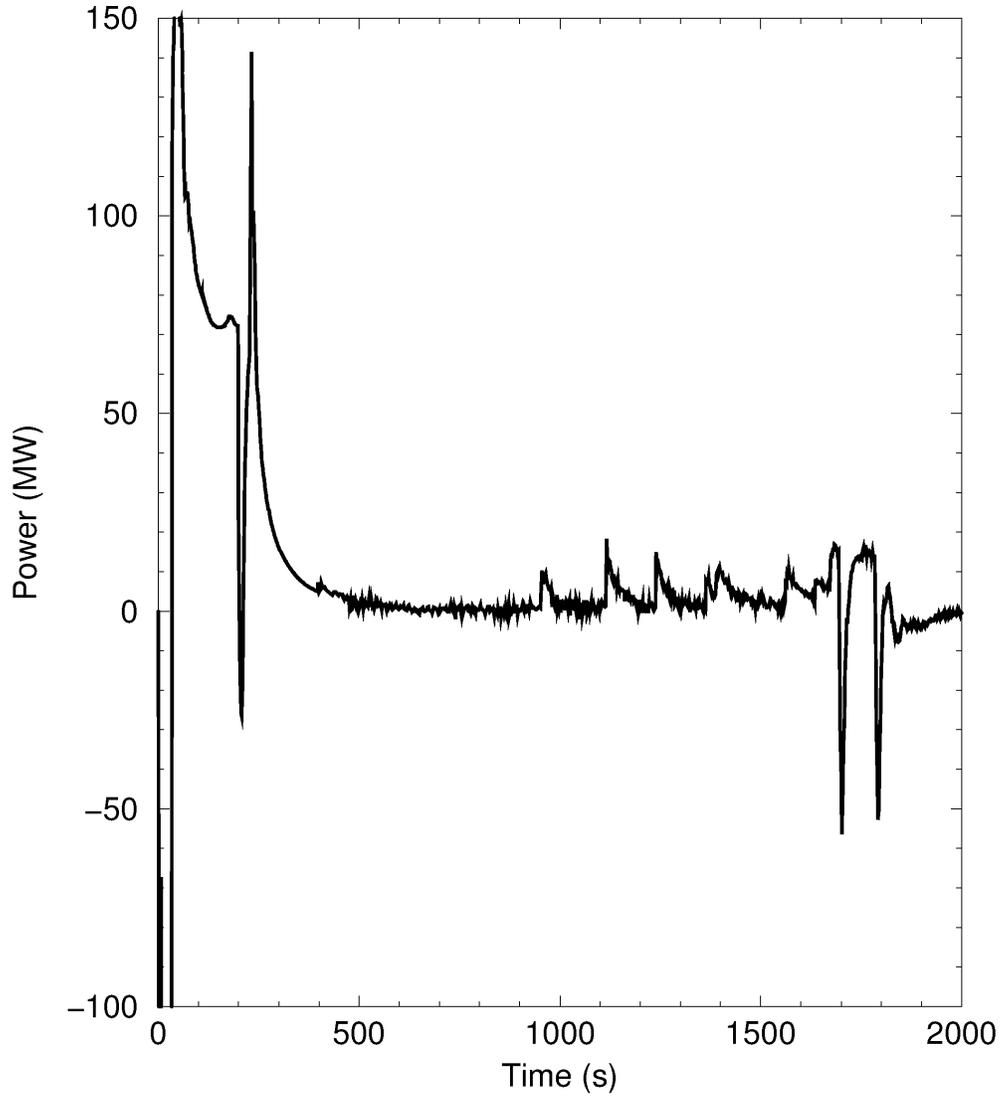
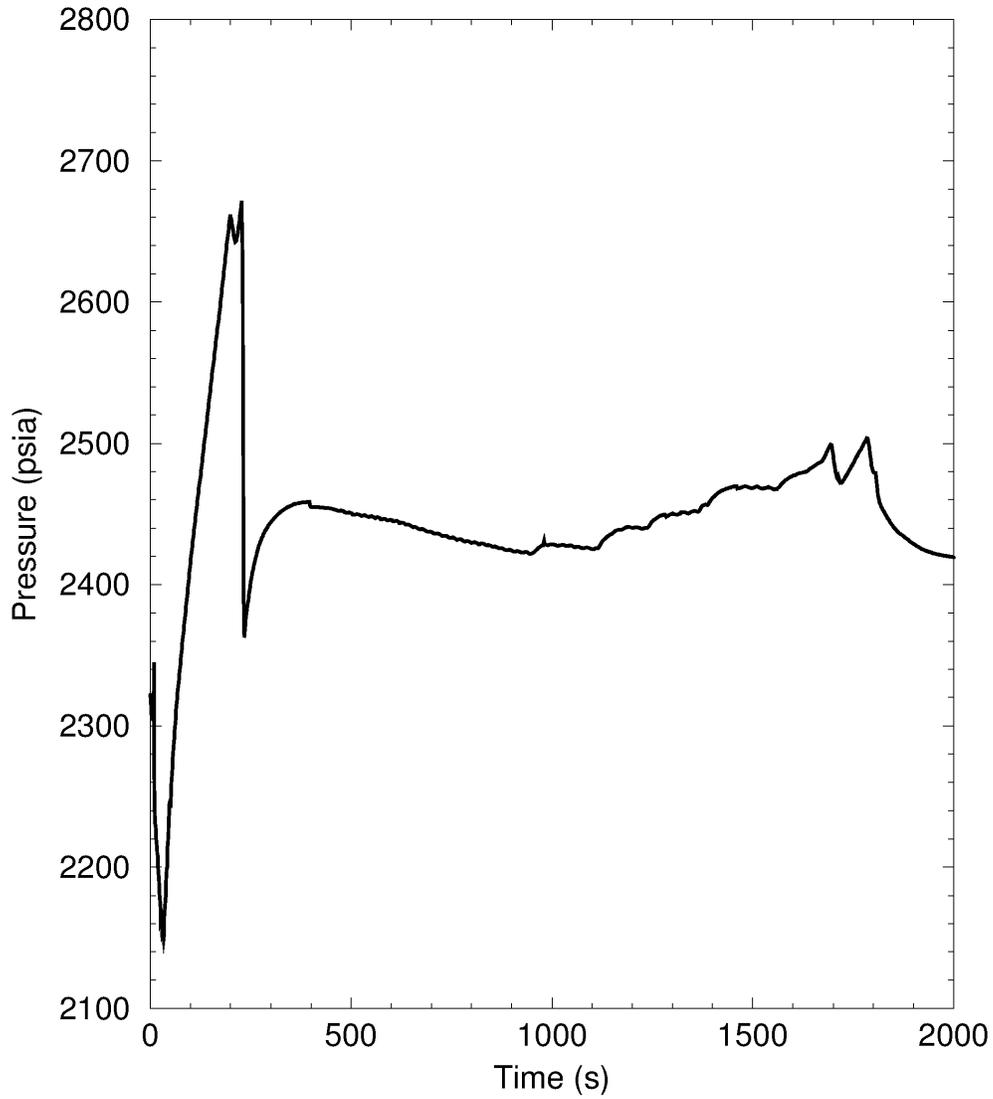
