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 of 15.2:

- 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 closure of main steam isolation valves (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 pipe break (FWLB).

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 (TCVs) causes a decrease in heat removal by the secondary system. The LOEL may also involve a temporary increase in 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 (MSRTs) and MSSVs 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 a 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 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.
- The secondary side overpressurization limit.
- Specified acceptable fuel design limits (SAFDLs), specifically, departure from nucleate boiling (DNB).

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.

15.2.1.3 Conclusions

The LOEL event is bounded by the TT event presented in Section 15.2.2.

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.
- The secondary side overpressurization limit.
- SAFDLs, specifically, DNB.

The TT event does not challenge SAFDLs because RCS pressure increases and there is little change in core power. The single main steam isolation valve (MSIV) closure event (Section 15.2.A) 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. 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). RCS temperature increases and the associated thermal expansion of the coolant causes an insurge into the PZR, thereby increasing PZR pressure and coolant level. RT occurs on high PZR pressure.

The TT event is initiated by rapidly closing the TSVs in 0.1 second. 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. The non-safety-related PZR spray, turbine bypass system, and PT (on TT) are not credited. The single failure for this event is assumed as failure of one MSRT to open.

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 BOC 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 reseat. 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 feedwater 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 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 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 reactor coolant system (RCS) pressure remains below 110 percent of the design value for the TT event. The single MSIV closure 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 (PWRs) 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 Part 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. 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.

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 Closure of Main Steam Isolation Valves

15.2.4.1 Identification of Causes and Event Description

The MSIV closure 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.
- The secondary side overpressurization limit.
- The MDNBR SAFDL.

In addition, an MSIVC event in combination with a stuck open MSRT may release enough steam to result in an offsite dose in excess of 10 CFR Part 20 limits.

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 core thermal-hydraulic computer code LYNXT is used to calculate the core flow, enthalpy distributions, MDNBR, and peak fuel centerline temperatures using the RCS response from S-RELAP5 as a boundary condition. It is described in the Incore Trip Setpoint and Transient Methodology for U.S. EPR (Reference 3). Section 4.4 describes the codes and methods used to evaluate SAFDLs. Section 15.0.3 describes the codes and methods used for the radiological analyses.

The MSIV closure 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 worst single failure is the failure of the MSRT to open in the affected loop.

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 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 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 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 5.2-23 MSIVC Secondary Overpressurization—Maximum Secondary Pressure).

Pressure in the affected SG reaches the high SG pressure RT setpoint at 5.54 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 MSRIV 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 9.05 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 1541 psia at 12.13 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. When evaluated realistically using 3-D kinetics, core power does not increase during the period prior to RT at approximately 6.44 s. During the same time period, 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 LOCV event (see Section 15.2.3).

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 (1541.0 psia) is below 110 percent of the secondary system design pressure (1593.2 psia). Therefore, the RCS and secondary overpressurization acceptance criterion for the MSIVC event are met.

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 feedwater 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 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; 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 Part 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 LOOP or LNEP is 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 LOOP, which causes a complete loss of RCS flow; see Section 15.2.7 for the long-term plant response to an LNEP event.

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 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 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.

- Decay heat removal. There is sufficient capacity for long-term decay heat removal for the plant to reach a stabilized condition.

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.

This analysis demonstrates that there is sufficient capacity for long-term decay heat and RCP heat removal for the plant to reach a stabilized condition. Non-safety-related systems are considered in the analysis if they make the event more severe. Cases are analyzed with LOOP at RT and without LOOP, with and without RCS pressure control (PZR sprays), and with zero percent and five percent SG tube plugging.

The worst single failure is the failure of an EFW train because it limits the heat removal capacity of EFW system. In addition, it is assumed that one EFW train is unavailable due to maintenance. SG inventories and emergency feedwater system (EFWS) capacity are challenged by primarily by assuming that LOOP does not occur, thereby maximizing the amount of power that must be removed by the MSRTs.

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 MSIV closure 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 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 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 GDCs 10 and 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 Part 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 Piping 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 Loss of Non-Emergency AC Power (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, SG high-pressure drop or, for the smallest breaks, on high PZR pressure. RT causes TT. If 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 ft^2, to the area of the MFW distribution half-ring inlet, 0.922 ft^2. 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.

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.

Tables 15.2-10—Feedwater Line Break - Key Input Parameters and 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 Part 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 ft^2, 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—FWLB Representative Small Break - Reactor and Total SG Power through Figure 15.2-68—FWLB Representative Small Break - Liquid Volume Fraction in Pressurizer Dome 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 PZR sprays are isolated automatically at 57 seconds on a PZR high-level signal. The PS initiates RT at 81 seconds on a high PZR pressure signal. LOOP is assumed to occur on the TT following RT. 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 96 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 MSRIV opening setpoint. The MSRTs release steam for less than 50 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 re-aligned to an unfed SG. After the EFW train is re-aligned, 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 ft^2 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 39 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 closes the MSIVs at 170 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 510 seconds and cycle to control pressure. The peak RCS pressure of 2676 psia is reached at the bottom of the reactor vessel when the PSRVs open for the first time.

The MSRIVs in the unaffected SGs open at 578 seconds and cycle to control to the high SG pressure setpoint. The PS actuates EFW in an unaffected SG at 588 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 re-aligns 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. The MSRIVs in the unaffected SGs close sixty seconds later. 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 ft^2 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 six 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 219 seconds to control pressure.

The MSRIVs open at 187 seconds in the two unaffected SGs where the MSRIVs are operable. Pressure continues to increase in the unaffected SG with the failed MSRT. At 308 seconds, the MSSV opens and controls pressure. The peak secondary system pressure of 1531.4 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 1293 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 2676 psia is below the 110 percent design value of 2803 psia. The main steam system peak pressure of 1531 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 2676 psia is below the 110 percent design value of 2803 psia. The main steam system peak pressure of 1531 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 pressurized-water reactors (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 Part 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., August 2007.

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

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

4. NUREG-1793, “Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design: Chapters 10-20,” NRC, Volume 2, September 2004.

Table 15.2-1—Turbine Trip - Key Input Parameters

Parameter	Analysis Value
Initial reactor power	4612 MWt
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 lbm/hr at 1229.7 psia per train
PSRV open setpoints	2600.4 psia
PSRV capacity	661,400 lbm/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 lbm/hr at 1518.5 psia 1,422,073 lbm/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 MWt
Initial RCS loop flow rate	119,692 gpm/loop
Initial reactor vessel average temperature	599°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	2,840,790 lbm/hr at 1449.7 psia per train
MSSV open setpoints	1518.5 psia 1549.4 psia
MSSV capacities	395.02 lbm/s at 1518.5 psia and 395.02 lbm/s at 1549.7 psia
TSV stroke time	0.1 s
Time of LOOP	1 s after RT

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
PZR pressure control (PZR spray)	Not operable
MFW	Isolated at RT
RCPs	Operating until LOOP
Turbine bypass system	Not credited
PT	Not credited
RT on TT	Not credited

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)	5.54
RT signal received	6.04
Rod insertion begins	6.44
Peak power (4732 MWt)	6.45
TT	7.04
Peak RCS pressure (2389 psia)	8.68
MSSV opens	9.05
Peak secondary pressure (1541 psia)	12.13

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

Parameter	Analysis Value
Initial reactor power	4612 MWt
Initial PZR liquid level	59.3%
Low SG level setpoint actuating RT	16.5% NR span
Low SG level setpoint delay	0.5 s
Low-low SG level setpoint for EFW actuation and normal SG blowdown isolation	38% WR span
Low-Low SG level setpoint delay	0.5 s
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, linearly ramping to 1245 gpm at EFW temperature (122°F) per credited train 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 lbm/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	2,840,790 lbm/hr at 1449.7 psia per train
MSRCV normal initial position	Fully open
MSRCV stroke time	40 s
MSRIV opening time	1.8 s

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

Parameter	Analysis Value
MSSV open setpoint pressures	1518.5 psia 1549.4 psia
MSSV capacities	1,422,073 lbm/hr at 1518.5 psia 1,422,073 lbm/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

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 RT	42.1
RT rod insertion begins	43.0
TT	43.6
Maximum PZR level	46
MSRTs open on high SG pressure	≈60
MSRCV closed	≈91
MSRCV open and modulating pressure to setpoint	≈146
SG water level reaches low WR level (SG 1/SG 2/SG 3/SG 4) Actuate EFW Isolate blowdown	307/317/300/311
EFW flow begins in unaffected SG 1/SG 2	323/333
Minimum liquid mass in unaffected SGs	324/334
Dryout in affected SG 3/SG 4	≈2380/≈2390
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 MWt
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
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	2,840,790 lbm/hr at 1449.7 psia per train
Low-low SG level EFW actuation and normal SG blowdown isolation setpoint for unaffected SGs	29% 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 active failure	1 unaffected SG EFW train fails to operate
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
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

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

Parameter	Analysis Value
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 reseat 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/l)	483.30/s
Core average U-238 capture-to-fission ratio	0.85

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

Plant Equipment or System	Status
PZR normal spray	Available for representative small break case Disabled for other cases
PZR auxiliary spray	Available for representative small break case 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, 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

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	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
PZR level reaches MAX2 level function setpoint Normal PZR sprays turned off	57.4
PZR Pressure reaches high pressure setpoint, RT	80.7
Scram rod insertion begins	81.0
TT	81.7
LOOP assumed on TT Trip RCPs	81.7
PSRVs open, start cycling	82.2
SG water level reaches low WR level in SG 1 Actuate EFW after 0.5 s delay for signal processing	95.8
EFW flow begins in SG 1	156
MSRIVs open on high SG pressure	268
MSRCVs closed	310
Maximum PZR level (111.45%[1])	1735
EFW from affected SG 4 is cross-connected to SG 2 at 30 minutes after RT (rod motion)	1881
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	0.00
SG 4 tube uncovery begins	16.0
SG 4 NR level reaches low level setpoint with delay, RT	39.4
TT	39.5
Scram rod insertion begins	39.7
SGs 1, 2, and 3 tube uncovery begins	48.0
MSIV closure signal on high SG pressure drop	170
MSIVs close	175
PSRVs open for the first time, peak RCS pressure of 2676.04 psia occurs	510
MSRIVs open on high SG pressure for SGs 1, 2 and 3	578
SG water level reaches low WR level in SG 1 Actuate EFW after 0.5 s delay for signal processing	588
MSRCVs close for SGs 2 and 3	596
MSRCVs close for SG 1	598
EFW flow begins in SG 1	603
MSRCVs open on high SG pressure for SGs 2 and 3	610
MSRCV opens on high SG pressure for SG 1	640
MSRCVs close for SGs 2 and 3	1555
MSRCVs open on high SG pressure for SGs 2 and 3	1571
EFW from affected SG 4 is cross-connected to SG 2 at 30 minutes after RT (rod motion)	1840
RCPs for SG 3 and SG 4 manually tripped at 30 minutes after RT (rod motion)	1840
MSRCV closes for SG 2	1846
MSRCV opens on high SG pressure for SG 2	1870
MSRCV closes for SG 3	1873
MSRCV opens on high SG pressure for SG 3	1880
MSRCV closes for SG 3	1899
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
TT	6.52
Scram rod insertion begins	6.68
SG 4 tube uncovery begins	9.80
MSIVs close	11.38
MSRIVs open on high SG pressure for SGs 1 and 2	187
MSRCVs close for SGs 1 and 2	204
MSRCVs open on high SG pressure for SGs 1 and 2 and remain open with flow reducing	213
PSRVs open for the first and only time	219
MSSV for SG 3 opens for the first time and continues to cycle	308
Maximum SG bottom pressure (1531.4 psia) is reached	308
SGs 1 and 2 tube uncovery begins	593
SG water level reaches low WR level in SG 1, Actuate EFW after delay for signal processing	1293
EFW flow begins in SG 1	1308
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

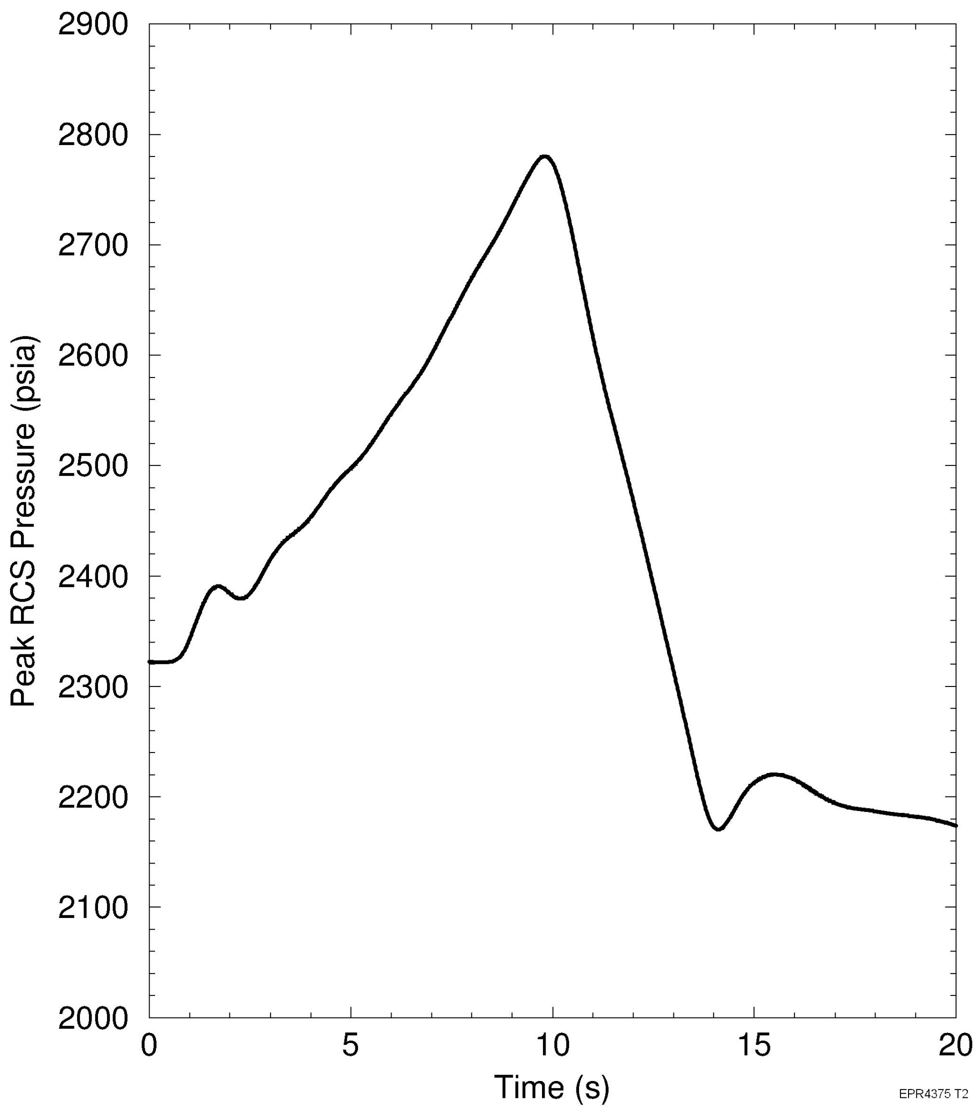

Figure 15.2-2—Turbine Trip RCS Overpressurization - Peak Secondary Pressure

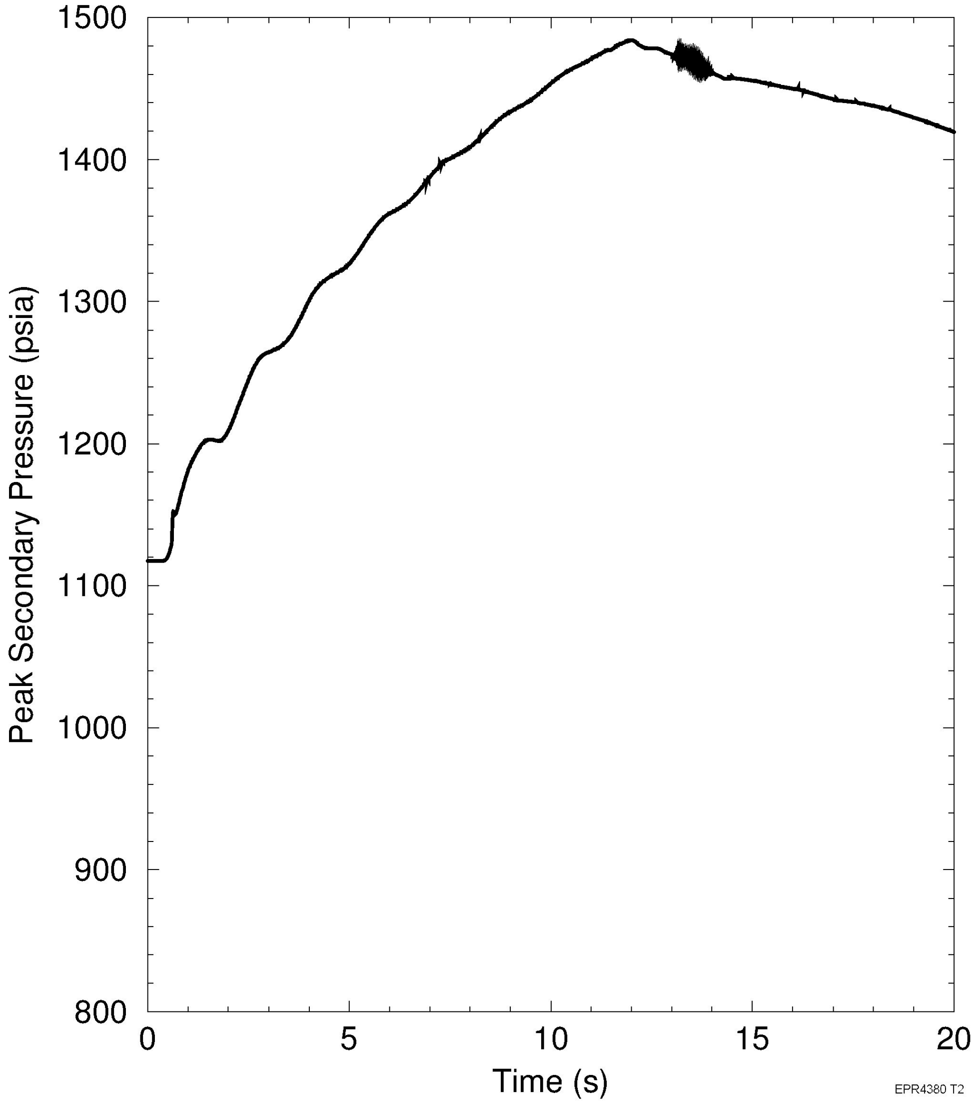

Figure 15.2-3—Turbine Trip RCS Overpressurization - Steam Generator Steam Dome Pressures

