15.3 Decrease in Reactor Coolant System Flow Rate

Descriptions are presented for each of the following reactor coolant system (RCS) flow decrease events; the first two events are anticipated operational occurrences (AOO), and the last two are postulated accidents (PA):

- Section 15.3.1 - Partial loss of forced reactor coolant flow.
- Section 15.3.2 - Complete loss of forced reactor coolant flow.
- Section 15.3.3 - Reactor coolant pump (RCP) rotor seizure.
- Section 15.3.4 - RCP shaft break.

15.3.1 Partial Loss of Forced Reactor Coolant Flow

15.3.1.1 Identification of Causes and Event Description

A partial loss of forced reactor coolant flow is an AOO that may be caused by a mechanical or electrical failure in a pump motor, a fault in the power supply to the pump motor, or a pump motor trip caused by anomalies such as overcurrent or phase imbalance. The resulting decrease in reactor coolant flow, which occurs while the plant is at power, degrades the core heat transfer and reduces the departure from nucleate boiling (DNB) margin.

The loss of forced flow in a single primary system loop causes a partial reactor trip (RT) at the low-flow setpoint. The partial RT is not credited because it is not safety related. The reactor is tripped when the flow in a single loop reaches the low-low flow protection system (PS) setpoint and the reactor power is above the P3 partial trip permissive. The loss of forced flow in multiple loops generates an RT at the low flow or low RCP speed setpoints. These RT functions are described in Section 7.2.1.2.

The two-pump loss of flow event is bounded by the one-pump loss of flow event because the PS generates an earlier RT at the higher low flow setpoint (in two loops) rather than the low-low flow setpoint (in one loop). This condition more than compensates for the coastdown of two pumps. The three-pump loss of flow event is not credible because there is no single fault that can de-energize three pumps.

The purpose of analyzing the partial loss of forced reactor coolant flow event for the U.S. EPR, as described in the Codes and Methods Applicability Report for the U.S. EPR, Section 5.2.2.3.1.1 (Reference 1), is to protect the DNB-specified acceptable fuel design limit (SAFDL). The fuel centerline melt SAFDL is not challenged because no significant increase in core power occurs for this event. This event does not challenge pressure limits.

15.3.1.2 Method of Analysis and Assumptions

The systems response methodology for this event (Reference 1) uses the S-RELAP5 computer code to simulate the responses of the primary and secondary coolant systems, reactor, protective equipment and systems, and automatic controllers. The change in DNB ratio (DNBR) is then calculated with the LYNXT core thermal-hydraulic computer code (see Section 15.0.2.3) using the RCS response from S-RELAP5 as a boundary condition. Section 15.0.2.4 provides a description of the S-RELAP5 code and the DNBR methodology is described in the Incore Trip Setpoint and Transient Methodology for U.S. EPR (Reference 3).

The analysis is biased for the conservative evaluation of the DNB SAFDL (Table 15.3.1—Decrease in Reactor Coolant System Flow Rate Events - Key Parameters). The event is initialized therefore at hot full power (HFP) plus measurement uncertainty in order to provide the largest initial power-to-flow ratio. Beginning-of-cycle (BOC) reactivity feedback coefficients are used to maximize reactor power during the event. The inertia of the RCPs is reduced to accelerate their coastdown. This condition makes the DNB results more severe and provides margin for the actual RCP coastdown performance. RT is delayed by using RT trip setpoints that have been reduced by the normal uncertainty (Table 15.0-7—Reactor Trip Setpoints and Delays Used in the Accident Analysis) and by using the maximum time delays. Conservative RT reactivity insertion characteristics (Figure 15.0-4—Normalized RCCA Reactivity Worth as a Function of Rod Drop Time) are assumed, including the maximum time delay with the most reactive rod held out of the core.

The S-RELAP5 model includes five percent steam generator (SG) tube plugging. Primary and secondary system pressures, average coolant temperature, and pressurizer (PZR) and SG levels are at their nominal hot full power (HFP) values. Uncertainties in RCS pressure and temperature are addressed in the DNBR analyses. Additionally, a loss of offsite power (LOOP) is assumed to occur coincident with turbine trip (TT). This assumption conservatively de-energizes the remaining RCPs. The PS logic generates a TT one second after the RT de-energizes the control rod drives (refer to Section 7.3.1.2.17).

The non-safety-related PZR spray system is modeled because it has the potential to make the DNB consequences more severe by limiting the RCS pressure increase during the event. There are no single failures that make this event worse. Table 15.3-2—Decrease in Reactor Coolant System Flow Rate Events - Status of Key Equipment presents the status of key non-safety-related equipment. Operator action is not credited to mitigate this event.

15.3.1.3 Results

Table 15.3-3—Partial Loss of Forced Reactor Coolant Flow Event - Sequence of Events presents the sequence of events for the partial loss of forced reactor coolant flow event. The transient is initiated by the assumed spurious trip of one RCP. The coastdown of the pump causes a decrease in flow in the affected loop and an increase in flow in the unaffected loops. The net core flow decreases (Figure 15.3-1—Partial Loss of Forced Reactor Coolant Flow Event - RCS Flow Rates). Reactor power and core heat flux stay relatively constant until RT (see Figures 15.3-2—Partial Loss of Forced Reactor Coolant Flow Event - Reactor Power and 15.3-3—Partial Loss of Forced Reactor Coolant Flow Event - Core Average Heat Flux, respectively).

The RCS pressure increase causes the PZR sprays to initiate (Figure 15.3-4—Partial Loss of Forced Reactor Coolant Flow Event - Pressurizer Pressure). An RT signal is generated when the flow rate in the affected loop decreases to the safety-related, low-low flow setpoint. The turbine is tripped one second later, and the remaining RCPs are de-energized due to the assumed LOOP. The time of maximum reduction in DNBR (DNBR) occurs before TT, demonstrating that it is not affected by the assumed LOOP. The ΔDNBR does not challenge the DNB SAFDL. Core inlet temperature is used in the determination of DNBR. A plot is not provided because it changes little during the period of minimum DNBR. A plot of PSRV flow rate is not provided because the PSRVs do not open during the period analyzed.

In the long term, the RCS pressure increases as the conditions for natural circulation are established. The PZR safety relief valves (PSRVs) open to control pressure. The main steam relief trains (MSRTs) and the emergency feedwater (EFW) system actuate to provide sufficient decay heat removal capability. The continued removal of decay heat establishes and maintains a stable, controlled condition until the operator initiates a normal plant cooldown.

15.3.1.4 Radiological Consequences

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs during this event.

15.3.1.5 Conclusions

The fuel-cladding integrity is maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.

15.3.1.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.3.1 events included in NUREG-0800, Section 15.3.1–15.3.2, (Reference 2) and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
   - Response: The RCS pressure increases as a result of the partial loss of flow, but the event does not challenge the design limit.
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 15.3.1.5.
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 a more serious plant condition.
4. The requirements stated in RG 1.105, "Instrument Spans and Setpoints," are evaluated 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 7.1.2.4.7. The instrument spans for the RT functions credited are provided in Table 7.2-1—Reactor Trip Variables.
5. Onsite and offsite electric power systems must be maintained so safety-related SSCs function during normal operation and AOOs.
   - Response: Section 8.1.4 presents the electric power design basis and describes how onsite and offsite electrical power is maintained so that the credited safety-related systems and components function.
6. The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, must be assumed in the analysis and should follow the guidance stated in RG 1.53.
   - Response: There are no single failures that make this event more severe.
7. The performance of non-safety-related systems during transients and accidents and of single failures of active and passive systems (especially the performance of check valves in passive systems), must be evaluated and verified by the guidance of SECY 77-439 as cited in Reference 2, SECY 94-084 as cited in Reference 2, and RG 1.206.
   - Response: The effect of non-safety-related systems has been evaluated, and the PZR spray system is credited because it has the potential to make the event more severe.

8. The applicant's analysis of the most limiting AOOs should use an acceptable model. Unapproved analytical methods proposed by the applicant are evaluated by the staff for acceptability.

   – Response: The analysis uses the models and methods described in Section 15.3.1.2.

9. 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, Reference 2). 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 (Section 15.3.1.2). 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 are assumed as described in Section 15.3.1.2.