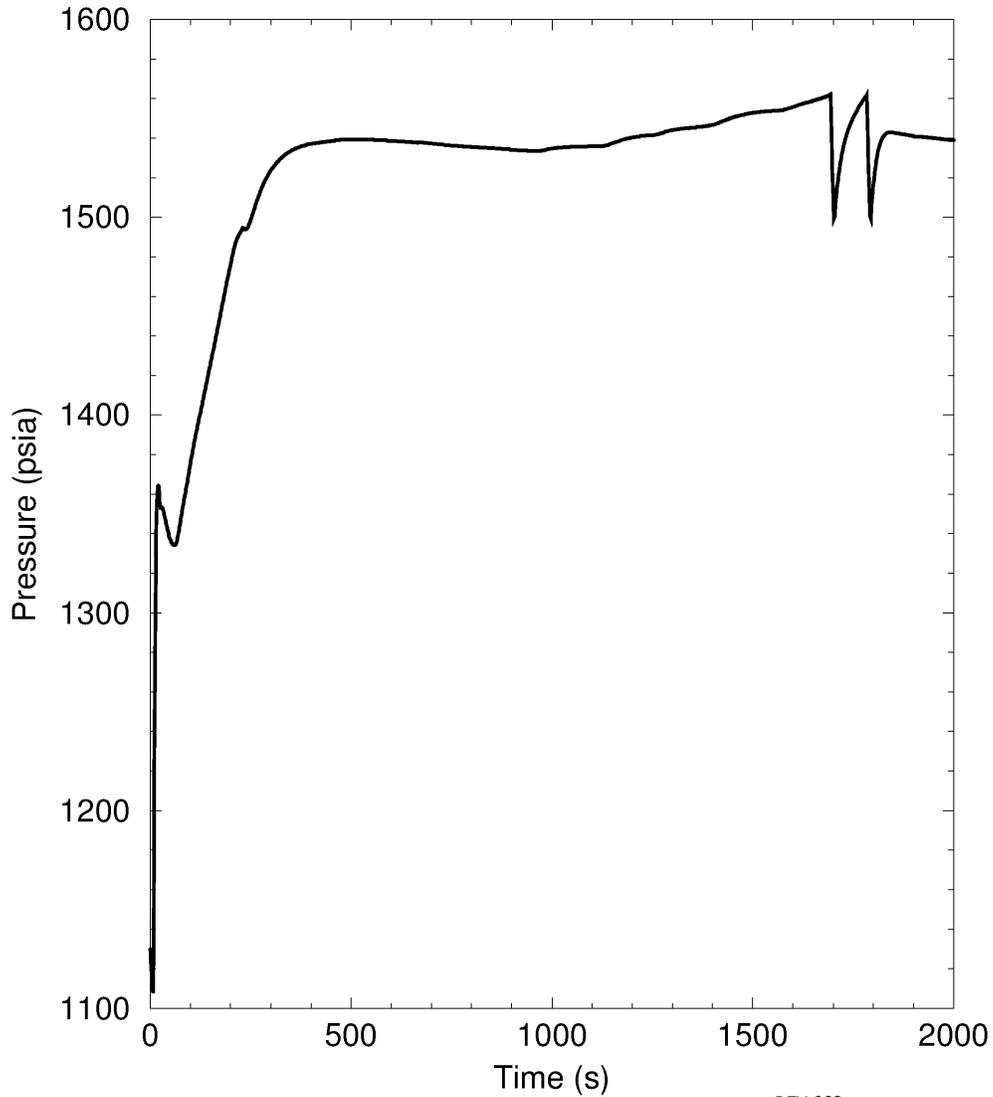


