15.1 Increase in Heat Removal by the Secondary System

Several anticipated operational occurrences (AOO) and one postulated accident (PA) result in an unplanned increase in heat removal by the secondary system. In these events, a decrease in reactor coolant temperature causes an increase in core reactivity that leads to an increase in core power. Detailed analyses of these reactor coolant system (RCS) cooldown events are presented in this section. The events include the following:

- Section 15.1.1 - Decrease in feedwater temperature.
- Section 15.1.2 - Increase in feedwater flow.
- Section 15.1.3 - Increase in steam flow.
- Section 15.1.4 - Inadvertent opening of a steam generator (SG) relief or safety valve.
- Section 15.1.5 - Steam system piping failures inside and outside of containment.

15.1.1 Decrease in Feedwater Temperature

15.1.1.1 Identification of Causes and Event Description

A decrease in feedwater temperature is an AOO that is postulated to occur because of the inadvertent opening of a feedwater heater bypass valve. The resulting bypass of a low-pressure or high-pressure feedwater heater train decreases feedwater temperature, thereby increasing heat removal from the RCS and lowering the temperature of the RCS. When the moderator temperature coefficient (MTC) is negative, it causes a positive reactivity insertion that increases reactor power and potentially leads to a reactor trip (RT), and RCS and SG pressure limits are not challenged.

The decrease in feedwater temperature event is classified as an AOO (as described in Section 15.0.0.1) that is expected to occur with moderate frequency. The unplanned power increase associated with this event potentially challenges specified acceptable fuel design limits (SAFDL).

15.1.1.2 Methods of Analysis and Assumptions

The analysis of the decrease in feedwater temperature event uses the approved non-loss-of-coolant accident (non-LOCA) analytical methodology described in the Codes and Methods Applicability Report for the U.S. EPR (Reference 1). The S-RELAP5 computer code (described in Section 15.0.2.4) is used to simulate the event and contains provisions to model the primary and secondary systems, as well as the core reactivity response.

The low departure from nucleate boiling (DNB) channel algorithm and high linear power density (LPD) channel algorithm are used to predict the RT and confirm the adequacy of the dynamic compensation of the algorithm consistent with the Incore Trip Setpoint and Transient Methodology for the U.S. EPR (Reference 2).

To determine the limiting event scenario for the specified acceptable fuel design limit (SAFDL) criteria, the minimum departure from nucleate boiling ratio (DNBR) and maximum LPD, a spectrum of decrease in feedwater temperature events is analyzed. The spectrum examines the range of initial conditions specified in Table 15.0-5 and Table 15.0-6. These initial conditions include hot full power (HFP), the time in cycle [beginning-of-cycle (BOC) or end-of-cycle (EOC)], with average coolant temperature (ACT) control or manual rod control (MRC), availability of offsite power [with loss-of-offsite power (LOOP) or without LOOP]. The analyses are performed with zero percent SG tube plugging to maximize the primary-to-secondary heat transfer and the overcooling effect.

Operation with one string of feedwater heaters out of service is analyzed for the decrease of feedwater temperature event, which results in a gradual heat up of the RCS. For the SAFDL acceptance criteria, HFP represents the limiting initial power because the temperature rise across a feedwater heater train is greatest at HFP. Losing the feedwater heater train at these conditions makes this overcooling event more severe.

The only mitigating equipment credited is the PS. A failure in a PS division is inconsequential because the protection system is designed as single failure proof due to its redundancy. There is no single failure which makes this event more severe.

A spectrum of analyses for HFP with zero percent SG tube plugging cases is performed to determine that the HFP EOC manual rod control with LOOP is the limiting case, as identified in Table 15.0-62. The limiting conditions are presented in Table 15.0-63.

The following limiting conditions apply to the analysis of the decrease in feedwater temperature event:

- At hot full power (HFP), a feedwater temperature reduction of 100°F is assumed to occur instantaneously at the feedwater inlet to each SG. This condition conservatively bounds possible physical reduction in temperature attributable to a single malfunction.
- A bounding end-of-cycle (EOC) MTC value is used.
- No operator actions are credited.

The applicable acceptance criteria for the decrease in feedwater temperature event are as follows:

- The thermal margin limit SAFDLs, DNB and fuel centerline melt (FCM), criteria are satisfied.
- Maximum RCS pressure limits are not exceeded.

This event meets the SAFDL criteria, which are satisfied by the combination of the low DNB and high LPD limiting conditions for operation (LCO) and RT setpoint described in Reference 2.

15.1.1.3 Results

Table 15.1-1—Decrease in Feedwater Temperature - Key Input Parameters provides the initial conditions for this event. Table 15.1-2—Decrease in Feedwater Temperature - Key Equipment Status lists parameters for the protection system functions that may mitigate this event. Table 15.1-3—Decrease in Feedwater Temperature - Sequence of Events provides the sequence of events. Figure 15.1-1—Decrease in Feedwater Temperature - Main Feedwater Flow Rate through Figure 15.1-8—Decrease in Feedwater Temperature - Pressurizer Level show the plant response for a representative decrease in feedwater temperature case. Figure 15.1-57—Decrease in Feedwater Temperature - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

The transient is initiated by instantly decreasing the main feedwater (MFW) temperature to the SGs to 346°F from the normal HFP temperature of 446°F. The decrease in feedwater temperature in the SGs causes a decrease in cold-leg temperature which, in combination with a negative MTC, increases core power. This condition quickly generates a low DNB ratio (DNBR) RT and subsequent turbine trip (TT). The TT closes the turbine stop valves (TSV) and is assumed to cause a loss of offsite power (LOOP). The LOOP terminates MFW flow and starts the coastdown of the reactor coolant pumps (RCP). The closing of the TSVs causes SG secondary pressure to increase to the main steam relief train (MSRT) setpoint, which reduces heat removal from the RCS. This condition causes RCS temperatures and pressure to increase. As the RCPs coast down, primary system temperatures increase further to establish natural circulation, which contributes to the increase in RCS pressure. The pressurizer (PZR) safety relief valves (PSRV) open to control RCS pressure. The plant is now in a stable, controlled state.

15.1.1.4 Radiological Consequences

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs, nor is there a release of radioactive materials to the environment.

15.1.1.5 Conclusions

The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

The analyses performed demonstrate that the system transitions to a stable, controlled state, with the peak reactor coolant and main steam system pressures remaining below 110 percent of their respective design values for the duration of the transient.

15.1.1.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.1.1 events included in NUREG-0800, Section 15.1.1–15.1.1.4, (Reference 3) and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
   - Response: Reactor coolant and main steam system pressures are maintained below 110 percent of their respective design values for the duration of the event as concluded in Section 15.1.1.5.
2. Fuel cladding integrity is maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.
   - Response: Fuel cladding integrity is maintained as concluded in Section 15.1.1.5.
3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
   - Response: The event is mitigated and the plant is maintained in a stable condition through the automated response of safety-related equipment, thus avoiding an aggravated plant condition.
4. The requirements stated in RG 1.105, Instrument Spans and Setpoints, are used with regard to their impact on the plant response to the type of AOOs addressed in this SRP section.
   - Response: The capability of the instrumentation and controls to meet the requirements of RG 1.105 is demonstrated in Section 7.1.3.4.7. The

instrument spans for the RT functions credited are provided in Table 7.2-1—Reactor Trip Variables.

5. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR 50, must be assumed in the analysis and should follow the guidance stated in RG 1.53.

   – Response: For 15.1.1 events, the only equipment credited is the RPS, which is designed single failure proof. Therefore, there are no single failure considerations for the 15.1.1 events.

6. Parameter values in the analytical model should be suitably conservative.

   A. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed operating plus an allowance of two percent to account for power measurement uncertainty unless a lower number can be justified through the measurement uncertainty methodology and evaluation or the uncertainty is accounted for otherwise (refer to SRP 4.4 as cited in Reference 3). 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 initial power level is rated output plus measurement uncertainty. The four loops are operating at the initiation of the event, as required by technical specifications.

   B. Conservative scram characteristics are assumed (e.g., maximum time delay with the most reactive rod held out of the core.

      - Response: Conservative scram characteristics were assumed.

   C. The core burnup is selected to yield the most limiting combination of MTC, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

      - Response: A conservative core burnup was selected for the analysis.

   D. Mitigating systems should be assumed as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 and as determined by the organization responsible for instrumentation and controls.

      - Response: The setpoints for the mitigating systems include instrument uncertainty.

15.1.2 Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Event Description

A failure or misoperation of the MFW control system is an AOO that can increase feedwater flow to an SG. The most severe event is a rapid full opening of an MFW high-load line (HL) control valve. The MFW low-load lines (LL) are open fully at HFP. The U.S. EPR MFW system consists of three MFW pumps with a total capacity of 105 percent of HFP flow (refer to Section 10.4.7).

The thermal-hydraulic response of an SG with internal feedwater preheating differs from one without this feature. The axial economizer section of the U.S. EPR SG design acts as a preheater. In this SG design, 100 percent of the feedwater flows into the downcomer feeding the axial economizer side of the tube bundle. This flow creates a subcooled region on the outlet side of the tube bundle in which the temperature and associated heat transfer are sensitive to changes in feedwater flow rate. An increase in feedwater flow therefore causes a decrease in the cold-leg temperature associated with the affected SG.

In conjunction with a negative MTC, this decrease in cold-leg temperature leads to an increase in core power. For beginning-of-cycle (BOC) cases with a zero or slightly positive MTC, the average coolant temperature (ACT) control function withdraws control rods to raise the coolant average temperature back into the control band. This withdrawal of control rods causes a power excursion. Regardless of MTC, core power increases.

The increase in MFW flow also reduces steam production in the affected SG. Because turbine demand is assumed constant, the turbine control valve opens to maintain flow to the turbine. Secondary pressure decreases and thereby increases overall heat transfer to the SGs.

A quasi-steady-state is established until the high SG level RT signal is reached. This condition terminates the event automatically by closing the MFW isolation valve. Should that isolation valve fail, the HL and LL isolation valves also close on this signal.

The increase in feedwater flow event is classified as an AOO (described in Section 15.0.0.1) that is expected to occur with moderate frequency. The event potentially challenges SAFDLs. It also can cause overfill of the SG, which could force liquid water into the steam lines and potentially create a more serious plant condition.

15.1.2.2 Methods of Analysis and Assumptions

The analysis of the increase in feedwater flow event uses the approved non-LOCA analytical methodology described in Reference 1. The S-RELAP5 computer code (described in Section 15.0.2.4) is used to simulate the event and contains provisions to

model the primary and secondary systems, as well as the core reactivity response.

The low DNB channel algorithm and high LPD channel algorithm are simulated to predict RT and adequacy of the dynamic compensation of the algorithms consistent with Reference 2. The focus for this event is meeting the SAFDLs. The SAFDLs are satisfied by the combination of the low DNB and high LPD LCO and RT setpoint described in Reference 2.

To determine the limiting event scenario for the SAFDL, the minimum DNBR and maximum LPD, a spectrum of feedwater flow events is analyzed. The spectrum examines the range of initial conditions specified in Table 15.0-5 and Table 15.0-6. These initial conditions include the HFP, 60 percent rated thermal power (RTP), 25 percent RTP and hot zero power (HZP) levels, the time in cycle (BOC or EOC), with ACT control or MRC, availability of offsite power (with or without LOOP). The analyses are performed with zero percent SG tube plugging to maximize the primary to secondary heat transfer and the overcooling effect.

Operation with a feedwater heater train out of service is analyzed for increase of feedwater flow events. This reduces the initial feedwater temperature at each power level. An additional feedwater temperature reduction is conservatively utilized for each power level that is potentially more severe for this overcooling event.

Cases at BOC are analyzed with action by the non-safety-related ACT control function because this condition makes the results for BOC cases more severe. Cases at EOC conditions are analyzed with and without action by the non-safety-related ACT control function to determine which condition makes the results more severe.

For the purposes of analyzing this event, the most severe single failure is the failure of the HL isolation valve to close on the affected SG. There is no impact on the results of the transient analysis because there is an MFW isolation valve upstream of the HL isolation valve that also closes to terminate the MFW flow.

A spectrum of analyses for various power levels with zero percent SG tube plugging is performed to determine that HFP BOC ACT control with LOOP is the limiting case identified in Table 15.0-62. The limiting conditions are presented in Table 15.0-63.

The following limiting conditions apply to this increase in feedwater flow analysis:

- The temperature of the MFW to the four SGs is reduced 100°F to account for one train of high-pressure feedwater heaters being out of service.
- MFW flow to the affected SG increases instantaneously to a bounding 150 percent of the HFP flow rate, regardless of initial power level.
- SG tube plugging is neglected in order to maximize heat transfer from the RCS to the SGs.

- The SG high-level setpoint for RT and isolation of MFW is biased to 100 percent of the narrow range sensor, which bounds the maximum measurement uncertainty.

15.1.2.3 Results

This event does not challenge SAFDLs or overpressure criteria. Presented as a representative case, the scenario that causes the highest core power is a BOC HFP scenario in which the non-safety-related ACT control function is simulated. Table 15.1-4—Increase in Feedwater Flow - Key Input Parameters presents the initial conditions for this event. Table 15.1-5—Increase in Feedwater Flow - Key Equipment Status presents the status of equipment for this event. Table 15.1-6—Increase in Feedwater Flow - Sequence of Events provides the sequence of events. Figure 15.1-9—Increase in Feedwater Flow - Reactor Power through Figure 15.1-21—Increase in Feedwater Flow - Reactivity show the plant response. Figure 15.1-58—Increase in Main Feedwater Flow - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

The transient is initiated by instantaneously increasing MFW flow to the affected SG to 150 percent of HFP flow rate. This immediately causes a rising water level in the affected SG and an increase in the heat transfer rate, but a reduction in steam production in that SG. Steam flow from the unaffected SGs increases to compensate and maintain a constant steam flow to the turbine as the TCV opens. The increase in heat transfer, particularly in the economizer region, causes the affected loop cold-leg temperature to decrease. Because the scenario is at BOC, the colder water reaching the core does not cause core power to increase.

After about 25 seconds, the ACT control function begins to withdraw control rods to compensate for the reduction in RCS temperatures. Power rises slowly as the ACT control function adds reactivity. The increase in power causes negative reactivity insertion because of Doppler feedback. These two effects tend to counter each other.

The high SG level trip signal occurs at 187 seconds, thereby isolating MFW and terminating the transient. The resulting TT is assumed to cause a LOOP and start the coastdown of the RCPs. RCS temperature and pressure increase as secondary system pressure increases to the MSRT setpoint and the RCS system transitions to natural circulation. The plant is now in a stable controlled state.

15.1.2.4 Radiological Consequences

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs, nor is there a release of radioactive materials to the environment.

15.1.2.5 Conclusions

The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and the high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

The analyses performed demonstrate that the system transitions to a stable, controlled state, with the peak reactor coolant and main steam system pressures remaining below 110 percent of their respective design values for the duration of the transient.

15.1.2.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.1.2 events included in NUREG-0800, Section 15.1.1–15.1.1.4, (Reference 3) and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
   - Response: Reactor coolant and main steam system pressures are maintained below 110 percent of their respective design values for the duration of the event as concluded in Section 15.1.2.5.
2. Fuel cladding integrity shall be maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.
   - Response: Fuel cladding integrity is maintained as concluded in Section 15.1.2.5.
3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
   - Response: The event is mitigated and the plant is maintained in a stable condition through the automated response of safety-related equipment, thus avoiding an aggravated plant condition.
4. The requirements stated in RG 1.105, "Instrument Spans and Setpoints," are used with regard to their impact on the plant response to the type of AOOs addressed in this SRP section.
   - Response: The capability of the instrumentation and controls to meet the requirements of RG 1.105 is demonstrated in Section 7.1.3.4.7. The

instrument spans for the RT functions credited are provided in Table 7.2-1—Reactor Trip Variables.

5. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR 50, must be assumed in the analysis and should follow the guidance stated in RG 1.53.

   – Response: For 15.1.2 events, the only equipment credited is the RPS, which is designed single failure proof. Therefore, there are no single failure considerations for the 15.1.2 events.

6. Parameter values in the analytical model should be suitably conservative.

   A. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed operating plus an allowance of two percent to account for power measurement uncertainty unless a lower number can be justified through the measurement uncertainty methodology and evaluation or the uncertainty is accounted for otherwise (refer to SRP 4.4 as cited in Reference 3). 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 initial power level is rated output plus measurement uncertainty. The four loops are operating at the initiation of the event, as required by technical specifications.

   B. Conservative scram characteristics are assumed (e.g., maximum time delay with the most reactive rod held out of the core).

      • Response: Conservative scram characteristics were assumed.

   C. The core burnup is selected to yield the most limiting combination of MTC, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

      • Response: A conservative core burnup was selected for the analysis.

   D. Mitigating systems should be assumed as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 and as determined by the organization responsible for instrumentation and controls.

      • Response: The setpoints for the mitigating systems include instrument uncertainty.

15.1.3 Increase in Steam Flow

15.1.3.1 Identification of Causes and Event Description

The increase in steam flow event is an AOO that occurs when main steam flow is increased above the steady-state demand. The magnitude can range from a small increase in flow caused by the opening of the turbine control valves to a large increase caused by the simultaneous opening of the turbine bypass valves. The increase in flow can be initiated by operator action or failure of a control system that results in spurious operation of the bypass system or turbine control valves.

A small increase in main steam flow can cause the RCS to stabilize at a higher core power configuration that does not cause an RT. A greater increase in steam flow produces a power excursion that is sufficient to produce a low DNBR trip or high LPD trip that terminates the event. For even larger increases in steam flow, the rapid drop in pressure in the main steam lines trips the reactor on the low SG pressure or high SG pressure drop trip within a few seconds. The rapid trip precludes a large power excursion.

15.1.3.2 Methods of Analysis and Assumptions

The analysis of the increase in steam flow event uses the approved non-LOCA analytical methodology described in Reference 1. The S-RELAP5 computer code (described in Section 15.0.2.4) is used to simulate the event and contains provisions to model the primary and secondary systems, as well as the core reactivity response. The low DNB channel algorithm and high LPD channel algorithm are used for predicting RT and the adequacy of the dynamic compensation of the algorithm consistent with Reference 2.

To determine the limiting event scenario for the SAFDL, the minimum DNBR, and maximum LPD, a spectrum of steam flow increase events is analyzed. The spectrum examines the range of initial conditions specified in Table 15.0-5 and Table 15.0-6, including the HFP, 25 percent RTP and HZP power levels; time in the cycle (BOC or EOC); with ACT control or MRC; availability of offsite power (with or without LOOP); amount of steam flow increase; and with zero percent SG tube plugging to maximize the primary to secondary heat transfer and the overcooling effect.

The maximum steam flow increase corresponds to the opening of the six turbine bypass valves. A spectrum of steam flow rates are analyzed, ranging from the opening of one through six of the bypass valves. Operation with a feedwater heater train out of service is also analyzed for increase of steam flow events. This reduces the initial feedwater temperature at each power level. An additional feedwater temperature reduction is conservatively utilized for each power level that is potentially more severe for this overcooling event. The only mitigating equipment credited is the RPS. A failure in an PS division is inconsequential because the protection system is designed as

single failure proof due to its redundancy. There is no single failure which makes this event more severe.

A spectrum of analyses for various power level with zero percent SG tube plugging cases is performed to determine that the HFP EOC ARC with LOOP is the limiting case identified in Table 15.0-62. The limiting conditions are presented in Table 15.0-63.

The following limiting conditions apply to the increase in steam flow analysis:

- Respective bounding values of MTC are used for BOC and EOC.
- Cases are analyzed with and without action by the non-safety-related ACT control function because this condition could make the results more severe at BOC.
- A 100°F reduction in MFW temperature is assumed, consistent with a high-pressure MFW heater train out of service.
- The only mitigating equipment credited is the PS, which is designed as single failure proof. There is no single failure that makes the consequences of this event more severe.
- No operator actions are assumed.

15.1.3.3 Results

Table 15.1-7—Increase in Steam Flow - Key Input Parameters provides the initial conditions for this event. Table 15.1-8—Increase in Steam Flow - Key Equipment Status lists parameters for the protection system functions that may mitigate this event. Table 15.1-9—Increase in Steam Flow - Sequence of Events provides the sequence of events. Figure 15.1-22—Increase in Steam Flow - Reactor Power through Figure 15.1-32—Increase in Steam Flow - Reactivity (Detail) show system response to a representative increase in steam flow case. Figure 15.1-59—Increase in Steam Flow - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

The event is initiated by simulating the simultaneous opening of the six turbine bypass valves. This causes a rapid decrease in pressure in the SGs. The decreased pressure increases heat transfer from the RCS to the SG. This causes the RCS cold-leg temperatures to decrease and reactor power to increase because of the highly negative MTC. The decrease in RCS coolant temperature causes the ACT control function to withdraw control rods to attempt to raise the RCS average temperature. Rods are withdrawn for about two seconds and reactor power continues to increase until the event is terminated by a low DNBR RT.

The resulting TT is assumed to cause a LOOP that terminates MFW flow and starts the coastdown of the RCPs. The increase in secondary system pressure to the MSRT

setpoint and the transition to natural circulation causes the heatup and pressurization of the RCS. The plant is now in a stable controlled state.

15.1.3.4 Radiological Consequences

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs, nor is there a release of radioactive materials to the environment.

15.1.3.5 Conclusions

The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

The analyses performed demonstrate that the system transitions to a stable, controlled state, with the peak reactor coolant and main steam system pressures remaining below 110 percent of their respective design values for the duration of the transient.

15.1.3.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.1.3 events included in NUREG-0800, Section 15.1.1–15.1.1.4, (Reference 3) and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
   - Response: Reactor coolant and main steam system pressures are maintained below 110 percent of their respective design values for the duration of the event as concluded in Section 15.1.3.5.
2. Fuel cladding integrity is maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.
   - Response: Fuel cladding integrity is maintained as concluded in Section 15.1.3.5.
3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
   - Response: The event is mitigated and the plant is maintained in a stable condition through the automated response of safety-related equipment, thus avoiding an aggravated plant condition.

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

   – Response: The capability of the instrumentation and controls to meet the requirements of RG 1.105 is demonstrated in Section 7.1.3.4.7. The instrument spans for the RT functions credited are provided in Table 7.2-1—Reactor Trip Variables.

5. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR 50, must be assumed in the analysis and should follow the guidance stated in RG 1.53.

   – Response: For 15.1.3 events, the only equipment credited is the RPS, which is designed single-failure proof. Therefore, no single failures are considered for the 15.1.3 events.

6. Parameter values in the analytical model should be suitably conservative.

   A. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed operating plus an allowance of two percent to account for power measurement uncertainty unless a lower number can be justified through the measurement uncertainty methodology and evaluation or the uncertainty is accounted for otherwise (refer to SRP 4.4 as cited in Reference 3). 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 initial power level is rated output plus measurement uncertainty. The four loops are operating at the initiation of the event, as required by technical specifications.

   B. Conservative scram characteristics are assumed (e.g., maximum time delay with the most reactive rod held out of the core.

      • Response: Conservative scram characteristics were assumed.

   C. The core burnup is selected to yield the most limiting combination of MTC, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

      • Response: A conservative core burnup was selected for the analysis.

   D. Mitigating systems should be assumed as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 and as determined by the organization responsible for instrumentation and controls.

      • Response: The setpoints for the mitigating systems include instrument uncertainty.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

15.1.4.1 Identification of Causes and Event Description

The inadvertent opening of the SG relief or safety valve is an AOO that increases steam flow and thus causes a mismatch between the energy being generated in the reactor core and the energy being removed by the secondary system. This condition causes a cooldown of the primary system. An increase in power occurs if the moderator temperature reactivity feedback coefficient is negative or if the rod control system is in an automatic mode and begins to remove control rods from the reactor core in response to the decrease in RCS temperature resulting from the increased steam flow. If the power increase is sufficiently large, a low DNBR RT or high LPD RT is initiated.

Overpressure protection is provided on each of the four main steam lines by an MSRT and two main steam safety valves (MSSVs). Each MSRT consists of a single-failure proof main steam relief isolation valve (MSRIV) and a downstream main steam relief control valve (MSRCV). The discharge capacity of each MSRT is approximately 50 percent of the full load steam flow of the SG. The discharge capacity of each MSSV is approximately 25 percent. Although the inadvertent opening of a relief valve occurs in one SG, the four SGs pick up and share the extra load within a few seconds because their steam lines are connected at the turbine bypass header. Consequently, conditions in the RCS remain symmetric until RT and closure of the MSIVs.

The MSRIVs are fast opening valves that are normally closed. The MSRCVs are control valves that are fully open above 50 percent RTP. They close linearly from full open to 40 percent open as RTP decreases from 50 percent to 20 percent. Below 20 percent RTP they remain 40 percent open. The opening of the MSRIV automatically switches the MSRCV to control SG pressure to a high relief valve setpoint. It closes when SG pressure falls below that setpoint.

When the low SG pressure or high SG pressure decrease setpoint is reached, the PS initiates RT and closes the MSIVs. This isolates the affected SG from the other three unaffected SGs. When the low-low SG pressure or high-high SG pressure drop setpoint is reached, the PS isolates the MSRIV and the low-load feedwater line in the affected SG.

Should an MSRIV inadvertently open as postulated, the associated MSRCV terminates steam flow through the affected MSRT automatically when SG pressure decreases below its setpoint. The MSRCV requires 40 seconds to close. The automatic closure of the MSRCV in the affected MSRT terminates the event, but is postulated to fail open as the most severe single failure. This allows the blowdown of the affected SG to continue until dryout or closure of the MSRIV on the low-low SG pressure PS signal. The MSRIV might cycle between its opening and closing setpoints until the operator

initiates a cool down in the unaffected SGs to transition to RHR cooling. Once RCS temperature decreases, the MSRIV in the affected SG remains closed.

The evolution of the stuck open MSSV scenario is similar to that of the inadvertent opening of MSRIV except for two differences: the capacity of the MSSV is half that of the MSRT and the MSSV cannot be isolated.

The inadvertent opening of an MSRIV and an MSSV are both AOOs evaluated for the following acceptance criteria:

- The event challenges SAFDLs by DNBR and LPD prior to RT and, in the long-term by a return to criticality because of the continued cooldown of the RCS.
- The event also challenges radiological release limits.

15.1.4.2 Methods of Analysis and Assumptions

The analysis of the inadvertent opening of the SG relief valve or safety valve events uses the approved non-LOCA analytical methodology described in Reference 1. The S-RELAP5 computer code, described in Section 15.0.2.4, is used to simulate the event. This code simulates the primary and secondary systems, as well as the core reactivity response. The low DNB channel algorithm and high LPD channel algorithm is used for predicting the RT and the adequacy of the dynamic compensation of the algorithm consistent with Reference 2. The inadvertent opening of the MSSV and MSRIV are different enough scenarios that both are evaluated. Although the MSSV has half the capacity of an MSRT, it cannot be isolated.

To determine the limiting event scenario for the SAFDL criteria, the minimum DNBR, and maximum LPD, a spectrum of inadvertent opening of a steam generator relief or a safety valve events is analyzed. The spectrum examines the range of initial conditions specified in Table 15.0-5 and Table 15.0-6. These initial conditions include the power level (HFP, 25 percent RTP and HZP), time in the cycle (BOC or EOC), with ACT control or MRC, and availability of offsite power (with or without LOOP). The analyses are performed with zero percent SG tube plugging to maximize the primary to secondary heat transfer and the overcooling effect.

Operation with a feedwater heater train out of service is analyzed for inadvertent opening of a steam generator relief or safety valve events. This reduces the initial feedwater temperature at each power level. An additional feedwater temperature reduction is conservatively utilized for each power level.

The single failure assumed in the analysis of the inadvertent opening of an MSRIV event is that one MSRCV fails to close. Since the capacity of a safety valve is half that of a relief valve, the inadvertent opening of an MSRIV is more severe than the inadvertent opening of an MSSV. The worst single failure which makes the opening of

an MSRIV event more severe is an MSRCV failing to close. The failed MSRCV is associated with the open MSRIV and causes the event to continue longer.

A spectrum of analyses for various power levels with zero percent SG tube plugging is performed to determine HFP EOC MRC with LOOP is the limiting case identified in Table 15.0-62. The limiting conditions are presented in Table 15.0-63.

The following limiting conditions apply to the analysis of an inadvertent opening of an MSRIV:

- The initiating failure is a spurious signal that opens the MSRIV. This failure does not prevent its later closing when it receives a signal from the PS to close.
- The most severe single failure is the failure of the associated MSRCV to close and terminate the event.
- The temperature of the MFW to the four SGs is reduced to account for one train of high-pressure feedwater heaters being out of service.
- Respective bounding values of MTC are used for BOC and EOC.
- Cases are analyzed with and without action by the non-safety-related ACT control function because it could make the results more severe at BOC.

The analysis of the inadvertent opening of an MSSV is the same as for the inadvertent opening of an MSRIV except:

- The MSSV is not isolatable.
- The most severe single failure is the failure of an MSRCV to close in another SG.

15.1.4.3 Results

15.1.4.3.1 Inadvertent Opening of an MSRIV

This event is mitigated by the low DNBR RT or the high LPD RT. The scenario that generates the earliest RT is presented as a representative case. This is an EOC HFP scenario in which the non-safety-related ACT control function is deactivated. Table 15.1-10—Inadvertent Opening of an SG Relief or Safety Valve - Key Input Parameters provides the initial conditions for this case. Table 15.1-11—Inadvertent Opening of an SG Relief or Safety Valve - Key Equipment Status presents the status of equipment. Table 15.1-12—Inadvertent Opening of an SG Relief or Safety Valve - Sequence of Events presents the sequence of events. Figure 15.1-33—Inadvertent Opening of a SG Relief or Safety Valve - MSRT Flow Rate through Figure 15.1-43—Inadvertent Opening of a SG Relief or Safety Valve - TSV Position show system response for a representative case for an inadvertent actuation of the MSRT. Figure 15.1-60—Inadvertent Opening of a SG Relief or Safety Valve -

Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

The transient is initiated by instantaneously fully opening the MSRIV in the affected SG. The MSRCV already is fully open during HFP operation. This condition causes a rapid decrease in pressure in the affected SG as well as the unaffected SGs via the common steam line header, which increases heat transfer from the RCS. As cold-leg temperatures decrease, reactor power increases because of the highly negative MTC. The reactor power continues to increase until the event is terminated by RT on a low DNBR signal. The subsequent TT is assumed to cause a LOOP that terminates MFW flow and starts the coastdown of the RCPs. Following TT, pressure in the secondary system initially increases as the TSVs close. Continued steam flow from the open MSRT train cools both the secondary system and, thereby, the RCS. This cooling causes SG and RCS pressures to start decreasing about 10 seconds after RT.

At about 100 seconds, an SG high-pressure drop signal is generated and the PS closes the MSIVs. This action isolates the three unaffected SGs from the affected SG. The affected SG continues to blow down until the SG low-low pressure setpoint is reached that closes the MSRIV. The MSRIV in the affected SG continues to slowly cycle open and closed between PS setpoints until the operator initiates a cool down in the unaffected SGs to transition to RHR cooling. Once RCS temperature decreases, the MSRIV in the affected SG remains closed. The unaffected SGs provide long-term heat removal for core decay heat. The plant enters a stable controlled state. The unaffected SGs provide long-term heat removal for core decay heat. The plant enters a stable controlled state.

The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

This scenario also is analyzed at EOC HZP to determine the post-RT peak return to power. The analysis shows that the peak return to power for this event is 314 MW—a power level that does not cause fuel damage. Peak steam system and RCS pressures do not exceed 110 percent of design value.

15.1.4.3.2 Inadvertent Opening of an MSSV

The inadvertent opening of an MSSV scenario that has the greatest post-RT return to power is an EOC HZP case. The analysis shows a post-RT return to a power to 216 MW. This does not cause fuel damage. This scenario provides thermal-hydraulic

conditions for evaluating the radiological consequences of long-term steam release through the unisolable failed-open MSSV.

The inadvertent opening of an MSSV event is bounded by the inadvertent opening of an MSRT event in regard to SAFDLs.

15.1.4.4 Radiological Consequences

The inadvertent opening of an MSSV is the limiting AOO event for radiological consequences. The dose acceptance criteria for such events are defined in 10 CFR 50, Appendix I, for the summation of radioactive releases during normal operation and the annual average radioactive releases due to an AOO event based on realistic assumptions. The RCS and secondary coolant concentrations correspond to normal operating conditions, and are determined through application of the ANSI/ANS-18.1-1999 standard (Reference 4).

The radiological consequences are determined at the EAB critical receptor. The analysis conservatively assumes that the entire ingestion pathway is located at this distance and the exposure is continuous during the entire event. Subsequent exposure continues for an entire year thereafter, accounting for submersion, inhalation, ingestion and ground-shine pathways.

The worst-case organ dose is determined to be to the infant thyroid based on the dose conversion factors from Federal Guidance Report 11 (Reference 5) and amounts to a small percentage of the 10 CFR 50, Appendix I limit of 15 mrem.

15.1.4.5 Conclusions

The DNB RT and high LPD RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm and high LPD channel algorithm, are adequate to protect the SAFDL for the conditions that cause the low DNB channel or high LPD channel to issue an RT. For conditions where the DNB and LPD degradation do not cause an RT, both the DNB LCO and LPD LCO are adequate to protect the SAFDL. This demonstrates that both the fuel cladding integrity is maintained and peak centerline temperatures remain below the fuel centerline melt limit.

The analyses performed demonstrate that the system transitions to a stable, controlled state, with the peak reactor coolant and main steam system pressures remaining below 110 percent of their respective design values for the duration of the transient. Additionally, the analysis of post-RT consequences show that the peak return-to-power value is below the threshold where fuel failure occurs.

15.1.4.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.1.4 events included in NUREG-0800, Section 15.1.1–15.1.1.4, (Reference 3) and descriptions of how these criteria are met are listed below:

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

   – Response: Reactor coolant and main steam system pressures are maintained below 110 percent of their respective design values for the duration of the event as concluded in Section 15.1.4.5.

2. Fuel cladding integrity shall be maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.

   – Response: Fuel cladding integrity is maintained as concluded in Section 15.1.4.5.

