15.4 Reactivity and Power Distribution Anomalies

Core reactivity and power distribution within the core are controlled for safe, efficient operation of the reactor. Several postulated events, however, can alter the core reactivity and power distribution and disrupt reactor operation. Core reactivity changes result from boron concentration changes, overcooling of the reactor coolant system (RCS) or addition of cold water to the RCS, spurious control rod motion, or total control rod ejection from the core region. Power distribution changes are caused by control rod movement, total control rod ejection, or mislocation of fuel assemblies. This section describes the following events and includes analyses that determine which of these events is most limiting:

- Section 15.4.1 - Uncontrolled control rod assembly withdrawal from a subcritical or low-power startup condition.
- Section 15.4.2 - Uncontrolled control rod assembly withdrawal at power.
- Section 15.4.3 - Control rod misoperation (system malfunction or operator error).
- Section 15.4.4 - Startup of an inactive loop at an incorrect temperature.
- Section 15.4.5 - Flow controller malfunction causing an increase in boiling water reactor (BWR) core flow rate (not applicable to the U.S. EPR).
- Section 15.4.6 - Inadvertent decrease in boron concentration in the RCS.
- Section 15.4.7 - Inadvertent loading and operation of a fuel assembly in an improper position.
- Section 15.4.8 - Spectrum of rod ejection accidents for a pressurized water reactor (PWR).
- Section 15.4.9 - Spectrum of rod drop accidents for a BWR (not applicable to the U.S. EPR).

Each of the above applicable events is analyzed for the U.S. EPR. The most severe radiological consequences occur for the rod ejection event (see Section 15.4.8) when a control rod drive mechanism fails. Radiological consequences for the bounding case are described in Section 15.0.3.9.

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition

15.4.1.1 Identification of Causes and Event Description

The uncontrolled control rod assembly withdrawal from a subcritical or low-power startup condition event is defined as an uncontrolled addition of reactivity due to the withdrawal of banks of rod cluster control assemblies (RCCAs) at hot shutdown or hot standby conditions. This event is an anticipated operational occurrence (AOO) of moderate frequency as described in Section 15.0.0.1.

For the U.S. EPR, the 89 RCCAs are divided into four banks spanning the reactor core. Banks 1, 2 and 3 contain 22 RCCAs each, while bank 4 contains 23 RCCAs and includes the center fuel assembly. Within each bank, the RCCAs are grouped further into a control bank and a shutdown bank. The four control banks are referred to as banks A, B, C, and D. Of the four banks, only banks C and D can trip the reactor to approximately 50 percent of rated thermal power (RTP) without a full insertion of the RCCAs in these banks. The rod position measurement instrumentation is divided into four redundant divisions. Sections 4.2 and 4.3 describe the RCCAs in detail.

RCCA bank withdrawal events are analyzed only for plant modes 1 (power operation) and 2 (startup). Events occurring in mode 1 are addressed in Section 15.4.2. Events occurring in mode 2 are initiated with the control rods at the hot zero power (HZP) rod sequence and overlap limits because plant technical specifications (TS) and operating procedures require that sufficient boron concentration is maintained to prevent a core criticality when the control rods are more deeply inserted. The rod sequence and overlap limits are TS that put restrictions on the allowed RCCS bank positions as a function of core power. In mode 3 (hot standby), the plant TS and operating procedures require that sufficient boron concentration is maintained to prevent the core from going critical, even if the control rods are fully withdrawn. Therefore, an RCCA bank withdrawal from mode 3 will not result in a core criticality. For an RCCA bank withdrawal transient to occur, the control rod breakers must be energized. In modes 4 (hot shutdown), 5 (cold shutdown), and 6 (refueling), the control rod breakers are not energized and an RCCA bank withdrawal event cannot occur.

The uncontrolled addition of reactivity to the reactor core by an uncontrolled RCCA bank withdrawal is postulated to result from a malfunction of the reactor control or RCCA control systems, which leads to a power excursion. The neutron flux response to the continuous reactivity insertion is characterized by a fast rise limited by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the power excursion is important because it limits the power during the delay for the safety system to respond. The neutron flux is measured during the transient. If the detected flux exceeds a threshold value, a reactor trip (RT) is initiated.

The transient is terminated by the following RT signals that are part of the PS:

- High positive neutron flux rate of change (intermediate and power range). This signal limits the consequences of an excessive reactivity increase from an intermediate power level as well as from hot full power (HFP) power. The signal is the nuclear flux derivative based on the excore instrumentation.

- Low doubling time of intermediate range neutron flux. This signal is the doubling time of the nuclear flux derived from the intermediate range detectors.
- Intermediate range high neutron flux. This signal is the nuclear flux derived from the intermediate range detectors.

These three trip setpoints are reached nearly simultaneously during uncontrolled RCCA bank withdrawals from HZP because of the fast power increase.

The applicable acceptance criteria for the uncontrolled RCCA bank withdrawal from a subcritical or low-power condition event are as follows:

- The DNBR thermal margin limit is met.
- Fuel centerline temperatures do not exceed the melting point.
- Uniform cladding strain does not exceed one percent.

This event primarily challenges the specified acceptable fuel design limits (SAFDL). The minimum calculated departure from nucleate boiling ratio (DNBR) is greater than the 95/95 safety limit of the applicable DNBR correlation, which demonstrates that a departure from nucleate boiling (DNB) is avoided. The peak central temperature of the fuel is maintained lower than the melting pointing during the event.

15.4.1.2 Method of Analysis and Assumptions

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

The core thermal-hydraulic computer code LYNXT is used to calculate the core flow, enthalpy distributions, DNBR, and peak fuel centerline temperatures using the RCS response from S-RELAP5 as a boundary condition as described in Incore Trip Setpoint and Transient Methodology for the U.S. EPR (Reference 2).

Table 15.4-1—Uncontrolled Control Bank Withdrawal from a Subcritical or Low-Power Startup Condition - Key Input Parameters lists the key input parameters for the limiting case. Table 15.4-2—Uncontrolled Control Bank Withdrawal from a Subcritical or Low-Power Startup Condition - Equipment Status lists the plant systems and equipment that are available to mitigate the effects of this event. The effect of non-safety-related equipment is considered in this analysis when it results in a more limiting transient. The analysis assumes that the most reactive RCCA is stuck in a fully withdrawn position. There is no single failure that can lead to more severe consequences. The uncontrolled RCCA bank withdrawals from HZP are simulated by withdrawing either the control banks or the shutdown banks with maximum RCCA velocity.

15.4.1.3 Results

Table 15.4-3—Uncontrolled Control Bank Withdrawal from a Subcritical or Low-Power Startup Condition – Sequence of Events presents the sequence of events for the limiting case. Figures 15.4.1— Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition – Reactor Power through 15.4-5—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition – Cold Leg Mass Flow present the response of the most important system parameters.

The minimum DNBR remains above the design limit value (refer to Section 4.4). The peak fuel centerline temperature remains below the fuel melting point. The fuel temperatures do not increase high enough to cause enough fuel expansion to exceed one percent uniform clad strain.

15.4.1.4 Radiological Conclusions

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs and no radioactive materials are released to the environment.

15.4.1.5 Conclusions

The analyses presented evaluate an uncontrolled control rod assembly withdrawal from a subcritical or low-power startup condition. During this event, the plant instrumentation, protection functions, and equipment provide an RT sufficiently early to preclude fuel or cladding damage. The core remains adequately cooled throughout this event.

15.4.1.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.1 events included in NUREG-0800, Section 15.4.1, (Reference 3) and descriptions of how these criteria are met are listed below:

1. The requirements of GDC 10, 20, and 25 concerning the SAFDL are assumed to be met for this event when:

   A. The thermal margin limits as specified in SRP Section 4.4 (Reference 3) are met.

      - Response: The results in Section 15.4.1.3 demonstrate that this requirement is met. The minimum DNBR remains above the design limit value

B. Fuel centerline temperatures for (PWRs) as specified in SRP Section 4.2 (Reference 3) do not exceed the melting point.

- Response: The results in Section 15.4.1.3 demonstrate that this requirement is met. The peak fuel centerline temperature remains below the fuel melting point.

C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 (Reference 3) does not exceed one percent.

- Response: This SRP requirement is for BWRs. The results in Section 15.4.1.3 demonstrate that the U.S. EPR PWR also meets this requirement.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.2.1 Identification of Causes and Event Description

The uncontrolled control rod assembly withdrawal at power event is defined as an uncontrolled addition of reactivity due to the withdrawal of RCCA banks (described in Sections 15.4.1.1, 4.2, and 4.3) during power operation. This event is postulated to result from either a failure in the RCCA position control system or an operator error that results in an uncontrolled withdrawal of a group of RCCA banks or sub-banks. This transient causes an increase in core power with a corresponding increase in heat flux. Due to the time lag in the response of the secondary system, the heat removal from the steam generators (SG) follows the heat increase in the primary system. Simultaneously, a net increase in the reactor coolant temperature and pressure occurs. Depending on the power level and point in the fuel cycle, the transient terminates from an RT signal due to either a high neutron flux rate of change, high pressurizer level, low DNBR, high core power level, or high SG pressure. This event is an AOO of moderate frequency as described in Section 15.0.0.1.

The uncontrolled withdrawal of an RCCA bank at power results in either a slow or a fast increase in reactivity. In a slow reactivity increase, the increase in coolant temperature follows the increase in reactor power. In a fast reactivity increase, the reactor power increases at a much faster rate than the coolant temperature. The slow reactivity transient is terminated by a high pressurizer level, low DNBR, high core power level, or high SG pressure trip. The fast reactivity transient is terminated by the high neutron flux rate of change RT.

In addition to the rate of reactivity increase, the limiting transients for the RCCA bank withdrawal at power depend on the starting reactor power level and the point in the fuel cycle burnup. The analyses considered power levels of 25 percent, 60 percent, and HFP to determine the limiting case. These power levels were selected because each of these power levels represents a breakpoint in the primary average temperature versus power curve for the U.S. EPR. The highest power level for each plateau of primary average temperature is analyzed. The limiting points in the fuel cycle burnup correspond to beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions. Therefore, the following matrix of cases is analyzed for this event:

- HFP at BOC and EOC conditions, with reactivity insertion rates spanning full range.
- 60 percent power at BOC and EOC conditions, with reactivity insertion rates spanning full range.
- 25 percent power at BOC and EOC conditions, with reactivity insertion rates spanning full range.

As determined from the transient events, the reactor system is protected during the RCCA bank withdrawal at power with the following RT setpoints:

- Low DNBR.
- Excore high neutron flux rate-of-change protection.
- High core power level protection.
- High pressurizer level protection.
- High SG pressure.

For fast reactivity transients, the excore neutron flux measurement provides a short response time to protect the core. The remaining trips provide core protection for all but the fastest reactivity transients. Only those trips that are credited in terminating the event are listed above.

The applicable acceptance criteria for the uncontrolled RCCA bank withdrawal at power event are as follows:

- The DNBR thermal margin limit is met.
- Fuel centerline temperatures do not exceed the melting point.
- Uniform cladding strain does not exceed one percent.

This event primarily challenges the SAFDL. The DNB SAFDL is satisfied by the combination of the low DNB limiting condition for operation (LCO) and RT setpoint, as described in Reference 2. The dynamic compensation of the low DNB channel algorithm is shown to be adequate to protect the SAFDL when the RT setpoint is reached. The high linear power density (HLPD) limits are not exceeded, which demonstrates that fuel centerline melt and one percent uniform clad strain is prevented.

15.4.2.2 Method of Analysis and Assumptions

This event uses the S-RELAP5 code and associated methodology described in Reference 1 to simulate the responses of the primary and secondary coolant systems, reactor, protective equipment and systems, and automatic controllers. The core thermal-hydraulic computer code LYNXT is used to calculate the core flow, enthalpy distributions, DNBR, and peak fuel centerline temperatures using the RCS response from S-RELAP5 as a boundary condition. The low DNB channel algorithm is simulated to predict RT and adequacy of the dynamic compensation of the algorithm in a manner consistent with Reference 2.

The low DNB channel algorithm uses the following measurements:

- The reactor power distribution is derived from the self-powered neutron detectors that are part of the nuclear incore instrumentation.
- The primary system pressure is derived from the primary pressure sensors.
- The core flow is derived from the reactor coolant pump (RCP) speed sensors.
- The reactor inlet temperature is derived from the cold-leg temperature sensors.

Cold-leg temperature measurements are treated in the S-RELAP5 model with a filtering module and a lead-lag module. The cold-leg temperature measurement is used for the DNBR calculations, while the core inlet temperature is the relevant parameter for the physical DNBR. A time delay occurs when the reactor coolant travels from the cold-leg measurement location to the core inlet. The lead-lag module compensates for this time delay.

Neither a single failure nor equipment taken out of service for maintenance makes this event more severe. The protection system (PS) is the only system or equipment that mitigates this event and its redundant design makes it single-failure proof. Following RT, it assumed that the highest worth rod is stuck above the core. It is also assumed that a loss of offsite power (LOOP) occurs coincident with turbine trip (TT). The opening and closing setpoints of the pressurizer safety relief valves (PSRVs) are biased conservatively low.

Table 15.4-4—Uncontrolled Control Bank Withdrawal at Power – Key Input Parameters presents the key input parameters used in the analyses. The control rod having the greatest worth is assumed to be stuck above the core. In addition, the most conservative negative reactivity insertion curve as a function of time is used in the transient analyses.

The power range considered for this event ranges from 25 percent power up to HFP. The uncontrolled withdrawal of a control bank below 10 percent power is described in Section 15.4.1. For each reactor power and burnup condition considered, the RCCA bank withdrawal rates that are analyzed range from a single RCCA to the maximum bank value for the particular burnup condition. The BOC condition is a minimum reactivity feedback case, while the EOC condition is a maximum reactivity feedback case. For the minimum reactivity feedback case, the moderator temperature coefficient is zero or larger depending on reactor power. For conservatism, the corresponding Doppler coefficient is minimized in absolute value. For the maximum reactivity feedback case, the most negative moderator temperature and Doppler coefficients are assumed.

The analyses minimize DNB margin by conservatively assuming that the pressure control system is operational, pressurizer spray flow is at its maximum value, and the PSRV opening and closing setpoints are biased low. Non-safety-related equipment is considered when it causes a more limiting transient. Table 15.4-5—Uncontrolled Control Bank Withdrawal at Power – Equipment Status lists the plant systems and equipment that are assumed available to mitigate this event.

15.4.2.3 Results

Table 15.4-6—Uncontrolled Control Bank Withdrawal at Power – Sequence of Events presents the sequence of events for a representative case. The analysis of this event considered a spectrum of reactivity insertion rates at power levels up to HFP. Table 15.4-6 provides the sequence of events for the maximum RCCA worth withdrawal at power and BOC conditions. Figures 15.4-6—Uncontrolled Control Bank Withdrawal at Power – Reactor Power through 15.4-14—Uncontrolled Control Bank Withdrawal at Power – Pressurizer Spray show the response of the most important system parameters.

The DNB RT setpoints as well as the dynamic compensation built into the low DNB channel algorithm are adequate to protect the DNB SAFDL for conditions that cause the low DNB channel to issue an RT. For conditions in which the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL. The peak fuel centerline temperature remains below the fuel melting point and uniform clad strain remains below one percent.

15.4.2.4 Radiological Consequences

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs and no release of radioactive materials to the environment occurs.

15.4.2.5 Conclusions

The analyses show that the plant instrumentation, protection functions, and equipment are sufficient to preclude fuel or cladding damage for the uncontrolled control bank withdrawal at power event. The core remains adequately cooled throughout this event.

15.4.2.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.2 events included in NUREG-0800, Section 15.4.2, (Reference 3) and descriptions of how these criteria are met are listed below:

1. The requirements of GDC 10, 17, 20, and 25 concerning the SAFDL are assumed to be met for this event when:

   A. The thermal margin limits as specified in SRP Section 4.4 (Reference 3) are met.

      - Response: The results in Section 15.4.2.3 demonstrate that this requirement is met. The DNB RT setpoints as well as the dynamic compensation built into the low DNB channel algorithm are adequate to protect the DNB SAFDL for conditions that cause the low DNB channel to issue an RT. For conditions in which the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL.

   B. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2 (Reference 3) do not exceed the melting point.

      - Response: The results in Section 15.4.2.3 demonstrate that this requirement is met. The peak fuel centerline temperature remains below the fuel melting point.

   C. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2 (Reference 3) does not exceed one percent.

      - Response: This SRP requirement is for BWRs. The results in Section 15.4.2.3 demonstrate that this requirement is additionally met for the U.S. EPR PWR by not exceeding the HLPD limits.

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

This section presents the analysis of three types of postulated control rod misoperation events:

- Dropped RCCA or RCCA sub-bank, described in Section 15.4.3.1.
- Statically misaligned RCCA, described in Section 15.4.3.2.
- Single RCCA withdrawal, described in Section 15.4.3.3.

These events are classified as AOOs, as described in Section 15.0.0.1. Results are presented for the most limiting cases. Sections 4.2, 4.3, and 15.4.1.1 describe the RCCAs and their control and shutdown banks in detail.

15.4.3.1 Dropped RCCA or RCCA Sub-Bank

15.4.3.1.1 Identification of Causes and Event Description

The dropped RCCA or RCCA sub-bank event is initiated by de-energizing an RCCA drive mechanism or by a malfunction associated with a control bank or sub-bank during power operation. The result is that a single RCCA, sub-bank or complete control bank falls into the core. Full insertion of the affected RCCAs decreases the reactor power and increases the radial power peaking commensurate with the worths and locations of the affected RCCAs. RCS temperature initially decreases in response to the initial decrease in reactor power. After the initial decrease in reactor power, power increases due to reactivity feedback and automatic withdrawal of control banks. The magnitude of this increase depends on the operating mode of the plant and the PS response.

The plant response to a dropped RCCA (single RCCA, sub-bank, or full bank) depends on the mode of operation of the RCCA system and the turbine control valve. The RCCAs are controlled using either average coolant temperature (ACT) control function or manual rod control (MRC). There are two possible response scenarios for the turbine control valve, depending on the turbine controls that are in effect:

1. If the turbine control valve is in automatic mode, the valve initially opens in an attempt to maintain load. Controllers then adjust the turbine steam flow to match reactor power and secondary system power.
2. If the turbine control valve is under operator control, the valve is not adjusted automatically to correct the load mismatch between the primary and secondary systems. The result is that reactor power decreases to a fraction of the pre-event power.

When the ACT control function is active, the reduction in reactor power due to a dropped RCCA is detected by the plant control system as a mismatch in primary to secondary system power. The ACT control function initiates an RCCA bank withdrawal to compensate for the drop in power. Power overshoot might occur due to the RCCA bank withdrawal, after which the ACT control function reinserts the control bank and returns the plant to the nominal operating power. The magnitude of the power overshoot is a function of the core reactivity coefficients, dropped rod worths, differential bank worths, and excore detector shadowing responses.

The plant response to a dropped RCCA event during ACT control function mode also depends on the PS processing of the excore and self-powered neutron detector (SPND) signals. Dropping a single RCCA near the detector may affect the ACT control function gain more than is proportionate to the change in core power corresponding to the dropped rod worth. This phenomenon is termed “shadowing” because the detector generates a false low-power signal. When a single RCCA is dropped near the control detector, the detector is shadowed and the reduction in the detected flux is greater than the corresponding core-average value.

To a lesser extent, during the RCS cooldown transient, the excore detector under-detects the reactor power because the density of the reactor vessel downcomer fluid is increased relative to the conditions at which the detector is calibrated. The detector therefore indicates a lower-than-actual power level and sends an output signal to the core control system. This condition causes overcompensation. While the excore detector closest to the dropped rod is shadowed, the excore detector on the opposite side of the reactor reads higher than the core average. In the U.S. EPR, the reactor control surveillance limitation (RCSL) design for handling excore shadowing auctioneers the second highest excore signal for use in the control system.

Since the PS system function that mitigates this event is redundant, no single failure makes the consequences of this event more severe.

The acceptance criteria for this event are as follows:

- The DNBR thermal margin limit is met.
- Fuel centerline temperatures do not exceed the melting point.
- Uniform cladding strain does not exceed one percent.

This event primarily challenges the SAFDL. The DNB SAFDL is satisfied by the combination of the low DNB LCO and RT setpoint described in Reference 2. The dynamic compensation of the low DNB channel algorithm is shown to be adequate to protect the SAFDL when the RT setpoint is reached. The HLPD limits are not exceeded, which demonstrates that fuel centerline melt and one percent uniform clad strain is prevented.

15.4.3.1.2 Method of Analysis and Assumptions

The S-RELAP5 computer code calculates the thermal hydraulic response of the primary and secondary systems using the methodology of Reference 1. Section 15.0.2 provides a description of the S-RELAP5 analysis methodology.

The low DNB channel algorithm is simulated to predict RT and adequacy of the dynamic compensation of the algorithm as described in Reference 2 (see Section 15.4.2.2 for more detail). Cases are analyzed for a spectrum of dropped rod reactivity worths, BOC and EOC neutronics feedbacks, and two core power levels. These analyses are performed at the maximum average RCS temperature and are initiated at power levels of 90 percent and HFP. The 90 percent power level was selected for analysis to confirm that HFP cases are limiting. The analyses assume the design maximum number of plugged SG tubes of five percent to minimize heat removal by the secondary system. A total of four categories of events are analyzed to assess the combinations of ACT control function operation (active or inactive) and steam turbine control (automatic or manual). Table 15.4-7—Dropped RCCA – Key Input Parameters presents key input parameters for these analyses and Table 15.4.-8—Dropped RCCA – Equipment Status presents the status of key equipment.

The insertion of an RCCA bank is detected by the PSRCCA position measurement function, which is a safety-related system that detects the position of the RCCAs within the reactor vessel. The drop of a single RCCA might not be detected if reactor control cluster acquisition unit is unavailable. Therefore, the RCCA position measurement system is credited for determining the low DNB and HLPD trip setpoints for RCCA bank drops, but not for the drop of a single RCCA.

The response of the non-safety-related rod drop limitation function to the drop in reactor power is to reduce the turbine and generator load setpoint to match the decreased reactor power level. In addition, the rod drop limitation function blocks automatic withdrawal of control RCCAs used to maintain the initial reactor power level and average coolant temperature. These protective actions are not credited in the analyses because the rod drop limitation function is not safety related.

The core protection functions that are available depend on the RCCA insertion failure and plant operating mode prior to the event. The low-DNBR and HLPD protection functions protect the core for events with a small increase in reactivity. If the reactivity increase or the rate of increase is large, the high neutron flux rate of change (power range), high neutron flux (intermediate range) or high neutron flux (source range) protection functions protect the core (Section 7.2.1.2.3).

It is assumed conservatively that offsite power is lost with RT. This condition causes the coastdown of the RCPs, which briefly reduces the margin to DNB. The limiting single active failure is the least-shadowed, highest-reading detector that provides input to the ACT control function. The S-RELAP5 model applies a conservative RCCA shadowing factor to the second least shadowed detector as a function of rod worth and rod location.

15.4.3.1.3 Results

The analyses demonstrate that the most limiting RCCA drop event is the one in which the ACT control function is active with automatic turbine valve control. Table 15.4-9—Dropped RCCA – Sequence of Events presents the sequence of events for a representative, 423 pcm, dropped RCCA bank case. Figures 15.4-15—Dropped RCCA – Reactor Power through 15.4-19—Dropped RCCA – Primary System Temperature present the plant response for this event.

The DNB RT setpoints as well as the dynamic compensation built into the low DNB channel algorithm are adequate to protect the DNB SAFDL for conditions that cause the low DNB channel to issue an RT. For conditions where the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL.

The peak fuel centerline temperature remains below the fuel melting point and uniform clad strain remains below one percent.

15.4.3.1.4 Radiological Consequences

No radiological consequences are calculated for this event because no fuel or cladding damage is predicted and no radiological releases to the environment occur.

15.4.3.1.5 Conclusions

The analyses show that the plant instrumentation, protection functions, and equipment are sufficient to preclude fuel or cladding damage for the dropped RCCA event. The core remains adequately cooled throughout this event.

15.4.3.2 Statically Misaligned RCCA

The effects of a statically misaligned RCCA are described in the setpoint methodology of Reference 2.

15.4.3.3 Single RCCA Withdrawal

15.4.3.3.1 Identification of Causes and Event Description

The accidental withdrawal of a single RCCA at power can occur in the following scenarios:

- When the reactor is operating in manual mode, the operator withdraws a single RCCA deliberately because of misinformation that the clusters are misaligned or that one cluster has dropped.
- When the reactor is operating in manual mode, several simultaneous electrical or mechanical failures cause withdrawal of a single RCCA.

The withdrawal of a single RCCA causes an insertion of reactivity and therefore an increase in average core power and temperature. Also, an increase in the local power peak occurs in the zone where the RCCA is located. DNBR limits may be violated during these conditions if the core is not adequately protected.

15.4.3.3.2 Method of Analysis and Assumptions

The RCS response to the withdrawal of a single RCCA at power is bounded by the RCS response of the withdrawal of an RCCA bank at power from Section 15.4.2. Therefore, no specific RCS response is required for the withdrawal of single RCCA at power.

The peaking augmentation due to the withdrawal of a single RCCA at power is covered in the setpoint methodology in Reference 2. Therefore, no specific transient in-core analysis is required for the withdrawal of single RCCA at power.

15.4.3.3.3 Results

The DNB RT setpoints, as well as the dynamic compensation built into the low DNB channel algorithm, are adequate to protect the DNB SAFDL for conditions that cause the low DNB channel to issue an RT. For conditions where the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL. The peak fuel centerline temperature remains below the fuel melting point and uniform clad strain remains below one percent.

15.4.3.3.4 Radiological Consequences

No radiological consequences are calculated for this event because no fuel or cladding damage is predicted and there are no radiological releases to the environment.

15.4.3.3.5 Conclusions

For a single RCCA withdrawal event the plant instrumentation, protection functions, and equipment provide an RT sufficiently early to preclude fuel or cladding damage. The core remains cooled throughout this event.

15.4.3.4 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.3 events included in NUREG-0800, Section 15.4.3, (Reference 3) and descriptions of how these criteria are met are listed below:

1. The thermal margin limits as specified in SRP Section 4.4, subsection II.1 (Reference 3), are met.

   – Response: The results in Sections 15.4.3.1.3 and 15.4.3.3.3 demonstrate that this requirement is met. The DNB RT setpoints as well as the dynamic compensation built into the low DNB channel algorithm are adequate to protect the DNB SAFDL for conditions that cause the low DNB channel to issue an RT. For conditions where the DNB degradation does not cause an RT, the DNB LCO is adequate to protect the DNB SAFDL.

2. Fuel centerline temperatures (for PWRs) as specified in SRP Section 4.2, subsection II.A.2(a) and (b) (Reference 3),do not exceed the melting point.

   – Response: The results in Sections 15.4.3.1.3 and 15.4.3.3.3 demonstrate that this requirement is met. The peak fuel centerline temperature remains below the fuel melting point.

3. Uniform cladding strain (for BWRs) as specified in SRP Section 4.2, subsection II.A.2(b) (Reference 3), does not exceed one percent.

   – Response: This SRP requirement is for BWRs. The results in Sections 15.4.3.1.3 and 15.4.3.3.3 demonstrate that this requirement is additionally met for the U.S. EPR PWR by not exceeding the HLPD limits.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

15.4.4.1 Identification of Causes and Event Description

This event is an AOO initiated from a condition in which the plant has undergone a partial scram due to the coastdown of one RCP. This condition reduces reactor power to approximately 50 percent. The time the reactor can operate at reduced power with only three RCPs running is limited by TS. The idle RCP must be restarted within an acceptable time limit, or the reactor must be shut down.

The idle loop initially has reverse flow and is entirely at a temperature near that of the cold legs of the active loops. This temperature is the same as the core inlet temperature. The challenge in restarting the fourth RCP while the reactor is at part power is the rapid increase in core flow, rather than a change in core inlet temperature. The increase in flow rate decreases the average temperature of water in the core, thereby increasing reactivity and core power.

This event returns the plant to full flow operation, after which the operator can increase core power level to full power. The PS does not intervene unless core protection limits are challenged. The PS is designed to terminate this transient before the DNB limits are reached. The principal protective trips for this event are the low DNBR trip and the high flux rate trip.

The acceptance criteria for this event are:

- Minimum DNBR remains above the design limit.
- Peak RCS pressure does not exceed 110 percent of design pressure.

Because this event causes only minor pressurization of the RCS, the analysis is biased to evaluate the minimum DNBR acceptance criterion.