   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: A conservative core burnup is selected as described in Section 15.3.1.2.

   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 as described in Section 15.3.1.2.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

15.3.2.1 Identification of Causes and Event Description

The complete loss of forced reactor coolant flow may result from a simultaneous fault in the electrical power supplies of the RCPs. The resulting decrease in reactor coolant flow (occurring while the plant is at power) degrades the core heat transfer and reduces the DNB margin.

The complete loss of forced reactor coolant flow causes an RT at either the low RCP speed setpoint or the low loop flow setpoint.

The purpose of analyzing the complete loss of forced reactor coolant flow event for the U.S. EPR (Section 5.2.2.3.1.1 of Reference 1) is to protect the DNB SAFDL. The fuel centerline melt SAFDL is not challenged because there is no significant increase in core power for this event. The pressure limits are not challenged by this event.

15.3.2.2 Method of Analysis and Assumptions

The methods of analysis and assumptions used in the analysis of this event are described in Section 15.3.1.2 except as follows:

- The initiating event is an assumed LOOP that is simulated by tripping the four RCPs and isolating main feedwater (MFW).
- TT is delayed 3.0 s after RT for conservatism so that the pressure increase caused by TT occurs after the time of maximum ΔDNBR.
- The DNBR associated with the free coastdown of the RCPs due to a LOOP bounds that caused by a frequency decay of up to 3.5 Hz/s. Therefore, only the results of the LOOP-initiated event are provided.
- The PS logic initiates RT when two out of the four loops generate a low RCP speed signal. In order to demonstrate that the necessary functionality is preserved when one PS division (coolant loop) is out of service for maintenance and another fails (single failure), it is assumed that the reactor trips only when the low RCP speed setpoint is reached in the coolant four loops.

15.3.2.3 Results

Table 15.3-4—Complete Loss of Forced Reactor Coolant Flow Event - Sequence of Events presents the sequence of events for the complete loss of forced reactor coolant flow event. The transient is initiated by an assumed LOOP, which de-energizes the four RCPs and causes a decrease in the core flow rate (Figure 15.3-5—Complete Loss of Forced Reactor Coolant Flow Event - RCS Flow Rates). Reactor power and core heat flux stay relatively constant until RT (Figures 15.3-6—Complete Loss of Forced Reactor Coolant Flow Event - Reactor Power and 15.3-7—Complete Loss of Forced Reactor Coolant Flow Event - Core Average Heat Flux, respectively).

RCS pressure increases (Figure 15.3-8—Complete Loss of Forced Reactor Coolant Flow Event - Pressurizer Pressure), which causes the PZR sprays to initiate. An RT signal is generated when the low RCP speed setpoint is reached in the four loops. The turbine is assumed to trip three seconds later to conservatively avoid an immediate increase in RCS pressure. The maximum reduction in ΔDNBR occurs before TT and does not challenge the DNB SAFDL. Core inlet temperature is used in the determination of DNBR. A plot is not provided because it changes little during the period of minimum

DNBR. A plot of PSRV flow rate is not provided because the PSRVs do not open during the period analyzed.

In the long term, the RCS pressure increases as the conditions for natural circulation are established. The PSRVs open to control pressure. The MSRTs and EFW system actuate to provide sufficient decay heat removal capability. The continued removal of decay heat establishes and maintains a stable, controlled condition until the operator initiates a normal plant cooldown.

15.3.2.4 Radiological Consequences

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs.

15.3.2.5 Conclusions

The fuel-cladding integrity is maintained by keeping the minimum DNBR above the 95 percent probability/95 percent confidence DNBR limit.

15.3.2.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.3.2 events included in NUREG-0800, Section 15.3.1–15.3.2, (Reference 2) and descriptions of how these criteria are met are listed below:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
   - Response: The RCS pressure increases as a result of the complete loss of flow, but the event does not challenge the design limit.
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 15.3.2.5.
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 a more serious plant condition.
4. The requirements stated in RG 1.105, "Instrument Spans and Setpoints," are evaluated 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 7.1.2.4.7. The instrument spans for the RT functions credited are provided in Table 7.2-1—Reactor Trip Variables.

5. Onsite and offsite electric power systems must be maintained so safety-related SSCs function during normal operation and AOOs.

   - Response: Section 8.1.4 presents the electric power design basis, and describes how onsite and offsite electrical power is maintained so that the credited safety-related systems and components function.

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

   - Response: The single failure of one PS division is assumed in the analysis.

7. The performance of non-safety-related systems during transients and accidents and of single failures of active and passive systems (especially the performance of check valves in passive systems), must be evaluated and verified by the guidance of SECY 77-439 as cited in Reference 2, SECY 94-084, as cited in Reference 2, and RG 1.206.

   - Response: The effect of non-safety-related systems has been evaluated, and the PZR spray system is credited because it has the potential to make the event more severe.

8. The applicant's analysis of the most limiting AOOs should use an acceptable model. Unapproved analytical methods proposed by the applicant are evaluated by the staff for acceptability.

   - Response: The analysis uses the models and methods described in Section 15.3.2.2.

9. 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, Reference 2). 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 (Section 15.3.2.2). 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 are assumed as described in Section 15.3.2.2.

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: A conservative core burnup is selected as described in Section 15.3.2.2.

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 as described in Section 15.3.2.2.

15.3.3 Reactor Coolant Pump Rotor Seizure

15.3.3.1 Identification of Causes and Event Description

The RCP rotor seizure (locked rotor) event is postulated to be an instantaneous seizure of an RCP rotor. The sudden decrease in core coolant flow while the reactor is at power degrades core heat transfer and can lead to fuel damage. The locked rotor causes flow in the faulted RCS loop to decrease rapidly, causing a partial RT when the flow reaches the low flow setpoint. This partial RT is not credited in the analysis because it is not safety related and makes the event less severe. An RT signal is generated when the flow reaches the safety-related, low-low flow setpoint and the reactor power is above the P3 permissive.

The purpose of analyzing the locked rotor event is to protect the DNB SAFDL. The fuel centerline melt SAFDL is not challenged because there is no significant increase in core power for this event. The pressure limits are not challenged by this event.

15.3.3.2 Method of Analysis and Assumptions

The methods of analysis and assumptions used in the analysis of this event are described in Section 15.3.1.2. The methods used to determine the radiological consequences are described in Section 15.0.3.8.

15.3.3.3 Results

Table 15.3-5—Reactor Coolant Pump Rotor Seizure Event - Sequence of Events presents the sequence of events for the rotor seizure event. The RCP rotor is assumed to lock instantaneously to initiate the event. This condition causes a rapid decrease in the affected loop flow rate and causes flow to reverse within 2.0 s (Figure 15.3-9—Reactor Coolant Pump Rotor Seizure Event - RCS Flow Rates). The reactor is tripped on low-low flow in the affected loop. This trip does not occur soon enough to maintain the DNBR above the 95 percent probability/95 percent confidence DNBR limit. It is conservatively predicted that eight percent of the core undergoes DNB-induced cladding failure. Core cooling capability is maintained.

The response of additional system parameters is shown in Figures 15.3-10 —Reactor Coolant Pump Rotor Seizure Event - Reactor Power through 15.3-12—Reactor Coolant Pump Rotor Seizure Event - Core Average Heat Flux. Core inlet temperature is used in the determination of DNBR. A plot is not provided because it changes little during the period of minimum DNBR. A plot of PSRV flow rate is not provided because the PSRVs do not open during the period analyzed.

15.3.3.4 Radiological Consequences

Section 15.0.3.8 addresses the radiological impact associated with locked rotor event. A bounding value of 9.5 percent fuel failure is assumed in the radiological analysis (Section 15.0.3.8.2).

15.3.3.5 Conclusions

The potential fuel damage is sufficiently limited to enable the core to remain in place and intact with no loss of core cooling capability. The calculated doses at the site boundary meet the acceptance criteria of 10 CFR Part 100.

15.3.3.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.3.3 events included in NUREG-0800, Section 15.3.3–15.3.4, (Reference 2) 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: The RCS pressure increases following an RCP rotor seizure, but does not challenge the design limit.
2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit. 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 (refer to SRP Section 4.2, Reference 2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system

pressure, then fuel rods may balloon shortly after entering DNB. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. The 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: The potential for core damage is evaluated in Section 15.3.3.3.