Figure 15.2-102—FWLB Maximum Secondary Pressure Case - RCS
Maximum Pressure



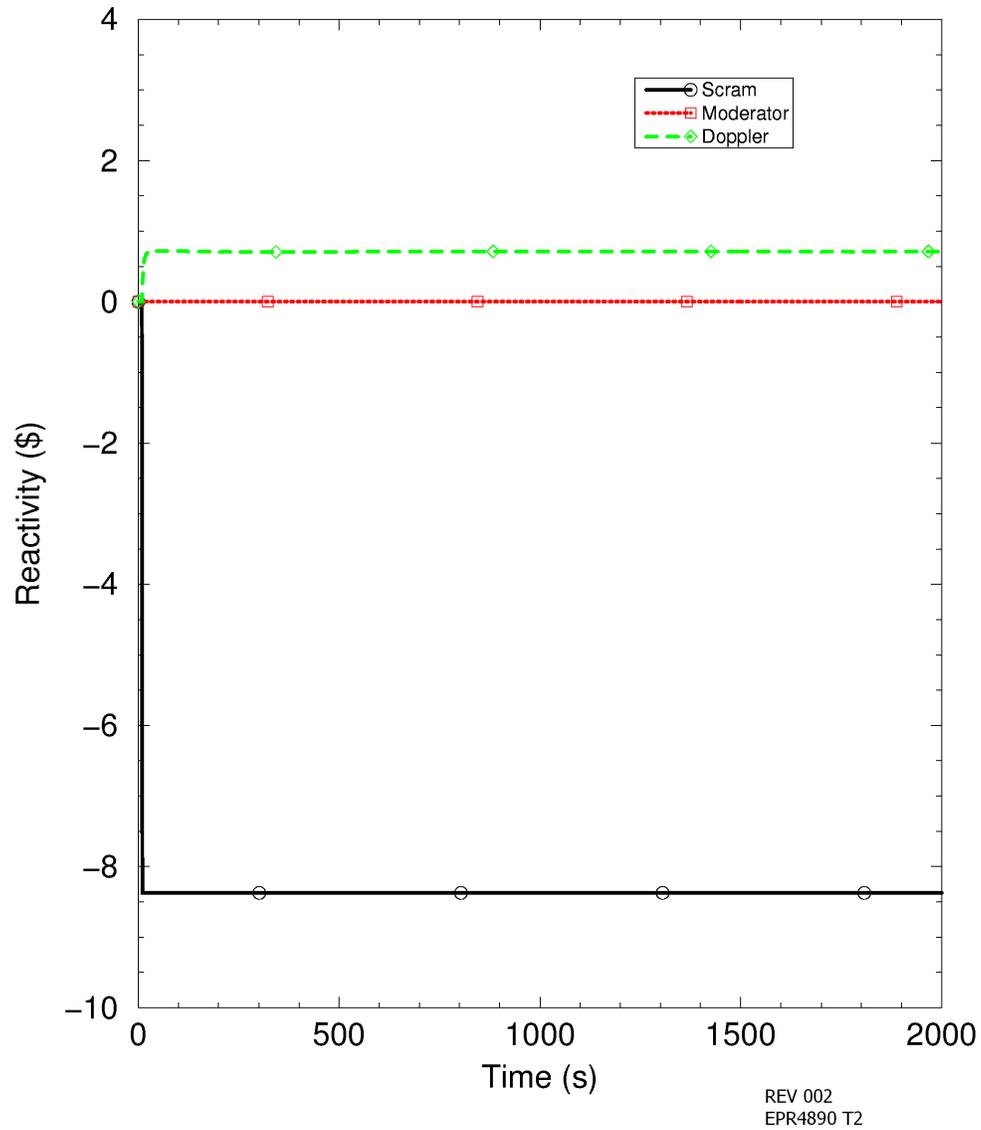
REV 002
EPR4880 T2

Figure 15.2-103—FWLB Maximum Secondary Pressure Case - Steam Generator Maximum Pressure



REV 002
EPR4885 T2

Figure 15.2-104—FWLB Maximum Secondary Pressure Case - Reactivities



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