15.4.4.2 Method of Analysis and Assumptions

The S-RELAP5 computer code calculates the thermal hydraulic response of the primary and secondary systems using the methodology of Reference 1. Section 15.0.2 provides a description of the S-RELAP5 analysis methodology.

The core thermal-hydraulic computer code LYNXT is used to calculate the core flow, enthalpy distributions, DNBR and peak fuel centerline temperatures. It uses the RCS response from S-RELAPS as a boundary condition as described in Reference 2. Administrative controls limit the power level at which the fourth RCP can be started. The event is analyzed from a 60 percent power EOC initial condition. The event is initiated by the startup of the idle fourth RCP. If the moderator temperature coefficient of reactivity is negative, an insertion of positive reactivity and an increase in reactor power occurs. It is assumed conservatively that rod control is in manual mode. The analysis of this event uses a conservatively large negative moderator temperature coefficient associated with the EOC.

Table 15.4-10—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Key Input Parameters presents key input parameters and Table 15.4.-11—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Equipment Status presents the status of key equipment available to mitigate this event.

There is no single failure that makes this event more severe. The only plant system that affects the response in this event is the non-safety-related pressurizer sprays, which are assumed active. This condition reduces the increase in RCS pressure and, therefore, reduces DNBR margin.

15.4.4.3 Results

Table 15.4-12—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Sequence of Events presents the sequence of events for this scenario. Figures 15.4-20—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Total RCS Loop Flow through 15.4-27—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – RCS Bottom Pressure present the plant response to this event.

RCS flow increases rapidly to full flow following the start of the fourth RCP (see Figure 15.4-20—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Total RCS Loop Flow). This condition causes a decrease in the core temperatures (see Figure 15.4.-23—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Core Temperatures). Reactor power increases to a maximum value of 75.7 percent during this time period due to the positive reactivity insertion associated with the decrease in the core average temperature (see Figure 15.4-24—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Indicated Reactor Power). No RT setpoints are reached. The combination of Doppler feedback (see Figure 15.4-25—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Reactivity) and increasing RCS cold leg temperatures (see Figure 15.4-26—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Cold Leg Temperatures) stop the power excursion once the pump reaches full speed. The DNB LCO is set sufficiently high that the startup of an inactive reactor coolant loop does not challenge the DNB SAFDL limits.

15.4.4.4 Radiological Consequences

No radiological consequences are calculated for this event because no fuel or cladding damage is predicted and there are no radiological releases to the environment.

15.4.4.5 Conclusions

The analysis of the startup of an inactive reactor coolant loop event demonstrates that minimum DNBR remains above the limit. The RCS pressure transient does not approach the peak RCS pressure limit.

15.4.4.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.4 events included in NUREG-0800, Section 15.4.4–15.4.5, (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 110 percent of the design values.

   – Response: The analysis described in Sections 15.4.4.2 and 15.4.4.3 indicates that there is acceptable design margin with respect to RCS pressure.

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

   – Response: The results in Section 15.4.4.3 demonstrate that this requirement is met. The DNB LCO is set sufficiently high so that the startup of an inactive RCP at an incorrect temperature event does not challenge the DNB SAFDL limits.

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

   – Response: The results in Section 15.4.4.3 demonstrate that this requirement is met. No RT setpoints are reached. The combination of Doppler feedback and increasing RCS cold leg temperatures stop the power excursion once the pump reaches full speed.

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

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

   – Response: There is no single failure for this event that makes its consequences more severe.

6. The guidance provided in SECY 77-439 as cited in Reference 3, SECY 94-084 as cited in Reference 3, and RG 1.206 with respect to the consideration of the performance of non-safety-related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems), must be evaluated and verified.

   – Response: Non-safety-related systems are modeled when they make the consequences of the event more severe. For this event, the non-safety-related PZR spray system is modeled because it reduces the increase in RCS pressure, which reduces DNBR margin.

15.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

This event is not applicable to the U.S. EPR.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

This section presents the analysis of the inadvertent boron dilution event. These events are classified as AOOs, as described in Section 15.0.0.1. Results are presented for the most limiting cases.

15.4.6.1 Identification of Causes and Accident Description

The main scenarios leading to a potential uncontrolled boron dilution are as follows:

- Operator addition of deborated water to the RCS. Administrative controls require close operator surveillance of the RCS during water addition. Additionally, procedures limit the rate of dilution and the duration of the dilution. Plant instrumentation allows the operator to monitor the RCS during the dilution process. If operator error occurs, instrumentation alarms alert the operator to the need for corrective action.

- Malfunction in the coolant degasification system. The evaporator and the degasifier column contain coolant during standby conditions from the last operating period. This coolant has a boron concentration that corresponds to the concentration from the last operating period, which can approach 0 ppm. If the degasifier is placed in service without discharging this coolant to the coolant storage tanks, the injection of unborated water into the RCS can occur. The delivery rate of this unborated water to the RCS is determined by the chemical and volume control system (CVCS) capacity. The volume of dilute is limited to the volume of the evaporator and the degasifier column.

- Malfunction in the coolant purification system. In this event, an insufficiently borated demineralizer in standby is placed into service with injection of deborated coolant to the RCS by the CVCS. The delivery rate of the deborated water to the RCS is determined by the CVCS capacity. The volume of dilute is limited to the volume of the demineralizer.

- Malfunction in the coolant storage and treatment system. A malfunction of the boron concentration measurements can cause a low boron concentration of the boric acid that is produced in the evaporator column. This boric acid is used for injection into the RCS. The worst-case assumption is an initially empty boric acid tank filled with a very low boron concentration product from the evaporator column. When an injection of boric acid is required, the RCS receives diluted boric acid, which can approach 0 ppm boron. The delivery rate to the RCS is limited by the delivery rate of the two boric acid pumps.

- Malfunction in the reactor boron water makeup system (RBWMS). Demineralized water can be inadvertently injected into the RCS via the CVCS with one or both of the demineralized water pumps from the RBWMS. It is considered an inadvertent injection because either no change in the RCS boron reactivity is required or the corresponding boric acid injection is missing. Demineralized water injection is initiated by either leakage make up to the RCS, rod position control, xenon compensation, load variations, or manual command. The maximum injection rate to the RCS is determined by the capacity of the two demineralized water pumps.

The last scenario leads to the worst combination of maximum possible dilution flow rate and minimum boron concentration and therefore bounds the other scenarios. This event is possible in the RCS operating modes.

The remainder of Section 15.4.6 analyzes the response of the U.S. EPR to a positive reactivity insertion event resulting from the addition of nonborated water to the RCS from a failure in the RBWMS or the CVCS. In order to cover the entire range of operating modes, three protection channels have been specifically defined to mitigate the consequences of the boron dilution event according to the reactor status:

- Anti-dilution at power conditions, modes 1 and 2.

- Anti-dilution in standard shutdown conditions, modes 3, 4, and 5 with at least one RCP in operation.

- Anti-dilution in shutdown conditions with RCPs secured, modes 5 or 6 with the reactor coolant pumps secured.

The sequence of events and operation of these protection channels is described in the following sections.

15.4.6.1.1 Operation at Power (Modes 1 and 2)

In this mode of operation, the reactor control rods are latched and at least one rod is partially withdrawn. An inadvertent boron dilution event is initiated either through operator error or through a malfunction in the RBWMS. The typical result of this action is that water with a reduced boron concentration (relative to the RCS) or no boron is injected into the RCS. This injection causes the RCS boron concentration to drop slowly, which leads to a positive reactivity insertion much like that seen during a rod or bank withdrawal event. Since RCPs are in operation, it can be assumed that the dilution flow is mixed instantaneously with the contents of the RCS.

When the plant is in MRC mode, the reactivity insertion leads to a relatively slow increase in core power and temperature. If no action is taken to discontinue the boron dilution event, dilution continues until a PS setpoint is reached. The actual setpoint reached is a function of core conditions and the rate of positive reactivity addition. The available trips and actions that can potentially intervene in this event are:

- DNB LCO alarm.
- DNBR RT.
- High core power level RT.
- Linear power density (LPD) LCO alarm.
- HLPD RT.
- High pressurizer level RT.
- Anti-dilution at power conditions trip (isolates CVCS).

In cases where an RT occurs first, the boron dilution might continue. Since the rods are on the bottom, protection against loss of shutdown margin is provided by the anti-dilution in standard shutdown conditions trip. Once the RCS boron concentration reaches the setpoint for the anti-dilution in standard shutdown conditions trip, then a signal is sent to isolate the charging pump suction. Three motor-operated valves (MOVs) automatically isolate the normal letdown line and the line from the volume control tank (VCT), and a valve from the in-containment refueling water storage tank (IRWST) automatically aligns to the charging pump suction providing borated water to the RCS. Simultaneously, the charging line isolation valves close and the three-way valve to the coolant storage and supply tanks fully opens. The charging flow to the RCP seal water system remains in service during this evolution. This action effectively isolates potential sources of dilution flow to the RCS. No credit is taken in transient analysis for the automatic lineup of boration flow to the RCS since this operation is performed with non-safety-grade equipment.

When the plant is in ACT control function mode, the positive reactivity insertion is balanced by control bank insertion, which maintains the RCS average temperature constant. This may continue until the control bank is inserted past its PDIL at which point a shutdown margin LCO alarm is issued to the control room. Actuation of this

LCO also initiates a signal to stop dilution and eventually to start boration but, since it is not a safety-grade signal, no credit is taken in the analysis of the event. If no further action is taken to discontinue the boron dilution event then dilution continues until the setpoint for the anti-dilution at power conditions trip is reached. This trip is based on a real-time derivation of the homogeneous RCS boron concentration and is designed to actuate prior to the RCS boron concentration dropping to a point where shutdown margin is lost. This trip automatically isolates the CVCS as described above, which terminates the boron dilution event. Other alarms and trips that might be activated prior to an anti-dilution trip are listed in the previous description of MRC.

15.4.6.1.2 Operation in Hot or Cold Shutdown States with RCPs in operation (Modes 3, 4, and 5)

In this mode of operation, the reactor is subcritical with the rods inserted. RCS boron concentrations are maintained high enough to provide sufficient shutdown margin. The initiation of boron dilution causes positive reactivity insertion leading to a continuous reduction in shutdown margin. Since RCPs are in operation, it can be assumed that the dilution flow is mixed instantaneously with the contents of the RCS. If no action is taken to discontinue the boron dilution event, dilution continues until the setpoint for the anti-dilution in standard shutdown conditions limitation trip is reached. Actuation of this limitation initiates a signal to stop dilution and eventually to start boration but, since it is not a safety-grade signal, no credit is taken in the analysis of the event. Eventually the anti-dilution in standard shutdown conditions PS trip is reached. This trip is based on a real-time derivation of the homogeneous RCS boron concentration and is designed to actuate prior to a loss of shutdown margin. This trip automatically isolates the CVCS, which terminates the boron dilution event. Another trip that might be activated prior to an anti-dilution trip is the source range high neutron flux trip, which generates an alarm in the control room.

15.4.6.1.3 Operation in Cold Shutdown with all RCPs Secured (Modes 5 and 6)

In this mode of operation, the reactor is subcritical with no operating RCPs. Heat removal is provided by the residual heat removal (RHR) system. The reactor vessel might be open or closed, and fuel may or may not be present. The RCS boron concentration is maintained at a refueling concentration level designed to provide adequate shutdown margin regardless of the core configuration. The initiation of boron dilution causes positive reactivity insertion, which leads to a continuous reduction in shutdown margin. If no action is taken to discontinue the boron dilution event, dilution continues until the setpoint for the anti-dilution in shutdown conditions with RCPs secured trip is reached. This trip is based on the measured boron concentration being injected into the core. This trip automatically isolates the CVCS, which terminates the boron dilution event. Another trip that might be activated prior to an anti-dilution trip is the source range high neutron flux trip, which generates an alarm in the control room.

15.4.6.2 Method of Analysis

The analyses determine values of the PS setpoints for the three anti-dilution protection channels that are based on boron concentration signals. Each of the three protection channels uses a slightly different algorithm for defining the measured boron concentration. The control logic for this trip system is described in Section 7.3.1.2.11.

The method for performing the analysis deterministically evaluates the uncertainty in each algorithm and applies it to the calculated critical boron concentration necessary to maintain shutdown margin. This method establishes the PS trip threshold in accordance with GDC 26. If the RCS boron concentration decreases below this threshold, indicating a possible dilution event or potential loss of shutdown margin, a signal is sent to isolate the CVCS thereby isolating the source of dilution. Additionally, the transient event is confirmed to be bounded by other events with regard to thermal margin (GDC 10) and reactor pressure boundary acceptance criteria (GDC 15).

The acceptance criteria for this event are as follows:

- Pressure in the reactor coolant and main steam systems is maintained below 110 percent of the design values.
- Fuel cladding integrity is maintained by keeping the minimum DNBR above the 95/95 DNBR limit.
- The event does not generate a more serious plant condition without other faults occurring independently.

If operator action is required to terminate the transient, it is not credited before 30 minutes during refueling and 15 minutes during startup, cold shutdown, hot shutdown, hot standby, and power operation.

15.4.6.2.1 Operation at Power (Modes 1 and 2)

In this mode of operation, the basis for the trip is the reconstructed transient boron concentration in the core based on charging flow rate, charging boron concentration, and primary system mass (assuming instantaneous mixing in the core). No temperature dependency of the primary system water mass with regard to average temperature (T_{AVG}) is included in this trip because a conservatively bounding mass occurs at HZP.

The instantaneous mixing algorithm used to derive the transient boron concentration is as follows:

$$BC_P^N = \frac{R}{1+R} BC_{inj}^N + \frac{1}{1+R} BC_P^{N-1}$$

where:

$$R = \frac{QF_{inj} * \Delta t}{M_P^N}$$

and:

BC_P^N =RCS boron concentration at time tN

BC_P^{N-1} =RCS boron concentration at time tN-1

BC_{inj}^N =Concentration of dilution flow

QF_{inj} =Charging rate of dilution flow

M_P^N =Mass of RCS system (fixed at HZP, 578°F conditions)

Δt =time from N-1 to N (system computational cycle time)

The results of this reconstruction are compared to a trip threshold, which represents the critical boron concentration for that HZP conditions with all rods inserted (ARI), the most reactive rod assumed stuck out of the core, and no xenon. This threshold represents the minimum boron concentration necessary to guarantee sufficient rod worth is available to shutdown the core if necessary. The PS threshold is defined by the following relationship:

Threshold =Critical boron concentration (HZP, ARI-1, xenon = 0)

+ Calculation uncertainty in critical boron concentration

+ Algorithm reconstruction uncertainty

+ TAVG uncertainty

+ Uncertainty due to trip response time

Various uncertainties must be accounted for in this threshold value including the following:

- Critical boron calculation uncertainty.
- Uncertainty in initialized RCS boron concentration.
- Uncertainty in the measured boron concentration of dilution flow.
- Uncertainty in RCS total mass (including CVCS system) used in the algorithm.
- Uncertainty in measured charging system flow rate.
- Uncertainty in T_{AVG} measurement.
- Uncertainty due to lag time from actuation of the trip signal to actual valve isolation of CVCS.

The uncertainties are applied in a deterministic manner that conservatively maximizes the required trip anti-dilution threshold. Both the algorithm reconstruction uncertainty and the uncertainty due to trip response time are a function of initial boron concentration, which changes with core burnup. Therefore, the trip threshold must adjust periodically with burnup. Table 15.4-13—Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant - Inputs for Anti-Dilution Analyses provides a list of applicable input parameters used in the uncertainty analyses.

15.4.6.2.2 Operation in Hot or Cold Shutdown States with RCPs in operation (Modes 3, 4, and 5)

This mode of operation is almost identical to the power operation mode above with the exception that the variation in the mass of the RCS as a function of the cold-leg temperature is accounted for. The PS allows for a table of trip thresholds to be input as a function of cold leg temperature. Therefore, the method for analysis is identical to that described in Section 15.4.6.2.1, except that the T_{AVG} uncertainty is replaced by the cold-leg temperature (T_{COLD}) uncertainty.

15.4.6.2.3 Operation in Cold Shutdown with all RCPs Secured (Modes 5 and 6)

This mode of operation is unique in that, with no RCPs in operation, it cannot be assumed that instantaneous mixing occurs. Instead, the RCS is maintained at a constant boron concentration equivalent to the IRWST. The PS trip threshold is defined, therefore, in a manner to detect charging from streams at a concentration less than the expected IRWST concentration. The setpoint also maintains the minimum allowed RCS boron concentration in excess of the maximum refueling mode critical boron concentration. The PS threshold for this mode of operation is defined by the following relationship:

Threshold = Minimum value of IRWST boron concentration - boron measurement uncertainty

This threshold is confirmed to satisfy the following relationship as well:

Threshold maximum refueling mode critical boron concentration + boron measurement uncertainty

15.4.6.3 Results

15.4.6.3.1 Operation at Power (Modes 1 and 2)

A boron dilution event at power does not present a challenge to DNB because the reactivity addition rates are bounded by those assumed for the uncontrolled control rod assembly withdrawal at power. Maximum dilution rates typically range from 0–0.1 ppm/s. Therefore, maximum reactivity addition rates remain below 2.0 pcm/s, which is within the rod worth spectrum analyzed in Section 15.4.2. As a result, the thermal margin and overpressure results for the boron dilution event are bounded by those reported in Section 15.4.2.

Calculation of the anti-dilution at power trip setpoints is based on determining the total trip uncertainties. The uncertainty calculations are performed over the entire cycle, spanning a range of minimum critical boron concentrations (based on using boron enriched to 37 atom percent boron-10 and HZP, ARI-1, xenon = 0) from 400 ppm down to 0 ppm. This data is fit using bounding linear regression techniques to develop the following equation that defines the trip setpoint as a function of the required minimum critical boron concentration (CBC):

Trip setpoint = 101.12 + 1.0265 x CBC

The trip setpoints are adjusted periodically throughout the cycle so that the PS intervenes prior to reaching the minimum critical boron concentration.

15.4.6.3.2 Operation in Hot or Cold Shutdown States with RCP's in operation (Modes 3, 4, and 5)

In this mode, the calculation of the anti-dilution in standard shutdown conditions is performed similar to the anti-dilution at power trip above. The exceptions are that the T_{AVG} uncertainty is replaced with cold-temperature ($T_{COLD)}$ uncertainty and the calculation is performed at several T_{COLD} temperatures ranging from HZP temperatures down to 200°F. This data is fit using bounding linear regression techniques to develop the following equation that defines the trip setpoint as a function of the required minimum CBC:

Trip setpoint = 115.90 + 1.0208 x CBC

The trip setpoints are adjusted periodically throughout the cycle so that the PS intervenes prior to reaching the minimum critical boron concentration.

15.4.6.3.3 Operation in Cold Shutdown with all RCPs Secured (Modes 5 and 6)

In this mode of operation, a single trip setpoint is based on the required minimum boron concentration of the IRWST. A bounding boron measurement uncertainty of 400 ppm is used with a minimum IRWST boron concentration (using boron enriched to 37 atom percent boron-10) of 1327 ppm. The resulting anti-dilution in shutdown conditions with RCPs secured trip is 927 ppm. Measured boron concentrations below this level indicate that some source of water other than the IRWST is being used and a boron dilution event is likely in progress. This setpoint is sufficiently high to maintain the RCS boron concentration higher than the minimum refueling mode critical boron concentration (using boron enriched to 37 atom percent boron-10) of 519 ppm after accounting for the uncertainty.

15.4.6.4 Radiological Conclusions

Radiological consequences are not calculated for this event because no fuel or cladding damage occurs and no release of radioactive materials to the environment occurs.

15.4.6.5 Conclusions

Anti-dilution, safety-related protection channels provide effective protection by automatically eliminating the dilution source prior to the loss of shutdown margin for all modes of operation. The analyses indicate that the PS anti-dilution functions intercede in time to prevent a complete loss of shutdown margin. This action preserves the capability to shutdown the reactor or maintain it in a shutdown condition at all times in accordance with GDC 26 criteria. In addition, fuel cladding integrity and fuel centerline melt temperature results are bounded by those for the uncontrolled control rod assembly withdrawal at power (Section 15.4.2), which meet the GDC 10 and 15 criteria.

The plant instrumentation, protection functions, and equipment are sufficient to preclude fuel or cladding damage. The core remains adequately cooled throughout this event.

15.4.6.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.6 events included in NUREG-0800, Section 15.4.6, (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: The RCS overpressure aspects of the inadvertent boron dilution event are bounded by the analyses of Section 15.5, “Increase in Reactor Inventory,” which addresses a malfunction of the CVCS system.

2. Fuel cladding integrity is maintained by keeping the minimum DNBR remains above the 95/95 DNBR limit based on acceptable correlations with SRP Section 4.4 (Reference 3).

   – Response: The results of Section 15.4.6.3.1 show that from a DNB/FCM margin standpoint, the boron dilution event at power conditions remains bounded by the uncontrolled control rod assembly withdrawal event at power (described in Section 15.4.2). Boron dilution events from subcritical conditions are precluded from reaching criticality by the anti-dilution trip system therefore no challenge to thermal design limits occurs.

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

   – Response: The successful mitigation of the inadvertent boron dilution event as shown in Section 15.4.6.3 demonstrates that the PS is adequate. The instrumentation and control for the RPS is presented in Sections 7.2 and 7.3. The analyses indicate that the anti-dilution trips intercede in time to prevent a complete loss of shutdown margin. This provides the capability to shut down or maintain a shutdown condition.

4. If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost:

   – During refueling: 30 minutes.

   – During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes.

     • Response: Operator action is not relied upon to terminate this event.

5. The applicant's analysis of moderator dilution events should use an acceptable analytical model. The following plant initial conditions should be considered in the analysis: refueling, startup, power operation (automatic control and manual modes), hot standby, hot shutdown and cold shutdown. Parameters and assumptions in the analytical model should be suitably conservative. The following values and assumptions are acceptable:

   A. For analyses during power operation, the initial power level is rated output (licensed core thermal power) plus an allowance of two percent to account for power-measurement uncertainty. The analysis may use a smaller power-measurement uncertainty if justified adequately.

      • Response: Power level is not a parameter of interest in this event.

   B. The boron dilution is assumed to occur at the maximum possible rate.

      • Response: Use of both demineralizer pumps in the evaluation of this event results in the maximum possible dilution flow rate.

C. Core burnup and corresponding boron concentration must yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution. The core burnup must be justified by either analysis or evaluation.

- Response: This evaluation covers the full spectrum of burnups and temperatures.

D. Fuel assemblies are installed in the core.

- Response: This condition is accounted for in the analyses.

E. A conservatively low value is assumed for the reactor coolant volume.

- Response: The RCS mass calculation conservatively neglects the volume of the pressurizer and surge line as well as the vessel upper head volume.

F. For analyses during refueling, control rods are withdrawn from the core. An alternate assumption requires adequate justification and delineation of necessary controls so the alternate assumption remains valid.

- Response: The anti-dilution trip methodology during refueling conditions is decoupled from the actual core conditions and is only a function of the required minimum allowable refueling boron concentration.

G. For analyses during power operation, the minimum shutdown margin allowed by the TS (usually one percent) is assumed prior to boron dilution.

- Response: This assumption is based on minimizing the time before a required manual action is required. The at-power anti-dilution trip is automatic and designed to prevent the loss of shutdown margin regardless of the starting condition.

H. A conservatively high reactivity addition rate is assumed for each analyzed event to take into account the effect of increasing boron worth with dilution.

- Response: The reactivity addition rates do not factor into this methodology. Instead, the anti-dilution trip is designed to maintain the RCS boron concentration above the level necessary to provide shutdown margin.

I. Conservative scram characteristics are assumed.

- Response: Scram characteristics are unnecessary here since it is shown that the boron dilution event is bounded by the uncontrolled rod withdrawal at power event.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 Identification of Causes and Accident Description

Fuel and core loading errors can lead to increased heat fluxes. These errors can arise from the inadvertent loading of one or more fuel assemblies into improper core locations, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment. Fuel design and fabrication controls combined with plant refueling procedures minimize the likelihood of fuel loading errors.

The aeroball measurement system (AMS) incore monitoring system provides additional protection against fuel loading errors by detecting power distribution anomalies. Administrative controls and the initial low-power flux map are the primary means used to determine if the core is loaded consistent with the design. Therefore, the operability status of the AMS is a major component in detecting fuel misleads. Despite these safeguards, however, the potential exists for an undetected fuel loading error to occur and present a challenge to fuel rod failure limits.

The inadvertent loading and operation of a fuel assembly in an improper location event is an AOO that is allowed to have fuel failures, provided several acceptance criteria are satisfied. These criteria include the requirement that the radiological consequences be a small fraction of the 10 CFR Part 100 criteria.

15.4.7.2 Method of Analysis and Assumptions

The misload analysis begins with the determination of whether or not a misload is detectable by the incore monitoring system. If misloaded assemblies are not detectable, the radiological consequences analysis must be applied for the most limiting misloads with respect to fuel design limits, and must determine that the number of fuel failures is within the regulatory acceptance criteria. The radiological acceptance criteria associated with the alternative source term (AST) methodology are found in 10 CFR 50.34(a)(1); for the offsite receptors the criterion is equal to 25 rem total effective dose equivalent (TEDE). 10 CFR 50, Appendix A, GDC 19, as incorporated by reference in 10 CFR 52.47(a)(1), includes the value for control room personnel, which is 5 rem TEDE.

The analysis to determine if misloaded assemblies are detected by the incore monitoring system employs steady-state power distributions in the x-y plane of the core, which are calculated using the three-dimensional nodal code PRISM described in Reference 1. Representative power distributions in the x-y plane for a correctly loaded core are described in Section 4.3. Calculations interchanging assembly locations are performed to swap fuel types within the core between a variety of interior and peripheral locations to model the worst-case assembly misloads. The calculations are modeled at beginning of life and 30 percent RTP to approximate the initial measured power distribution during startup of the cycle. The minimum number of operable aeroball detectors is used to provide a bounding analysis.

If assemblies are not detectable when misloaded, an analysis is performed to determine the peaking limitations that are required to prevent fuel failures due to DNB, centerline fuel melt (CFM), and one percent transient clad strain (TCS). No consideration is given to limiting conditions for operation on DNB, LPD, or peaking because these rely on accurate information from the incore instrumentation. The $F_{\Delta H}$ limit is set so that the limiting fuel rod in the core does not experience DNB with 95 percent probability at 95 percent confidence (95/95). The LYNXT computer code described in Reference 1 is used to determine the maximum hot pin $F_{\Delta H}$ that results in the minimum DNB ratio being equal to the design limit of the ACH-2 critical heat flux (CHF) correlation described in "The ACH-2 CHF Correlation for the U.S. EPR" (Reference 4). The results are evaluated deterministically and include the impacts of uncertainties and operational control bands.

The calculated F_Q limit is set so that the highest peaked fuel pin in the core does not have a peak linear heat generation rate (LHGR) that is greater than the limit for CFM and TCS. The calculated F_Q peaking limit is used with the results of the analysis of misload assemblies to determine how many failed fuel pins or fuel assemblies need to be considered in the offsite radiological dose evaluations.

The methodology for U.S. EPR design basis accident (DBA) radiological evaluations is described in Section 15.0.3. The DBA analyses follow the guidance of SRP 15.0.3 (Reference 3) and RG 1.183. This methodology addresses the submersion and inhalation doses and the direct shine doses from contained or external sources.

15.4.7.3 Results

Results of the misload analyses show that the AMS in the EPR cannot detect all potential misloaded assemblies. The calculated F_Q and $F_{\Delta H}$ limits for fuel failure due to the misloading of assemblies are 3.20 and 1.71, respectively. The F_Q for the undetected misload cases were found to remain below the F_Q limit. Therefore, no fuel failures occur due to CFM or TCS. Cases found to exceed the fuel failure $F_{\Delta H}$ limit are evaluated to determine the maximum number of potential failed rods.

The maximum number of assemblies with pins exceeding the $F_{\Delta H}$ limit during an undetected misload is four. Four assemblies include conservatively up to 1060 failed rods, or less than 2 percent of the pins in the core. Including all pins is highly conservative because it is likely that only a small portion of each assembly actually violate the $F_{\Delta H}$ limit.

The radiological analysis for offsite radiological consequences and main control room habitability includes an evaluation of the locked-rotor accident scenario with clad failure of up to 9.5 percent of the pins in the core. The 9.5 percent clad failure is the maximum fuel damage that was determined to result in TEDE doses within 90 percent of the dose acceptance criteria.

15.4.7.4 Conclusions

Loading errors in assembly fabrication or core loading are considered extremely unlikely because of strict procedural control used during manufacturing and core loading. A computer database with material traceability information and operation sequencing and completion is used to link the fuel cycle design and manufacturing processes. Fuel cycle design data is tied with identifiers for batch, the associated part numbers, and the fuel assembly serial numbers with automated processes.