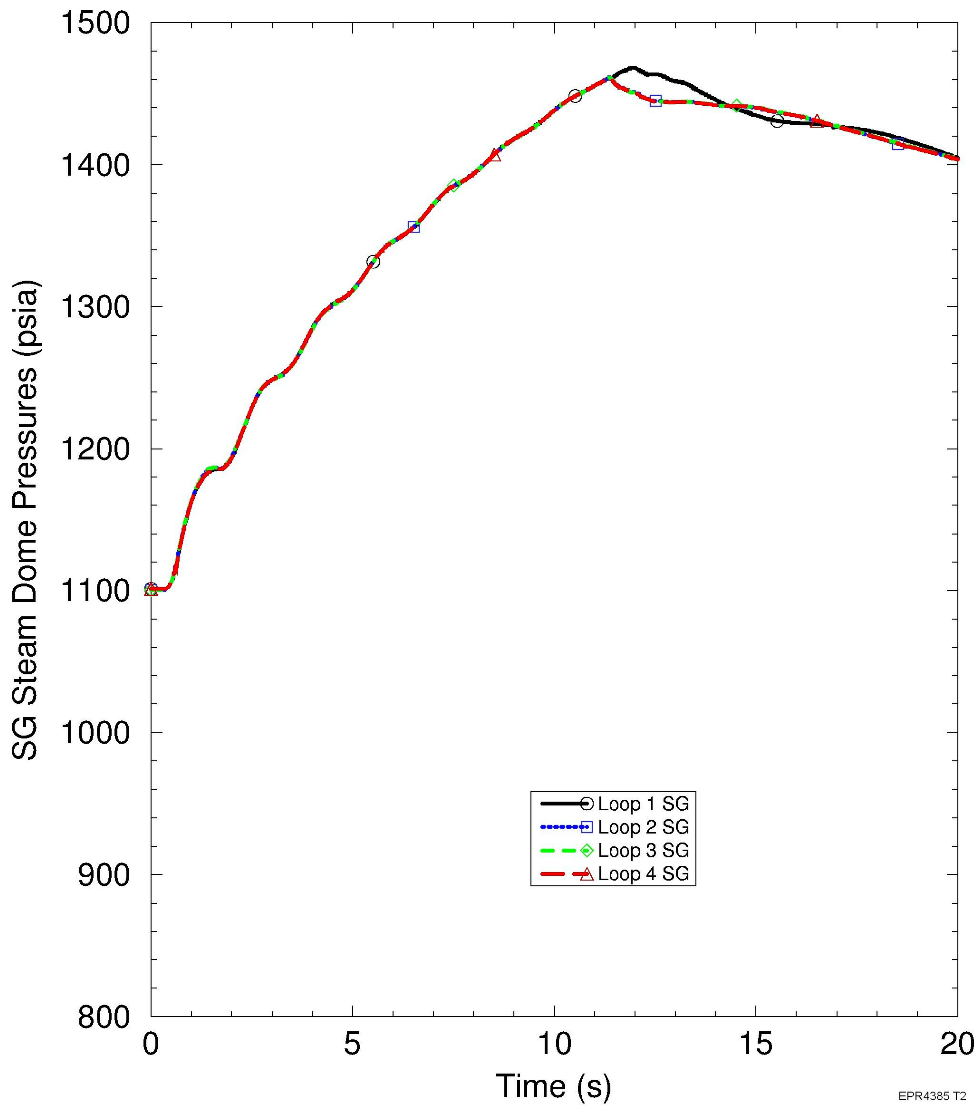

Figure 15.2-4—Turbine Trip RCS Overpressurization - % Reactor Power

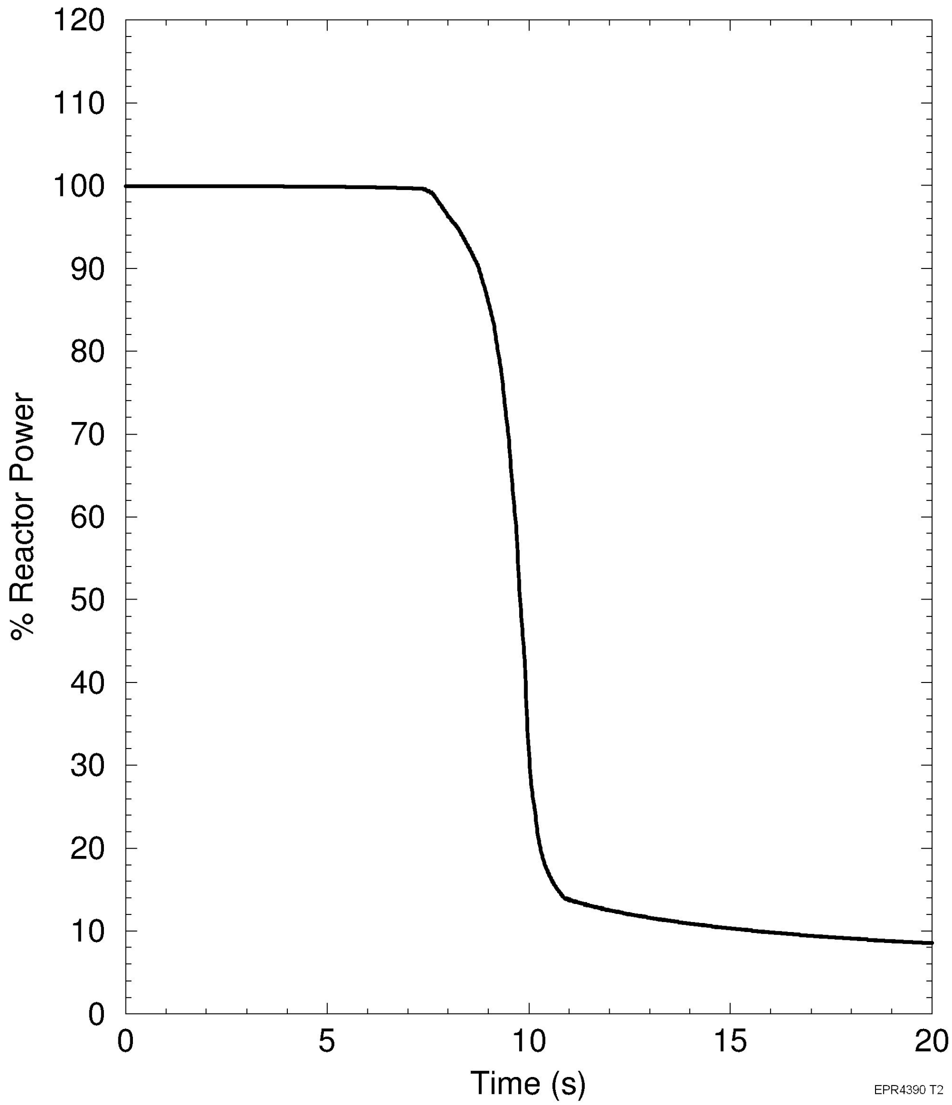

Figure 15.2-5—Turbine Trip RCS Overpressurization - Total Reactivity

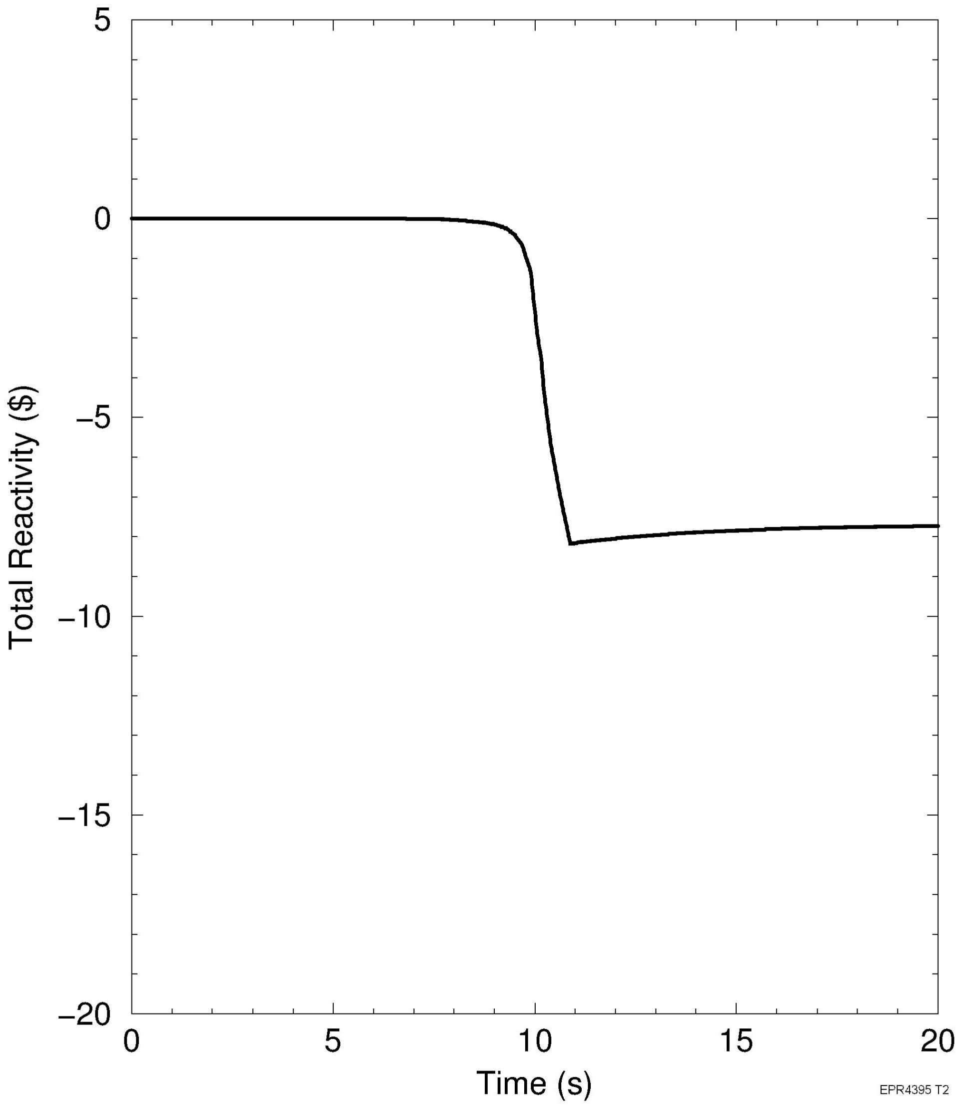

Figure 15.2-6—Turbine Trip RCS Overpressurization - Total RCS Flow

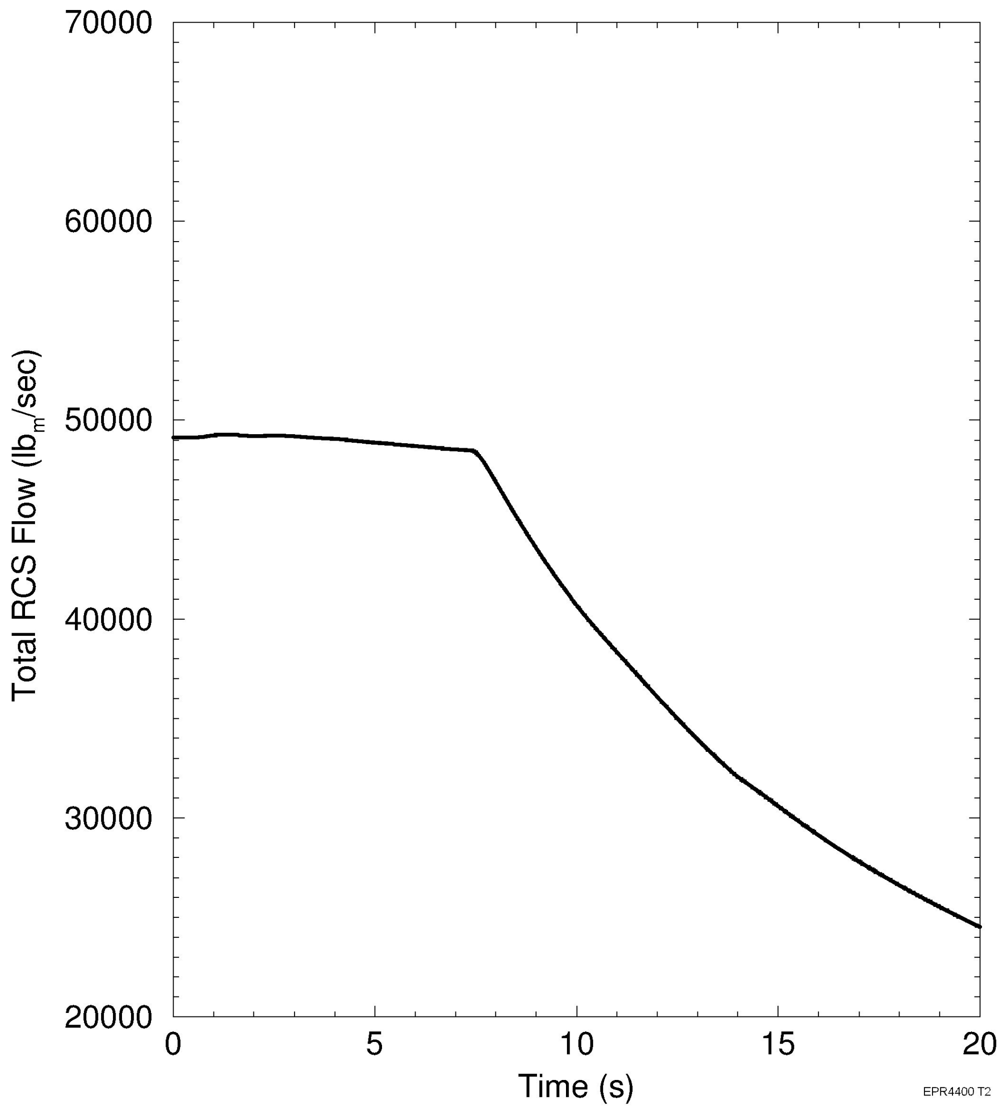

Figure 15.2-7—Turbine Trip RCS Overpressurization - Core Fluid Temperatures

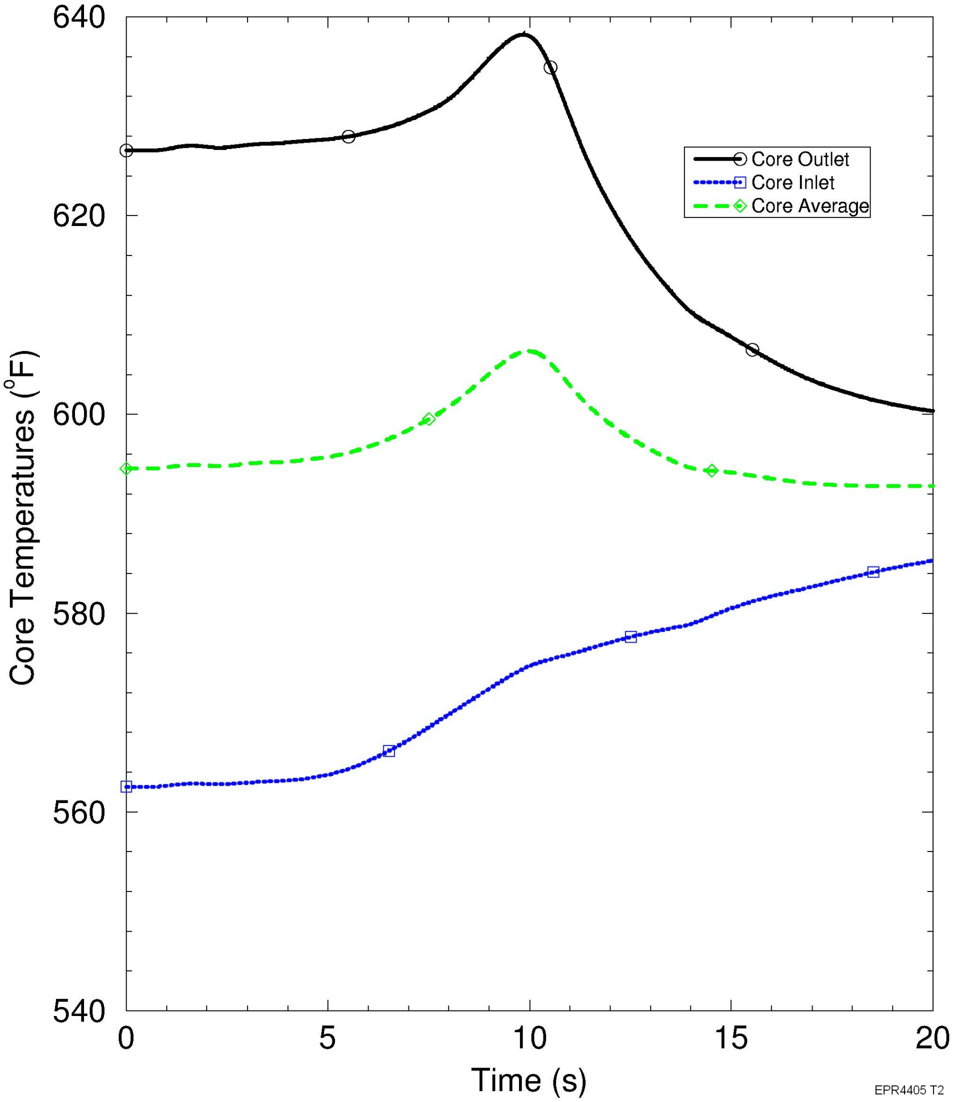

Figure 15.2-8—Turbine Trip RCS Overpressurization - RCS Loop Temperature

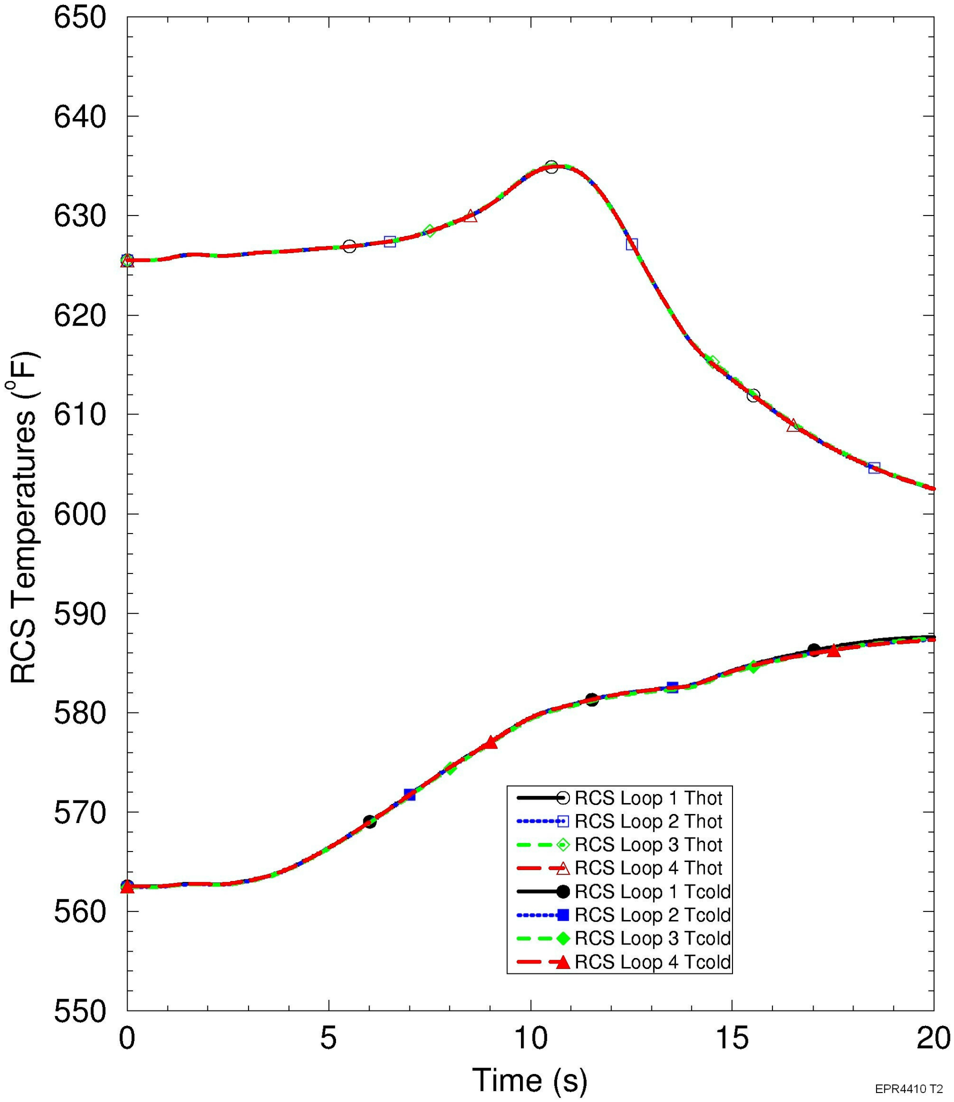

Figure 15.2-9—Turbine Trip RCS Overpressurization - PZR Pressure

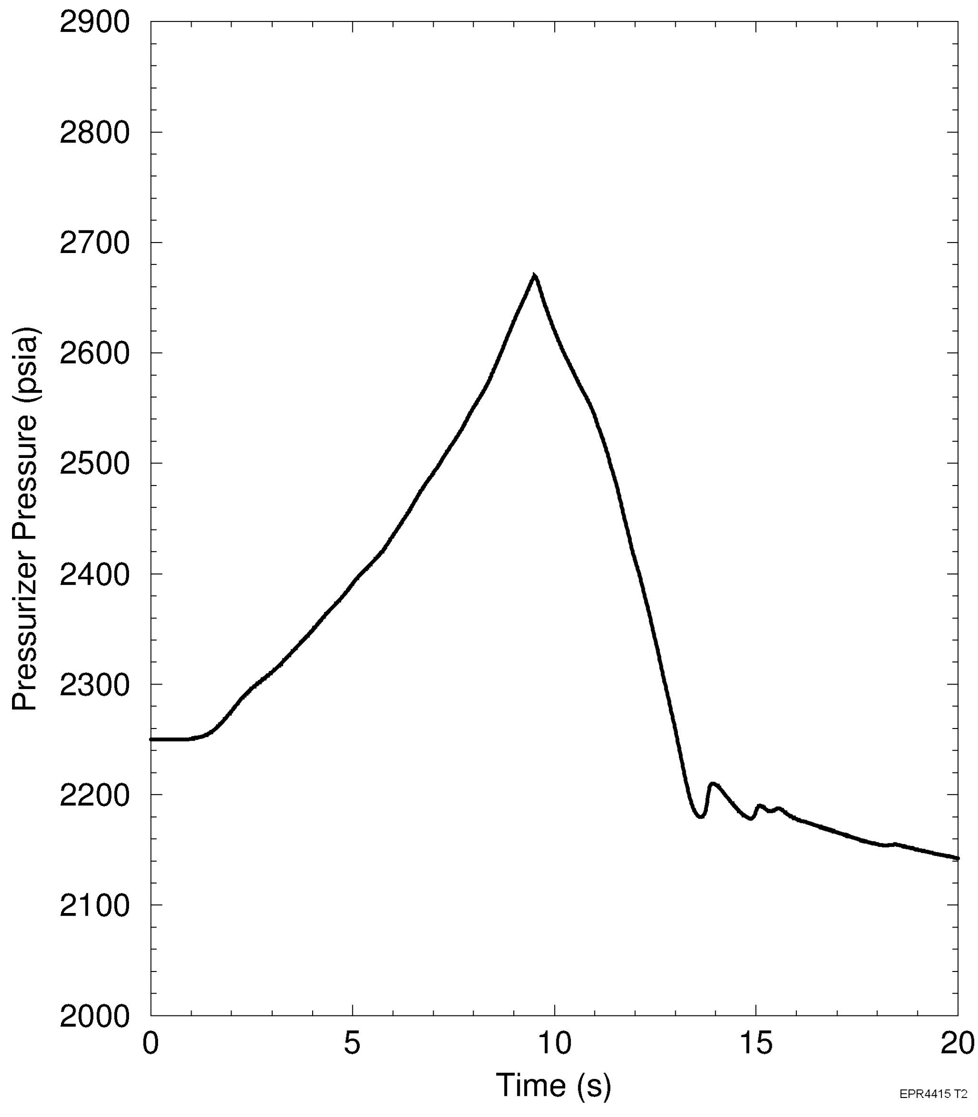

Figure 15.2-10—Turbine Trip RCS Overpressurization - PZR Level

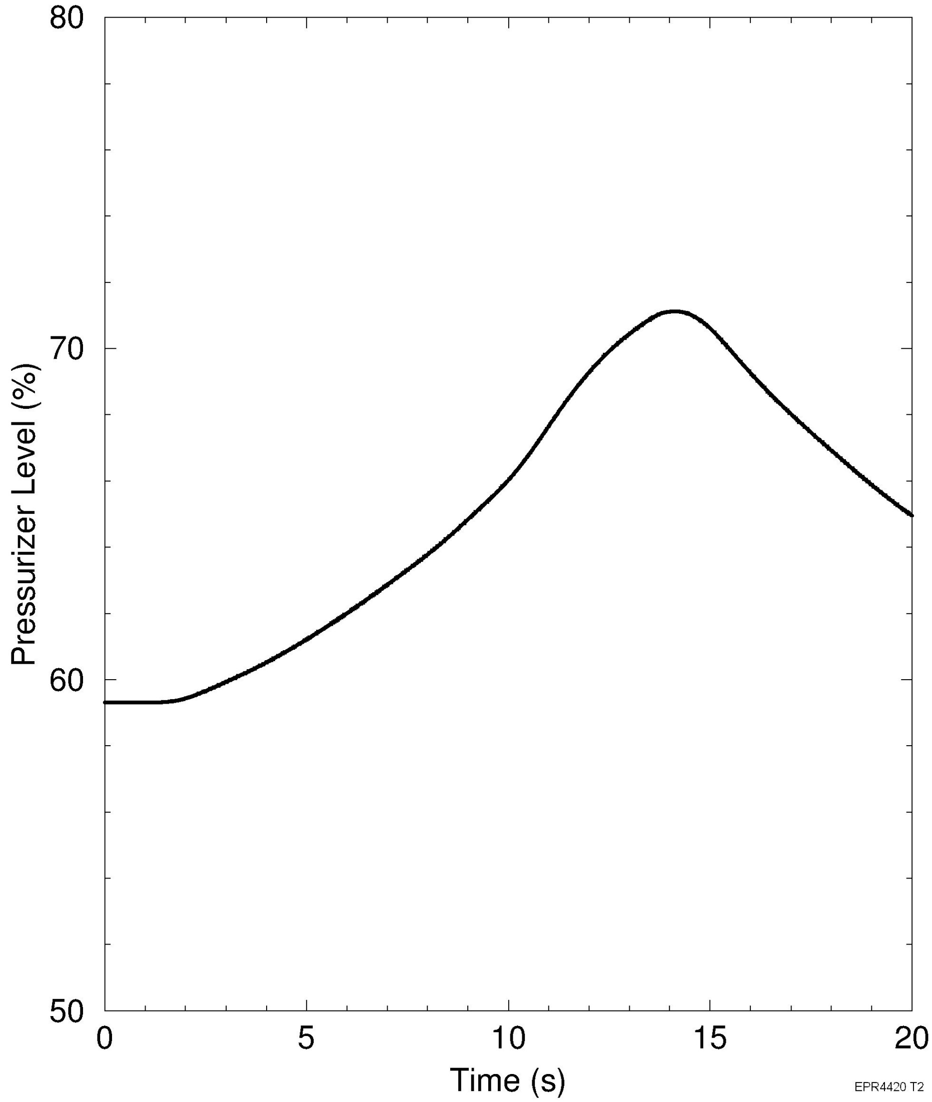

Figure 15.2-11—Turbine Trip RCS Overpressurization - PSRV Flow Rates

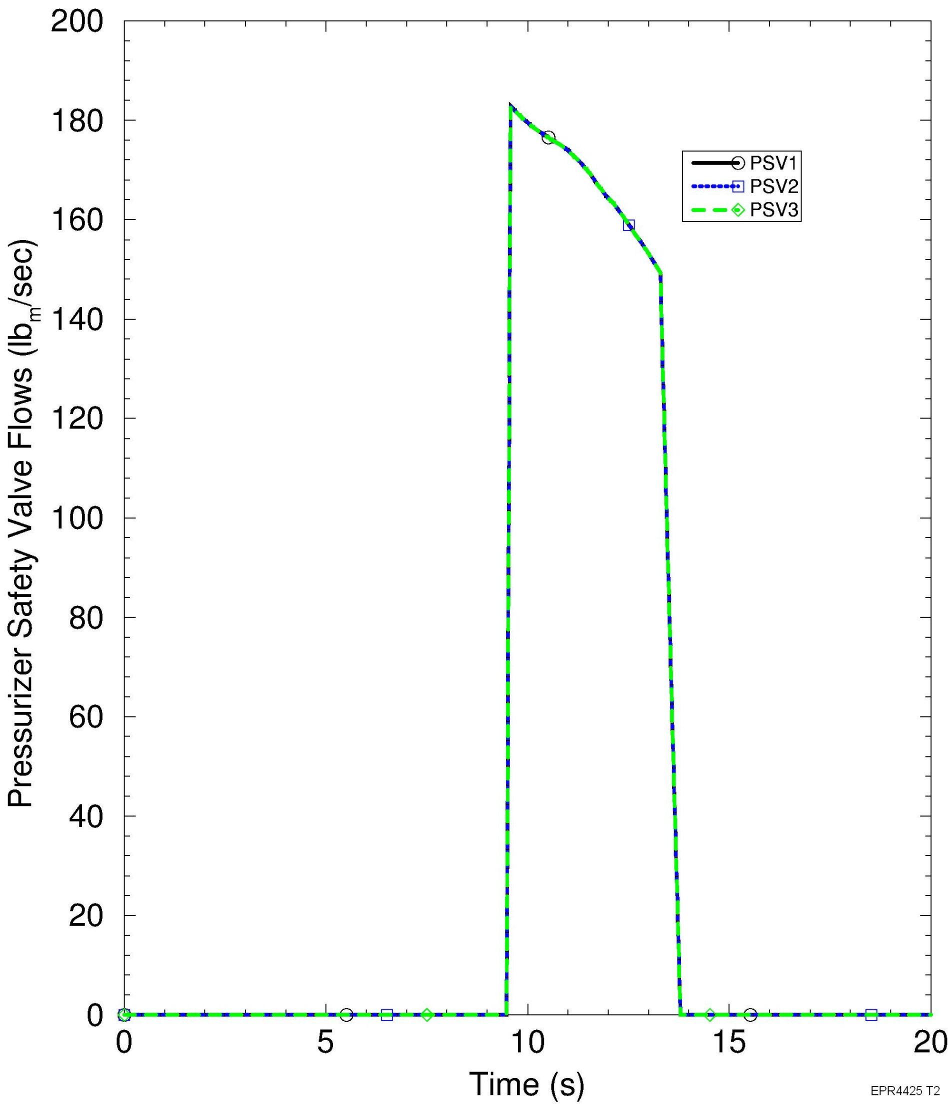

Figure 15.2-12—Turbine Trip RCS Overpressurization - MSRT Flow Rates

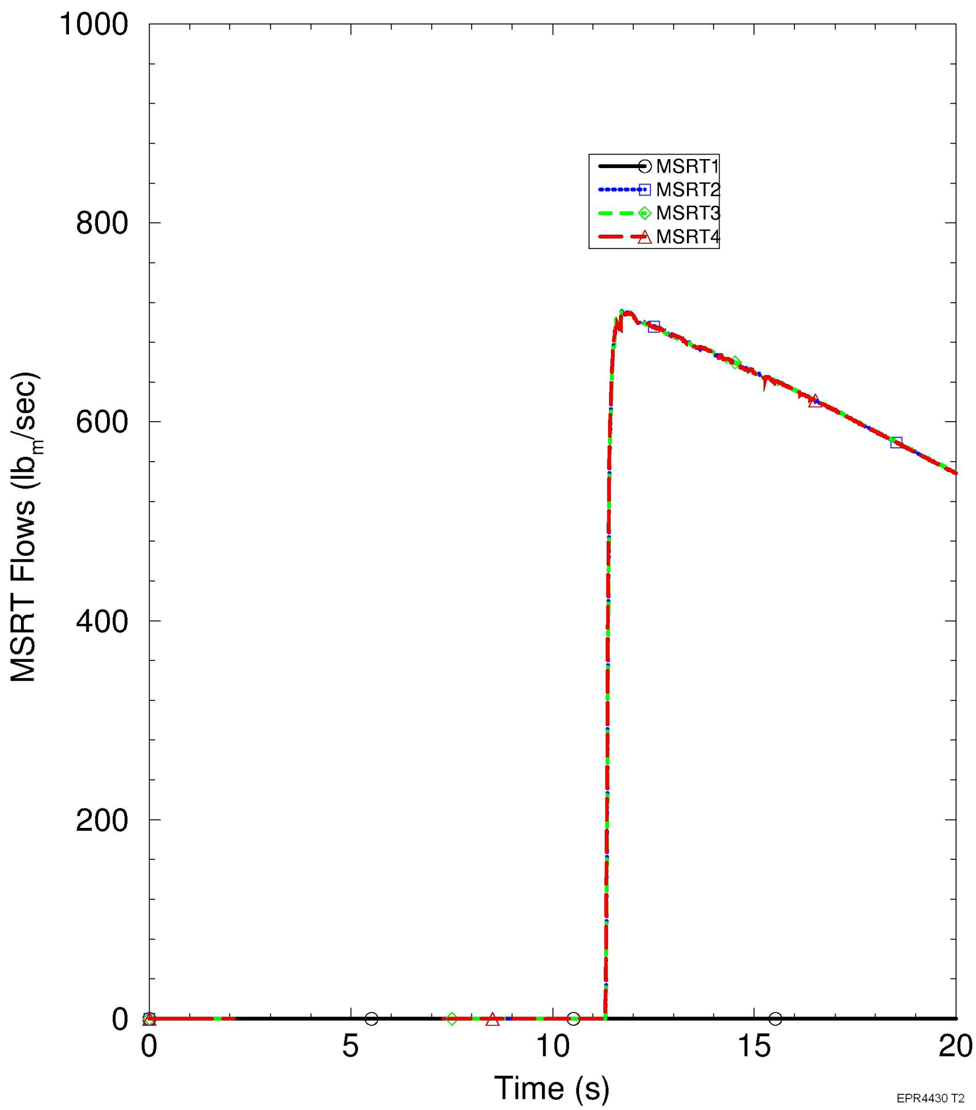

Figure 15.2-13—Turbine Trip RCS Overpressurization - MFW Flows

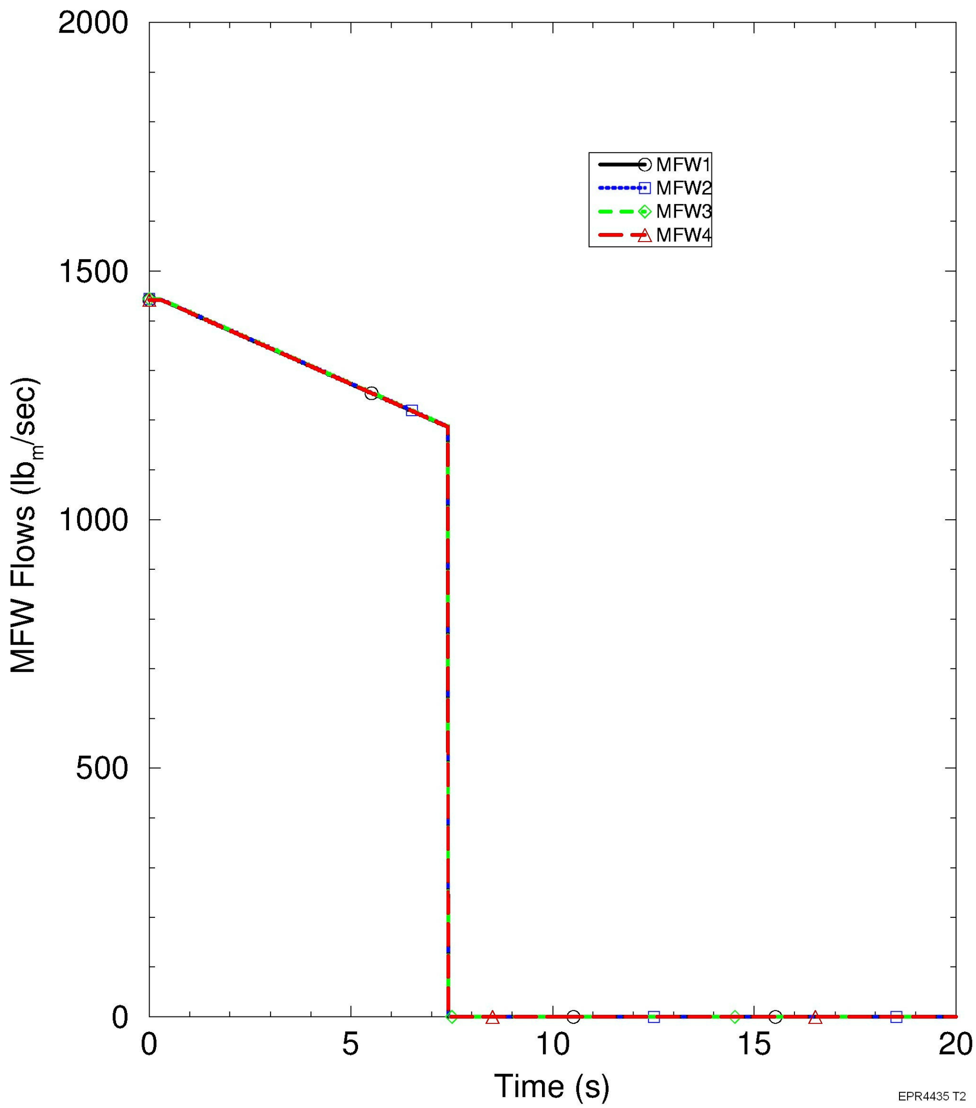

Figure 15.2-14—MSIVC Secondary Overpressurization—MSIV Flow Rates

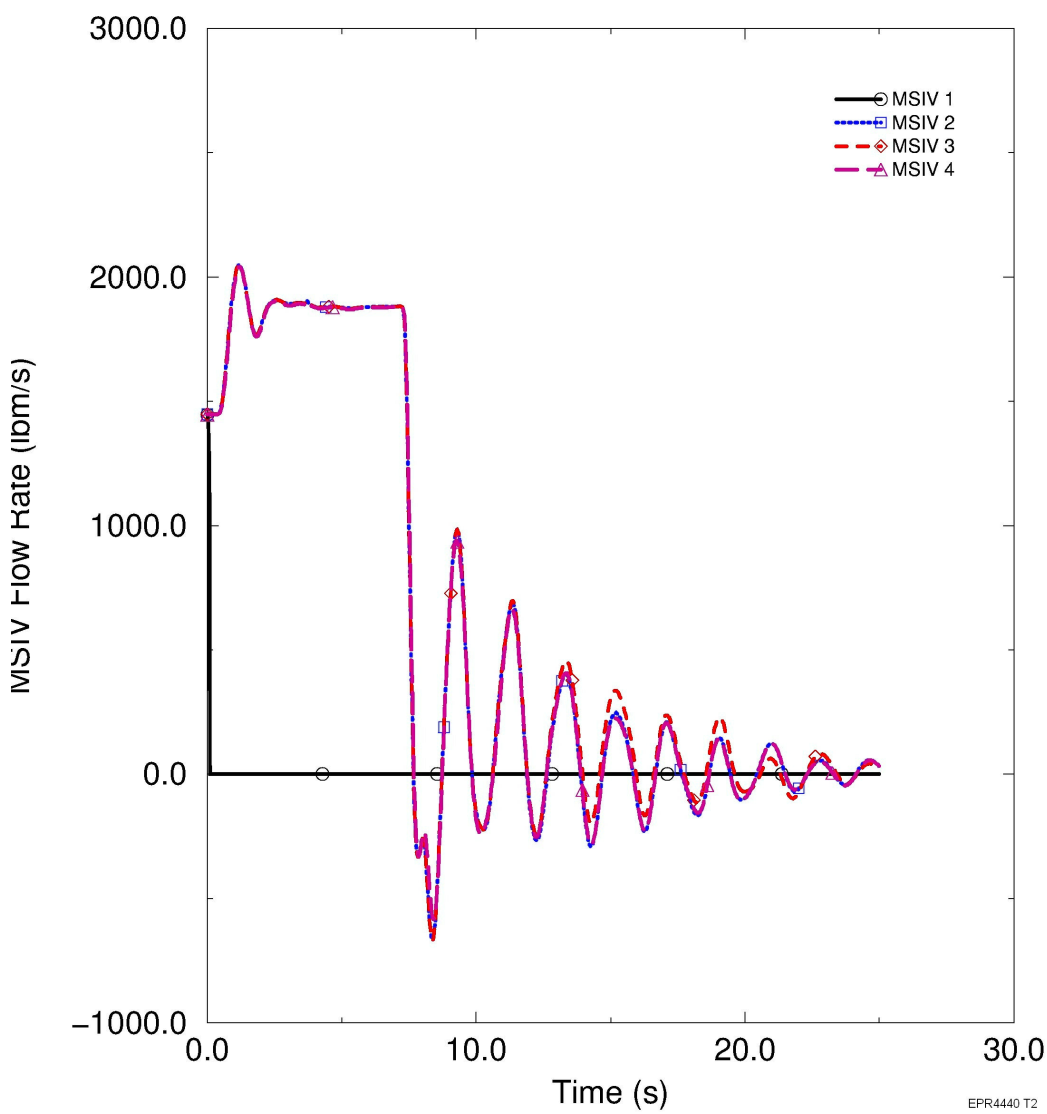

Figure 15.2-15—MSIVC Secondary Overpressurization - Safety Valve Flows for the Affected Loop

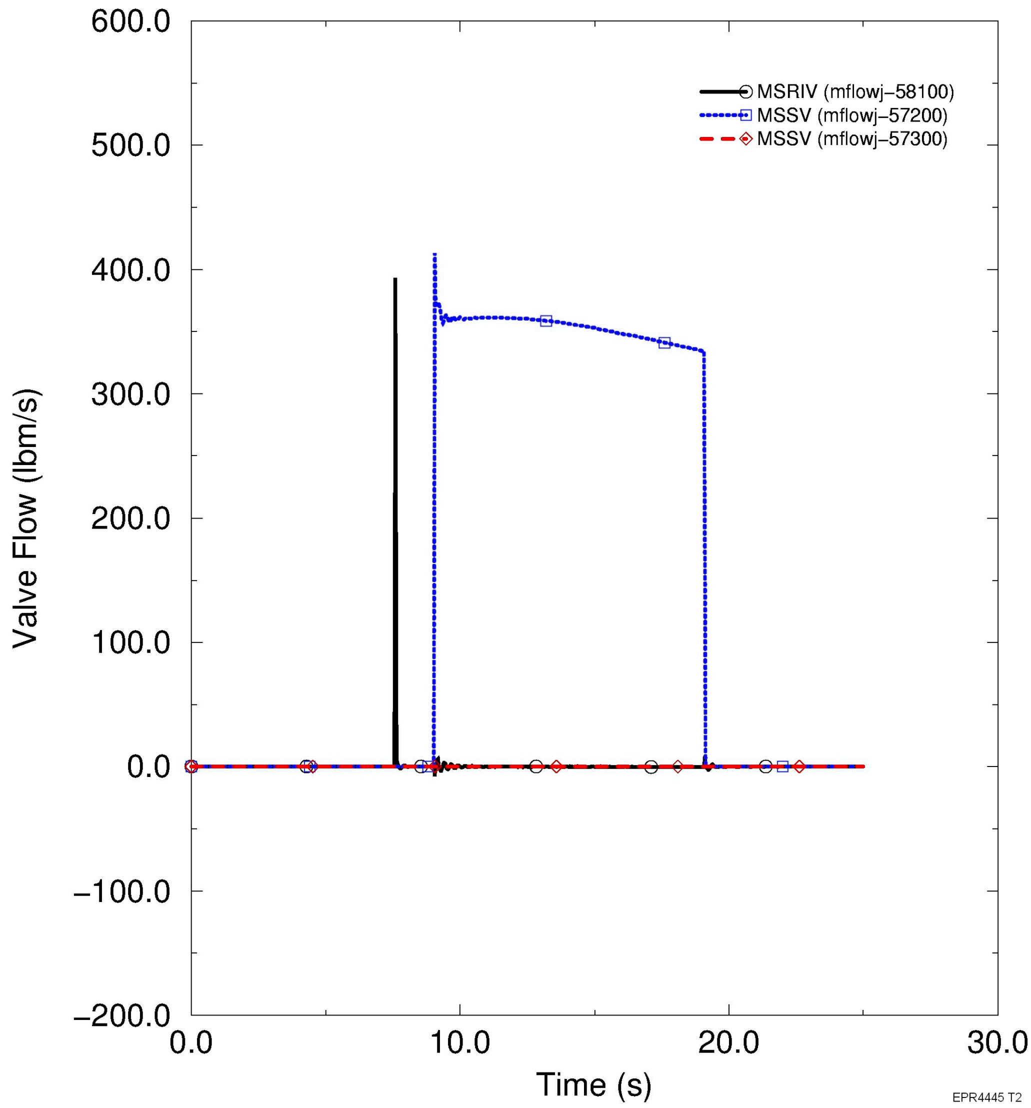

Figure 15.2-16—MSIVC Secondary Overpressurization—Cold Leg Temperatures

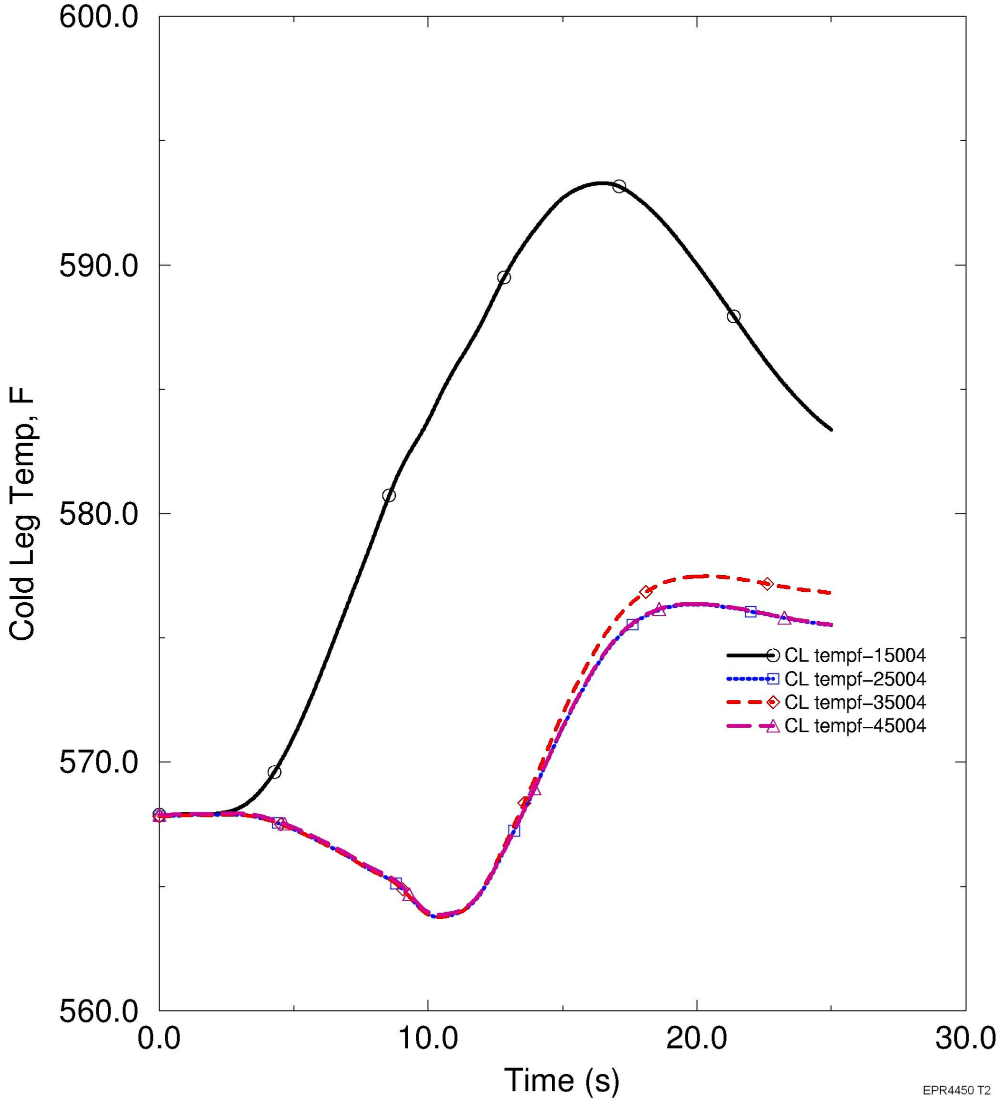

Figure 15.2-17—MSIVC Secondary Overpressurization — Average Core Heat Flux

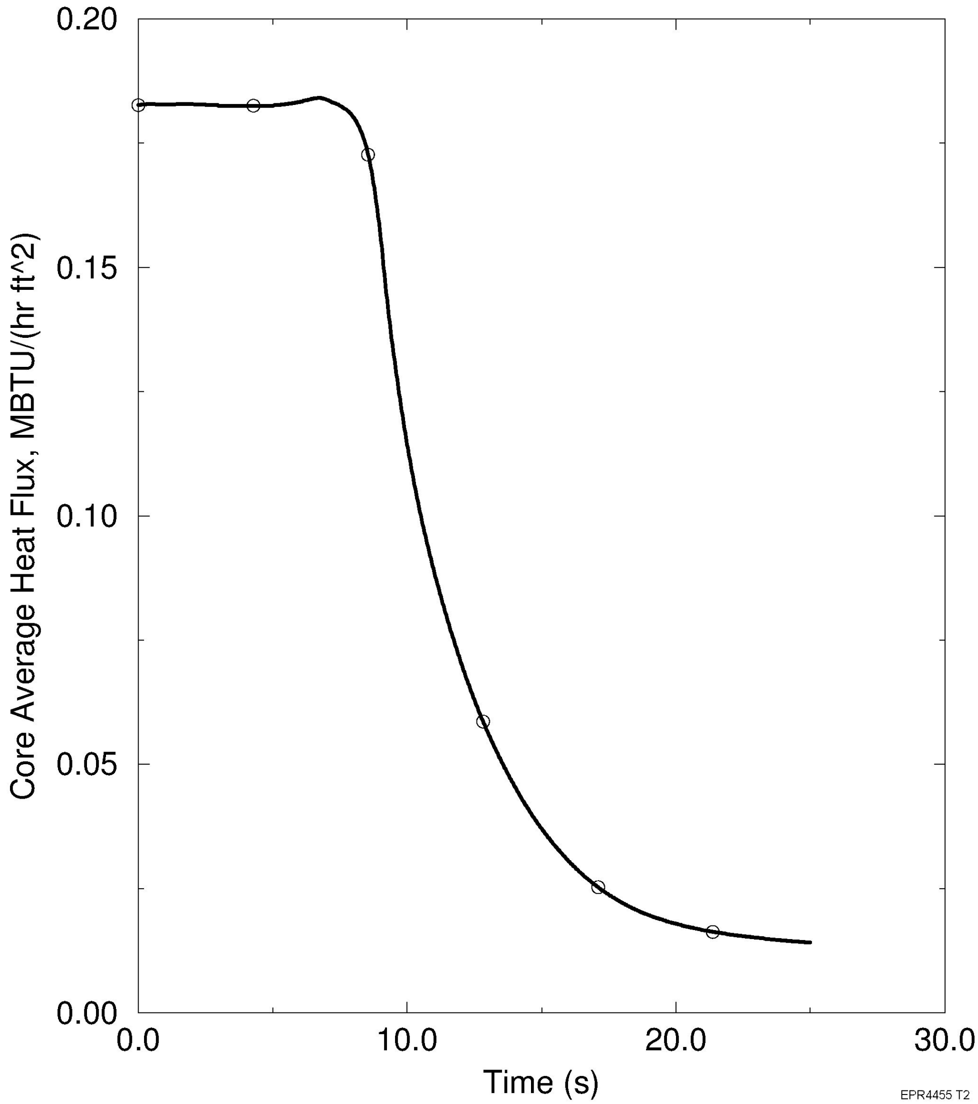

Figure 15.2-18—MSIVC Secondary Overpressurization — Maximum Primary Pressure

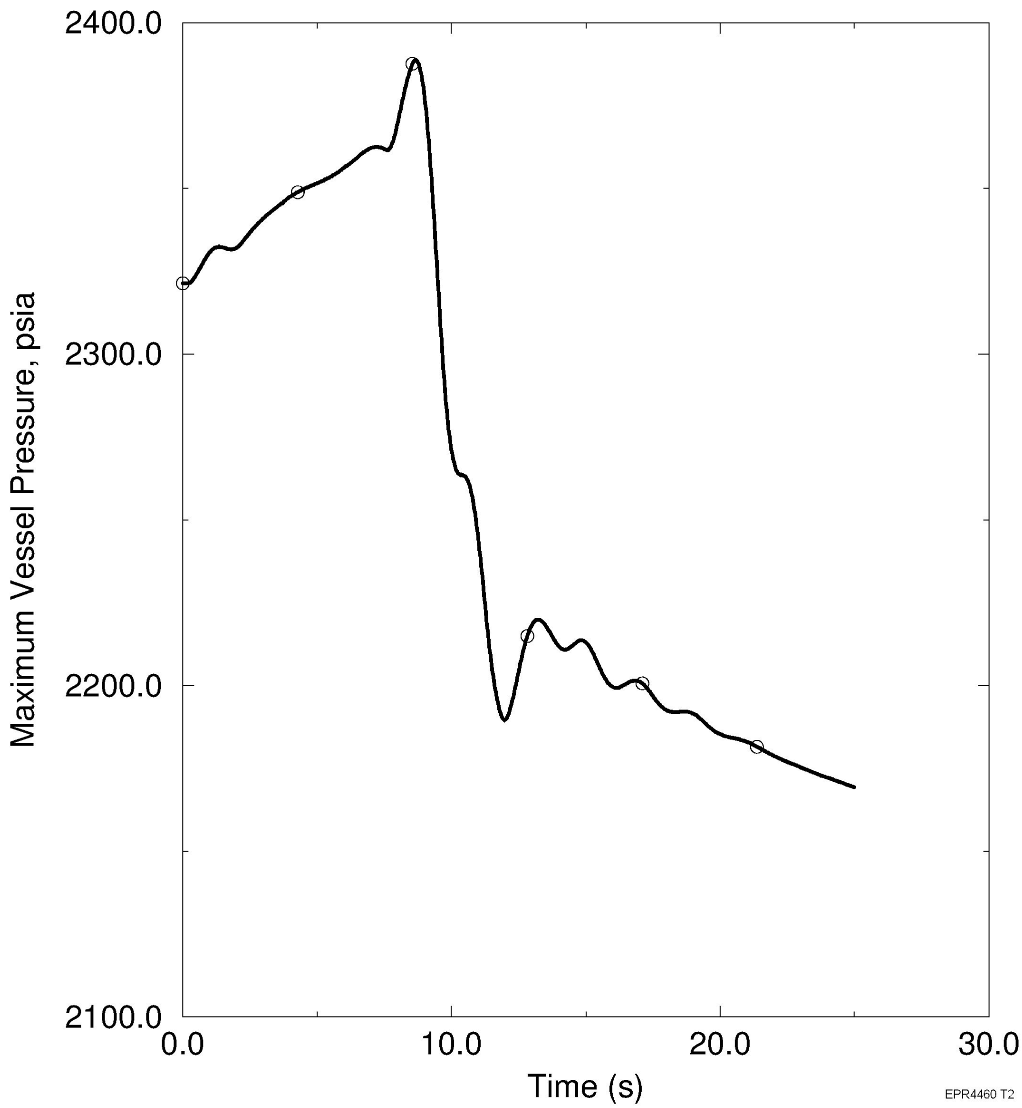

Figure 15.2-19—MSIVC Secondary Overpressurization — Reactor Power

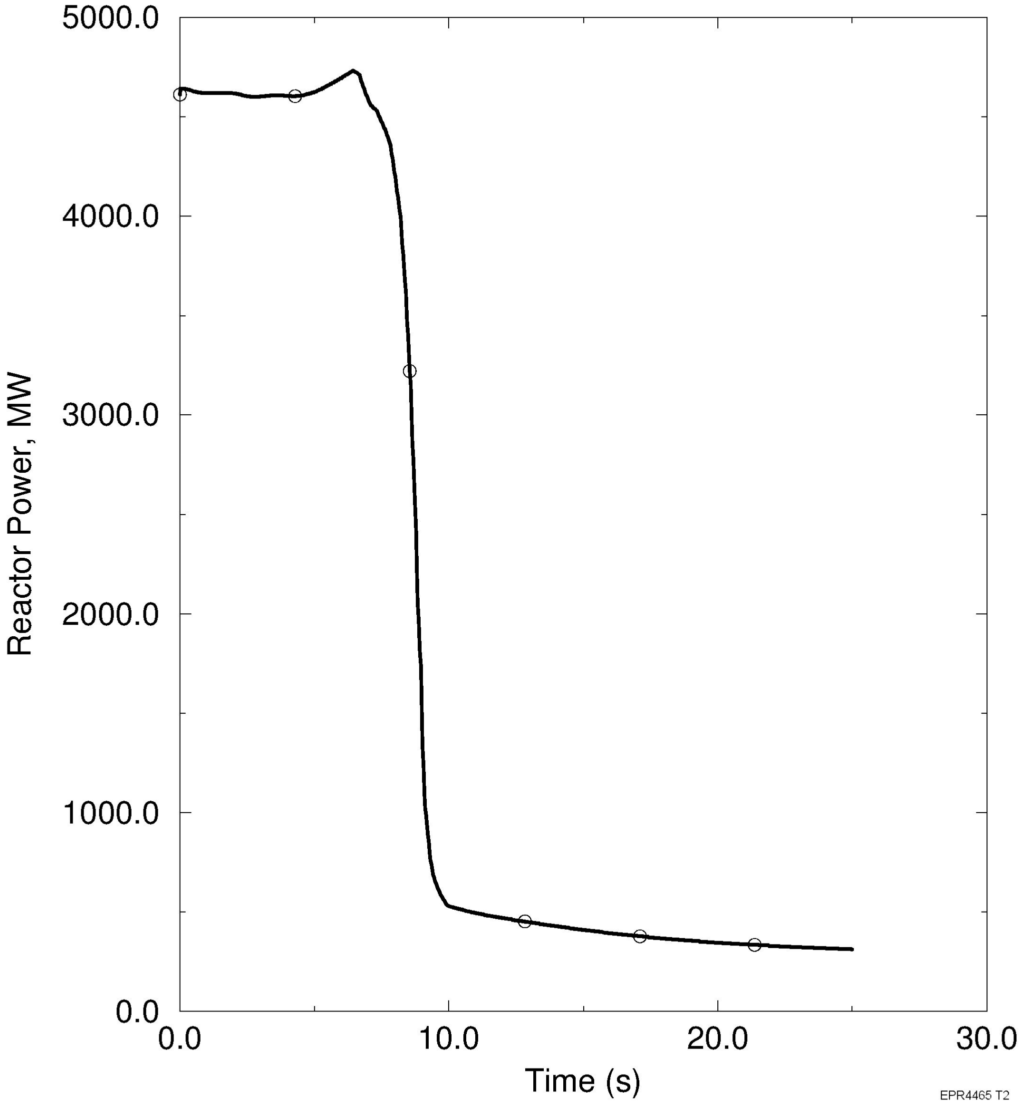

Figure 15.2-20—MSIVC Secondary Overpressurization — Pressurizer Level

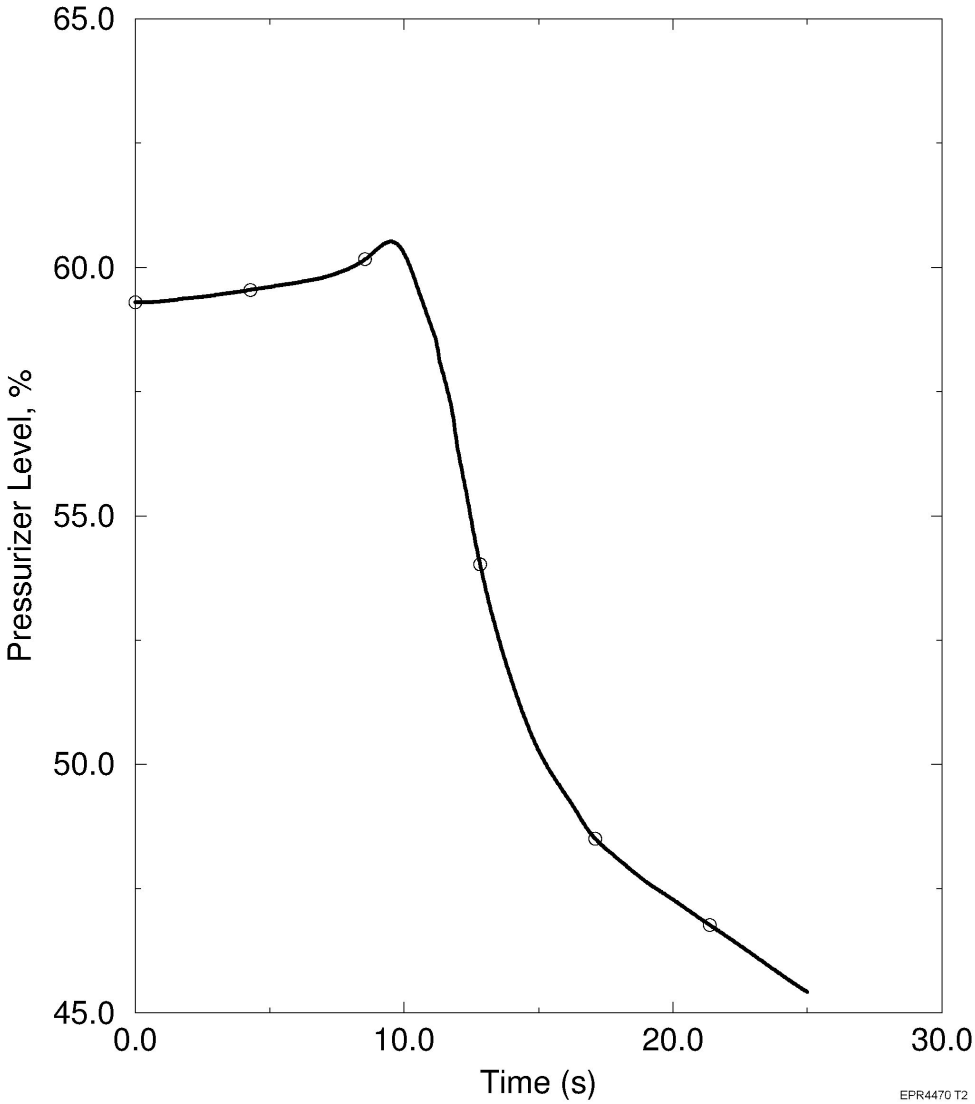

Figure 15.2-21—MSIVC Secondary Overpressurization — Pressurizer Relief Valve Flows

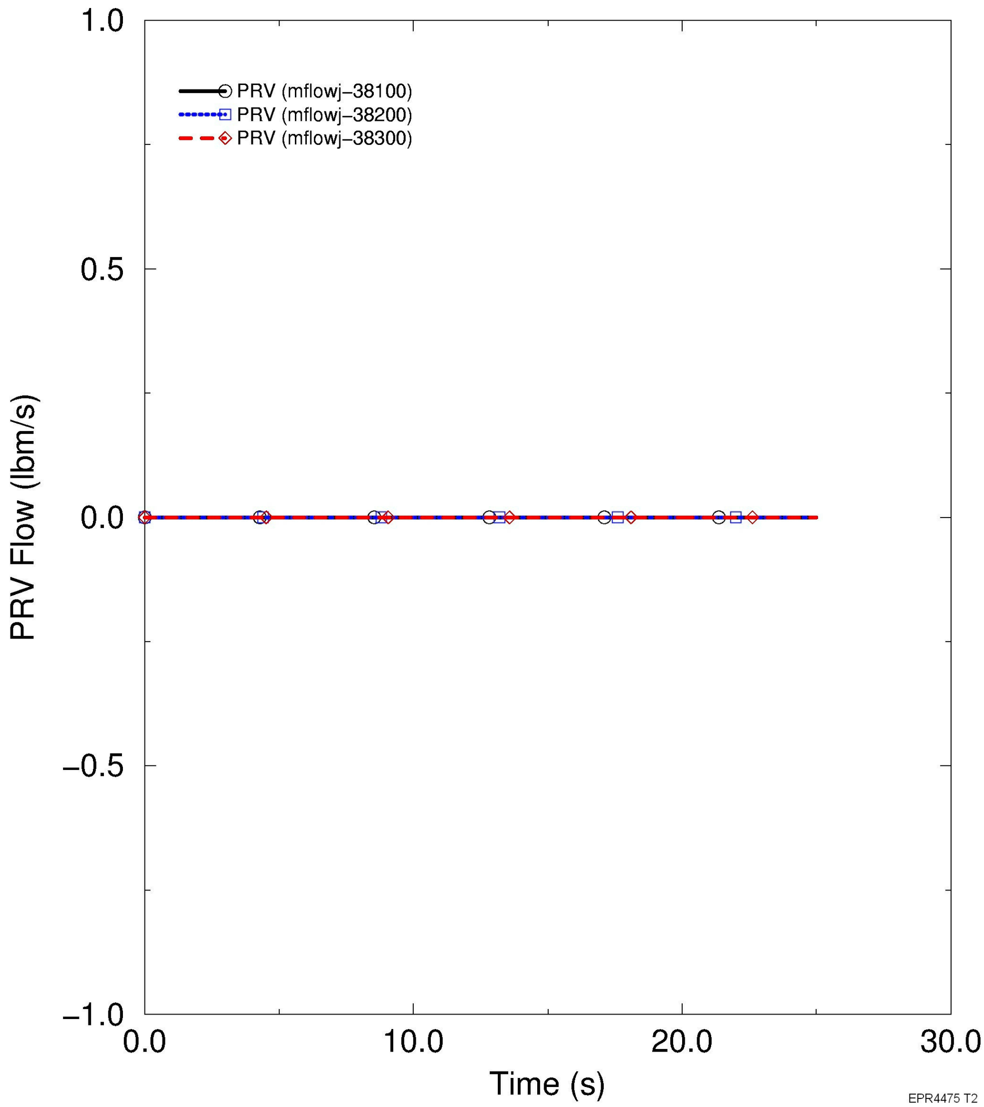

Figure 15.2-22—MSIVC Secondary Overpressurization—Total Reactivity

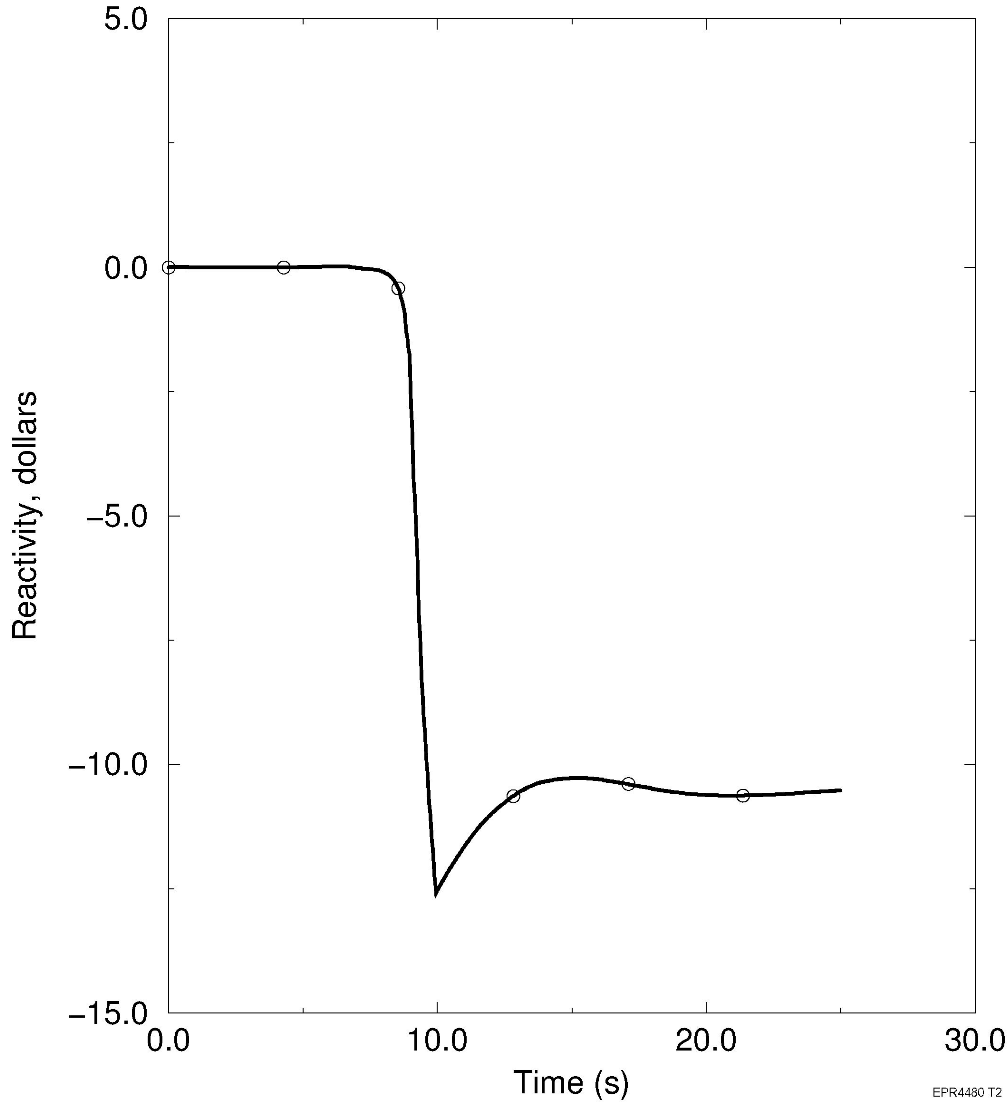

Figure 15.2-23—MSIVC Secondary Overpressurization — Maximum Secondary Pressures

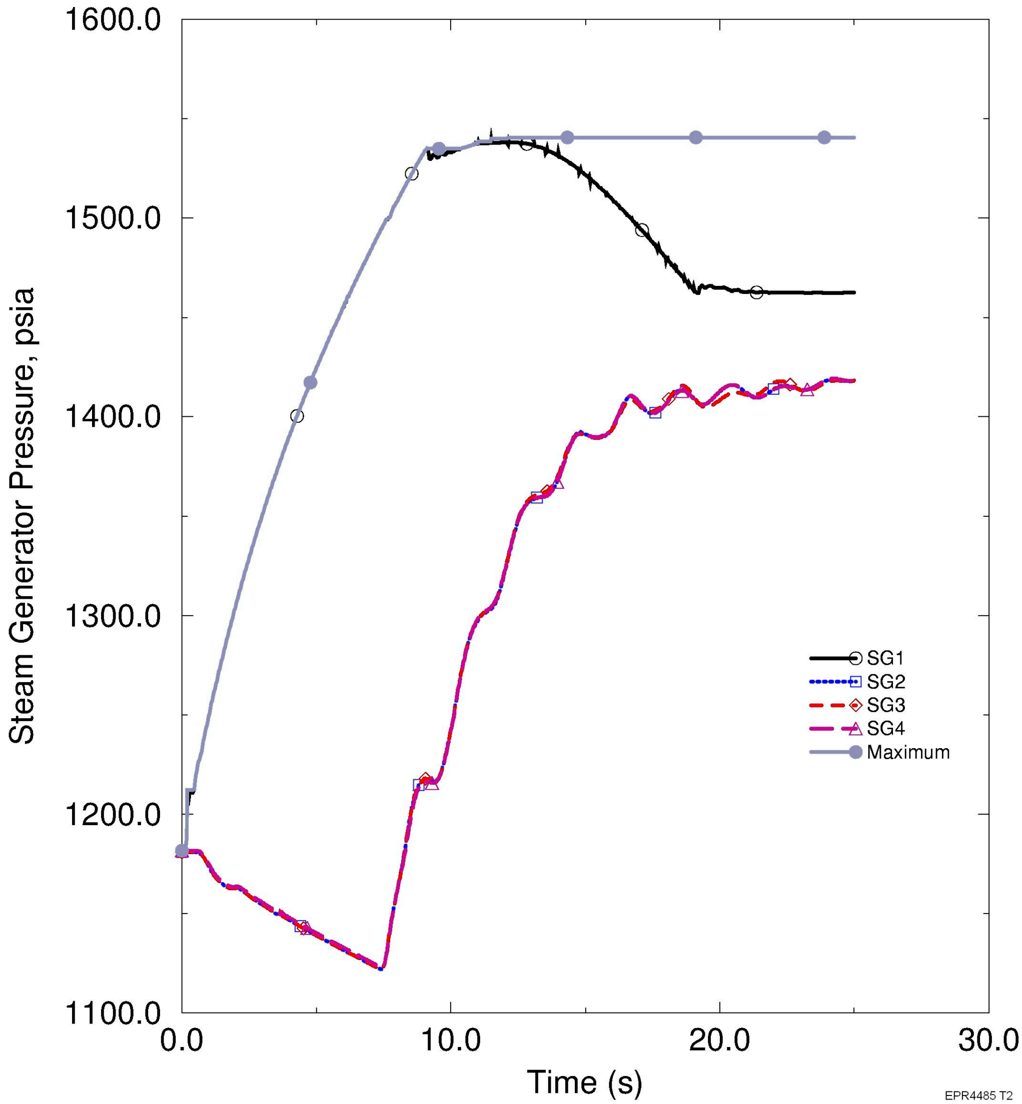

Figure 15.2-24—MSIVC Secondary Overpressurization — Average Core Fluid Temperature