3. The release of radioactive material must be such that the calculated doses at the site boundary are a small fraction of the 10 CFR Part 100 guidelines.

   – Response: The calculated doses at the site boundary meet the acceptance criteria of 10 CFR Part 100 (Section 15.3.3.4).

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: The integrity of the RCPs is maintained during the event. RCP seal integrity is described in Section 5.4.1.2.1.

5. The auxiliary feedwater system must be safety grade and, when required, automatically initiated.

   – Response: The auxiliary feedwater system (EFW) is safety grade and, when required, automatically initiates (Section 7.3.1.2.2).

6. A rotor seizure or shaft break in an RCP should not, by itself, generate a more serious condition or result in a loss of function of the RCS or containment barriers.

   – Response: This event is mitigated and the plant is maintained in a stable condition through the automated response of safety-related equipment, thus avoiding a more serious condition or a loss of function of the RCS or containment barriers.

7. Only safety-grade equipment should be used to mitigate the consequences of the event. Safety functions should be accomplished assuming the worst single failure of a safety system active component. For new applications, LOOP should not be considered a single failure. RCP rotor seizures and shaft breaks should be analyzed with a loss of off-site power (see Item 9 below) in combination with a single active failure.

   – Response: Only safety-grade equipment is used to mitigate the event, and a LOOP is assumed independently of the single failure evaluation.

8. The ability to achieve and maintain long-term core cooling should be verified.

   – Response: The ability to achieve and maintain long-term core cooling is verified (Section 15.0.4.1.2).

9. This event should be analyzed assuming TT and coincident LOOP and coastdown of undamaged pumps.

- Response: The analysis uses the models and methods described in Section 15.3.3.2. The parameter values in the analytical model are suitably conservative as described in Section 15.3.3.2.

According to the SRP (Reference 2) the applicant's analysis should be performed using an acceptable analytical model.

1. 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 to account for power measurement uncertainties. 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 (Section 15.3.3.2). The four loops are operating at the initiation of the event, as required by technical specifications.

2. The local flow conditions used in the core thermal-hydraulics model should be calculated based upon an inlet flow distribution corresponding to N-1 RCPs (initial minus faulted pump) and a conservative time-dependent flow coastdown. Note that the inlet flow distribution changes as more pumps begin to coast down following TT and coincident LOOP.

   - Response: Local flow conditions are considered in the core thermal-hydraulics model as described in Section 4.4.

3. Conservative scram characteristics are assumed, i.e., for a pressurized water reactor maximum time delay with the most reactive rod held out of the core and, for a boiling water reactor, a design conservatism factor of 0.8 times the calculated negative reactivity insertion rate.

   - Response: Conservative scram characteristics are assumed as described in Section 15.3.3.2.

4. 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: A conservative core burnup is selected as described in Section 15.3.3.2.

15.3.4 Reactor Coolant Pump Shaft Break

15.3.4.1 Identification of Causes and Event Description

The broken shaft event is a PA that is postulated to reduce the RCP inertia to that of the impeller. This results in a slower decrease in flow rate when compared to the rotor seizure event. An RT is initiated on a low-low loop flow rate signal. The shaft break separates the RCP anti-reverse rotation device from the rotor, thereby rendering it ineffective when flow in the affected loop reverses direction.

15.3.4.2 Method of Analysis and Assumptions

Although the shaft break event leads to a higher reverse flow rate in the affected loop compared to the rotor seizure event, maximum ΔDNBR occurs prior to significant reverse flow in either event. Because core flow decreases faster for the rotor seizure event, it bounds the shaft break event for ΔDNBR (see Section 15.3.3).

15.3.5 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 Transient Methodology Topical Report," AREVA NP Inc., December 2007.

Table 15.3-1—Decrease in Reactor Coolant System Flow Rate Events - Key Input Parameters

Parameter	Analysis Value
Initial reactor power	4612 MWt
Average RCS temperature	594°F
Initial PZR pressure	2250 psia
Initial RCS loop flow rate	119,692 gpm
Moderator temperature coefficient	0.0 pcm/°F
Doppler temperature coefficient	-1.17 pcm/°F
RT on low-low RCS flow rate	50% NF
RT on low RCP speed	92% NS

Table 15.3-2—Decrease in Reactor Coolant System Flow Rate Events - Key Equipment Status

Plant Equipment or System	Status
Turbine control valves position control mode	Automatic
PZR sprays	Available for DNB event
Rod cluster control assembly position control mode	Manual
Non-faulted RCPs	Tripped off on LOOP
MFW	Isolated on RT or LOOP

Table 15.3-3—Partial Loss of Forced Reactor Coolant Flow Event - Sequence of Events

Event	Time (s)
Transient initiated (one RCP tripped)	0
RT on low-low flow	8.4
Rods begin to drop	8.8
Maximum DNBR occurs	9.1
Turbine tripped	9.4
LOOP	9.4

Table 15.3-4—Complete Loss of Forced Reactor Coolant Flow Event - Sequence of Events

Event	Time (s)
Transient initiated (LOOP)	0
RT on low RCP speed	1.4
Rods begin to drop	1.8
Maximum DNBR Occurs	3.0
Turbine tripped	4.4

Table 15.3-5—Reactor Coolant Pump Rotor Seizure Event - Sequence of Events

Event	Time (s)
Transient initiated (rotor seizure)	0
RT on low-low flow	1.0
Rods begin to drop	1.4
Turbine tripped	2.0
LOOP	2.0
Maximum DNBR occurs	2.5