To reduce the probability of core loading errors, each fuel assembly is marked with an identification number and loaded in accordance with a core-loading diagram. Prior to core loading, the identification number of each assembly is checked before it is moved into the core. Serial numbers read during or after fuel movement are subsequently recorded on the loading diagram as a further check on proper placement after the loading is completed.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNB and increased fuel and clad temperatures are limited to the incorrectly loaded pin or pins and perhaps the immediately adjacent pins.

Assembly loading errors that cannot be detected by the incore instrumentation are evaluated with the radiological consequences of the fuel-loading error and are well within the required criteria. Meeting these criteria provides assurance that, in the event of an undetected fuel-loading error, radiation exposures at the site boundary does not exceed a small fraction of the reference values specified in 10 CFR Part 100.

15.4.7.5 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.7 events included in NUREG-0800, Section 15.4.7, (Reference 3) and descriptions of how these criteria are met are listed below:

The primary safeguards against fuel-loading errors are procedures and design features to minimize the likelihood of the event. Additional safeguards include incore instrumentation systems which detects errors. However, should an error be made and go undetected, it is possible in some reactor designs for fuel rod failure limits to be exceeded. Therefore, the following acceptance criteria cover the event of operation with misloaded fuel caused by loading errors:

1. To meet requirements of GDC 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel-loading errors after fueling operations.

   – Response: Fuel design and fabrication controls combined with plant refueling procedures minimize the likelihood of fuel loading errors. In addition, the incore monitoring system must provide additional protection against fuel loading errors. The AMS identifies power distribution anomalies to assist in the detection of misload assemblies.

2. In the event the error is not detectable by the instrumentation system and fuel rod failure limits are exceeded during normal operation, the offsite consequences should be a small fraction of 10 CFR Part 100 criteria. A small fraction is interpreted to be less than 10 percent of 10 CFR Part 100 reference values. For the purpose of this review, the radiological consequences of fuel-loading error should include consideration of the containment, confinement, and filtering systems. The applicant's source terms and methodologies with respect to gap release fractions, iodine chemical form, and fission product release timing should reflect NRC-approved source terms and methodologies.

   – Response: If misloaded assemblies are not detectable, the radiological consequences analysis must be applied for the most limiting misloads with respect to fuel design limits and the number of fuel failures must not exceed a small fraction (i.e., be less than 10 percent of 10 CFR Part 100 reference values). The results in Section 15.4.7.3 demonstrate that, in the event of an undetected fuel-loading error, radiation exposures at the site boundary are within the regulatory acceptance criteria. The methodology for U.S. EPR design basis accident (DBA) radiological evaluations is described in Section 15.0.3.

15.4.8 Spectrum of Rod Ejection Accidents in a PWR

15.4.8.1 Identification of Causes and Accident Description

The rod ejection accident is defined as the postulated rupture of a control rod drive mechanism housing that results in the complete ejection of an RCCA from the reactor core. The consequences of the mechanical failure are a rapid positive reactivity insertion and an increase in the local power peaking with high local energy deposition in the fuel assembly, accompanied by an initial pressure increase in the RCS. It is postulated that the ejection occurs over a 0.1-second interval, and the reactivity increase during this time is nearly linear. An RCCA ejection event is considered a PA according to the classification system described in Section 15.0.0.1.

The power spike resulting from the RCCA ejection is quickly countered by Doppler reactivity feedback, when the fuel temperatures begin to increase, and may be terminated by RT. The probable trips are the high positive neutron flux rate, low neutron flux doubling time (intermediate range) or high neutron flux (intermediate range) signals.

Although the initial increase in power occurs too fast to be impacted by the RT, the RT terminates the event and limits the damage from the pulse. For the ejection of lower worth control rods that produce small power increases (less than approximately 20 percent) the ex-core neutron detector signals may not be high enough to activate the trip signals. In this case, the power is limited to the prompt jump and the power stabilizes at a level that balances the reactivities due to the ejected rod and the fuel and moderator temperatures. For this situation, the reactor trips on either low PZR pressure, low saturation margin, high SG pressure or low DNBR.

In addition to the overall reactor power excursion, an increase in the radial and axial power peaking factors occurs because of the skewed power distribution near the ejected rod. The magnitude of the reactivity insertion and the power peaking shift depends on the insertion depth of the bank to which the RCCA belongs, as well as the number of RCCAs that are inserted, the point in the fuel cycle, and the core loading.

The ejection of an RCCA could cause an opening in the reactor vessel upper head, which is a small-break LOCA. Mitigation of this event is bounded by the small-break LOCA analyses described in Section 15.6.5.2.

15.4.8.2 Method of Analysis and Assumptions

Different methodologies are used to evaluate the core thermal-hydraulic and neutronics response and the primary system pressure response.

15.4.8.2.1 Core Thermal-Hydraulics and Neutronics

The approach for analyzing the spectrum of rod ejection accidents is outlined in the U.S. EPR Rod Ejection Accident Methodology Topical Report (Reference 5). This method combines the results from neutronic, thermal-hydraulic, and plant simulations.

A spectrum of initial conditions for the event is considered by evaluating BOC and EOC conditions. For the purposes of neutronic and thermal-hydraulic analyses, the system pressure, inlet temperature, and mass flux boundary conditions are kept constant. The fuel rod model for the heat transfer allows the use of temperature dependent thermo-mechanical properties for the fuel-to-clad gap heat conductance and the fuel and clad thermal conductivities and specific heats. The fuel conductivity, gap conductance, and pellet radial power profiles are evaluated at different pellet burnups to account for the changes to those properties at BOC and EOC conditions. The hot fuel rod channel is included in the assembly of interest and is evaluated for the fuel melting limit, fuel enthalpy criteria, and the DNBR performance.

The transient is characterized by the following integral and local core phenomena:

- Very fast increase of the neutron flux and reactor power with strong spatial redistribution axially and radially near the fuel assembly with the ejected rod.
- Heat conductance in the fuel rods and different heat transfer regimes from the cladding to the coolant.
- Coolant flow behavior dependent on pressure drop and crossflow.
- Maximum heat flux typically reached before RT.

A spectrum of RCCA worths is evaluated. Some events might not cause RT, in which case the inherent reactivity feedback from fuel temperatures acts to stabilize the reactor power at some elevated power level, whereupon the balance of plant systems begin to respond. Most of the transients with high worth control rods and negative moderator reactivity coefficients occur rapidly, so that the changes in primary system and secondary system parameters are limited and do not influence the core transients.

Fuel Cladding Failure Criteria for Radiological Assessment

The total number of fuel rods that must be considered in the radiological assessment is equal to the sum of the fuel rods failing one or more of the criteria from Reference 3, Appendix B, Section B, page 4.2-33.

Core Cooling Acceptance Criteria

Fuel rods are evaluated using design-specific criteria that account for manufacturing tolerances and modeling uncertainties using NRC-approved methods (References 1,2, and 5), including acceptance criteria for burnup-enhanced effects on pellet power distribution, fuel thermal conductivity and fuel melting temperature. Table 15.4-14—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Limits for U.S. EPR provides a summary of the analysis limits for the U.S. EPR.

15.4.8.2.2 Overpressurization

The plant simulation computer code S-RELAP5 (Reference 1) is used to determine the peak pressure response of the primary system to the RCCA ejection event. For the overpressurization analysis, the assumption is that the RCCA is ejected, but not totally released from the drive mechanism. Boundary conditions for this analysis are biased in order to provide the largest magnitude pressure response. The resulting maximum pressure is verified not to exceed the system design pressure limit for postulated accidents.

Table 15.4-15—Rod Ejection Accident Overpressurization Analysis –Key Input Parameters presents the initial conditions used in the S-RELAP5 analysis. Table 15.4-16—Rod Ejection Accident Overpressurization Analysis – Equipment Status presents the status of key plant systems and equipment. The maximum reactor pressure during the excursion must remain below Service Limit C as defined by the ASME Boiler and Pressure Vessel Code (Reference 6). The S-RELAP5 analysis is used to demonstrate that the primary pressure does not exceed 120 percent of the system design pressure (i.e., 3056 psia).

15.4.8.3 Results

15.4.8.3.1 Core Thermal-Hydraulics and Neutronics

The rod configurations are evaluated to determine the most limiting cases from a spectrum of RCCA worths and initial power levels ranging from HZP to HFP. In all cases evaluated, the inherent negative fuel Doppler reactivity coefficient is adequate to limit the power excursion.

For the scenarios that cause DNBR to fall below the DNB SAFDL limit, the percent of fuel rods that fail remains below the radiological release limit. If the reactor does not initially trip on the neutron signals, the event then becomes similar to a single rod withdrawal event described in Section 15.4.3. The potential for the additional depressurization of the primary due to leakage from the ejected rod flange housing makes this scenario similar to the depressurization events described in Section 15.6. These events are protected by the low DNB LCO and RT on low PZR pressure, low saturation margin or high SG pressure. The power event is terminated before fuel failure limits for radiological release are exceeded.

Tables 15.4-17—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Results for BOC conditions and 15.4-18—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Results for EOC conditions provide a summary of the spectrum of ejected rod transients. Figure 15.4-28—Rod Ejection Accident DNBR Analysis – BOC HFP Transient Power Fraction shows the nuclear power Figure 15.4-29 Rod Ejection Accident DNBR Analysis – BOC HFP Transient Peak Fuel and Cladding Temperatures shows the fuel clad temperature behavior for the limiting case in terms of DNBR and fuel temperatures (at BOC HFP). Figures 15.4-30 Rod Ejection Accident DNBR Analysis – EOC HZP Transient Power Fraction and 15.4-31— Rod Ejection Accident DNBR Analysis – EOC HZP Transient Peak Fuel and Cladding Temperatures show the same parameters for the limiting case in terms of enthalpy rise (at EOC HZP).

15.4.8.3.2 Overpressurization

The overpressure analysis is performed in a separate set of analyses using the point kinetics reactivity simulations in S-RELAP5 (Reference 1). The rod ejection event is modeled conservatively as ejecting the highest worth RCCA within 0.1 s. The result is a rapid reactivity insertion with large local power peaking. The power peak is limited by the fuel temperature reactivity feedback due to increased fuel temperatures while the transient is eventually terminated by the PS. In the cases analyzed, the primary system pressure does not exceed 120 percent of the design pressure (3056 psia, or the Service Limit C as defined in the ASME Code (Reference 6).

The following cases are analyzed using conservatively bounding ejected RCCA worths at each power level:

- Rod ejection at HFP, ejected worth: 300 pcm.
- Rod ejection at 60 percent NP, ejected worth: 500 pcm.
- Rod ejection at HZP, ejected worth: 700 pcm.

Table 15.4-19—Rod Ejection Accident Overpressurization Analysis – Sequence of Events presents the sequence of events for the HFP, 60 percent NP and HZP cases. Figures 15.4-32—HFP Rod Ejection Accident Overpressurization Analysis – Percent Reactor Power through 15.4-43—HZP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature present the transient response.

15.4.8.4 Radiological Consequences

The radiological consequences of the rod ejection accident are evaluated in Section 15.0.3.9.

15.4.8.5 Conclusions

For the spectrum of rod ejection accidents evaluated, none of the power excursions caused the fuel temperatures to reach either the limiting fuel melt temperature or the fuel enthalpy limits. For the events which exceeded the DNBR limit, the number of fuel failures was less than the value allowed for the radiological release limit. The stresses due to the primary pressure response during the transients did not exceed Service Limit C defined in the ASME Code (Reference 6).

15.4.8.6 SRP Acceptance Criteria

A summary of the SRP acceptance criteria for Section 15.4.8 events included in NUREG-0800, Section 15.4.8, (Reference 3) and descriptions of how these criteria are met are listed below:

1. GDC 13, describes the availability of instrumentation to monitor variables and systems over their anticipated ranges to provide adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.
   - Response: The successful mitigation of the spectrum of rod ejection accidents as shown in Section 15.4.8.3.1 demonstrates that the PS is adequate. The instrumentation and control for the RPS is presented in Sections 7.2 and 7.3. Initial conditions for the analysis took into account the limiting rod worths

from rod insertion limits, reactivity coefficients for three-dimensional nodal calculations, and a breadth of operating conditions.

2. Acceptance criteria are based on meeting GDC 28 requirements as to the effects of postulated reactivity accidents that result in neither damage to the reactor coolant pressure boundary greater than limited local yielding nor sufficient damage to impair significantly core-cooling capacity. Regulatory positions and specific guidelines necessary to meet the relevant requirements of GDC 28 are in RG 1.77 and SRP Section 4.2 (Reference 3). The maximum reactor pressure during the assumed excursion should be less than the value that result in stresses that exceed Service Limit C as defined in Reference 6.

   – Response: The analysis described in Section 15.4.8.3.2 indicates that acceptable design margin is present with respect to RCS pressure. Using the analysis techniques described in References 1 and 5, the calculations indicate that no fuel temperatures reach the melting condition, nor do they exceed the enthalpy rise criteria.

3. 10 CFR 100.11 and 10 CFR 50.67 establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone. The fission product inventory released from the failed fuel rods is an input to the radiological evaluation under SRP Section 15.0.3 (Reference 3). SRP Section 4.2 (Reference 3) describes fuel rod failure mechanisms. Guidance for calculating radiological consequences is in RGs 1.183 and 1.195.

   – Response: The analysis described in Section 15.4.8.3.1 indicates that the predicted number of fuel failures remains below that which causes a violation of the dose limits for a radiological release.

15.4.9 Spectrum of Rod Drop Accidents (BWR)

This event is not applicable to the U.S. EPR.

15.4.10 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., December 2007.

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

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

5. ANP-10286P, Revision 0, “U.S. EPR Rod Ejection Accident Methodology Topical Report,” AREVA NP Inc., November 2007.

6. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article 3224, "Level C Service Limits," American Society of Mechanical Engineers, 2004.

Table 15.4-1—Uncontrolled Control Bank Withdrawal from a Subcritical or Low-Power Startup Condition - Key Input Parameters

Parameter	Analysis Value
Initial reactor power	4.590 W (10^{-9} of 4590 MW)
Average RCS temperature	578°F at HZP
Initial PZR pressure	2250 psia
Initial RCS loop flow rate	119,692 gpm per loop
Maximum possible RCCA bank differential worth	12.0 pcm/sec
Moderator temperature coefficient	+5.73 pcm/°F at BOC
Doppler temperature coefficient	-1.17 pcm/°F at most reactive exposure
Bounding value for fraction of delayed neutrons (β)	0.007358
Core reactivity	3000 pcm
U^{-238} capture to fission ratio	0.85
Time when LOOP is assumed	At TT

Table 15.4-2—Uncontrolled Control Bank Withdrawal from a Subcritical or Low-Power Startup Condition – Equipment Status

Plant equipment or system	Status
RPS	Operable (single failure proof)
RCPs	Operating until LOOP
PZR spray	Operable

Table 15.4-3—Uncontrolled Control Bank Withdrawal from a Subcritical or Low-Power Startup Condition – Sequence of Events

Event	Time (s)
Withdrawal beginning	1.00
RT setpoint reached, high neutron flux rate of change (ROC)	53.86
RT signal issued	54.16
LOOP and TT	54.31
Rod insertion beginning	54.56
Peak of total power	56.00
PZR spray set point reached	57.00
RCS pressure peak	59.50

Table 15.4-4—Uncontrolled Control Bank Withdrawal at Power – Key Input Parameters

Parameter	Analysis Value
Initial reactor power	4612, 2767, 1153 MW cases
Average RCS temperature	594°F at 100% 594°F at 60% 587°F at 25%
Initial PZR pressure	2250 psia
Initial RCS loop flow rate	119,692 gpm per loop
Maximum possible RCCA bank differential worth	5.59 pcm/sec at BOC 7.09 pcm/sec at EOC
Moderator temperature coefficient	0.0 pcm/°F to +5.73 pcm/°F at BOC -42.0 pcm/°F to -50 pcm/°F at EOC
Doppler temperature coefficient	-1.17 pcm/°F at BOC -1.85 pcm/°F at EOC
Bounding value for fraction of delayed neutrons (β)	0.007358 at BOC 0.005151 at EOC
U^{-238} capture to fission ratio	0.85
Time when LOOP is assumed	At TT
Initial PZR Level	54.3% of span at 100% 54.3% of span at 60% 44.3% of span at 25%

Table 15.4-5—Uncontrolled Control Bank Withdrawal at Power – Equipment Status

Plant Equipment or System	Status
RCCA position control mode	Manual
Turbine control valves position control mode	Automatic
PZR spray	Available
RCPs	Operating until LOOP

Table 15.4-6—Uncontrolled Control Bank Withdrawal at Power – Sequence of Events

Event	Time (s)
5.59 pcm/s RCCA withdrawal event starts	0.0
Pressurizer spray on	6.30
RT setpoint reached, high neutron flux ROC	25.1
RT	25.1
Peak core power occurs	25.7
SG MSRVs open	34.8
PSRV opens	27.5
End of transient calculation	60.0

Table 15.4-7—Dropped RCCA – Key Input Parameters

Parameter	Value
Initial reactor power	4612 MW at 100%
Average RCS temperature	594°F
Initial PZR pressure	2250 psia
Initial RCS loop flow rate	119,692 gpm per loop
Maximum possible RCCA bank withdrawal rate.	30 in/min
Most positive moderator temperature coefficient (BOC cases)	0.0 pcm/°F
Most negative moderator temperature coefficient (EOC cases)	-50.0 pcm/°F
Most reactive exposure (MRE) Doppler temperature coefficient (BOC cases)	-1.17 pcm/°F
EOC Doppler temperature coefficient (EOC cases)	-1.85 pcm/°F
Dropped RCCA worth	12 pcm minimum 91 pcm maximum
Dropped RCCA bank worth	423 pcm minimum 2167 pcm maximum
Fraction of core reactivity vs. fraction of insertion distance	HFP curve used
Time when LOOP is assumed	At TT

Table 15.4-8—Dropped RCCA – Equipment Status

Plant Equipment or System	Status
RCCA control mode	ACT control function or manual
Turbine control valves position control mode	Open consistent with full-power operation, fixed-position, and automatically controlled cases
RCPs	Operating until LOOP
SPNDs	Cases with low DNBR RT: No signal from SPND in location with lowest DNBR. Cases with HLPD RT: No signal from SPND in location with highest (or alternately, second-highest) LPD.
Excore detectors	Cases with no low DNBR or HLPD RT: No signal from excore detector in core quadrant with highest (or alternately, second-highest) power
Pressurizer heaters	Disable
Pressurizer spray	Available
Main feedwater	Automatic
Auxiliary feedwater	Available
Rod block system	Disable
Initial RCCA position	PDIL
Rod position control system NI feedback	Sensor-shadowed
RPS NI signals	Decalibrated and sensor-shadowed

Table 15.4-9—Dropped RCCA – Sequence of Events

Event	Time (s)
Dropped control rod initiated	0
Control bank withdrawal begins	2.2
Rod completely dropped	3.5
Low DNBR trip signal	13.1
Low DNBR trip (with sensor delay)	13.6
TT signal	13.7
LOOP	13.7
Control rods drop for core reactivity (with delays for sensor, breaker opening and gripper release)	13.9

Table 15.4-10—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Key Input Parameters

Parameter	Analysis Value
Initial reactor power	2,754 MW
Average RCS temperature	594°F[1]
Initial PZR pressure	2250 psia
Initial RCS loop flow rate	119,692 gpm per loop[2]
Moderator temperature coefficient	-42 pcm/°F
Doppler temperature coefficient	-1.512 pcm/°F
Bounding value for fraction of delayed neutrons (β)	0.005151
U^{-238} capture to fission ratio	0.85
Time when LOOP is assumed	At TT

Notes for Table 15.4-10:

1. Primary average temperature does not include idle loop temperatures.
2. RCS loop flow shown for operating loops only.

Table 15.4-11—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Equipment Status

Plant equipment or system	Status
RCCA position control model	Manual
PZR heaters	Disabled
PZR spray	Operable
RPS	Operable (single failure proof)
RCPs	Three operating until restart of fourth RCP
Offsite power	No LOOP unless RT due to restart of fourth RCP

Table 15.4-12—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Sequence of Events

Event	Time (s)
RCP 1 is restarted	0.0
Peak RCS bottom pressure, 2371 psia	12.5
RCP 1 at full speed	16.5
Peak core power occurs, 75.7%	16.7
End of analysis	100

Table 15.4-13—Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant - Inputs for Anti-Dilution Analyses

Parameter	Value
Charging flow rate[1]	57.4 lbm/s
Charging flowmeter uncertainty	± 4.41 lbm/s
T_{AVG} uncertainty and control band	± 3.0°F (uncertainty) ± 2.0°F (control band)
T_{COLD} uncertainty	± 7.62°F
Nominal RCS volume[2]	12229.8 ft^3
Anti-dilution trip response time ($T_1 + T_2$)	66 s
CVCS isolation valve stroke time (T_7)	40 s
Boron meter uncertainty	MAX (60, 20 + 0.06 * BC[3])
Critical boron calculation uncertainty	100 ppm
Most negative MTC	-50 pcm/°F
Minimum boron worth	-7.25 pcm/ppm

Notes for Table 15.4-13:

1. Based on both demineralizer pumps operating at maximum capacity.
2. Determined by reducing the total RCS volume by the volume of the PZR and surge line as well as the vessel upper head.
3. Measured boron concentration.

Table 15.4-14—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Limits for U.S. EPR

Criterion Description	Limit
Maximum enthalpy of the fuel	<150 cal/g
Maximum energy deposition during prompt power pulse for core powers <5%	<110 cal/g
Fuel melt during prompt power pulse	=0.00%
After power pulse, limit on # of pins effectively failed due to DNBR or fuel melt	<30%

Table 15.4-15—Rod Ejection Accident Overpressurization Analysis – Key Input Parameters

Parameter	Value
Initial reactor power	4.6 W at HZP 4590 MW at 100%
Average RCS temperature	578°F at 0% 587°F for 25% to 35% 594°F for 60% to 100%
Initial PZR pressure	2250 psia
Initial RCS loop flow rate	119,692 gpm per loop
Ejected RCCA worth	300 pcm at HFP 500 pcm at 60% 700 pcm at HZP
Moderator temperature coefficient	0.0 pcm/°F at HFP 3.73 pcm/°F at HZP and at 60%
Doppler temperature coefficient	-1.17 pcm/°F
Bounding value for fraction of delayed neutrons (β)	0.007358
Scram reactivity	6161 pcm at HFP (BOC) 5964 at 60% BOC 3000 pcm at HZP
U^{238} capture to Fission Ratio	0.85
Fraction of scram reactivity vs. fraction of insertion distance	HFP curve used for starting powers over 50%. Under 50%, use the HZP curve.
Time when LOOP is assumed	LOOP is not required

Table 15.4-16—Rod Ejection Accident Overpressurization Analysis – Equipment Status

Plant Equipment or System	Status
RCCA position control model	Manual
PZR heaters	Disabled
PZR spray	Disabled
RCPs	Operating

Table 15.4-17—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Results for BOC

Parameter	Criterion	0	25	35	60	100
Maximum ejected rod worth, pcm	-	433	362	346	286	64
Delayed neutron fraction	-	0.0055	0.0055	0.0055	0.0055	0.0055
MTC, pcm/°F	-	2.16	1.32	1.35	0.34	0.01
DTC, pcm/°F	-	-1.22	-1.14	-1.11	-1.05	-0.96
Initial F_Q	-	NA[1]	3.01	2.88	2.63	2.36
Maximum transient F_Q	-	9.46	5.75	5.23	5.06	2.70
Initial F_H	-	NA[1]	2.15	2.09	1.94	1.70
Maximum transient F_H	-	5.21	3.75	3.58	3.01	2.11
Maximum neutron power, F_{OP}	-	0.32	0.55	0.69	0.98	1.10
Maximum cal/g	< 150	-	70.4	50.4	63.9	109.4
Maximum cal/g, prompt	< 110	-	10.0	10.9	11.8	7.2
Maximum fuel temperature, °F	<rim melt	-	2655	1901	2529	4014
Maximum cladding temperature, °F	-	-	1098	727	951	1461
MDNBR/SAFDL normalized	< 1.0 For failure	-	0.71	1.86	0.96	0.33
Time of trip (initiation of safety bank insertion), s	-	No trip	No trip[2]	0.850	0.825	No trip
Equivalent nominal rods failed, %	< 30%[3]	0	1.7	0	0	7.2

Notes for Table 15.4-17:

1. Not applicable since initial stored energy above the coolant temperature is zero.
2. Trip is disabled to bound consequences of powers lower than 25%.
3. This is a sample value.