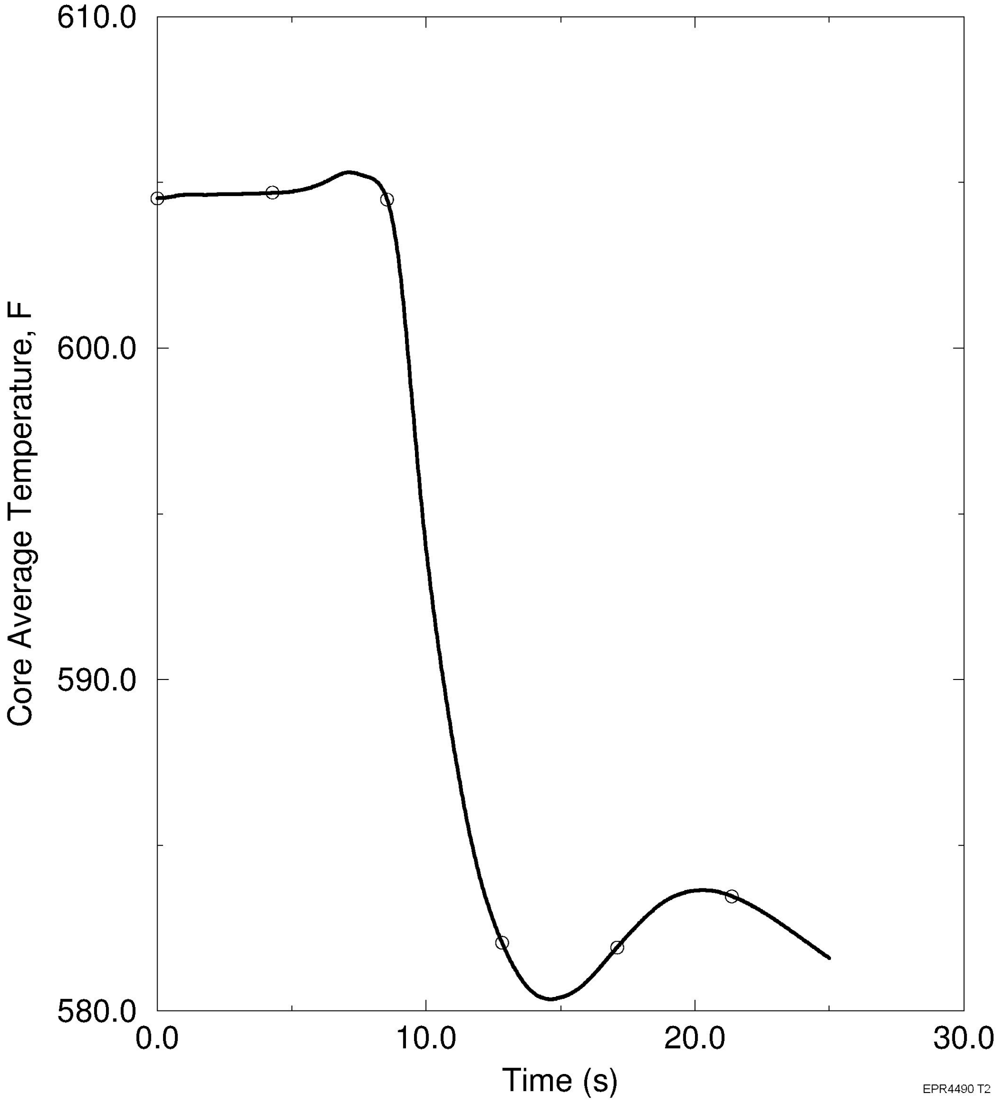

Figure 15.2-25—MSIVC Secondary Overpressurization — Turbine Control Valve Flow

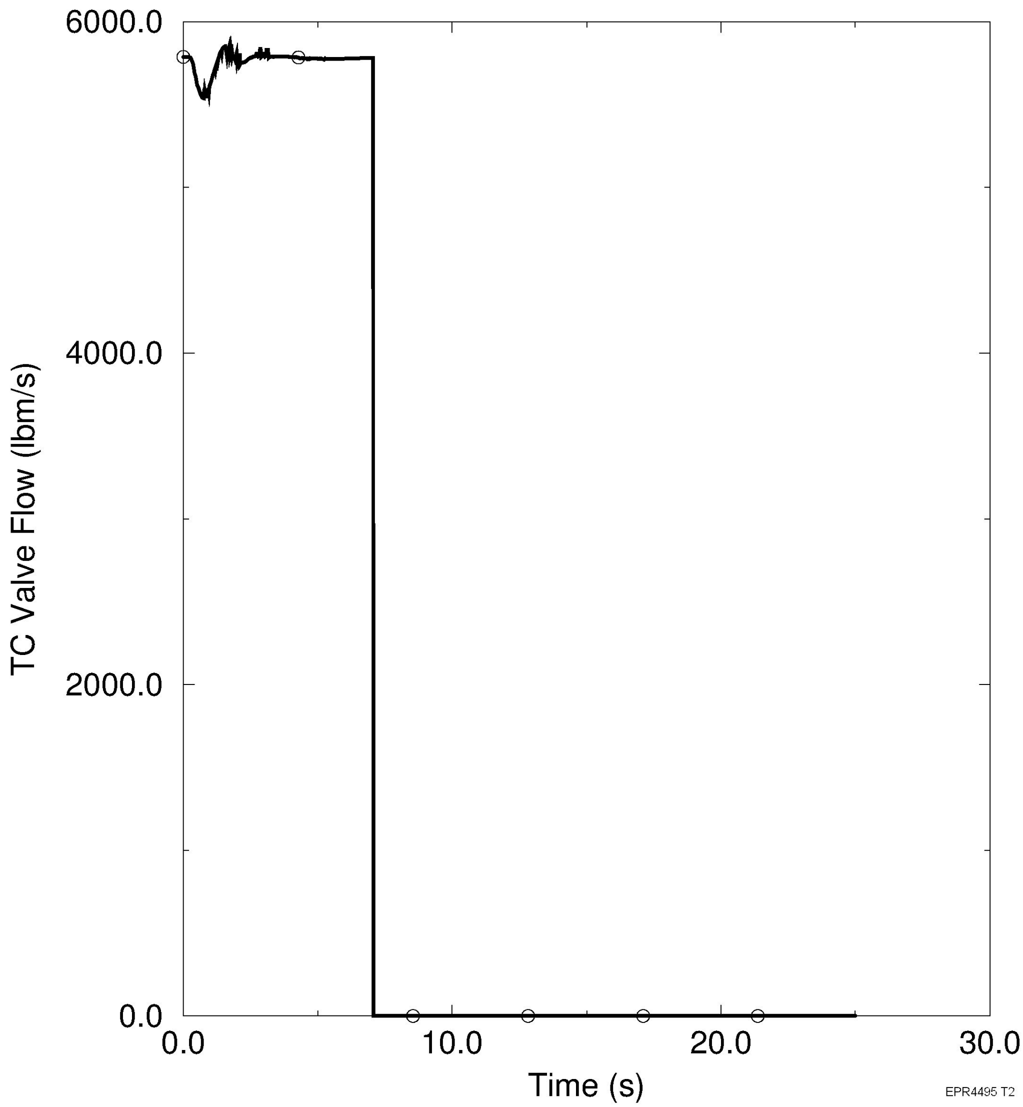

Figure 15.2-26—MSIVC Secondary Overpressurization — Safety Valve Flows for an Unaffected Loop

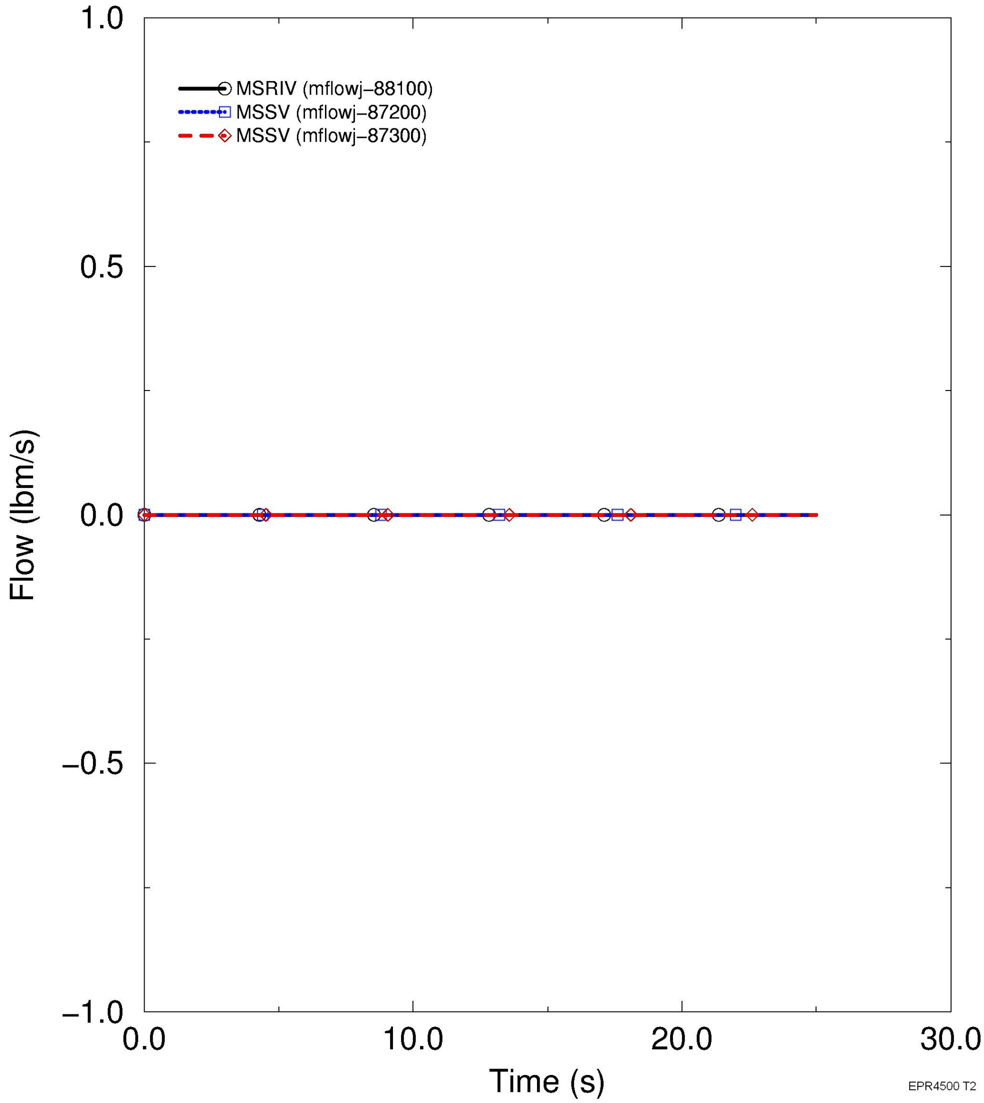

Figure 15.2-27—Loss of Normal Feedwater - Reactor and Total SG Power

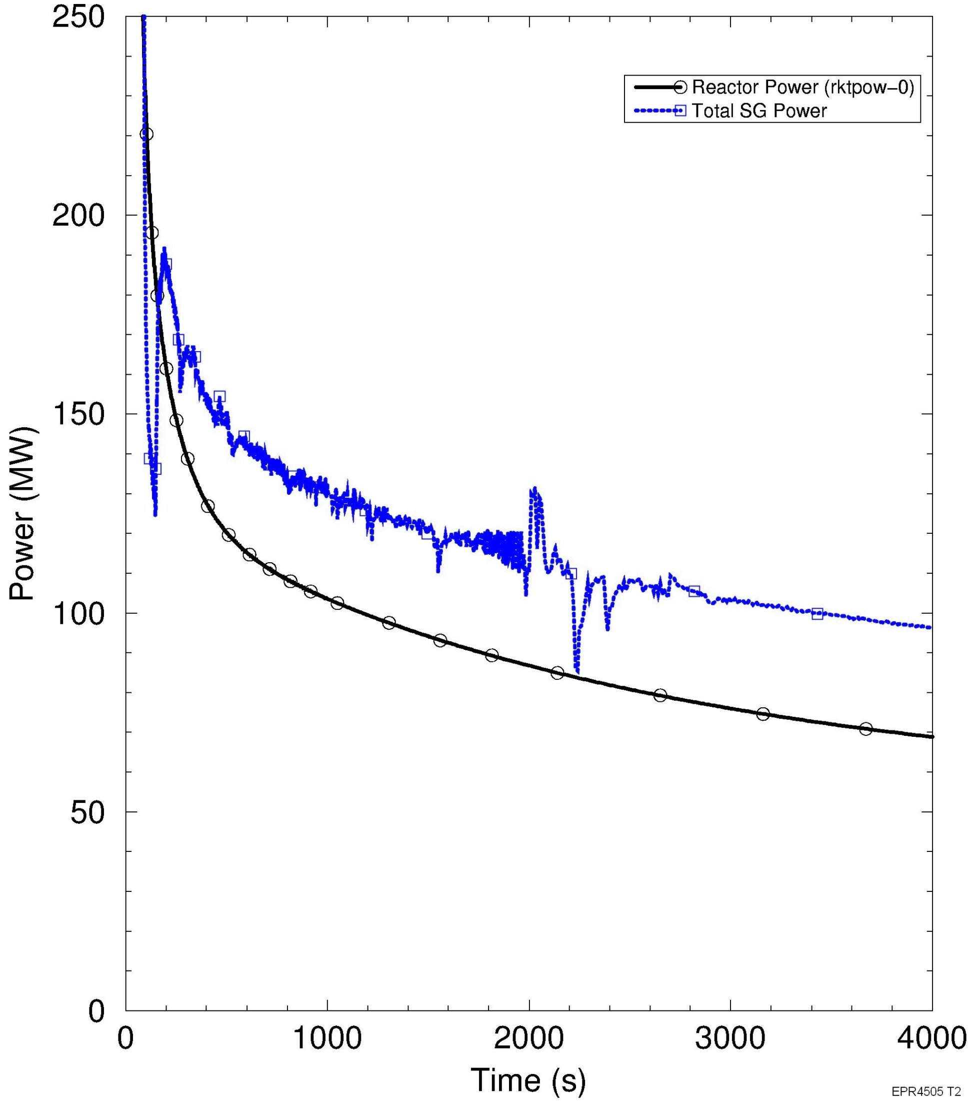

Figure 15.2-28—Loss of Normal Feedwater - SG Power

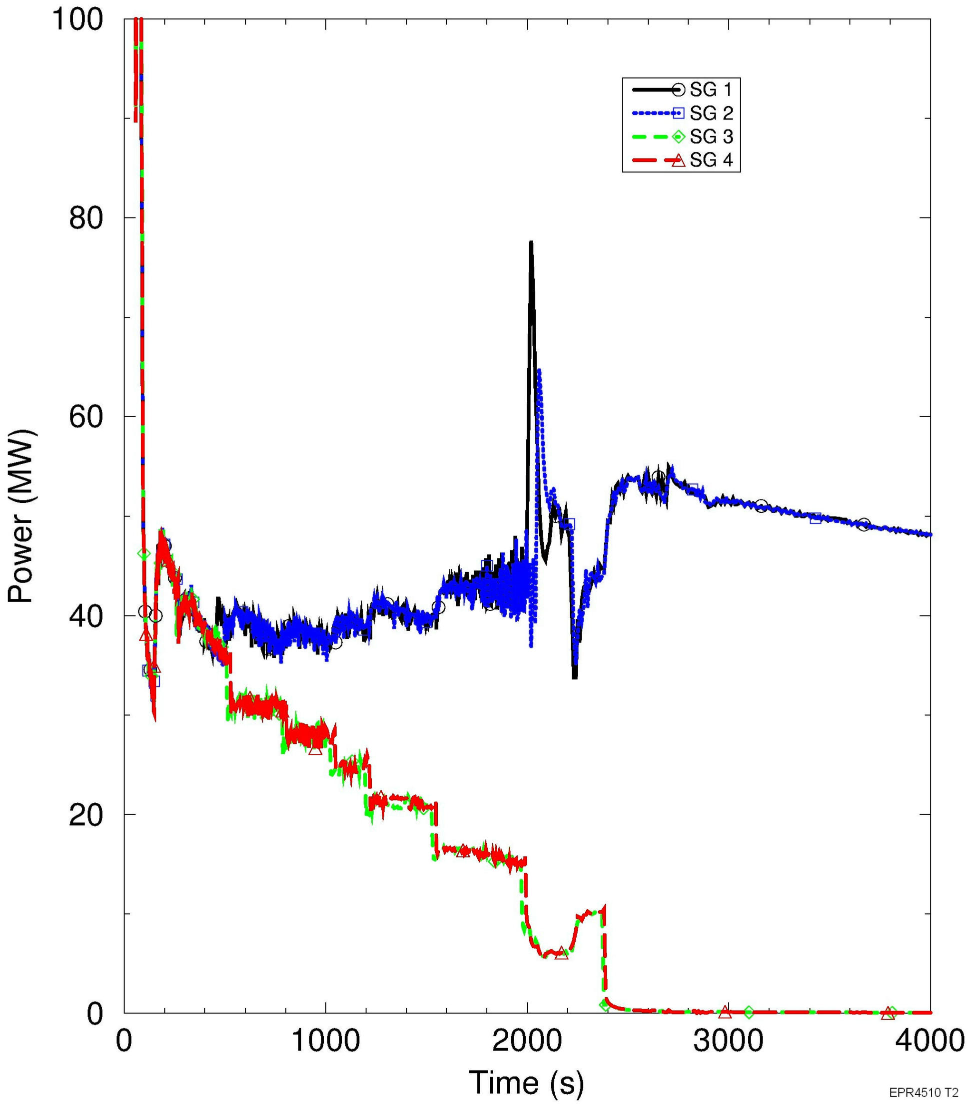

Figure 15.2-29—Loss of Normal Feedwater - PZR Pressure

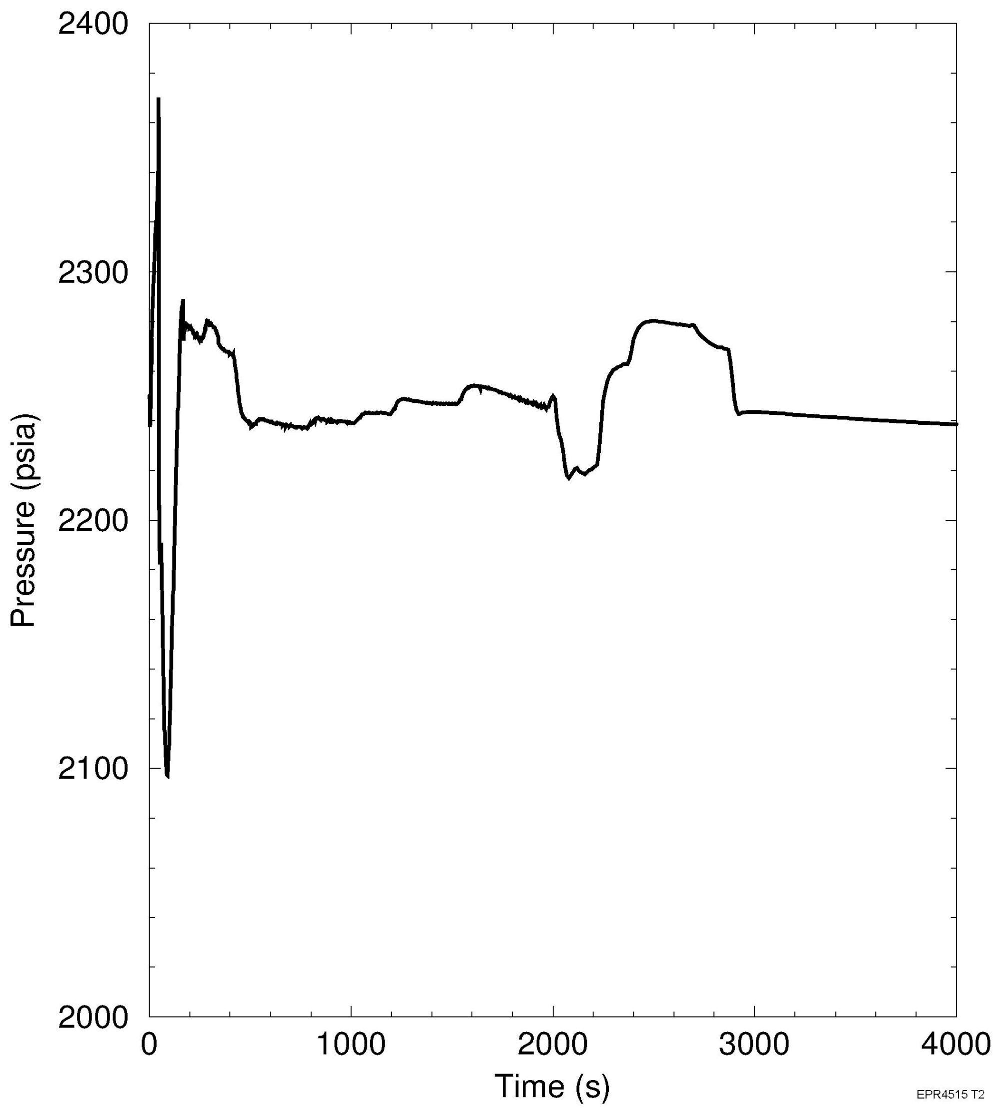

Figure 15.2-30—Loss of Normal Feedwater - PZR Spray Flow

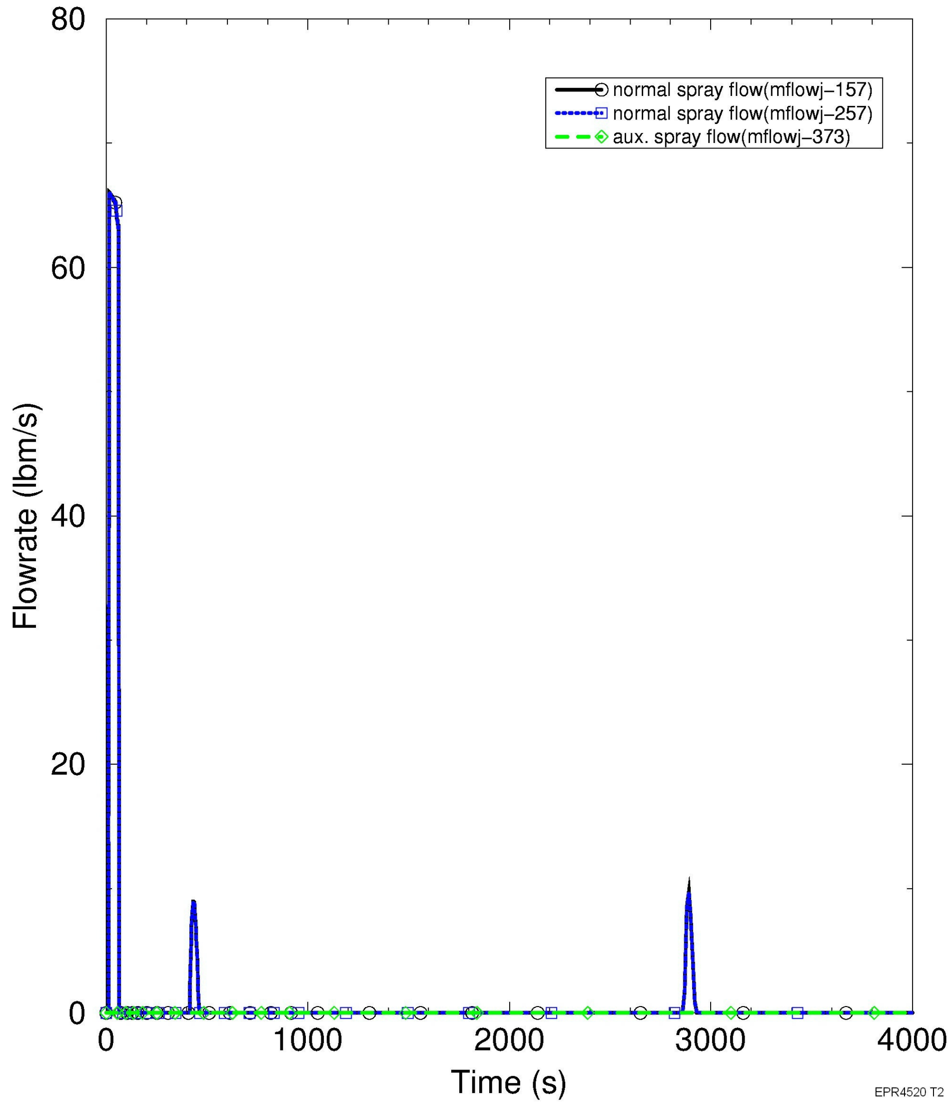

Figure 15.2-31—Loss of Normal Feedwater - PSRV Flow

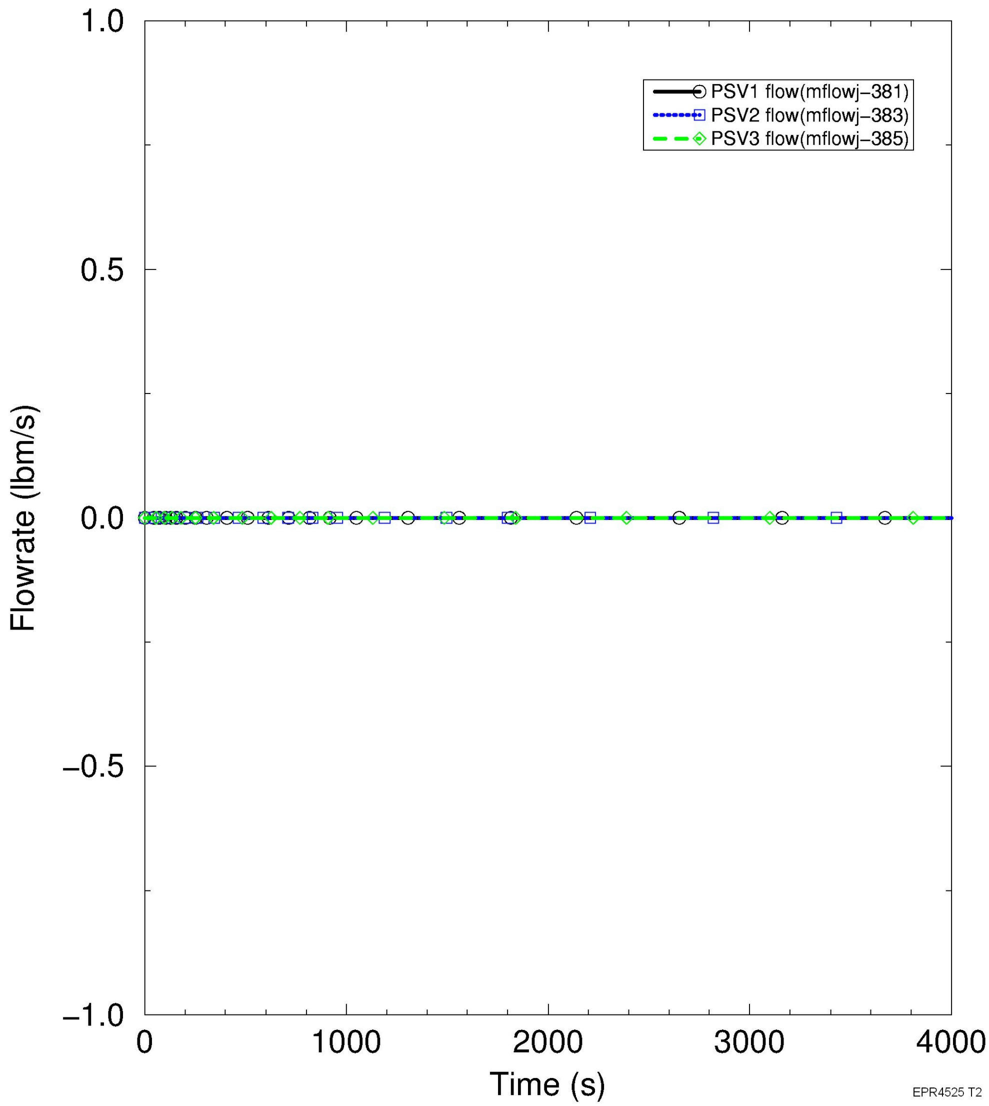

Figure 15.2-32—Loss of Normal Feedwater - PZR Liquid Level

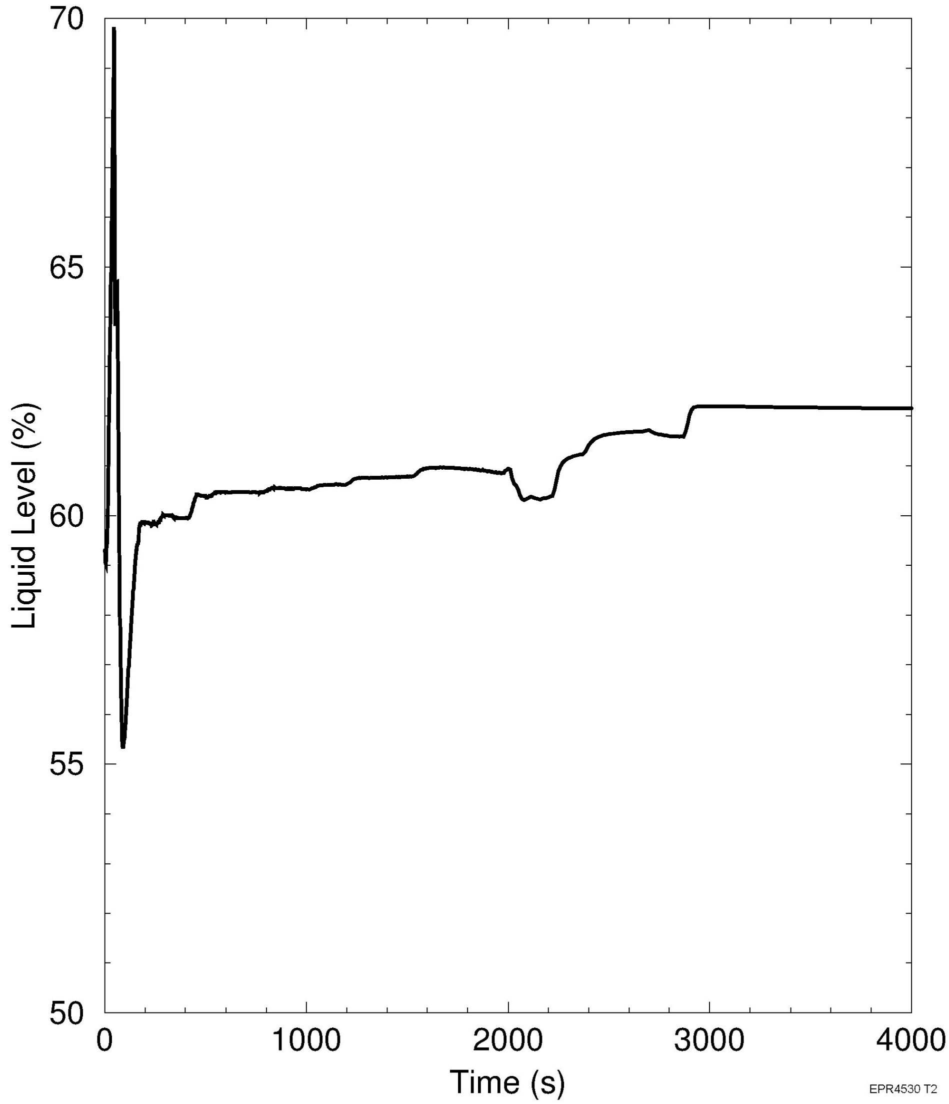

Figure 15.2-33—Loss of Normal Feedwater - RCS Loop Mass Flow

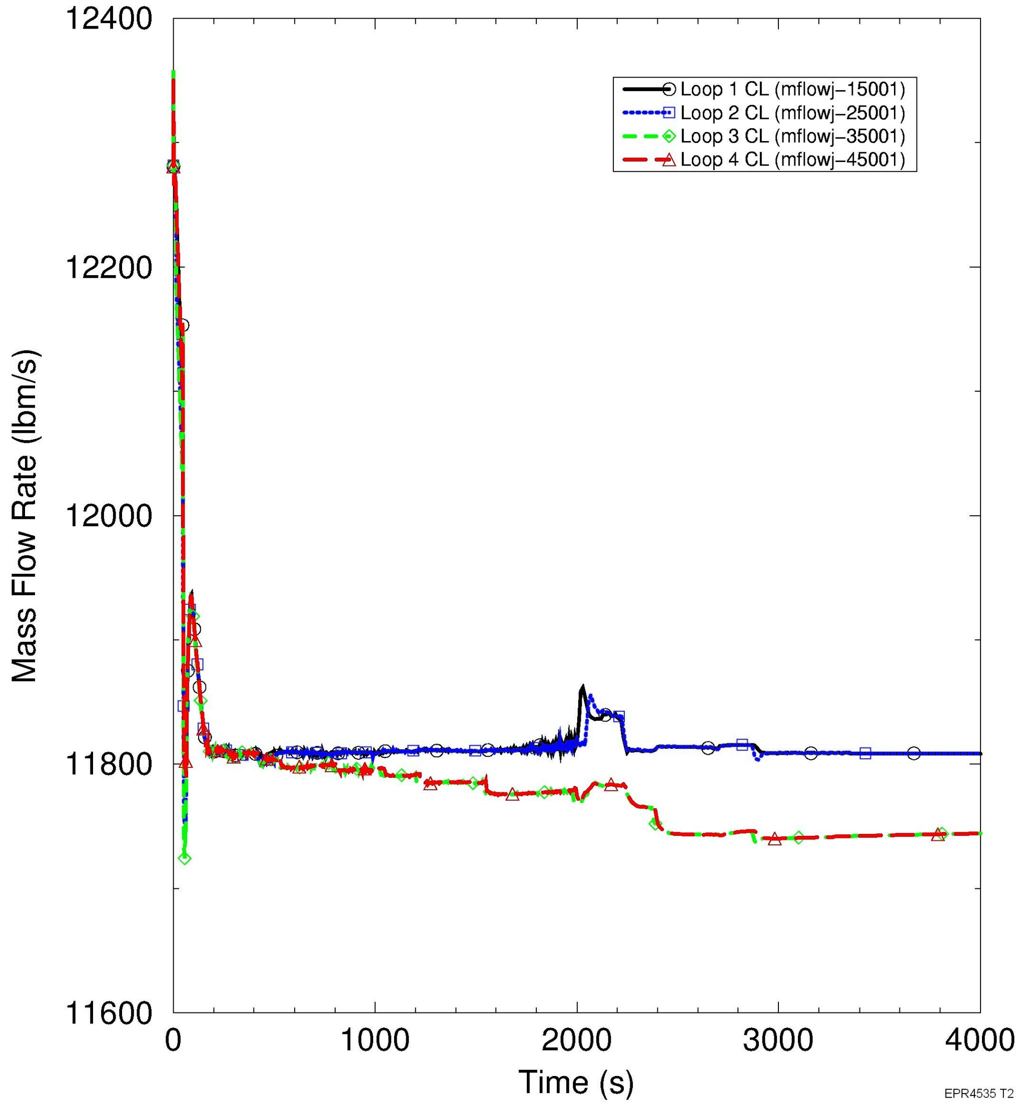

Figure 15.2-34—Loss of Normal Feedwater - RCS Temperatures

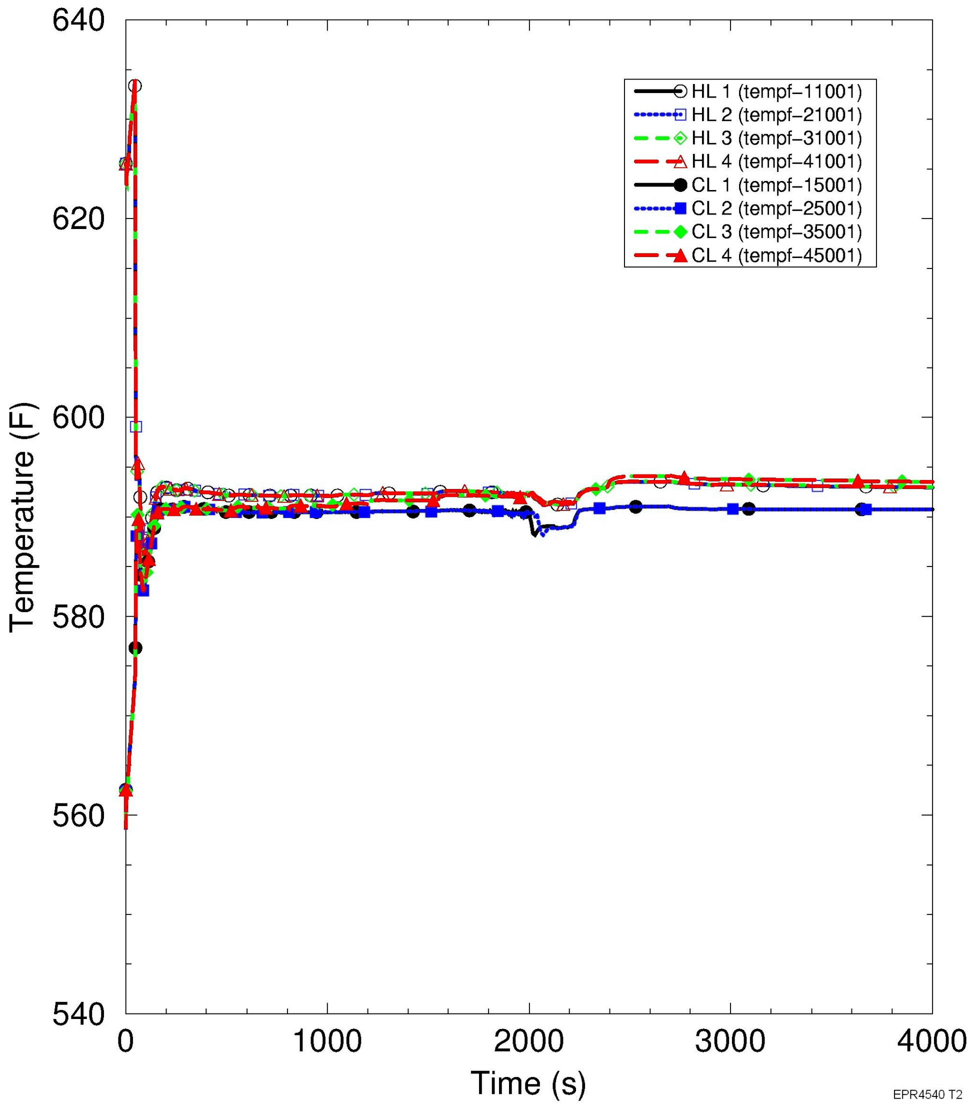

Figure 15.2-35—Loss of Normal Feedwater - Core Exit Subcooling

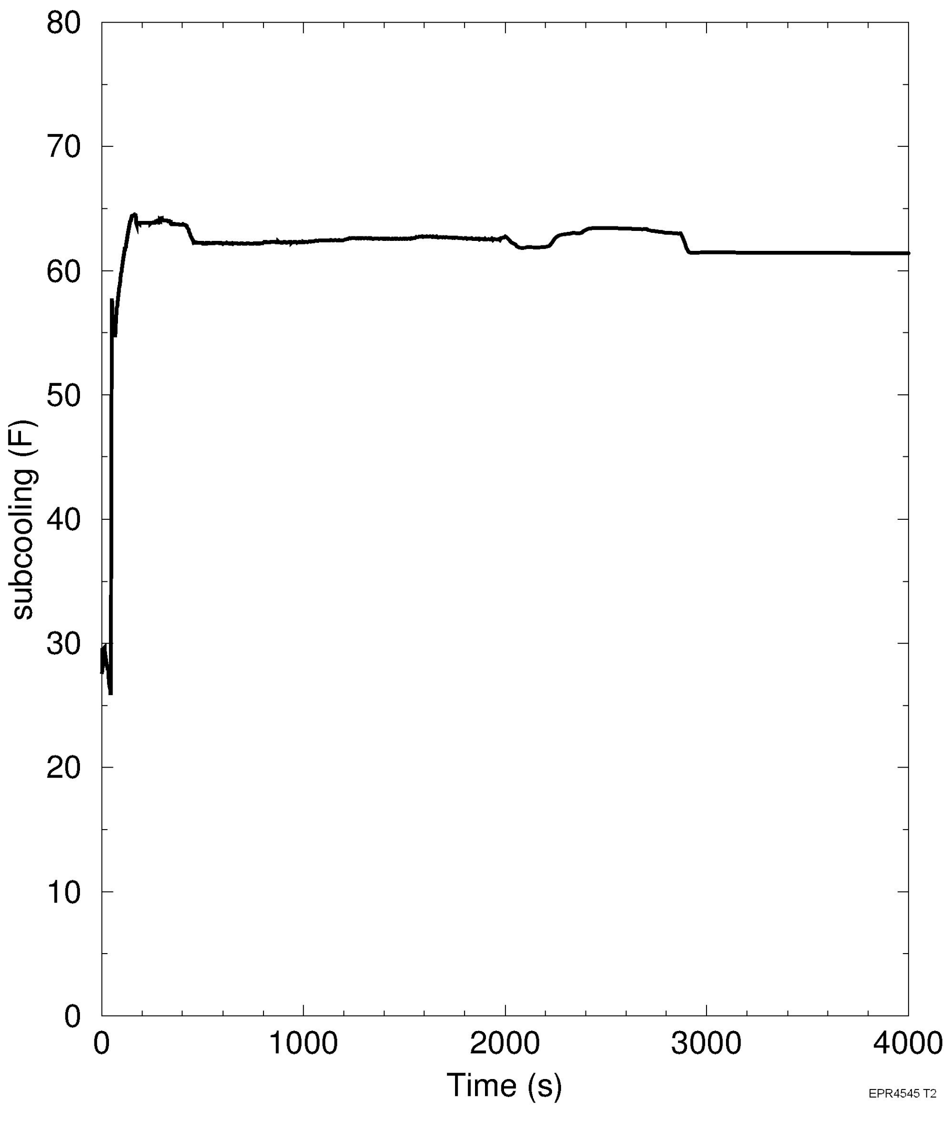

Figure 15.2-36—Loss of Normal Feedwater - Core Flow Rate

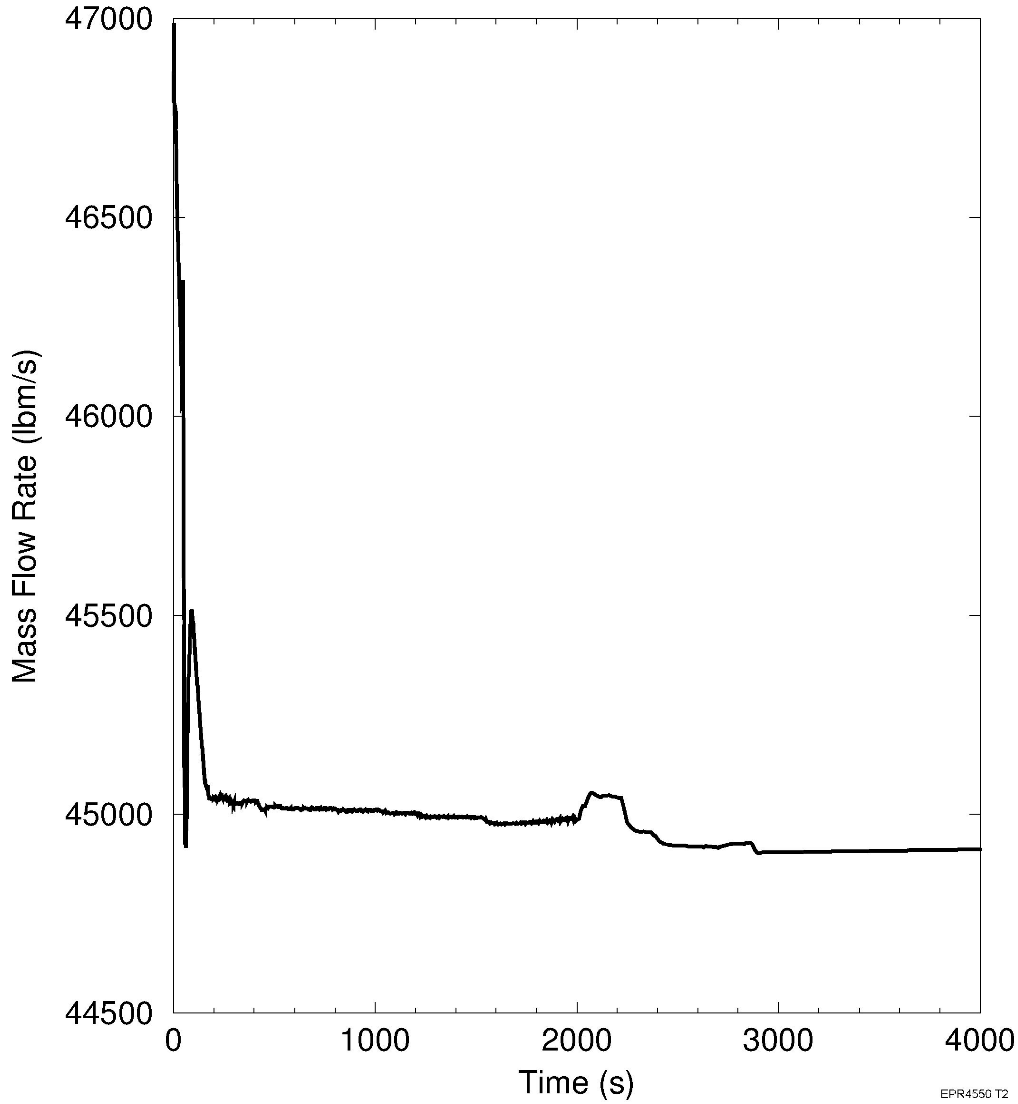

Figure 15.2-37—Loss of Normal Feedwater - Flow Rates through Main Steam Relief Train, Unaffected SGs

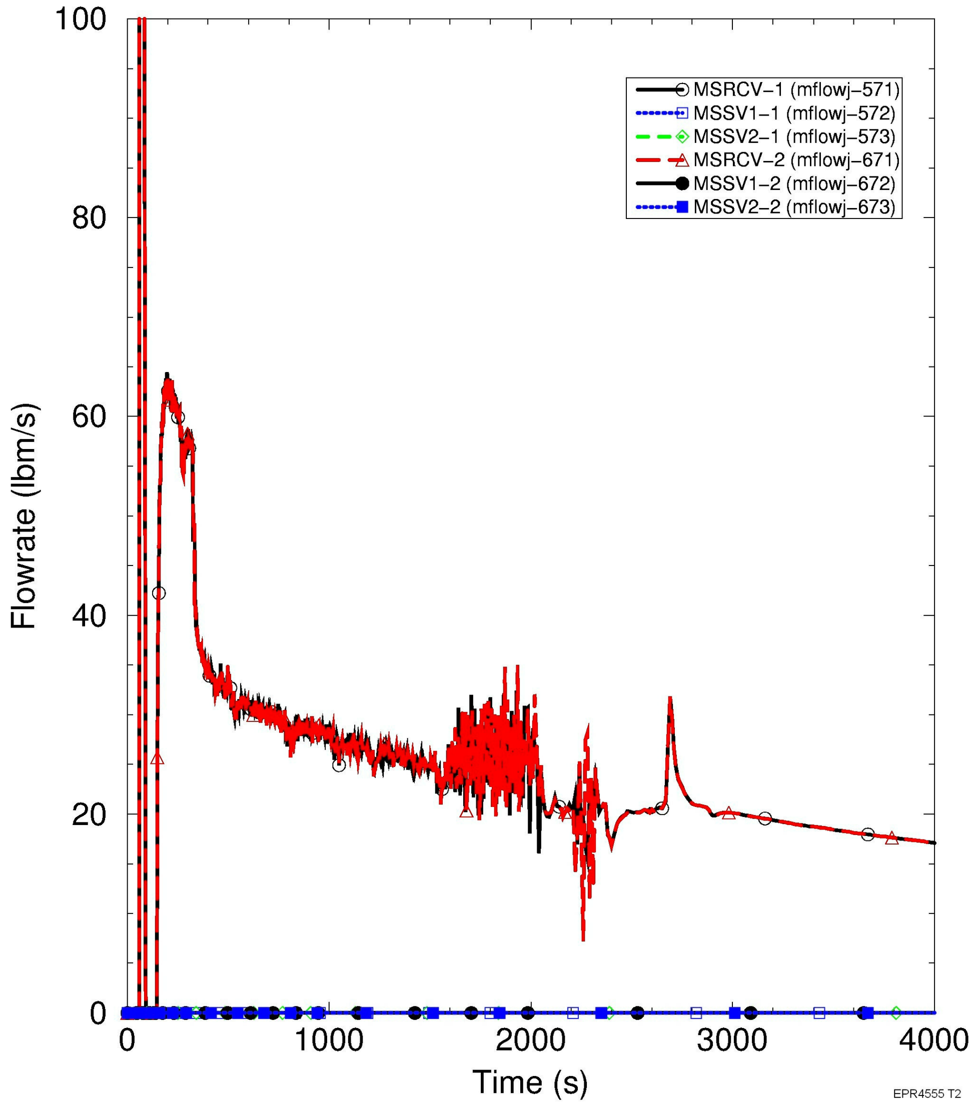

Figure 15.2-38—Loss of Normal Feedwater - Flow Rates through Main Steam Relief Train, Affected SGs