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

   – Response: The event is mitigated and the plant is maintained in a stable condition through the automated response of safety-related equipment, thus avoiding an aggravated plant condition.

4. The requirements stated in RG 1.105, “Instrument Spans and Setpoints,” are used with regard to their impact on the plant response to the type of AOOs addressed in this SRP section.

   – Response: The capability of the instrumentation and controls to meet the requirements of RG 1.105 is demonstrated in Section 7.1.3.4.7. The instrument spans for the RT functions credited are provided in Table 7.2-1—Reactor Trip Variables.

5. The most limiting plant systems single failure, as defined in the “Definitions and Explanations” of Appendix A to 10 CFR 50, must be assumed in the analysis and should follow the guidance stated in RG 1.53.

   – Response: The limiting single failure was assumed as described in Section 15.1.4.2.

6. Parameter values in the analytical model should be suitably conservative.

   A. Initial power level is rated output (licensed core thermal power) for the number of loops initially assumed operating plus an allowance of two percent to account for power measurement uncertainty unless a lower number can be justified through the measurement uncertainty methodology and evaluation or the uncertainty is accounted for otherwise (refer to SRP 4.4 as cited in Reference 3). 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 initial power level is rated output plus measurement uncertainty. The four loops are operating at the initiation of the event, as required by technical specifications.

B. Conservative scram characteristics are assumed (e.g., maximum time delay with the most reactive rod held out of the core).

- Response: Conservative scram characteristics were assumed.

C. The core burnup is selected to yield the most limiting combination of MTC, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

- Response: A conservative core burnup was selected for the analysis.

D. Mitigating systems should be assumed as actuated in the analyses at setpoints with allowance for instrument uncertainty in accordance with RG 1.105 and as determined by the organization responsible for instrumentation and controls.

- Response: The setpoints for the mitigating systems include instrument uncertainty.

15.1.5 Steam System Piping Failures Inside and Outside of Containment (PWR)

The analysis of postulated failures of steam system piping are described in this section. The most severe scenario from a radiological standpoint is a break located in the valve room outside the Reactor Building but upstream of the MSIVs. Pipe failures inside the Reactor Building are not analyzed because their radiological consequences are less severe.

In accordance with the classification of events described in Section 15.0.0.1, minor steam system piping failures are considered AOOs, and larger breaks are treated as PAs. The analyses described in Section 15.1.5 address large main steam line breaks (MSLB), which are PAs.

15.1.5.1 Identification of Causes and Event Description

The rupture of a main steam pipe increases the rate of energy removal from the RCS and lowers RCS temperatures and pressures. Initially, the rate of steam flow in the failed pipe increases, but it subsequently decreases with time as the steam pressure drops. Because the SGs are connected via the turbine bypass header, the four SGs depressurize through the break. The four RCS loops cool down, although the rate is greater in the loop with the affected SG.

When the low SG pressure or high SG pressure decrease setpoint is reached, the PS initiates RT and closes the MSIVs. Because the break is assumed to be located upstream of an MSIV, their closure isolates the unaffected SGs from the break. When the low-low SG pressure or high-high SG pressure decrease setpoint is reached, the PS isolates the low-load feedwater line in the affected SG.

Once the MSIVs close, temperatures in the unaffected loops recover while temperature continues to decrease in the affected loop as the SG continues to blow down. Positive reactivity is inserted because of the assumed negative MTC. This condition erodes core shutdown margin, which can lead to re-criticality.

When the low-low SG level setpoint is reached in the affected SG, the PS actuates EFW to that SG. This action prolongs the release of steam through the break, the cooldown of the RCS, and the associated erosion of the shutdown margin. SAFDLs, including cladding strain, can be challenged if shutdown margin is eroded sufficiently for the reactor to return to power, particularly in conjunction with augmented radial power peaking near the stuck-out rod cluster control assembly (RCCA). Where SAFDLs are exceeded, the fuel is assumed to fail. The resulting fuel failure fractions are used to determine radiological consequences.

The event ends when the operator terminates EFW delivery to the affected SG, after which the RCS gradually reheats and reactor shutdown margin recovers.

15.1.5.2 Methods of Analysis and Assumptions

MSLB is analyzed using the approved computer codes and methods described in Reference 1. The system response to the MSLB event is analyzed using the S-RELAP5 computer code (see Section 15.0.2.4). This computer code simulates neutron kinetics, the RCS, the PZR, the SGs, the main steam lines and valves, and the feedwater system. The S-RELAP5 code computes system variables, including temperatures, pressures, flows, and power level. The LYNXT computer code (refer to Section 15.0.2.3) is used to determine the minimum DNBR for the event, and the PRISM computer code (refer to Section 15.0.2.1) is used to calculate the core power distributions. A spectrum of break size cases is analyzed for conservative combinations of core power levels, break locations (upstream and downstream of MSIV), offsite power conditions and single failures. Small break pre-scram events use the low DNB channel algorithm and high LPD channel algorithm for predicting the RT and the adequacy of the dynamic compensation of the algorithm, consistent with Reference 2.

To determine the limiting event, a spectrum of main steam line break events was performed. The parameters of interest examined in this spectrum were initial power level, break size and location, status of offsite power, and single failure.

A main steam line break event represents an uncontrolled cooldown of the RCS and can result in a return to power as a result of the addition of positive reactivity to the

reactor core by this cooldown. For the acceptance criteria, HZP represents the limiting initial power because the absence of appreciable decay heat exacerbates the RCS cooldown. EOC conditions are more limiting because they represent the most negative MTC. The availability of offsite power is more limiting because the operation of the RCPs enhances the RCS cooldown more than the RCP heat generation offsets it. The limiting combination of break location and single failure is identified by a sensitivity study. The limiting break size is influenced by the timing of MS isolation and MFW termination and can be identified only by a sensitivity study.

Because the event is aggravated by maximizing the RCS cooldown, it is assumed that the MSRCV on one of the unaffected main steam lines fails in the fully open position. With this assumption, when the unaffected SGs repressurize sufficiently following MSIV closure for their MSRIVs to open, the SG with the faulted MSRCV depressurizes more than it would otherwise, until the corresponding MSRIV closure setpoint is reached. This situation exacerbates the RCS cooldown and represents the worst-case single failure.

In summary, the limiting postulated MSLB event scenario is a 1.72 ft^2 break in a main steam line outside the Reactor Building, upstream of the MSIV, and at EOC and HZP conditions. Offsite power is assumed to be available and the MSRCV on one of the unaffected main steam lines is assumed to fail in the fully open position.

Additional assumptions for the analysis include the following:

- One train of each mitigating safety system is assumed to be unavailable due to maintenance.
- The highest worth RCCA is assumed to be located in the affected core sector and stuck in the fully withdrawn position, which reduces shutdown margin and augments power peaking.
- EOC conditions are assumed as they yield the most limiting combination of reactivity coefficients and control rod worth.
- Mixing of fluid between core sectors is modeled as described in Reference 1.
- MFW flow is treated conservatively as described in Reference 1.

15.1.5.3 Results

The limiting MSLB event scenario is initiated at EOC HZP conditions by a postulated 1.72 ft^2 break in a main steam line outside the Reactor Building. Offsite power remains available to operate the RCPs. Table 15.1-13—MSLB - Key Input Parameters for Limiting Case presents the input for the limiting case. Table 15.1-14—MSLB - Key Equipment Status for Limiting Case presents the status of key plant equipment and systems. Table 15.1-15—MSLB - Sequence of Events presents the sequence of events

for the limiting case. Figure 15.1-44—MSLB - Break Flow Rate through Figure 15.1-56—MSLB - Longer-Term Reactor Power present plots of key system variables. Figure 15.1-61—MSLB (small break, pre-scram) - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL presents a representative case of DNB and LPD normalized to their respective SAFDLs.

The MSLB rapidly reduces pressure in the four SGs generating an SG high pressure drop signal that closes the MSIVs. This closure isolates the three unaffected SGs from the break. As the inventory of the affected SG flashes and is discharged through the break, the RCS cools down, adding positive reactivity. The RCS cooldown is exacerbated once the MSRIVs on the unaffected SGs open, permitting uncontrolled steam release from the unaffected SG, whose MSRCV is assumed to fail in the fully open position. The resultant RCS cooldown and associated reactivity addition causes the reactor to return to critical. As the affected SG approaches dry out, reactor power peaks at approximately 23 percent RTP and begins to decrease. The limiting conditions with respect to SAFDL acceptance criteria occur at the time of peak reactor power.

Following the power peak, reactor power stabilizes at approximately three percent RTP due to the stabilization of the core inlet temperature of the affected core region. Upon the injection of MHSI that results from the RCS depressurization, the core power decreases and again stabilizes at approximately 2.5 percent RTP. The power decrease is the result of the increased boron concentration in the core region from the MHSI flow. RCS pressure begins to recover after the MSRIV of the failed open MSRCV SG closes, halting the additional primary system overcooling. MHSI injection begins decreasing at this point due to the approach to the MHSI shutoff head. The boron injected by the MHSI does not contribute to the mitigation of the event.

The event is terminated when the operator stops EFW delivery to the affected SG and it dries out. The RCS gradually reheats, reactor shutdown margin recovers and the plant enters a stable controlled condition with heat removal via the unaffected SGs.

The analysis results indicate that the fuel cladding strain limits for AOOs are exceeded for this PA, causing a small number of fuel failures. Table 15.1-16—MSLB - Calculated Fuel Parameters provides the calculated fuel parameters for the limiting case.

15.1.5.4 Radiological Consequences

Radiological consequences for the MSLB event are described in Section 15.0.3.7.

15.1.5.5 Conclusions

The results presented in Section 15.1.5.3 demonstrate that RCS and main steam system pressures are maintained below acceptable design limits. Analysis of the post-RT

consequences shows that there is only limited fuel damage. The radiological consequences are within the limits of 10 CFR 50.34(a)(1) and 10 CFR 50, Appendix A.

15.1.5.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.1.5 events included in NUREG-0800, Section 15.1.5, (Reference 3) and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

   – Response: Steam line break events do not challenge pressure criteria. RCS and main steam system pressures remain below acceptable design limits.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations. If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for the rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (which includes the potential adverse effects of hydraulic instabilities), that fewer failures occur. Fuel damage that is 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: Section 15.1.5.3 presents the results of the analysis of the steam line break scenario that produces the most severe consequences for fuel failure. The analysis demonstrates that the extent of fuel failure is limited so that the core remains in place and intact with no loss of core cooling capability.

3. The radiological consequences meet the requirements of the following regulations:

   A. Section 50.34(a)(1) of 10 CFR 50, “Contents of applications; technical information,” as it relates to the evaluation and analysis of the offsite radiological consequences of PAs with fission product release.

      • Response: The radiological consequences of MSLB events are within the limits of 10 CFR 50.34(a)(1) and 10 CFR 50, Appendix A (see Sections 15.0.3.1 and 15.0.3.2).

   B. GDC 19 of Appendix A to 10 CFR 50, “Control room,” as it relates to maintaining the control room in a safe condition under accident conditions by providing adequate protection against radiation.

      • Response: The radiological consequences of MSLB events are within the limits of 10 CFR 50.34(a)(1) and 10 CFR 50, Appendix A (see Sections 15.0.3.1 and 15.0.3.2).

4. The integrity of the RCPs should be maintained such that loss of ac power and containment isolation does not result in pump seal damage.

– Response: As described in Section 5.4.1.2.1, RCP seal integrity is maintained in the event of a loss of seal cooling or isolation of seal leak-off lines, both of which could occur because of an MSLB event.

5. The auxiliary feedwater system or other means of decay heat removal must be safety related and, when required, automatically initiated.

   – Response: The EFW system is safety related and is actuated automatically by the safety related PS. Refer to Section 10.4.9 for a description of the removal of decay heat.

6. Tripping of the RCPs should be consistent with the resolution to Task Action Plan item II.K.3.5.

   – Response: The PS automatically trips the RCPs when a safety injection system signal is generated and the pressure differential across the RCPs decreases below a setpoint value. These conditions trip the RCPs early during an SBLOCA event. These conditions do not occur during an MSLB so the RCPs are not tripped.

15.1.6 References

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

2. ANP-10287P, Revision 0, “Incore Transient Methodology Topical Report,” AREVA NP, Inc., November 2007.

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

4. ANS/ANSI-18.1-1999, “American National Standard-Radioactive Source Term for Normal Operation of Light Water Reactors,” American Nuclear Society, September 21, 1999.

5. EPA 520/1-88-020, Federal Guidance Report No. 11, “Limiting Values of Radionuclide Intake and Air Concentration, and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” U.S. Environmental Protection Agency, September 1988.

Table 15.1-1—Decrease in Feedwater Temperature - Key Input Parameters

Parameter	Analysis Value
Reactor power	4612 MW_t
RCS loop volumetric flow rate	119,692 gpm
Reactor vessel average temperature	594°F
PZR pressure	2250 psia
PZR level	54.3%
Main steam pressure	1113 psia
MFW flow rate	1444 lb_m/s
MFW temperature	446°F
SG level	48.9%
SG tube plugging	0%
MSRT opening setpoints	1414.7 psia
Pilot-operated PSRV opening setpoints	2538.20 psia 2615.45 psia 2692.70 psia
Fuel-to-cladding gap conductance	13,445.20 BTU/(hr-ft^2-°F)
MTC	-50 pcm/°F
Doppler reactivity feedback	-1.512 pcm/°F
Scram worth	-7353 pcm

Table 15.1-2—Decrease in Feedwater Temperature - Key Equipment Status

Plant Equipment or System	Status
RT RCCAs	Most reactive RCCA stuck out of core
PZR heaters	Not credited
MSRTs	Available
MSIVs	Functional
RCPs	Operating
Feedwater pumps	Operating
EFW	Available
MHSI pumps	One MHSI pump out of service for maintenance; remaining three MHSI pumps available
PSRVs	Available

Table 15.1-3—Decrease in Feedwater Temperature - Sequence of Events

Event	Time (s)
Decrease in MFW temperature to the SGs	0.0
Low DNBR trip signal	15.1
Low DNBR trip	15.6
RT signal	15.7
MFW isolation signal	15.7
TT signal	15.7
RCPs trip	15.7
Peak reactor power (106.1%) occurs	15.8
RT (rod drop) time	15.9
MFW flows terminate	55.7
Minimum PZR level (31%)	58.3
Peak RCS pressure (2539 psia)	283.7
Calculation terminated	300.0
Low setpoint PSRV actuates (2538.20 psia)	283.8
Low setpoint PSRV closes (2388.82 psia)	290.8

Table 15.1-4—Increase in Feedwater Flow - Key Input Parameters

Parameter	Analysis Value
Reactor power	4612 MW_t
RCS loop volumetric flow rate	119,692 gpm
Reactor vessel average temperature	594°F
PZR pressure	2250 psia
PZR level	54.3%
Main steam pressure	1176 psia
Main steam/feed flow rate	1267 lb_m/s
MFW temperature	346°F
SG level	48.9%
SG tube plugging	0%
MSRT opening setpoints	1414.7 psia
Pilot-operated PSRV opening setpoints	2538.20 psia 2615.45 psia 2692.70 psia
MTC	-0 pcm/°F
Doppler reactivity feedback	-1.260 pcm/°F
Scram worth	-6161 pcm

Table 15.1-5—Increase in Feedwater Flow - Key Equipment Status

Plant Equipment or System	Status
RT RCCAs	Most reactive RCCA stuck out of core
PZR heaters	Not credited
MSRTs	Available
MSIVs	Functional
RCPs	Operating
Feedwater pumps	Operating
EFW	Available
MHSI pumps	One MHSI pump out of service for maintenance; remaining three MHSI pumps available

Table 15.1-6—Increase in Feedwater Flow - Sequence of Events

Event	Time (s)
Increase in MFW flow to SG Four	0.0
ACT control function begins withdrawing control rods	25
High SG level trip	186.6
Peak reactor power (105.1%)	187.5
RT	187.6
TT	188.2
MFW isolation	188.2
All MSRTs Open	214
All MSRTs Shut	238
Calculation terminated	500

Table 15.1-7—Increase in Steam Flow - Key Input Parameters

Parameter	Analysis Value
Reactor power	4612 MW_t
RCS loop volumetric flow rate	119,692 gpm
Reactor vessel average temperature	594°F
PZR pressure	2250 psia
PZR level	54.3%
Main steam pressure	1176 psia
Main steam/feed flow rate	1267 lb_m/s
MFW temperature	346°F
SG level	48.9%
SG tube plugging	0%
MSRT opening setpoints	1414.7 psia
Pilot-operated PSRV opening setpoints	2538.20 psia 2615.45 psia 2692.70 psia
Fuel-to-cladding gap conductance	13445.20 BTU/(hr-ft^2-°F)
MTC	-50 pcm/°F
Doppler reactivity feedback	-1.512 pcm/°F
Scram worth	-7353 pcm

Table 15.1-8—Increase in Steam Flow - Key Equipment Status

Plant Equipment or System	Status
RT RCCAs	Most reactive RCCA stuck out of core
PZR heaters	Not credited
MSRTs	Available
MSIVs	Functional
RCPs	Operating
Feedwater pumps	Operating
EFW	Available
MHSI pumps	One MHSI pump out of service for maintenance; remaining three MHSI pumps available

Table 15.1-9—Increase in Steam Flow - Sequence of Events

Event	Time (s)
Increase in steam flow to the SGs	0.0
Low DNBR trip signal	7.1
TT	7.8
MFW isolation	7.8
RT	7.9
Peak reactor power (108.7%)	7.9
MSIVs shut	7.9
Calculation terminated	300

Table 15.1-10—Inadvertent Opening of an SG Relief or Safety Valve - Key Input Parameters

Parameter	Analysis Value
Reactor power	4612 MW_t
RCS loop volumetric flow rate	119,692 gpm
Reactor vessel average temperature	594°F
PZR pressure	2250 psia
PZR level	54.3%
Main steam pressure	1176 psia
Main steam/feed flow rate	1267 lb_m/s
MFW temperature	346°F
SG level	48.9%
SG tube plugging	0%
MSRT opening setpoints	1414.7 psia
Pilot-operated PSRV opening setpoints	2538.20 psia 2615.45 psia 2692.70 psia
Fuel-to-cladding gap conductance	13445.20 BTU/(hr-ft^2-°F)
MTC	-50 pcm/°F
Doppler reactivity feedback	-1.512 pcm/°F
Scram worth	-7353 pcm

Note:

1. Parameter is nominal for the decreased MFW temperature value.

Table 15.1-11—Inadvertent Opening of an SG Relief or Safety Valve - Key Equipment Status

Plant Equipment or System	Status
RT RCCAs	Most reactive RCCA stuck out of core
PZR heaters	Not credited
MSRTs	MSRTs for unaffected SGs available—SG 3 MSRCV fails to close
MSIVs	Functional
RCPs	Operating
Feedwater pumps	Operating
EFW	Available
MHSI pumps	One MHSI pump out of service for maintenance; remaining three MHSI pumps available

Table 15.1-12—Inadvertent Opening of an SG Relief or Safety Valve - Sequence of Events

Event	Time (s)
Inadvertent opening MSRIV of SG 3 and failure of the associated MSRCV to close	0.0
Low DNBR trip signal	6.2
Low DNBR trip	6.7
TT signal	6.9
MFW isolation signal	6.9
RCP Trip	6.9
Peak reactor power (108.0%)	7.0
RT	7.0
Peak RCS pressure (2291 psia)	9.7
MSIVs shut	100
Minimum PZR level (22.2%)	200

Table 15.1-13—MSLB - Key Input Parameters for Limiting Case

Parameter	Analysis Value
Initial reactor power	1 W
Initial RCS loop flow rate	119,692 gpm/loop
Initial vessel average temperature	578°F
Initial PZR pressure	2250 psia
Initial PZR liquid level	34.0% of span
Initial SG pressure	1290.6 psia
Initial SG secondary-side total fluid inventory	240,649 lb_m/SG
MFW temperature	250°F
Initial shutdown margin	3000 pcm
Moderator reactivity feedback	EOC curve, biased to support -50 pcm/°F most-negative full-power MTC limit
EFW actuation for affected SG	At event initiation
EFW flow rate	572 gpm (to affected SG)
EFW temperature	50°F
High MS pressure decrease MSIV closure setpoint	$P_{initial}$–177 psia–29 psia/min
MSIV closure time	5 s (after signal)
High-high MS pressure decrease MFW-LL isolation setpoint	$P_{initial}$–322 psia–29 psia/min
MFW-LL isolation time	20 s (after signal)

Table 15.1-14—MSLB - Key Equipment Status for Limiting Case

Plant Equipment or System	Status
RT RCCAs	Most reactive RCCA stuck out of core
PZR heaters	Not credited
MSRTs	MSRTs for unaffected SGs available—with one of their MSRCVs failed open
MSIVs	Functional
RCPs	Operating
Feedwater pumps	Operating
EFW	Available
MHSI pumps	One MHSI pump out of service for maintenance; remaining three MHSI pumps available

Table 15.1-15—MSLB - Sequence of Events
Sheet 1 of 2

Event	Time (s)
Break occurs in main steam line upstream of MSIV.	0.0
Affected SG EFW delivery begins coincident with break.	0.0
Affected MS line pressure reaches “high MS pressure decrease” setpoint actuating MS isolation.	6.9
MSIVs begin to close.	7.4
Affected MS line pressure reaches “high-high MS pressure decrease” setpoint for isolating MFW from affected SG.	11.9
Isolation of MFW from affected SG is actuated.	12.4
MSIVs are fully closed.	12.4
Affected SG MFW isolation valves are fully closed.	32.4
Unaffected MS line pressures reach MSRTs opening setpoint.	≈243
Unaffected MSRIVs open, with MSRCV for MS line 1 assumed to be failed in fully open position and MSRCVs for MS lines 2 and 3 beginning to close (attempting to maintain pressures at setpoint).	245.0 to 246.0
MSRCVs for MS lines 2 and 3 reach closed positions.	≈262
Affected SG begins to dry out (as signified by sudden decrease in primary-to-secondary heat transfer rate, in conjunction with very low secondary-side liquid inventory).	264.8 to 265.0
Affected core sector inlet temperature reaches minimum and then begins to increase.	273.0
Reactor power peaks (23.14% of RTP) and then begins to decrease.	273.2
Unaffected SG 1 liquid level reaches “low-low SG level” EFW actuation setpoint.	418.6
Unaffected SG 1 EFW delivery begins.	434.1
PZR pressure reaches “low-low PZR pressure” setpoint actuating MHSI and partial depressurization of unaffected SGs.	457.6
MHSI is actuated, and partial depressurization of unaffected SGs begins.	458.7

Table 15.1-15—MSLB - Sequence of Events
Sheet 2 of 2

Event	Time (s)
Credited MHSI pumps are running at full speed but do not yet begin filling their SI lines with borated fluid, because RCS cold-leg pressures are well above MHSI pump shutoff head.	473.7
RCS cold-leg pressures reach MHSI pump shutoff head, and credited MHSI pumps begin filling their SI lines with borated fluid.	595.0 to 597.0
Borated fluid has filled credited SI lines and begins to enter RCS cold legs.	720.5 to 722.6
Unaffected MS line 1 pressure reaches MSRT closing setpoint.	792.6
Unaffected SG 1 MSRIV closes.	795.0 to 799.0
Unaffected core sector inlet temperature reaches minimum and then begins to increase.	802.0
Operator takes control of plant to bring it to safe shutdown condition.	1807.4

Table 15.1-16—MSLB - Calculated Fuel Parameters

Parameter	Value
Peak post-scram reactor power	1062.3 MWt
Peak LHGR	19.1 kW/ft
FCM fuel failure	0.00%
Fuel failure at clad strain limit	1.24%

Figure 15.1-1—Decrease in Feedwater Temperature - Main Feedwater Flow Rate

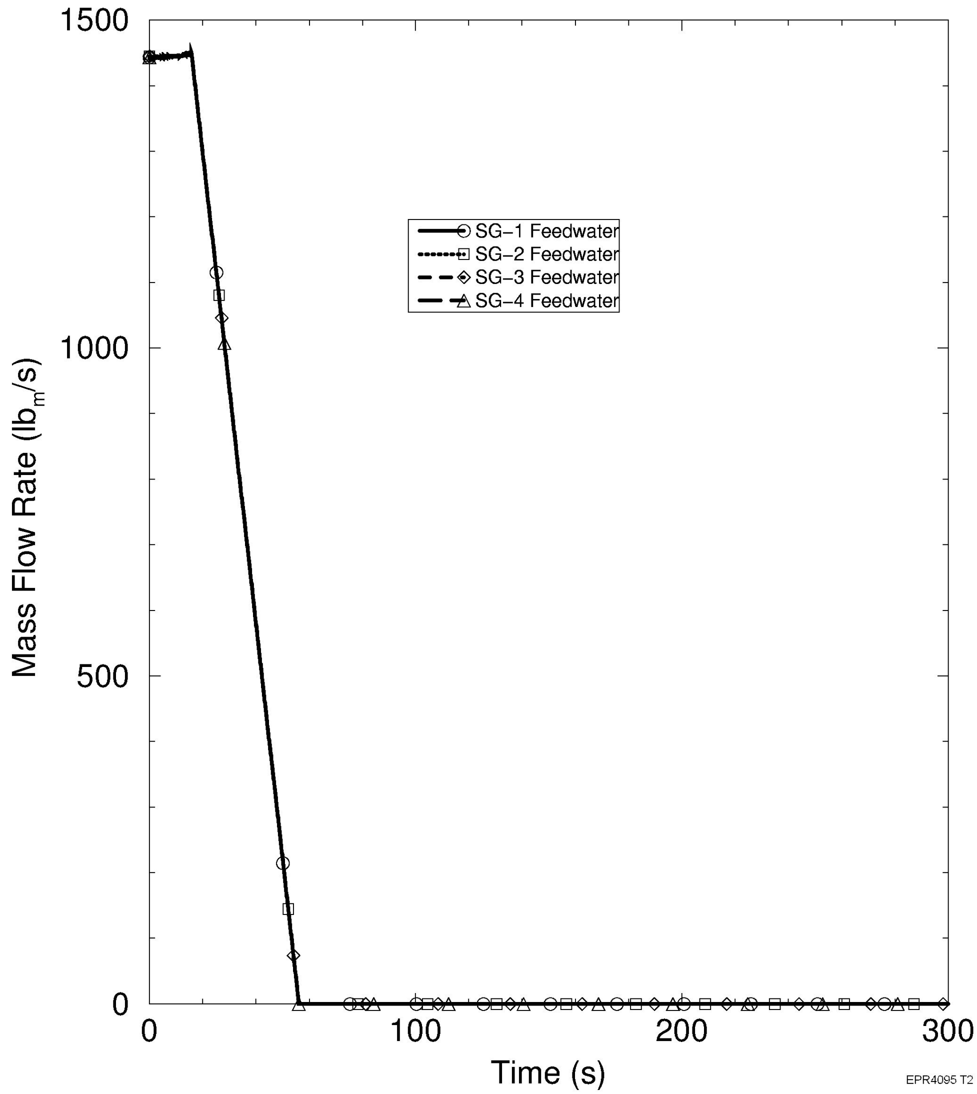

Figure 15.1-2—Decrease in Feedwater Temperature - Steam Generator Narrow Range Liquid Level

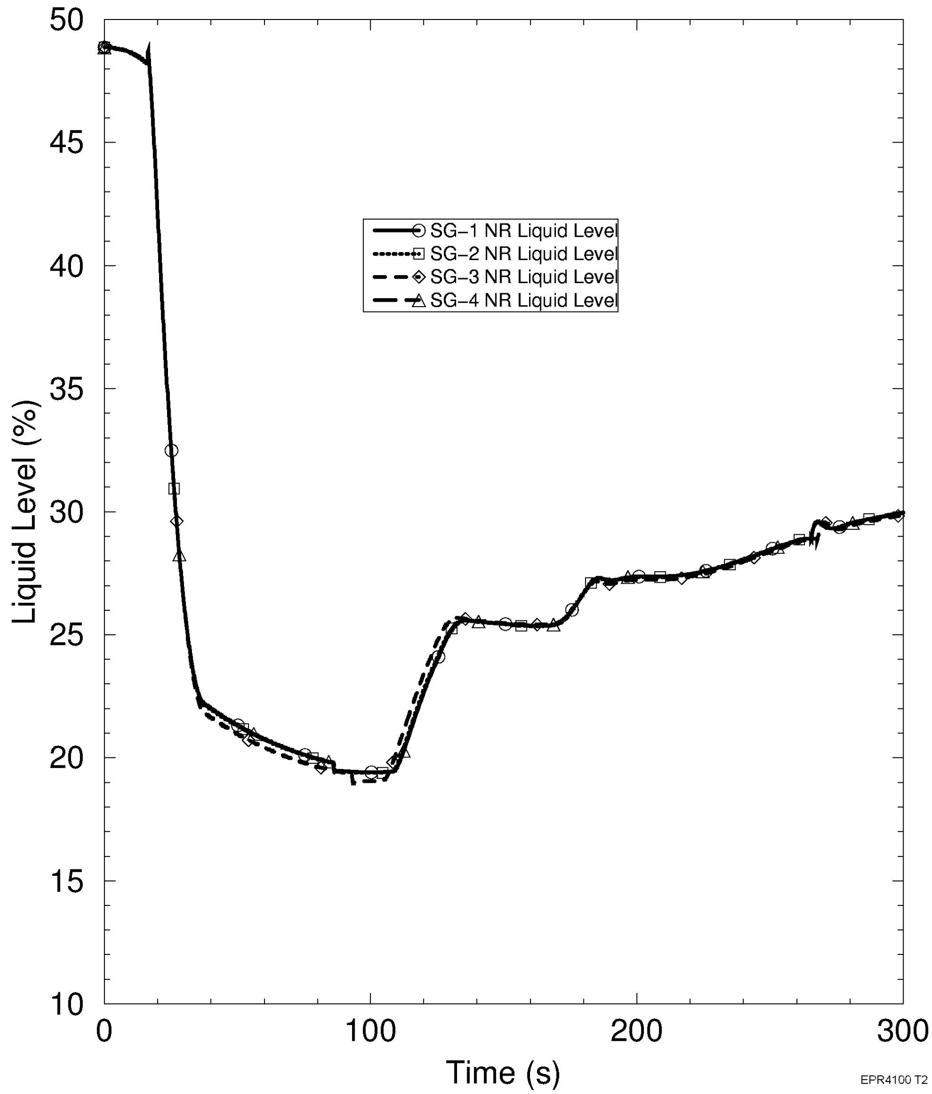

Figure 15.1-3—Decrease in Feedwater Temperature - Heat Transfer to Steam Generators