Figure 15.3-1—Partial Loss of Forced Reactor Coolant Flow Event - RCS Flow Rates

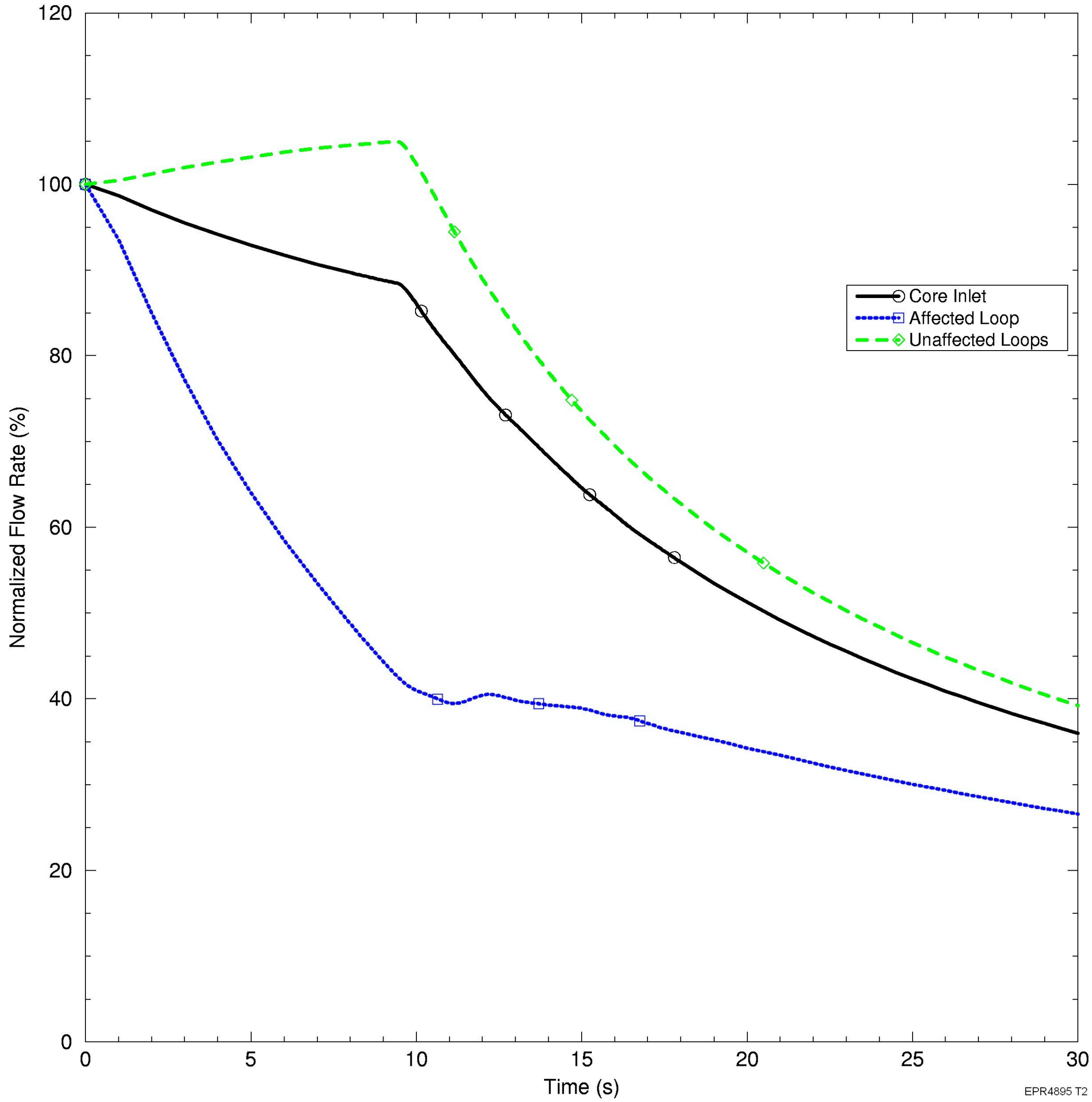

Figure 15.3-2—Partial Loss of Forced Reactor Coolant Flow Event - Reactor Power

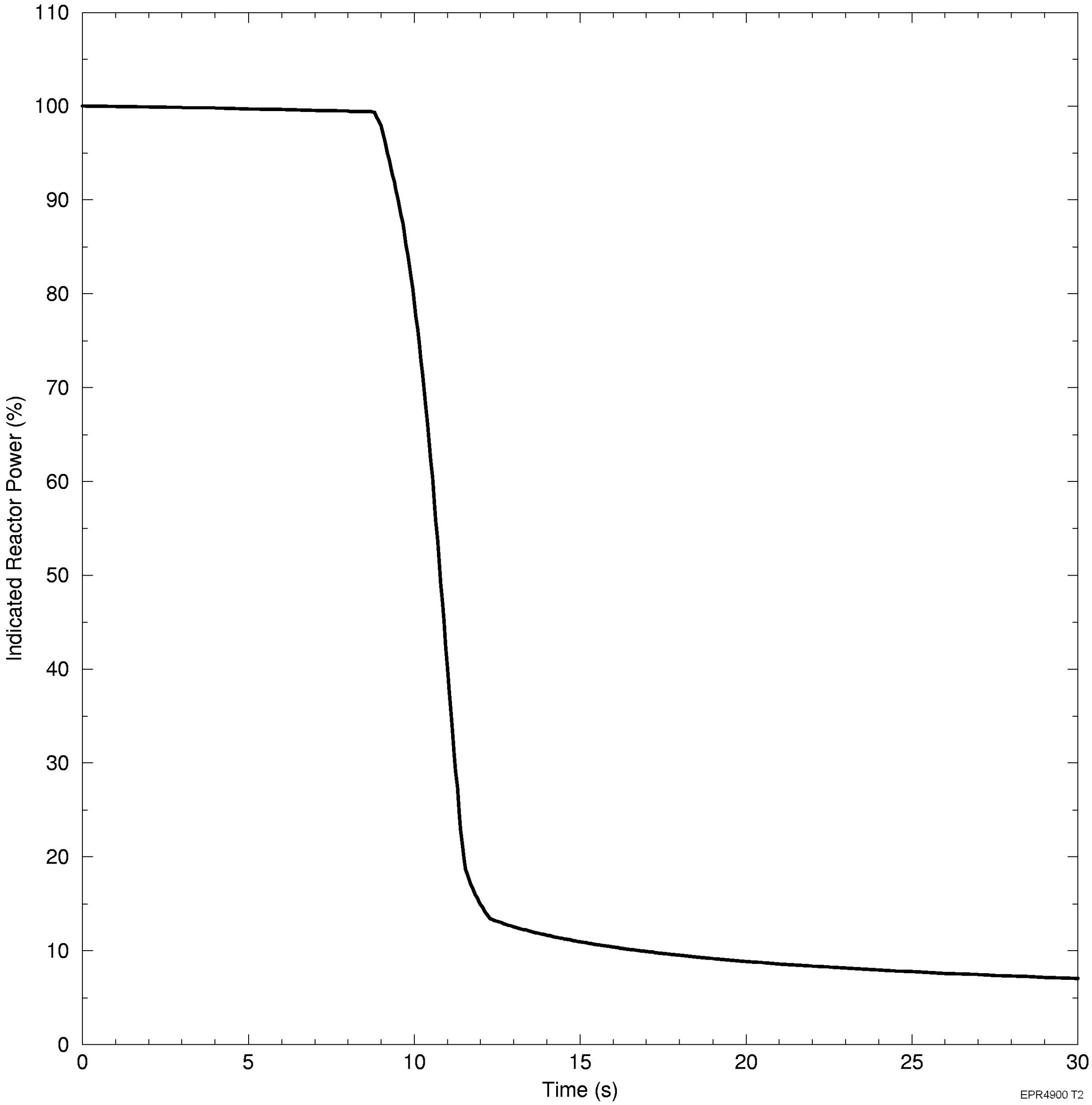

Figure 15.3-3—Partial Loss of Forced Reactor Coolant Flow Event - Core Average Heat Flux

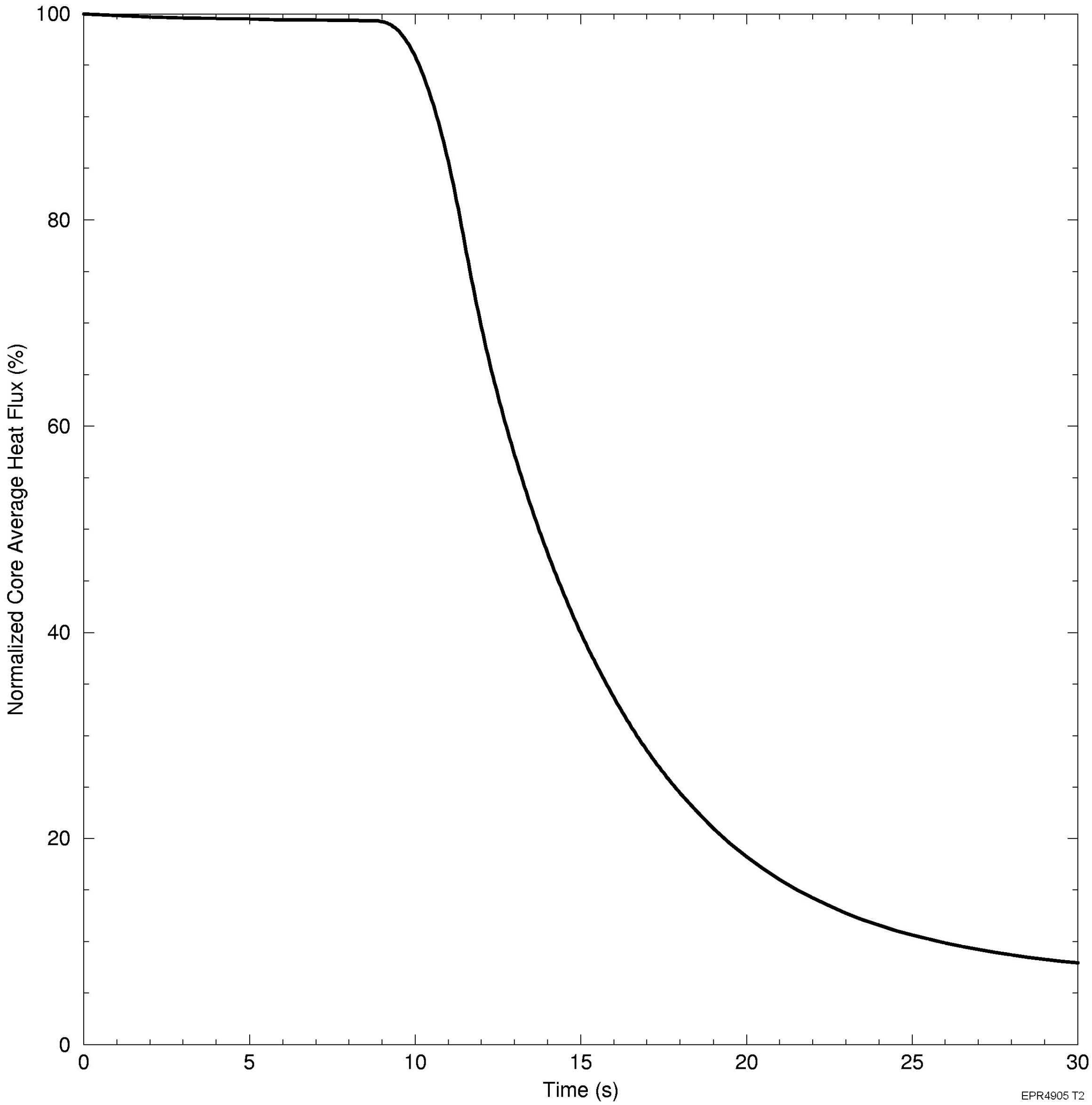

Figure 15.3-4—Partial Loss of Forced Reactor Coolant Flow Event - Pressurizer Pressure