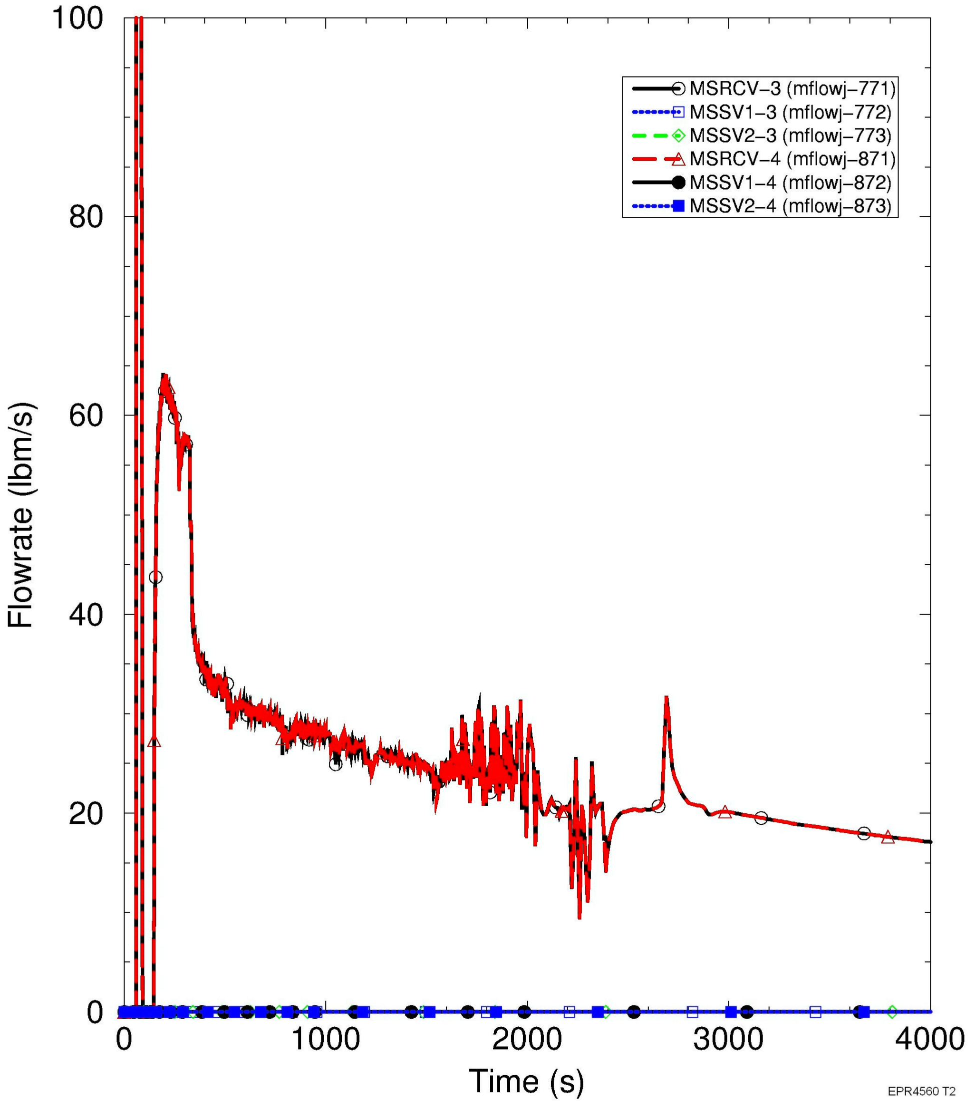

Figure 15.2-39—Loss of Normal Feedwater - SG Pressure

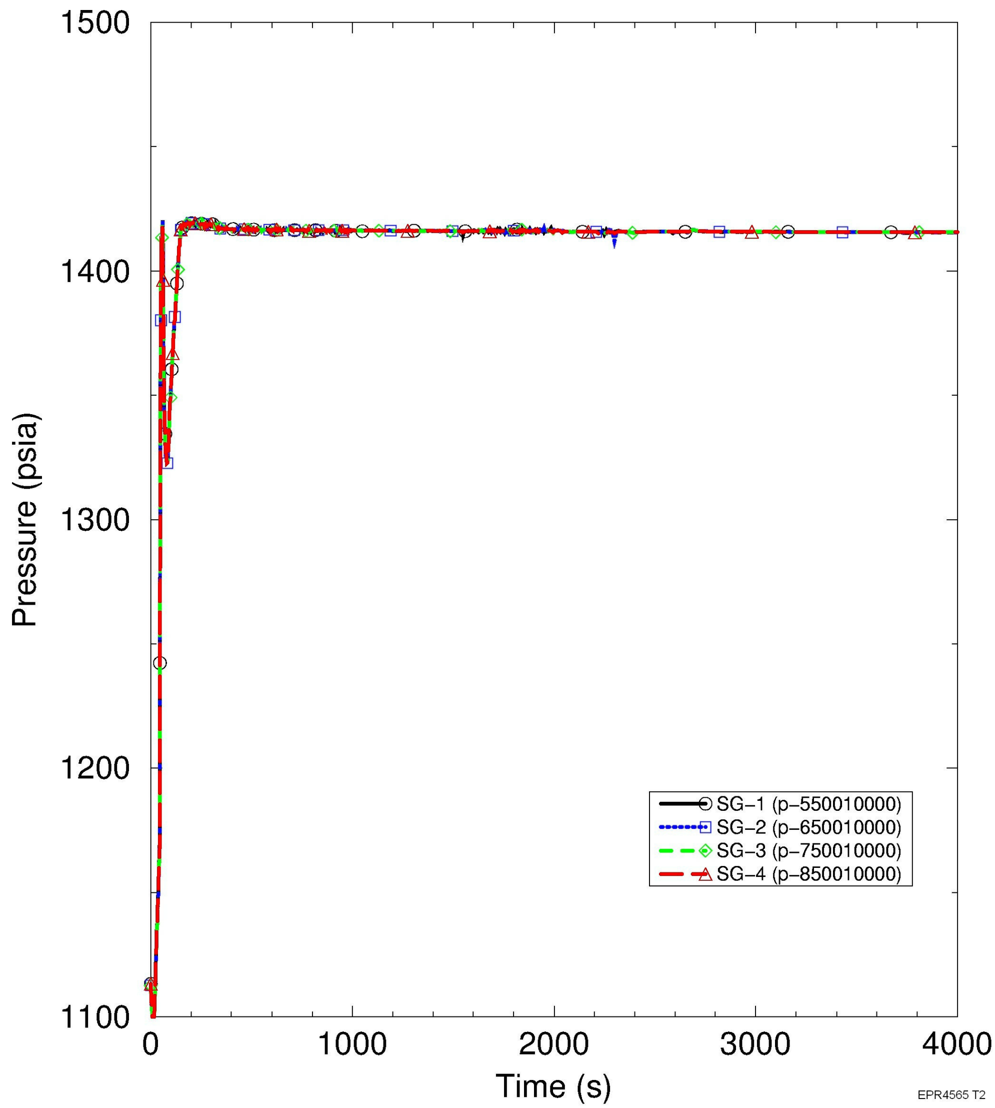

Figure 15.2-40—Loss of Normal Feedwater - SG NR Liquid Level

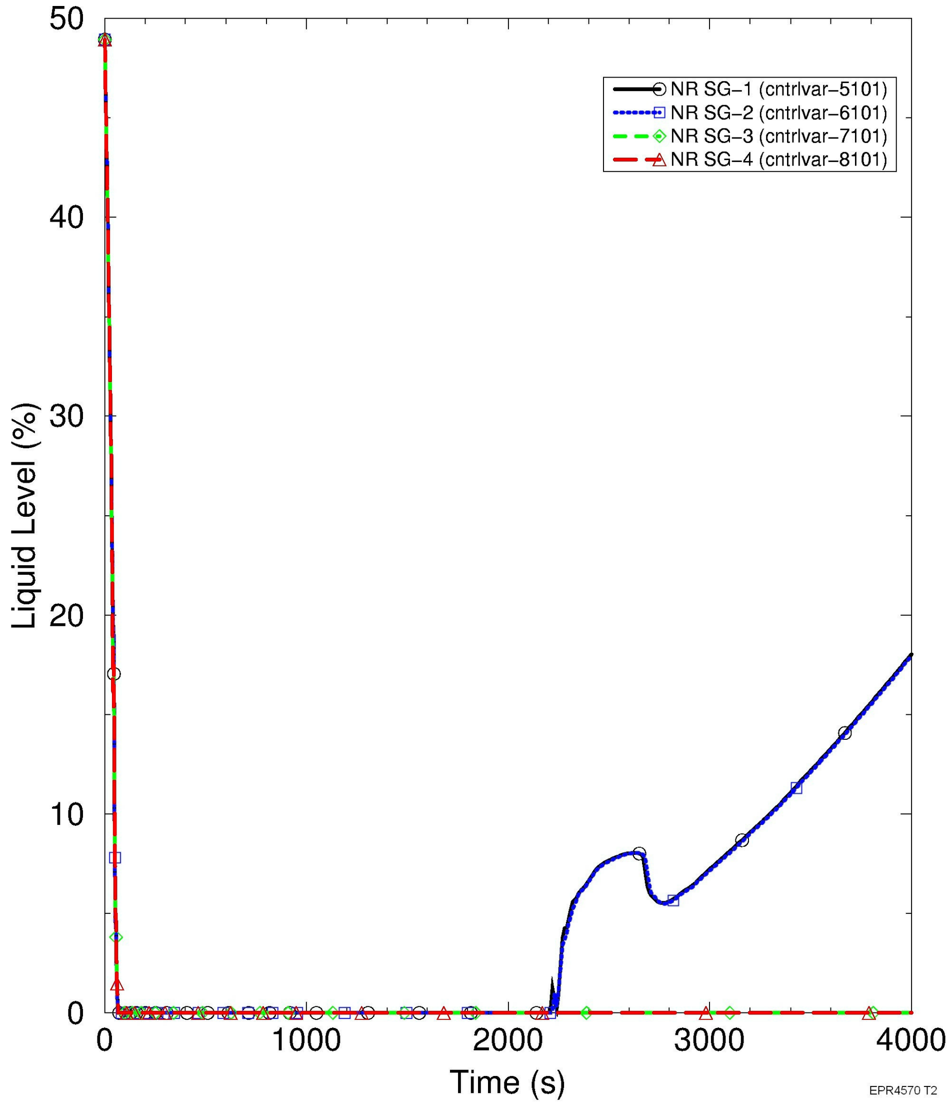

Figure 15.2-41—Loss of Normal Feedwater - SG WR Liquid Level

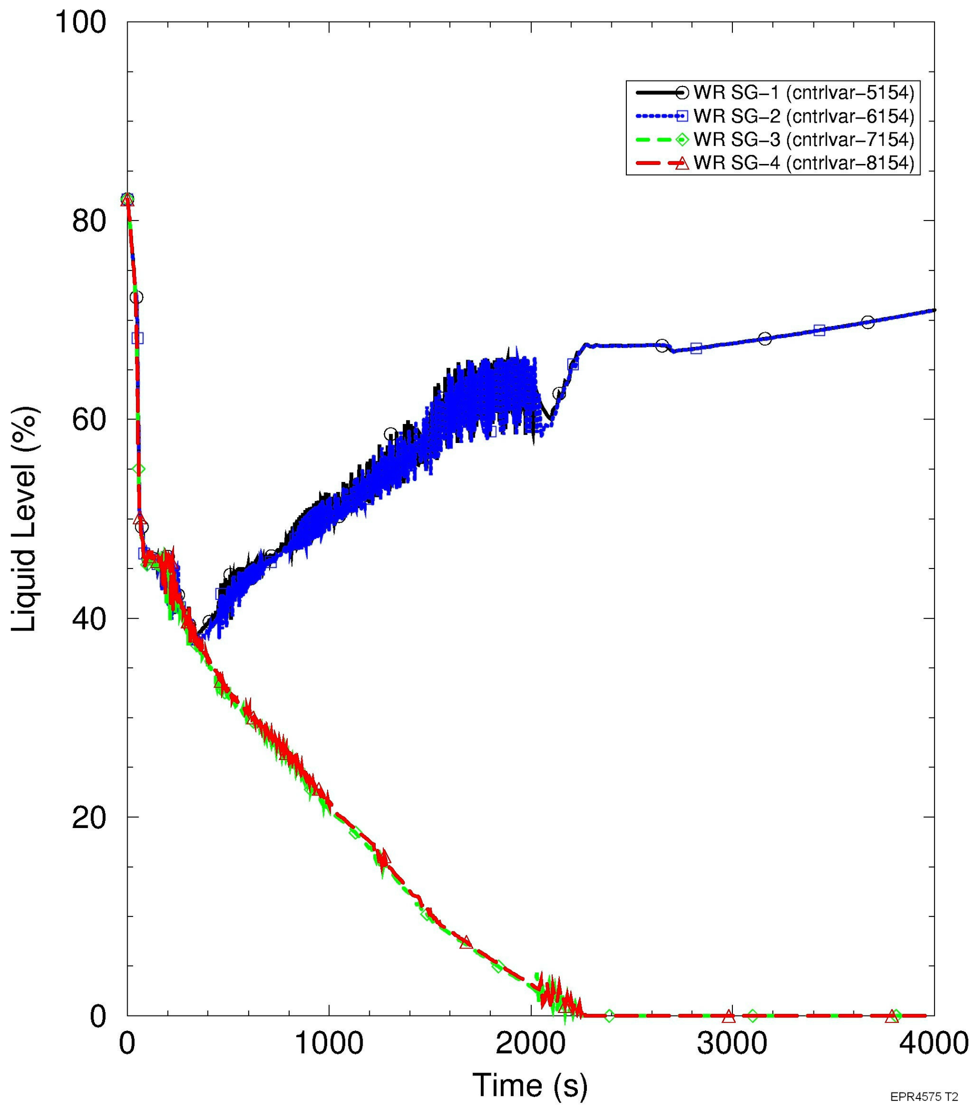

Figure 15.2-42—Loss of Normal Feedwater - SG Total Mass Inventory

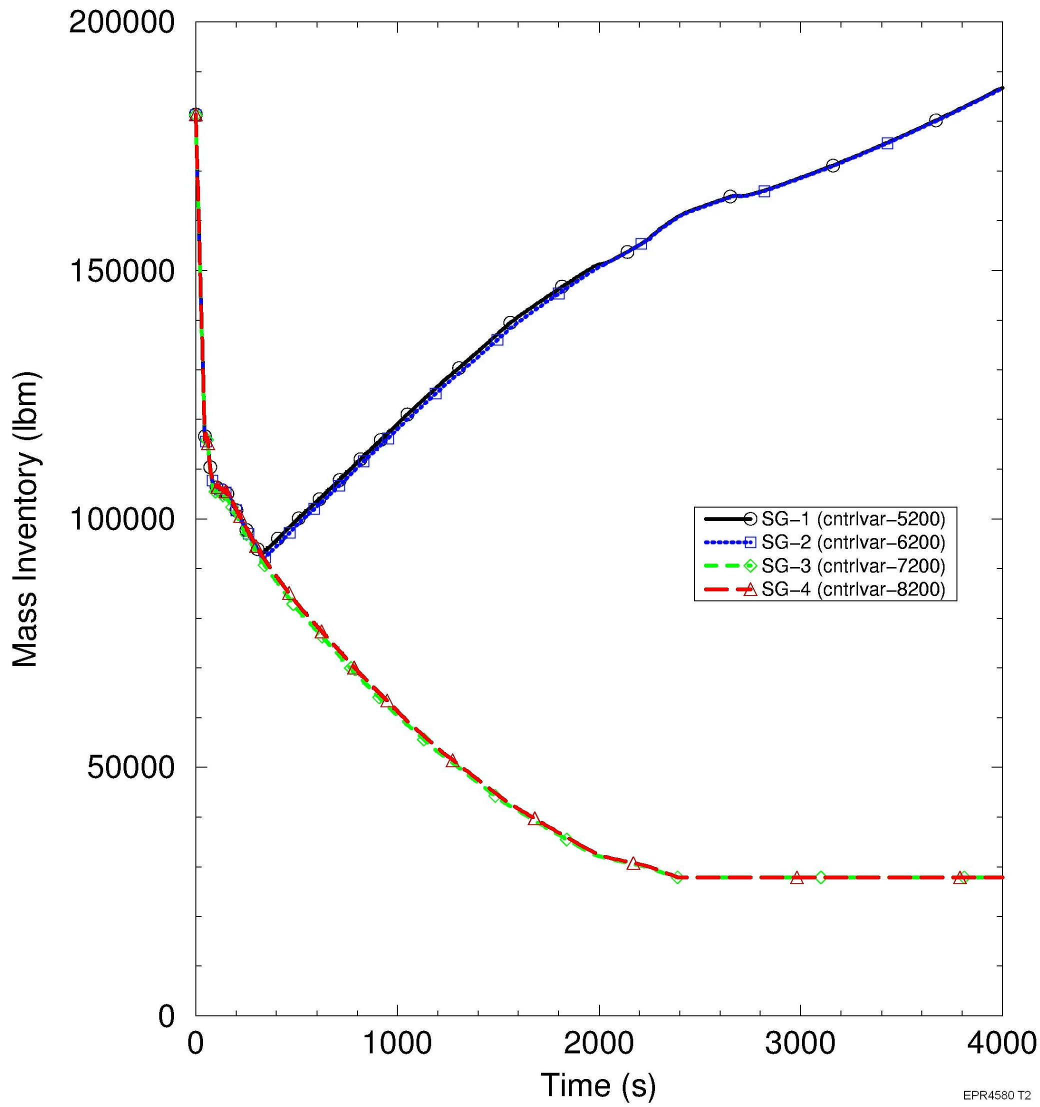

Figure 15.2-43—Loss of Normal Feedwater - SG Liquid Mass Inventory

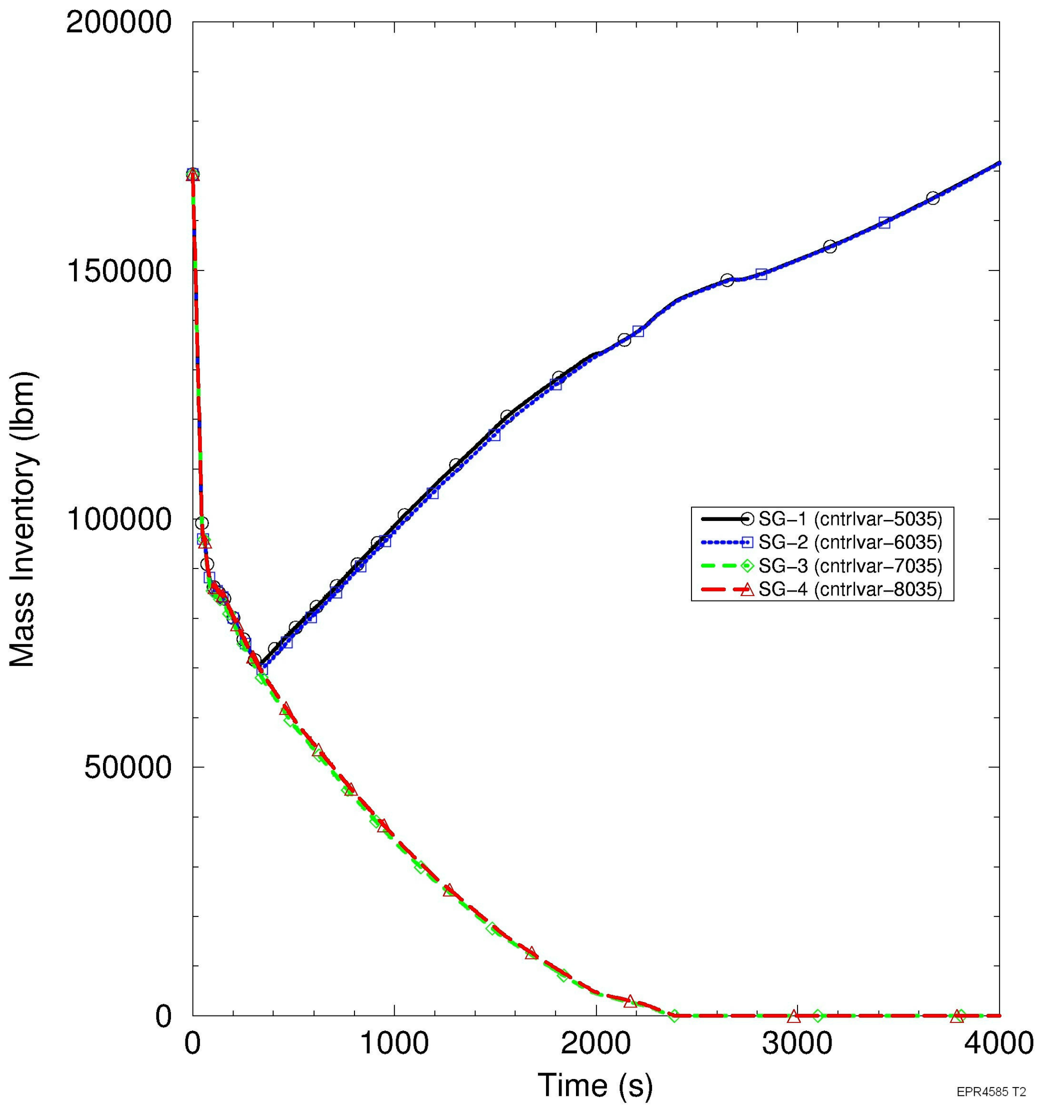

Figure 15.2-44—Loss of Normal Feedwater - Steam and EFW Flow, Unaffected

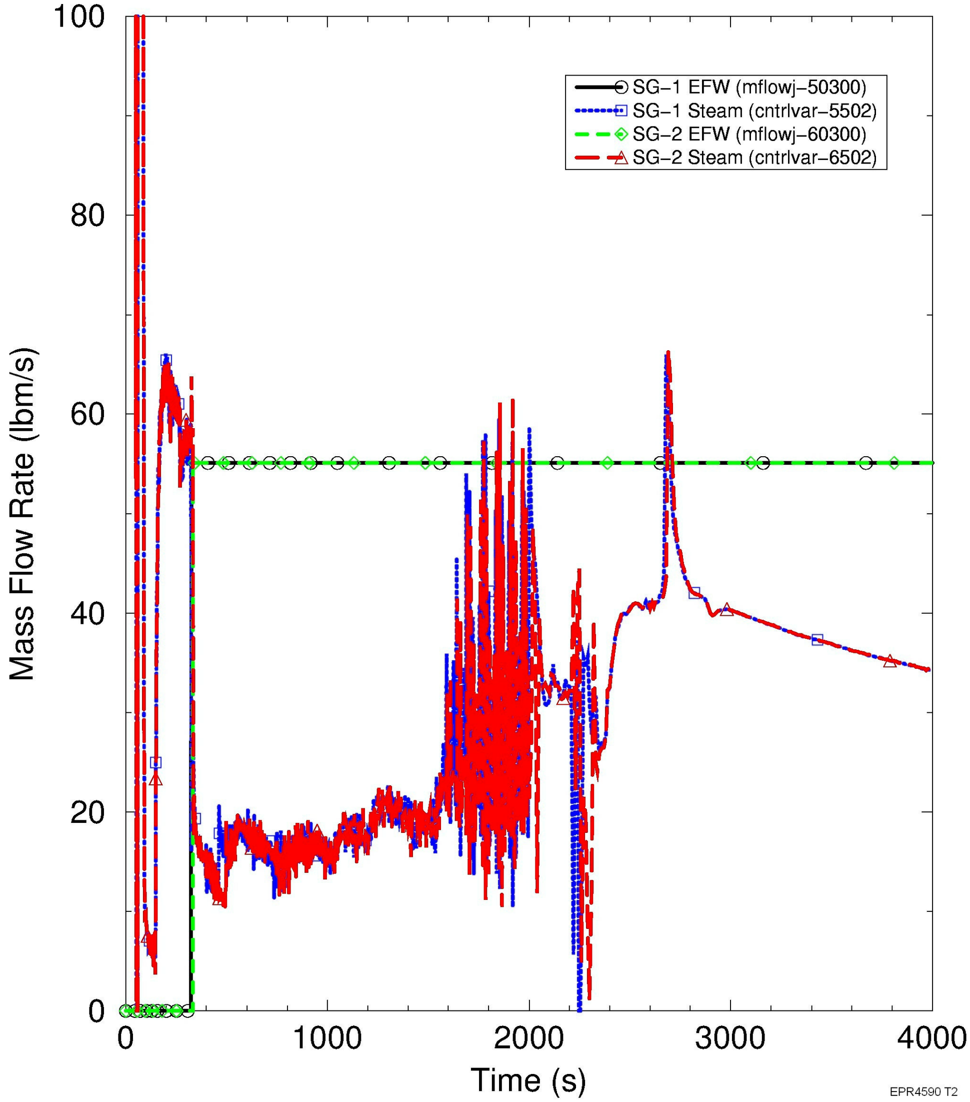

Figure 15.2-45—Loss of Normal Feedwater - Steam and EFW Flow, Affected

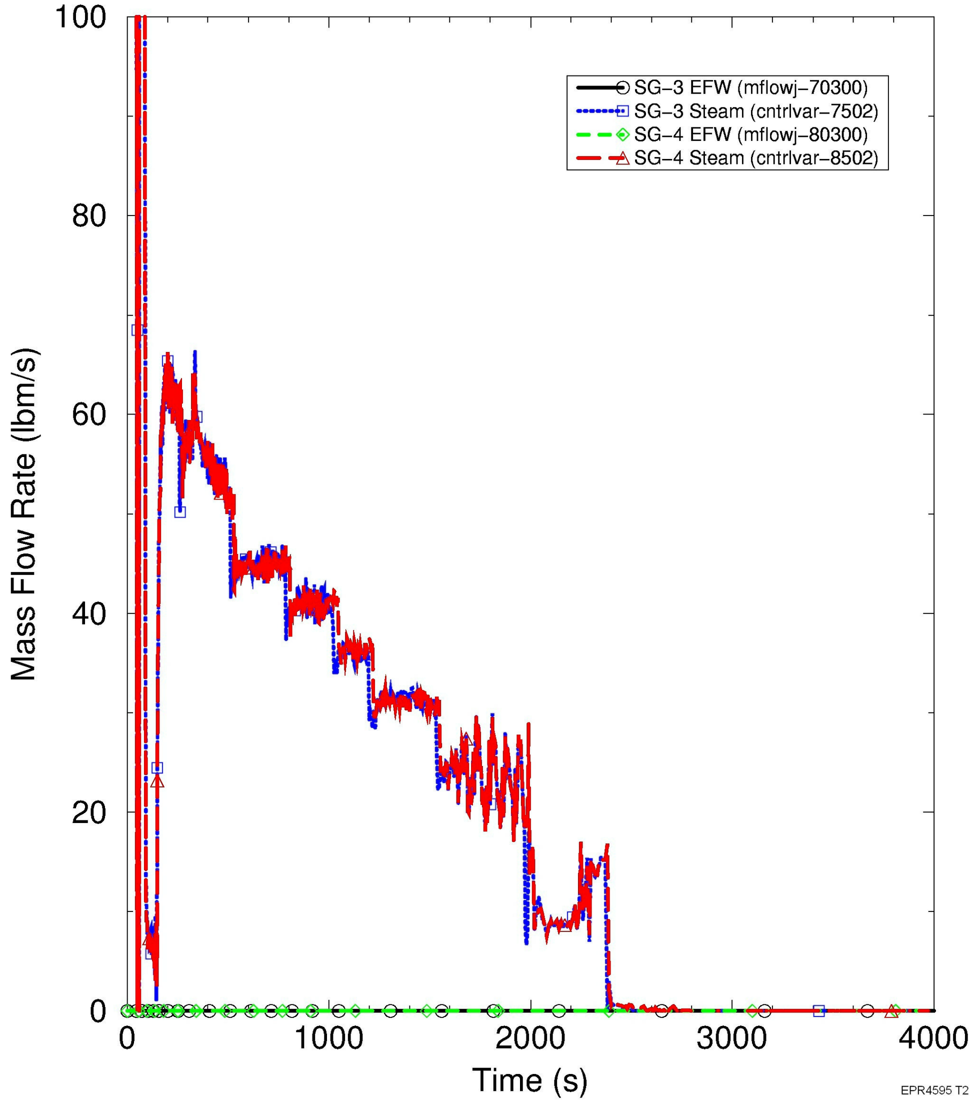

Figure 15.2-46—Loss of Normal Feedwater - MFW Flow

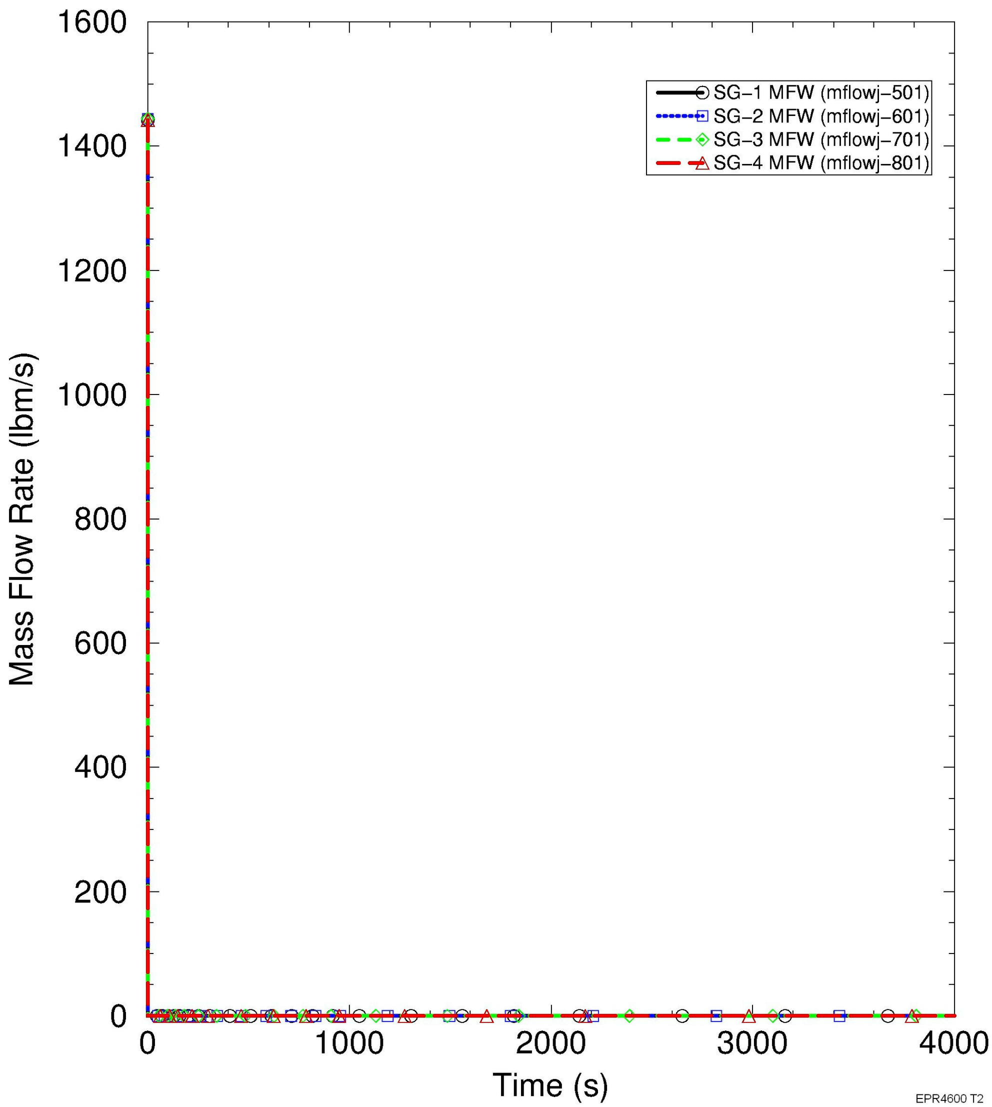

Figure 15.2-47—Loss of Normal Feedwater - Blowdown Flow

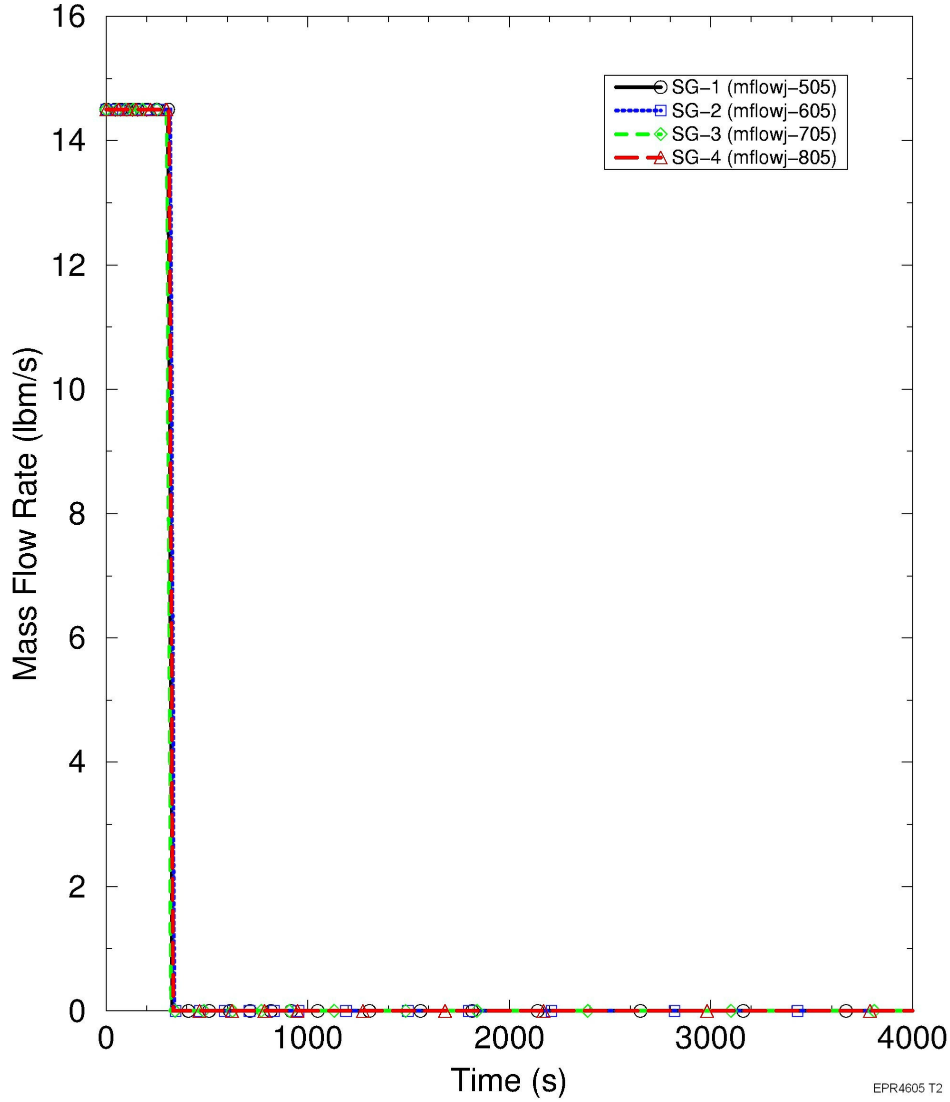

Figure 15.2-48—Loss of Normal Feedwater - Net Heat Addition to RCS

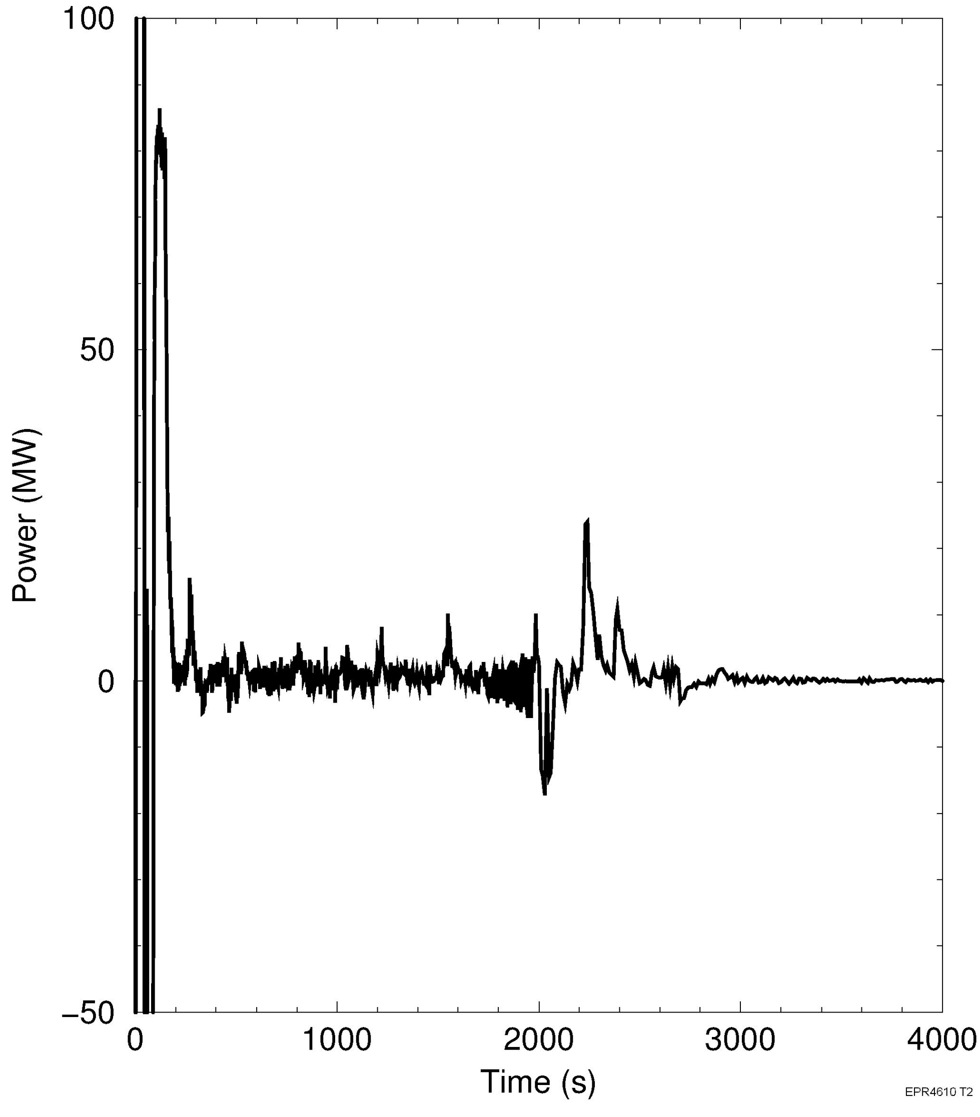

Figure 15.2-49—FWLB Representative Small Break - Reactor and Total Steam Generator Power