Table 15.4-18—Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Results for EOC

Parameter	Criterion	0	25	35	60	100
Maximum ejected rod worth, pcm	-	634	516	484	389	97
Delayed neutron fraction	-	0.0047	0.0047	0.0047	0.0047	0.0047
MTC, pcm/°F	-	-19.40	-23.44	-23.31	-26.68	-28.47
DTC, pcm/°F	-	-1.52	-1.41	-1.40	-1.35	-1.28
Initial F_Q	-	NA[1]	5.28	4.24	3.28	2.10
Maximum transient F_Q	-	20.10	13.32	10.91	7.38	3.30
Initial F_H	-	NA[1]	2.15	2.09	1.94	1.70
Maximum transient F_H	-	6.51	4.87	4.53	3.61	2.22
Maximum neutron power, F_{OP}	-	2.04	1.75	1.75	1.58	1.17
Maximum cal/g	≤150	33.9	62.5	64.6	73.1	103.4
Maximum cal/g, prompt	≤ 110	13.8	10.3	9.0	6.0	7.9
Maximum fuel temperature, °F	<rim melt	1140	2402	2534	2987	3856
Maximum cladding temperature, °F	-	741	789	774	1062	1337
MDNBR/SAFDL normalized	≤ 1.0 For failure	1.82	1.36	1.33	0.97	0.46
Time of trip (initiation of safety bank insertion)	-	1.000	0.850	0.850	0.825	No Trip
Equivalent nominal rods failed, %	≤ 30%[2]	0	0	0	0	1.9

Notes for Table 15.4-18:

1. Not applicable since initial stored energy above the coolant temperature is zero.
2. This is a sample value.

Table 15.4-19—Rod Ejection Accident Overpressurization Analysis – Sequence of Events

Event	Time (s)
Rod Ejection at HFP	
Ejection beginning	1.000
RT setpoint reached (high neutron flux ROC) (13%)	1.034
RT signal issued	1.334
Rod insertion beginning	1.634
Cold-leg pressure peak	4.000
PZR pressure peak	5.000
Rod Ejection at 60%NP	
Ejection beginning	1.000
RT setpoint reached (high neutron flux ROC) (13%)	1.031
RT signal issued	1.331
Rod insertion beginning	1.631
Cold-leg pressure peak	4.000
Rod Ejection at HZP	
Ejection beginning	1.000
RT setpoint reached (high neutron flux ROC) (13%)	4.110
RT signal issued	4.410
Rod insertion beginning	4.710
Cold-leg pressure peak	7.500
PZR pressure peak	9.500

Figure 15.4-1—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Reactor Power

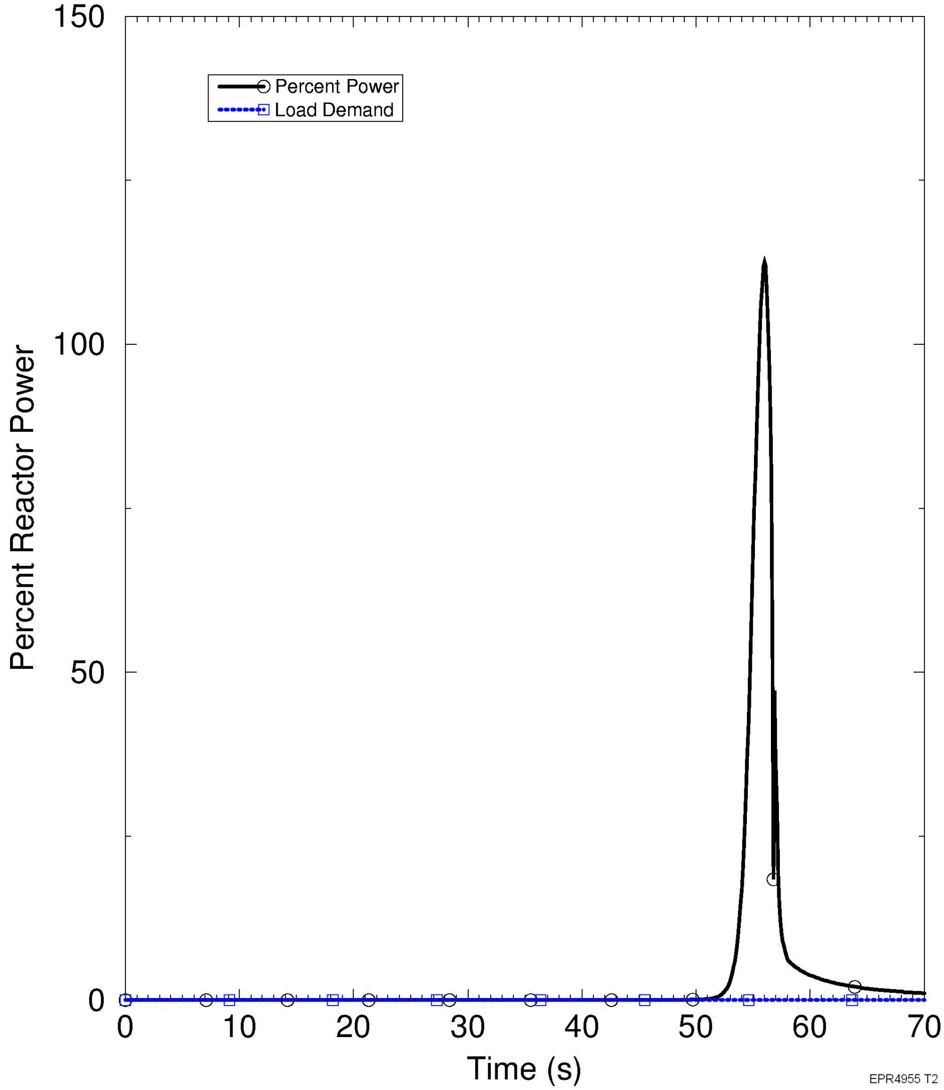

Figure 15.4-2—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Reactivity

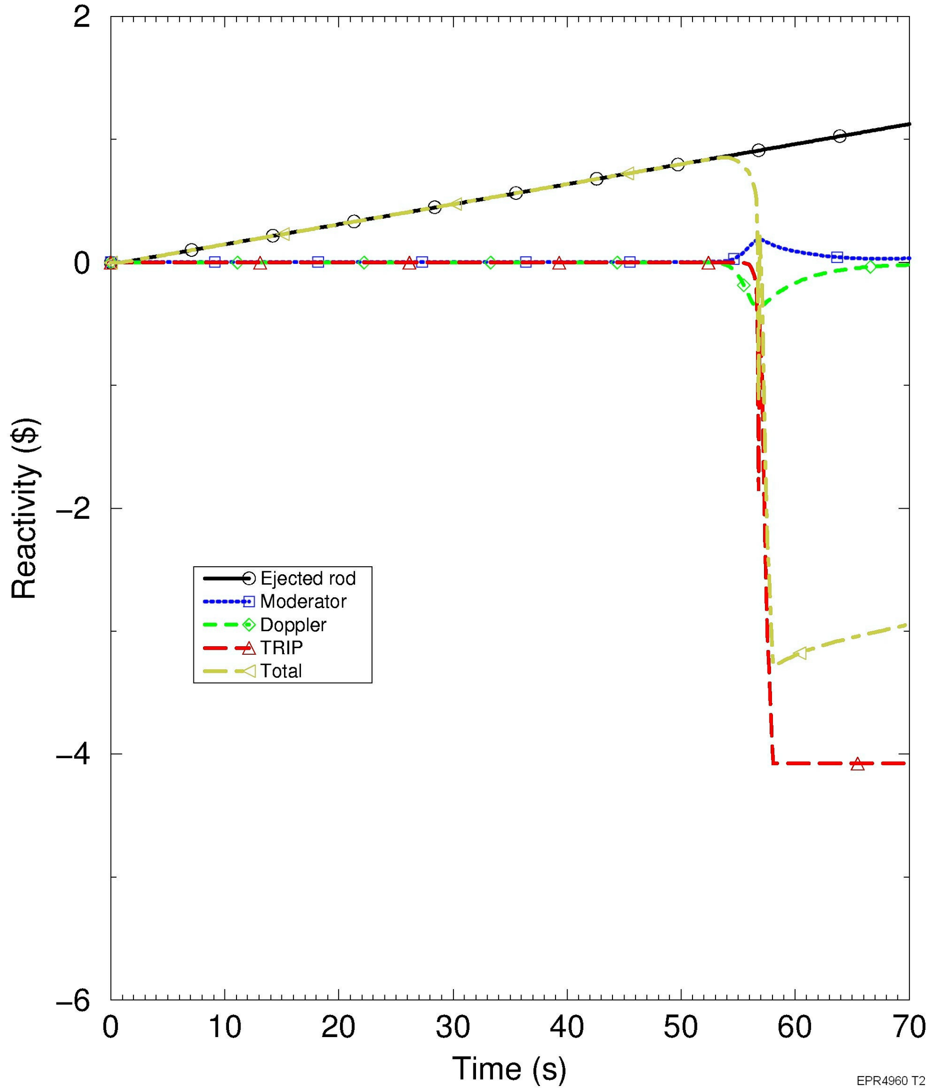

Figure 15.4-3—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary System Pressure

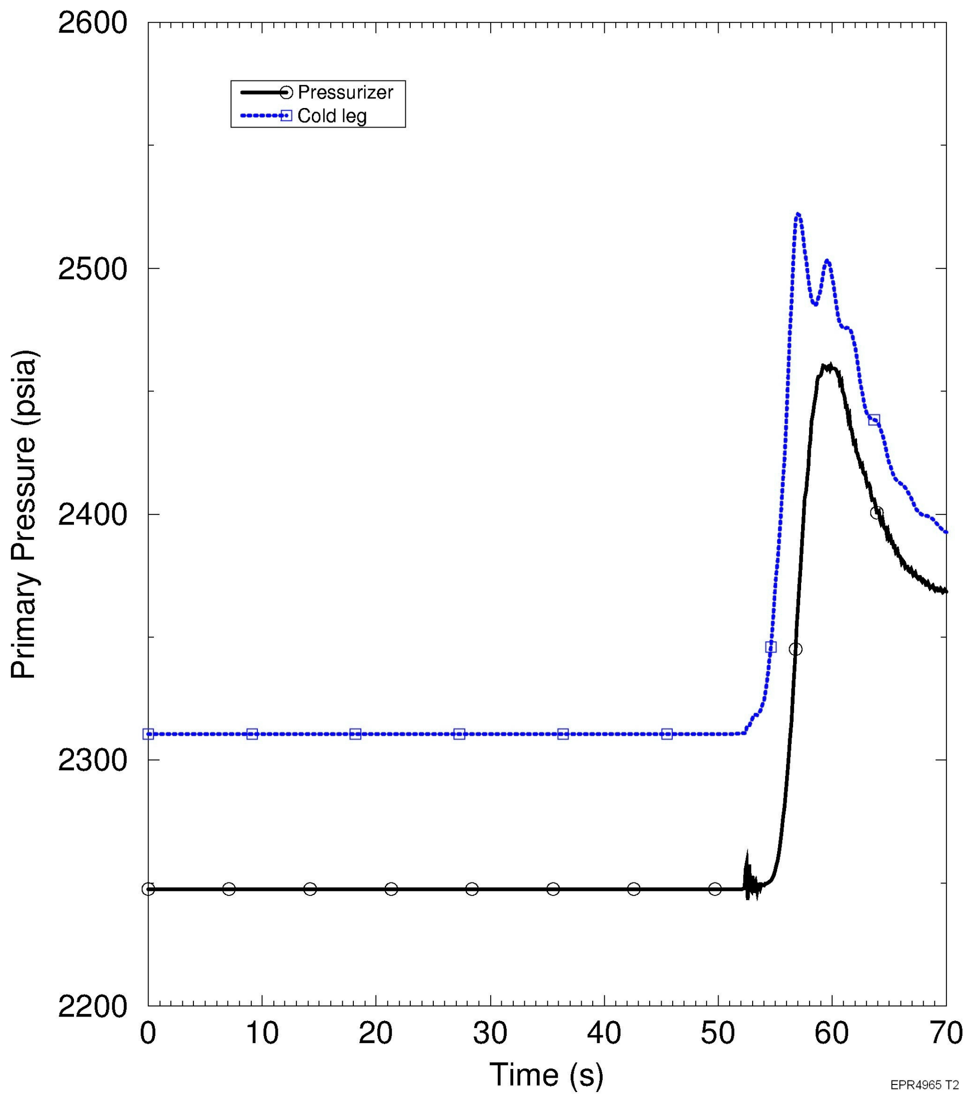

Figure 15.4-4—Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Startup Condition - Primary System Temperature

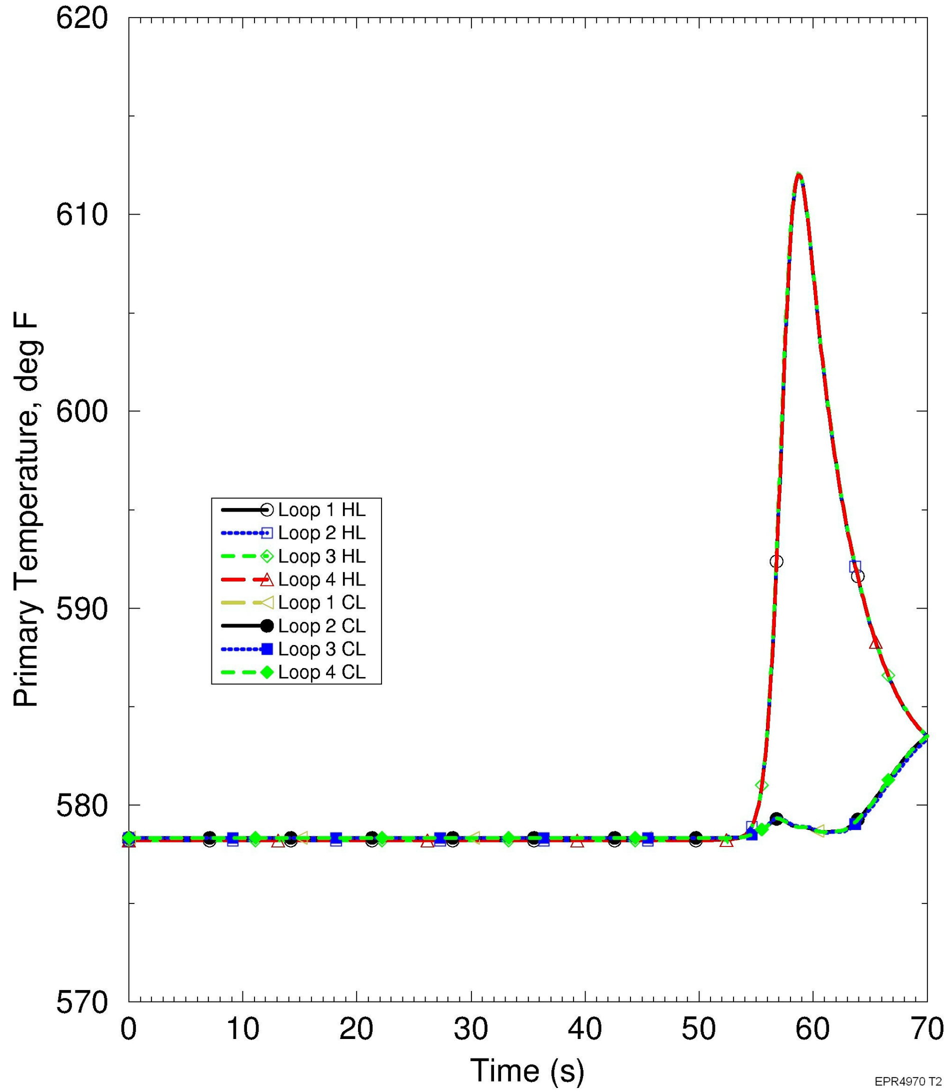

Figure 15.4-5—Uncontrolled Control Bank Withdrawal from a Subcitical or Low Power Startup Condition - Cold Leg Mass Flow

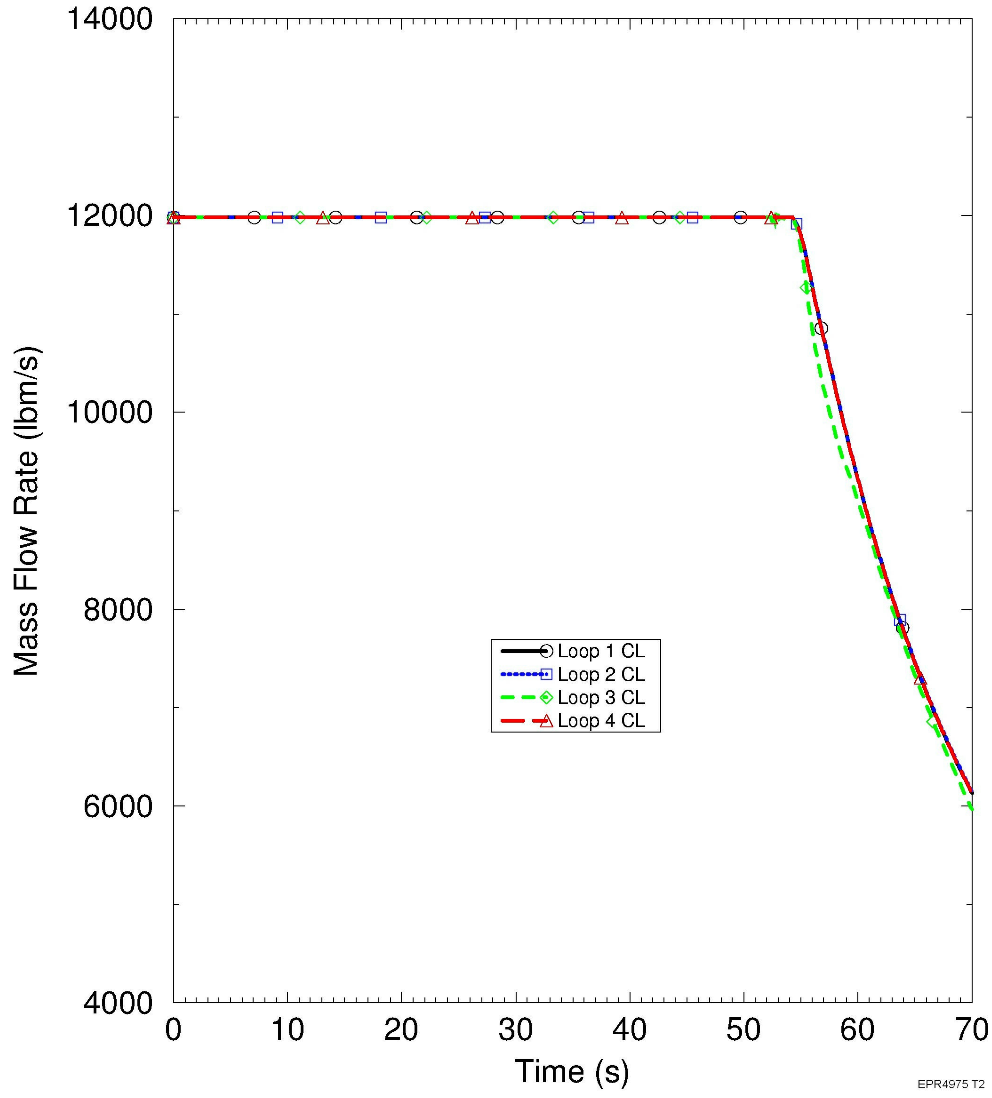

Figure 15.4-6—Uncontrolled Control Bank Withdrawal at Power - Reactor Power

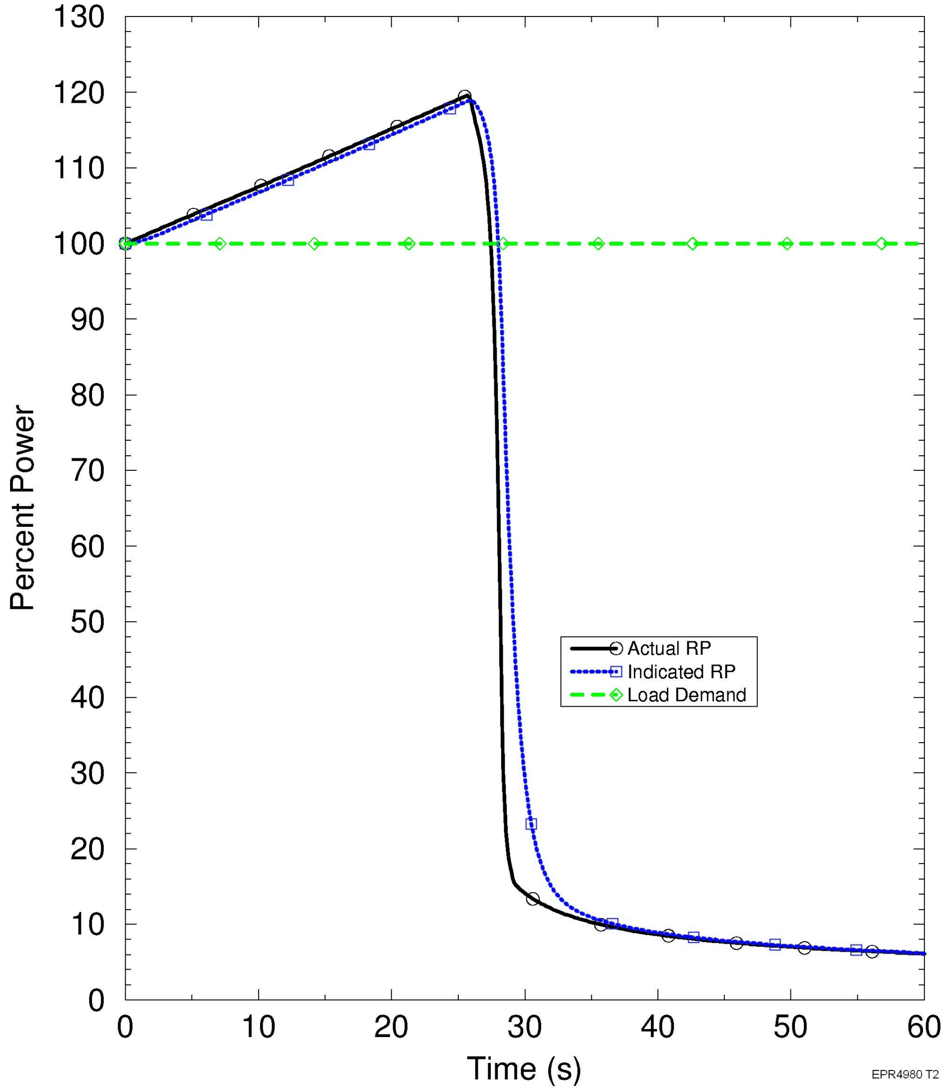

Figure 15.4-7—Uncontrolled Control Bank Withdrawal at Power - Reactivity

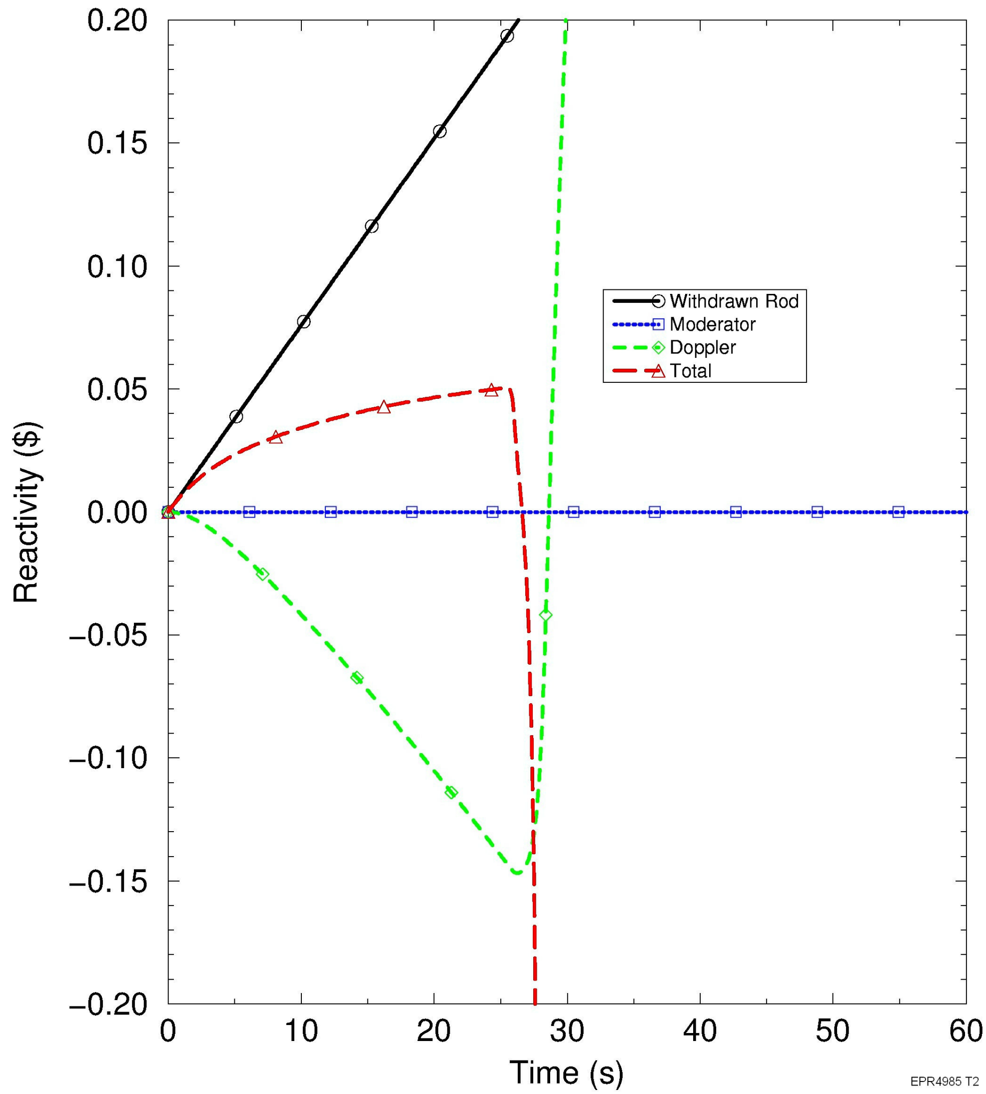

Figure 15.4-8—Uncontrolled Control Bank Withdrawal at Power - Average Core Heat Flux

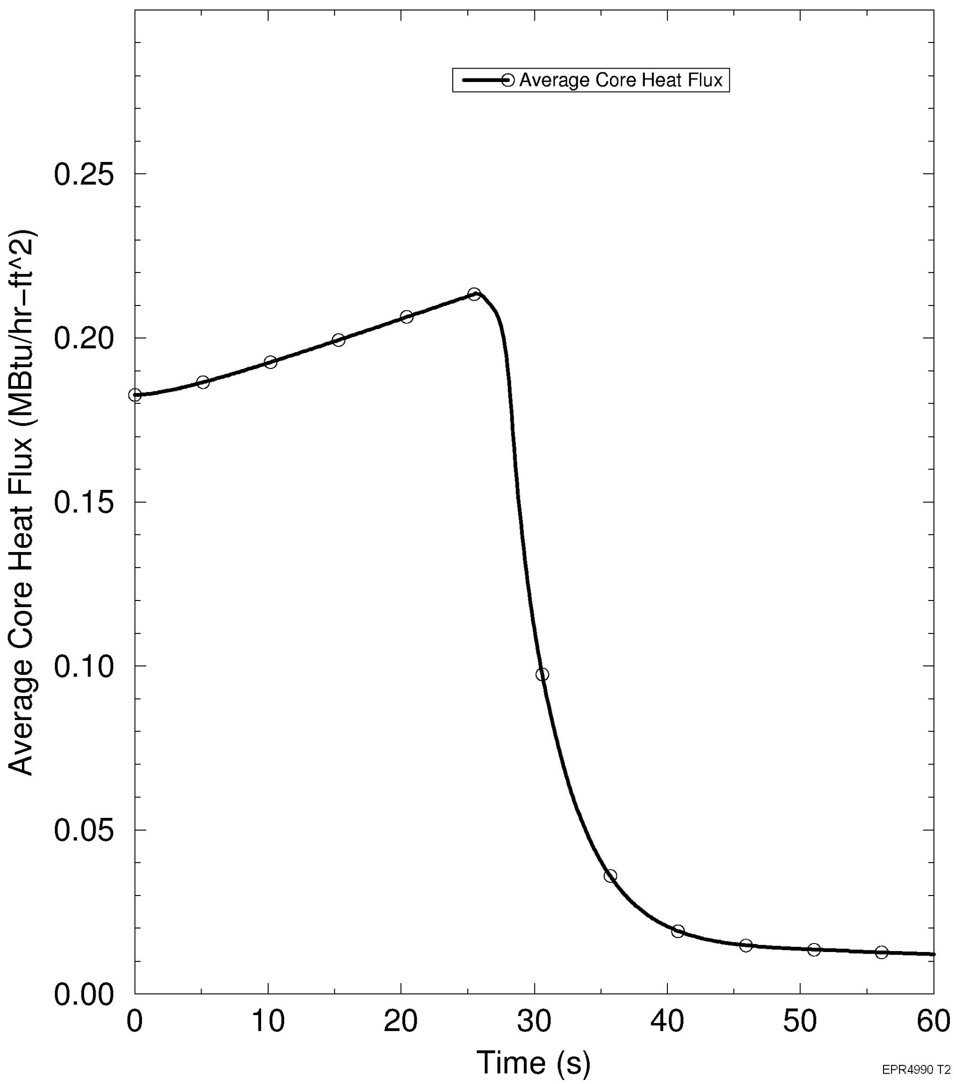

Figure 15.4-9—Uncontrolled Control Bank Withdrawal at Power - Primary Pressure

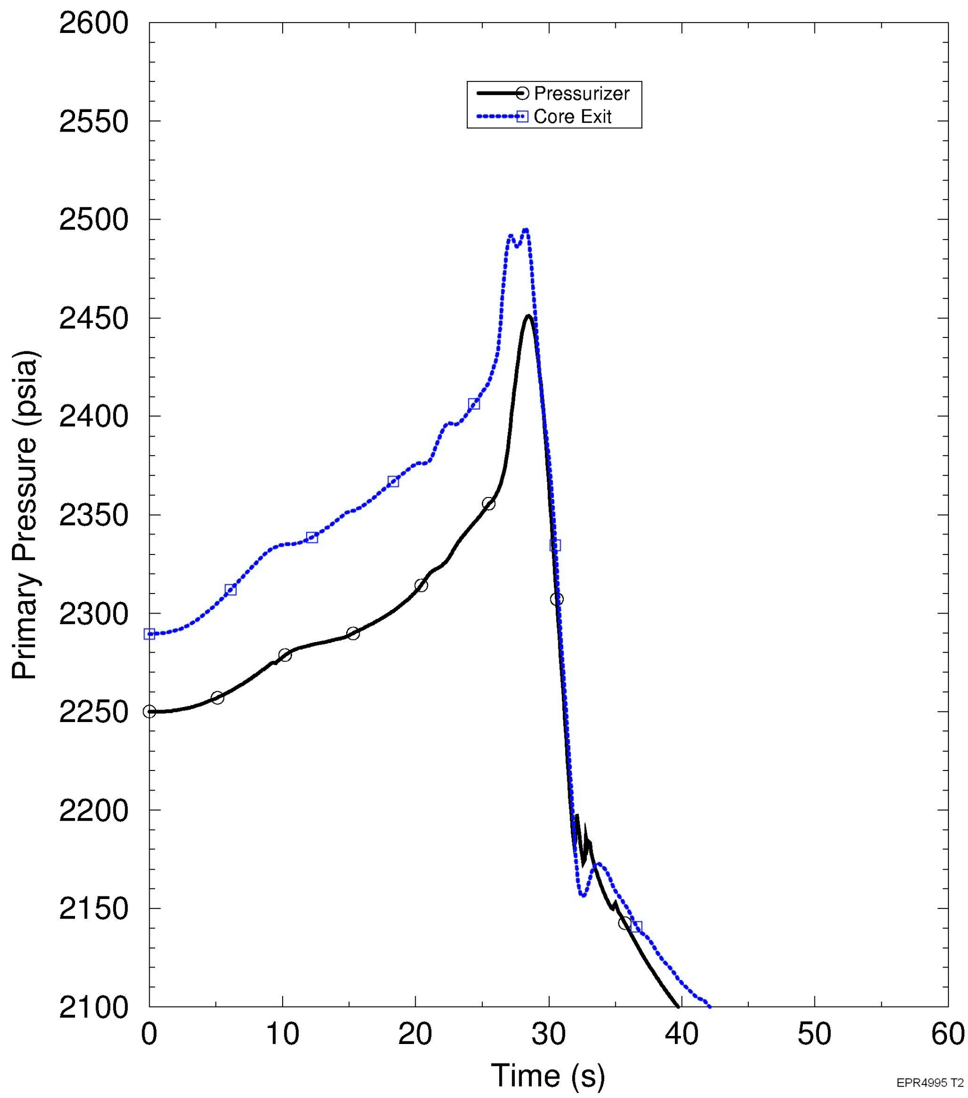

Figure 15.4-10—Uncontrolled Control Bank Withdrawal at Power - Cold Leg Flow

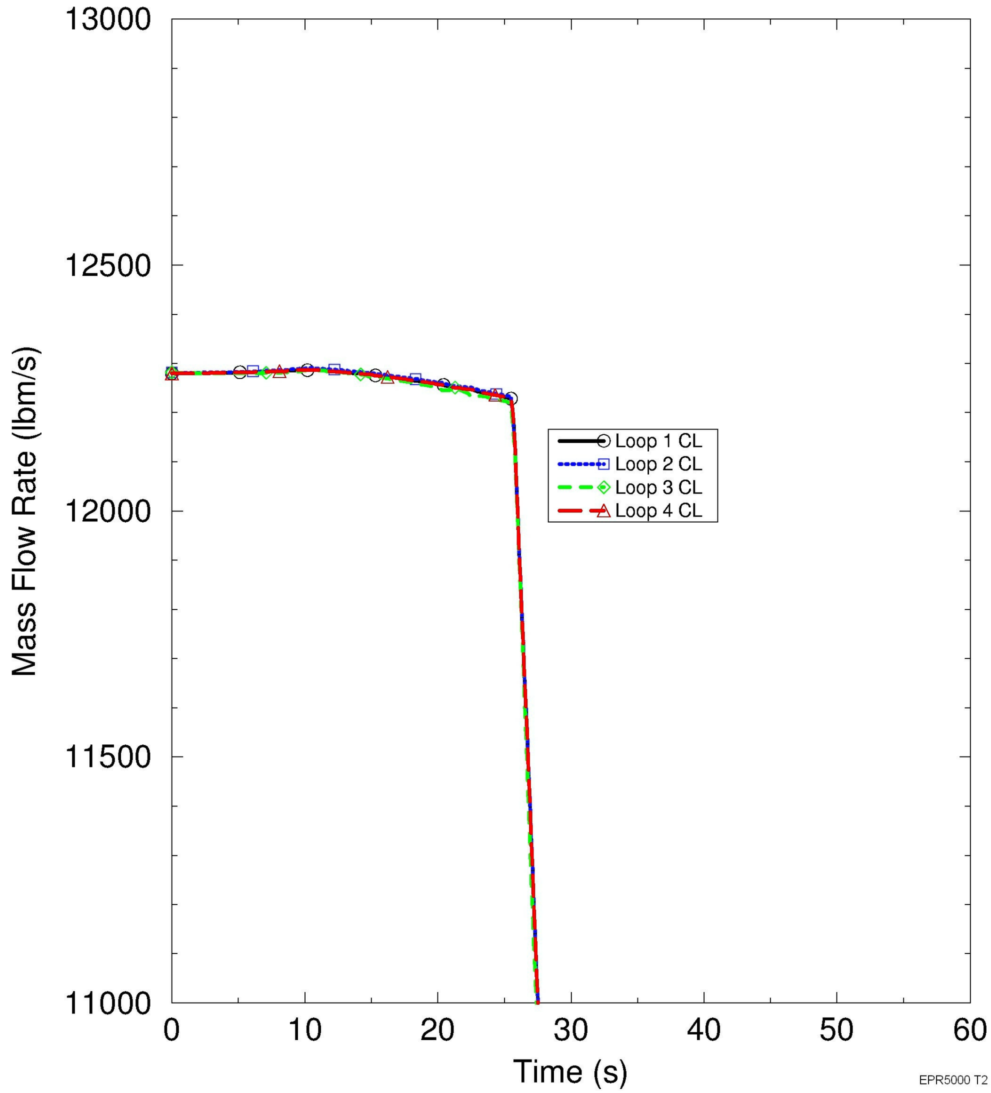

Figure 15.4-11—Uncontrolled Control Bank Withdrawal at Power - Primary System Temperature