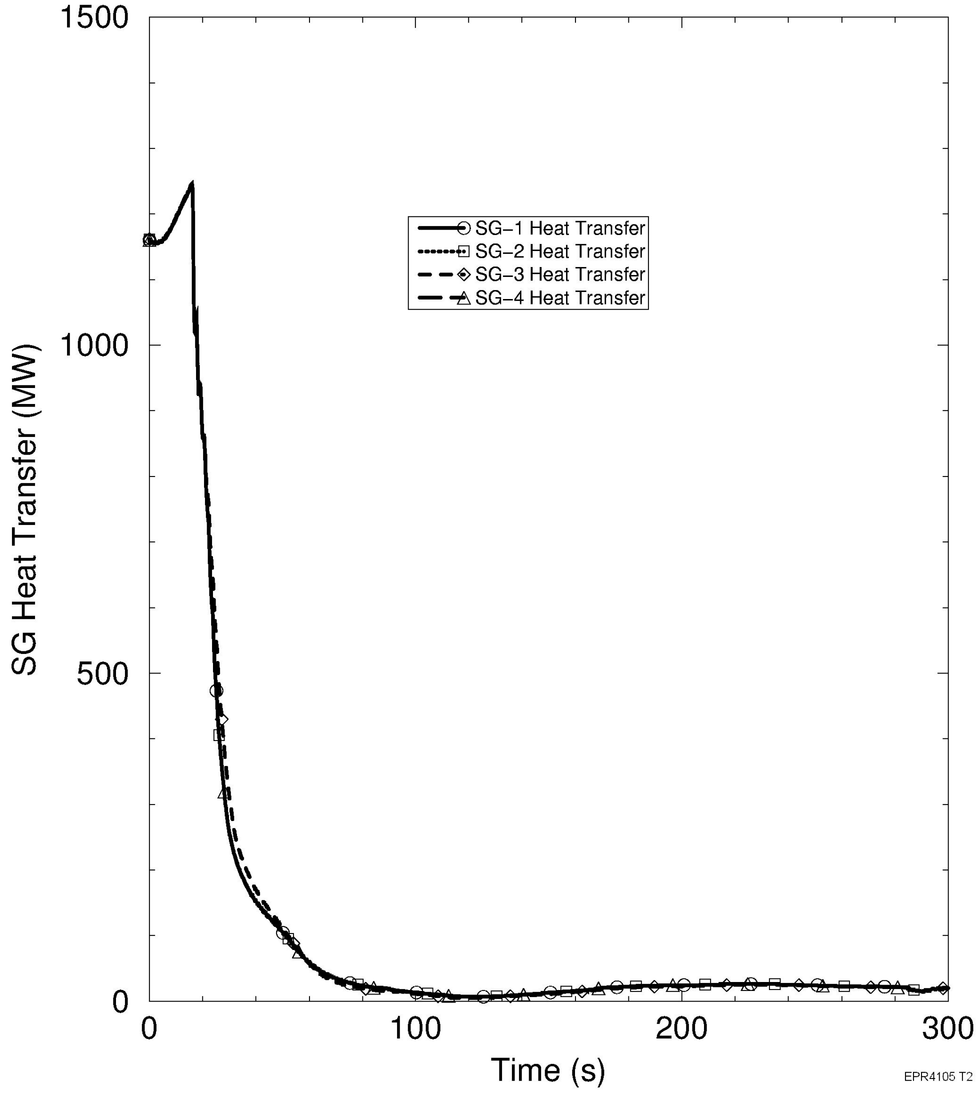

Figure 15.1-4—Decrease in Feedwater Temperature - Cold Leg Temperatures

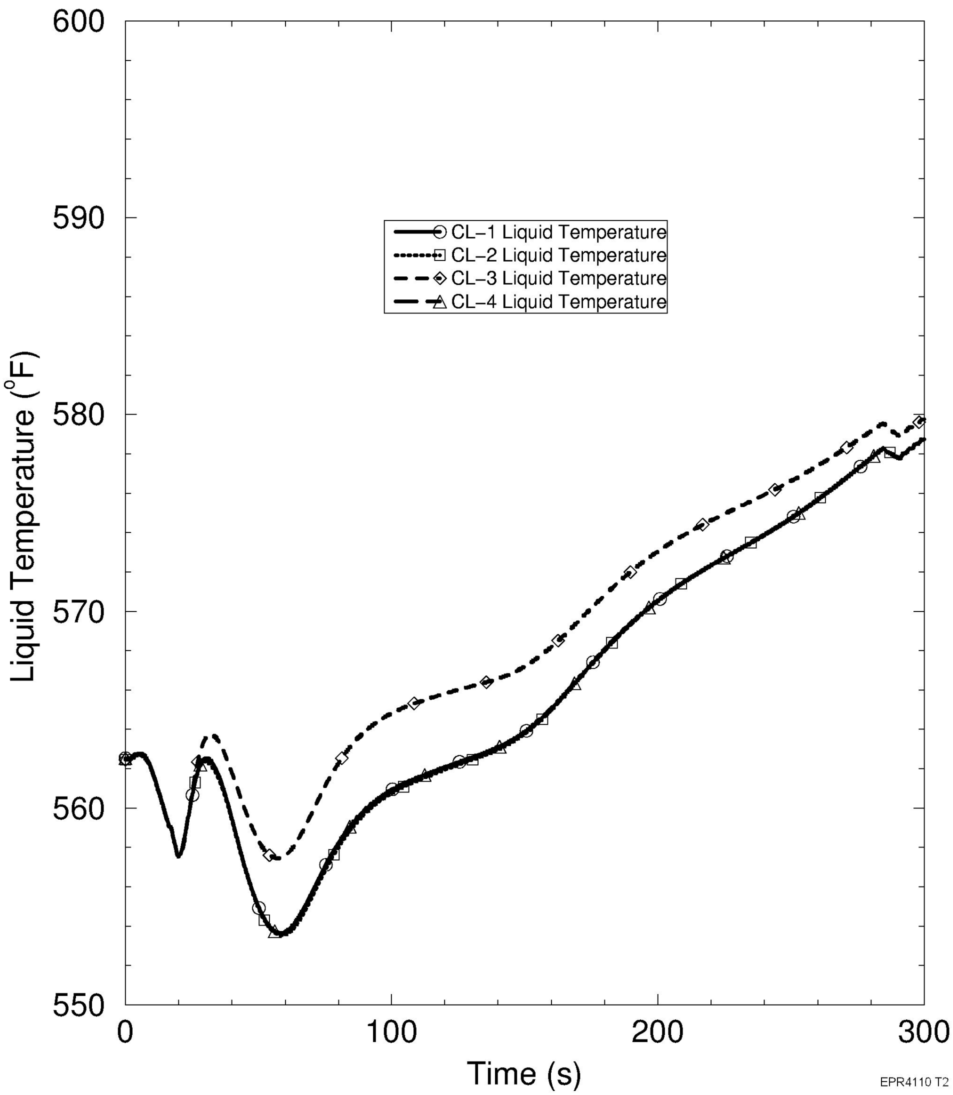

Figure 15.1-5—Decrease in Feedwater Temperature - Reactor Power

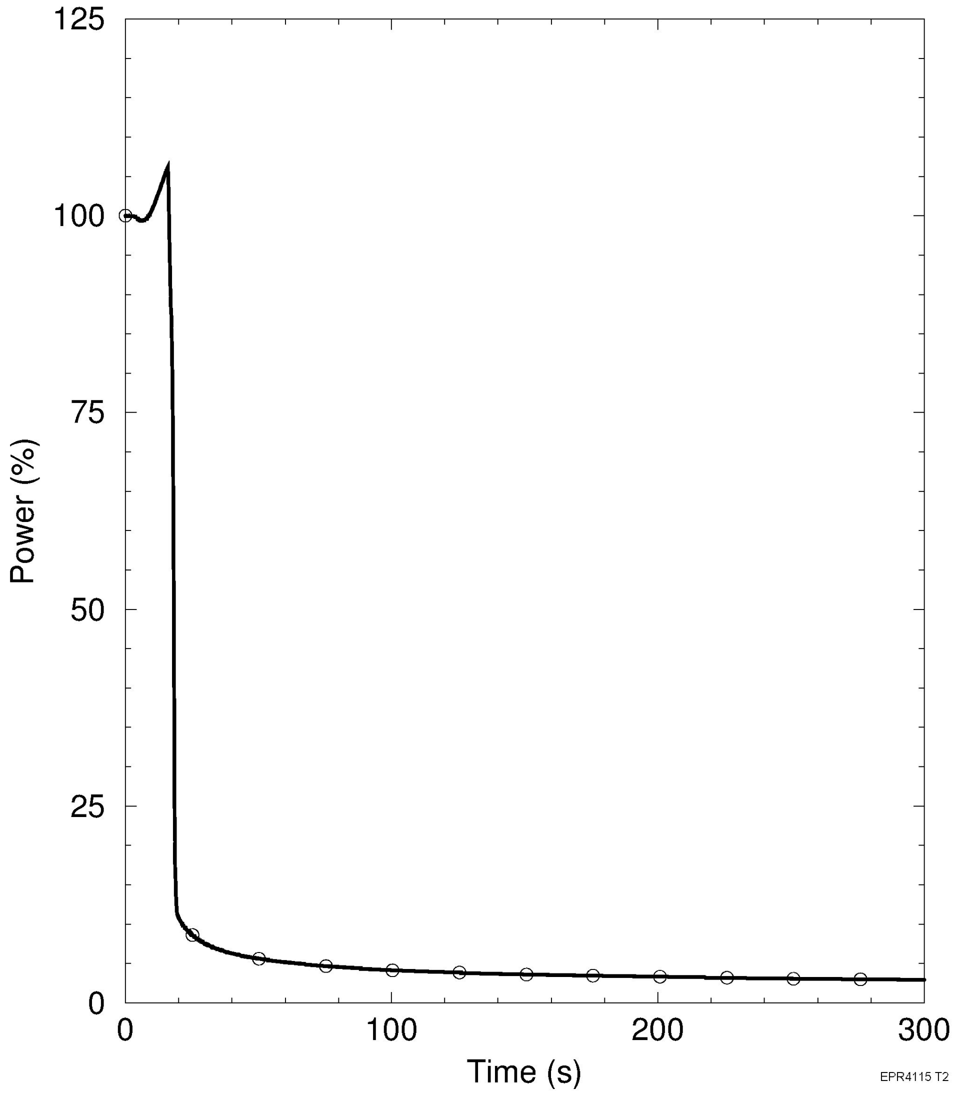

Figure 15.1-6—Decrease in Feedwater Temperature - Reactivity

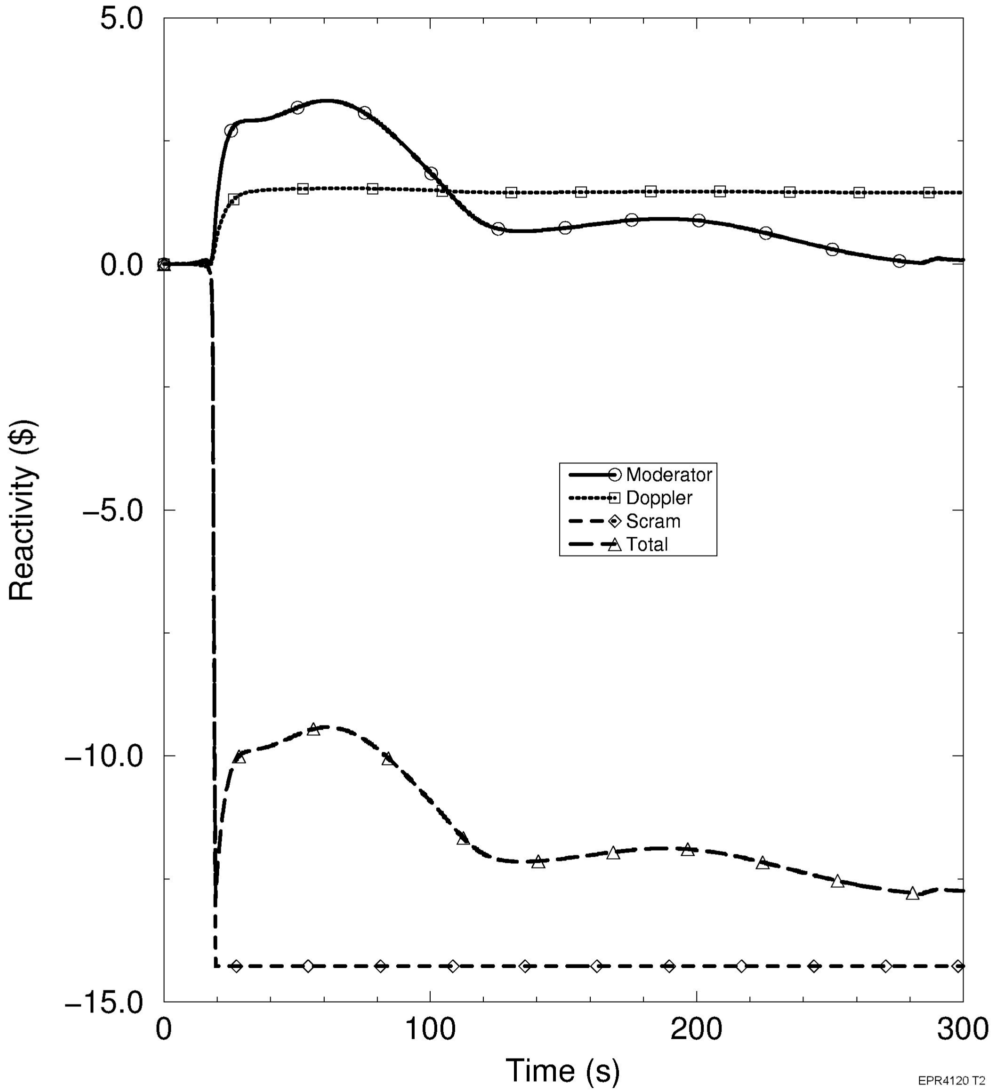

Figure 15.1-7—Decrease in Feedwater Temperature - RCS and Secondary System Pressures

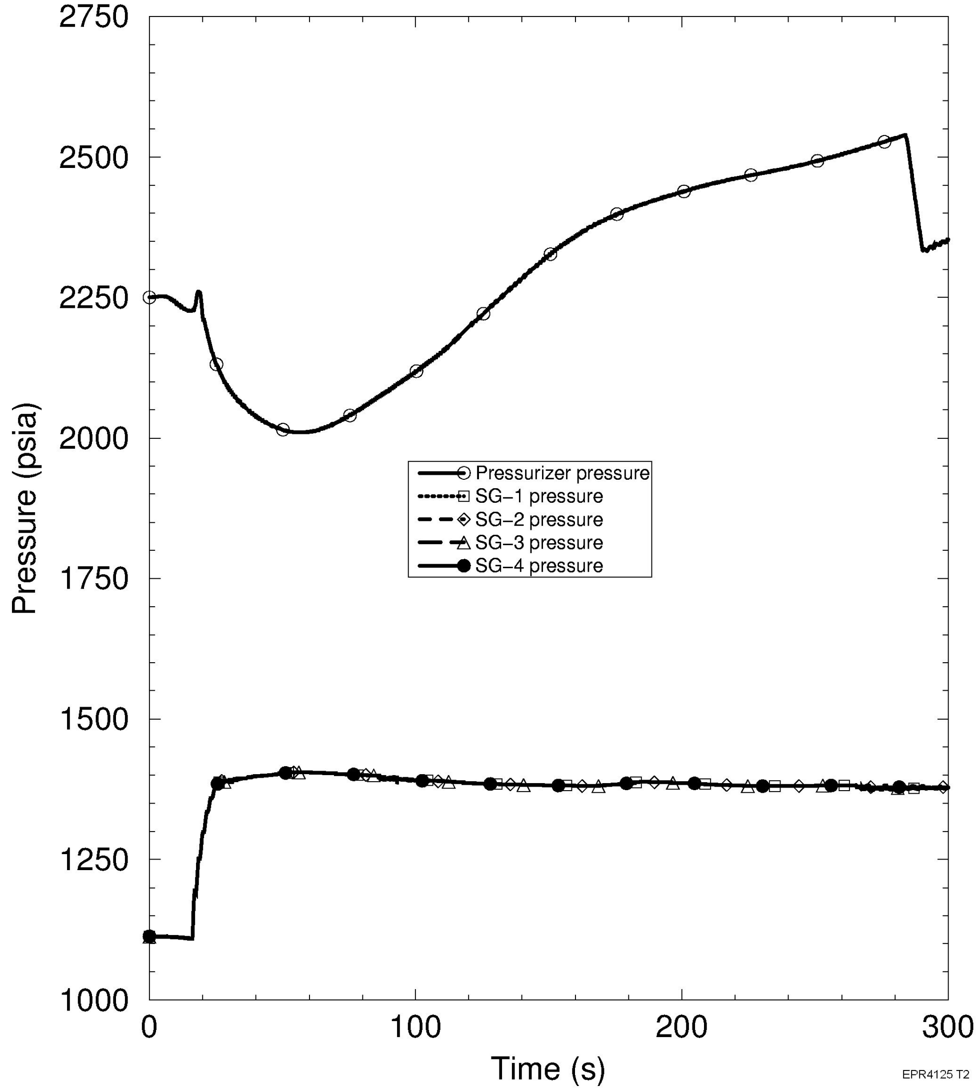

Figure 15.1-8—Decrease in Feedwater Temperature - Pressurizer Level

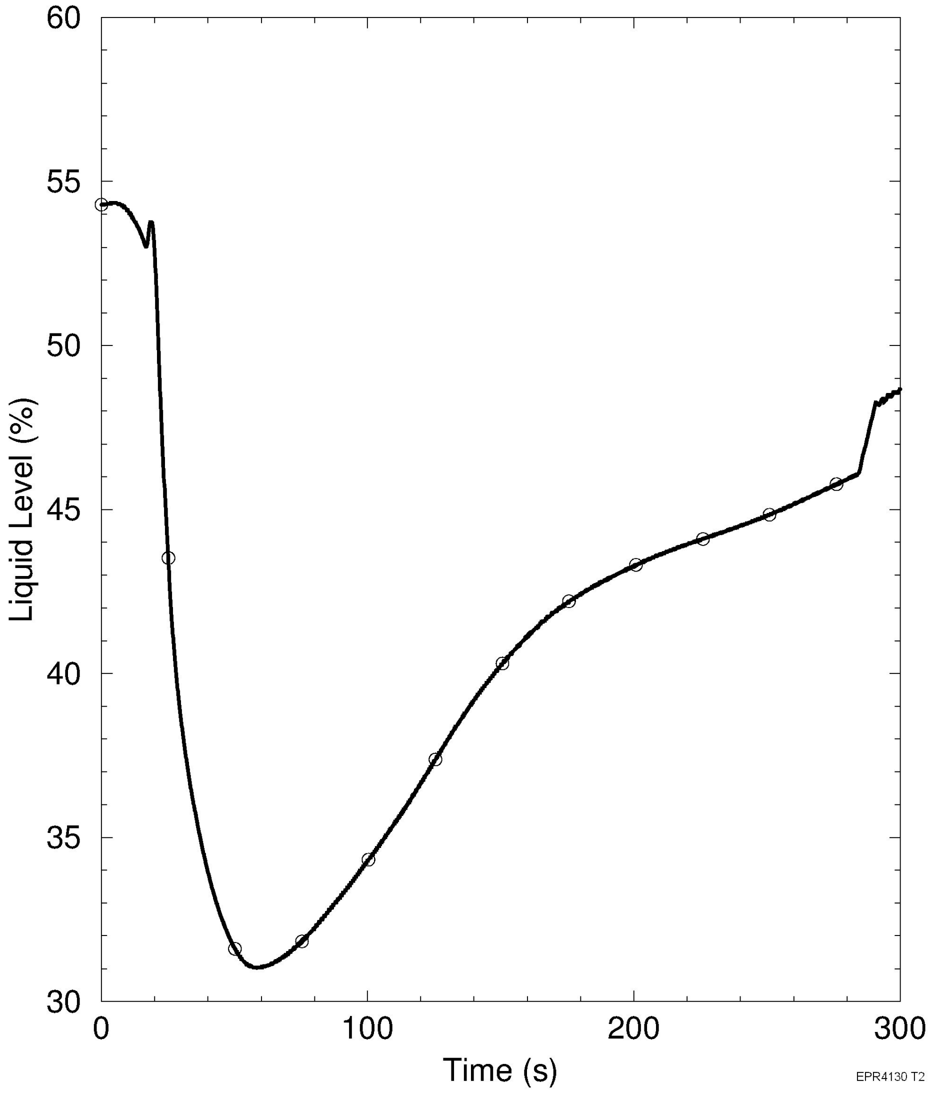

Figure 15.1-9—Increase in Feedwater Flow - Reactor Power

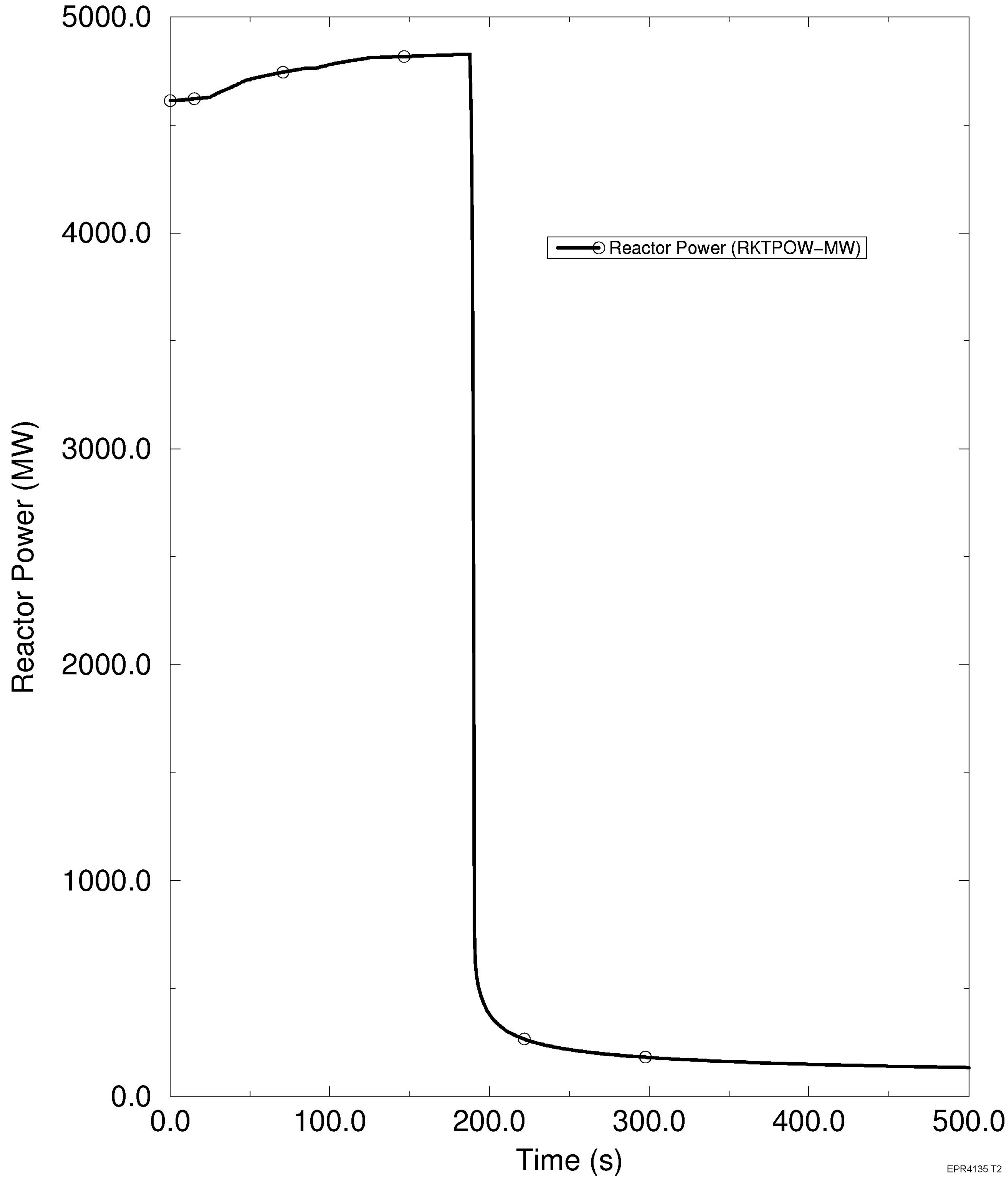

Figure 15.1-10—Increase in Feedwater Flow - Reactivity

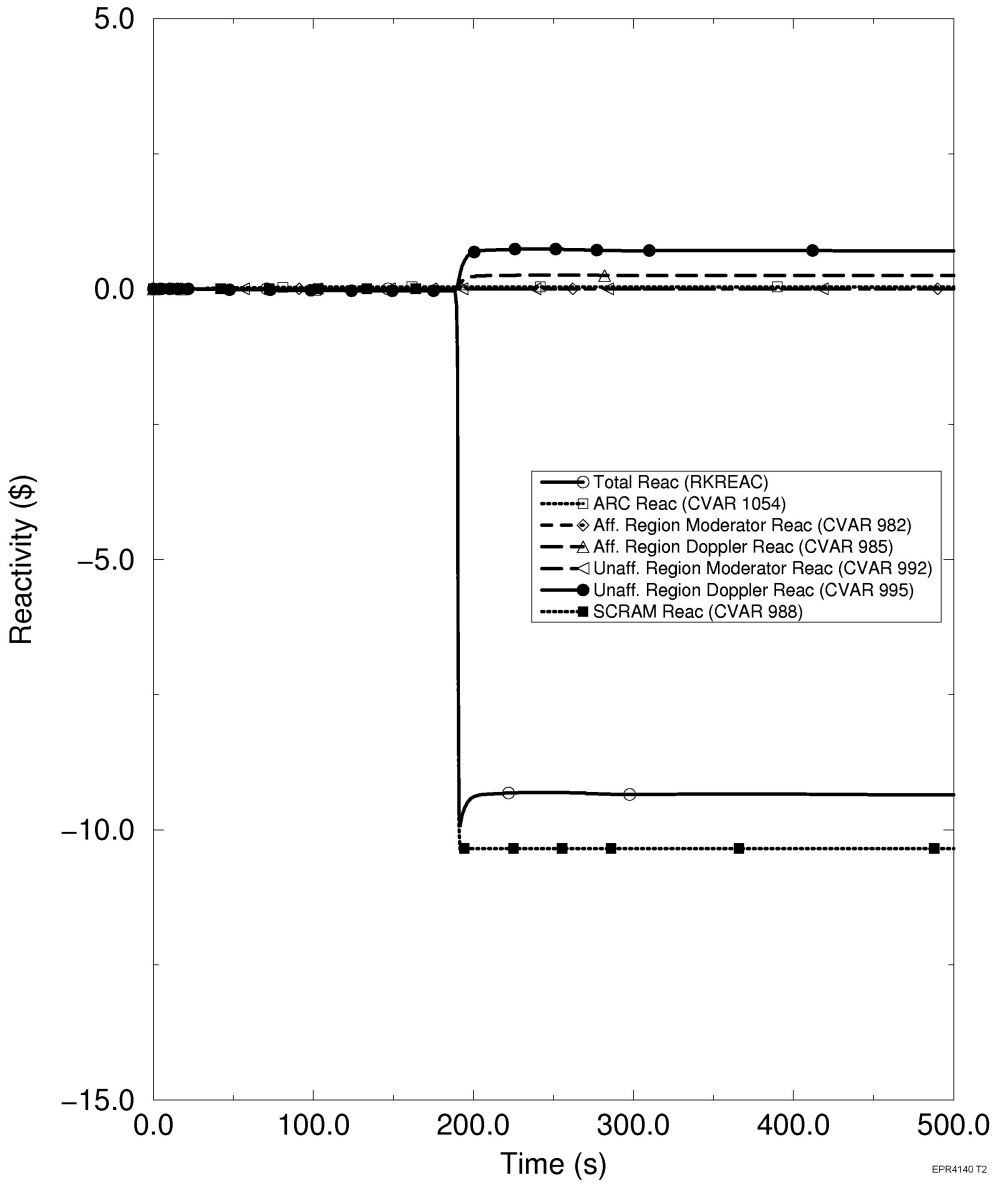

Figure 15.1-11—Increase in Main Feedwater Flow - RCS Pressures

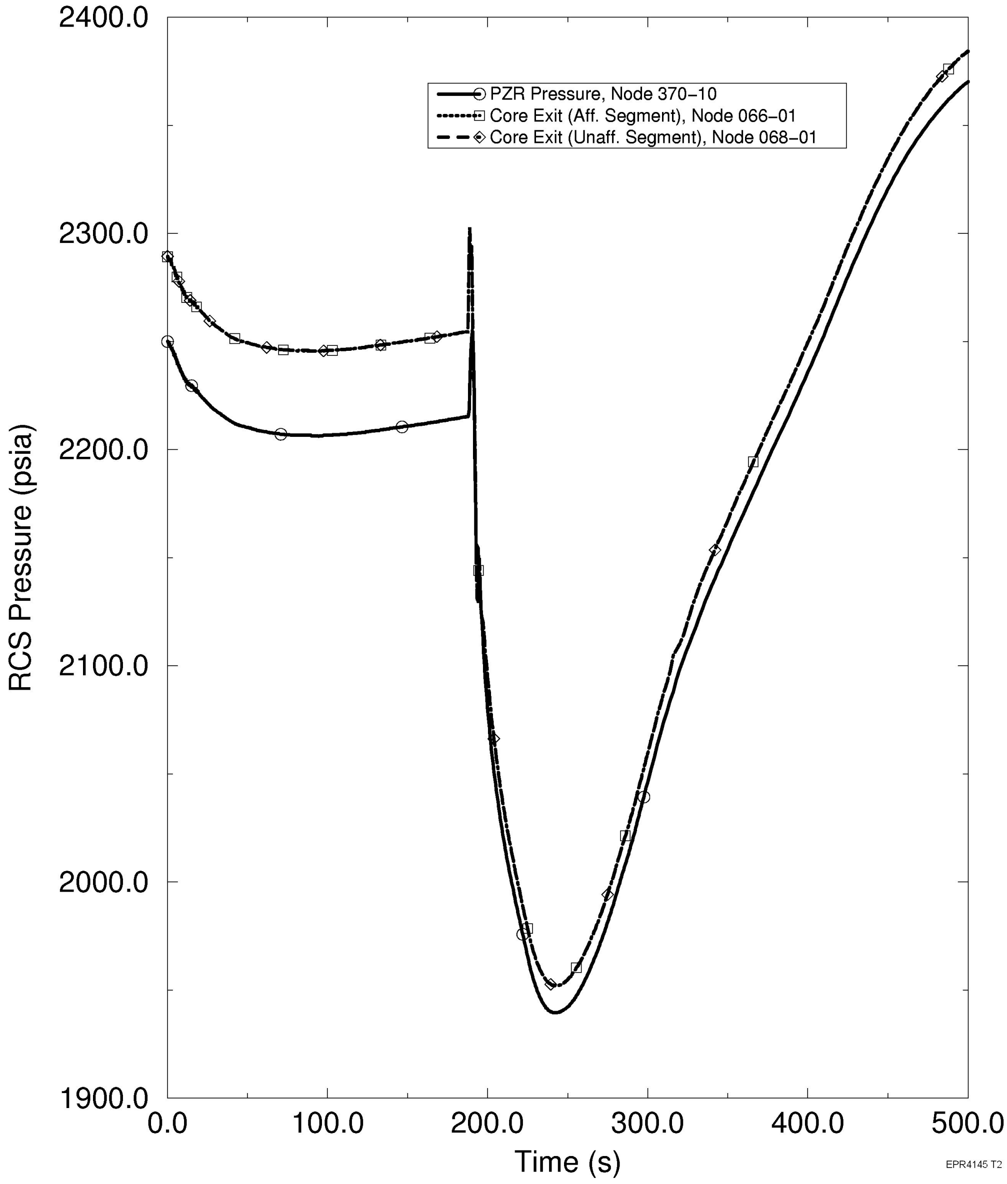

Figure 15.1-12—Increase in Main Feedwater Flow - RCS Flow

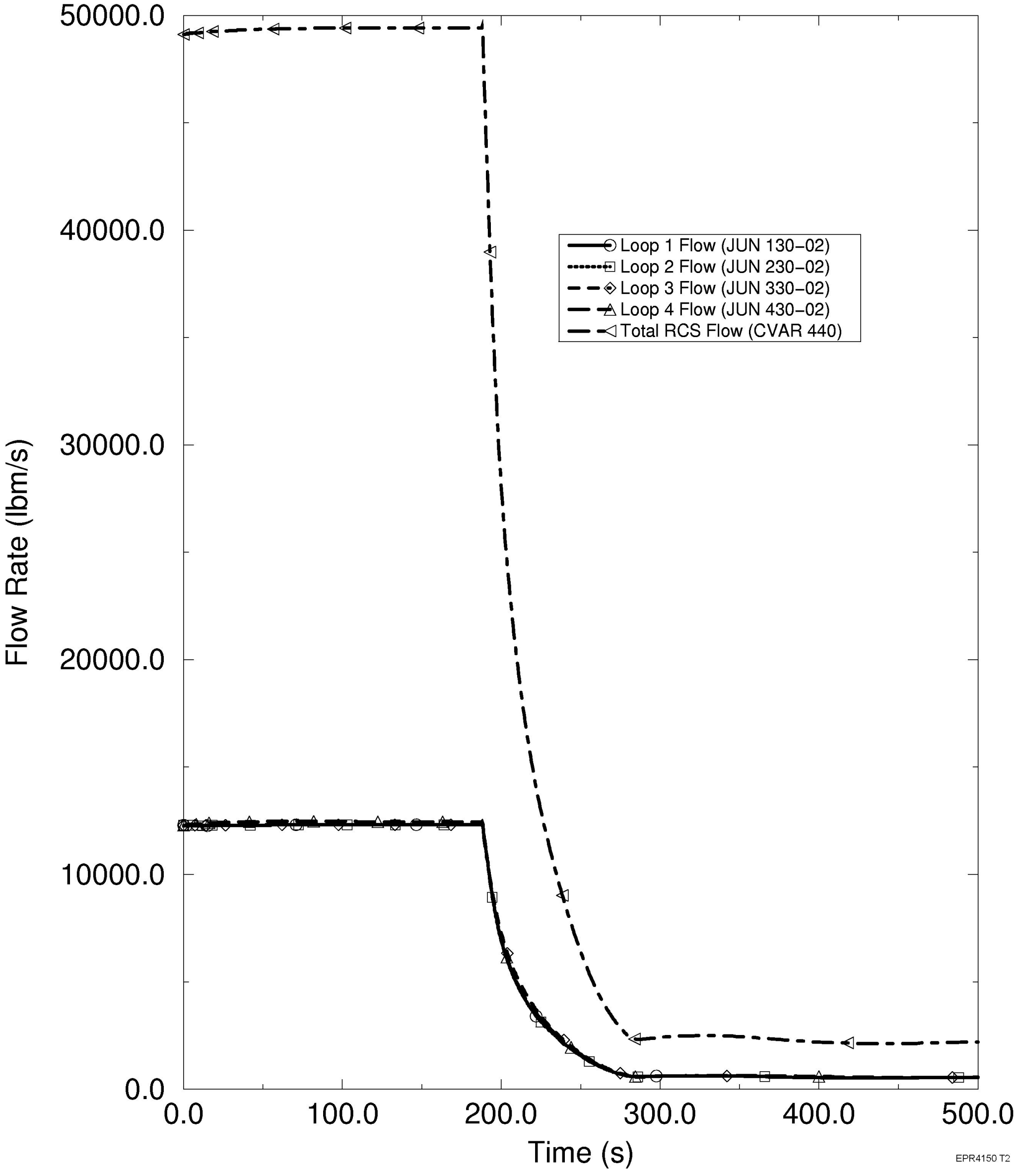

Figure 15.1-13—Increase in Main Feedwater Flow - RCS Loop Temperatures

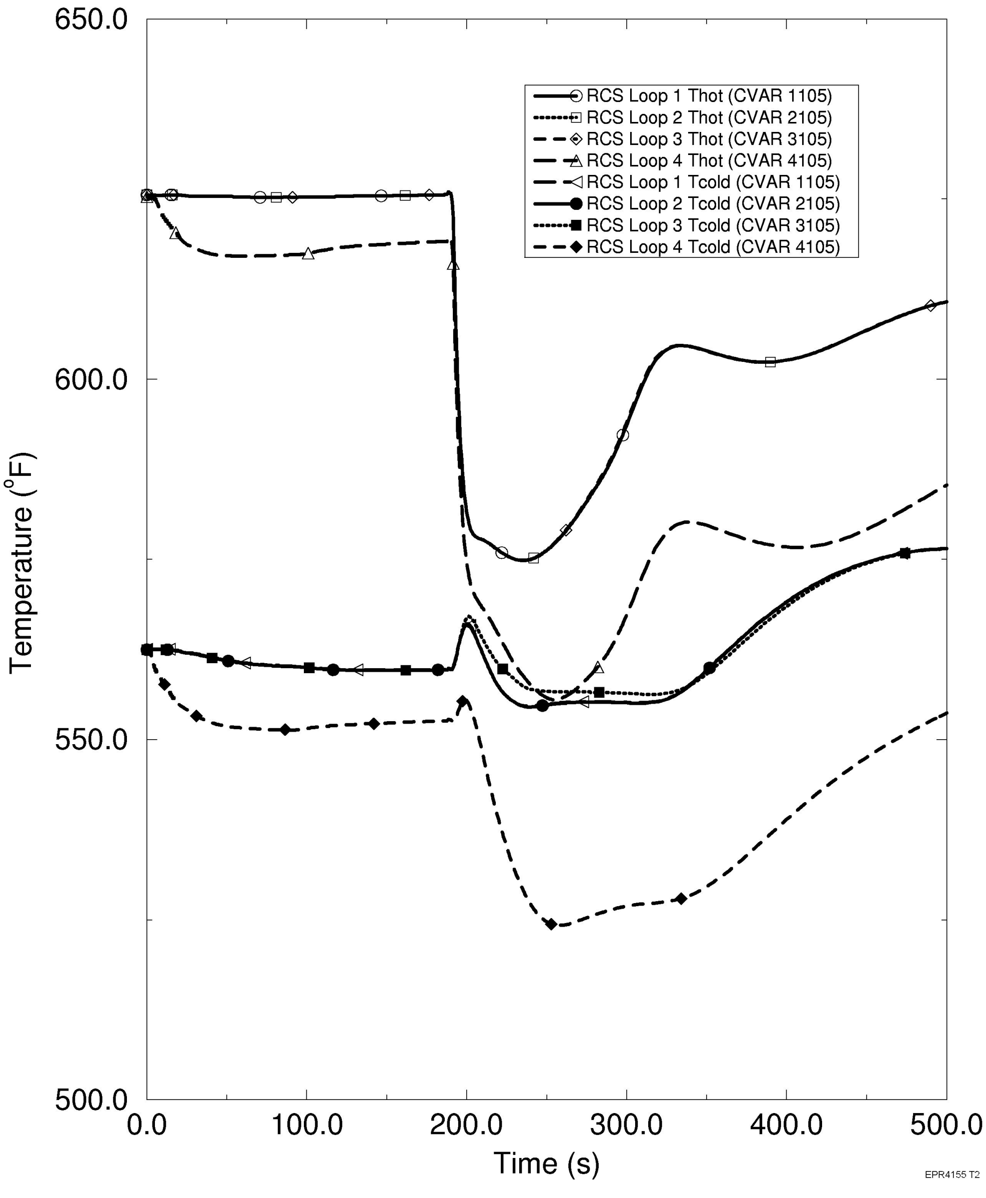

Figure 15.1-14—Increase in Main Feedwater Flow - Affected Core Segment Temperatures