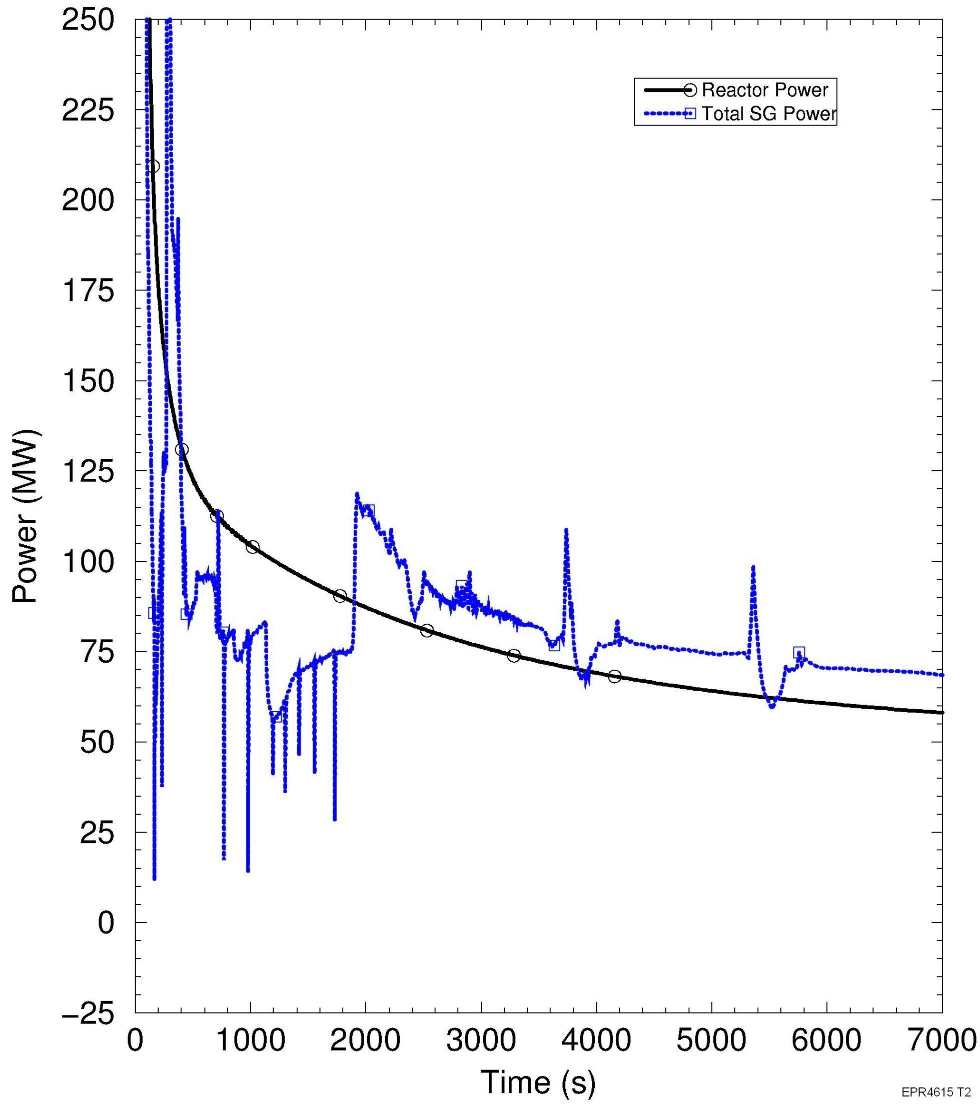

Figure 15.2-50—FWLB Representative Small Break - Pressurizer Pressure

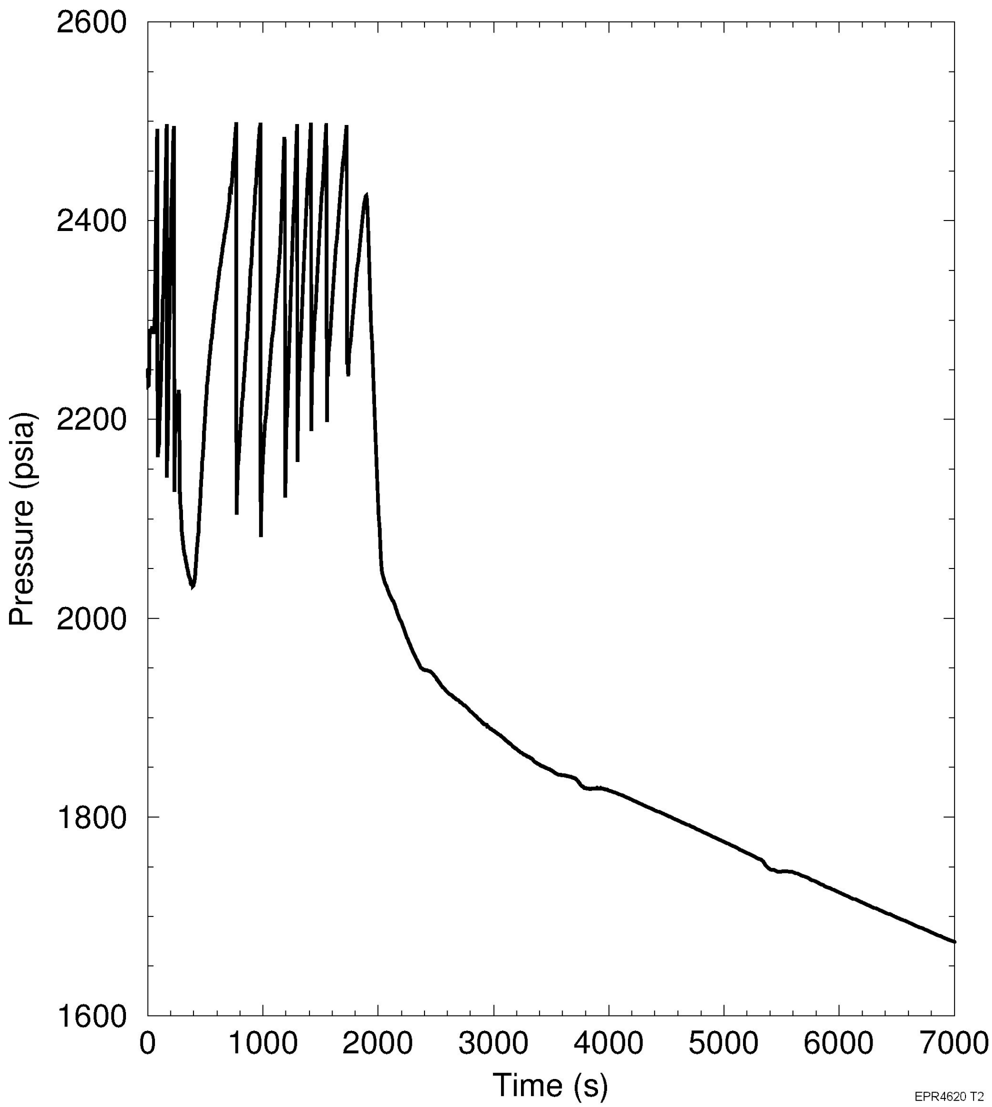

Figure 15.2-51—FWLB Representative Small Break – Pressurizer Spray Flow

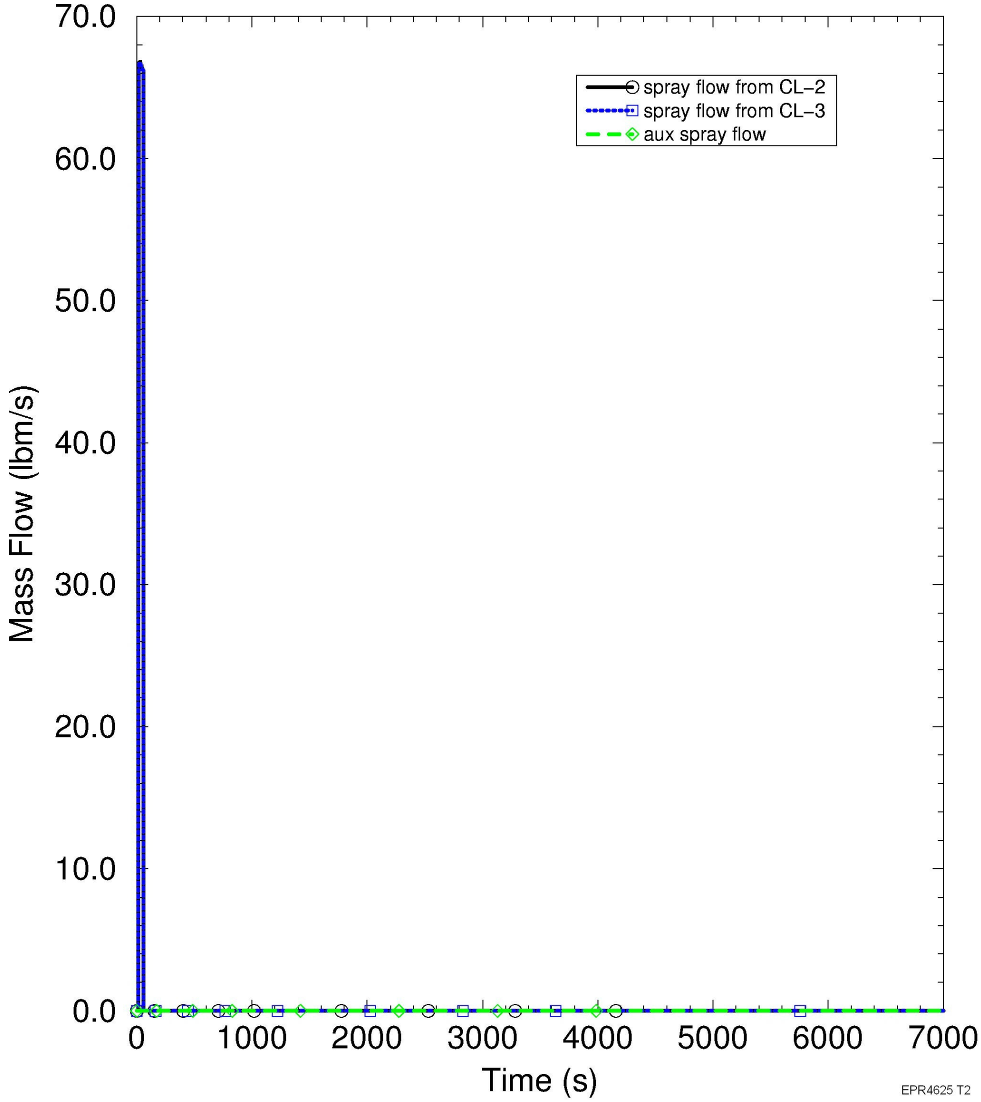

Figure 15.2-52—FWLB Representative Small Break – PSRV Flow

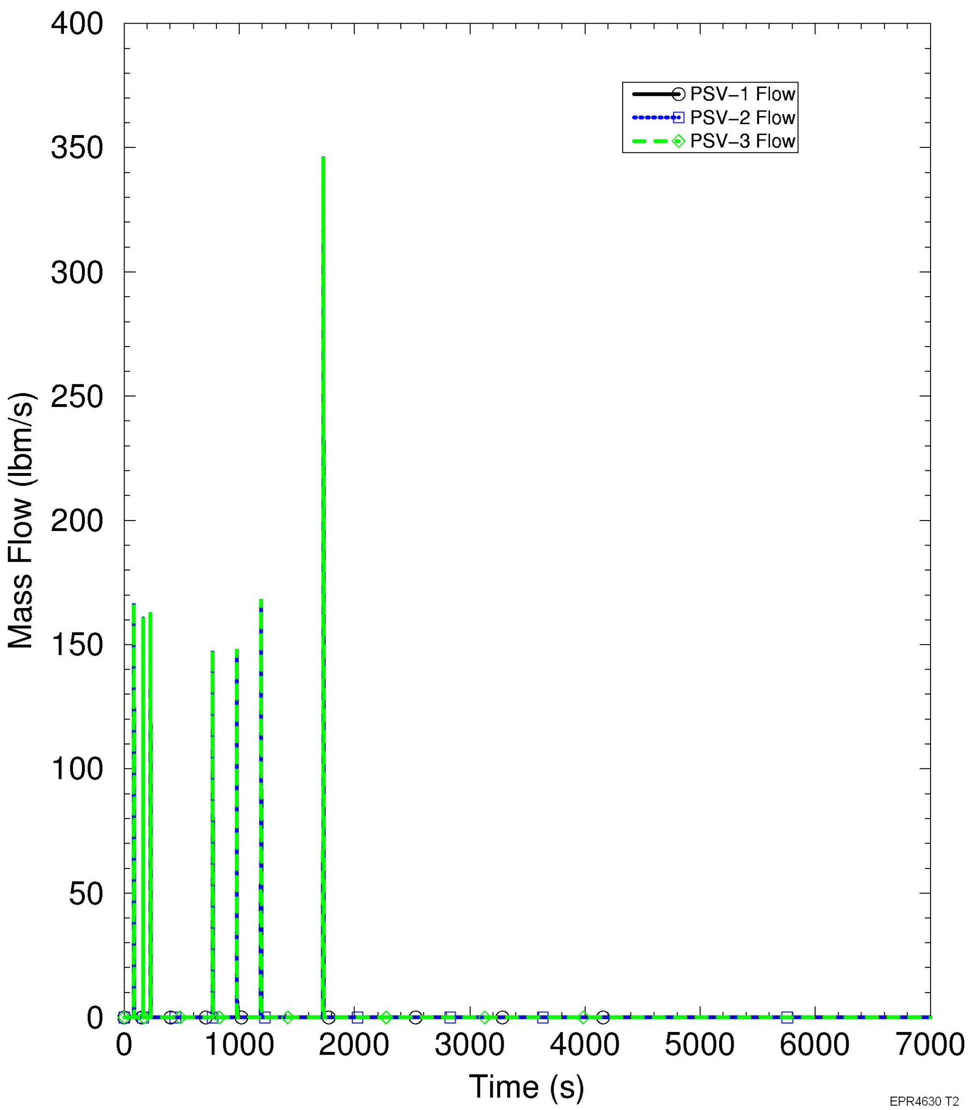

Figure 15.2-53—FWLB Representative Small Break – Pressurizer Liquid Level

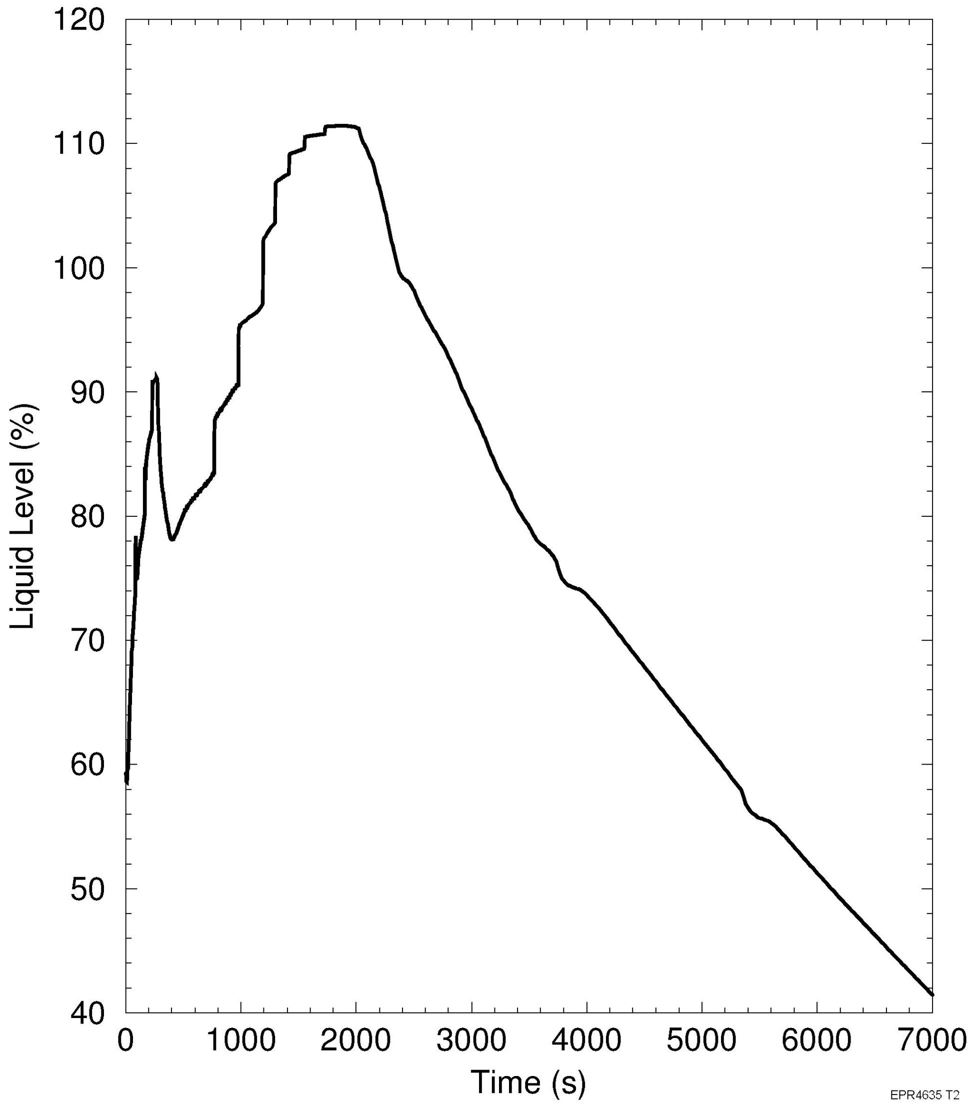

Figure 15.2-54—FWLB Representative Small Break – RCS Cold Leg Temperatures

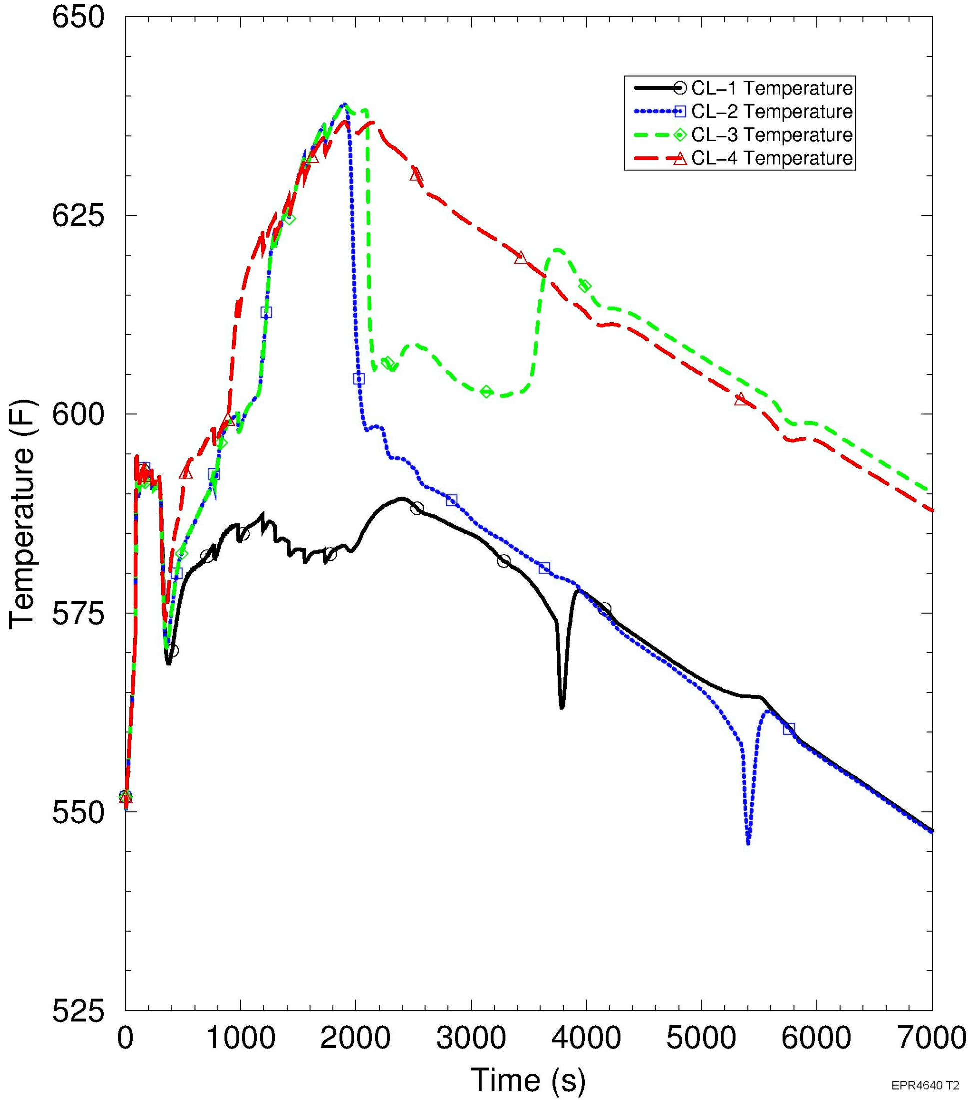

Figure 15.2-55—FWLB Representative Small Break – RCS Hot Leg Temperatures

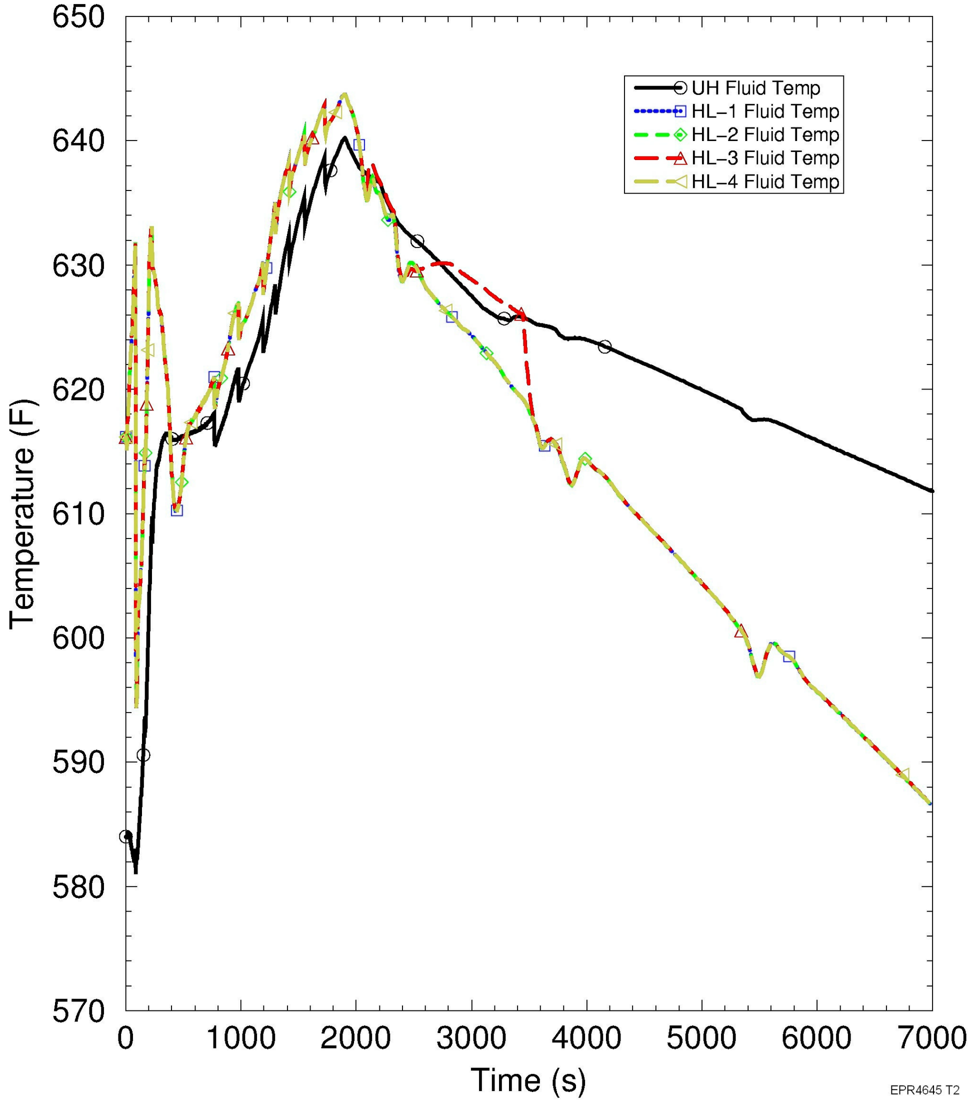

Figure 15.2-56—FWLB Representative Small Break – Core Exit Subcooling

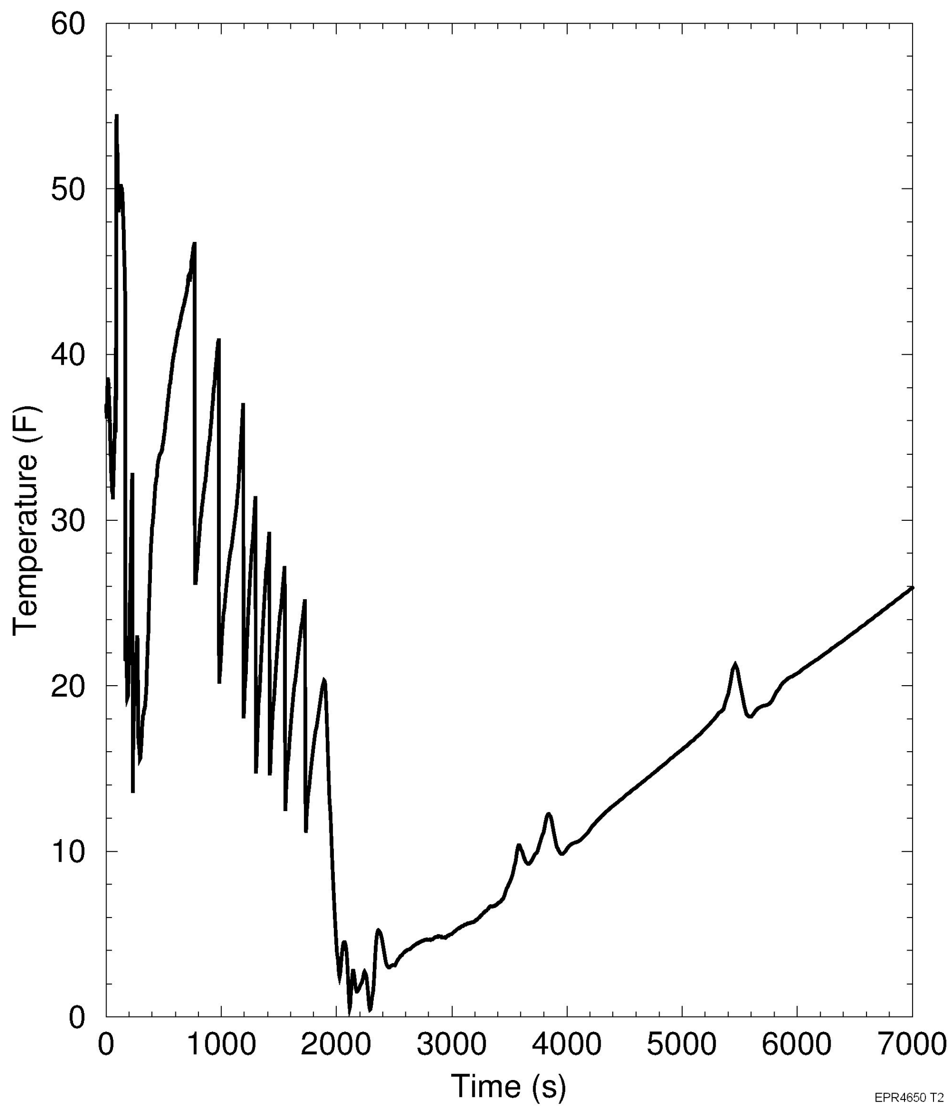

Figure 15.2-57—FWLB Representative Small Break – Core Flow

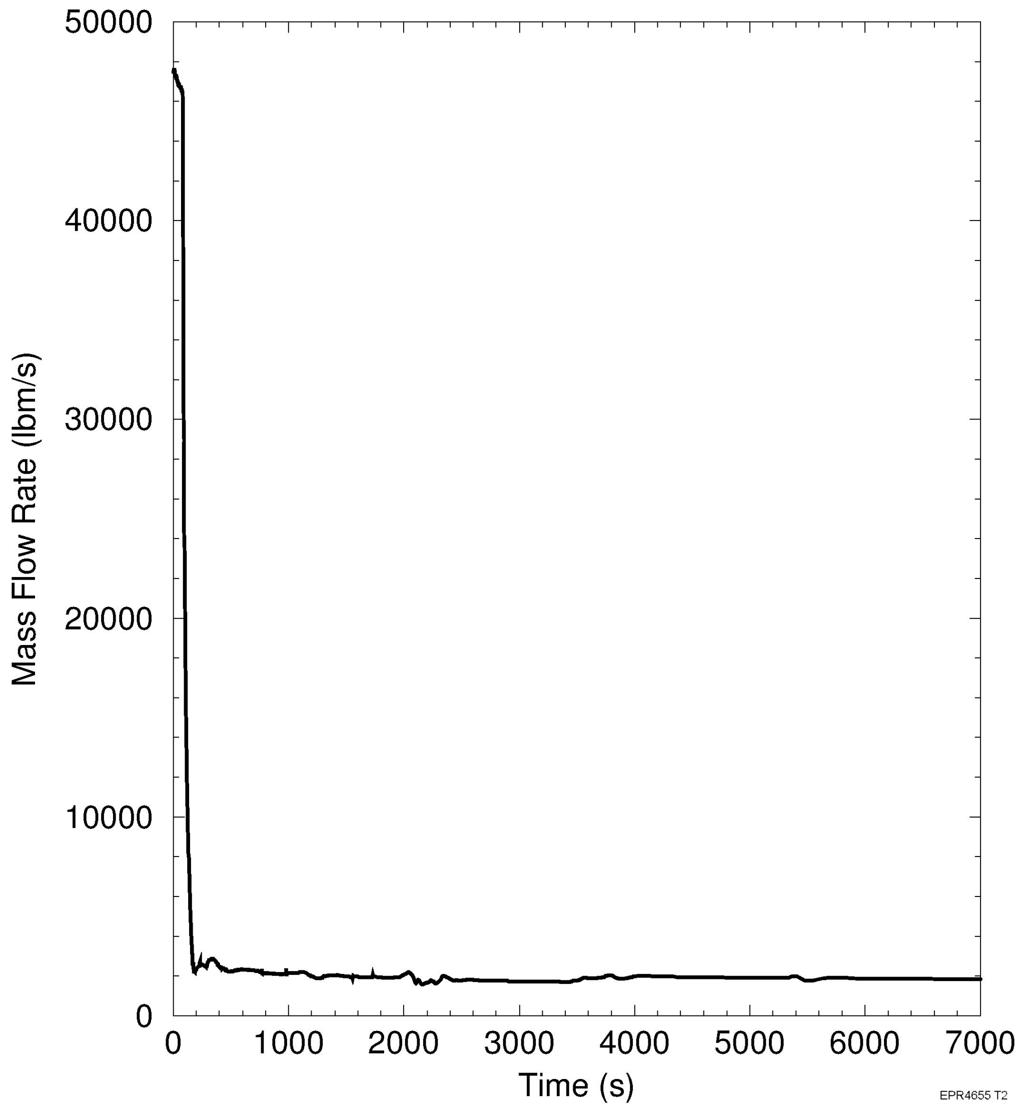

Figure 15.2-58—FWLB Representative Small Break – Break Flow

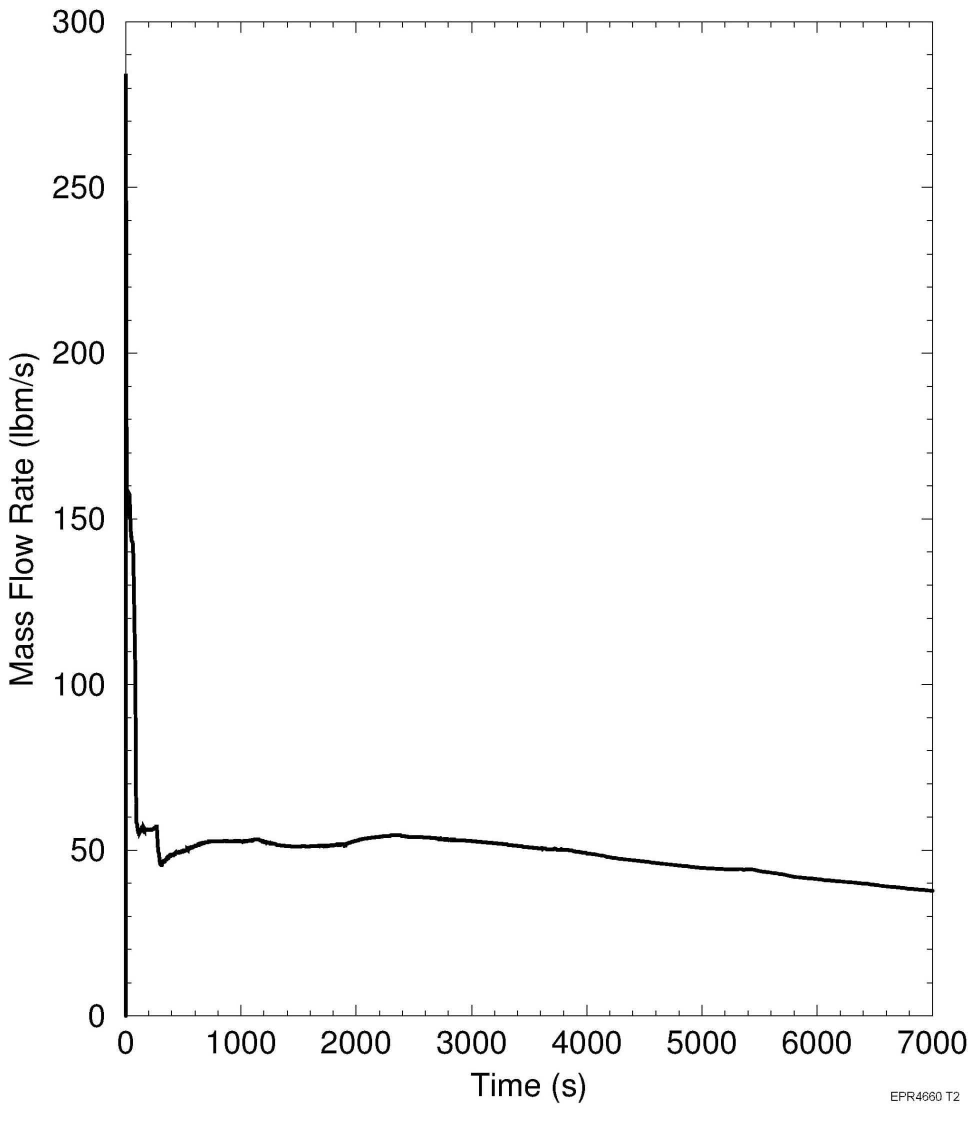

Figure 15.2-59—FWLB Representative Small Break – Main Steam Relief Loops 1 and 2

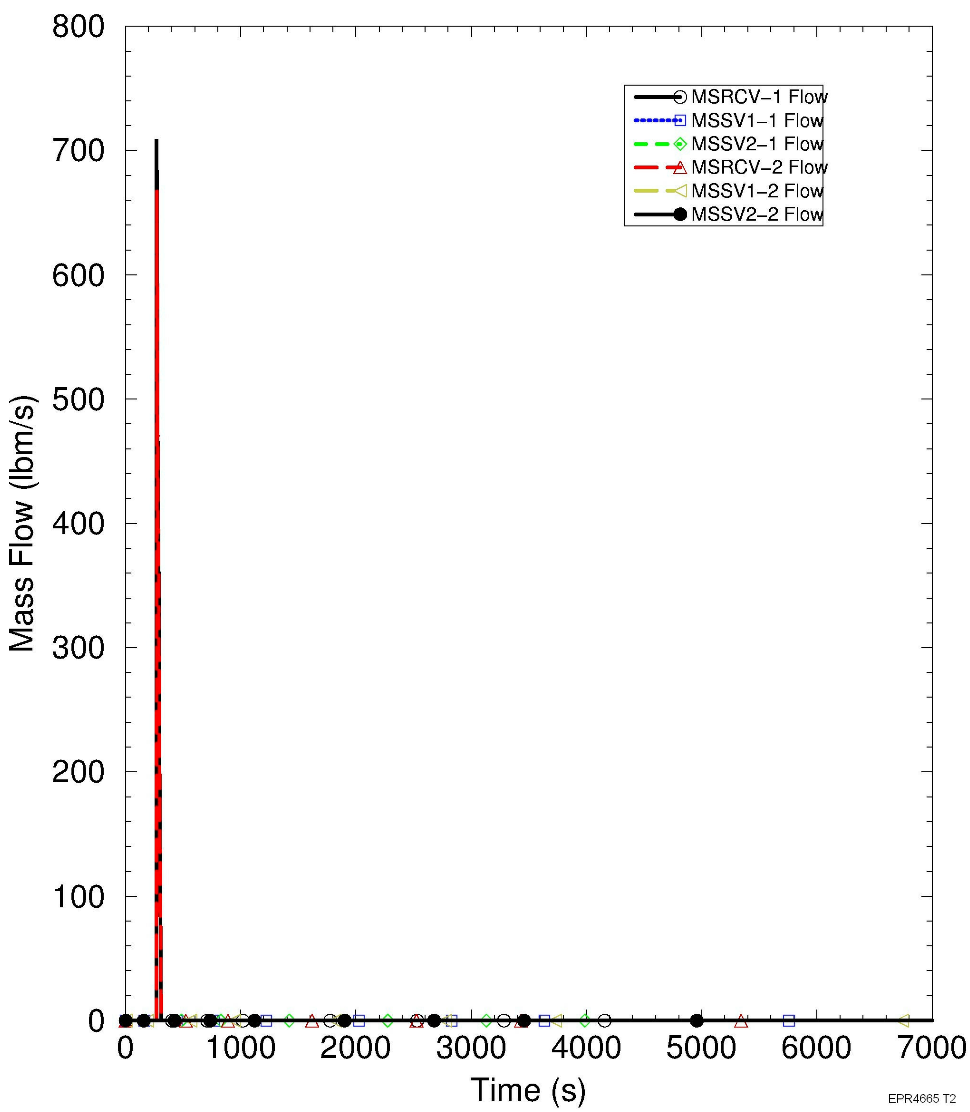

Figure 15.2-60—FWLB Representative Small Break – Main Steam Relief Loops 3 and 4

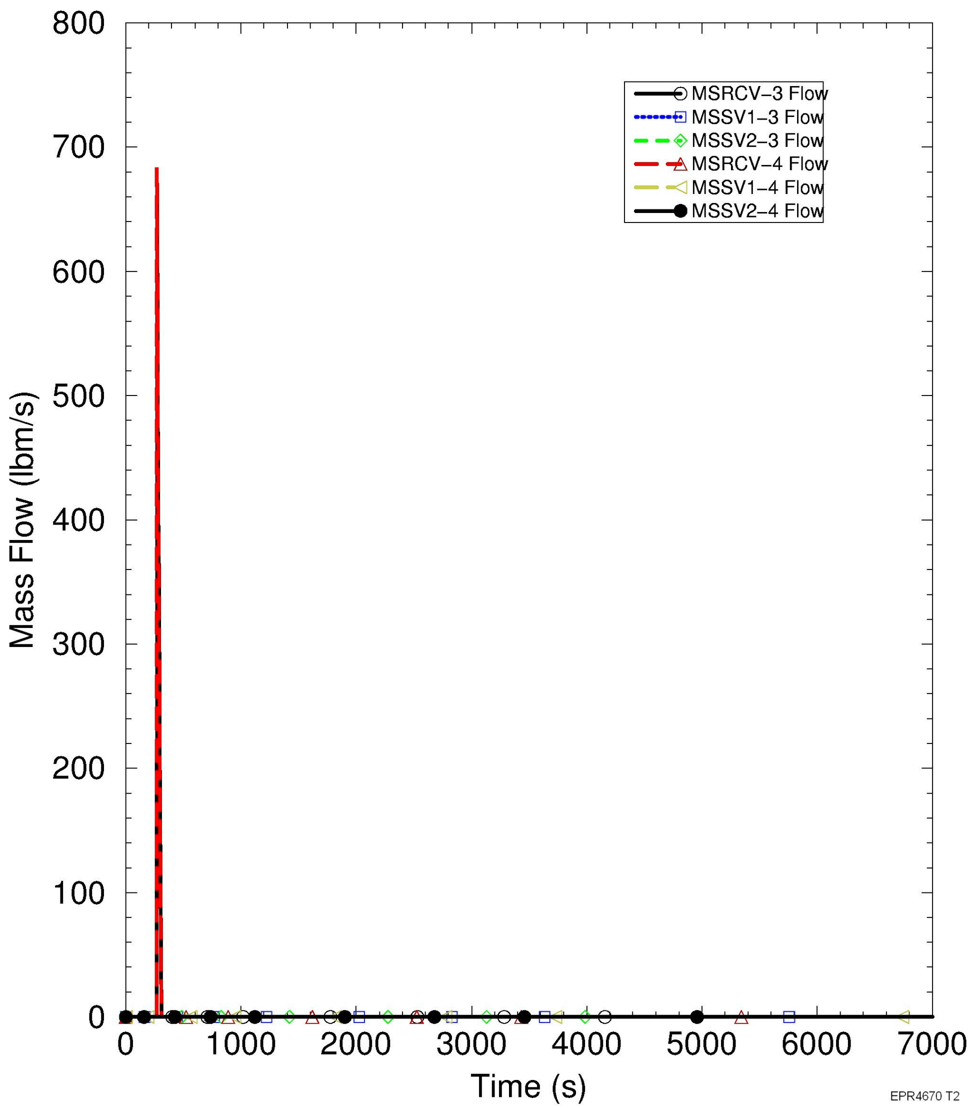

Figure 15.2-61—FWLB Representative Small Break – Steam Generator Dome Pressure

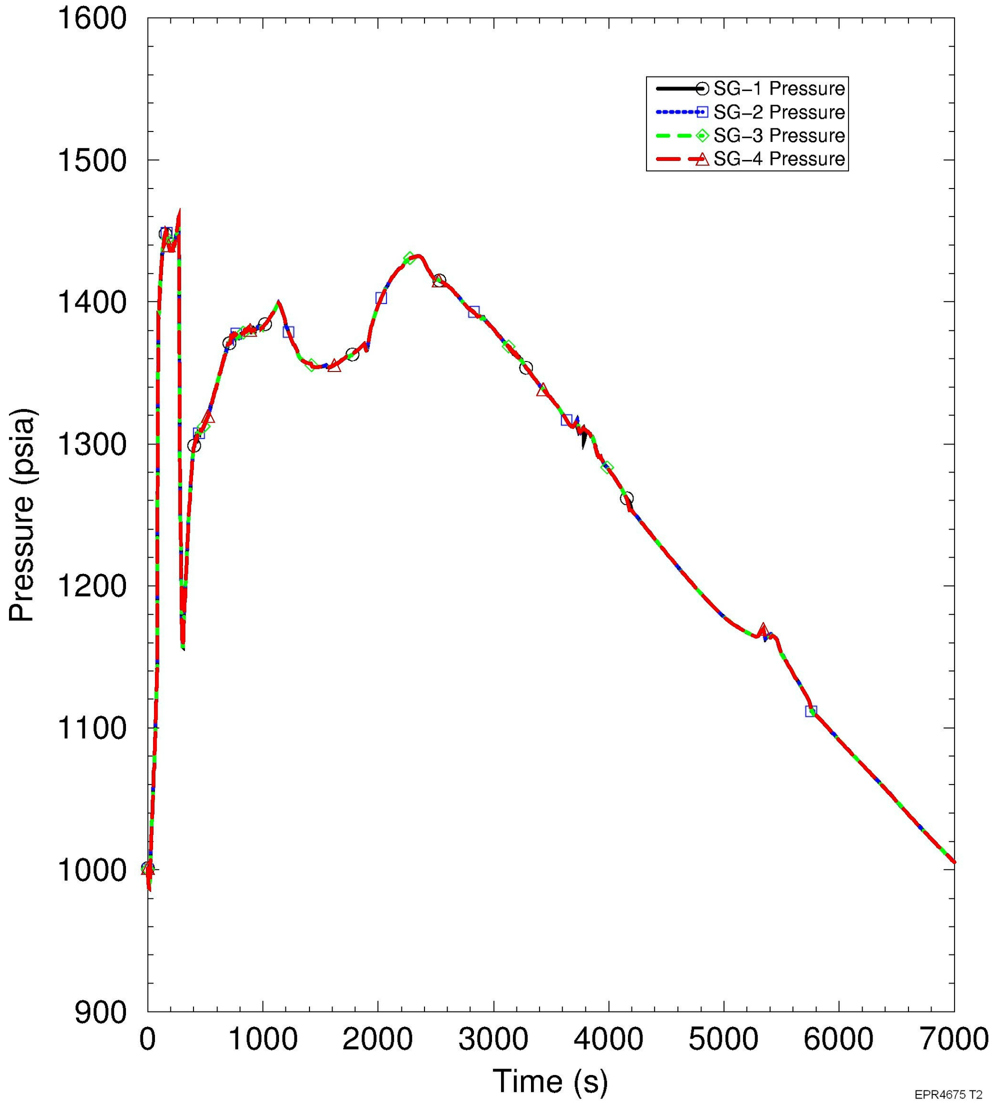

Figure 15.2-62—FWLB Representative Small Break – Steam Generator Total Mass

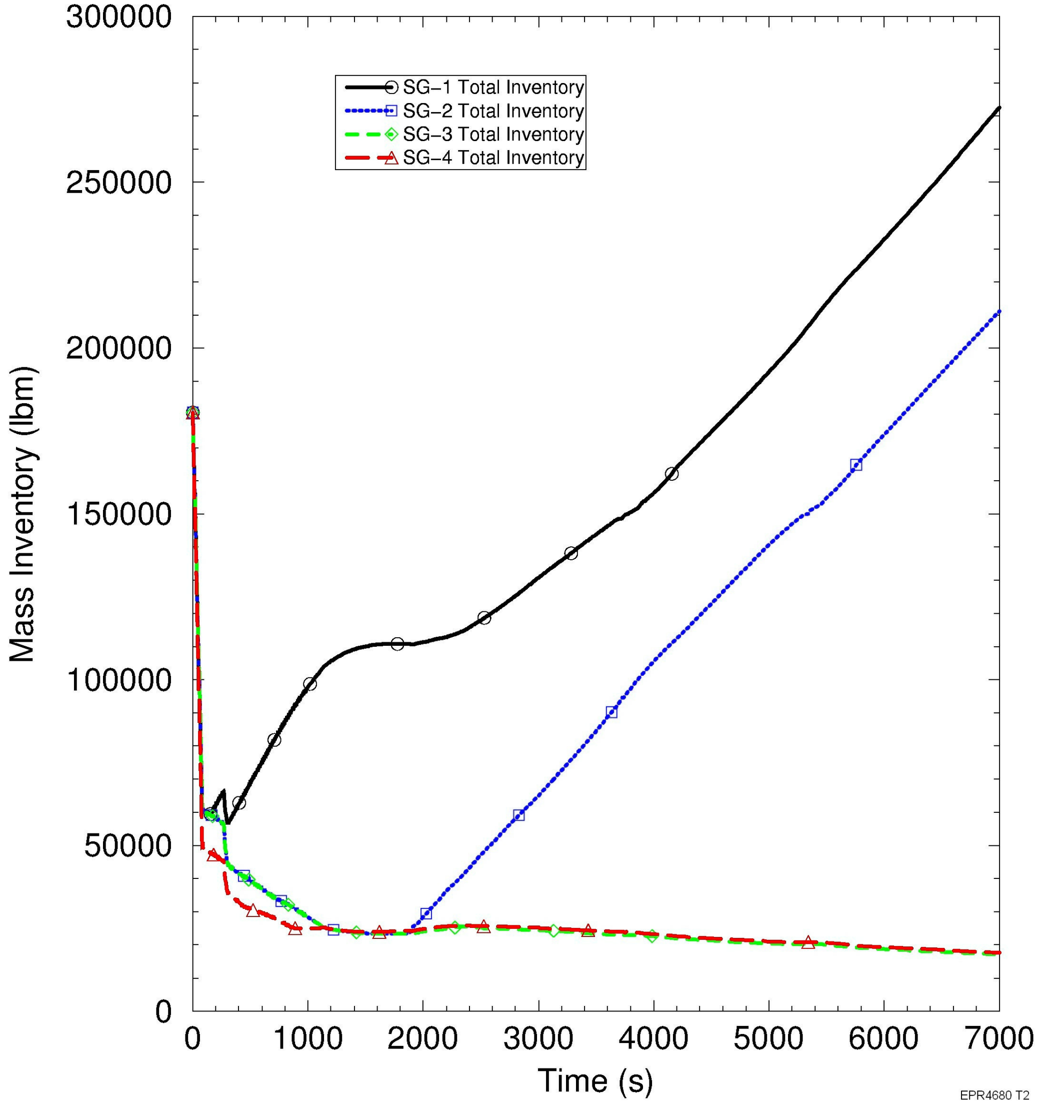

Figure 15.2-63—FWLB Representative Small Break – Steam Generator Liquid Mass

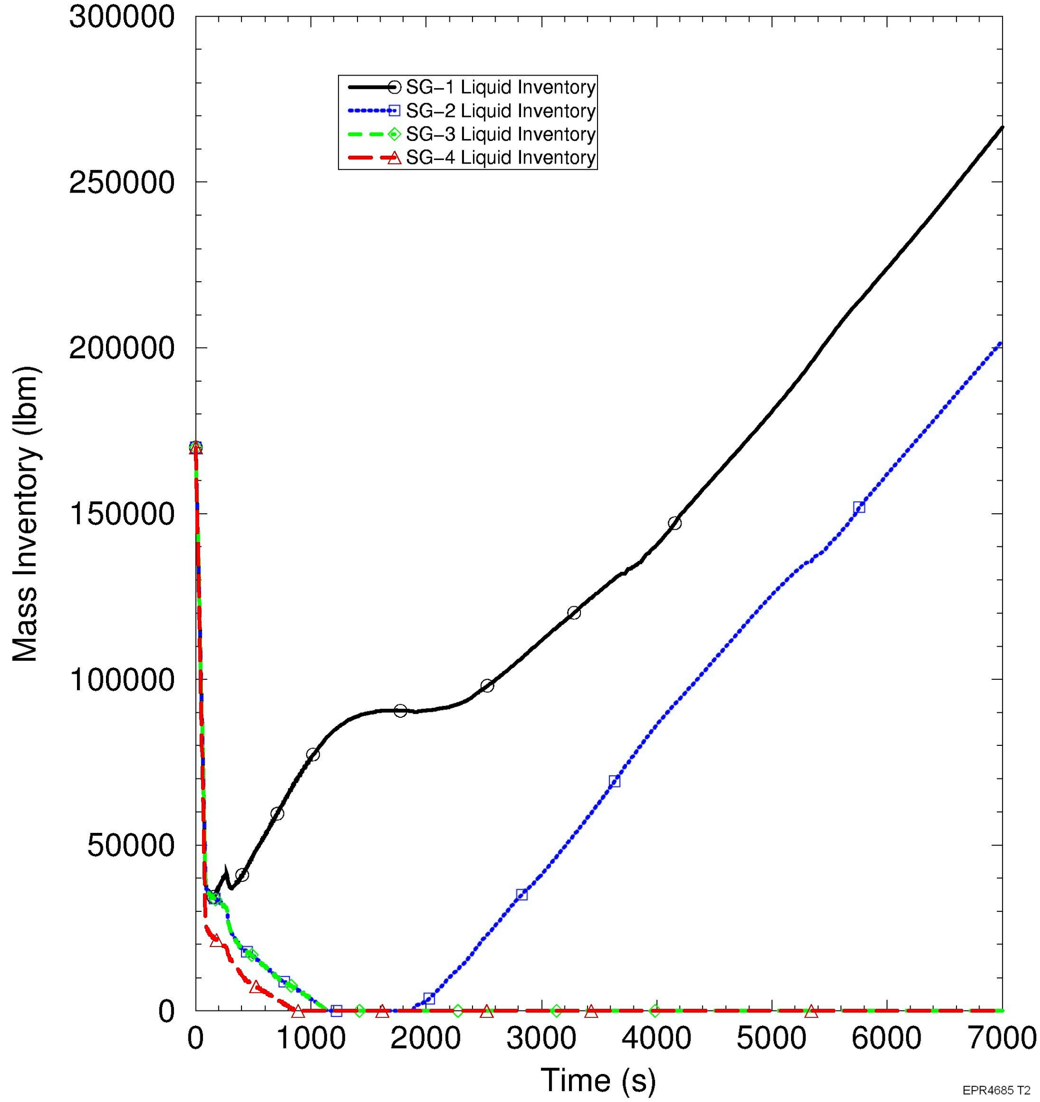

Figure 15.2-64—FWLB Representative Small Break – Net Heat Addition to RCS

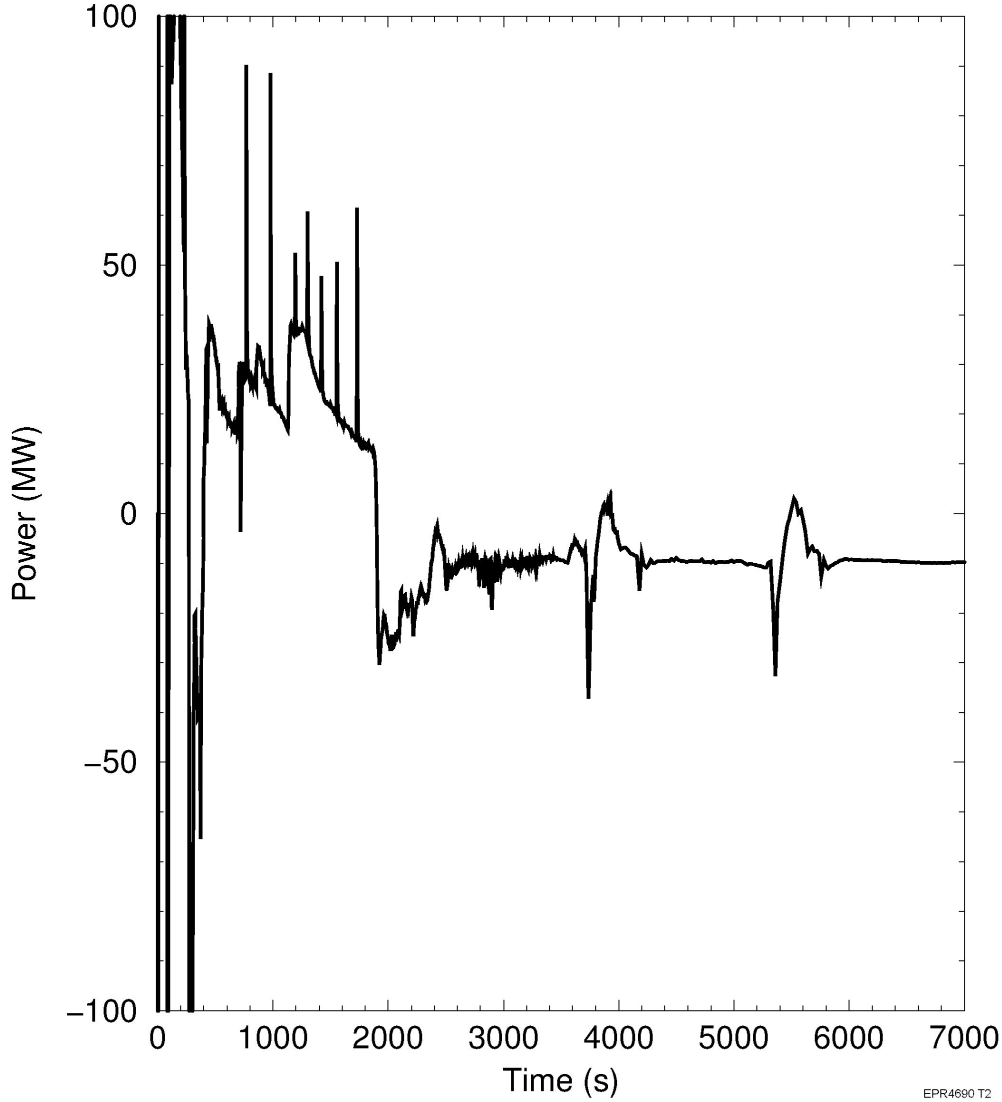

Figure 15.2-65—FWLB Representative Small Break – RCS Maximum Pressure

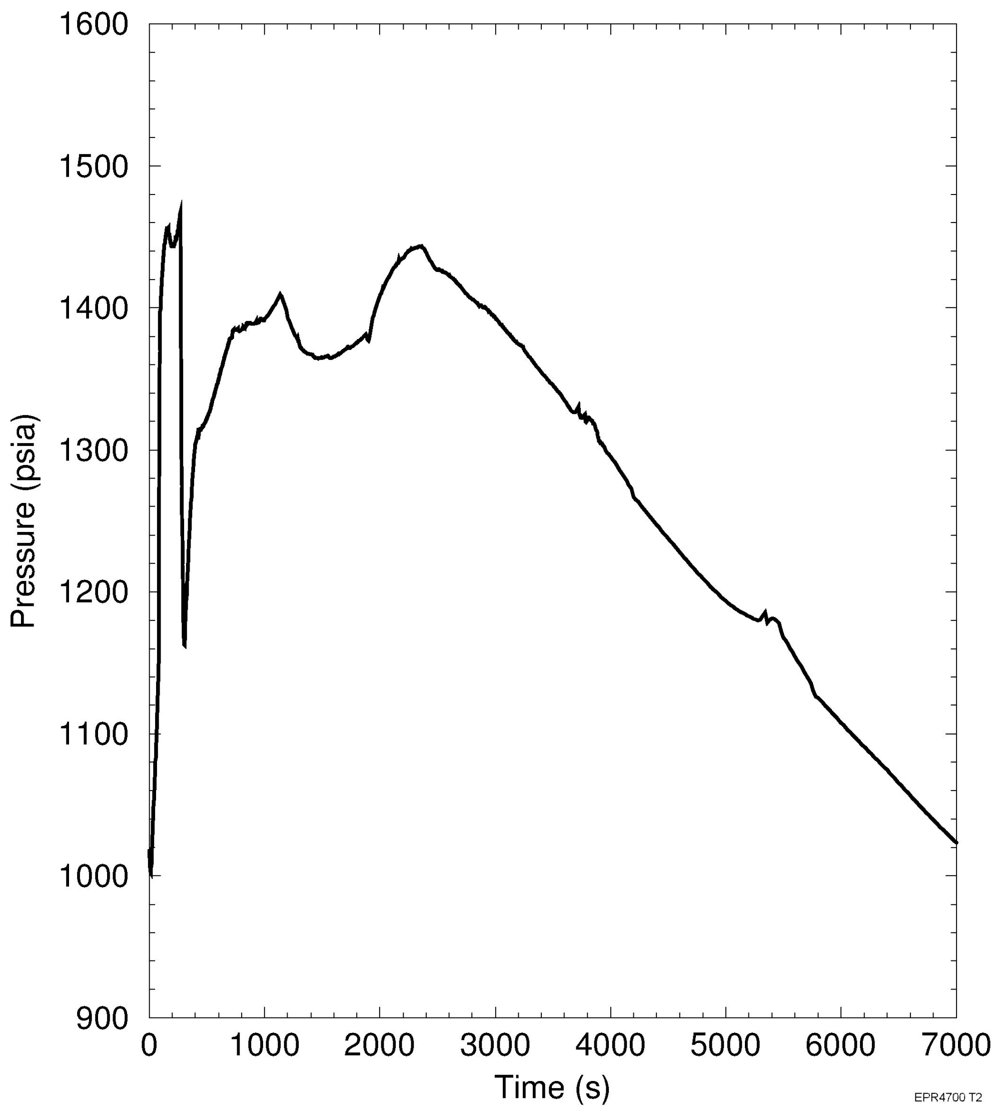

Figure 15.2-66—FWLB Representative Small Break – Steam Generator Maximum Pressure