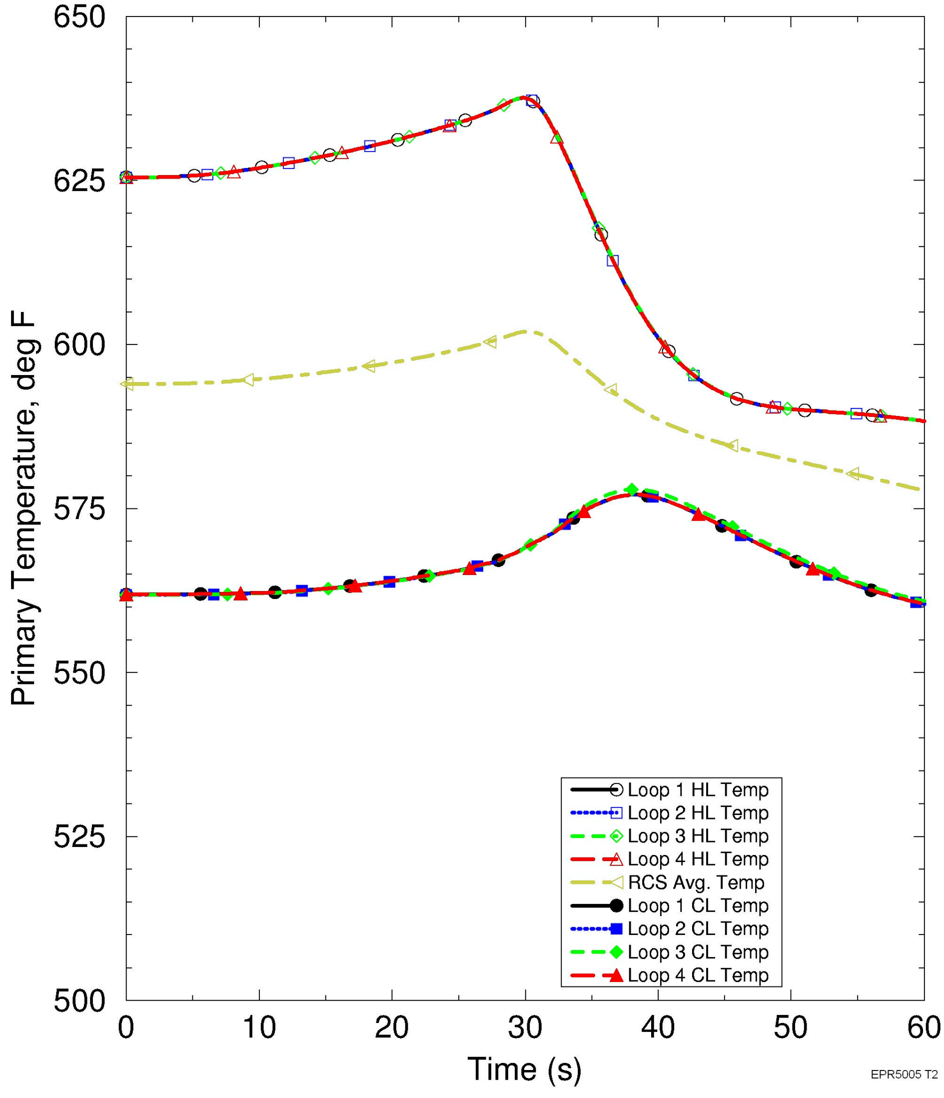

Figure 15.4-12—Uncontrolled Control Bank Withdrawal at Power - Pressurizer Level

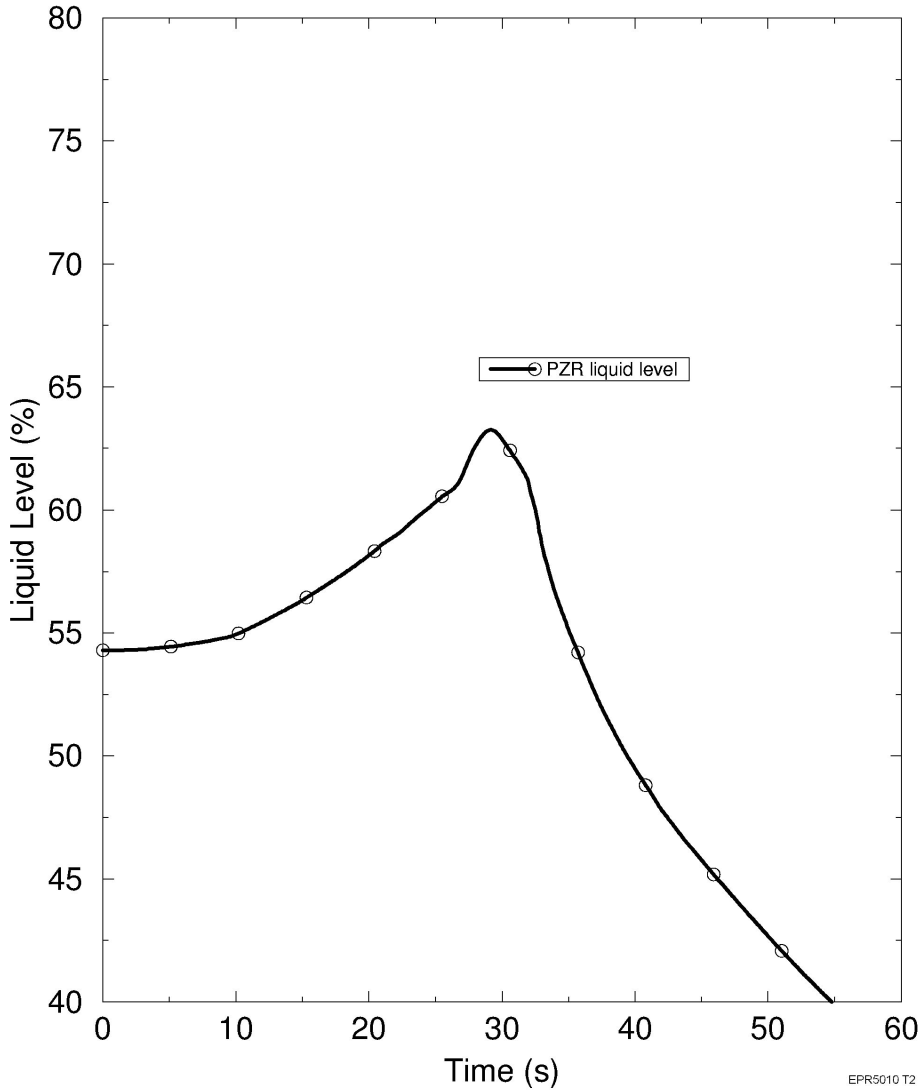

Figure 15.4-13—Uncontrolled Control Bank Withdrawal at Power - SG Secondary Pressure