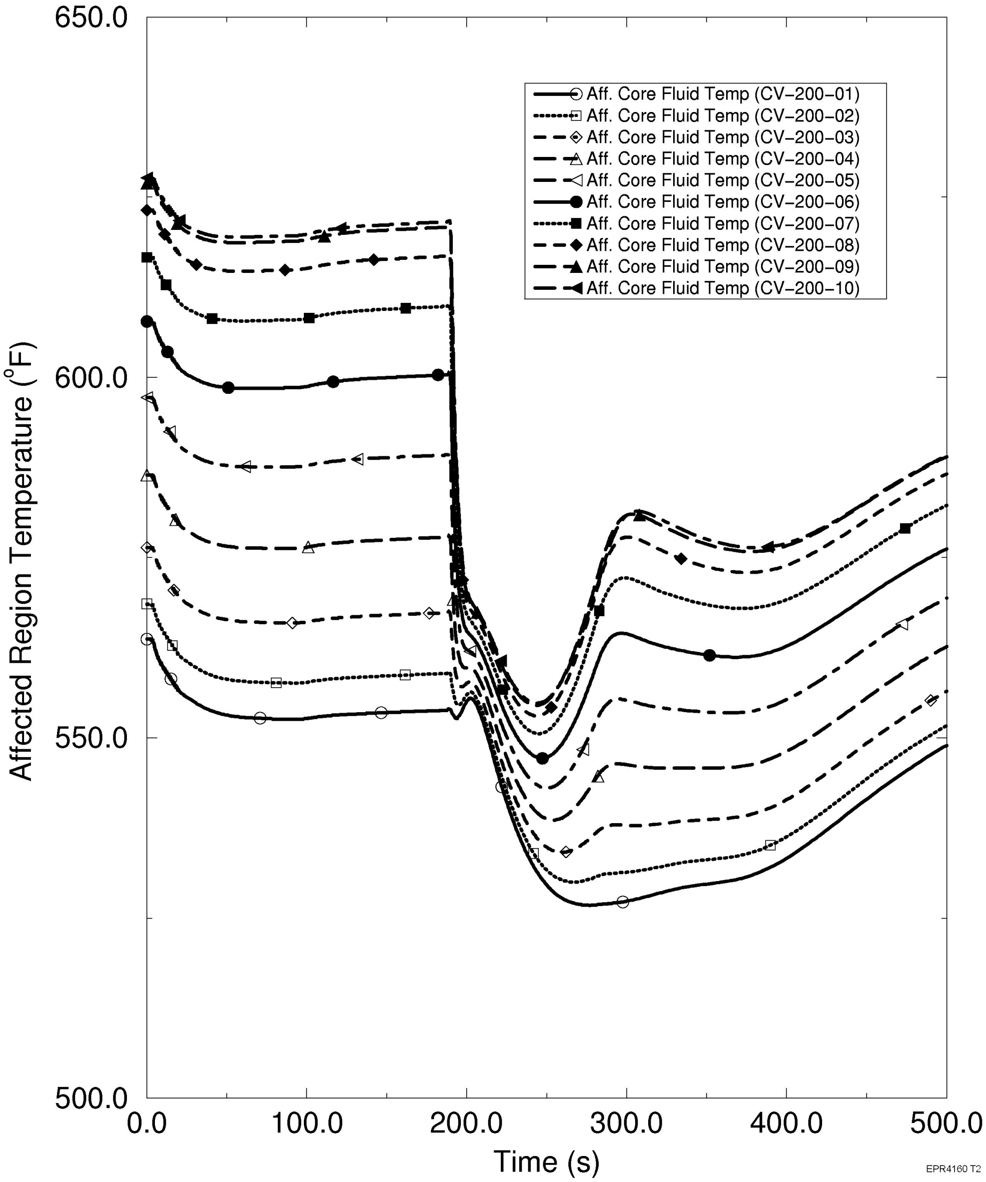

Figure 15.1-15—Increase in Main Feedwater Flow - Unaffected Core Segment Temperatures