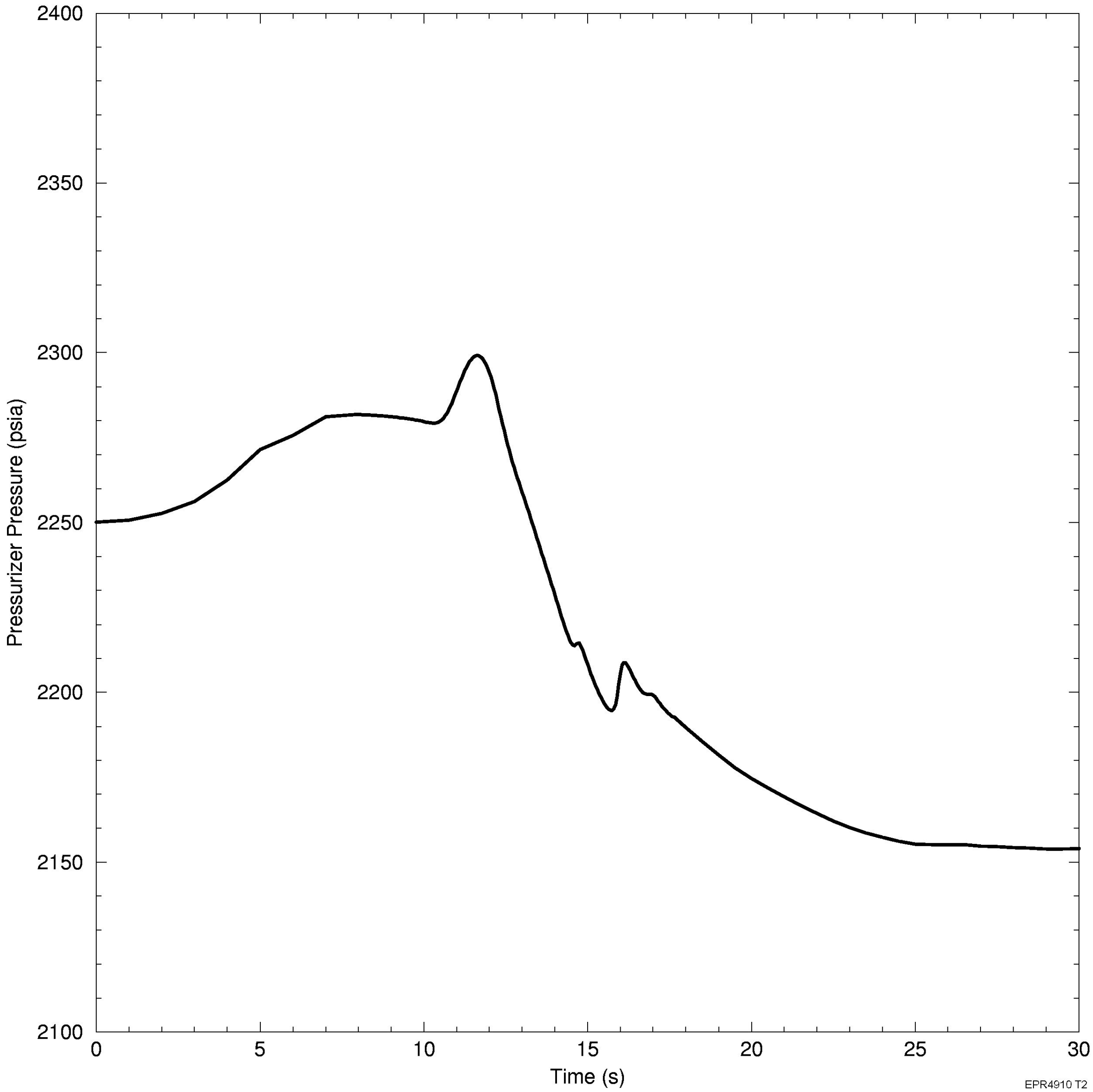

Figure 15.3-5—Complete Loss of Forced Reactor Coolant Flow Event - RCS Flow Rates