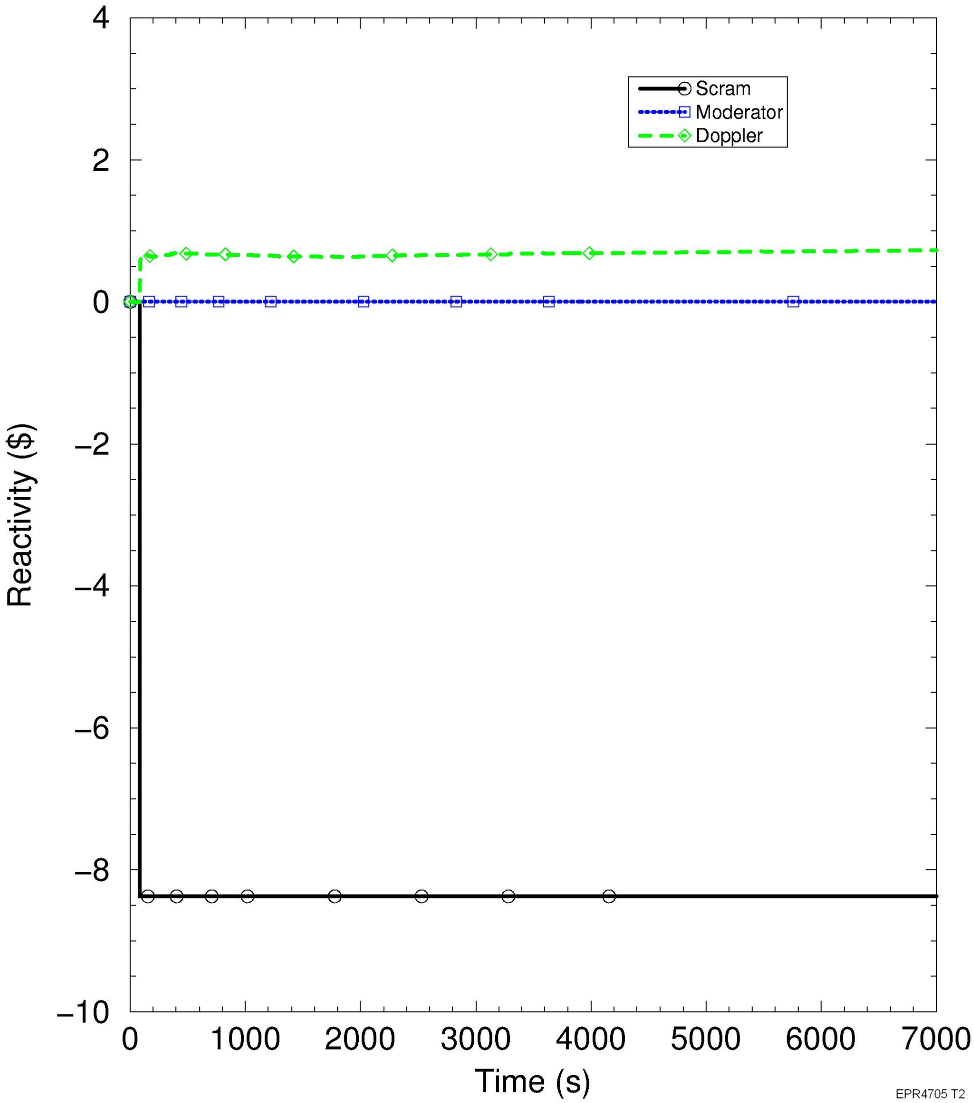

Figure 15.2-67—FWLB Representative Small Break – Reactivities

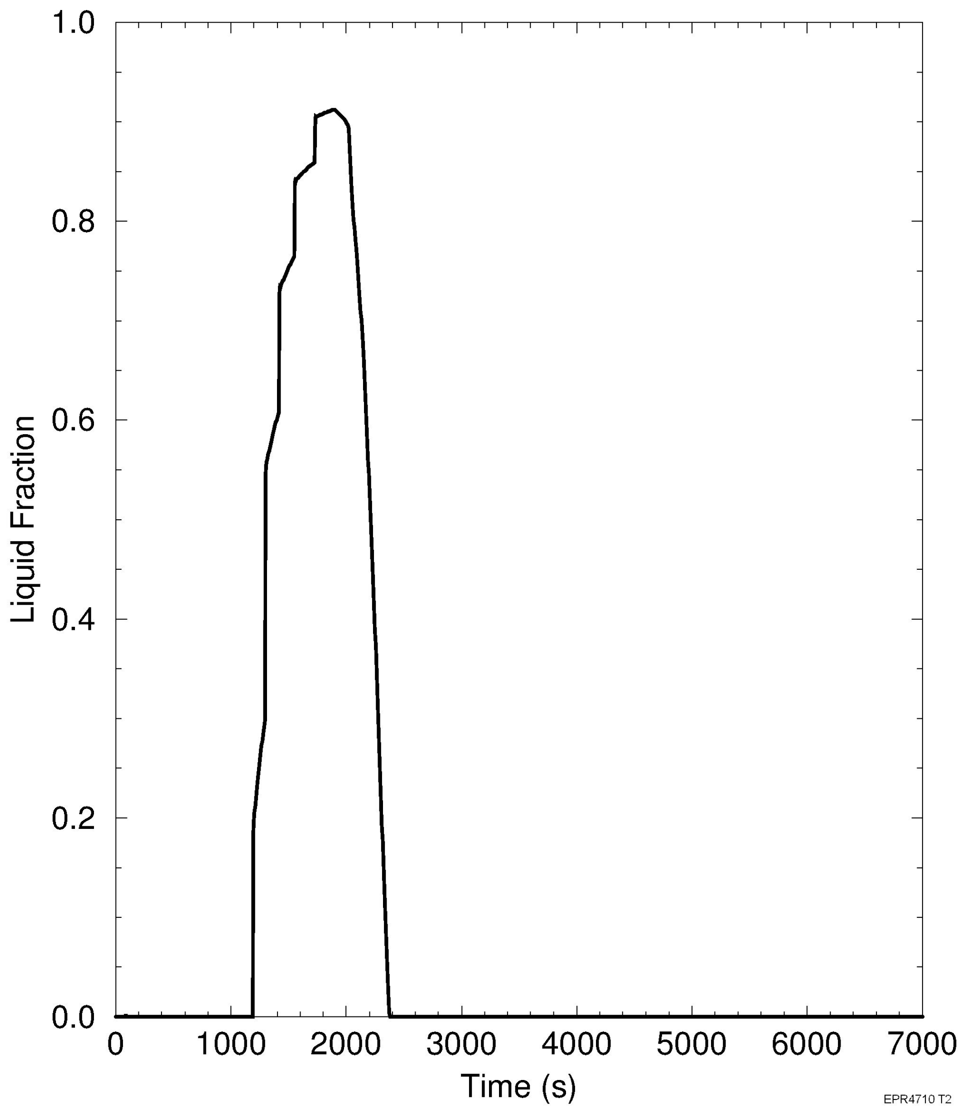

Figure 15.2-68—FWLB Representative Small Break – Liquid Volume Fraction in Pressurizer Dome

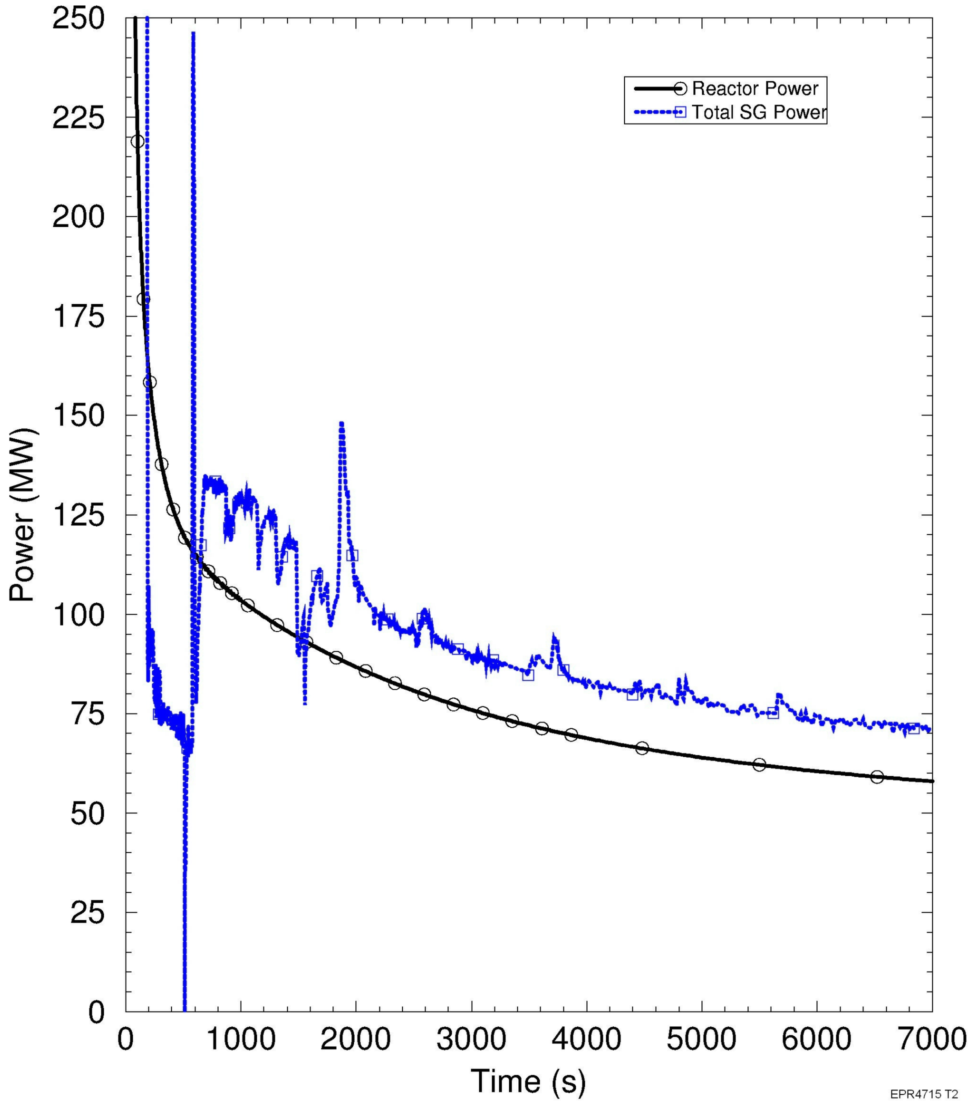

Figure 15.2-69—FWLB Maximum RCS Pressure Case – Reactor and Total Steam Generator Power

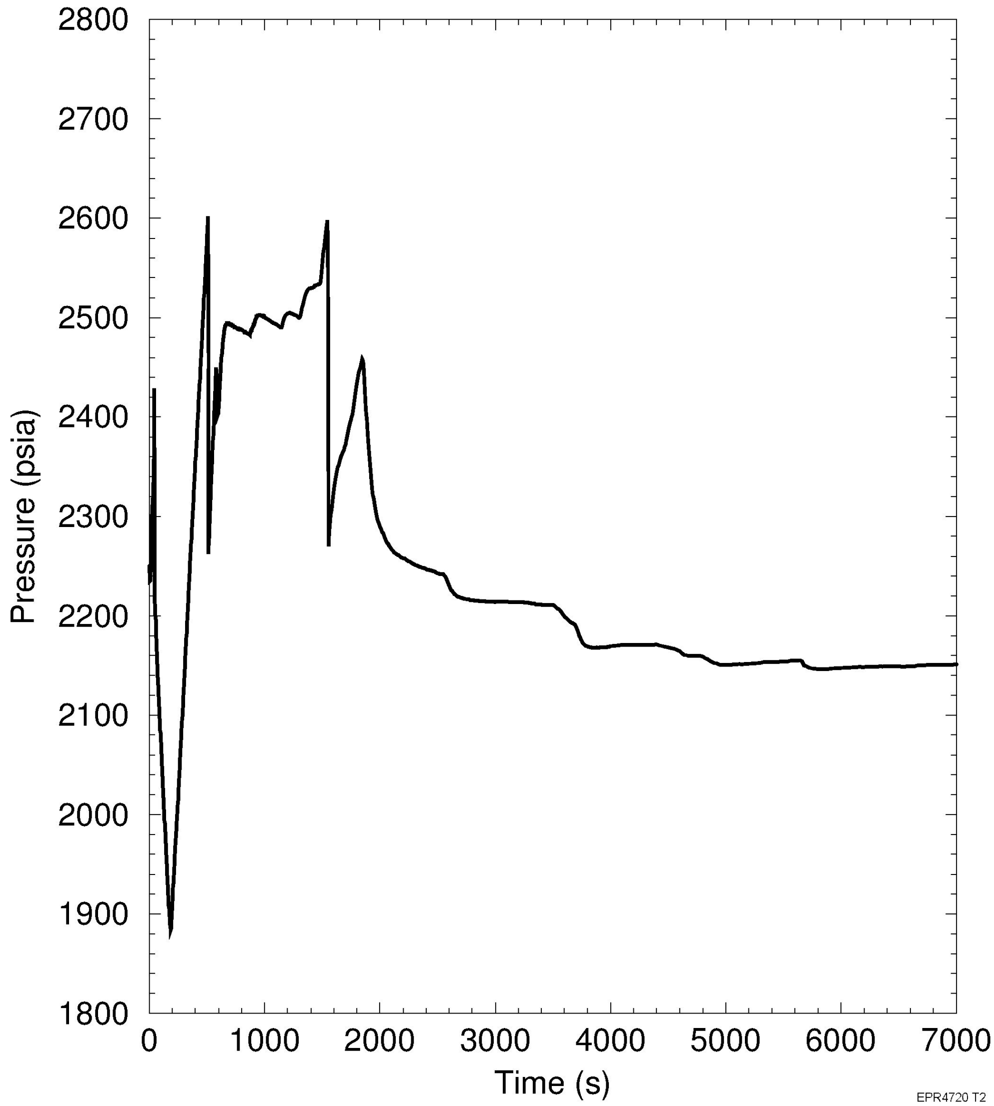

Figure 15.2-70—FWLB Maximum RCS Pressure Case - Pressurizer Pressure

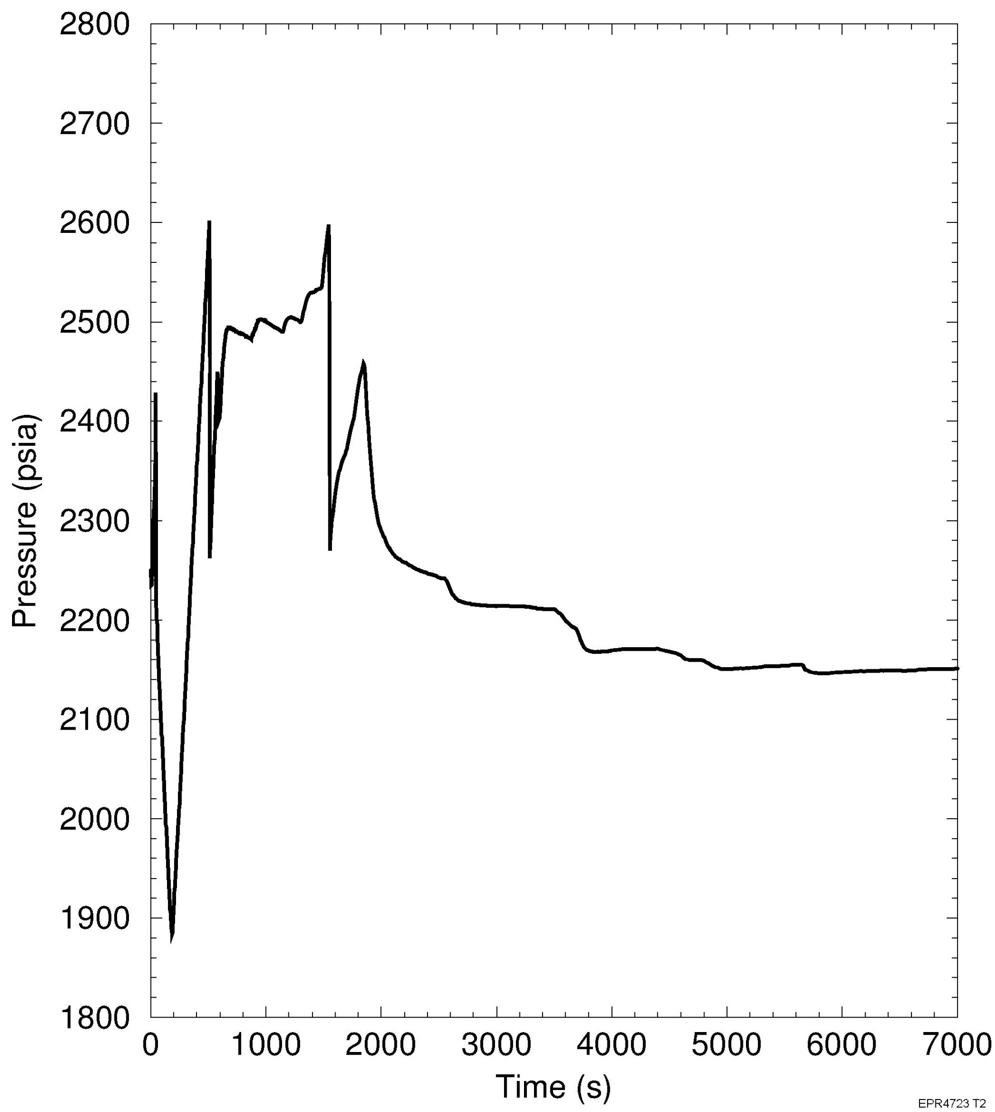

Figure 15.2-71—FWLB Maximum RCS Pressure Case – PSRV Flow

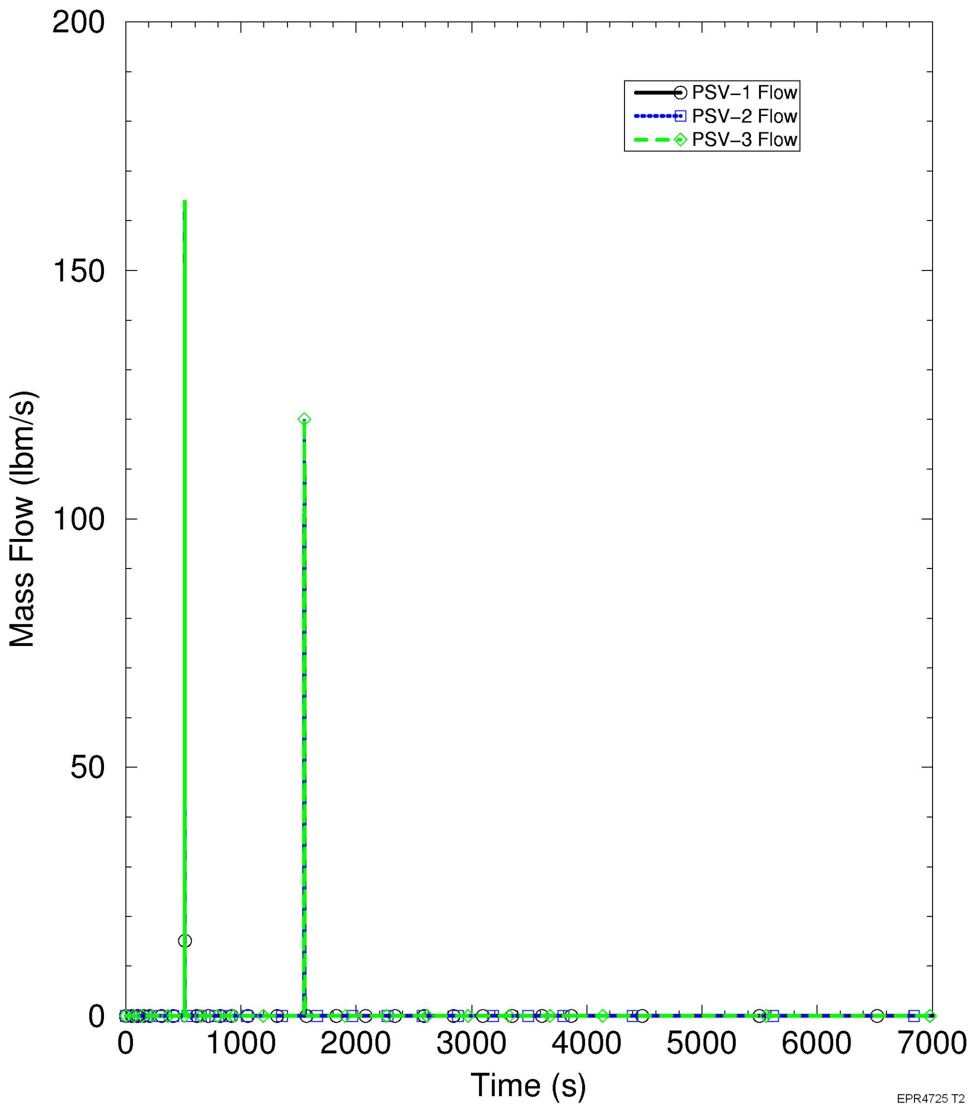

Figure 15.2-72—FWLB Maximum RCS Pressure Case – Pressurizer Liquid Level

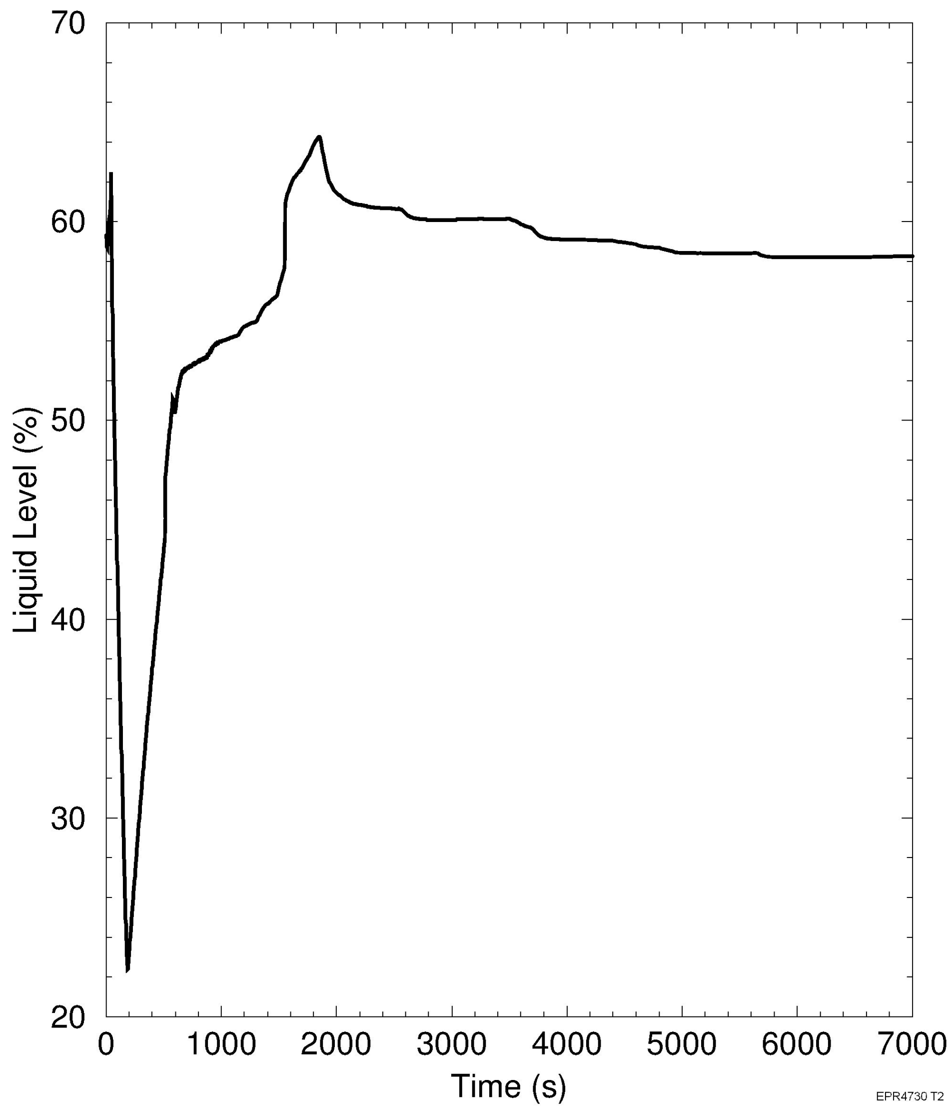

Figure 15.2-73—FWLB Maximum RCS Pressure Case – RCS Cold Leg Temperatures

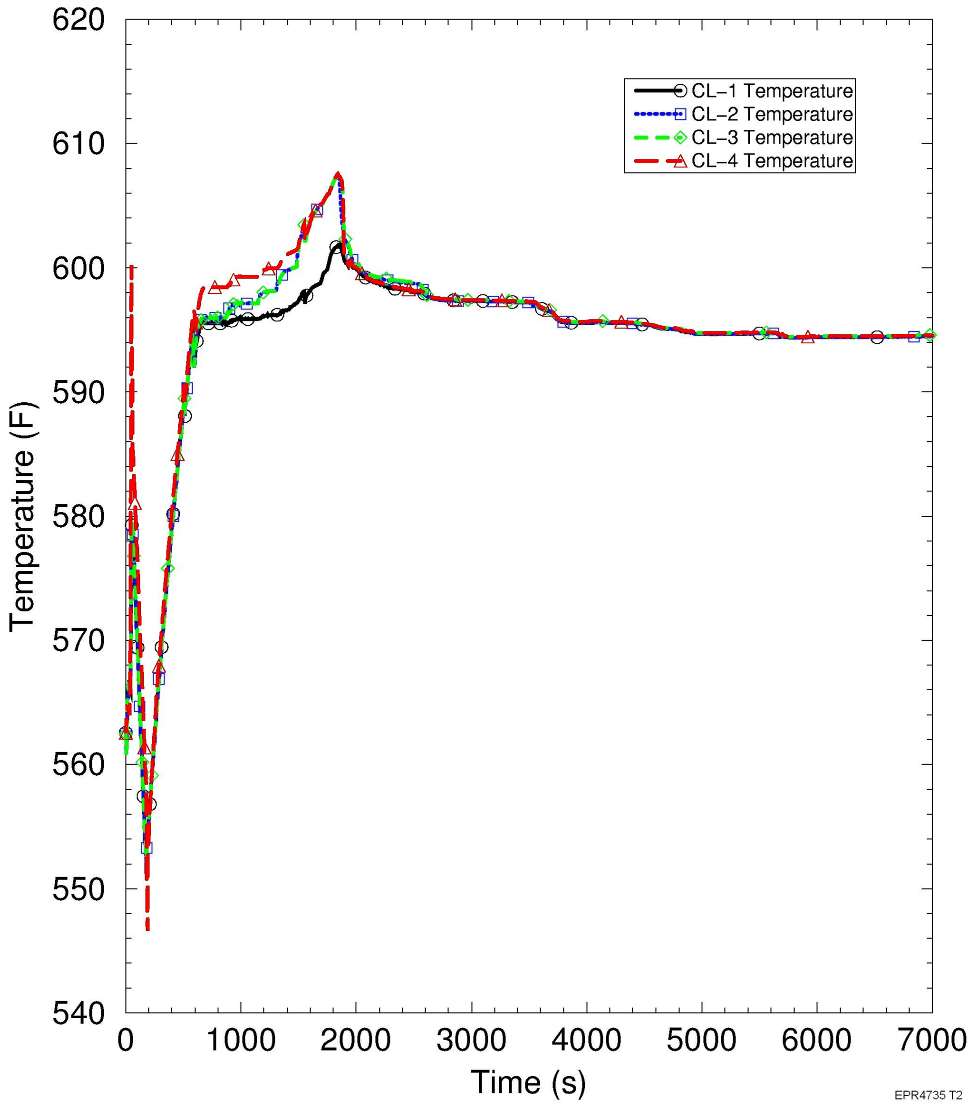

Figure 15.2-74—FWLB Maximum RCS Pressure Case – RCS Hot Leg and Upper Head Temperatures