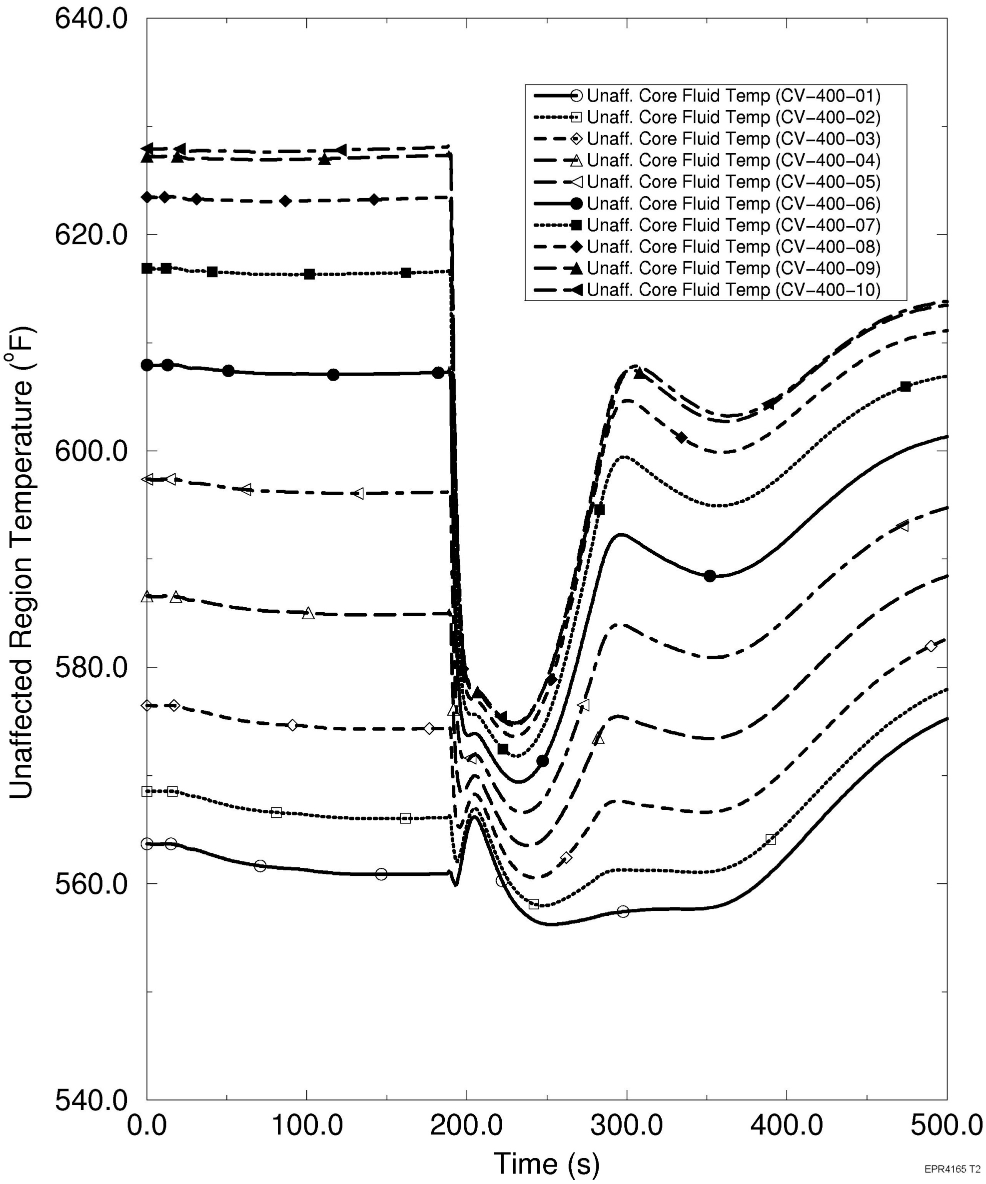

Figure 15.1-16—Increase in Main Feedwater Flow - Heat Flux

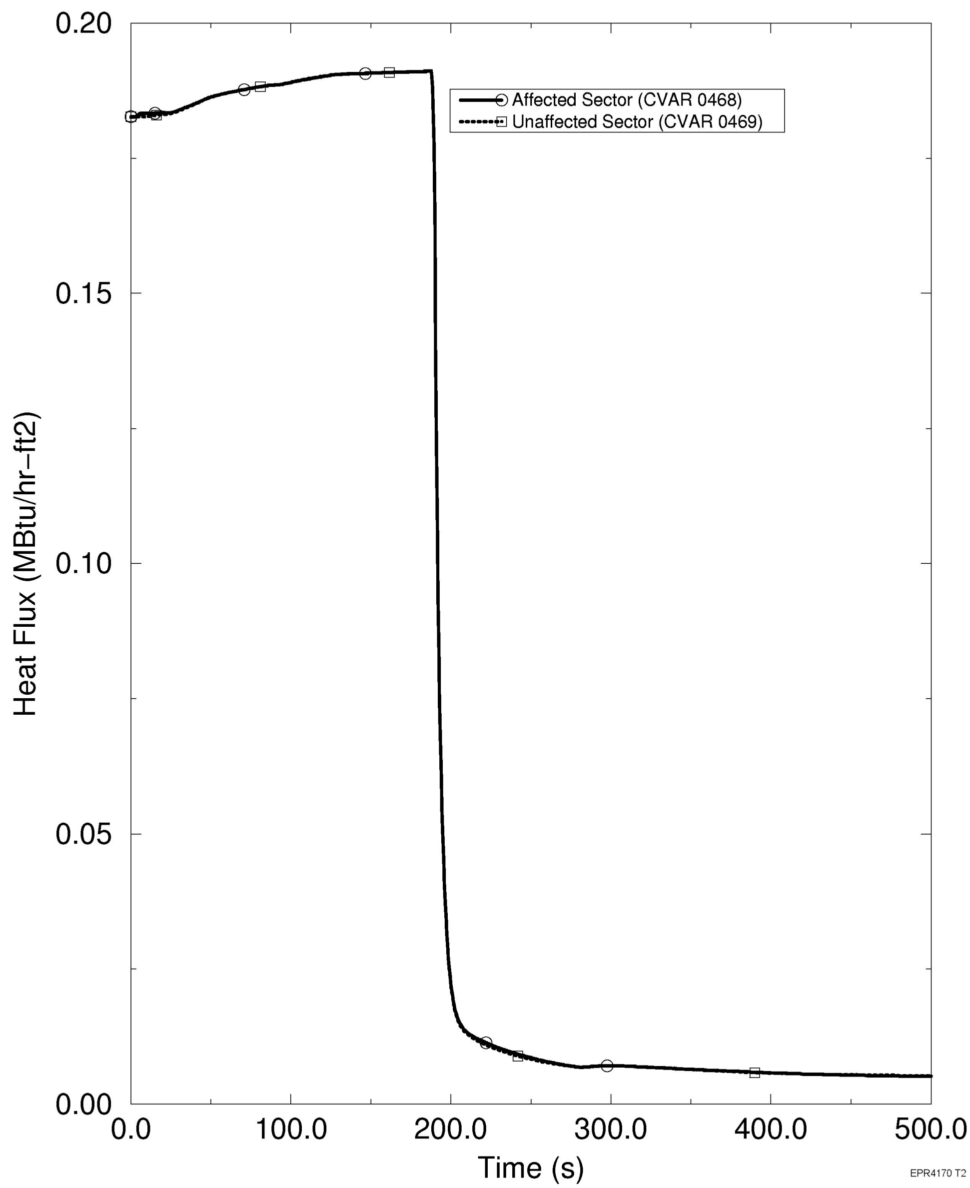

Figure 15.1-17—Increase in Main Feedwater Flow - MFW and Main Steam Flows

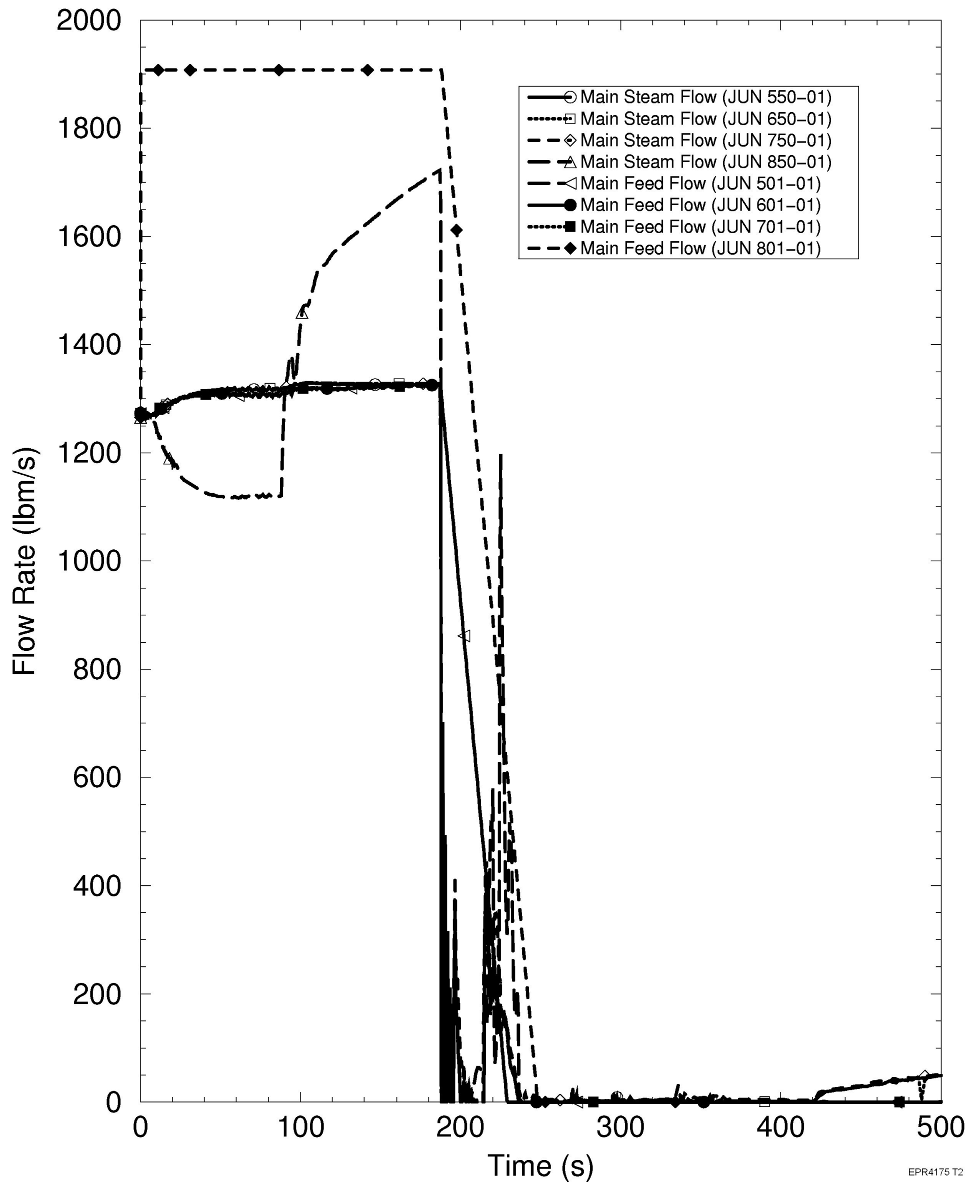

Figure 15.1-18—Increase in Main Feedwater Flow - SG Pressures

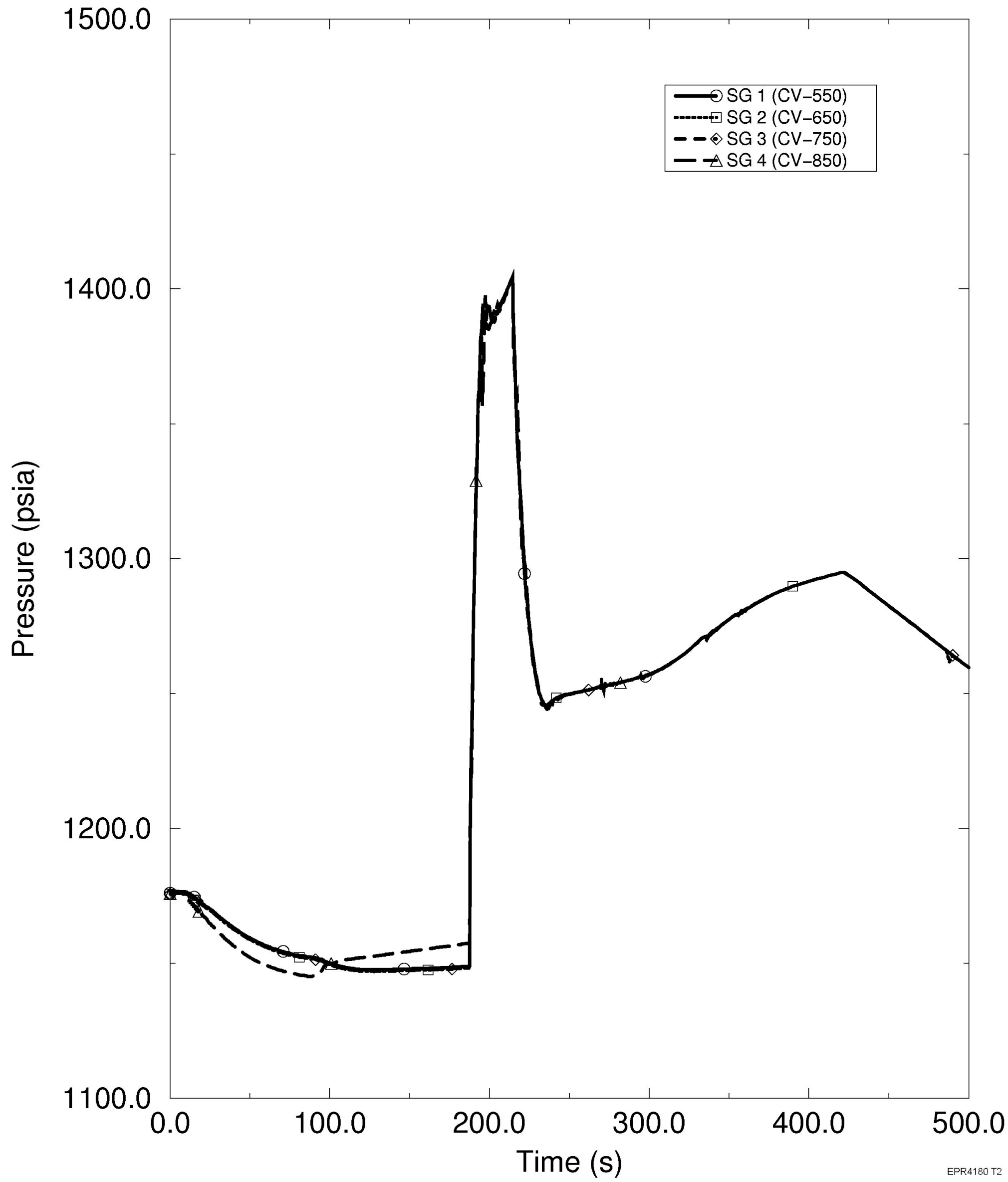

Figure 15.1-19—Increase in Main Feedwater Flow - SG NR Liquid Levels

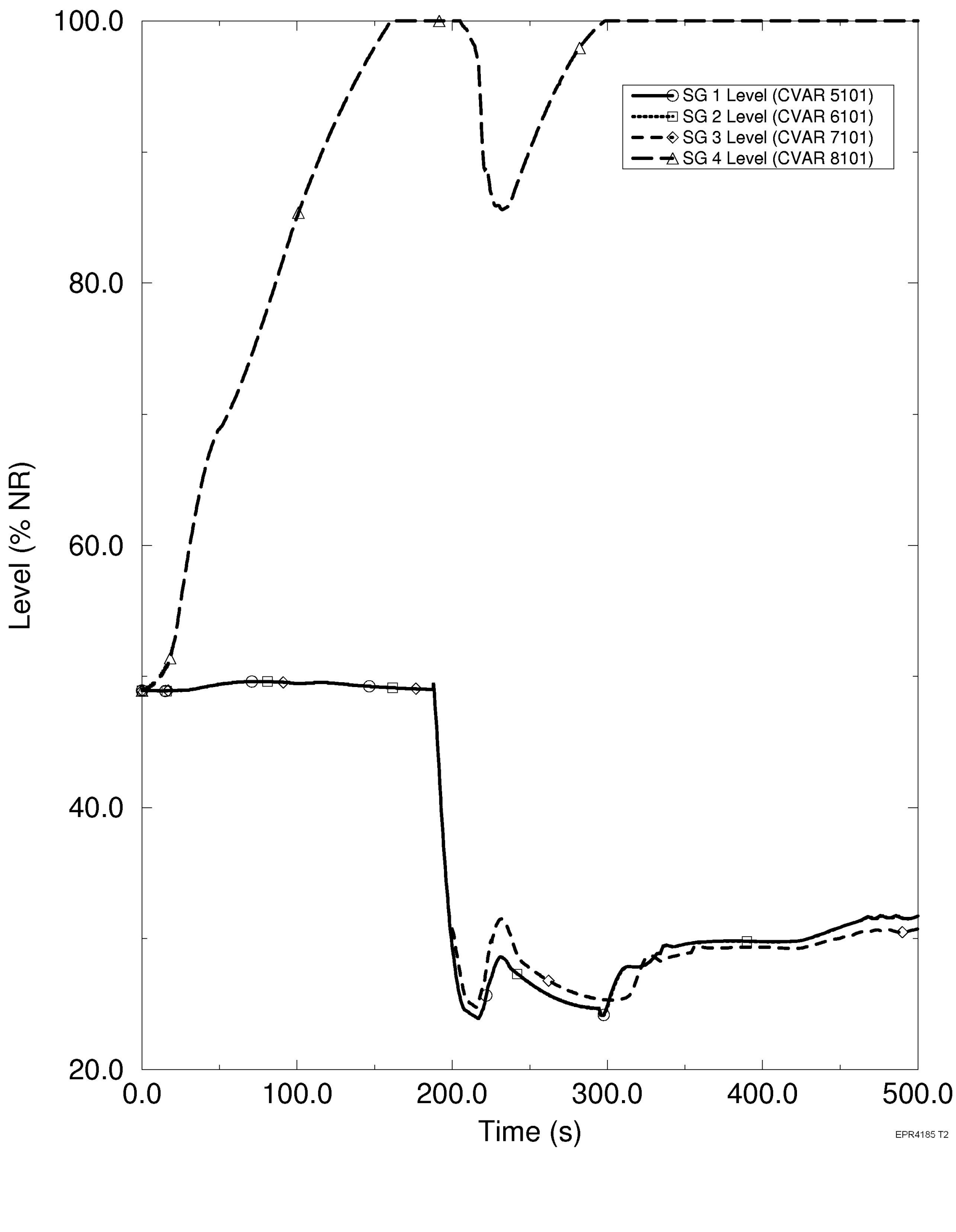

Figure 15.1-20—Increase in Feedwater Flow - SG Collapsed Level

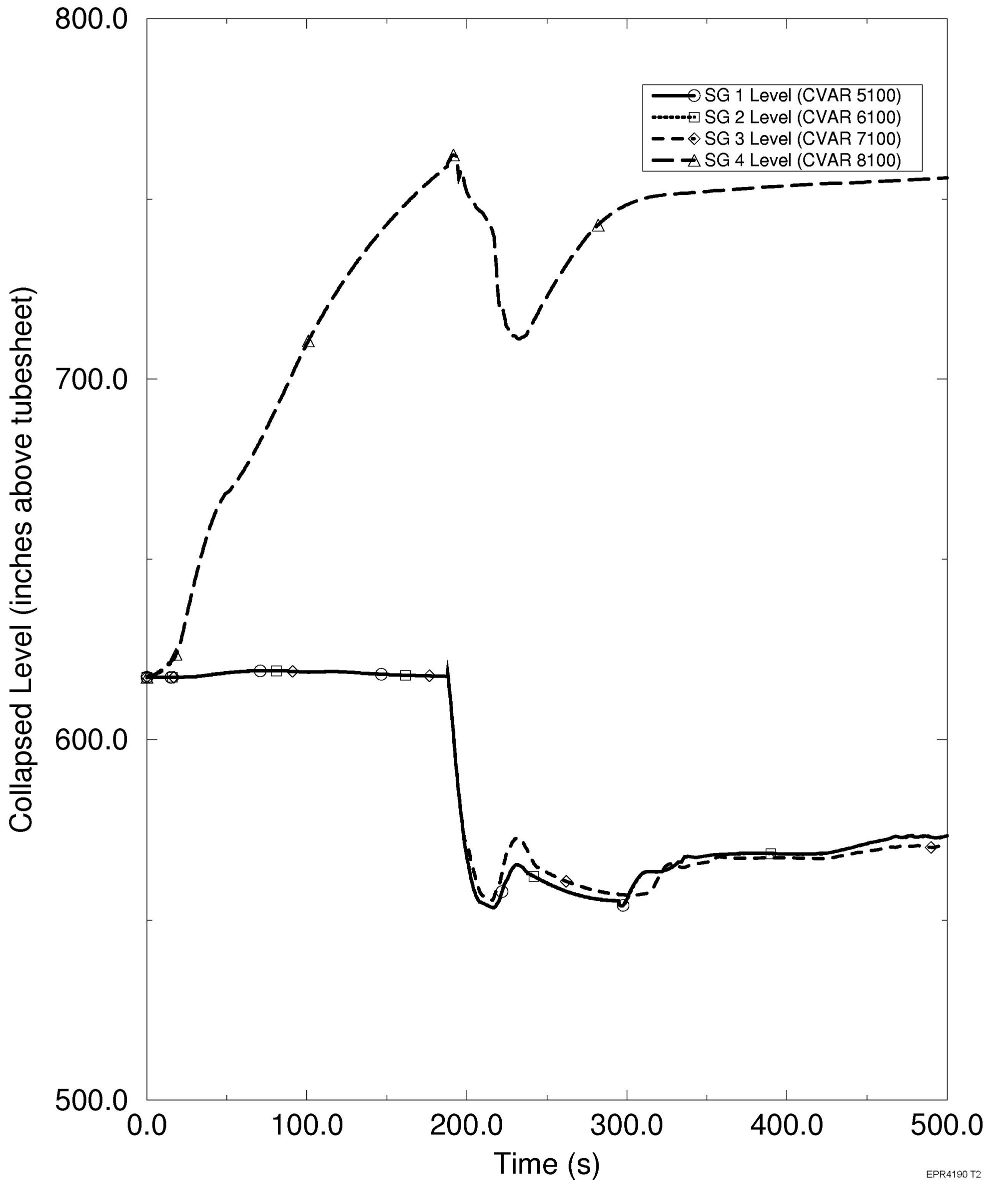

Figure 15.1-21—Increase in Feedwater Flow - Reactivity

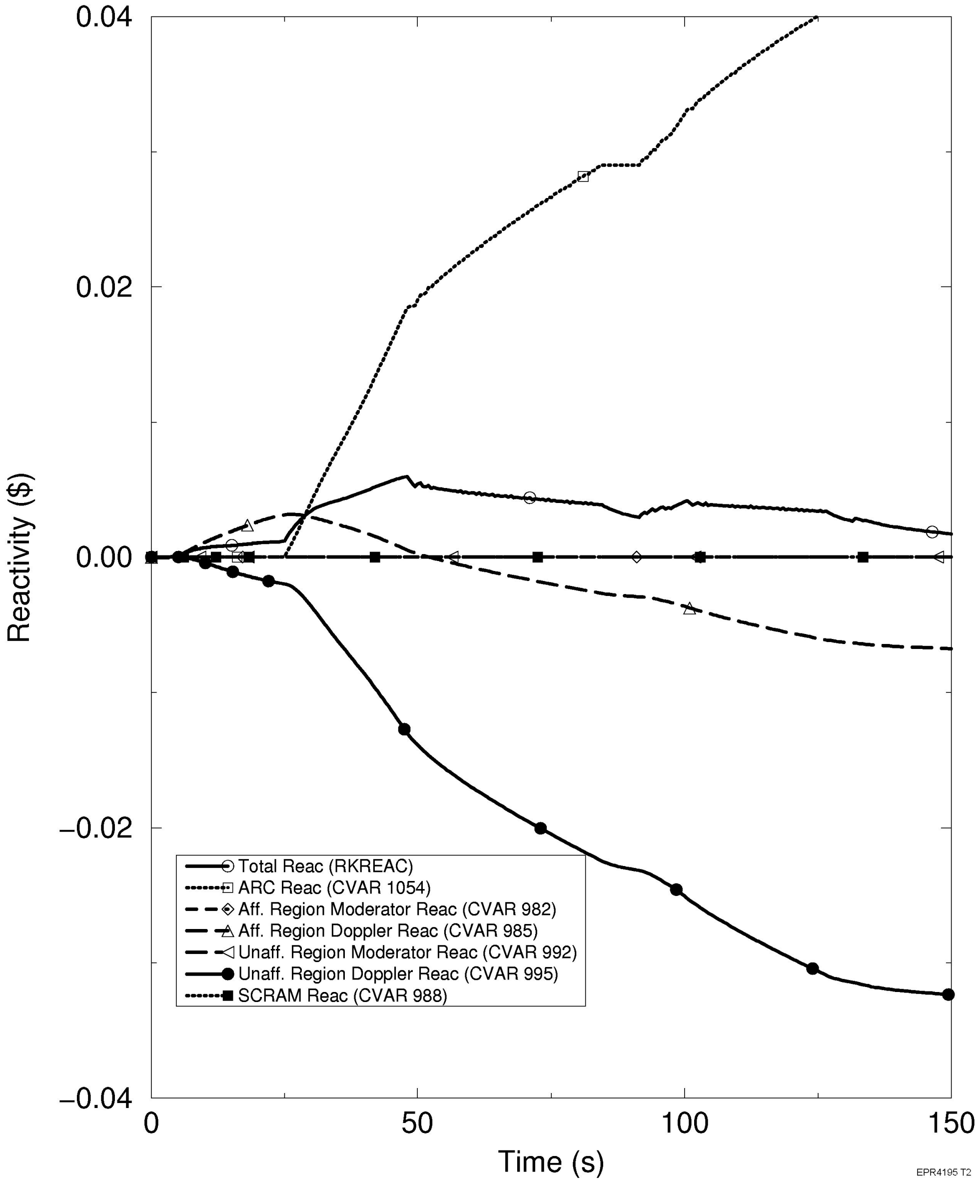

Figure 15.1-22—Increase in Steam Flow - Reactor Power

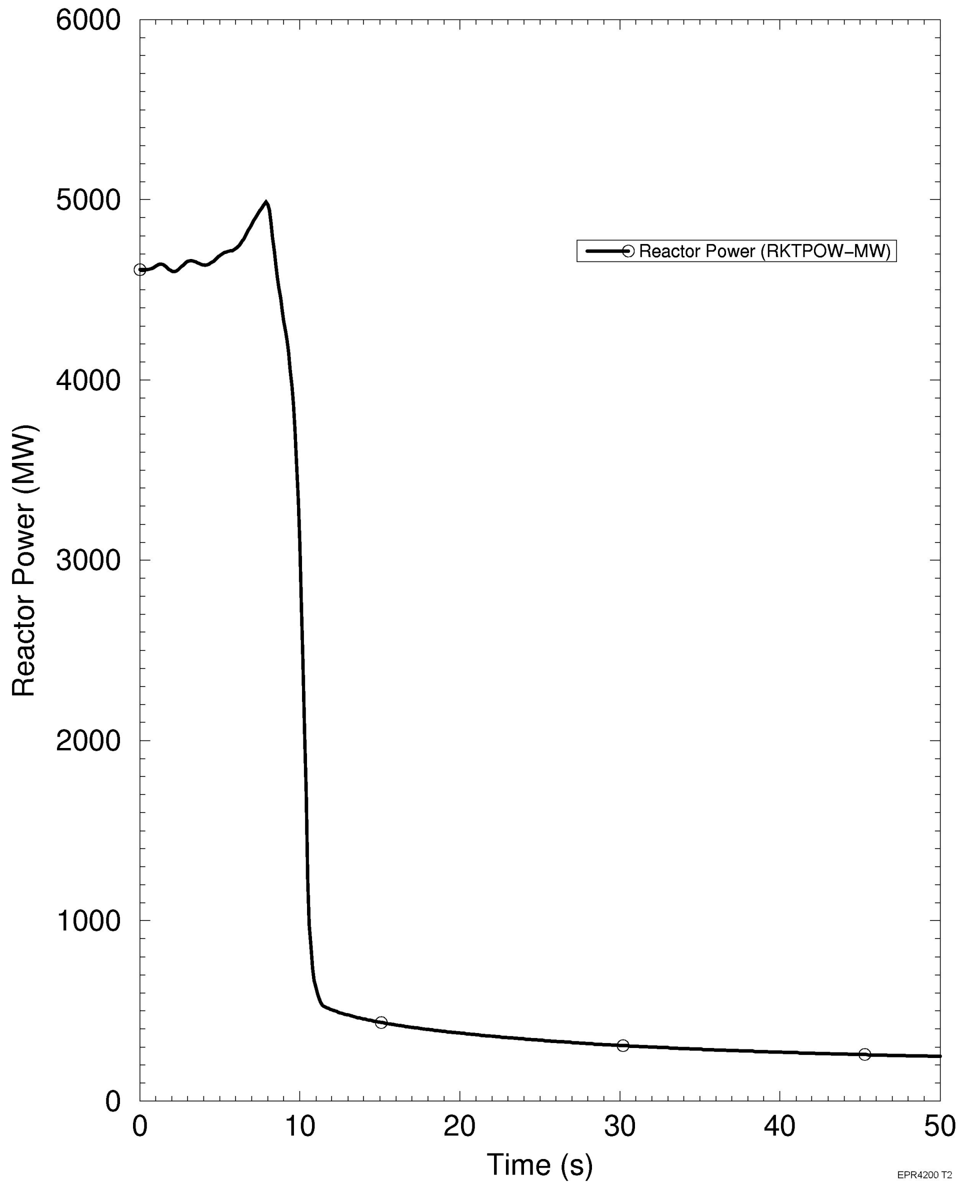

Figure 15.1-23—Increase in Steam Flow - Reactivity

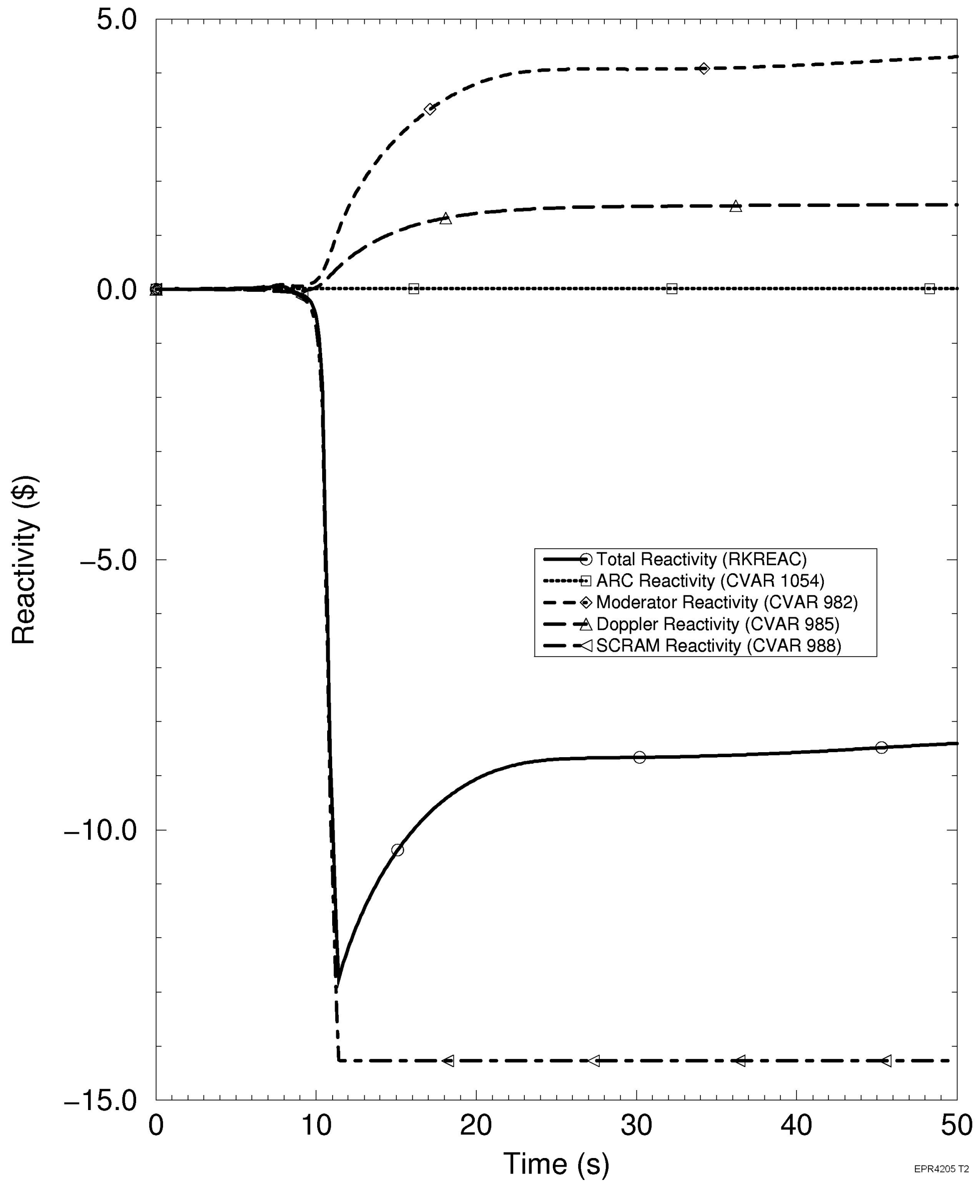

Figure 15.1-24—Increase in Steam Flow - RCS Pressures

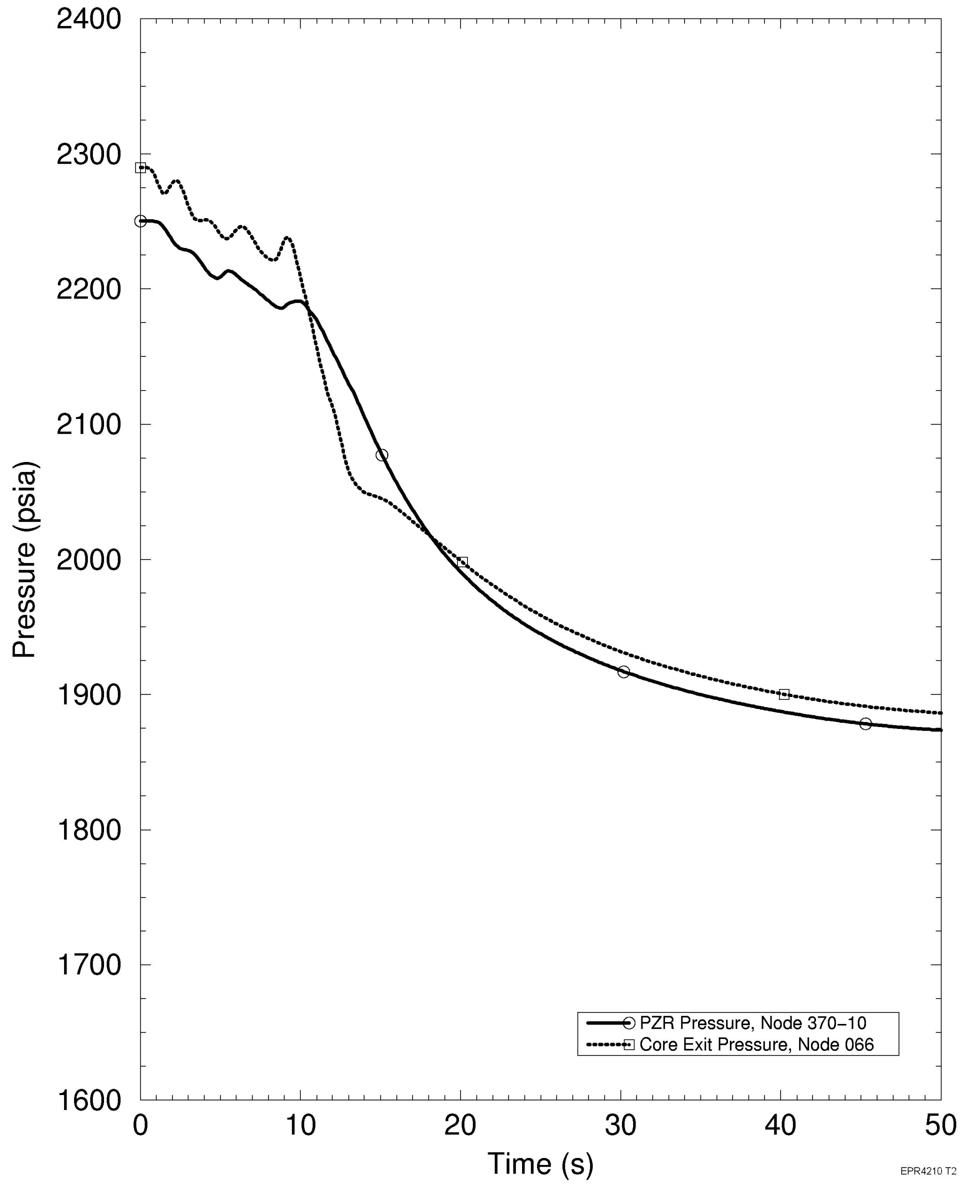

Figure 15.1-25—Increase in Steam Flow - RCS Flow Rate

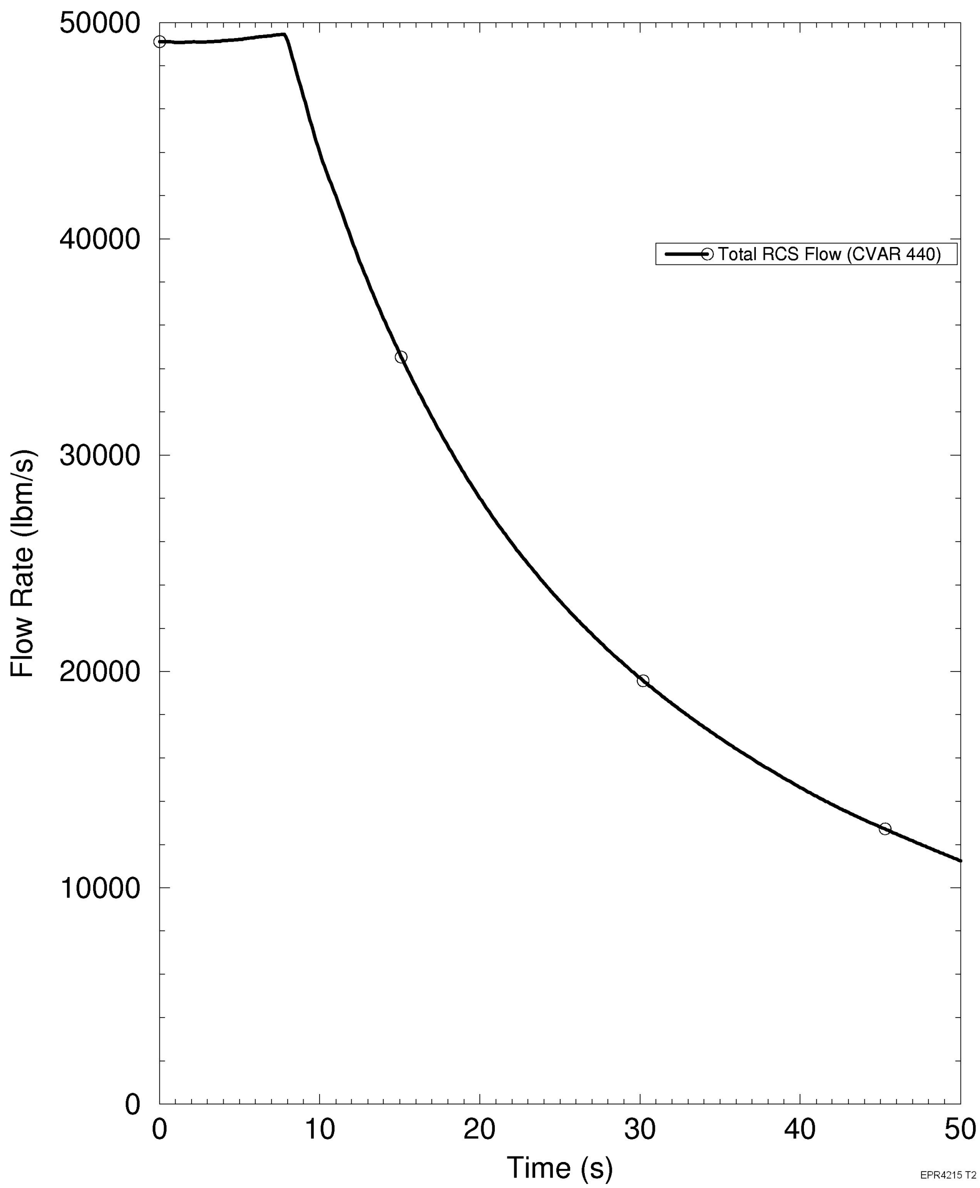

Figure 15.1-26—Increase in Steam Flow - RCS Loop Temperatures

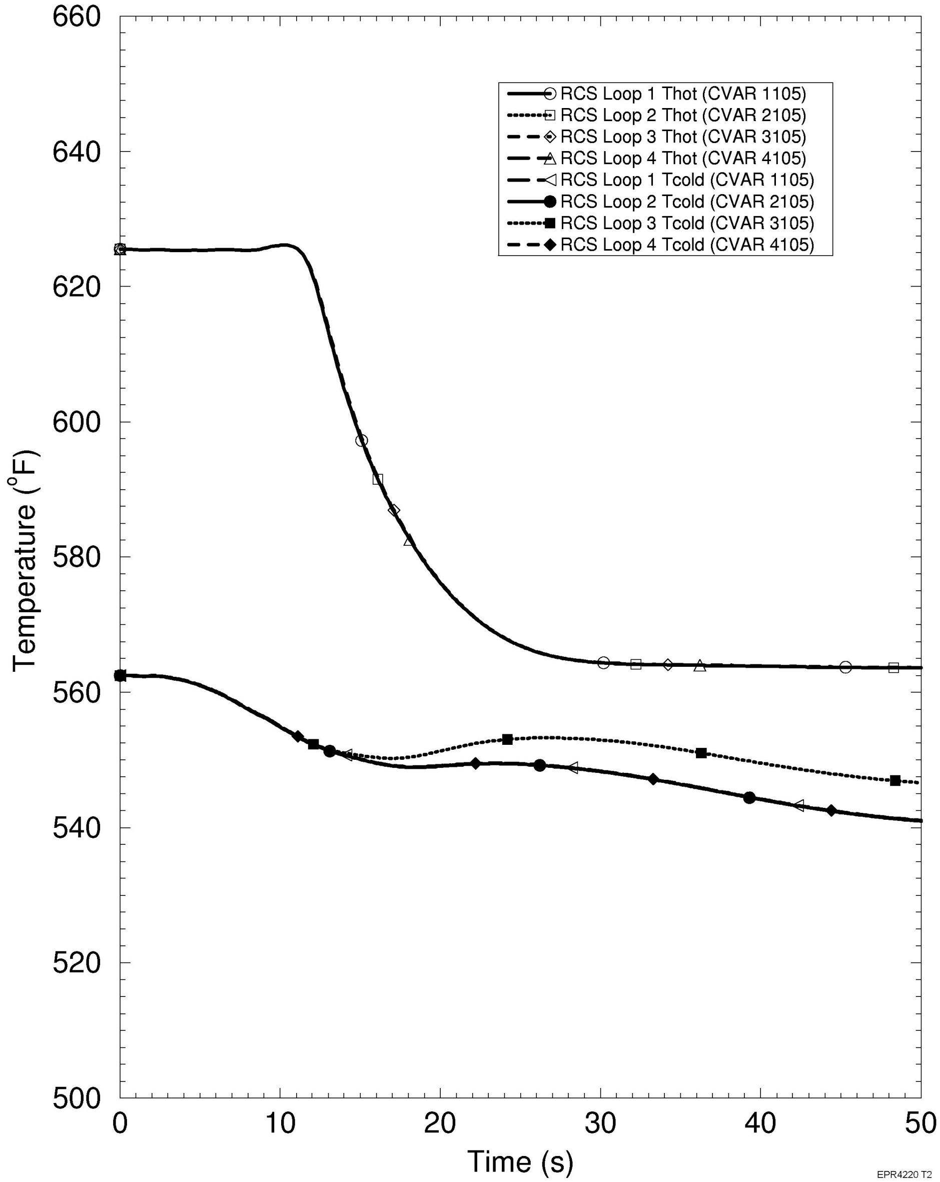

Figure 15.1-27—Increase in Steam Flow - Core Fluid Temperatures

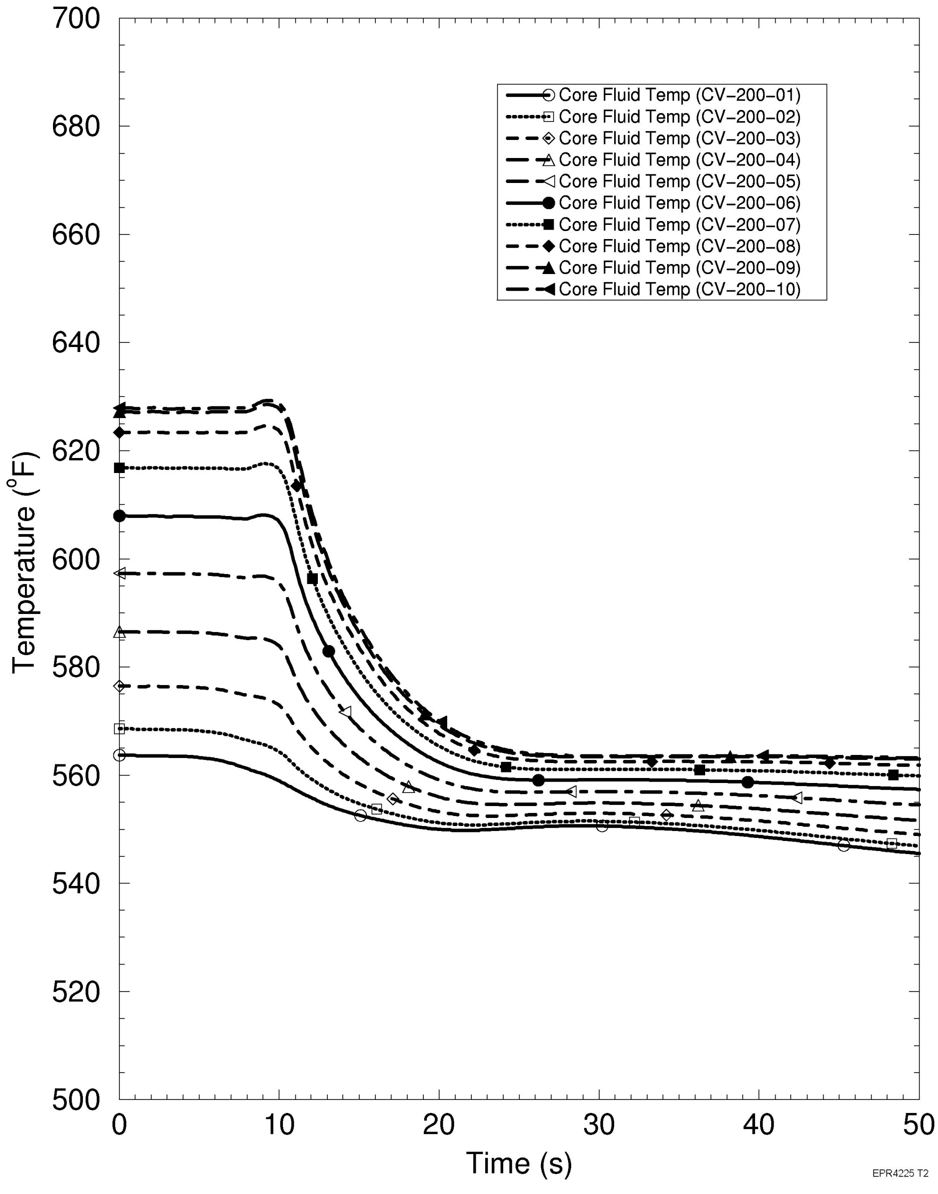

Figure 15.1-28—Increase in Steam Flow - Heat Flux

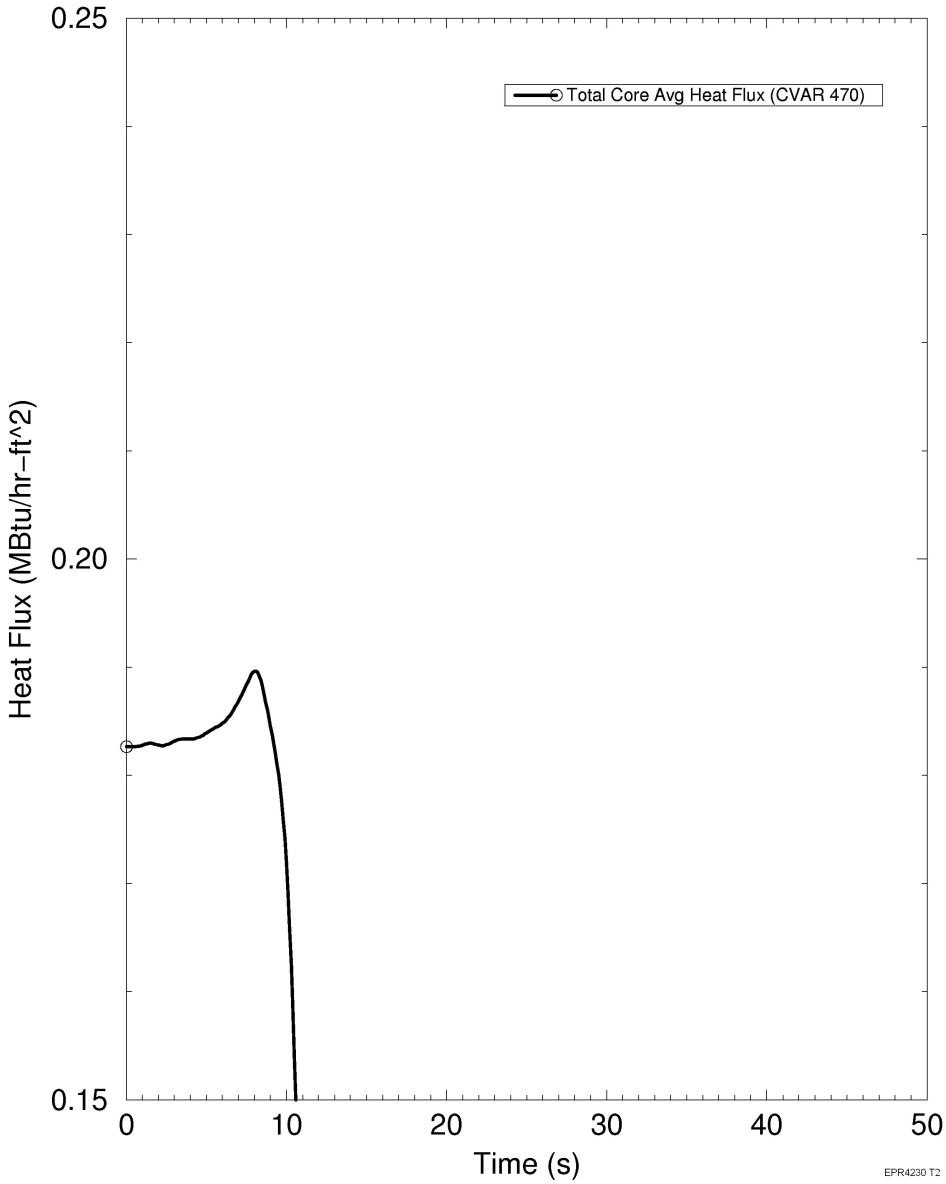

Figure 15.1-29—Increase in Steam Flow - MFW and Main Steam Flow Rates

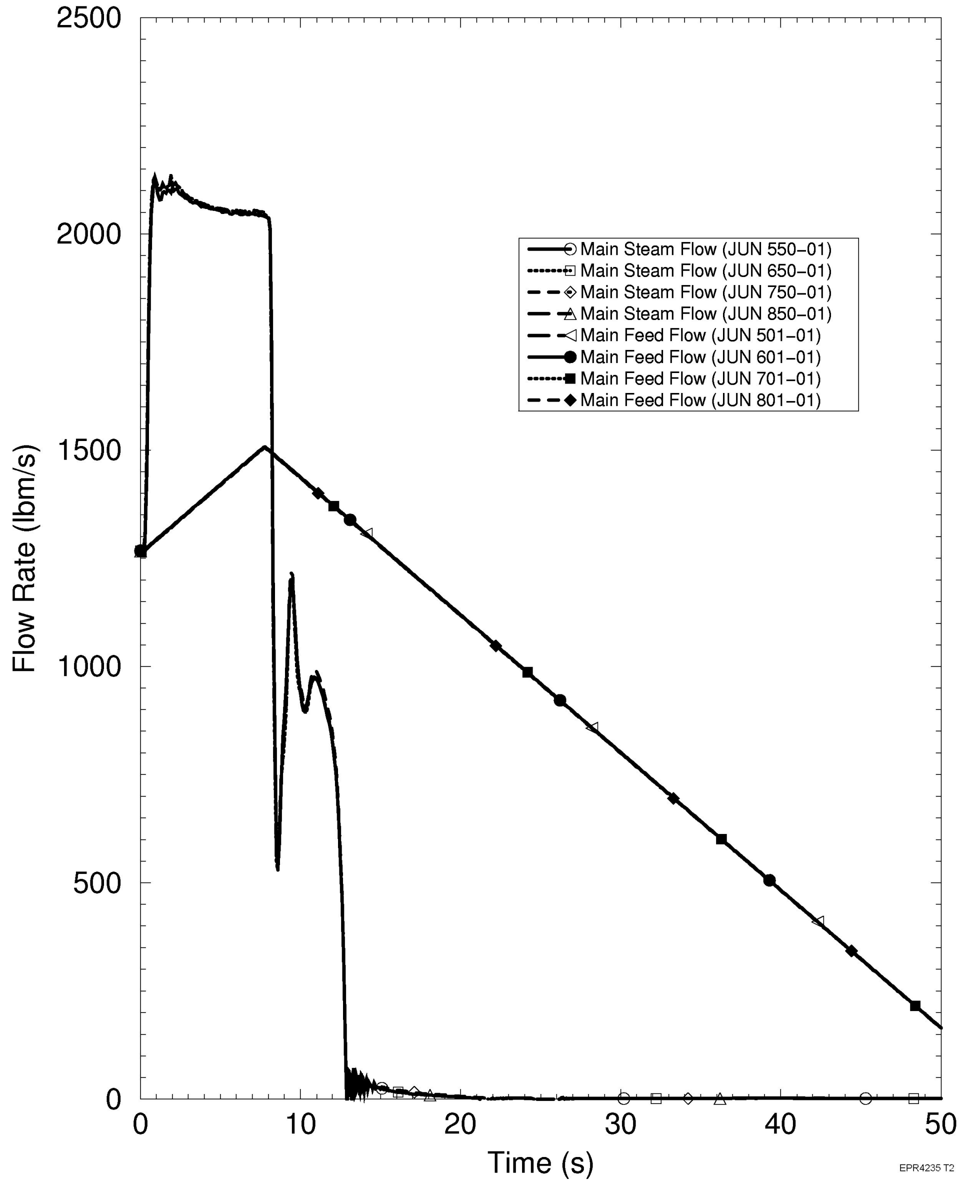

Figure 15.1-30—Increase in Steam Flow - SG Pressures

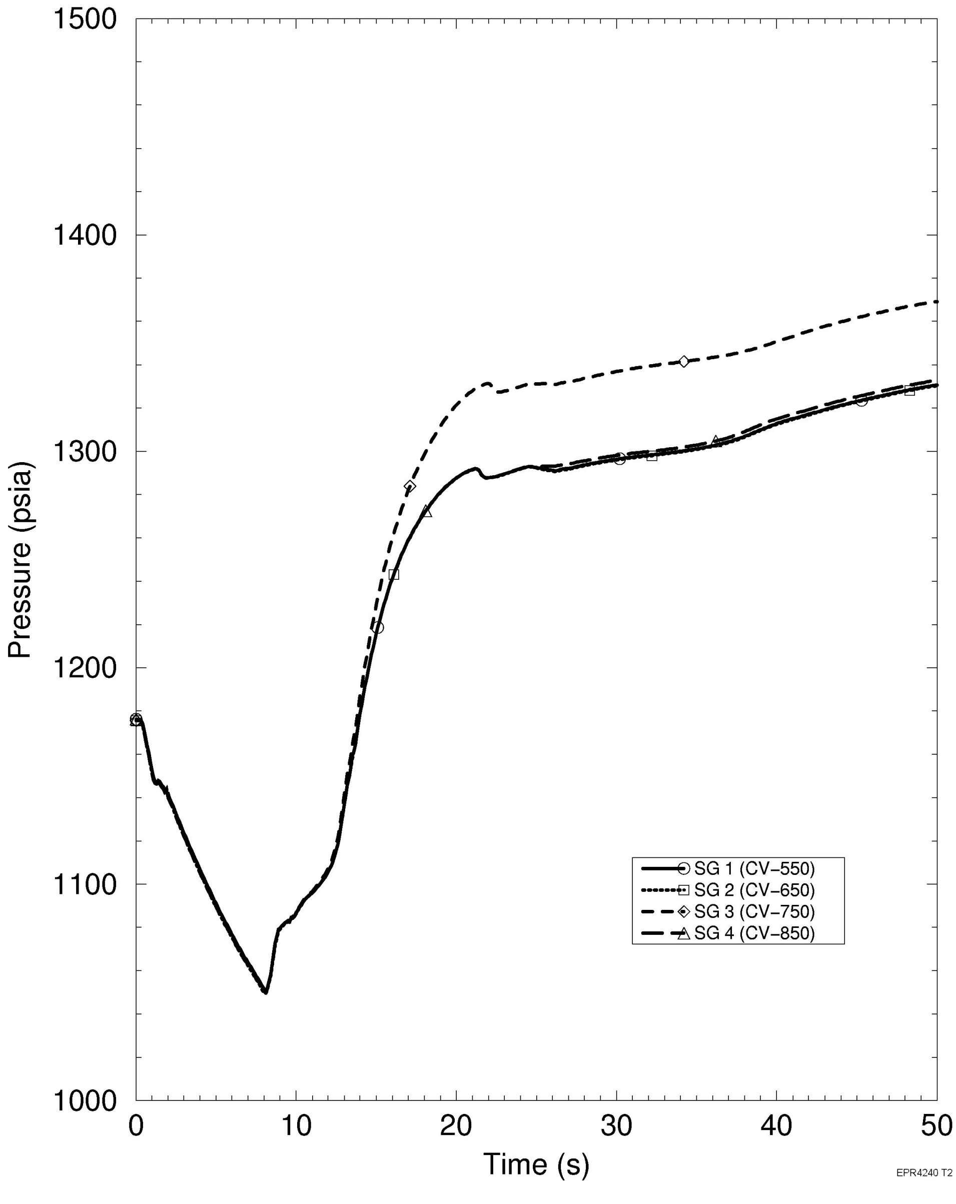

Figure 15.1-31—Increase in Steam Flow - SG WR Liquid Level

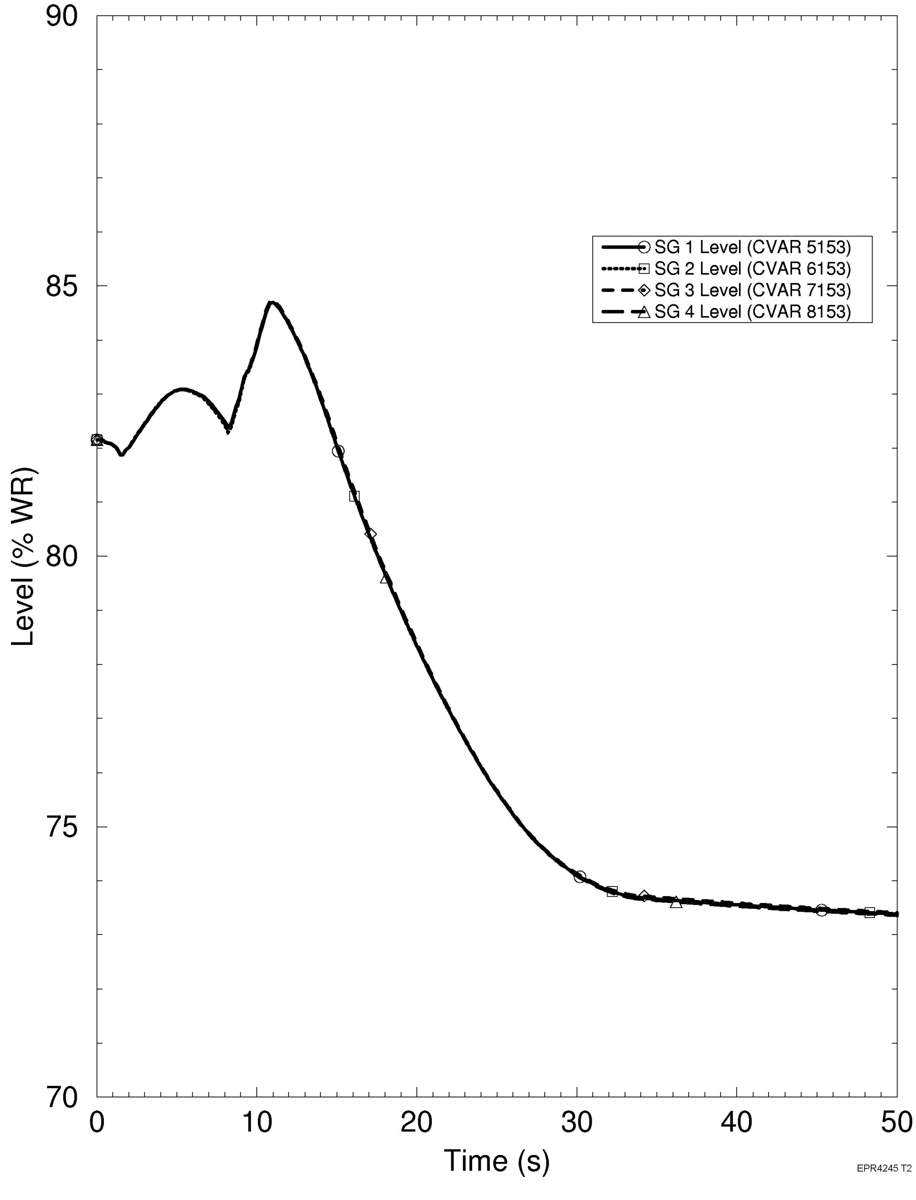

Figure 15.1-32—Increase in Steam Flow - Reactivity (Detail)