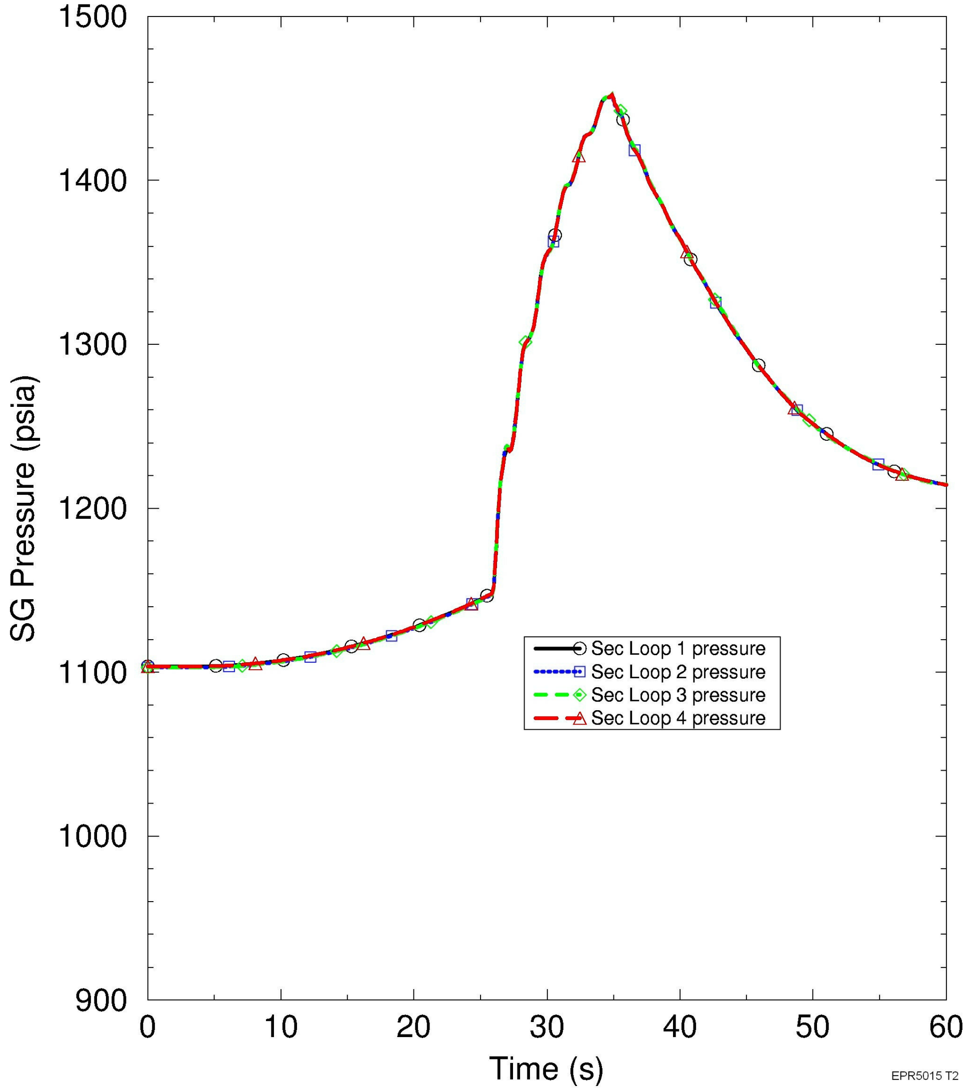

Figure 15.4-14—Uncontrolled Control Bank Withdrawal at Power - Pressurizer Spray

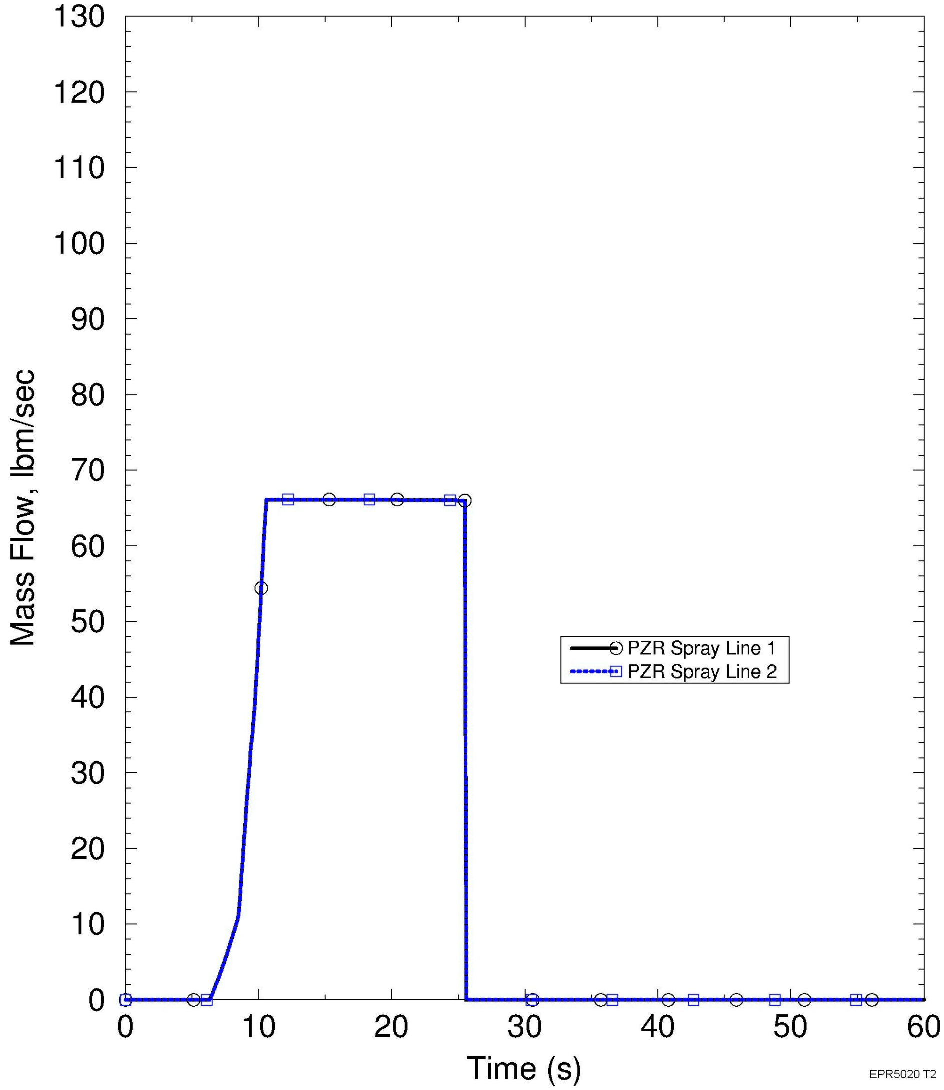

Figure 15.4-15—Dropped RCCA - Reactor Power

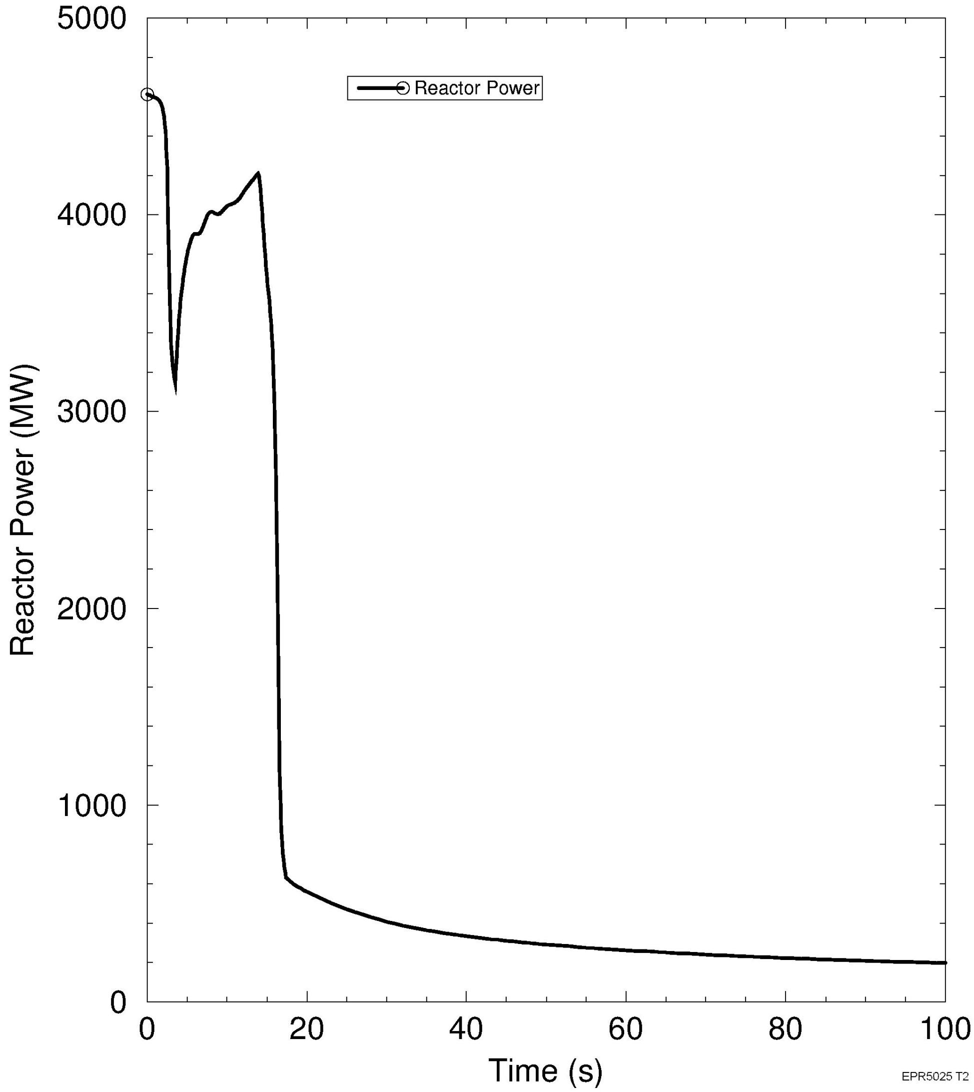

Figure 15.4-16—Dropped RCCA - Reactivity

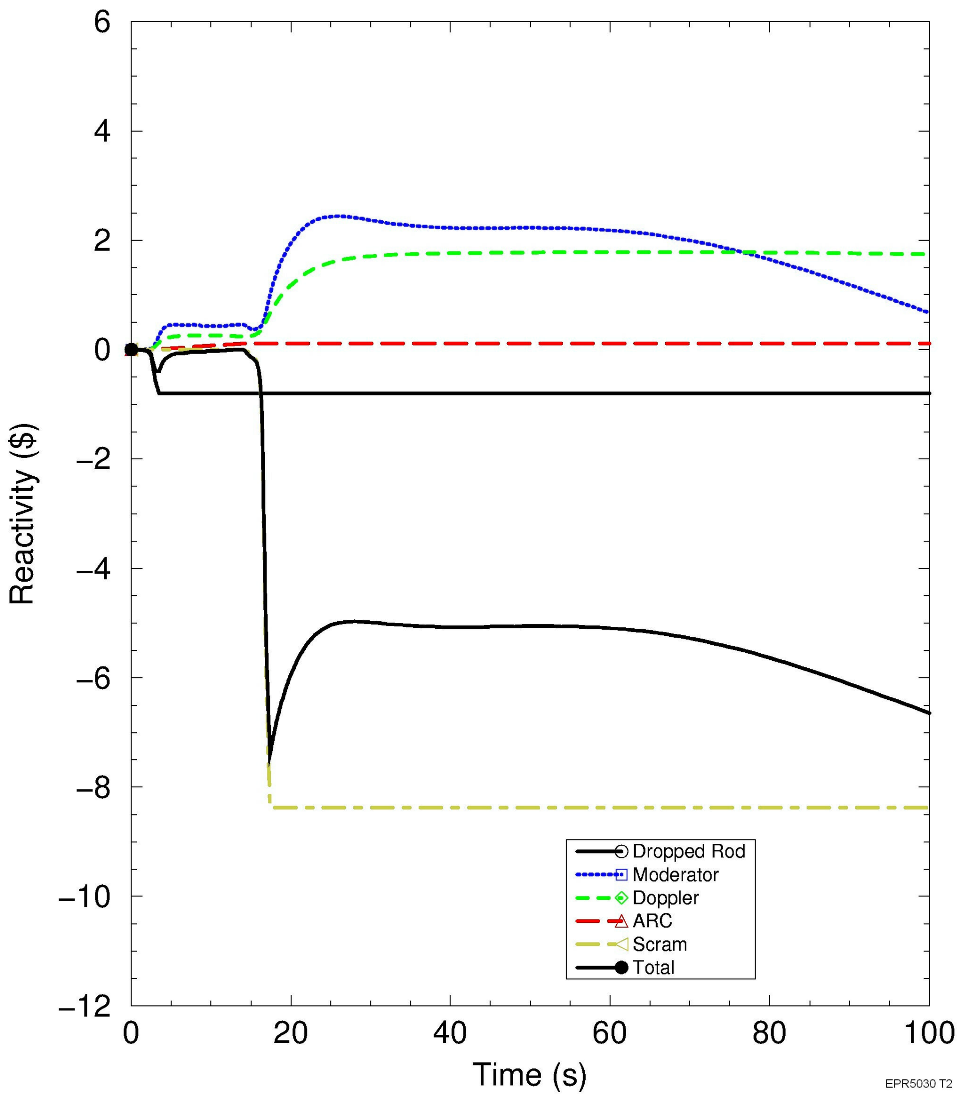

Figure 15.4-17—Dropped RCCA - Average Core Heat Flux

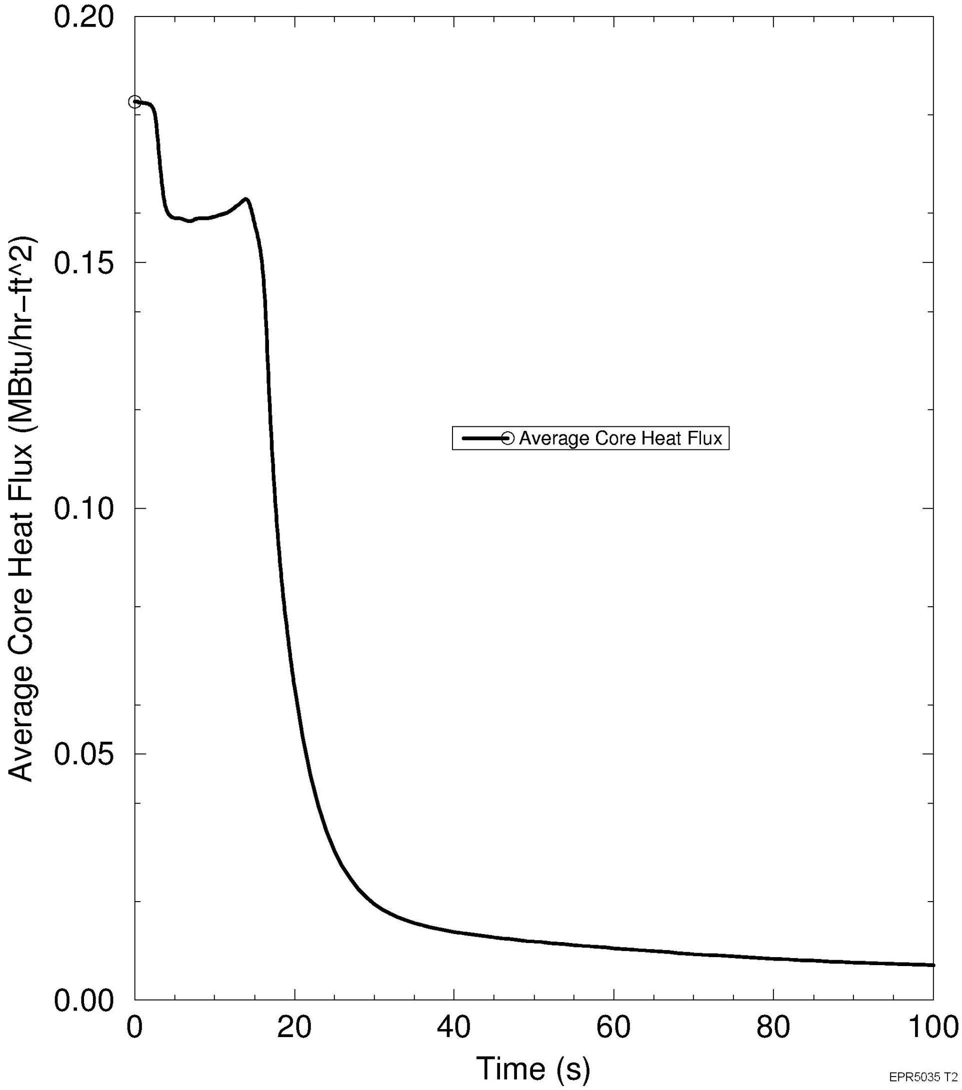

Figure 15.4-18—Dropped RCCA - Primary System Pressure

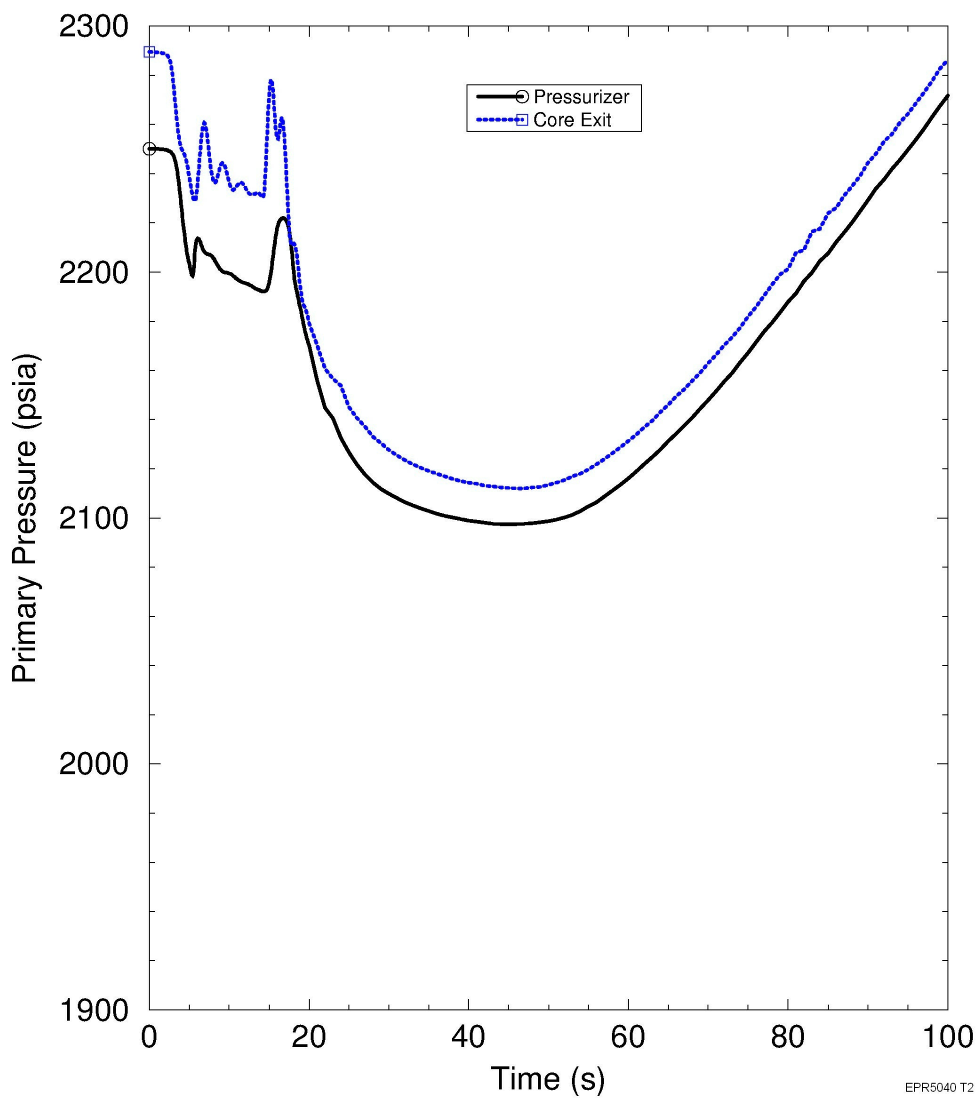

Figure 15.4-19—Dropped RCCA - Primary System Temperature

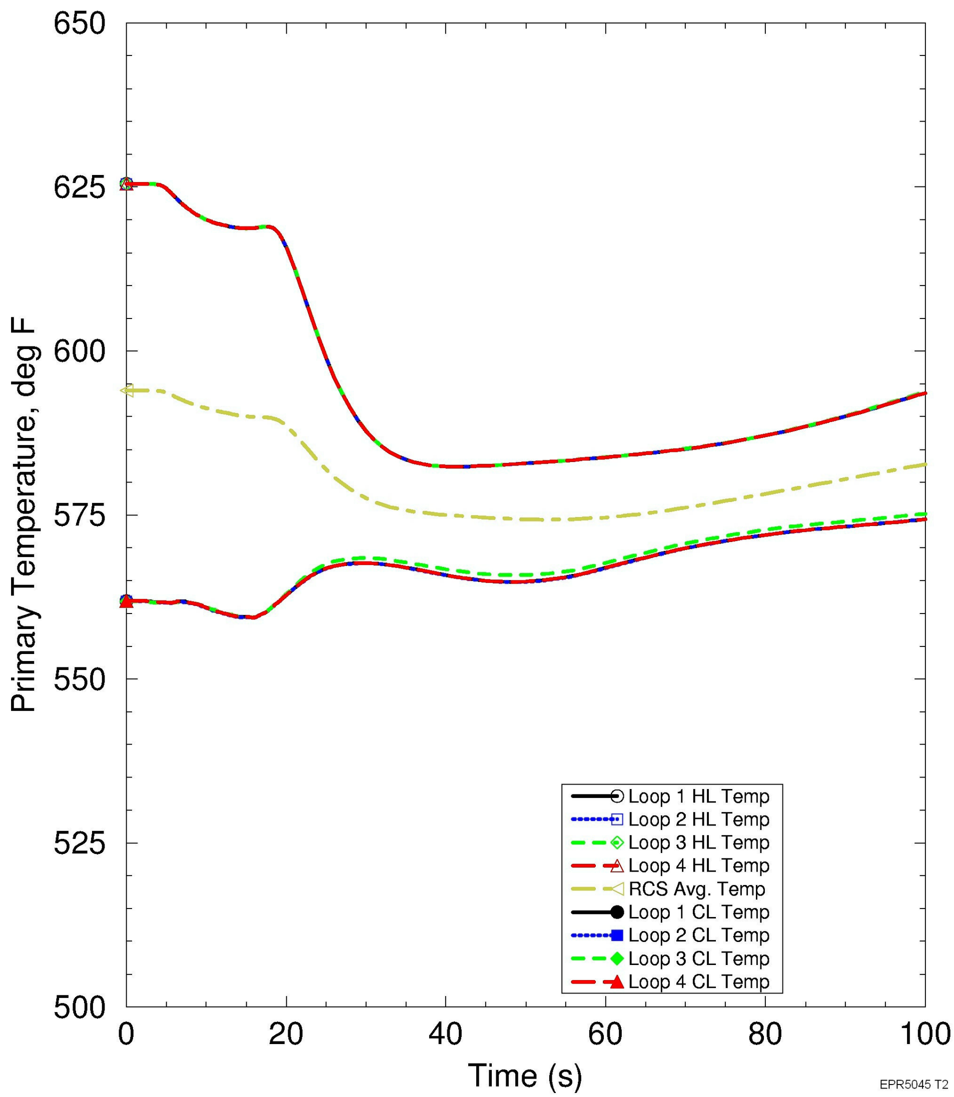

Figure 15.4-20—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature - Total RCS Loop Flow

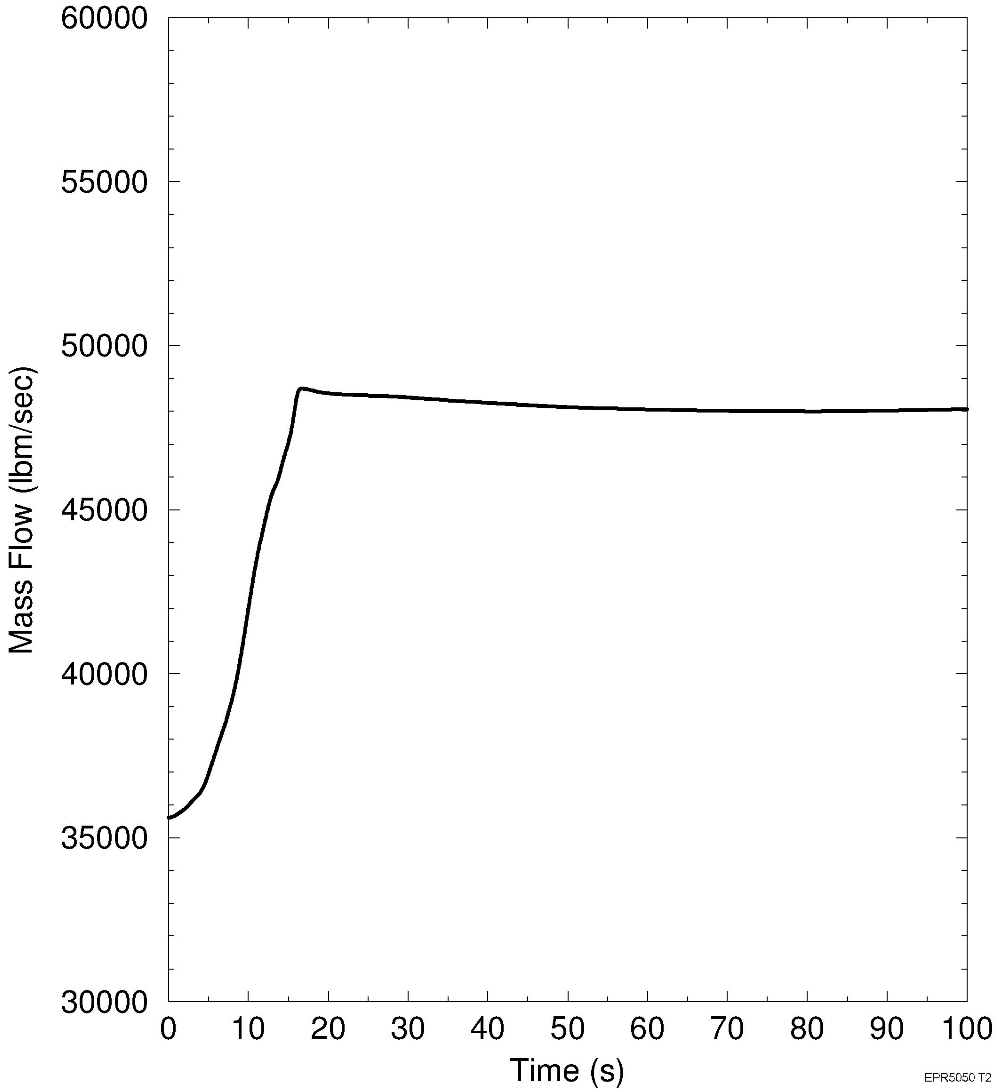

Figure 15.4-21—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature - RCS Loop Flows

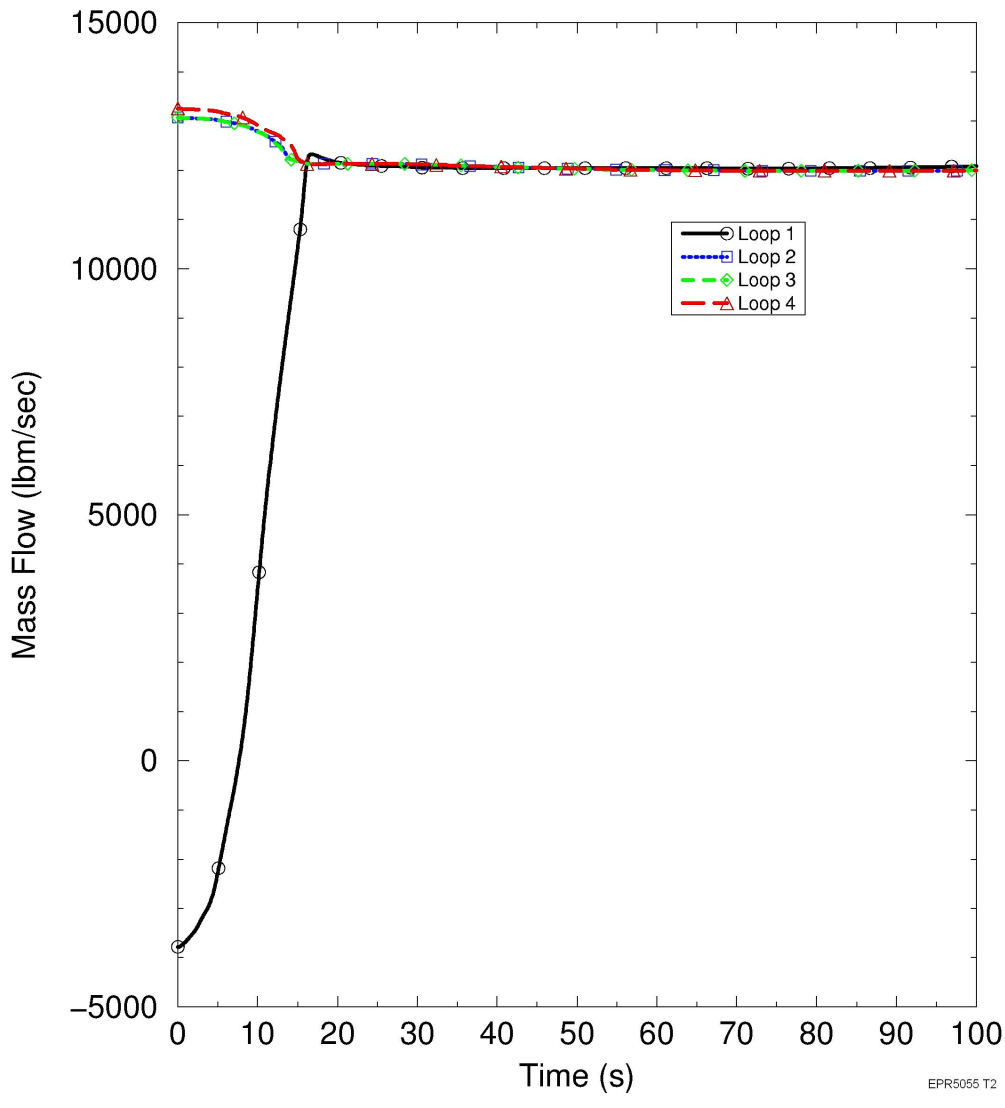

Figure 15.4-22—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – RCP Speeds

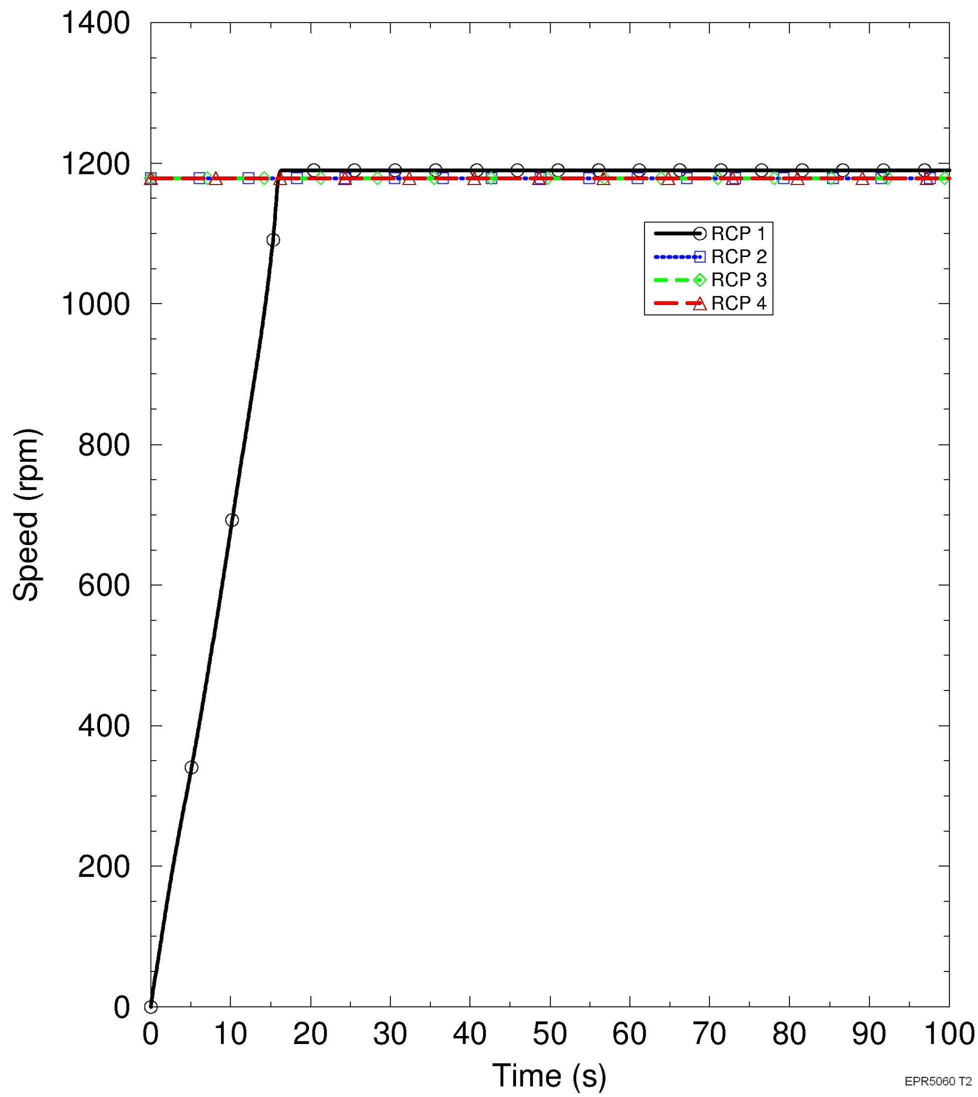

Figure 15.4-23—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Core Temperatures

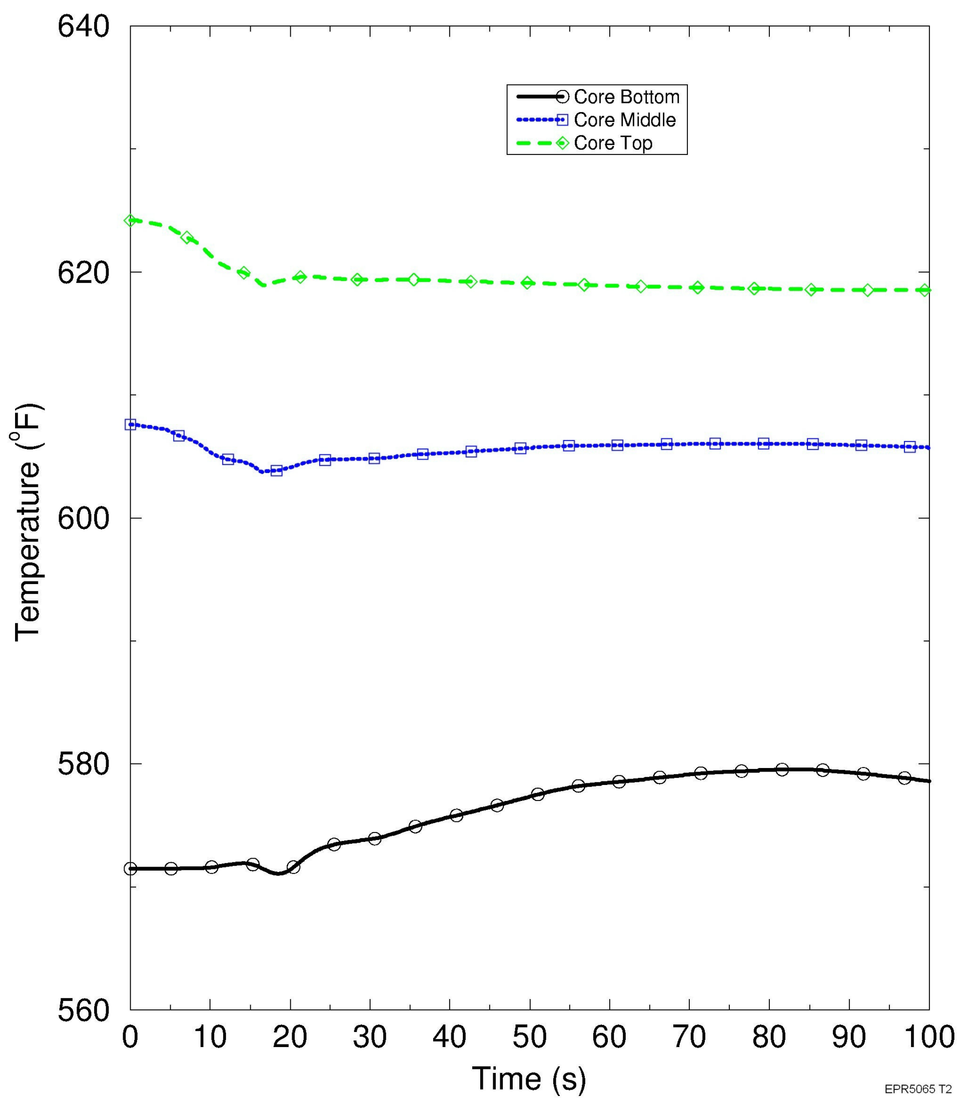

Figure 15.4-24—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Indicated Reactor Power

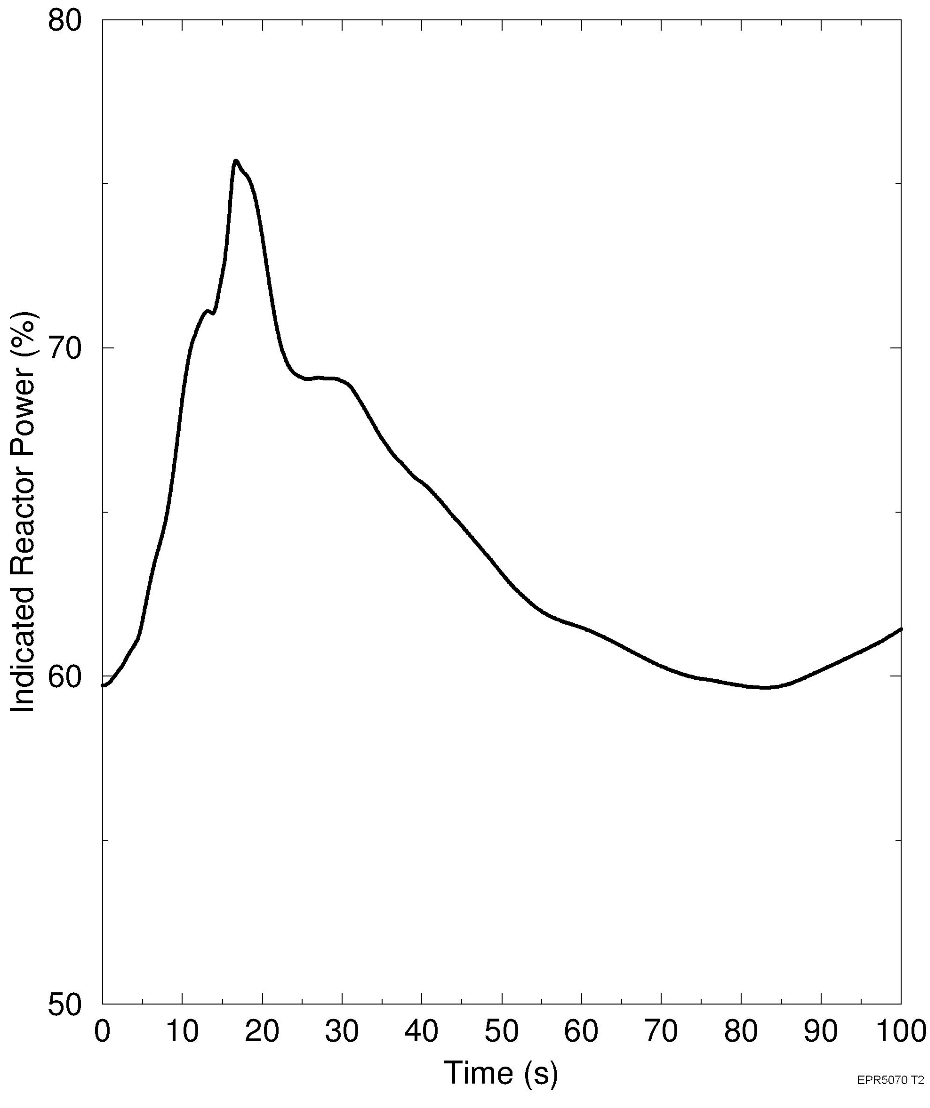

Figure 15.4-25—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Reactivity

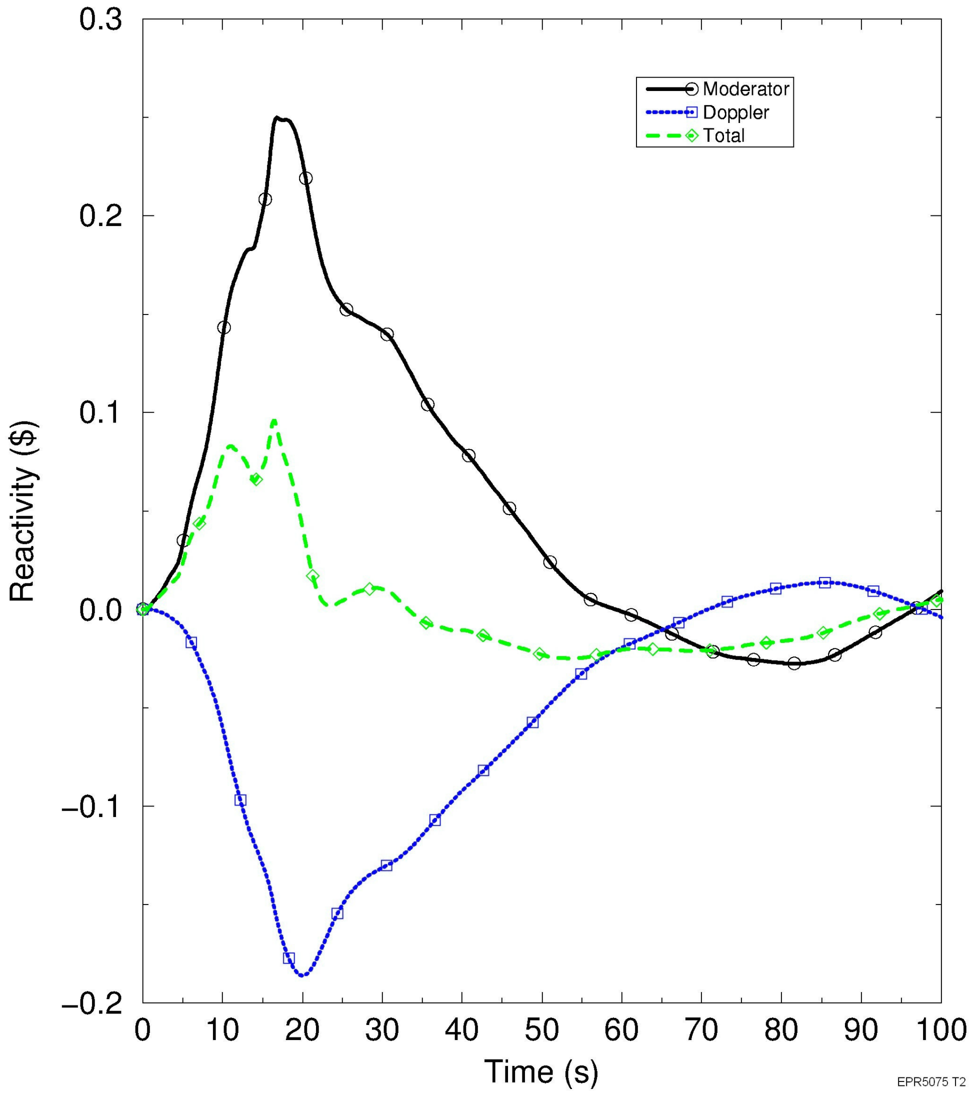

Figure 15.4-26—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – Cold Leg Temperatures

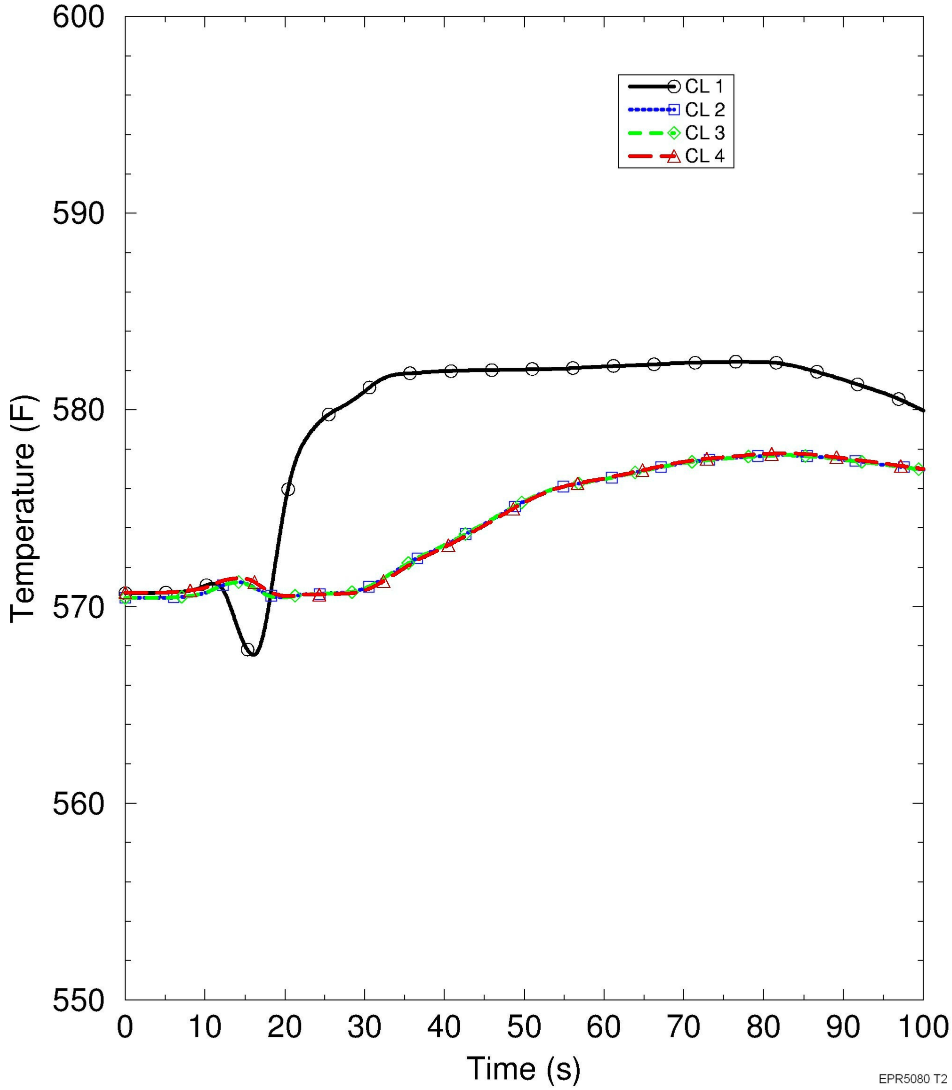

Figure 15.4-27—Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature – RCS Bottom Pressure