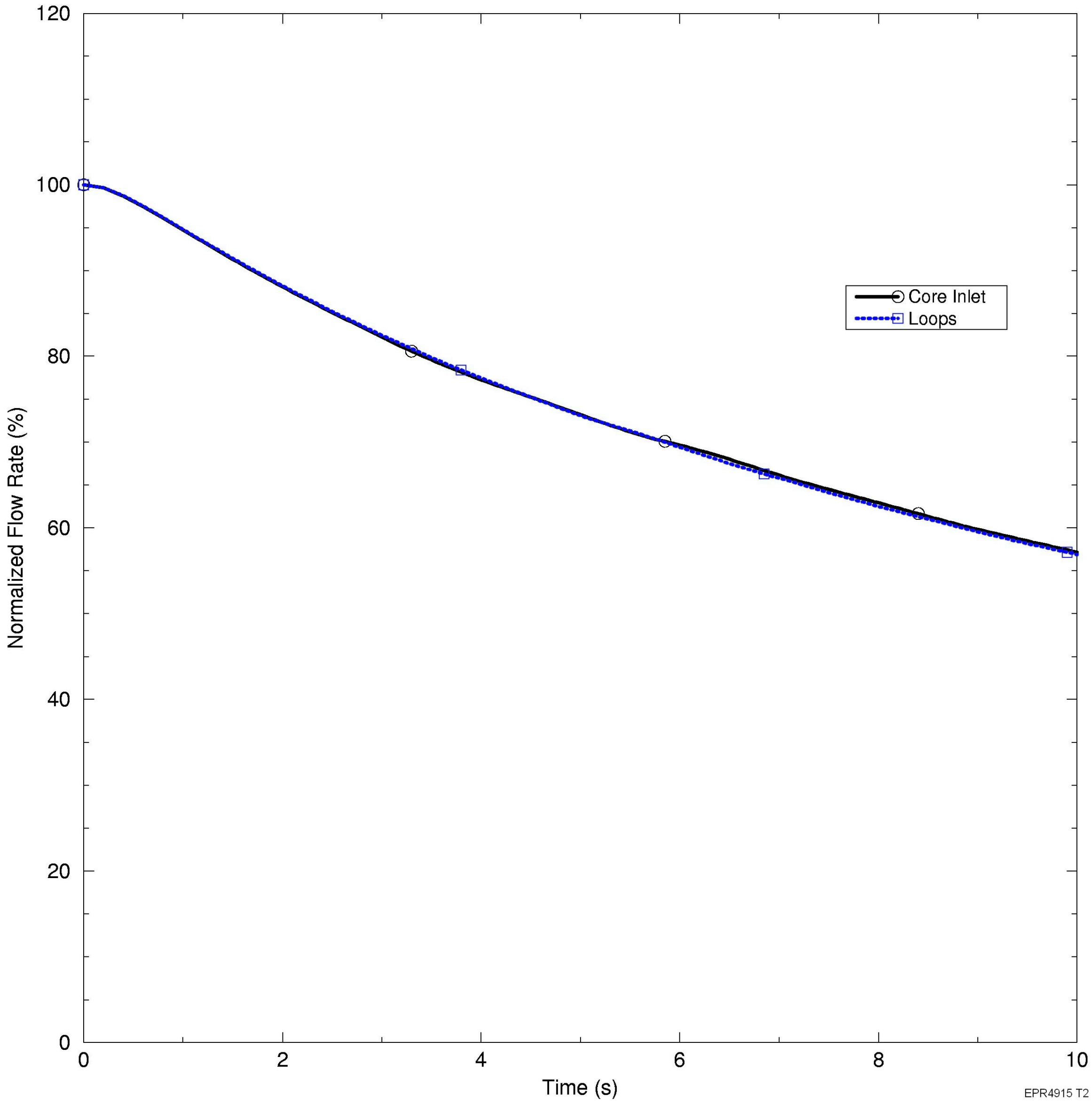

Figure 15.3-6—Complete Loss of Forced Reactor Coolant Flow Event - Reactor Power

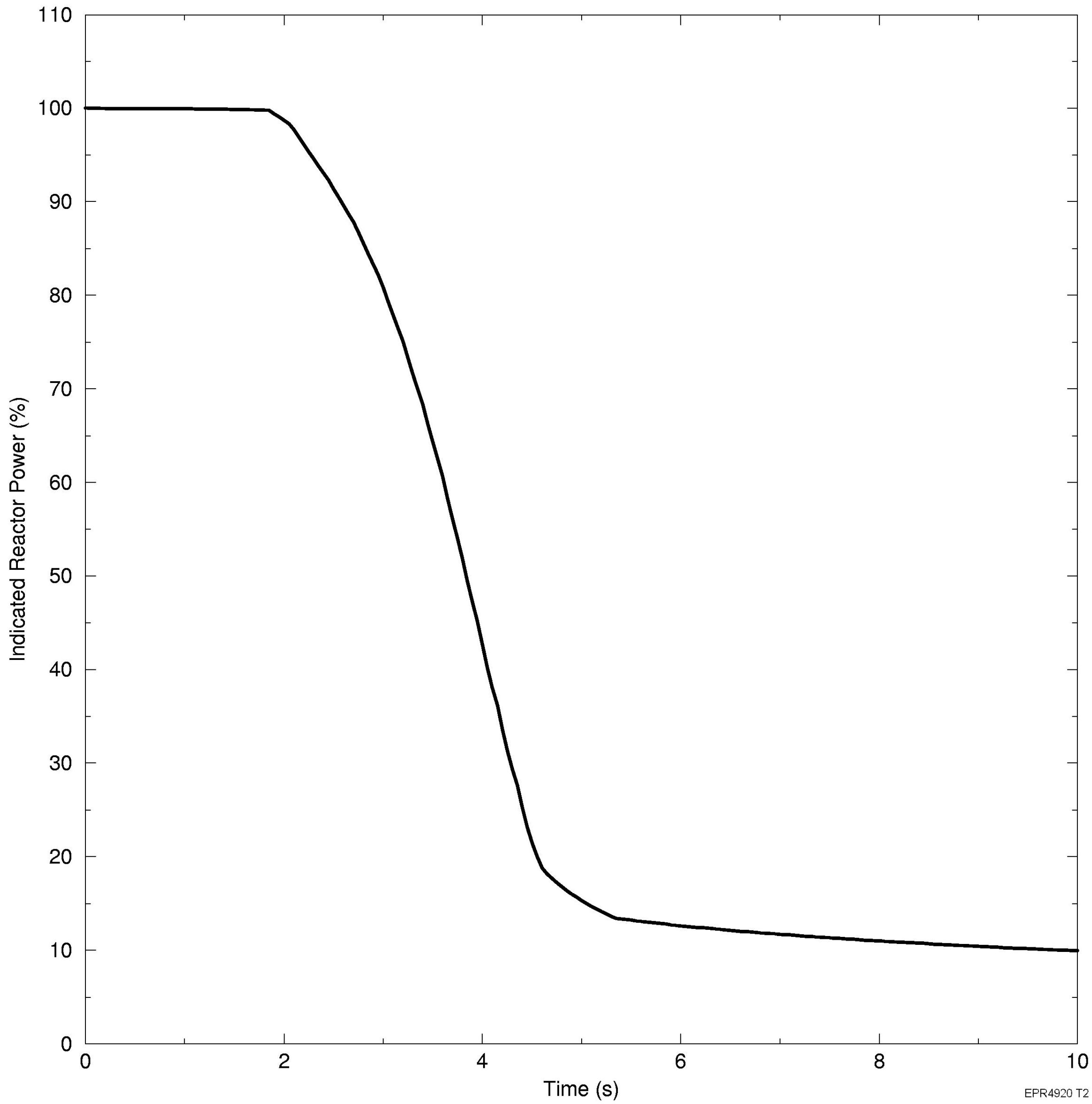

Figure 15.3-7—Complete Loss of Forced Reactor Coolant Flow Event - Core Average Heat Flux

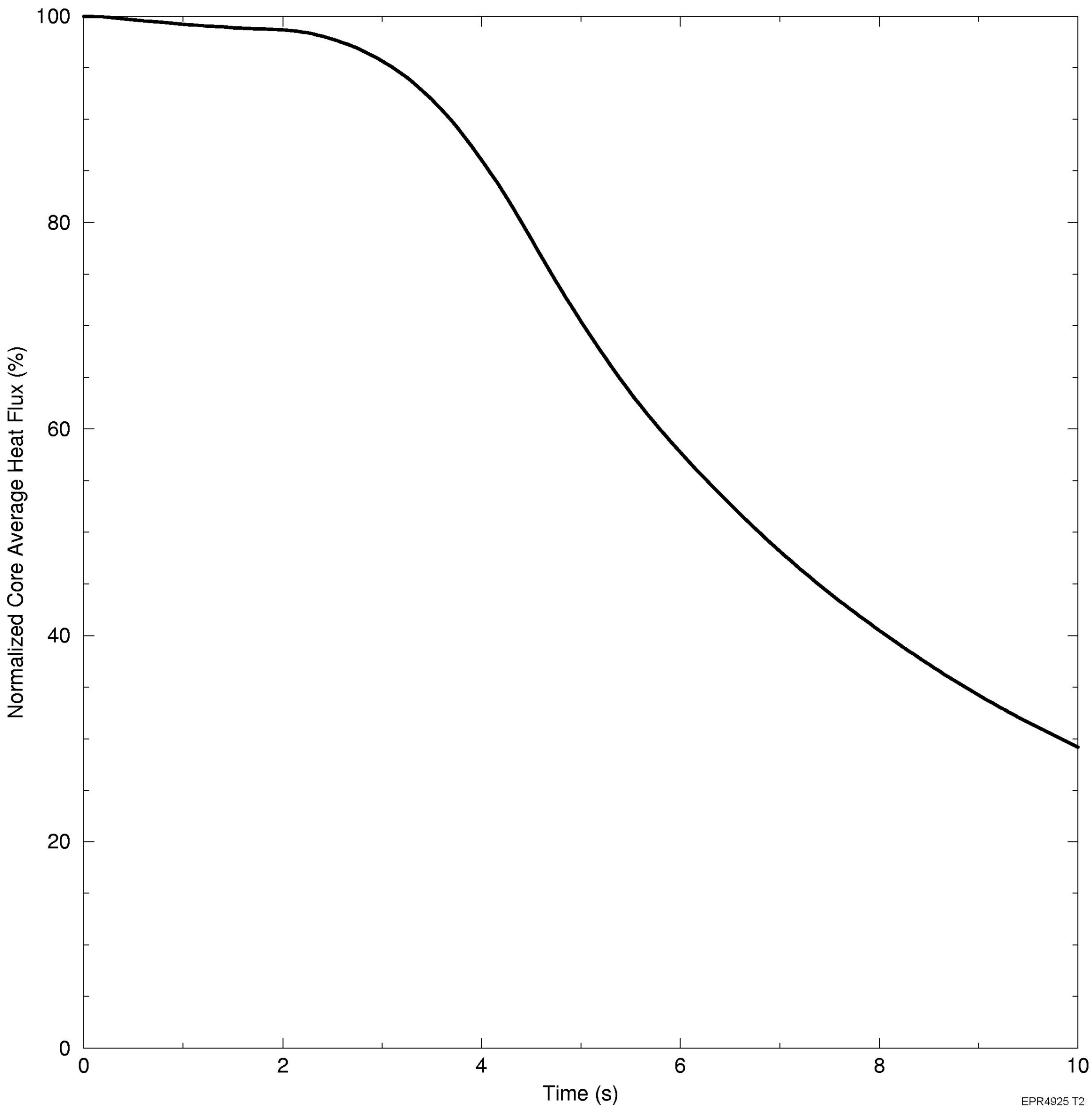

Figure 15.3-8—Complete Loss of Forced Reactor Coolant Flow Event - Pressurizer Pressure