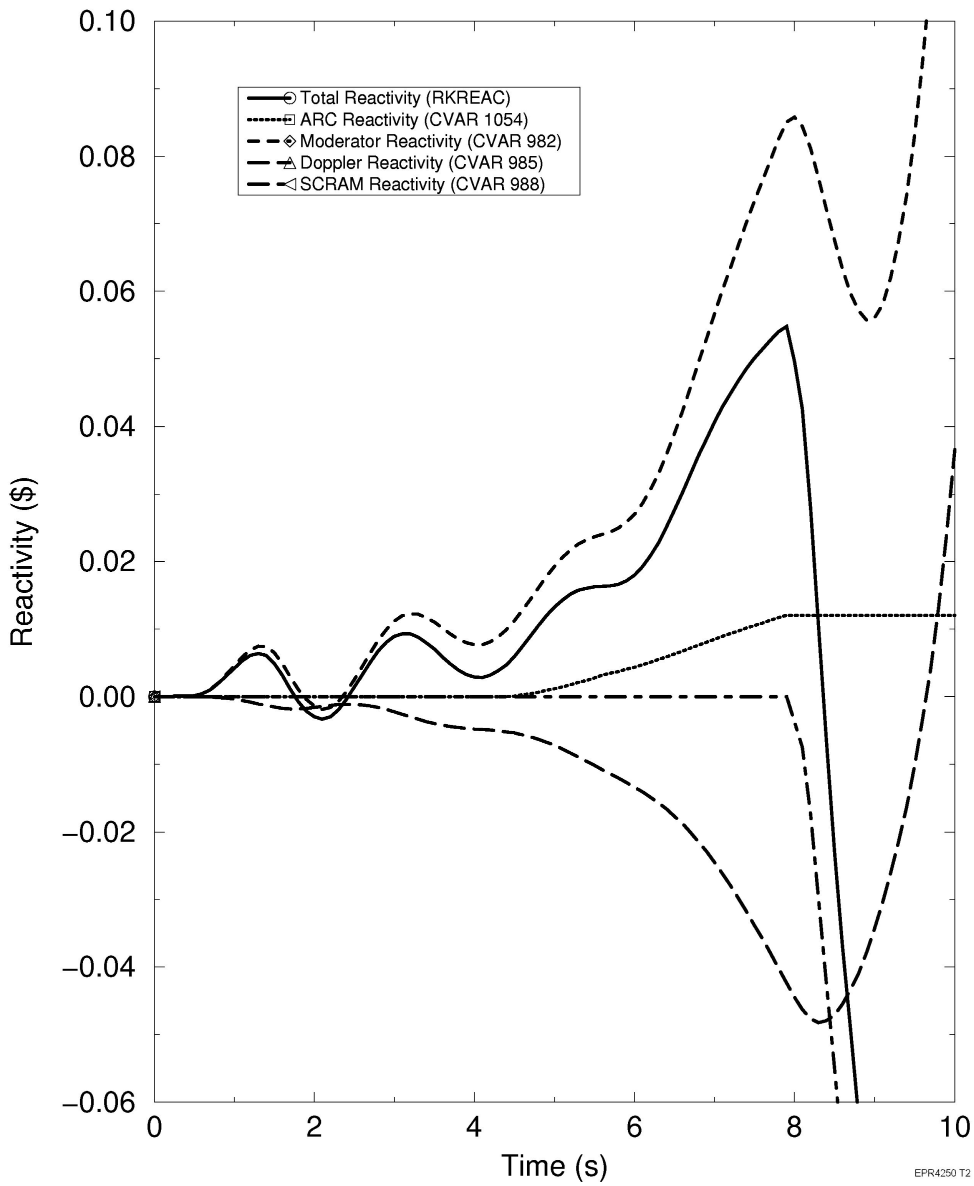

Figure 15.1-33—Inadvertent Opening of a SG Relief or Safety Valve - MSRT Flow Rate

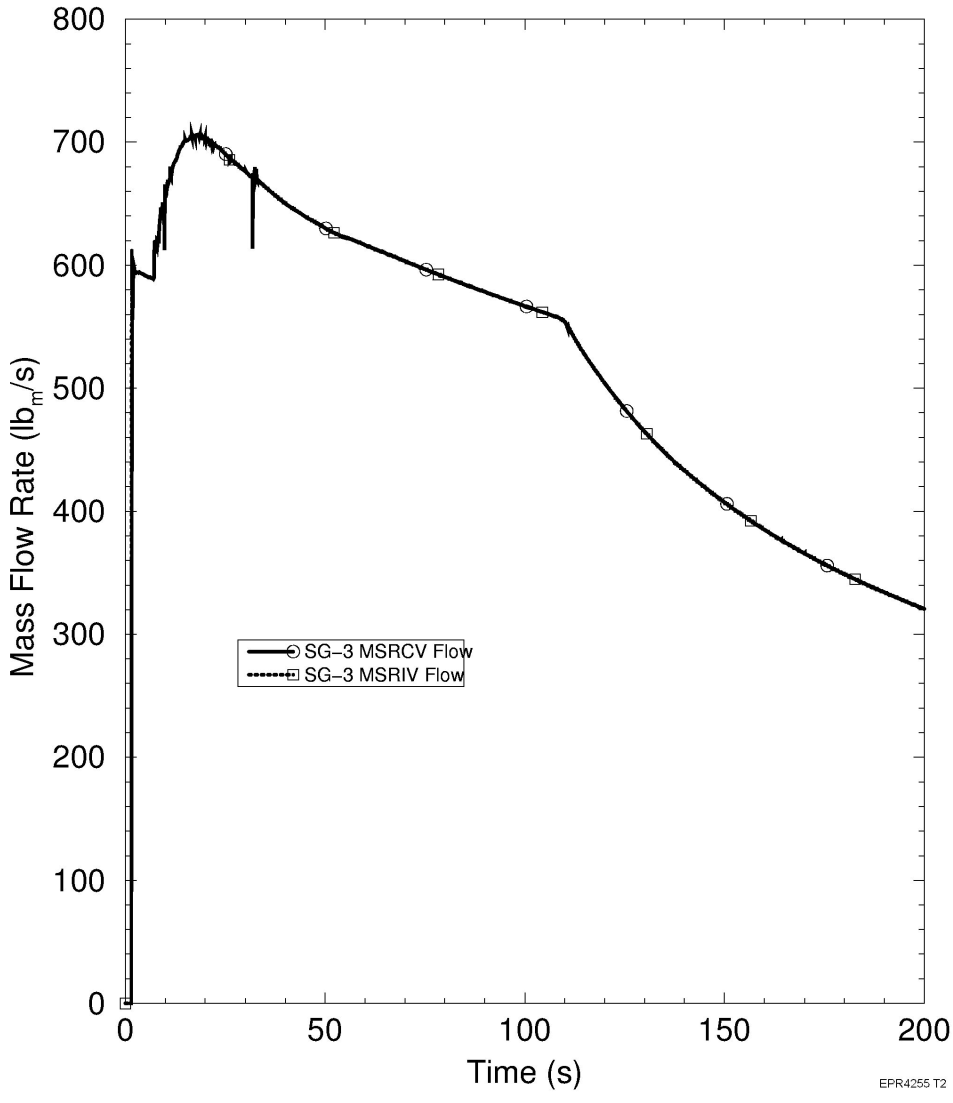

Figure 15.1-34—Inadvertent Opening of a SG Relief or Safety Valve - MFW Flow Rate

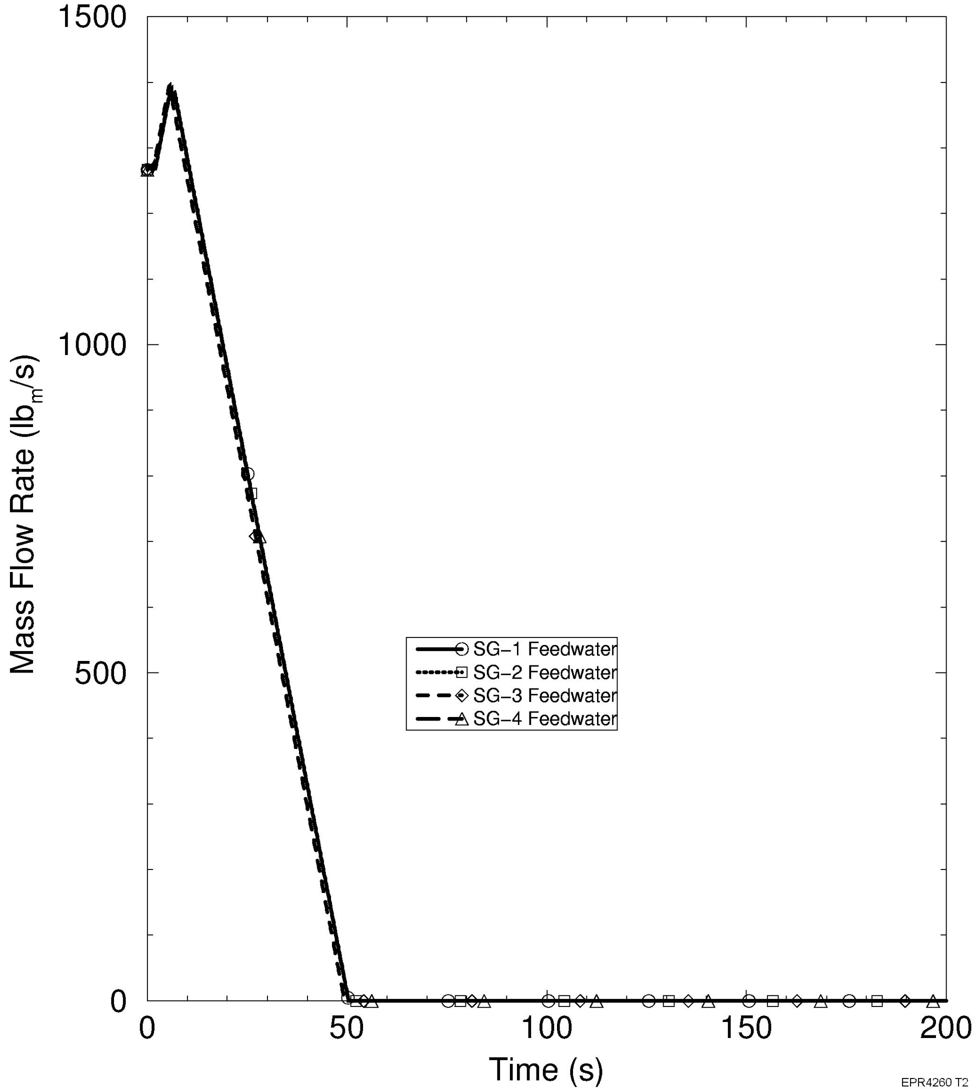

Figure 15.1-35—Inadvertent Opening of a SG Relief or Safety Valve - SG NR Liquid Levels

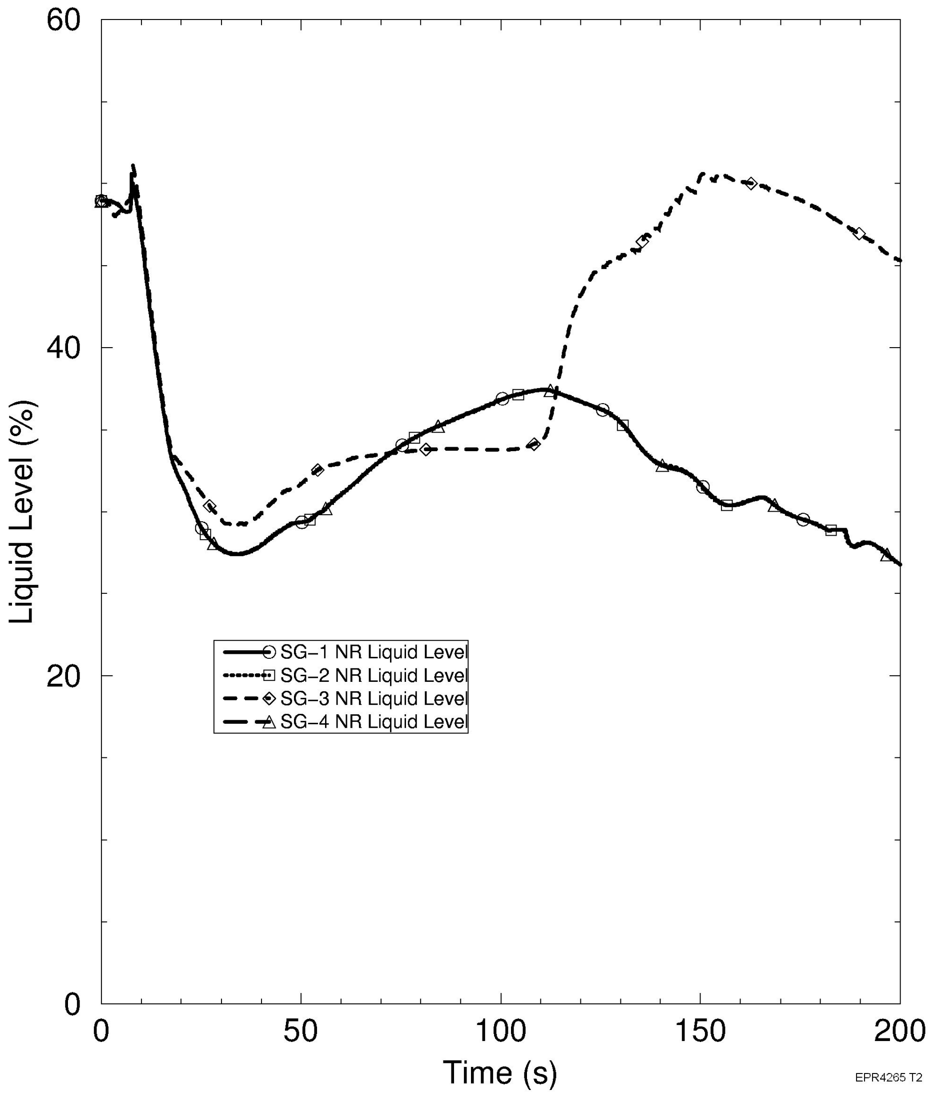

Figure 15.1-36—Inadvertent Opening of a SG Relief or Safety Valve - Heat Transfer to SGs

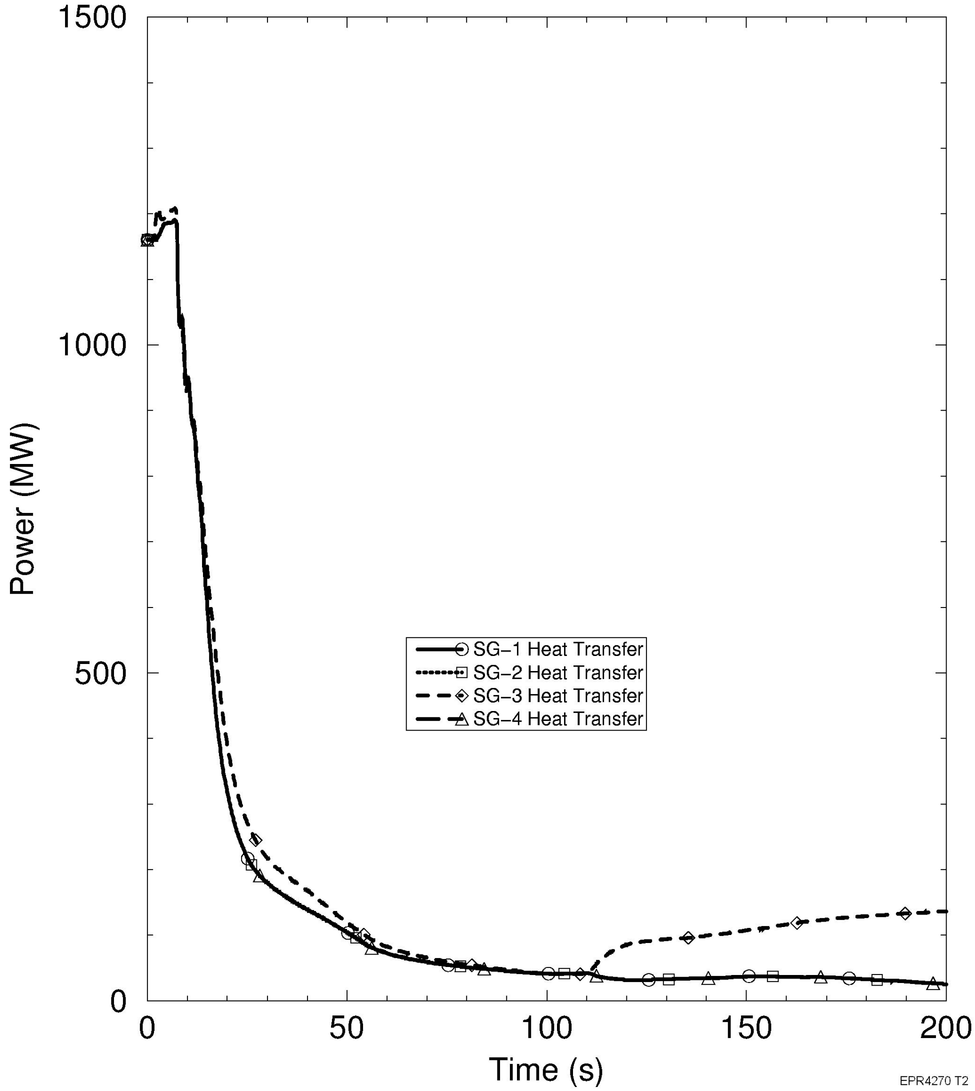

Figure 15.1-37—Inadvertent Opening of a SG Relief or Safety Valve - Cold Leg Temperatures

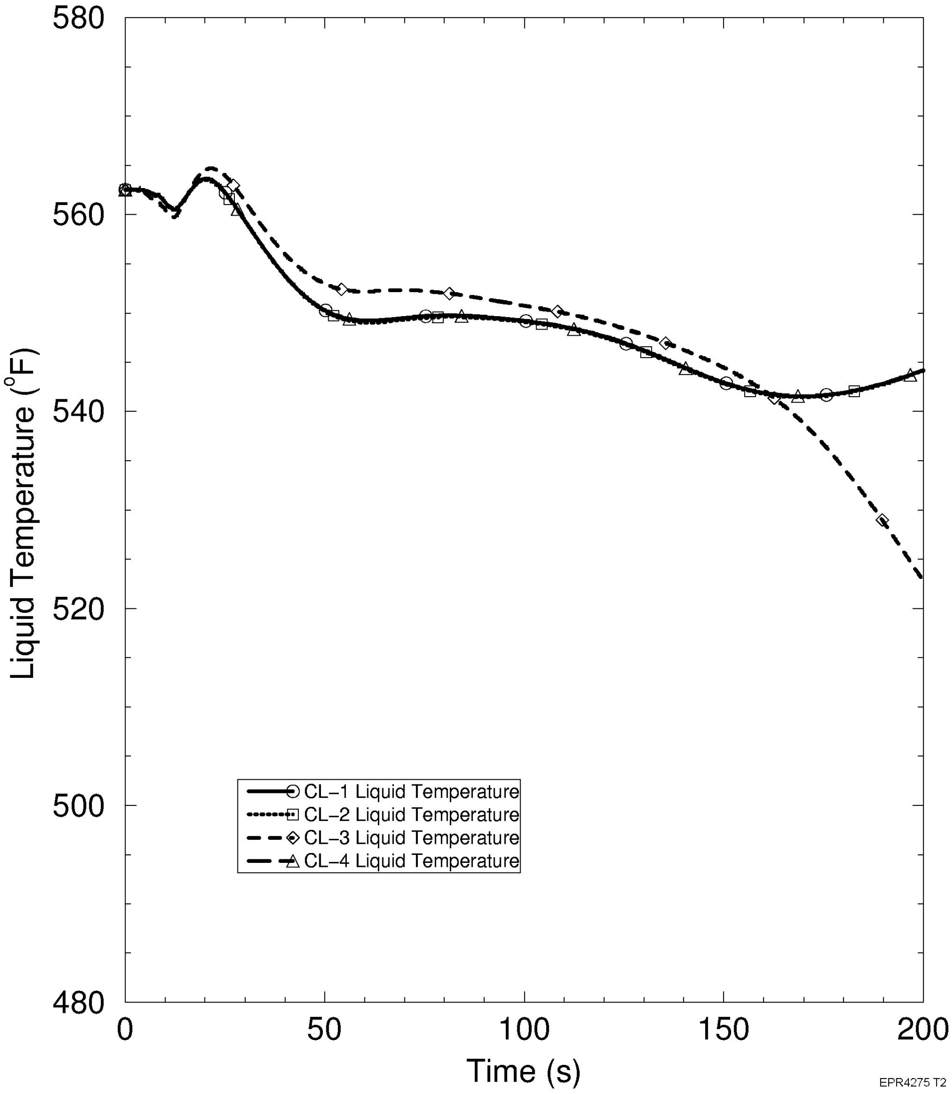

Figure 15.1-38—Inadvertent Opening of a SG Relief or Safety Valve - Reactor Power

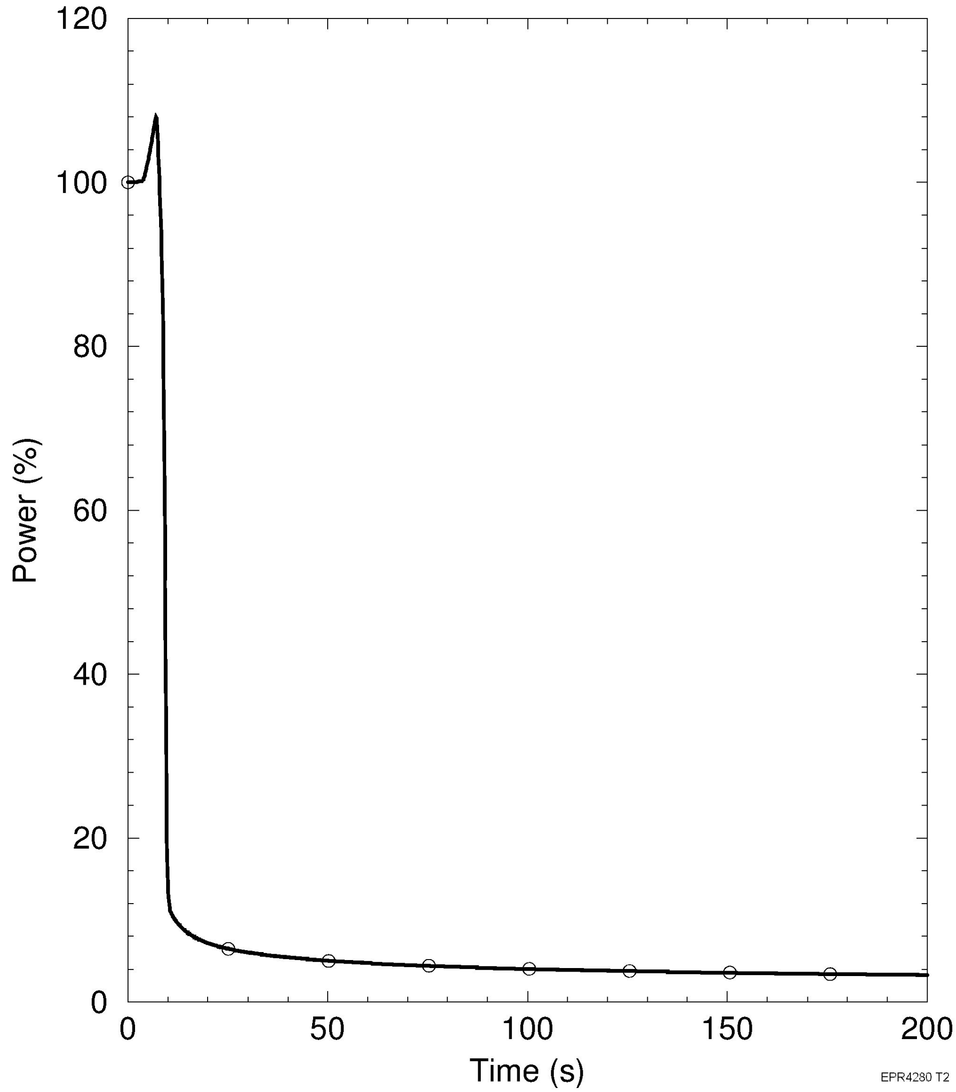

Figure 15.1-39—Inadvertent Opening of a SG Relief or Safety Valve - Reactivity

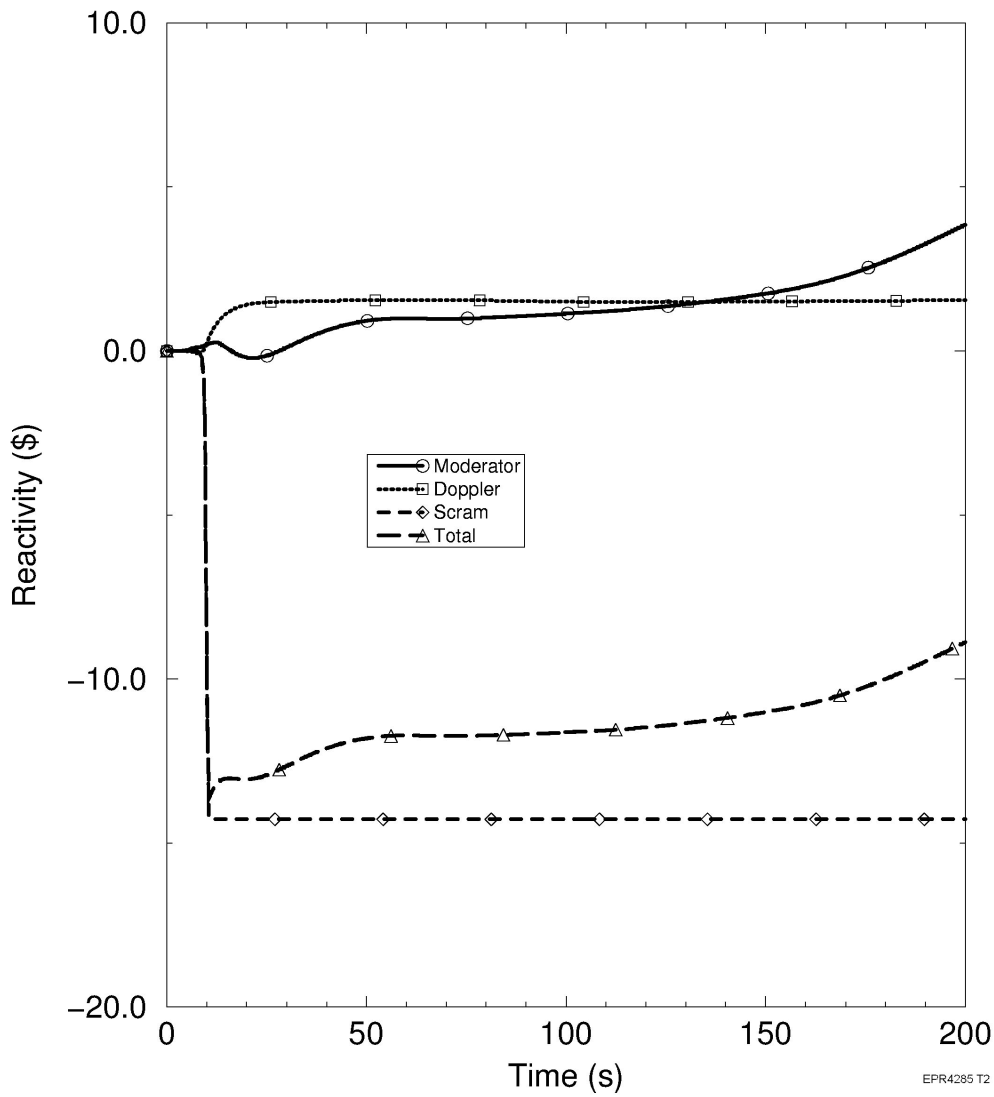

Figure 15.1-40—Inadvertent Opening of a SG Relief or Safety Valve - RCS and Secondary System Pressures

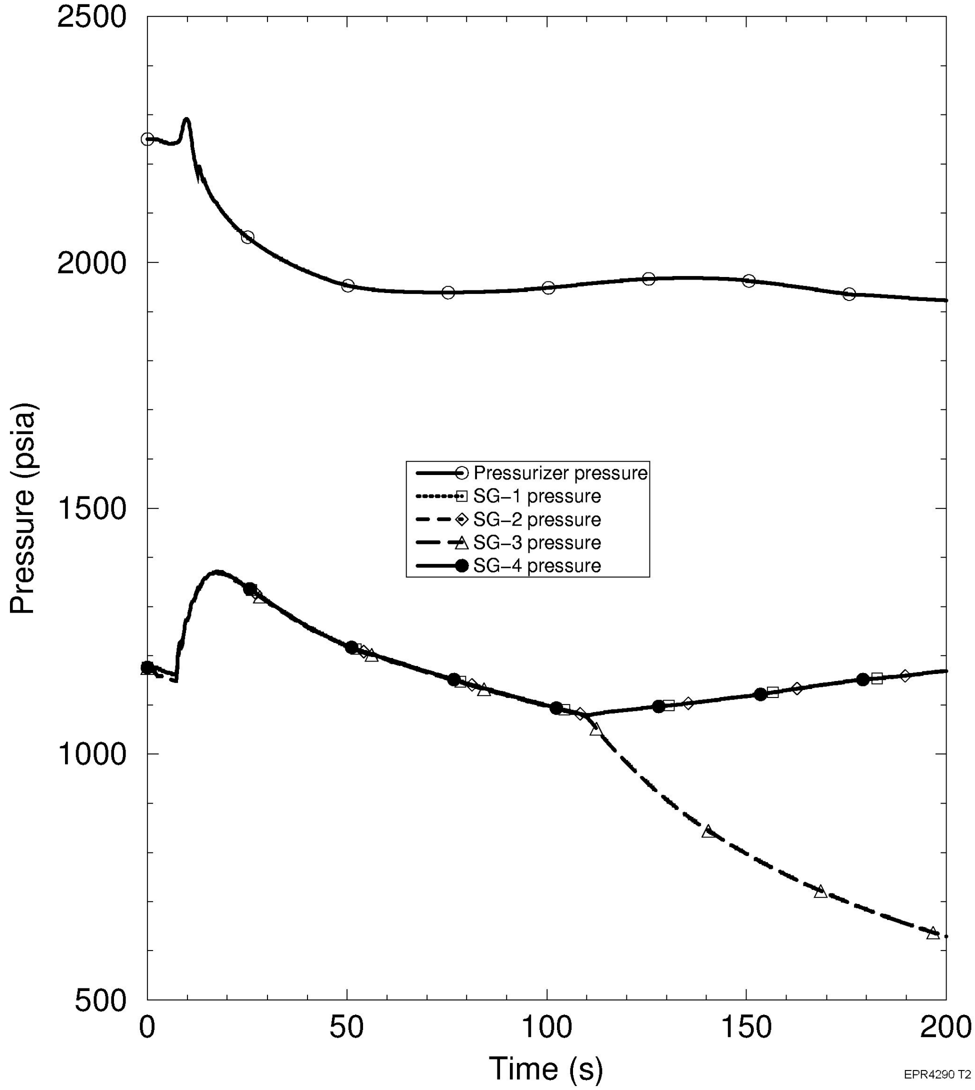

Figure 15.1-41—Inadvertent Opening of a SG Relief or Safety Valve - Pressurizer Level