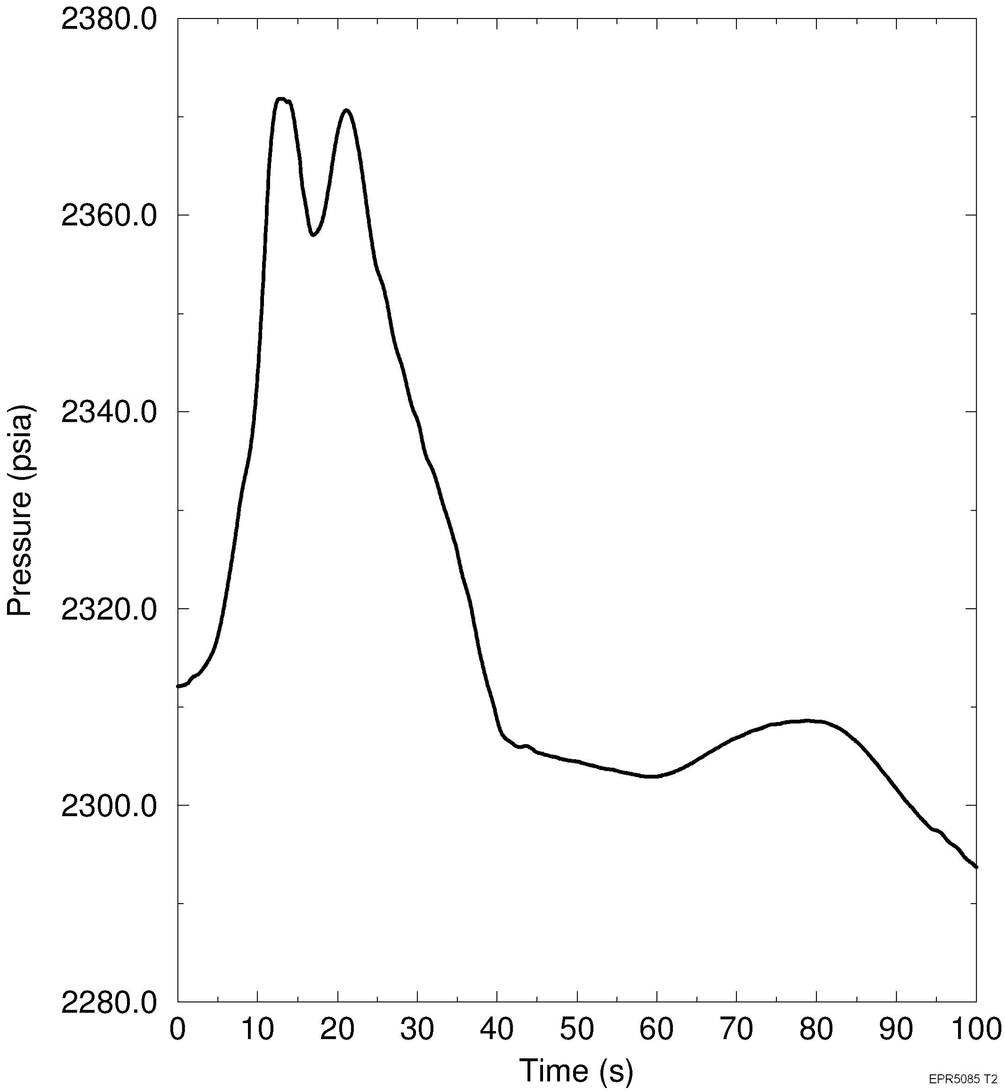

Figure 15.4-28—Rod Ejection Accident DNBR Analysis – BOC HFP Transient Power Fraction

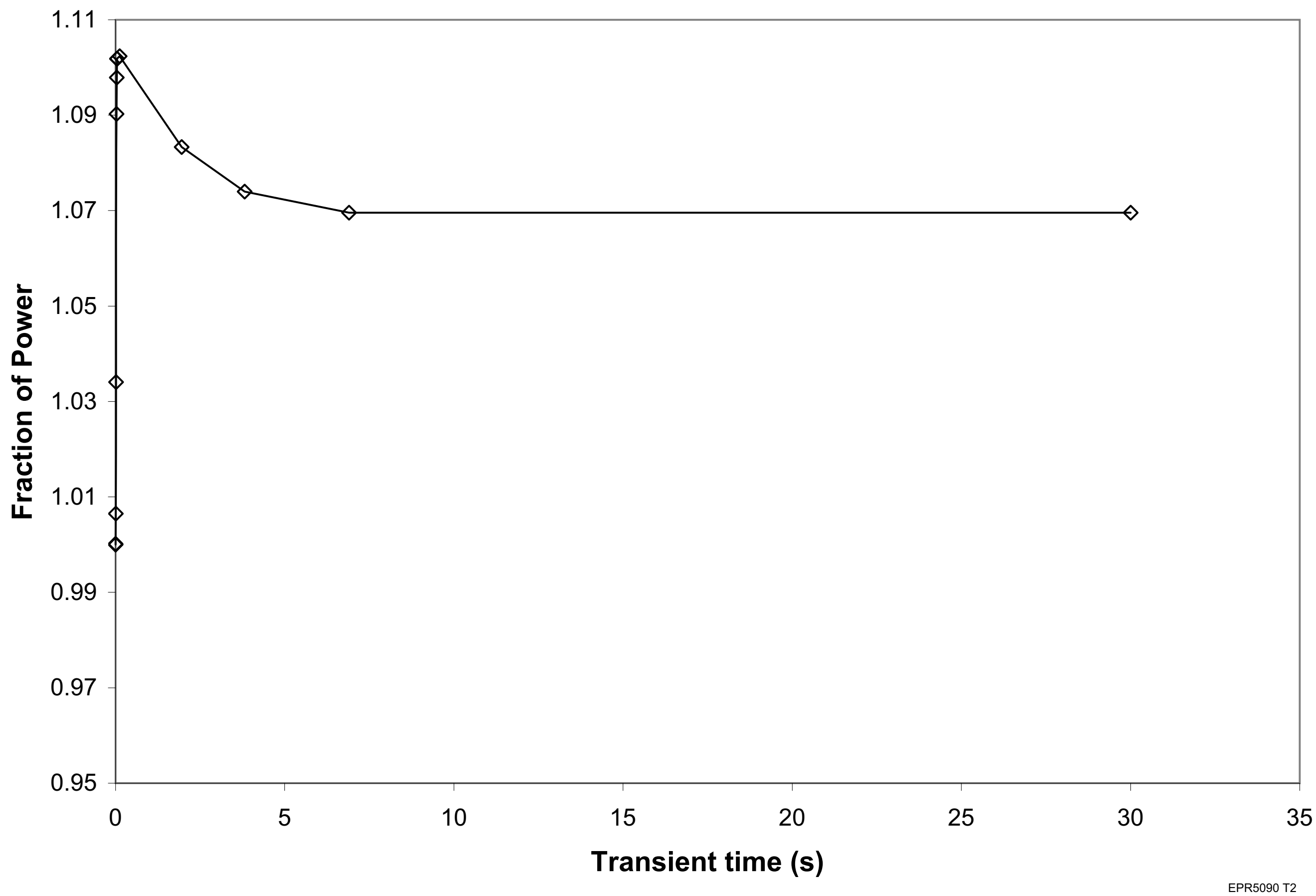

Figure 15.4-29—Rod Ejection Accident DNBR Analysis – BOC HFP Transient Peak Fuel and Cladding Temperatures

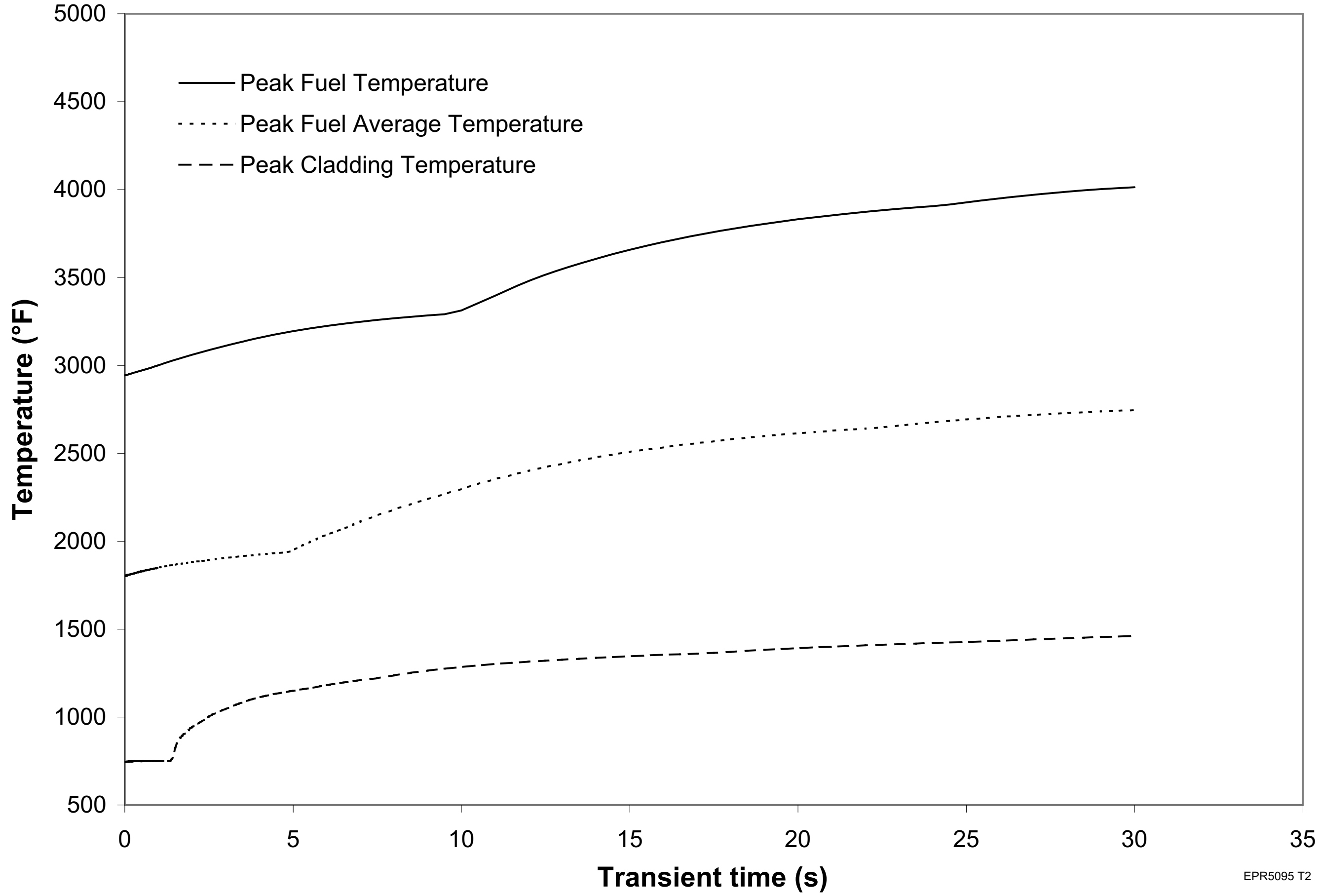

Figure 15.4-30—Rod Ejection Accident DNBR Analysis – EOC HZP Transient Power Fraction

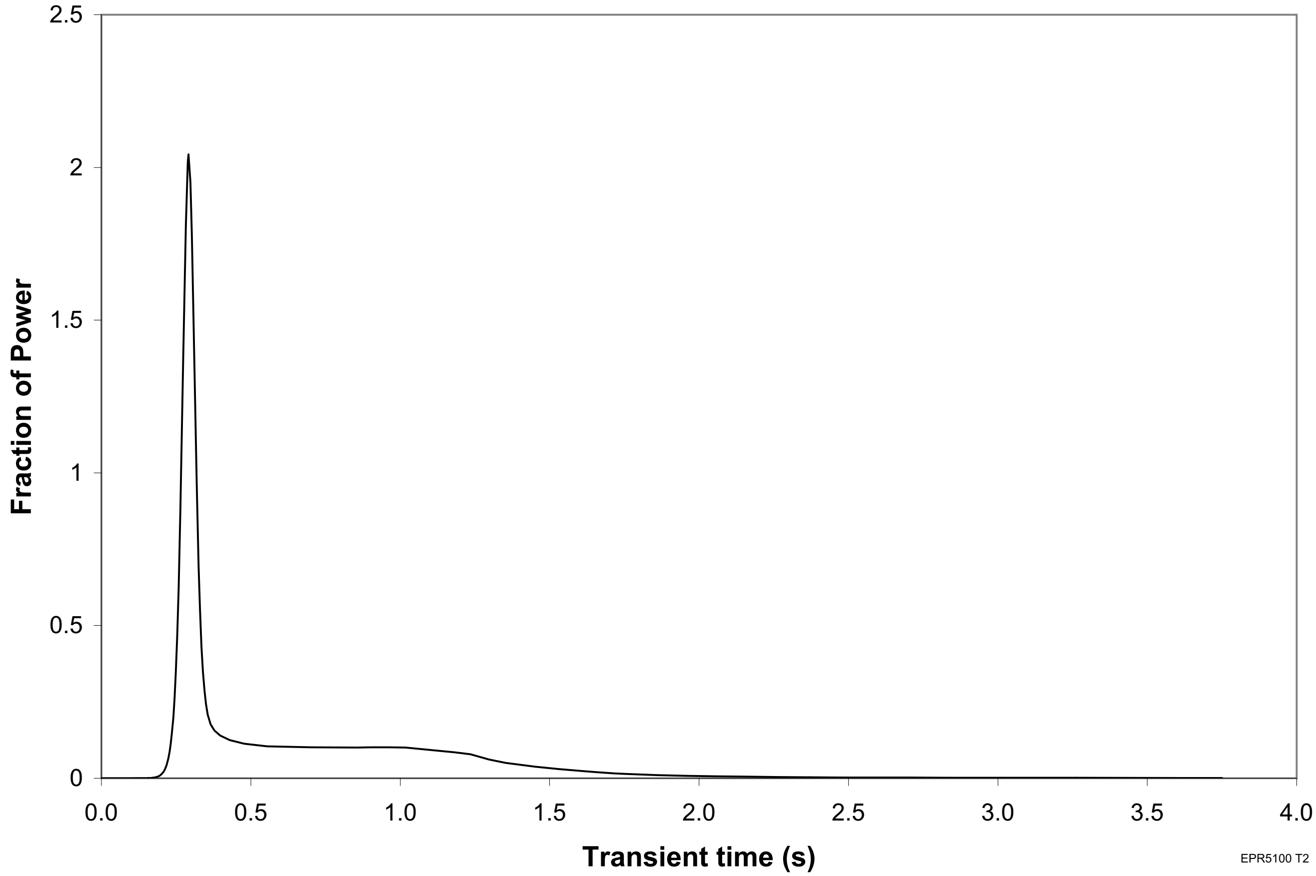

Figure 15.4-31—Rod Ejection Accident DNBR Analysis – EOC HZP Transient Peak Fuel and Cladding Temperatures

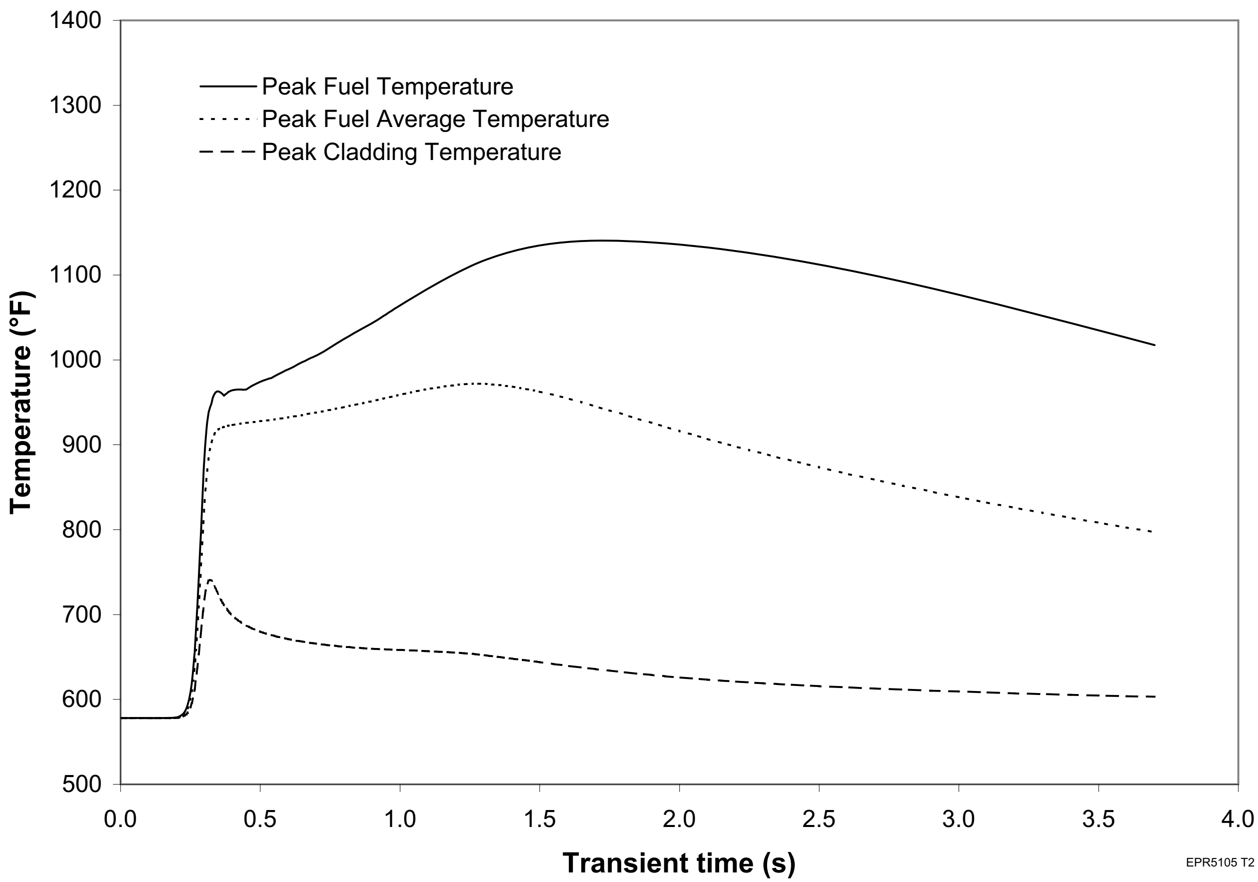

Figure 15.4-32—HFP Rod Ejection Accident Overpressurization Analysis – Percent Reactor Power

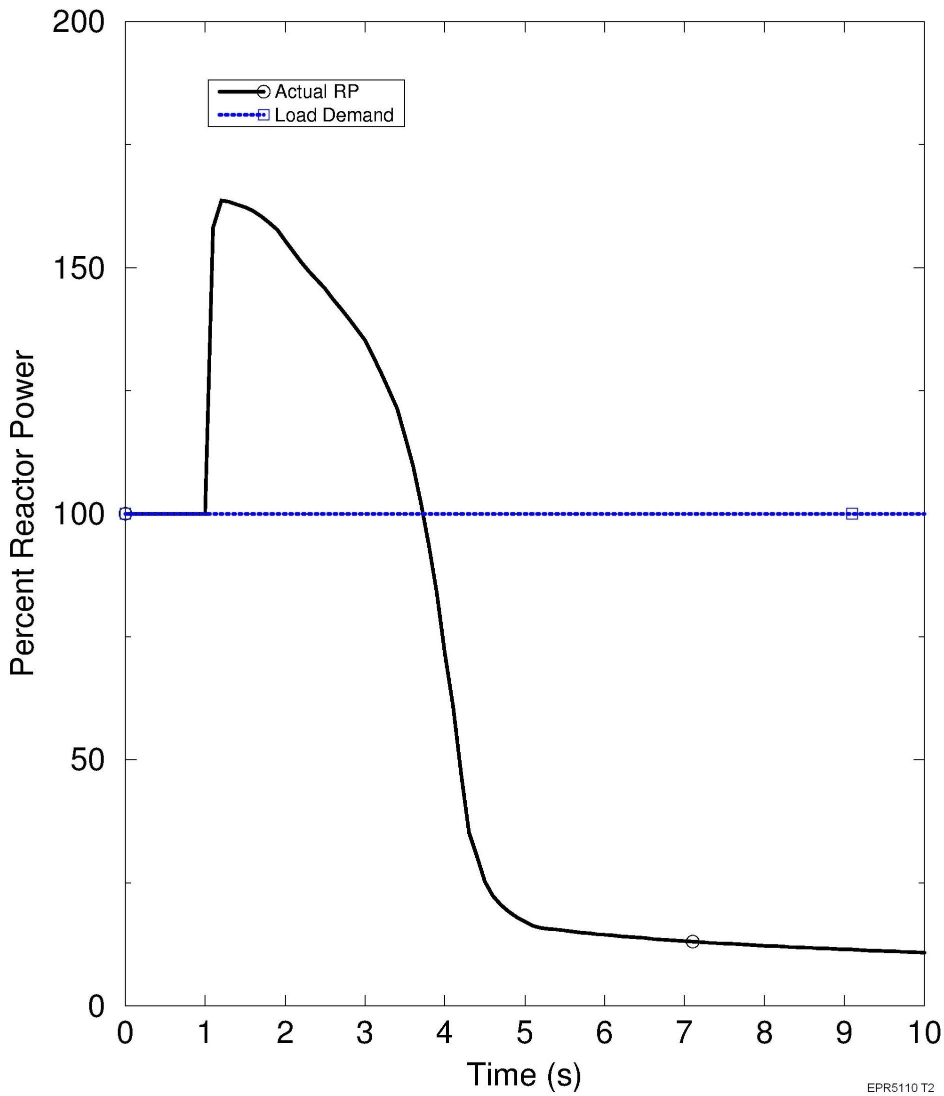

Figure 15.4-33—HFP Rod Ejection Accident Overpressurization Analysis – Reactivity

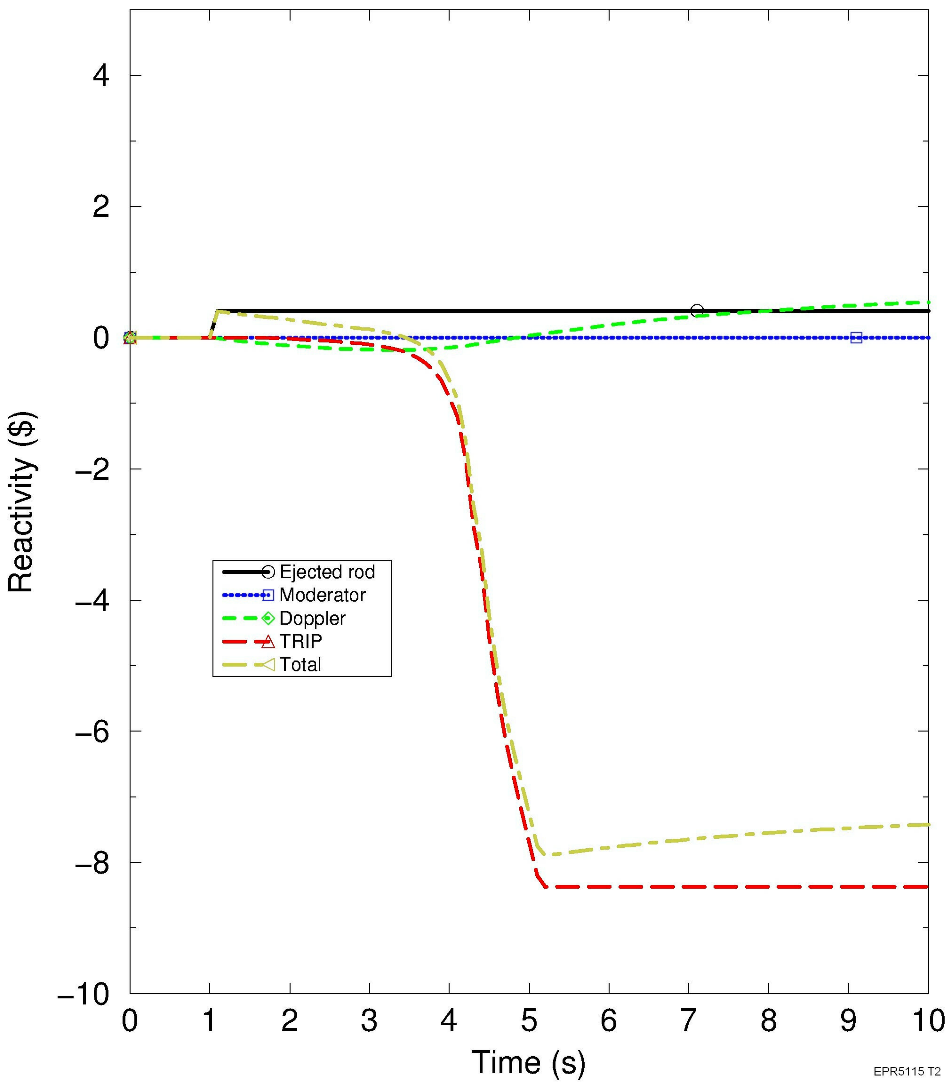

Figure 15.4-34—HFP Rod Ejection Accident Overpressurization Analysis – Primary System Pressure