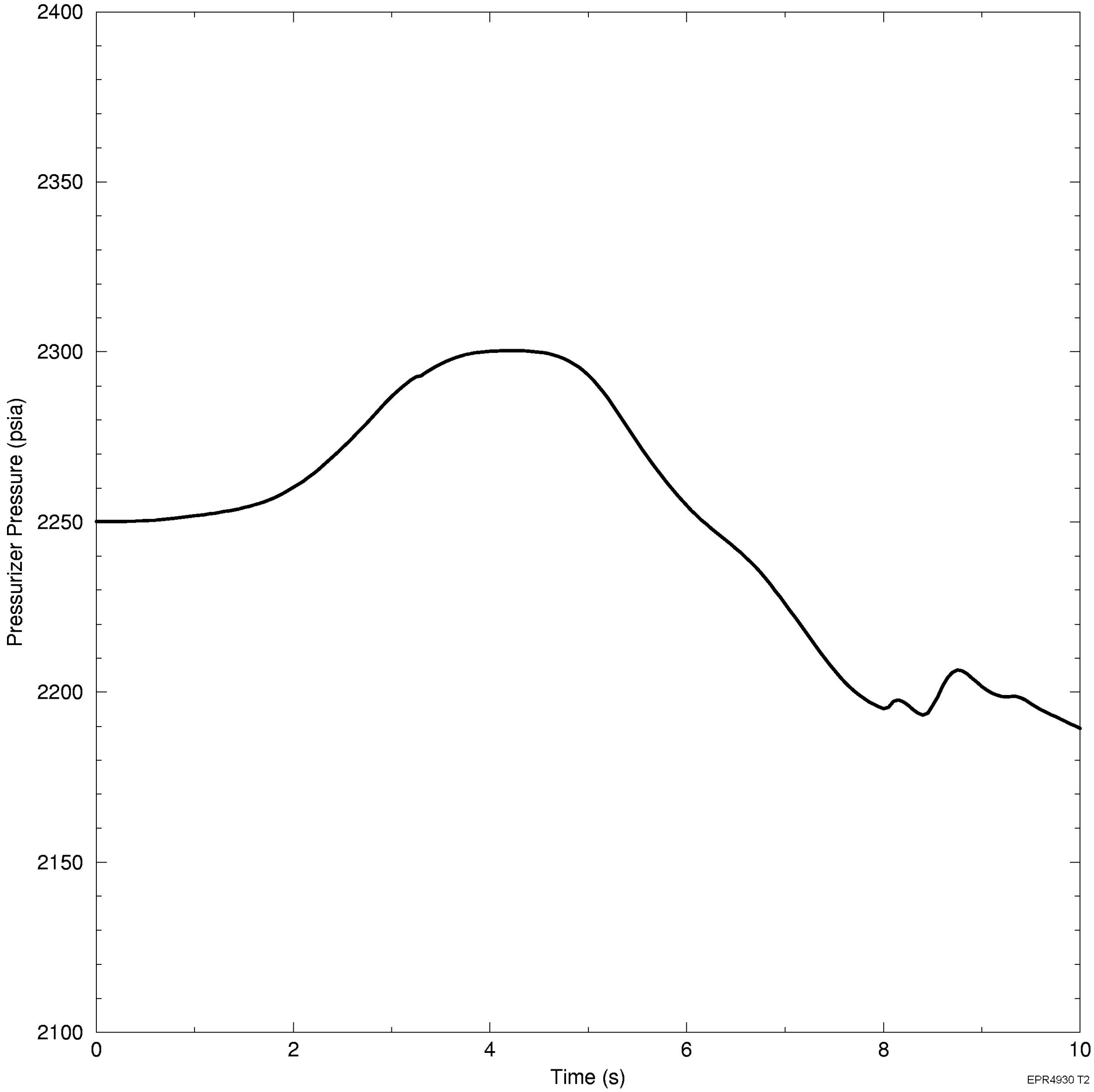

Figure 15.3-9—Reactor Coolant Pump Rotor Seizure Event - RCS Flow Rates

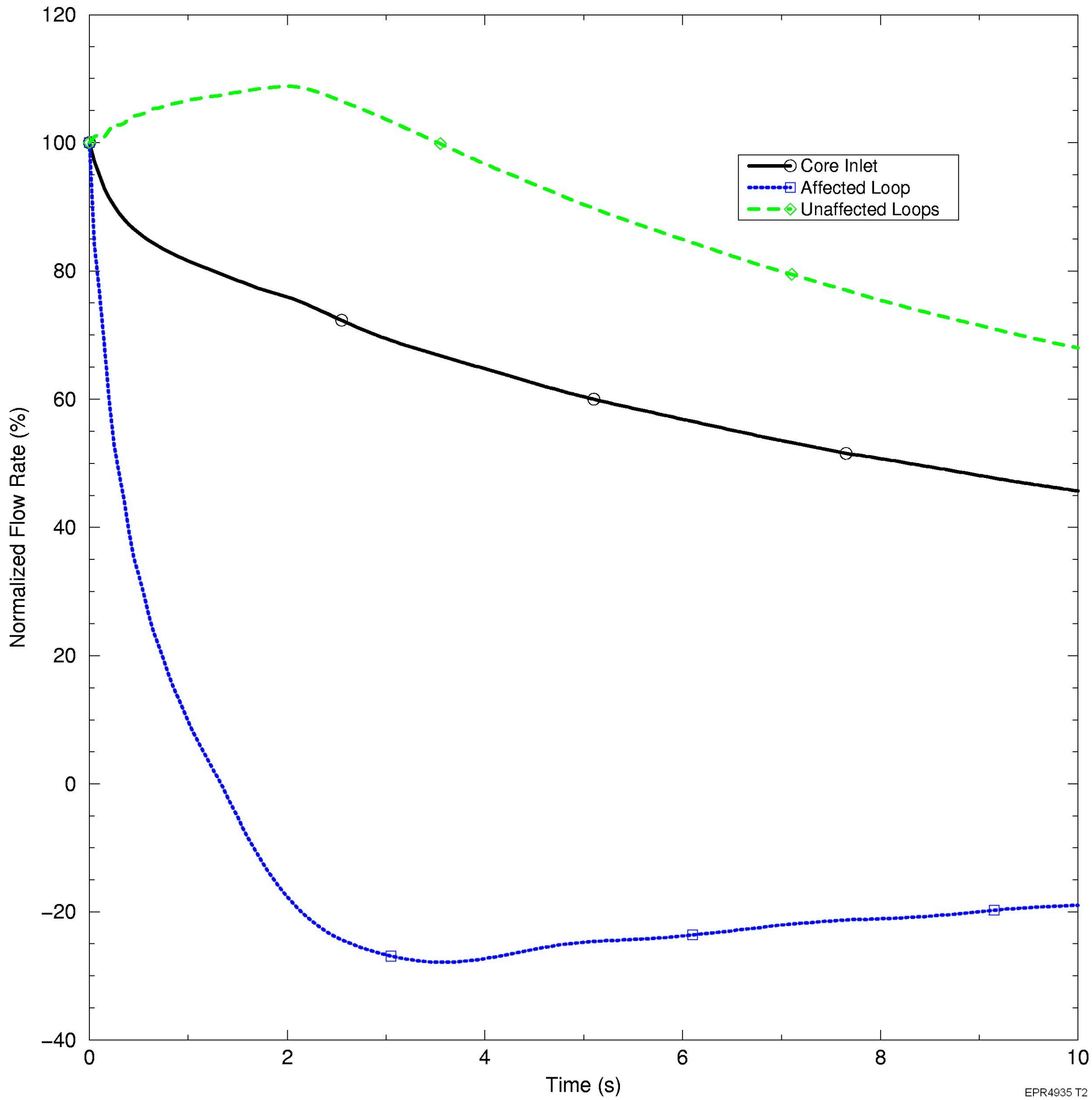

Figure 15.3-10—Reactor Coolant Pump Rotor Seizure Event - Reactor Power