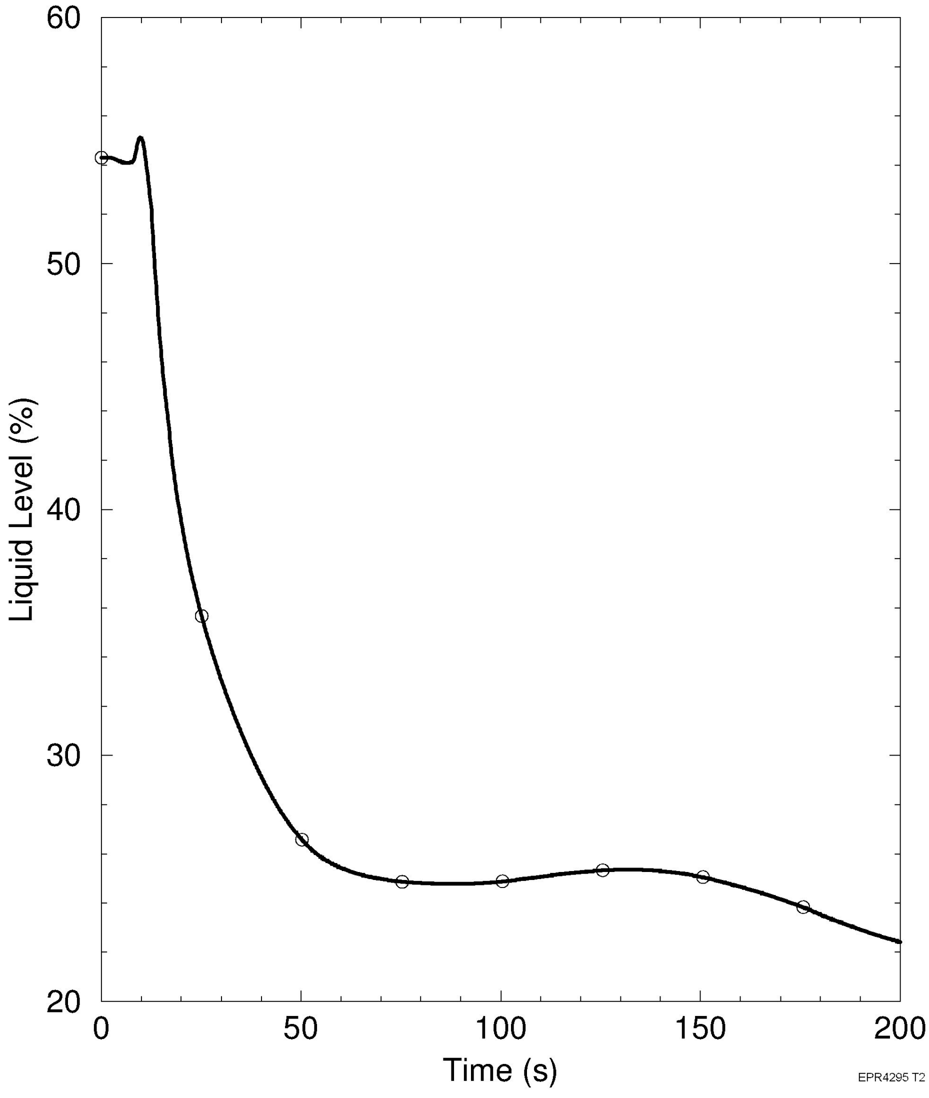

Figure 15.1-42—Inadvertent Opening of a SG Relief or Safety Valve - Steam Flow Rate