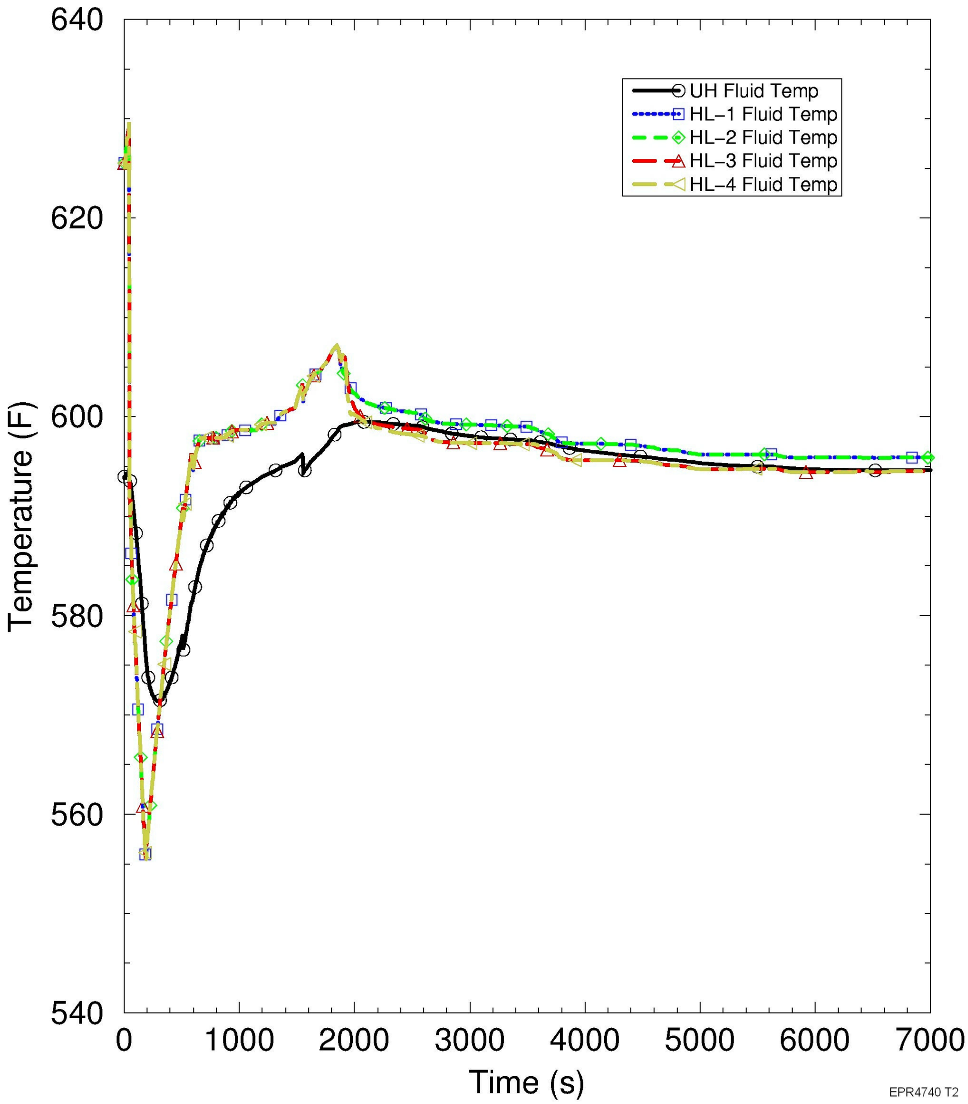

Figure 15.2-75—FWLB Maximum RCS Pressure Case – Core Exit Subcooling

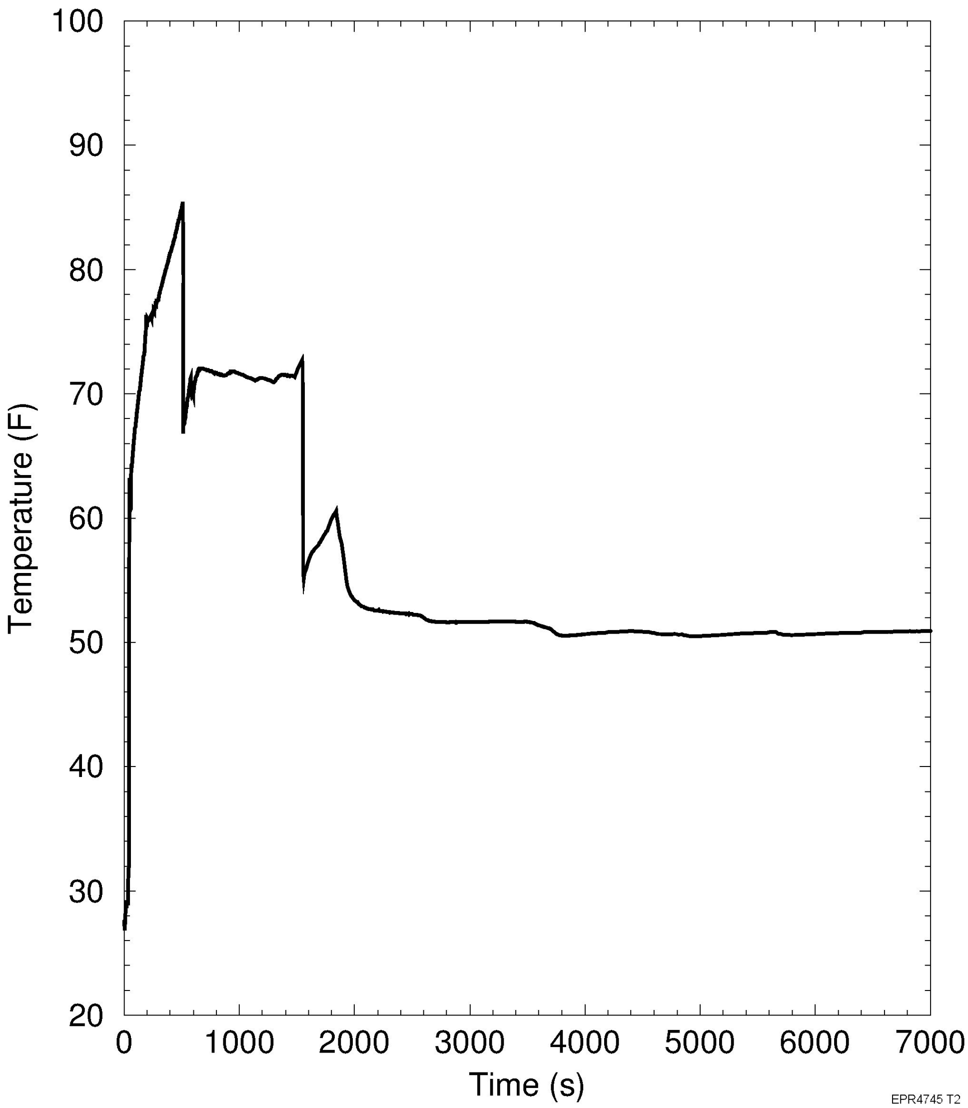

Figure 15.2-76—FWLB Maximum RCS Pressure Case – Core Flow

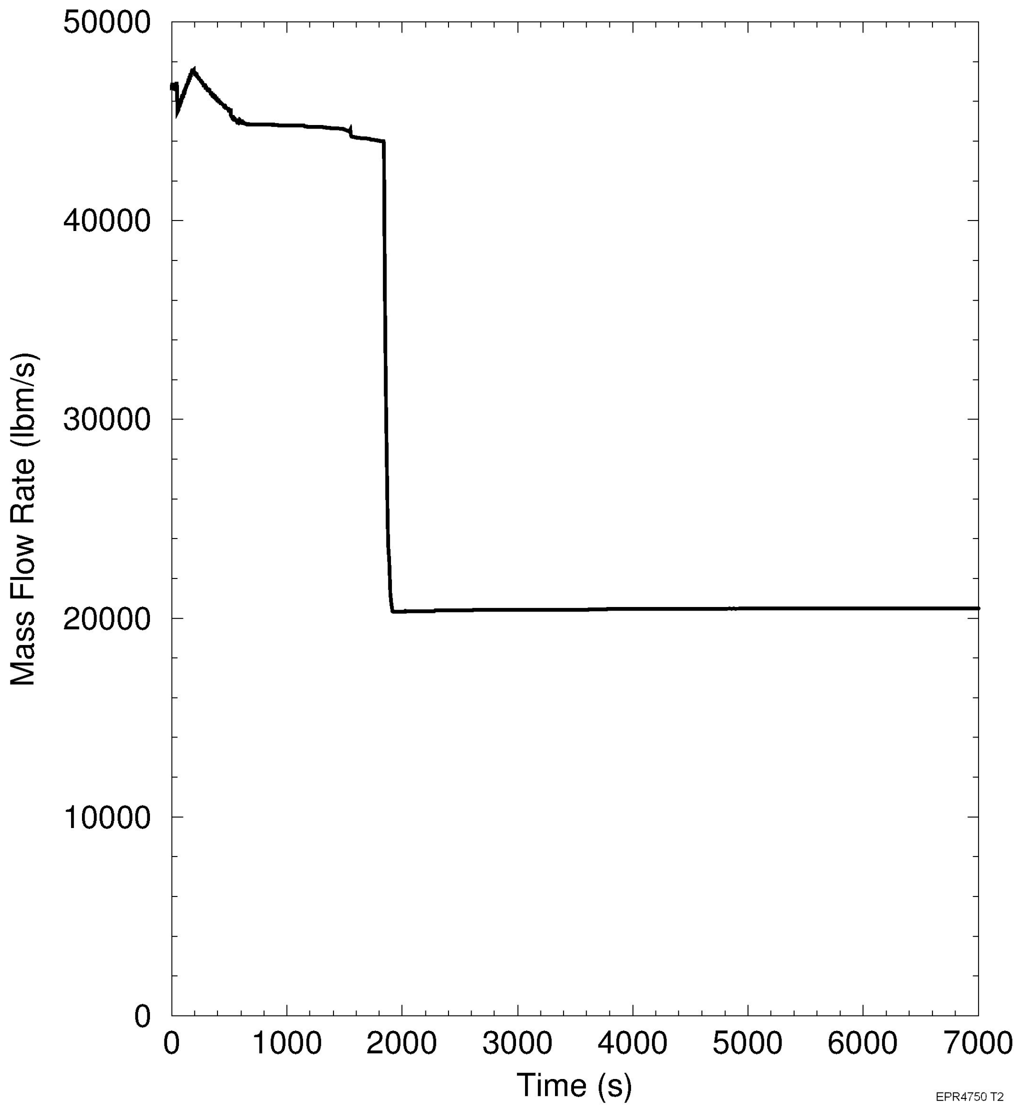

Figure 15.2-77—FWLB Maximum RCS Pressure Case – Break Flow

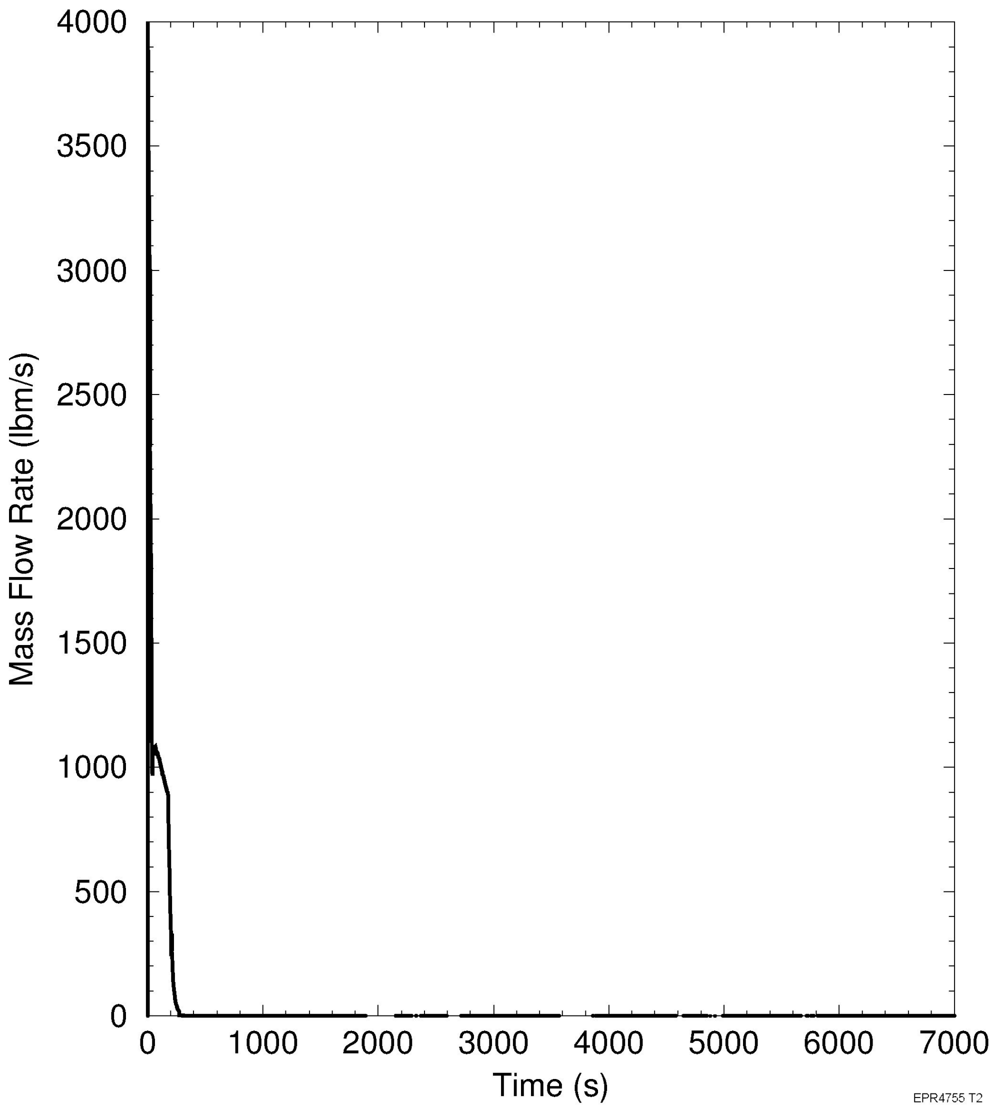

Figure 15.2-78—FWLB Maximum RCS Pressure Case – Main Steam Relief Loops 1 and 2

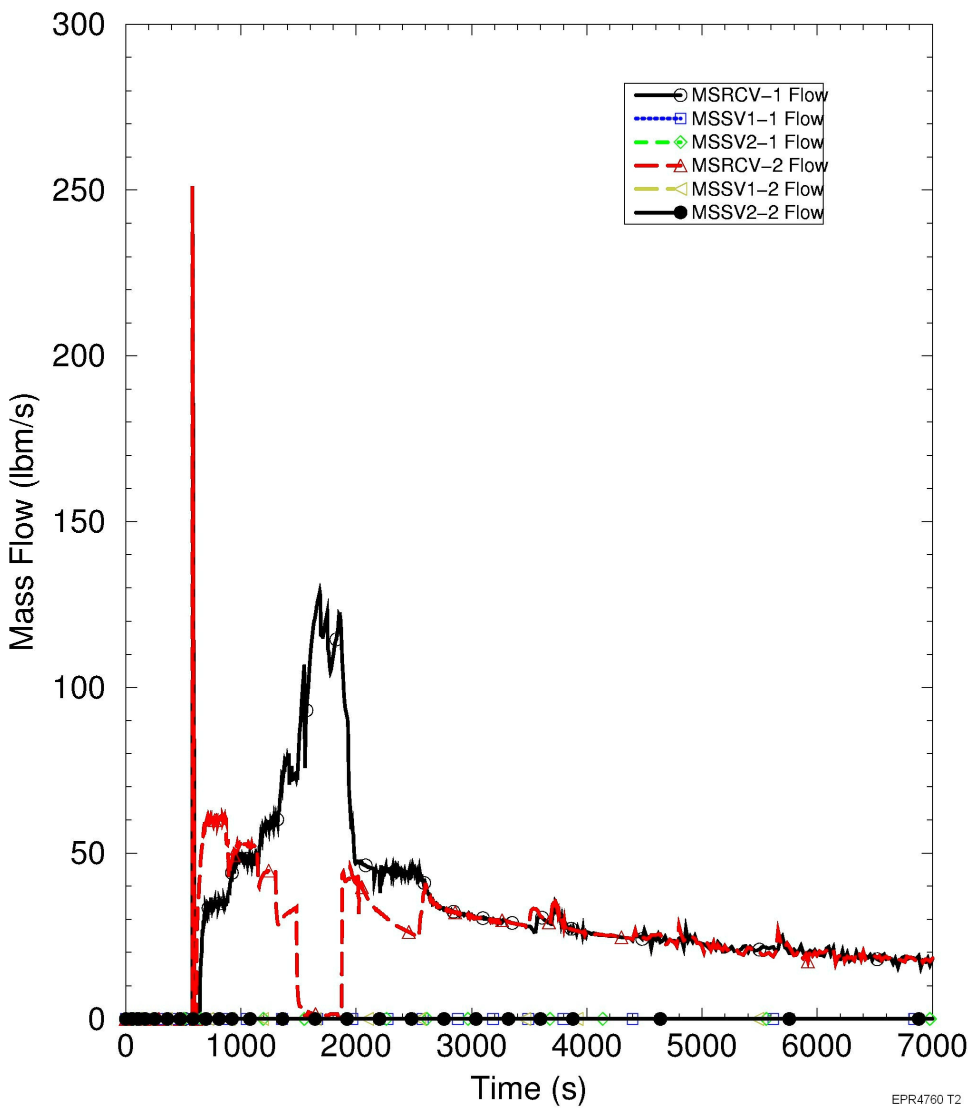

Figure 15.2-79—FWLB Maximum RCS Pressure Case – Main Steam Relief Loops 3 and 4

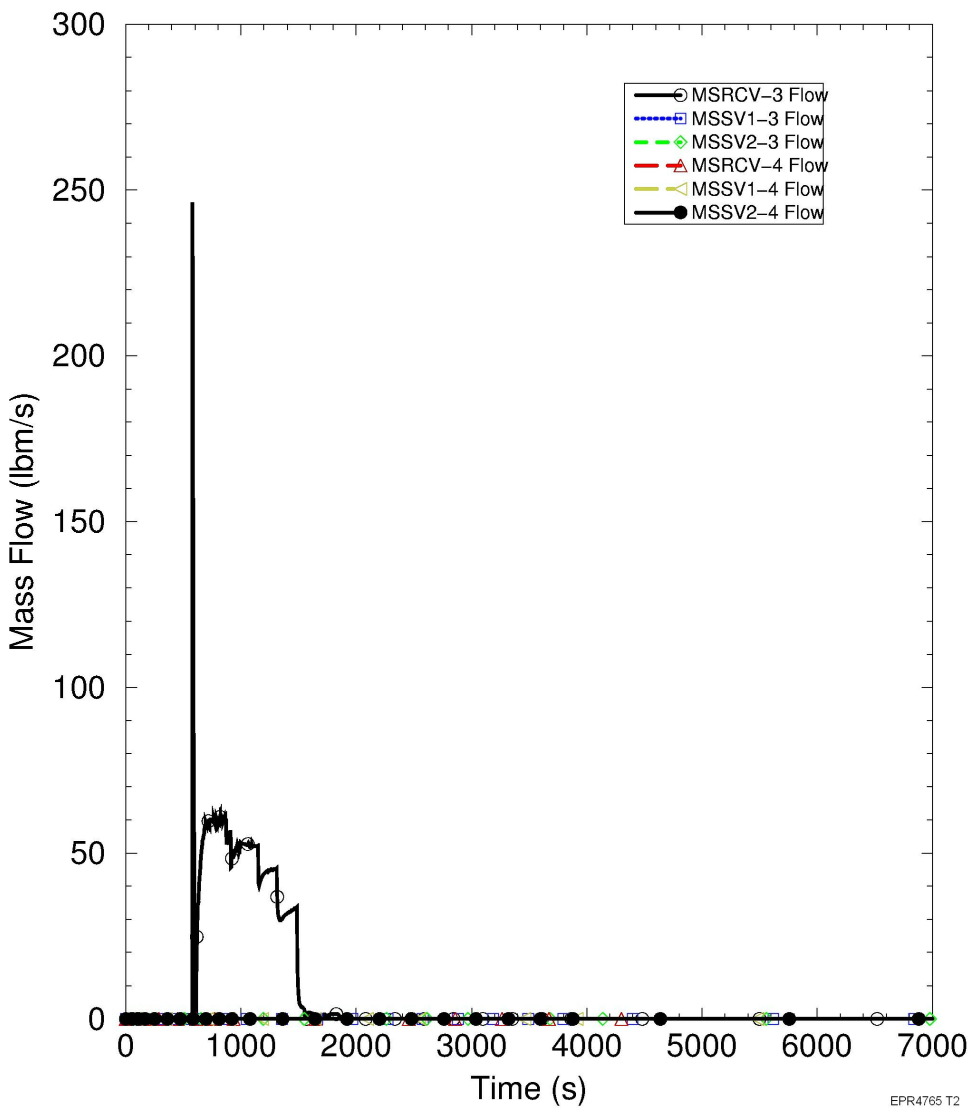

Figure 15.2-80—FWLB Maximum RCS Pressure Case – Steam Generator Dome Pressures

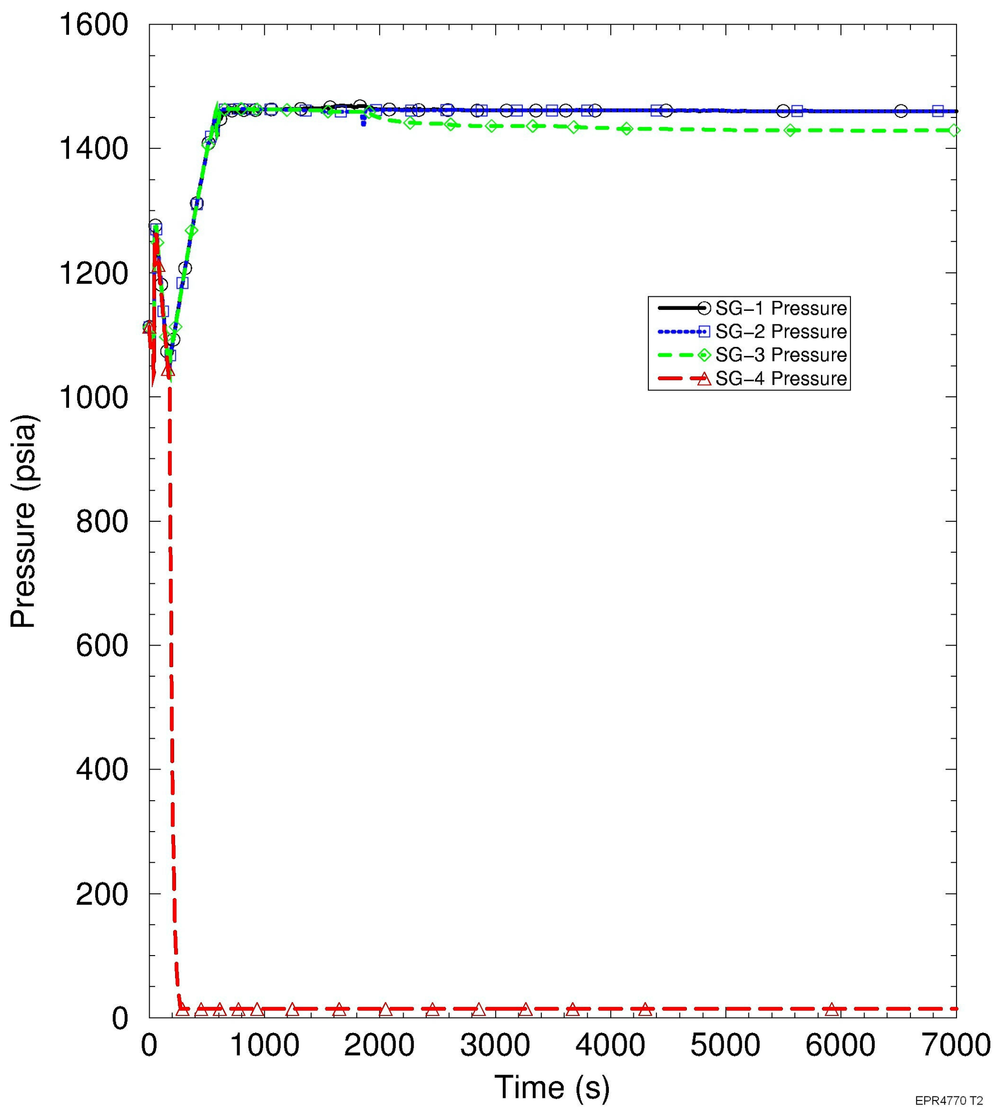

Figure 15.2-81—FWLB Maximum RCS Pressure Case – Steam Generator Total Mass

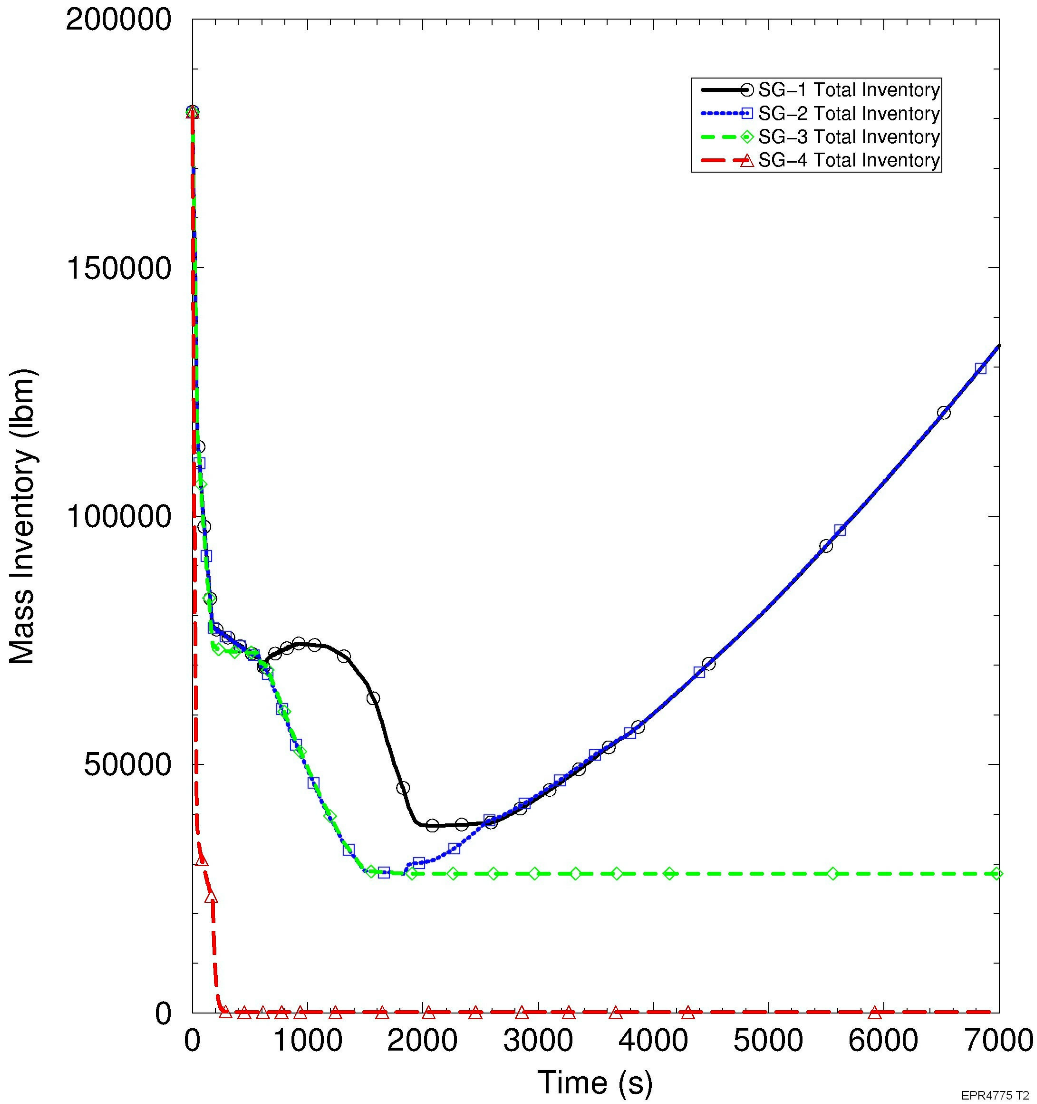

Figure 15.2-82—FWLB Maximum RCS Pressure Case –Steam Generator Liquid Mass

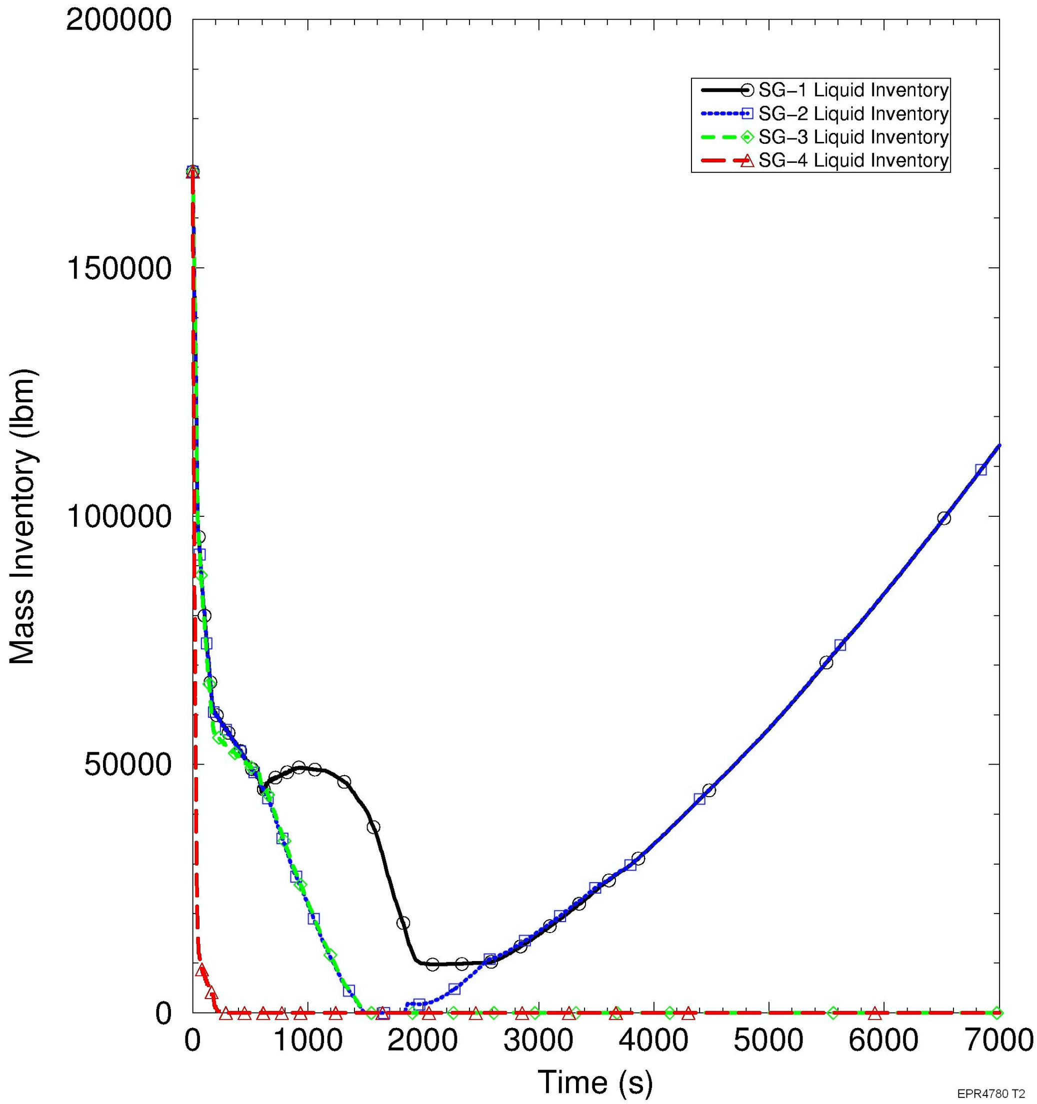

Figure 15.2-83—FWLB Maximum RCS Pressure Case –Net Heat Addition to RCS

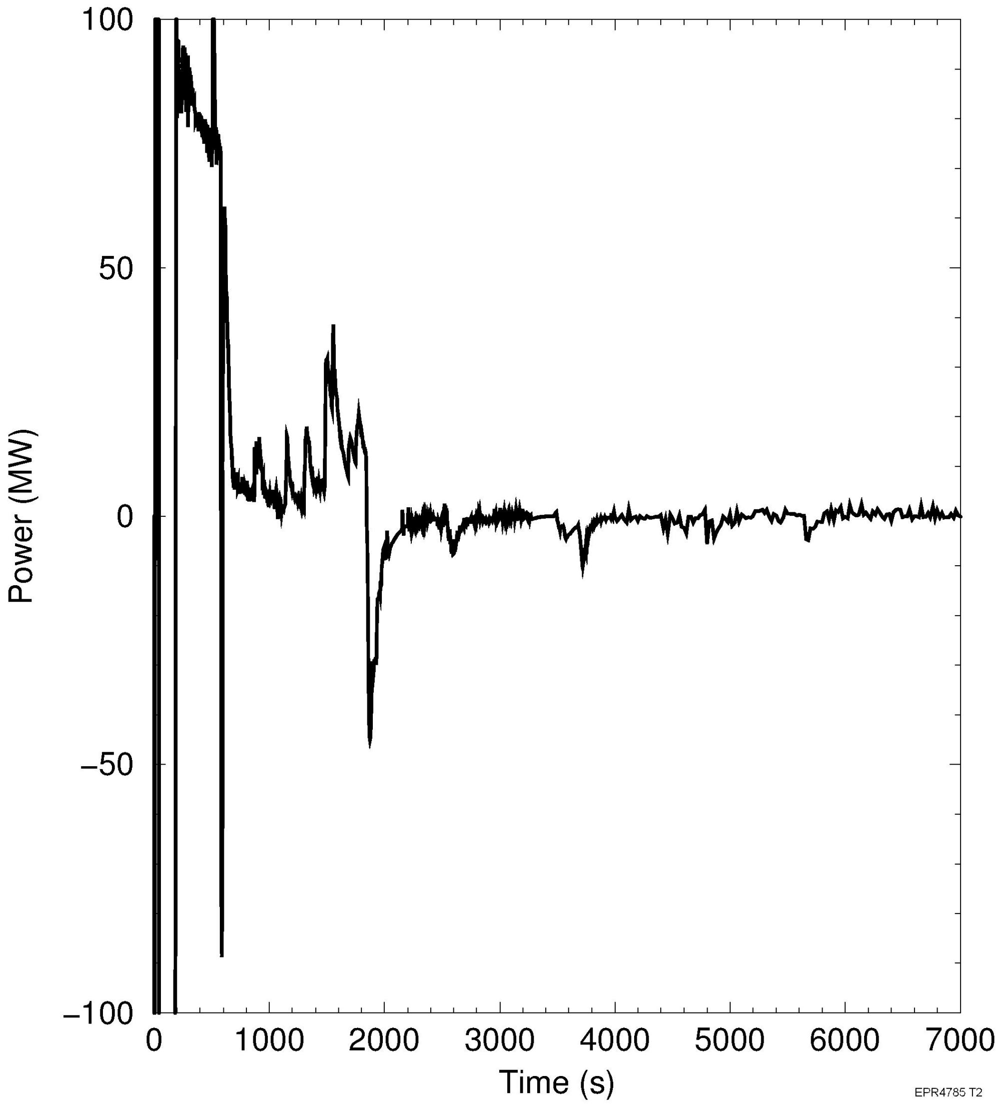

Figure 15.2-84—FWLB Maximum RCS Pressure Case – RCS Maximum Pressure

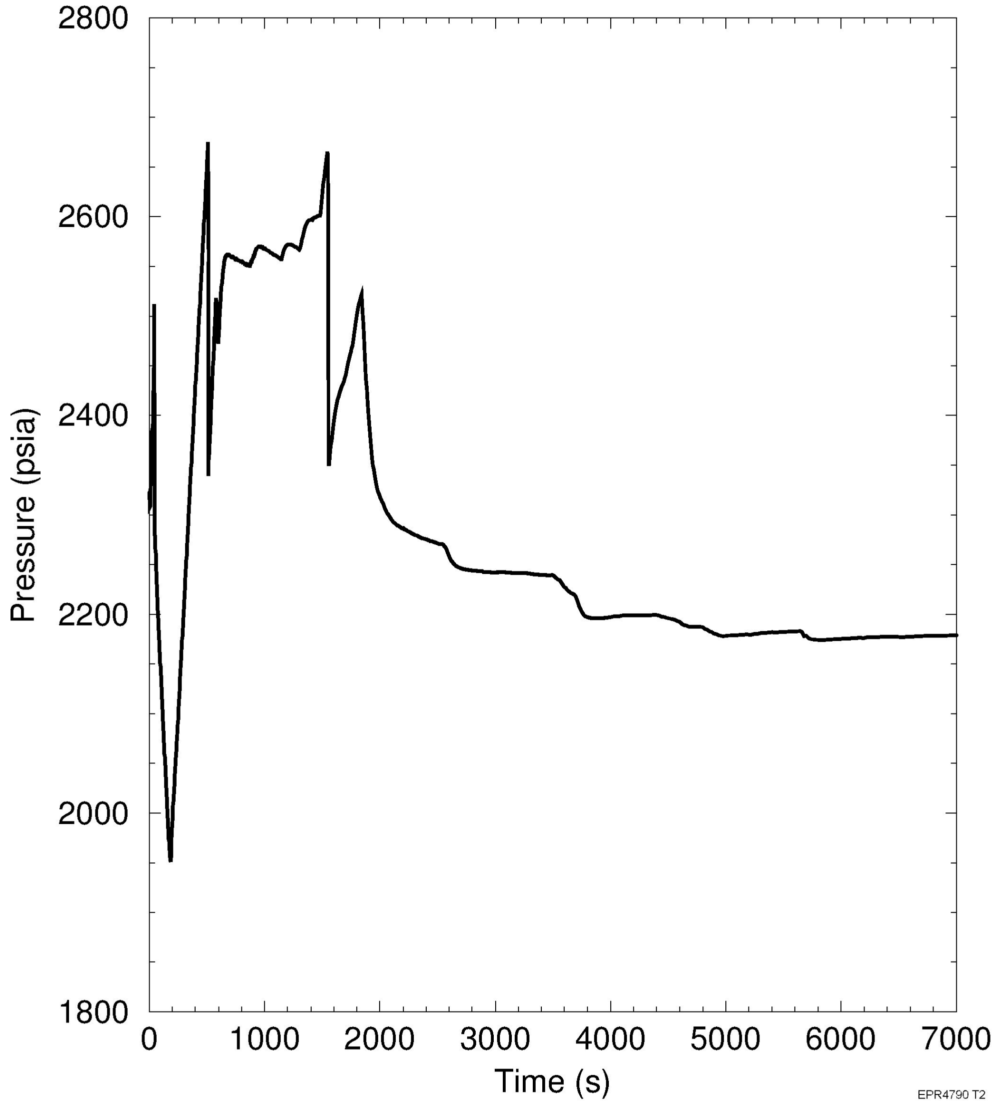

Figure 15.2-85—FWLB Maximum RCS Pressure Case – Steam Generator Maximum Pressure

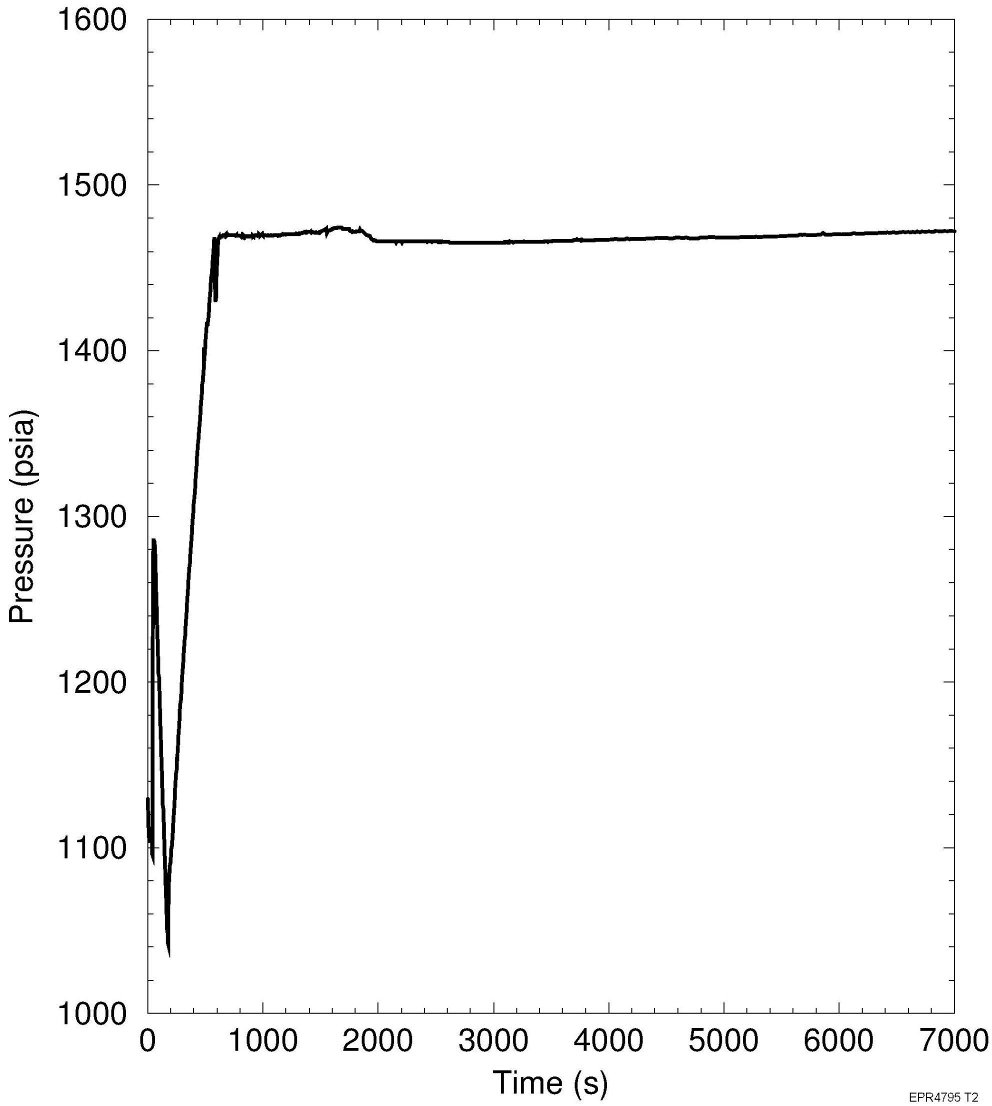

Figure 15.2-86—FWLB Maximum RCS Pressure Case – Reactivities

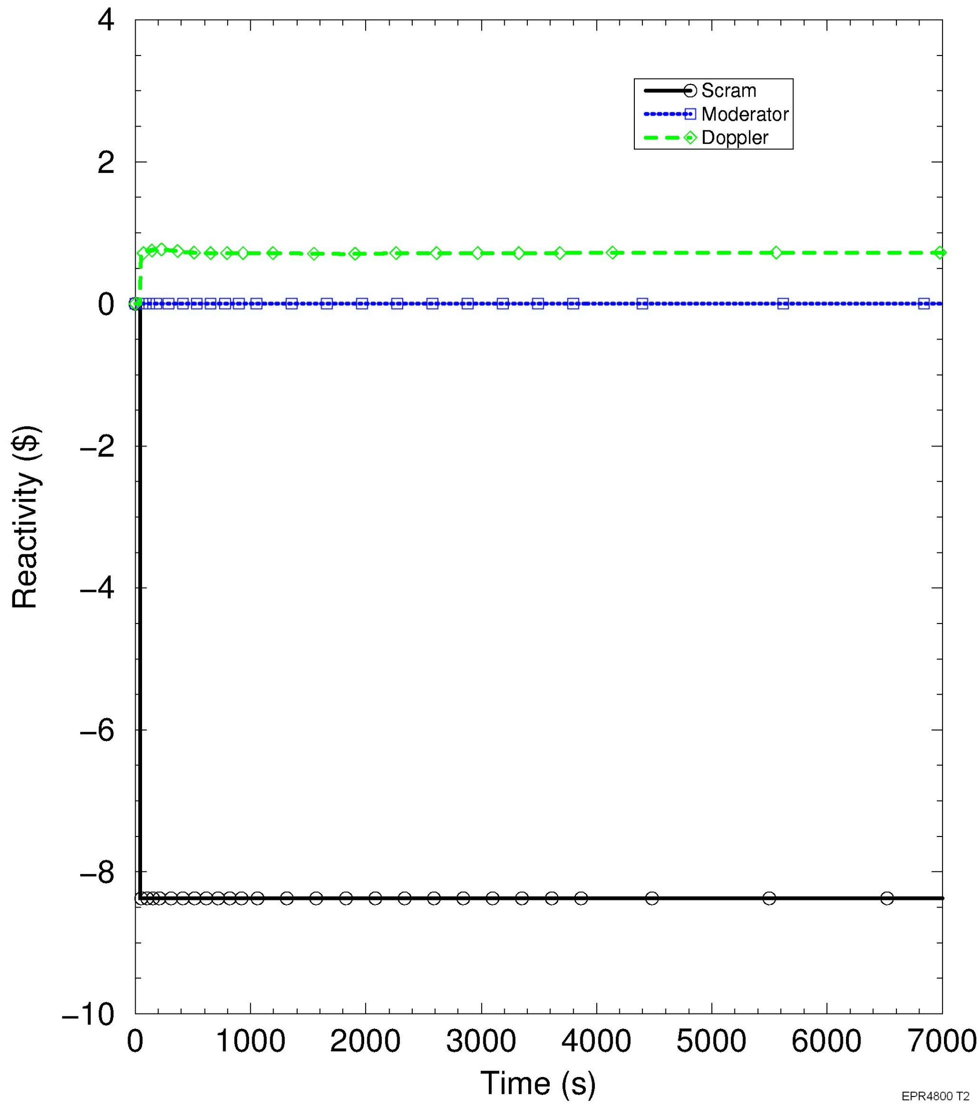

Figure 15.2-87—FWLB Maximum Secondary Pressure Case – Reactor and Total Steam Generator Power

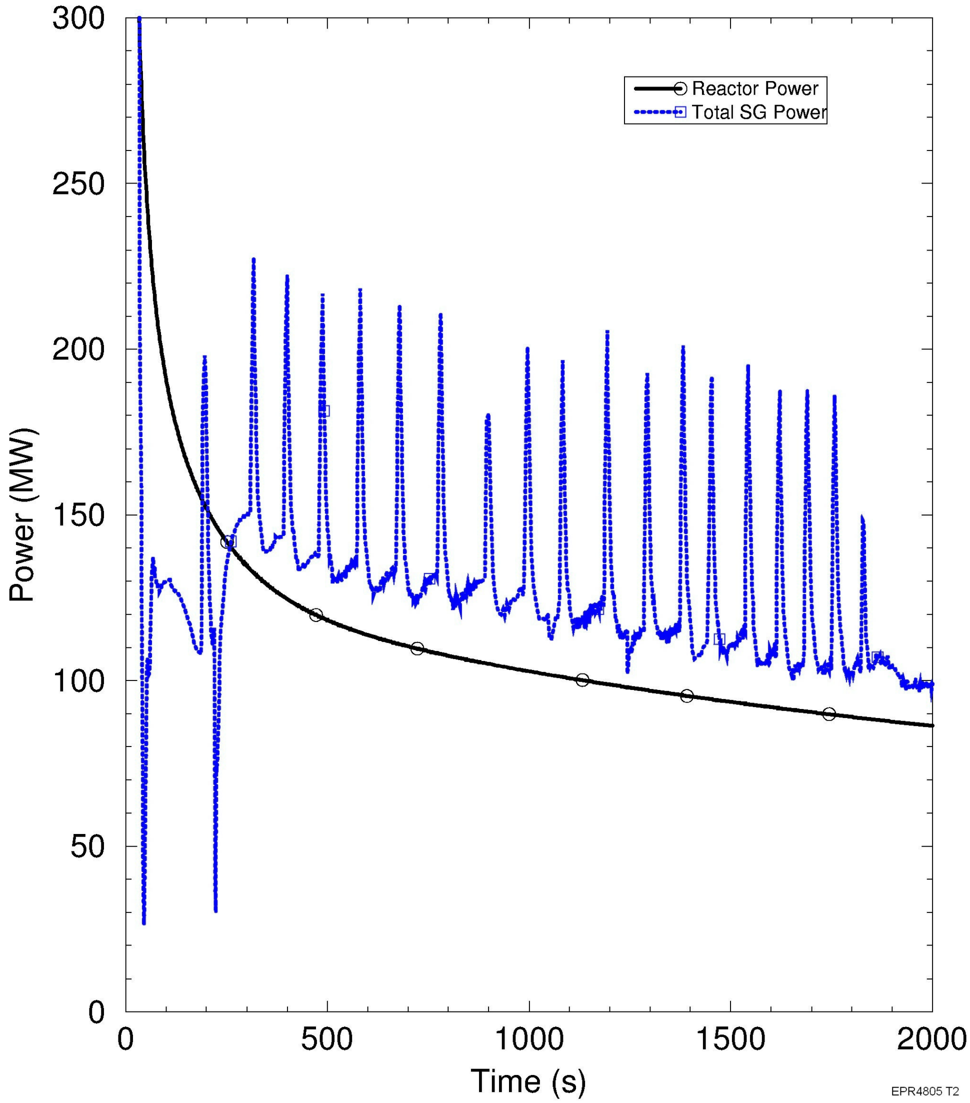

Figure 15.2-88—FWLB Maximum Secondary Pressure Case – Pressurizer Pressure

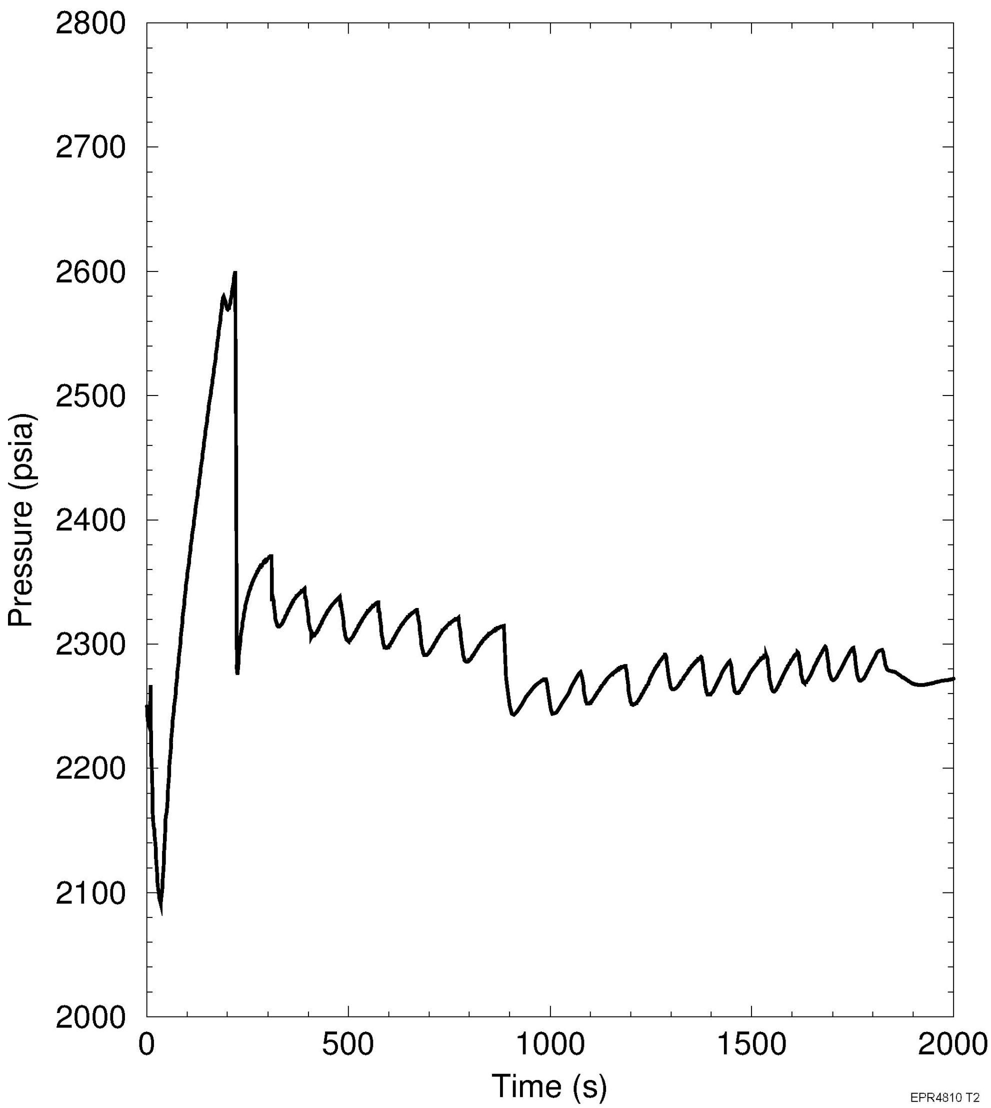

Figure 15.2-89—FWLB Maximum Secondary Pressure Case – PSRV Flow

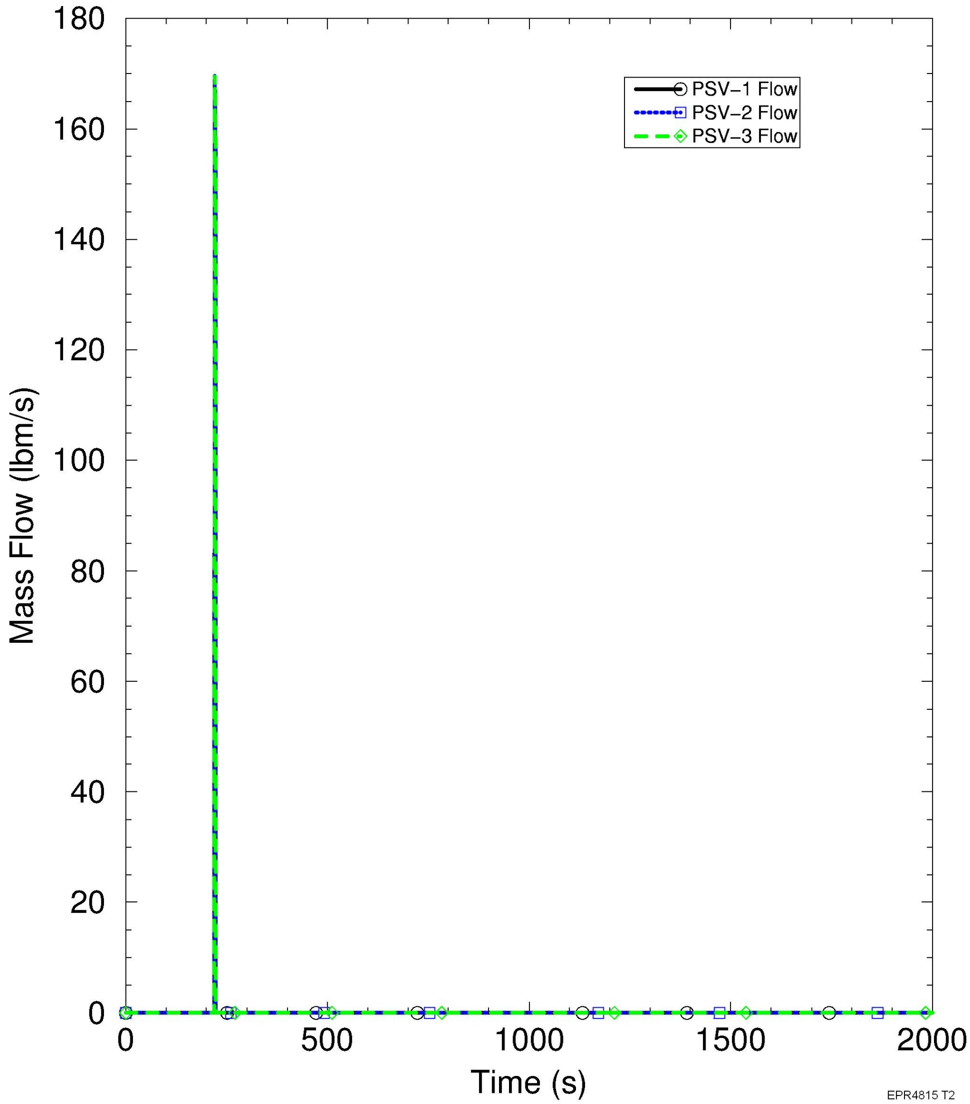

Figure 15.2-90—FWLB Maximum Secondary Pressure Case – Pressurizer Liquid Level

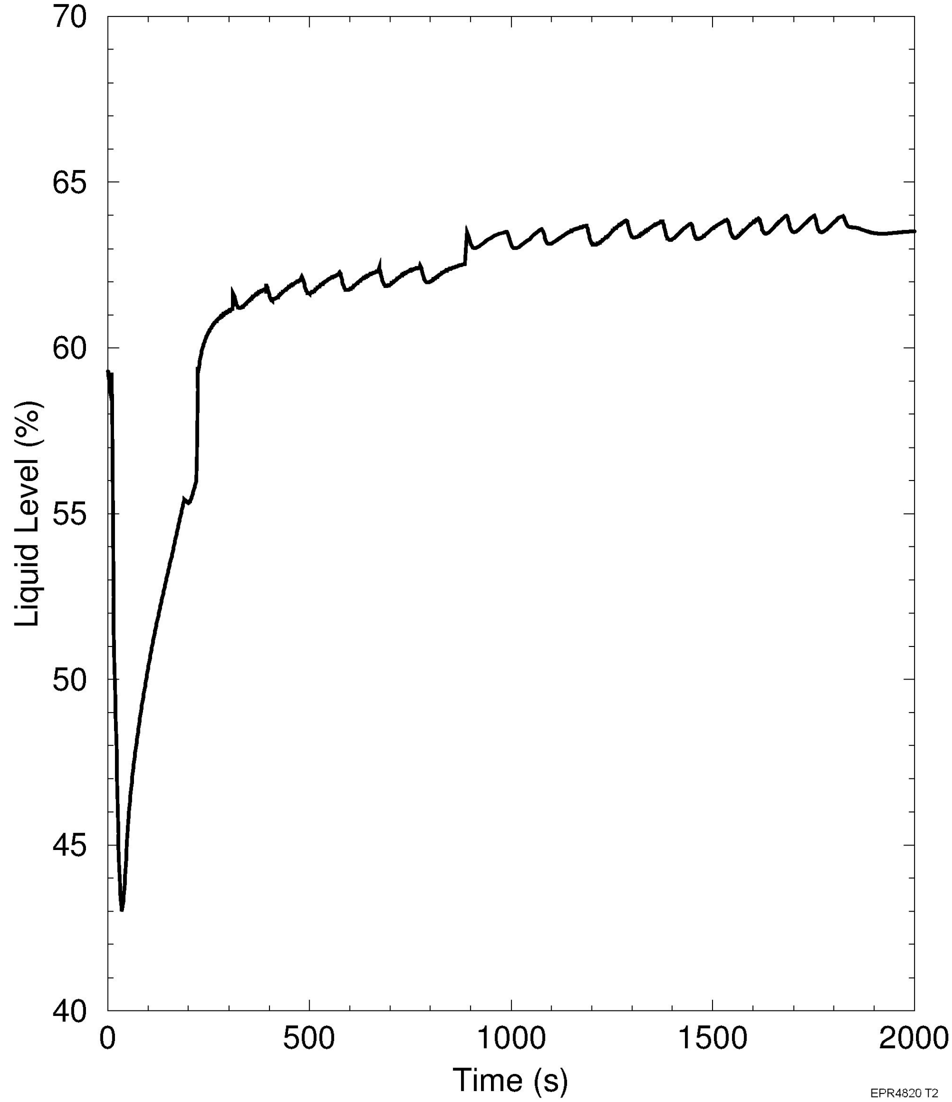

Figure 15.2-91—FWLB Maximum Secondary Pressure Case – RCS Cold Leg Temperatures

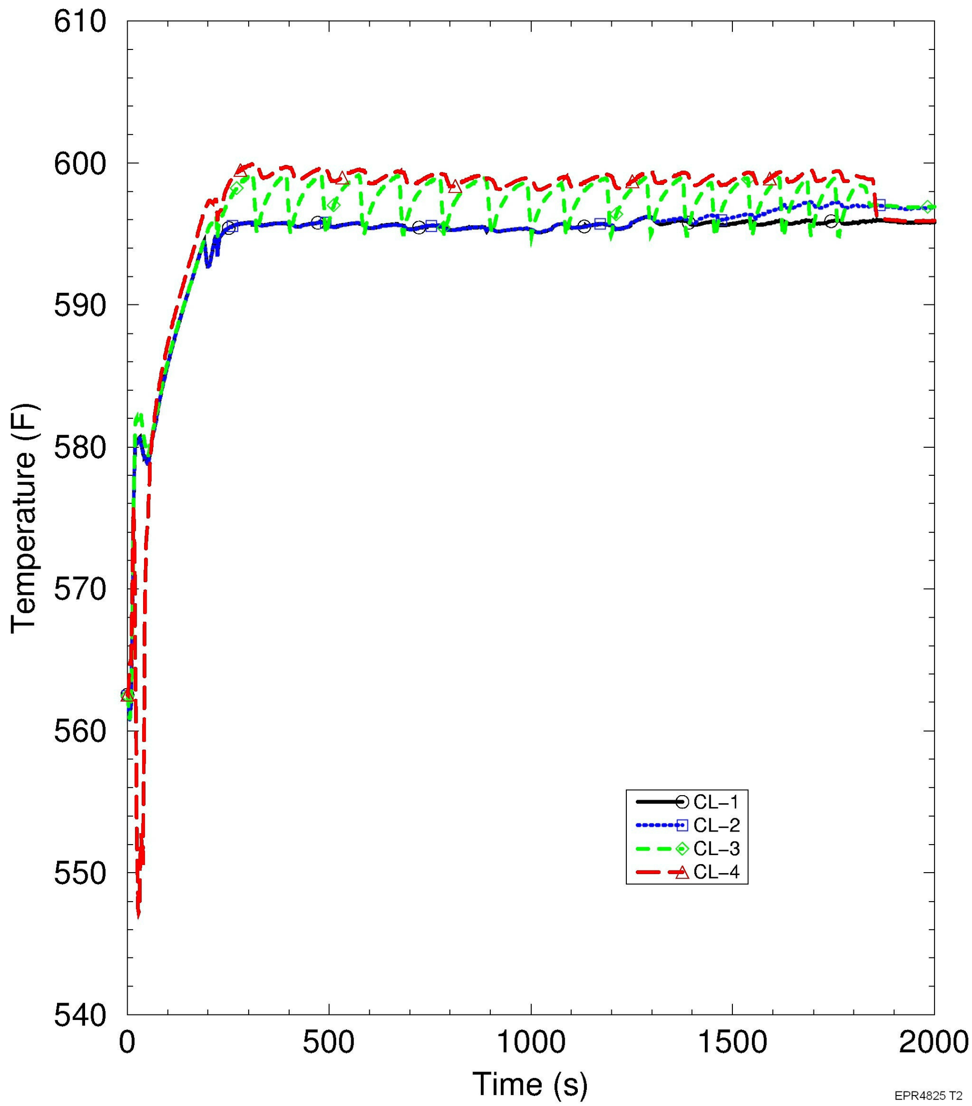

Figure 15.2-92—FWLB Maximum Secondary Pressure Case – RCS Hot Leg and Upper Head Temperature