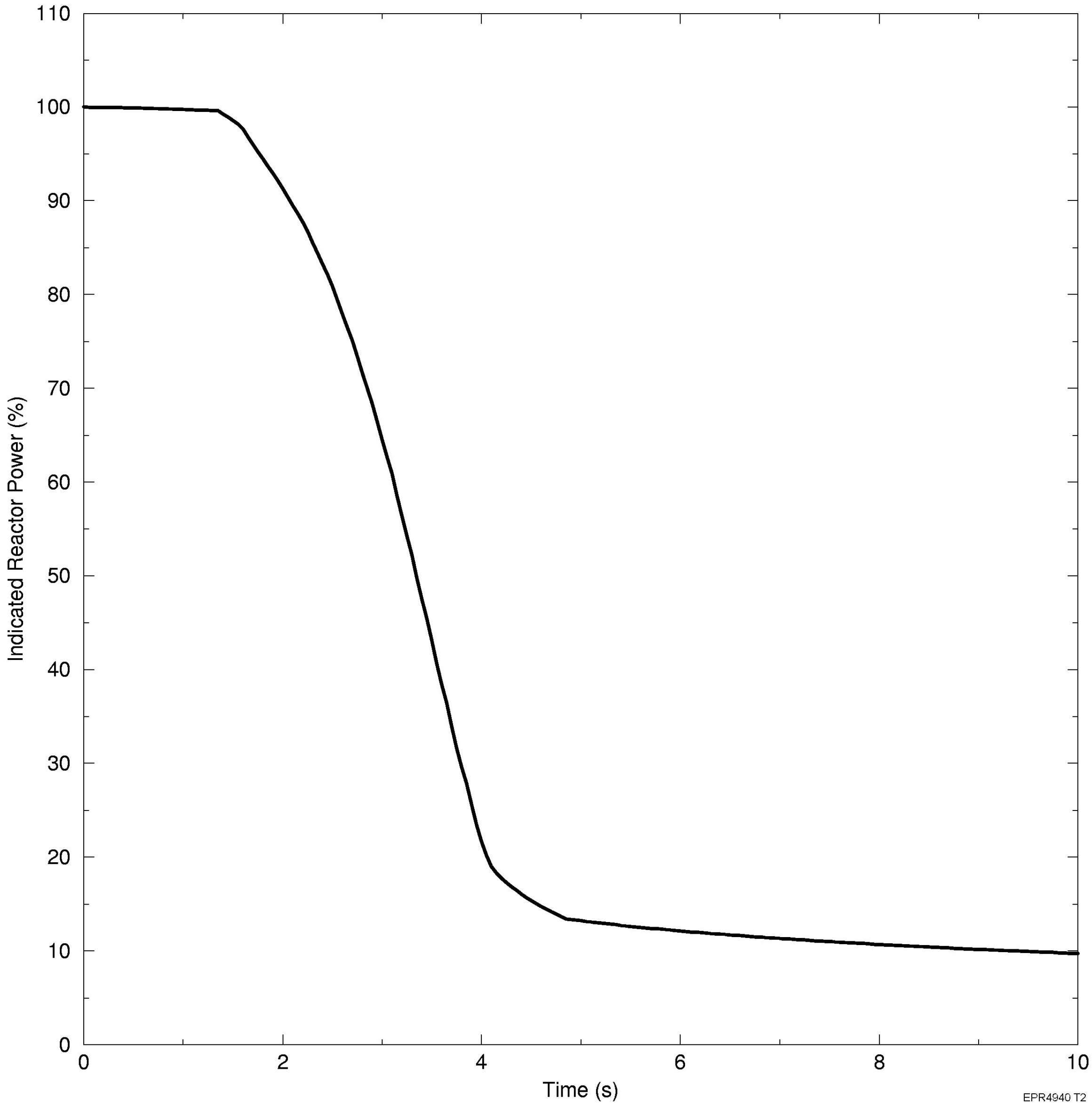

Figure 15.3-11—Reactor Coolant Pump Rotor Seizure Event - Pressurizer Pressure

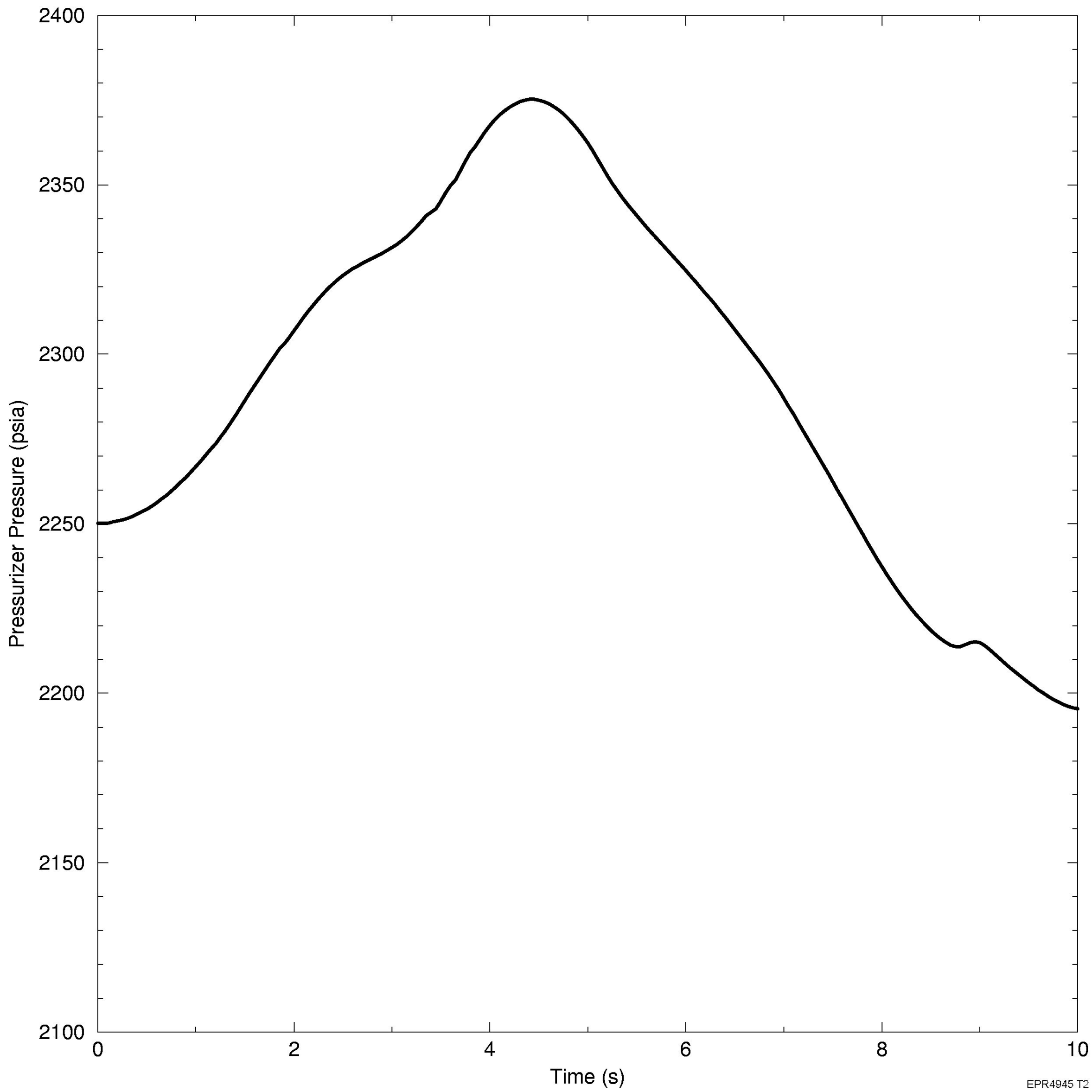

Figure 15.3-12—Reactor Coolant Pump Rotor Seizure Event - Core Average Heat Flux