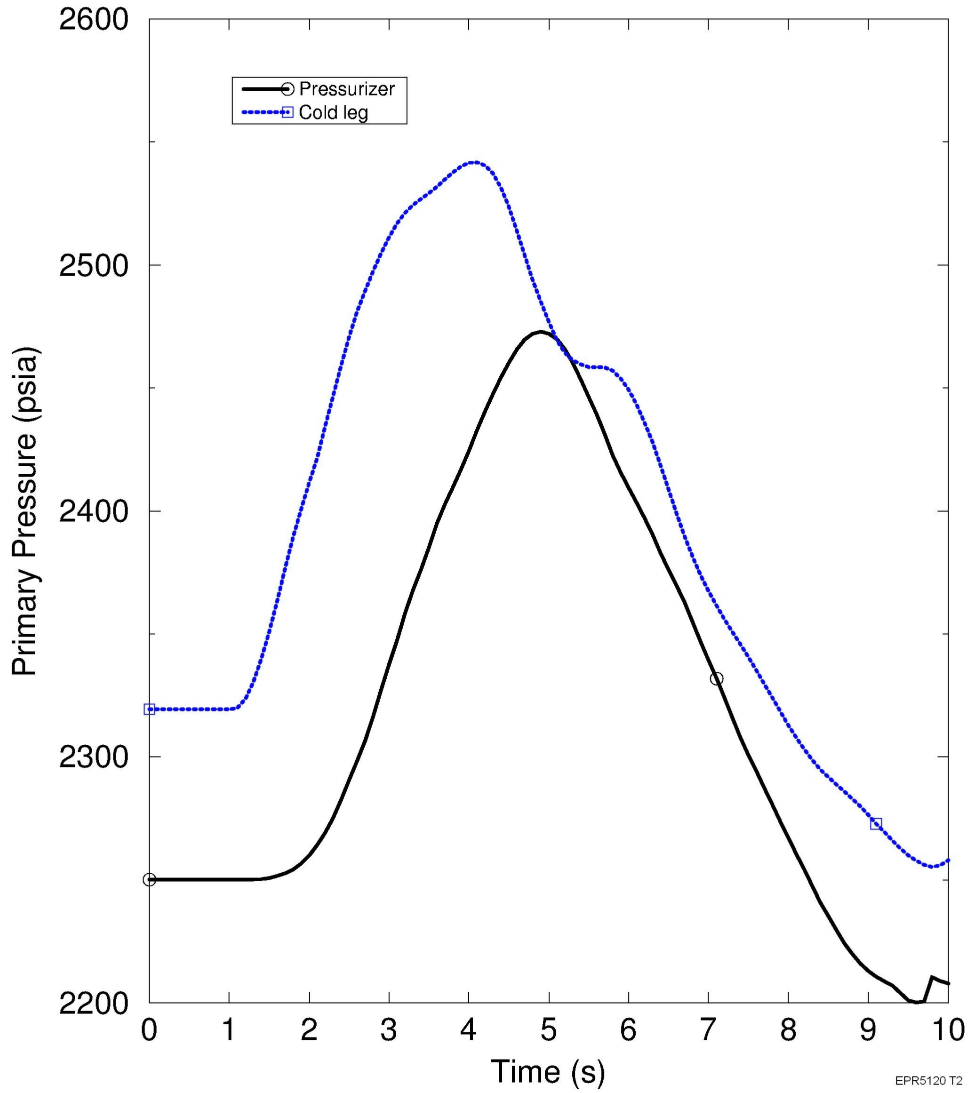

Figure 15.4-35—HFP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature

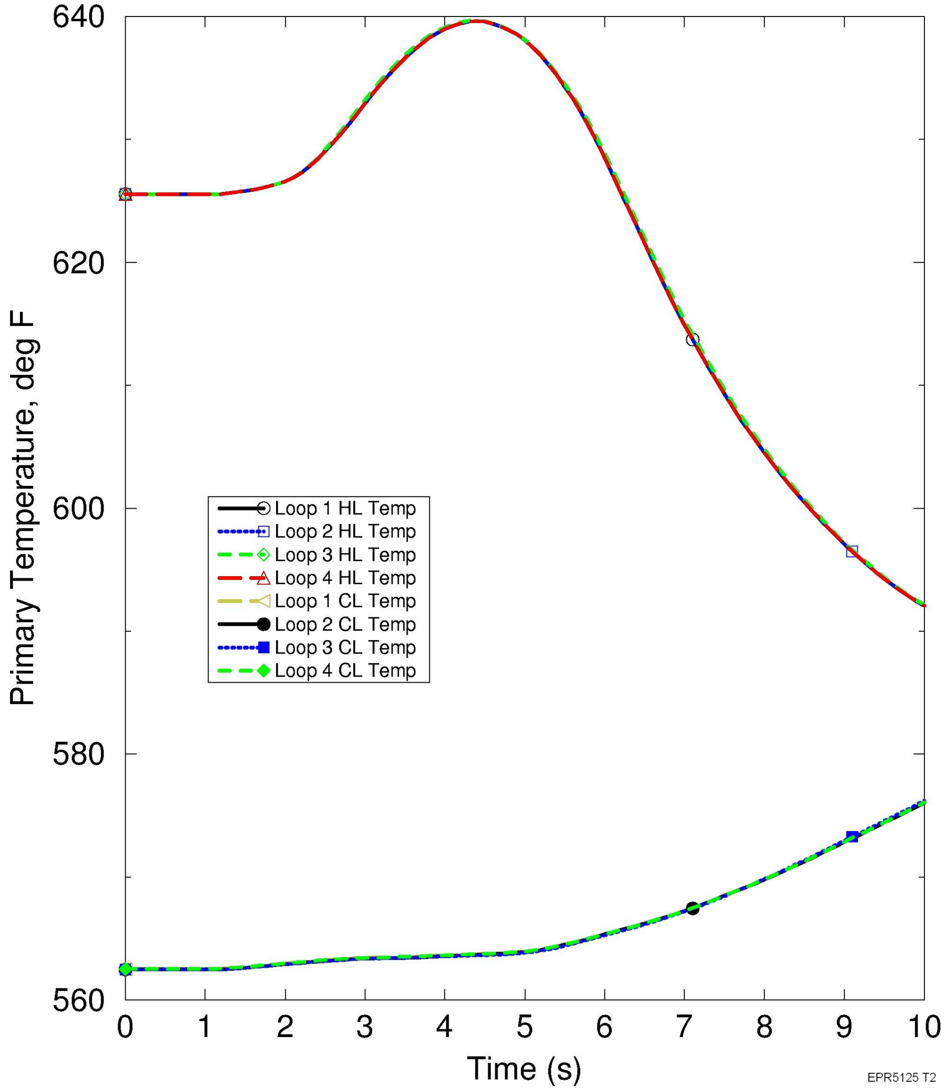

Figure 15.4-36—60% NP Rod Ejection Accident Overpressurization Analysis – Percent Reactor Power

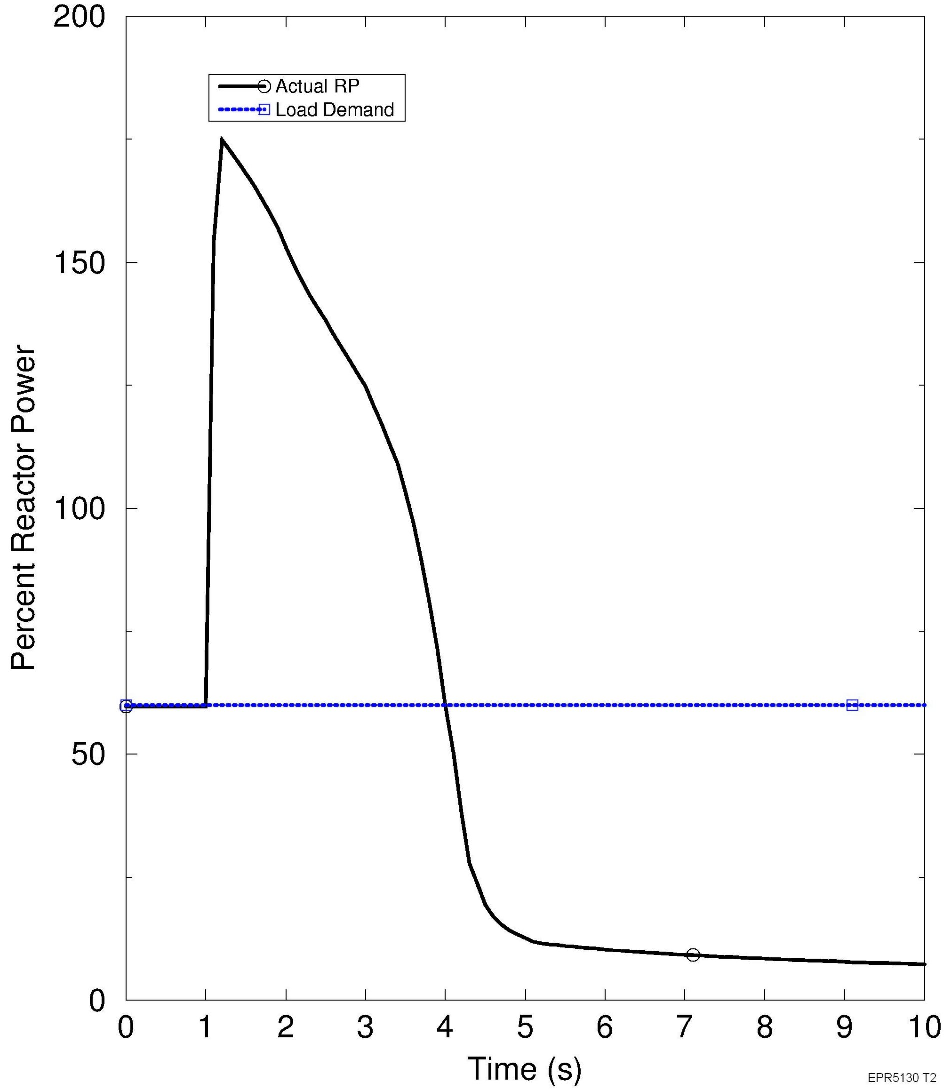

Figure 15.4-37—60% NP Rod Ejection Accident Overpressurization Analysis – Reactivity

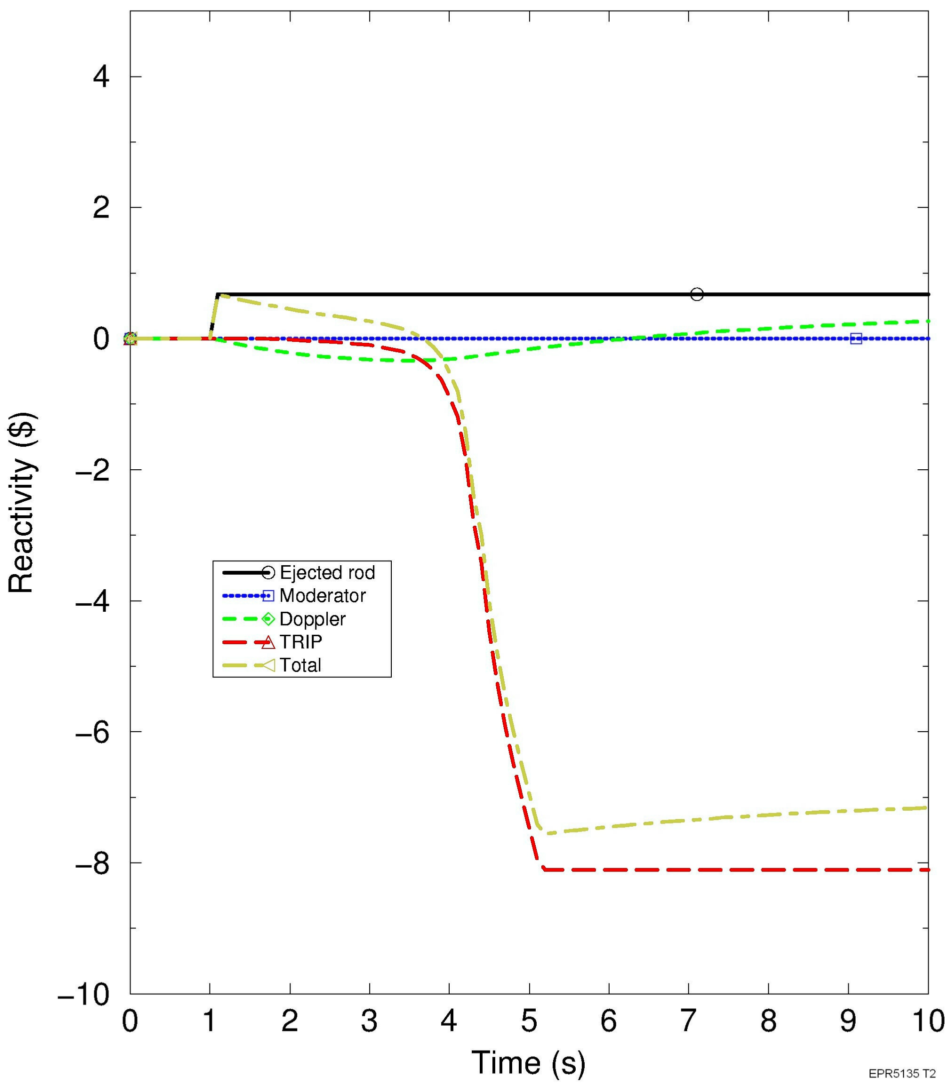

Figure 15.4-38—60% NP Rod Ejection Accident Overpressurization Analysis – Primary System Pressure

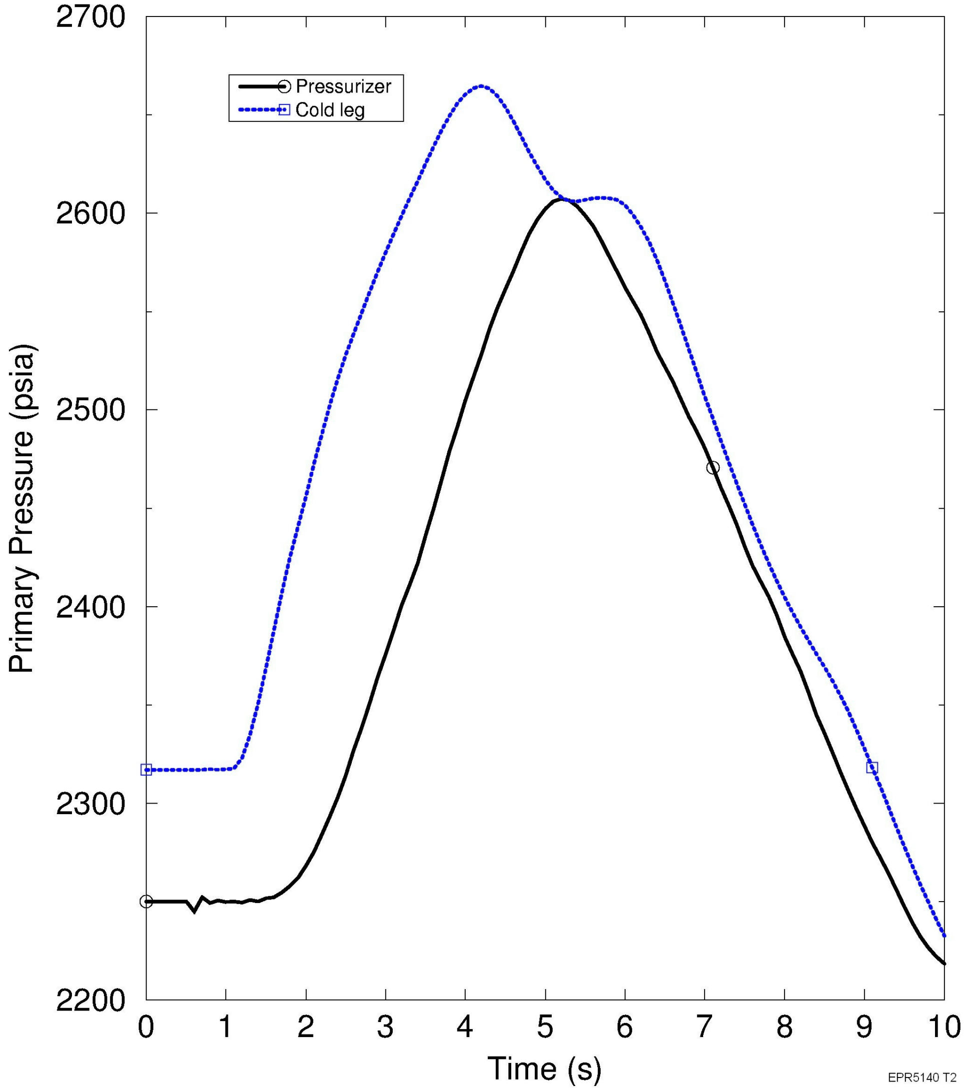

Figure 15.4-39—60% NP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature

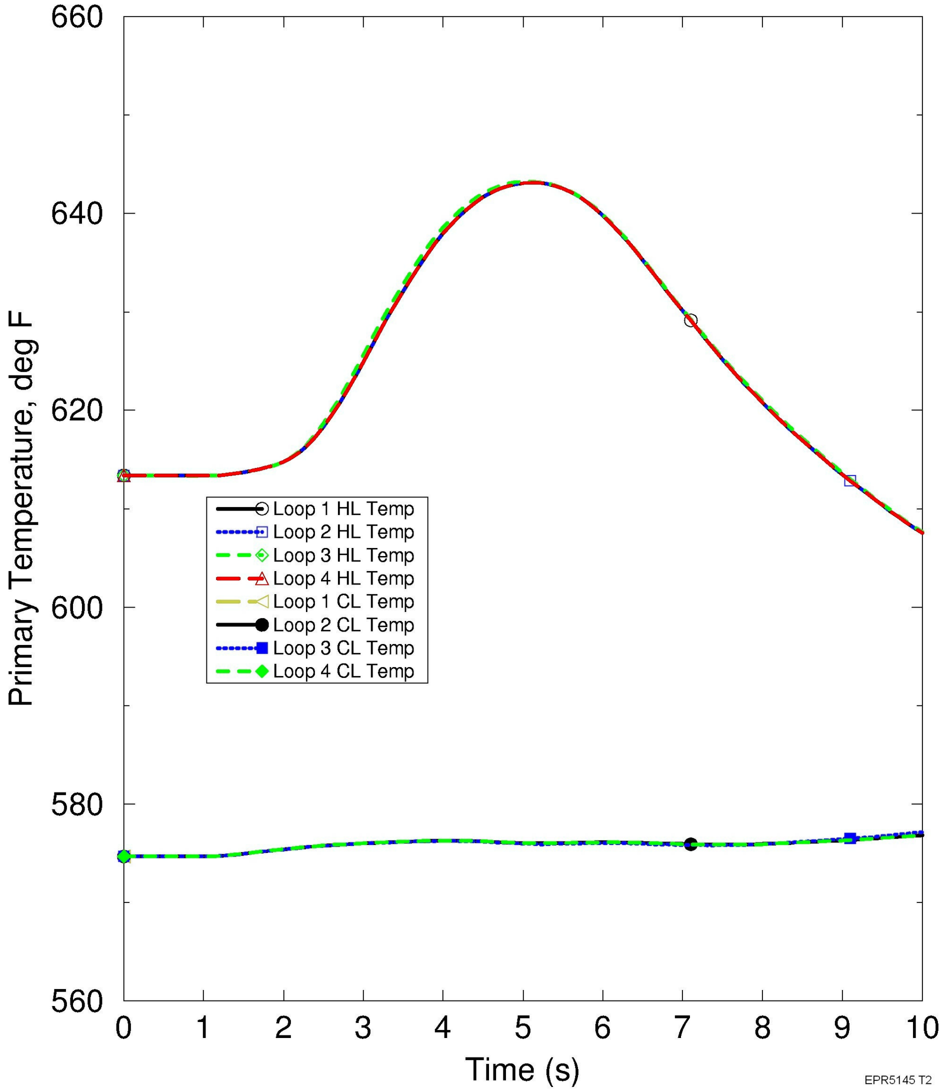

Figure 15.4-40—HZP Rod Ejection Accident Overpressurization Analysis – Percent Reactor Power

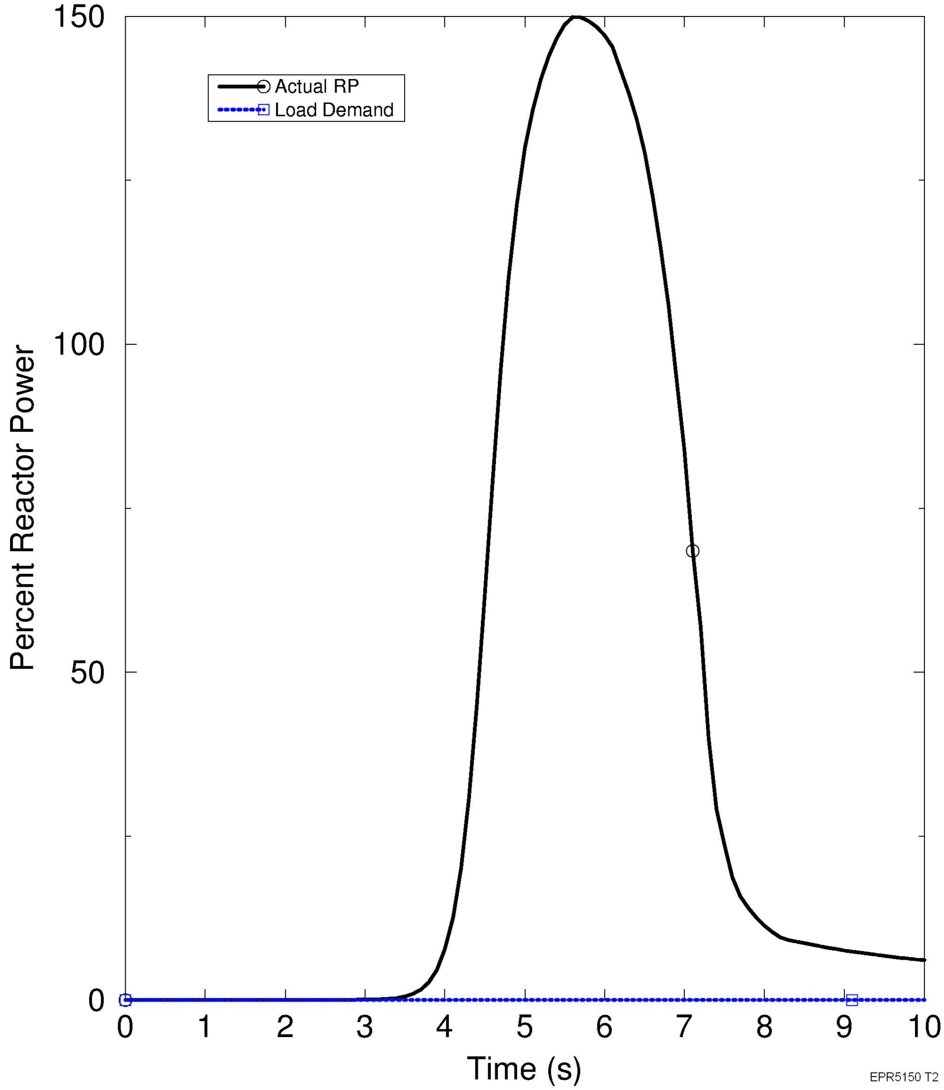

Figure 15.4-41—HZP Rod Ejection Accident Overpressurization Analysis – Reactivity

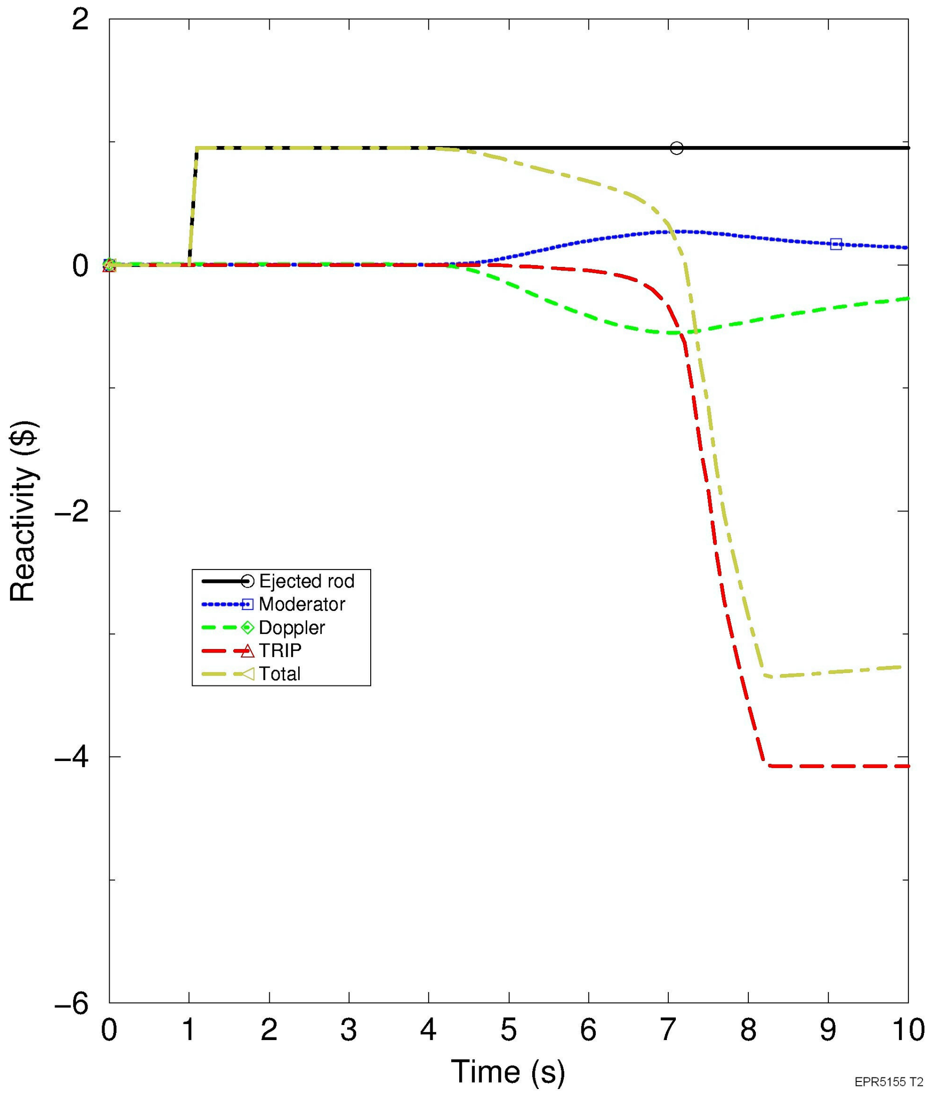

Figure 15.4-42—HZP Rod Ejection Accident Overpressurization Analysis – Primary System Pressure

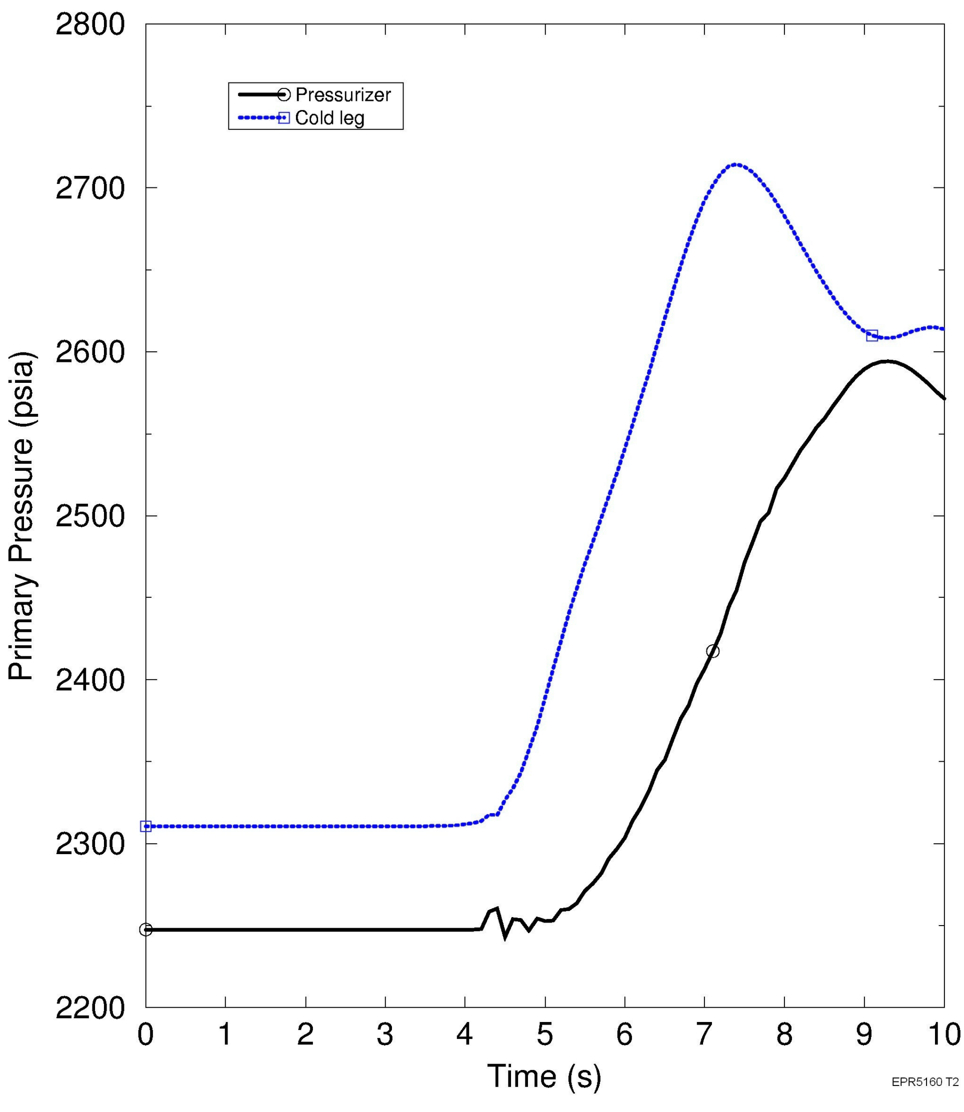

Figure 15.4-43—HZP Rod Ejection Accident Overpressurization Analysis – Primary System Temperature