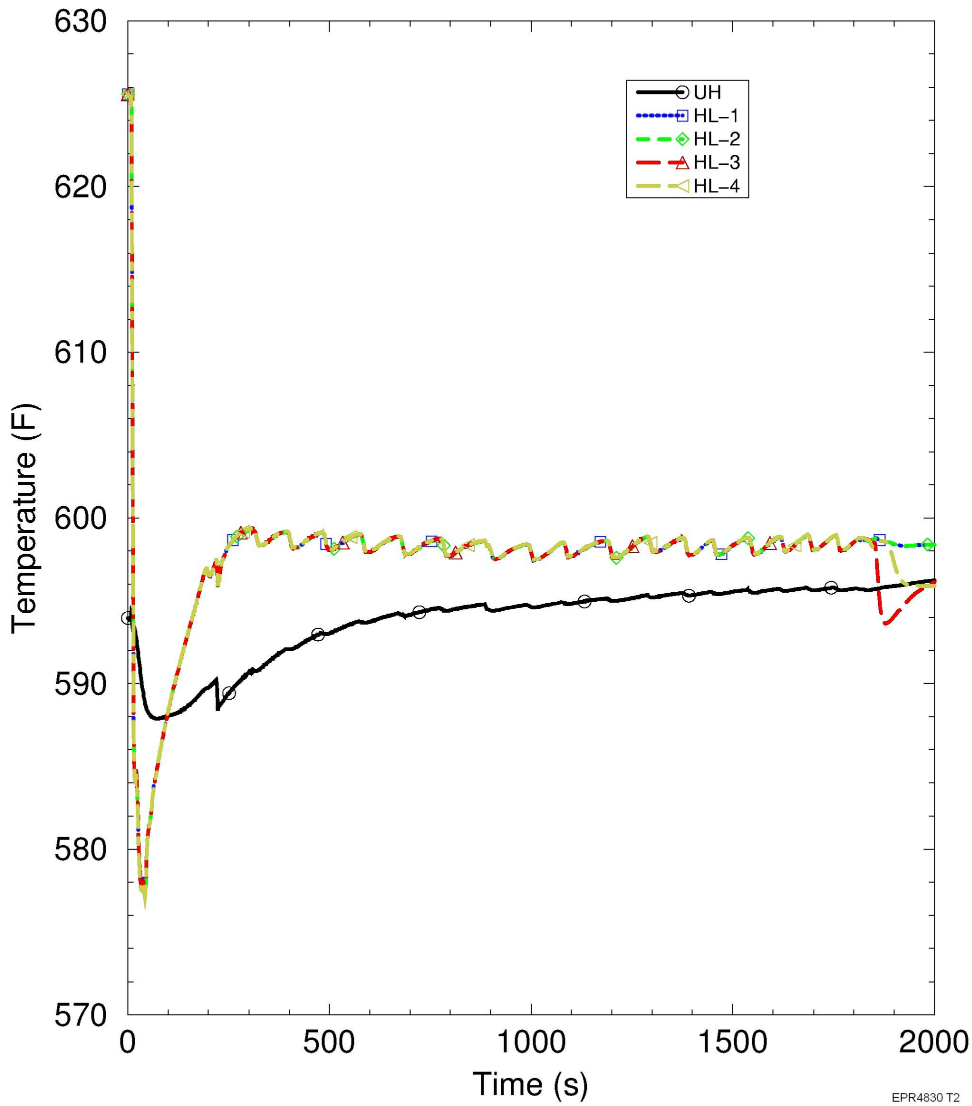

Figure 15.2-93—FWLB Maximum Secondary Pressure Case – Core Exit Subcooling

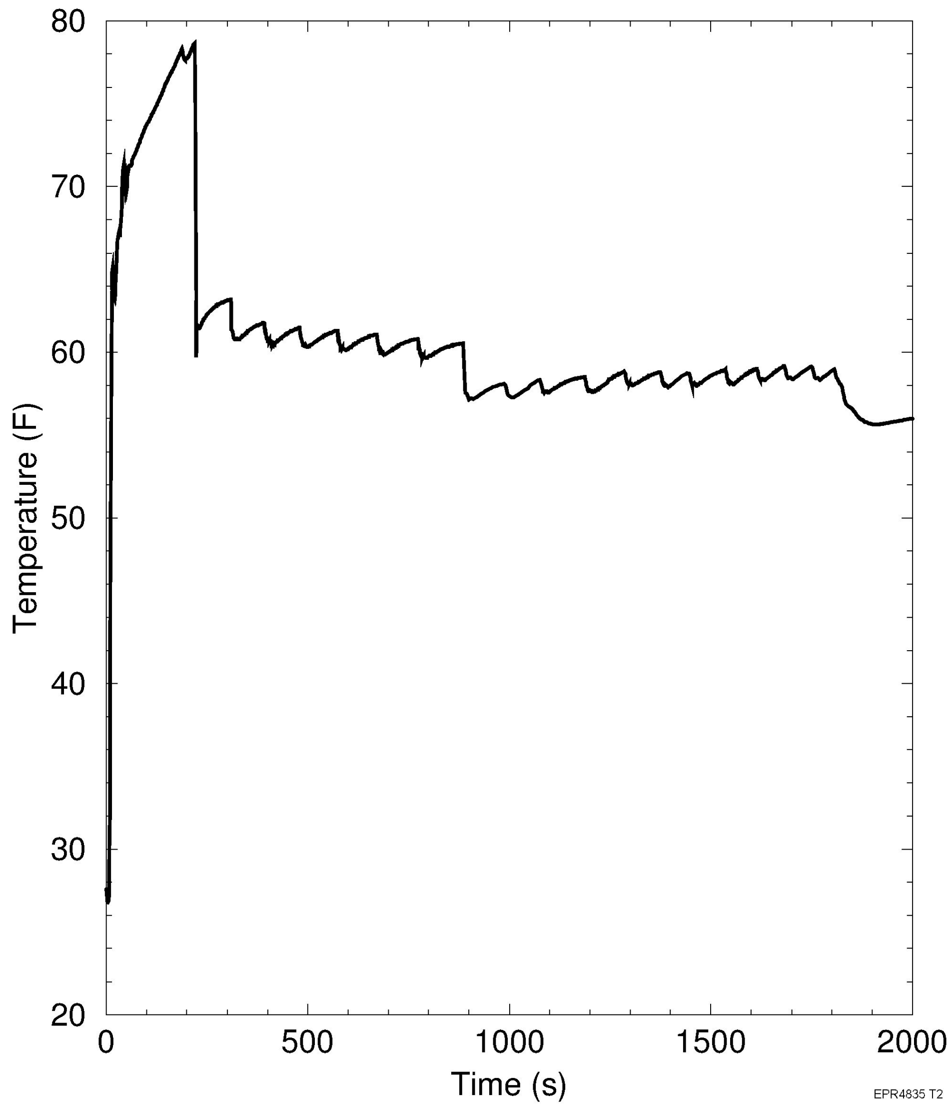

Figure 15.2-94—FWLB Maximum Secondary Pressure Case – Core Flow

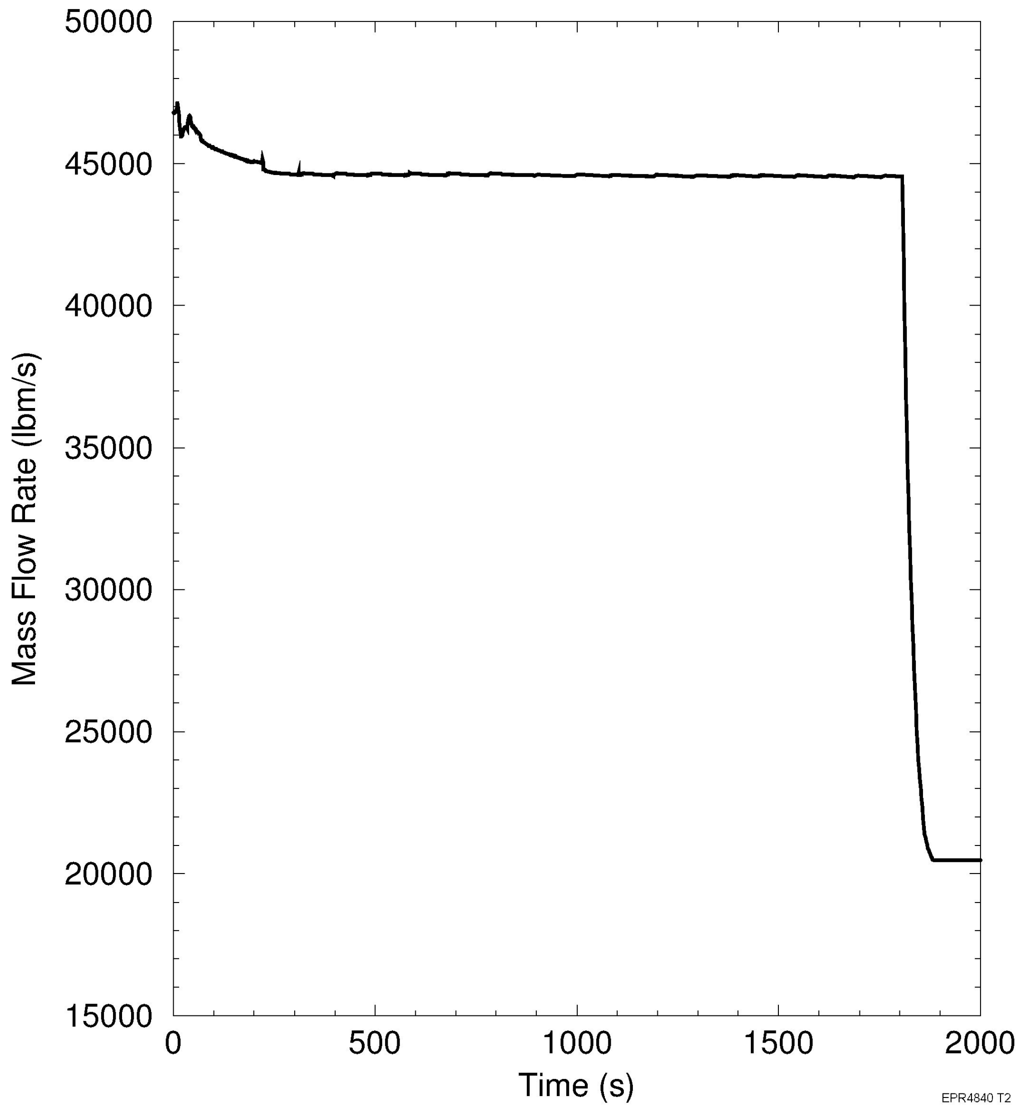

Figure 15.2-95—FWLB Maximum Secondary Pressure Case – Break Flow

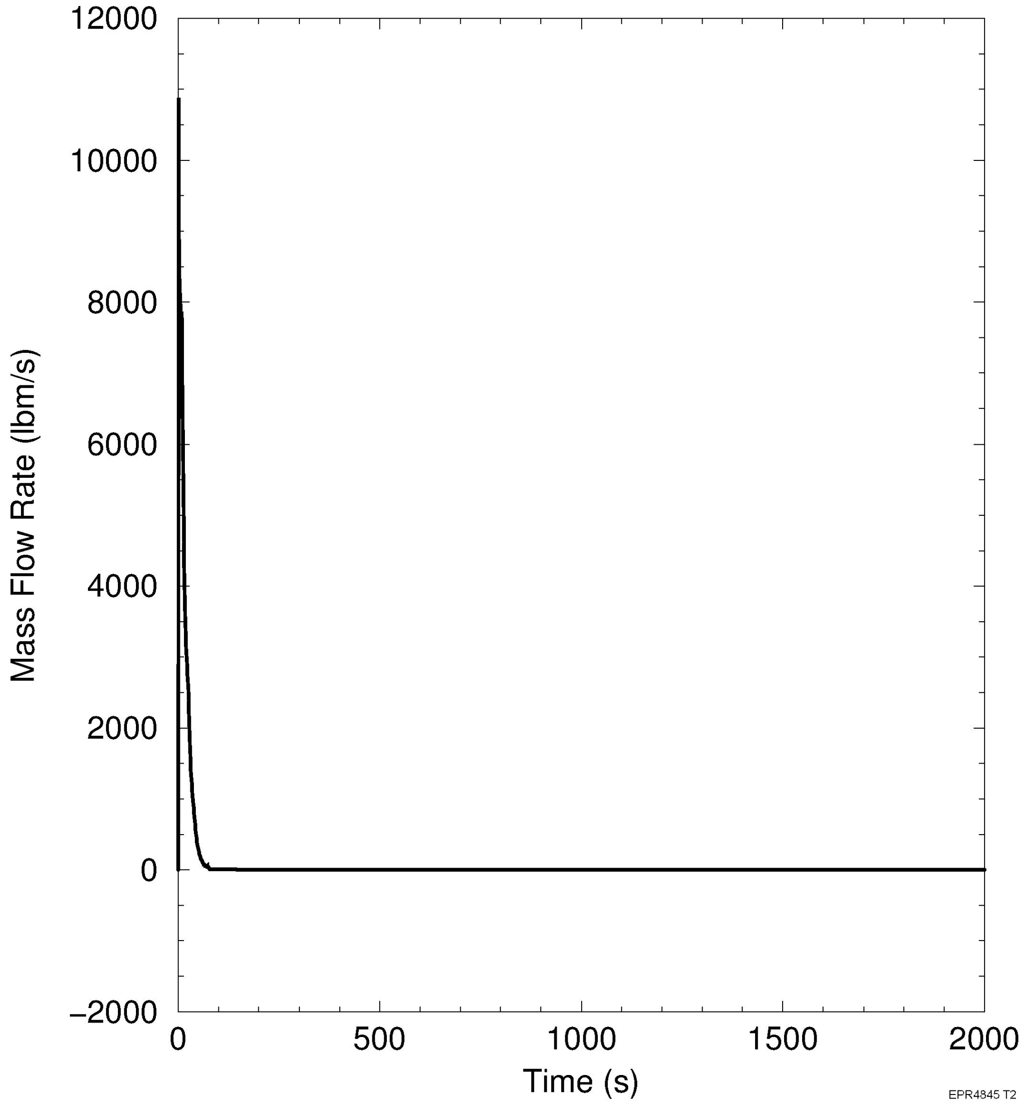

Figure 15.2-96—FWLB Maximum Secondary Pressure Case – Main Steam Relief Loops 1 and 2

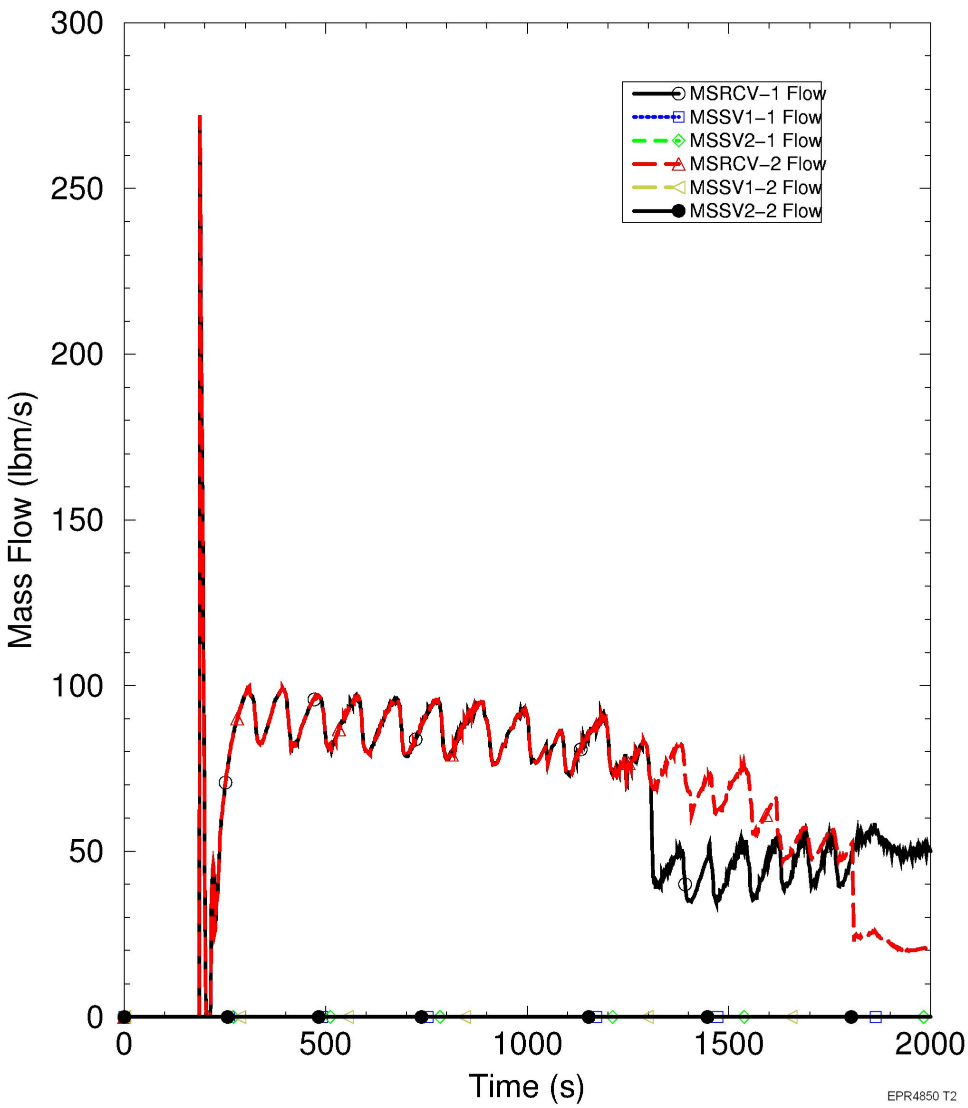

Figure 15.2-97—FWLB Maximum Secondary Pressure Case – Main Steam Relief Loops 3 and 4

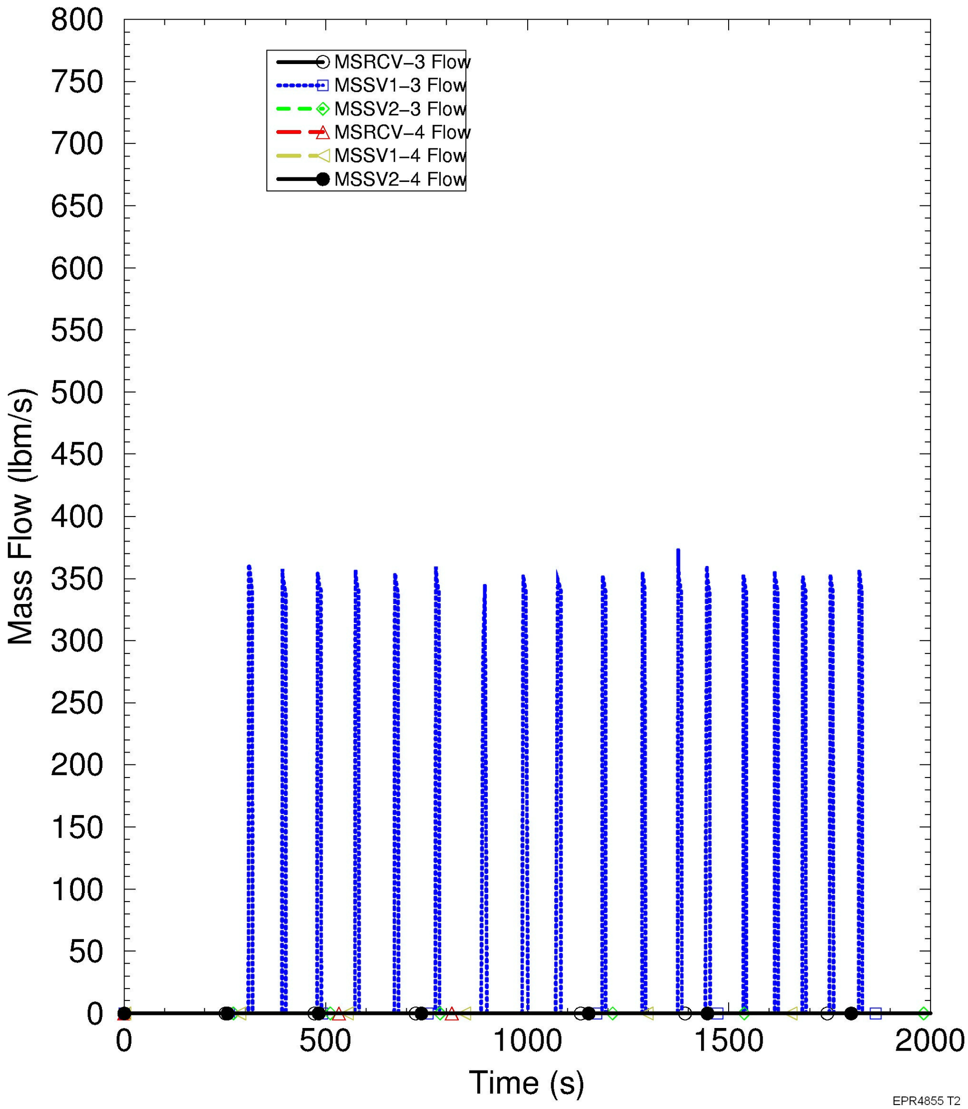

Figure 15.2-98—FWLB Maximum Secondary Pressure Case – Steam Generator Dome Pressures

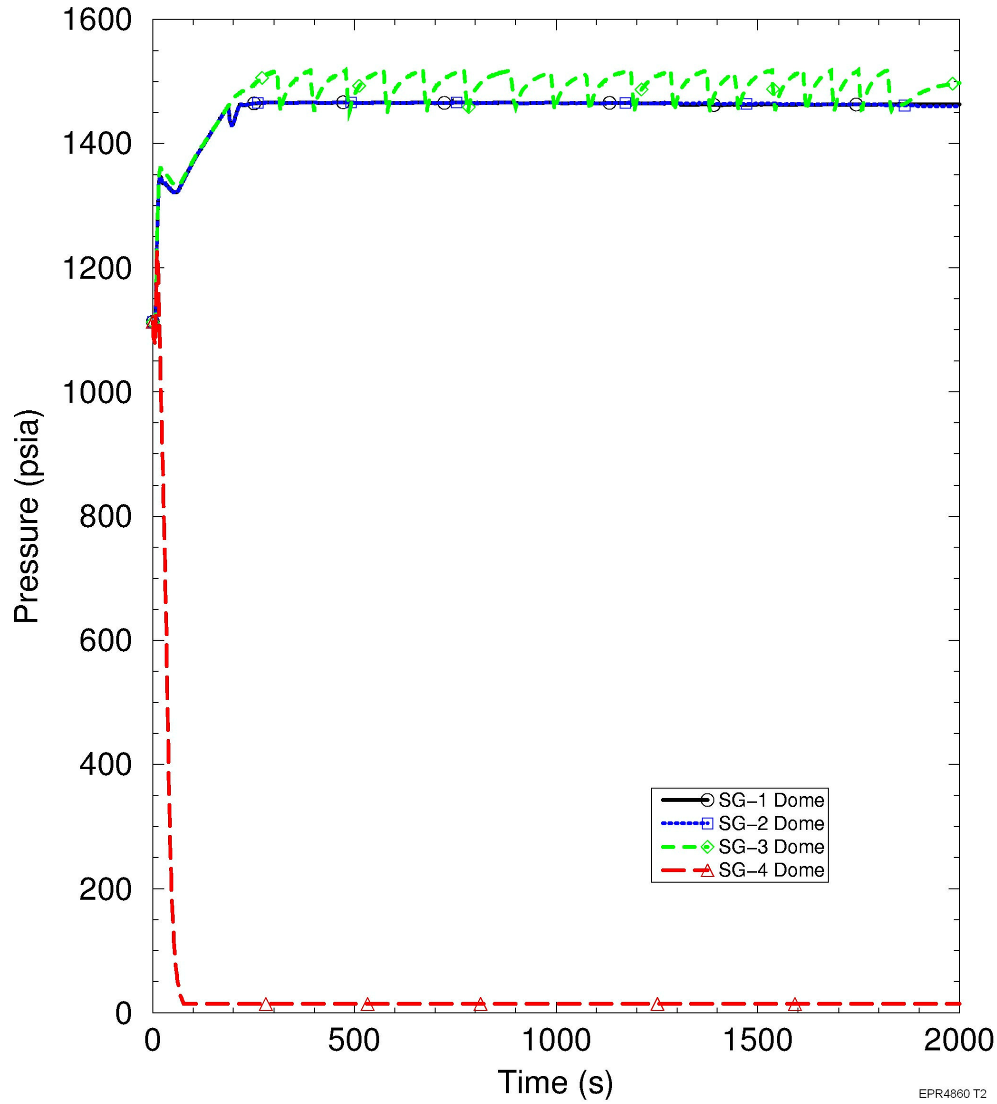

Figure 15.2-99—FWLB Maximum Secondary Pressure Case – Steam Generator Total Mass

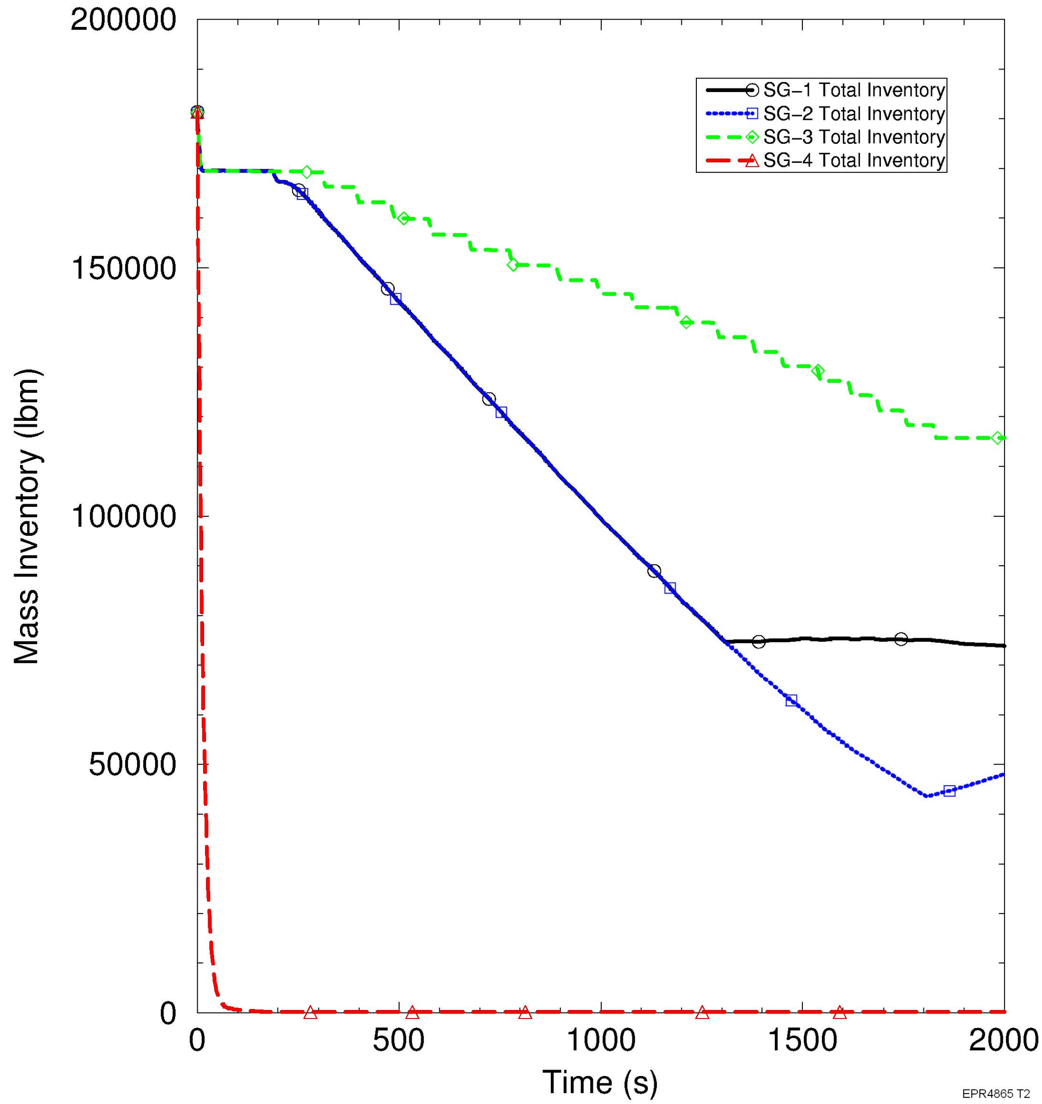

Figure 15.2-100—FWLB Maximum Secondary Pressure Case – Steam Generator Liquid Mass

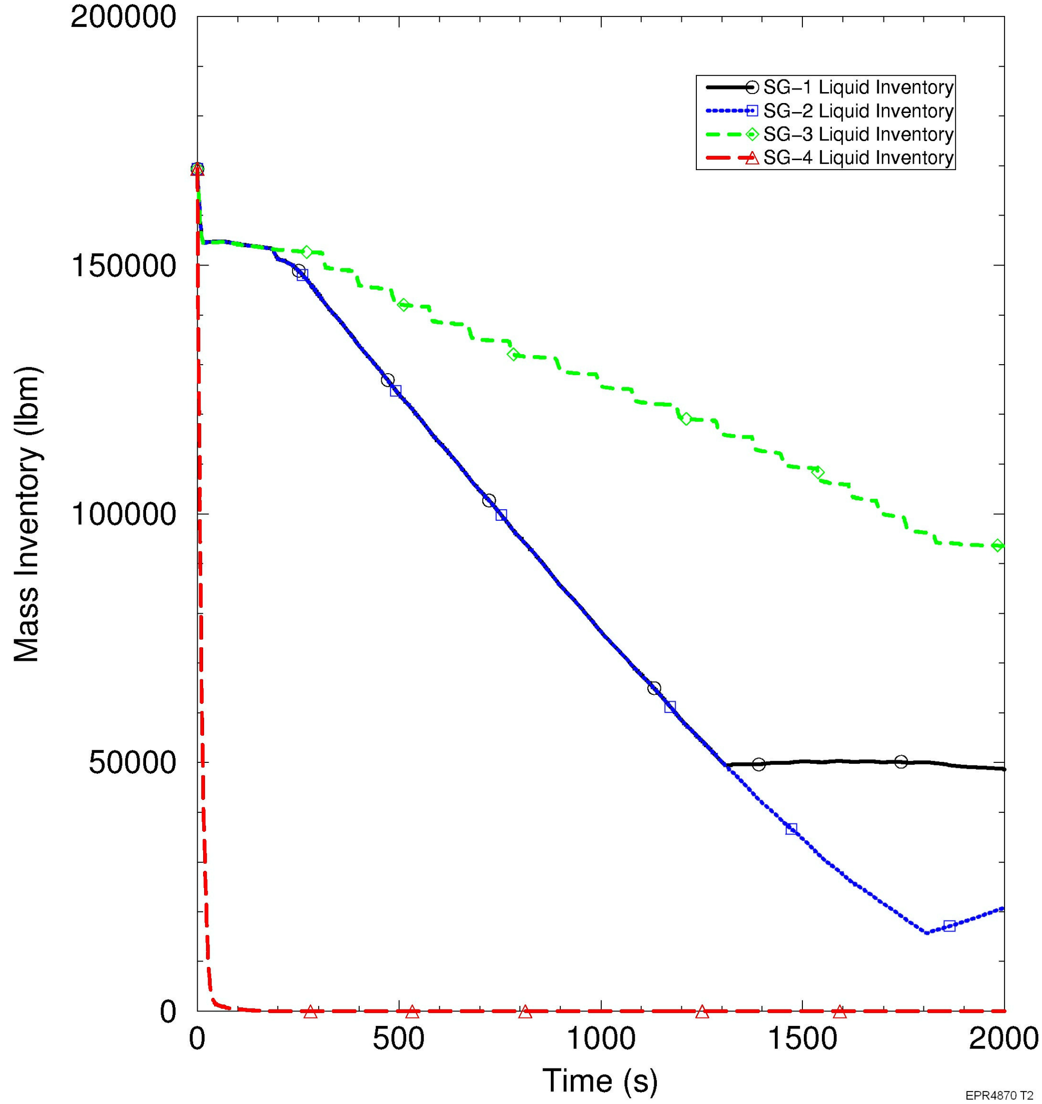

Figure 15.2-101—FWLB Maximum Secondary Pressure Case – Net Heat Addition to RCS

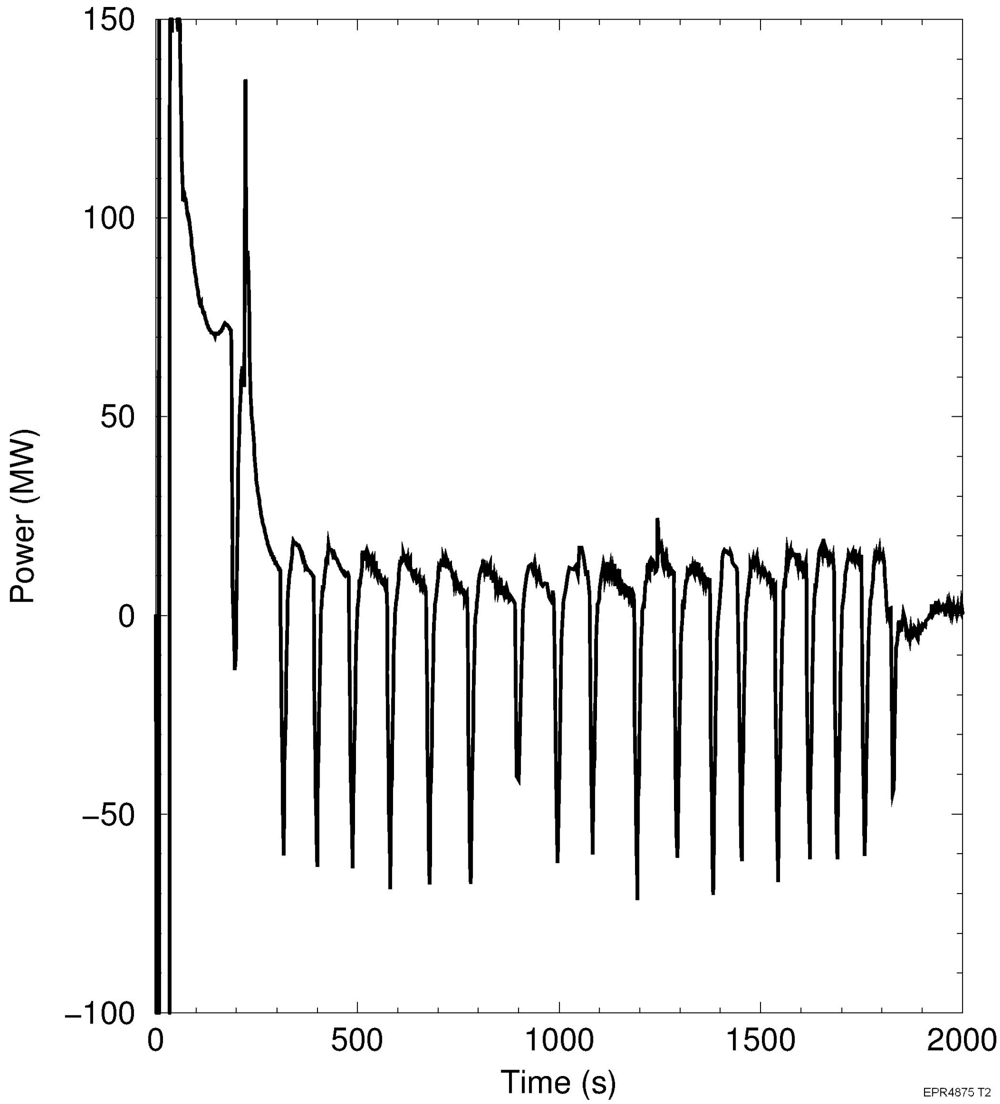

Figure 15.2-102—FWLB Maximum Secondary Pressure Case – RCS Maximum Pressure

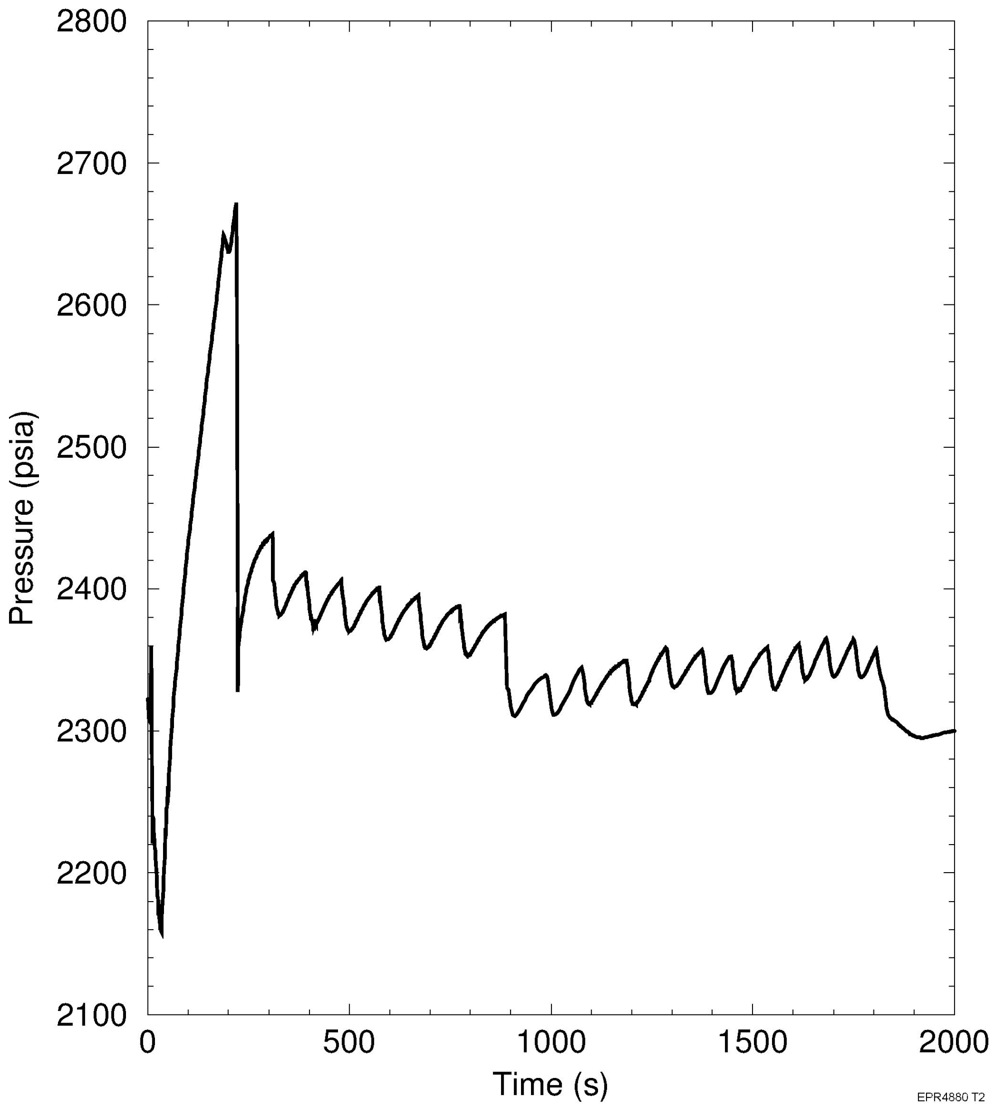

Figure 15.2-103—FWLB Maximum Secondary Pressure Case – Steam Generator Maximum Pressure

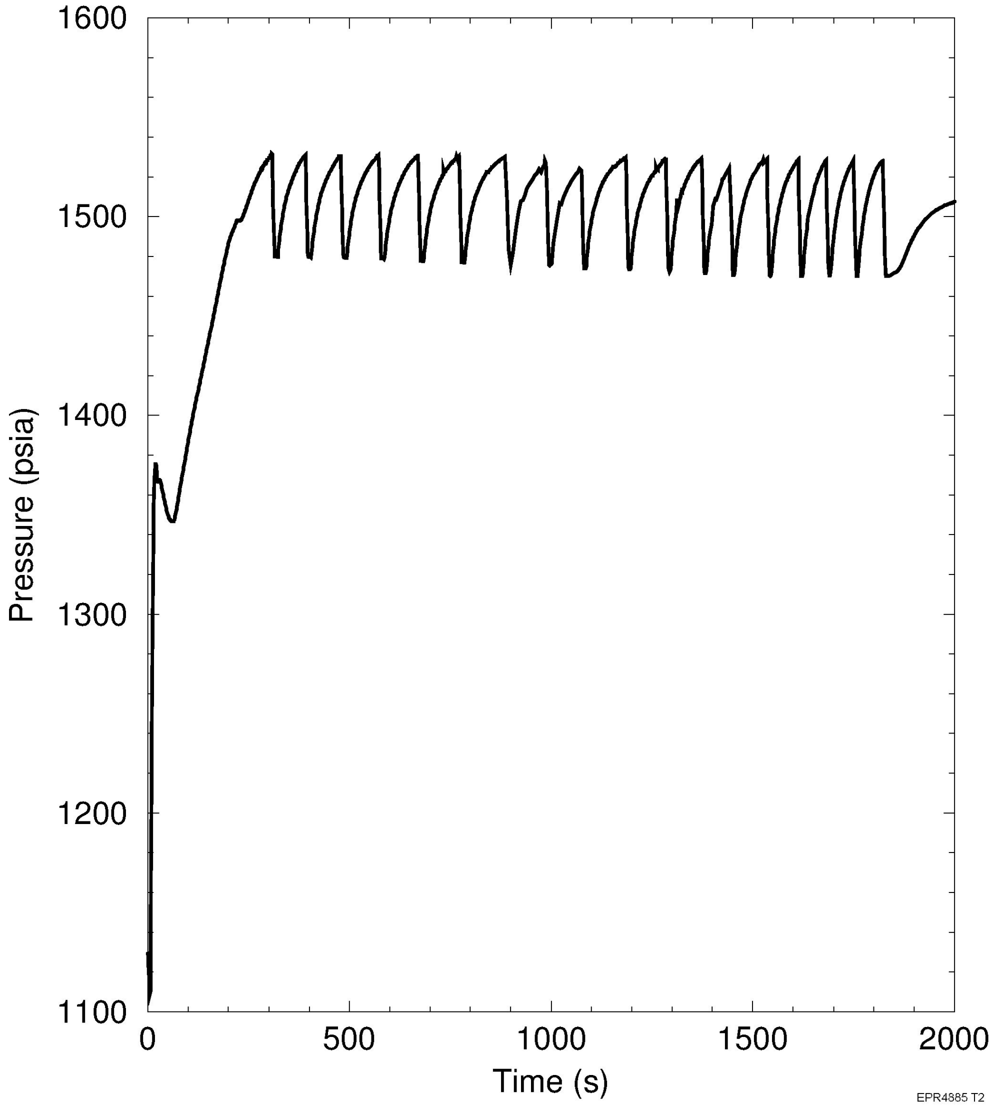

Figure 15.2-104—FWLB Maximum Secondary Pressure Case – Reactivities