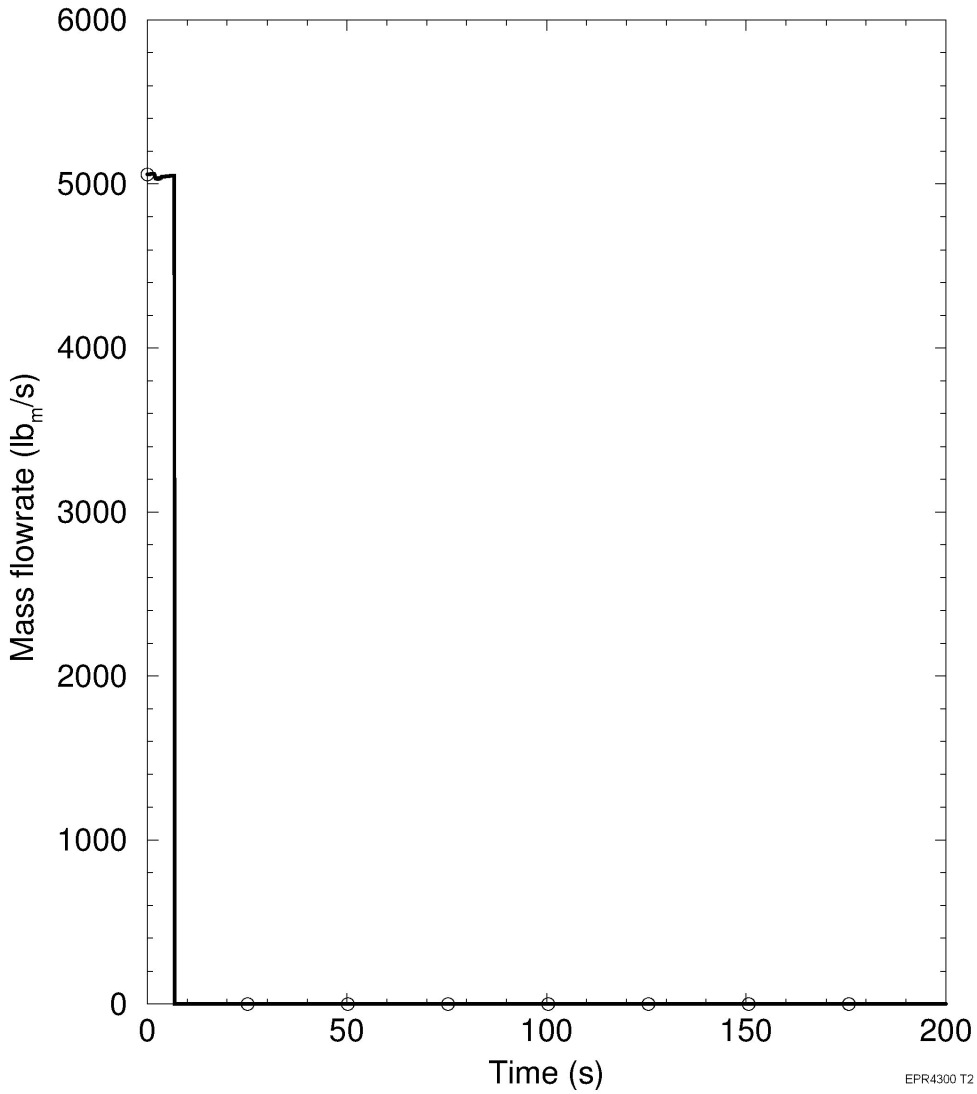

Figure 15.1-43—Inadvertent Opening of a SG Relief or Safety Valve - TSV Position

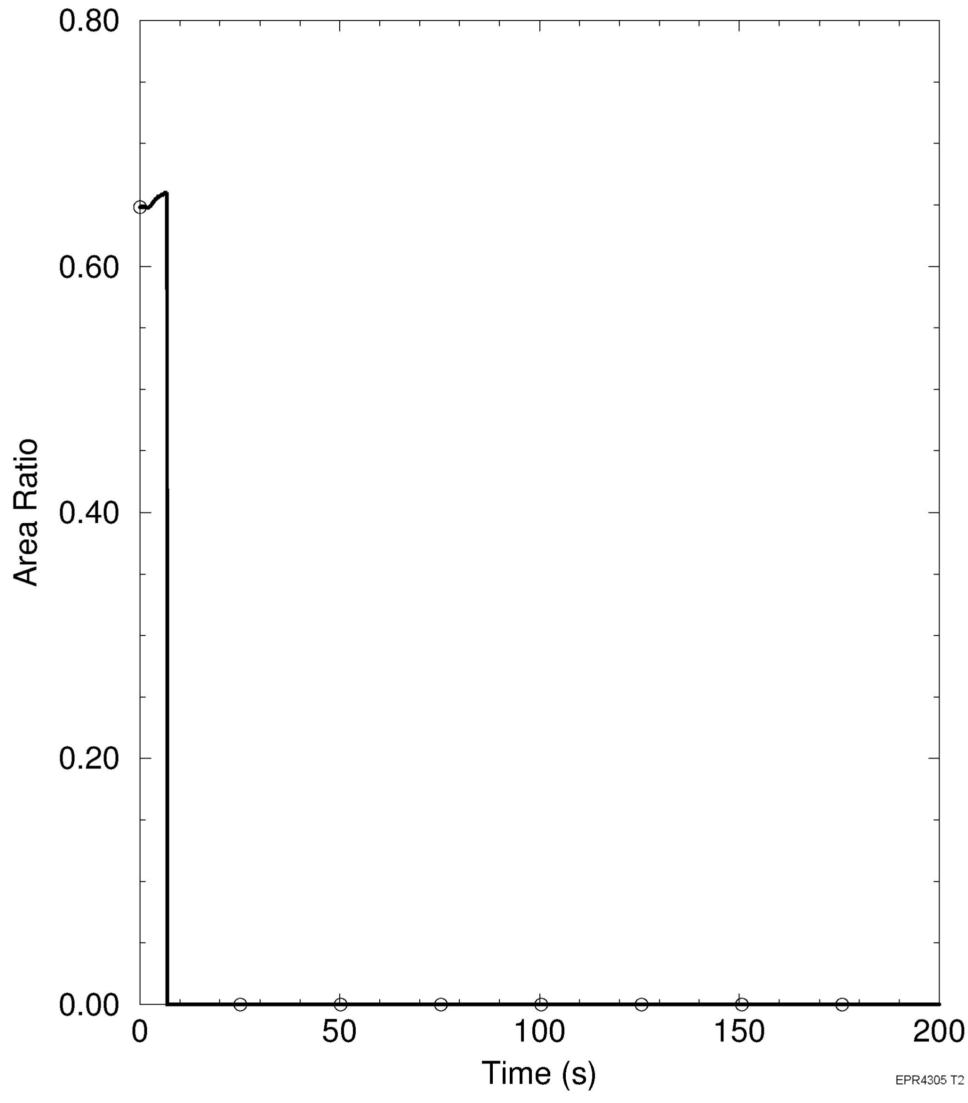

Figure 15.1-44—MSLB - Break Flow Rate

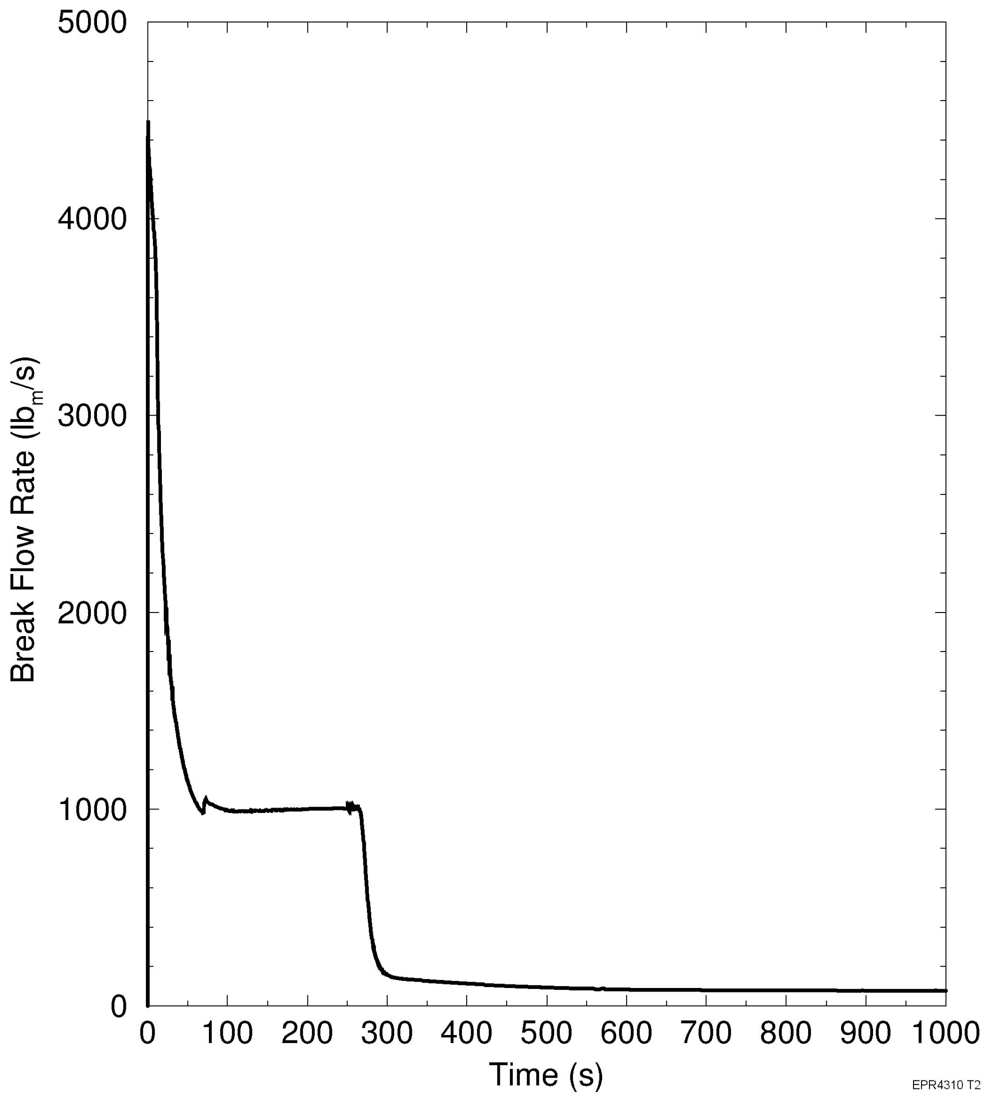

Figure 15.1-45—MSLB - Steam Generator Pressures

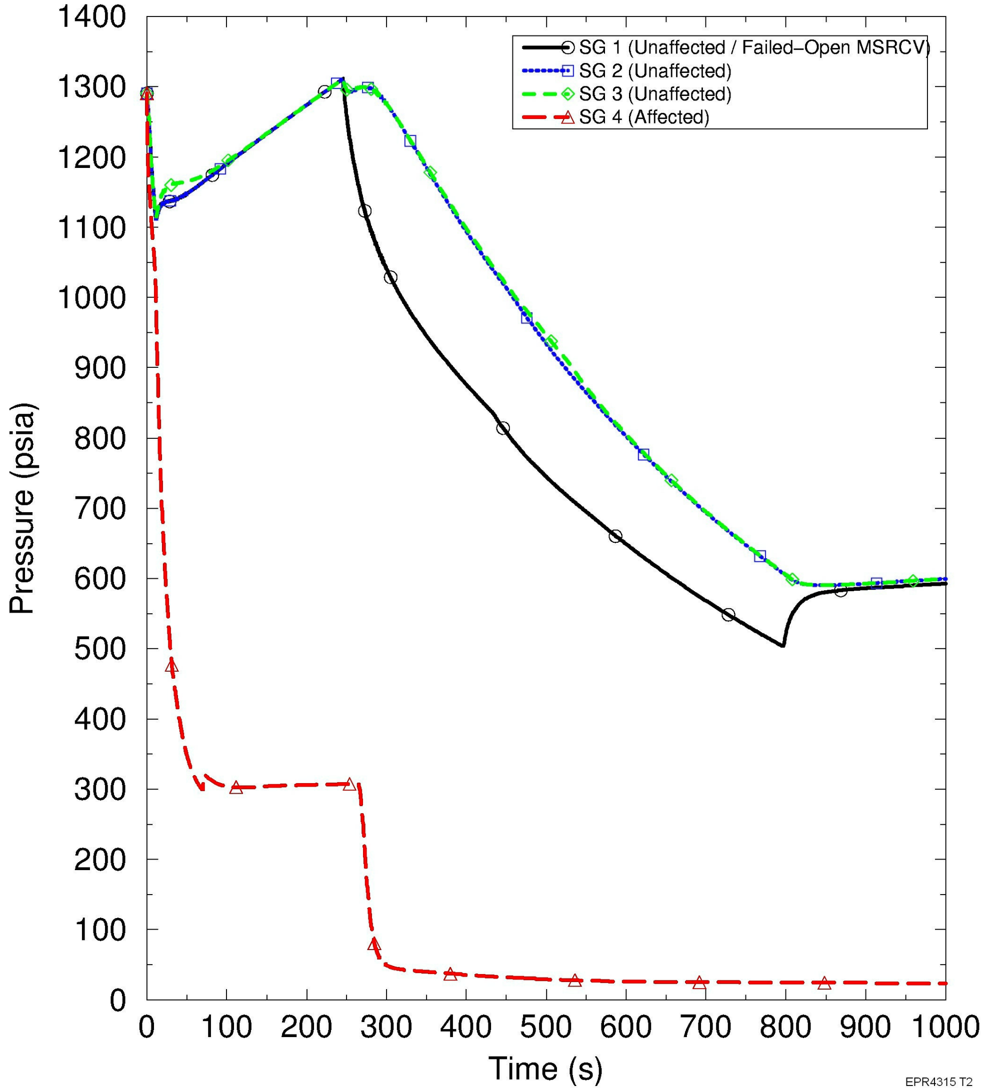

Figure 15.1-46—MSLB - Main Steam Relief Train Flow Rate

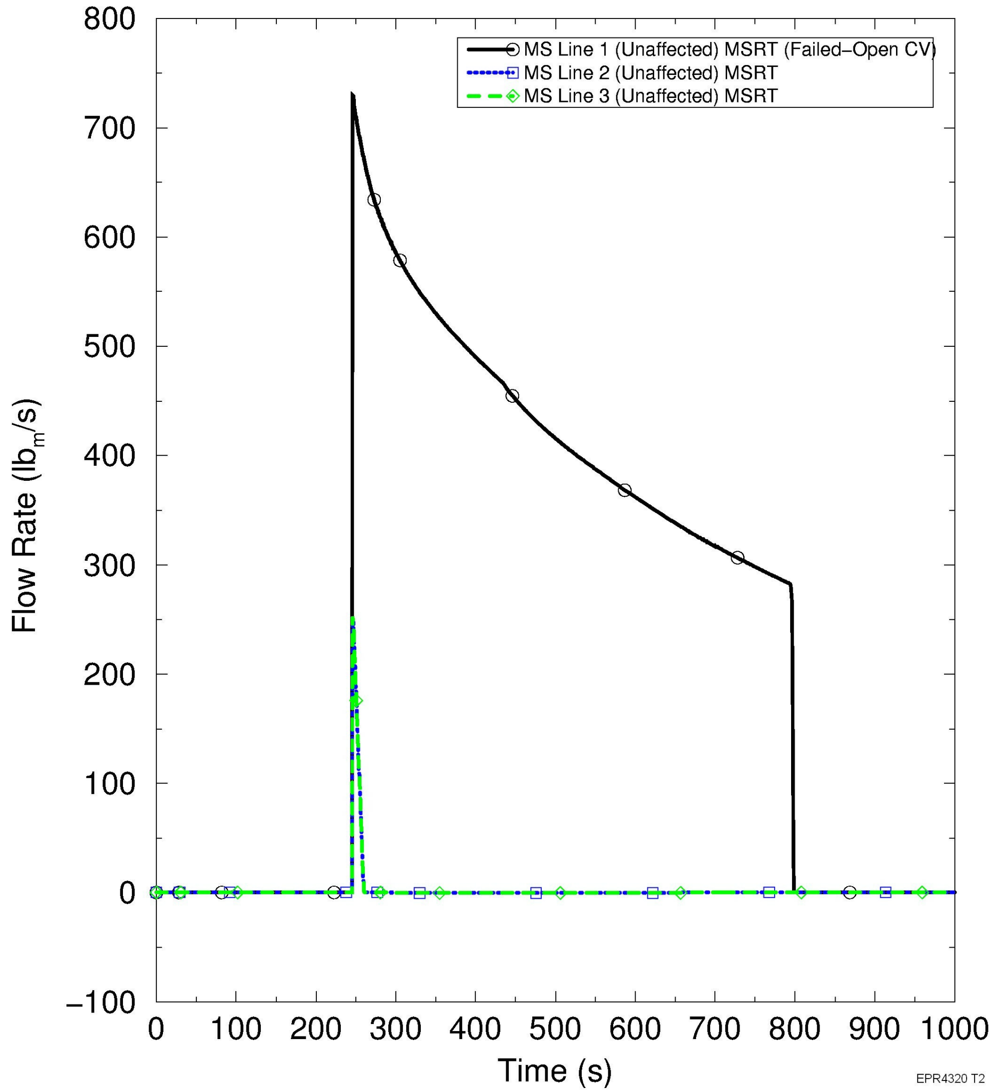

Figure 15.1-47—MSLB - Steam Generator Heat Transfer Rates

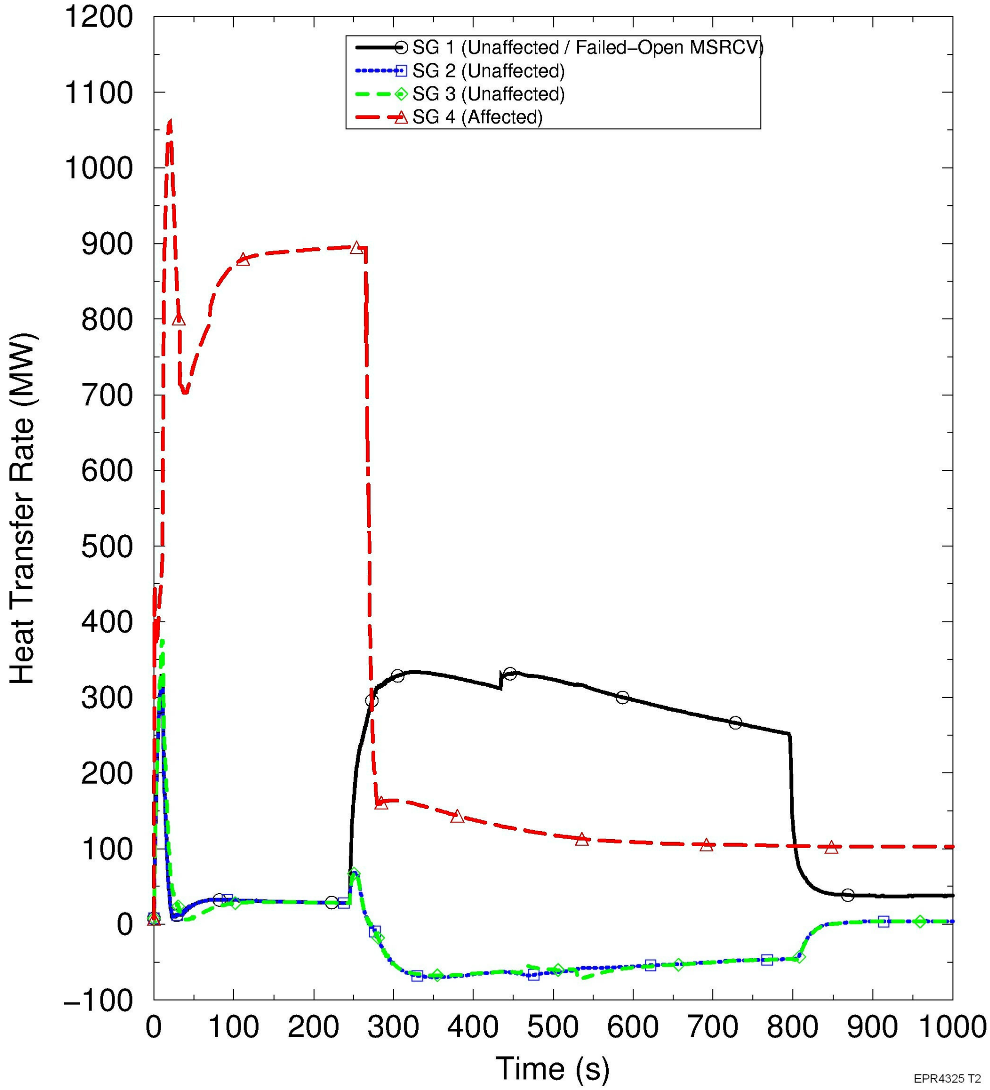

Figure 15.1-48—MSLB - Feedwater Flow Rates

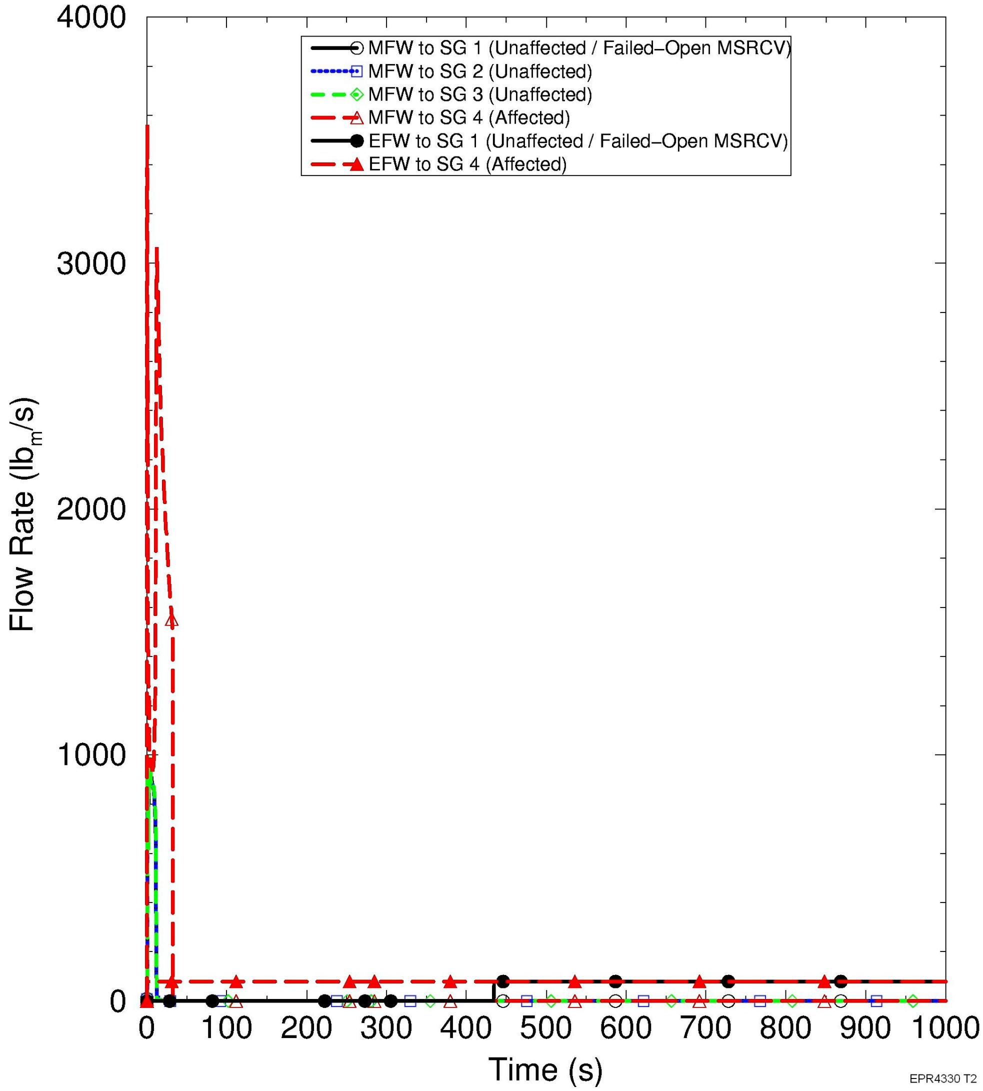

Figure 15.1-49—MSLB - Steam Generator Secondary-Side Liquid Inventories

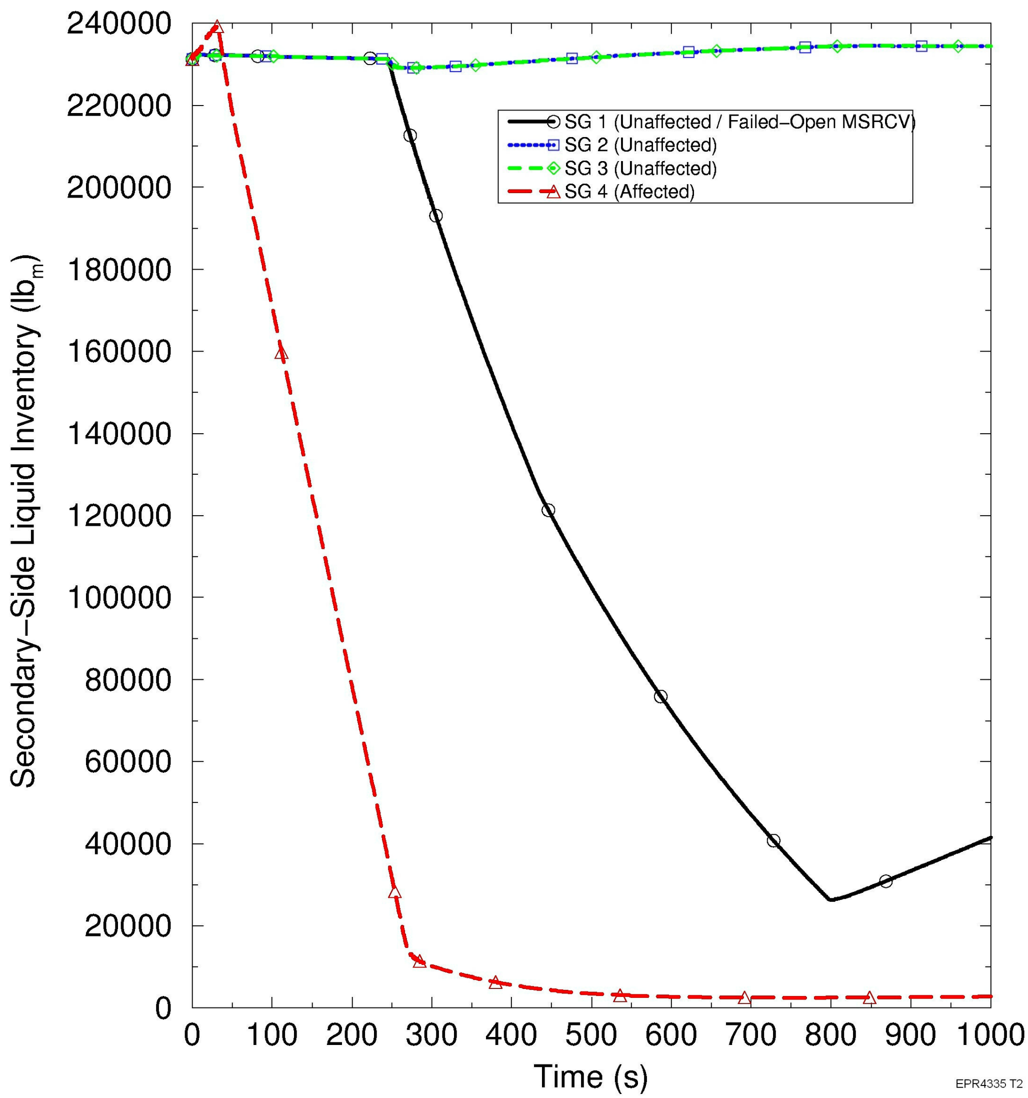

Figure 15.1-50—MSLB - Core Inlet Temperatures

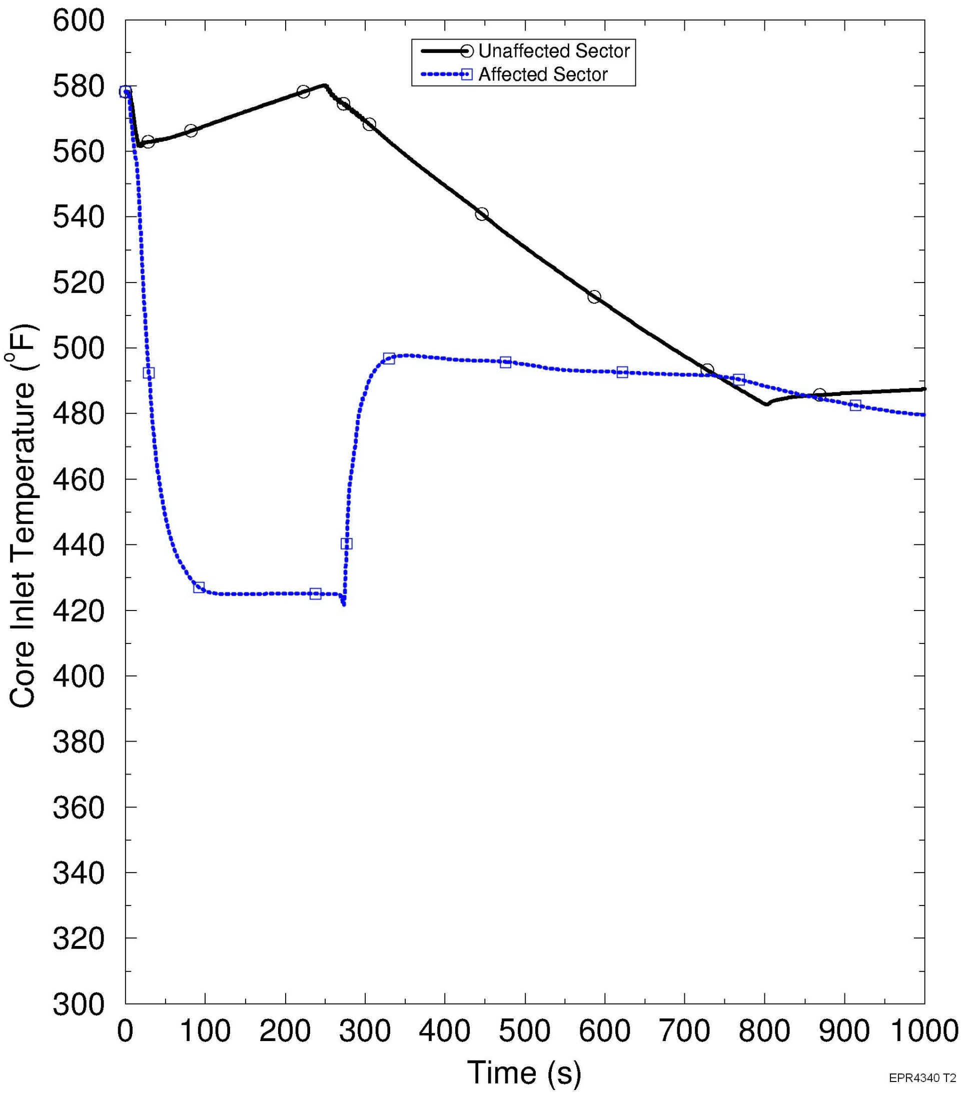

Figure 15.1-51—MSLB - Pressurizer Pressure

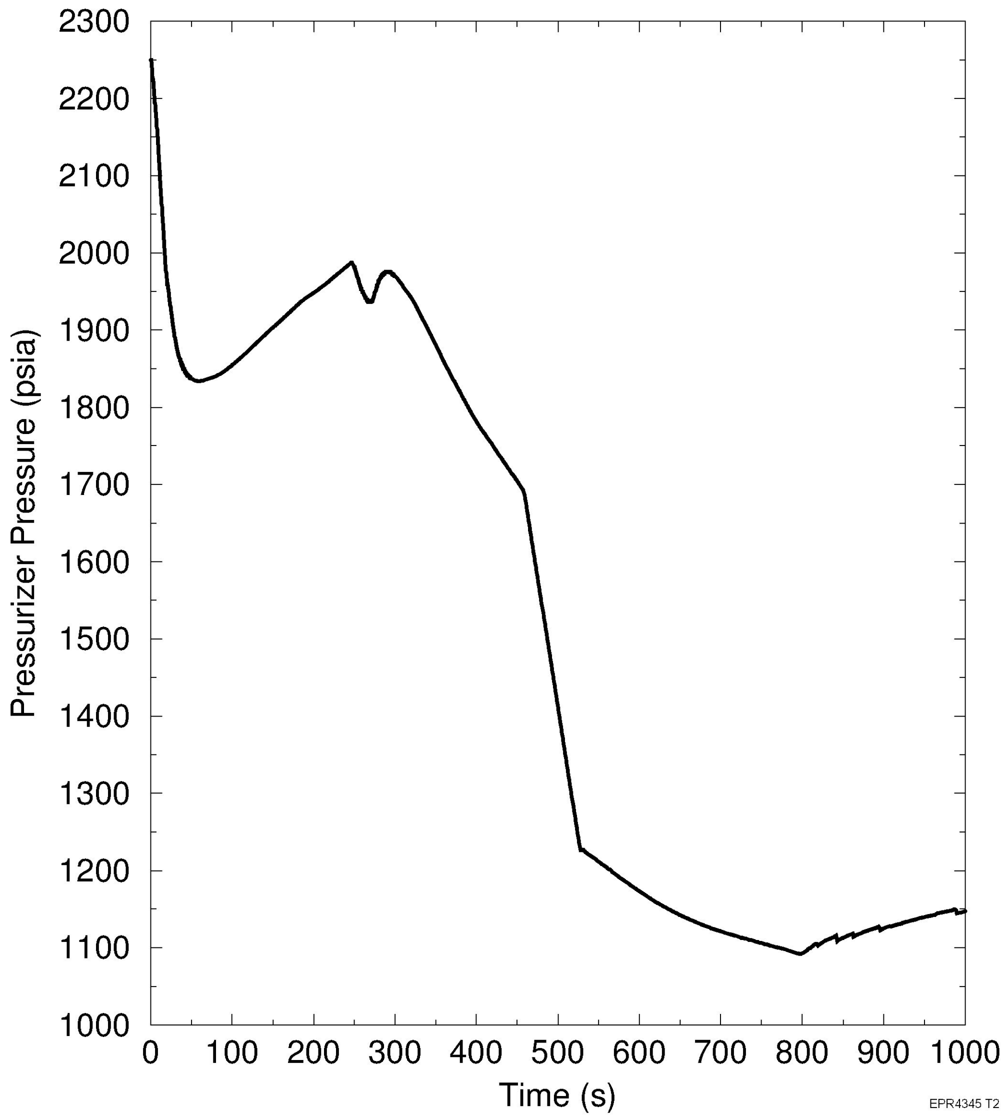

Figure 15.1-52—MSLB - Pressurizer Level

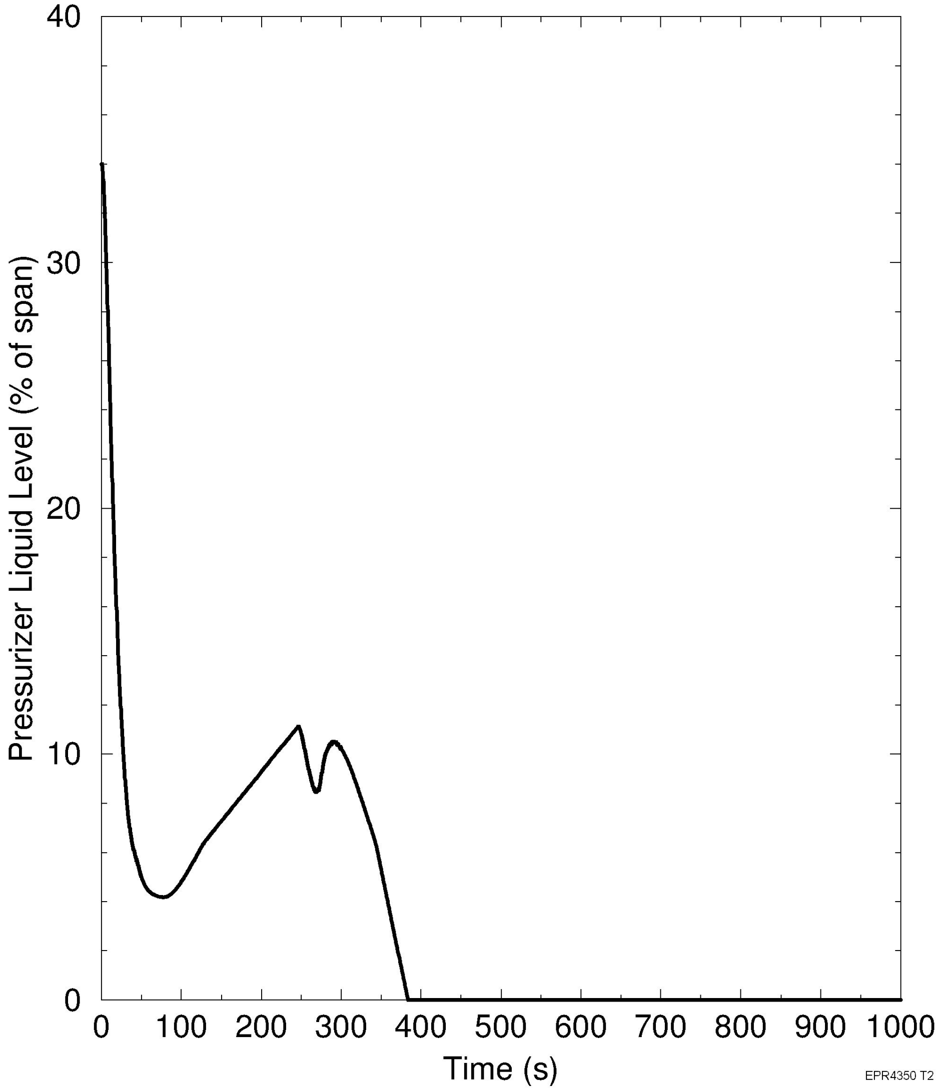

Figure 15.1-53—MSLB - Medium-Head Safety Injection Flow Rates

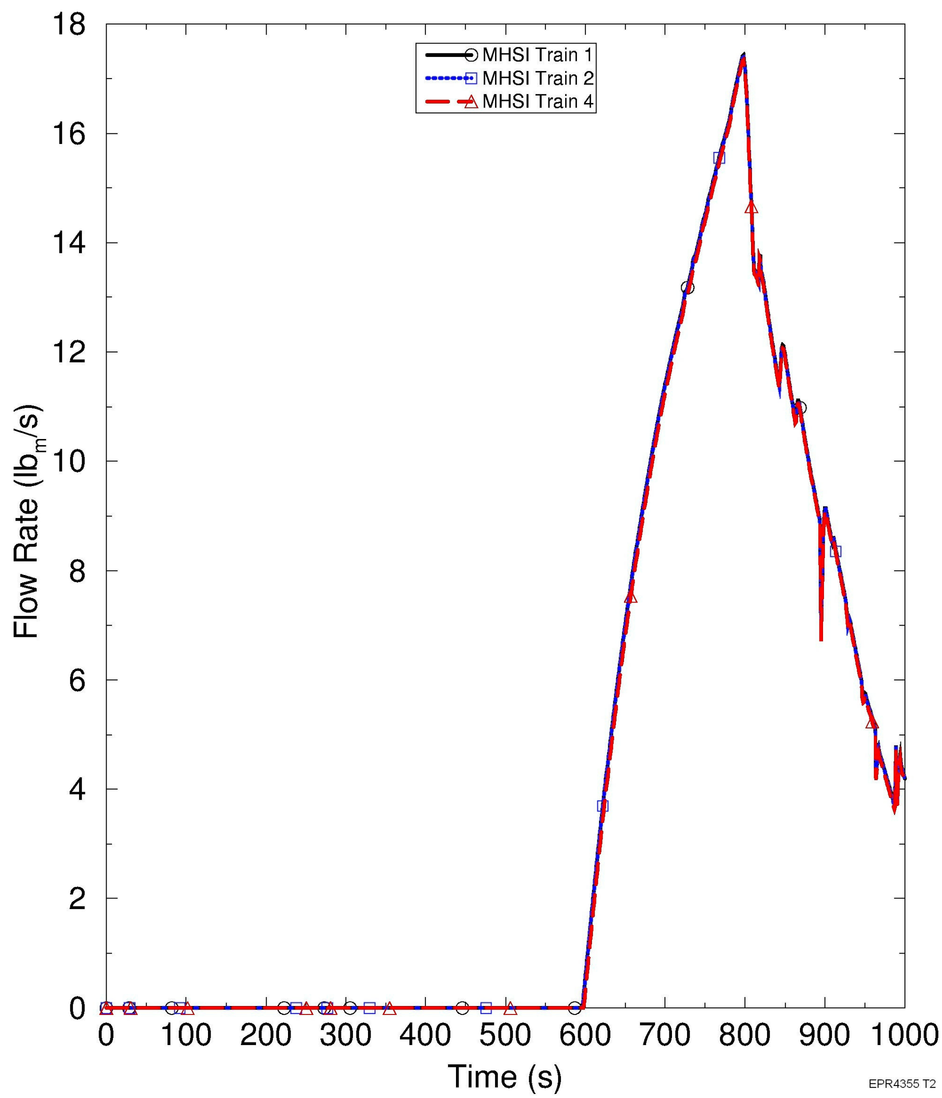

Figure 15.1-54—MSLB - Reactivity

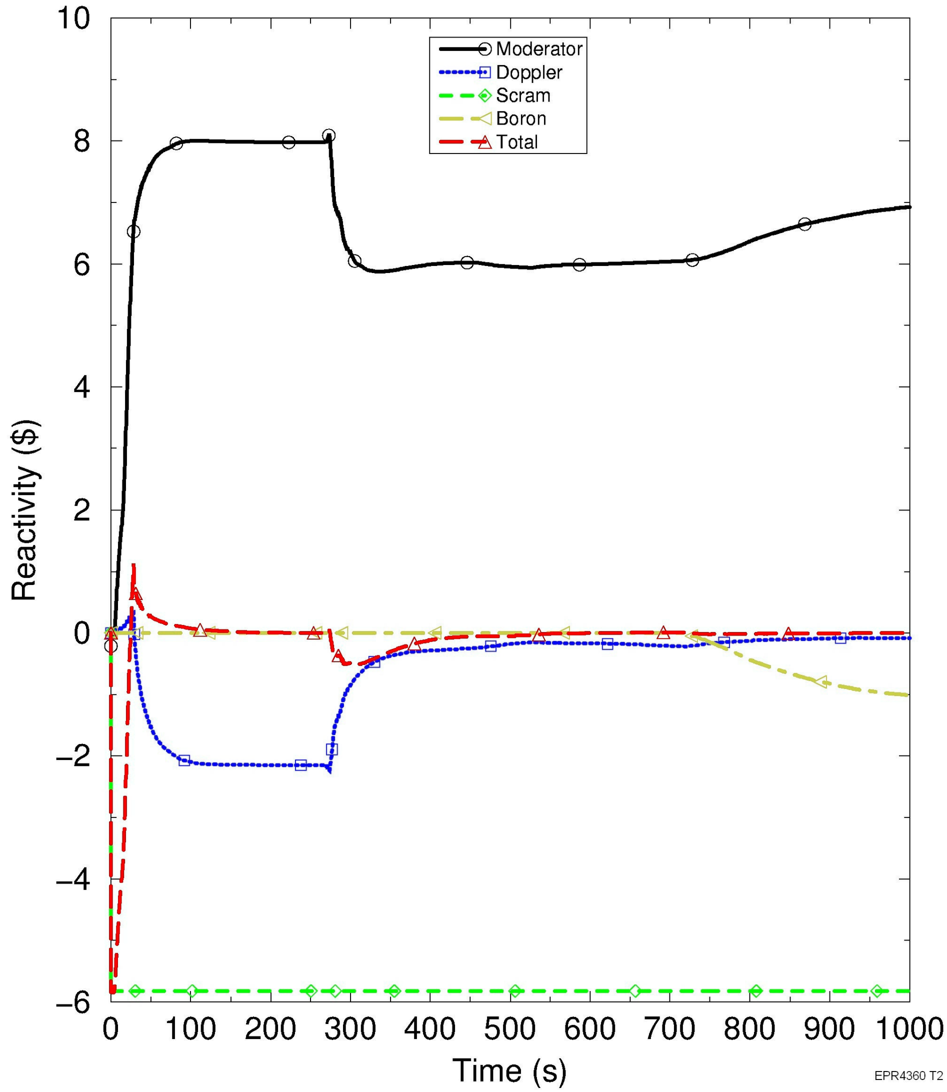

Figure 15.1-55—MSLB - Reactor Power

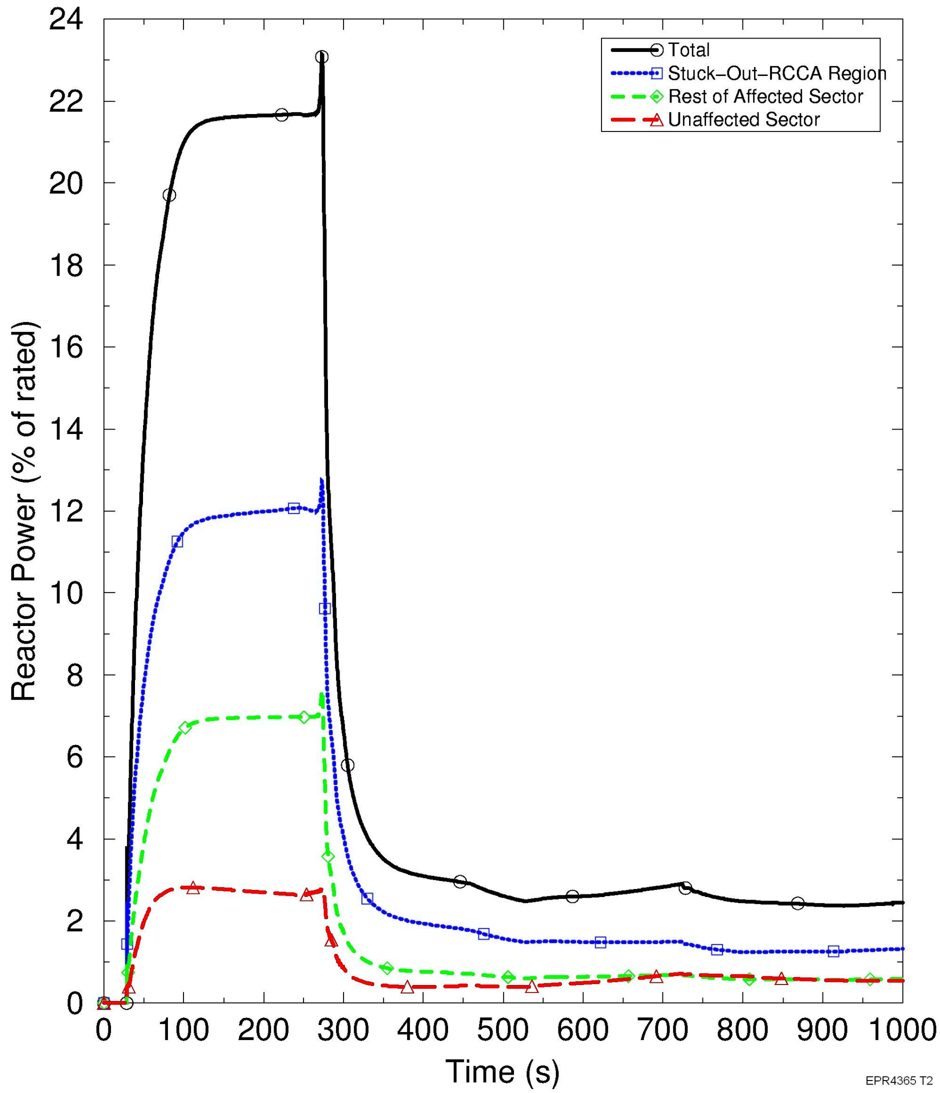

Figure 15.1-56—MSLB - Longer-Term Reactor Power

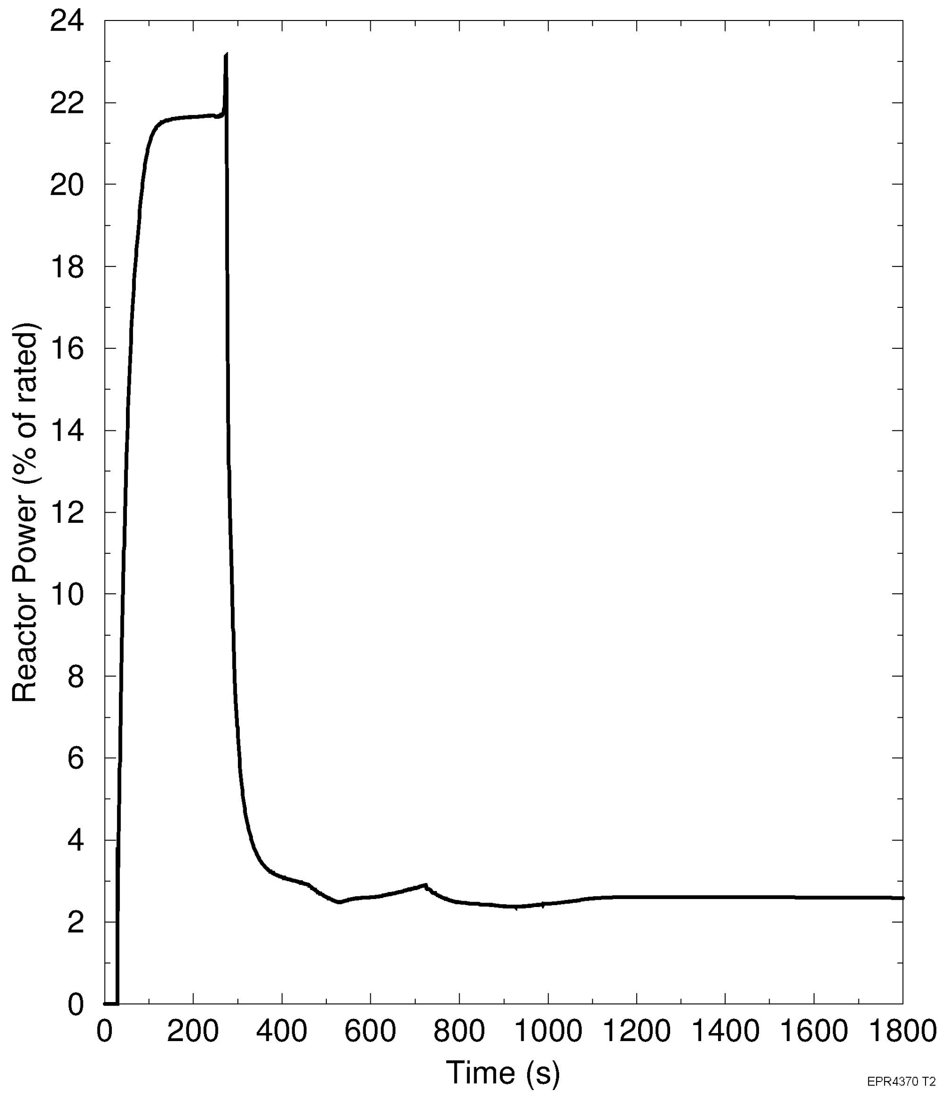

Figure 15.1-57—Decrease in Feedwater Temperature - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL

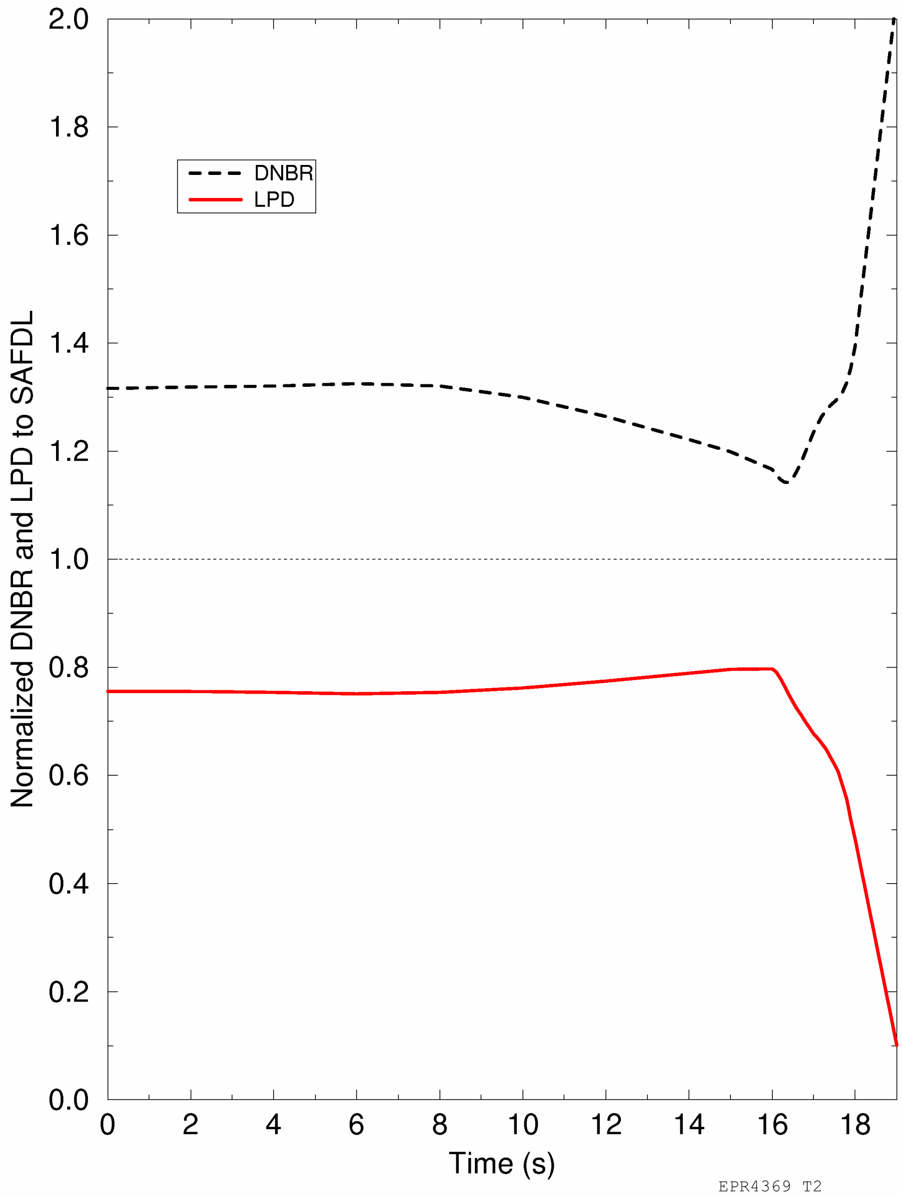

Figure 15.1-58—Increase in Main Feedwater Flow - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL

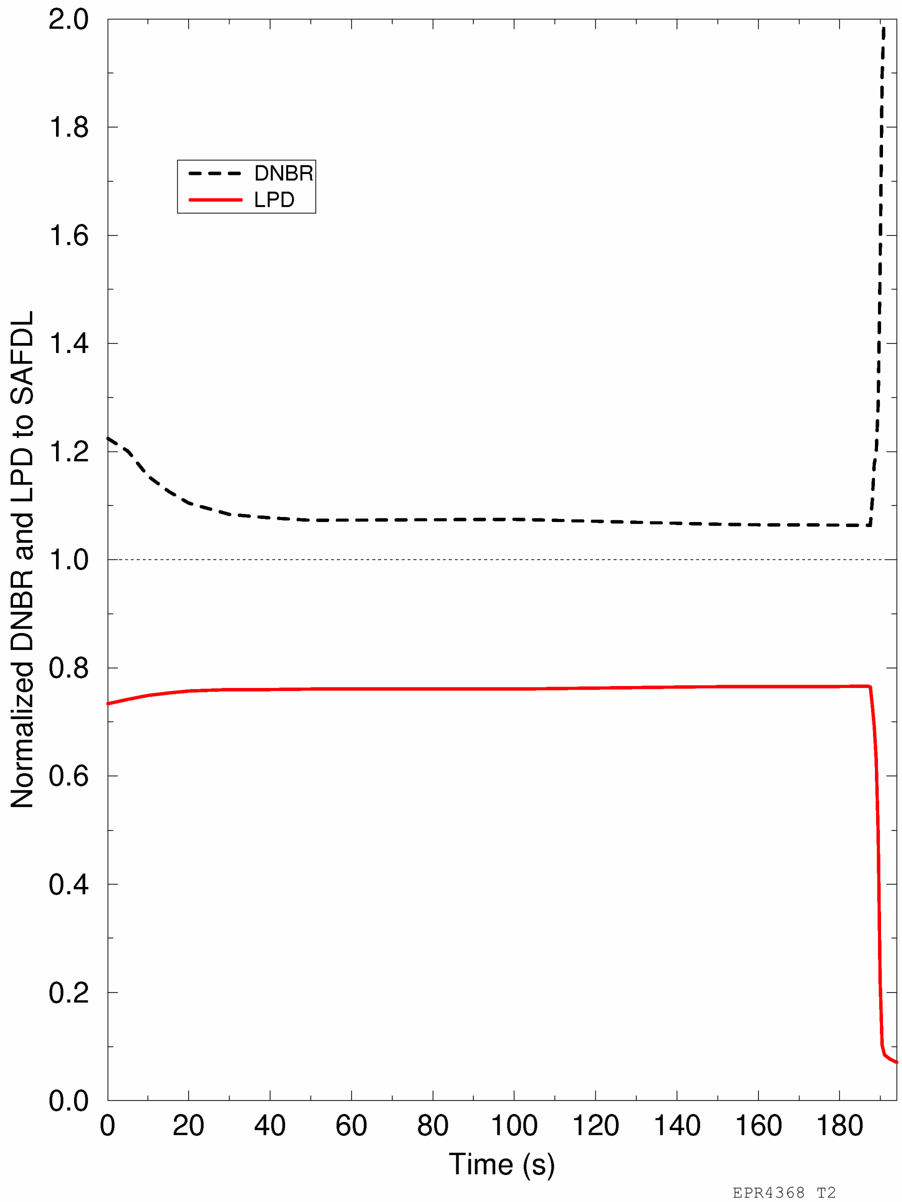

Figure 15.1-59—Increase in Steam Flow - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL

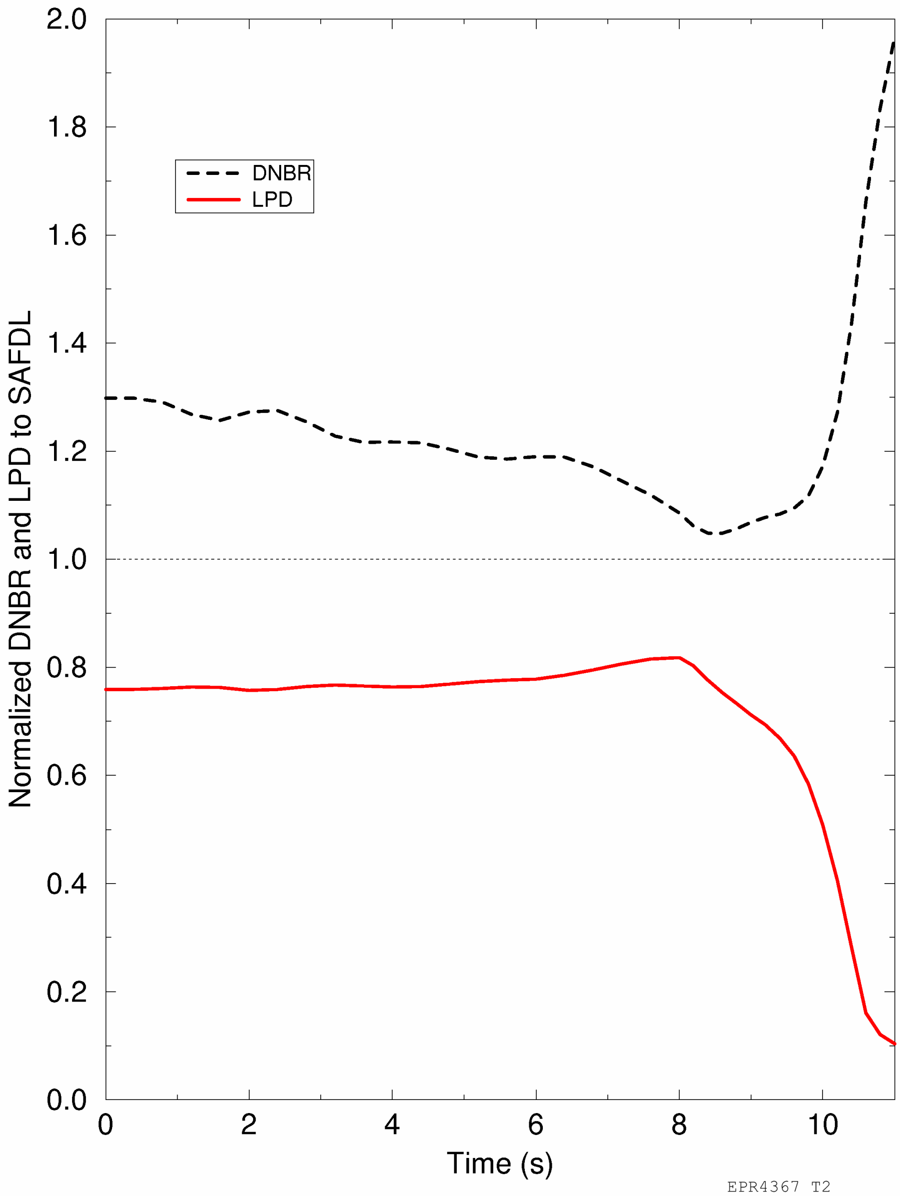

Figure 15.1-60—Inadvertent Opening of a SG Relief or Safety Valve - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL

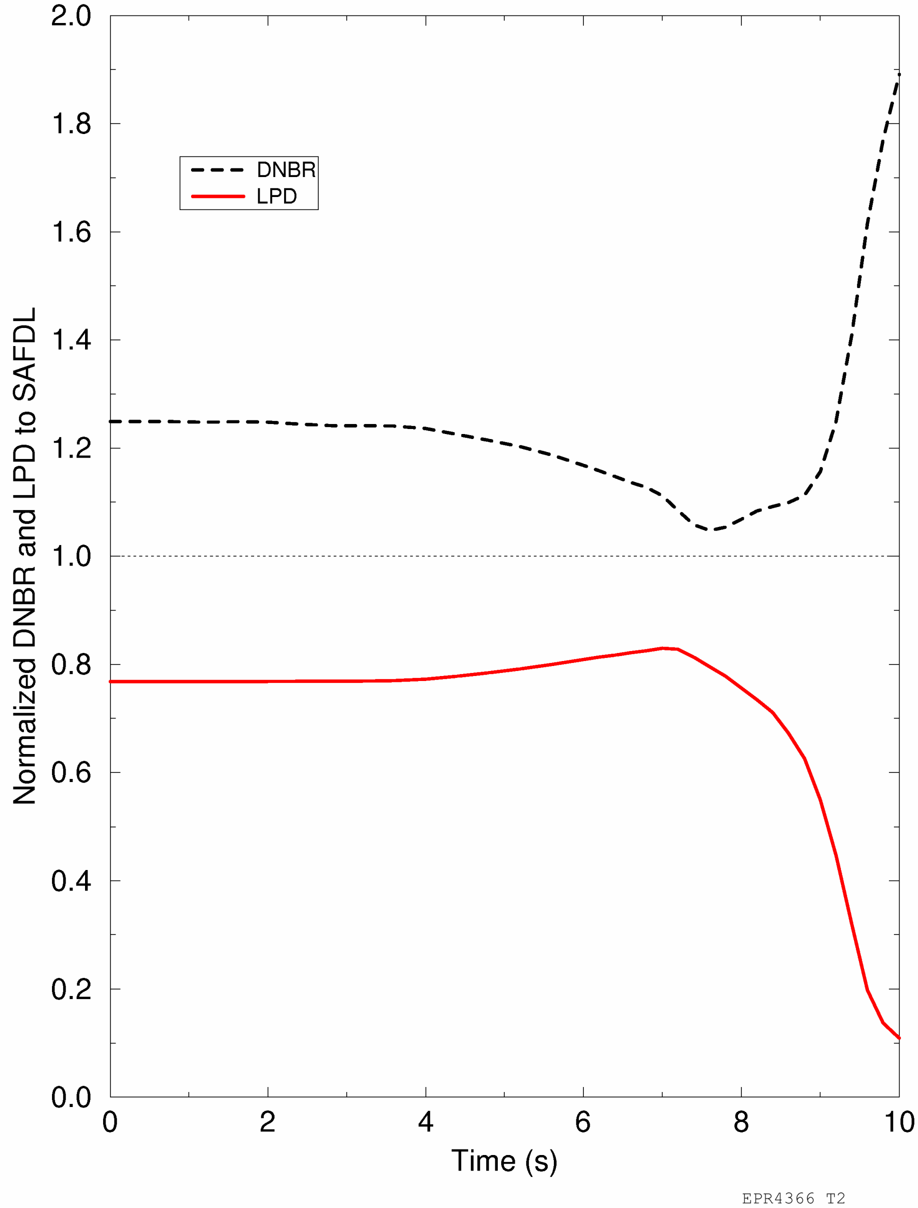

Figure 15.1-61—MSLB (small break, pre-scram) - Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL