

December 31, 2022

Docket No. 52-050

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of the NuScale Standard Design Approval Application Part 2 – Final Safety Analysis Report, Chapter 15, “Transient and Accident Analyses,” Revision 0

REFERENCES:

1. NuScale letter to NRC, “NuScale Power, LLC Submittal of Planned Standard Design Approval Application Content,” dated February 24, 2020 (ML20055E565)
2. NuScale letter to NRC, “NuScale Power, LLC Requests the NRC staff to conduct a pre-application readiness assessment of the draft, ‘NuScale Standard Design Approval Application (SDAA),’” dated May 25, 2022 (ML22145A460)
3. NRC letter to NuScale, “Preapplication Readiness Assessment Report of the NuScale Power, LLC Standard Design Approval Draft Application,” Office of Nuclear Reactor Regulation dated November 15, 2022 (ML22305A518)
4. NuScale letter to NRC, “NuScale Power, LLC Staged Submittal of Planned Standard Design Approval Application,” dated November 21, 2022 (ML22325A349)

NuScale Power, LLC (NuScale) is pleased to submit Chapter 15 of the Standard Design Approval Application, “Transient and Accident Analyses,” Revision 0. This chapter supports Part 2, “Final Safety Analysis Report,” (FSAR) of the NuScale Standard Design Approval Application (SDAA), described in Reference 1. NuScale submits the chapter in accordance with requirements of 10 CFR 52 Subpart E, Standard Design Approvals. As described in Reference 4, the enclosure is part of a staged SDAA submittal. NuScale requests NRC review, approval, and granting of standard design approval for the US460 standard plant design.

From July 25, 2022 to October 26, 2022, the NRC performed a pre-application readiness assessment of available portions of the draft NuScale FSAR to determine the FSAR’s readiness for submittal and for subsequent review by NRC staff (References 2 and 3). The NRC staff reviewed draft Chapter 15. NuScale is enclosing information in this submittal that: 1) closes gaps identified between the draft SDAA Chapter 15 and technical content generally expected by the NRC; and 2) resolves identified technical issues that may have adversely impacted acceptance, docketing, or technical review of the application. Section B of the enclosures provide NuScale’s responses to Reference 3 for Chapter 15 observations.

Enclosure 1 contains SDAA Part 2 Chapter 15, “Transient and Accident Analyses,” Revision 0, proprietary version. NuScale requests that the proprietary version (Enclosure 1), be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The

enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 contains the nonproprietary version.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 31, 2022.

Sincerely,



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Enclosure 1: SDAA Part 2 Chapter 15, "Transient and Accident Analyses," Revision 0, (proprietary)

Enclosure 2: SDAA Part 2 Chapter 15, "Transient and Accident Analyses," Revision 0, (nonproprietary)

Enclosure 3: Affidavit of Carrie Fosaaen, AF-132467

Enclosure 1:

SDAA Part 2 Chapter 15, "Transient and Accident Analyses," Revision 0, (proprietary)

Enclosure 2:

SDAA Part 2 Chapter 15, "Transient and Accident Analyses," Revision 0, (nonproprietary)

Contents

<u>Section</u>	<u>Description</u>
A	Chapter 15, “Transient and Accident Analyses,” Revision 0, nonproprietary
B	Readiness Assessment Report responses for Chapter 15
C	Technical Report(s)

Section A

A decorative graphic on the left side of the page consists of three overlapping circles. The top circle contains a mountain range. The middle circle contains a city skyline at night. The bottom circle contains a city skyline at night.

NuScale US460 Plant Standard Design Approval Application

Chapter Fifteen **Transient and Accident Analyses**

Final Safety Analysis Report

Revision 0

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CHAPTER 15 TRANSIENT AND ACCIDENT ANALYSES

15.0 Transient and Accident Analyses

15.0.0 Classification and Key Assumptions

This section addresses the classification of potential plant events that are considered in the design basis. This section identifies key assumptions used in the design-basis event (DBE) evaluations including initial conditions, trip setpoints, and the limiting failure identification process.

15.0.0.1 Initiating Event Selection

Initiating events are considered for internal events occurring in a single NuScale Power Module (NPM) while at power. Design-basis analyses consider the range of power operation because events from a low-power or zero-power state could be more limiting than from full-power conditions. Hazards such as floods and fires in the plant, earthquakes, severe weather conditions, external fires, and external floods are evaluated in Chapters 3, 9, and 19. Section 19.1.7 discusses the consideration of multi-module events.

Initiating events are identified by considering the similarities and differences of the NuScale Power Plant US460 standard design to current-generation plants and the NPM US600 approved design. The probabilistic risk assessment (PRA) in Section 19.1 evaluates the risk associated with the operation of a single NPM at full power as well as at low-power and shutdown modes of operation for internal and external initiating events. Several sources, including the PRA analyses, are included to identify possible DBEs. The events are categorized as follows:

- increase in heat removal by the secondary system
- decrease in heat removal by the secondary system
- decrease in reactor coolant system (RCS) flow rate
- reactivity and power distribution anomalies
- increase in reactor coolant inventory
- decrease in reactor coolant inventory
- radioactive release from a subsystem or component

Table 15.0-1 lists the events selected for evaluation and computer codes used for analyzing each event. Special events, such as the beyond-design-basis core damage event (CDE), are also included in Table 15.0-1.

15.0.0.2 Design-Basis Event Classification

NuScale DBEs are classified by frequency of occurrence, which is then used to assign the analysis acceptance criteria for the event.

The set of DBEs establishes the design adequacy of the NPM and NuScale Power Plant US460 standard design to limit radiological releases below regulatory guidelines.

15.0.0.2.1 Classification by Event Frequency and Type

Design-basis event classification by frequency is based on three distinct categories:

- anticipated operational occurrences (AOOs)
- infrequent events (IEs)
- postulated accidents

Events that are expected to occur one or more times during an NPM lifetime are classified as AOOs. Events that are not expected to occur during an NPM lifetime are classified as IEs, as postulated accidents, or conservatively as AOOs. In general, events that are not considered to be within the design basis are evaluated in Chapter 19; however, those beyond-design-basis events that are explicitly defined by regulation are addressed in this chapter. These events are termed special events. For example, the CDE described in Section 15.10 is a special event.

Special events also encompass defense-in-depth and common cause failures (CCFs) of digital control systems. Digital control system errors caused by a single instance error are treated in this chapter as AOOs. The multiple instance events (i.e., CCFs) are treated as beyond-design-basis events, and addressed with realistic methods and alternate acceptance criteria as described in Section 7.1.

The NuScale Power Plant US460 standard design life is 60 years and the criterion "one or more times in NPM life" is conservatively interpreted as including transients with a frequency of $1\text{E-}2$ per year or more. Because of the increased reliability of plant systems, notably the digital control system, many event initiators traditionally categorized as AOOs have a frequency of occurrence longer than an NPM lifetime. However, such events remain conservatively classified as AOOs in the NuScale Power Plant US460 standard design. The classification of IE is used to identify transients that have a frequency of less than $1\text{E-}2$ per year, but radiological analysis acceptance criteria applied to the event are more restrictive than accident criteria. Design-basis accidents that have very low frequencies and are not expected to occur during plant lifetime are used to establish design criteria for safety-related structures, systems, and components (SSC). Lowering the initiating event frequency reduces the probability of initiating events that can lead to more serious transients and accidents. Chapter 19 contains discussions regarding the reliability of nonsafety-related equipment that contribute to the event frequency.

The single-NPM events are categorized both by the thermal-hydraulic effects on the NPM and the radiological consequences. These basic conditions relate

to the capability to remove heat from the RCS and retain radionuclide inventory within the fission product barriers. Infrequent events have more restrictive radiological acceptance criteria than postulated accidents to maintain the overall product of risk times consequence approximately constant.

To establish the appropriate event frequency for DBEs, typical data bases and industry event classifications are examined for applicability to the NuScale Power Plant US460 standard design. The initiating event frequency is only calculated for a subset of the AOOs. In most cases, AOO consequences are too low to warrant quantification of the event frequency. The event classification is simplified by substituting a deterministic classification where the consequences are small. Quantification of frequency is performed with NPM-specific PRA data to inform event classification where the design is unique or where event consequences are large enough to warrant a quantification of the event frequency.

15.0.0.2.2 Acceptance Criteria

Acceptance criteria relate to the plant thermal-hydraulic and neutronic response and to potential radiological consequence associated with a DBE. Table 15.0-2 provides thermal-hydraulic acceptance criteria.

Thermal-hydraulic acceptance criteria for the rod ejection accident (REA) and loss-of-coolant accidents (LOCAs) are provided in Table 15.0-3 and Table 15.0-4, respectively. Table 15.0-5 provides radiological acceptance criteria.

These acceptance criteria provide the bases for conclusions regarding the integrity of the three radiological barriers (fuel cladding, reactor coolant pressure boundary, and containment) and whether the occurrence of a DBE can generate a more serious condition.

15.0.0.3 Licensing Methodology

The set of DBEs in Table 15.0-1 establishes the capability of the NPM to limit radiological releases within regulatory guidelines. The analyses of these events are performed with assumptions about plant equipment availability, combinations with external events, acceptance criteria, and other parameter information necessary to meet the requirements of the General Design Criteria (GDC) of 10 CFR 50, Appendix A. These conditions, assumptions, and conclusions establish the acceptability of the evaluation models used for analysis as discussed in Section 15.0.2. The GDCs used in developing these conditions, assumptions, and conclusions are discussed in Section 3.1.

The application of the GDCs and principal design criteria described in Section 3.1 establishes the SSC available for event mitigation of a DBE.

15.0.0.4 Initial Conditions

The NPM parameters used in DBE evaluation are identified in Table 15.0-6 and in the Chapter 15 sections detailing the analyzed events. These parameters represent limiting analysis conditions and, unless otherwise noted in a specific evaluation, are common to the evaluation of the DBEs evaluated in this chapter. Parameter uncertainty to bound the range of potential plant conditions at the start of an event is also identified. The spectrum of initial power conditions and time in life are considered for each event to ensure limiting initial conditions are identified.

The module protection system (MPS) is credited in the DBE analyses. Table 15.0-7 lists the signals, the analytical limits, and their associated time delays used in the Chapter 15 analyses.

The NPM response to DBEs is dependent on core power distribution and reactivity characteristics provided by the moderator temperature coefficient and the fuel (Doppler) coefficient. Section 4.3.2 provides details about the calculation of core coefficients and power distributions. Table 4.3-2 provides the reactivity coefficients calculated for the core equilibrium design. The DBE analyses generally use reactivity coefficients associated with the time in cycle analyzed; beginning of cycle analyses use the most positive (least negative) value while end of cycle analyses use the most negative value. The moderator temperature coefficients also vary with power as shown in Table 4.3-2 and Figure 4.3-15. The DBE analyses can also use conservative reactivity coefficients not strictly based on time in cycle or power. The specific treatment of reactivity coefficients in the DBE analyses is described in their respective Chapter 15 sections.

The spectrum of plant conditions is evaluated for development of the radiological source term presented in Section 15.0.3.

15.0.0.5 Limiting Single Failures

10 CFR 50, Appendix A describes a single failure as an occurrence that results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. A component that changes position or state to achieve its safety function is considered an "active" component while a component that does not change position or state to achieve its safety function is considered a "passive" component.

The principal considerations in applying the single-failure criterion to NPM design-basis event evaluations are discussed below.

- 1) Active failures are considered for mechanical components.
 - Design-basis event mitigation credits valves that are classified as safety-related. Valves move to their "fail safe" position when the externally-applied motive force is removed.

- Check valves in the condensate and feedwater system (FWS) are used to mitigate the consequences of a DBE.
 - There is one safety-related check valve and one nonsafety-related backup check valve in each feedwater line (four total check valves per NPM). The FWS check valves are credited to mitigate the consequences of the feedwater line break event.
 - The FWS check valves are not credited for containment isolation but for short-term retention of decay heat removal system (DHRS) inventory, until the feedwater isolation valve (FWIV) or its backup, the nonsafety-related feedwater regulating valve (FWRV), closes. The FWIV performs the containment isolation function and the FWRV serves as a backup to the FWIV.
- 2) Passive failure of a single system, structure or component is considered a potential event initiator, but not as a single failure in the short term.
- Passive failures of fluid systems are considered only on a long-term basis except for check valves whose failure must be postulated coincident with its required response to a DBE.
 - For the purpose of considering passive single failures, the short term is defined as the period up to 24 hours following an initiating event.
 - Components whose proper function is demonstrated and documented are not considered a credible single failure (e.g., American Society of Mechanical Engineers code safety valves).
 - The emergency core cooling system (ECCS) valve inadvertent actuation block (IAB) is a sub-component feature of the ECCS reactor recirculation valve (RRV). The IAB feature prevents the RRV from opening until RCS pressure drops below the threshold established by the IAB opening spring. The IAB operates based on stored energy resulting from the differential pressure between the RCS and containment. The IAB requires no alternating current (AC) or direct current (DC) power, actuation signals, or external motive force to maintain the ECCS valves closed. The IAB provides a simple safety-related mechanical backup to avoid inadvertent actuation of an RRV, in the event DC power to trip valve solenoids is lost and depressurizes the trip line. As such, the IAB closing function is determined not to be subject to consideration as a single active failure for Chapter 15 analyses. This determination is consistent with the position in SECY-19-0036 (Reference 15.0-13).
- 3) Active and passive failures are considered for electrical components.
- Protective actions must be accomplished in the presence of a single detectable failure. The effects of non-detectable failures are considered concurrently as part of the most-limiting single failure.
 - Both loss of and availability of power is assumed in the analysis of each event and are not considered as a single failure.

- 4) Single human errors are considered. In this regard, a key consideration is whether the potential error is an error of "omission" or an error of "commission." An error of omission is one in which an operator action is required to achieve a safety function, but the operator fails to perform the function. A error of commission is one in which no operator action is required, but an erroneous action is taken.
 - Operator errors are considered in identifying event initiators. The NuScale Power Plant US460 standard design limits operator errors to consequences that are no more severe than the worst-case single failure.

Therefore, there are no operator errors that have to be analyzed in the accident analysis. Because the NuScale Power Plant US460 standard design is a passive plant, no operator actions are required for 72 hours. Operator actions allowed by procedure make the consequences less severe. Failure to take one of these actions cannot make the consequences worse than the bounding Chapter 15 analysis.

 - Multiple operator errors or errors that result in common mode failures are beyond the design basis. These events are analyzed in Chapter 19.
- 5) The effect of the single failure on a particular transient is assessed in the context of the transient calculation. The assessment is qualitative, with the use of engineering assessment, quantitative with the use of sensitivity calculations, or both. The assessment considers applicable acceptance criteria. For a particular transient, the limiting single failure for one acceptance criterion can be different than the limiting single failure for a different acceptance criterion for the same DBE. The limiting single failures for Chapter 15 events are described with the event analysis and are identified in Table 15.0-8.

15.0.0.6 Equipment Response and Physical Parameter Assumptions

The following sections address control rod assembly insertion characteristics, decay heat, engineered safety feature characteristics, and required operator actions.

15.0.0.6.1 Control Rod Assembly Insertion Characteristics

The time for inserting control rods directly affects the amount of heat that has to be removed from the core in response to a DBE. Section 4.3 describes the analytical basis for the control rod assembly insertion rates and reactivity effect as a function of time. The analyses for the Chapter 15 DBEs apply additional conservatism to the reactivity insertion rate provided in Section 4.3 to bound potential plant conditions. Figure 4.3-20 shows the control rod insertion versus time and Figure 15.0-1 shows the relative cumulative reactor trip reactivity worth versus time. The use of bounding insertion times provides conservative results for DBE analyses. Drop time testing requirements are specified in the technical specifications.

15.0.0.6.2 Decay Heat

Bounding values for decay heat are designated to represent the maximum decay heat of the core following an event. The 1973 American Nuclear Society (ANS) decay heat standard is used in NRELAP5 to represent bounding decay heat. The LOCA methodology calculates fission product decay heat using a bounding form of the 1973 ANS decay heat standard with a 20 percent uncertainty added to the base value. A bounding form of the 1973 ANS standard in NRELAP5 is conservative relative to the 1971 ANS standard specified in 10 CFR 50, Appendix K. The NuScale Power Plant US460 standard design supports an exemption from 10 CFR 50, Appendix K. The LOCA Evaluation Model topical report (Reference 15.0-5) provides additional information.

For non-LOCAs, the model also uses the conservative 1973 ANS decay heat standard, which is varied by using different decay heat multipliers and specifying whether or not to include the actinide contribution.

The following decay heat contribution values are examples for the NPM. A review of the applicable core physics parameters is performed each cycle to confirm the bounding nature of the values used for the non-LOCA transient analyses.

- Minimum = use multiplier of 0.8 while excluding the actinide contribution
- Maximum = use multiplier of 1.0 while including the actinide contribution

15.0.0.6.3 Engineered Safety Features Characteristics

Engineered safety feature systems include the containment systems (Section 6.2), ECCS (Section 6.3), and DHRS (Section 5.4.3). The DHRS provides cooling for non-LOCA design-basis events when normal secondary-side cooling is unavailable or otherwise not used. The DHRS is designed to remove post-reactor trip residual and core decay heat from operating conditions and transition the NPM to safe shutdown conditions without reliance on external power. Section 5.4.3 provides additional description of the DHRS.

In conjunction with the containment heat removal function, the ECCS provides a means of core decay heat removal for LOCAs that exceed makeup capability. The DHRS provides an additional capacity to remove decay heat during the initial blowdown period of a LOCA and transfers heat to the reactor pool, particularly for smaller LOCAs, where the steam generators are covered for a longer period of time.

The ECCS, in conjunction with the containment vessel (CNV), has unique design provisions that ensure sufficient coolant inventory is retained to maintain the core covered and cooled. The ECCS consists of two reactor vent valves (RVVs) mounted on the head of the reactor pressure vessel (RPV) and two RRVs mounted on the side of the RPV in the downcomer region at a height above the core, as shown in Figure 6.3-1. The ECCS valves are closed

during normal operation and open to actuate the system during accident conditions. Upon ECCS actuation, the RVVs and the RRVs open creating a steam flow path from the pressurizer to containment and an RPV downcomer flow path to and from containment. Water that is vaporized in the core leaves as steam through the RVVs, is condensed and collected in the CNV, and is returned to the downcomer region inside the RPV through the RRVs. The CNV is sized such that the displacement of liquid from the RPV into containment establishes a liquid level above the RRVs, establishing the natural circulation loop. The natural circulation loop removes decay and residual heat from the core and RPV into containment. Heat in the containment is then transferred by conduction and convection to the water in the reactor pool. The ECCS supplemental boron (ESB) function provides a passive source of boron in the CNV that compensates for reactivity addition during reactor cooldown. Section 15.6.5 provides additional information on the NPM response during a LOCA.

The ECCS valves and the DHRS valves do not rely on electrical power or on nonsafety-related support systems for actuation. After actuation, the valves do not require a subsequent change of state or continuous availability of power to maintain their intended safety functions. The RVVs and RRVs are the only active components in the ECCS. No single failure prevents the ECCS from performing its safety function, including single failures in electrical power (single failures in onsite power and offsite power, busses, electrical and mechanical parts, cabinets and wiring), initiation logic, and single active component failure. One RRV and one RVV are required for successful ECCS operation. If the augmented DC power system (EDAS) supply to the MPS or the ECCS and DHRS valve actuators is lost, the valves actuate. The RVVs open immediately and the RRVs open once differential pressure is below the IAB pressure release threshold.

An ECCS actuation would occur in the event of an AOO or IE such as an inadvertent opening of a reactor safety valve (RSV), an inadvertent opening of an ECCS valve, inadvertent ECCS actuation, or a loss of AC power for more than 24 hours. An ECCS actuation would also occur during an AOO that includes an assumption of a loss of EDAS power. An analysis of these events is conducted to quantify the frequency for which a combination of an AOO and an actuation of the ECCS is expected to occur, and the analysis concludes that ECCS actuation in response to an AOO or IE is not expected to occur in the lifetime of an NPM.

15.0.0.6.4 Required Operator Actions

There are no operator actions credited in the evaluation of NuScale Power Plant US460 standard design DBEs. After a DBE, automated actions place the NPM in a safe state and it remains in the safe-state condition for at least 72 hours without operator action, even with assumed failures.

15.0.0.6.5 Availability of Power

Normal AC power systems are nonsafety-related and not credited to mitigate Chapter 15 events. The normal AC power systems consist of the:

- high voltage (13.8 kV) AC electrical distribution system (EHVS)
- medium voltage (4.16 kV) AC electrical distribution system (EMVS)
- low voltage (480V and 120V) AC electrical distribution system (ELVS)

The onsite DC power systems are nonsafety-related and not credited to mitigate Chapter 15 events. The DC power systems consist of the:

- augmented DC power system that supplies essential loads (EDAS)
- normal DC power system that supplies non-essential loads (EDNS)

Treatment of the AC and DC power systems in the Chapter 15 analyses is discussed separately below. Analyses of the DBEs both with and without power from the AC or DC power supplies, as described below, demonstrate these SSC are not relied upon to remain functional during a DBE to perform safety-related functions, and the NuScale Power Plant US460 standard design has appropriate margin to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including effects of AOOs.

15.0.0.6.5.1 Treatment of Alternating Current Power

As discussed in Section 8.2.1, offsite power is not needed to achieve and maintain safety functions and a failure within EHVS does not prevent safety-related functions. A loss of the plant external power connections is typically referred to as a loss of offsite power. As discussed in Section 8.2.1, a transmission grid connection is not the normal power source for the NuScale Power Plant US460 standard design; rather, a connection to an external transmission grid represents a primary plant load. Therefore, failure within an AC electrical distribution system in the NuScale Power Plant US460 standard design is referred to as a loss of normal AC power supply.

The analysis of DBEs considers that a connection to normal AC power is unavailable if the event involves a turbine trip. Although loss of a single NPM is a small disturbance unlikely to cause external connection instability, an initiating event with a reactor trip and subsequent automatic turbine trip is conservatively assumed to disrupt the electrical connection to an external transmission grid. As discussed in Section 8.3.1, the NuScale Power Plant US460 standard design has the capability to become self-sustaining while separated from the offsite electrical grid. However, this capability is not credited in the DBE analyses.

Based on the electrical system design for the NPM, a loss of normal AC power means a loss of power from the ELVS. Such a condition could be

due to: 1) a failure within the ELVS; or, 2) a loss of power from the EHVS or EMVS. Because a failure of the EHVS or EMVS results in the same response from the plant electrical system for transient analyses, these failures are not considered separately. Therefore, the following scenarios are considered for normal AC power assumptions in DBEs: normal AC is available, loss of normal AC at the time of the initiating event, or loss of normal AC at the time of the turbine trip.

A loss of normal AC at the time of the initiating event results in the following:

- control rods begin to fall due to loss of power to the control rod drive mechanisms
- turbine trip
- feedwater pumps stop
- chemical and volume control system (CVCS) recirculation and injection pumps stop
- pressurizer heaters turn off
- the MPS actuates reactor trip, containment isolation, secondary system isolation, and DHRS within 60 seconds of sensing the low AC voltage
- the MPS actuates ECCS 8 hours after reactor trip, or 24 hours after the loss of AC if action is taken to manually block the actuation in the first 8 hours after reactor trip

Although a loss of normal AC power does result in immediate drop of the control rods, the Chapter 15 analyses do not directly credit control rod insertion from the loss of power to the control rod drive mechanisms. The impact of the above responses typically result in plant conditions reaching an MPS setpoint within the first 60 seconds after the loss of normal AC power. If an MPS setpoint is not reached by then, MPS actuations occur at 60 seconds. The Chapter 15 analyses also do not credit the MPS actuations at 60 seconds after the loss of normal AC power unless the event states otherwise.

The response to a loss of normal AC power at the time of turbine trip is the same as described above, except that reactor trip and turbine trip have already taken place (i.e., they are assumed to cause the loss of normal AC) and other MPS actuations may also have already taken place due to the event progression.

The Chapter 15 analyses consider the AC power scenarios described above and present the limiting case. Unless the event states otherwise, the limiting case is for normal AC power available.

15.0.0.6.5.2 Treatment of Normal Direct Current Power

The EDNS is designed with battery backup to allow a continuous power supply in the event the normal AC power supply to the chargers is not available, as described in Section 8.3. The EDNS, including its battery backup, is not safety-related. The EDNS supplies power to the module control system. A loss of EDNS causes a loss of the module control system which results in the following:

- turbine trip
- feedwater pumps stop
- CVCS recirculation and injection pumps stop
- pressurizer heaters turn off

The response to the loss of EDNS is the same as the response to a loss of normal AC except that the control rods do not drop immediately and MPS actuations do not occur based on a set time from the loss of EDNS. For the Chapter 15 analyses that do not credit immediate control rod drop or MPS actuations based directly on loss of normal AC power, loss of EDNS is equivalent to a loss of normal AC power. Therefore, the loss of normal AC power assessment for the event addresses potential loss of EDNS and separate assessment of the loss of EDNS is not performed. For Chapter 15 analyses that do credit immediate control rod drop or MPS actuations based directly on loss of normal power, a separate assessment of loss of EDNS is performed.

15.0.0.6.5.3 Treatment of Augmented Direct Current Power

The EDAS is designed with battery backup to allow a continuous power supply in the event the normal AC power supply to the chargers is not available, as described in Section 8.3. The battery backup for EDAS is sized to supply loads, with the exception of the ECCS valves, for greater than 24 hours. The ECCS valves are unique because the MPS acts to shed the load for these valves at 24 hours after the loss of normal AC power to the EDAS battery chargers. The loss of normal AC power to the EDAS chargers causes the MPS to initiate a reactor trip, actuate DHRS, and close the containment isolation valves 60 seconds after the loss of AC power is detected. The loss of normal AC power to the EDAS battery chargers also causes MPS to actuate ECCS 8 hours after reactor trip or 24 hours after the loss of AC if action is taken to manually block the actuation in the first 8 hours after reactor trip. The EDAS, including its battery backup, is not safety-related.

A loss of normal AC power removes power from the EDAS battery chargers but does not cause a loss of the EDAS because the batteries have independent feeds to the DC buses. The failure modes and effects analysis (FMEA) of the EDAS is summarized in Section 8.3. The FMEA concludes that if EDAS is operating with AC power available and then AC power is lost, the EDAS batteries continue to supply power to their loads,

such as to the MPS. For the transition from EDAS operating with AC power available to AC power unavailable, the FMEA identifies no failure mode that would cause the EDAS batteries to fail to supply power to their loads.

The augmented quality requirements applied to the EDAS are summarized in Section 8.3. Using the FMEA, the EDAS is deterministically evaluated to identify whether power supply from the EDAS would fail during a DBE. No failure modes associated with a DBE progression are identified. This conclusion is consistent with the functions performed by EDAS and the augmented quality requirements applied to ensure the EDAS performs its required functions. The EDAS battery backup has an independent connection to the DC bus that provides power to MPS. The loss of normal AC power does not have a direct causal link to failure of the DC bus supported by the EDAS batteries. If the battery chargers fail due to an electrical fault, the fault is electrically separated from the DC bus. There are no single component failures that could cause loss of power supply from EDAS. Loss of a single module-specific power channel does not result in loss of power supply, reactor trip, or inadvertent ECCS actuation. Failure of two module-specific power channels is required to result in reactor trip and ECCS actuation. The EDAS is designed with independent and diverse monitoring devices such that no single monitoring device failure goes undetected and prevents detection of the failure mechanisms evaluated in the FMEA. Based on these deterministic evaluations, failure of the EDAS would not occur during a DBE. In addition, the probabilistic evaluation of the EDAS concludes that failure of the EDAS is not expected to occur during an NPM lifetime.

A loss of EDAS results in removal of power from the MPS and causes the following:

- reactor trip
- closure of containment isolation valves
- closure of secondary system isolation valves
- closure of demineralized water system isolation valves
- loss of power to the pressurizer heaters
- actuation of DHRS
- opening of RVVs
- opening of RRVs after differential pressure drops below the IAB release pressure threshold

The timing of a potential loss of EDAS is assessed based on the conclusions of the deterministic system evaluation and insights from the probabilistic evaluations. A loss of EDAS at event initiation is considered and addressed in the Chapter 15 analyses as described below. A loss of EDAS after reactor trip is completed is non-limiting because the control rods are fully inserted. A loss of EDAS after event initiation, but before

control rods are fully inserted, is not considered because deterministic evaluation of the system design does not identify a failure mode that would result in failure during this period. The probability of a DBE combined with a random failure of EDAS during this period is so low that this postulated scenario is classified as a beyond-design-basis event. The evaluation of beyond-design-basis events is performed in Chapter 19. Therefore, evaluation of DBEs with an assumed loss of EDAS at event initiation is appropriate to demonstrate that the EDAS is not relied upon to remain functional during a DBE to perform safety-related functions.

For the DBEs and associated acceptance criteria evaluated with the LOCA evaluation model, described in Section 15.0.2, a loss of EDAS at event initiation is explicitly addressed in the Chapter 15 section for that DBE.

For the DBEs that do not use the LOCA evaluation model, a generic assessment of the loss of EDAS at event initiation is performed to confirm EDAS power being available is more limiting for the acceptance criteria being evaluated. A loss of EDAS at event initiation results in immediate reactor trip, containment isolation, secondary system isolation, and DHRS actuation. These MPS actuations immediately mitigate the initiating event. Therefore, the primary and secondary pressure responses are less limiting than if EDAS is assumed to be available and reactor trip is delayed. For evaluation of the SAFDLs, the non-LOCA DBEs are compared to Section 15.6.6, Inadvertent Operation of Emergency Core Cooling System, and found to be bounded by the inadvertent operation of ECCS with EDAS available. Based on the conclusion of this generic assessment, further evaluation of a loss of EDAS power is not provided in the Chapter 15 section of these non-LOCA DBEs.

Reference 15.0-14 provides additional information regarding the treatment of EDAS power in the safety analysis.

15.0.0.6.6 Treatment of Nonsafety-Related Systems

Nonsafety-related systems are considered in establishing the initial plant conditions for DBEs and during the initial plant response to those events. The treatment of nonsafety-related equipment in DBEs is as follows:

- Nonsafety-related system normal operation that increases the consequences of the event is modeled.
- Nonsafety-related system normal operation that improves (decreases) the consequences of the event is not modeled.
- Nonsafety-related system normal operation that does not significantly alter the consequences of the event can be modeled.
- Nonsafety-related equipment is evaluated considering the licensing basis assumptions defining the event. These assumptions can include external events, environmental effects, offsite power availability, and onsite power availability.

- A nonsafety-related system failing to perform its function is considered, but not the failure to a worst-state condition, or worst time of occurrence, except as an event initiator or as a consequence of the event progression.

The reliability of nonsafety-related systems is also considered when categorizing events by frequency (Section 15.0.0.2). Nonsafety-related mechanical and electrical systems are treated in an analogous manner.

Nonsafety-related equipment can be used for event mitigation for the following circumstances:

- when a detectable and nonconsequential random and independent failure must occur in order to disable the system.
- when nonsafety-related components are used as backup protection.

There are occurrences where nonsafety-related equipment is credited for event mitigation because the nonsafety-related component is used for backup protection. Listed below is the equipment associated with these occurrences. Table 15.0-8 identifies the events in which nonsafety-related equipment is credited for event mitigation.

- The nonsafety-related secondary main steam isolation valve (MSIV) serves as the backup isolation device to the safety-related MSIV for isolation of the main steam piping penetrating containment when the safety-related MSIV is assumed to fail.
- The nonsafety-related FWRV serves as the backup isolation device to the safety-related FWIV for isolation of the FWS piping penetrating the containment when the FWIV is assumed to fail.
- The nonsafety-related backup feedwater check valve serves as the backup isolation device to the safety-related feedwater check valve for isolation of the DHRS when reverse flow is experienced during a break in the FWS piping.

The nonsafety-related secondary main steam isolation bypass valve (MSIBV) serves a similar function. The nonsafety-related secondary MSIBV serves as a backup isolation device to the safety-related MSIBV for isolation of the main steam piping penetrating containment. The MSIBVs are normally open during power operations. Although failure of an MSIV bounds MSIBV failure as the MSIV is a larger valve, the nonsafety-related secondary MSIBV is credited as backup protection for event mitigation based on potential failure of the safety-related MSIBVs. For these reasons, the nonsafety-related secondary MSIBV is identified in Table 15.0-8 as nonsafety-related equipment credited for event mitigation when the nonsafety-related secondary MSIV is credited.

The secondary MSIVs and MSIBVs are described in Section 10.3.2. The FWRVs and nonsafety-related backup feedwater check valves are described in Section 10.4.6.

The reactor pool liner, described in Section 9.2.5, is a nonsafety-related component of the reactor pool used as the ultimate heat sink (UHS). Section 9.2.5 describes how the pool liner meets the criteria for event mitigation in that water leakage from the UHS is detectable and leakage is a nonconsequential random and independent failure. Therefore, any event that progresses to using DHRS or convection cooling through the CNV to the reactor pool with the use of RVVs and RRVs uses the UHS and the pool liner.

15.0.0.7 Multiple Module Events

The short-term response of Chapter 15 DBEs are analyzed for a single NPM. The UHS is the only safety-related system for decay heat removal that is shared amongst modules. The extended passive cooling analyses consider a range of UHS pool level and temperature boundary conditions to account for one module, or multiple modules rejecting decay heat to the UHS. Section 19.1.7 discusses consideration of multi-module events.

15.0.1 Radiological Consequence Analyses Using Alternative Source Terms

A modified version of the alternative source term methodology is used to evaluate radiological consequences of DBEs and the beyond-design-basis CDE. The source term methodology and the application of that methodology are described in Section 15.0.3.

15.0.2 Review of Transient and Accident Analysis Methods

This section summarizes the principal computer codes used in transient analyses and describes the evaluation model development and assessment process (EMDAP). A roadmap with references to NuScale topical and technical reports required to develop those evaluation models is provided in Table 15.0-9.

Several different licensing methodologies are required to provide the neutronic, thermal-hydraulic, and radiological response of the plant to postulated accidents, IEs, and AOOs. The licensing methodologies include the computer programs and the calculation framework for a specific transient or accident such as the mathematical models used, assumptions included in the programs, and procedures for treating the program input and output information. The licensing methodology also includes required assumptions about the plant equipment availability, combinations with external events, and other information necessary to specify the calculation procedure and to meet the requirements of the GDC of 10 CFR 50, Appendix A.

The licensing methodologies address the following elements:

- identification of the DBEs to be analyzed
- design-basis licensing assumptions
- determination of the requirements for the licensing methodology, code assessment, uncertainty analysis, and framework qualification

The events that are considered for evaluation in the NuScale Power Plant US460 standard design, the categorization of DBEs, and the description of the licensing assumptions used in the analyses of these events are described in Section 15.0.0.

The design-basis analyses of accidents and transients presented in this chapter are performed in accordance with the requirements of 10 CFR 50.34, 10 CFR 50.46, and where applicable, NUREG-0737. This section describes the review process and acceptance criteria for analytical models and computer codes used for analyzing the accident and transient behavior. The models simulate the event under consideration and demonstrate conservatism of the analysis by one of several methods.

- 1) Demonstrating from the analysis results that the chosen bounding parameters and licensing assumptions are conservative. The 10 CFR 50 Appendix K LOCA analyses are performed in this manner.
- 2) Using sensitivity analyses in the calculation to demonstrate that values are conservative. Many of the AOOs and postulated accident analyses performed in Chapter 15 use this approach.
- 3) Performing best-estimate analyses.

Table 15.0-1 provides a summary of the computer codes used. Acceptance criteria for the events analyzed in Chapter 15 are listed in Table 15.0-2 through Table 15.0-5.

15.0.2.1 Licensing Methodology (Evaluation Models)

The computer codes represent the important phenomena and components necessary to simulate the events identified in Section 15.0.0. Mathematical models and the numerical solution of those models predict the important physical phenomena to calculate the safety consequences of the events being analyzed and to determine if adequate safety margin is provided. The LOCA mathematical models and ECCS evaluation models meet the applicable requirements contained in Appendix K to 10 CFR 50. The NuScale Power Plant US460 standard design supports an exemption from 10 CFR 50 Appendix K requirements that are not applicable. Non-LOCA methodologies are treated by choosing suitably conservative inputs.

The EMDAP establishes the adequacy of a methodology for evaluating complex events that are postulated to occur in nuclear power plant systems. The EMDAP described here is developed for simulating LOCAs in the NPM. While there are differences between the NuScale Power Plant US460 standard design and current generation pressurized water reactors (PWRs), many of the basic physical phenomena are essentially the same. Also, the EMDAP for LOCA and non-LOCA uses a computer code based on the well-established RELAP5 computer code.

By design, the margins to regulatory limits for the NPM are greater than for current generation PWRs, such that a number of phenomena that occur and are important in the PWR loss-of-coolant accident do not occur in the NPM. Examples of phenomena that do not occur for NPM design-basis LOCAs include loop seal clearing, pump coastdown, two-phase pump performance, entry of significant

amounts of noncondensable gases into the system, core reflooding at low reflood rates, clad swelling and rupture, metal-water reaction, and ECCS bypass. A more detailed examination of differences and a detailed proposed reconciliation of existing light water reactor (LWR) regulatory requirements and guidance with the characteristics of the NPM design are presented in Reference 15.0-5.

Reference 15.0-5 provides a description of the LOCA evaluation model, which meets the applicable requirements of 10 CFR 50 Appendix K and 10 CFR 50.46 acceptance criteria. Differences in geometry and operating mode between current PWRs and the NuScale Power Plant US460 standard design include natural circulation, a high pressure containment, and helical coil steam generators. Therefore, a graded approach to development and assessment of the EMDAP for LOCA is appropriate with emphasis on the differences between the NuScale Power Plant US460 standard design and current PWRs. Reference 15.0-5 demonstrates the applicability and assessment of the NRELAP5 code for application to the LOCA event and the inadvertent operation of the ECCS event.

The non-LOCA analysis methodology follows the intent of the EMDAP and builds on the NRELAP5 LOCA model, as described in the Non-LOCA Analysis Methodology topical report (Reference 15.0-7). A non-LOCA phenomena identification and ranking table (PIRT) is used to establish the requirements for this model.

Non-LOCA analysis covers analysis of AOOs, IEs, and postulated accidents with the exception of the inadvertent operation of ECCS discussed in Section 15.6.6. Non-LOCA methodology interfaces with the radiological assessment methodology described in Section 15.0.3 and the Accident Source Term Methodology topical report (Reference 15.0-6), the subchannel methodology described in Section 4.4, the Subchannel Analysis Methodology topical report (Reference 15.0-1), the Statistical Subchannel Analysis Methodology topical report (Reference 15.0-2), and the containment methodology described in Section 6.2 and Reference 15.0-5.

Extended passive cooling following LOCA or non-LOCA events is discussed in Section 15.0.5 and the Extended Passive Cooling and Reactivity Control Methodology topical report (Reference 15.0-8).

Additional details about the code capabilities, limitations, modeling details, and verification and validation are provided in the topical or technical reports listed in Table 15.0-9.

Chapter 15 design-basis analyses require licensing evaluation models for subchannel analysis, radiological analysis, and prediction of neutronic behavior. Model development and qualification for neutronic and subchannel analysis is discussed in Chapter 4, and application of these models in transient and accident analyses is discussed in this chapter.

15.0.2.2 Thermal-Hydraulic Response

15.0.2.2.1 Loss-of-Coolant Accident Methodology

The evaluation model used to analyze design-basis LOCAs and ECCS valve opening events uses the NRELAP5 code, which addresses the unique features and phenomena of the NPM design and complies with the applicable requirements of 10 CFR 50, Appendix K. The design supports an exemption from the requirements of 10 CFR 50, Appendix K that are not applicable. Details of the LOCA methodology and code qualification are in Reference 15.0-5.

10 CFR 50 Appendix K

The ECCS cooling performance must be calculated in accordance with an acceptable evaluation model as specified in Appendix K to 10 CFR 50, and must be calculated for a number of cases to ensure the most limiting LOCAs are identified. Two options for ECCS performance calculations allow for ensuring the most limiting design-basis accident is evaluated.

10 CFR 50.46(a)(i) endorses the best-estimate approach detailed in Regulatory Guide (RG) 1.157 and 10 CFR 50.46(a)(ii) endorses the conservative deterministic approach detailed in Appendix K. In view of the large safety margins in the NPM with respect to 10 CFR 50.46 LOCA acceptance criteria, the deterministic bounding approach in 10 CFR 50.46(a)(ii) is used.

10 CFR 50.46 Loss-of-Coolant Accident Acceptance Criteria

10 CFR 50.46 requires LWRs fueled with uranium oxide pellets within zircaloy cladding be provided with an ECCS such that their calculated core cooling performance after a LOCA conforms to the following acceptance criteria:

- 1) Peak fuel cladding temperature shall not exceed 2200 degrees F.
- 2) Maximum cladding oxidation shall not exceed 0.17 times the total cladding thickness before oxidation.
- 3) The maximum hydrogen generation shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding were to react.
- 4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- 5) The calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for an extended period of time after successful initial operation of the ECCS.

The LOCA licensing methodology (Reference 15.0-5) addresses the first four criteria and the extended passive cooling methodology (Reference 15.0-8)

addresses the fifth criterion. The methodology described in Reference 15.0-5 covers ECCS performance in the NPM up to the time when recirculation flow is established, and the pressures and levels in containment and RPV approach a stabilized condition (i.e., flow is recirculating in through the RRVs, core heat is removed by boiling in the core, and steam exits through the RRVs).

The NPM is designed so there is no core uncover or heatup for a design-basis LOCA and peak cladding temperature (PCT) is below the acceptance criterion. For the LOCA evaluation model, the relevant figures of merit are the collapsed liquid water level in the core, the critical heat flux (CHF) ratio, and containment pressure and temperature. These figures of merit are more sensitive than PCT in the NPM design. Maintaining primary inventory, ensuring the core does not go into post-CHF heat transfer, and ensuring the containment pressure and temperature for the limiting collapsed liquid level case remains below design limits ensures the Appendix K limits for peak cladding temperature, oxidation, and hydrogen production are not violated.

There is no oxidation of the cladding as a result of a LOCA in the NPM. There are no changes in core geometry resulting from a LOCA that would prevent the core from being amenable to cooling. Therefore, the first four 10 CFR 50.46 acceptance criteria are met when the collapsed liquid level remains above the top of the core and the CHF ratio is greater than the analysis limit. The NuScale-specific LOCA acceptance criteria are listed in Table 15.0-4. The minimum collapsed liquid level is discussed in Section 15.0.5.

The calculated core temperature is maintained at an acceptably low value and decay heat is removed in both the short-term and long-term of a LOCA in the NPM. The long-term evaluation of core temperature and decay heat removal is assessed in Section 15.0.5.

15.0.2.2.2 Non-Loss-of-Coolant Accident Methodology

The main steps of the non-LOCA system transient analysis process are:

- 1) Perform steady state and transient system analysis calculation with NRELAP5.
- 2) Evaluate results to confirm margin to RCS and steam generator pressure acceptance criteria.
- 3) Identify if a subchannel analysis is necessary based on system response.
- 4) Perform a subchannel analysis for those events identified in step 3.

For step 1, NRELAP5 is the thermal-hydraulics code used to calculate the NPM system response short-term transient event progression. The NuScale non-LOCA evaluation model starts with the LOCA evaluation model and

modifies it for use for non-LOCA events, as described in Reference 15.0-7. The requirements of the non-LOCA evaluation model capability are established based on the analysis purpose and plant design.

The EMDAP provides a four-element structured process to establish the adequacy of a methodology for evaluating complex events that are postulated to occur in nuclear power plant systems. The evaluation model simulates the NPM system transient response to non-LOCA events.

The short-term non-LOCA transient calculations cover transient initiation and reactor trip, and demonstrate stable natural circulation is achieved and effective DHRS operation is established. The transient progression from this point is similar regardless of the specific initiating event, and the subsequent transient progression is treated as part of long-term decay and residual heat removal analysis discussed in Section 15.0.5.

The NPM parameters used in DBE evaluations are identified in Table 15.0-6. Table 15.0-7 lists the analytical limits and the associated time delays used in the Chapter 15 DBEs. Results of the DBE analyses are compared to the acceptance criteria identified in Table 15.0-2 through Table 15.0-5. System response is evaluated with respect to depressurization rates, to determine if the event is bounded by another DBE. Events with a slow depressurization rate that tend toward increasing CHF ratio, which are bounded by events with a rapid depressurization rate, are not specifically analyzed for CHF with a subchannel analysis. VIPRE-01 is used to perform the subchannel analysis, and Table 15.0-1 identifies the DBEs for which a subchannel analysis is performed.

For the rod ejection calculations, a combination of CASMO5, SIMULATE5, and SIMULATE-3K are used to calculate the core response and reactivity-related inputs. SIMULATE-3K is used to calculate fuel energy deposition and temperatures. The power response for the accident is determined by SIMULATE-3K for both NRELAP5 and VIPRE-01.

NRELAP5 is used to calculate system response including data such as flow rates, pressures, and temperatures. NRELAP5 results are used as boundary conditions for the subchannel analysis.

VIPRE-01 is used to perform the detailed subchannel calculations to determine the minimum critical heat flux ratio (MCHFR) for each event. Peak fuel temperatures, clad temperatures, radially averaged fuel enthalpy, and CHF calculations are performed with VIPRE-01 to address the event acceptance criteria. The description of the codes and the methodology for the REA analysis is provided in the Rod Ejection Accident Methodology topical report (Reference 15.0-12).

15.0.2.2.3 Flow Stability

The NPM system response is obtained by the NuScale proprietary computer code, PIM, which is used to demonstrate system stability at steady-state

operation. The PIM code is described in Section 4.4.7. The Evaluation Methodology for Stability Analysis of the NPM topical report (Reference 15.0-11) provides details about the process used to select and qualify the PIM code.

15.0.2.3 Subchannel Analysis

VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity.

VIPRE-01 is used to generate local thermal-hydraulic conditions for CHF tests in developing a CHF correlation. VIPRE-01 provides local thermal-hydraulic conditions throughout the reactor core used in calculating the MCHFR. VIPRE-01 also provides more realistic boundary conditions, such as the axial profiles for the coolant temperature and wall heat transfer coefficient in a limiting subchannel, for the fuel rod performance analyses. VIPRE-01 is validated against applicable test data that spans the plant range and establishes the code accuracy and uncertainty. Reference 15.0-1 and Reference 15.0-2 provide an applicability assessment of the models, correlations, and features in the VIPRE-01 code for the NPM design.

NuScale-specific CHF correlations have been developed to better represent the core and fuel assembly design, as described in Section 4.4.4, the NuScale Power CHF Correlations topical report (Reference 15.0-3), and the Applicability Range Extension of NSP4 CHF Correlation topical report (Reference 15.0-4). These correlations have been added to the existing suite of VIPRE-01 CHF correlations as an enhancement to VIPRE-01.

15.0.2.4 Radiological Analyses Methodology

The computer codes used in calculating DBE and beyond-design-basis CDE doses are described in the Accident Source Term Methodology topical report (Reference 15.0-6).

15.0.3 Design-Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors

This section summarizes the methodology used to perform the calculations associated with the radiological consequences of the design-basis source terms (DBSTs) and the core damage source term (CDST) associated with the beyond-design-basis CDE; details are given in the Accident Source Term Methodology topical report (Reference 15.0-6). The accident source term methodology assumes one NPM is impacted as there is no credible DBE or maximum hypothetical accident that would involve multiple NPMs. Table 2-1 of Reference 15.0-6 identifies the events analyzed for radiological consequences. Results are in Table 15.0-10.

The regulatory requirements for design certification applicants are described in Reference 15.0-6. The equivalent standard design approval requirements of

10 CFR 52.137(a)(2)(iv), including footnote 9, are applicable to the NuScale Power Plant US460 standard design. The results in Table 15.0-10 meet acceptance criteria.

15.0.3.1 Methodology Overview

The DBE radiological consequence analyses follow the guidance of RG 1.183 methodology modified to reflect the difference in the NuScale Power Plant US460 standard design from large PWRs, as described in Reference 15.0-6. This methodology addresses the submersion and inhalation doses and the direct shine doses from contained or external sources. The key elements of this methodology are:

- Thermal-hydraulic conditions are modeled using NRELAP or bounded using conservative assumptions.
- Source term and dose evaluations are calculated using RADTRAD.
- Meteorological dispersion is calculated using ARCON96.

Reference 15.0-6 summarizes the computer codes used for calculating DBE doses.

15.0.3.2 General Dose Analysis Inputs

The following sections summarize the key aspects for calculating the radiological consequences of the DBEs.

15.0.3.2.1 Core Radionuclide Inventory

The isotopic inventories of fuel assemblies are calculated using SCALE 6.1 as described in Reference 15.0-6. Table 11.1-1 provides the maximum end of cycle core isotopic inventory.

15.0.3.2.2 Primary Coolant Radionuclide Inventory

Table 15.0-11 provides the primary coolant radionuclides and nominal inventory assumed in the dose analyses. The iodine appearance rates, including the pre-incident appearance rates, are described in Reference 15.0-6, where used.

15.0.3.2.3 Secondary Coolant Activity

A sensitivity study was performed in Reference 15.0-6 for the steam generator tube failure (SGTF) and main steam line break (MSLB) events assuming the liquid secondary coolant in the steam generator (SG) was at the primary coolant design-basis limit concentration. The sensitivity study demonstrated dose results are not sensitive to the initial secondary side activity. This conclusion is supported by comparing the secondary coolant source terms shown in Table 11.1-5 with the primary coolant source terms shown in Table 11.1-4.

15.0.3.2.4 Atmospheric Dispersion Factors (χ/Q), Breathing Rates, and Occupancy Factors

Atmospheric dispersion factor (χ/Q) inputs to RADTRAD are derived as described in Reference 15.0-6 with assumptions shown in Table 15.0-12 and Table 15.0-13. Table 2.0-1 provides the accident release χ/Q values.

Main Control Room (MCR) and offsite breathing rate and MCR occupancy factor inputs to RADTRAD, consistent with RG 1.183, are listed in Table 15.0-14.

15.0.3.2.5 Dose Conversion Factors

Consistent with RG 1.183, dose conversion factors from Environmental Protection Agency Federal Guidance Report No. 11 (Reference 15.0-9) and Report No. 12 (Reference 15.0-10) are used for dose analysis.

15.0.3.3 Containment Leakage

Containment leakage is described in Reference 15.0-6 and is consistent with the recommendations of RG 1.183. The design-basis containment leak rate is provided in Table 6.5-1.

15.0.3.4 Secondary-Side Decontamination

The helical coil SGs of the NuScale Power Plant US460 standard design are different than that of a large PWR because the primary coolant is on the outside of the tubes. As a result, there is no bulk water volume in which decontamination can easily occur. Reference 15.0-6 provides the details about the decontamination factor used in the helical coil SGs as well as the treatment of iodine deposition in the main steam piping and the condenser.

15.0.3.5 Reactor Building Decontamination Factors

Reactor Building (RXB) decontamination factors are described in Reference 15.0-6.

15.0.3.6 Receptor Location Considerations

Potential onsite radiological receptor locations considered are the MCR and Technical Support Center (TSC); potential offsite locations considered are the exclusion area boundary (EAB) and low population zone (LPZ). Figure 15.0-2 shows the schematic of a typical RADTRAD code nodalization used to model these locations.

A summary of MCR and TSC characteristics are provided in Section 15.0.3.6.1 and Section 15.0.3.6.2, respectively. The inputs associated with these receptors are presented in Table 15.0-14.

15.0.3.6.1 Main Control Room Design

Accident analyses are performed for two MCR emergency modes as described below.

- Uninterrupted power supply with continuous filtered airflow to the MCR envelope for the event duration.
- Immediate control room habitability system (CRHS) initiation and restored filtered airflow to the MCR envelope at the time of CRHS depletion (after 72 hours of CRHS operation).

Simplifying assumptions are made for the MCR ventilation system design. The key design features assumed for the MCR are summarized below.

- The nonsafety-related normal MCR ventilation system inlet filters remove 99 percent of iodine.
- The nonsafety-related normal MCR ventilation is redirected through the air filtration unit by a nonsafety-related control system once the radioactivity measured at the duct intake reaches the setpoint of 10-times background. The time between when the radiation concentration reaches the detector setpoint and when normal MCR ventilation airflow is credited as being filtered is assumed to be 30 seconds
- The nonsafety-related normal MCR ventilation is isolated and the CRHS is initiated by a nonsafety-related control system once the radioactivity measured downstream of the air filtration unit reaches the setpoint. The setpoint for CRHS initiation and MCR envelope isolation is 10-times the expected radiation out of the filtration unit following the limiting event (i.e., a CDE), which indicates a failure of the filtration unit to remove sufficient radioactivity. The time between when the radiation concentration reaches the detector setpoint and when CRHS is credited as being in operation is assumed to be 30 seconds. Ten times the expected post-accident radiation analytical limits for noble gases, particulate, and iodine are shown in Table 15.0-15.
- The CRHS provides an emergency source of clean pressurized air for 72 hours.
- After 72 hours of CRHS operation, the normal MCR ventilation system is available for use except after a seismic event.
- After a seismic event the normal MCR ventilation system is not available for use. However, dose consequences are not applicable because mitigating SSC are Seismic Category I and capable of performing their safety-related and nonsafety-related functions during and following the event.
- The MCR is habitable during a loss of normal AC power as the CRHS automatically activates without normal AC power, as described in Section 6.4.3.
- The MCR ventilation is designed to minimize inleakage.

- The MCR is designed with a two-door air lock system. Therefore, leakage of 5 cfm is assumed for ingress and egress. An additional 10 cfm of leakage is also assumed.

The MCR ventilation system design modeling assumptions are provided in Table 15.0-16. Details about system operation with CRHS are provided in Section 6.4 and Section 9.4.1.

No credit is taken for the use of personal protective equipment, such as beta radiation resistant protective clothing, eye protection, or self-contained breathing apparatus. No credit is taken for prophylactic drugs such as potassium iodide pills.

15.0.3.6.2 Technical Support Center Design

Accident analyses are performed for one emergency mode: an uninterrupted power supply with continuous filtered airflow to the TSC envelope for the event duration. In the event of CRHS activation, TSC personnel are evacuated and the TSC functions are transferred to an alternate site-specific location because the TSC is not serviced by the CRHS.

The key design features assumed for the TSC are summarized below.

- The nonsafety-related normal TSC ventilation system filters remove 99 percent of iodine under accident conditions.
- The nonsafety-related normal TSC ventilation is redirected through the air filtration unit by a nonsafety-related control system once the radioactivity measured at the duct intake reaches the setpoint of 10-times background. The time between when the radiation concentration reaches the detector setpoint and when normal TSC ventilation airflow is credited as being filtered is assumed to be 30 seconds.
- Leakage of 10 cfm is assumed for ingress and egress. An additional 10 cfm of leakage is also assumed.

The TSC ventilation system design modeling assumptions are provided in Table 15.0-17.

No credit is taken for the use of personal protective equipment, such as beta radiation resistant protective clothing, eye protection, or self-contained breathing apparatus. No credit is taken for prophylactic drugs such as potassium iodide pills.

15.0.3.6.3 Reactor Building Pool Boiling Radiological Consequences

Without available power for the active cooling systems, the addition of makeup water, or operator action, the sensible and decay heat from the NPMs and spent fuel would heat the UHS pool water and could eventually cause the water to boil. Table 9.2.5-2 provides the time for the pool to reach boiling after a loss of normal AC power event. If the pool is assumed to boil, the dose for a

30-day period would be less than 0.1 rem total effective dose equivalent (TEDE) at each of the onsite and offsite receptor locations.

15.0.3.6.4 Radiological Exposures from Shine

Potential radiological exposures from shine following a radiological release event are evaluated. Direct shine from the NPM, sky shine, and shine from ventilation filters are evaluated for the onsite (MCR and TSC) receptor locations. Reference 15.0-6 describes the use of the computer code MCNP for calculating these shine components for the onsite receptors. For the NuScale Power Plant US460 standard design, an alternative approach is used based on other computer codes described in Reference 15.0-6. Direct shine from the NPM is calculated by using RADTRAD to determine the dose within the NPM, and then applying geometric and linear attenuation between the NPM and the onsite receptor locations. Similarly, filter shine is calculated by using RADTRAD to determine the dose at the filter, and then applying geometric and linear attenuation between the filter and the onsite receptor locations. Sky shine is calculated by first using NARCON to account for atmospheric dispersion over the minimum distance between the RXB and the Control Building where the onsite receptors are located. RADTRAD is next used to determine the dose at the Control Building exterior and then linear attenuation is applied between the Control Building exterior and the onsite receptor locations. To provide simplified, bounding shine doses, the source term used for calculating each of these shine components is that associated with the beyond-design-basis CDE. The total shine dose to personnel in the MCR is negligible for the beyond-design-basis CDE. Nevertheless, the 30-day cumulative MCR doses in Table 15.0-10 reflect the combination of the CDE-based total shine dose and the event-specific doses calculated as described in Section 15.0.3.7. The total shine dose to personnel in the TSC is non-negligible for the beyond-design-basis CDE because of the proximity of the TSC to the ventilation filter. However, the 30-day cumulative TSC doses considering the combination of the CDE-based shine dose and the event-specific doses calculated as described in Section 15.0.3.7 remain below the limit of 5 rem TEDE.

Reference 15.0-6 describes that direct shine from the NPM through the CNV, reactor pool water, and then through the RXB walls or ceiling to the environment is assumed to be negligible for the portion of the CNV that is submerged in the reactor pool for the majority of the thirty-day evaluation period. However, direct shine from the NPM through the RXB airspace is explicitly evaluated for the offsite (EAB and LPZ) receptor locations. Direct shine from the NPM is calculated using RADTRAD to determine the dose within the NPM and applying geometric and linear attenuation between the NPM and the offsite receptor locations. The 30-day cumulative EAB and LPZ doses in Table 15.0-10 reflect the combination of the CDE-based direct shine dose and the event-specific doses calculated as described in Section 15.0.3.7.

15.0.3.7 Consequence Analyses of Design-Basis Source Terms**15.0.3.7.1 Failure of Small Lines Carrying Primary Coolant Outside Containment**

Failure of small lines carrying primary coolant outside containment is described in Section 3.2.5 of Reference 15.0-6. Section 15.6.2 describes the sequence of events and thermal-hydraulic response to a failure of small lines carrying primary coolant outside containment. The analysis in Section 15.6.2 shows the reactor core remains covered and no fuel failures occur.

A bounding release of primary coolant from the RPV is defined for the dose consequence analysis rather than using results from the transient analysis described in Section 15.6.2. The assumed primary coolant released from the RPV is 23,000 lbm. This value is calculated to bound the maximum possible decrease in pressurizer level before the containment isolation setpoint is reached, including uncertainty, plus allowance for continued mass release during signal processing and valve closure times. The total mass released from the event is the sum of the mass released from the RPV and the primary coolant from CVCS equipment and piping. Primary coolant in the CVCS equipment (e.g., heat exchangers and filters) and piping within the RXB contribute less than 15,000 lbm additional primary coolant to the potential release.

Containment isolation is assumed to occur at 30 minutes. The primary coolant masses described above are released over this duration. The release of activity is conservatively modeled as a direct release to the environment. After containment isolation, primary coolant leaks through one containment isolation valve (the redundant in-series valve is assumed to fail open) at the maximum leak rate allowed by design-basis limits. The activity from this leak path is assumed to flow directly to the environment with no mitigation or reduction by intervening structures. After 30 hours, the reactor is assumed to be shut down and depressurized, and releases through the containment isolation valve stop.

Iodine spike assumptions for this event are listed in Reference 15.0-6.

The primary coolant in the reactor vessel and CVCS equipment and piping in the RXB is assumed to initially contain the allowable concentration of dose equivalent (DE) I-131 of $5.8\text{E-}02 \mu\text{Ci/gm}$ and DE Xe-133 of $16 \mu\text{Ci/gm}$.

There are no single failures for this event that could result in more severe radiological consequences.

Doses are determined at the EAB, the LPZ, and for personnel in the MCR and the TSC. The MCR model is described in Section 15.0.3.6.1. The potential radiological consequences of the failure of small lines carrying primary coolant outside containment event are presented in Table 15.0-10.

15.0.3.7.2 Steam Generator Tube Failure

Radiological consequences of the SGTF are calculated based on the guidance provided in Appendix F of RG 1.183 as described in Section 3.2.4 of Reference 15.0-6. Section 15.6.3 describes the sequence of events and thermal-hydraulic response to an SGTF. The analysis in Section 15.6.3 shows the reactor core remains covered and no fuel failures occur.

For the faulted SG, a bounding release of primary coolant is defined for the dose consequence analysis rather than using results from the transient analysis described in Section 15.6.3. The assumed primary coolant released from the reactor is 23,000 lbm. This value is calculated to bound the maximum possible decrease in pressurizer level before the secondary system isolation setpoint is reached, including uncertainty, plus allowance for continued mass release during signal processing and valve closure times. Secondary system isolation, including closure of MSIVs and MSIBVs, is assumed to occur at 30 minutes. The primary coolant release associated with the faulted SG is conservatively modeled as a direct release to the environment over this duration. The secondary system is not modeled and no credit is taken for holdup or dilution in the secondary system.

For the intact SG, the primary coolant associated with the maximum primary-to-secondary leak rate allowed by design-basis limits is assumed to be released directly to the environment. The secondary system is not modeled and no credit is taken for holdup or dilution in the secondary system. The direct release of primary coolant to the environment associated with the intact SG is assumed to be terminated after 30 hours when the RCS is depressurized and primary and secondary system pressures equalize.

Iodine spike assumptions for this event are listed in Reference 15.0-6. The primary coolant contains an assumed concentration of $5.8\text{E-}02 \mu\text{Ci/gm}$ DE I-131 for the coincident iodine spike scenario and $3.5 \mu\text{Ci/gm}$ DE I-131 for the pre-incident iodine spike scenario. For both iodine spiking scenarios, the primary coolant is assumed to contain $16 \mu\text{Ci/gm}$ DE Xe-133.

A single failure of an MSIV or MSIBV to close at the time of secondary system isolation is mitigated by closure of the secondary MSIV or secondary MSIBV and does not result in more severe radiological consequences.

Doses are determined at the EAB, the LPZ, and for personnel in the MCR and the TSC. The MCR model is described in Section 15.0.3.6.1. The potential radiological consequences of the SGTF event are presented in Table 15.0-10.

15.0.3.7.3 Main Steam Line Break Outside Containment Accident

Radiological consequences of the MSLB outside containment accident are calculated based on the guidance provided in Appendix E of RG 1.183 as described in Section 3.2.3 of Reference 15.0-6. Section 15.1.5 describes the sequence of events and thermal-hydraulic response to an MSLB outside

containment. The analysis in Section 15.1.5 shows the reactor core remains covered and no fuel failures occur.

A bounding release of primary coolant is defined for the dose consequence analysis rather than using results from the transient analysis described in Section 15.1.5. The primary coolant associated with the maximum primary-to-secondary leak rate allowed by design-basis limits is assumed to be released directly to the environment. The secondary system is not modeled and no credit is taken for holdup or dilution in the secondary system. The direct release of primary coolant to the environment is assumed to be terminated after 30 hours when the RCS is depressurized and primary and secondary system pressures equalize.

Iodine spike assumptions for this event are listed in Reference 15.0-6. The primary coolant contains an assumed concentration of $5.8\text{E-}02 \mu\text{Ci/gm DE I-131}$ for the coincident iodine spike scenario and $3.5 \mu\text{Ci/gm DE I-131}$ for the pre-incident iodine spike scenario. For both iodine spiking scenarios, the primary coolant is assumed to contain $16 \mu\text{Ci/gm DE Xe-133}$.

There are no single failures for this event that could result in more severe radiological consequences.

Doses are determined at the EAB, the LPZ, and for personnel in the MCR and the TSC. The MCR model is described in Section 15.0.3.6.1. The potential radiological consequences of a steam system piping failure outside the primary containment are presented in Table 15.0-10.

15.0.3.7.4 Rod Ejection Accident

Radiological consequences of an REA are calculated based on the guidance provided in Appendix H of RG 1.183. Section 15.4.8 describes the sequence of events and thermal-hydraulic response to an REA, which shows the REA does not result in fuel failure. Therefore, per Appendix H of RG 1.183, a radiological analysis is not required as the consequences of this event are bounded by the consequences of other analyzed events.

15.0.3.7.5 Fuel Handling Accident

A fuel handling accident is postulated to occur during movement of fuel resulting in a dropped assembly onto the spent fuel racks, in the reactor vessel during refueling, in a spent fuel cask during loading, or on the weir wall between the reactor pool and the spent fuel pool. The weir wall provides the highest point in the reactor pool on which a fuel assembly could come to rest. Therefore, it is assumed the dropped fuel assembly lands on the top of the weir wall. The minimum water depth above the dropped assembly is greater than 23 feet. The methodology for determining fuel handling accident radiological consequences, described in Section 3.2.2 of Reference 15.0-6, is consistent with the guidance provided in Appendix B of RG 1.183.

The inventory of fission products available for release at the time of the accident is dependent on a number of factors, such as the power history of the fuel assembly, the time delay between reactor shutdown and the beginning of fuel handling operations, the volatility of the nuclides, and the number of fuel rods damaged in a fuel handling accident. The activity available for release is based on 102 percent power, bounding core inventory provided in Table 11.1-1, and a 1.4 radial peaking factor with 48 hours decay from time of reactor shutdown to the beginning of fuel handling operation. Activity is instantaneously released into the pool water from all fuel rods in the dropped assembly. The assumed release fractions are from RG 1.183, Table 3, and the assumed iodine chemical form released from the pool is 57 percent elemental iodine and 43 percent organic iodide.

There are no single failures assumed for this event that could result in more severe radiological consequences. Noble gases and iodines are released from the pool, while the cesiums and rubidiums are particulates and remain in the pool. The activity is assumed to be instantaneously released from the pool to the RXB atmosphere without holdup or mitigation and then released to the environment over a two-hour period.

Doses are determined at the EAB, the LPZ, and for personnel in the MCR and the TSC. The MCR model is described in Section 15.0.3.6.1. The potential radiological consequences of a fuel handling accident are presented in Table 15.0-10.

15.0.3.7.6 Radiological Analysis of the Iodine Spike Design-Basis Source Term

Section 15.6.5 presents the LOCA analysis, which shows no fuel failures occur. The design has DBEs that result in primary coolant entering an intact containment and the iodine spike DBST is used to bound the radiological consequences of these events. The design-basis iodine spike DBST and the beyond-design-basis CDST described in Section 15.10 are each assessed against the radiological criteria of 10 CFR 52.137(a)(2)(iv). If both analyses show acceptable dose results, then 10 CFR 52.137(a)(2)(iv) is met.

Section 3.2.6 of Reference 15.0-6 provides the methodology for the radiological consequences of the iodine spike DBST, with design-specific details listed below.

- 1) A generic failure is assumed to occur inside the CNV, resulting in the release of all 100,000 lbm of primary coolant from the RCS to the CNV.
- 2) The iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design-basis limits for each of the iodine spiking scenarios. The primary coolant contains an assumed concentration of $5.8\text{E-}02 \mu\text{Ci/gm DE I-131}$ for the coincident iodine spike scenario and $3.5 \mu\text{Ci/gm DE I-131}$ for the pre-incident iodine spike scenario. For both iodine spiking scenarios, the primary coolant is assumed to contain $16 \mu\text{Ci/gm DE Xe-133}$.

- 3) At 30 hours, it is assumed the reactor is shut down and depressurized and releases through the containment to the environment stop.

There are no single failures for this event that could result in more severe radiological consequences.

Doses are determined at the EAB, the LPZ, and for personnel in the MCR and the TSC. The MCR model is described in Section 15.0.3.6.1. The potential radiological consequences of the iodine spike DBST are presented in Table 15.0-10.

15.0.4 Safe, Stabilized Condition

Safety analyses of DBEs are performed from event initiation until a safe, stabilized condition is reached. A safe, stabilized condition is reached when the initiating event is mitigated, the acceptance criteria are met, and system parameters (for example inventory levels, temperatures, and pressures) are trending in the favorable direction. For events that involve a reactor trip, system parameters continue changing slowly as decay and residual heat are removed and the RCS continues to cool down. No operator action is required to reach or maintain a safe, stabilized condition.

Additional considerations are discussed to show Chapter 15 acceptance criteria are not challenged beyond the safe, stabilized condition. Extended passive cooling for decay and residual heat removal is discussed in Section 15.0.5.

15.0.5 Extended Passive Cooling for Decay and Residual Heat Removal

There are two systems that perform the safety-related function of decay and residual heat removal from the NPM following a DBE. The DHRS, described in Section 5.4.3, provides decay and residual heat removal while RCS inventory is retained inside the RPV, the containment is maintained in partially evacuated dry conditions, and power is available. The ECCS, described in Section 6.3, provides decay and residual heat removal when RCS inventory is redistributed between the RPV and the CNV after the RRVs and RVVs are opened.

Design-basis events requiring passive cooling from DHRS or ECCS operation progress from initiation of the event to effective DHRS or ECCS operation demonstrating the NPM reaches a safe, stabilized condition, as described in Section 15.0.4. The decay heat removal process continues into the extended passive cooling phase, either with DHRS, natural circulation between the CNV and RPV through the RRVs and RVVs, or a combination of the two.

15.0.5.1 Decay and Residual Heat Removal Scenarios

The following are the decay and residual heat removal scenarios:

- 1) DHRS with ECCS actuation blocked by operators after verifying acceptable reactivity conditions

- 2) DHRS with the RVVs and RRVs opening 24 hours after a loss of normal AC power
- 3) DHRS with the RVVs and RRVs opening 8 hours after a reactor trip (if ECCS actuation not blocked by operator action)
- 4) ECCS actuation following a LOCA, inadvertent ECCS operation, or DBE with loss of normal AC and EDAS power

Significant boron redistribution before ECCS actuation and unacceptable positive reactivity insertion is precluded as shown by analyses performed in accordance with Reference 15.0-8 and discussed below.

Scenario 1 - Decay and Residual Heat Removal using DHRS

Non-LOCA events progress from event initiation to the point at which DHRS actuation valves open and MSIVs and FWIVs close to allow DHRS operation. This scenario assumes AC power is available and the post-trip reactivity balance for cold conditions is acceptable. Once reactivity conditions are verified by the operators, the 8-hour ECCS timer is blocked. DHRS cools the NPM and provides long-term removal of decay heat while the RRVs and RVVs remain closed. Section 5.4.3 describes operation of DHRS, including actuation, cooling to the safe, stabilized condition, and long-term residual and decay heat removal.

In some scenarios, DHRS can cool the RCS such that the level drops below the top of the riser. Condensation of steam on the outside of the steam generator tubes could reduce the downcomer boron concentration. Diverse flow paths in the riser allow continued primary coolant flow and promote mixing to preclude unacceptable positive reactivity insertion in the event that ECCS actuates.

Scenario 2 - Decay and Residual Heat Removal using DHRS followed by Natural Circulation through the RVVs and RRVs after 24 hours

For non-LOCA events that result in reactor trip and DHRS actuation, the ECCS actuates 8 hours after reactor trip unless the operators block the ECCS timer. Operators block the ECCS timer if reactivity conditions indicate the additional negative reactivity provided by ESB is not needed to maintain subcriticality under cold conditions (Scenario 3 addresses the case where the 8-hour timer is not blocked by operators). If onsite AC power is lost for an extended time and the 8-hour timer actuation is blocked, EDAS power to the RVVs and RRVs is automatically removed after 24 hours and the valves go to a fail-safe open position. The RVVs open, steam condenses in the CNV, and natural circulation is established through the RRVs.

Opening the RVVs and RRVs to continue to depressurize the RCS and establish extended passive cooling with the ECCS is not considered an event escalation because the functions of the RCS barrier are not lost. Progression of the cooling function from DHRS to natural circulation using the RVVs and RRVs is an inherent function in the passive design of the NPM. The RCS barrier continues to provide a confined volume for reactor coolant, which allows a flow path for cooling the core

and thus, confining fission products to the fuel and preventing an escalation of a DBE, including an AOO.

In some scenarios, DHRS can cool the RCS such that the level drops below the top of the riser. Condensation of steam on the outside of the steam generator tubes could reduce the downcomer boron concentration. Diverse flow paths located in the riser allow continued primary coolant flow and promote mixing to preclude unacceptable positive reactivity insertion when ECCS is actuated.

Scenario 3 - Decay and Residual Heat Removal using DHRS followed by Natural Circulation through the RVVs and RRVs after 8 hours

For non-LOCA events that result in reactor trip and DHRS actuation, the ECCS actuates in 8 hours after a reactor trip. If reactivity conditions indicate that negative reactivity from ESB is needed to maintain subcriticality under cold conditions, operators do not block the ECCS 8-hour timer. In this scenario ECCS actuates after 8 hours, the RVVs open immediately and RRVs open when the IAB release differential pressure is reached, typically at the same time as the RVVs as DHRS cooldown depressurizes the RCS. Natural circulation is established through the RRVs.

In some scenarios, DHRS can cool the RCS such that the level drops below the top of the riser and the natural circulation loop is interrupted. Without natural circulation flow, condensation of steam on the outside of the steam generator tubes could reduce the downcomer boron concentration. Diverse flow paths through four holes located in the riser allow continued primary coolant flow and promote mixing to preclude unacceptable positive reactivity insertion when ECCS is actuated.

Scenario 4: Decay and Residual Heat Removal using ECCS following an Inadvertent ECCS Operation or LOCA

The system response in terms of potential challenge to the fuel from inadvertent ECCS operation, as described in Section 15.6.6, bounds other RPV valve opening events. After the RVVs and RRVs open following a LOCA or inadvertent ECCS operation, RCS inventory is redistributed between the RPV and CNV, and the NPM enters the same cooling configuration, irrespective of the initiating event. The results of the extended passive cooling analysis are summarized in Section 15.0.5.3.

Loss-of-coolant accidents or inadvertent ECCS operation events can result in condensation of unborated water in the RPV downcomer once level drops below the riser and the SG tubes become uncovered. The diverse flow paths in the upper riser promote mixing to preclude unacceptable positive reactivity insertion when ECCS is actuated, and ECCS actuation signals are designed to occur before the upper riser flow paths uncover. The lower riser holes provide a flow path for boron mixing for all extended ECCS cooling events.

For events in which ECCS is actuated, the ESB function provides a passive source of boron that compensates for the positive reactivity added by the cooldown.

15.0.5.2 Decay Heat Removal System Cooldown

15.0.5.2.1 Riser Flow Characteristics

During DHRS operation with the riser covered, primary side single phase natural circulation flow is driven by the heat source in the core region and the steam generator heat sink. While the riser is covered during DHRS operation, flow is driven by buoyancy-driven natural circulation flow, determined by the heat source and the total loop resistance. As the riser uncovers, the driving force to sustain flow from riser to downcomer, through the riser flow paths, transitions to the hydrostatic head across the riser. As total flow through the core and over the steam generator decreases, the coolant temperature difference increases to sustain decay heat removal.

The riser holes are divided into one group of lower riser holes and four groups of upper riser holes at discrete elevation levels along the riser wall. The upper riser flow paths are designed to maintain a substantial amount of liquid flow over the steam generator during extended DHRS operation after the top of the riser uncovers. If the mass flow rate decreases sufficiently that fluid in the core or riser region becomes saturated, then some void generation can increase the hydrostatic head driver for flow from the hot to cold side. The upper riser flow paths are sized to allow sufficient mass flow rate to support continued decay heat removal. The flow paths and ECCS actuations are designed so that at least one elevation of upper riser flow paths remains covered by fluid in the riser during extended DHRS operation.

The lower riser holes are designed to allow recirculation of liquid between the riser and downcomer during extended DHRS and ECCS. During DHRS operation, flow through the lower riser holes is driven by the momentum balance, which is affected by the relative pressure drop of fluid above the riser holes. While the riser is covered and the fluid height is the same on both the hot side and cold side, flow is expected to be from the downcomer to riser because the fluid density is higher on the cold side than hot side. When the riser uncovers and flow rate decreases, the temperature gradient from hot side to cold side increases, and the relative collapsed liquid heights of the water columns change since the pressure at the lower plenum must be equal on both sides. In this case, the hot side water column height is higher with lower density compared to the cold side. Additional information about the riser flow path characteristics and analysis is included in Reference 15.0-8.

15.0.5.2.2 Decay Heat Removal System Extended Passive Cooling

During extended DHRS cooling, RCS inventory is retained inside the RPV such that maintaining collapsed liquid level over top of the core is not challenged. ECCS cooling conditions bound extended DHRS cooling with respect to boron precipitation evaluations because ECCS cooling results in

lower coolant temperatures with higher boron concentration. Therefore, demonstration that decay heat removal is sustained, and downcomer concentration remains higher than the critical concentration prior to ECCS actuation, are sufficient to demonstrate acceptance criteria are met during extended DHRS cooling. The methodology for the extended passive cooling analysis and reactivity control is described in Reference 15.0-8.

As described in Section 5.4, the DHRS has redundancy such that two trains of DHRS are available for most transients, even with a single failure. However, there are some transients involving secondary system piping failures or system malfunctions that degrade DHRS capability. DHRS heat transfer capacity is limited in cases with a single DHRS train unavailable or the SGs have high secondary side inventory. Therefore, the following initiating events are specifically evaluated because they are the bounding design-basis events for DHRS extended cooling:

- Initiating event increases secondary side inventory:
 - Increase in feedwater flow (Section 15.1.2)
- Initiating event disables one train of DHRS:
 - Feedwater line break (Section 15.2.8)
 - Main steam line break (Section 15.1.5)
 - Steam generator tube failure (Section 15.6.3)

Each of the above referenced FSAR sections provides analysis results that demonstrate that accident and transient acceptance criteria are met and DHRS cooling is established for initial phase of the event.

For the limiting events for DHRS heat removal capacity, the reactor safety valves provide over-pressure protection until DHRS cooling is sufficient to reduce primary pressure and temperatures. During boiling-condensing DHRS operation, the maximum steam generator pressure is limited to the saturation pressure at the maximum RCS primary coolant temperature. Analyses demonstrate that intact steam generator loops do not overfill. Therefore, primary and secondary pressure limits are not challenged during extended DHRS operation.

The non-LOCA transient analyses presented in Chapter 15 demonstrate that adequate decay heat removal is established. Effects of pool heatup on extended DHRS heat removal is considered. The pool liquid level remains above the top of the DHRS condensers for at least 72 hours. After stable DHRS cooling is established, no changes to the system are expected that would impede the performance of the DHRS to remove heat. Sensitivity calculations with different reactor pool temperature boundary conditions (e.g., 120°F compared to 200°F) demonstrate that eventually the same DHRS heat transfer rate is established, although with a different RCS temperature to maintain the temperature difference from RCS to pool necessary for the decay heat removal. Therefore, after stable DHRS cooling exceeds decay heat and

downward trends in primary pressure and temperature are established during the short-term transient response, RCS pressure and temperatures remain acceptable and cooling continues for 72 hours. Therefore, the US460 standard design provides sufficient cooling via the DHRS for 72 hours.

The upper riser flow paths provide borated water from the riser to mix with the condensate in the downcomer. Calculations demonstrate that primary coolant flow through the riser flow paths exceeds condensation generation and the downcomer concentration remains higher than the critical concentration. The downcomer concentration remains above the critical concentration prior to ECCS actuation, including accounting for the highest-worth control rod assembly stuck out of the core. Whether the module remains on DHRS cooling or ECCS actuates, the core remains subcritical. Therefore, the criteria for extended passive cooling for 72 hours with DHRS are met.

15.0.5.3 Emergency Core Cooling System Extended Passive Cooling

15.0.5.3.1 Boron Transport during Emergency Core Cooling System Cooldown

An ECCS cooldown occurs following a LOCA or inadvertent ECCS operation events or can occur following a non-LOCA event if required to maintain subcriticality. There are two factors that determine limiting cases for maintaining adequate boron concentration in the core to preclude criticality. The first is that an injection line break inside containment presents a continual flow path for boron to transport from the core and riser into containment. During ECCS cooling, this event is unique as it presents the only break where sustained liquid flow is from the RPV to the CNV. Other ECCS events result in increasing the concentration of boron in the RPV and are non-limiting for reactivity analysis during extended ECCS cooldown scenarios. The second factor is ensuring subcriticality for events that transition from DHRS cooling to ECCS cooling on the 8-hour timer. The transients that transition at 8 hours rapidly cool after ECCS actuation. The boron added from the ESB must be transferred into the core by recirculating liquid to offset the positive reactivity addition from decreasing temperature and xenon decay. This scenario is non-limiting for final boron concentration at the end of the analysis period, but can be limiting in the hours immediately following ECCS actuation. Events with slow-biased boron dissolution from the ESB, and events with fast-biased boron dissolution from the ESB are evaluated to determine limiting event progressions. A break on piping connected to the reactor component cooling water system is evaluated because steam generation and condensation in the CNV can dissolve the ESB boron oxide into the liquid below the CNV flange before ECCS actuation at 8 hours. These events are most limiting at end of cycle (EOC), when initial boron concentration is low. The result for the limiting cases for the boron transport analysis are given in Table 15.0-18. Results for beginning of cycle (BOC) and middle of cycle (MOC) are also given for the injection line break for comparison. The methodology for calculating boron transport is presented in Reference 15.0-8; calculation of the critical boron concentration accounts for the highest-worth control rod assembly remaining stuck out of the core.

The margin to boron precipitation limits is evaluated. The minimum difference between core or downcomer concentration and the precipitation limit is recorded along with the time of the minimum in Table 15.0-19. The limiting event is the 100 percent high point vent line outside the CNV. The break outside the CNV reduces RCS inventory available for ECCS cooldown, however, margin to boron precipitation is maintained. The lower riser holes maintain mixing during ECCS operation and boron precipitation is precluded for 72 hours. Therefore, coolable geometry is maintained during ECCS cooldown as required by 10 CFR 50.46 (b)(4).

15.0.5.3.2 Minimum Collapsed Liquid Level during Emergency Core Cooling System Cooldown

This analysis evaluates a number of transients with minimum level biasing applied, consistent with the methodology described in Reference 15.0-8. This biasing is focused on maximizing the differential pressure across the ECCS valves during ECCS conditions, and minimizing the initial RCS inventory. The transients evaluate a wide range of break types, break areas, and power availabilities including LOCA breaks, inadvertent valve opening events, non-LOCA breaks outside containment, and leakage scenarios with small break areas. The initiating events considered are inadvertent opening of an RVV, inadvertent opening of an RRV, injection line break inside containment, injection line break outside containment, discharge line break inside containment, discharge line break outside containment, high point vent line break inside containment, high point vent line break outside containment and SGTF. Each of the line breaks are evaluated with six different break area percentages (100, 33, 11, 3, 1, 0.3). The impact of power availability is also evaluated. The model includes ECCS valve opening logic, including the ECCS actuation 8 hours after reactor trip. Figure 15.0-3 shows the collapsed liquid level for a spectrum of inadvertent RVV and RRV opening events. The collapsed liquid level for these events are representative of the minimum level reached for ECCS cooldown progression with biases set for minimum level.

For collapsed liquid level, the mass retained in the RCS and CNV is the most significant factor. Breaks outside the CNV lose RCS fluid that is not available for ECCS cooling. The greatest amount of mass lost is in the SGTF event. SGTF cases lose primary inventory until the secondary system isolates on low pressurizer level. Once the secondary system isolates, additional mass is transferred as the pressure equalizes between the primary and the failed secondary loop. Once the pressures equalize, some mass drains back from the failed secondary into the primary, but a portion of the mass transferred is retained in the secondary piping. The collapsed liquid level for the limiting SGTF event is shown in Figure 15.0-4.

The collapsed liquid levels for the limiting events are shown in Table 15.0-20. The results show that the core remains covered in all events. The 24-hour, 48-hour, and 72-hour levels for the inadvertent RVV opening event with a loss of AC and EDAS are also given. These levels are typical for ECCS cooldown events and show that collapsed liquid level is on an increasing trend after

24 hours. Therefore, the core remains cooled for 72 hours and the long-term cooling requirement for 10 CFR 50.46(b)(5) is met.

15.0.5.3.3 Maximum Temperature during Emergency Core Cooling System Cooldown

This analysis evaluates extended ECCS cooling with maximum temperature biasing applied, consistent with the methodology described in Reference 15.0-8. Results of this analysis demonstrate that during extended ECCS cooling, vessel integrity is maintained, and module pressures and temperatures decrease and are maintained at low values. The initiating event and short-term transient effects do not impact the long-term ECCS conditions in the module. The limiting analysis case is the inadvertent opening of an RVV with loss of AC and EDAS power with one RVV failing to open.

The maximum temperature analysis accounts for multi-module impacts by specification of high pool temperature and low pool level conditions in the transient analysis. Effects of reactor pool temperature increase and level decrease over the 72-hour period, considering spent fuel heat loads and all modules rejecting decay heat to the reactor pool, are bounded.

The maximum temperature case is not limiting for collapsed liquid level. With pool temperature conditions biased high, collapsed liquid level in the riser remains approximately 10 feet above the top of active fuel throughout the 72-hour period. Within three hours of ECCS actuation, the CNV pressure decreases from its maximum value to less than 200 psia and continues to decrease. Moderator temperatures decreases similarly. The CNV pressure and moderator temperature results shown in Table 15.0-21 demonstrate that module pressures and temperatures are maintained at acceptable values during extended ECCS operation.

15.0.6 References

- 15.0-1 NuScale Power, LLC, "Subchannel Analysis Methodology," TR-0915-17564-P-A, Revision 2.
- 15.0-2 NuScale Power, LLC, "Statistical Subchannel Analysis Methodology, Supplement 1 to TR-0915-17564-P-A, Revision 2," TR-108601-P, Revision 2.
- 15.0-3 NuScale Power, LLC, "NuScale Power Critical Heat Flux Correlations," TR-0116-21012-P-A, Revision 1.
- 15.0-4 NuScale Power, LLC, "Applicability Range Extension of NSP4 Critical Heat Flux Correlation, Supplement 1 to TR-0116-21012-P-A, Revision 1," TR-107522-P, Revision 1.
- 15.0-5 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422-P, Revision 3.

- 15.0-6 NuScale Power, LLC, "Accident Source Term Methodology," TR-0915-17565-P-A, Revision 4.
- 15.0-7 NuScale Power, LLC, "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-P, Revision 4.
- 15.0-8 NuScale Power, LLC, "Extended Passive Cooling and Reactivity Control Methodology," TR-124587-P, Revision 0.
- 15.0-9 U.S. Environmental Protection Agency, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11, EPA-520/1-88-020, 1988.
- 15.0-10 U.S. Environmental Protection Agency, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, 1993.
- 15.0-11 NuScale Power, LLC, "Evaluation Methodology for Stability Analysis of NuScale Power Module," TR-0516-49417-P-A, Revision 1.
- 15.0-12 NuScale Power, LLC, "NuScale Rod Ejection Accident Methodology," TR-0716-50350-P, Revision 2.
- 15.0-13 U.S. Nuclear Regulatory Commission, "Staff Requirements - SECY-19-0036 - Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," July 2, 2019.
- 15.0-14 NuScale Power, LLC, "Treatment of DC Power in Safety Analysis," TR-102621-P, Revision 0.

Table 15.0-1: Design-Basis Events

Section	Type	Classification	Computer Code Used
15.0	Transient and Accident Analysis		
15.0.3	Iodine Spike Design-Basis Source Term (10 CFR 52.137(a)(2)(iv))	N/A ⁽⁶⁾	RADTRAD ORIGEN ARCON96
15.1	Increase in Heat Removal by Secondary System		
15.1.1	Decrease in Feedwater Temperature	AOO	NRELAP5 VIPRE-01
15.1.2	Increase in Feedwater Flow	AOO	NRELAP5 VIPRE-01
15.1.3	Increase in Steam Flow	AOO	NRELAP5 VIPRE-01
15.1.4	Inadvertent Opening of Steam Generator Relief or Safety Valve	AOO	NRELAP5 VIPRE-01
15.1.5	Steam Piping Failures Inside and Outside of Containment	Postulated Accident	NRELAP5 VIPRE-01 RADTRAD ORIGEN ARCON96
15.1.6	Loss of Containment Vacuum/Containment Flooding	AOO	NRELAP5 VIPRE-01
15.2	Decrease in Heat Removal by the Secondary System		
15.2.1	Loss of External Load	AOO	NRELAP5 VIPRE-01
15.2.2	Turbine Trip	AOO	NRELAP5 VIPRE-01
15.2.3	Loss of Condenser Vacuum	AOO	NRELAP5 VIPRE-01
15.2.4	Closure of Main Steam Isolation Valve	AOO	NRELAP5 VIPRE-01
15.2.5	Steam Pressure Regulator Failure (Closed)	N/A ⁽¹⁾	N/A
15.2.6	Loss of Non-Emergency AC to the Station Auxiliaries	AOO	NRELAP5 VIPRE-01
15.2.7	Loss of Normal Feedwater Flow	AOO	NRELAP5 VIPRE-01
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Postulated Accident	NRELAP5 VIPRE-01

Table 15.0-1: Design-Basis Events (Continued)

Section	Type	Classification	Computer Code Used
15.2.9	Inadvertent Operation of the Decay Heat Removal System	AOO	NRELAP5 VIPRE-01
15.3	Decrease in RCS Flow Rate (not applicable)		
15.4	Reactivity and Power Distribution Anomalies		
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	AOO	NRELAP5 VIPRE-01
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	AOO	NRELAP5 VIPRE-01
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	AOO	NRELAP5 VIPRE-01 SIMULATE5
15.4.4	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature	N/A ⁽¹⁾	N/A
15.4.5	Flow Controller Malfunction Causing an Increase in Core Flow Rate (Boiling Water Reactor)	N/A ⁽¹⁾	N/A
15.4.6	Inadvertent Decrease in Boron Concentration in Reactor Coolant System	AOO	NRELAP5
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	IE	SIMULATE5 VIPRE-01
15.4.8	Spectrum of Rod Ejection Accidents	Postulated Accident	SIMULATE-3K NRELAP5 VIPRE-01
15.4.9	Spectrum of Rod Drop Accidents	N/A ⁽¹⁾	N/A
15.5	Increase in Reactor Coolant Inventory		
15.5.1	Chemical and Volume Control System Malfunction	AOO	NRELAP5 VIPRE-01
15.6	Decrease in Reactor Coolant Inventory		
15.6.1	Inadvertent Opening of Reactor Safety Valve	AOO	NRELAP5
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	IE	NRELAP5 RADTRAD ORIGEN ARCON96
15.6.3	Steam Generator Tube Failure	Postulated Accident	RADTRAD NRELAP5 ORIGEN ARCON96
15.6.4	Main Steam Line Failure Outside Containment (BWR)	N/A ⁽¹⁾	N/A
15.6.5	Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Postulated Accident	NRELAP5

Table 15.0-1: Design-Basis Events (Continued)

Section	Type	Classification	Computer Code Used
15.6.6	Inadvertent Operation of Emergency Core Cooling System	AOO	NRELAP5
15.7	Radioactive Release from a Subsystem or Component⁽²⁾		
15.7.4	Fuel Handling Accidents	Postulated Accident	RADTRAD ORIGEN ARCON96
15.7.5	Spent Fuel Cask and NuScale Power Module Drop Accidents	Postulated Accident ⁽³⁾	Not analyzed
Special Events			
15.8	Anticipated Transient Without Scram (10 CFR 50.62)	Special Event	No analysis required
15.9	Stability - note that stability is not an event. The NPM is protected from this phenomenon by MPS trips and technical specification initial conditions.	N/A ⁽⁴⁾	PIM
15.10	Core Damage Source Term (10 CFR 52.137(a)(2)(iv))	Special Event	RADTRAD ORIGEN STARNAUA pH _T ARCON96 MELCOR
8.4	Station Blackout (10 CFR 50.63)	N/A ⁽⁵⁾	NRELAP5

Notes:

(1) Design feature is not part of NuScale Power Plant US460 standard design.

(2) Releases from waste processing systems are described in Chapter 11.

(3) An NPM drop is considered a beyond design-basis-event.

(4) Event is analyzed to AOO acceptance criteria.

(5) Event is included in the loss of non-emergency AC power analysis described in Section 15.2.6 and extended passive cooling analyses described in Section 15.0.5.

(6) The iodine spike DBST is not an event, rather it serves as a bounding surrogate for design-basis loss of primary coolant into containment events described in Section 15.6.

Table 15.0-2: Acceptance Criteria - Thermal Hydraulic and Fuel

Classification⁽⁵⁾	Fuel Clad⁽¹⁾	RCS Pressure	Main Steam System Pressure	Containment	Event Progression
AOO	Fuel cladding integrity shall be maintained by ensuring that SAFDLs are met. ⁽⁶⁾	$\leq 110\%$ of system design pressure	$\leq 110\%$ of system design pressure	Peak pressure $\leq$ design pressure ⁽⁴⁾	An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant.
IE	Fuel cladding integrity shall be maintained by ensuring that SAFDLs are met. ⁽⁶⁾	$\leq 120\%$ of system design pressure	$\leq 120\%$ of system design pressure	Peak pressure $\leq$ design pressure ⁽⁴⁾	Shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.
Postulated Accidents ^{(2),(3)}	Fuel cladding integrity shall be maintained by ensuring that SAFDLs are met. ⁽⁶⁾	$\leq 120\%$ of system design pressure	$\leq 120\%$ of system design pressure	Peak pressure $\leq$ design pressure ⁽⁴⁾	Shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.
Special Event (SBO)	Core cooling maintained	N/A	N/A	Maintain CNV integrity	N/A

Notes:

- (1) Minimum CHF ratio is used instead of minimum departure from nucleate boiling ratio (DNBR), as described in Section 4.4.2.
 (2) Table 15.0-3 provides acceptance criteria for the rod ejection accident.
 (3) Table 15.0-4 provides acceptance criteria for LOCAs.
 (4) Section 6.2.1 provides containment pressure design limits.
 (5) The iodine spike DBST and CDE associated CDST do not have thermal hydraulic or fuel acceptance criteria.
 (6) The SAFDLs are met by assuring MCHFR is maintained above the CHF analysis limit.

Table 15.0-3: Acceptance Criteria Specific to Rod Ejection Accidents

Purpose	Parameter	Acceptance Criteria
To ensure no fuel failure occurs	Maximum peak radial average fuel enthalpy	< 100 cal/g
	Minimum DNBR ⁽¹⁾	≥ CHF analysis limit ⁽¹⁾
	Change in radial average fuel enthalpy	< 33 Δcal/g
	Maximum peak fuel temperature	< Incipient fuel melt limit
For assessment of core coolability	Maximum peak radial average fuel enthalpy	< 230 cal/g
	Maximum peak fuel temperature	< Incipient fuel melt limit
	Fuel pellet cladding fragmentation and dispersal	Coolable geometry maintained
	Fuel rod ballooning	Coolable geometry maintained

Notes:

(1) Minimum CHF ratio is used instead of minimum DNBR, as described in Section 4.4.2.

Table 15.0-4: Acceptance Criteria Specific to Loss-of-Coolant Accidents

Parameter ⁽¹⁾	Acceptance Criteria
Maximum fuel element cladding temperature	MCHFR > 1.20 and collapsed water level above the top of active fuel.
Maximum total oxidation of cladding	MCHFR > 1.20 and collapsed water level above the top of active fuel.
Maximum total hydrogen generated from chemical reaction of cladding with water or steam	MCHFR > 1.20 and collapsed water level above the top of active fuel.
Coolable geometry	Core boron concentration remains below precipitation limit. ⁽²⁾
Long-term cooling	Following initial successful operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended time required by the long-lived radioactivity remaining in the core. Core is subcritical, collapsed water level above the top of active fuel, coolable geometry is maintained.

Notes:

(1) From 10 CFR 50.46.

(2) Note that this is typically met by demonstrating the acceptance criteria are met and by assuring the fuel deformation due to combined LOCA and seismic loads is specifically addressed.

Table 15.0-5: Acceptance Criteria - Radiological

Event	Analysis Release Duration	Exclusion Area Boundary and Low Population Zone⁽¹⁾⁽²⁾ (rem TEDE)	Main Control Room⁽³⁾ (rem TEDE)
Loss-of-Coolant Accident (iodine spike DBST)	Until reactor shut down and depressurized	25	5
Rod Ejection Accident	Per Appendix H of RG 1.183, a radiological analysis is not required as the consequences of this event are bounded by the consequences of other analyzed events	6.3	5
Steam Generator Tube Failure	Affected steam generator: until isolation	25 (Fuel damage or pre-incident spike)	5
	Unaffected steam generator: until reactor shut down and depressurized	2.5 (Coincident iodine spike)	
Main Steam Line Break	Until reactor shut down and depressurized	25 (Fuel damage or pre-incident spike)	5
		2.5 (Coincident iodine spike)	
Fuel Handling Accidents	2 hours	6.3	5
Small Lines Carrying Primary Coolant Outside Containment	Until isolation, if capable, or until reactor shut down and depressurized	2.5	5
Feedwater System Pipe Breaks Inside and Outside Containment	Radiological analysis is not required as the radiological consequences of the event are bounded by the consequences of a steam line break discussed in Section 15.0.3	2.5 ⁽⁴⁾	5
Core Damage Event	30 days for leakage pathways	25	5

Notes:

(1) Based on 10 CFR 52.137 (LOCA), RG 1.183, and 10 CFR 20.1301.

(2) Individual at the EAB shall not receive dose limit for any 2 hour period flowing the onset of release.

(3) Based on 10 CFR 52.137 and is for the duration of the event.

(4) Small fraction (10%) of regulatory dose reference value (25 rem TEDE).

Table 15.0-6: Module Initial Conditions Ranges for Design-Basis Event Evaluation

Plant Parameter	Units	Value	Uncertainty (Bias)	Basis
Design core power	%	100	+ 2	Maximum initial core power is assumed to be 102% power because of uncertainty. Rated power is 250 MW.
RCS T _{avg} at operating conditions.	°F	540	± 5	RCS average temperature is held constant above 20% power as shown in Table 5.1-2.
RCS T _{avg} at startup	°F	340 - 545	N/A	The minimum temperature for criticality is 345° F. The temperature range is given as the minimum temperature for criticality minus the low bias up to the RCS T _{avg} at operating conditions plus the high bias (540+5).
Pressurizer pressure	psia	2000	± 70	Nominal operating pressure is 2000 psia in the pressurizer. The analysis range is 70 psia from the nominal condition.
Pressurizer level at T _{avg} ≥ 530°F	%	60	± 8	Hot full power (HFP) pressurizer level is 60%. The analysis range of ±8% is applied to the HFP level.
Pressurizer level at core power < 15% rated thermal power (RTP)	%	42 - 68	N/A	The low range is 8% below the nominal level at low power. The upper level is defined as the level at HFP plus the high bias (60+8).
Containment pressure	psia	0 - 3	N/A	Nominal operating pressure is less than 1 psia. The maximum of the analytical range is 3.0 psia, which is chosen to bound the initial pressure at which the leak detection system is not available.
Main steam pressure at 100% RTP	psia	475	± 35	Main steam pressure decreases approximately linearly from 700 psia at 20% RTP to 475 psia at 100% RTP.
Feedwater temperature at 100% RTP	°F	250	± 20	This value is a function of reactor power.
RCS flow at 100% RTP	lbm/s	1426 - 1810*	N/A	Flow range intended to slightly bound the minimum and maximum flowrates calculated. RCS natural circulation flow is primarily a function of operating power and the loop hydraulic resistances. The bias direction of 'minimum' or 'maximum' indicates that the RCS loop hydraulic resistance is biased high or low, respectively, to adequately minimize or maximize the initial RCS flow rate to generate the worst case event analysis for comparison against applicable acceptance criteria (primarily for MCHFR). The specified flow range is not related to control deadband or instrumentation uncertainty.
RCS flow minimum	ft ³ /s	1.0	N/A	Minimum flow rate for boron dilution. This flow rate limits the maximum RCS loop time. The specified flow rate is not related to control deadband or instrumentation uncertainty.

Note:

*Flow rate 647-821 kg/s converted to lbm/s.

Table 15.0-7: Analytical Limits and Time Delays

Signal ⁽¹⁾	Analytical Limit	Basis and Event Type	Actuation Delay
High Power	115% ⁽²⁾ RTP (≥ 15% RTP) 25% RTP (<15% RTP)	This signal is designed to protect against exceeding CHF limits for reactivity and overcooling events.	2.0 sec
Source and Intermediate Range Log Power Rate	3 decades/min	This signal is designed to protect against exceeding CHF and energy deposition limits during startup power excursions.	Variable
High Power Rate	±7.5% RTP/30 sec	This signal is designed to protect against exceeding CHF limits for reactivity and overcooling events.	2.0 sec
High Source Range Count Rate	5.0 E+05 counts per second ⁽³⁾	This signal is designed to protect against exceeding CHF and energy deposition limits during rapid startup power excursions.	3.0 sec
High Subcritical Multiplication	3.2	This signal is designed to detect and mitigate inadvertent subcritical boron dilutions in operating Modes 2 and 3.	150.0 sec
High RCS Hot Temperature	620°F	This signal is designed to protect against exceeding CHF limits for reactivity and heatup events.	8.0 sec
High RCS Average Temperature	555°F	This signal is designed to protect against exceeding CHF limits for reactivity events.	8.0 sec
High Containment Pressure	9.5 psia	This signal is designed to detect and mitigate RCS or secondary leaks above the allowable limits to protect RCS inventory and ECCS function during these events.	2.0 sec
High Pressurizer Pressure	2100 psia	This signal is designed to protect against exceeding RPV pressure limits for reactivity and heatup events.	2.0 sec
High Pressurizer Level	80%	This signal is designed to detect and mitigate CVCS malfunctions to protect against overfilling the pressurizer.	3.0 sec
Low Pressurizer Pressure	1850 psia ⁽⁴⁾	This signal is designed to detect and mitigate high-energy line break (HELB) events from the pressurizer vapor space and protect RCS subcooled margin for protection against instability events.	2.0 sec
Low-Low Pressurizer Pressure	1200 psia ⁽⁵⁾	This signal is designed to protect RCS subcooled margin for protection against instability events.	2.0 sec
Low Pressurizer Level	35%	This signal is designed to detect and mitigate pipe breaks to protect RCS inventory and ECCS functionality during LOCAs, primary HELB outside containment events, or SGTF, and to protect the pressurizer heaters from uncovering and overheating during decrease in RCS inventory events.	3.0 sec
Low-Low Pressurizer Level	15%	This signal is designed to detect and mitigate pipe breaks to protect RCS inventory and ECCS functionality during LOCAs, primary HELB outside containment events, or SGTF.	3.0 sec
Low-Low Main Steam Pressure	20 psia	This signal is designed to detect and mitigate secondary HELB outside containment to protect steam generator inventory and DHRS functionality.	2.0 sec

Table 15.0-7: Analytical Limits and Time Delays (Continued)

Signal ⁽¹⁾	Analytical Limit	Basis and Event Type	Actuation Delay
Low Main Steam Pressure	300 psia ⁽⁶⁾	This signal is designed to detect and mitigate secondary HELB outside containment to protect steam generator inventory and DHRS functionality.	2.0 sec
High Main Steam Pressure	1200 psia	This signal is designed to detect and mitigate loss of main steam demand to protect primary and secondary pressure limits during heatup events.	2.0 sec
High Main Steam Superheat	150°F	This signal is designed to detect and mitigate steam generator boil off to protect DHRS functionality during at power and post trip conditions.	8.0 sec
Low Main Steam Superheat	0.0°F	This signal is designed to detect and mitigate steam generator overfilling to protect DHRS functionality during at power and post trip conditions.	8.0 sec
Low RCS Flow	1.0 ft ³ /s	This signal is designed to ensure boron dilution cannot be performed at low RCS flow rates where the loop time is too long to be able to detect the reactivity change in the core within sufficient time to mitigate the event.	6.0 sec
Low-Low RCS Flow	0.0 ft ³ /s	This signal is designed to ensure flow remains measureable and positive during low power startup conditions.	6.0 sec
Low RPV Riser Level Range	540-552 in. ⁽⁷⁾	This signal is designed to actuate ECCS upon riser uncover for LOCA events.	60.0 sec
Low-Low RPV Riser Level Range	460-472 in. ⁽⁷⁾	This signal is designed to actuate ECCS before the upper riser holes uncover.	60.0 sec
Low AC Voltage	Note 8	This signal is designed to ensure appropriate load shedding occurs to EDAS in the event of extended loss of normal AC power to the EDAS battery chargers.	60.0 sec
High Under-the-Bioshield Temperature	250°F	This signal is designed to detect high energy leaks or breaks at the top of the NPM under the bioshield to reduce the consequences of HELBs on the safety-related equipment located on top of the module.	8.0 sec

Notes:

- Interlocks, permissives, and overrides for these signals are described in Table 7.1-5.
- The overcooling event analyses account for uncertainty due to decreasing downcomer temperature. The uncertainty is 7% for downcomer temperature decreases up to 10°F and is scaled upwards from 7% for downcomer temperature decreases beyond 10°F.
- The high count rate trip is treated as a source range over power trip that occurs at a core power analytical limit of 500 kW, which functionally equates neutron monitoring system counts per second to core power in watts. This trip is bypassed once the intermediate range signal is established.
- If RCS hot temperature is above 500°F as shown in Figure 4.4-2.
- If RCS hot temperature is below 500°F as shown in Figure 4.4-2.
- If RCS hot temperature is above 500°F.
- The RPV water level is presented in terms of elevation where reference zero is the bottom of the module assembly (at the bottom of the reactor pool). The range accommodates instrumentation uncertainty.
- Normal AC voltage is monitored at the battery chargers for the EDAS. The analytical limit is based on an average voltage below 80% of normal.

Table 15.0-8: Assumed Single Failures and Credited Nonsafety-Related Systems

Section	Design-Basis Event	Assumed Single Failure	Credited Nonsafety-Related System
15.1	Increase in Heat Removal by the Secondary System		
15.1.1	Decrease in Feedwater Temperature	No adverse single failures	None
15.1.2	Increase in Feedwater Flow	Failure of one FWIV to close	FWRV
15.1.3	Increase in Steam Flow	No adverse single failures	None
15.1.4	Inadvertent Opening of Steam Generator Relief or Safety Valve	No adverse single failures	None
15.1.5	Steam System Piping Failures Inside and Outside of Containment	Failure of FWIV to close on the impacted secondary train or Failure of MSIV to close on the impacted secondary train	FWRV or Secondary MSIV, Secondary MSIBV
15.1.6	Loss of Containment Vacuum/ Containment flooding	No adverse single failures	None
15.2	Decrease in Heat Removal by the Secondary System		
15.2.1	Loss of External Load	Failure of one FWIV to close	FWRV
15.2.2	Turbine Trip	Failure of one FWIV to close	FWRV
15.2.3	Loss of Condenser Vacuum	Failure of one FWIV to close	FWRV
15.2.4	Closure of Main Steam Isolation Valve	Failure of one FWIV to close	FWRV
15.2.5	Steam Pressure Regulator Failure (Closed)	N/A	N/A
15.2.6	Loss of Non-Emergency AC to the Station Auxiliaries	No adverse single failures	None
15.2.7	Loss of Normal Feedwater Flow	No adverse single failures	None
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Failure of one FWIV to close	FWRV
15.2.9	Inadvertent Operation of the Decay Heat Removal System	No adverse single failures	None
15.3	Decrease in RCS Flow Rate (not applicable)		
15.4	Reactivity and Power Distribution Anomalies		
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition	No adverse single failures	None
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	No adverse single failures	None
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	Failure of an ex-core flux detector with respect to power-related trips	None

Table 15.0-8: Assumed Single Failures and Credited Nonsafety-Related Systems (Continued)

Section	Design-Basis Event	Assumed Single Failure	Credited Nonsafety-Related System
15.4.4	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature	N/A	N/A
15.4.5	Flow Controller Malfunction Causing an Increase in Core Flow Rate (BWR)	N/A	N/A
15.4.6	Inadvertent Decrease in Boron Concentration in Reactor Coolant System	No adverse single failures	None
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	No adverse single failures	None
15.4.8	Spectrum of Rod Ejection Accidents	Failure of an ex-core detector with respect to power-related trips	None
15.4.9	Spectrum of Rod Drop Accidents	N/A	N/A
15.5	Increase in Reactor Coolant Inventory		
15.5.1	Chemical and Volume Control System Malfunction	No adverse single failures	None
15.6	Decrease in Reactor Coolant Inventory⁽¹⁾		
15.6.1	Inadvertent Opening of Reactor Safety Valve	No adverse single failures	None
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	No adverse single failures	None
15.6.3	Steam Generator Tube Failure	Failure of MSIV or MSIBV to close on faulted steam generator	Secondary MSIV, Secondary MSIBV
15.6.4	Main Steam Line Failure Outside Containment (BWR)	N/A	N/A
15.6.5	Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure	No adverse single failures	None
15.6.6	Inadvertent Operation of Emergency Core Cooling System	No adverse single failures	None
15.7	Radioactive release from a subsystem or component		
15.7.4	Fuel Handling Accidents	No adverse single failures	None
15.7.5	Spent Fuel Cask and NuScale Power Module Drop Accidents	N/A	N/A

Table 15.0-8: Assumed Single Failures and Credited Nonsafety-Related Systems (Continued)

Section	Design-Basis Event	Assumed Single Failure	Credited Nonsafety-Related System
Special Events			
15.8	Anticipated Transient Without Scram	N/A	N/A
15.9	Stability	No adverse single failures	None
15.10	Core Damage Event	N/A	N/A

Note:
(1) The iodine spike DBST, which is a surrogate for the Section 15.6 DBEs that result in primary coolant entering the containment, does not have assumed single failures or credited nonsafety-related systems.

Table 15.0-9: Referenced Topical and Technical Reports

Topical or Technical Report	Report	Description
LOCA Evaluation Model (Topical)	TR-0516-49422-P	Summarizes the LOCA PIRT, the NRELAP5 code features and modifications, assessments of NRELAP5 against separate effects tests (SETs) and integral effects tests (IETs), and the applicability evaluation of NRELAP5 to LOCA and inadvertent RPV valve opening analyses. Provides sample cases to demonstrate application of the evaluation model to the NPM design. The transition between the initial LOCA event and long-term cooling is defined in this document. Summarizes the CNV pressure and temperature analysis methodology.
Extended Passive Cooling and Reactivity Control Methodology (Topical)	TR-124587-P	This report summarizes the long-term design-basis event progression following DHRS and ECCS actuation, the regulatory requirements and NuScale-specific design requirements applicable to long-term core cooling, the extended passive cooling (XPC) acceptance criteria, the XPC PIRT, the analysis tools, qualification of the tools, and methodology for demonstrating that the XPC acceptance criteria are met, and example results of XPC analyses for collapsed liquid level above top of core, subcriticality, and boron transport.
Evaluation Methodology for Stability Analysis of NuScale Power Module (Topical)	TR-0516-49417-P-A	Presents a methodology for addressing thermal-hydraulic stability in the NPM. The basis of the NPM stability study is a detailed phenomenological review. Provides generic representations of anticipated transients where unstable oscillations can occur. Identifies the limiting instability mode as natural circulation instability. The adiabatic riser response dominates the response rather than wave propagation in the core. The dynamics of the SG and the fission power response to reactivity feedback influence stability. Describes the computational method for the analysis of the postulated instability modes of the NPM during steady state normal operation and anticipated transients. Identifies that potential instability from loss of subcooling in the riser is excluded by MPS protective actions.
Accident Source Term Methodology (Topical)	TR-0915-17565-P-A	Describes assumptions, codes, and methodologies used to calculate the radiological consequences of design-basis accidents, including the iodine spike DBST, and the beyond-design-basis core damage event. Describes the methodologies associated with performing the beyond-design-basis CDST radiological analysis and associated aerosol transport and iodine re-evolution assessment methodologies. Describes the STARNAUA aerosol modeling to the range of post-accident containment conditions and justifies the assumption that no elemental iodine decontamination factor limit should be applied to natural aerosol removal phenomenon in containment. Describes the use of ARCON96 for establishing offsite atmospheric dispersion.

Table 15.0-9: Referenced Topical and Technical Reports (Continued)

Topical or Technical Report	Report	Description
Subchannel Analysis Methodology (Topical)	TR-0915-17564-P-A	Discusses how the NRC requirements for use of VIPRE-01 safety evaluation reports are met, the modelling methodology for performing steady state and transient subchannel analyses, and the qualification of the code for application to the design. Explains why the methodology is independent of any one CHF correlation and can be used for NPM applications if methodology requirements are satisfied. Describes methodology for treatment of uncertainties in the subchannel methodology.
Statistical Subchannel Analysis Methodology (Topical)	TR-108601-P (Supplement 1 to TR-0915-17564-P-A)	Discusses the modeling approach applied in steady-state and transient subchannel analysis. Describes the methodology for treatment of uncertainties in the statistical subchannel methodology.
NuScale Power Critical Heat Flux Correlations (Topical)	TR-0116-21012-P-A	Provides the bases for use of CHF correlations in VIPRE-01 within its range of applicability, along with its associated correlation limit. The report describes the tests, test facilities, statistical methods, base CHF correlation development, NSPX factor development, and final validation for the development of the CHF correlation.
Applicability Range Extension of NSP4 Critical Heat Flux Correlation (Topical)	TR-107522-P (Supplement 1 to TR-0116-21012-P-A)	Provides the bases for an extension to the range of applicability for the NSP4 CHF correlation. The correlation and correlation limit that are justified and previously approved remain unchanged.
NuScale Rod Ejection Accident Methodology (Topical)	TR-0716-50350-P	Describes the codes and methodology used to analyze the REA. Describes the three-dimensional behavior using SIMULATE5 and SIMULATE-3K, the reactor system response using NRELAP5, and the subchannel thermal-hydraulic behavior and fuel response using VIPRE-01.

Table 15.0-9: Referenced Topical and Technical Reports (Continued)

Topical or Technical Report	Report	Description
Non-LOCA Analysis Methodology (Topical)	TR-0516-49416-P	Describes evaluation model that simulates the NPM transient response to non-LOCA events. Addresses the EMDAP process used to establish the adequacy of the non-LOCA methodology. Uses a graded approach to the EMDAP for development of the non-LOCA system transient evaluation model considering the overlap in high-ranked phenomena and conservatism applied to input and boundary conditions of the LOCA evaluation model in the non-LOCA plant transient calculations. Describes the non-LOCA PIRT assessment of the relative importance of phenomena and processes that can occur in the NPM during non-LOCA events in relation to specified figures of merit. Describes the requirements for evaluation model capability developed from the non-LOCA PIRT. Explains how NRELAP5 assessments performed for LOCA evaluation model development demonstrate NRELAP5 qualification for high rank/low knowledge-level non-LOCA PIRT phenomena.
Treatment of DC Power in Safety Analysis (Technical)	TR-102621-P	Describes the design-review process to determine the safety and risk classification of SSC, including the application of the process to DC power electrical systems. Provides a deterministic technical basis for the loss of power considerations in the safety analysis. Provides an evaluation of an RPV valve opening event with a loss of all power. Provides a summary of the probabilistic risk assessment, including metrics pertaining to the potential for ECCS valve opening during AOOs.

Table 15.0-10: Radiological Dose Consequences for Chapter 15 Analyses

Event ⁽¹⁾	Location	Acceptance Criteria (rem TEDE)	Dose (rem TEDE)
Failure of Small Lines Carrying Primary Coolant Outside Containment	EAB	2.5	0.62
	LPZ	2.5	0.63
	MCR	5.0	0.19
Steam Generator Tube Failure (pre-incident iodine spike)	EAB	25.0	0.55
	LPZ	25.0	0.56
	MCR	5.0	0.25
Steam Generator Tube Failure (coincident iodine spike)	EAB	2.5	0.48
	LPZ	2.5	0.51
	MCR	5.0	0.14
Main Steam Line Break (pre-incident iodine spike)	EAB	25.0	0.20
	LPZ	25.0	0.21
	MCR	5.0	0.02
Main Steam Line Break (coincident iodine spike)	EAB	2.5	0.20
	LPZ	2.5	0.26
	MCR	5.0	0.05
Fuel Handling Accident	EAB	6.3	1.40
	LPZ	6.3	1.40
	MCR	5.0	0.55
Iodine Spike Design-Basis Source Term ⁽²⁾ (pre-incident iodine spike)	EAB	25.0	0.20
	LPZ	25.0	0.20
	MCR	5.0	0.02
Iodine Spike Design-Basis Source Term ⁽²⁾ (coincident iodine spike)	EAB	25.0	0.20
	LPZ	25.0	0.21
	MCR	5.0	0.02
Core Damage Event ⁽³⁾	EAB	25.0	2.19
	LPZ	25.0	4.75
	MCR	5.0	1.31

Notes:

- (1) The dose shown for the event reflects the combination of the dose for that specific event described in Section 15.0.3.7 with the CDE-based shine dose described in Section 15.0.3.6.4.
- (2) The iodine spike DBST is not an event, rather it serves as a bounding surrogate for design-basis loss of primary coolant into containment events described in Section 15.6.
- (3) The CDE is a beyond-design-basis special event.

Table 15.0-11: Primary Coolant Source Term

Nuclide	Primary Activity (Ci)
I-131	2.021E+00
I-132	9.379E-01
I-133	3.077E+00
I-134	5.518E-01
I-135	1.945E+00
Kr-85m	1.481E+00
Kr-85	2.698E+02
Kr-87	8.066E-01
Kr-88	2.348E+00
Xe-133	3.944E+02
Xe-135	1.055E+01

Table 15.0-12: Assumptions for Accident Airborne Effluent Release Point Characteristics for Offsite Receptors

Parameter	Value
Release location	Any point on Reactor Building or Turbine Generator Building wall
Release height	Ground level (0.0 meters)
Intake height	0.0 meters
Adjacent building cross-sectional area	Negligible (0.01 square meters)
Distance from source to receptor	101.2 meters

Table 15.0-13: Assumptions for Main Control Room χ/Q

Parameter	Value
Release location	Reactor Building wall
Release height	Ground level (0.0 meters)
Intake height	0.0 meters
Adjacent building cross-sectional area	Negligible (0.01 square meters)
Distance from source to receptor	47.4 meters

Table 15.0-14: Onsite and Offsite Receptor Variables

	Main Control Room	Technical Support Center	Exclusion Area Boundary	Low Population Zone
Atmospheric dispersion				
0-8 hours	Table 2.0-1	Table 2.0-1	Table 2.0-1	Table 2.0-1
8-24 hours	Table 2.0-1	Table 2.0-1	Table 2.0-1	Table 2.0-1
24-720 hours	Table 2.0-1	Table 2.0-1	Table 2.0-1	Table 2.0-1
Breathing Rates (sec/m³)				
0-8 hours	3.50E-04	3.50E-04	3.50E-04	3.50E-04
8-24 hours	3.50E-04	3.50E-04	1.80E-04	1.80E-04
24-720 hours	3.50E-04	3.50E-04	2.30E-04	2.30E-04
Occupancy factor %				
0-24 hours	100	100	N/A	N/A
24-96 hours	60	60	N/A	N/A
96-720 hours	40	40	N/A	N/A

Table 15.0-15: 10x Limiting Accident Radiation Post-Filtration Monitor Analytical Limits

Principal Radionuclides Measured	Value (μCi/cc)
Noble gas (Kr-85)	3.73E-03
Noble gas (Xe-133)	6.06E-02
Particulate (Cs-137)	1.27E-07
Iodine (I-131)	7.10E-06

Table 15.0-16: Main Control Room Parameters

Parameter	Value
MCR Envelope Volume	48,929 ft ³
MCR Emergency Duration	72 hr
MCR Emergency Flow Rate	107.33 cfm
MCR Normal Flow Rate	455 cfm
MCR Unfiltered Ingress/Egress	5 cfm
MCR Unfiltered Inleakage	10 cfm

Table 15.0-17: Technical Support Center Parameters

Parameter	Value
TSC normal flow rate	700 cfm
TSC unfiltered ingress/egress	10 cfm
TSC unfiltered inleakage	10 cfm
TSC volume	84,291 ft ³

Table 15.0-18: Margin to Critical Boron Concentration for Emergency Core Cooling System Cooldown Events

Event (Time in Cycle)	Minimum Margin to Critical Boron Concentration (ppm) ⁽¹⁾	Core Concentration at Time of Minimum Margin (ppm)	Time of Minimum Margin to Critical (seconds)
Injection line break in CNV, RVV fails (BOC)	514	1014	31
Injection line break in CNV, RVV fails (MOC)	538	589	31
Injection line break in CNV, RVV fails (EOC)	203	5	1
High point vent break outside CNV (EOC)	102	26	37,070
Reactor component cooling water pipe break (EOC)	25	128	118,600

(1) For cases with the margin to critical boron concentration greater than the core boron concentration, criticality can not be achieved by boron dilution without other positive reactivity addition.

Table 15.0-19: Margin to Boron Precipitation Limit for Emergency Core Cooling System Cooldown Events

Event	Minimum Margin to Boron Precipitation Limit (ppm)	Maximum Core Concentration (ppm)	Time of Minimum Margin (seconds)
High point vent break outside CNV	5416	1890	27,890
Inadvertent RVV opening	6128	9485	232,980

Table 15.0-20: Limiting Collapsed Liquid Level above Top of Active Fuel during Emergency Core Cooling System Cooldown

Event	Minimum Collapsed Liquid Level above Top of Fuel (ft)	Time of Minimum Level (seconds)
Discharge line break inside CNV	3.13	49,295
Discharge line break outside CNV	1.57	59,930
Injection line break inside CNV	3.12	46,385
Injection line break outside CNV	1.81	71,975
High point vent break line inside CNV	3.18	45,395
High point vent break line outside CNV	3.40	32,165
SGTF	0.23	69,740
Inadvertent RRV	3.17	47,300
Inadvertent RVV	3.17	42,905
Inadvertent RVV with loss of AC and EDAS power - 24 hours after event	4.00	N/A
Inadvertent RVV with loss of AC and EDAS power - 48 hours after event	4.44	N/A
Inadvertent RVV with loss of AC and EDAS power - 72 hours after event	5.24	N/A

Table 15.0-21: Maximum Temperature Case Containment Pressure and Moderator Temperature Statepoint Results

Time after Event Initiation (hr)	Containment Pressure (psia)	Moderator Temperature (°F)
12	72	311
24	62	301
48	54	293
72	49	288

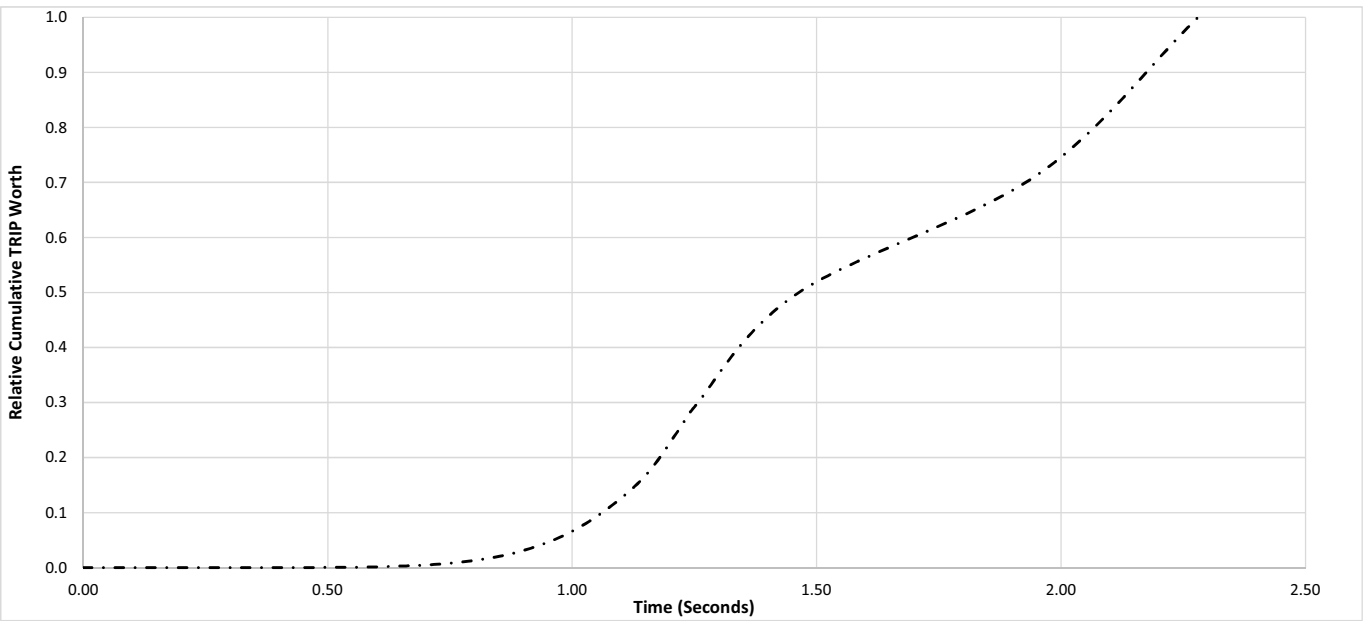
Figure 15.0-1: Reactor Trip Reactivity versus Time after Trip

Figure 15.0-2: Schematic of a Typical RADTRAD Model Nodalization

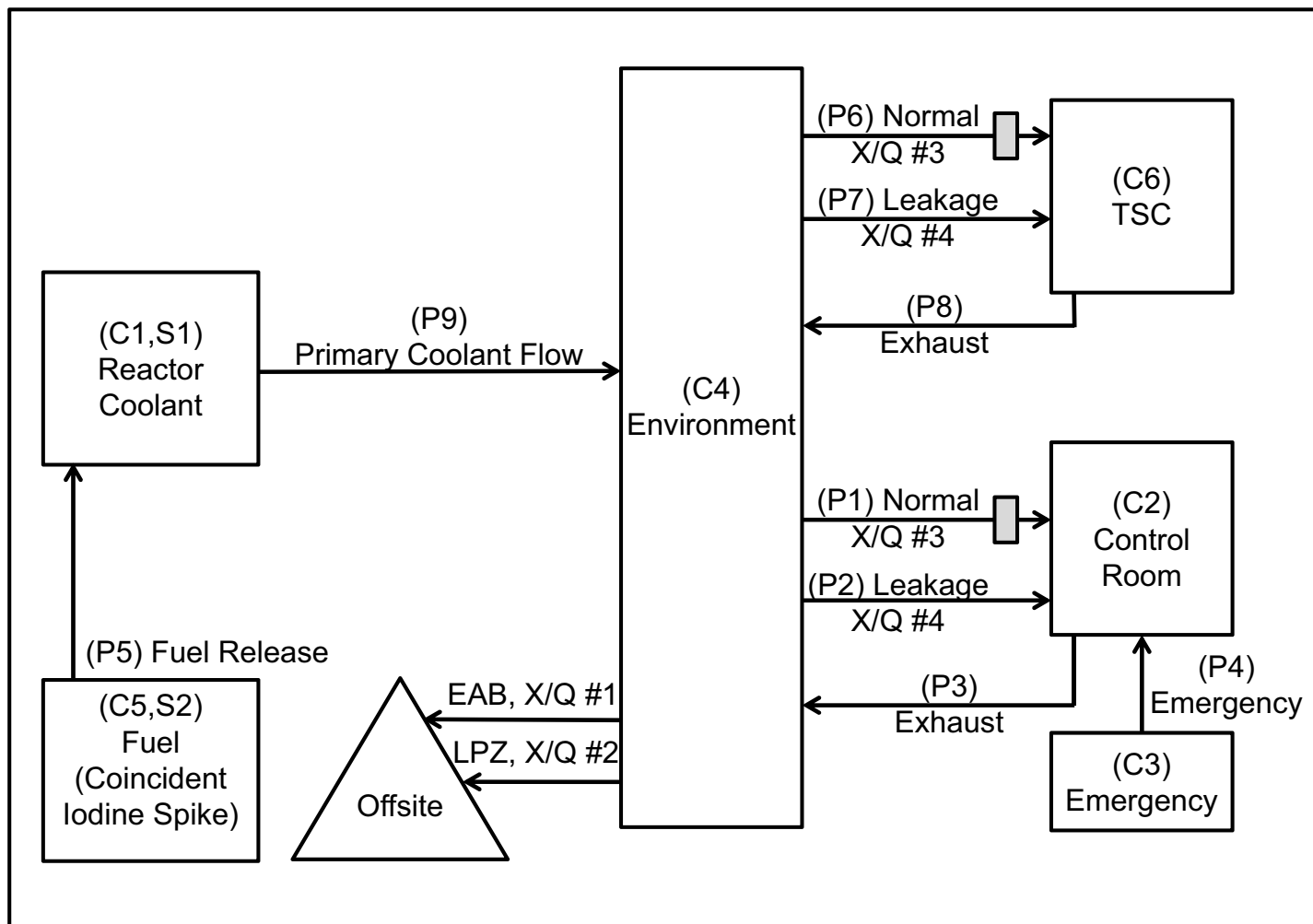


Figure 15.0-3: Collapsed Liquid Level for Inadvertent Operation of Emergency Core Cooling System Events

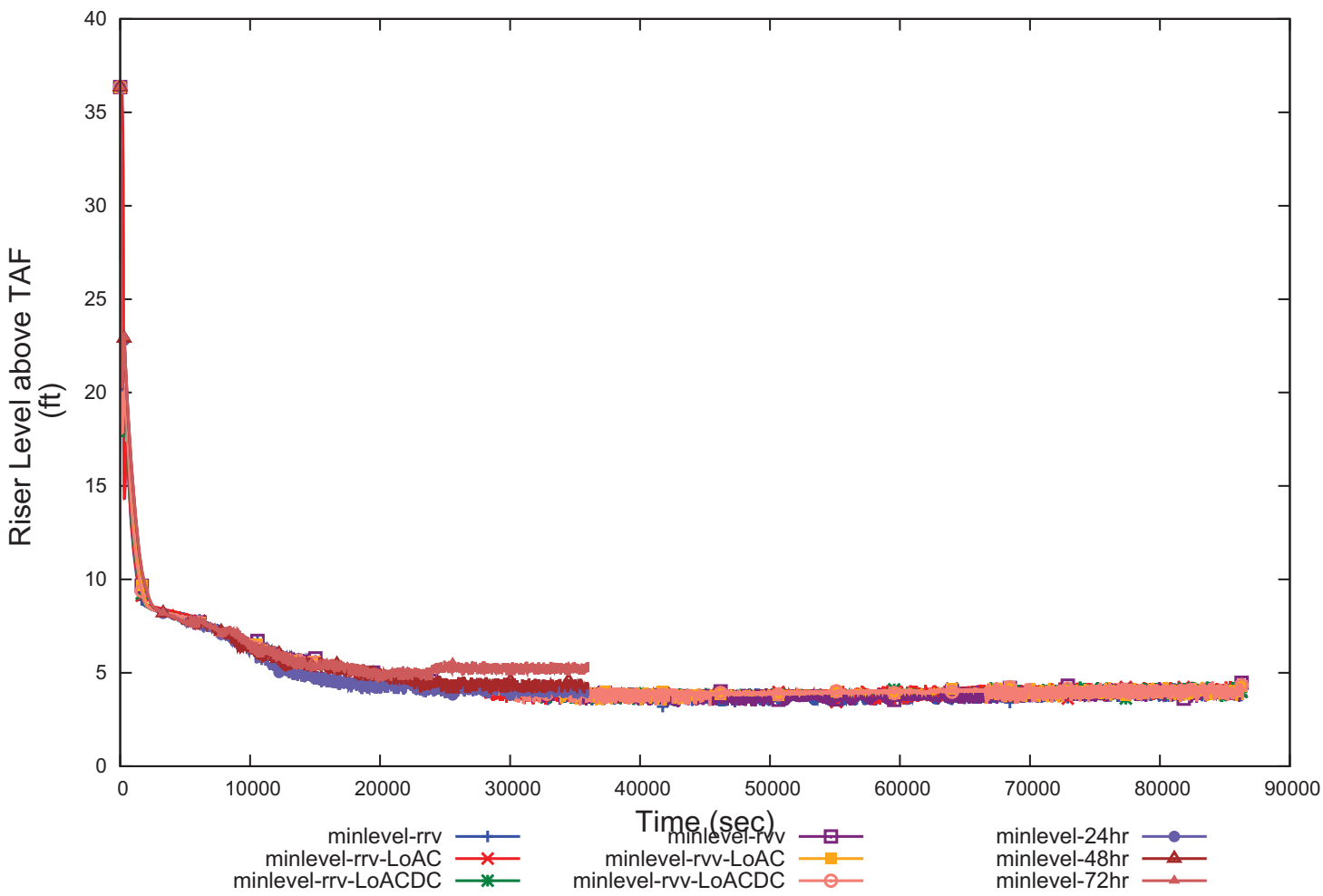
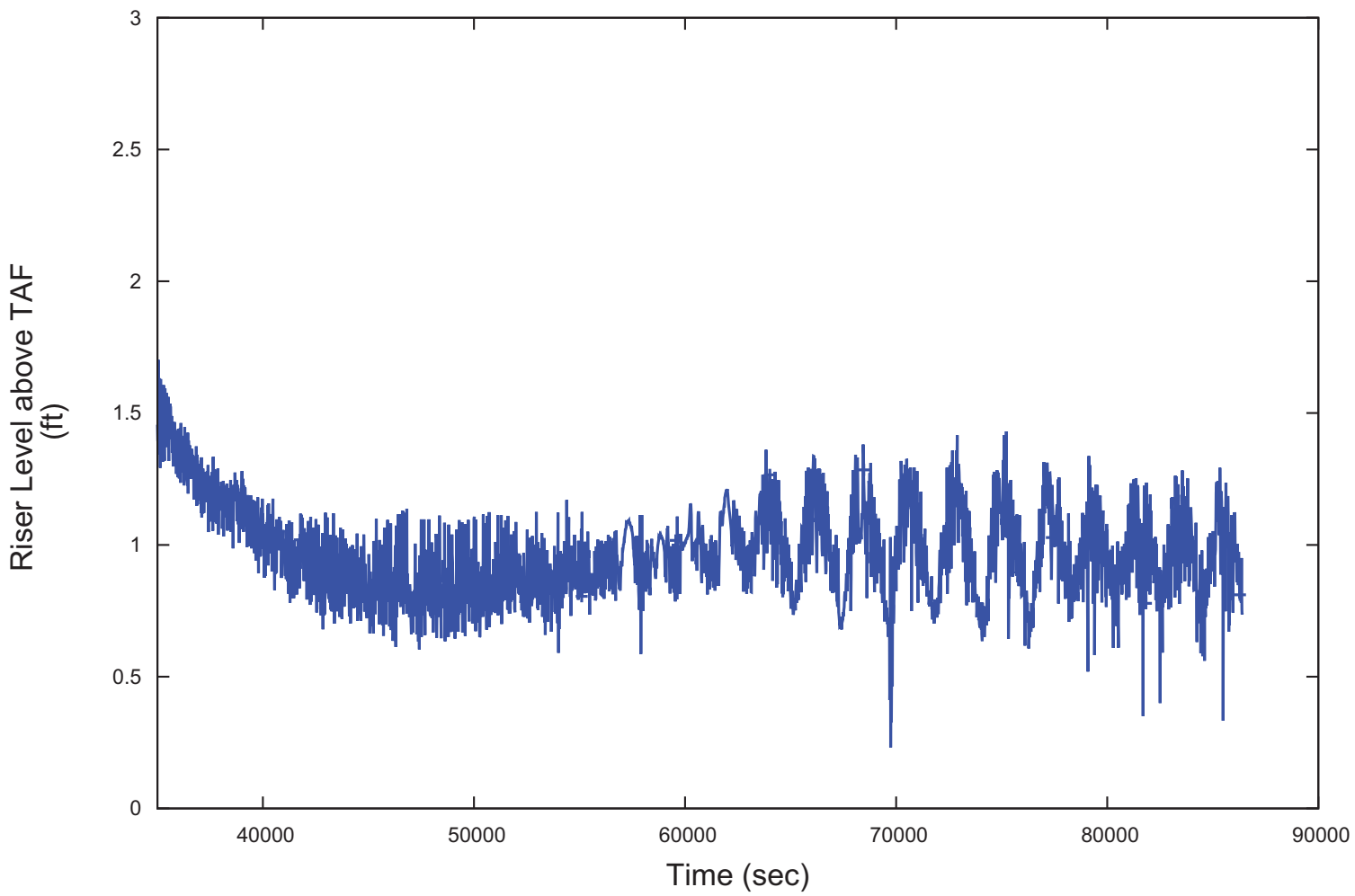


Figure 15.0-4: Collapsed Liquid Level for Limiting Steam Generator Tube Failure Case

15.1 Increase in Heat Removal by the Secondary System

There are several events that could result in an increase in primary heat removal by the secondary system. There is also a NuScale Power Plant US460 standard design-specific event that causes an overcooling of the primary system due to a loss of containment vacuum or flooding of containment. These events are classified as anticipated operational occurrences (AOOs), infrequent events, or accidents as shown in Table 15.0-1. An increase in primary heat removal results in an increase in core reactivity, which leads to an increase in power. The increase in reactor power leads to a decrease in the minimum critical heat flux ratio (MCHFR). The increase in heat removal events do not present a challenge to primary or secondary pressure limits. Sensitivities to maximize primary or secondary pressure are not run for the AOOs in this section. Therefore, figures of merit are only provided for the limiting MCHFR case for this class of events. The maximum reactor coolant system (RCS) and steam generator (SG) pressures from the cases analyzed are provided in the results summary table for each section.

15.1.1 Decrease in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

A decrease in feedwater (FW) temperature could be caused by a failure in the feedwater system (FWS). A lower FW temperature increases the heat removal from the primary system, leading to a higher moderator density. As the RCS cools, if the reactivity control system is in an automatic mode, it inserts positive reactivity by pulling the regulating control rods from the core in an attempt to maintain RCS temperature. The increase in reactor power due to the insertion of positive reactivity results in a decrease in the MCHFR.

A decrease in FW temperature is classified as an AOO as discussed in Section 15.0.

15.1.1.2 Sequence of Events and Systems Operation

The sequence of events for the limiting decrease in FW temperature case is provided in Table 15.1-1.

Unless specified below, the analysis of a decrease in FW temperature assumes the plant control systems and engineered safety features (ESFs) perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a decrease in FW temperature event.

Normally, as the colder FW enters the SG, the FW controls reduce FW flow, helping to mitigate the overcooling event. This event is conservatively analyzed with the FW controls disabled, forcing a constant FW flow rate, and FW control is not credited to mitigate the event.

The reactivity control system is assumed to be in automatic mode. Operator action is not credited for regulating control rod movement or increasing boron concentration. This control ensures the maximum reactivity insertion is reached

as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The module protection system (MPS) is credited to protect the plant in the event of a decrease in FW temperature. If the FW temperature were to drop to a level that causes a high enough power excursion, the MPS high power signal trips the reactor, preventing the reactor from reaching a power level where the acceptance criteria could be challenged. The following MPS signals protect the plant during a decrease in FW temperature:

- high power
- high main steam pressure
- low main steam superheat
- high power rate

Because of the cooling of the RCS during a decrease in FW temperature event, the coolant in the downcomer increases in density. This increase in density can affect the power level detection by the excore neutron detectors. In order to account for this effect, the high power trip is adjusted to account for downcomer coolant temperature.

In a decrease in FW temperature event that results in a reactor trip, closure of the feedwater isolation valves (FWIVs) and main steam isolation valves (MSIVs) through a secondary system isolation signal isolates the FW source, ending the overcooling event. The decay heat removal system (DHRS) is actuated as a result of high main steam pressure. Passive DHRS cooling is established and the event is terminated with the NuScale Power Module (NPM) in a stable condition.

There are no single failures that could make the limiting decrease in FW temperature MCHFR case more severe because any single active failure occurs after MCHFR occurs. Therefore, a single failure is not modeled in the case presented in this section, which demonstrates the limiting decrease in FW temperature with respect to MCHFR.

Normal alternating current (AC) power is assumed to be available for this event. A loss of AC power, either at event initiation or at reactor trip, is not a conservative condition for a decrease in FW temperature event because it de-energizes the FW pumps, reducing the severity of the overcooling event.

15.1.1.3 Thermal Hydraulic and Subchannel Analyses

15.1.1.3.1 Evaluation Models

The thermal hydraulic analysis of the plant response to a decrease in FW temperature is performed using NRELAP5. The NRELAP5 model is based on the design features of an NPM. The non-loss-of-coolant accident (non-LOCA) NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the subsequent subchannel critical heat flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.1.1.3.2 Input Parameters and Initial Conditions

The lowest FW temperature that could result from a failure in the FWS is 100 degrees F. This is the temperature of the FW before it passes through the FW heaters. Sensitivities on a spectrum of FW temperature reductions reveal the limiting case with respect to MCHFR is a rapid reduction to 100 degrees F.

The initial conditions used in the evaluation of the limiting decrease in FW temperature event result in a conservative calculation. Table 15.1-2 provides key inputs for the limiting decrease in FW temperature case. The following initial conditions are assumed in the analysis to ensure the results have sufficient conservatism.

- Initial power level is assumed to be 102 percent of rated thermal power to account for a 2-percent instrumentation uncertainty.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and using a bounding control rod drop rate.
- The most limiting end-of-cycle (EOC) reactivity feedback is used to provide a limiting power response.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation."

The results from the thermal-hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.

15.1.1.3.3 Results

The sequence of events for the limiting decrease in FW temperature case is provided in Table 15.1-1. Figure 15.1-1 through Figure 15.1-7 show the transient behavior of key parameters for the decrease in FW temperature event. A spectrum of FW temperature cases are analyzed and the limiting temperature decrease is a case in which the cooldown rate yields a reactor trip on high power. The limiting event initiates with a rapid decrease in FW temperature to the minimum possible temperature of 100 degrees F. The RCS response to the overcooling event begins once the cold FW front propagates through the secondary system piping and reaches the SG. This decrease in FW temperature leads to an increase in the heat removal rate from the RCS via the SG. During the overcooling phase of the transient, the RCS temperature decreases while the RCS density increases. The colder, denser

RCS causes the regulating control rod bank to withdraw in an attempt to maintain RCS temperature. The withdrawal of the control rods causes a positive reactivity insertion that increases reactor power. If the regulating control rod bank is disabled, a similar power response is driven by moderator feedback instead.

The decrease in reactor coolant temperature causes the primary coolant volume to shrink, which initially reduces the pressure of the RCS. However, as core power increases, the RCS pressure begins to rise. The rise in reactor power initiates a high power reactor trip. The peak RCS pressure occurs just after the reactor trip. High main steam pressure actuates the DHRS valves to open. The FWIVs and MSIVs close, isolating the SGs from the rest of the secondary system.

Steam generator pressure does not change significantly during the initial phase of the transient. However, after the reactor trip SG pressure increases until secondary system isolation (SSI) actuation. After DHRS and SSI actuation, the closure of the FWIVs and MSIVs causes additional pressurization of the SG. Steam generator pressure increase resulting from main steam isolation is expected and is not a direct consequence of the decrease in FW temperature event itself. The maximum secondary pressure is reached after main steam and FW isolation.

The critical heat flux ratio (CHFR) decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the CHF analysis limit. The MCHFR for the limiting decrease in FW temperature does not violate the CHF analysis limit.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor trip causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures reequilibrate. After DHRS actuation, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of the DHRS, and stabilization of RCS flow demonstrate the plant response to a decrease in FW temperature, and a return to a stable condition with no operator actions.

15.1.1.4 Radiological Consequences

The normal leakage-related radiological consequences of this event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.1.1.5 Conclusions

The Design Specific Review Standard (DSRS) acceptance criteria are met for the limiting case.

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

- The pressure responses in the reactor pressure vessel (RPV) and in the main steam system (MSS) are less severe than those of the limiting AOOs presented in Section 15.2, Decrease in Heat Removal by the Secondary System. Therefore, this acceptance criterion is met for the decrease in FW temperature event. The maximum pressure values for the cases analyzed are shown in Table 15.1-3.
- 2) Fuel cladding integrity shall be maintained by ensuring the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.
- The MCHFR for this event is above the CHF analysis limit as shown in Table 15.1-3. Therefore, this acceptance criterion is met.
- 3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
- The analysis presented for this event shows that stable DHRS cooling is reached, and the acceptance criterion is met.
- 4) The guidance provided in RG 1.105 can be used to analyze the impact of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of General Design Criteria (GDC) 10, 13, 15, 20, and 26.
- The instrument spans and setpoints discussed in Section 15.1.1.3, address the guidance in RG 1.105.
- 5) The most limiting plant systems single failure shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53, "Application of the Single-Failure Criterion to Safety Systems."
- There is no limiting single failure that could occur during a decrease in FW temperature event that could result in more severe conditions with respect to the acceptance criteria.

15.1.2 Increase in Feedwater Flow

15.1.2.1 Identification of Causes and Accident Description

An increase in FW flow event could occur because of a failure in the FWS. For the NuScale Power Plant US460 standard design, the increased FW flow increases heat transfer from the primary side, which results in a power increase due to the moderator density increase. As the RCS cools, if the reactivity control system is in an automatic mode, it inserts positive reactivity by pulling the regulating control rods from the core in an attempt to maintain RCS temperature. The increase in reactor power due to the insertion of positive reactivity results in a decrease in the MCHFR.

An increase in FW flow is classified as an AOO as discussed in Section 15.0.

15.1.2.2 Sequence of Events and Systems Operation

The sequence of events for an increase in FW flow event is provided in Table 15.1-4 for the limiting MCHFR case and Table 15.1-5 for the limiting SG overfill case.

Unless specified below, the analysis of an increase in FW flow event assumes the plant control systems and the ESFs perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an increase in FW event.

The FWS could malfunction and increase FW flow by increasing the speed of normally operating FW pumps, turning on a FW pump, opening a FW regulating valve (FWRV), or opening a DHRS valve at low RCS power. The inadvertent opening of a DHRS valve at low RCS power is addressed by the analysis in Section 15.2.9. In order to bound the possible FW flow increase scenarios, a spectrum of FW flow increases are analyzed to demonstrate that limiting conditions for MCHFR are reached. For the limiting MCHFR case, the steam outlet is modeled as a constant steam pressure, allowing the steam flow to increase providing an increase in SG heat transfer in response to the increase in FW flow.

An increase in FW flow can lead to overfilling the SG. For the SG overfill scenario, the maximum FW pump flow curve is used, as this results in the greatest increase in secondary inventory and the highest SG levels. Increased SG levels degrade DHRS condensers by increasing DHRS condenser static head liquid levels and causing a reduction in heat transfer surface area. The single failure of an FWIV leads to the most limiting SG overfill condition.

Operator action is not credited for regulating control rod movement or increasing boron concentration, which ensures the maximum reactivity insertion is reached as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The MPS is credited to protect the plant in the event of an increase in FW flow. If the FW flow increases to a level that causes a high enough power excursion, the MPS high power signal trips the reactor, preventing the reactor from reaching a power level at which the acceptance criteria could be challenged. The following MPS signals protect the plant during an increase in FW flow:

- high power
- high main steam superheat
- high main steam pressure
- low main steam superheat
- high pressurizer (PZR) pressure

Because of the cooling of the RCS during an increase in FW flow event, the coolant in the downcomer increases in density. This increase in density can affect

the power level detection by the excore neutron detectors. In order to account for this effect, the high power rate trip is not credited in the analyses, and the high power trip is adjusted to account for downcomer coolant temperature.

In increase in FW flow events that result in a reactor trip, the subsequent actuation of SSI is credited for mitigation of changes to the secondary system inventory and the subsequent actuation of the DHRS is credited with maintaining reactor cooling. The MPS signals credited for SSI actuation are high main steam superheat, low main steam superheat, high main steam pressure and high PZR pressure. The MPS signals credited for DHRS actuation are high PZR pressure and high main steam pressure.

There are no single active failures that could occur during the limiting increase in FW flow event that could result in more severe conditions with respect to the acceptance criteria. In order to evaluate the potential for an increase in FW flow to overfill the SG, a case is analyzed with the failure of one FWIV to close. This failure contributes to SG overfill and impacts DHRS performance.

Normal AC power is assumed to be available for this event. A loss of AC power, either at event initiation or at reactor trip, is not a conservative condition for an increase in FW flow event because FW is lost.

15.1.2.3 Thermal Hydraulic and Subchannel Analyses

15.1.2.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to an increase in FW flow is performed using NRELAP5. The NRELAP5 model is based on the design features of an NPM. The non-LOCA transient modifications to the NRELAP5 model are discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.1.2.3.2 Input Parameters and Initial Conditions

The initiating amount of FW flow increase is varied in sensitivity calculations, up to the maximum FW pump curve, to find the limiting increase. The FW flow is assumed to increase linearly from its initial steady state value to its final value over a time span of 0.1 seconds.

The initial conditions used in the evaluation of the limiting increase in FW flow event result in a conservative calculation. Table 15.1-6 provides key inputs for the limiting cases for the increase in FW flow event. The following initial

conditions are assumed in the analysis to ensure the results have sufficient conservatism for the limiting MCHFR case.

- Initial power level is assumed to be 102 percent of rated thermal power to account for a 2-percent measurement uncertainty.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The most limiting EOC reactivity feedback is used to provide a limiting power response.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.

15.1.2.3.3 Results

The sequence of events for the limiting MCHFR increase in FW flow event is provided in Table 15.1-4. Figure 15.1-8 through Figure 15.1-16 show the transient behavior of key parameters for the limiting MCHFR case. A spectrum of FW flow increase cases are analyzed and the limiting FW flow increase is a near instantaneous 15-percent increase of normal FW flow.

An FWS malfunction that causes an increase in FW flow results in an unplanned overcooling of the RCS. The subsequent decrease in RCS temperature increases core reactivity because of moderator feedback, which raises reactor power. Decreasing average RCS temperature also prompts the control rod drive controller to withdraw the regulating bank from the core if automatic control is enabled. Rising reactor power potentially causes CHF conditions to develop in the core. The FW flow increase causes RTS actuation on low main steam superheat, high main steam pressure, high PZR pressure, or high power. Closure of the FWIVs and MSIVs on SSI isolates the SGs from the FW source, ending the overcooling event. The DHRS actuates on the high PZR pressure or high main steam pressure signals. Core decay heat drives natural circulation, which transfers thermal energy from the RCS to the reactor pool via the DHRS. Passive DHRS cooling is established and the transient is terminated with the NPM in a safe, stable condition.

Steam generator overfill cases are evaluated with initial conditions including primary flow and RCS temperature biased to maximize SG level. Boundary conditions such as pool temperature are biased to minimize DHRS heat removal, high decay heat is assumed, the maximum FW pump curve is modeled, the maximum containment isolation valve closure time is assumed, and single failure of one FWIV to close is assumed. The SG level calculated during an increase in FW flow demonstrates the SG does not overfill as shown in Figure 15.1-18. Evaluation of cases biased for SG overfill demonstrate

adequate DHRS heat removal capability is available under these conditions as shown in Figure 15.1-17.

The MCHFR for the limiting increase in FW flow event does not fall below the CHF analysis limit.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor trip causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures reequilibrate. The reactor trip, subsequent actuation of SSI and the DHRS, and stabilization of RCS flow demonstrate the plant response to increase in FW flow, and a return to a stable condition with no operator actions.

15.1.2.4 Radiological Consequences

The normal leakage-related radiological consequences of this event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.1.2.5 Conclusions

The DSRS acceptance criteria are met for the limiting case.

- 1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
 - The pressure responses in the RPV and in the MSS are less severe than those of the limiting AOOs presented in Section 15.2. Therefore, this acceptance criterion is met for the increase in FW flow event. The maximum pressure values for the cases analyzed are shown in Table 15.1-7.
- 2) Fuel cladding integrity shall be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit.
 - The MCHFR for this event is above the CHF analysis limit as shown in Table 15.1-7. Therefore, this acceptance criterion is met.
- 3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
 - The analysis presented for this event shows that stable DHRS cooling is reached, and the acceptance criterion is met.
- 4) The guidance provided in RG 1.105 can be used to analyze the impact of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDC 10, 13, 15, 20, and 26.
 - The instrument spans and setpoints discussed in Section 15.1.2.3 address the guidance in RG 1.105.

- 5) The most limiting plant systems single failure shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.
 - There is no limiting single failure that could occur during an increase in FW flow event that could result in more severe conditions with respect to the acceptance criteria. Single failure of an FWIV is evaluated for impact on SG overfill and DHRS performance.

15.1.3 Increase in Steam Flow

15.1.3.1 Identification of Causes and Accident Description

An increase in steam flow event could occur because of a spurious opening of the turbine bypass valve. The inadvertent opening of the main steam safety valves (MSSVs), described in Section 15.1.4, could also cause an increase in steam flow. The opening of an MSSV creates a similar plant response to the inadvertent opening of the turbine bypass valve and is bounded by the analysis presented in this section. For the NuScale Power Plant US460 standard design, the increased steam flow increases heat transfer from the primary side, which in turn results in a power increase due to the moderator density increase. As the RCS cools, the reactivity control system inserts positive reactivity by pulling the regulating control rods from the core in an attempt to maintain RCS temperature. The insertion of positive reactivity results in an increase in reactor power and a decrease in the MCHFR.

An increase in steam flow is classified as an AOO as discussed in Section 15.0.

15.1.3.2 Sequence of Events and Systems Operation

The sequence of events for the limiting increase in steam flow case is provided in Table 15.1-8.

Unless specified below, the analysis of an increase in steam flow event assumes the plant control systems and ESFs perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an increase in steam flow event.

The maximum possible increase in steam flow is a 100-percent increase of normal full power steam flow, following a spurious opening of the turbine bypass valve. In order to bound the maximum steam flow increase, a maximum increase of 125 percent of normal steam flow is analyzed. However, this condition results in rapid reactor trip system (RTS) actuation before limiting CHF conditions can develop. Less extreme increases in steam flow delay the RTS actuation, given the slower power response and slower SG depressurization. Smaller steam flow increases allow more time for the system thermal response to develop, which is more limiting in terms of MCHFR. The FWS is not credited with mitigating the overcooling effect of an increase in steam flow. These assumptions maximize the overcooling event by increasing the available source of secondary coolant.

Operator action is not credited for regulating control rod movement or increasing boron concentration, which ensures the maximum reactivity insertion is reached as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The MPS is credited to protect the plant in the event of an increase in steam flow. When the steam flow increases to a level that causes a high enough power excursion, the MPS high power signal trips the reactor, preventing the reactor from reaching a power level at which the acceptance criteria could be challenged. A 25-percent flow increase yields the limiting combination of reactor power and hot leg temperature to minimize CHF. The following MPS signals protect the plant during an increase in steam flow:

- high power
- high power rate
- high main steam superheat
- high PZR pressure
- low main steam superheat
- low main steam pressure

Because of the cooling of the RCS during an increase in steam flow event, the coolant in the downcomer increases in density. This increase in density can affect the power level detection by the excore neutron detectors. In order to account for this effect, the high power trip is adjusted to account for downcomer coolant temperature.

In increased steam flow events that result in a reactor trip, the subsequent actuation of SSI and the DHRS are credited with maintaining reactor cooling. The MPS signals credited for SSI actuation are high PZR pressure, low main steam pressure, high main steam superheat, and low main steam superheat. The MPS signal credited for DHRS actuation is high PZR pressure.

There are no single failures that could result in a more severe outcome of the limiting increase in steam flow event with respect to the acceptance criteria. Two possible failures that could affect the transient are the failure of one of the MSIVs or one of the FWIVs. However, these single failures could only occur after secondary side isolation or DHRS actuation, which occurs coincident with or after the reactor trip, and the MCHFR has already occurred. Therefore, neither single failure is modeled in the case presented in this section.

Normal AC power is assumed to be available for this event. A loss of AC power, either at event initiation or at reactor trip, is not a conservative condition for an increase in steam flow event because FW is lost, which reduces the overcooling event.

15.1.3.3 Thermal Hydraulic and Subchannel Analyses**15.1.3.3.1 Evaluation Model**

The thermal hydraulic analysis of the plant response to an increase in steam flow is performed using NRELAP5. A description of the NRELAP5 model is provided in Section 15.0.2. The NRELAP5 model is based on the design features of an NPM. The non-LOCA transient modifications to the NRELAP5 model are discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.1.3.3.2 Input Parameters and Initial Conditions

As discussed in Section 15.1.3.2, the limiting increase in steam flow is an increase of 25 percent normal steam flow. Steam flow is assumed to linearly increase from its initial steady state value to its final value over a time span of 0.1 seconds.

The initial conditions used in the evaluation of the limiting increase in steam flow event result in a conservative calculation. Table 15.1-9 provides key inputs for the increase in steam flow limiting MCHFR case. The following initial conditions are assumed in the analysis to ensure the results have sufficient conservatism.

- Initial power level is assumed to be 102 percent of rated thermal power to account for a 2-percent measurement uncertainty.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The most limiting EOC reactivity feedback is used to provide a limiting power response.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel model is discussed in Section 15.0.2.

15.1.3.3.3 Results

An increase in steam flow increases secondary system heat removal causing RCS temperature to decrease. In response, the regulating bank controller is assumed to withdraw the regulating bank to compensate for positive moderator feedback, and reactor power increases. As power and subsequently core outlet temperature increase, margin to the CHF limit decreases.

Figure 15.1-19 through Figure 15.1-28 show the transient behavior of key parameters for an increase in steam flow event. A spectrum of steam flow increase cases is analyzed. In general, low steam flow increases are limiting for MCHFR because hot leg temperature and pressure increase to reactor trip actuation limits slower given the slow increase in power. When the steam flow increase becomes greater in magnitude, RTS actuation occurs earlier on high reactor power or low main steam pressure signals. These cases are not limiting because hot leg temperature remains at a relatively low value.

The limiting event initiates with an increase in steam flow by approximately 25 percent of full power steam flow in 0.1 seconds. The FW flow rate is allowed to increase with the steam flow to maximize the overcooling effect. The RCS response to the overcooling event begins once the steam flow through the SGs begins to increase. This increase in steam flow leads to an increase in the heat removal rate from the RCS via the SGs. During the overcooling phase of the transient, the RCS temperature steadily decreases, while the RCS density increases. The colder, denser RCS causes the regulating control rod bank to withdraw in an attempt to maintain RCS temperature. The withdrawal of the control rods causes a positive reactivity insertion that increases reactor power. This increase leads to an RTS actuation on high power.

The decrease in reactor coolant temperature causes the primary coolant volume to shrink, which initially reduces the pressure of the RCS. However, as core power increases, the RCS pressure begins to rise. The hot leg temperature rises in response to the increase in reactor power. The peak RCS pressure occurs around the time of the reactor trip.

An SSI actuation causes closure of the FWIVs and MSIVs, which causes pressurization of the SG. Steam generator pressure increase resulting from main steam isolation is expected and is not a direct consequence of the increase in steam flow. The maximum secondary pressure is reached after main steam and FW isolation and DHRS actuation.

The CHFR decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the CHF analysis limit. The MCHFR for the limiting increase in steam flow case does not violate the CHF analysis limit.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor trip causes a rapid decrease in flow as the

heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures reequilibrate. The reactor trip, subsequent SSI and actuation of the DHRS, and stabilization of RCS flow demonstrate the plant response to an increase in steam flow, and a return to a stable condition with no operator actions.

The highest RCS pressure of the analyzed cases occurs when a 50-percent increase in steam flow is evaluated. This case results in an earlier reactor trip on high power rate and thus has no effect on the limiting MCHFR case.

The highest secondary pressure of the analyzed cases results from a 10 percent increase in steam flow. The event progression terminates on a low steam line superheat signal, which trips the reactor and isolates the secondary system at the same time.

15.1.3.4 Radiological Consequences

The normal leakage-related radiological consequences of this event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.1.3.5 Conclusions

The DSRS acceptance criteria are met for the limiting case.

- 1) Pressure in the RCS and in the MSS should be maintained below 110 percent of the design values.
 - The pressure responses in the RPV and in the MSS are less severe than those of the limiting AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met for the increase in steam flow event. The maximum pressure values for the cases analyzed are shown in Table 15.1-10.
- 2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit.
 - The MCHFR for this event is above the CHF analysis limit as shown in Table 15.1-10. Therefore, this acceptance criterion is met.
- 3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
 - The analysis presented for this event shows that stable DHRS cooling is reached, and the acceptance criterion is met.
- 4) The guidance provided in RG 1.105 can be used to analyze the impact of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDC 10, 13, 15, 20, and 26.
 - The instrument spans and setpoints discussed in Section 15.1.3.3 address the guidance in RG 1.105.

- 5) The most limiting plant systems single failure shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.
 - There is no limiting single failure that could occur during an increase in steam flow event that could result in more severe conditions with respect to the acceptance criteria as discussed in Section 15.1.3.2.

15.1.4 Inadvertent Opening of Steam Generator Relief or Safety Valve

This section is typically used to provide an analysis for an inadvertent opening of an SG relief or an MSSV. The NuScale Power Plant US460 standard design does not have an SG relief valve, but does have two MSSVs. The event is initiated by the spurious opening of one of the two main MSSVs, which are located downstream of the secondary MSIVs. The MSSVs are sized to accommodate 100 percent of the full power steam flow. As both valves are required to meet this flow requirement, the spurious opening of one MSSV following a mechanical failure yields a steam flow increase less than 100 percent of the flow at full power conditions. A spurious opening of the turbine bypass valve yields a similar system response as it is also located downstream of the secondary MSIVs. However, the increase in steam flow due to a spurious opening of the turbine bypass valve could result in a 100-percent increase in steam flow. Therefore, the spectrum of possible steam increases because of a full or partial turbine bypass valve opening covers and bounds the steam increases because of an inadvertent opening of an MSSV. The analysis of a limiting increase in steam flow is presented in Section 15.1.3.

15.1.5 Steam Piping Failures Inside and Outside of Containment

15.1.5.1 Identification of Causes and Accident Description

A steam line break (SLB) event for the NuScale Power Plant US460 standard design could range from a small break to a double-ended rupture of the main steam line. This event could occur inside or outside of the containment vessel (CNV). A spectrum of SLB locations with varied core and plant conditions are analyzed to determine the scenarios with the most severe results.

An SLB inside the CNV increases the pressure inside containment, reaching the high containment pressure MPS setpoint. The high containment pressure signal actuates RTS, isolates the CNV and chemical and volume control system, deenergizes PZR heaters, and actuates SSI and DHRS. The break flow decreases because of SG depressurization and increasing CNV backpressure until dryout from FW isolation. The containment pressure is sensitive to any SLB size, so the MPS detects the break earlier than a comparable break outside of containment. A spectrum of breaks inside containment is evaluated to ensure containment pressure is acceptable. The peak containment pressure remains below the design limit for all postulated events, as shown in Section 6.2, Containment Systems. Aside from containment pressure, the plant conditions for an SLB inside containment are bounded by the analysis presented in this section for an SLB outside of containment.

An SLB outside the CNV causes an increase in steam flow event that could either cause a low SG pressure signal or a high core power trip because of the reactor power response from the decreased RCS temperature. The break flow is stopped by closure of the MSIV and depressurization of the steam system piping. The largest SLB outside containment could occur from a double-ended rupture of the portion of the main steam line located outside the CNV. However, a double-ended rupture results in a low steam pressure signal that occurs earlier in the transient than for a small SLB outside containment. A small SLB outside containment provides the longest event progression before detection by the MPS. A smaller break can result in a significant delay in detection time relative to a larger break, potentially producing more limiting primary and secondary pressures, MCHFR, and integral mass and energy release.

A significant steam piping failure is not expected to occur during the life of the plant, so the event is classified as a postulated accident. A smaller or secondary piping failure has a higher probability of occurrence, so the SLB accident is evaluated against the AOO acceptance criteria for MCHFR and primary and secondary pressures.

15.1.5.2 Sequence of Events and Systems Operation

There are separate SLB cases outside containment that are limiting with respect to primary pressure, SG pressure, and MCHFR. The SLB event is a secondary depressurization event. The sequences of events for the limiting SLB cases are provided in Table 15.1-11, Table 15.1-12, and Table 15.1-13.

Unless specified below, the analysis of an SLB event assumes the plant control systems and ESFs perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a steam piping failure.

The FWS is not credited with mitigating the overcooling effect of an increase in steam flow. The FW flow increases to the maximum pump capacity as a function of flow and pressure for two trains of FW aligned in parallel. The analysis of an increase in steam flow event assumes the FW pump speed remains constant. These simplifying assumptions maximize the overcooling event by increasing the available source of secondary coolant.

Operator action is not credited for regulating control rod movement or increasing boron concentration, which ensures the maximum reactivity insertion is reached as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The MPS is credited to protect the plant in the event of an SLB. The following MPS signals protect the plant during an SLB:

- high power
- low main steam pressure
- low PZR level

- high CNV pressure
- high main steam superheat
- low main steam superheat
- high RCS hot temperature
- high PZR pressure
- low PZR pressure
- high main steam pressure
- low AC voltage

Because of the cooling of the RCS during an SLB, the coolant in the downcomer increases in density. This increase in density can affect the power level detection by the excore neutron detectors. In order to account for this effect, the high power rate trip is not credited in the analyses, and the high core power trip is adjusted to account for downcomer coolant temperature.

The DHRS is a safety-related system credited to actuate and mitigate the effects of this transient. The redundancy and passive nature of the DHRS ensure the system performs its intended function despite a single failure. The operation of a DHRS train is challenged when an SLB is located on the steam line downstream of the MSIV, but upstream of the secondary MSIV, with a single failure of the MSIV, or when an SLB occurs inside containment. These scenarios prevent one train of the DHRS from functioning by preventing isolation of one SG train.

The typical limiting single failure for an SLB inside containment is a failure of the FWIV to close on the impacted secondary train. In this scenario, the FWRV is credited to perform the isolation function. However, the FWRV has a longer closure time than the FWIV. This longer time maximizes the runout of the FW pumps, which increases the mass and energy released and increases containment pressure. This scenario also increases the energy removed by the break making the overcooling event worse.

The typical limiting single failure for an SLB outside of containment is the failure of the MSIV to close in the affected line. If the break location is between the MSIV and the secondary MSIV, a failure of the MSIV to isolate on demand results in a complete blowdown of the affected SG train. If the break location is downstream of the secondary MSIV, the single failure of either the MSIV or the secondary MSIV is inconsequential because the valves have the same closure time.

Normal AC power is assumed to be available for the limiting SLB cases. A loss of AC power, either at event initiation or at reactor trip, is not a conservative condition because FW is lost, which reduces the severity of the overcooling event.

Because the low AC voltage signal is credited with DHRS actuation in some of the non-limiting cases, a potential loss of the normal direct current power system (EDNS) is assessed based on the discussion in Section 15.0.0. However, the loss

of EDNS is determined to have no impact on the results of those cases. The limiting SLB cases assume EDNS is available.

Section 15.0.0 describes the impact of a loss of the augmented direct current power system (EDAS), which includes the opening of the reactor vent valves. The limiting SLB cases assume EDAS is available, consistent with the discussion in Section 15.0.0. However, an SLB inside containment with a loss of EDAS creates a temporary pathway for primary coolant to be released through the open reactor vent valves into containment and then out of containment through the broken steam line. This potential release pathway is isolated by the MSIV closure which also occurs on a loss of EDAS. The amount of primary coolant release is therefore inherently limited by the MSIV closure time and is bounded by the assumed primary coolant releases analyzed in Section 15.0.3. A single failure of the MSIV in this scenario is inconsequential because the secondary MSIV has the same closure time.

15.1.5.3 Thermal Hydraulic and Subchannel Analyses

15.1.5.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to an SLB is performed using NRELAP5. A description of the NRELAP5 model is provided in Section 15.0.2. The NRELAP5 model is based on the design features of an NPM. The non-LOCA transient modifications to the NRELAP5 model are discussed in Section 15.0.2. The steam piping breaks are modeled in NRELAP5 as valves that instantly open at transient initiation and have a sudden infinite expansion loss. This modeling is appropriate for an SLB because the break vents either to a relatively large CNV or to an even larger Reactor Building or Turbine Building. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

The radiological analysis of the SLB event is provided in Section 15.0.3.

15.1.5.3.2 Input Parameters and Initial Conditions

Each of the SLB cases for the limiting RCS pressure, limiting SG pressure, and limiting MCHFR are provided below. Table 15.1-14 provides key inputs including biases for each of the limiting SLB cases. The initial conditions and assumptions used in the evaluation of an SLB event result in a conservative calculation.

Steam Line Break Case Resulting in a Limiting Reactor Coolant System Pressure

The limiting RCS pressure case results from a 6-percent split break located outside containment between the MSIV and secondary MSIV. The following initial conditions are assumed in the analysis of this case.

- Initial power level is assumed to be 102 percent of rated thermal power to account for a 2-percent measurement uncertainty.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and using a bounding control rod drop rate.
- The most limiting EOC reactivity feedback is used to provide a limiting power response.
- The beginning-of-life SG is assumed with no tube plugging.
- The single failure identified for this case is the failure of an MSIV to close.

Steam Line Break Case Resulting in Limiting Steam Generator Pressure

The SLB event is a secondary depressurization event; however, because of the nature of DHRS operation, the combination of secondary inventory and RCS temperature can cause significant SG pressurization when DHRS actuates. This is a normal behavior of the system and does not challenge the SG design pressure. The limiting case is a double-ended guillotine break of the steam line outside containment between the MSIV and secondary MSIV. This case includes the following assumptions:

- The limiting case is initialized at 102 percent reactor power with biased high RCS average temperature and SG pressure.
- End of cycle reactivity feedback with a low fuel temperature bias.
- This case assumes a failed FWIV on the impacted SG train, which allows additional FW into the impacted SG until the FWRV closes.

Steam Line Break Cases Resulting in the Limiting Minimum Critical Heat Flux Ratio

A thermal hydraulic analysis is performed to provide the limiting boundary conditions for the downstream subchannel analysis, which evaluates the final CHF value. The limiting MCHFR case results from a 12-percent split break located outside containment between the MSIV and secondary MSIV. The following initial conditions are assumed in the analysis of the limiting MCHFR case.

- Initial power level is assumed to be 102 percent of rated thermal power to account for a 2-percent measurement uncertainty.
- Beginning of cycle reactivity feedback with a high fuel temperature bias.

- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The beginning-of-life SG is assumed, with no SG tube plugging.
- The single failure identified for this case is the failure of an MSIV to close.

The results from the thermal hydraulic evaluations are used as input to the subchannel analysis to determine the limiting MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.

15.1.5.3.3

Results

Steam Line Break Case Resulting in the Limiting Reactor Coolant System Pressure

The sequence of events for the limiting RCS pressure SLB case event is provided in Table 15.1-11. Figure 15.1-29 through Figure 15.1-35 show the transient behavior of key parameters for this SLB case. The effects of the postulated SLB on other systems are considered in Section 3.6 consistent with Branch Technical Position (BTP) 3-3 and BTP 3-4. Most SLB events have a cooldown effect on the RCS, where there is little or no pressurization of the RCS. However, this SLB case represents a small break in the steam line that causes core power to increase until reactor trip on high power. Feedwater is terminated because of SSI actuation on low PZR pressure, causing a subsequent RCS heatup and pressurization until the DHRS actuates on high PZR pressure. This scenario results in the highest RCS pressure for the steam piping failure events analyzed.

Because of reactivity feedback from the cooldown, core power rises until reactor trip. The reactor safety valve (RSV) lifts during the re-pressurization following SSI and maintains RCS pressure within limits as shown in Table 15.1-11 and Figure 15.1-30.

After the DHRS is actuated, heat removal on the steam side is limited by the DHRS condenser, causing a pressurization of the secondary side. This pressurization is an expected function of the DHRS actuation and not a direct consequence of the steam pipe break accident.

The limiting single failure assumed in this SLB case is the failure of the primary MSIV on the impacted train to close on demand, which allows the impacted SG to completely empty and depressurize after reactor trip and DHRS actuation. This single failure disables the DHRS functionality on the impacted SG and causes a higher heat load on the intact DHRS train. Reactor coolant system average temperature decreases as shown in Figure 15.1-32, demonstrating stable natural circulation cooling and successful functionality of the intact DHRS train. Figure 15.1-34 and Figure 15.1-35 demonstrate DHRS flow for this case.

The result of this SLB case is a stable plant condition, where the DHRS maintains core cooling.

Steam Line Break Case Resulting in Limiting Steam Generator Pressure

The sequence of events for the limiting SG pressure SLB case event is provided in Table 15.1-12. The limiting SG pressure case is a double-ended guillotine break of the steam line between the MSIV and secondary MSIV. An FWIV failure to close is assumed as the worst case single failure. The FWIV failure on the impacted SG allows additional inventory to be added to the impacted loop following the reactor and turbine trips. The additional SG inventory in the impacted loop exacerbates the SG pressure increase following turbine trip and MSIV closure. The acceptance criterion for peak secondary pressure is satisfied for the steam pipe break accident as shown in Figure 15.1-36 and Figure 15.1-37.

Steam Line Break Case Resulting in the Limiting Minimum Critical Heat Flux Ratio

The sequence of events for the limiting MCHFR case is provided in Table 15.1-13. Figure 15.1-38 through Figure 15.1-43 show the transient behavior of key parameters for this SLB case. The transient initiates with a 12-percent split break of the main steam piping outside containment between the MSIV and secondary MSIV. The effects of the postulated SLB on other systems are considered in Section 3.6 consistent with BTP 3-3 and BTP 3-4. This small break minimizes MCHFR by allowing the system to reach the high power MPS limit.

The SLB causes an increase in steam flow through the SG. The FW pump flow rate is allowed to increase with the steam flow to maximize the overcooling effect. This increase in steam flow leads to an increase in the heat removal rate from the RCS via the SG. During the overcooling phase of the transient, the RCS temperature steadily decreases, while the RCS density increases. The colder, denser RCS causes the regulating control rod bank to withdraw in an attempt to maintain RCS temperature. The withdrawal of the control rods causes a positive reactivity insertion that increases reactor power. If the regulating control rod bank is disabled or kept in insertion-only mode, a similar power response is driven by moderator feedback instead. Reactor power reaches the high power limit, tripping the reactor.

In this SLB case, the overcooling effect initially lowers RCS pressure. However, as core power rises, pressure increases until the reactor trip. The pressure in the intact SG increases following SSI and DHRS actuation for the same reasons as the main steam pressure increase shown in the limiting RPV pressure SLB case. The peak primary and secondary pressures are bounded by the pressures presented for the maximum pressure cases.

The CHFR decreases as reactor power and RCS pressure increase. The MPS terminates this transient before the CHFR reaches the CHF analysis limit. The

MCHFR for the limiting steam pipe break case does not violate the CHF analysis limit as shown in Figure 15.1-43.

As in the SLB case with limiting RPV pressure, the limiting single failure assumed in this SLB case is failure of the MSIV on the impacted train to close on demand, which allows the impacted SG to completely empty and depressurize after reactor trip and DHRS actuation. This single failure disables the DHRS functionality on the impacted SG and causes a higher heat load on the intact DHRS train. However, this failure does not affect the limiting CHF because it occurs after the time of MCHFR.

The result of this SLB case is a stable plant condition in which the DHRS maintains core cooling.

15.1.5.4 Radiological Consequences

The radiological consequences of the SLB event are discussed in Section 15.0.3.

15.1.5.5 Conclusions

The DSRS acceptance criteria for this accident are met for the limiting cases.

- 1) Pressure in the RCS and in the MSS should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.
 - The limiting RPV pressure for an SLB is under the more conservative AOO acceptance criterion of 110 percent of design values. The limiting MSS pressures for an SLB must be less than or equal to 110 percent of the design value. The calculated RPV and MSS pressures demonstrate margin to the acceptance criterion. Therefore, the acceptance criterion for pressure is met for this event. The maximum primary and SG pressure values for the cases analyzed are shown in Table 15.1-15.
- 2) The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit.
 - The MCHFR for this event is above the CHF analysis limit as shown in Table 15.1-15. Therefore, this acceptance criterion is met.
- 3) The radiological evaluation is in Section 15.0.3 and demonstrates the acceptance criteria are met.
- 4) System(s) provided for decay heat removal must be highly reliable and, when required, automatically initiated. For the NuScale Power Plant US460 standard design, the DHRS provides the safety-related means of decay heat removal.
 - The results of the analysis show the DHRS actuates and provides heat removal during an SLB, ensuring acceptance criteria are not challenged.

15.1.6 Loss of Containment Vacuum and Containment Flooding**15.1.6.1 Identification of Causes and Accident Description**

Loss of containment vacuum and containment flooding events that result in an increase in RCS cooling are NuScale Power Plant US460 standard design-specific events. The containment net volume is less than conventional designs and the NPM is partially immersed in a pool of borated water during normal operation. Because the containment operates at a vacuum during normal operation, air or water ingress into containment could increase heat transfer from the RPV to the reactor pool. This overcooling could lead to higher reactor power, higher RCS pressure, and reduced MCHFR.

The containment evacuation system maintains the containment volume at a vacuum during normal operation. A failure in the containment evacuation system could result in loss of vacuum because containment pressure increases due to evaporation of RCS fluid leaking into containment. If the failure of the containment evacuation system or RCS fluid leakage is sufficiently severe, it could result in a loss of containment vacuum event. If the containment vacuum is lost, heat transfer from the reactor vessel increases. The analysis of a loss of containment vacuum shows a negligible effect on reactor power, and is therefore bounded by a containment flooding event.

The reactor component cooling water system (RCCWS) provides heat removal to the control rod drive system (CRDS). The RCCWS supplies reactor component cooling water to the CNV that then conducts reactor component cooling water to the CRDS piping that passes through containment to provide this function. If piping containing reactor component cooling water leaks or ruptures inside the CNV, a containment flooding event occurs. Other potential containment flooding sources include: FW containing line break, main steam containing line break, chemical and volume control system fluid containing line break, high-point vent fluid containing pipe break, and RCCWS fluid containing line break. The FW fluid containing line break event is evaluated in Section 15.2.8, the SLB event is evaluated in Section 15.1.5, and the chemical and volume control system fluid containing line break is evaluated in Section 15.6.2. The RCCWS fluid line break is a more limiting containment flooding event than a high-point vent fluid pipe break because it has a temperature lower than the containment saturation temperature. If the lower temperature RCCWS fluid line ruptures, there is no immediate boiling, preventing the high containment pressure limit from being reached. The flooding of the CNV could cause an increase in heat transfer from the RPV to containment, cooling the RCS. As the RCS cools, reactor power increases because of the negative moderator temperature coefficient. This unexpected rise in core power decreases the MCHFR and leads to overpressurization of the RPV.

A loss of containment vacuum event is categorized as an AOO. Typically, pipe system failures are categorized as accidents, but the containment flooding event is conservatively categorized as an AOO. The categorization of the design-basis events is discussed in Section 15.0.

15.1.6.2 Sequence of Events and Systems Operation

The event sequence table for the limiting containment flooding case is provided in Table 15.1-18.

Unless specified below, the analysis of the containment flooding event assumes the plant control systems and ESFs perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a containment flooding event.

The containment flooding is initiated by a break in a CRDS line that transports reactor component cooling water inside containment. A conservative total RCCWS volume is assumed. The containment flooding cases assume one or more RCCWS pumps continue operating after the break. The number of operating RCCWS pumps assumed in the analysis is varied to determine the most limiting case.

The containment evacuation pumps could malfunction to cause a loss of containment vacuum scenario, but are assumed to operate at nominal capacity for the limiting containment flooding scenario in order to delay reaching the high containment pressure MPS setpoint.

Operator action is not credited for regulating control rod movement or increasing boron concentration, which ensures the maximum reactivity insertion is reached as the control system attempts to maintain RCS temperature by pulling the regulating control rods from the core.

The MPS high containment pressure signal is credited to provide protection against loss of containment vacuum and containment flooding events. In cases that result in a reactor trip, the same high containment pressure signal actuates SSI and DHRS to maintain reactor cooling.

There are no single failures that could make a containment flooding event more severe with respect to the acceptance criteria.

Normal AC power is assumed to be available for this event. A loss of AC power, either at event initiation or at reactor trip, is not a conservative condition for a containment flooding event. The loss of AC power results in a reactor trip, DHRS actuation, and containment isolation on other MPS signals earlier than the high containment pressure signal.

15.1.6.3 Thermal Hydraulic and Subchannel Analyses**15.1.6.3.1 Evaluation Model**

The thermal hydraulic analysis of the plant response to containment flooding is performed using NRELAP5. A description of the NRELAP5 model is provided in Section 15.0.2. The NRELAP5 model is based on the design features of an NPM. The non-LOCA transient modifications to the NRELAP5 model are discussed in Section 15.0.2. The relevant boundary conditions from

the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a complete discussion of the VIPRE-01 code and evaluation model.

15.1.6.3.2 Input Parameters and Initial Conditions

Table 15.1-16 provides key inputs for the limiting containment flooding case for MCHFR. The initial conditions used in the evaluation of this containment flooding case ensure the results have sufficient conservatism.

- RCCWS break flow conditions:
 - One RCCWS pump is operating.
 - The RCCWS water is assumed to be 100 degrees F.
- Initial power level is assumed to be 102 percent of rated thermal power to account for a 2-percent measurement uncertainty.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The most limiting EOC reactivity feedback is used to provide a limiting power response.
- The containment conditions are biased to provide the limiting MCHFR. The containment initial conditions are biased so as to empty the entire volume of reactor component cooling water into containment.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.

15.1.6.3.3 Results

Figure 15.1-44 through Figure 15.1-48 show the transient behavior of key parameters for the containment flooding case with the limiting MCHFR.

The containment flooding event is initiated by a rupture in a CRDS line that conducts reactor component cooling water inside the CNV. As the RCCWS fluid enters containment, there is a rise in containment pressure. The rise in containment pressure does not reach the MPS setpoint for high containment pressure. The RCCWS fluid raises the liquid level in the CNV to a level that

allows fluid to replace the space between the RPV and the CNV walls. The fluid allows an increase in heat transfer from the RPV to the CNV.

As the heat transfer increases from the RPV to the CNV, the RCS is overcooled. The cooler core inlet temperature introduces an increase in reactivity because of the moderator temperature reactivity feedback, resulting in an increase in reactor power. The magnitude of the power increase is smaller than the other overcooling events presented in Section 15.1. The smaller power increase ensures the pressures of the RCS and the MSS do not challenge the design pressures of the RPV and main steam piping, respectively. As with other overcooling events, if a containment flooding event trips the reactor, the subsequent pressure rises in the primary and secondary systems are a result of the isolation functions and not a direct consequence of the containment flooding.

The containment flooding case that is limiting for MCHFR does not reach MPS setpoints. The MCHFR calculated for this event does not challenge the CHF analysis limit as shown in Table 15.1-17 and is bounded by the other overcooling events presented in Section 15.1.

For containment flooding cases in which the high containment pressure MPS setpoint is reached, reactor trip, containment isolation, and DHRS actuation ensure a return to a stable condition with no operator actions.

15.1.6.4 Radiological Consequences

The normal leakage-related radiological consequences of this event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.1.6.5 Conclusions

The DSRS acceptance criteria are met for the limiting case.

- 1) Pressure in the RCS and in the MSS should be maintained below 110 percent of the design values.
 - The pressure responses in the RPV and in the MSS are less severe than those of the limiting AOOs presented in Section 15.2. Therefore, this acceptance criterion is met. The maximum primary and SG pressure values for the cases analyzed are shown in Table 15.1-17.
- 2) Fuel cladding integrity shall be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit.
 - The MCHFR for this event is above the CHF analysis limit as shown in Table 15.1-17, and bounded by the other overcooling events presented in Section 15.1. Therefore, this acceptance criterion is met.

- 3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
 - The analysis presented for this event demonstrates stable DHRS cooling is reached or MPS actuation is not required, and the acceptance criterion is met.
- 4) The guidance provided in RG 1.105 can be used to analyze the impact of instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDC 10, 13, 15, 20, and 26.
 - The instrument spans and setpoints discussed in Section 15.1.6.3 address the guidance in RG 1.105.
- 5) The most limiting plant systems single failure shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.
 - There is no single failure that adversely affects the transient results of a containment flooding event.

**Table 15.1-1: Sequence of Events for Limiting Minimum Critical Heat Flux Ratio Case
(15.1.1 Decrease in Feedwater Temperature)**

Event	Time [s]
FW temperature begins to decrease	0
FW temperature reaches 100 °F	2
Control rod insertion begins (high power)	95
Limiting MCHFR	95
SSI valves closed	191
DHRS actuation valves open	211

Table 15.1-2: Decrease in Feedwater Temperature - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	250 MWt	+2%
Pressurizer pressure	2000 psia	+70 psia
RCS flow rate	Range in Table 15.0-6	Low
RCS average temperature	540 °F	+5 °F
SG pressure	475 psia	+35 psia
FW temperature	250 °F	+20 °F
Pressurizer level	60%	+8%
SG heat transfer bias	Nominal	N/A

Table 15.1-3: Decrease in Feedwater Temperature (15.1.1) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2126 psia
Maximum SG pressure	2420 psia	1475 psia
MCHFR	1.33	1.46

**Table 15.1-4: Sequence of Events for Limiting Minimum Critical Heat Flux Ratio Case
(15.1.2 Increase in Feedwater Flow)**

Event	Time [s]
FW flow begins to increase	0
Reactor trips on high reactor power	54
Control rods begin to insert	54
Lowest MCHFR	54
SSI actuation on high main steam superheat	63
SSI valves fully closed	73
DHRS actuation on high main steam pressure	76
DHRS valves fully open	106

**Table 15.1-5: Sequence of Events for Limiting Steam Generator Overfill Case
(15.1.2 Increase in Feedwater Flow)**

Event	Time [s]
FW flow begins to increase	0
Reactor trip and SSI actuation on low main steam superheat	8
SSI valves fully closed (single failure of one FWIV)	18
DHRS actuation on high main steam pressure	25
Maximum SG level (SG with failed FWIV)	88

Table 15.1-6: Increase in Feedwater Flow - Inputs

Parameter	Nominal	Bias (Limiting MCHFR)	Bias (SG Overfill)
Core power	250 MWt	+2%	+2%
Pressurizer pressure	2000 psia	+70 psia	Nominal
RCS flow rate	Range in Table 15.0-6	Low	Nominal
RCS average temperature	540 °F	+5 °F	Nominal
SG pressure	475 psia	-35 psia	Nominal
Pressurizer level	60%	+8%	Nominal
SG heat transfer	Nominal	+30%	-30%
FW temperature	250 °F	-150 °F	Nominal
Reactor pool temperature	100 °F	Nominal	Nominal

Table 15.1-7: Increase in Feedwater Flow (15.1.2) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2191 psia
Maximum SG pressure	2420 psia	1507 psia
MCHFR	1.33	1.52

Table 15.1-8: Sequence of Events (15.1.3 Increase in Steam Flow)

Event	Time [s]
Steam flow begins to increase	0
Peak reactor power is reached	39
Reactor trips on high reactor power	39
Lowest MCHFR	39
Secondary system isolation and DHRS actuation (high main steam pressure)	190

Table 15.1-9: Increase in Steam Flow (15.1.3) - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	250 MWt	+2%
Pressurizer pressure	2000 psia	+70 psia
RCS flow rate	Range in Table 15.0-6	Low
RCS average temperature	540 °F	+5 °F
SG pressure	475 psia	+35 psia
Pressurizer level	60%	+8%
SG heat transfer	Nominal	None
FW temperature	250 °F	None
Reactor pool temperature	100 °F	120 °F

Table 15.1-10: Increase in Steam Flow (15.1.3) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2185 psia
Maximum SG pressure	2420 psia	1507 psia
MCHFR	1.33	1.45

Table 15.1-11: Sequence of Events (15.1.5 Steam Piping Failure - Limiting Reactor Coolant System Pressure Case)

Event	Time [s]
SLB occurs	0
Reactor trip actuated (high power)	56
SSI actuated (low pressurizer pressure)	142
SSI valves closed except assumed failed MSIV	152
DHRS actuated (high pressurizer pressure)	901
RSV lifts	1838
Peak RCS pressure reached	1838

Table 15.1-12: Sequence of Events (15.1.5 Steam Piping Failure - Limiting Steam Generator Pressure Case)

Event	Time [s]
SLB occurs	0
Low main steam pressure limit is reached	0.1
Reactor trip actuated	2
SSI actuated	2
SSI valves closed (except failed FWIV)	12
DHRS actuation	32
Peak SG pressure reached	82

Table 15.1-13: Sequence of Events (15.1.5 Steam Piping Failure - Limiting Minimum Critical Heat Flux Ratio Case)

Event	Time [s]
SLB occurs	0
Reactor trip actuated (high power)	33.5
Peak reactor power reached	34
MCHFR reached	34
SSI actuation (low PZR pressure)	85
DHRS actuation (high PZR pressure)	816

Table 15.1-14: Steam Piping Failure - Inputs

Parameter	Nominal	Biases		
		RCS Pressure Limiting Case	SG Pressure Limiting Case	MCHFR Limiting Case
Core power (MWt)	250	+2%	+2%	+2%
Pressurizer pressure (psia)	2000	+70	+70	+70
Pressurizer level (%)	60	+8	+8	+8
RCS flow rate	Range in Table 15.0-6	Low	Low	Low
RCS average temperature (°F)	540	+5	+5	+5
SG pressure (psia)	475	+0	+35	+0
FW temperature (°F)	250	+20	-20	+20

Table 15.1-15: Steam Piping Failure (15.1.5) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2284 psia
Maximum SG pressure	2420 psia	1421 psia
MCHFR	1.33	1.44

Table 15.1-16: Loss of Containment Vacuum/Containment Flooding - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	250 MWt	+2%
Initial pressurizer pressure	2000 psia	Nominal
Initial pressurizer level	60%	Nominal
RCS flow rate	Range in Table 15.0-6	Low
RCS average temperature	540 °F	+5 °F

Table 15.1-17: Loss of Containment Vacuum/Containment Flooding (15.1.6) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2087 psia
Maximum SG pressure	2420 psia	1407 psia
MCHFR	1.33	2.09

Table 15.1-18: Sequence of Events (15.1.6 Loss of Containment Vacuum/Containment Flooding) Minimum Critical Heat Flux Ratio Limiting Case

Event	Time [s]
Malfunction that causes CNV flooding	0
MCHFR occurs	60
Maximum RCS pressure	645
Maximum SG pressure	1713

Figure 15.1-1: Reactor Coolant System Temperatures (15.1.1 Decrease in Feedwater Temperature)

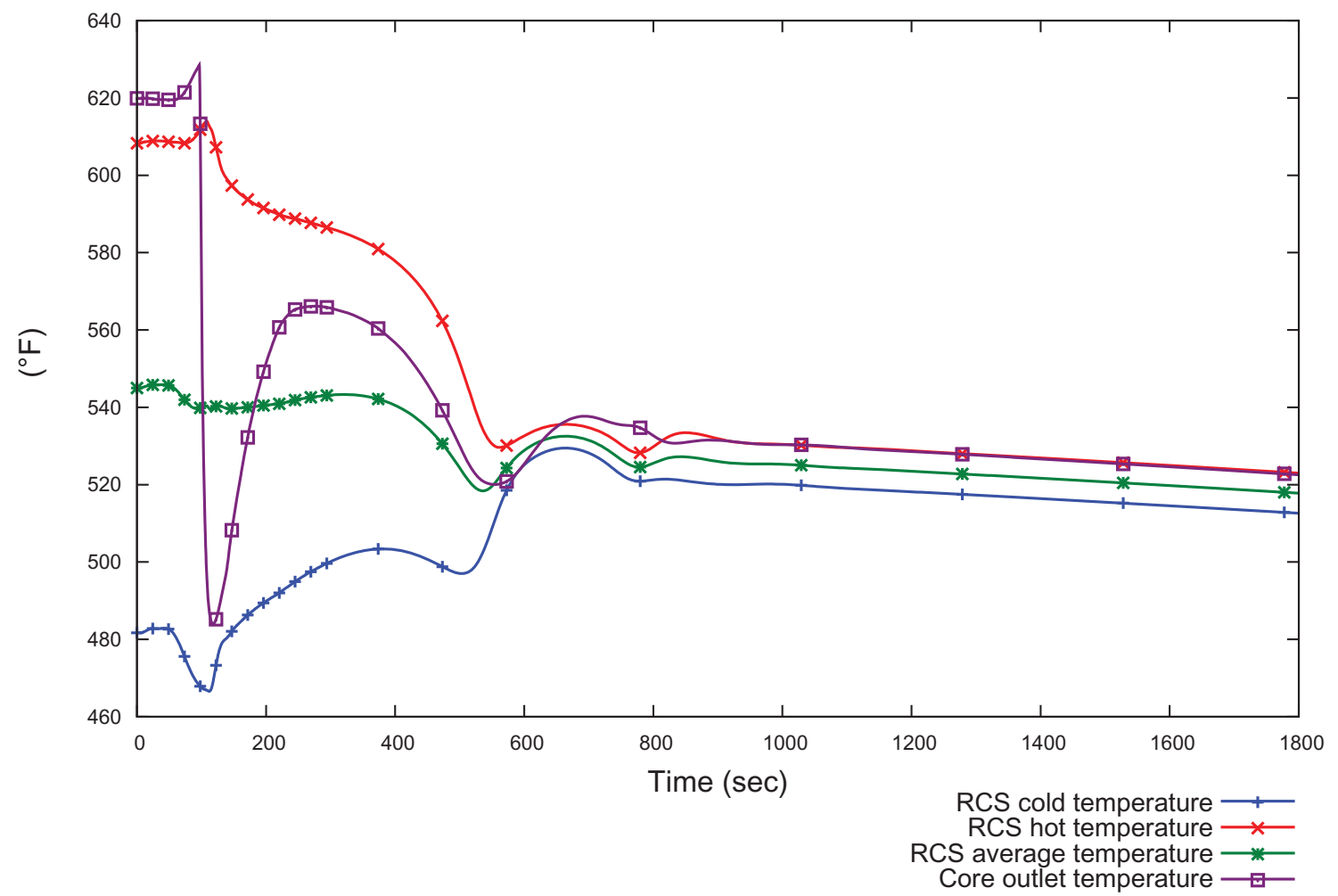


Figure 15.1-2: Reactivity (15.1.1 Decrease in Feedwater Temperature)

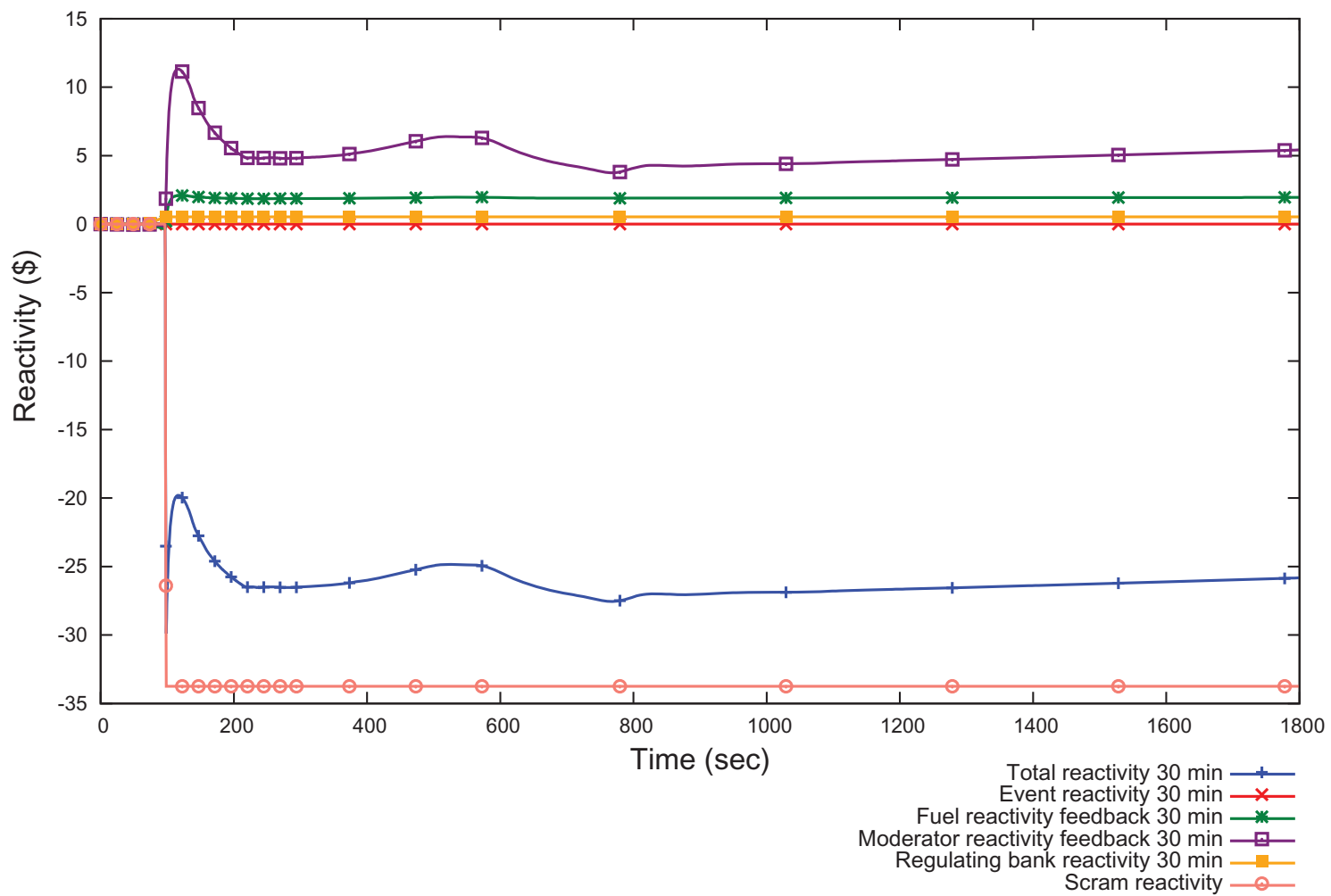


Figure 15.1-3: Reactor Power (15.1.1 Decrease in Feedwater Temperature)

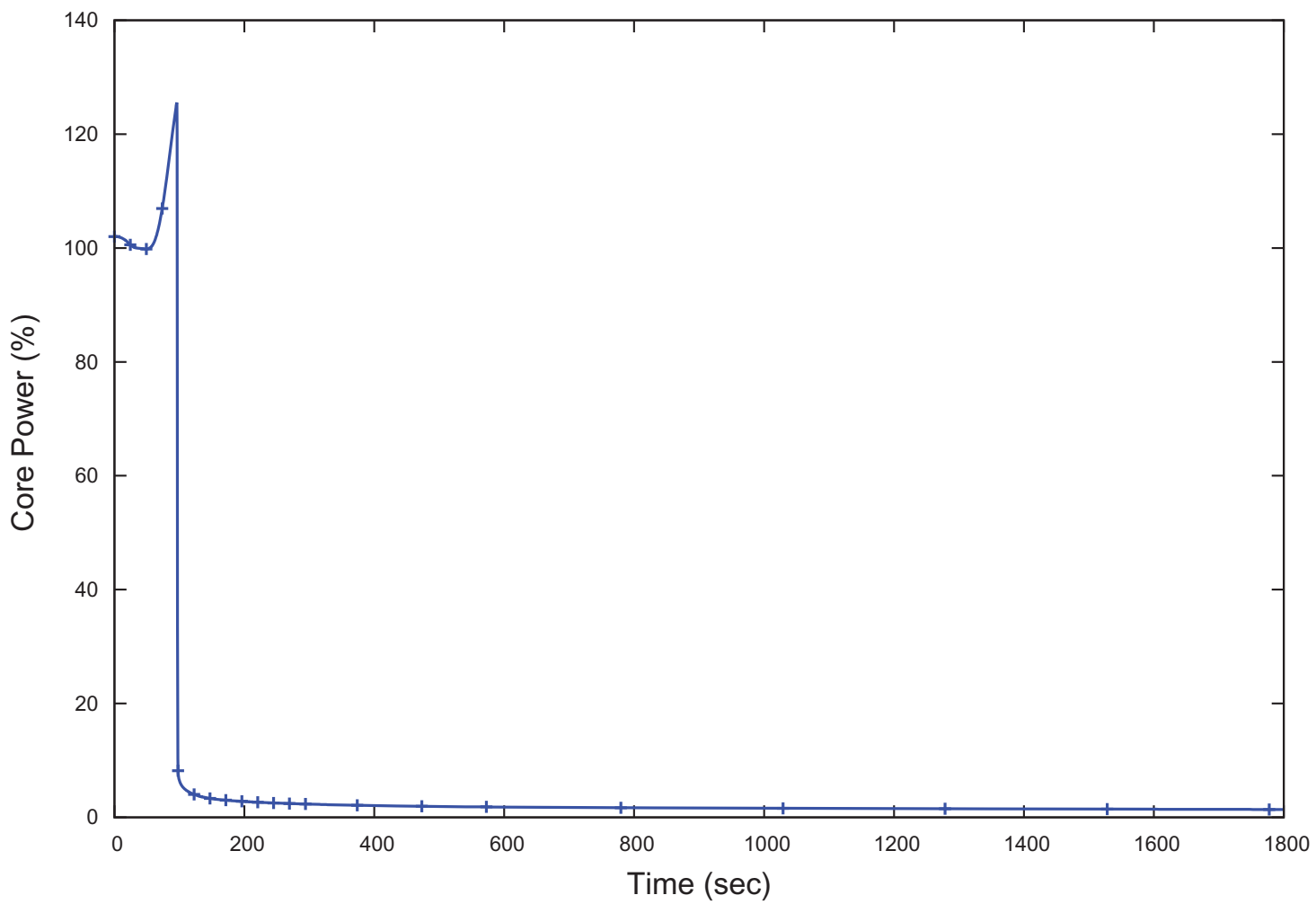


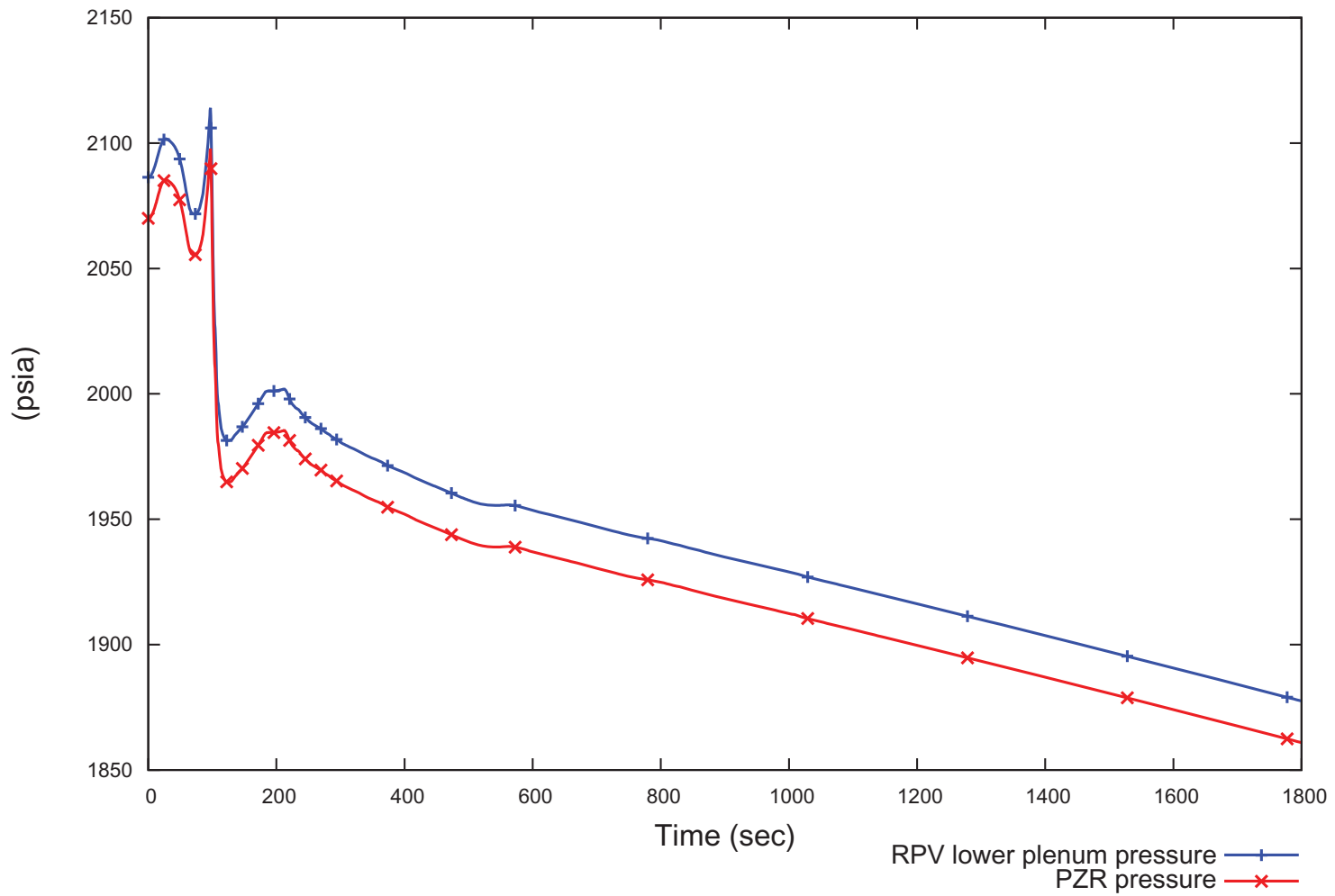
Figure 15.1-4: Reactor Coolant System Pressures (15.1.1 Decrease in Feedwater Temperature)

Figure 15.1-5: Main Steam System Pressure (15.1.1 Decrease in Feedwater Temperature)

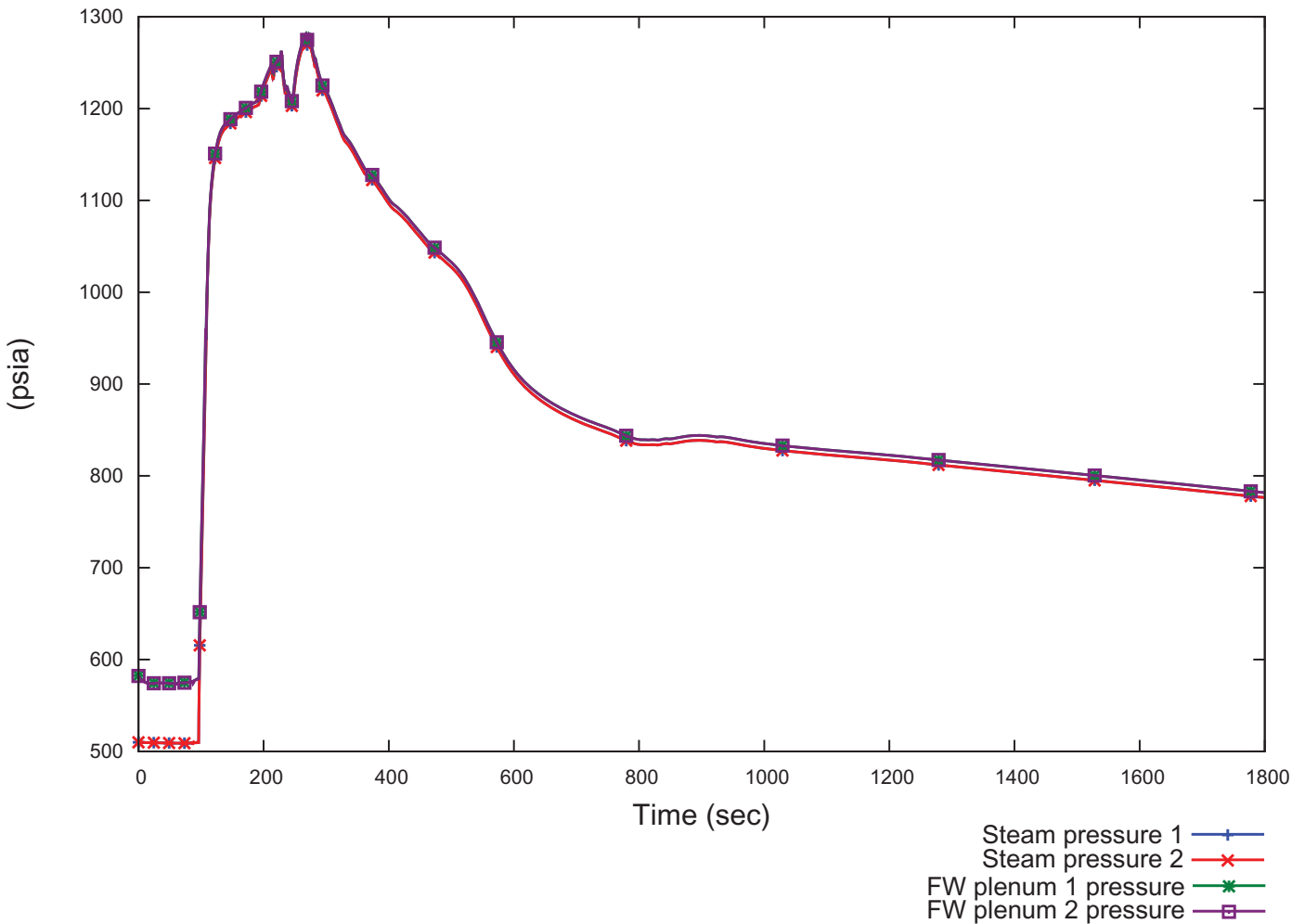
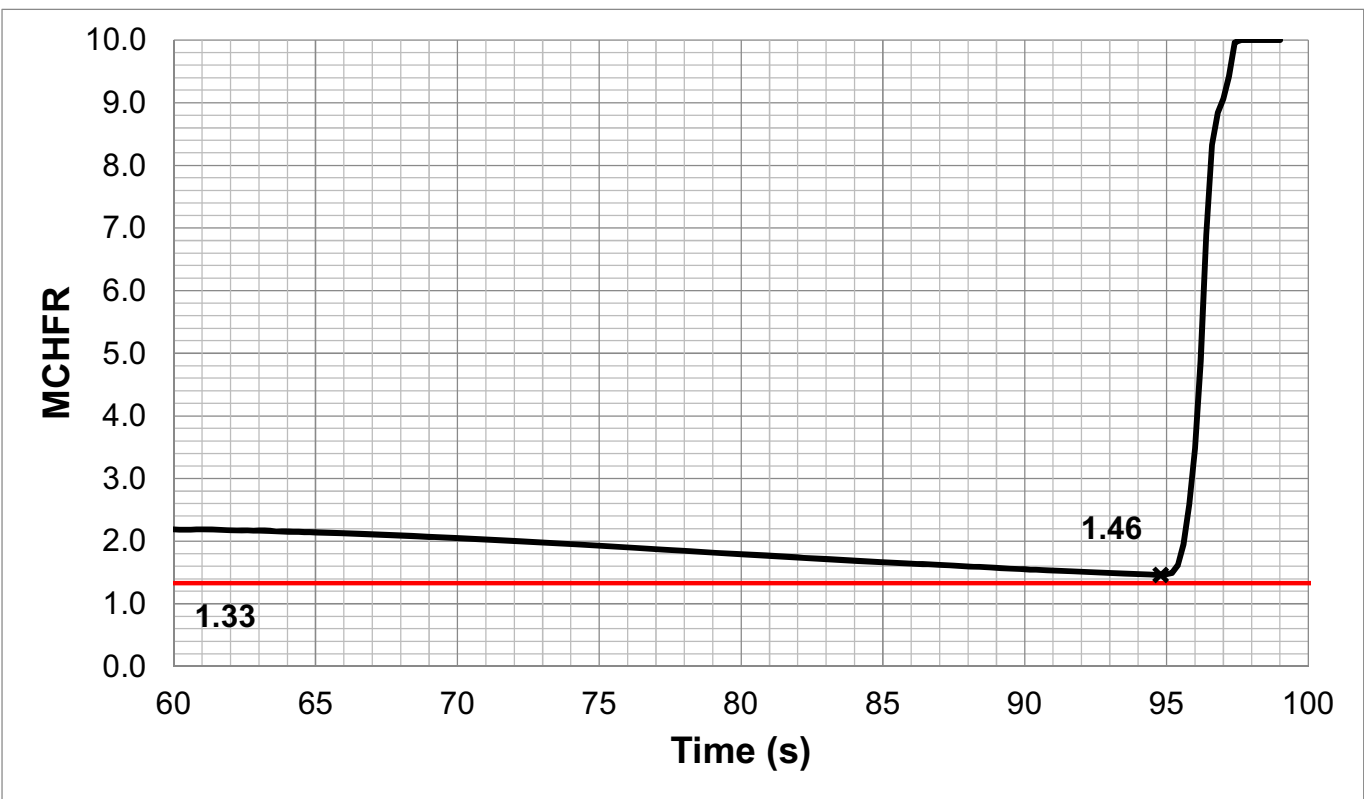


Figure 15.1-6: Critical Heat Flux Ratio (15.1.1 Decrease in Feedwater Temperature)



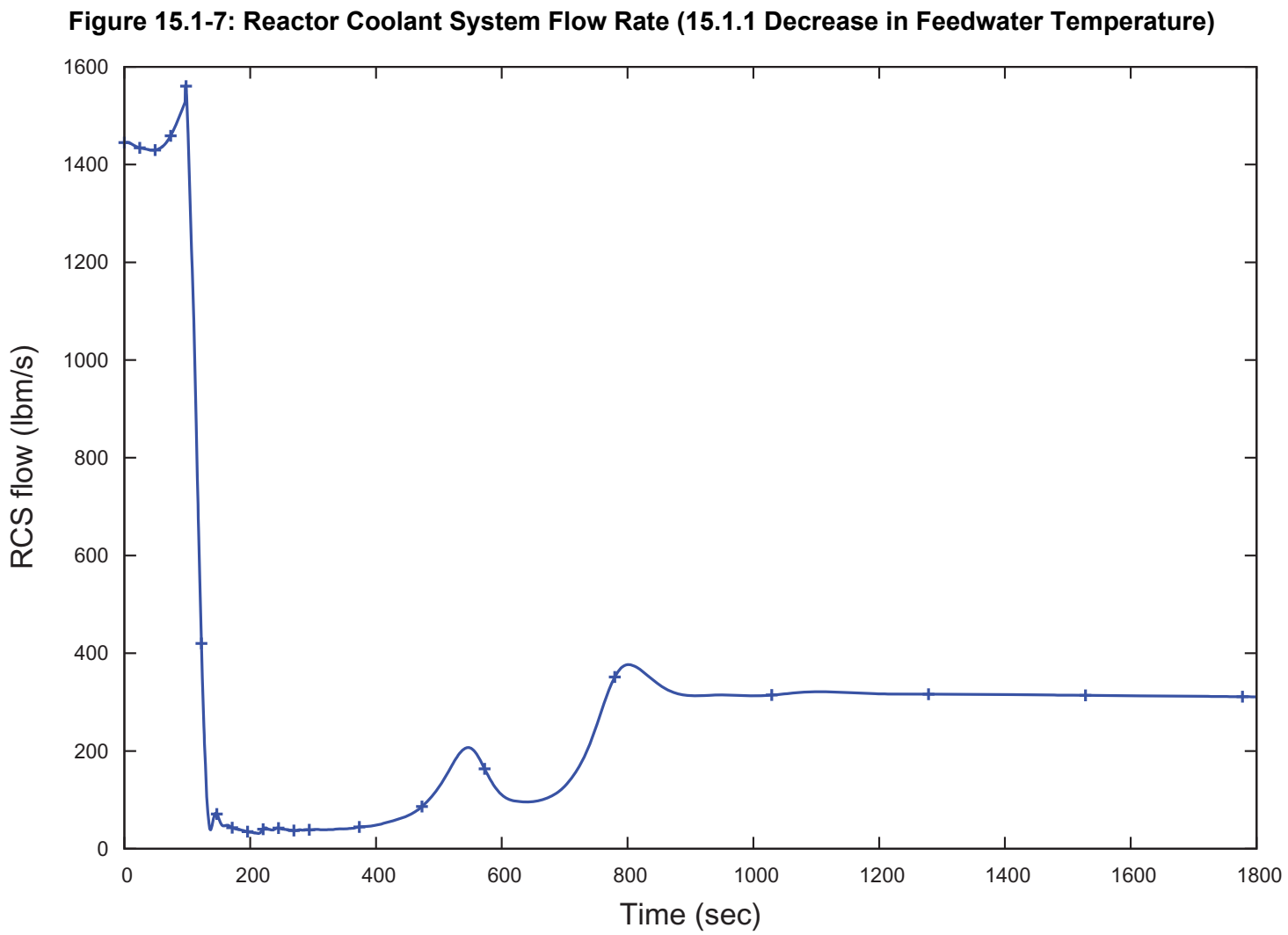
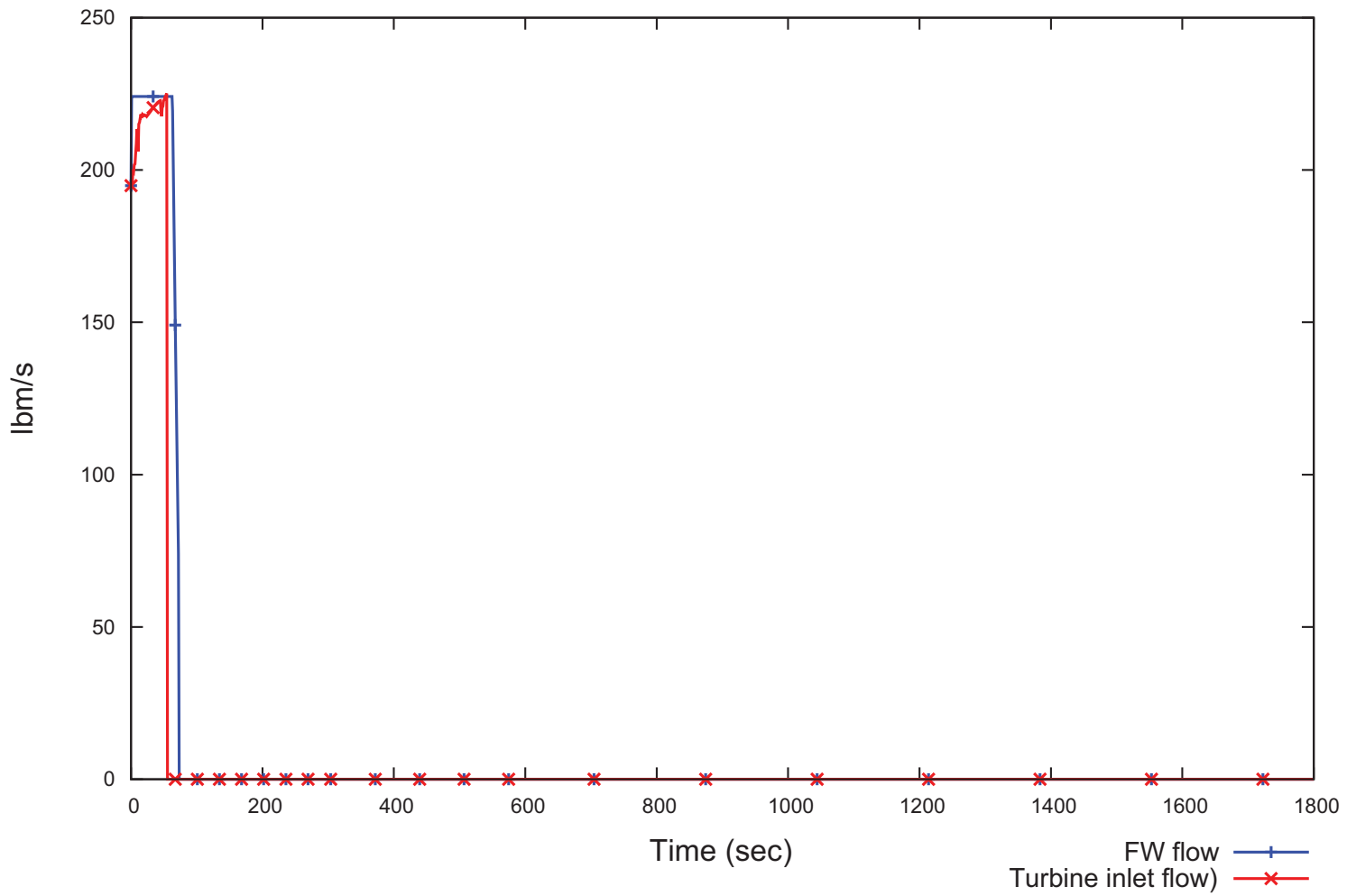


Figure 15.1-8: Feedwater Flow Rate (15.1.2 Increase in Feedwater Flow)



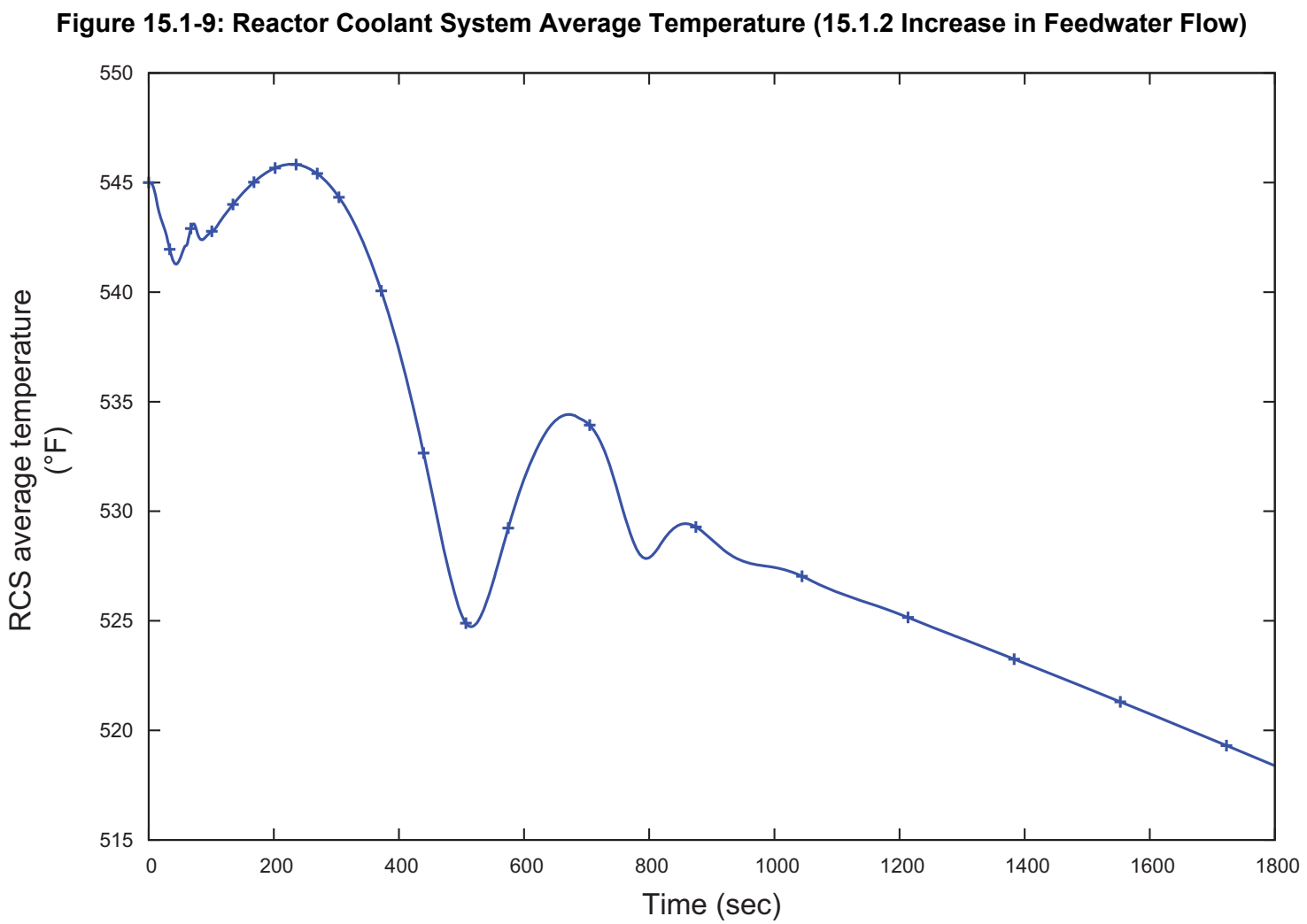


Figure 15.1-10: Reactor Coolant System Flow (15.1.2 Increase in Feedwater Flow)

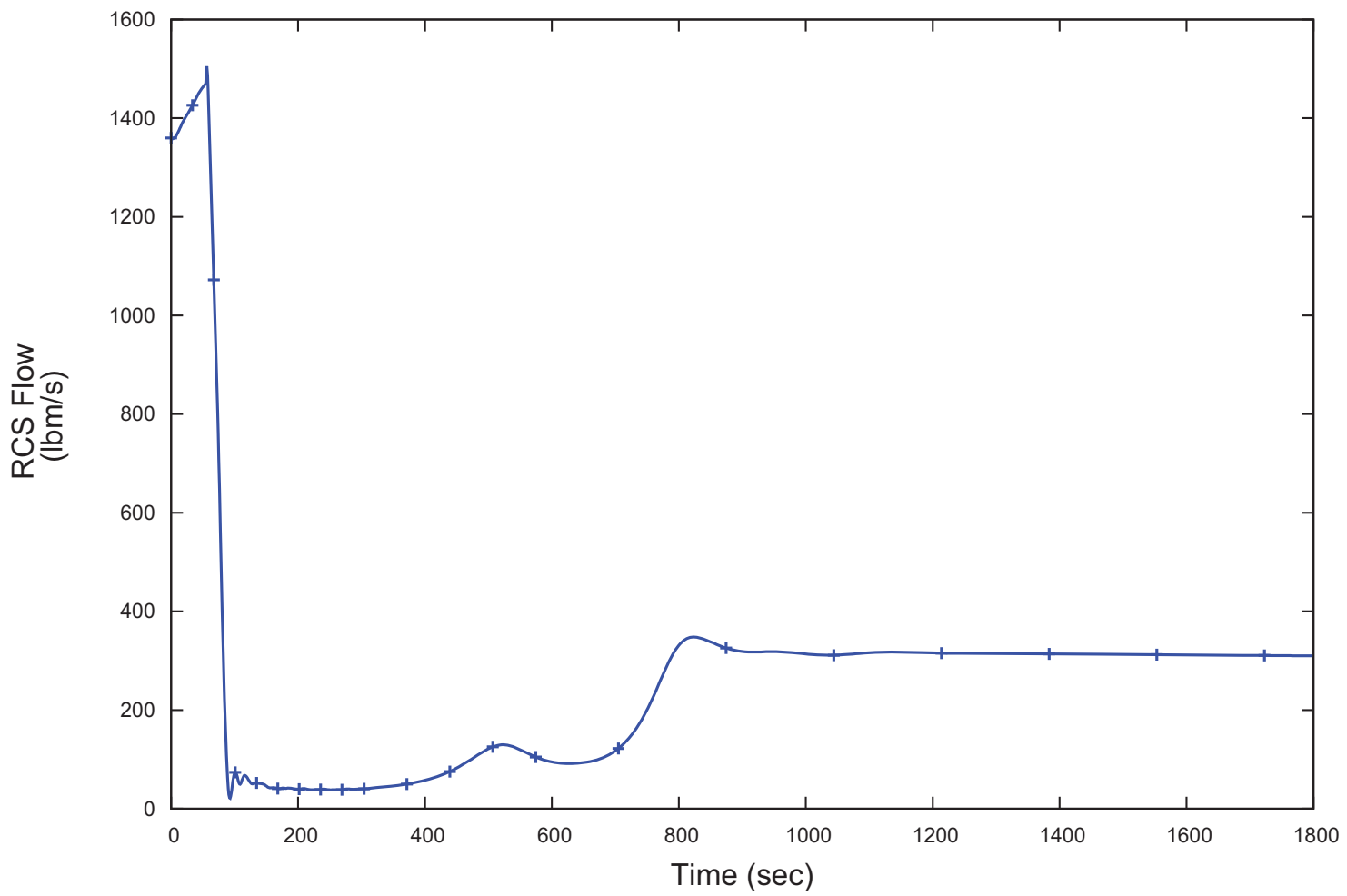


Figure 15.1-11: Total Reactivity (15.1.2 Increase in Feedwater Flow)

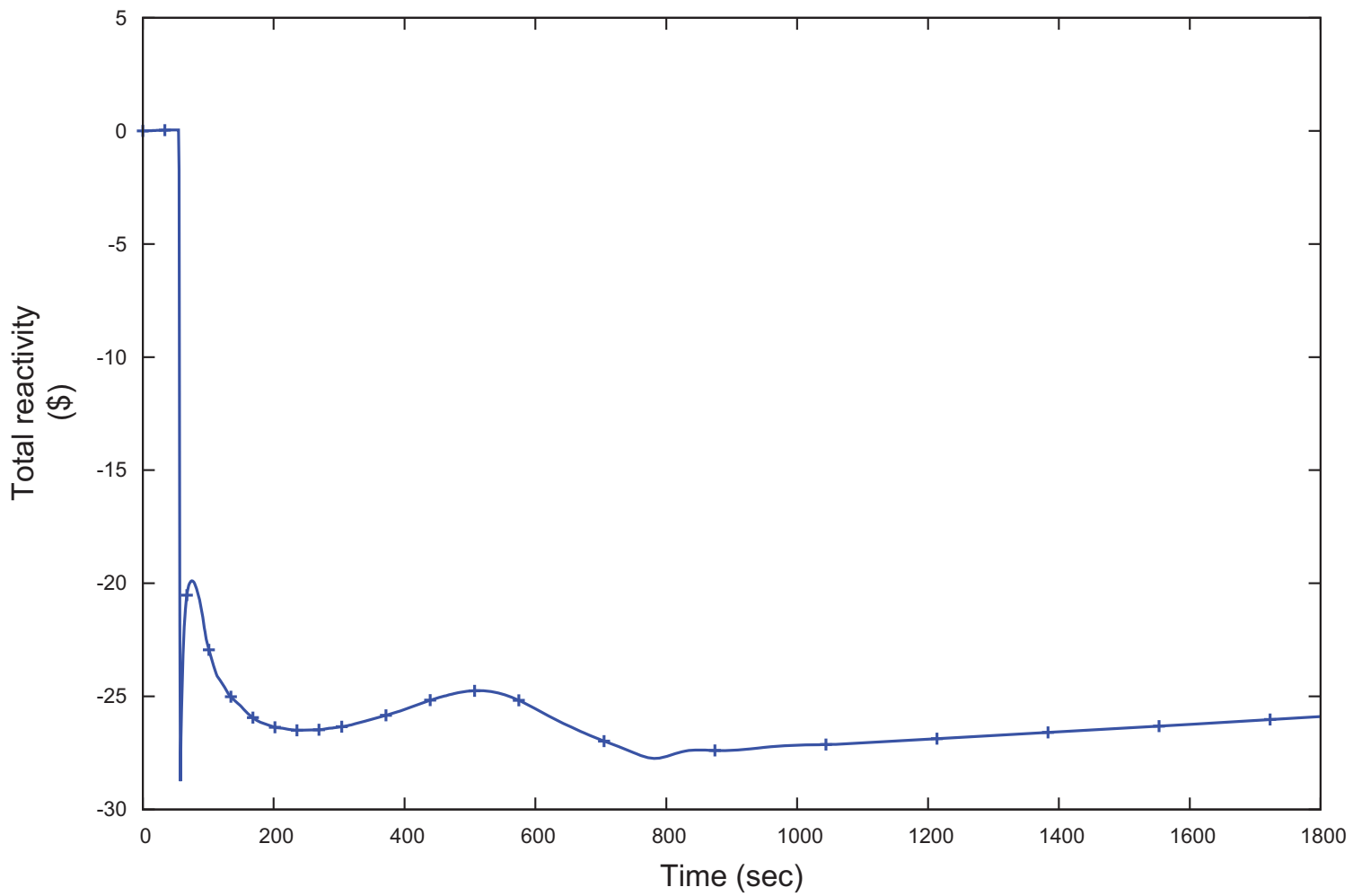


Figure 15.1-12: Reactor Power (15.1.2 Increase in Feedwater Flow)

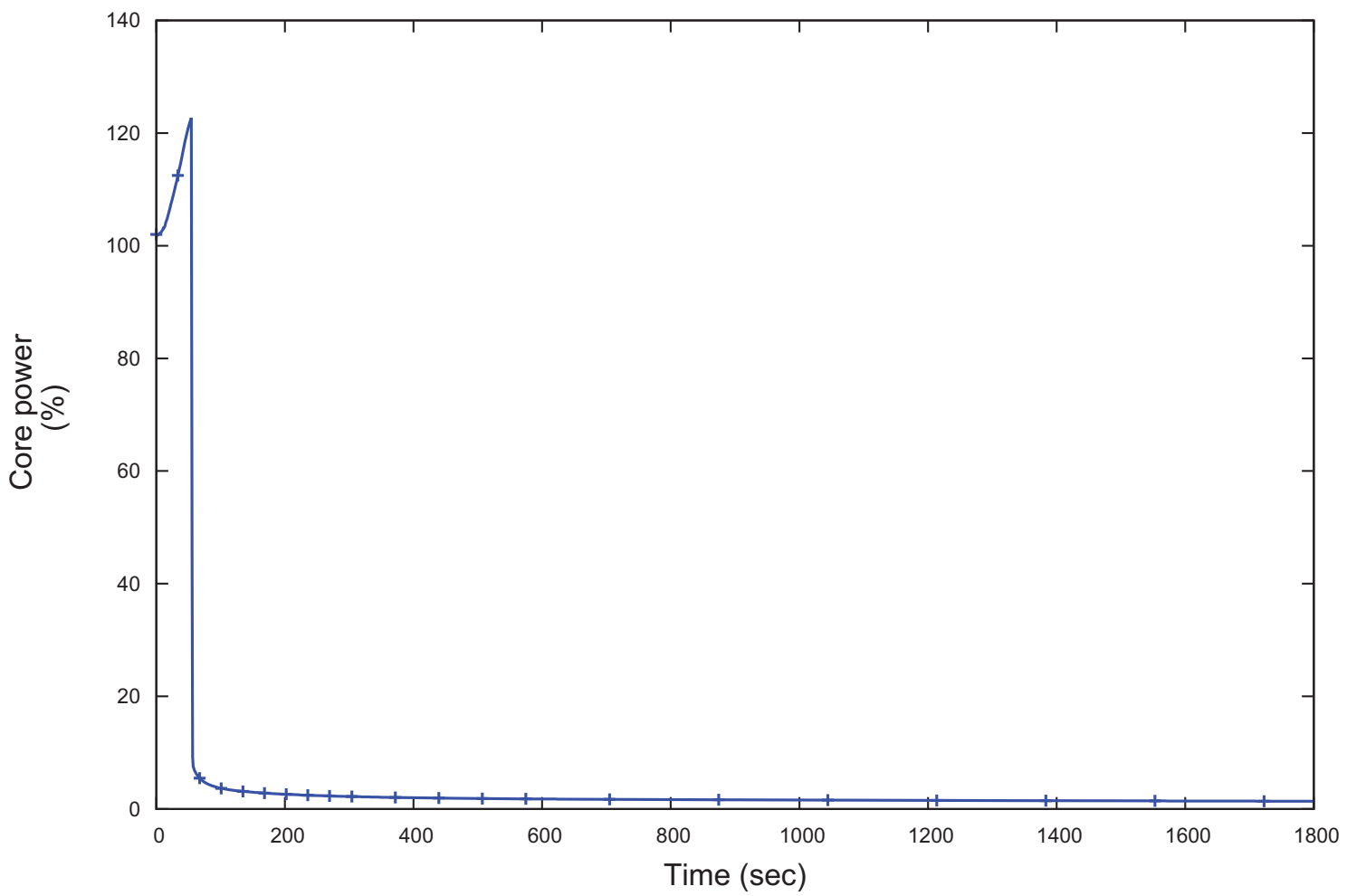


Figure 15.1-13: Reactor Coolant System Pressures (15.1.2 Increase in Feedwater Flow)

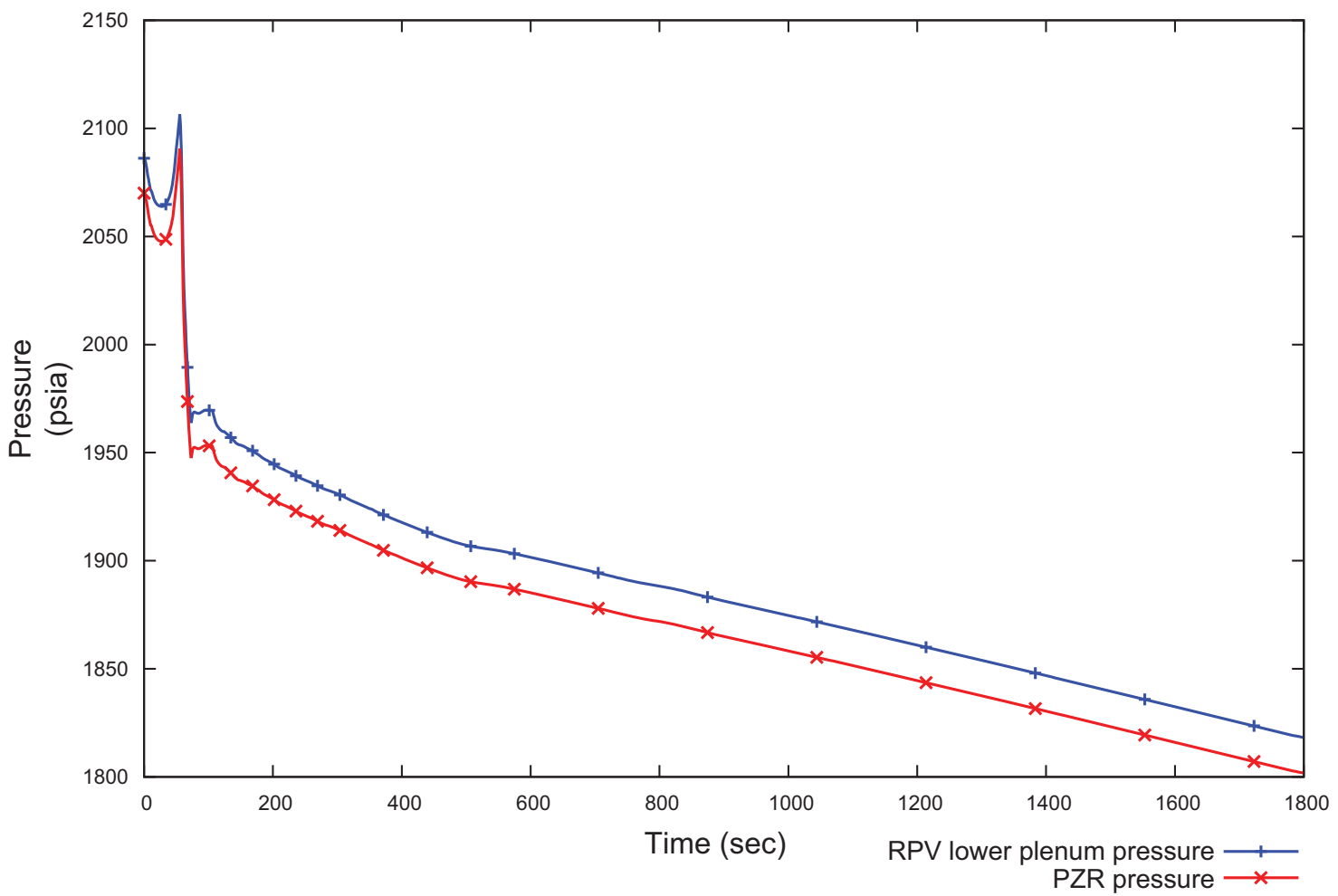


Figure 15.1-14: Secondary Pressures (15.1.2 Increase in Feedwater Flow)

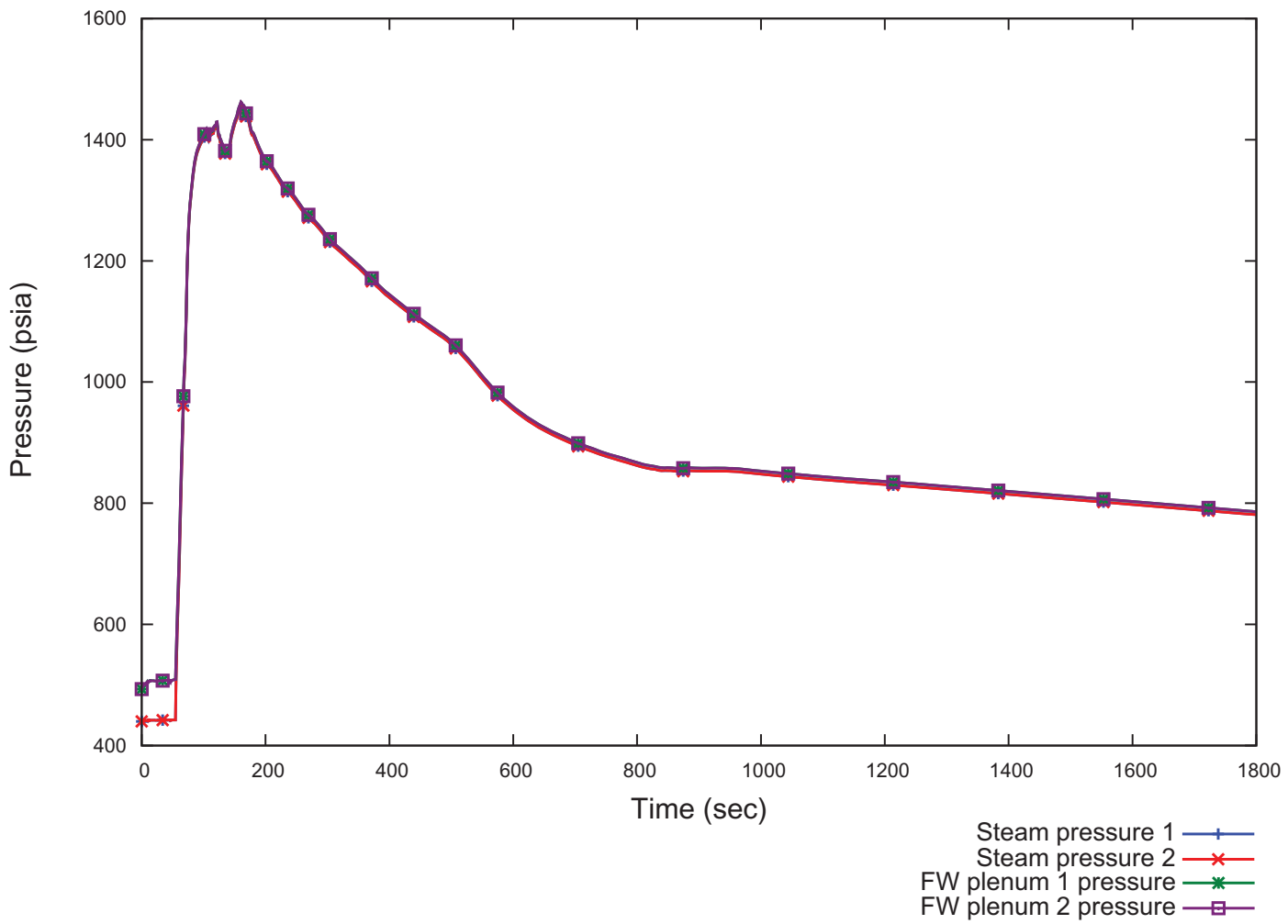


Figure 15.1-15: Steam Generator Level (15.1.2 Increase in Feedwater Flow)

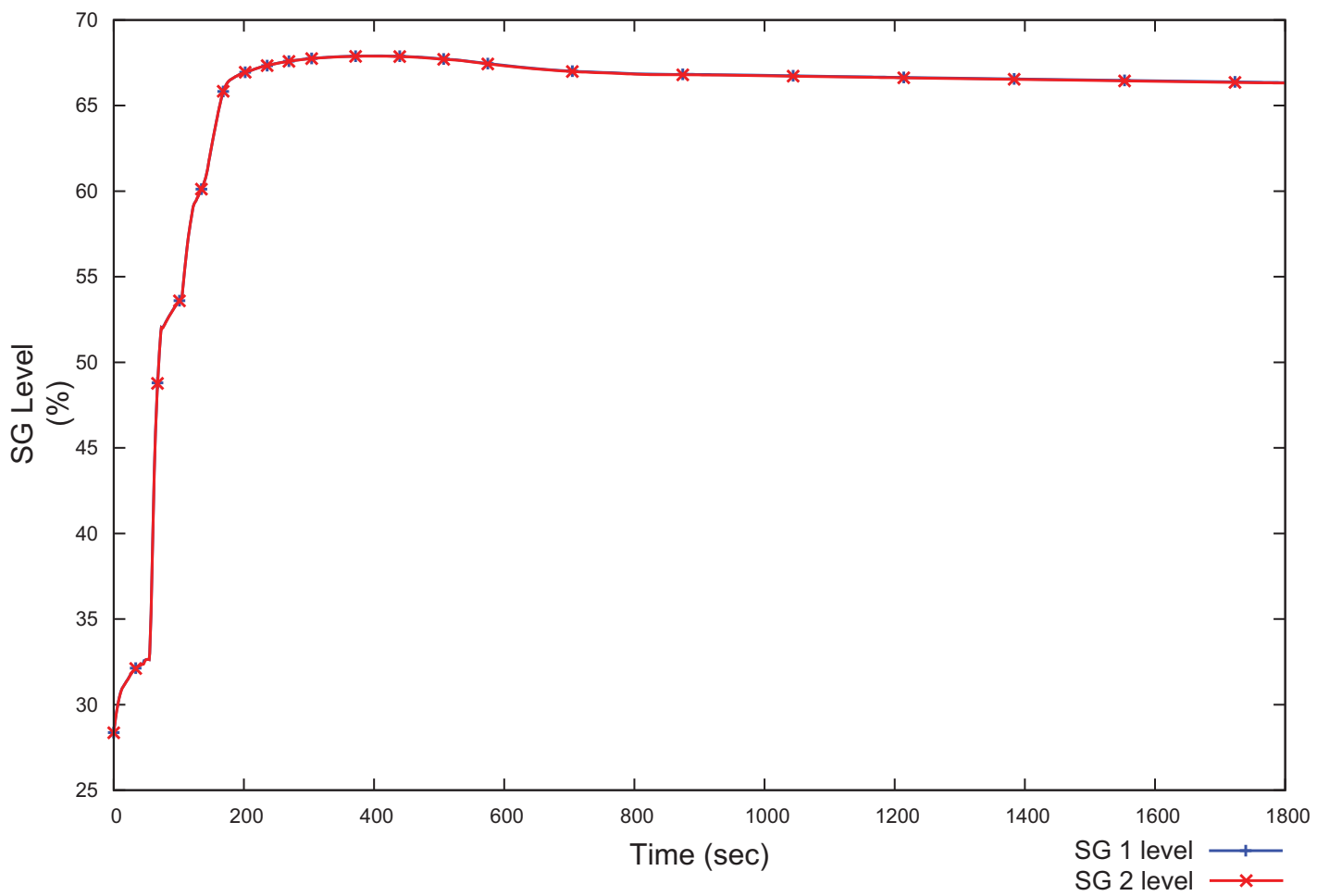


Figure 15.1-16: Critical Heat Flux Ratio (15.1.2 Increase in Feedwater Flow)

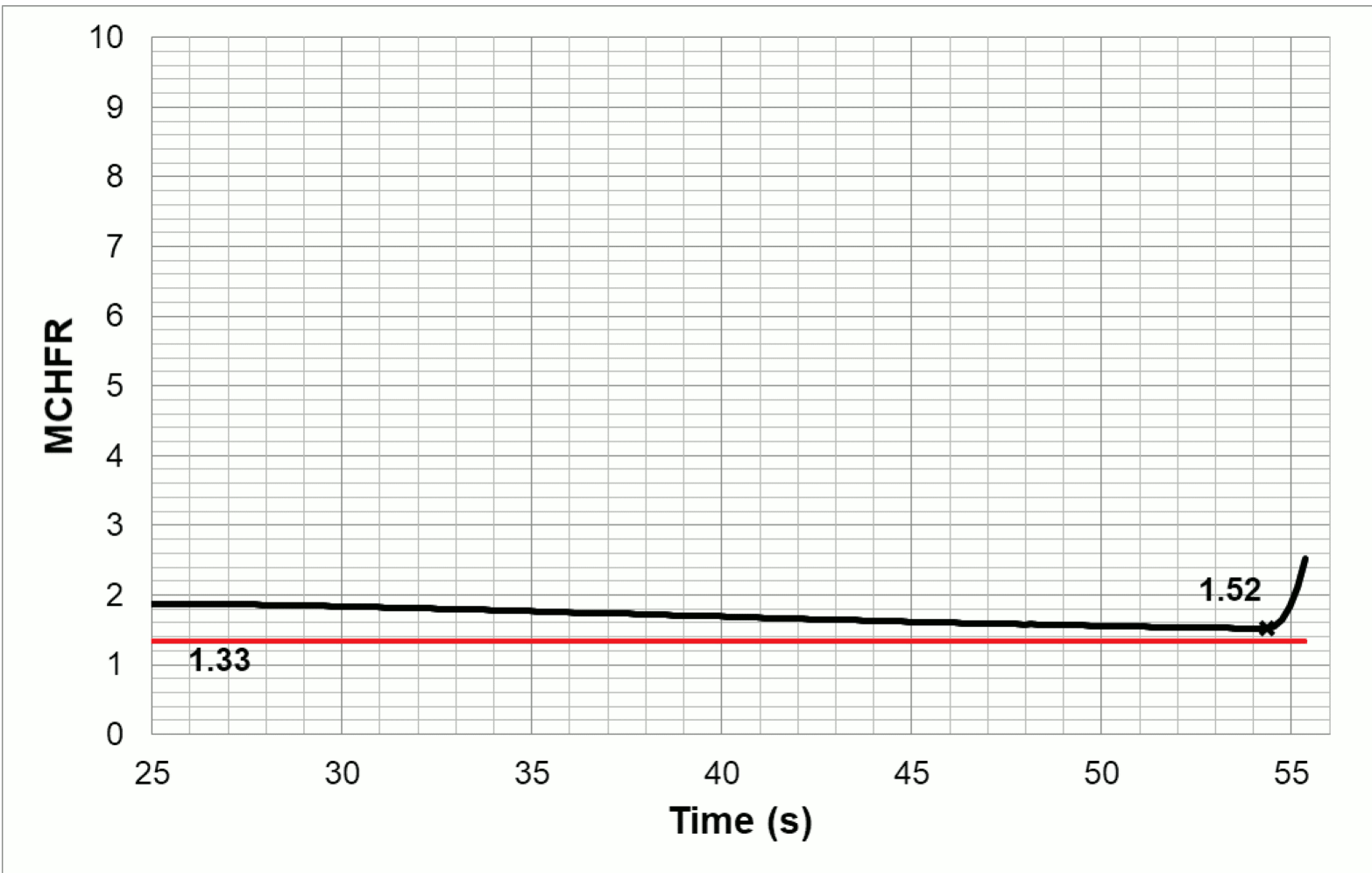


**Figure 15.1-17: Decay Heat Removal System Inlet Flows - Limiting Steam Generator Overfill Case
(15.1.2 Increase in Feedwater Flow)**

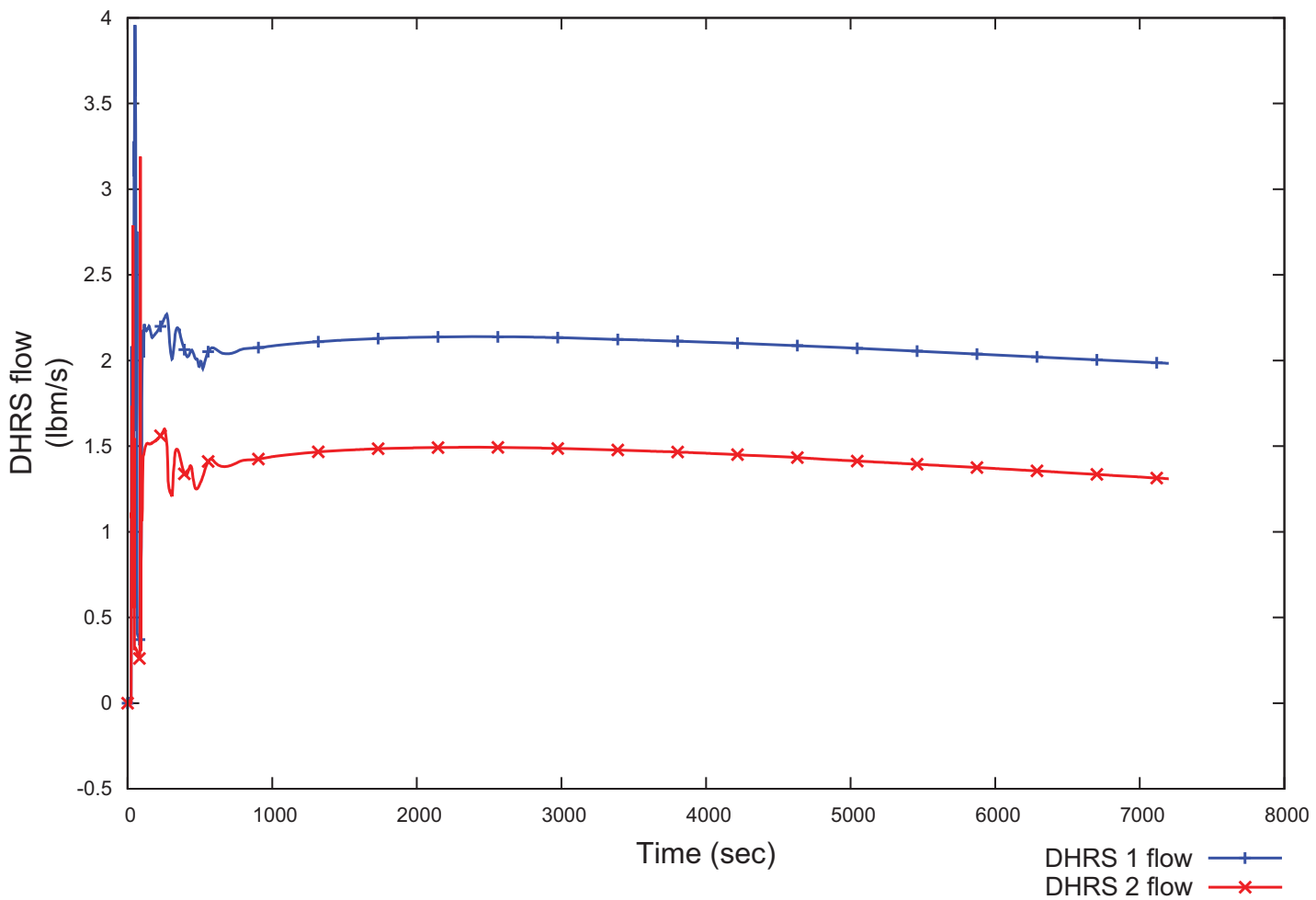


Figure 15.1-18: Steam Generator Levels - Steam Generator Overfill Case (15.1.2 Increase in Feedwater Flow)

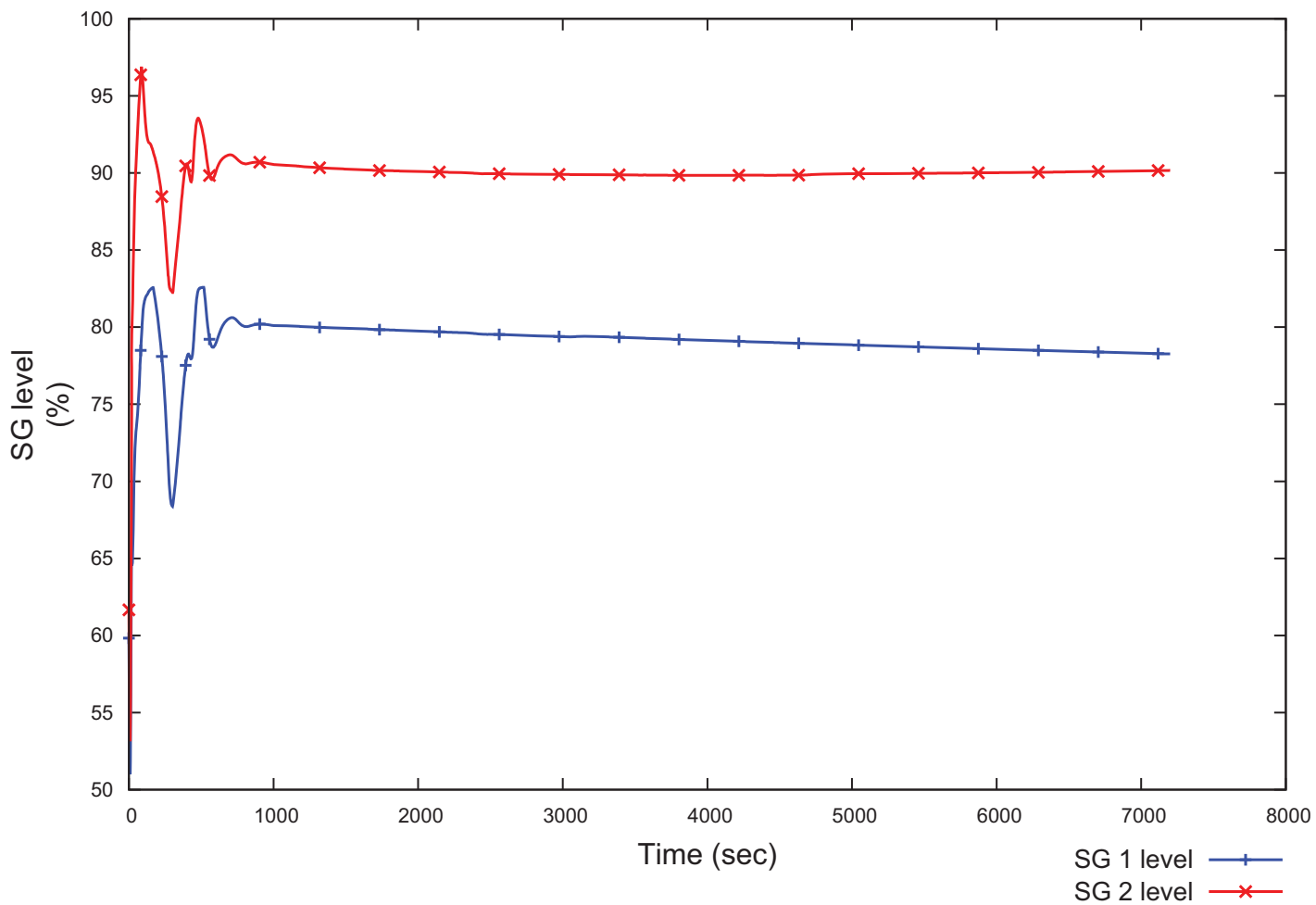


Figure 15.1-19: Secondary Flow (15.1.3 Increase in Steam Flow)

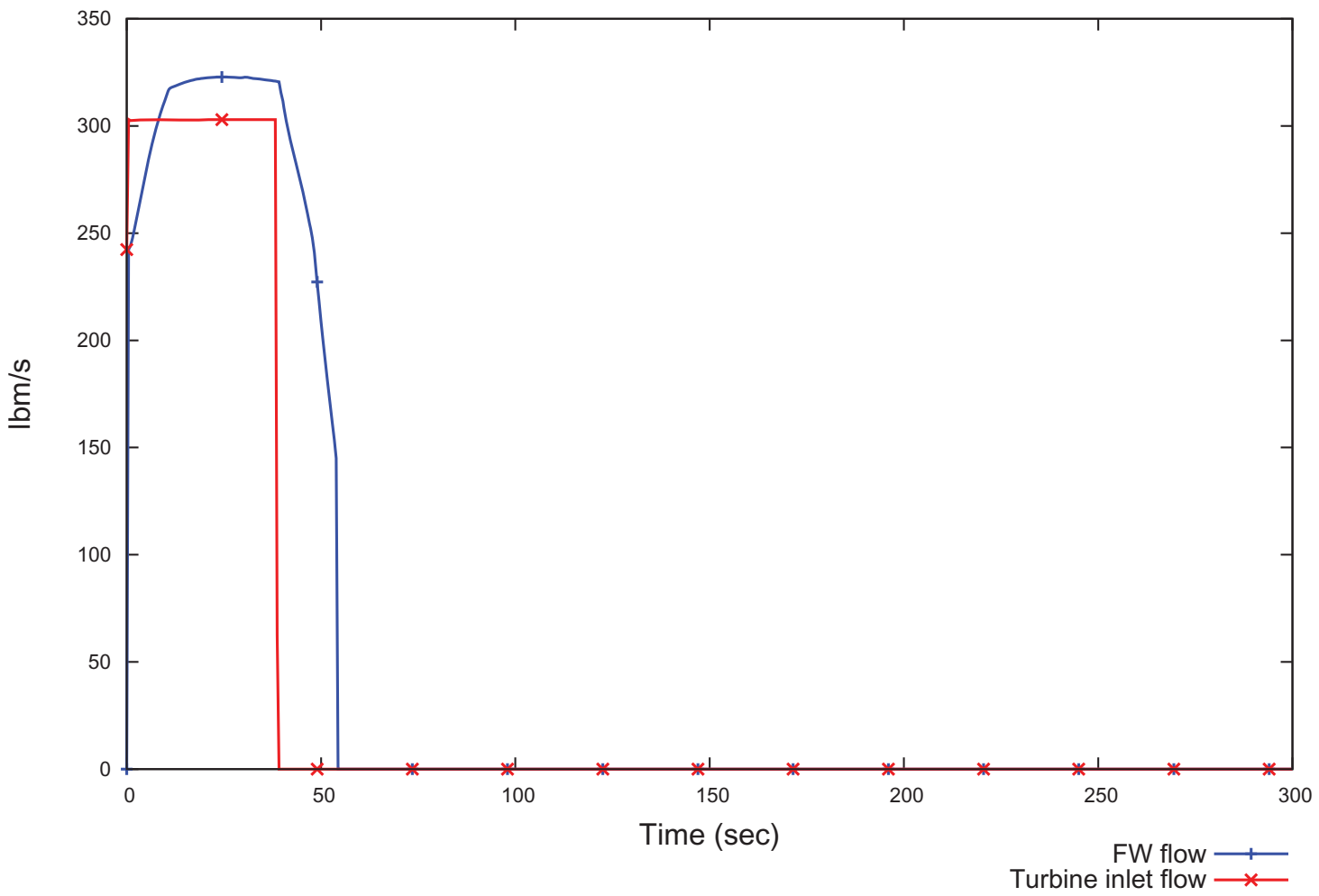


Figure 15.1-20: Primary Temperatures (15.1.3 Increase in Steam Flow)

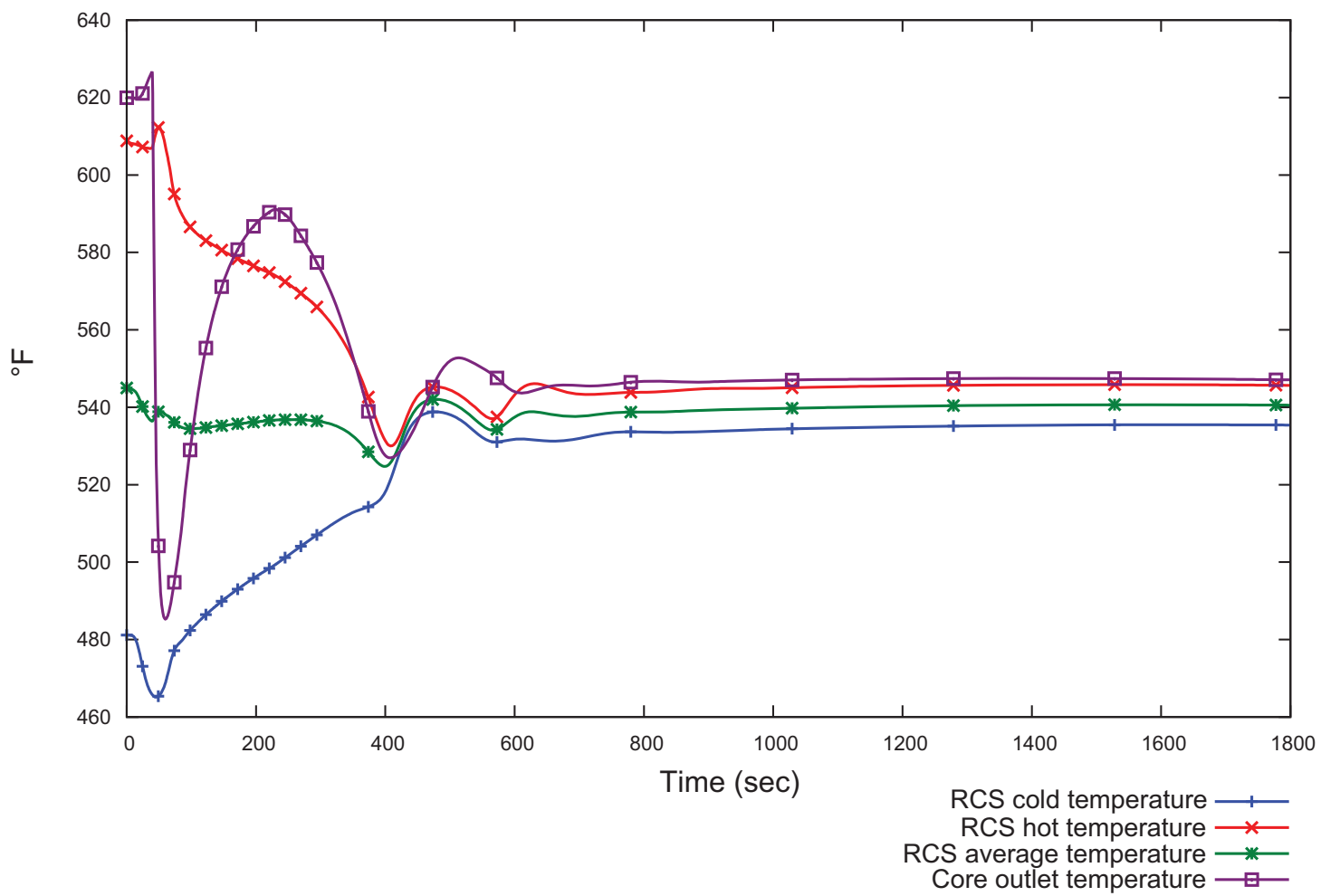


Figure 15.1-21: Pressurizer Level (15.1.3 Increase in Steam Flow)

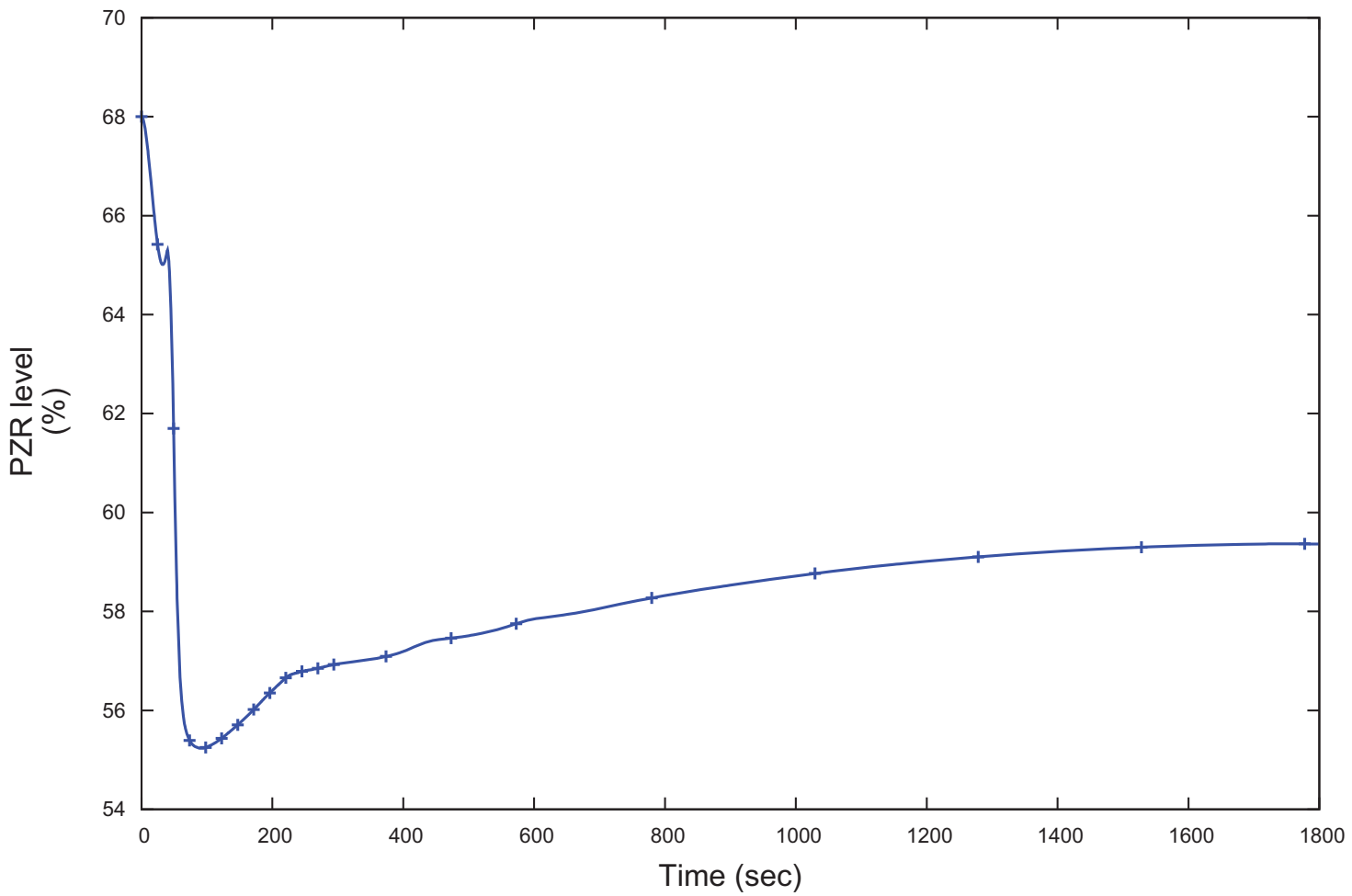


Figure 15.1-22: Event Reactivity (15.1.3 Increase in Steam Flow)

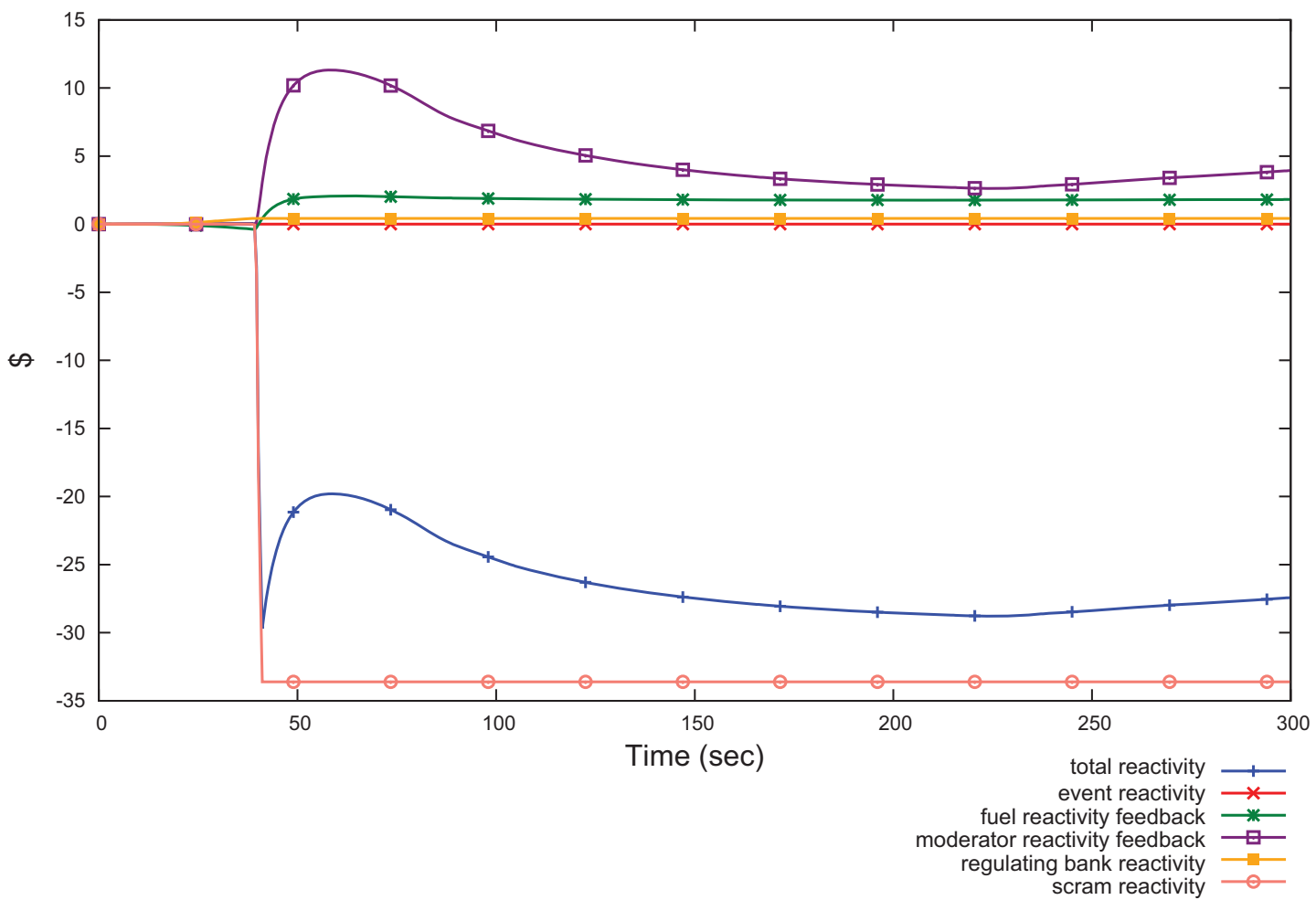


Figure 15.1-23: Reactor Power (15.1.3 Increase in Steam Flow)

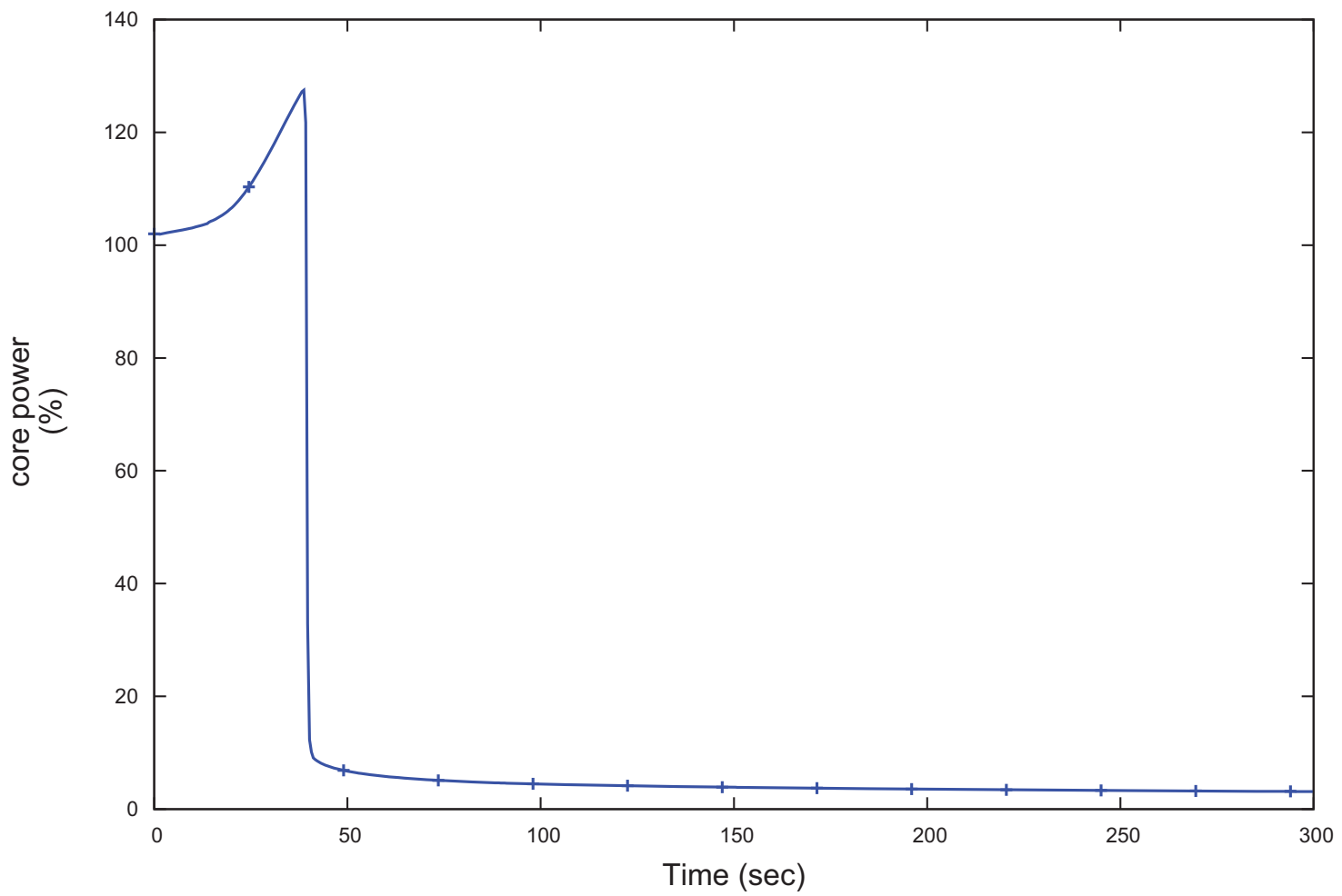


Figure 15.1-24: Primary Pressures (15.1.3 Increase in Steam Flow)

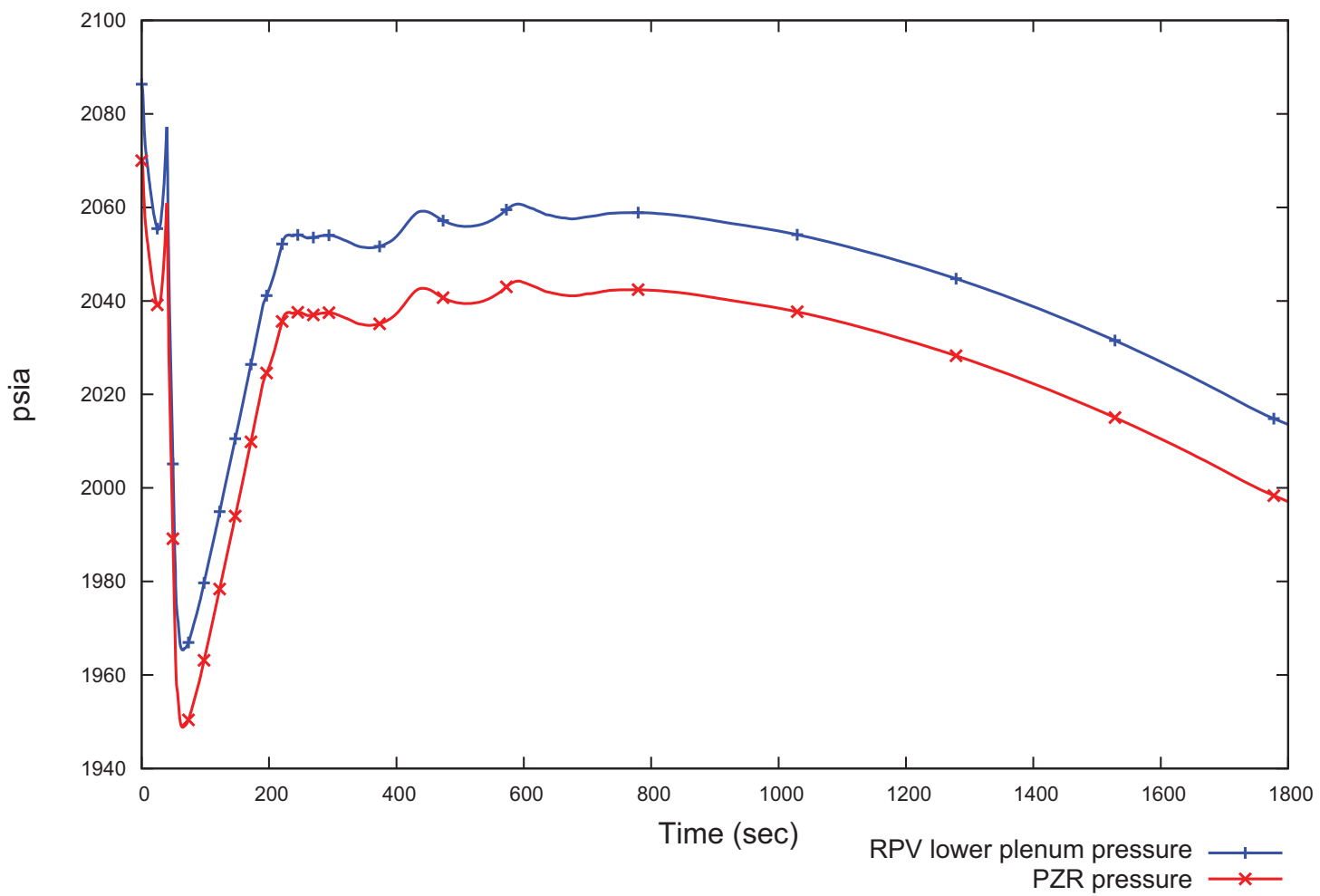


Figure 15.1-25: Steam Generator Superheat (15.1.3 Increase in Steam Flow)

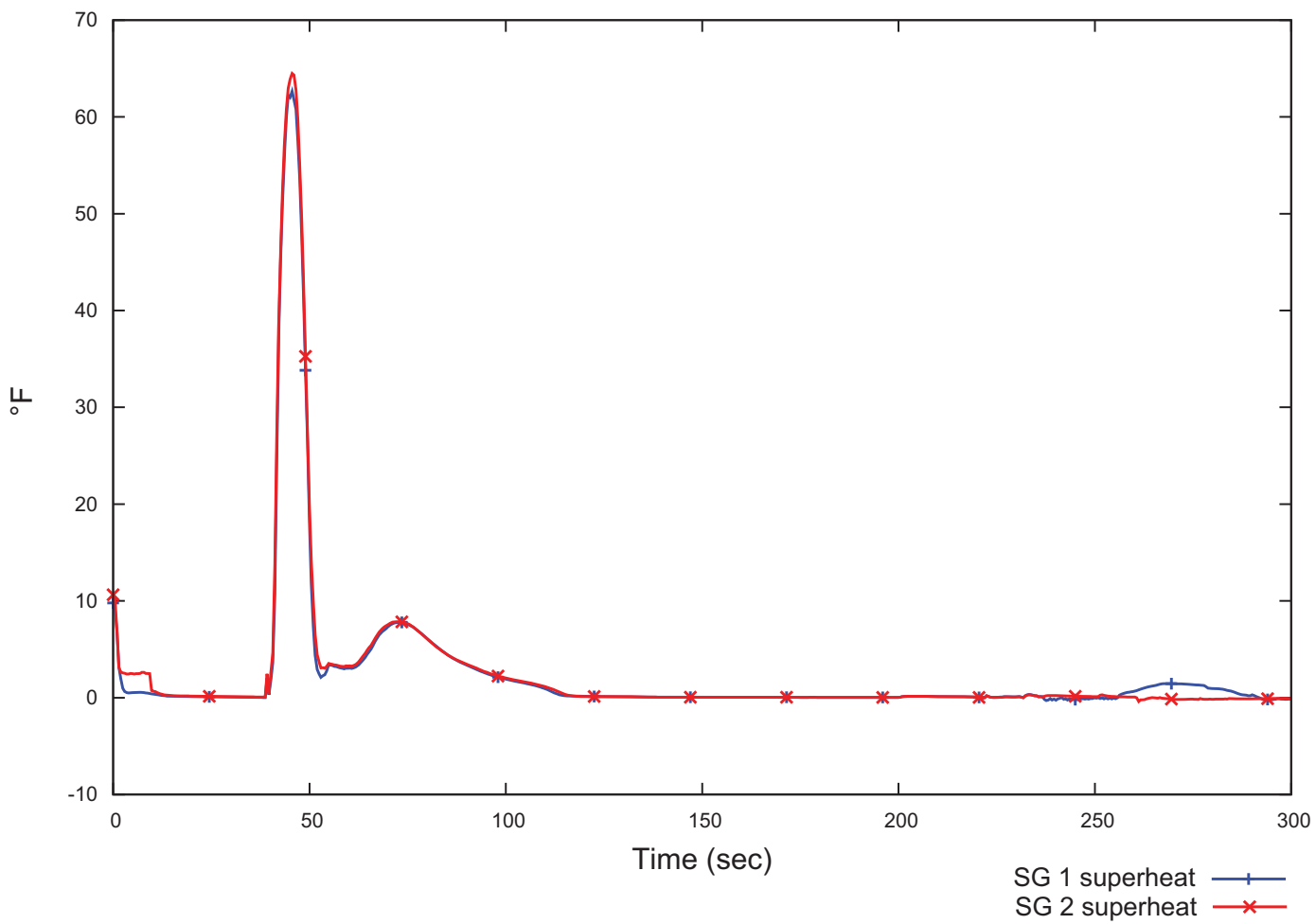


Figure 15.1-26: Reactor Coolant System Flow (15.1.3 Increase in Steam Flow)

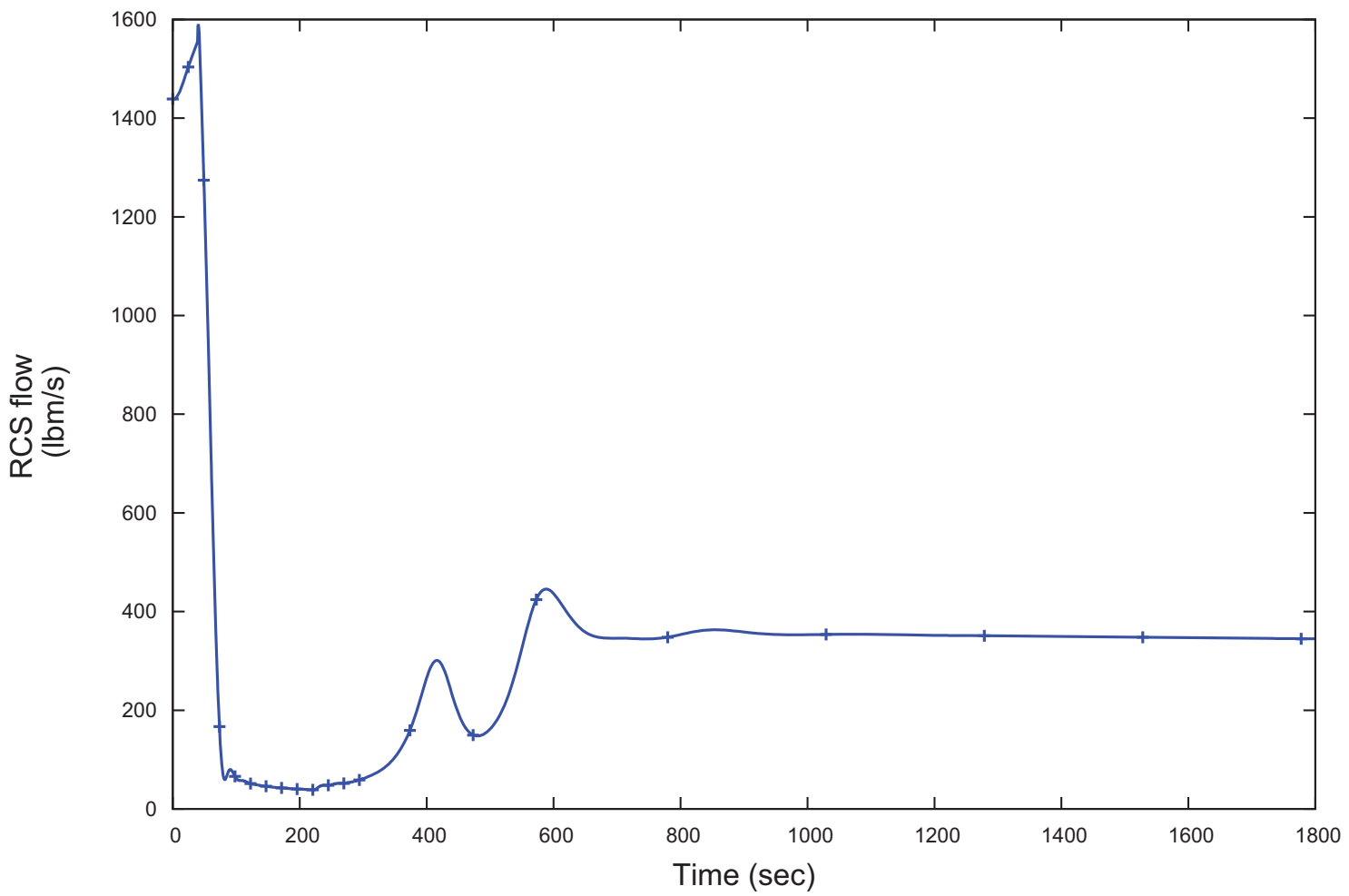
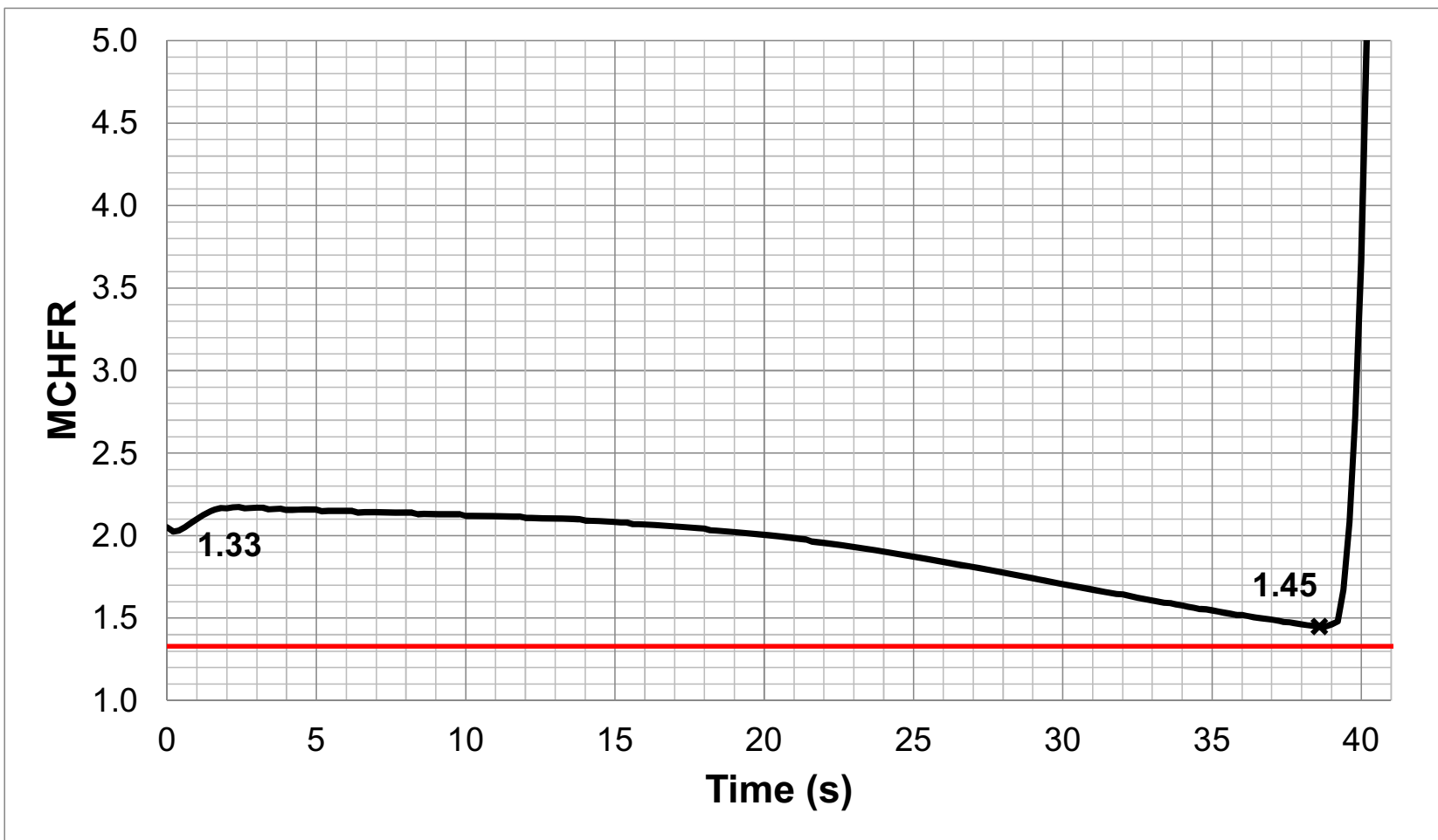


Figure 15.1-27: Critical Heat Flux Ratio (15.1.3 Increase in Steam Flow)



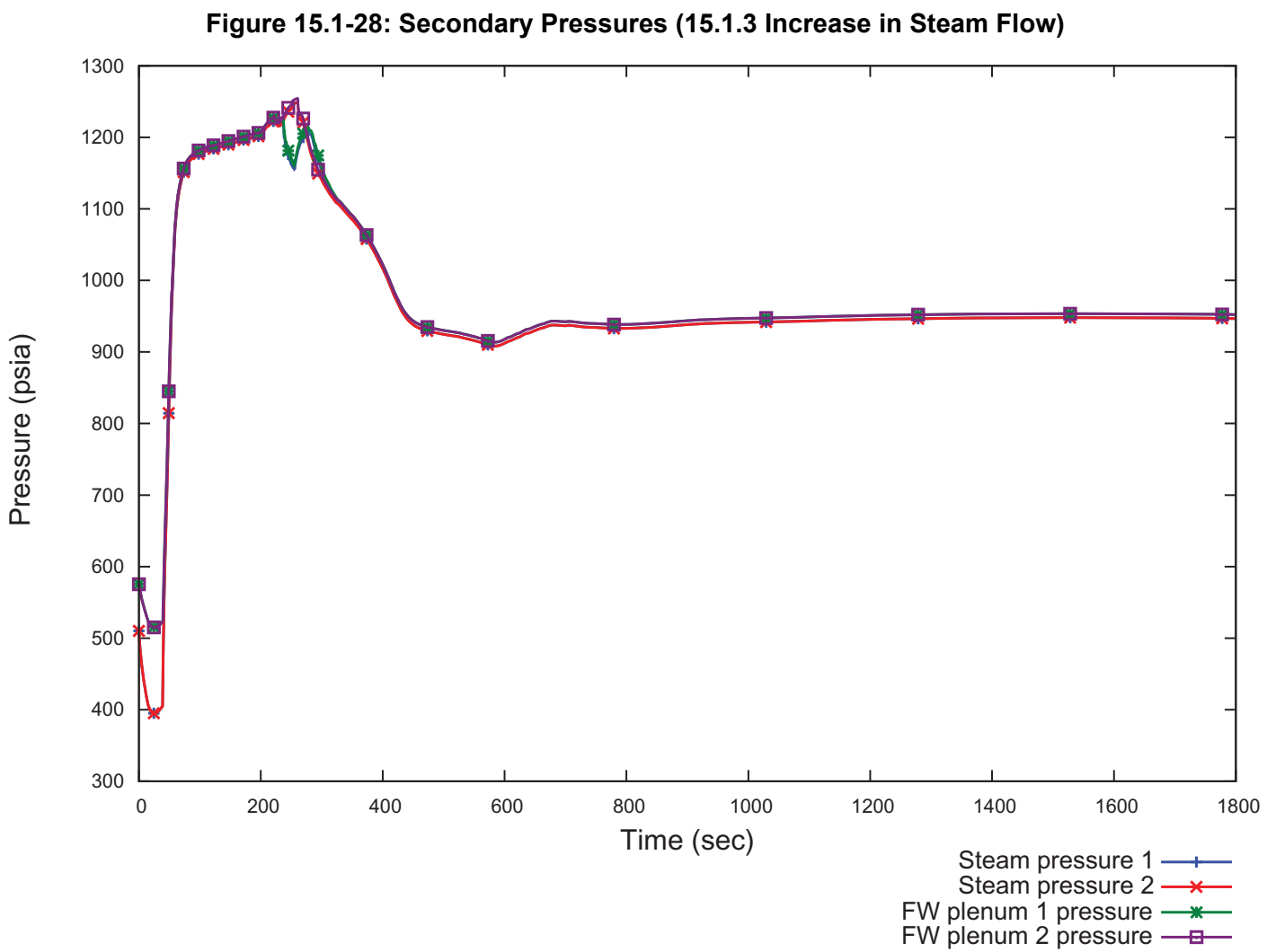
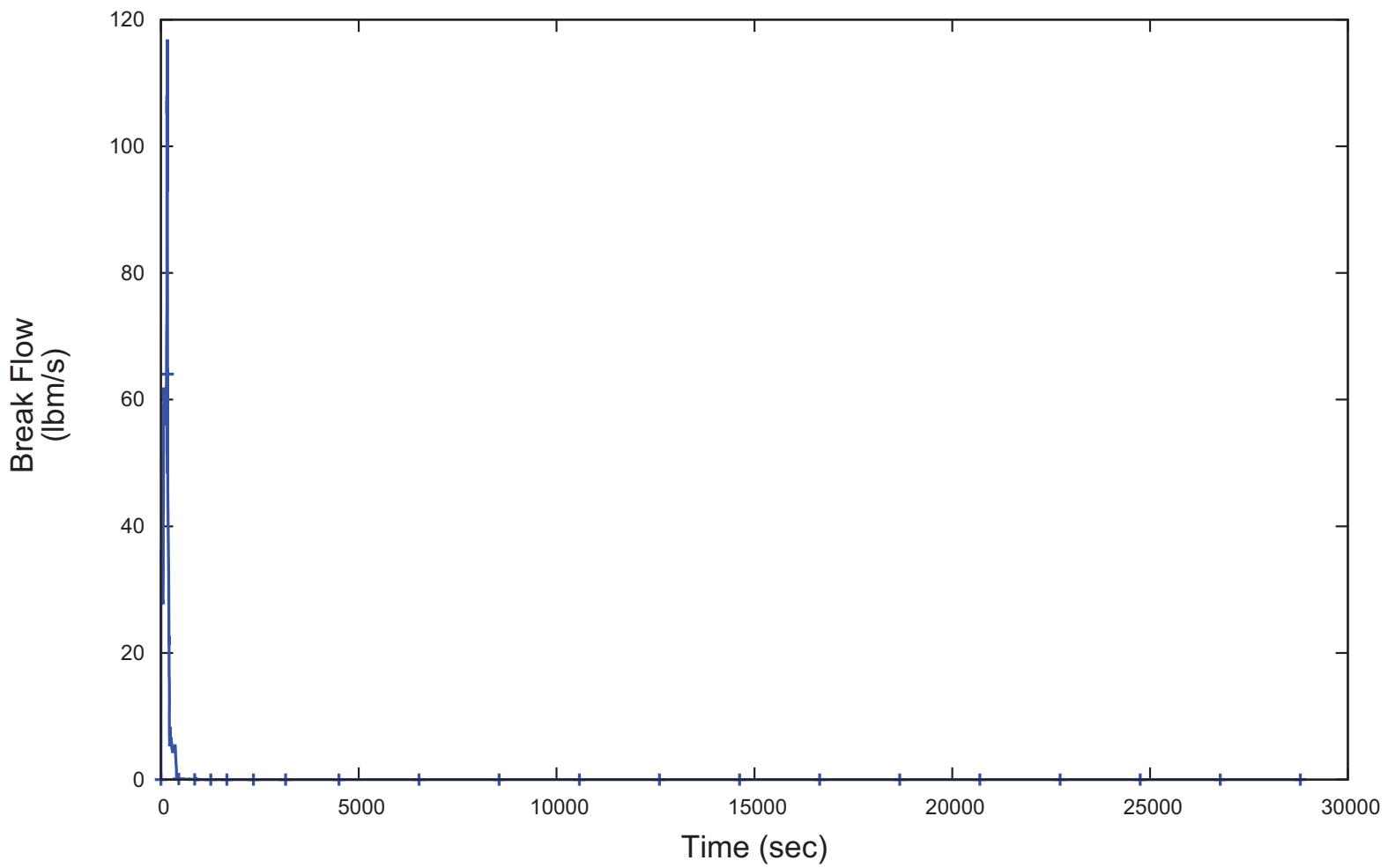


Figure 15.1-29: Break Flow Rate (15.1.5 Steam Piping Failure - Limiting Reactor Coolant System Pressure Case)



EC-116465 Rev. 1 (Attachment 1, Figure 25)

Figure 15.1-30: Reactor Pressure Vessel Lower Plenum Pressure (15.1.5 Steam Piping Failure - Limiting Reactor Coolant System Pressure Case)

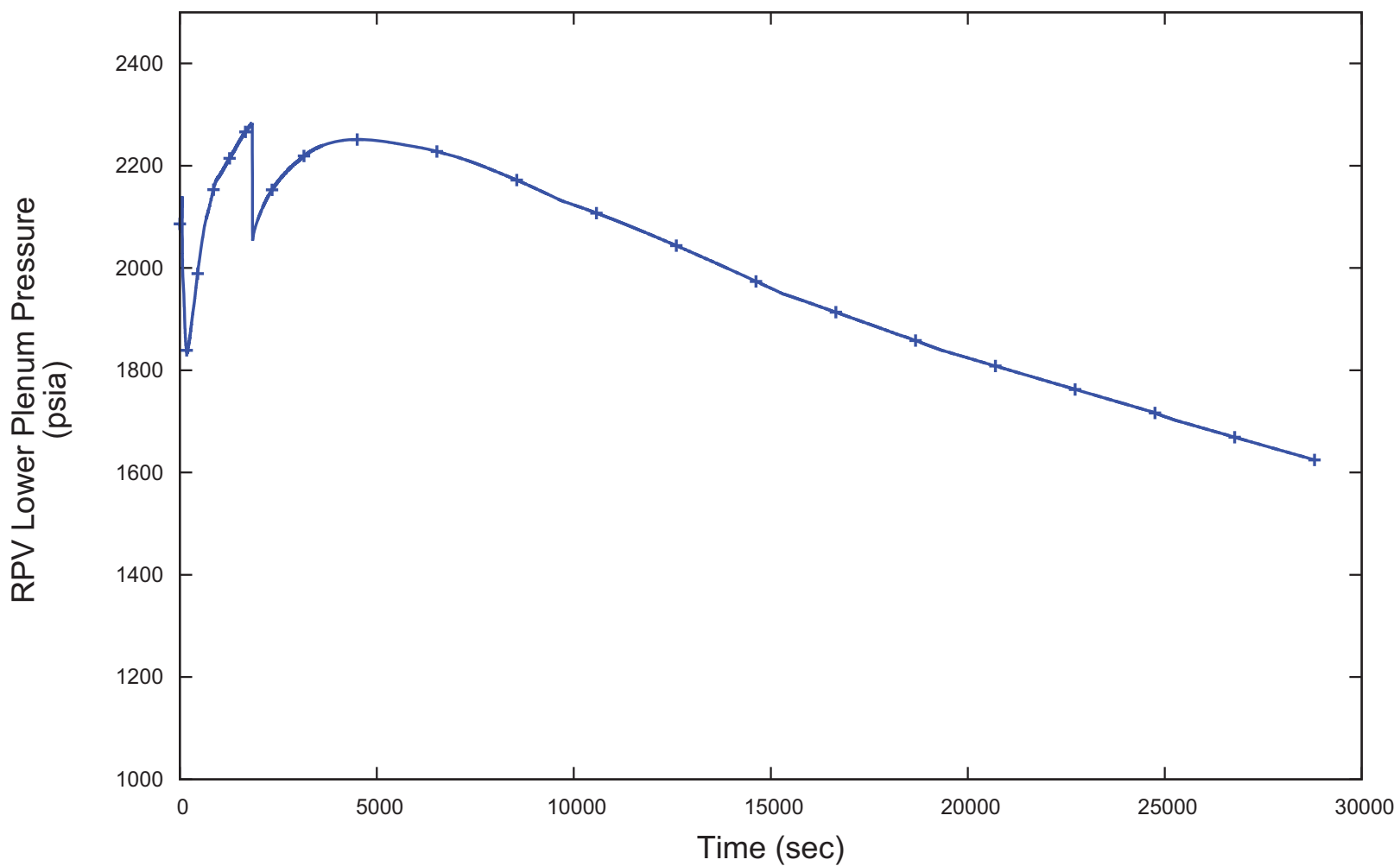


Figure 15.1-31: Reactor Power (15.1.5 Steam Piping Failure - Limiting Reactor Coolant System Pressure Case)

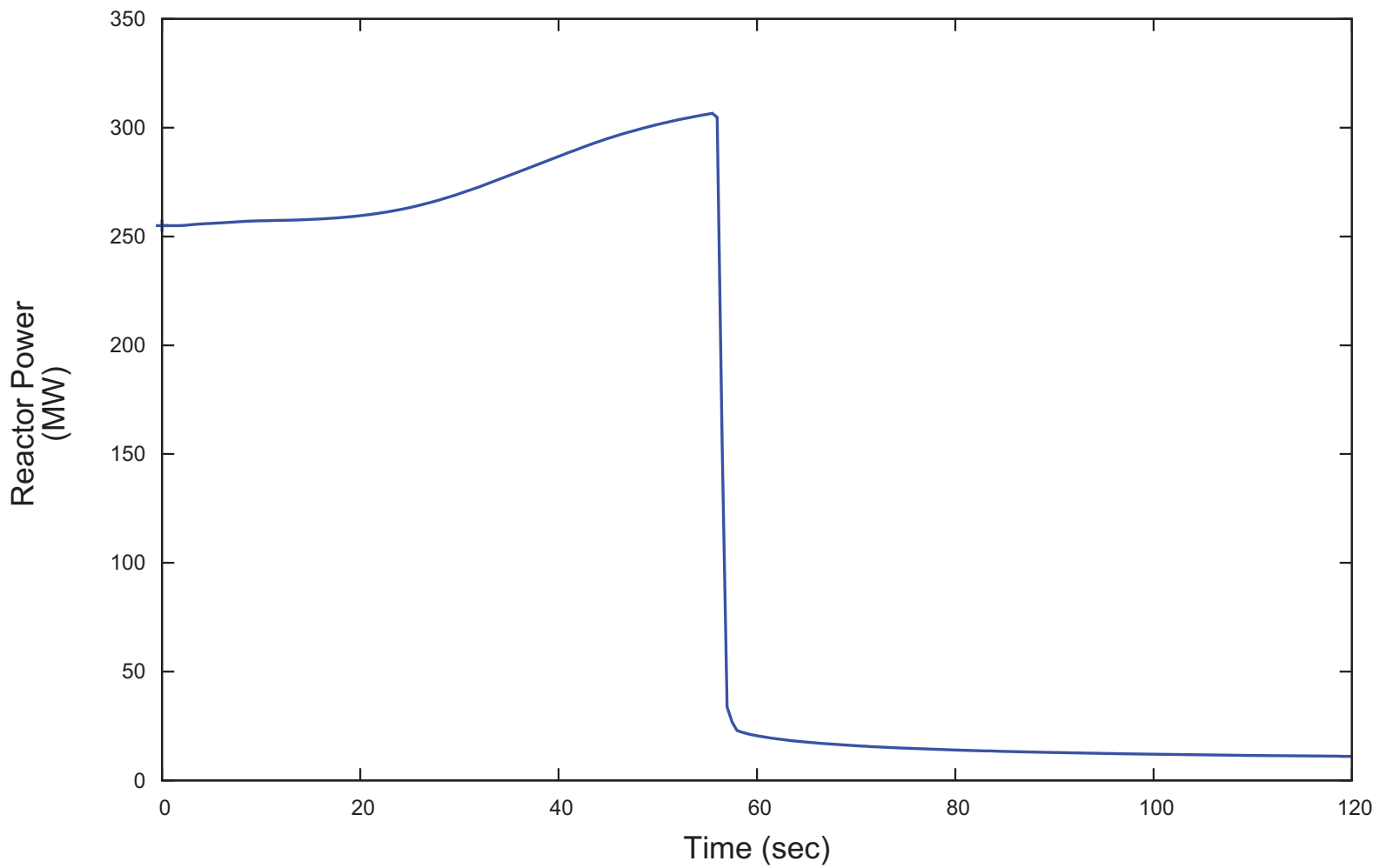


Figure 15.1-32: Reactor Coolant System Average Temperature (15.1.5 Steam Piping Failure - Limiting Reactor Coolant System Pressure Case)

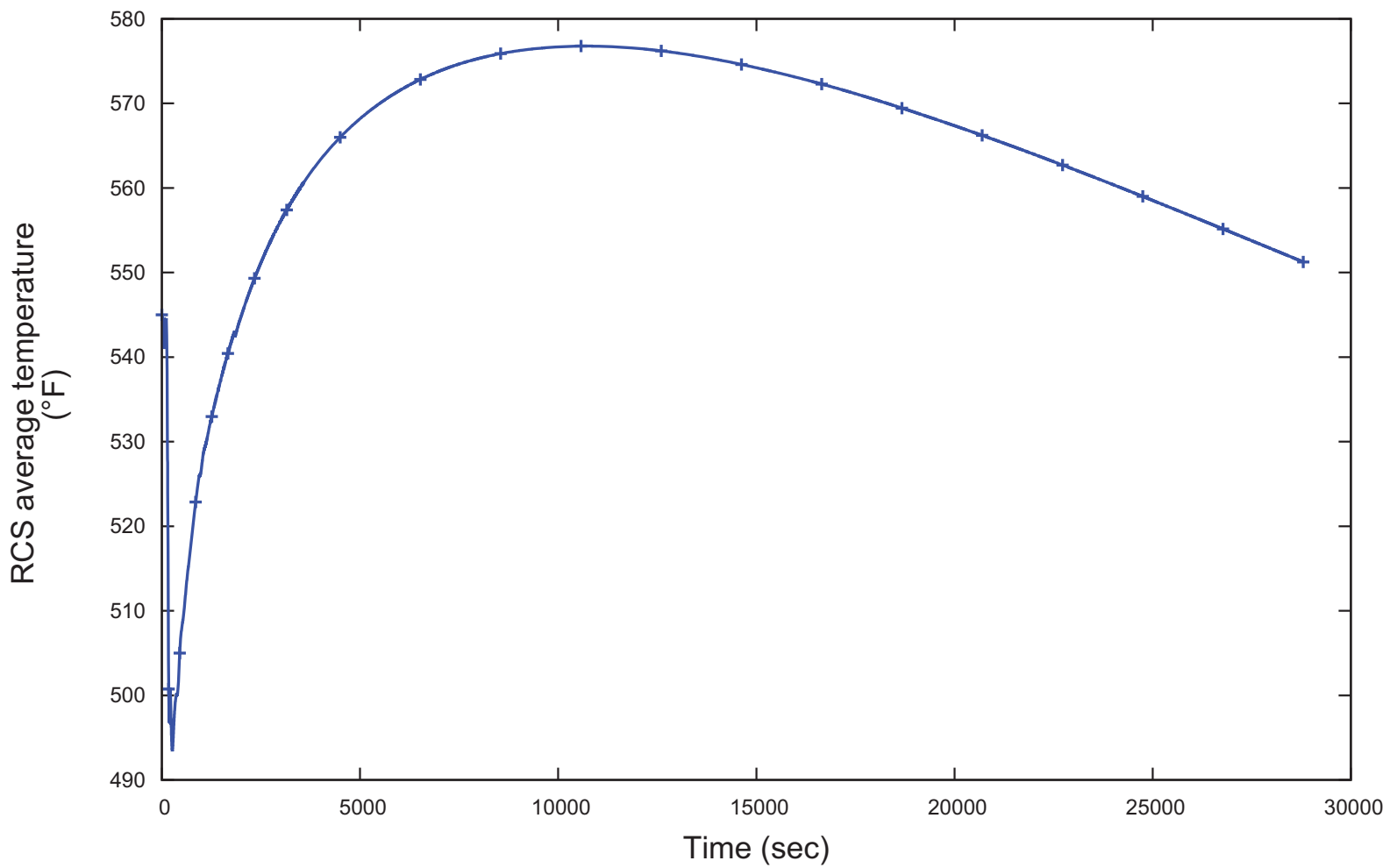


Figure 15.1-33: Reactor Coolant System Flow (15.1.5 Steam Piping Failure - Limiting Reactor Coolant System Pressure Case)

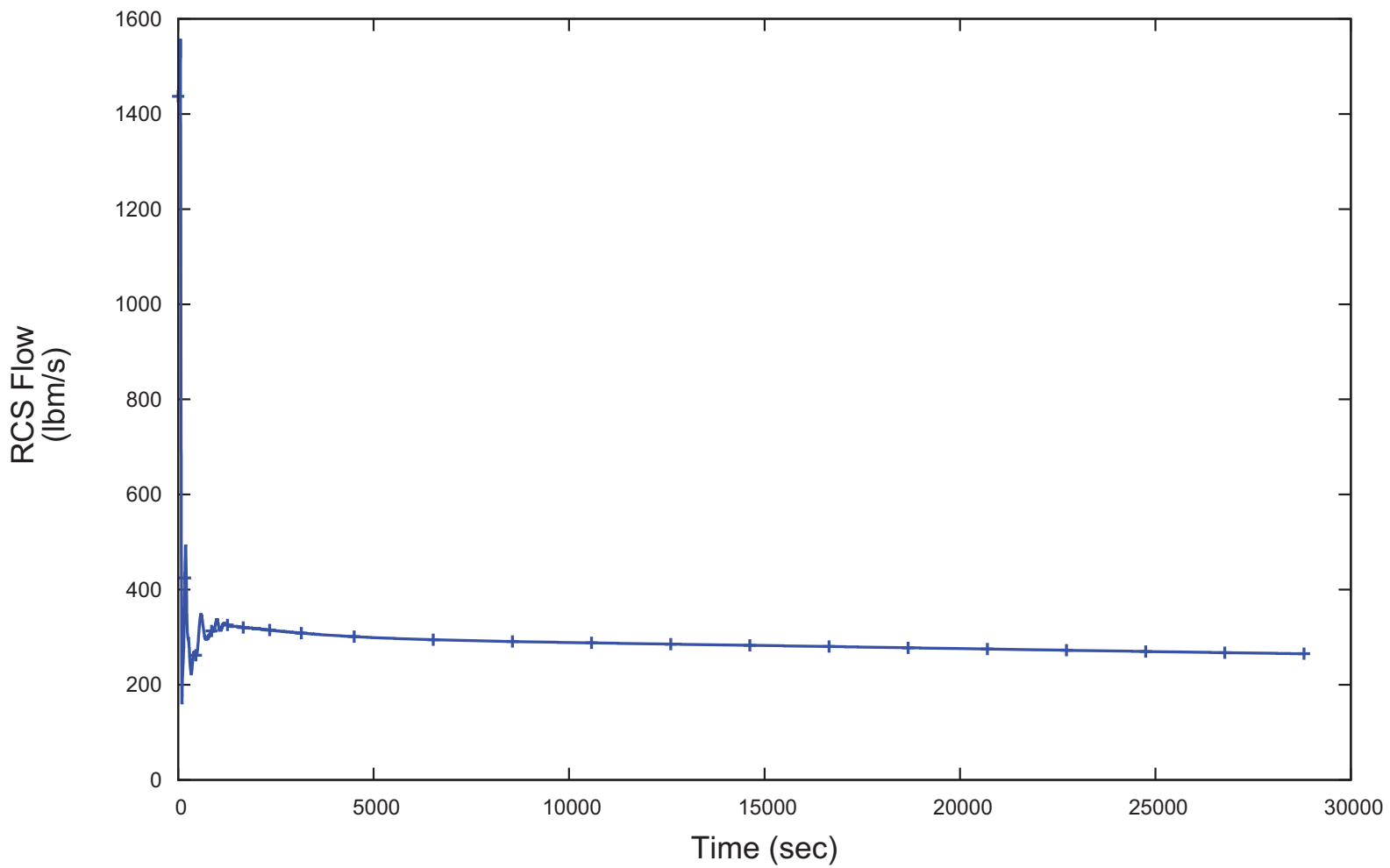


Figure 15.1-34: Decay Heat Removal System Flow Train 1 (15.1.5 Steam Piping Failure - Limiting Reactor Coolant System Pressure Case)

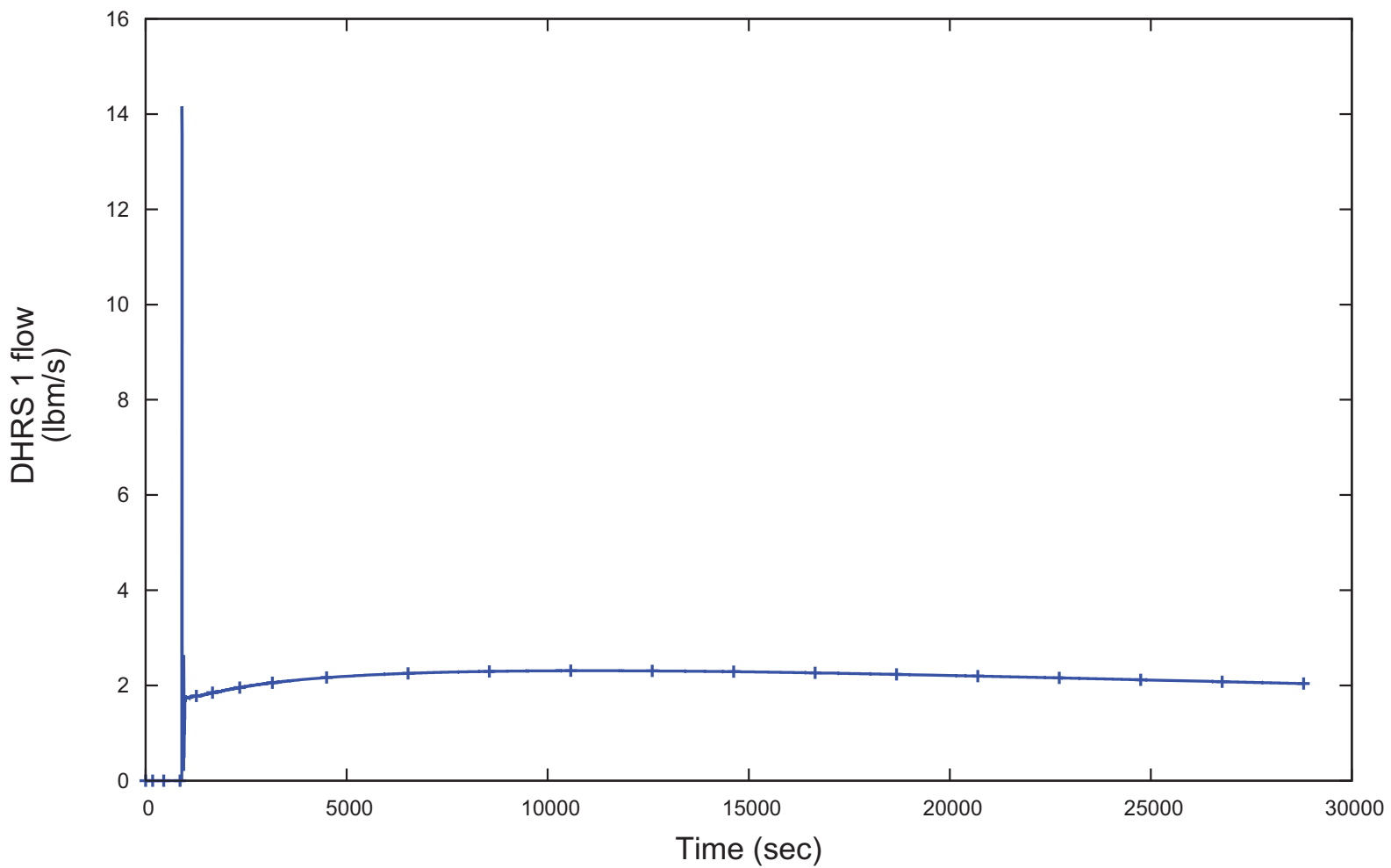


Figure 15.1-35: Decay Heat Removal System Flow Train 2 (15.1.5 Steam Piping Failure - Limiting Reactor Coolant System Pressure Case)

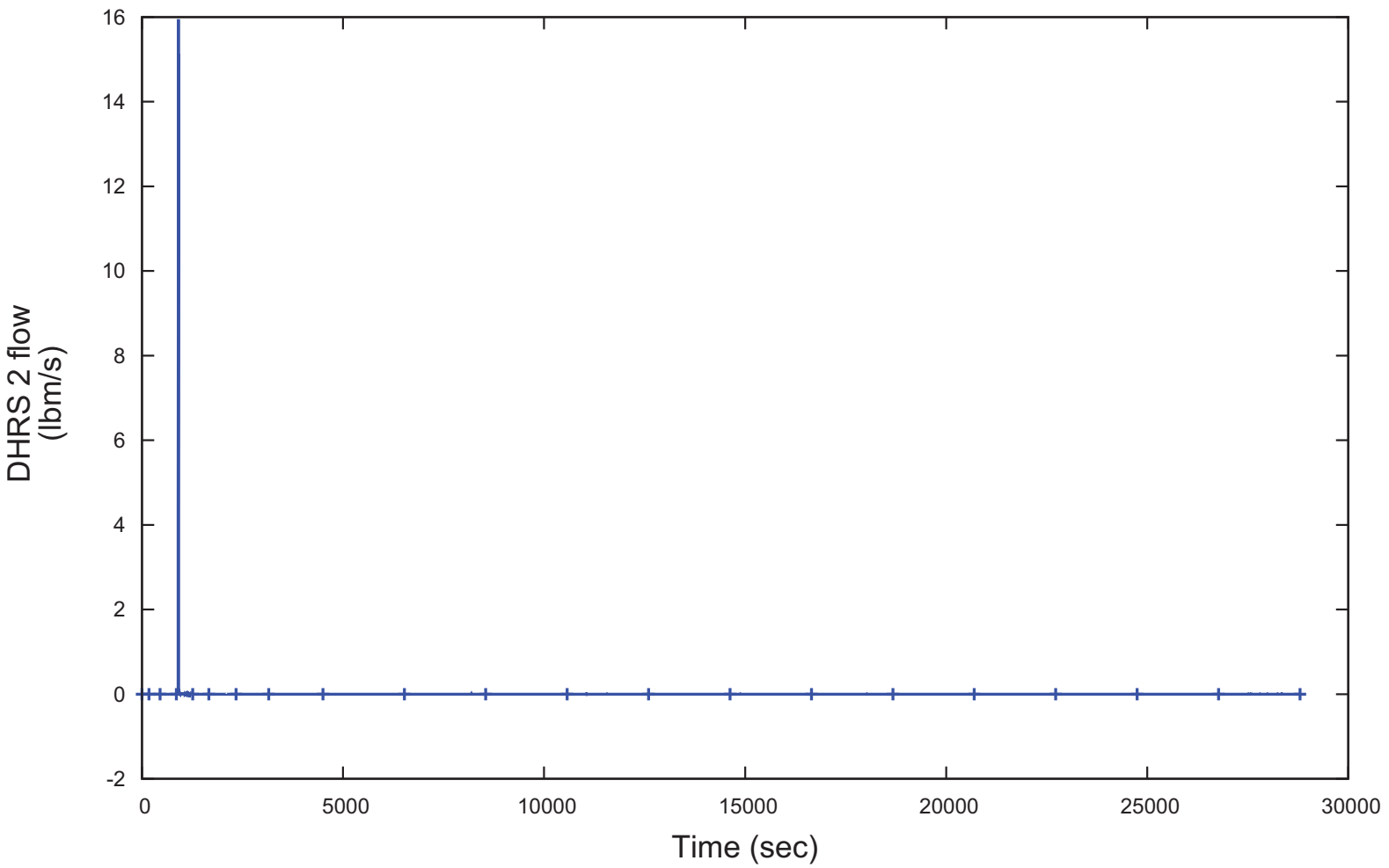


Figure 15.1-36: Pressure Steam Generator 1 (15.1.5 Steam Piping Failure - Limiting Steam Generator Pressure Case)

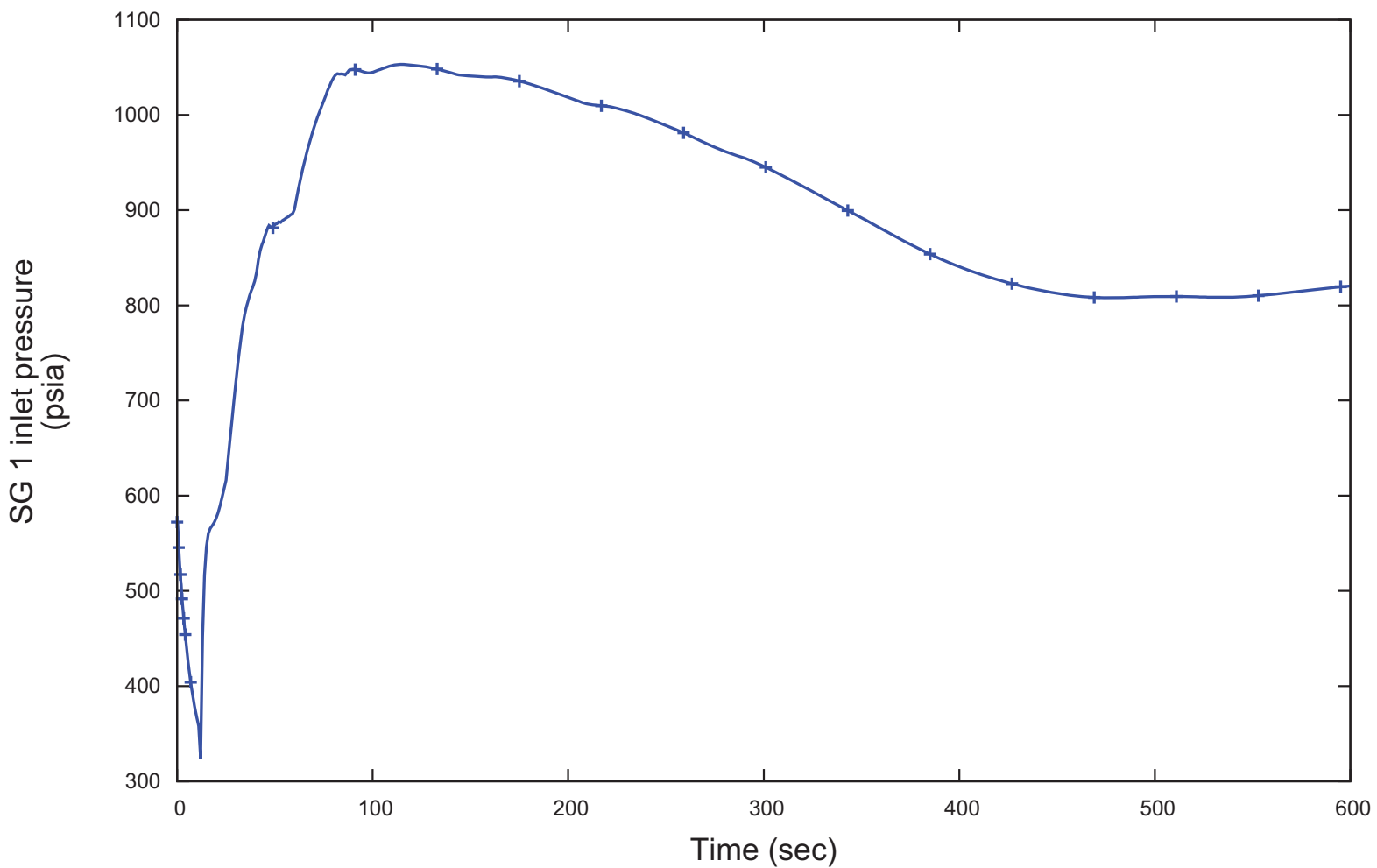


Figure 15.1-37: Pressure Steam Generator 2 (15.1.5 Steam Piping Failure - Limiting Steam Generator Pressure Case)

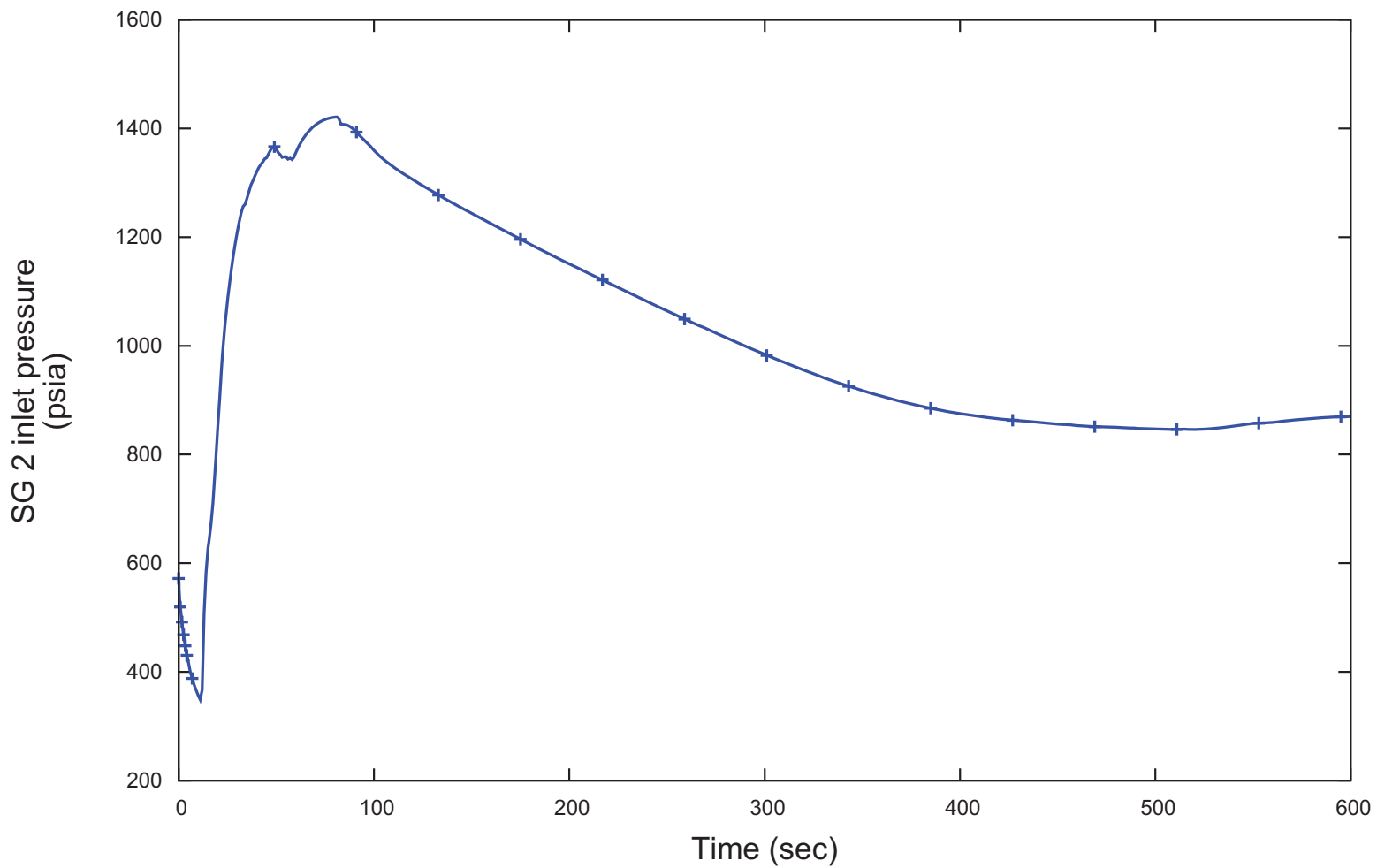


Figure 15.1-38: Break Flow Rate (15.1.5 Steam Piping Failure - Limiting Minimum Critical Heat Flux Ratio Case)

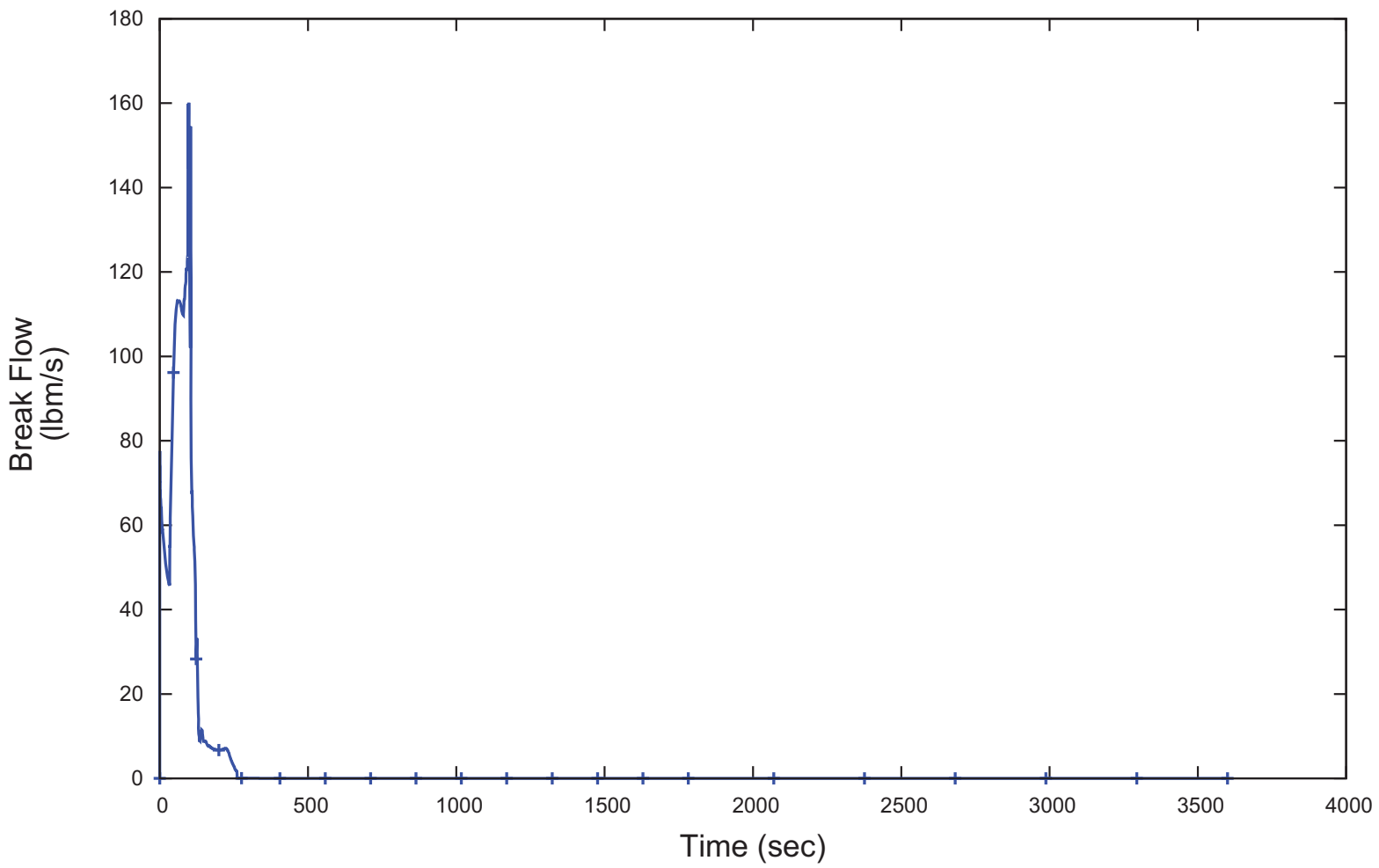


Figure 15.1-39: Event Reactivity (15.1.5 Steam Piping Failure - Limiting Minimum Critical Heat Flux Ratio Case)

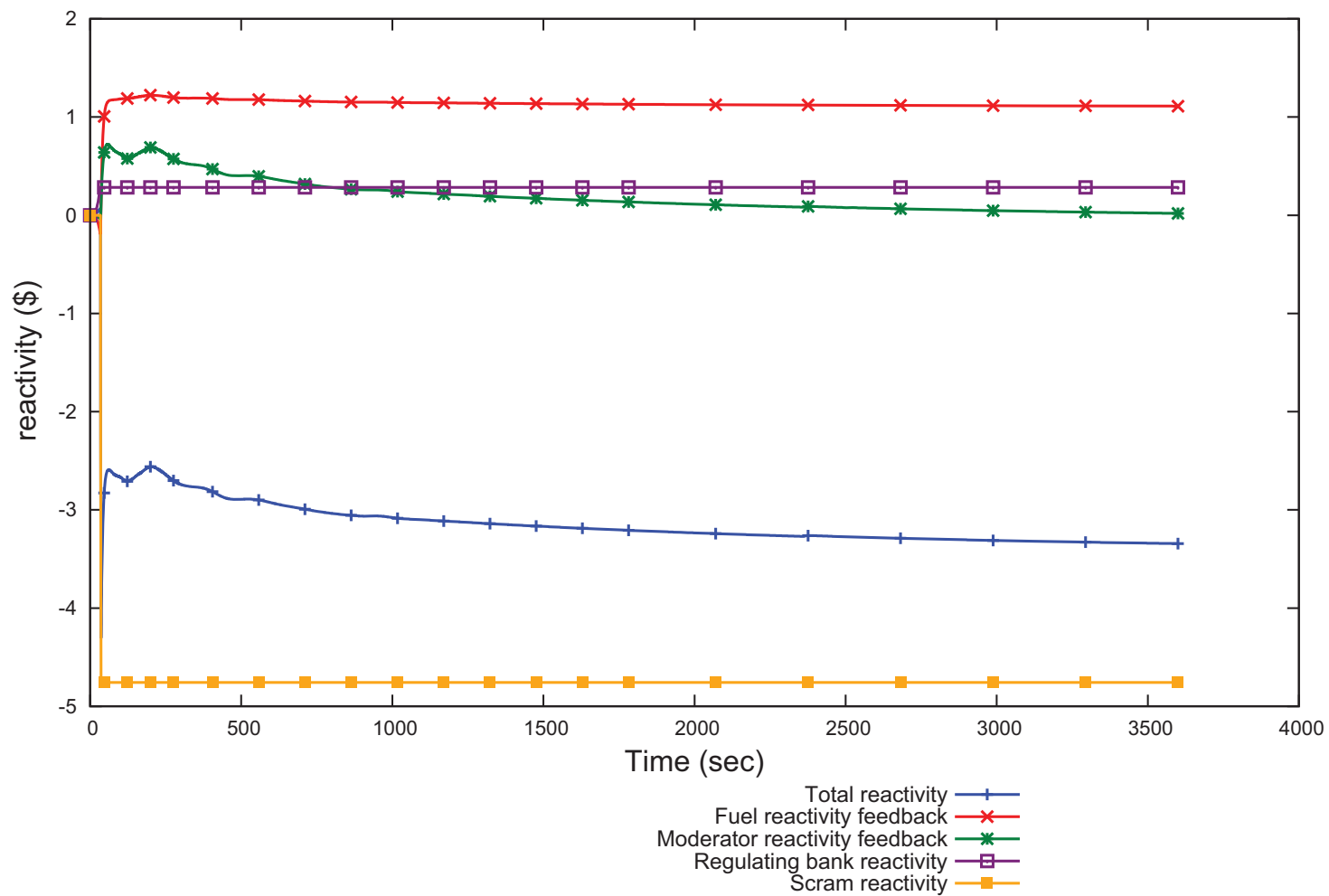


Figure 15.1-40: Reactor Power (15.1.5 Steam Piping Failure - Limiting Minimum Critical Heat Flux Ratio Case)

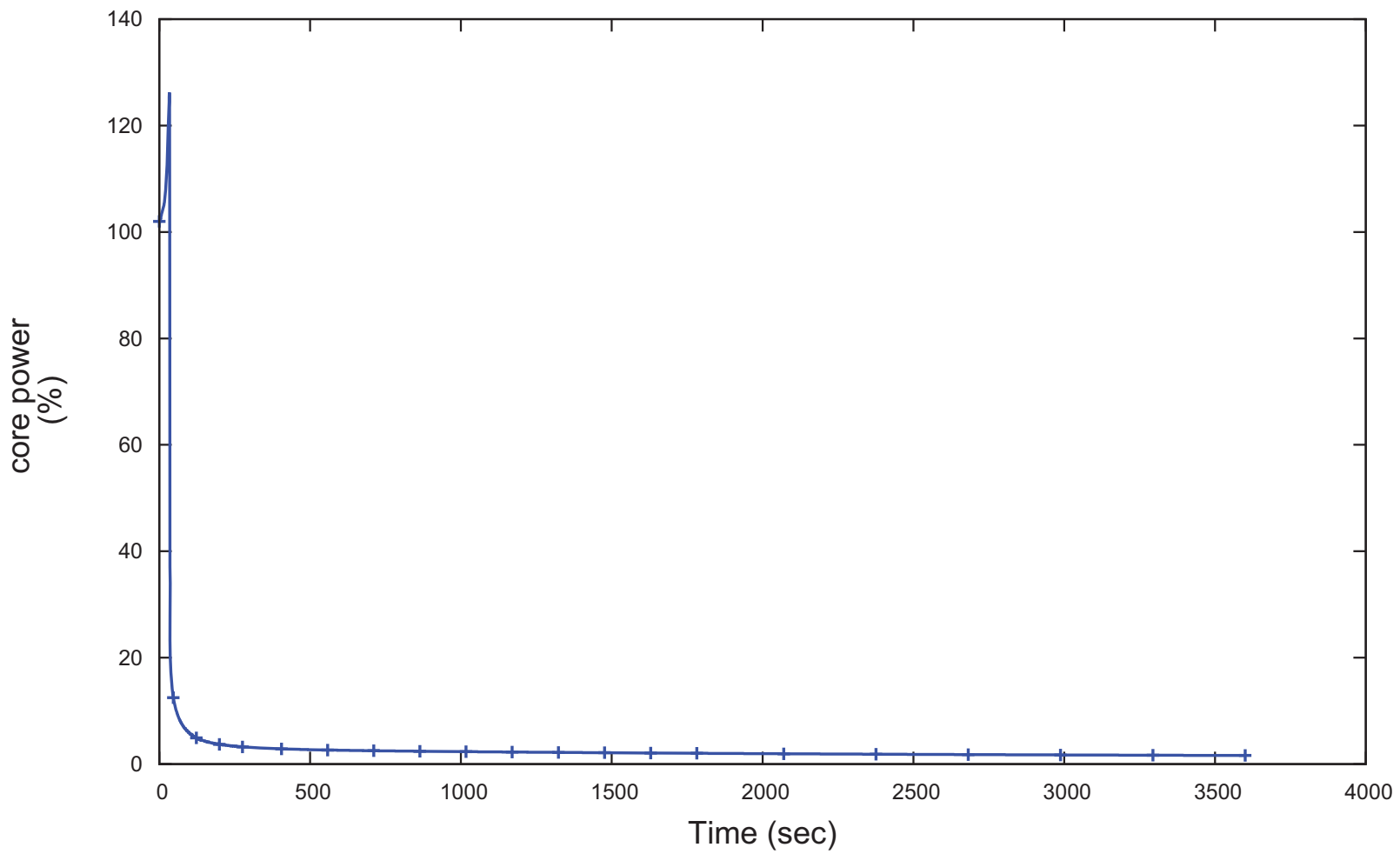


Figure 15.1-41: Volume Average Fuel Temperature (15.1.5 Steam Piping Failure - Limiting Minimum Critical Heat Flux Ratio Case)

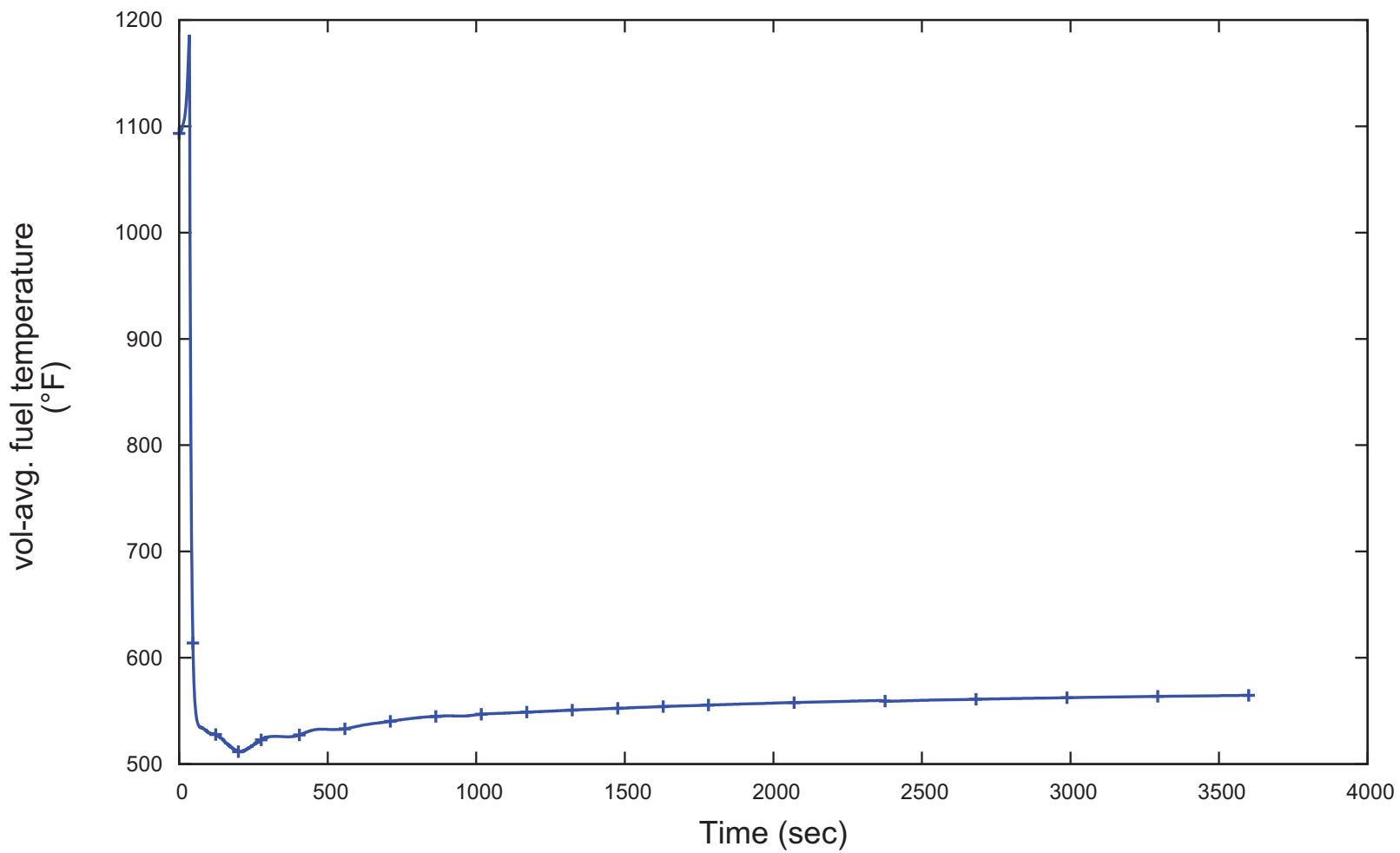


Figure 15.1-42: Secondary Pressures (15.1.5 Steam Piping Failure - Limiting Minimum Critical Heat Flux Ratio Case)

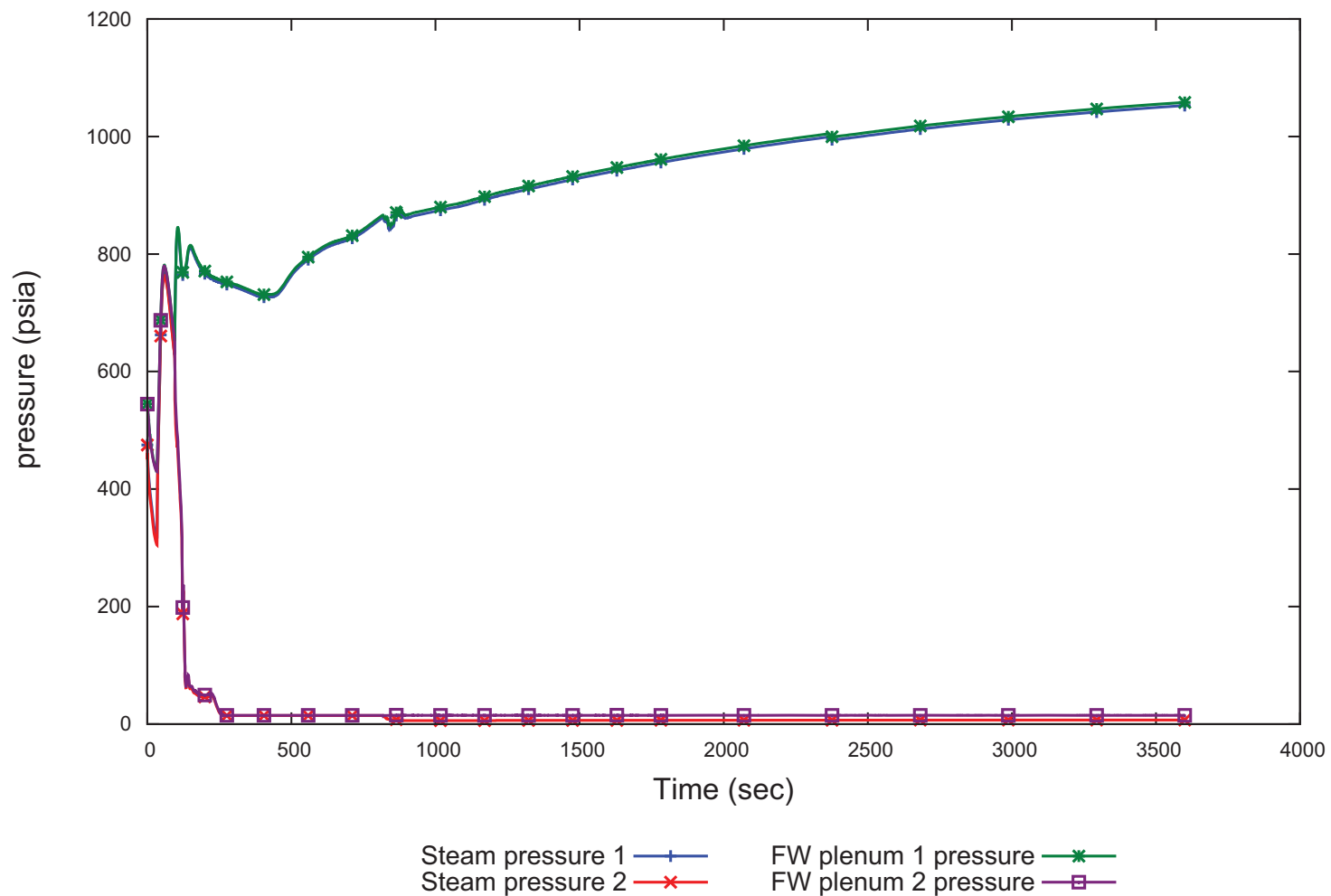


Figure 15.1-43: Critical Heat Flux Ratio (15.1.5 Steam Piping Failure - Limiting Minimum Critical Heat Flux Ratio Case)

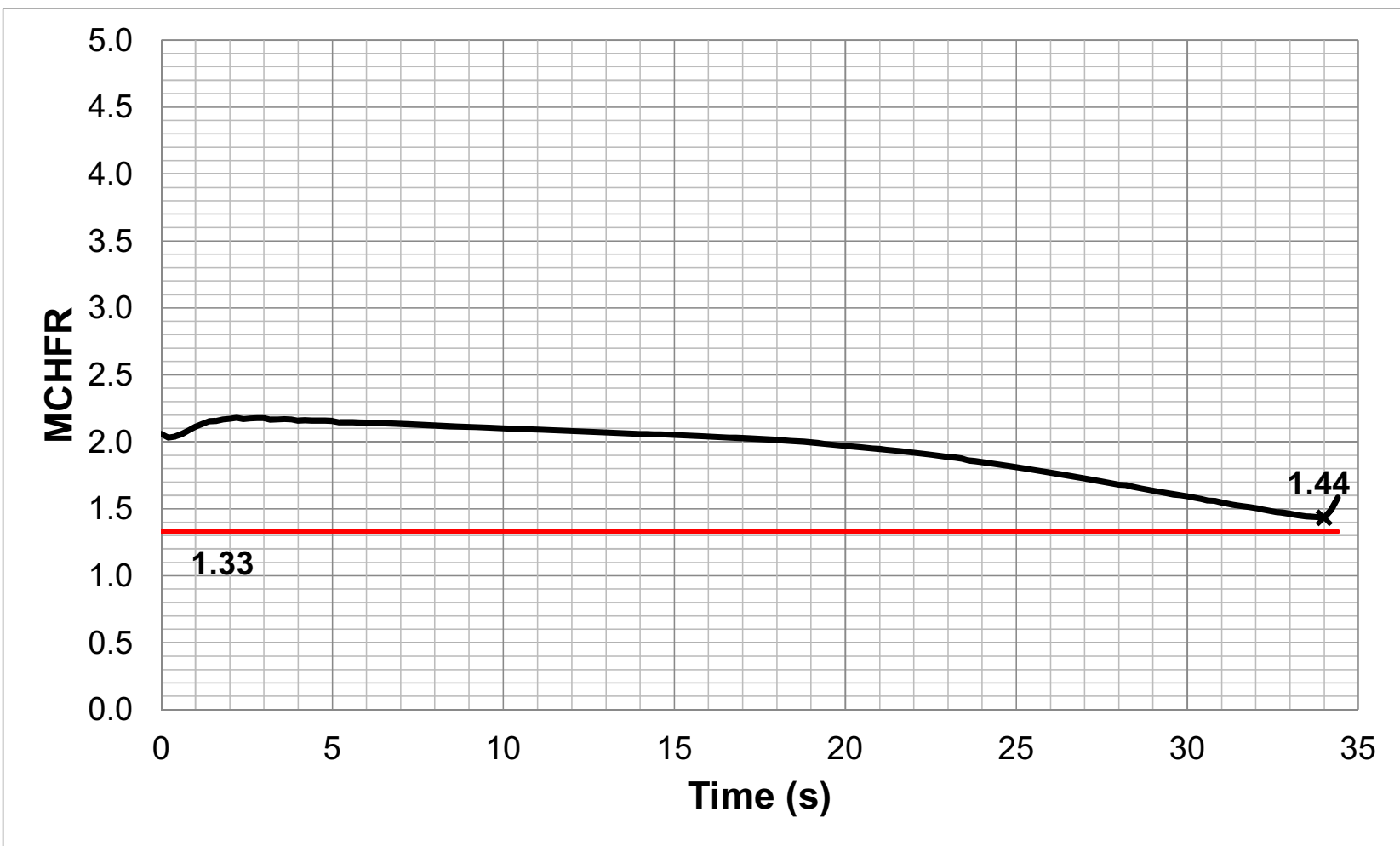


Figure 15.1-44: Reactor Power (15.1.6 Containment Flooding)

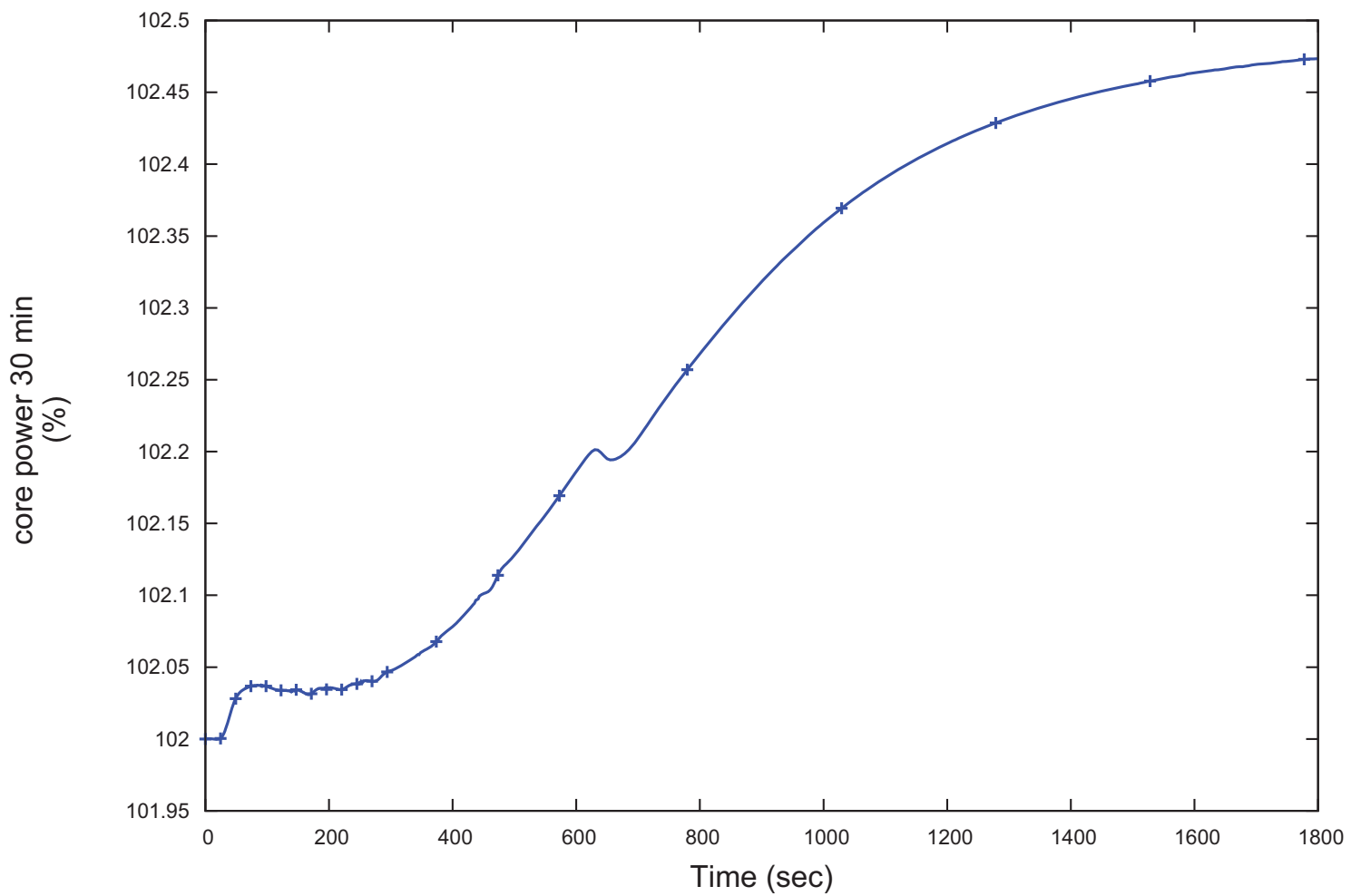
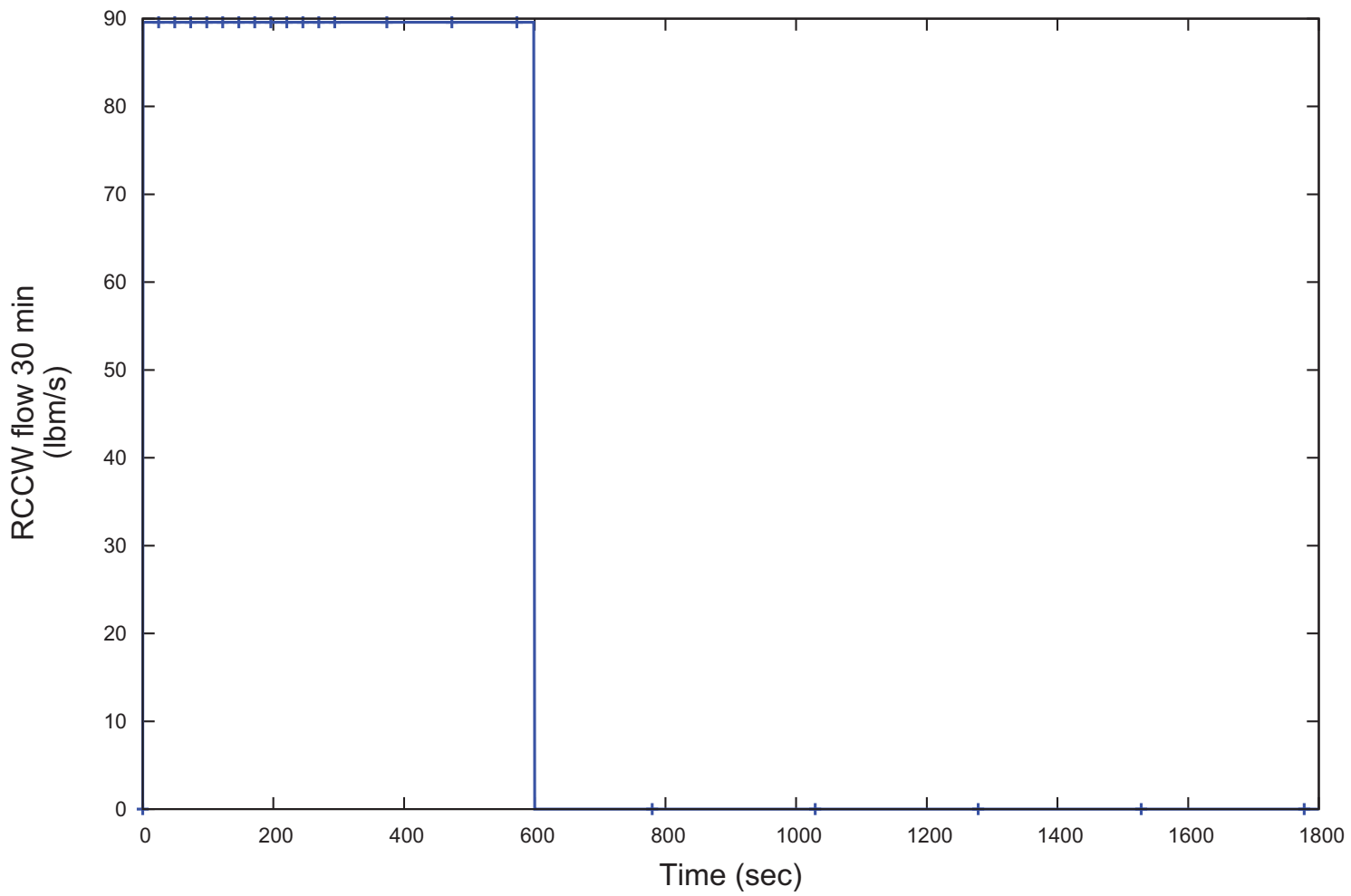
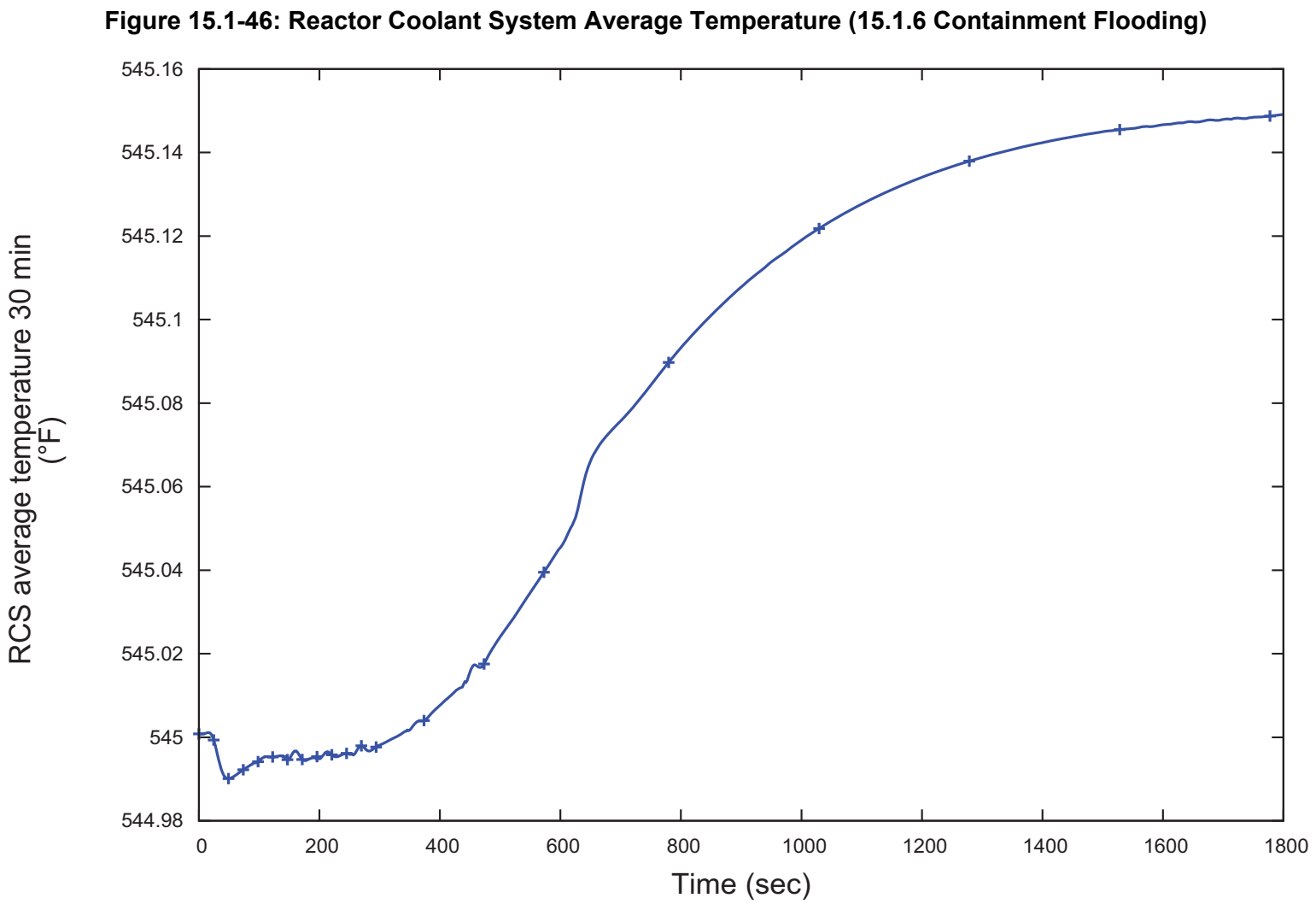


Figure 15.1-45: Reactor Component Cooling Water System Break Flow Rate (15.1.6 Containment Flooding)





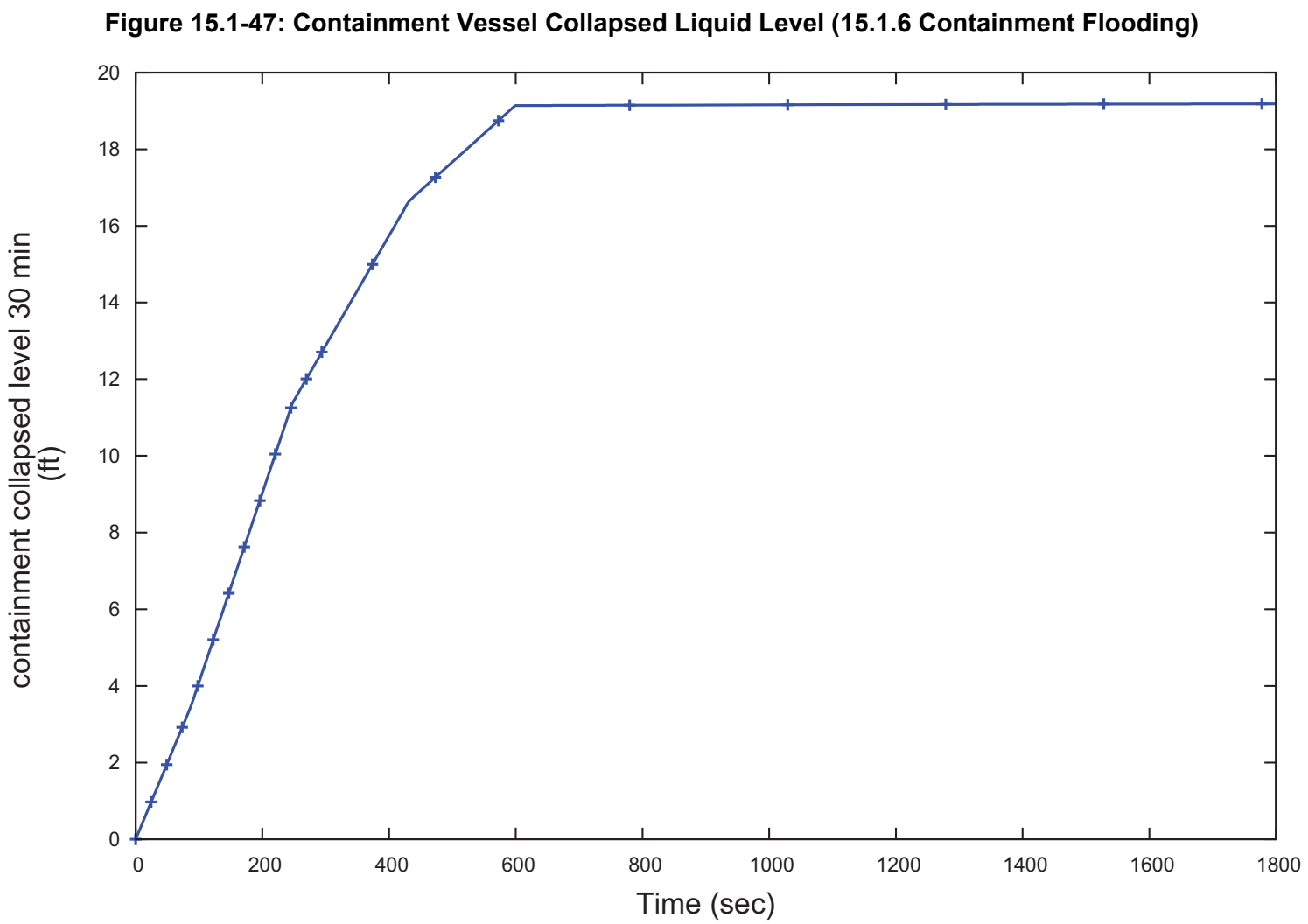
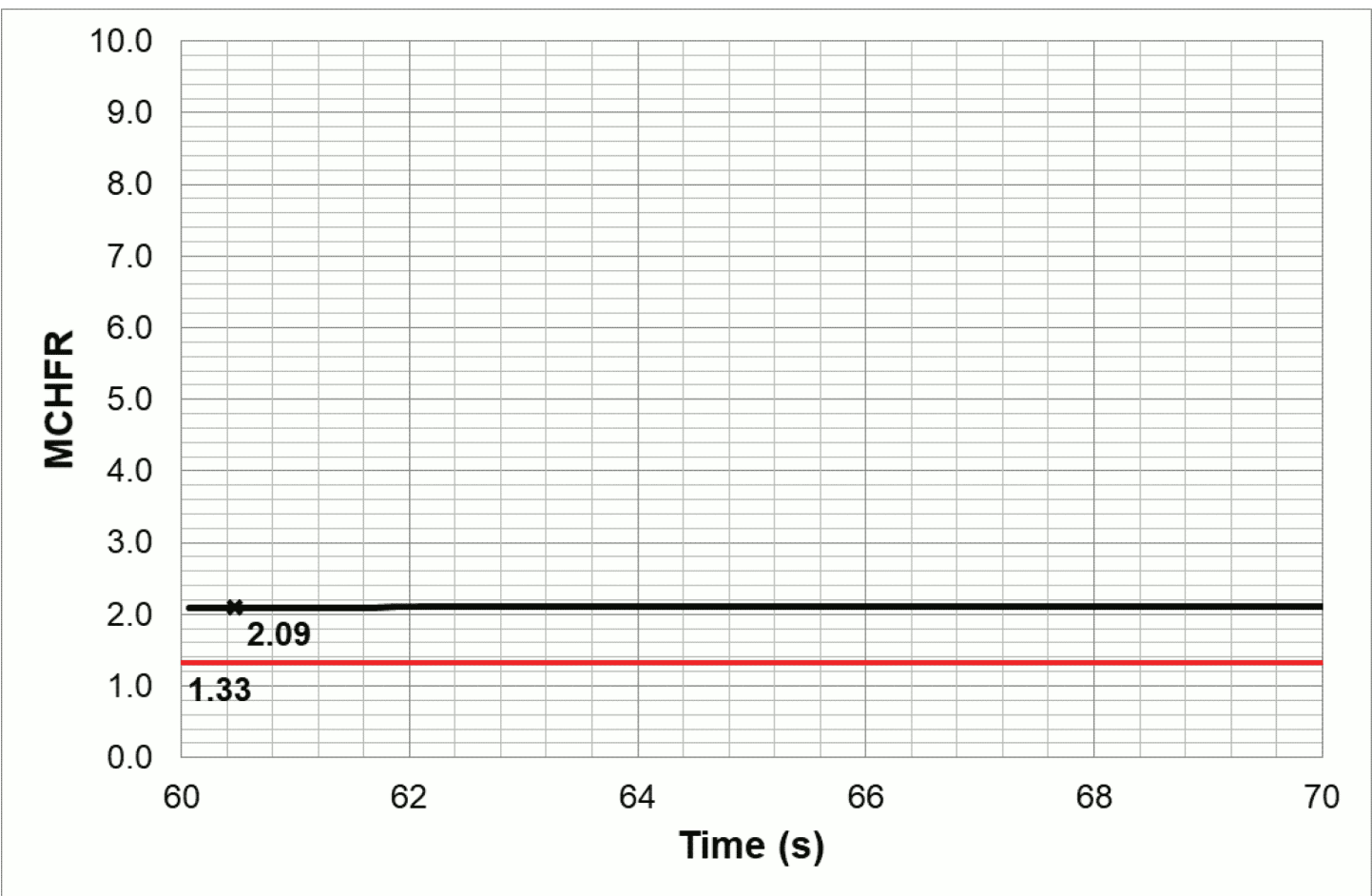


Figure 15.1-48: Critical Heat Flux Ratio (15.1.6 Containment Flooding)



15.2 Decrease in Heat Removal by the Secondary System

This section addresses design-basis events associated with a potential unplanned decrease in primary system heat removal through the steam generators (SGs). The decrease in heat removal causes the primary side temperature and pressure to rise and the pressurizer (PZR) level to increase. The SG pressure also increases. There are eight events that are defined for this category by the NuScale Design Specific Review Standard (DSRS). One unique NuScale Power Module (NPM) event is identified for this event type involving the decay heat removal system (DHRS) (Section 15.2.9).

The evaluation of the first three events in Section 15.2, loss of external load (LOEL), turbine trip (TT), and loss of condenser vacuum (LOCV), produced essentially identical responses for the primary and secondary system pressure and minimum critical heat flux ratio (MCHFR). Therefore, a single set of figures based on an enveloping analysis of these events is provided to show the bounding cases for these events.

The inadvertent closure of a main steam isolation valve (MSIV) figures are presented separately. Section 15.2.5, Steam Pressure Regulator Failure, is not applicable to the NPM and is kept to maintain section organization and numbering. Figures for Section 15.2.6 through Section 15.2.9 are presented individually for each section.

The NuScale DSRS states that for new applications the loss of offsite power (LOOP) must be considered in addition to a single active failure. For NuScale, a LOOP is considered by a loss of alternating current (AC) power at the low voltage AC electrical distribution system (ELVS) level (480 V). The augmented direct current (DC) power system (EDAS) and the normal DC power (EDNS) are not safety-related and their loss of function to provide power is considered as described in Section 15.0.0.

While there are various integrated, automatic control systems that are expected to keep the reactor at power when a TT or LOEL occurs, sensors, signal processing, and final control elements that support these automated controls are not safety-related. Therefore, mitigating control system responses are not credited for the events in this section, but their potential adverse impact to safety functions are considered.

15.2.1 Loss of External Load

15.2.1.1 Identification of Causes and Event Description

An LOEL event is initiated by an electrical disturbance that results in the loss of a portion, or all, of the turbine generator load, leading to a TT. The TT causes the primary and secondary side temperatures and pressures to increase because energy is not being removed through the SGs to the condenser. The reactor trip signal and secondary system isolation (SSI) are initiated on high PZR pressure. DHRS actuates on high PZR pressure. The reactor trip reduces power to decay heat levels. The SSI isolates the feedwater (FW) and steam systems and DHRS actuates and transfers decay heat to the reactor pool.

An LOEL event is an anticipated operational occurrence (AOO) as indicated in Table 15.0-1.

15.2.1.2 Sequence of Events and Systems Operation

The severity of an LOEL event is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the turbine control valves. However, following the valve closures, SG pressure continues to increase until DHRS establishes natural circulation. The DHRS establishes cooling and begins to depressurize the SGs and the reactor coolant system (RCS). Key parameters are shown in Figure 15.2-1 through Figure 15.2-7 for the peak RCS pressure case.

If AC and DC power are available, the plant control systems (PCSs) and the engineered safety features (ESFs) perform as designed with allowances for instrument inaccuracy. Some PCSs are disabled in the analysis because their operation is beneficial with respect to the consequences for an LOEL event. The disabled PCSs are for RCS temperature control, PZR pressure control, PZR level control, steam pressure control, and FW and turbine load control. No operator action is credited to mitigate the effects of an LOEL event.

For the limiting RCS pressure case, the module protection system (MPS) initiates a reactor trip, actuates DHRS and SSI, and deenergizes the PZR heaters on a high PZR pressure signal. The feedwater regulating valves (FWRVs) and secondary MSIVs are nonsafety-related but are credited as having redundant isolation capability in the case of the failure of the feedwater isolation valves (FWIVs) or MSIVs to close. The limiting assumption for RCS pressure during an LOEL event is a loss of AC power at event initiation, with DC power available, primarily due to the heatup caused by an immediate loss of FW. No single failure results in a more limiting RCS pressure for the LOEL event. The RCS pressure increase is mitigated by opening one of the two redundant reactor safety valves (RSVs).

For the limiting SG pressure case, the MPS initiates a reactor trip, actuates SSI and DHRS, and deenergizes the PZR heaters on high PZR pressure. The limiting single failure for an LOEL for peak SG pressure is the failure of a FWIV to close with AC power available until reactor trip. If a FWIV fails to close, FW flow is provided to the SG until the FW pumps are secured or the FWRVs close. The FWRVs are nonsafety-related but are credited to close within 30 seconds in the event of a failure of the safety-related FWIV. The FWRVs get a close signal on SSI actuation. The FWIV failure to close results in the highest SG peak pressure. The peak SG pressure is shown on Figure 15.2-8.

For the limiting MCHFR case, the MPS initiates a reactor trip, actuates the DHRS and SSI, and deenergizes the PZR heaters on high PZR pressure. The limiting assumption for MCHFR during an LOEL event is a loss of AC power at event initiation with DC power available. No single failure results in a more limiting MCHFR for the LOEL event. The RCS pressure reaches the RSV actuation setpoint in this case. The limiting MCHFR versus time is shown in Figure 15.2-9.

The sequence of events for the LOEL, TT, or LOCV transients are described in Table 15.2-4 for the limiting RCS pressure case, Table 15.2-5 for the limiting secondary pressure case, and Table 15.2-6 for the limiting MCHFR case.

15.2.1.3 Thermal Hydraulic and Subchannel Analyses**15.2.1.3.1 Evaluation Models**

The thermal hydraulic analysis of the NPM response to an LOEL is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel critical heat flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.2.1.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator temperature and fuel temperature is taken from the beginning of cycle (BOC). The reactivity coefficients for moderator temperature are least negative at BOC. Thus, they are conservative for undercooling events such as LOEL because they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102 percent. The most reactive rod is assumed to remain out of the core, with a delay of two seconds between the reactor trip signal and trip initiation. The most limiting combination of reactivity coefficients for moderator temperature and fuel temperature is applied. Instrument inaccuracy is accounted for in the setpoints.

The values for key input parameters and initial conditions for the evaluation of the LOEL event are listed in Table 15.2-1 for the maximum RCS pressure case, Table 15.2-2 for the maximum steam pressure case, and Table 15.2-3 for the MCHFR case.

15.2.1.3.3 Results

As the RCS heats up, the expansion of water volume increases PZR level and pressure as shown in Figure 15.2-1 and Figure 15.2-7, respectively. Upon the reactor trip, power decreases as shown in Figure 15.2-2. Figure 15.2-3 presents the transient reactivity response. The RCS temperature increases due to the heatup from loss of secondary cooling followed by a decrease due to the reactor trip and then increases due to the reduction in heat removal until DHRS begins to cool the primary system as shown in Figure 15.2-4. The RCS flow (Figure 15.2-5) drops due to the reactor trip and is reestablished as DHRS flow is established (Figure 15.2-6). The SG pressure for the peak SG pressure case is presented in Figure 15.2-8.

An LOEL results in increased temperatures in the RCS, which could potentially challenge fuel parameters. Although RCS fluid and fuel temperatures increase, the core remains covered throughout the event, such that the CHF analysis limit is not challenged. The limiting MCHFR is demonstrated in Figure 15.2-9.

15.2.1.4 Radiological Consequences

The radiological consequences of an LOEL event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.2.1.5 Conclusions

The DSRS acceptance criteria are met for the enveloping analysis, which includes LOEL, TT, and LOCV cases.

- 1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
 - The limiting RCS pressure for this event, shown in Table 15.2-7, is below 110 percent of the design value for the RCS.
 - The limiting SG pressure, shown in Table 15.2-7, is below 110 percent of the design value for the main steam system (MSS) up to the MSIVs.
- 2) Fuel cladding integrity should be maintained by ensuring the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.
 - The MCHFR for this event, shown in Table 15.2-7, is above the CHF analysis limit.
- 3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
 - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met. The LOEL event does not lead to a more serious event.
- 4) The guidance provided in Regulatory Guide (RG) 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response in order to meet the requirements of GDCs 10 and 15.
 - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

- 5) The most limiting plant systems single failure shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.
 - The limiting single failure is a failure of an FWIV to close for the limiting SG pressure case. Results from this scenario do not challenge the identified limits. No single failure results in greater challenges to the RCS pressure and MCHFR acceptance criteria.
- 6) The guidance provided in SECY-77-439 and SECY-94-084 with respect to the consideration of the performance of nonsafety-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.
 - The inputs and assumptions for the operation of nonsafety-related systems and single failures, as discussed in Section 15.2.1.2 and Section 15.2.1.3, ensure the guidance provided is met.

15.2.2 Turbine Trip

15.2.2.1 Identification of Causes and Event Description

A TT event is initiated by the closure of the turbine stop valves (TSVs). For the NPM design, the effect of a TT is similar to LOEL and LOCV. No credit is taken for the turbine bypass system or other control systems and, therefore, numerous secondary side or electrical perturbations can result in a trip of the turbine generator. The TT causes the primary and secondary side temperatures and pressures to increase because energy is not being removed through the SGs to the condenser. The reactor trip signal, DHRS and SSI actuation signals are initiated on high PZR pressure or high steam line pressure. The reactor trip reduces power to decay heat levels. The DHRS actuates and transfers decay heat to the reactor pool. If offsite power is lost at event initiation, with a coincident loss of DC power, the reactor trip, DHRS actuation and SSI occur simultaneously.

A TT event is an AOO as indicated in Table 15.0-1.

15.2.2.2 Sequence of Events and Systems Operation

The severity of a TT is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the TSV. Following the valve closure, SG pressure continues to increase until DHRS establishes natural circulation. The DHRS establishes cooling and begins to depressurize the SGs and the RCS.

If AC and DC power are available, the PCSs and the ESFs perform as designed with allowances for instrument inaccuracy. Some PCSs are disabled in the analysis because their operation is beneficial with respect to the consequences for a TT event. The disabled PCSs are for RCS temperature control, PZR pressure control, PZR level control, steam pressure control, and FW and turbine load control. No operator action is credited to mitigate the effects of a TT event.

The description for the remaining sequence of the TT event is the same as presented for the LOEL event in Section 15.2.1.2, including the loss of power scenarios and also including the failure of one FWIV to close. The enveloping sequence of events for either the LOEL, TT, or LOCV transients are described in Table 15.2-4 for the limiting RCS pressure case, Table 15.2-5 for the limiting secondary pressure case, and Table 15.2-6 for the limiting MCHFR case.

15.2.2.3 Thermal Hydraulic and Subchannel Analyses

15.2.2.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a TT event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.2.2.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator temperature and fuel temperature is taken from BOC. The reactivity coefficients for moderator temperature are least negative at BOC. Thus, they are conservative for undercooling events such as a TT because they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102 percent. The most reactive rod is assumed to remain out of the core, along with a delay of two seconds between the reactor trip signal and the trip initiation. The most limiting combination of reactivity coefficients for moderator temperature and fuel temperature is applied. Instrument inaccuracy is accounted for in the setpoints. The key parameters are listed in Table 15.2-1 for the maximum RCS pressure case, Table 15.2-2 for the maximum SG pressure case, and Table 15.2-3 for the limiting MCHFR case.

15.2.2.3.3 Results

The results for the TT event are the same as those presented in Section 15.2.1.3.3 for the LOEL event. As shown in Table 15.2-7, the TT event does not challenge the limits for RCS pressure, SG pressure, or MCHFR.

15.2.2.4 Radiological Consequences

The radiological consequences of a TT event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.2.2.5 Conclusions

The regulatory acceptance criteria for an AOO are met for the limiting TT event. The results show margin between the system response and the design limits. The MCHFR for the limiting TT event remains above the CHF analysis limit, demonstrating that this event does not result in any fuel damage. The high PZR pressure reactor trip and actuation of the SSI and DHRS terminates this transient by inserting the control rods and removing the decay heat, resulting in a safe stable condition following the event. No operator actions are credited for mitigation of the TT event. A TT does not lead to a more serious event.

The analysis for the TT shows that the NPM design meets the acceptance criteria to maintain RCS and secondary piping pressure below 110 percent of the design value and maintain MCHFR above the CHF analysis limit. Details of the DSRS acceptance criteria in Section 15.2.1 for the LOEL event are also applicable for the TT event. The numerical values for the acceptance criteria are listed in Table 15.2-7.

15.2.3 Loss of Condenser Vacuum**15.2.3.1 Identification of Causes and Event Description**

An LOCV event involves a disturbance that results in an increase in condenser pressure due to air inleakage or a reduction in cooling to the condenser. For the NPM design, the effect of an LOCV results in a trip of the turbine generator and a loss of FW flow, which causes the primary side temperature and pressure to increase because energy is not being removed through the SGs to the condenser. The reactor trips on high PZR pressure or high steam line pressure, reducing power to decay heat levels. The DHRS and SSI actuate to isolate the steam and FW systems and transfer decay heat to the reactor pool.

An LOCV event is an AOO as indicated in Table 15.0-1.

15.2.3.2 Sequence of Events and Systems Operation

The severity of an LOCV event is dictated by the time it takes DHRS to initiate and establish a stable cooldown rate. Secondary pressure is initially driven by the speed of the closure of the TSV. Following the valve closure, SG pressure continues to increase until DHRS establishes natural circulation. The DHRS establishes cooling and begins to depressurize the SGs and the RCS.

If AC and DC power are available, the PCSs and the ESFs perform as designed with allowances for instrument inaccuracy. Some PCSs are disabled in the analysis because their operation is beneficial with respect to the consequences for an LOCV event. The disabled PCSs are for RCS temperature control, PZR

pressure control, PZR level control, steam pressure control, and FW and turbine load control. No operator action is credited to mitigate the effects of an LOCV event.

The description for the remaining sequence of the LOCV event is the same as presented for the LOEL event in Section 15.2.1.2, including the loss of power scenarios and also including the failure of one FWIV to close. The enveloping sequence of events for either the LOEL, TT, or LOCV transients are presented in Table 15.2-4 for the limiting RCS pressure case, Table 15.2-5 for the limiting secondary pressure case, and Table 15.2-6 for the limiting MCHFR case.

15.2.3.3 Thermal Hydraulic and Subchannel Analyses

15.2.3.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to an LOCV event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.2.3.3.2 Input Parameters and Initial Conditions

The reactivity feedback from the moderator temperature and fuel temperature is taken from BOC. The reactivity coefficients for moderator temperature are least negative at BOC. Thus, they are conservative for undercooling events such as an LOCV because they minimize the negative reactivity insertion from the increase in coolant temperature.

Input conditions include an assumed power level of 102 percent. The most reactive rod is assumed to remain out of the core, along with a delay of two seconds between the reactor trip signal and trip initiation. The most limiting combination of reactivity coefficients for moderator temperature and fuel temperature is applied. Instrument inaccuracy is accounted for in the setpoints. The key parameters are listed in Table 15.2-1 for the maximum RCS pressure case, Table 15.2-2 for the maximum SG pressure case, and Table 15.2-3 for the limiting MCHFR case.

15.2.3.3.3 Results

The results for the LOCV event are the same as those presented in Section 15.2.1.3.3 for the LOEL event. As shown in Table 15.2-7, the LOCV event does not challenge the limits on RCS pressure, SG pressure, or MCHFR.

15.2.3.4 Radiological Consequences

The radiological consequences of an LOCV event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.2.3.5 Conclusions

The regulatory acceptance criteria for an AOO are met for the limiting LOCV event. The results show margin between the system response and the design limits. The MCHFR for the limiting LOCV event remains above the CHF analysis limit, demonstrating that this event does not result in any fuel damage. The high PZR pressure reactor trip and actuation of the DHRS and SSI terminates this transient by inserting the control rods and removing the decay heat, resulting in a safe stable condition following the event. No operator actions are credited for mitigation of the LOCV event. An LOCV event does not lead to a more serious event.

The analysis for the LOCV shows that the NPM design meets the acceptance criteria to maintain RCS and secondary piping pressure below 110 percent of the design value and maintain MCHFR above the CHF analysis limit.

Details of the DSRS acceptance criteria in Section 15.2.1 for the LOEL event are also applicable to the LOCV event. The numerical values for the acceptance criteria are listed in Table 15.2-7.

15.2.4 Closure of Main Steam Isolation Valve(s)**15.2.4.1 Identification of Causes and Accident Description**

Closure of one or more MSIVs increases secondary side pressure and temperature in the affected SGs. For the NPM design, the pressure and temperature also increase on the primary side due to the decrease in cooling. Closure of either one or two MSIVs causes primary pressure to increase rapidly to the MPS setpoint for pressurizer pressure, resulting in a reactor trip and actuation of the DHRS and SSI.

Unintended actuation of a single MSIV could be caused by an inadvertent MPS actuation signal to the valve or a loss of DC (EDAS) power to the valve. A single valve could also close on a valid MSIV closure signal with a failure of the opposite train MSIV to close. An inadvertent MPS signal to both trains of valves or a loss of EDAS power to both MSIVs could cause both valves to close. The MSIVs could also be inadvertently closed due to operator error.

An MSIV closure event is an AOO as indicated in Table 15.0-1.

15.2.4.2 Sequence of Events and Systems Operation

The inadvertent MSIV closure event is analyzed for one and both MSIVs in parallel lines closing. The safety-related MSIVs are designed to close within ten seconds. For conservatism, this analysis assumes the MSIVs close

immediately at the event initiation, which maximizes the primary and secondary temperature and pressure transients. Each steam line also has a backup nonsafety-related MSIV that is designed to close within ten seconds.

The peak RCS pressure occurs following closure of two MSIVs with a LOOP at event initiation. The peak SG pressure occurs following closure of two MSIVs with offsite power available. The differences between the peak primary and secondary pressure cases is due to the assumed biases and initial conditions. The lowest MCHFR also occurs with the closure of two MSIVs with AC power available. The DC power sources (EDNS and EDAS) are assumed to be available for all scenarios as the loss of DC power results in earlier reactor trip and actuation of ESF mitigating functions.

The sequence of events for the limiting scenarios is shown in Table 15.2-10 and Table 15.2-11. The single MSIV closure is not limiting for the acceptance criteria.

The MPS is credited to protect the plant in the event of MSIV closure. The MPS actuation trips the reactor and initiates SSI and DHRS, preventing the plant from reaching conditions where the acceptance criteria could be challenged. The high PZR pressure MPS signal is credited in this event. The sensitivity of the transient response to the initial PZR pressure and level bias is evaluated. One of the two redundant RSVs is required to open to mitigate the increase in RCS pressure.

The analysis of the MSIV closure event assumes that plant control systems perform as designed, with allowance for instrument inaccuracy and biases as provided in Table 15.2-8 and Table 15.2-9.

No single failures result in higher RCS pressure. Single failure of an FWIV leads to the limiting SG pressure as the FW system continues filling the SG until the FWRVs close. The limiting SG pressure case also results in the lowest MCHFR. No operator action is credited for the MSIV closure event.

For cases with AC power available, the chemical and volume control system is assumed to maintain normal recirculation flow with no makeup. The PZR spray system is not assumed to function for this event, as it reduces the severity of RCS pressurization.

The inadvertent MSIV closure event is a module-specific event and does not affect other modules. The trip of the turbine is not expected to affect the electrical grid or result in LOOP.

15.2.4.3 Thermal Hydraulic and Subchannel Analyses

15.2.4.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to an MSIV closure event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.2.4.3.2 Input Parameters and Initial Conditions

The initial power level is assumed to be 102 percent of reactor thermal power. A signal delay of two seconds is assumed on the reactor trip, SSI, DHRS actuation (30-second valve opening time), and FWIV closure. MSIV closure is assumed to occur at the beginning of the event with no signal delay. The input parameters for the evaluation model for the MSIV closure events are provided in Table 15.2-8 and Table 15.2-9. The input parameters are developed to provide a conservative result for the events evaluated for RCS pressure, SG pressure, and MCHFR. No credit is taken for PZR spray or decreasing PZR heater power.

15.2.4.3.3 Results

For all of the MSIV closure events pressurizer pressure rises quickly and the MPS initiates a reactor trip, SSI, and DHRS actuation. No single failures cause the results of the limiting RCS pressure event to become more severe. Failure of an FWIV to close results in the limiting SG pressure and lowest MCHFR. The assumption of a LOOP does result in the limiting primary pressure. For the other acceptance criteria, offsite power available is limiting.

Following the closure of both MSIVs, the RCS heats up and RCS pressure increases as shown in Figure 15.2-11. The expansion of water volume increases PZR level as shown in Figure 15.2-15. Upon the reactor trip, power decreases as shown in Figure 15.2-10. Figure 15.2-12 presents the volume average fuel temperature. RCS flow is shown in Figure 15.2-13. The RCS temperature initially increases followed by a decrease due to the reactor trip and then increases due to the reduction in heat removal until DHRS begins to cool the primary system as shown in Figure 15.2-14. The SG pressure and level for the peak SG pressure case are presented in Figure 15.2-16 and Figure 15.2-17.

For each limiting case, the RSV lifts to maintain primary pressure below design limits. The RSV lifts for a short time and containment pressure increases to the containment isolation setpoint.

The single MSIV closure is not limiting for any of the acceptance criteria. The results of the events for the inadvertent closure of MSIVs are within the acceptance criteria.

The subchannel analysis indicates that the MCHFR is above the limit and is acceptable. Figure 15.2-18 shows the MCHFR for the most limiting event, concurrent closure of both MSIVs with a loss of AC power.

15.2.4.4 Radiological Consequences

The radiological consequences of a MSIV closure event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.2.4.5 Conclusions

The DSRS acceptance criteria are met for the MSIV closure event.

- 1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
 - The limiting RCS pressure for this event, listed in Table 15.2-12, is below 110 percent of the design value for the RCS.
 - The limiting SG pressure, listed in Table 15.2-12, is below 110 percent of the design value for the MSS up to the MSIVs.
- 2) Fuel cladding integrity should be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit.
 - The MCHFR for this event, listed in Table 15.2-12, is above the CHF analysis limit.
- 3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
 - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met.
- 4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response in order to meet the requirements of GDCs 10 and 15.
 - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.
- 5) The most limiting plant systems single failure shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.
 - No single failure for an MSIV closure event results in greater challenges to the RCS pressure acceptance criteria. Failure of a FWIV results in the peak SG pressure and limiting MCHFR but margin to the acceptance criteria is shown in Table 15.2-12.

6) The guidance provided in SECY-77-439 and SECY-94-084 with respect to the consideration of the performance of nonsafety-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.

- The inputs and assumptions for the operation of nonsafety-related systems and single failures, as discussed in Section 15.2.4.2 and Section 15.2.4.3, ensure the guidance provided is met.

15.2.5 Steam Pressure Regulator Failure (Closed)

There is no steam pressure regulator in the NuScale Power Plant US460 standard design.

15.2.6 Loss of Non-Emergency Alternating Current Power to the Station Auxiliaries

15.2.6.1 Identification of Causes and Event Description

A loss of AC power to station auxiliaries can occur from the following:

- failures in the electrical grid
- failures in the plant equipment
- failures in switchyard equipment
- external weather events

For the NuScale US460 standard design, the loss of AC power results in the turbine generator tripping and a loss of pumps in the secondary. The primary side temperature and pressure increase due to heat no longer being removed through the SGs. The pressure increase in the primary system initiates a reactor trip, which reduces power to decay heat levels and actuates SSI and DHRS to transfer heat from the primary system to the reactor pool.

A loss of AC power event is classified as an AOO as discussed in Table 15.0-1.

15.2.6.2 Sequence of Events and Systems Operation

Unless specified below, the analysis of a loss of AC power event assumes the plant control systems and MPS perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a loss of AC power event.

The loss of AC power event for the NuScale US460 standard design assesses the loss of the ELVS, which supplies power to plant motors, heaters, packaged equipment and battery chargers. The severity of the loss of AC power event is dictated by the timing of the reactor trip and the loss of power. The transients in this section are evaluated assuming there is a loss of AC power with both the EDAS and EDNS DC power supplies still available. Loss of DC power initiates a

reactor trip and ESF actuations earlier in the transient and results in less limiting conditions.

In the limiting RCS pressure case, power is lost to PZR heaters; condensate, FW, and chemical and volume control system pumps; and the turbine trips immediately. The loss of cooling in the secondary causes a heatup and increase in pressure in the RCS, which initiates a reactor trip and actuates SSI and DHRS on the high PZR pressure MPS signal. Following closure of the MSIVs and FWIVs, the peak RCS pressure occurs coincident with the lifting of an RSV, after which system pressure quickly decreases. Secondary pressure increases until the DHRS valves are fully open and natural circulation is established.

Single failures are considered, however, there is no limiting failure that adversely affects either primary or secondary pressure. No single failure of a FWIV or MSIV to close increases peak RCS pressure as secondary flow is lost at event initiation. No single failure reduces the MCHFR as the limiting value occurs before the SSI signal is generated.

Event sequences for the loss of AC power event are provided in Table 15.2-13 for the limiting RCS pressure case and Table 15.2-14 for the limiting SG pressure and MCHFR case.

15.2.6.3 Thermal Hydraulic and Subchannel Analyses

15.2.6.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a loss of AC power event is performed using NRELAP5. The NRELAP5 model is based on the design features of an NPM. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.2.6.3.2 Input Parameters and Initial Conditions

The initial conditions used in the evaluation of the limiting loss of AC power event result in a conservative calculation. The following initial conditions are assumed in the analysis to ensure that the results have sufficient conservatism:

- Initial power level is assumed to be 102 percent of nominal to account for a two percent measurement uncertainty.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.

- The most limiting BOC core parameters are used to provide a limiting power response.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event. Other key inputs and assumptions used in the thermal hydraulic and subchannel analyses are provided in Table 15.2-15.

15.2.6.3.3 Results

Sensitivity studies on a loss of AC power event are performed. Cases are performed with different loss of power scenarios based on nominal and biased boundary conditions in an attempt to maximize the primary and secondary pressures. For cases that did not result in an immediate reactor trip upon loss of power, the pressurization is found to be higher.

Upon the reactor trip, power decreases as shown in Figure 15.2-19. Figure 15.2-20 presents the event reactivity components. As the RCS heats up, the expansion of water volume increases PZR level (Figure 15.2-21). The RCS pressure increases as shown in Figure 15.2-22. The increase in reactor coolant temperature causes the primary coolant volume to expand, raising RCS pressure. The peak RCS pressure occurs with the lifting of the RSV, which reduces pressure below the RSV reset pressure and ensures RCS pressure is maintained below the acceptance criteria. The RCS temperature decreases due to the reactor trip and then increases due to the reduction in heat removal due to SSI until DHRS begins to cool the primary system as shown in Figure 15.2-24. The RCS flow (Figure 15.2-23) drops initially due to the reactor trip and is reestablished as DHRS flow is established.

The SG pressure increases significantly during the initial phase of the transient following the tripping of the turbine stop valves and SSI, but is limited by the cooling provided by DHRS. The maximum secondary pressure is below the design pressure of the steam system piping. The SG pressure for the peak SG pressure case is presented in Figure 15.2-25.

The predicted lowest MCHFR is greater than the acceptance criteria for this event, no fuel failure is predicted to occur. The limiting MCHFR versus time is shown in Figure 15.2-26.

15.2.6.4 Radiological Consequences

The radiological consequences of a loss of AC power event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.2.6.5 Conclusions

The DSRS acceptance criteria are met for the limiting loss of AC power cases.

- 1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
 - The limiting RCS pressure for this event, listed in Table 15.2-16, is below 110 percent of the design value for the RCS.
 - The limiting SG pressure, listed in Table 15.2-16, is below the 110 percent design value for the MSS up to the MSIV.
- 2) Fuel cladding integrity should be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit.
 - The MCHFR for this event, listed in Table 15.2-16, is above the CHF analysis limit.
- 3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
 - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met.
- 4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response in order to meet the requirements of GDCs 10 and 15.
 - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.
- 5) The most limiting plant systems single failure shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53.
 - No single failures result in increased RCS or SG pressure or decreased MCHFR for the limiting events.

The results of this analysis show that a loss of AC power event does not adversely affect the core, the RCS, or the steam system, and the regulatory acceptance criteria are met.

15.2.7 Loss of Normal Feedwater Flow**15.2.7.1 Identification of Causes and Event Description**

A loss of normal FW flow could occur from the following scenarios:

- pump failures
- valve malfunctions
- loss of AC power

A loss of normal FW is a decrease in heat removal via the SGs resulting in an increase in RCS temperature and pressure, which leads to a reactor trip. A loss of FW can occur due to an FW system pipe break; however, such an event is addressed separately as an accident in Section 15.2.8.

The loss of normal FW flow event is an AOO as discussed in Table 15.0-1.

15.2.7.2 Sequence of Events and Systems Operation

Unless specified below, the analysis of a loss of normal FW flow event assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a loss of normal FW flow event. DC power (EDNS and EDAS) is assumed to be available.

Different event sequences provide limiting results for the applicable acceptance criteria. The first limiting case results in the limiting RCS pressure. Water level in the SG decreases causing primary- to-secondary heat transfer rates to decrease and RCS temperature and pressures to increase. The reactor is tripped and DHRS and SSI are actuated on high PZR pressure. The loss of normal AC power is assumed coincident with the reactor trip. Insertion of control rods reduces reactor power and subsequently, core flow. The conservative delay in DHRS valve actuation results in continued pressure increase in the primary side. The peak RCS pressure coincides with the lifting of the RSV, which decreases primary pressure and maintains it less than 110 percent of the design pressure. The establishment of DHRS flow results in cooling the RCS. The event sequence is provided in Table 15.2-17.

The second limiting case results in the peak secondary pressure. The event is initiated from a fault that results in a partial loss of FW flow. The reduced FW flow causes the SG water level to fall, resulting in a decrease in primary-to-secondary heat transfer, increasing both RCS temperature and pressure. The reactor is tripped and DHRS and SSI are actuated on high PZR pressure. Normal power is assumed available during the transient. RCS pressure reaches a maximum and begins to decrease before DHRS is fully actuated. RSV lifting does not occur. The maximum secondary pressure is reached while stable DHRS flow is being established. The establishment of DHRS flow results in restored cooling of the module. The event sequence is provided in Table 15.2-18.

The case that results in the limiting MCHFR has an event sequence that is similar to the case that results in the peak secondary pressure. The reactor is tripped and SSI actuated on high main steam superheat. DHRS is actuated on high PZR pressure. The MCHFR is reached early in the transient, before the reactor trip. The establishment of DHRS flow results in restored cooling.

The MPS is credited to protect the plant in the event of a loss of normal FW flow. The high PZR pressure signal and high main steam superheat signal are the MPS signals that are credited in providing the plant protection for the limiting cases. One of the two RSVs is credited for mitigating RCS overpressure.

No single failures are found to have adverse impact on the primary or secondary peak pressure or the MCHFR.

15.2.7.3 Thermal Hydraulic and Subchannel Analysis

15.2.7.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to a loss of FW event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general purpose thermal hydraulic analysis under normal operation conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.2.7.3.2 Input Parameters and Initial Conditions

The initial conditions used in the evaluation of the limiting loss of normal FW flow events result in a conservative calculation. These parameters are listed in Table 15.2-19. The following initial conditions are assumed in the analysis to ensure that the results have sufficient conservatism:

- Initial power level is assumed to be 102 percent of nominal to account for a two percent measurement uncertainty.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The most limiting combination of BOC and EOC reactivity feedback is used to provide a limiting power response.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.
- Loss of normal AC power is assumed coincident with reactor trip for the limiting RCS pressure case.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR for this event.

15.2.7.3.3 Results

Sensitivity studies on the loss of FW are performed in an attempt to maximize the primary and secondary pressure response and minimize the CHF ratio. The cases are assessed with normal boundary conditions as well as with biased conditions.

Different cases are identified as providing limiting results: the first case includes a complete loss of FW flow and results in the maximum RCS pressure and MCHFR. A second case assesses a partial loss of FW flow, and results in the maximum SG pressure. A third case with a partial loss of FW flow results in the limiting MCHFR.

Upon the reactor trip, power decreases as shown in Figure 15.2-27. As the RCS heats up, the expansion of water volume increases PZR level (Figure 15.2-32). The RCS pressure increases as shown in Figure 15.2-28. The increase in reactor coolant temperature causes the primary coolant volume to expand, raising RCS pressure. The peak RCS pressure occurs with the lifting of the RSV, which quickly reduces pressure below the RSV reset pressure and ensures RCS pressure is maintained below the acceptance criteria. The RCS temperature decreases due to the reactor trip and then increases due to the reduction in heat removal until DHRS begins to cool the primary system as shown in Figure 15.2-31. The RCS flow (Figure 15.2-30) drops initially due to the reactor trip and stabilizes as DHRS flow is established (Figure 15.2-29). The limiting SG pressure is presented in Figure 15.2-33.

The predicted lowest MCHFR is greater than the acceptance criteria for this event as shown in Figure 15.2-34. No fuel failure is predicted to occur.

15.2.7.4 Radiological Consequences

The radiological consequences of a loss of FW event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.2.7.5 Conclusions

The DSRS acceptance criteria are met for the limiting loss of normal FW flow cases.

- 1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
 - The limiting RCS pressure for this event, listed in Table 15.2-20, is below 110 percent of the design value for the RCS.
 - The limiting MSS pressure, listed in Table 15.2-20, is below 110 percent of the design value for the MSS.
- 2) Fuel cladding integrity should be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit.
 - The MCHFR for this event, listed in Table 15.2-20, is above the CHF analysis limit.
- 3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
 - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met.

- 4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response in order to meet the requirements of GDCs 10 and 15.
 - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.
- 5) The most limiting plant systems single failure shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.
 - No single failures are identified that have adverse impact on the acceptance criteria.
- 6) The guidance provided in SECY-77-439 and SECY-94-084 with respect to the consideration of the performance of nonsafety-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.
 - The inputs and assumptions for the operation of nonsafety-related systems and single failures as discussed in Section 15.2.7.2 and Section 15.2.7.3 ensure the guidance provided is met.

15.2.8 Feedwater System Pipe Breaks Inside and Outside of Containment

15.2.8.1 Identification of Causes and Event Description

A feedwater line break (FWLB) event could range from a small split crack to a double ended rupture of the FW line. This event can occur both inside and outside of the containment vessel (CNV) due to seismic events, thermal stress, or cracking of the FW piping. A spectrum of FWLB locations and break sizes, with varied core and plant conditions, are analyzed to determine the scenarios with the most severe results.

Large FWLBs inside the CNV increase the pressure in the evacuated atmosphere, resulting in a loss of containment vacuum and actuating the high containment pressure MPS signal. The high containment pressure MPS signal actuates the reactor trip system (RTS), isolates containment, and actuates SSI and DHRS. The break depressurizes the impacted SG system and drains the DHRS piping and condenser in the affected loop. The non-impacted SG system and DHRS loop continues to provide cooling to the RCS. Smaller FWLBs inside containment reach the high PZR pressure MPS signal, resulting in reactor trip, SSI, and DHRS actuation, before the high containment pressure MPS signal is reached.

The FW breaks outside of containment cause a loss of FW flow to the SGs and a heatup and subsequent pressure increase in the RCS. Breaks outside containment cause a rapid heatup and trip the reactor and actuate SSI and DHRS on high pressurizer pressure. Reactor trip on high power rate occurs earlier than high PZR pressure for some breaks, but the high power rate MPS signal does not

actuate SSI or DHRS. Similarly, reactor trip and SSI on low main steam pressure occurs earlier than high PZR pressure for some breaks, but the low main steam pressure MPS signal does not actuate DHRS. The DHRS provides cooling and depressurization to the RCS via the intact loop.

A break in the FW line is classified as a postulated accident as shown in Table 15.0-1.

15.2.8.2 Sequence of Events and Systems Operation

Unless specified below, the analysis of a FWLB event assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the consequences of a FWLB event.

For FWLB events where AC power is lost, the turbine and FW pumps trip immediately resulting in a loss of RCS cooling and rapid pressurization.

The transients in this section are evaluated assuming DC power supplies are available.

Breaks inside containment cause a rapid pressurization of the CNV. The CNV response is addressed in Section 6.2.1.

The limiting FWLB for RCS pressure is a double ended guillotine break in an FW line inside containment with a coincident loss of AC power. The combination of turbine trip, FW pump trip and the FWLB result in a loss of cooling to the primary and a heatup and pressurization of the RCS. Reactor trip, DHRS, and SSI actuation occur from the high containment pressure MPS signal followed by the lifting of an RSV that quickly reduces primary pressure below the RSV reset pressure. Secondary pressure increases during the event until stable DHRS cooling is established. Table 15.2-21 provides the sequence of events for the limiting RCS pressure case.

The limiting FWLB for minimum critical heat flux ratio is a double ended guillotine break in an FW line outside of containment coincident with a loss of AC power. The limiting MCHFR sequence of events is provided in Table 15.2-23.

The limiting FWLB for SG pressure occurs from a double ended guillotine break in an FW line inside containment coincident with a loss of AC power at the time of the break. The peak secondary pressurization is a function of the delay in DHRS actuation and establishing heat removal. Table 15.2-22 provides the sequence of events for the limiting secondary pressure case.

The limiting DHRS cooling case involves a double ended guillotine break of an FW line inside containment. This break results in the complete loss of one train of DHRS. No loss of AC power is assumed. Upon break initiation, pressure inside of the CNV rapidly increases, reaching the high CNV pressure analytical limit and actuating reactor trip, containment isolation, DHRS, and SSI. The remaining DHRS loop provides cooling to the module and is sufficient to remove 100 percent

of decay heat and drive flow through the core. This event is not limiting for any of the acceptance criteria. The sequence of events for this case is provided in Table 15.2-24.

The MPS is credited to protect the NPM in the event of a FWLB. The following MPS signals provide the plant with protection during a FWLB:

- high power rate
- low main steam pressure
- high PZR pressure
- high CNV pressure

The actuation of a single RSV is credited for ensuring pressures in the RCS do not exceed the acceptance criteria.

The failure of the FWIV to close on the failed SG results in a limiting value for RCS pressure and MCHFR. A single failure of the FWIV is also assumed in the limiting DHRS case. The nonsafety-related FWRV is credited to close when single failure of the FWIV is assumed. Failure of the FWIV is not limiting for SG pressure.

15.2.8.3 Thermal Hydraulic and Subchannel Analyses

15.2.8.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to an FWLB is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.2.8.3.2 Input Parameters and Initial Conditions

The initial conditions and assumptions used in the evaluation of a FWLB event result in a conservative calculation.

The limiting peak RCS pressure case is a double ended guillotine break in an FW line inside of containment. The following initial conditions are assumed in the analysis of the FWLB to ensure that the transient results in the limiting RCS pressure.

- Initial power level is assumed to be 102 percent of nominal to account for a two percent measurement uncertainty.

- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The limiting BOC reactivity feedback is used to provide a limiting power response.
- A loss of AC power is assumed to occur at the time of the break, causing an immediate turbine and FW pump trip.
- The FWIV is assumed to fail to close on the faulted FW line.
- The SSI valves are assumed to close, and the DHRS valves are assumed to open, at their maximum times.
- System biases include high RCS temperature, high fuel temperature, high PZR pressure, high PZR level, and minimum RCS flow.

The limiting MCHFR event is a double ended guillotine break of an FW line outside of containment. The following initial conditions are assumed in the analysis to ensure that the transient results in the limiting MCHFR.

- Initial power level is assumed to be 102 percent of nominal to account for a two percent measurement uncertainty
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The limiting BOC reactivity feedback is used to provide a limiting power response.
- A loss of AC power is assumed to occur at the time of the break, causing an immediate turbine and FW pump trip.
- The FWIV is assumed to fail close on the faulted FW line.
- SSI valves are assumed to close, and the DHRS valves are assumed to open, at their maximum times.
- System biases include high RCS temperature, high fuel temperature, low PZR pressure, low PZR level, and minimum RCS flow.

The limiting event for SG pressure is the double ended guillotine break in an FW line inside of containment. The following initial conditions are assumed in the analysis to ensure that the transient results in the limiting secondary pressure.

- Initial power level is assumed to be 102 percent of nominal to account for a two percent measurement uncertainty.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The limiting BOC reactivity feedback is used to provide a limiting power response.

- A loss of AC power is assumed to occur at the time of the break, causing an immediate turbine and FW pump trip.
- The SSI valves are assumed to close at their minimum time while the DHRS valves are assumed to open at their maximum time.
- No single failure is assumed.
- System biases include high RCS temperature, high fuel temperature, high PZR pressure, high PZR level, and minimum RCS flow.
- The end of life SG is assumed, which includes the design limit tube plugging and fouling in addition to a negative 30 percent uncertainty for the primary to secondary heat transfer.

The limiting DHRS cooling case is a double ended guillotine break in an FW line inside containment. The following initial conditions are assumed in the analysis to ensure that the transient results are conservative in the assessment of DHRS functionality.

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and utilizing a bounding control rod drop rate.
- The limiting BOC reactivity feedback is used to provide a limiting power response.
- The SSI valves are assumed to close at their minimum time while the DHRS valves are assumed to open at their maximum time.
- The FWIV is assumed to fail to close on the faulted FW line.
- System biases include high RCS temperature, high fuel temperature, high PZR pressure, high PZR level, and minimum RCS flow.
- Normal AC power is assumed to be available with the TT occurring at the time of reactor trip.
- The end of life steam SG is assumed, which includes the design limit tube plugging and fouling in addition to a negative 30-percent uncertainty for the primary-to-secondary heat transfer.

The results from the thermal hydraulic evaluations are used as input to the subchannel analysis to determine the limiting MCHFR for this event. Other key inputs and assumptions used in the analysis are included in Table 15.2-25.

15.2.8.3.3 Results

For the limiting RCS pressure case, the FW line is assumed to have a double ended guillotine break in an FW line inside containment. The AC power is assumed to be lost, causing the turbine and FW pumps to immediately trip, resulting in a rapid loss of cooling and large RCS pressurization. In this scenario, the reactor is tripped by the high containment pressure MPS signal

also leading to SSI and DHRS actuation. Pressure in the RCS continues to increase following reactor trip until the RSV lift setpoint is reached. Once the RSV lifts, system pressure returns to below the RSV reset pressure. The limiting peak RCS pressure from FWLB is provided in Figure 15.2-35. The DHRS cooling is sufficient to begin to depressurize the system such that a second RSV lift does not occur. The reactor power drops sharply following the reactor trip and follows the typical decay heat curve as shown in Figure 15.2-36. The SG pressure rises in the unaffected (intact FW loop) SG as shown in Figure 15.2-37. The affected (FW pipe break loop) SG pressure response is also shown in Figure 15.2-37. The RCS temperature and RCS flow stabilize as shown in Figure 15.2-38 and Figure 15.2-39, respectively. The SSI and DHRS actuation closes the FWIVs and stops the loss of secondary fluid. The integrated break flow for the event is shown in Figure 15.2-40.

The peak SG pressure case results from a double ended guillotine break in an FW line inside containment. The AC power is assumed to be lost at the time of the break. This condition results in pressure in the SGs as shown in Figure 15.2-41. The intact SG has the peak SG pressure.

In the limiting MCHFR case, the FWLB is a double ended guillotine break in an FW line outside containment. The AC power is assumed to be lost at the time of the break. The MCHFR versus time is presented in Figure 15.2-42.

For the limiting DHRS cooling case, the FWLB is a doubled ended guillotine break of an FW line inside containment. The break flow flashes in the evacuated CNV atmosphere resulting in a loss of containment vacuum and a high containment pressure signal. The high CNV pressure signal causes an actuation of the reactor trip system, SSI, DHRS, and a containment isolation signal. The break causes a loss of FW pressure and decreased flow to both SGs causing a heatup of the RCS. The break results in the loss of an entire DHRS loop as the fluid from the FW line and SG empty into containment. The second loop of DHRS remains intact throughout the event providing cooling to the RCS. The DHRS heat removal is shown in Figure 15.2-43. The RCS flow rate decreases and then stabilizes as shown in Figure 15.2-45. The intact and depressurized (faulted) SG pressures are shown in Figure 15.2-44.

15.2.8.4 Radiological Consequences

The radiological consequences of the FWLB event are bounded by the consequences of a steam line break discussed in Section 15.0.3.

15.2.8.5 Conclusions

The DSRS acceptance criteria are met for the limiting cases.

- 1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures for low probability events and below 120 percent for very low probability events.
 - The pressure responses in the RPV and in the MSS for this accident are shown in Table 15.2-26 to be under the more conservative AOO acceptance criterion of 110 percent of design values. Therefore, the acceptance criteria for pressure are met for this event.
- 2) The potential for core damage is evaluated for an acceptable minimum DNBR remaining above the 95/95 DNBR limit.
 - The MCHFR for this event is above the CHF analysis limit as presented in Table 15.2-26.
- 3) Calculated doses at the site boundary from any activity release must be a small fraction of the 10 CFR Part 100 (10 CFR 50.34(a)(1)) guidelines.
 - The radiological analysis of the SLB accident bounds the consequences for the FWLB and is presented in Section 15.0.3 and demonstrates that the acceptance criteria are met.
- 4) The DHRS must be safety grade and automatically initiated when required.
 - The results of this analysis demonstrate that, even with a failed DHRS train (due to FWLB inside containment), the second DHRS actuates and provides heat removal during a FWLB, ensuring that acceptance criteria are met.
- 5) Certain assumptions should be in the analysis of important parameters that describe initial plant conditions and postulated system failures.
 - The assumptions included in this analysis for biasing and sensitivities are developed to consider sufficient conservatism to ensure the acceptance criteria are met.

15.2.9 Inadvertent Operation of the Decay Heat Removal System**15.2.9.1 Identification of Causes and Event Description**

This section addresses actuation of DHRS at higher power, which is a heatup event due to the decrease in cooling. The limiting cases for this event are at full power. The events evaluated include the following scenarios:

- inadvertent opening of a single DHRS actuation valve
- inadvertent isolation of one SG and actuation of one DHRS train
- inadvertent isolation of both SGs and actuation of both DHRS trains

- inadvertent isolation of one SG
- inadvertent isolation of both SGs

At low power and reduced FW flow rates, the inadvertent operation of decay heat removal system (IODHRS) is a cooldown event. The pressure drop along the secondary decreases such that it no longer exceeds the hydrostatic pressure of the liquid inventory in the DHRS piping. If the DHRS actuation valve opens under these conditions, a portion of the DHRS liquid inventory drains into the FW line and momentarily increases SG flow. This unique variant of the IODHRS event leads to an increase in heat removal from the RCS. This IODHRS event is bounded by more limiting overcooling events addressed in Section 15.1, such as an increase in FW flow.

The DHRS has several scenarios that can initiate part or all of the system functions. The DHRS actuation valves are normally energized and powered by independent DC power sources from the EDAS system. Loss of power or an inadvertent control signal to one actuation valve on either DHRS system opens the flow path to the associated DHRS heat exchanger, providing a short circuit flow path for FW through the DHRS piping instead of through the SG. A spurious actuation could occur on one or both trains of DHRS.

A full actuation of DHRS opens the two actuation valves on each of the two trains and closes the FWIVs and the MSIVs. An inadvertent signal to isolate one or both SGs by closure of the FWIV and MSIV on the affected train(s) is also evaluated.

The inadvertent operation of DHRS is classified as an AOO as discussed in Table 15.0-1.

15.2.9.2 Sequence of Events and Systems Operation

Unless specified below, the analysis of the inadvertent operation of the DHRS event assumes the plant control systems and engineered safety features perform as designed, with the allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of the inadvertent operation of the DHRS event.

The DHRS actuates on a loss of AC (ELVS) or DC (EDAS) power. A loss of ELVS or EDAS also causes a reactor trip and containment isolation. Therefore, loss of power scenarios are not addressed further in this section.

The inadvertent opening of a single DHRS actuation valve provides a short circuit fluid pathway between the FW system and MSS that bypasses the SG. When operating at full power, this causes a portion of the subcooled FW flow to flow through the DHRS piping instead of through the SG. This redirection of flow leads to a reduction in heat removal from the RCS but is normally compensated for with the turbine load controller, which increases the total FW flow rate to return to the demanded turbine load. Under these nominal conditions, this is a relatively minor transient that does not result in a MPS actuation of safety related systems.

If the turbine load controller is unable to compensate for the reverse DHRS flow, a TT occurs due to the reduced temperature steam coming from the MSS. This event progression (not crediting turbine bypass) results in an MPS high steam superheat or high steam pressure reactor trip signal that also initiates a full SSI actuation, which closes the FWIVs and MSIVs.

Limiting RCS Pressure Case

The maximum RCS pressure case occurs with inadvertent isolation of both SGs. In this case all the FWIVs, FWRVs, MSIVs and backup MSIVs close, but the DHRS actuation valves do not open. RCS pressure continues to increase until the high PZR pressure MPS reactor trip is initiated. Control rods are inserted and then the RSV lifts relieving the RCS pressure. The RSV is sized to relieve the maximum pressure and inventory ramp of the RCS that occurs in this event. Subsequently, the DHRS valves open and RCS cooling begins.

Limiting SG Pressure Case

The inadvertent isolation of a single SG results in the reduction in heat removal leading to a continuous heatup and pressurization of the RCS until the high PZR pressure MPS reactor trip and SSI signals are generated. This gradual RCS heatup is limiting for addressing peak SG pressure. Therefore, this scenario results in the most limiting case for peak SG pressure. After the DHRS valves are fully opened, SG pressure begins to decline.

Limiting MCHFR Case

An inadvertent isolation of both SGs rapidly leads to an MPS reactor trip signal for high PZR pressure. Because the loss of heat transfer to both trains of the SG occurs simultaneously, this scenario provides the most rapid reduction in RCS heat removal and is the limiting MCHFR scenario.

The timing of the sequence of events for the limiting cases is provided in Table 15.2-27 through Table 15.2-29.

15.2.9.3 Thermal Hydraulic and Subchannel Analyses

15.2.9.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to an inadvertent operation of the DHRS event is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general purpose thermal hydraulic analysis under normal operation conditions, operational transients, and events

of moderate severity. Section 15.0.2 contains a discussion of the VIPRE-01 code and evaluation model.

15.2.9.3.2 Input Parameters and Initial Conditions

Limiting MCHFR Case

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent measurement uncertainty.
- Limiting BOC reactivity coefficients are applied, which minimizes the decrease in reactor power following the increase in RCS temperature.
- Regulating rod bank reactivity control is disabled to prevent the insertion of negative reactivity and decrease in reactor power following the increase in RCS temperature.
- The initial SG pressure is biased low.
- Minimum RCS design flow is assumed, which minimizes MCHFR.
- Initial fuel temperature and decay heat are biased high.
- The RCS coolant temperature is initialized to the high condition to result in the highest temperature during the transient calculation.
- High steam superheat signal is disabled to prevent an MPS actuation before the maximum RCS temperature is reached.
- The PZR pressure and PZR level are biased low for the event.
- No loss of AC power or failure of a FWIV is considered as this reduces the consequences of the event.
- Initial reactor pool temperature is assumed to be 120 degrees F.
- The PZR spray is disabled and PZR heater output is constant.

Limiting SG Pressure Case

- Core power is initialized at 102 percent of rated power.
- Limiting BOC reactivity coefficients are applied, which minimizes the decrease in reactor power following the increase in RCS temperature.
- Initial RCS design flow is biased low.
- Initial PZR pressure and PZR level are biased high.
- The RCS coolant temperature is initialized to the high condition to facilitate reaching the maximum analytical temperature during the transient calculation.
- Loss of AC power is not assumed to occur.
- Single failure of a FWIV to close is not considered. The impact of an FWIV failure on SG pressure is bounded by the SG response to the event in Section 15.2.4, which does consider single failure of an FWIV.
- Initial fuel temperature and decay heat are biased high.

- The PZR spray is disabled and PZR heater output is constant.
- The reactor pool temperature is assumed to be 120 degrees F.

Limiting RCS Pressure Case

- Core power is initialized at 102 percent of rated power.
- Limiting BOC reactivity coefficients are applied, which minimizes the decrease in reactor power following the increase in RCS temperature.
- Initial RCS pressure is biased to the high condition to maximize the peak RCS pressure during the transient.
- The RCS temperature is biased high and RCS flow is biased low.
- Initial PZR level is biased to the high condition to maximize the peak RCS pressure following the in-surge of coolant into the PZR during the transient.
- Pressure control is disabled during the transient. This includes disabling the PZR heater response (i.e., set heater power to a constant value) and disabling the PZR spray.
- No loss of AC power or failure of a FWIV is considered as this reduces the consequences of the event.
- Initial fuel temperature and decay heat are biased high.
- Initial SG pressure is biased low.
- Initial reactor pool temperature is assumed to be 120 degrees F.

15.2.9.3.3 Results

The limiting RCS pressure event occurs in the inadvertent isolation of both SGs at full power. In this event RCS pressure escalates and leads to a high PZR pressure trip of the reactor. Subsequently the RSV lifts to reduce RCS pressure. The maximum RCS pressure reached in this analysis occurs after the RSV lift setting is reached but is below the RCS maximum design pressure as shown in Figure 15.2-46. Core power is shown in Figure 15.2-47. The RCS heatup is shown in the average RCS temperature provided in Figure 15.2-48. RCS flow is shown in Figure 15.2-49. The total reactivity is negative throughout the event as shown in Figure 15.2-50. Secondary pressures are shown in Figure 15.2-51. The PZR level is shown in Figure 15.2-52. This event is terminated after the MPS reactor trip, actuation of DHRS, and establishing of stable DHRS cooling. The DHRS flow is shown in Figure 15.2-53.

The limiting case identified for MCHFR occurs for the inadvertent isolation of both SGs. As shown in Figure 15.2-54, the limit for MCHFR is not challenged.

The inadvertent isolation of a single SG results in the highest SG pressure because it results in a gradual RCS heatup, which is limiting for SG pressure.

This case results in higher SG pressures than those shown in Figure 15.2-51, but SG pressure is maintained below the SG design pressure.

Based on the above evaluation, these inadvertent operation of DHRS events do not result in pressure or temperature transients that exceed the criteria for which the reactor pressure vessel, SG, containment vessel, or fuel are designed. Therefore, these barriers to the transport of radionuclides to the environment function as designed.

15.2.9.4 Radiological Consequences

The radiological consequences of an inadvertent operation of the DHRS event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.2.9.5 Conclusion

The DSRS acceptance criteria are met for the inadvertent operation of the DHRS.

- 1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
 - The limiting RCS pressure for this event, listed in Table 15.2-31, is below 110 percent of the design value for the RCS.
 - The limiting MSS pressure, listed in Table 15.2-31, is below 110 percent of the design value for the MSS.
- 2) Fuel cladding integrity should be maintained by ensuring the minimum DNBR remains above the 95/95 DNBR limit.
 - The MCHFR for this event, listed in Table 15.2-31, is above the CHF analysis limit.
- 3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
 - The analyses presented for this event shows that stable DHRS cooling is reached, and the acceptance criteria are met.
- 4) The guidance provided in RG 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response in order to meet the requirements of GDCs 10 and 15.
 - Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.
- 5) The most limiting plant systems single failure shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.
 - Postulated single failures for this event would provide results that are less limiting than those identified in the limiting case. As such, there are no

limiting single failures identified for this event that would adversely affect the results.

This evaluation demonstrates that RCS pressure, SG pressure, and MCHFR limits are not challenged. The regulatory acceptance criteria are met for the inadvertent operation of DHRS event cases evaluated. A comparison of the limiting pressure values and the associated acceptance criteria is shown in Table 15.2-31. The system stabilizes to a safe condition.

Table 15.2-1: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions Peak Reactor Coolant System Pressure Case

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+20°F	120°F
FW Temperature	nominal	250°F
Decay Heat	high	various
RCS Flow Rate	minimum	647 kg/sec
PZR Level	+8%	68%
RSV Lift Setpoint	+3%	2266 psia
RCS Average Coolant Temperature	nominal	540°F
SG Pressure	nominal	475 psia
PZR Pressure	-70 psia	1930 psia
Turbine Stop Valve Stroke Time	n/a	0.001 s
Time of Loss of AC Power	n/a	event initiation

Table 15.2-2: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions Peak Steam Generator Pressure Case

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+20°F	120°F
FW Temperature	nominal	250°F
Decay Heat	high	various
RCS Flow Rate	minimum	647 kg/sec
PZR Level	+8%	68%
RSV Lift Setpoint	+3%	2266 psia
RCS Average Coolant Temperature	nominal	540°F
SG Pressure	nominal	475 psia
Turbine Stop Valve Stroke Time	n/a	0.001 s
PZR Pressure	nominal	2000 psia

Table 15.2-3: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Initial Conditions Minimum Critical Heat Flux Ratio Case

Parameter	Bias	Analysis Value
Initial Reactor Power	+2%	102%
Reactor Pool Temperature	+20°F	120°F
FW Temperature	nominal	250°F
Decay Heat	high	various
RCS Flow Rate	minimum	647 kg/sec
PZR Level	+8%	68%
RSV Lift Setpoint	+3%	2266 psia
RCS Average Coolant Temperature	+5°F	545°F
SG Pressure	nominal	475 psia
PZR Pressure	nominal	2000 psia
Turbine Stop Valve Stroke Time	n/a	0.001 s

Table 15.2-4: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Reactor Coolant System Maximum Pressure Case Sequence of Events

Event	Time (sec)
Event initiator - TT and loss of FW flow due to assumed LOOP	0
Turbine stop valves fully closed (assumption)	0
FW flow is secured (assumption)	0
Reactor trip, SSI and DHRS actuation, PZR heater trip (high PZR pressure)	9
RSV lifts	10
Peak RCS pressure	11
DHRS actuation valves full open	39

Table 15.2-5: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Steam Generator Maximum Pressure Case Sequence of Events

Event	Time (sec)
Event initiator - TT	0
Turbine stop valves fully closed (assumption)	0
Reactor trip, SSI and DHRS actuation, PZR heater trip (high PZR pressure)	9
Assumed loss of normal AC power	9
Secondary system isolation complete	19
DHRS actuation valves full open	39
Time of peak secondary pressure	93

Table 15.2-6: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Minimum Critical Heat Flux Ratio Sequence of Events

Event	Time (sec)
TT and loss of FW flow due to assumed LOOP	0
Turbine stop valves fully closed (assumption)	0
FW flow is secured (assumption)	0
MCHFR reached	<1
Reactor trip, SSI and DHRS actuation, PZR heater trip (high PZR pressure)	8
RSV lifts	8
DHRS actuation valves full open	37

Table 15.2-7: Loss of External Load -Turbine Trip - Loss of Condenser Vacuum - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2420 psia	2310 psia
Maximum SG Pressure	2420 psia	1426 psia
MCHFR	1.33	2.09

Table 15.2-8: Main Steam Isolation Valve Closure - Initial Conditions for Reactor Coolant System Pressure Limiting Case

Parameter	Analysis Value
Initial Reactor Power (includes 2% uncertainty)	102%
Initial RCS T _{ave} (high bias)	545°F
Initial PZR Pressure (high bias)	2070 psia
Initial PZR Level (high bias)	68%
FW Temperature (nominal)	250°F
SG Pressure (nominal)	475 psia
Initial RCS Flow Rate (low bias)	647 kg/s
Safety Relief Valve Open Setpoint	2266 psia
Pool Temperature (high bias)	120°F

Table 15.2-9: Main Steam Isolation Valve Closure - Initial Conditions for Steam Generator Pressure and Minimum Critical Heat Flux Limiting Case

Parameter	Analysis Value
Initial Reactor Power (includes 2% uncertainty)	102%
Initial RCS Tave (high bias)	545°F
Initial PZR Pressure (low bias)	1930 psia
Initial PZR Level (low bias)	52%
FW Temperature (nominal)	250°F
SG Pressure (high bias)	510 psia
Initial RCS Flow Rate (low bias)	647 kg/s
Safety Relief Valve Open Setpoint	2266 psia
Pool Temperature (high bias)	120°F

Table 15.2-10: Main Steam Isolation Valve Closure - Sequence of Events for Reactor Coolant System Pressure Limiting Case

Event	Time (s)
Event initiator - MSIV closure - concurrent loss of AC power assumed	0
Reactor trip, SSI and DHRS actuation, PZR heater trip (high PZR pressure)	5
RSV lifts	6
Peak RPV pressure is reached	7
CNV isolation	19
DHRS actuation valves are fully open	35

Table 15.2-11: Main Steam Isolation Valve Closure - Sequence of Events for Steam Generator Pressure and Minimum Critical Heat Flux Limiting Case

Event	Time (s)
Event initiator - MSIV closure	0
MCHFR occurs	<1
Reactor trip, SSI and DHRS actuation, PZR heater trip (high PZR pressure)	11
RSV lifts	13
CNV isolation	27
DHRS actuation valves are fully open	40
Peak secondary pressure	41

Table 15.2-12: Main Steam Isolation Valve Closure - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2313
Maximum SG pressure	2420 psia	1597
MCHFR	1.33	2.03

Table 15.2-13: Loss of Non-Emergency Alternating Current Power- Reactor Coolant System Pressure Limiting Case Sequence of Events

Event	Time [s]
Loss of AC power	0
Control rod insertion begins (reactor trip on high PZR pressure)	9
Secondary system isolation occurs	9
DHRS actuation valves begin to open	9
RSV lifts	10
Maximum RCS pressure	11
CNV isolation	60

Table 15.2-14: Loss of Non-Emergency Alternating Current Power - Steam Generator Pressure and Minimum Critical Heat Flux Ratio Limiting Case - Sequence of Events

Event	Time [s]
Loss of AC power	0
MCHFR occurs	<1
Control rod insertion begins (reactor trip on high PZR pressure)	11
Secondary system isolation occurs	11
DHRS actuation valves begin to open	11
RSV lifts	12
CNV isolation	60
Maximum SG pressure	91

Table 15.2-15: Input Parameters Loss of Non-Emergency Alternating Current Power - Limiting Cases

Parameter	Limiting RCS Pressure	Limiting SG Pressure and MCHFR
Initial PZR pressure	1930 psia	1930 psia
Initial RCS temperature	545°F	545°F
Initial PZR level	68%	68%
Initial FW temperature	250°F	250°F
Initial SG pressure	510 psia	510 psia
RSV setpoint	2266 psia	2266 psia
RCS flow rate	647 kg/s	647 kg/s
Pool temperature	100°F	100°F
SG tube heat transfer	Nominal	-30%

Table 15.2-16: Loss of Non-Emergency Alternating Current Power - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2314 psia
Maximum SG pressure	2420 psia	1459 psia
MCHFR	1.33	1.93

Table 15.2-17: Loss of Feedwater Event - Maximum Reactor Coolant System Pressure - Sequence of Events

Event	Time [s]
Loss of FW initiation	0
Control rod insertion begins (reactor trip on high PZR pressure)	13
Loss of AC power	13
Secondary system isolation valves fully closed	14
RSV lift setpoint reached	14
Maximum RCS pressure	15
RSV reseal setpoint reached	23
DHRS actuation valves fully open	43

Table 15.2-18: Loss of Feedwater Event - Maximum Steam Generator Pressure - Sequence of Events

Event	Time [s]
Loss of FW initiation	0
Control rod insertion begins (reactor trip on high PZR pressure)	126
Secondary system isolation valves fully closed	136
DHRS actuation valves fully open	156
Maximum SG pressure	210

Table 15.2-19: Input Parameters Loss of Feedwater - Limiting Cases

Parameter	RCS Pressure	SG Pressure	MCHFR
Initial RCS pressure	1930 psia	1930 psia	1930 psia
Initial RCS temperature	545°F	545°F	545°F
Initial PZR level	68%	52%	52%
Initial FW temperature	270°F	230°F	230°F
Initial SG pressure	440 psia	510 psia	440 psia
RSV setpoint	2266 psia (+3%)	2266 psia (+3%)	2266 psia (+3%)
RCS flow rate ⁽¹⁾	1784 lbm/s	1416 lbm/s	1394 lbm/s
Pool temperature	65°F	120°F	120°F
Loss of FW flow at initialization	100%	12%	12%

¹ Table 15.0-6 contains the range of RCS flow rates at 100 percent power.

Table 15.2-20: Loss of Feedwater - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2314 psia
Maximum SG pressure	2420 psia	1548 psia
MCHFR	1.33	2.03

Table 15.2-21: Feedwater Line Break Sequence of Events - Peak Reactor Coolant System Pressure Case

Event	Time [s]
Failure that initiates event	0
Loss of AC power	0
Maximum SG level	1
Maximum power	3
Control rod insertion begins (reactor trip on high containment pressure)	4
Secondary system isolation valves fully closed	14
RSV lifts	22
Maximum RCS pressure	22
DHRS actuation valves fully open	34

Table 15.2-22: Feedwater Line Break Sequence of Events - Peak Steam Generator Pressure Case

Event	Time [s]
Failure that initiates event	0
Loss of AC power	0
Maximum power	3
Control rod insertion begins (reactor trip on high containment pressure)	4
Secondary system isolation valves fully closed	5
DHRS actuation valves fully open	34
Maximum SG pressure	90
Maximum SG level	3689

Table 15.2-23: Feedwater Line Break Sequence of Events - Limiting Minimum Critical Heat Flux Ratio Case

Event	Time [s]
Failure that initiates event	0
Loss of AC power	0
MCHFR occurs	<1
Maximum power	3
Control rod insertion begins (reactor trip on high PZR pressure)	11
Secondary system isolation valves fully closed	21
DHRS actuation valves fully open	41

Table 15.2-24: Feedwater Line Break Sequence of Events - Limiting Decay Heat Removal System Case

Event	Time [s]
Failure that initiates event	0
Maximum power	3
Control rod insertion begins (reactor trip on high containment pressure)	4
Secondary system isolation valves fully closed	4
DHRS actuation valves fully open	34
Maximum SG pressure	88
Maximum RCS pressure	380
Maximum SG level	3439

Table 15.2-25: Biases and Uncertainties - Feedwater Line Break

Region	Parameter	Value
Core	Maximum power	102%
	Reactivity feedback	BOC
	Decay heat	High
	Core flow rate	Low
RCS	Max core average temperature	545°F
	PZR pressure ⁽¹⁾	2070 psia
	PZR level ⁽¹⁾	68%
	SG heat transfer ⁽²⁾	Nominal
	FW temperature	230°F
Reactor Pool	Temperature	200°F
DHRS	DHRS heat transfer	-30%
CNV	CNV pressure	1.0 psia

(1) For the limiting MCHFR case, PZR pressure and level are biased low to 1930 psia and 52 percent, respectively.

(2) For the limiting SG pressure and DHRS cases, SG heat transfer is biased low (-30 percent).

(3) For the limiting MCHFR case, FW temperature is biased high to 270°F.

Table 15.2-26: Feedwater Line Break - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2320 psia
Maximum SG pressure	2420 psia	1424 psia
MCHFR	1.33	2.01

Table 15.2-27: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting Minimum Critical Heat Flux Ratio Case

Event	Time (s)
Transient initiation (inadvertent isolation of both SGs)	0
MCHFR occurs	<1
High PZR pressure analytical limit reached	7
RTS actuation - control rod insertion begins	9
RSV lift setpoint reached	10
RSV reseal setpoint reached	23
DHRS actuation valves fully open	39

Table 15.2-28: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting Peak Steam Generator Pressure Case

Event	Time (s)
Transient initiation (inadvertent isolation of one SG)	0
High pressurizer pressure analytical limit reached	6
RTS actuation - control rod insertion begins	8
Secondary system isolation valves closed	9
DHRS actuation valves fully open	38
Maximum SG pressure reached	97

Table 15.2-29: Sequence of Events for Inadvertent Operation of Decay Heat Removal System - Limiting Peak Reactor Coolant System Pressure Case

Event	Time (s)
Transient initiation (inadvertent isolation of both SGs)	0
Maximum power	2
High PZR pressure analytical limit reached	3
RTS actuation - control rod insertion begins	5
RSV lift point is reached	5
Maximum RCS pressure reached	7
DHRS actuation valves fully open	35

Table 15.2-30: Inadvertent Operation of Decay Heat Removal System Inputs, Uncertainties, and Biases

Parameter	Minimum MCHFR Case	Maximum SG Pressure Case	Maximum RCS Pressure Case
Reactor power	102%	102%	102%
Initial RCS average temperature (bias high)	545°F	545°F	545°F
RCS flow (bias low)	646 kg/s	644 kg/s	644 kg/s
SG pressure	440 psia	440 psia	440 psia
PZR pressure	1930 psia	2070 psia	2070 psia
PZR level	52%	68%	68%

Table 15.2-31: Inadvertent Operation of Decay Heat Removal System - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Maximum RCS pressure	2420 psia	2317 psia
Maximum SG pressure	2420 psia	1490 psia
MCHFR	1.33	2.03

Figure 15.2-1: Pressurizer Level - Peak Reactor Coolant System Pressure Case (15.2.1-15.2.3 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum)

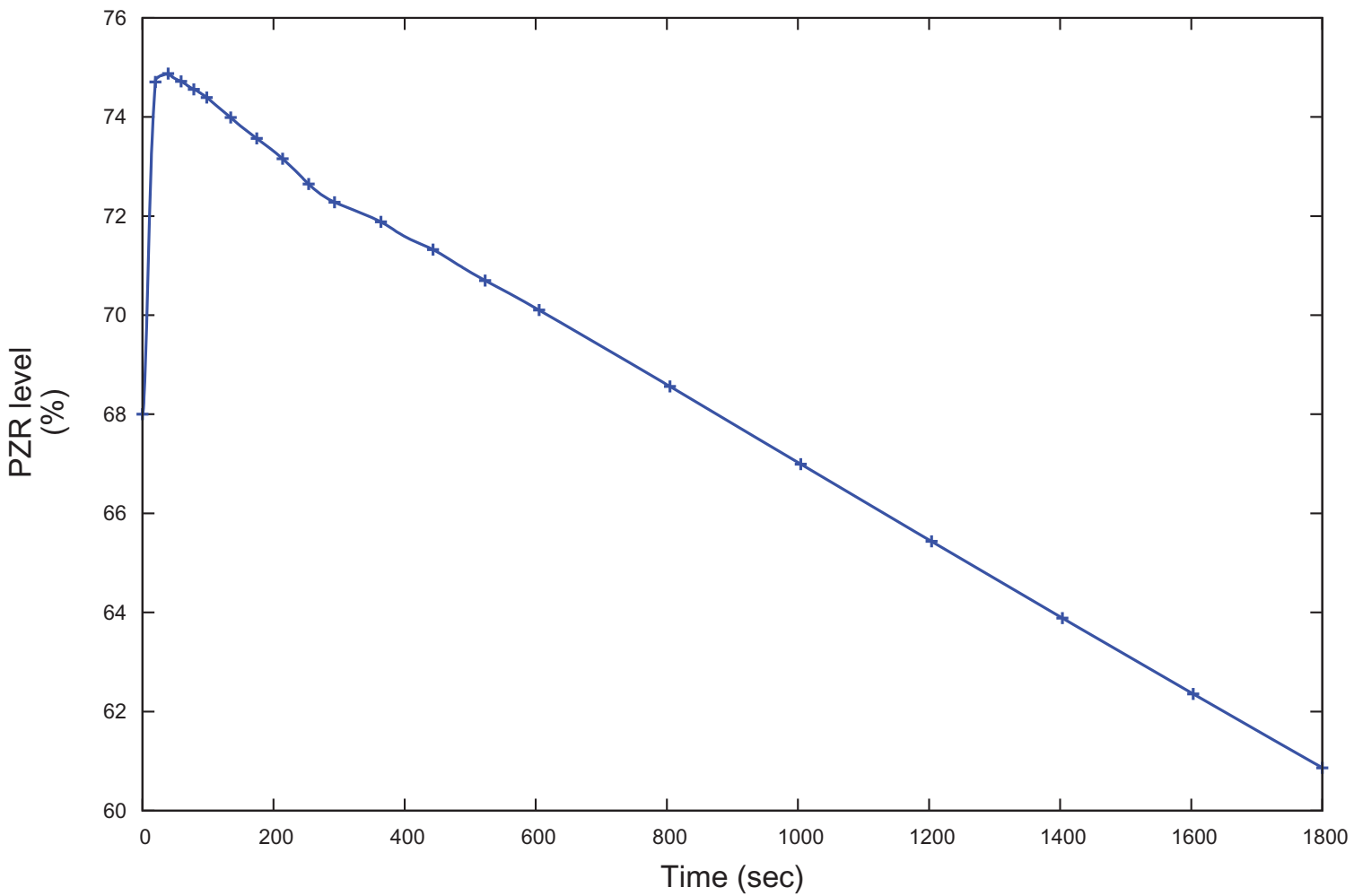


Figure 15.2-2: Reactor Power - Peak Reactor Coolant System Pressure Case (15.2.1-15.2.3 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum)

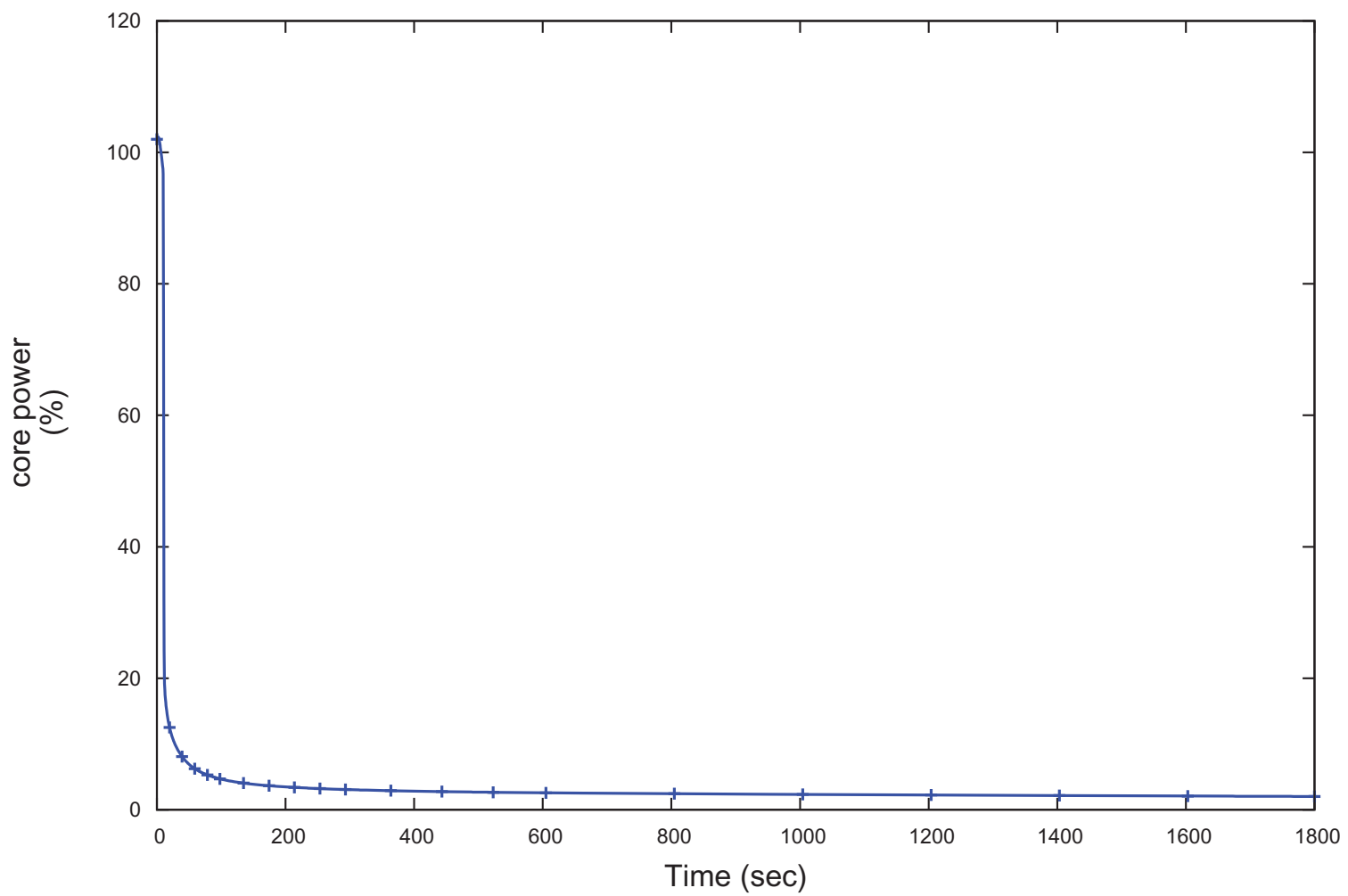


Figure 15.2-3: Event Reactivity - Peak Reactor Coolant System Pressure Case (15.2.1-15.2.3 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum)

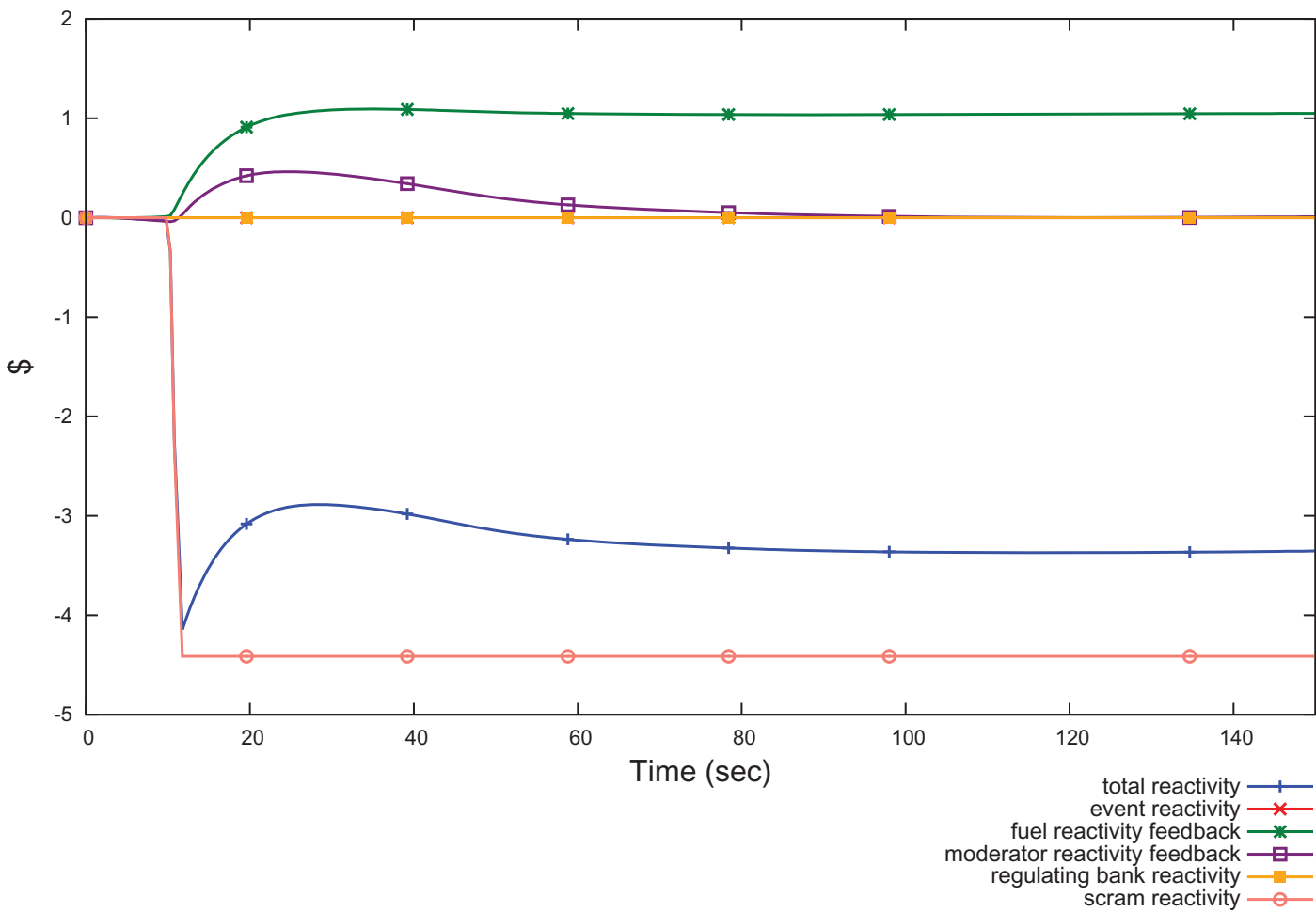


Figure 15.2-4: Reactor Coolant System Temperatures - Peak Reactor Coolant System Pressure Case (15.2.1-15.2.3 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum)

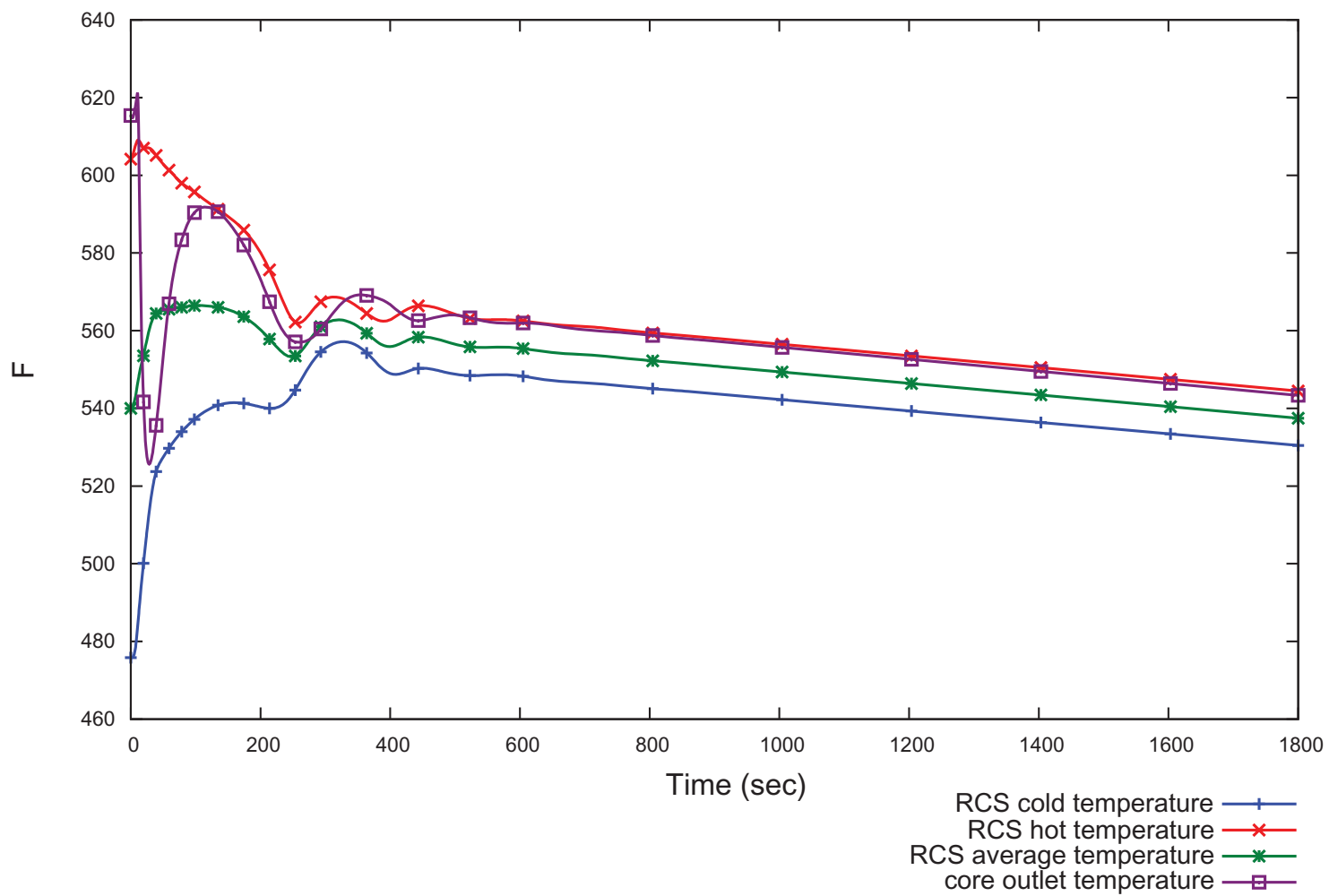


**Figure 15.2-5: Reactor Coolant System Flow Rate - Peak Reactor Coolant System Pressure Case
(15.2.1-15.2.3 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum)**

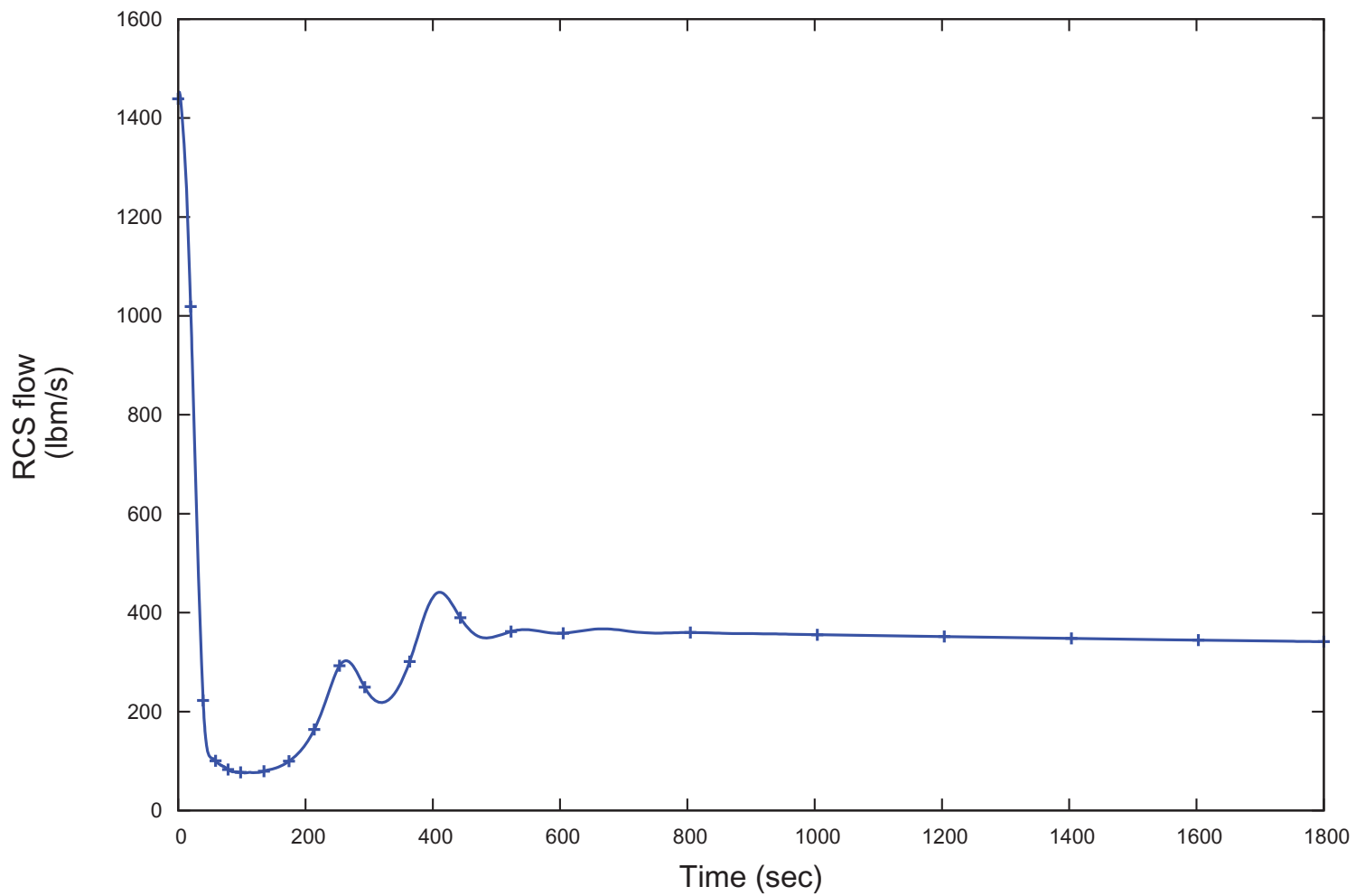


**Figure 15.2-6: Decay Heat Removal System Flow - Peak Reactor Coolant System Pressure Case
(15.2.1-15.2.3 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum)**

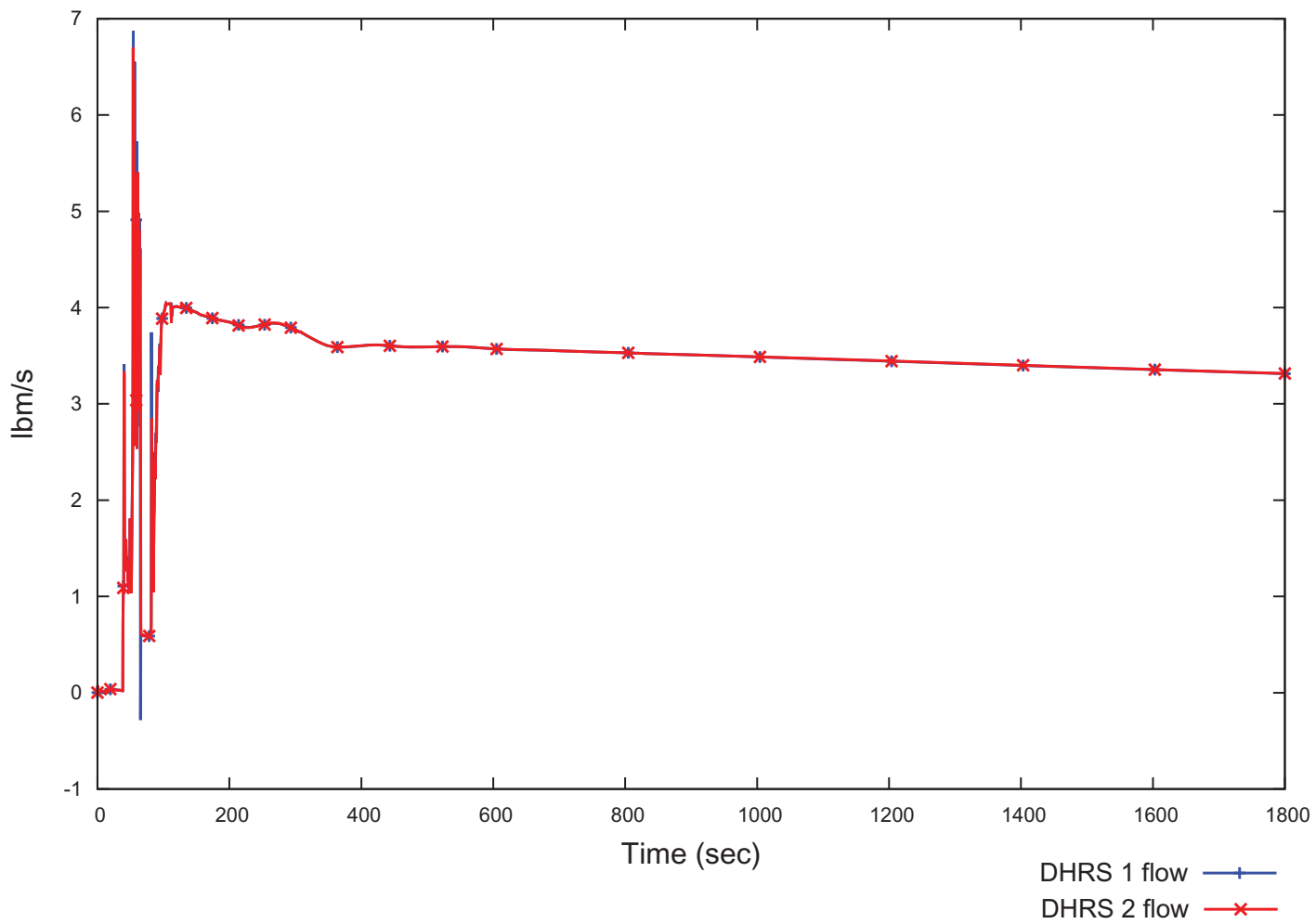


Figure 15.2-7: Reactor Pressure Vessel Pressure - Peak Reactor Coolant System Pressure Case (15.2.1-15.2.3 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum)

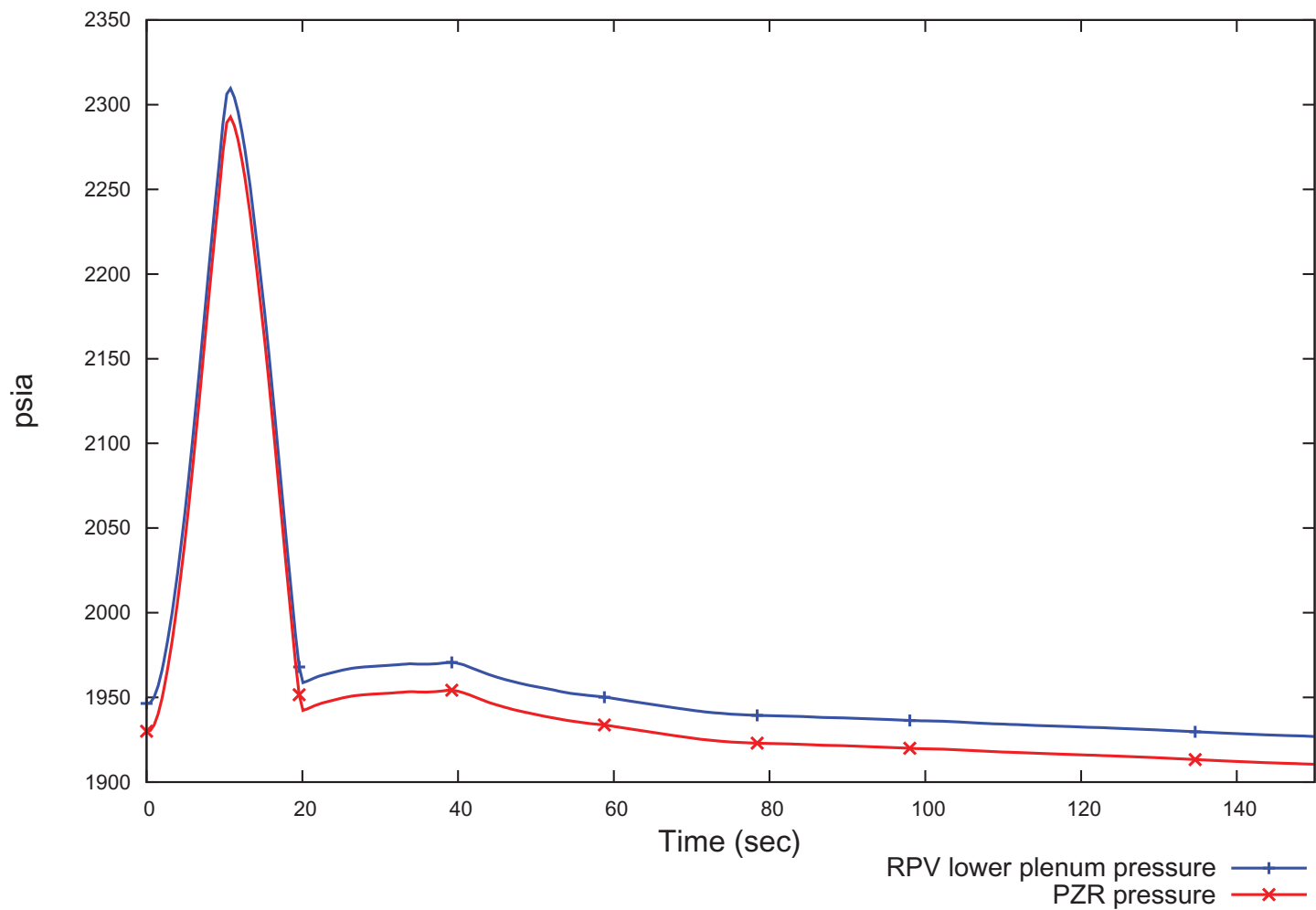


Figure 15.2-8: Steam Generator Pressure - Peak Steam Generator Pressure Case (15.2.1-15.2.3 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum)

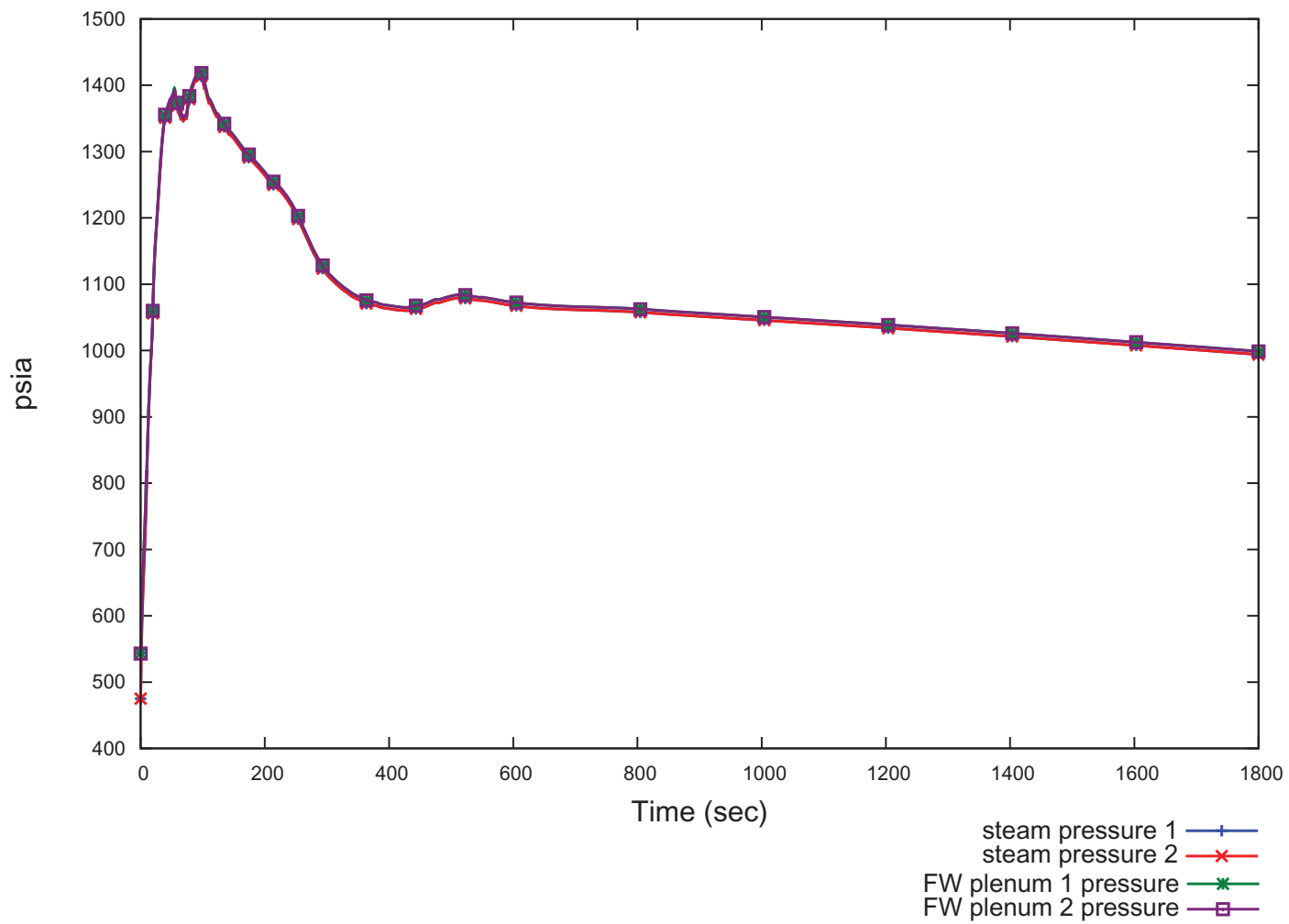
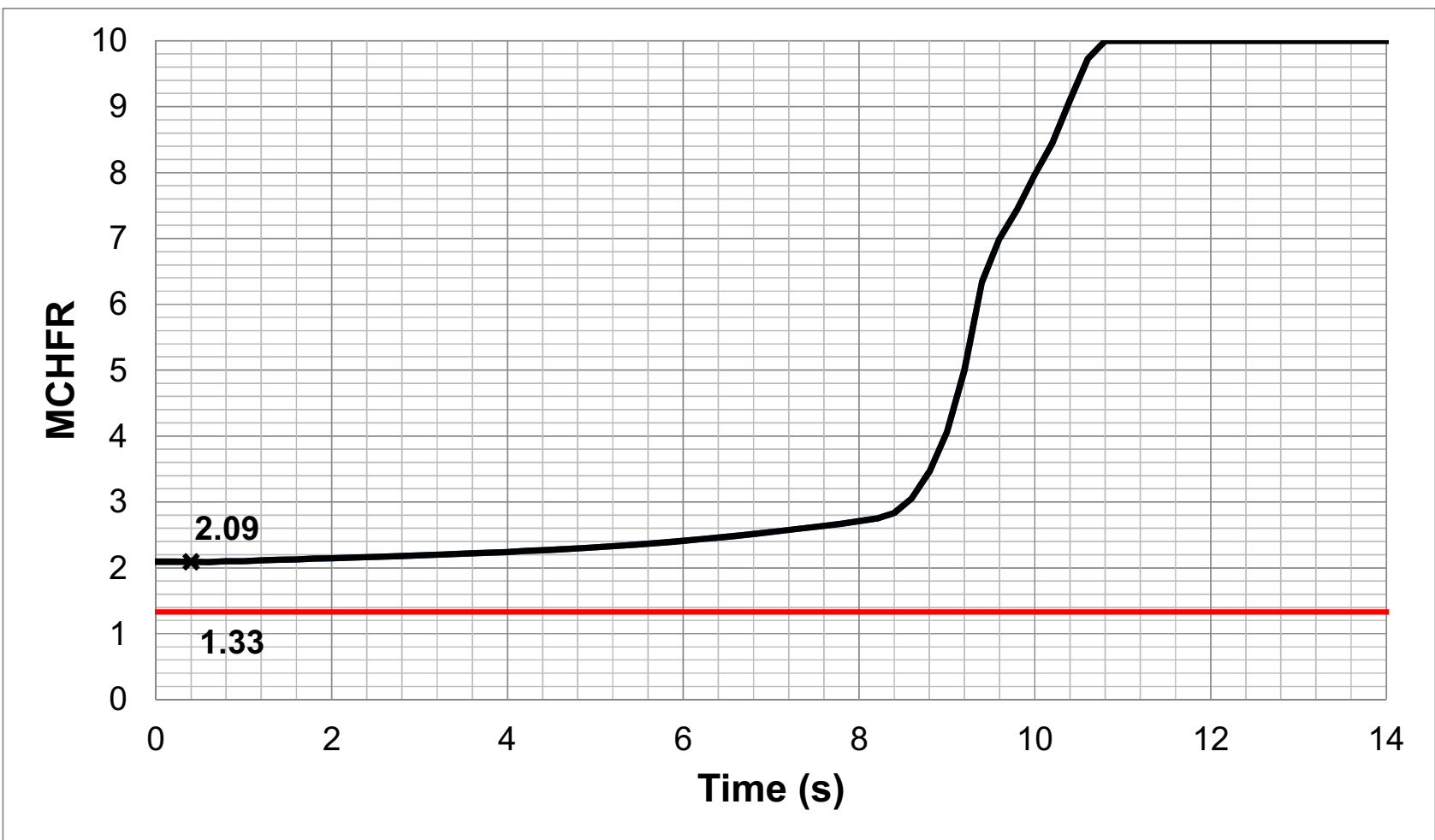


Figure 15.2-9: Minimum Critical Heat Flux Ratio - Limiting Minimum Critical Heat Flux Ratio Case
(15.2.1-15.2.3 Loss of External Load, Turbine Trip, and Loss of Condenser Vacuum)



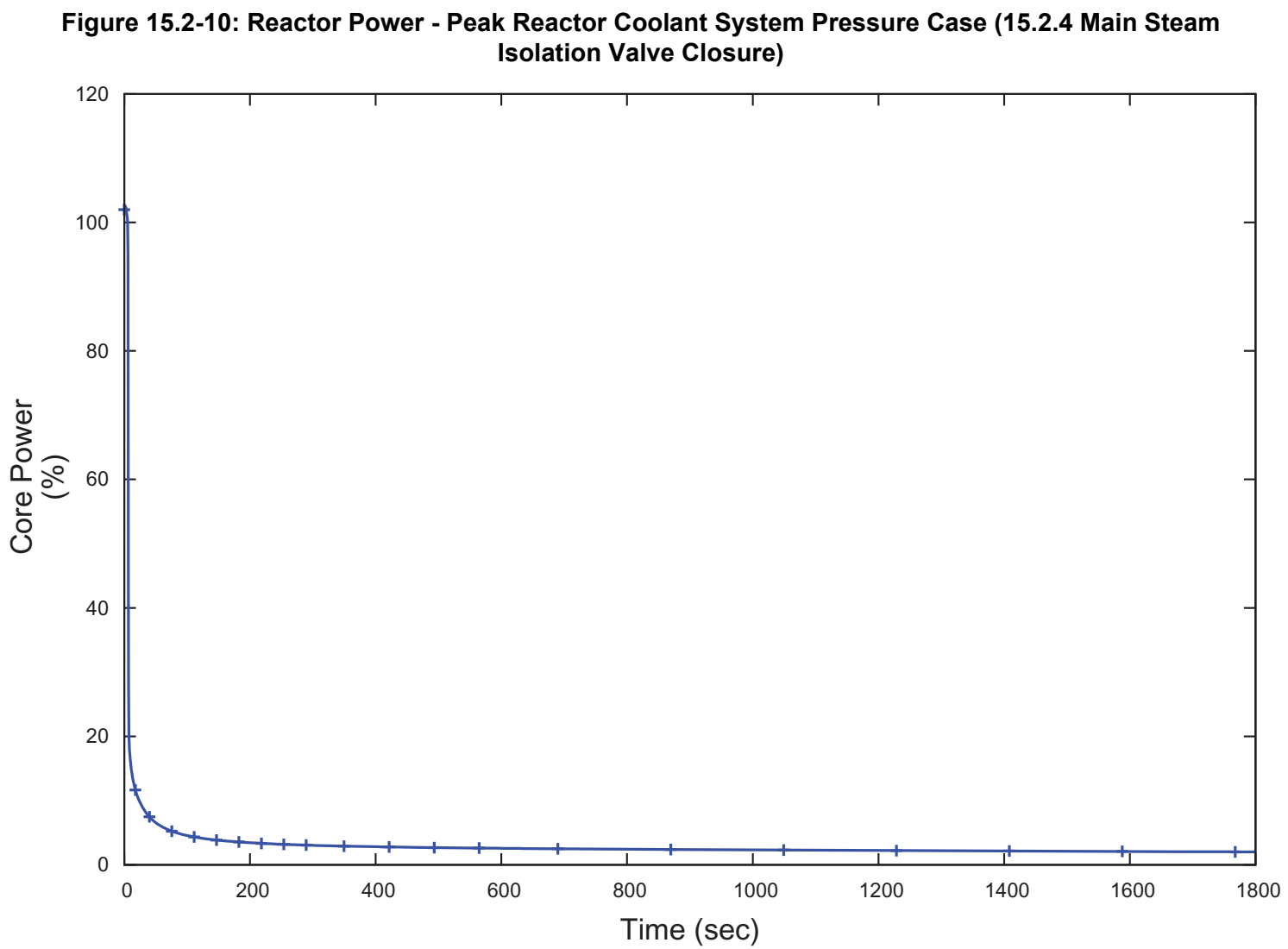
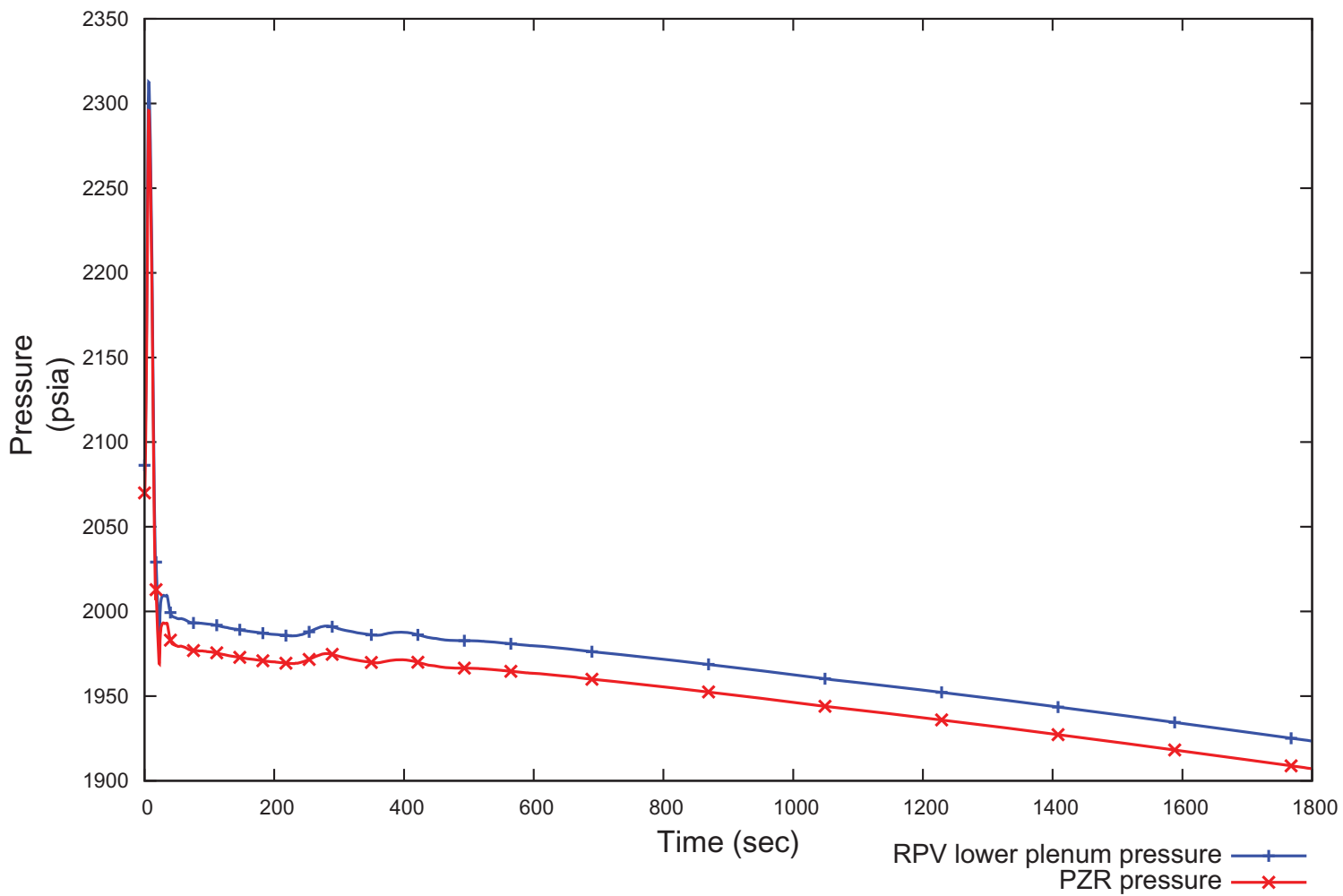


**Figure 15.2-11: Reactor Pressure Vessel Pressure - Peak Reactor Coolant System Pressure Case
(15.2.4 Main Steam Isolation Valve Closure)**



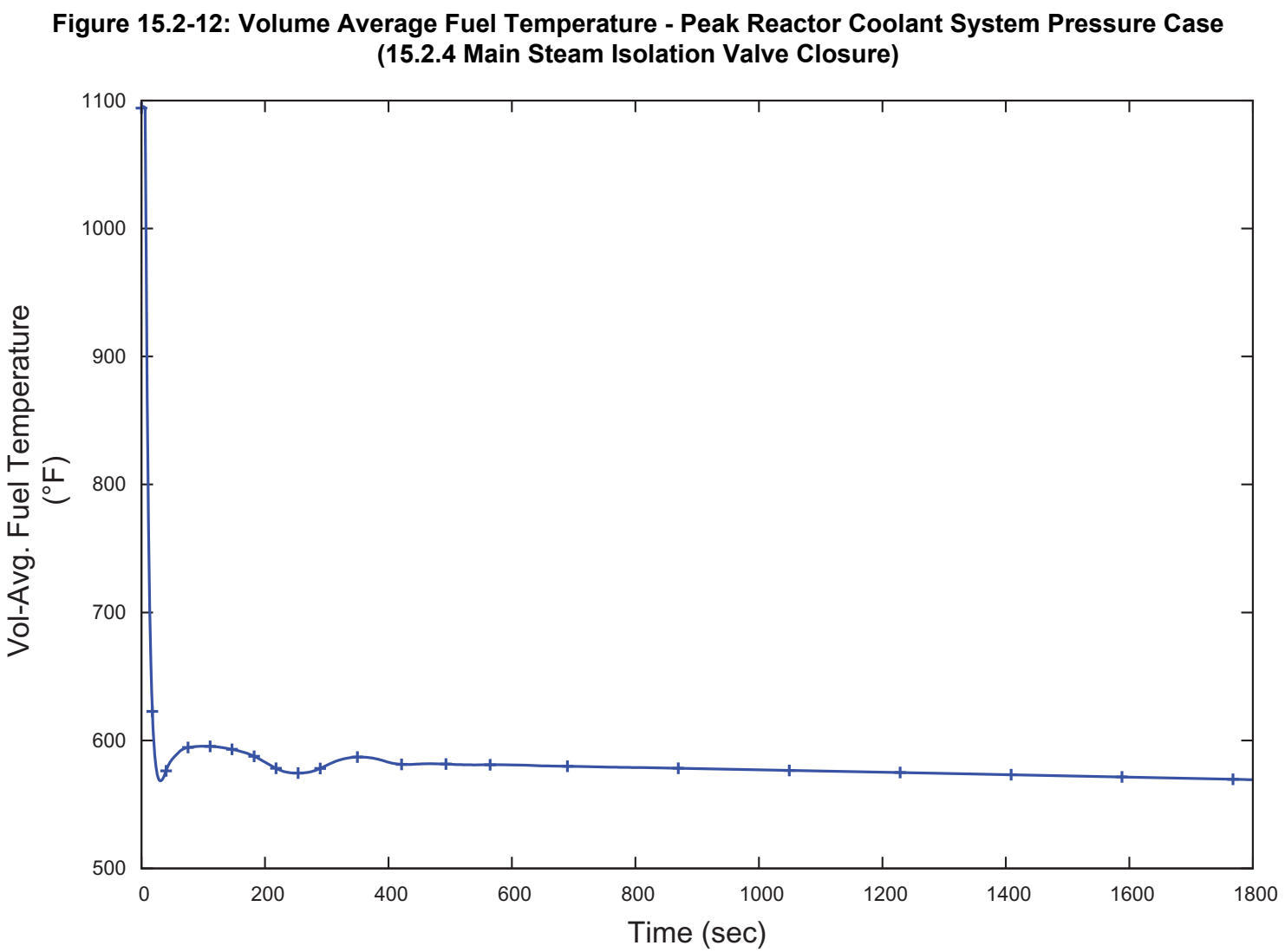


**Figure 15.2-13: Reactor Coolant System Flow Rate - Peak Reactor Coolant System Pressure Case
(15.2.4 Main Steam Isolation Valve Closure)**

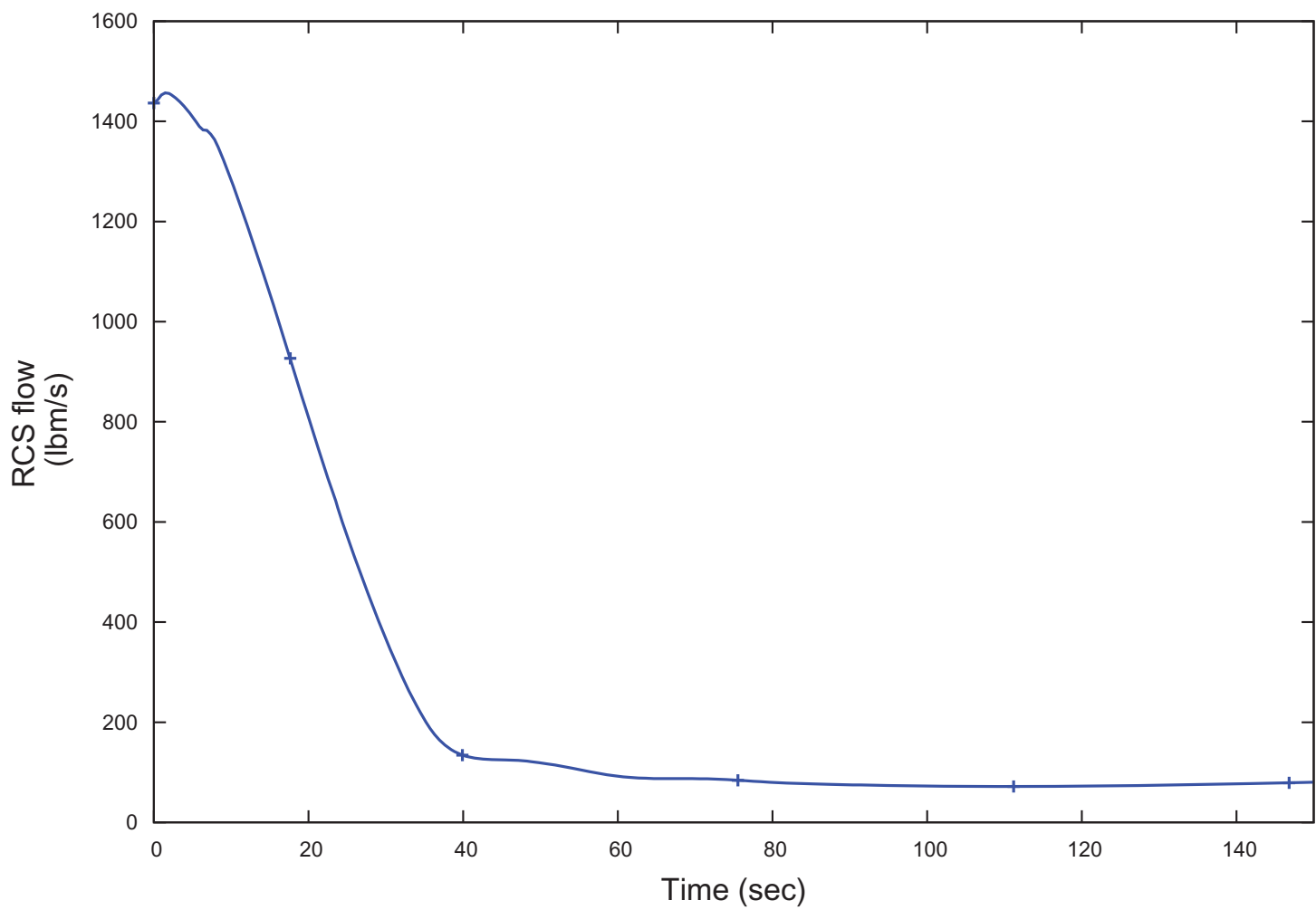


Figure 15.2-14: Primary Temperatures - Peak Reactor Coolant System Pressure Case (15.2.4 Main Steam Isolation Valve Closure)

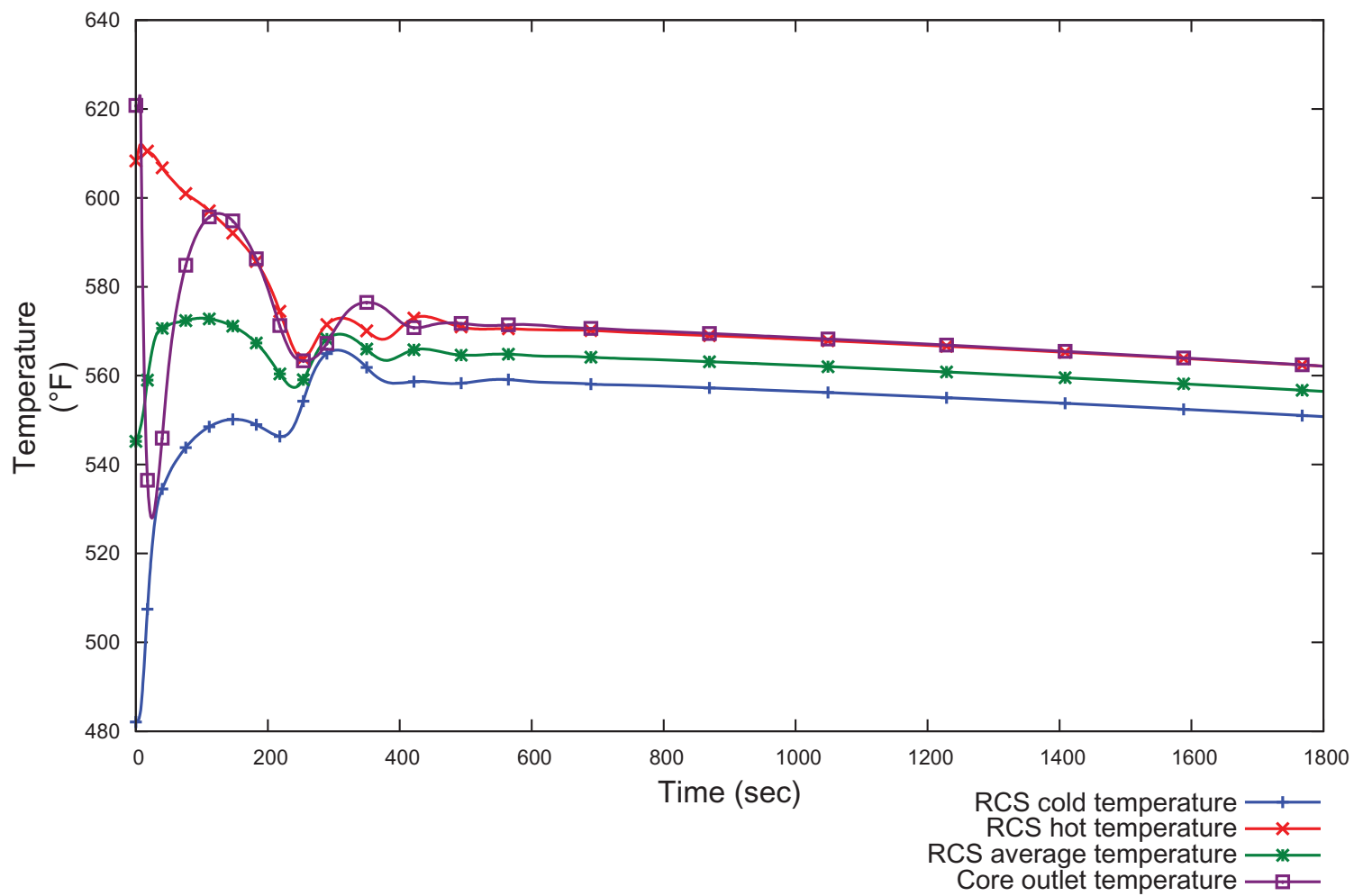


Figure 15.2-15: Pressurizer Level - Peak Reactor Coolant System Pressure Case (15.2.4 Main Steam Isolation Valve Closure)

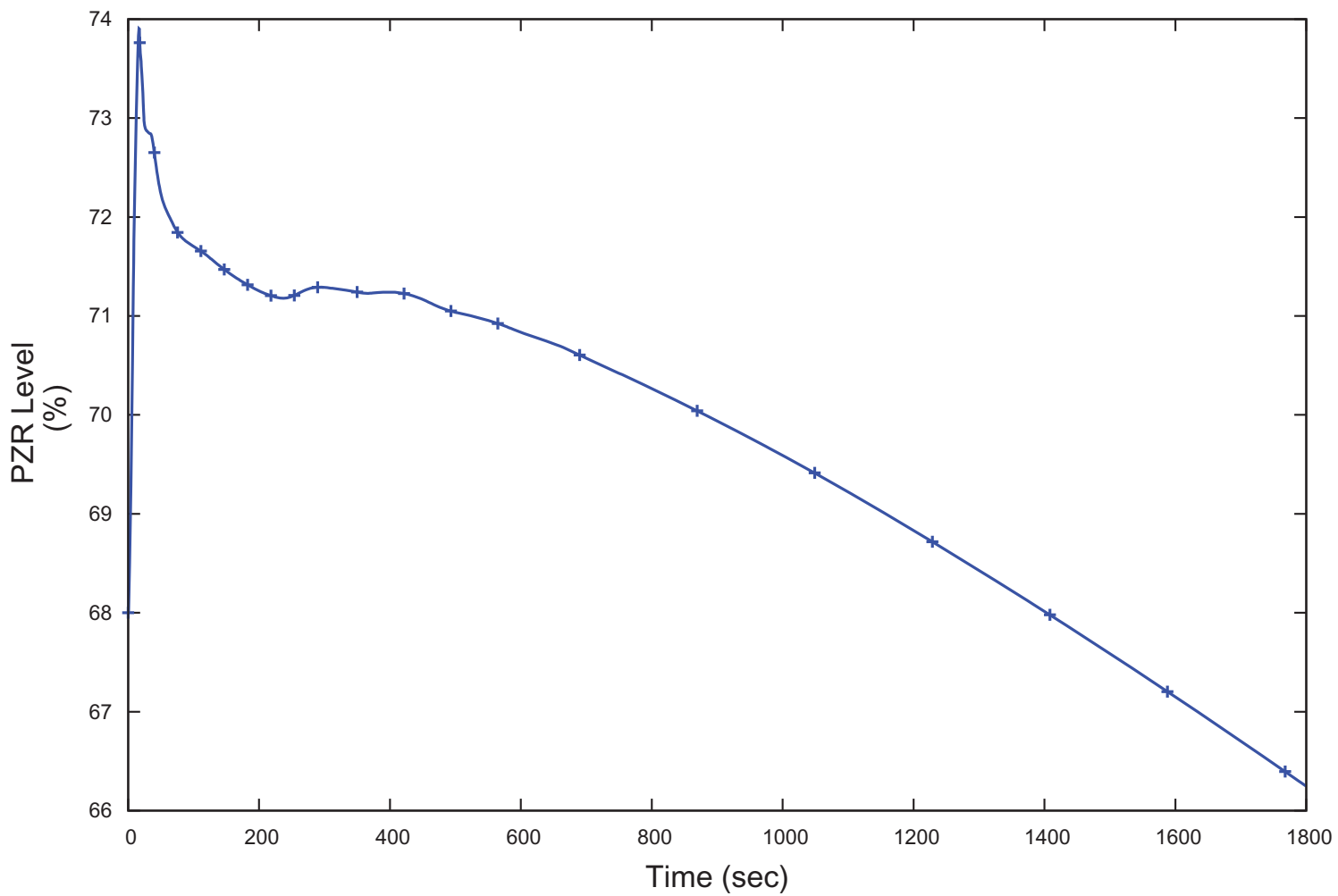
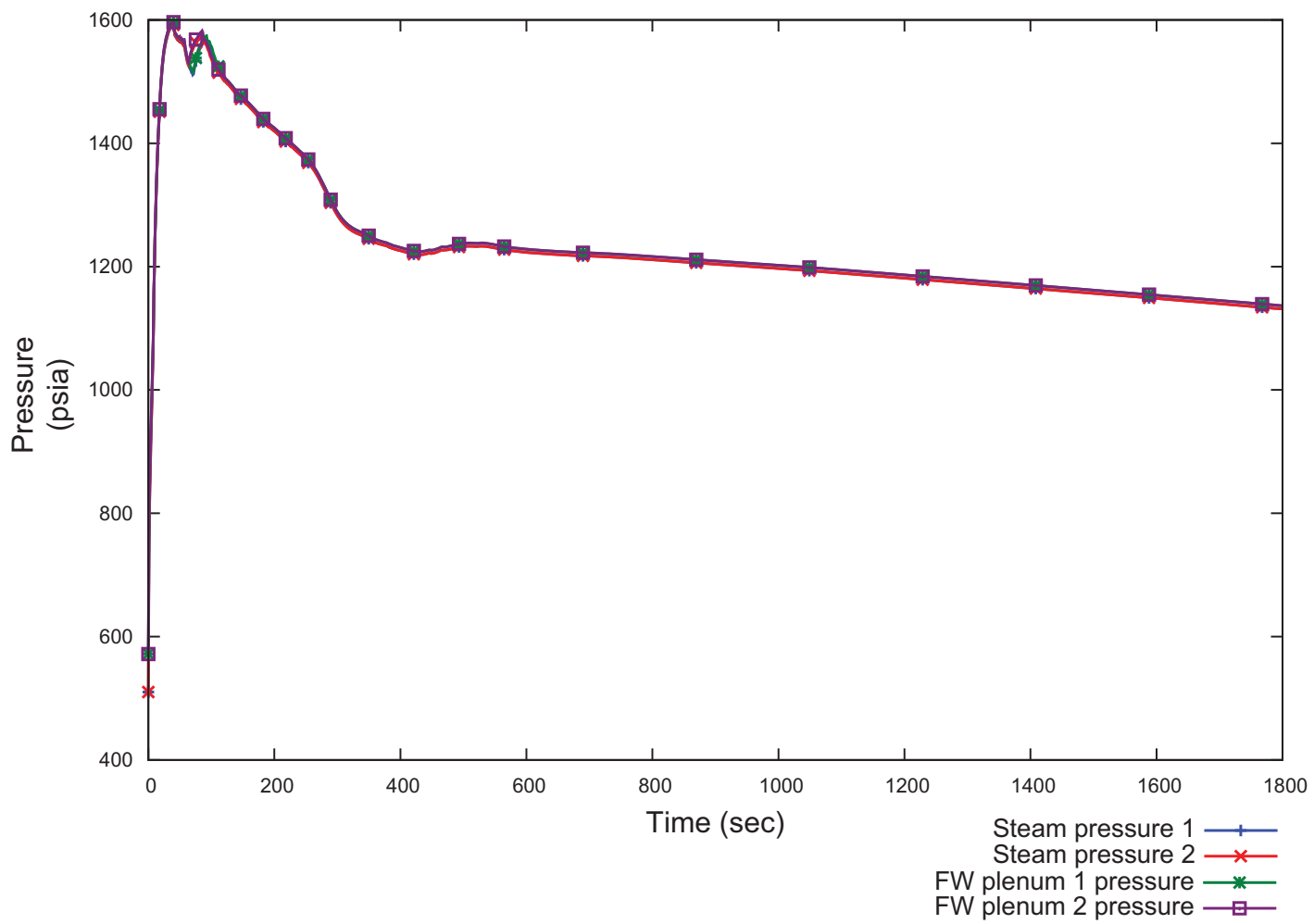


Figure 15.2-16: Steam Generator Pressure - Peak Steam Generator Pressure Case (15.2.4 Main Steam Isolation Valve Closure)



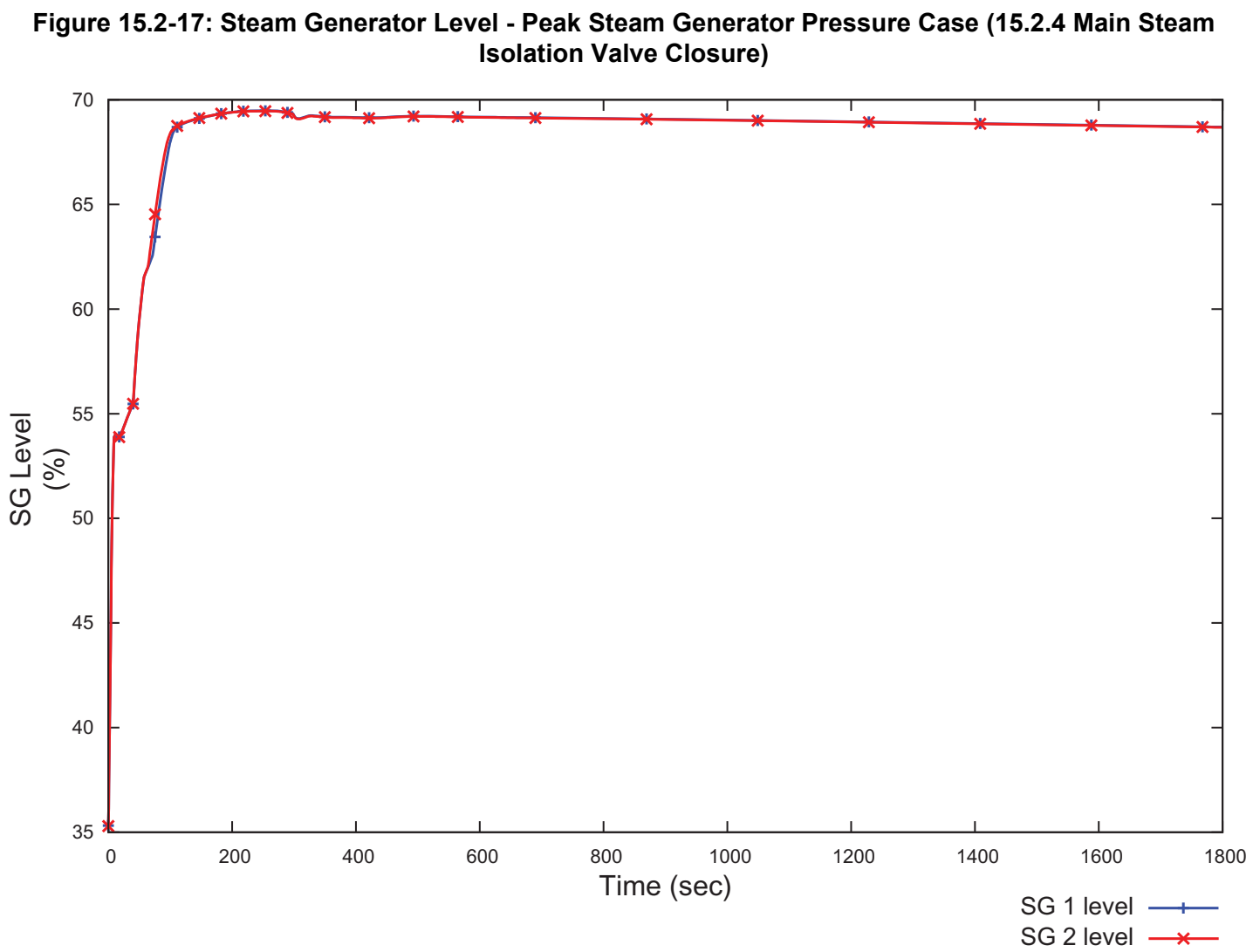


Figure 15.2-18: Minimum Critical Heat Flux Ratio - Limiting Minimum Critical Heat Flux Ratio Case
(15.2.4 Main Steam Isolation Valve Closure)

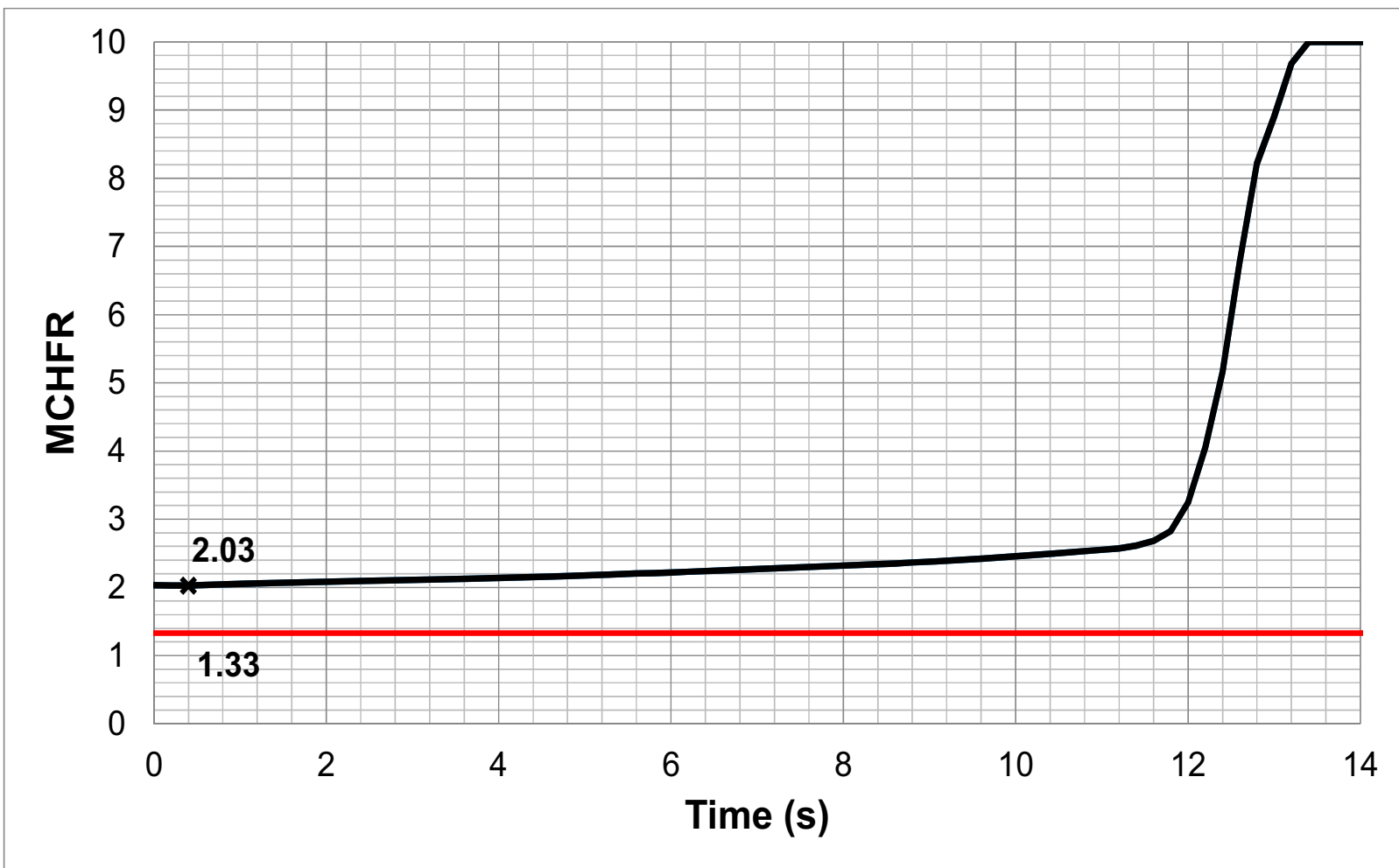
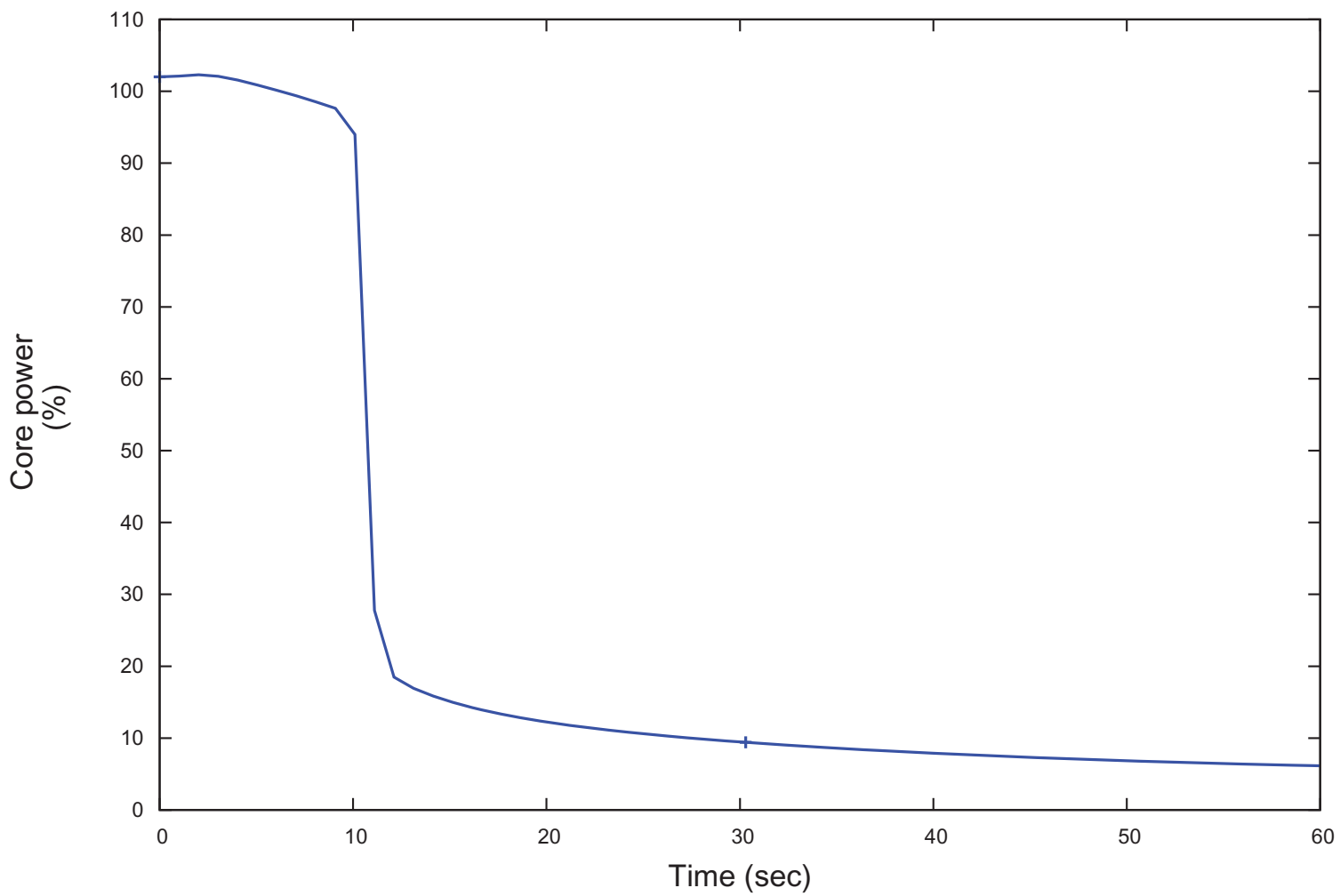


Figure 15.2-19: Reactor Power - Peak Reactor Coolant System Pressure Case (15.2.6 Loss of Alternating Current Power)



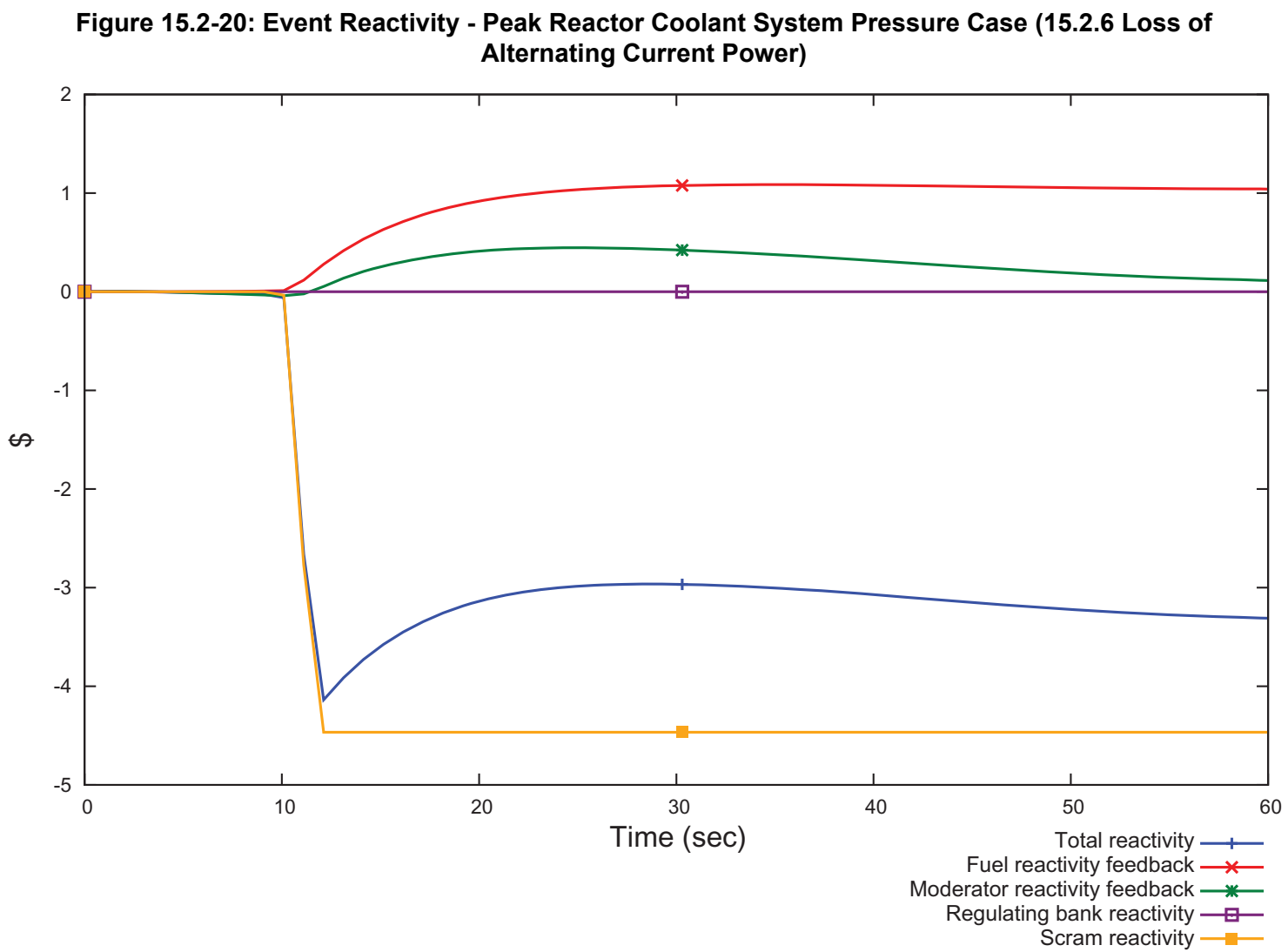


Figure 15.2-21: Pressurizer Level - Peak Reactor Coolant System Pressure Case (15.2.6 Loss of Alternating Current Power)

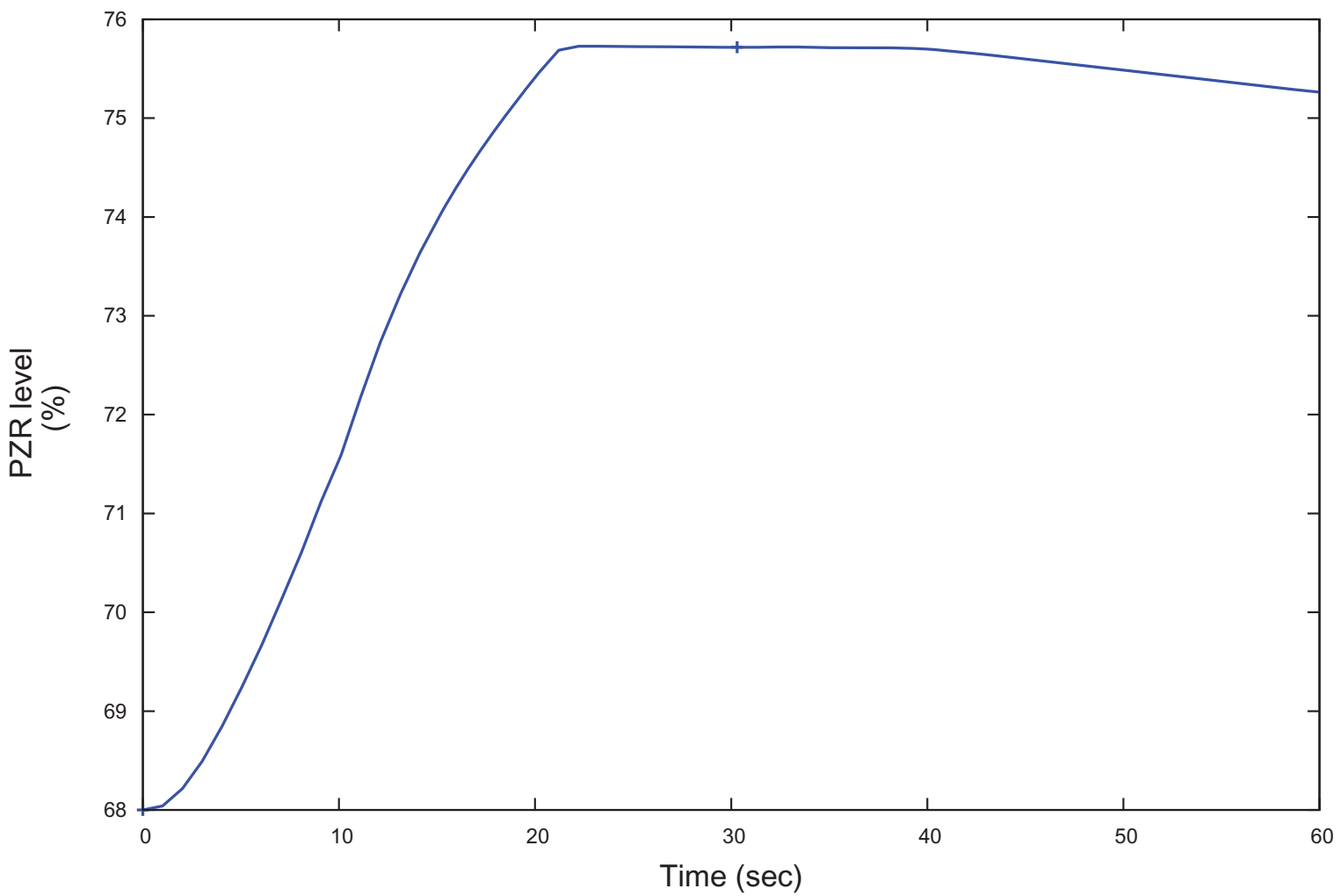


**Figure 15.2-22: Reactor Pressure Vessel Pressure - Peak Reactor Coolant System Pressure Case
(15.2.6 Loss of Alternating Current Power)**

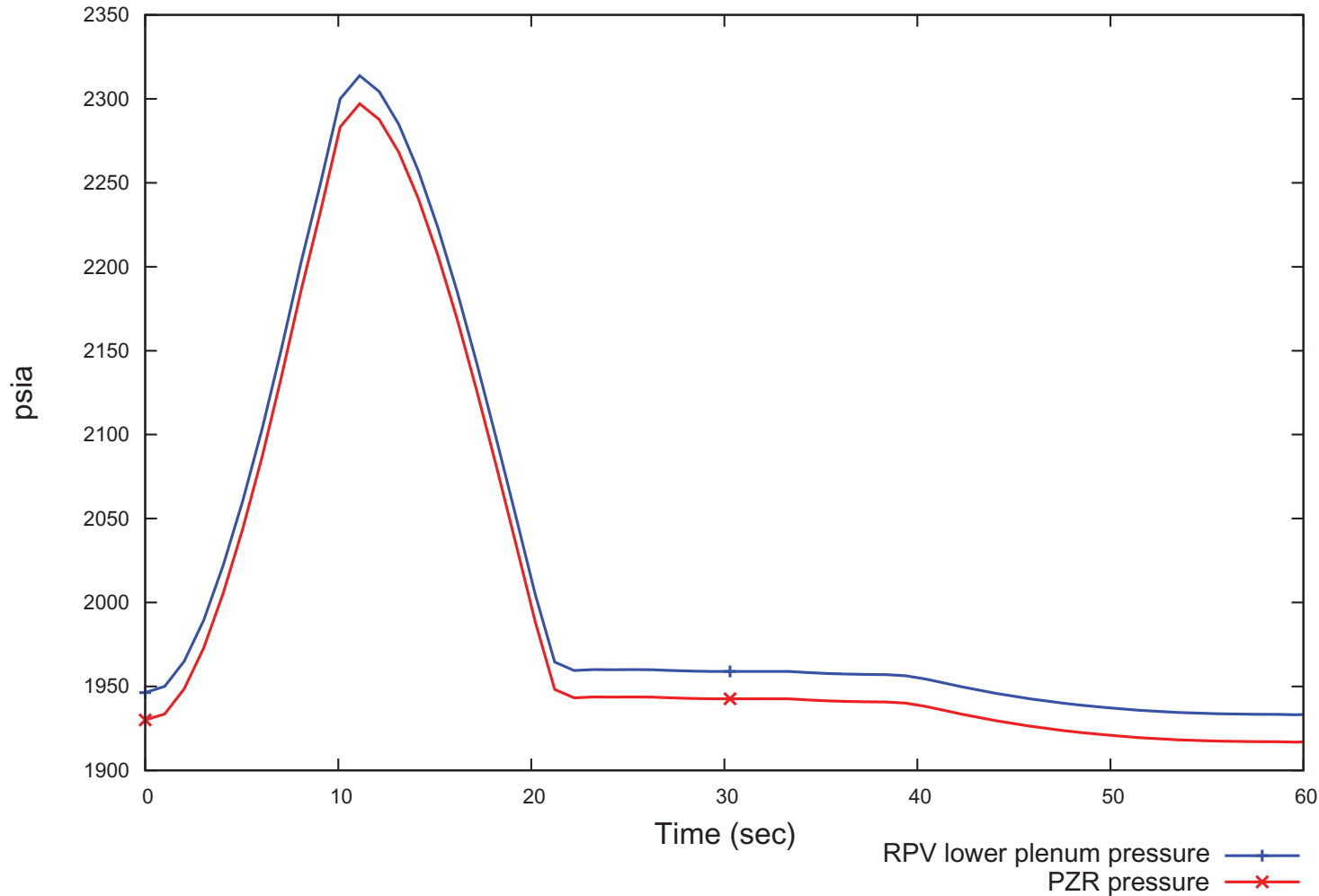


**Figure 15.2-23: Reactor Coolant System Flow Rate - Peak Reactor Coolant System Pressure Case
(15.2.6 Loss of Alternating Current Power)**

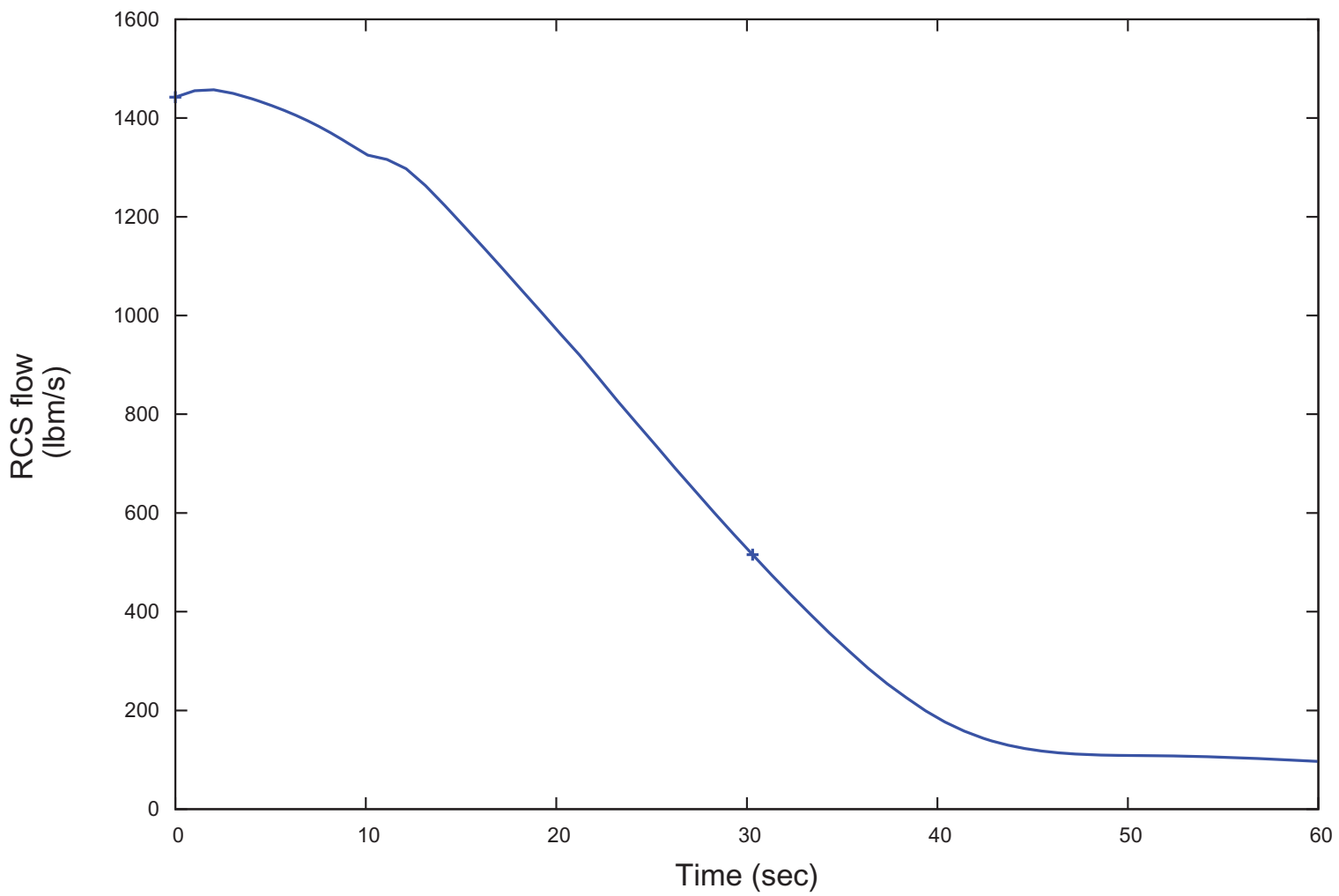


Figure 15.2-24: Reactor Coolant System Temperatures - Peak Reactor Coolant System Pressure Case (15.2.6 Loss of Alternating Current Power)

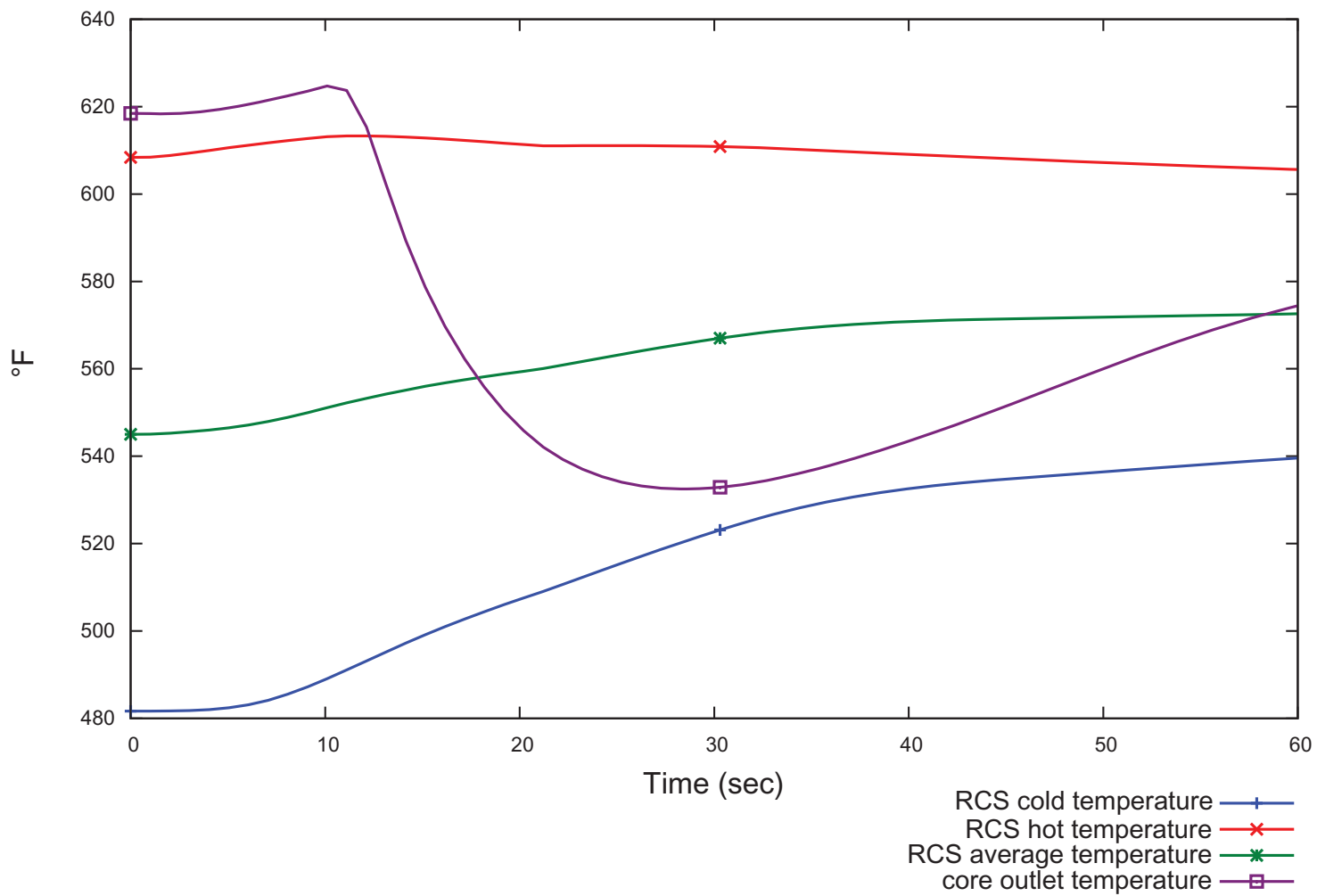


Figure 15.2-25: Steam Generator Pressure - Peak Steam Generator Pressure Case (15.2.6 Loss of Alternating Current Power)

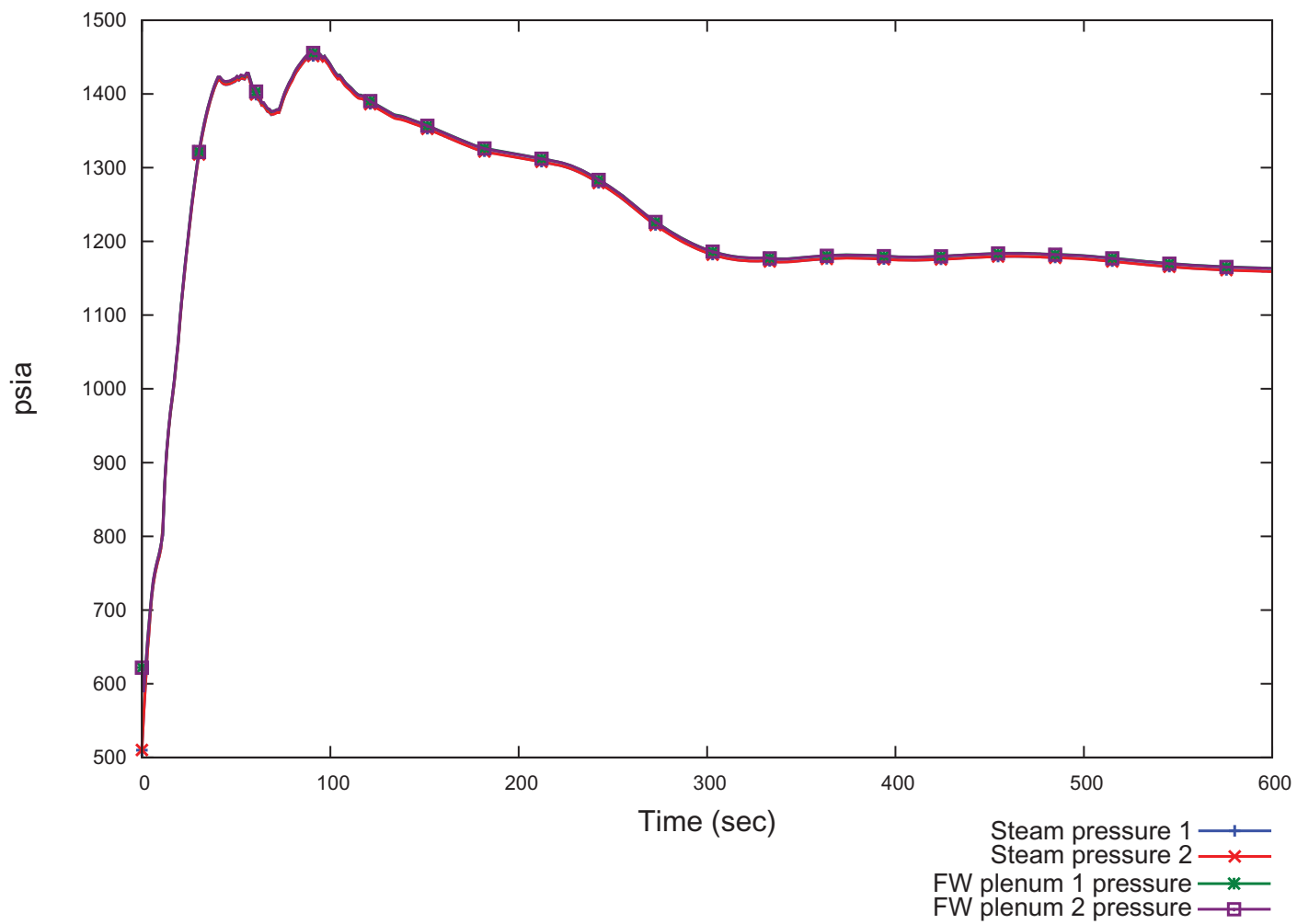


Figure 15.2-26: Critical Heat Flux Ratio - Minimum Critical Heat Flux Ratio Case (15.2.6 Loss of Alternating Current Power)

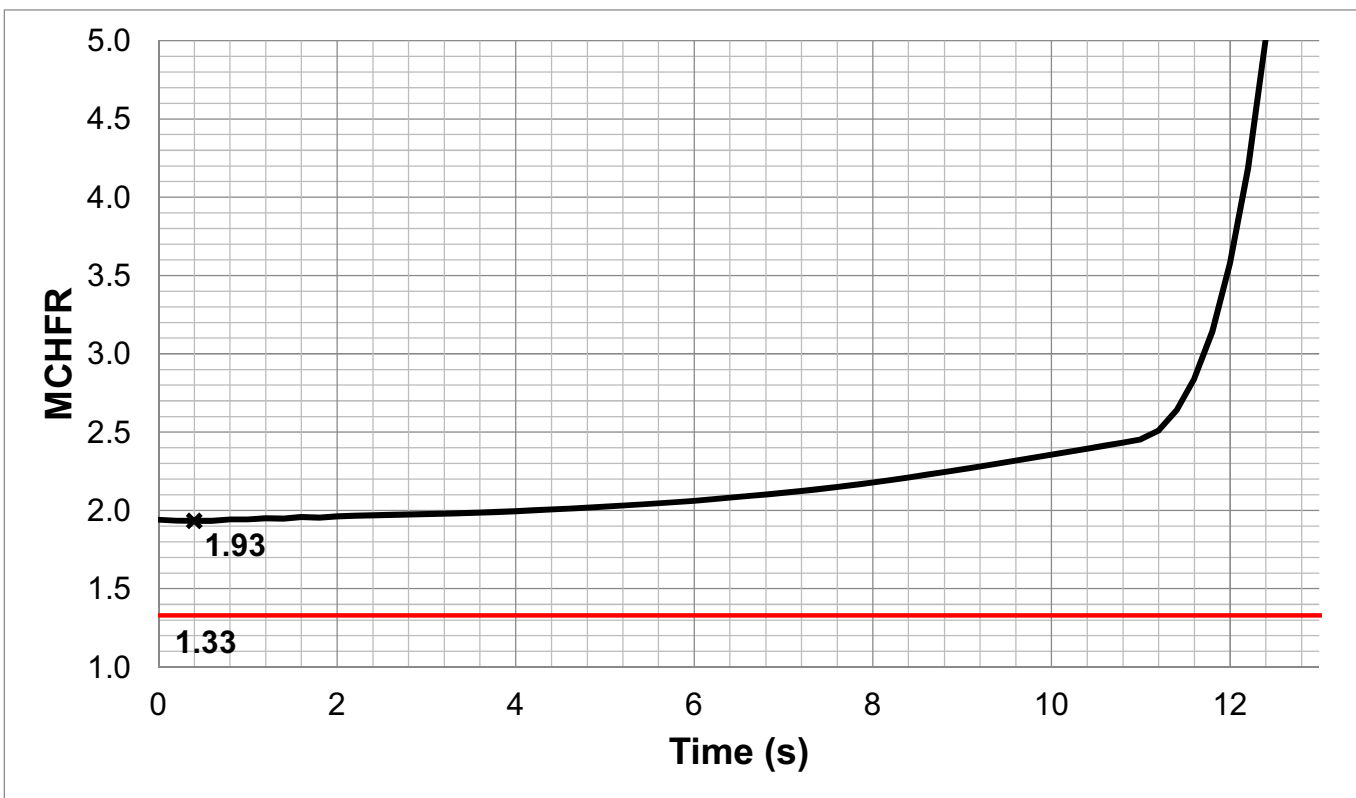


Figure 15.2-27: Reactor Power - Peak Reactor Coolant System Pressure Case (15.2.7 Loss of Feedwater)

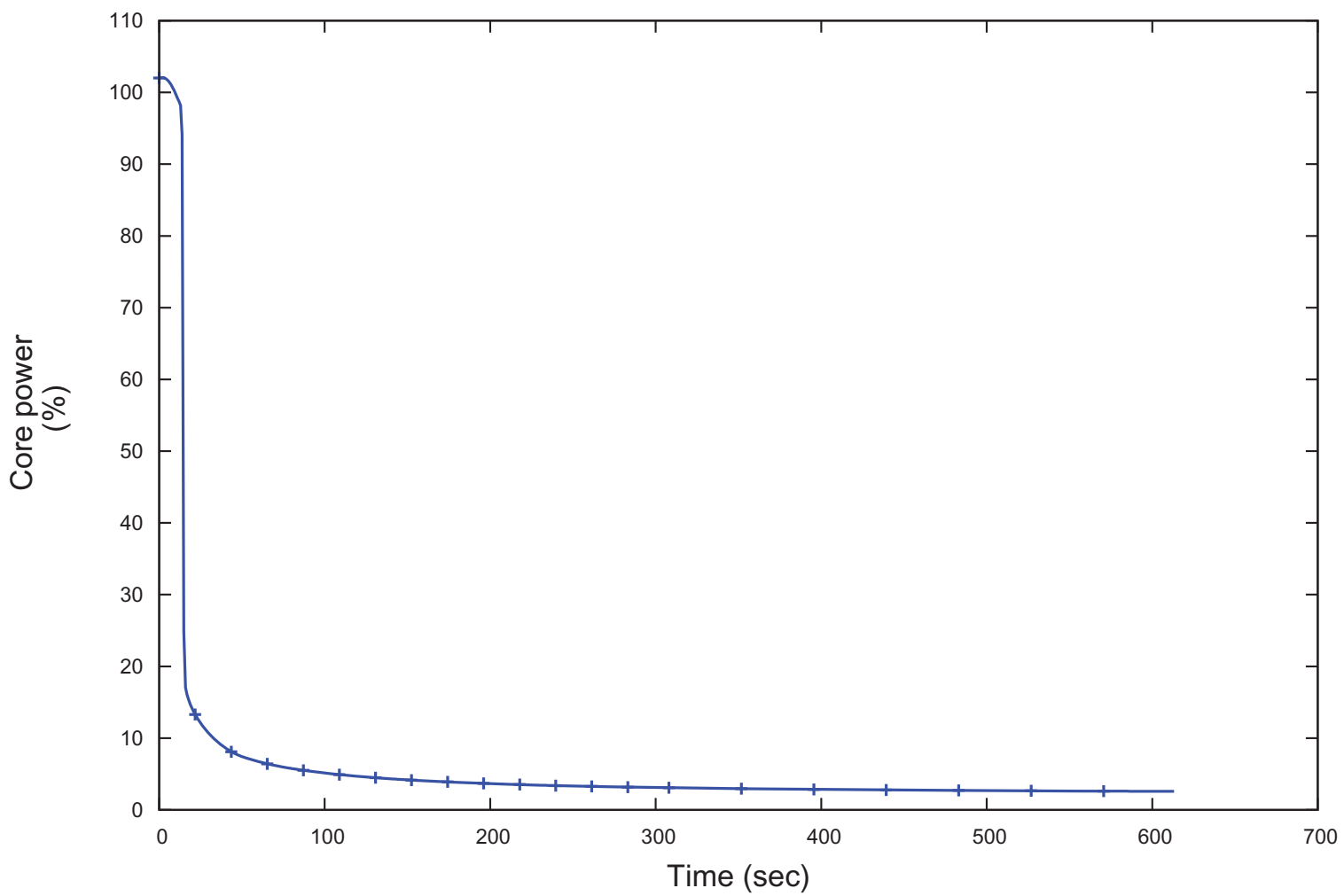
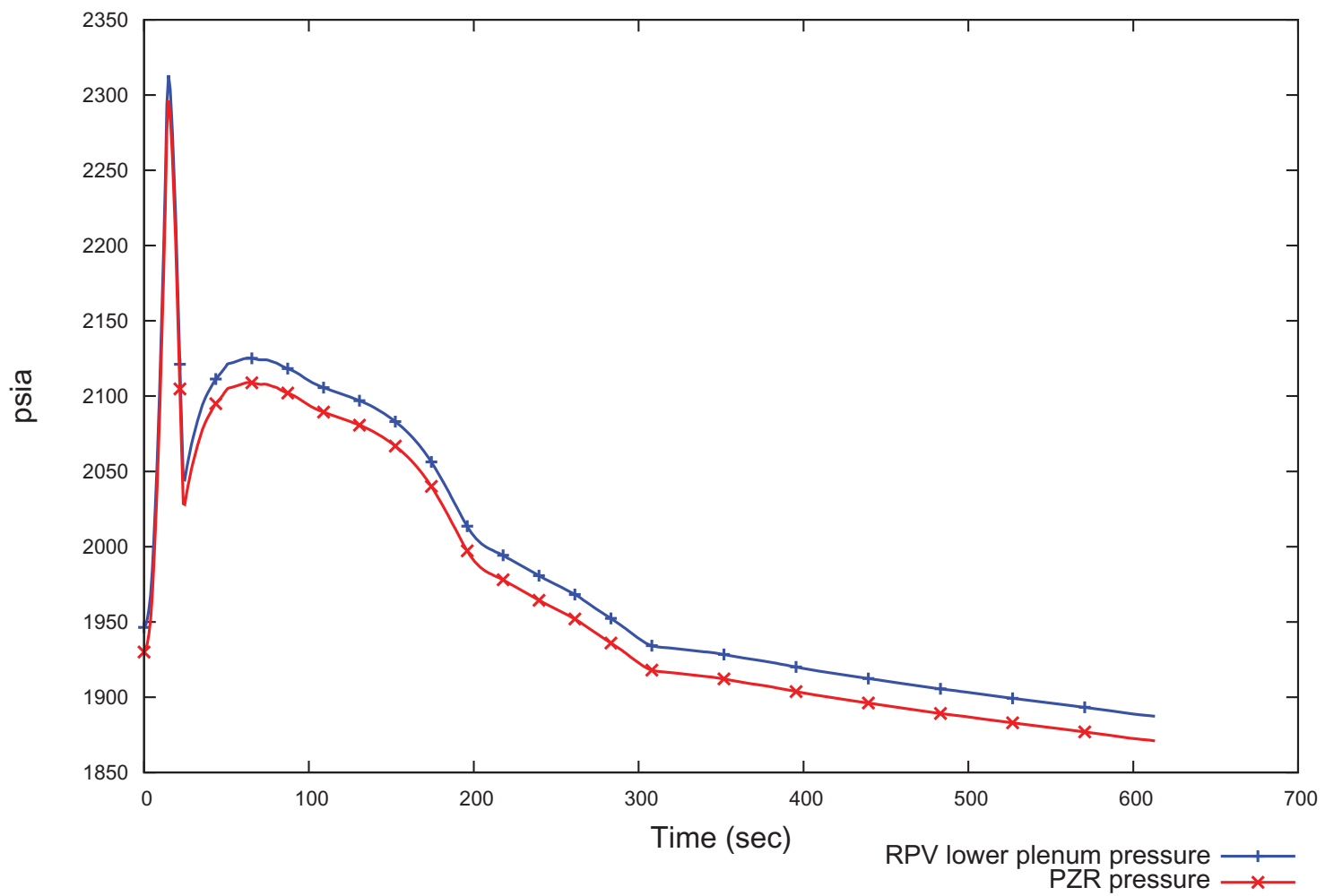


Figure 15.2-28: Reactor Pressure Vessel Pressure - Peak Reactor Coolant System Pressure Case (15.2.7 Loss of Feedwater)



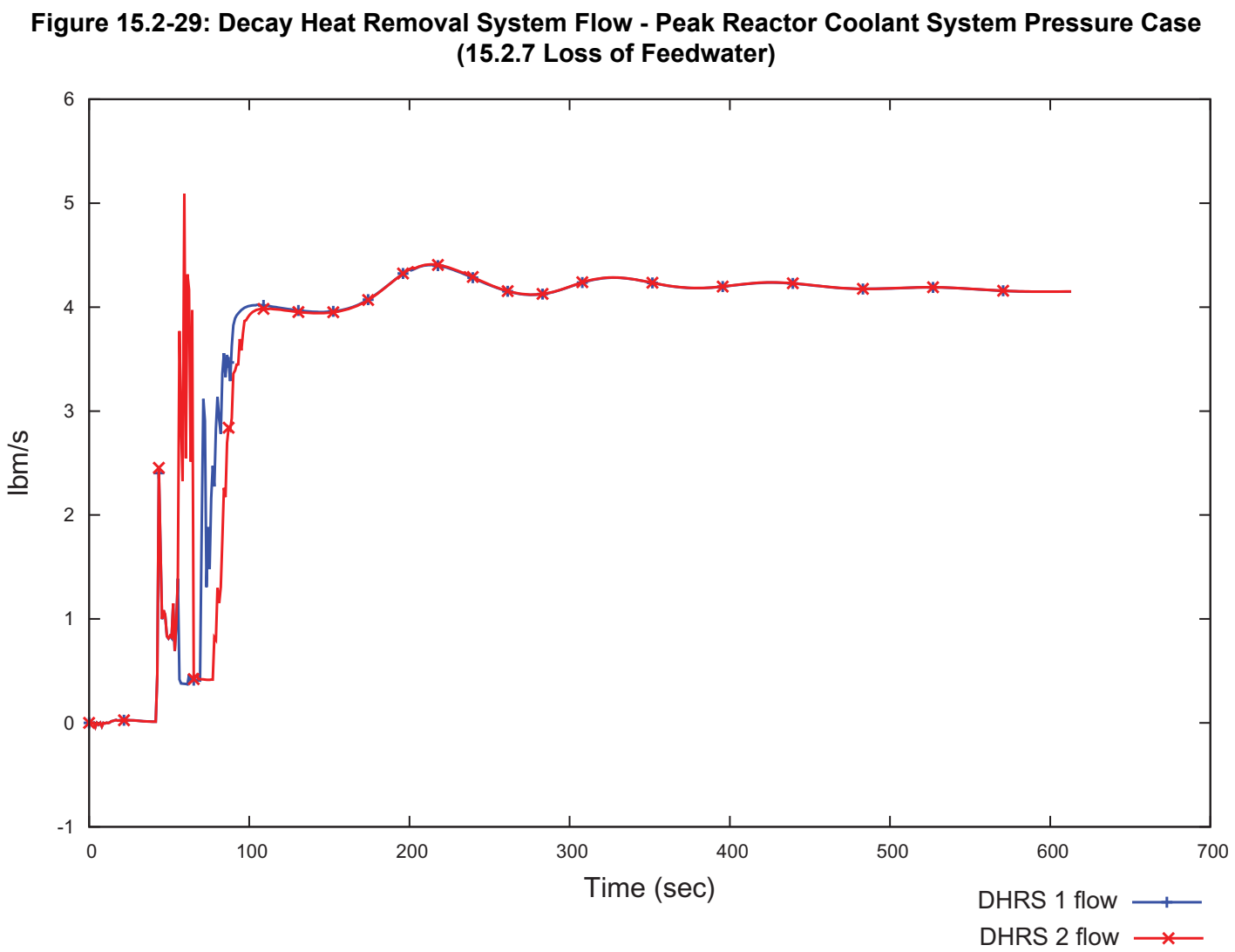


**Figure 15.2-30: Reactor Coolant System Flow Rate - Peak Reactor Coolant System Pressure Case
(15.2.7 Loss of Feedwater)**

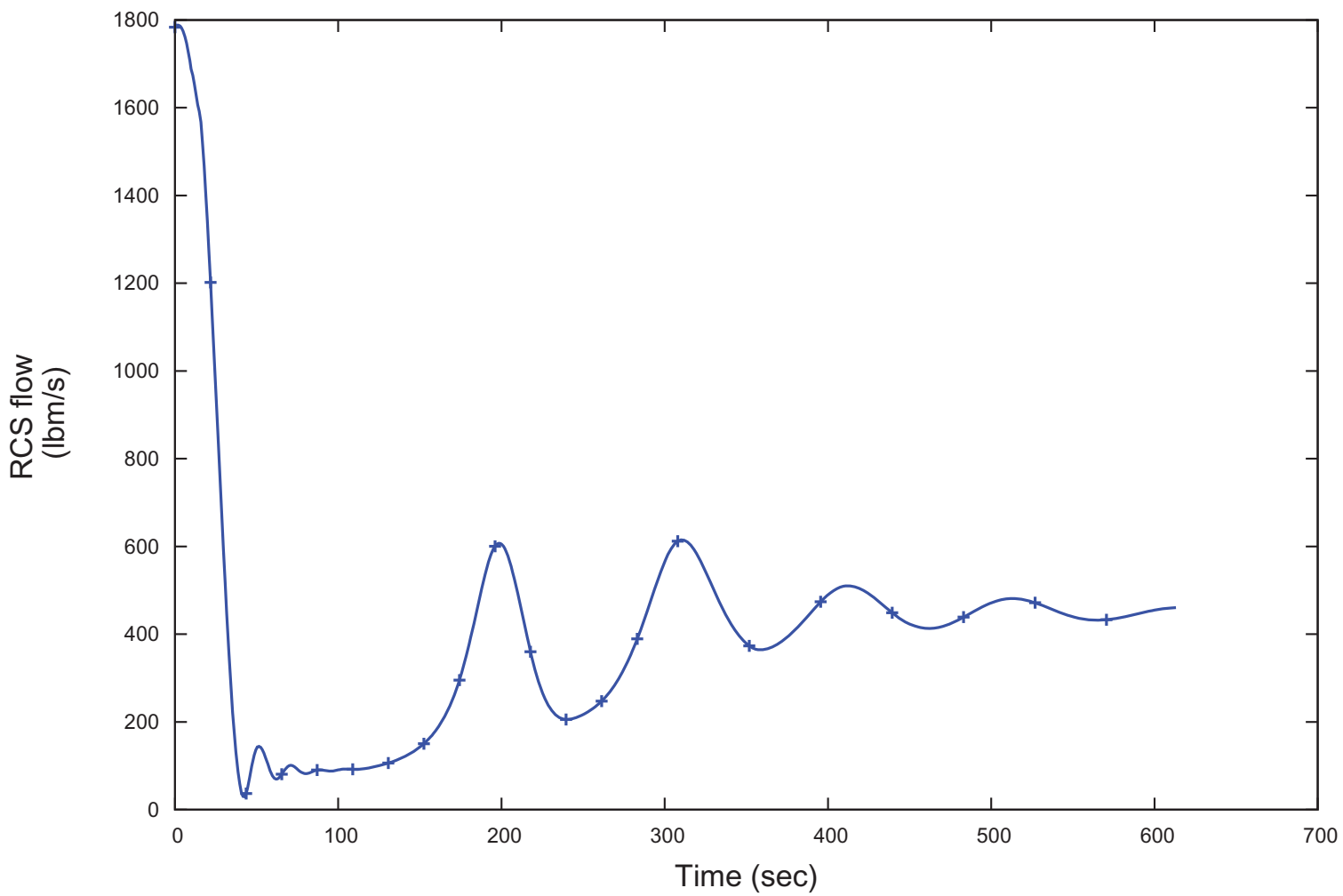


Figure 15.2-31: Reactor Coolant System Average Temperature - Peak Reactor Coolant System Pressure Case (15.2.7 Loss of Feedwater)

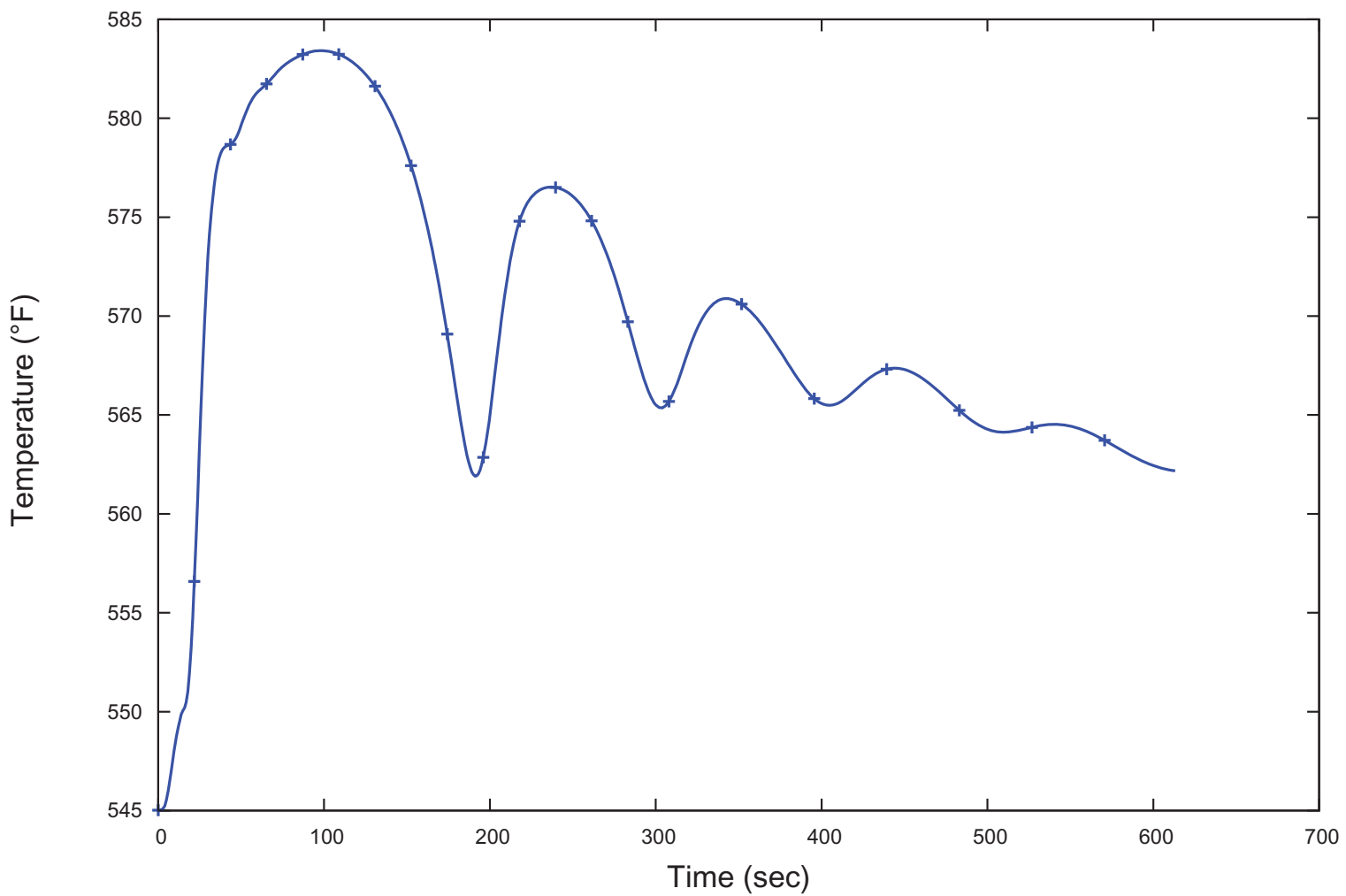


Figure 15.2-32: Pressurizer Level - Peak Reactor Coolant System Pressure Case (15.2.7 Loss of Feedwater)

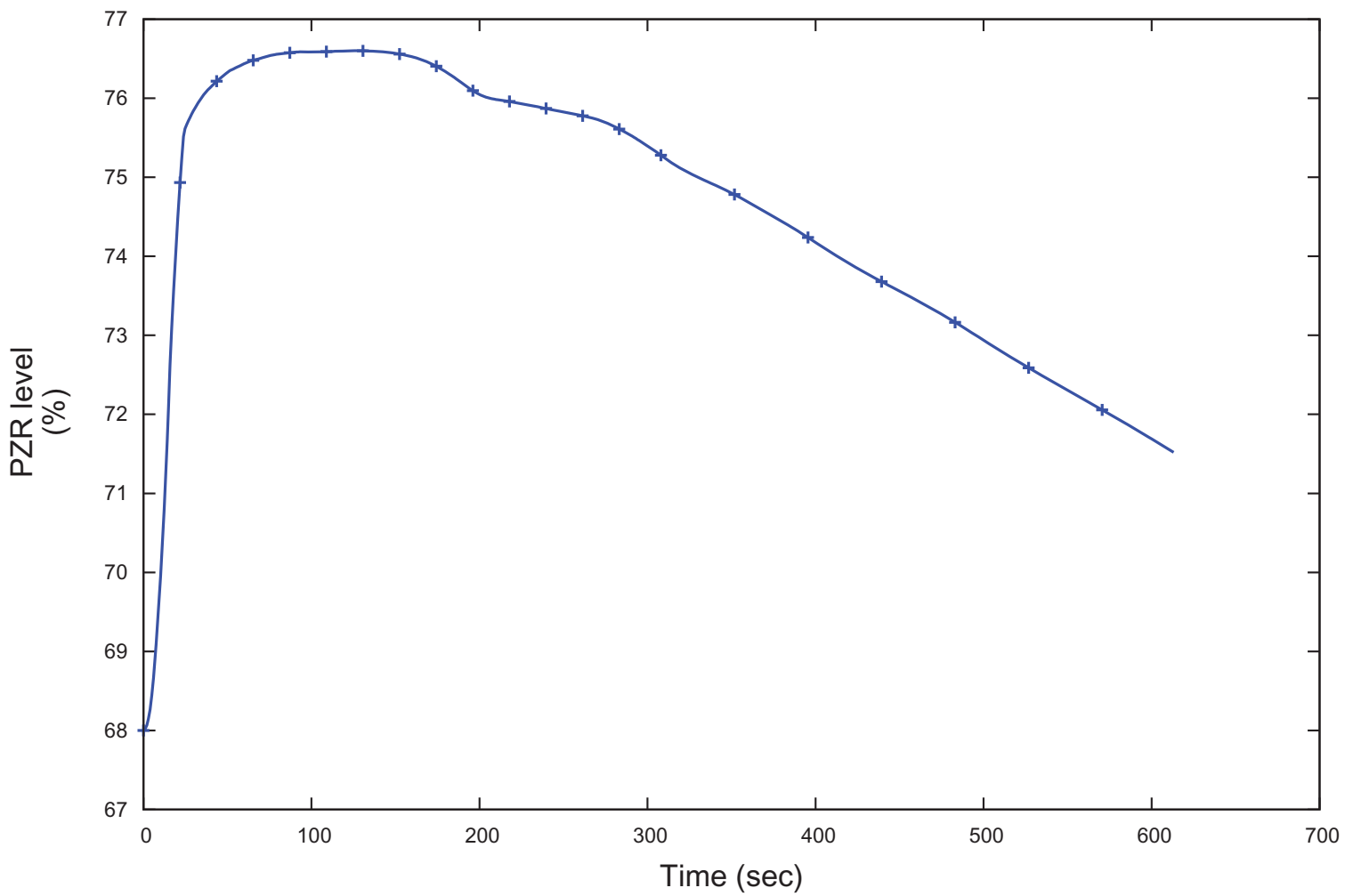


Figure 15.2-33: Steam Generator Pressure - Peak Steam Generator Pressure Case (15.2.7 Loss of Feedwater)

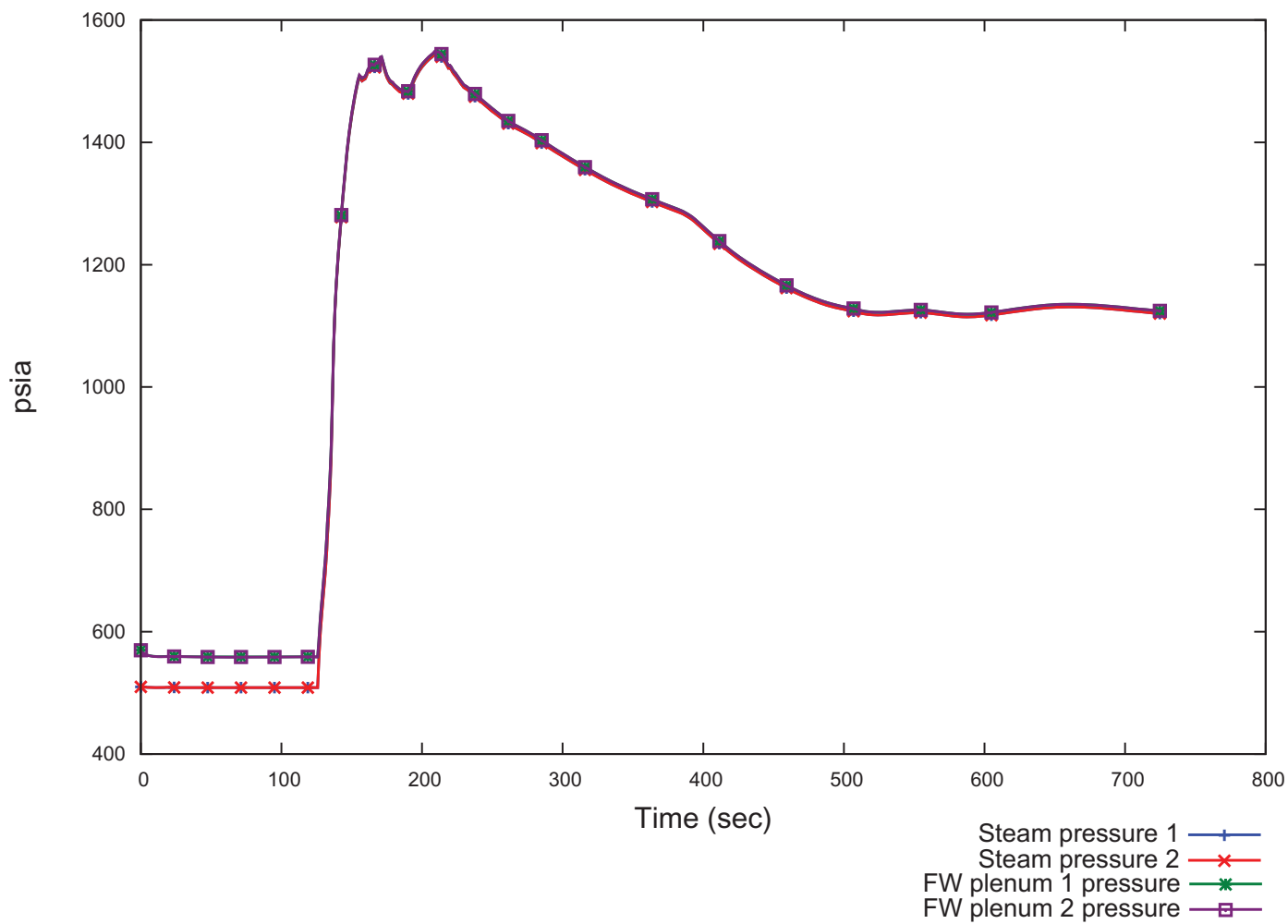


Figure 15.2-34: Minimum Critical Heat Flux Ratio - Limiting Minimum Critical Heat Flux Ratio Case
(15.2.7 Loss of Feedwater)

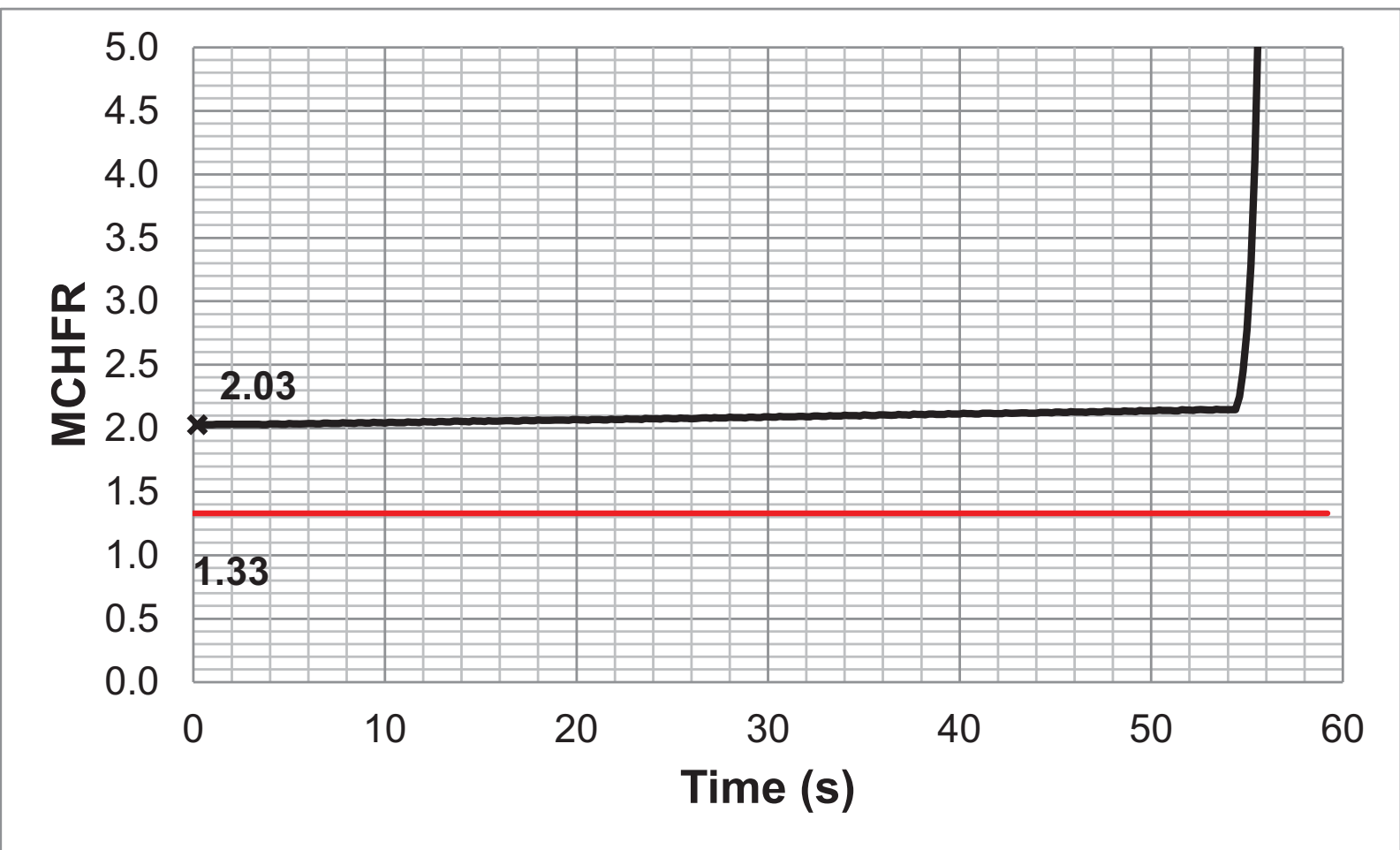


**Figure 15.2-35: Reactor Pressure Vessel Pressure - Peak Reactor Coolant System Pressure Case
(15.2.8 Feedwater Line Break)**

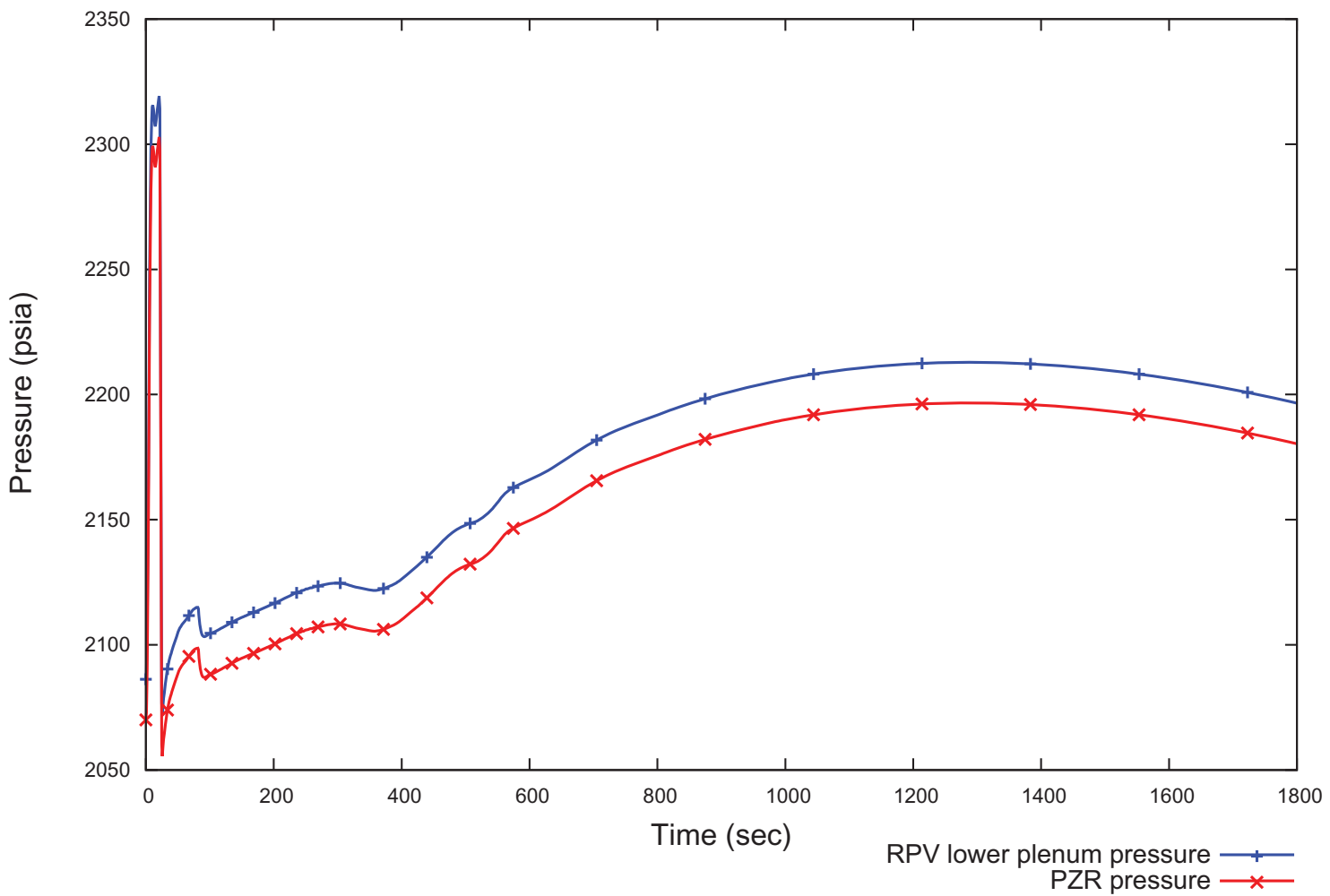


Figure 15.2-36: Reactor Power - Peak Reactor Coolant System Pressure Case (15.2.8 Feedwater Line Break)

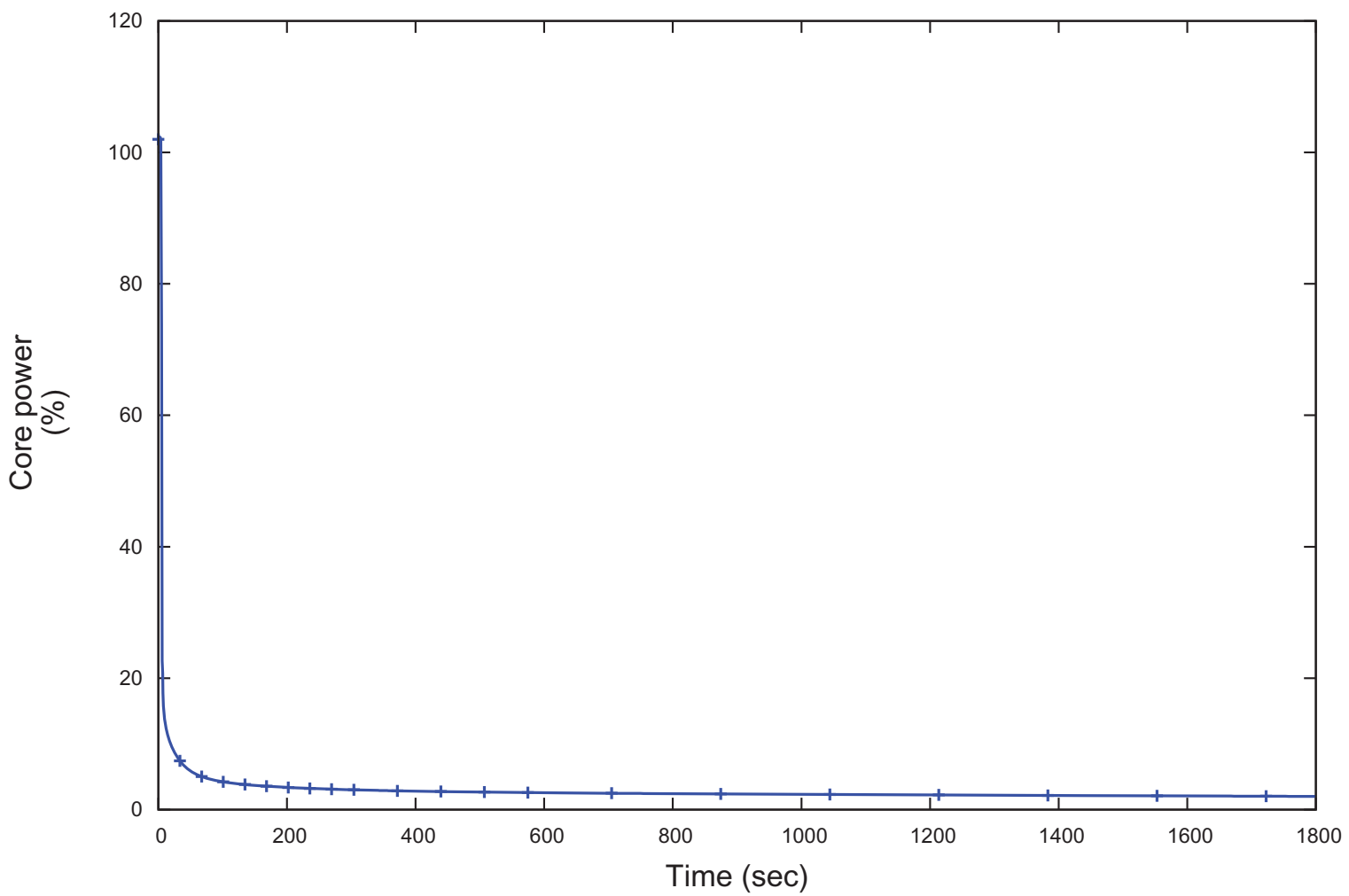


Figure 15.2-37: Secondary Pressure - Peak Reactor Coolant System Pressure Case (15.2.8 Feedwater Line Break)

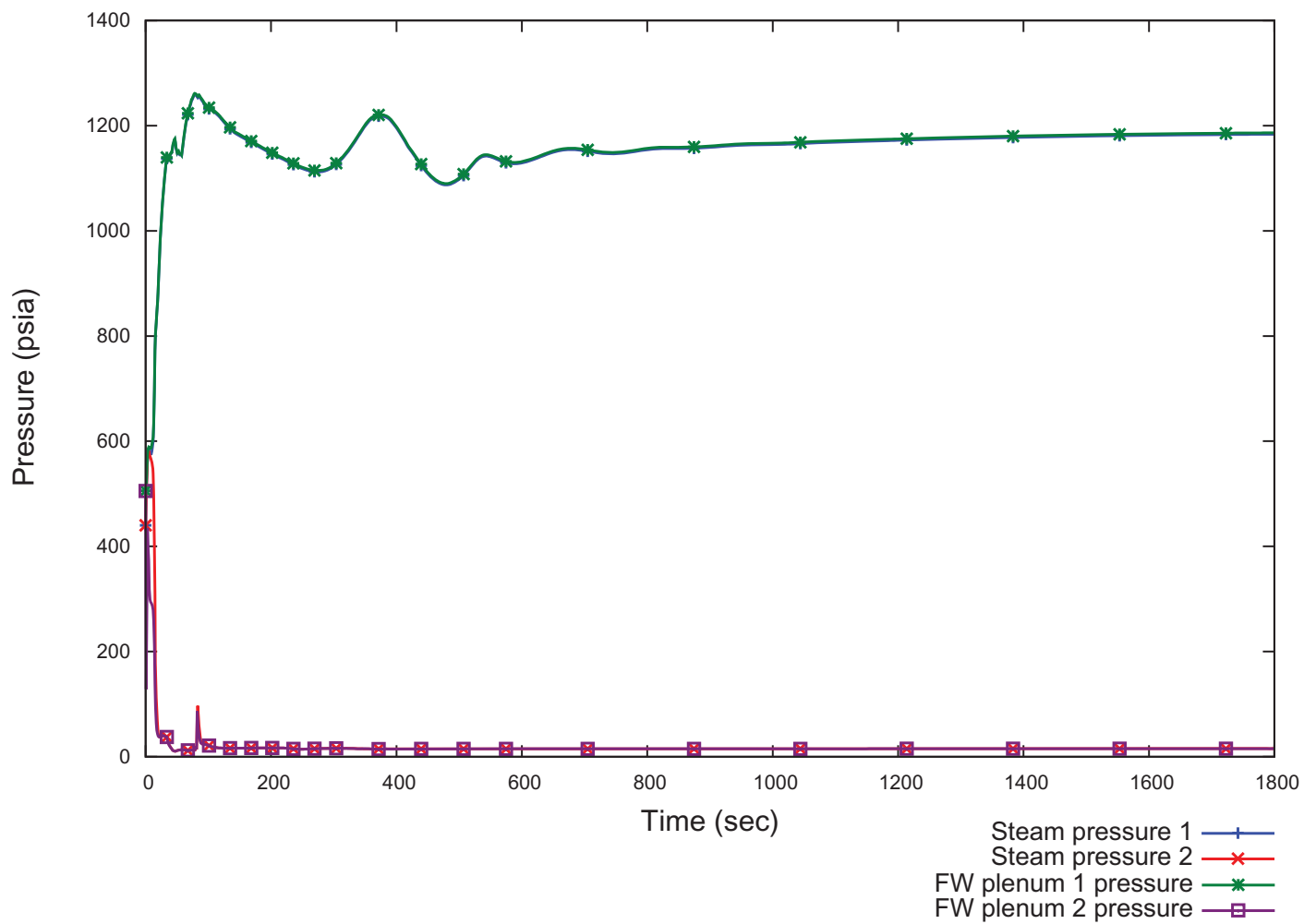


Figure 15.2-38: Reactor Coolant System Temperatures - Peak Reactor Coolant System Pressure Case (15.2.8 Feedwater Line Break)

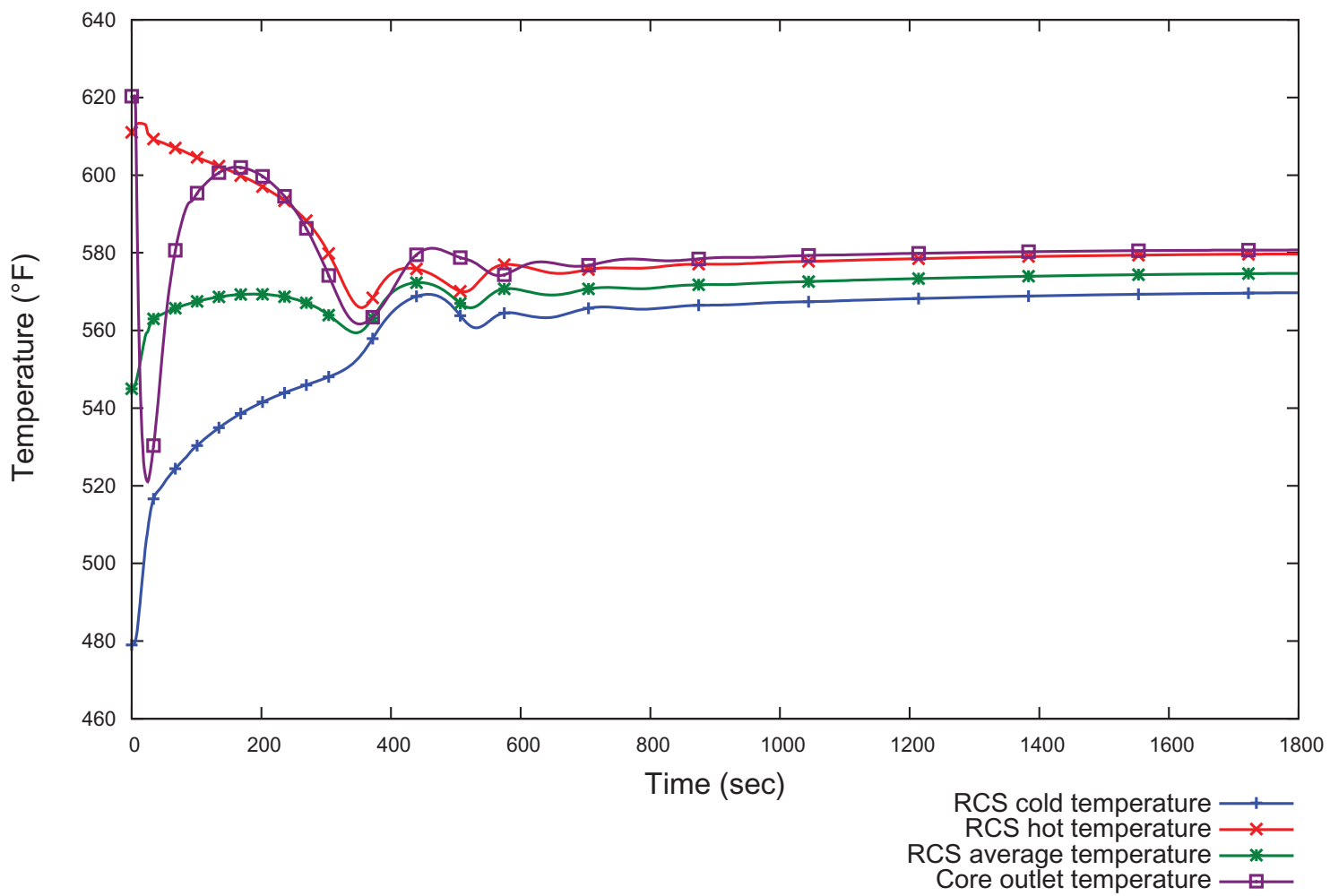


**Figure 15.2-39: Reactor Coolant System Flow Rate - Peak Reactor Coolant System Pressure Case
(15.2.8 Feedwater Line Break)**

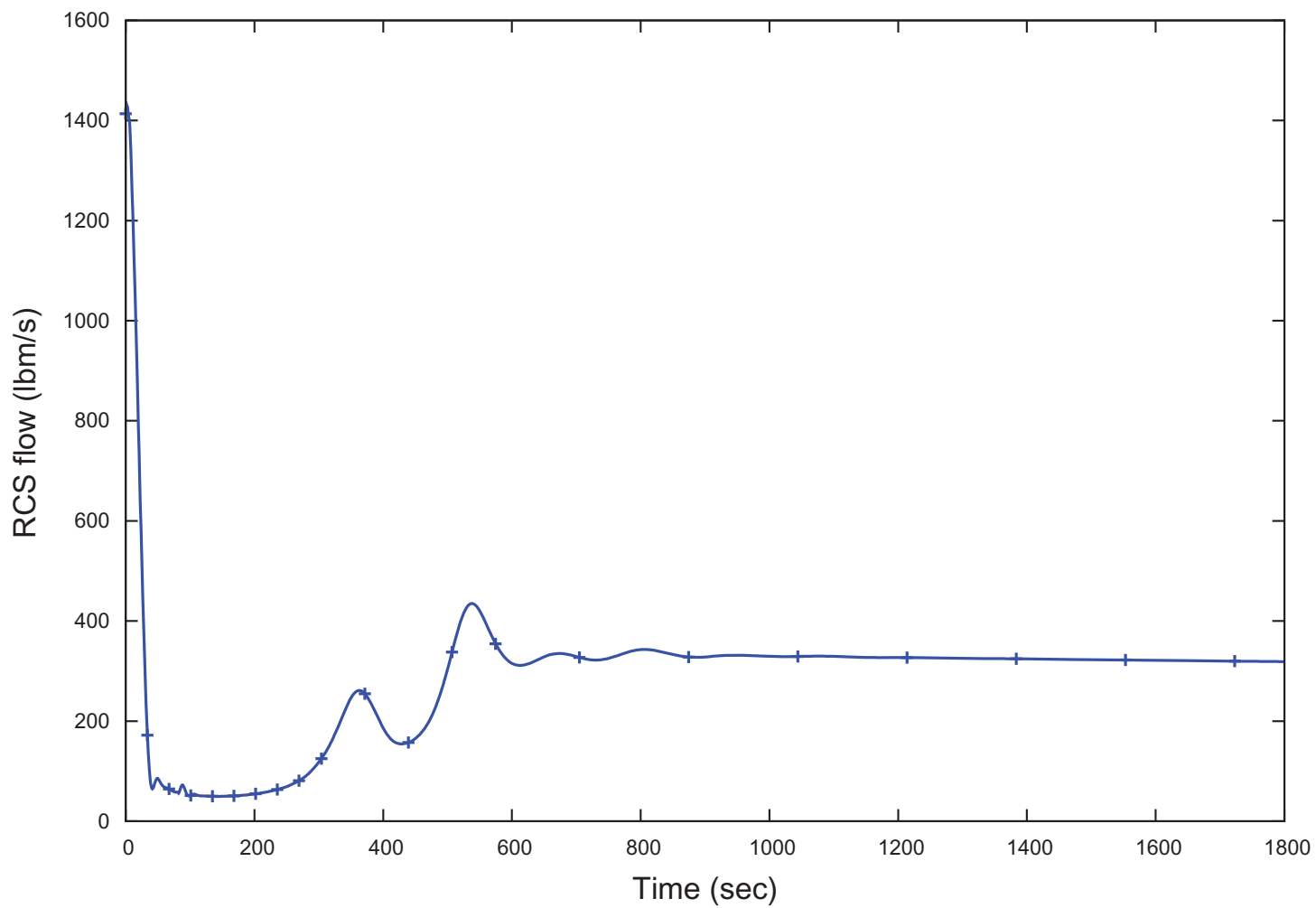


Figure 15.2-40: Integrated Break Flow - Peak Reactor Coolant System Pressure Case (15.2.8 Feedwater Line Break)

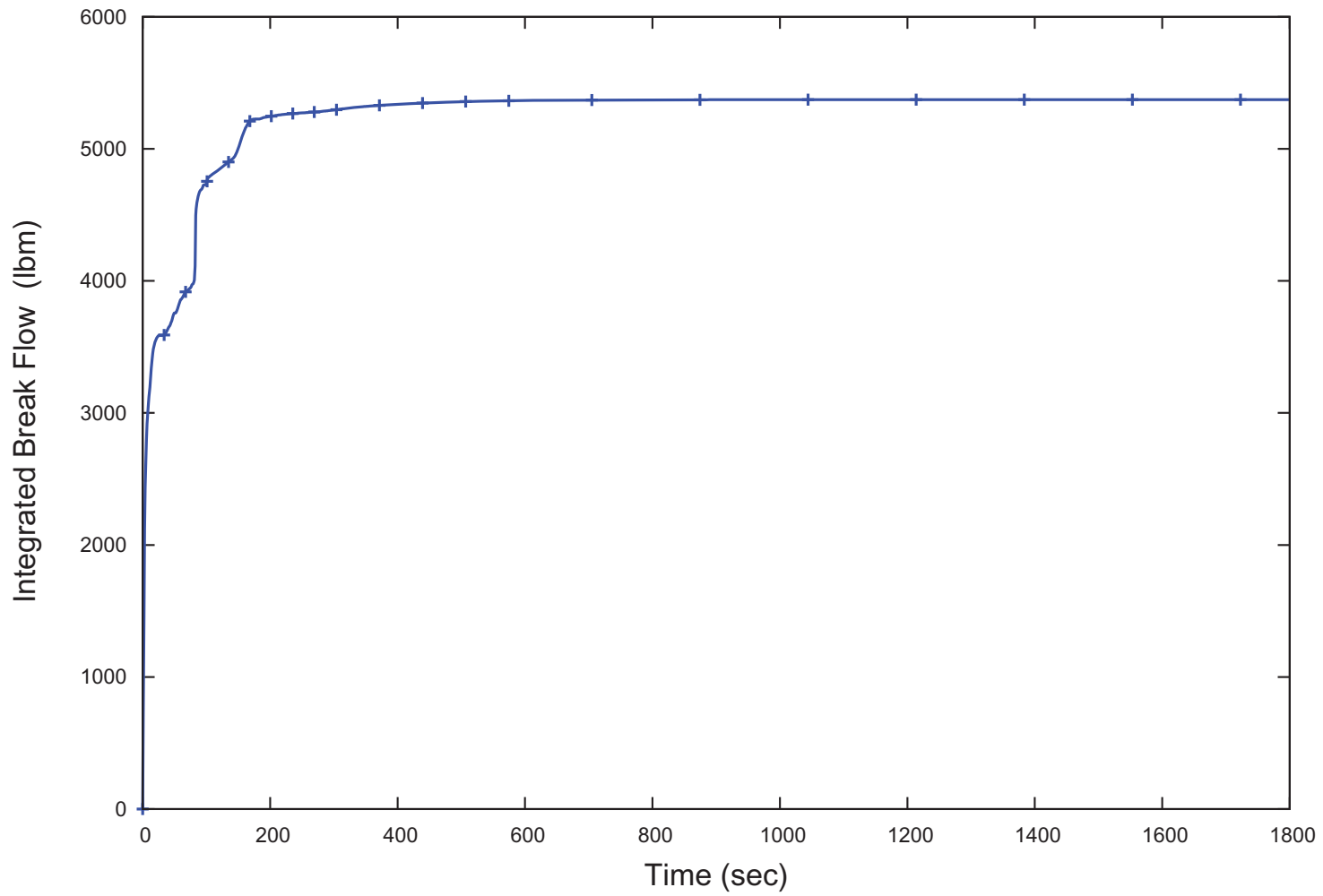


Figure 15.2-41: Secondary Pressure - Peak Steam Generator Pressure Case (15.2.8 Feedwater Line Break)

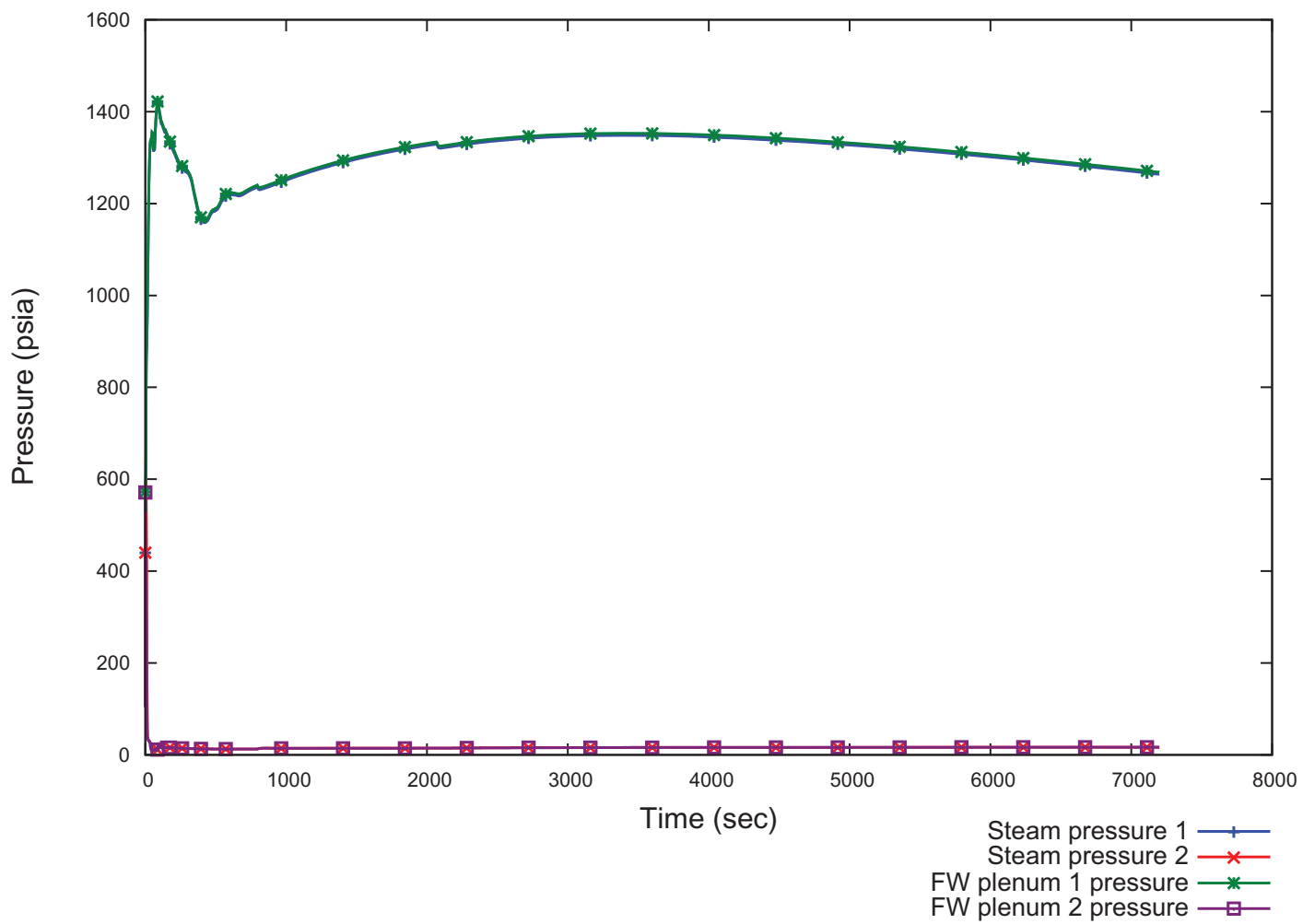


Figure 15.2-42: Minimum Critical Heat Flux Ratio Limiting Minimum Critical Heat Flux Ratio Case
(15.2.8 Feedwater Line Break)

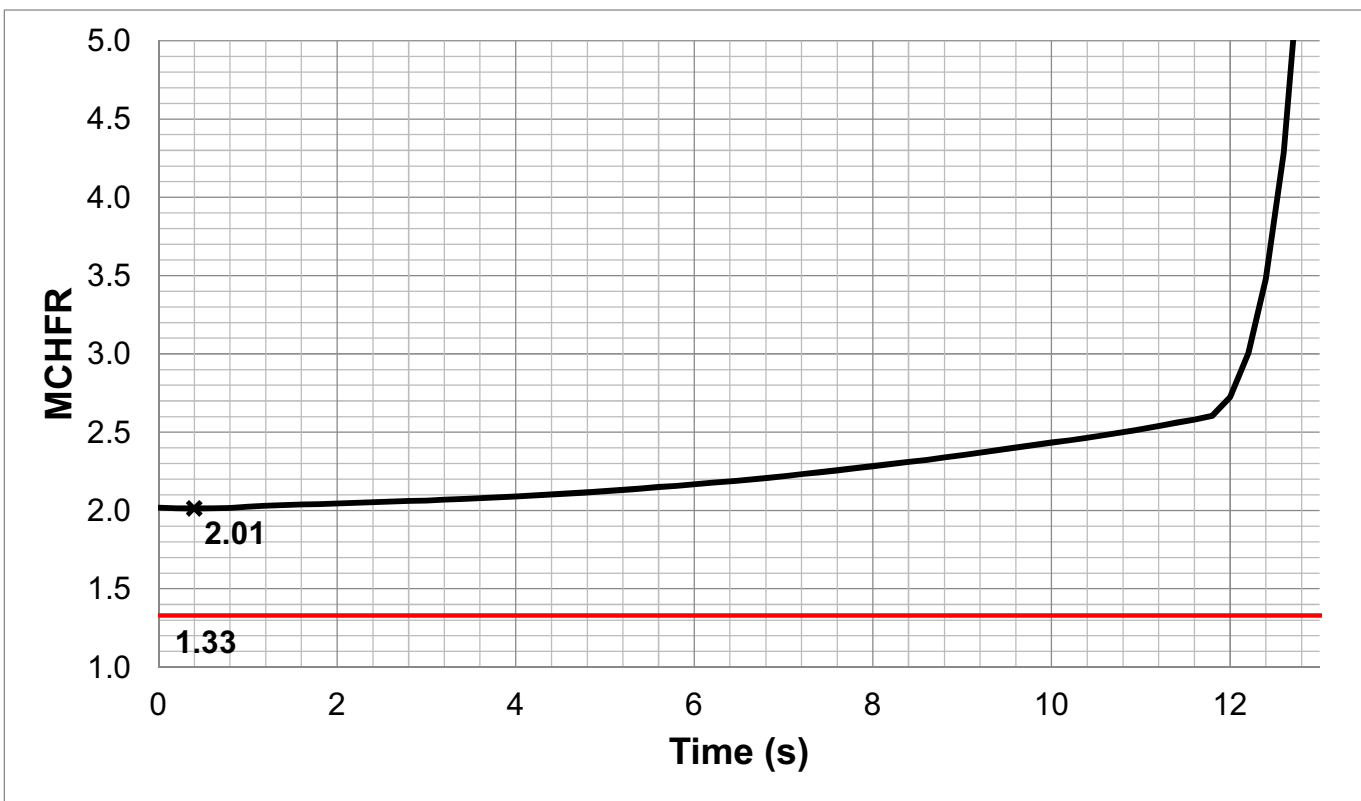


Figure 15.2-43: Decay Heat Removal System Heat Removal - Limiting Decay Heat Removal System Case (15.2.8 Feedwater Line Break)

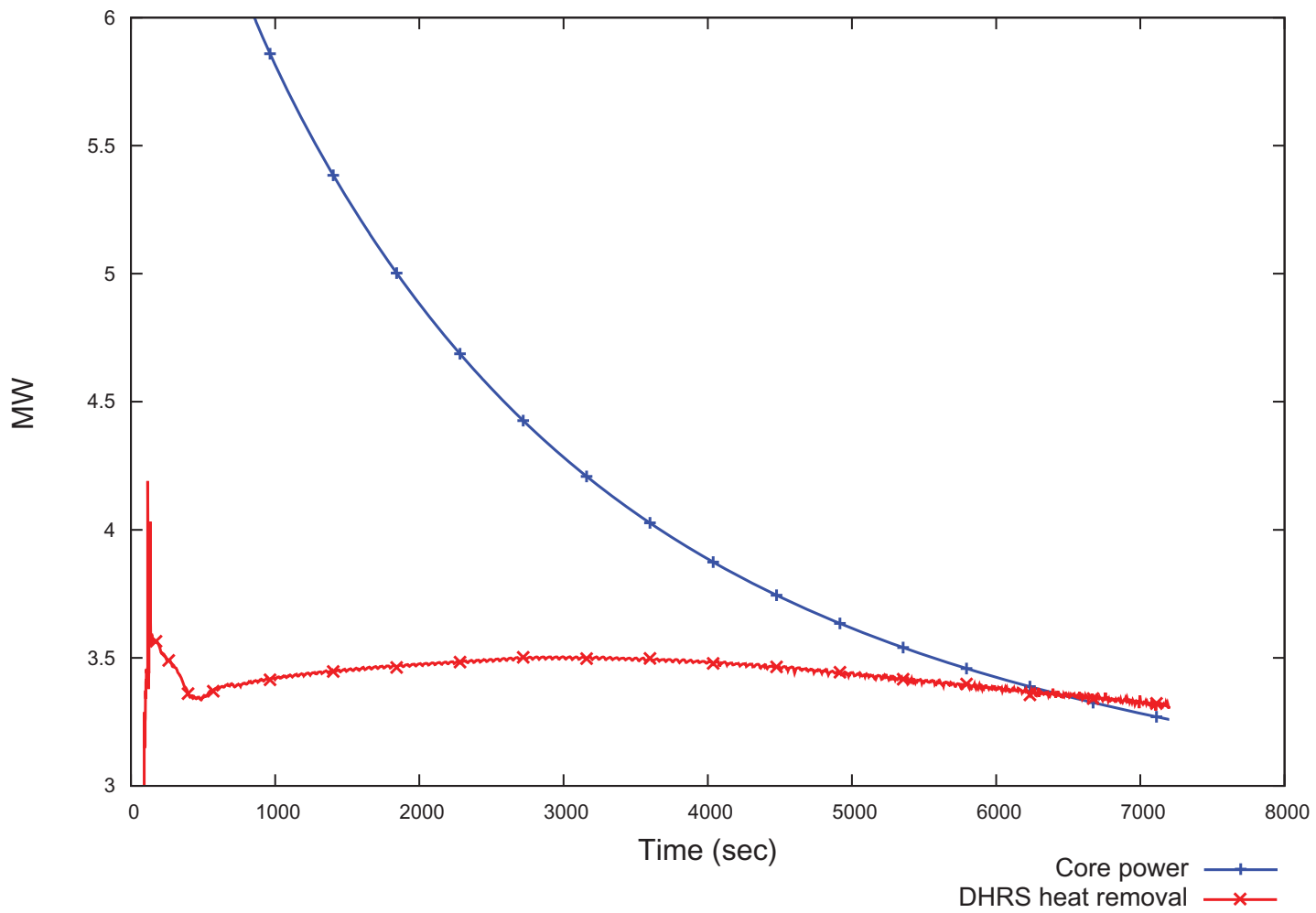


Figure 15.2-44: Secondary Pressure - Limiting Decay Heat Removal System Case (15.2.8 Feedwater Line Break)

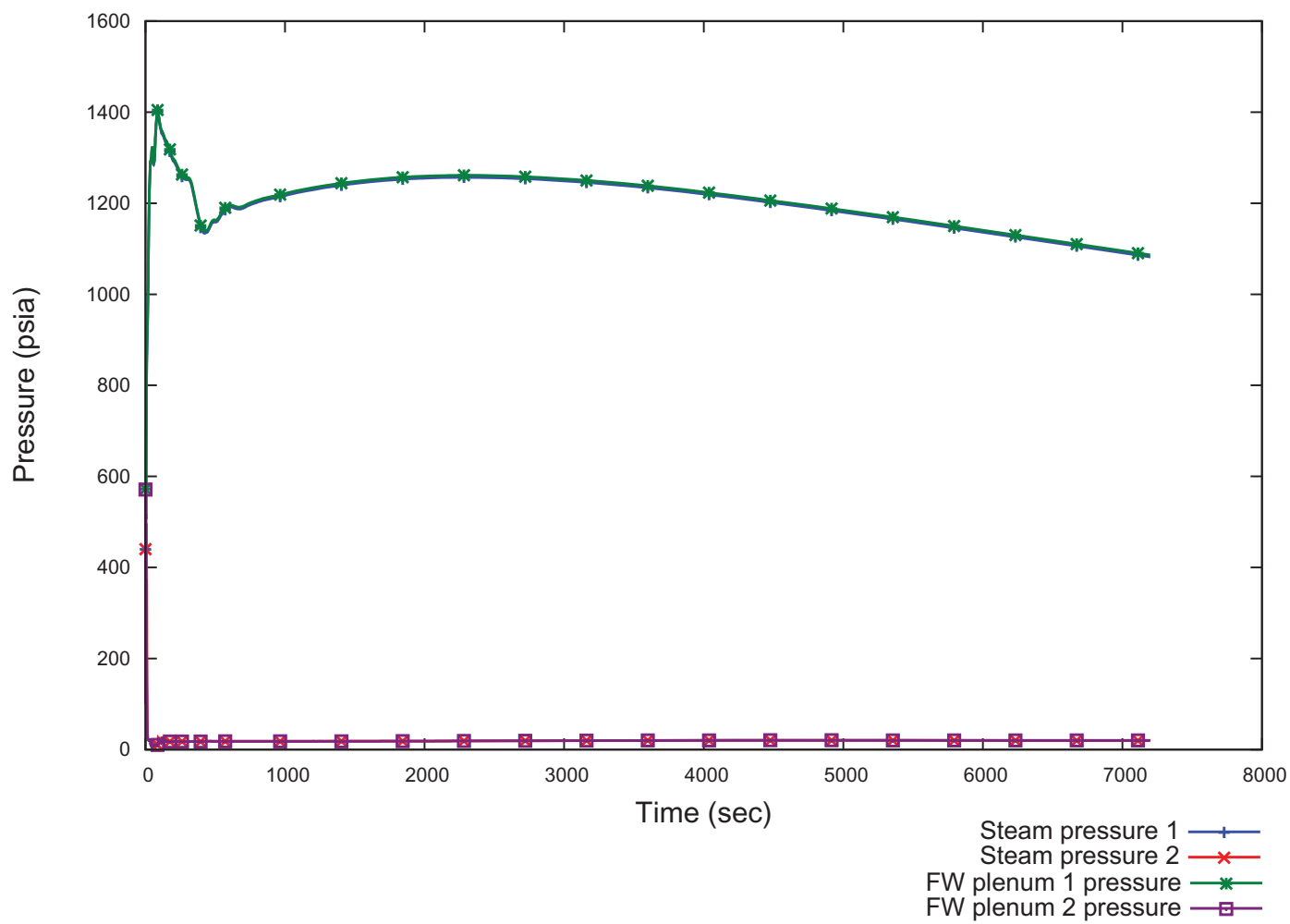


Figure 15.2-45: Reactor Coolant System Flow Rate - Limiting Decay Heat Removal System Case (15.2.8 Feedwater Line Break)

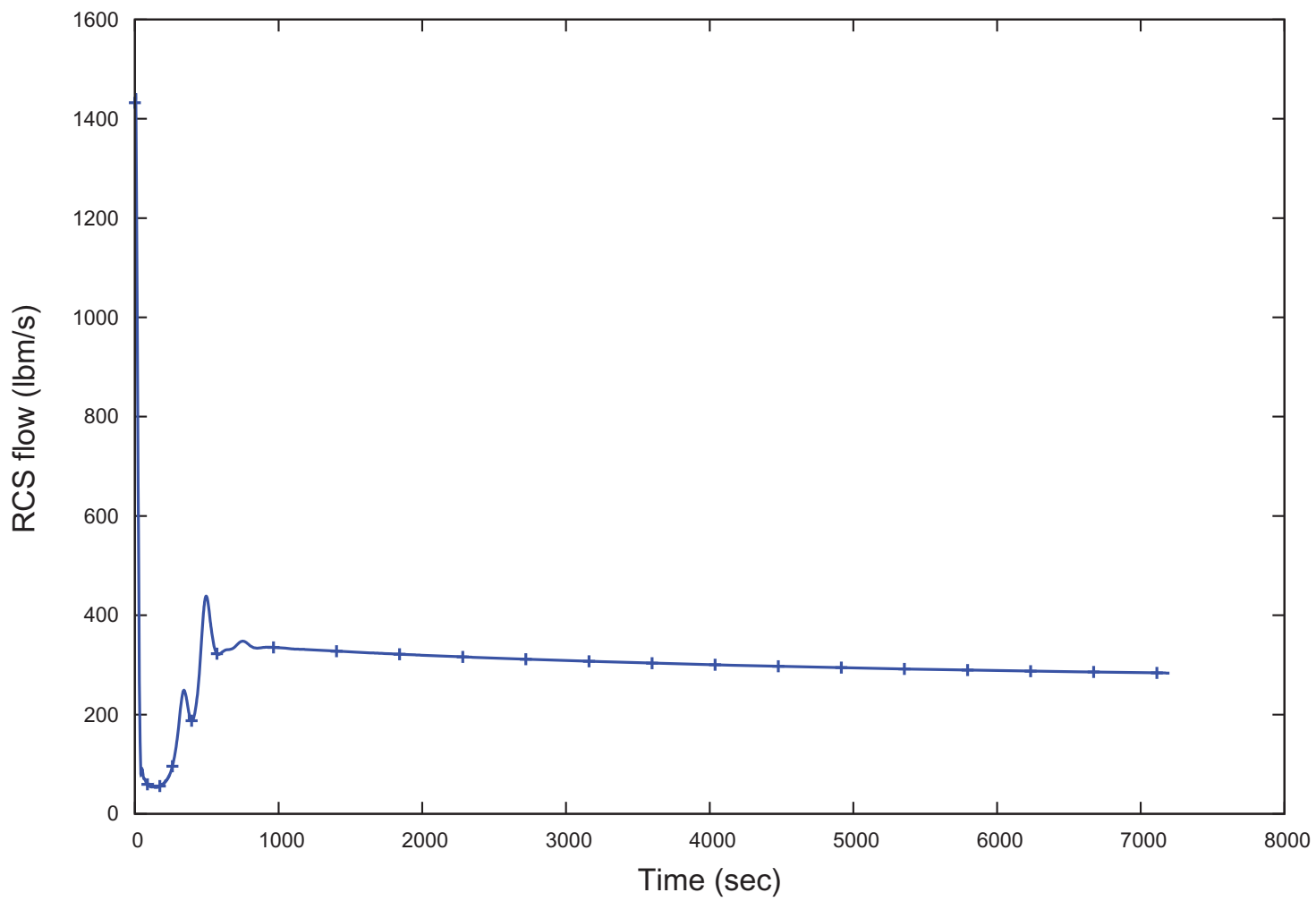
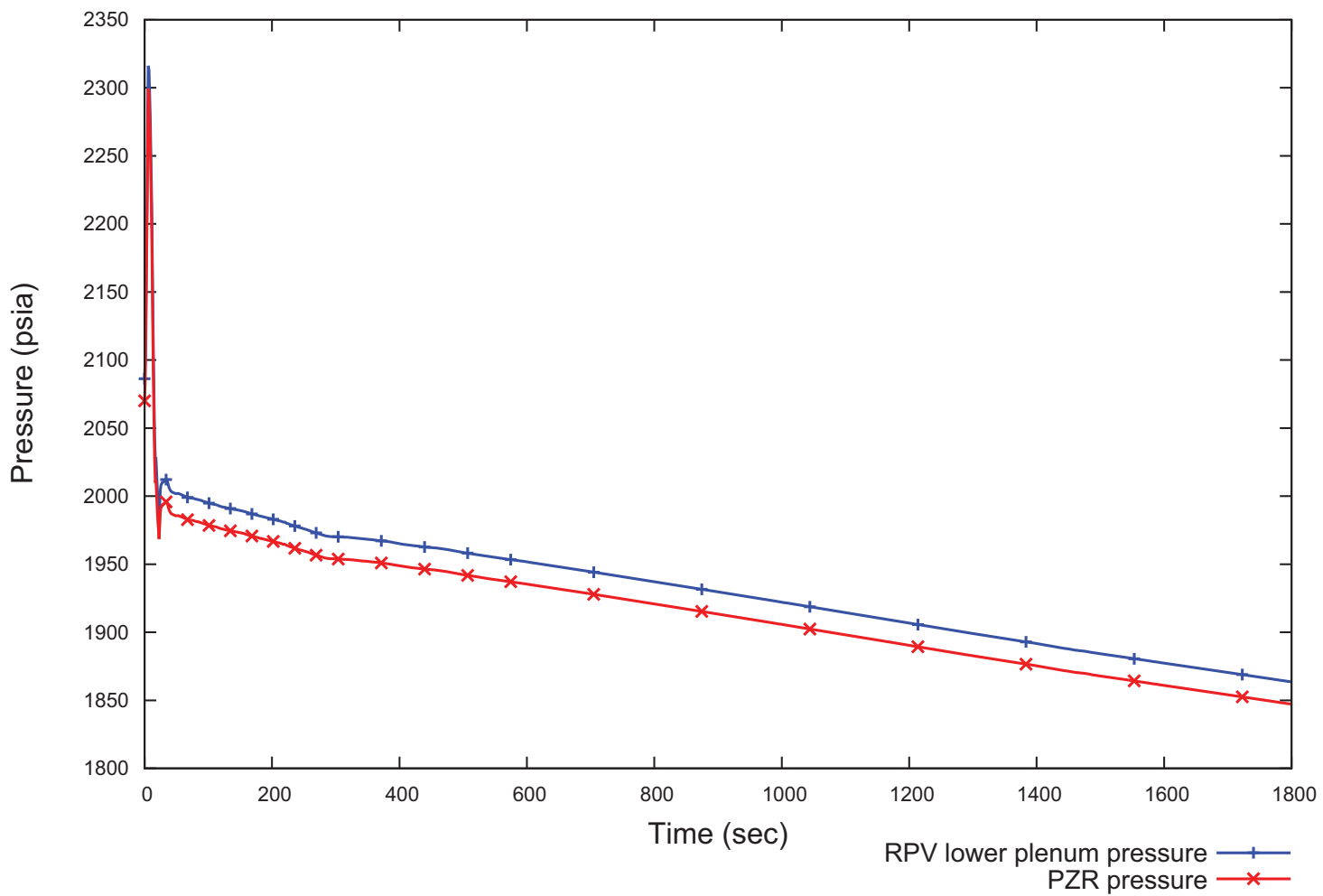


Figure 15.2-46: Primary Pressures - Peak Reactor Coolant System Pressure Case (15.2.9 Inadvertent Operation of Decay Heat Removal System)



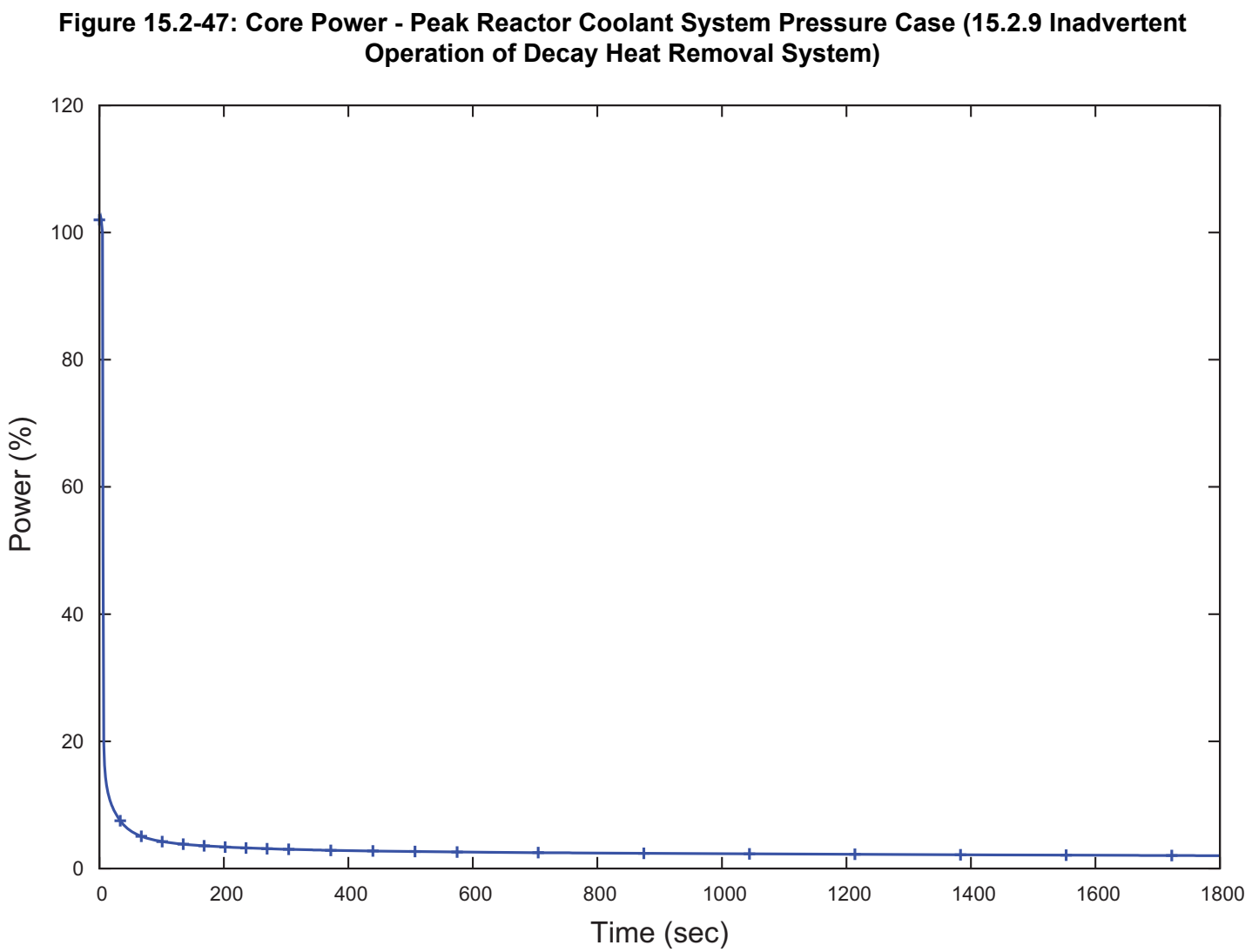
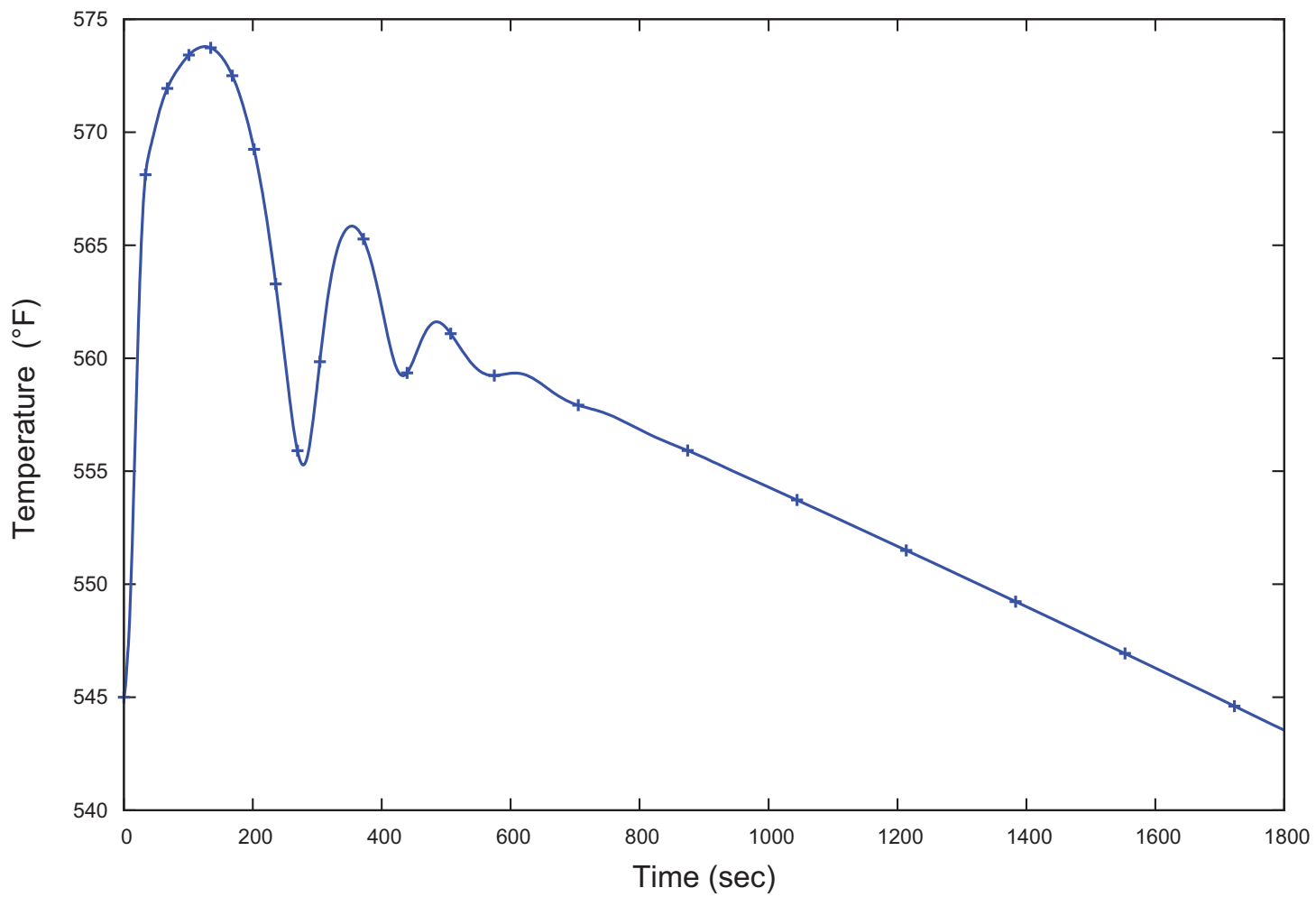


Figure 15.2-48: Reactor Coolant System Average Temperature - Peak Reactor Coolant System Pressure Case (15.2.9 Inadvertent Operation of Decay Heat Removal System)



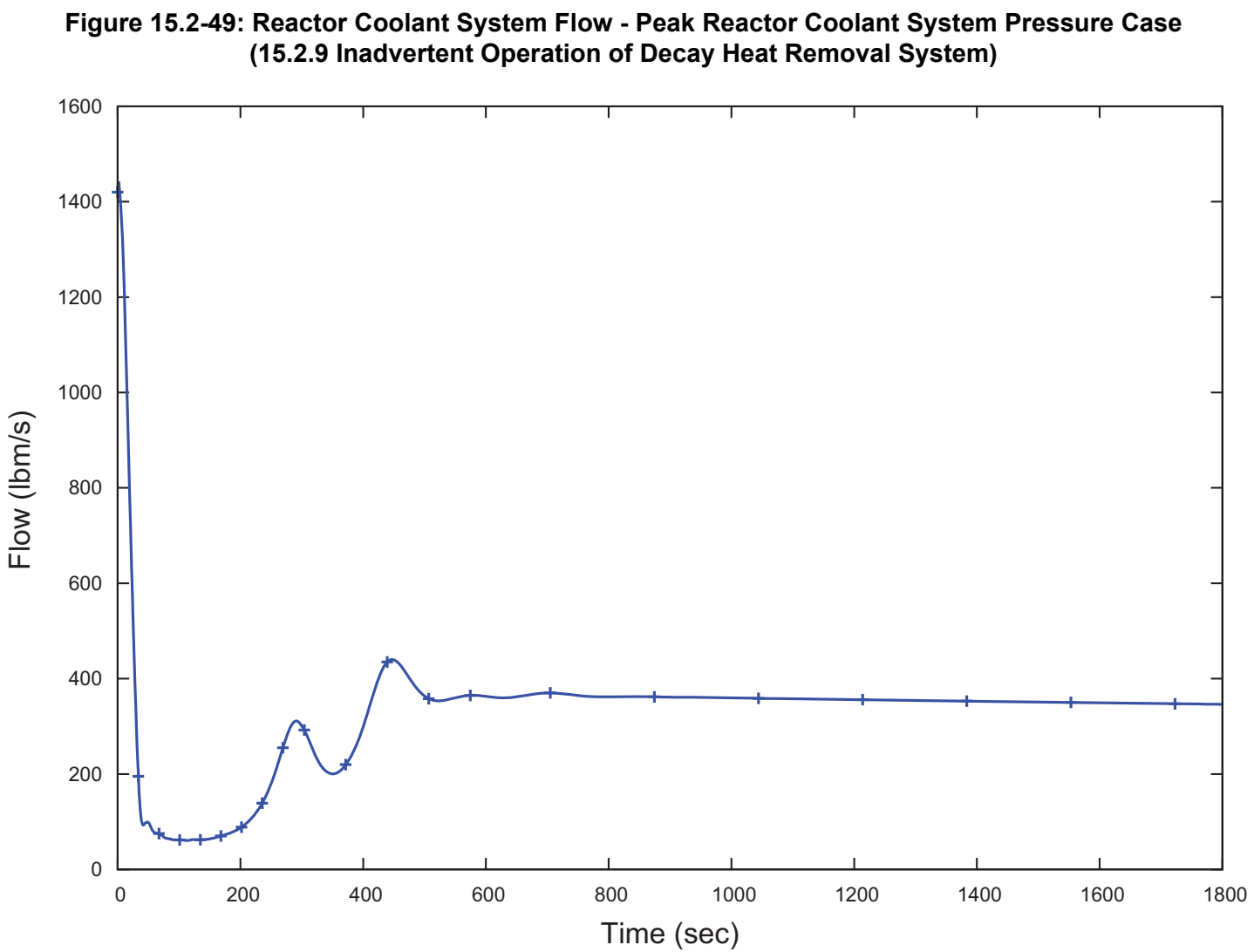


Figure 15.2-50: Total Reactivity - Peak Reactor Coolant System Pressure Case (15.2.9 Inadvertent Operation of Decay Heat Removal System)

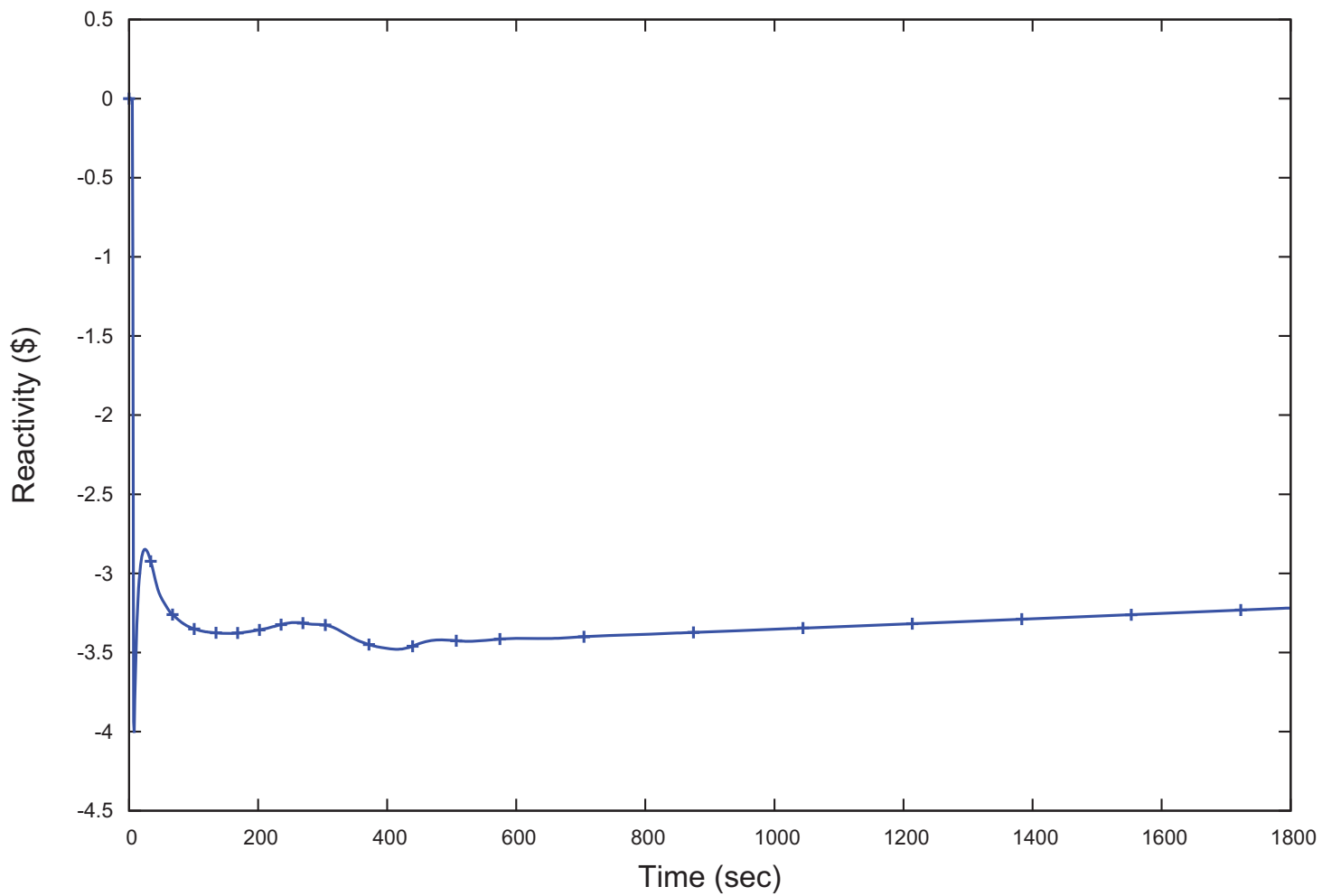


Figure 15.2-51: Secondary Pressures - Peak Reactor Coolant System Pressure Case (15.2.9 Inadvertent Operation of Decay Heat Removal System)

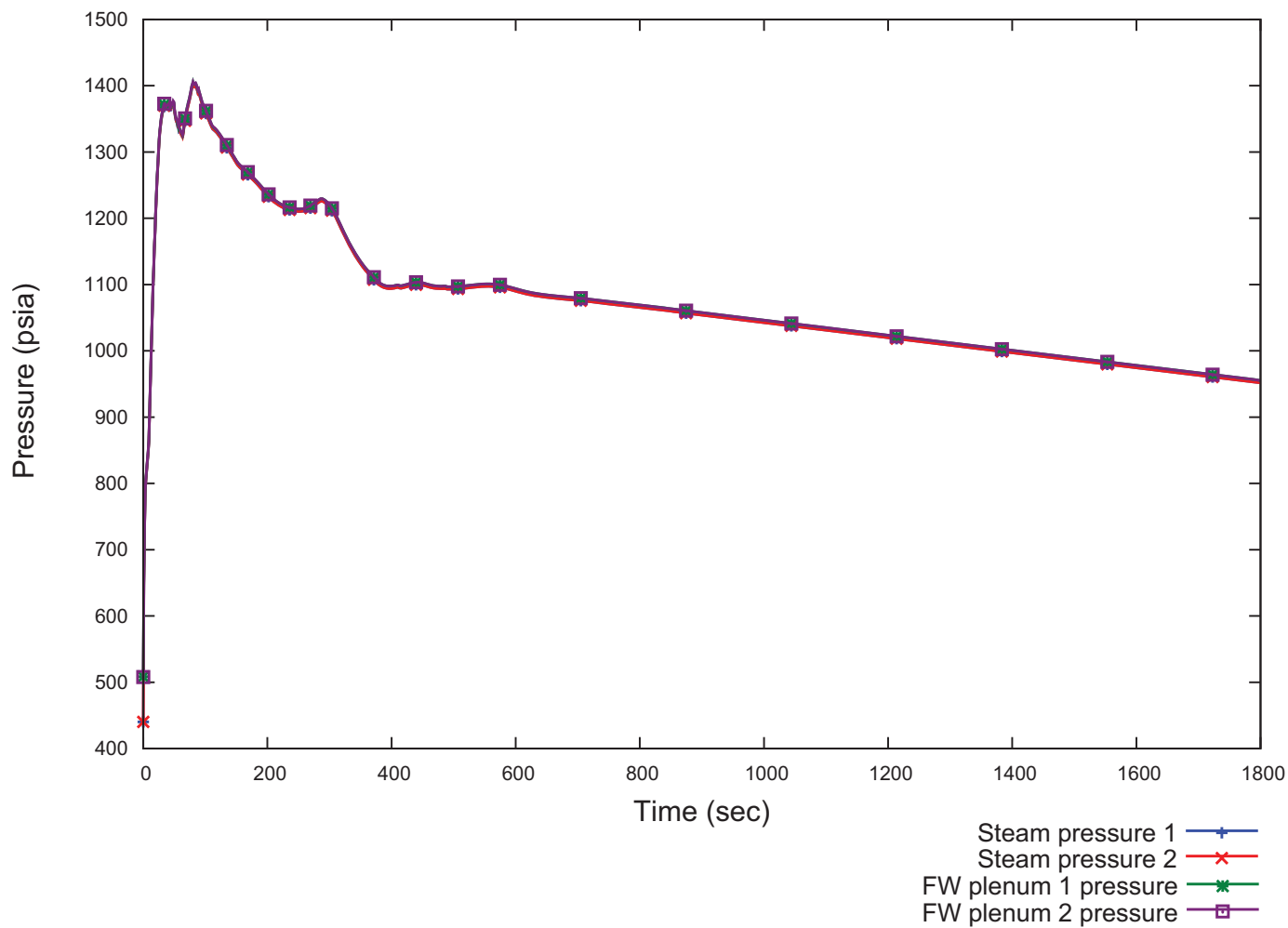


Figure 15.2-52: Pressurizer Level - Peak Reactor Coolant System Pressure Case (15.2.9 Inadvertent Operation of Decay Heat Removal System)

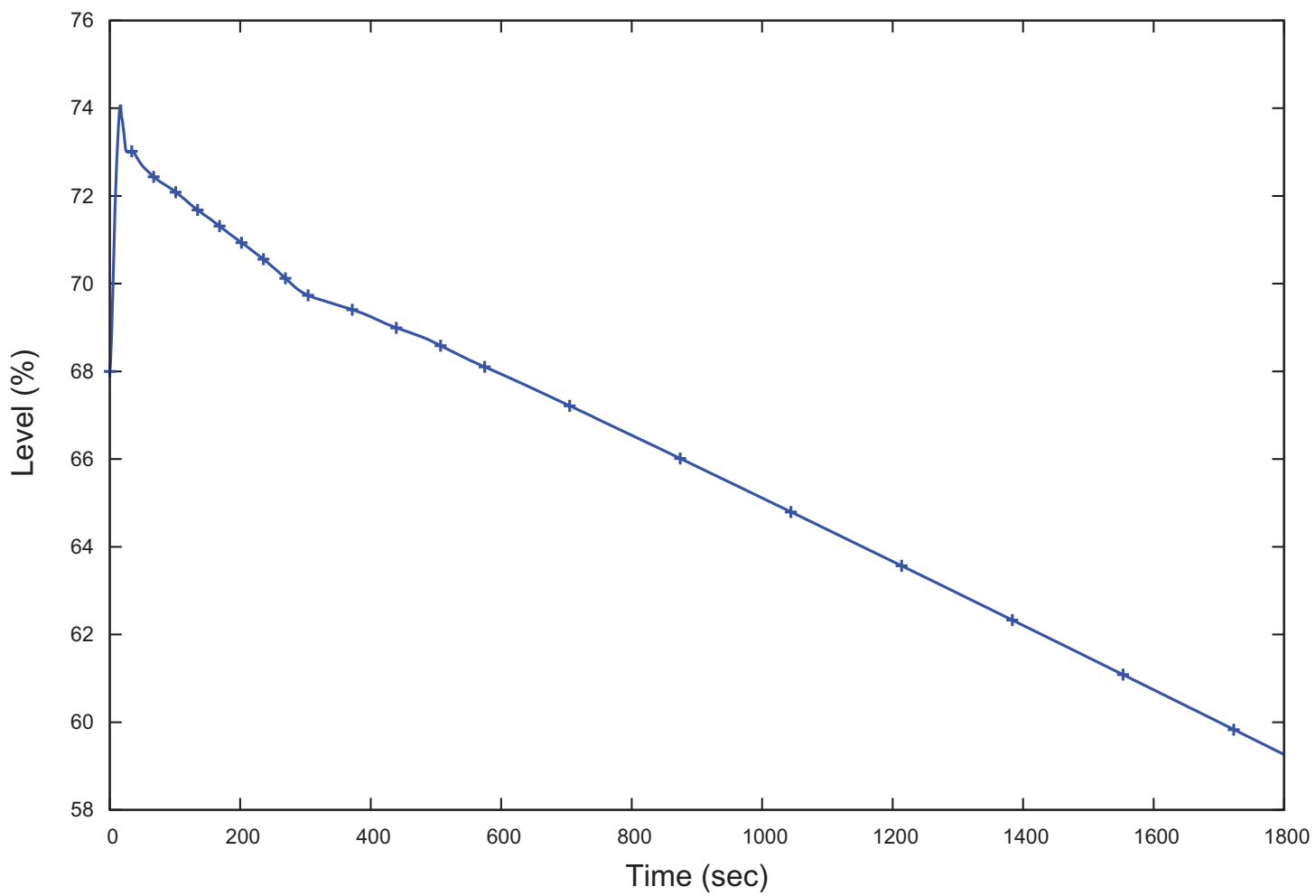


**Figure 15.2-53: Decay Heat Removal System Inlet Flows - Peak Reactor Coolant System Pressure Case
(15.2.9 Inadvertent Operation of Decay Heat Removal System)**

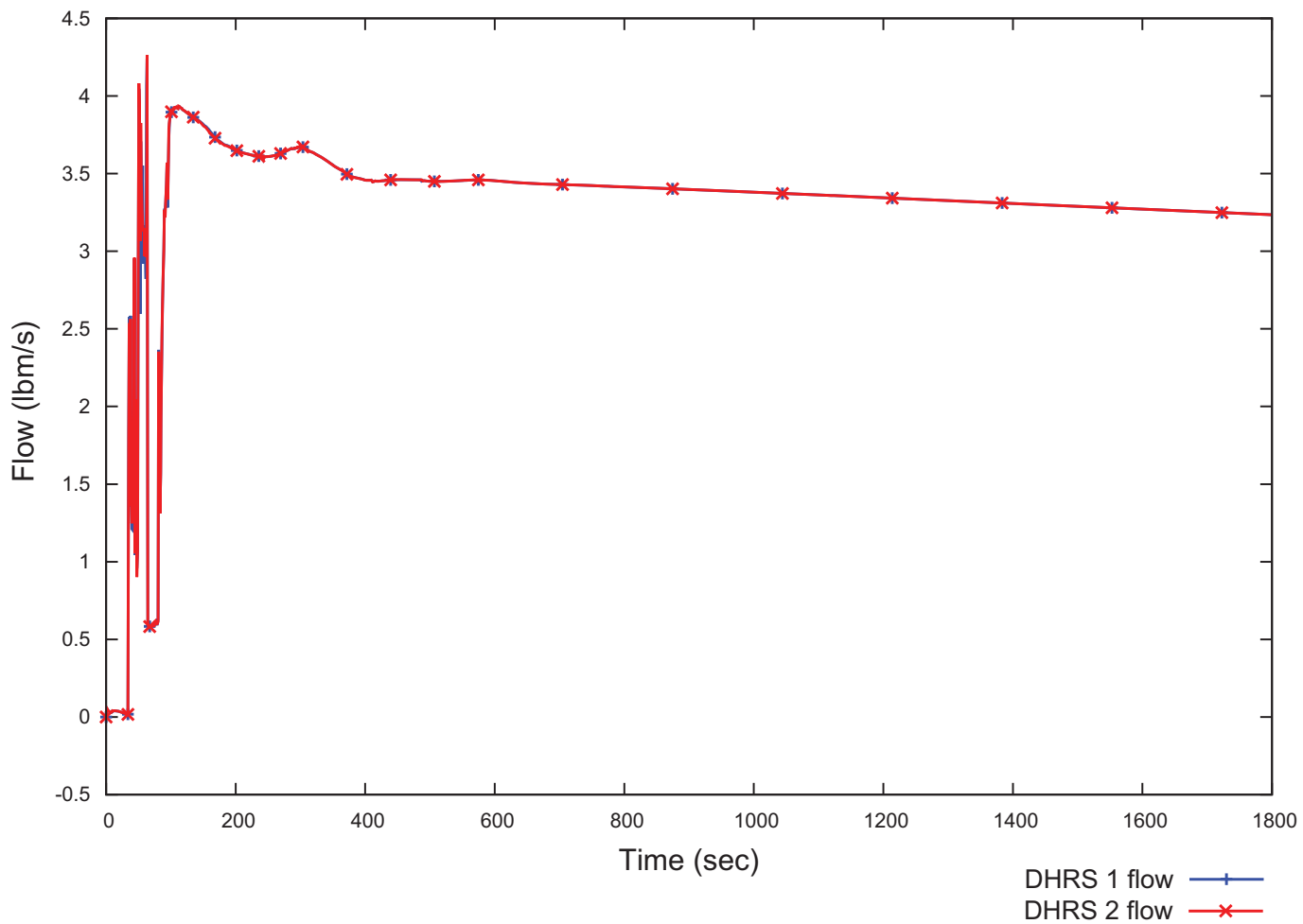
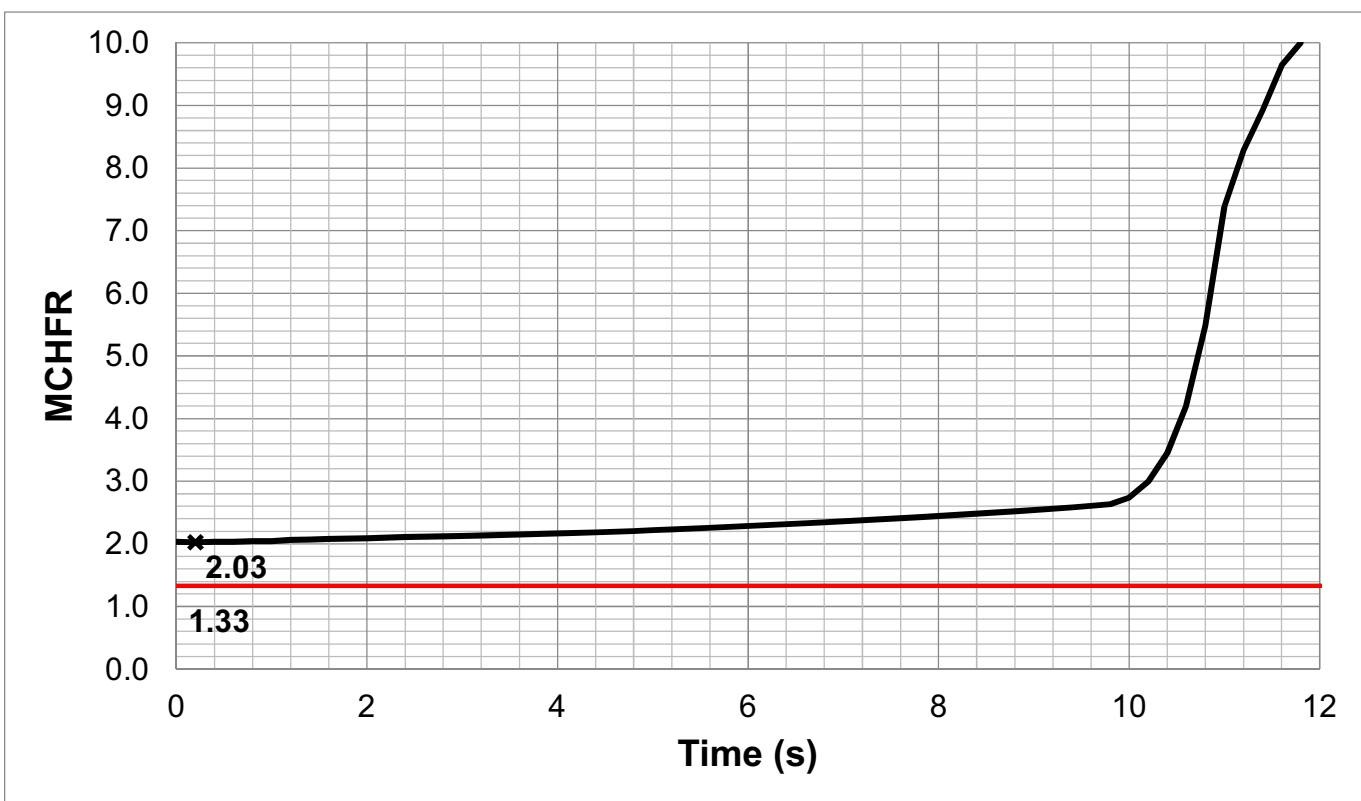


Figure 15.2-54: Minimum Critical Heat Flux Ratio - Limiting Minimum Critical Heat Flux Ratio Case
(15.2.9 Inadvertent Operation of Decay Heat Removal System)



15.3 Decrease in Reactor Coolant System Flow Rate

These events do not apply to the NuScale Power Plant US460 standard design, and no new or unique NuScale Power Module events are identified for this event type.

These events are not applicable because there are no forced flow mechanisms such as reactor coolant pumps.

15.4 Reactivity and Power Distribution Anomalies**15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition****15.4.1.1 Identification of Causes and Accident Description**

An uncontrolled control rod assembly (CRA) withdrawal from a subcritical or low power startup condition event could result in a rapid insertion of reactivity into the reactor core. There is an increase in reactor power due to the unexpected addition of reactivity as the CRA bank is withdrawn from the core. The core power increases at a faster rate than heat can be removed, resulting in an increase in reactor coolant system (RCS) temperature and a decrease in minimum critical heat flux ratio (MCHFR).

An uncontrolled CRA withdrawal from a subcritical or low power startup condition is classified as an anticipated operational occurrence (AOO) as indicated in Table 15.0-1.

15.4.1.2 Sequence of Events and Systems Operation

The sequence of events for an uncontrolled CRA withdrawal from a subcritical or low power startup condition is provided in Table 15.4-1 for the limiting MCHFR and linear heat generation rate (LHGR) case. The RCS pressure and secondary pressure are not acceptance criteria for this event. The RCS and secondary pressure are bounded by other AOO events.

Unless specified below, the analysis of an uncontrolled CRA withdrawal from a subcritical or low power startup condition event assumes the control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of the CRA withdrawal.

The rod control function of the control rod drive system (CRDS) provides reactivity control to compensate for rapid, short-term variations in the reactivity of the core. The rod control function is also used to maintain the measured RCS temperature at or near the programmed average coolant temperature. The CRDS rod control operational modes include manual mode, automatic mode, and insertion-only automatic mode. The CRDS rod control function could be in manual mode during startup conditions. An operator error or malfunction in the CRDS would have to occur to initiate a uncontrolled CRA withdrawal from a subcritical or low power startup condition.

The expected normal travel rate of the CRAs is 6 in/min. However, the maximum allowed withdrawal rate of a CRA is 15 in/min, with a step size no greater than three-eighths inch. A spectrum of constant reactivity insertion rates that bounds this maximum rate and also bounds possible boron dilution scenarios is included in the analysis.

The module protection system (MPS) is credited to protect the NuScale Power Module (NPM) in the event of an uncontrolled CRA withdrawal from a subcritical or low power startup condition. The following MPS signals protect the NPM in the event of an uncontrolled CRA withdrawal from a subcritical or low power startup condition:

- high power (at 25 percent of full power for startup conditions)
- source range (SR) and intermediate range (IR) log power rate
- high SR count rate

The RCS pressure control with heaters and spray is assumed to function normally to delay the trip on high pressurizer pressure. Pressurizer level control is disabled.

No single failure could occur during an uncontrolled CRA withdrawal from a subcritical or low power startup condition event that would result in more severe conditions for the limiting case.

15.4.1.3 Thermal Hydraulic and Subchannel Analyses

15.4.1.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to an uncontrolled CRA withdrawal from a subcritical or low power startup condition is performed using NRELAP5. The NRELAP5 model is based on the design features of the NPM. The non-loss-of-coolant accident (non-LOCA) NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel critical heat flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative MCHFR result, in accordance with the methodology described in the Subchannel Analysis Methodology Topical Report (Reference 15.4-1) and the Statistical Subchannel Analysis Methodology Topical Report (Reference 15.4-2). Section 15.0.2 includes a discussion of the VIPRE-01 code and evaluation model.

15.4.1.3.2 Input Parameters and Initial Conditions

A spectrum of initial conditions is analyzed to find the limiting reactivity insertion due to an uncontrolled CRA withdrawal from a subcritical or low power startup condition. The initial conditions of the transient evaluation result in a conservative calculation. Table 15.4-2 provides key inputs and associated biases for the limiting MCHFR and LHGR case. The following initial conditions and assumptions ensure the results have sufficient conservatism.

- The minimum initial power assumed for this analysis is 1 watt. The transient analyses for this event evaluate cases with initial power ranging

from this minimum power of 1 watt to 15 percent of full power (consistent with use of the low setting for the high power analytical limit). The Standard Review Plan (SRP) guidance states that minimizing initial power provides the most conservative conditions because it provides the maximum power peak. However, this is not the case for the NuScale Power Plant US460 standard design. The SR log power rate trip signal prevents a significant power increase at lower powers.

- The least negative/most positive reactivity feedback coefficients are used to minimize the reactivity feedback.
- No maximum bank worth is credited to limit the total reactivity insertion.
- The positive reactivity inserted by the CRA withdrawal is modeled as a constant reactivity addition beginning at the transient initiation. To bound the reactivity insertion at the maximum rod speed and the reactivity insertion from possible boron dilution scenarios, a range of reactivity insertion rates from 0.0001 \$/sec to 0.0500 \$/sec is analyzed.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and using a bounding control rod drop rate.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with Regulatory Guide (RG) 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR and LHGR for this event. The subchannel model is discussed in Section 15.0.2.

15.4.1.3.3 Results

The sequence of events for a representative uncontrolled CRA withdrawal from a low power or startup condition is provided in Table 15.4-1 for the limiting MCHFR and LHGR case. Figure 15.4-1 through Figure 15.4-4 show the transient behavior of key parameters for the case that is limiting with respect to MCHFR and LHGR.

The CRA bank begins to withdraw at the transient initiation, which begins to raise power, RCS temperature, and RCS pressure. The case that is limiting for MCHFR and LHGR has an initial power and reactivity insertion rate that initially avoids the high power rate trip and allows a heatup and pressurization before reactor trip. The reactor power rises until the high power (25 percent) limit is reached and MPS initiates a reactor trip after the delay. The peak power, maximum RCS pressure, and MCHFR occur near the time of reactor trip.

The limiting cases for an uncontrolled CRA withdrawal from a low power or startup condition demonstrate margin to the acceptance criteria. The limiting MCHFR for this event is above the CHF analysis limit and the limiting LHGR is

below the design limit. The limiting values for these acceptance criteria are shown in Table 15.4-3.

15.4.1.4 Radiological Consequences

The leakage-related radiological consequences of this event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.4.1.5 Conclusions

The acceptance criteria for this AOO are met for the limiting cases.

- The thermal margin limits are met.
 - The MCHFR for the limiting case is above the CHF analysis limit, as shown in Table 15.4-3.
- Fuel centerline temperatures do not exceed the melting point.
 - As discussed in Reference 15.4-1, a steady-state LHGR protection limit can be applied to an uncontrolled CRA withdrawal from a subcritical or low power startup condition event to ensure the fuel centerline temperatures do not exceed the melting point. The LHGR for the limiting case is below the limit, as shown in Table 15.4-3.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.2.1 Identification of Causes and Accident Description

A spurious CRA withdrawal that occurs when the reactor is at power leads to an addition of positive reactivity into the reactor, an increase in core power, and a corresponding increase in heat flux. Because of the time lag in the response of the secondary system, the heat removal from the steam generators follows the heat increase in the primary system. The result is an increase in RCS temperature and pressure. These conditions could challenge design pressures and the specified acceptable fuel design limits (SAFDLs). The power range neutron excor detectors provide high power and high power rate core protection. For cases where the reactivity insertion is sufficiently slow, the high pressurizer pressure, high RCS hot temperature, and high RCS average temperature limits provide protection. These MPS limits are analyzed for a spectrum of uncontrolled CRA withdrawal conditions to ensure protection functions are actuated to prevent the violation of the design safety limits.

An uncontrolled CRA withdrawal is classified as an AOO as indicated in Table 15.0-1.

15.4.2.2 Sequence of Events and Systems Operation

The sequence of events for a representative uncontrolled CRA withdrawal at power is provided in Table 15.4-4 for the limiting MCHFR case.

Unless specified below, the analysis of an uncontrolled CRA withdrawal event assumes the control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an uncontrolled CRA withdrawal event.

The regulating CRA banks contain the only CRAs that are not fully withdrawn during power operation. The power dependent insertion limit (PDIL), plus uncertainty, restricts the amount of insertion steps the regulating banks can achieve during power operation. The PDIL-imposed restrictions are not modeled in the uncontrolled CRA withdrawal analysis in order to bound possible boron dilution scenarios. The expected normal travel rate of the CRAs is 6 in/min. However, the maximum allowed withdrawal rate of a CRA is 15 in/min, with a step size no greater than three-eighths inch. This withdrawal rate corresponds to the maximum possible reactivity insertion rate. A spectrum of constant reactivity insertion rates that bounds this maximum rate and also bounds possible boron dilution scenarios is included in the uncontrolled CRA withdrawal analysis.

The effect of a reactivity insertion event on the RCS is an increase in temperature, which decreases density and causes flow into the pressurizer, increasing RCS pressure. As a result, the normal module control system response would be to decrease pressurizer heater power and increase pressurizer spray. In some uncontrolled CRA withdrawal cases, the pressurizer pressure control with heaters and spray is assumed to function normally to delay the trip on high pressurizer pressure; in other cases pressurizer spray is disabled.

The MPS is credited to protect the NPM in the event of an uncontrolled CRA withdrawal. The following MPS signals protect the NPM during an uncontrolled CRA withdrawal:

- high power
- high power rate
- high RCS hot temperature
- high pressurizer pressure
- high RCS average temperature

In uncontrolled CRA withdrawal events that result in a reactor trip, the subsequent actuation of the decay heat removal system (DHRS) is credited with maintaining reactor cooling. The MPS signal credited for DHRS actuation in the limiting MCHFR case is high main steam pressure.

No single failure could occur during an uncontrolled CRA withdrawal event that results in a more severe outcome for the limiting uncontrolled CRA withdrawal cases.

15.4.2.3 Thermal Hydraulic and Subchannel Analyses

15.4.2.3.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to an uncontrolled CRA withdrawal is performed using NRELAP5. The NRELAP5 model is based on the design features of the NPM. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative MCHFR result, in accordance with the methodology described in Reference 15.4-1 and Reference 15.4-2. Section 15.0.2 includes a discussion of the VIPRE-01 code and evaluation model.

15.4.2.3.2 Input Parameters and Initial Conditions

A spectrum of initial conditions is analyzed to find the limiting reactivity insertion due to an uncontrolled CRA withdrawal. Key inputs of the uncontrolled CRA withdrawal evaluation are provided in Table 15.4-5 for the limiting MCHFR case and Table 15.4-6 for the limiting LHGR case. The following initial conditions and assumptions ensure the results have sufficient conservatism.

- Initial power level: 15 percent, 50 percent, 75 percent, and 102 percent of nominal power are analyzed in the uncontrolled CRA withdrawal evaluation. The power level for the limiting MCHFR and LHGR cases is 102 percent of nominal power.
- Reactivity insertion rate: The positive reactivity inserted by CRA withdrawal is modeled as a constant reactivity addition beginning at the transient initiation. To bound the reactivity insertion at the maximum rod speed and the reactivity insertion from possible boron dilution scenarios, a range of reactivity insertion rates from 0.0001 \$/sec to 0.0500 \$/sec is analyzed.
- Reactivity feedback: The least negative reactivity coefficients are implemented in the limiting uncontrolled CRA withdrawal cases. The least negative reactivity coefficients provide the least amount of feedback to mitigate the power increase due to an uncontrolled CRA withdrawal.
- The turbine bypass system is not credited in this analysis to minimize heat removal by the secondary side.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and using a bounding control rod drop rate.

- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with RG 1.105.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the MCHFR and LHGR for this event. The subchannel evaluation model is discussed in Section 15.0.2.

15.4.2.3.3 Results

The sequence of events for a limiting uncontrolled CRA withdrawal with respect to MCHFR is provided in Table 15.4-4. Figure 15.4-5 through Figure 15.4-10 show the transient behavior of key parameters for an uncontrolled CRA withdrawal.

The withdrawal of the regulating bank results in a reactivity insertion that increases reactor power. The power increase leads to a rise in RCS temperature, pressurizer level, and RCS pressure. Feedback from the rising fuel and moderator temperatures partially counteracts the reactivity insertion, slowing the power increase. For uncontrolled CRA withdrawal cases with higher reactivity insertion rates, the MPS trips the reactor on high pressurizer pressure or high power rate. These cases are non-limiting for MCHFR because the reactor is tripped before the maximum amount of reactivity can be inserted. The limiting combination of reactivity insertion and reactivity feedback produces the maximum possible power increase before trip. The power increase in the limiting MCHFR case is terminated by a reactor trip after a signal delay. The MPS trips the reactor and actuates secondary system isolation and the DHRS during this event. The most limiting MCHFR occurs near the time of the power peak. The MCHFR remains above the CHF analysis limit.

The maximum LHGR case for an uncontrolled CRA withdrawal at power occurs for the case with the highest reactivity insertion rate. The rapid increase in power causes a high power rate trip. This case results in the limiting power peak and thus a higher fuel temperature as evidenced by the LHGR. The LHGR remains below the design limit, so no fuel centerline melting is predicted for the uncontrolled CRA withdrawal.

The uncontrolled CRA withdrawal at power cases that result in a reactor trip, secondary system isolation, and DHRS actuation, demonstrate stable core cooling is maintained.

15.4.2.4 Radiological Consequences

The leakage-related radiological consequences of this event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.4.2.5 Conclusions

The acceptance criteria for this AOO are met for the limiting cases.

- 1) The thermal margin limits are met.
 - The MCHFR for the limiting case is above the CHF analysis limit, as shown in Table 15.4-7.
- 2) Fuel centerline temperatures do not exceed the melting point.
 - As discussed in Reference 15.4-1, a steady-state LHGR protection limit can be applied to an uncontrolled CRA withdrawal at power event to ensure the fuel centerline temperatures do not exceed the melting point. The LHGR for the limiting case is below the limit, as shown in Table 15.4-7.

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

15.4.3.1 Identification of Causes and Accident Description

A control rod misoperation (CRM) could introduce an unexpected change of reactivity in the core. The resulting change in power distribution could lead to a decrease in the CHF ratio. The CRM events analyzed in this section are

- A single CRA or multiple CRAs are displaced in a position relative to the bank while the other rods in the bank are in another allowed position.
- A single CRA is inadvertently withdrawn from the bank insertion limit. The withdrawal of an entire bank of CRAs is analyzed in Section 15.4.1 and Section 15.4.2.
- A single CRA or multiple CRAs are dropped inadvertently into the reactor core.

The CRA misalignment is an event where a single or multiple CRAs are out of alignment with the remaining rods in the bank. The alignment may be higher or lower than the expected rod position. A misalignment can occur based on the uncertainty in the rod position from its indicated or expected position. A limiting misalignment occurs when the core is operating at steady-state full power with the rods inserted to the PDILs except one rod is left withdrawn. This limiting misalignment is a postulated condition as the rod position indicators alarm when the rods are out of alignment beyond the position uncertainty. The effects of this misalignment are bounded by the analysis of the withdrawal of a single CRA. Another limiting misalignment that is postulated is where all CRAs are withdrawn, except one that is misaligned in to the 20 percent rated power PDIL position. The analysis of this misalignment is presented in this section. The postulated misalignment of the rods inserted to the PDIL but with one CRA fully inserted is not a credible condition. Reactor hold points prohibit the movement of rods for that severe peaking distortion and therefore is not analyzed. The CRA misalignments are classified as AOOs.

The single rod withdrawal transient occurs when a control rod is set at the bank position PDIL and is postulated to withdraw. This event may occur due to wiring failures or operator error where one rod is pulled with disregard for rod position information. The single rod withdrawal adds reactivity and initiates a power

increase transient. The power distribution in the core becomes asymmetric and peaking can challenge the MCHFRR safety limit. A CRM that results in a withdrawal is classified as an AOO.

The CRA or bank drop occurs when a single rod or entire group from the control or shutdown banks drops into the core. The CRA or bank drop can be caused by a mechanical or electrical failure. The event is characterized by a sudden drop in reactor core power. When the rod worth is not significant enough to shut the core down or cause a reactor trip, the constant demand of the secondary side causes a decrease in core inlet temperature, which could result in a power increase. The control system for the regulating control rod bank could withdraw the regulating bank to restore power. The resulting power overshoot with asymmetric peaking could challenge the MCHFRR safety limit. A single CRA drop event is classified as an AOO.

15.4.3.2 Sequence of Events and Systems Operation

The sequence of events for each CRM transient is discussed in the respective results section.

Unless specified below, the analysis of a CRM event assumes the control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a CRM event.

The regulating CRA groups contain the only CRAs that are not fully withdrawn during power operation. The PDIL restricts the amount of insertion steps that the regulating groups can achieve during power operation. The CRM analyses assume that the regulating bank CRAs have a position uncertainty of six insertion or withdrawal steps. The expected normal travel rate of the CRAs is 6 in/min. However, the maximum allowed withdrawal rate of a CRA is 15 in/min, with a step size no greater than three-eighths inch.

The effect of a reactivity insertion event on the RCS is an increase in temperature, which decreases density and causes flow into the pressurizer, increasing RCS pressure. As a result, the normal module control system response would be to decrease pressurizer heater power and increase pressurizer spray. Pressurizer pressure control with heaters and spray is allowed to function normally to delay the trip on high pressurizer pressure.

The MPS is credited to protect the NPM in the event of a CRM. The following MPS reactor trip signals protect the NPM during a CRM:

- high pressurizer pressure
- high power
- high power rate
- high RCS hot temperature
- high RCS average temperature

In CRM events that result in a reactor trip and secondary system isolation, the subsequent actuation of the DHRS is credited with maintaining reactor cooling. The MPS signals available for secondary system isolation and DHRS actuation are identified in Table 7.1-4.

The CRM event analyses assume a single failure of an ex-core flux detector. This failure could affect the power-related reactor trips (high power and high power rate). Because of the power asymmetry resulting from the CRA drop and withdrawal transients, it is possible for a failed detector to cause an error in the detected power. This error could occur if the remaining detectors are located in regions with relative flux that has either increased or decreased due to the power asymmetry. This effect is captured in the analyses by assuming that the remaining detectors see the lowest possible flux in the CRA withdrawal cases, and the highest possible flux in the CRA drop cases.

15.4.3.3 Control Rod Assembly Misalignment Thermal Hydraulic and Subchannel Analysis

15.4.3.3.1 Evaluation Models

There is no NPM thermal hydraulic analysis required for the CRA misalignments given the steady-state nature of the NPM thermal hydraulic conditions of the events. The steady-state core analyses are performed using SIMULATE5 to provide power distributions as input to the subchannel analyses. A discussion of SIMULATE5 is provided in Section 4.3.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic core analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 includes a discussion of the VIPRE-01 code and evaluation model.

15.4.3.3.2 Input Parameters and Initial Conditions

The key inputs for the primary side conditions for the subchannel analysis of the CRA misalignment are provided in Table 15.4-8. The limiting CRA misalignment presented in this section occurs when all CRAs are withdrawn, except for one that is inserted to the 20 percent power PDIL. The following input parameters and initial conditions are considered in this CRA misalignment analysis to ensure a conservative calculation.

- The misaligned CRA is assumed to be inserted 6 steps past the PDIL to account for CRA position indication uncertainty. The added insertion steps maximize the misalignment, providing the most conservative power asymmetry.
- The full range of initial power conditions are analyzed for this misalignment scenario, including: 20, 50, 75, and 100 percent initial power.
- The most positive and negative axial offset values are considered in the analysis.

- The beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC) core conditions are analyzed.

15.4.3.3.3 Results

A static misalignment results in a change in local power shapes and peaking, but the overall power of the core does not change. The thermal hydraulic boundary conditions of the core are at steady-state such that there is no NPM thermal hydraulic transient analysis required for input to the subchannel analysis. Instead, the misalignment scenarios are analyzed to ensure that the applicable SAFDLs are not violated.

The MCHFR for the limiting static misalignment of a CRA is above the CHF analysis limit as indicated in Table 15.4-11.

15.4.3.4 Single Control Rod Assembly Withdrawal Thermal Hydraulic and Subchannel Analysis

15.4.3.4.1 Evaluation Models

The thermal hydraulic analysis of the NPM response to a single CRA withdrawal is performed using NRELAP5. The NRELAP5 model is based on the design features of the NPM. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel CHF analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 includes a discussion of the VIPRE-01 code and evaluation model.

15.4.3.4.2 Input Parameters and Initial Conditions

A spectrum of initial conditions is analyzed to find the limiting reactivity insertion due to a single CRA withdrawal. Key inputs and the associated biases for the limiting single CRA withdrawal analyses are provided in Table 15.4-9 with respect to MCHFR and LHGR. The following initial conditions and assumptions ensure the results have sufficient conservatism.

- Initial power level: 20 percent, 35 percent, 45 percent, 50 percent, 55 percent, 65 percent, 75 percent, 85 percent, 100 percent, and 102 percent of nominal power are analyzed to find the limiting cases. The initial power level for the limiting MCHFR and LHGR cases is 65 percent of nominal power.
- Reactivity insertion rate: The positive reactivity inserted by the CRA withdrawal is modeled as a constant reactivity addition beginning at the transient initiation. The reactivity insertion continues until the time of the reactor trip or the CRA is fully withdrawn. A range of reactivity insertion rates is considered for each initial power level.

- Reactivity feedback: The BOC (least negative) reactivity coefficients are implemented in the limiting CRA withdrawal cases. The least negative reactivity coefficients provide the least amount of feedback to mitigate the power increase due to a CRA withdrawal.
- Conservative reactor trip characteristics are used, including a maximum time delay, holding the most reactive rod out of the core, and using a bounding control rod drop rate.
- The turbine bypass system is not credited in this analysis to minimize heat removal by the secondary side.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with RG 1.105.
- The limiting axial and radial power shapes are used in the subchannel analysis to ensure a conservative evaluation of the SAFDLs.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine the limiting MCHFR and LHGR for this event. The subchannel evaluation model is discussed in Section 15.0.2.

15.4.3.4.3 Results

The sequence of events for a single CRA withdrawal that results in the minimum MCHFR is provided in Table 15.4-10. Figure 15.4-11 through Figure 15.4-16 show the transient behavior of key parameters for this event. The limiting MCHFR and LHGR are provided in Table 15.4-11.

The withdrawal of a single CRA that results in a limiting MCHFR has an initial power of 65 percent. The withdrawal of the CRA results in a reactivity insertion that increases reactor power. The power increase leads to a rise in RCS temperature, pressurizer level, and RCS pressure. The CRA misalignment with the rest of the bank causes an asymmetry in the core, where power peaking increases in the location of the withdrawn CRA. Reactivity feedback from the rising fuel and moderator temperatures partially counteracts the reactivity insertion, slowing the power increase. For CRM cases with higher reactivity insertion rates, the MPS trips the reactor on high power or high power rate. These cases are non-limiting because the reactor is tripped before the maximum amount of reactivity can be inserted. The limiting combination of reactivity insertion and reactivity feedback produces the maximum possible power increase without reaching the high power or high power rate limits. The power increase is terminated by MPS actuation of reactor trip when the high RCS hot temperature limit, high pressurizer pressure limit, or high RCS average temperature limit is reached.

The MPS high pressurizer pressure signal trips the reactor and actuates the DHRS for the limiting MCHFR case. The most limiting MCHFR (Figure 15.4-16) occurs at the moment before the power begins to decrease. The MCHFR remains above the CHF analysis limit and no fuel centerline melting is predicted for the withdrawal of a single CRA. The LHGR calculated for the single CRA withdrawal is below the calculated limits for cladding strain

and fuel centerline melting. The maximum RCS pressure occurs after MCHFR occurs and is followed by decreasing RCS temperature and pressure. The limiting LHGR occurs in a single CRA withdrawal case where the initial power and limiting reactivity insertion rate is the same as that for the MCHFR case, and does not cause a high power rate trip like cases at lower initial powers. The pressurizer spray is assumed to function normally, which delays the trip on high pressure. This case maximizes power in combination with asymmetric peaking, resulting in a limiting LHGR. The LHGR remains below the design limit, so no fuel centerline melting is predicted for the single CRA withdrawal.

15.4.3.5 Control Rod Assembly Drop Thermal Hydraulic and Subchannel Analysis

15.4.3.5.1 Evaluation Models

There is no NPM thermal hydraulic analysis required for the CRA drop. The Non-LOCA Analysis Methodology Topical Report (Reference 15.4-3) identifies that the CRA drop is bounded by either steady-state conditions at event initiation or by the single CRA withdrawal in Section 15.4.3.4. SIMULATE5 is used to calculate CRA worth and power distributions. A discussion of SIMULATE5 is provided in Section 4.3.

The subchannel core CHF analysis of the events bounding the CRA drop are performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 includes a discussion of the VIPRE-01 code and evaluation model.

15.4.3.5.2 Input Parameters and Initial Conditions

A spectrum of possible CRA drop scenarios is analyzed to ensure the scenarios are bounded by either steady-state conditions at event initiation or by the single CRA withdrawal in Section 15.4.3.4. The following initial conditions and assumptions ensure the results have sufficient conservatism.

- The initial power levels of 20 percent, 50 percent, 75 percent, and 100 percent of nominal power are analyzed.
- Scenarios are analyzed for drop of a single CRA within a group and for drop of a whole group.
- The BOC, MOC, and EOC core conditions are analyzed.
- Other initial conditions varied in the analyzed cases include initial CRA position, axial offset, flow, and temperature.
- For the initial CRA drop, a representative drop time is used for all cases. For cases where a reactor trip occurs, conservative reactor trip characteristics are used, including a maximum time delay and holding the most reactive rod out of the core.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with RG 1.105.

15.4.3.5.3 Results

Following a CRA drop, there is a rapid decrease in core reactivity and power. The high power rate limit is reached in the first few seconds of the transient. The MPS sends a reactor trip signal, terminating the event. At lower powers, the power decrease is less pronounced, and the reactor does not trip. In these cases, the regulating CRA bank brings the reactor back to the initial power after an initial power overshoot.

Of the analyzed cases, most are confirmed to reach the high power rate limit by 1.5 seconds. Although local peaking increases due to the asymmetry caused by the CRA drop, total power decreases throughout the event due to the initial CRA drop and subsequent reactor trip. For these cases, the decrease in total power is greater than the increase in local peaking at all times such that local power decreases throughout the event. With the transient decrease in local power, the MCHFR and LHGR are limiting at the initial conditions and therefore bounded by the limiting steady-state subchannel analysis.

The cases that do not reach the high power rate limit by 1.5 seconds are characterized by low initial power and low CRA worth. Because of the lower initial power, these cases are also likely bounded by the limiting steady-state subchannel analysis. However, because power overshoot may occur during the event, these cases are instead compared to the single CRA withdrawal event. The single CRA withdrawal limits for CRA worth, radial peaking, and quadrant tilt are confirmed to bound the analyzed cases that do not reach the high power rate limit by 1.5 seconds. The MCHFR and LHGR are therefore bounded by the limiting single CRA withdrawal analysis in Section 15.4.3.4.

15.4.3.6 Radiological Consequences

The leakage-related radiological consequences of these events are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.4.3.7 Conclusions

The CRM events meet the SRP 15.4.3 acceptance criteria.

1) The thermal margin limits are met.

- The MCHFR for the limiting CRA misalignment is above the CHF analysis limit, as shown in Table 15.4-11.
- The MCHFR for the limiting single CRA withdrawal is above the CHF analysis limit, as shown in Table 15.4-11.
- The MCHFR for the limiting CRA drop is above the CHF analysis limit, as shown in Table 15.4-11.

- 2) Fuel centerline temperatures do not exceed the melting point.
 - As discussed in Reference 15.4-1, a steady-state LHGR protection limit can be applied to the CRM events to ensure the fuel centerline temperatures do not exceed the melting point. The LHGR for the limiting CRA misalignment is below the limit for fuel melt, as shown in Table 15.4-11.
 - The LHGR for the limiting single CRA withdrawal is below the limit for fuel melt, as shown in Table 15.4-11.
 - The LHGR for the limiting CRA drop is below the limit for fuel melt, as shown in Table 15.4-11.
- 3) Uniform cladding strain as specified in SRP Section 4.2, subsection II.A.2(b), does not exceed 1 percent.
 - A core design evaluation described in Section 4.2.3 provides cladding strain limits for the fuel in terms of LHGR. The LHGR values for each of the CRM events is below these limits, demonstrating that cladding strain is not predicted.

15.4.4 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature

There are not multiple coolant loops in the NuScale Power Plant US460 standard design. This event is not applicable to the design.

15.4.5 Flow Controller Malfunction Causing an Increase in Core Flow Rate (Boiling Water Reactor)

There is no steam pressure regulator in the NuScale Power Plant US460 standard design. This event is not applicable to the design.

15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System

15.4.6.1 Identification of Causes and Accident Description

A malfunction in the chemical volume and control system (CVCS) or an operator error could result in unborated or diluted water being inadvertently added to the RCS. A boric acid blend system allows the operator to match or adjust the boron concentration of the reactor coolant makeup water during normal operation. Boron dilution can be either an automatic or a manual operation. In either case, the dilution is governed by administrative controls with procedures that establish the limits on the rate and duration of dilution. An unintended decrease in boron concentration increases the reactivity of the core and decreases the shutdown margin.

The MPS is designed to isolate the demineralized water system (DWS) before the loss of a significant portion of the technical specification minimum shutdown margin. The MPS automatically isolates the DWS on high subcritical multiplication, low RCS flow, and any reactor trip system actuation. The CVCS is

disconnected from the RCS during Mode 4 (transition) and Mode 5 (refueling), but the refueling mode is evaluated to make sure that it bounds the effects of other possible dilution sources present during the refueling process.

An inadvertent decrease in boron concentration in the RCS is classified as an AOO.

15.4.6.2 Sequence of Events and Systems Operation

An inadvertent decrease in boron concentration in the RCS is evaluated for Modes 1, 2, 3, and 5. Boron dilution causes an increase in reactivity, and the NPM response to the event is similar to an uncontrolled CRA withdrawal, presented in Section 15.4.1 and Section 15.4.2. The limiting CVCS dilution source considered in this analysis is the DWS supply. To reduce the overall probability of boron dilution events, administrative controls are placed on the boron addition system supply to the CVCS makeup pumps, assuring that it is not a dilution source for the RCS or the refueling pool.

Unless specified in this section, the RCS boron dilution evaluation assumes the control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an RCS boron dilution event.

The CVCS has two makeup pumps that supply the RCS with makeup coolant and change RCS boron concentration by supplying blended makeup water. Each makeup pump has a maximum capacity of 20 gpm as shown in Table 9.3.4-1. Normally one pump is operated at a time and the other is provided for redundancy. Administrative controls prohibit operation with two pumps early in cycle when critical boron concentrations are high. Table 15.4-12 provides the critical boron concentration below which two pump operation is allowed. The analysis uses a maximum makeup flow rate of 25 gpm. A makeup flow rate of 50 gpm (two pumps) is considered when critical boron concentration is below the limit in Table 15.4-12. To prevent reactor trip on low or high pressurizer level, the CVCS letdown mass flow rate is maintained equal to the makeup flow rate.

The regulating CRA bank is not credited with mitigating the reactivity insertion associated with a boron dilution of the RCS. Each of the two regulating bank groups is assumed to be at their respective PDIL so that rods do not insert automatically as a result of the reactivity addition of an RCS boron dilution.

Technical specifications preclude the possibility of boron dilution when a CRA is stuck out during Modes 2 and 3 by enforcing shutdown margin requirements.

A boron dilution event could occur during refueling operation (Mode 5) if unborated water is unexpectedly introduced to the Reactor Building pool, such as due to internal flooding sources. It is assumed that the flooding source immediately enters the pool and mixes perfectly.

A loss of normal power is considered. The loss of alternating current (AC) power during Mode 5 results in the loss of the reactor pool cooling and recirculation

system. To accommodate the loss of pool cooling circulation, the initial minimum pool mixing volume assumed in the analysis is further reduced for conservatism. In Mode 1 operation, the loss of power scenarios are non-limiting.

During Modes 2 and 3, the reactor is subcritical and there is no power produced by the turbine. Therefore, a loss of AC power due to a possible grid disturbance following a turbine trip is not postulated to occur during Modes 2 and 3.

The MPS signals to trip the reactor and isolate the CVCS are credited with protecting the NPM in the event of a boron dilution of the RCS. The DWS supply to the CVCS makeup pumps is isolated by two in-series safety-related isolation valves on the following MPS signals:

- any reactor trip system actuation
- high subcritical multiplication
- low RCS flow

These MPS signals provide protection in Modes 1 through 3.

No single failure could occur during a boron dilution of the RCS that results in a more severe outcome for the limiting cases. The diversity, redundancy, and independence of the MPS and CVCS isolation valves ensure the NPM is protected from a boron dilution of the RCS despite a single failure.

15.4.6.3 Boron Mixing, Thermal Hydraulic, and Subchannel Analyses

15.4.6.3.1 Evaluation Models

Two calculation techniques are used to analyze the boron dilution event that provide conservative boron dilution assumptions for the evaluation of both reactivity insertion and loss of shutdown margin. The first method evaluates the boron dilution by assuming an instantaneous perfect (complete) mixing model. The second method evaluates the boron dilution by assuming a slug flow or dilution front (wave front) mixing model. In the instantaneous perfect mixing model, unborated water injected into the RCS is assumed to mix instantaneously with the effective system volume. The change in core boron concentration with time is continuous and homogeneous, corresponding to the increasing amount of dilution water entering the RCS. In the dilution front model, unborated water injected into the RCS is assumed to mix with a slug of borated water at the injection point. The diluted slug is assumed to move through the RCS (i.e., through the riser, steam generators, downcomer, and finally through the reactor core). The change in core boron concentration with time depends on the location of the diluted slug.

The two calculation techniques provide the reactivity insertion rate due to the boron dilution. To ensure that the SRP 15.4.6 acceptance criteria are met, the reactivity insertion rate in Mode 1 operation is compared to the spectrum of reactivity insertion rates evaluated in the uncontrolled CRA withdrawal from a subcritical or low power startup condition and uncontrolled CRA withdrawal at

power analyses in Section 15.4.1 and Section 15.4.2, respectively. The reactivity insertion rates are also used as input to NRELAP5 thermal-hydraulic analyses that provide reactor trip timing to determine when the MPS terminates the boron dilution event. The NRELAP5 model is based on the design features of the NPM. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. Adequate shutdown margin must remain in the mixing model analyses at the time when the boron dilution event is terminated in the NRELAP5 analyses. For Mode 2 and Mode 3 operation, the boron dilution scenarios are evaluated at the time of DWS isolation to ensure that adequate shutdown margin remains at the time of automatic DWS isolation. Boron dilution of the reactor pool is estimated using the perfect mixing equation (Equation 15.4-1). The boron dilution scenarios in Mode 5 operation are performed with conservative assumptions of water volumes to demonstrate that shutdown margin is maintained.

15.4.6.3.1.1 Boron Dilution Assuming Perfect Mixing

The perfect (complete) mixing method evaluates the boron concentration of the RCS with the following equation:

$$\frac{dC}{dt} = \frac{-Q_{in}\rho_{in}}{V_r\rho_r}C(t) \quad \text{Eq. 15.4-1}$$

where,

Q_{in} = dilution flow rate of unborated water (gpm). The maximum dilution flow rate is used for this parameter based on the ability of the makeup pumps to deliver water to the CVCS injection line.

ρ_{in} = dilution water density (lb_m/cu.ft). The density value at 14.7 psia, 40 degrees F (minimum temperature value) is used in all cases. The heat addition by the regenerative heat exchangers is not credited, so the analysis assumes that the recirculation pumps are not operational.

V_r = effective water volume of the RCS (gal). A conservatively small value is used that removes the volume of the pressurizer.

ρ_r = density of the water in the RCS (lb_m/cu.ft), and

$C(t)$ = time dependent concentration of boron in the RCS (ppm).

The reactivity insertion rate associated with a given boron dilution rate is calculated with the following equation:

$$\frac{dR}{dt} = \alpha_B \frac{dC}{dt} = -\frac{\alpha_B Q_{in} \rho_{in}}{60 V_r \rho_r} C(t) \quad \text{Eq. 15.4-2}$$

where,

dR/dt = reactivity insertion rate (pcm/sec), and

α_B = differential boron worth (pcm/ppm).

15.4.6.3.1.2

Boron Dilution Assuming Dilution Front or Slug Flow Model

The dilution front or slug (wave front) model uses the following equation for boron concentration:

$$C(N) = C_0 \left[\frac{W_{NC}}{(W_D + W_{NC})} \right]^N \quad \text{Eq. 15.4-3}$$

$$t = \frac{M_{RCSI}}{W_D + W_{NC}} + (N - 1) \frac{M_{RCS}}{W_D + W_{NC}} \quad \text{Eq. 15.4-4}$$

where,

$C(N)$ = the Nth front boron concentration,

C_0 = initial boron concentration,

W_D = dilution mass flow rate,

W_{NC} = natural circulation mass flow rate,

M_{RCS} = RCS fluid mass minus the pressurizer,

M_{RCSI} = initial pass RCS fluid mass (mass between the CVCS injection point to core inlet), and

N = number of times the wave front passes through the core.

In this model, the boron concentration in the RCS is reduced in discrete steps at each time, t , corresponding to the time the wave front passes through the core. Using these equations, the ratio $C_0/C(t)$ is calculated that corresponds to discrete times after dilution begins.

15.4.6.3.2 Input Parameters and Initial Conditions

The initial conditions and input parameters for the boron dilution of the RCS analysis are selected to ensure a conservative calculation.

- The shutdown margin threshold in this analysis for Mode 1 is when $k_{\text{eff}} = 0.995$. The shutdown margin threshold for Modes 2 and 3 in this analysis is when $k_{\text{eff}} = 0.93$. The shutdown margin threshold in this analysis for Modes 4 and 5 is when $k_{\text{eff}} = 0.90$. Therefore, the shutdown margin reactivity credited in this analysis is 503 pcm for Mode 1, 7527 pcm for Modes 2 and 3 and 11,112 pcm for Modes 4 and 5.
- For Mode 1 operation, the initial power levels considered for a boron dilution of the RCS include: hot zero power (HZP), 25 percent power, 50 percent power, 75 percent power and full (100 percent) power. The BOC and EOC conditions are also considered. The Mode 1 cases of HZP and full power at BOC are provided in this section.
- Maximum critical boron concentrations and boron coefficients are assumed because the rate of change of concentration and associated reactivity is greater for an initially higher concentration. The critical boron concentrations and boron reactivity coefficients assumed for each mode of operation are provided in Table 15.4-12.
- The makeup flow rates assumed in the analysis are 5 gpm and 25 gpm. A makeup flow rate of 50 gpm is assumed for critical boron concentrations below the limit for two pump operation in Table 15.4-12. The letdown flow rates are assumed to be equal to the makeup flow rates assumed in the analysis.
- A minimum makeup temperature of 40 degrees F is assumed for the analysis of boron dilution of the RCS.
- The minimum RCS flow rates are assumed to increase loop transit time, which increases the timing for detection and isolation.
- A conservatively smaller reactor pool volume is used to provide a limiting boron dilution for Mode 5.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with RG 1.105.

15.4.6.3.3 Results

The results for a boron dilution of the RCS during Mode 1 operation are presented in Table 15.4-13 for hot full power and Table 15.4-14 for HZP. The tabulated results for the hot full power scenario demonstrate that the reactivity insertion rates are bounded by the range of the reactivity insertion rates that are evaluated in the uncontrolled CRA withdrawal at power analysis, presented in Section 15.4.2. The tabulated results for the HZP scenario demonstrate that the reactivity insertion rates are bounded by the range of the reactivity insertion rates that are evaluated in the uncontrolled CRA withdrawal from a subcritical or low power startup condition analysis, presented in

Section 15.4.1. The tabulated results for the Mode 1 scenarios also demonstrate that shutdown margin is maintained at the time of reactor trip and DWS isolation.

The results for a boron dilution of the RCS during Mode 2 operation and Mode 3 operation are presented in Table 15.4-15 and Table 15.4-16, respectively. The tabulated results for the Mode 2 and Mode 3 scenarios demonstrate that shutdown margin is maintained at the time of DWS isolation.

The results for a boron dilution of the RCS during Mode 5 operation are presented in Table 15.4-17. The total dilution volume necessary to achieve criticality is greater than the volume necessary to raise the reactor pool level from its technical specification minimum allowed level to its technical specification maximum allowed level. Therefore, the technical specifications for pool level preclude the potential for a boron dilution of the RCS resulting in a loss of shutdown margin during Mode 5 operation.

15.4.6.4 Radiological Consequences

The NPM conditions after the limiting decrease in boron concentration cases during Mode 1 operation are bounded by the uncontrolled CRA withdrawal from a subcritical or low power startup condition and uncontrolled CRA withdrawal at power analyses presented in Section 15.4.1 and Section 15.4.2, respectively. Based on the uncontrolled CRA withdrawal from a subcritical or low power startup condition and uncontrolled CRA withdrawal at power results, no fuel failures are predicted and radionuclide barriers maintain integrity during a decrease in boron concentration event. The results for the non-power modes of operation show that shutdown margin is maintained for a decrease in boron concentration event. The normal leakage related radiological consequences of this event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.4.6.5 Conclusions

The NPM conditions after the decrease in boron concentration cases during Mode 1 operation are bounded by the uncontrolled CRA withdrawal from a subcritical or low power startup condition and uncontrolled CRA withdrawal at power analyses presented in Section 15.4.1 and Section 15.4.2, respectively. Shutdown margin is also maintained for Mode 1 cases. The results for the non-power modes of operation (Modes 2 and 3) show that shutdown margin is maintained for a decrease in boron concentration event. The results of reactor pool dilution during Mode 5 operation demonstrate that the technical specifications for pool level preclude the potential for a boron dilution of the RCS resulting in a loss of shutdown margin. Based on these results, it is concluded that the SRP 15.4.6 acceptance criteria are met.

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

An inadvertent loading and operation of a fuel assembly in an improper position is an event that could affect the power distribution and power peaking of the reactor core. If undetected, a fuel assembly loading error could lead to a reduced CHF ratio and reduced margin to fuel centerline melt.

Fuel loading controls are established to prevent a fuel assembly loading error. The fuel loading operation is conducted in accordance with detailed approved procedures. The fuel loading safety measures and procedures are discussed in Section 14.2.

An inadvertent loading and operation of a fuel assembly in an improper position is classified as an infrequent event.

15.4.7.2 Sequence of Events and Systems Operation

The core monitoring system detects a fuel loading error if it causes a relative power shape deviation larger than a detection threshold. The overpower fraction detection threshold is 1.06 and the underpower fraction detection threshold is 0.94. These fractions mean that if an assembly is 6 percent above its predicted power or 6 percent below its predicted power it is detected by the core monitoring system. Some assembly misloads can be detected before these power fraction detection thresholds are reached. The entire spectrum of potential shuffle and rotational fuel assembly misloads are considered.

Shuffle Misloads

The spectrum of potential fuel assembly misloads for the reactor core is examined to assess the impacts of an undetectable fuel assembly misload being present during normal operations. Figure 15.4-17 shows the full spectrum of fuel misloads considered.

The fuel assembly misloads are evaluated in the three categories shown in Figure 15.4-17: quarter-core, half-core, and cross-core. Each of the assembly locations that are evaluated as misloads are numbered for each of the three categories. For the quarter-core misloads, assembly locations 1 through 13 could potentially be shuffled into any of the other numbered assembly locations, which results in a total of 78 potential fuel misloads to be considered on a quarter-core basis. For the half-core misloads, assembly locations 1 through 9 could potentially be shuffled into assembly locations 10 through 18, a total of 81 potential fuel misloads to be considered on a half-core basis. For the cross-core misloads, assembly locations 1 through 12 could potentially be shuffled into assembly locations 13 through 18, resulting a total of 72 potential fuel misloads to be considered on a half-core basis. For the three categories, there are 231 potential misloads that are analyzed.

The center assembly is a fresh assembly or previously irradiated in the center location. Therefore, exchanges of the center assembly are only examined on a quarter-core basis because the faces of the center assembly do not have a different depletion history than each other and exchanges in the other quadrants are consistent with those performed in a single quadrant.

Rotational Misloads

Section 4.2 describes the design features that prevent fuel assembly rotational misloads. Nevertheless, 180 degree rotational misloads are examined.

15.4.7.3 Thermal Hydraulic and Subchannel Analyses

15.4.7.3.1 Evaluation Model

There is no NPM thermal hydraulic analysis required for the fuel assembly misloads given the steady-state nature of the NPM thermal hydraulic conditions of the events. SIMULATE5 is used to determine the limiting undetectable fuel misload, and to provide peaking factors to the subchannel analysis. A discussion of SIMULATE5 is provided in Section 4.3.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 includes a discussion of the VIPRE-01 code and evaluation model.

15.4.7.3.2 Input Parameters and Initial Conditions

The fuel misload event changes the power distribution of the core, but the thermal hydraulic boundary conditions remain the same. Therefore, there is no need for an NRELAP5 analysis to ensure that the RCS pressure remains below the design limit of the reactor pressure vessel (RPV). The power distribution of the equilibrium core analysis is discussed in Section 4.3.

The power peaking augmentation factors for the limiting undetectable misload are calculated using SIMULATE5, and provided as input to the steady-state subchannel analysis to determine the MCHFR for this event. The limiting undetectable misload is a cross-core misload. This limiting misload power peaking augmentation factor is bounded by the analysis value of 1.15. Other key inputs and assumptions used in the subchannel analysis are provided in Reference 15.4-1 and Reference 15.4-2.

15.4.7.3.3 Results

The limiting undetectable fuel misloading event results in an MCHFR that is above the CHF analysis limit. Fuel temperature margin to centerline melt is calculated for the worst case fuel assembly misloading event. The calculated value of LHGR for the worst misload is below the limiting LHGR. These results are provided in Table 15.4-18. Because MCHFR is above the limit and fuel

centerline melting is not expected to occur, no fuel damage is expected. These events change the power distribution within the core, not overall core power. Therefore, there is no power increase associated with the fuel misloading events that could challenge the radionuclide barriers.

15.4.7.4 Radiological Consequences

The normal leakage related radiological consequences of this event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.4.7.5 Conclusions

The results from the evaluation of the limiting undetectable fuel misloading events show that no fuel damage is expected. There is no pressure transient associated with this event, so the RCS pressure boundary is not challenged. With no fuel damage and no challenge to radionuclide boundaries, the normal leakage related radiological consequences of this event are bounded by the design-basis accident analyses in Section 15.0.3. Therefore, SRP 15.4.7 acceptance criteria are met.

15.4.8 Spectrum of Rod Ejection Accidents

15.4.8.1 Identification of Causes and Accident Description

A postulated failure of the control rod drive mechanism (CRDM) pressure housing could cause a control rod to be ejected from the core. The unexpected and rapid increase in positive reactivity demonstrates the effects of a limiting reactivity insertion event.

The power spike resulting from the CRA ejection is quickly countered by the fuel reactivity feedback as the fuel temperature begins to increase. The sudden increase in power is detected by the MPS, resulting in a reactor trip. The sudden ejection of a CRA adds positive reactivity to a localized region of the core in a very short period of time. This CRA ejection results in a power excursion in the region near the affected fuel assembly and results in a highly asymmetric power distribution in the radial dimension. This adverse power distribution subsequently leads to overheating of the affected fuel assemblies and possible fuel damage.

The rod ejection accident (REA) is classified as a postulated accident.

15.4.8.2 Sequence of Events and Systems Operation

The sequence of events for the REA case with the maximum power increase is shown in Table 15.4-19.

Unless specified in this section, the analysis of an REA assumes the control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of an REA.

Pressure and inventory control is disabled to ensure maximum pressure, which is conservative with respect to MCHFR.

The feedwater flow control would normally raise feedwater flow in response to increasing RCS temperature. In order to minimize the CHF ratio, the feedwater flow is held constant.

Steam pressure is held constant. Steam pressure controls would normally relieve steam pressure as reactor power increases which would enhance heat transfer and lower both the RCS temperature and pressure.

If the reactivity control system is in an automatic mode, the regulating CRAs could insert before a reactor trip in response to an REA. The regulating CRAs are not credited to mitigate the reactivity insertion of the REA before a reactor trip, to provide the most conservative power response.

The REA analyses assume AC and DC power is available and assume a single failure of an ex-core flux detector. This single failure could affect the power-related reactor trips (high power and high power rate). Because of the power asymmetry resulting from the REA, it is possible for a failed detector to cause an error in the detected power. This error could occur if the remaining detectors are located in regions with relative flux that has either increased or decreased due to the power asymmetry. This effect is captured in the analyses by requiring all ex-core flux detectors to reach the power-related trip setpoint before a trip signal is generated.

15.4.8.2.1 Mechanical Design

To prevent a failure of the CRDM pressure housings, the CRDM is made as a single-piece housing, with a top plug. The CRDM pressure housing is an American Society of Mechanical Engineers Class 1 pressure boundary. The control rod drive system functional design is discussed in Section 4.6.

15.4.8.2.2 Effects on Other Control Rod Housings

The damaged control rod housing is postulated to provide a limiting reactivity insertion event. However, the mechanical failure of a control rod housing that would result in a missile is non-credible, as discussed in Section 3.5. Nevertheless, the reactivity effects of a neighboring CRDM housing being damaged are bounded by assuming a rod is stuck out for a reactor trip.

15.4.8.2.3 Nuclear Design

The NPM uses two reactivity control mechanisms: power regulating control rods and RCS boron concentration. The use of boron in the RCS to control reactivity limits the insertion of regulating control rods during power operation to a predefined PDIL. With the rods mostly withdrawn from the core, the amount of reactivity associated with an REA is limited.

The core is also designed with a negative Doppler temperature coefficient (DTC) that limits the magnitude of a power pulse associated with an REA. A

discussion of the negative reactivity feedback design of the core is provided in Section 4.3.

15.4.8.2.4 Module Protection System

The MPS protects the NPM by tripping the reactor in the event of an REA. The MPS reactor trip signals that provide protection during an REA are the following:

- high power
- high power rate
- high main steam superheat
- high pressurizer pressure
- high RCS hot temperature
- high main steam pressure

In the event of a reactor trip, the subsequent actuation of the DHRS would maintain reactor cooling.

15.4.8.3 Nuclear, Thermal Hydraulic, and Subchannel Analyses

15.4.8.3.1 Evaluation Models

The nuclear analyses for an REA are performed using SIMULATE-3K (S3K). A discussion of the S3K code can be found in Section 4.3.3. The S3K output provides the reactor power as a function of time to the downstream NRELAP5 and VIPRE-01 analyses.

The thermal hydraulic analysis of the NPM response to an REA is performed using NRELAP5. The NRELAP5 model is based on the design features of the NPM. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel analysis.

The subchannel core analysis is performed using VIPRE-01 to predict any fuel failure due to CHF, fuel temperature, or fuel enthalpy. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. Section 15.0.2 includes a discussion of the VIPRE-01 code and evaluation model.

15.4.8.3.2 Input Parameters and Initial Conditions

SIMULATE-3K Model

The power and reactivity data for an REA is calculated using S3K for the downstream thermal hydraulic and CHF analyses. The power response calculated by S3K is also used in the downstream analyses to determine the

temperature and enthalpy responses of the fuel. The CHF analysis and the temperature and enthalpy responses of the fuel indicate if there is any fuel failure during an REA. The inputs, initial conditions, and conservatism of the S3K rod ejection accident model are discussed in this section.

In order to maximize the possible reactivity insertion from an ejected rod, the non-shutdown CRAs are assumed to be at the PDIL with an uncertainty of 6 steps. The shutdown bank is positioned all rods out. Conservative reactor trip characteristics are applied to the REA model including:

- highest worth CRA (other than the ejected rod) remains stuck out of the core
- reactor trip delay of 2 seconds
- maximum CRA drop time after reactor trip

Conservative core characteristics are applied to the REA model to ensure the maximum reactivity insertion with minimum feedback. A top peaked power shape is applied to the REA model to maximize the effects of the ejected rod. The uncertainty values of the DTC and moderator temperature coefficient (MTC) in the REA analysis are applied to the S3K values in the conservative (less negative) direction to minimize the fuel feedback effects that could mitigate the power response of an REA. A discussion of specific core parameter values and the associated biases used in the REA methodology is provided in the Rod Ejection Methodology Topical Report (Reference 15.4-4).

The S3K analysis provides REA power response calculations at the following power levels and times in cycle:

- Power (%) - 0, 20, 50, 75, 100
- Time in life - BOC, MOC, EOC

NRELAP5 Model

The NRELAP5 thermal hydraulic analysis uses the power response calculated by S3K to simulate the power pulse associated with an REA.

The NRELAP5 analysis provides boundary conditions to the downstream subchannel analysis to evaluate CHF, fuel temperature, and fuel enthalpy. The NRELAP5 cases provided for the downstream subchannel analysis are initialized with the following inputs:

- Initial power - The highest S3K power increase at each initial power level is selected.
- BOC and EOC conditions - The EOC conditions result in the highest S3K power increase at each initial power level, as shown in Table 15.4-20. However, BOC conditions at hot full power (HFP) are also included as they may be limiting for the downstream subchannel analysis.
- Average RCS temperature biased low and high - Although higher temperatures increase the vaporization and pressurization rate, lower

temperatures are also included as they may be limiting for the downstream subchannel analysis.

- The RCS pressure and pressurizer level biased high - This combination maximizes the RCS pressurization rate.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with RG 1.105.

Key inputs and assumptions used in the subchannel analysis are provided in Reference 15.4-1 and Reference 15.4-2. The results of the subchannel analysis determine if there is any potential fuel damage resulting from an REA. The REA event-specific methodology is provided in Reference 15.4-4.

NRELAP5 is also used to evaluate the maximum pressure response to an REA. The methodology in Reference 15.4-4 identifies that a generic assessment demonstrates the pressure acceptance criteria are generally satisfied for the MPS analytical limits of power, power rate, and pressure. The generic assessment using NRELAP5 is applicable to an REA for the NuScale Power Plant US460 standard design based on the MPS analytical limits in Section 7.1 and the REA peak power calculated in S3K.

15.4.8.3.3 Regulatory Criteria

Reference 15.4-4 discusses the various REA regulatory acceptance criteria and how they apply. A summary of these acceptance criteria are provided in this section.

Fuel Cladding Failure

- The high temperature cladding failure threshold is expressed in cladding differential pressure. The peak radial average fuel enthalpy must be below 100 cal/g. For the NuScale Power Plant US460 standard design, the 100 cal/g limit is applied at all peak rod differential pressures.
- Fuel cladding failure is presumed if local heat flux exceeds the CHF analysis limit.
- The pellet-cladding mechanical interaction failure limit is a change in radial average fuel enthalpy of 33 cal/g or greater, bounding the excess hydrogen-dependent limit depicted in Figure 5-3 of Reference 15.4-4.
- If fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions, then fuel cladding failure is presumed.

Core Coolability

- Peak radial average fuel enthalpy shall remain below 230 cal/g.
- No fuel melt shall occur.
- Local heat flux shall not exceed the CHF analysis limit.

Reactor Coolant System Pressure

The maximum RCS pressure must remain below 120 percent of design pressure. Therefore, the peak pressure during an REA is limited below 2640 psia.

15.4.8.3.4 Fuel and Cladding Integrity Results

SIMULATE-3K provides the power and reactivity response to an REA for each statepoint discussed in Section 15.4.8.3.2. The S3K analysis assumes the maximum reactivity insertion from ejecting the highest worth CRA for each of these statepoints. Table 15.4-20 provides the maximum power in percent HFP as well as the inserted reactivity of the ejected rod for a spectrum of initial power levels and times in cycle. Figure 15.4-18 provides the maximum power pulse, which occurs for EOC hot zero power conditions. The figure shows a rapid reactivity excursion when the rod is ejected, but the power pulse is mitigated due to fuel feedback effects. The reactor is tripped when the power reaches the setpoint, and the rods are inserted after the trip delay.

The subchannel analysis evaluates a spectrum of REA conditions that are provided by the S3K and NRELAP5 analyses. Table 15.4-21 provides the peak fuel temperature, change in enthalpy, peak fuel enthalpy, and MCHFR for the spectrum of REA conditions. A comparison of the limiting values of these parameters to the acceptance criteria is provided in Table 15.4-22. The REA case that results in the limiting MCHFR is an REA that occurs at HFP at BOC with a power increase that does not result in reactor trip. The limiting MCHFR is above the CHF analysis limit as demonstrated in Figure 15.4-19.

The maximum possible mass and energy release to containment due to a postulated control rod housing failure is bounded by an inadvertent opening of a reactor vent valve (RVV). A postulated control rod housing failure would represent a maximum break size of 3.125 inches (inner diameter). This break size is smaller than the venturi of an RVV. The additional energy from the power excursion of an REA is not sufficient to exceed the energy release of an inadvertent opening of an RVV. The inadvertent opening of an RVV is discussed in Section 15.6.6. The limiting containment peak pressures and temperatures for design-basis events are discussed in Section 6.2.

The REA acceptance criteria for fuel cladding failure and core coolability are met, which indicates that no fuel failures are predicted in the event of an REA.

15.4.8.3.5 Maximum Reactor Coolant System Pressure Results

The maximum pressure based on the generic assessment of the REA using the Reference 15.4-4 methodology is shown in Table 15.4-22. The REA maximum pressure is bounded by the maximum pressures of the decrease in heat removal by the secondary system events in Section 15.2.

15.4.8.4 Radiological Consequences

No fuel damage is predicted for the limiting REA. Therefore, the radiological consequences of an REA are bounded by the consequences of other accidents presented in Section 15.0.3.

15.4.8.5 Conclusions

The applicable acceptance criteria for this accident are met for the limiting REA cases. These acceptance criteria are provided in Section 15.4.8.3.3. The NPM design meets these criteria as discussed in the summary below.

Fuel Cladding Failure

- The limiting peak radial average fuel enthalpy is below the fuel cladding failure limit, as shown in Table 15.4-22.
- The limiting MCHFR is above the CHF analysis limit, as shown in Table 15.4-22.
- The limiting change in radial average fuel enthalpy is below the pellet-cladding mechanical interaction failure threshold, as shown in Table 15.4-22.
- The limiting peak fuel temperature is below the fuel melting temperature, as shown in Table 15.4-22.

These limiting fuel cladding results indicate no fuel failures.

Core Coolability

- The limiting peak radial average fuel enthalpy is below the limit, as shown in Table 15.4-22.
- The limiting peak fuel temperature is below the fuel melting temperature, as shown in Table 15.4-22.

These limiting results indicate that core coolability is maintained for the limiting REA.

Reactor Coolant System Pressure

- The limiting RCS peak pressure is below the RPV design limit, as shown in Table 15.4-22.

This limiting result indicates that RPV integrity is maintained during the limiting REA.

15.4.9 Spectrum of Rod Drop Accidents

15.4.9.1 Identification of Causes and Accident Description

This event is specific to boiling water reactors and not applicable to the NuScale Power Plant US460 standard design. The pressurized water reactor equivalent of

a rod drop, the rod ejection, is addressed in Section 15.4.8. Control rod misoperations, including a dropped CRA, are addressed in Section 15.4.3.

15.4.10 References

- 15.4-1 NuScale Power, LLC, "Subchannel Analysis Methodology," TR-0915-17564-P-A, Revision 2.
- 15.4-2 NuScale Power, LLC, "Statistical Subchannel Analysis Methodology, Supplement 1 to TR-0915-17564-P-A, Revision 2," TR-108601-P, Revision 2.
- 15.4-3 NuScale Power, LLC, "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-P, Revision 4.
- 15.4-4 NuScale Power, LLC, "Rod Ejection Methodology," TR-0716-50350-P, Revision 2.

Table 15.4-1: Sequence of Events for Limiting Minimum Critical Heat Flux Ratio and Linear Heat Generation Rate Case (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from Subcritical or Low Power Startup Condition)

Event	Time [s]
Rod withdrawal initiates	0
High power (25%) limit reached	201
Reactor trip actuated	203
Maximum power occurs	205
MCHFR occurs	205
Maximum RCS pressure occurs	205

Table 15.4-2: Key Inputs for Limiting Minimum Critical Heat Flux Ratio and Linear Heat Generation Rate Case (15.4.1 Uncontrolled Control Rod Assembly Withdrawal from Subcritical or Low Power Startup Condition)

Parameter	Nominal	Bias
Initial power	500 kW	N/A ¹
Initial RCS flow rate	80 kg/sec	Low ²
Pressurizer pressure	2000 psia	Nominal
Pressurizer level	50%	Nominal
RCS average temperature	351.9 °F	Nominal
MTC	N/A ³	BOC (most positive/least negative)
FTC	N/A ³	BOC (least negative)
Reactivity insertion rate	0.00094 \$/sec	N/A

¹A spectrum of initial powers is analyzed, and this value provided the limiting results.

²The initial RCS flow rate varies as a function of the initial power. RCS flow is biased low for the initial power level.

³Nominal values are not specified; only most negative and most positive/least negative values are specified.

Table 15.4-3: Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition (15.4.1) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR	1.33	10.0
LHGR	14.2 kW/ft	6.0 kW/ft

**Table 15.4-4: Sequence of Events for Limiting Minimum Critical Heat Flux Ratio
Case - 102% Power (15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)**

Event	Time [s]
CRA bank begins to withdraw	0
Maximum RCS pressure occurs	548
High power limit reached	1280
Reactor trip actuated	1282
MCHFR occurs ¹	1282
DHRS actuated on high main steam pressure	1298
DHRS valves fully open	1328

¹ Because of conservative modeling, the time of MCHFR shown in Figure 15.4-10 is before reactor trip. Without this conservative modeling, the MCHFR occurs at the time of reactor trip as shown in this table. Table 15.4-7 provides the more conservative MCHFR value consistent with Figure 15.4-10.

**Table 15.4-5: Key Inputs for Limiting Minimum Critical Heat Flux Ratio Case
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)**

Parameter	Nominal	Bias
Initial power	250 MW	+2%
RCS flow rate	Table 15.0-6 ²	1441 lbm/sec (low ¹)
RCS pressure	2000 psia	-70 psia
Pressurizer level	60%	+8%
RCS average temperature	540 °F	-5 °F
MTC	N/A ³	BOC (most positive/least negative)
FTC	N/A ³	BOC (least negative)
Reactivity insertion rate	0.0002 \$/sec	N/A

¹ RCS flow rate is near the minimum for 100 percent power.

² Table 15.0-6 provides the range for RCS flow rate at 100 percent power.

³ Nominal values are not specified; only most negative and most positive/least negative values are specified.

**Table 15.4-6: Key Inputs for Limiting Linear Heat Generation Rate Case
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)**

Parameter	Nominal	Bias
Initial power	250 MW	+2%
RCS flow rate	Table 15.0-6 ²	1818 lbm/sec (high ¹)
RCS pressure	2000 psia	-70 psia
Pressurizer level	60%	+8%
RCS average temperature	540 °F	+5 °F
MTC	N/A ³	BOC (most positive/ least negative)
FTC	N/A ³	BOC (least negative)
Reactivity insertion rate	0.0500 \$/sec	N/A

¹ RCS flow rate is conservatively above the maximum for 100 percent power.

² Table 15.0-6 provides the range for RCS flow rate at 100 percent power.

³ Nominal values are not specified; only most negative and most positive/least negative values are specified.

Table 15.4-7: Uncontrolled Control Rod Assembly Withdrawal at Power (15.4.2) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR	1.33	1.47
Peak LHGR	14.2 kW/ft	12.1 kW/ft

**Table 15.4-8: Key Inputs for Limiting Control Rod Assembly Misalignment
(15.4.3 Control Rod Misoperation, Control Rod Assembly Misalignment)**

Parameter	Nominal	Bias
Initial power	250 MW	+2%
RCS flow rate	Table 15.0-6 ²	647 kg/sec (low ¹)
RCS pressure	2000 psia	-70 psia
Core inlet temperature	479 °F ³	Nominal

¹ RCS flow rate is near the minimum for 100 percent power.

² Table 15.0-6 provides the range for RCS flow rate at 100 percent power.

³ This core inlet temperature corresponds to the biased high RCS average temperature of 545 °F and biased low RCS flow rate.

Table 15.4-9: Key Inputs for Limiting Minimum Critical Heat Flux Ratio and Linear Heat Generation Rate Cases (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

Parameter	Nominal	Bias
Initial power	65% of full power	Nominal
RCS flow rate	Table 15.0-6 ²	539 kg/sec (low ¹)
Pressurizer level	60%	-8%
RCS pressure	2000 psia	-70 psia
RCS average temperature	540 °F	+5 °F
MTC	N/A ³	BOC (most positive/least negative)
DTC	N/A ³	BOC (least negative)
Reactivity insertion rate	0.0034 \$/sec	N/A

¹ RCS flow rate is near the minimum for 65 percent power.

² Table 15.0-6 provides the range for RCS flow rate at 100 percent power.

³ Nominal values are not specified; only most negative and most positive/least negative values are specified.

**Table 15.4-10: Sequence of Events for Limiting Minimum Critical Heat Flux Ratio Case
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)**

Event	Time [s]
Single CRA begins to withdraw	0
High pressurizer pressure limit reached	79
Reactor trip actuation	81
MCHFR occurs	82
Maximum RCS pressure occurs	84
DHRS valves fully open	111

Table 15.4-11: Control Rod Misoperation (15.4.3) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR CRA misalignment	1.33	1.45
MCHFR single CRA withdrawal	1.33	1.46
MCHFR CRA drop	1.33	1.46 ¹
Peak LHGR CRA misalignment	14.2 kW/ft	12.1 kW/ft
Peak LHGR single CRA withdrawal	14.2 kW/ft	12.6 kW/ft
Peak LHGR CRA drop	14.2 kW/ft	12.6 kW/ft ¹

¹ The CRA drop is bounded by steady state conditions or by the single CRA withdrawal. The analysis value is shown for the limiting case for the single CRA withdrawal.

**Table 15.4-12: Bounding Critical Boron Concentrations and Boron Reactivity Coefficients
(15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)**

Operation Mode	Critical Boron Concentration ^{1,2} (ppm)	Boron Reactivity Coefficient (pcm/ppm)
Mode 1, $\geq 25\%$ power	1600 / 600	-10
Mode 1, hot zero power	1900 / 1000	-10
Mode 2	600	-11
Mode 3	650	-12.5
Modes 4 and 5	1900	-11.5

¹Where two values are provided for critical boron concentration, the two values correspond to BOC and EOC, respectively.

²Operation with two makeup pumps is prohibited when critical boron concentration is above 600 ppm.

**Table 15.4-13: Mode 1, Hot Full Power Results, Beginning of Cycle²
(15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)**

Parameter	Value	Value	Value
Dilution rate (gpm)	5	25	50
Reactivity insertion rate - complete mixing model (pcm/sec)	0.1182	0.5908	0.4310
Initial reactivity insertion rate - wave front model (pcm/sec)	3.44	17.19	12.89
Range of reactivity insertion rates assumed in Section 15.4.2 (pcm/sec) ¹	0.064 to 24		
Time to loss of shutdown margin - complete mixing model (minutes)	70.9	14.1	19.4
Time to reactor trip/DWS isolation actuation signal - complete mixing model (minutes)	20.8	3.9	6.3
Shutdown margin remaining at time of reactor trip/DWS isolation actuation signal - complete mixing model (pcm)	352.5	359.6	330.0

¹Reactivity insertion rates from all dilution rates are bounded by the range of reactivity insertion rates assumed in uncontrolled control rod assembly withdrawal at power condition (Section 15.4.2).

²The dilution flow rate of 50 gpm is associated with low critical boron concentrations instead of a time in cycle.

Table 15.4-14: Mode 1, Hot Zero Power Results, Beginning of Cycle²
(15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Parameter	Value	Value	Value
Dilution rate (gpm)	5	25	50
Initial reactivity step - wave front model (pcm)	234.91	1119.19	667.53
Initial reactivity insertion rate - wave front model (pcm/sec)	3.43	17.15	10.83
Reactivity insertion rate - complete mixing model (pcm/sec)	0.1131	0.5653	0.3076
Duration of the reactivity insertion rate for each wave - wave front model (seconds)	68.5	65.3	61.6
Range of reactivity insertion rates assumed in Section 15.4.1 (pcm/sec) ¹	0.064 to 24		
Time to loss of shutdown margin - wave front model (minutes)	321.0	57.0	112.5
Time to reactor trip/DWS isolation actuation signal - wave front model (minutes)	27.2	25.7	24.4
Shutdown margin remaining at time of reactor trip/DWS isolation actuation signal - wave front model (pcm)	1889.3	1147.7	1541.3

¹Reactivity insertion rates from all dilution rates are bounded by the range of reactivity insertion rates assumed in uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition (Section 15.4.1).

²The dilution flow rate of 50 gpm is associated with low critical boron concentrations instead of a time in cycle.

Table 15.4-15: Mode 2 Results
(15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Parameter	Value	Value
Dilution rate (gpm)	5	25
Initial wave reactivity step (pcm)	207.37	979.34
Time to loss of shutdown margin (minutes)	1690	334
Time of DWS isolation (minutes)	1592	241
Shutdown margin remaining at DWS isolation (pcm)	245	1351

Table 15.4-16: Mode 3 Results
(15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Parameter	Value	Value
Dilution rate (gpm)	5	25
Initial wave reactivity step (pcm)	203.34	966.48
Time to loss of shutdown margin (minutes)	1660	336
Time of DWS isolation (minutes)	1562	243
Shutdown margin remaining at DWS isolation (pcm)	231	1275

Table 15.4-17: Mode 5 Results
(15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

Parameter	Value
Assumed initial mixing volume (ft ³)	104,210
Time to loss of shutdown margin (minutes) ¹	309
Time to loss of shutdown margin (hours) ¹	5.14
Total dilution volume to reduce shutdown margin to zero (gallons)	308,633
Total volume to raise pool level from minimum allowed to maximum allowed (gallons)	117,609

¹ Based on assumed dilution flow rate of 1000 gpm.

Table 15.4-18: Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (15.4.7) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
MCHFR	1.33	1.45
Peak LHGR	14.2 kW/ft	12.1 kW/ft

**Table 15.4-19: Sequence of Events - Maximum Increase in Power Case
(15.4.8 Rod Ejection Accident)**

Event	Time [s]
Rod ejection begins	0
High power rate limit reached	< 1
Maximum power occurs	< 1
Reactor trip actuates on high power rate signal	2
Reactor trip complete	4

Table 15.4-20: Rod Ejection Accident (15.4.8) - Maximum Power Pulses and Reactivity Insertions

Time in Cycle	Initial Power (%)	Maximum Power (%)	Maximum Reactivity Insertion (\$)
BOC	100	105.4	0.053
BOC	75	87.8	0.151
BOC	50	71.9	0.312
BOC	20	71.7	0.720
BOC	0	19.3	0.854
MOC	100	106.0	0.059
MOC	75	88.9	0.163
MOC	50	72.9	0.324
MOC	20	63.2	0.686
MOC	0	13.8	0.777
EOC	100	116.0	0.141
EOC	75	97.1	0.237
EOC	50	92.7	0.473
EOC	20	258.1	0.968
EOC	0	299.5	1.139

Table 15.4-21: Rod Ejection Accident (15.4.8) - Fuel Temperatures, Enthalpies, and Minimum Critical Heat Flux Ratios for Limiting S3K Cases

Time in Cycle	Initial Power (%)	Peak Fuel Temperature (°F)	Change in Radially Averaged Fuel Enthalpy (cal/g)	Peak Radially Averaged Fuel Enthalpy	MCHFR
BOC	100	2139	0	58	3.58
EOC	0	610	20	22	10.00
EOC	20	1269	13	40	4.32
EOC	50	1601	3	46	5.13
EOC	75	1800	1	50	4.54
EOC	100	2015	1	55	3.88

Table 15.4-22: Spectrum of Rod Ejection Accidents (15.4.8) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Peak radial average fuel enthalpy ¹	100 cal/g	58 cal/g
Change in radial average fuel enthalpy	33 cal/g	20 cal/g
Maximum RCS pressure ²	2640 psia	2231 psia
Peak fuel temperature	4791 °F	2139 °F
MCHFR	1.33	3.58

¹ Meeting the 100 cal/g limit on peak radial average fuel enthalpy for high-temperature cladding failure at any power also inherently meets the 230 cal/g limit for core coolability.

² The maximum RCS pressure value is from a generic assessment based on the Reference 15.4-4 methodology.

Figure 15.4-1: Reactor Power
(15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition)

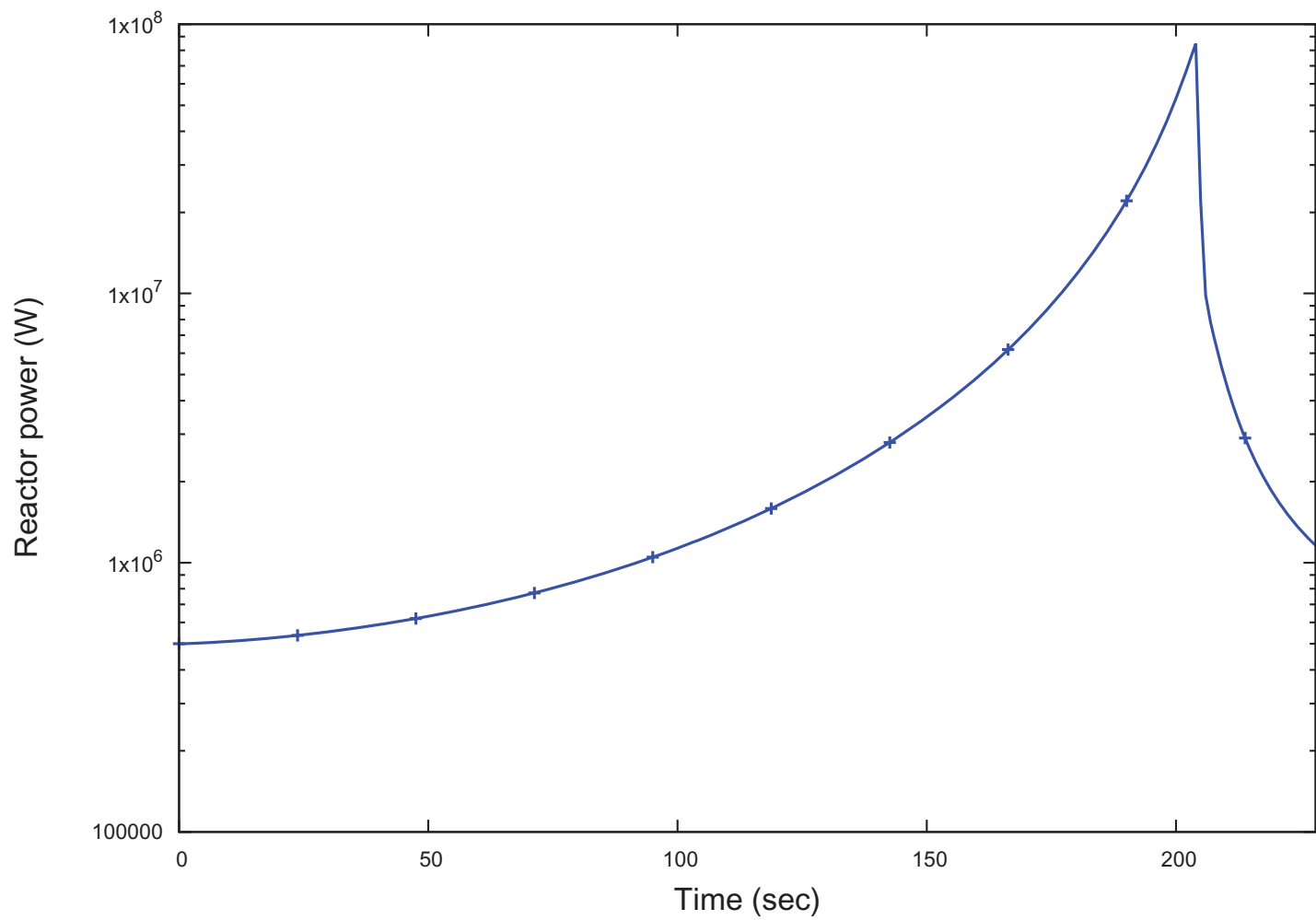


Figure 15.4-2: Net Reactivity
(15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition)

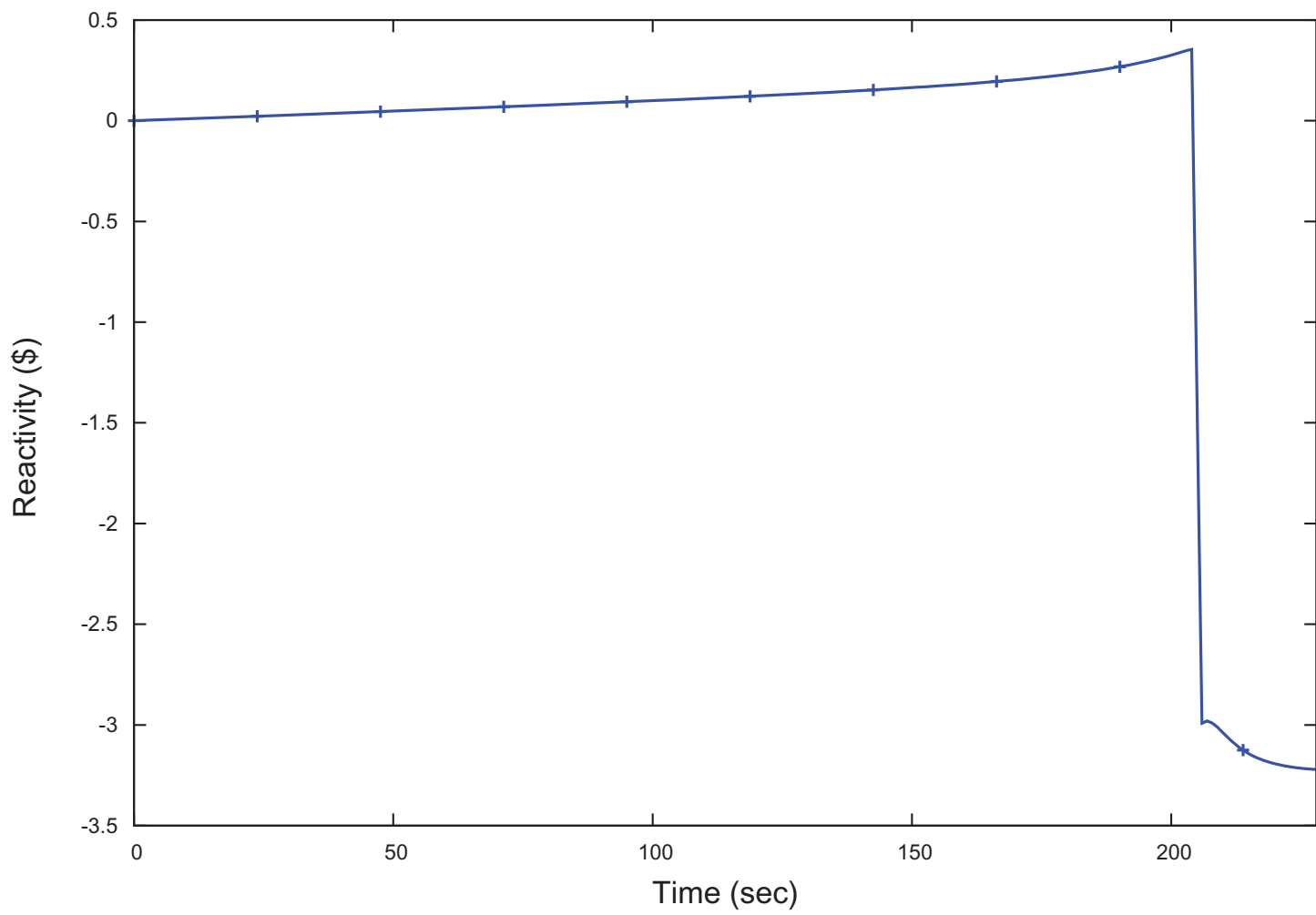


Figure 15.4-3: Core Outlet Temperature
(15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition)

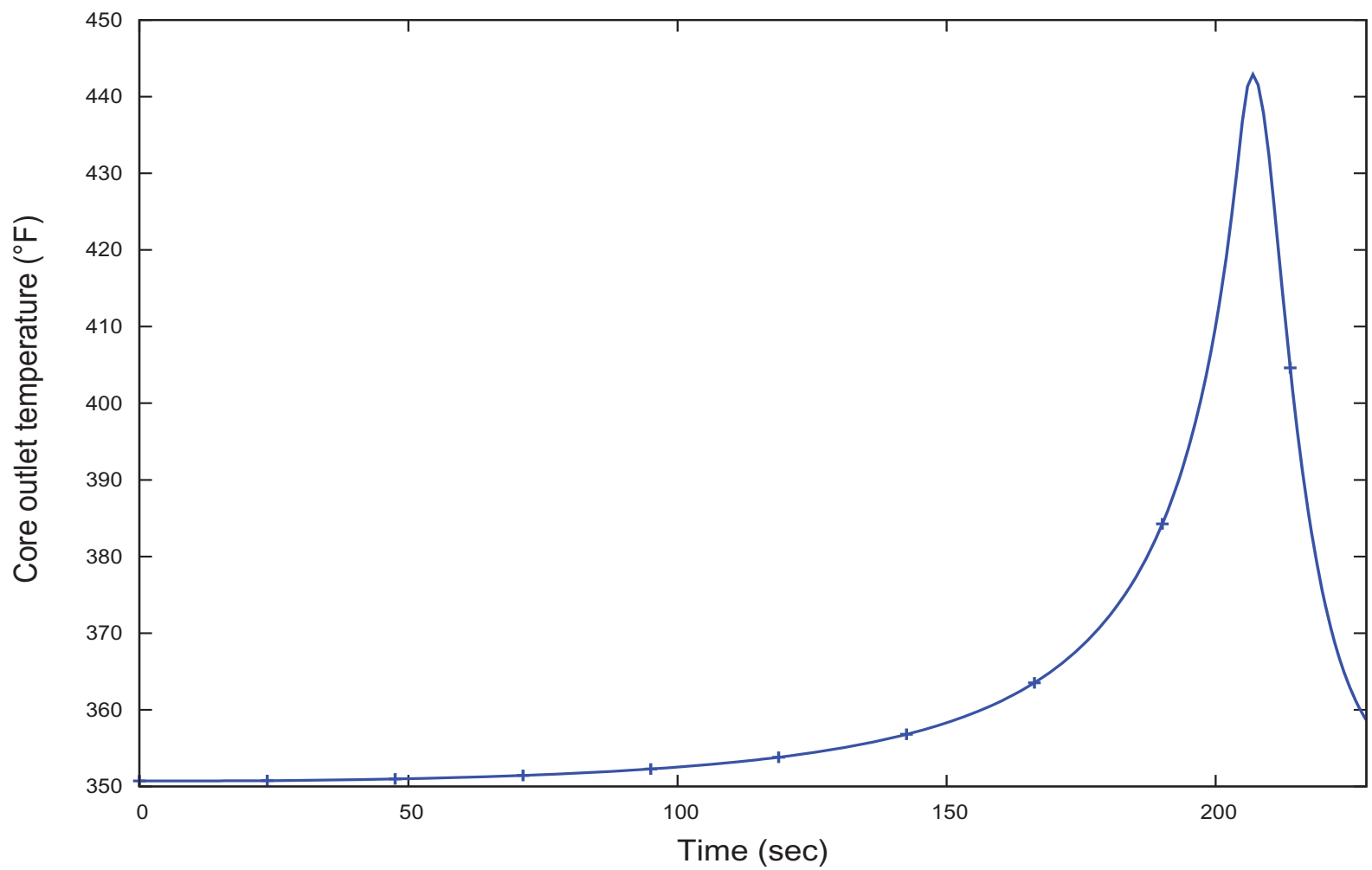


Figure 15.4-4: Primary Pressure
(15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition)

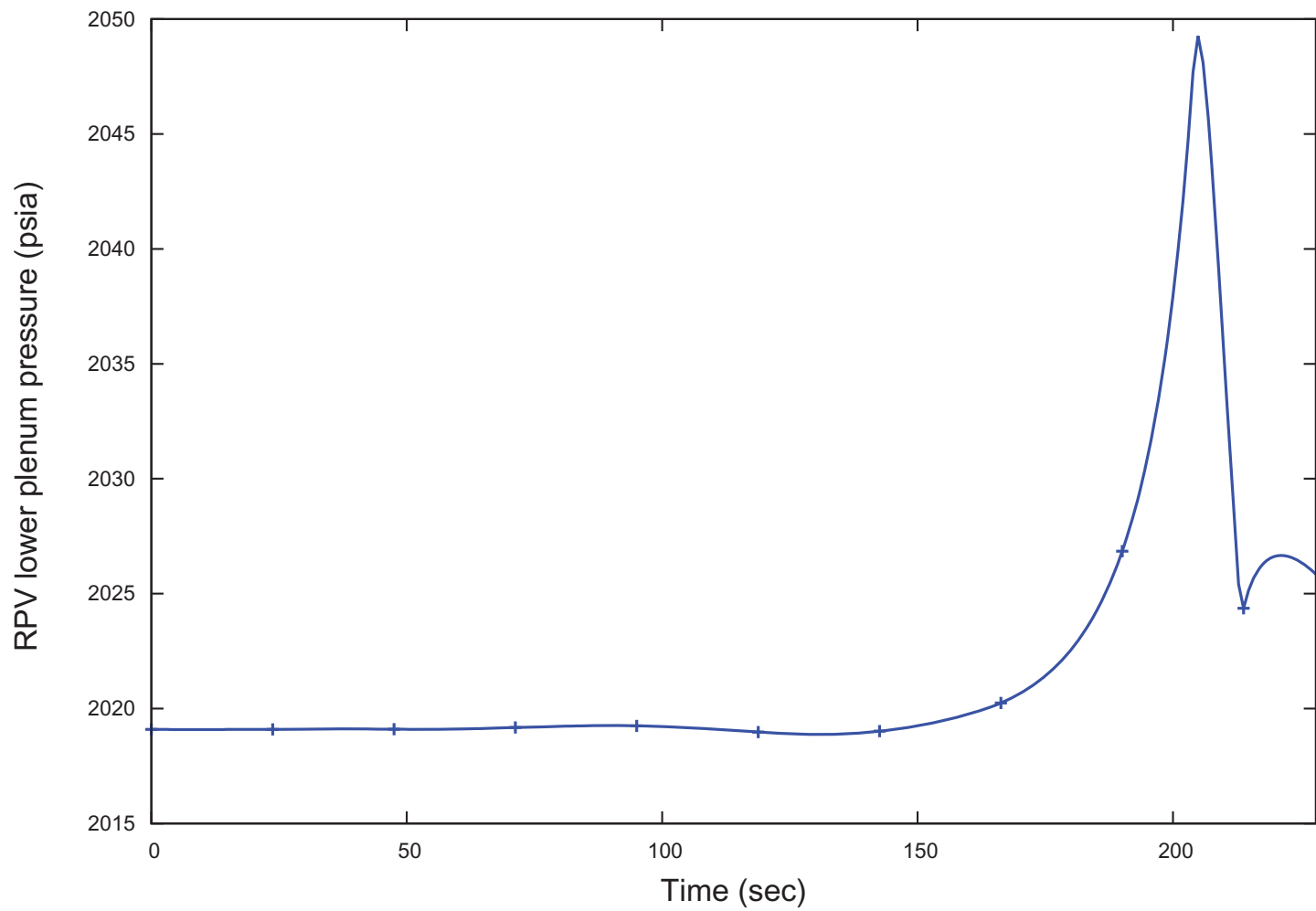


**Figure 15.4-5: Withdrawn Control Rod Assembly Reactivity Insertion
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)**

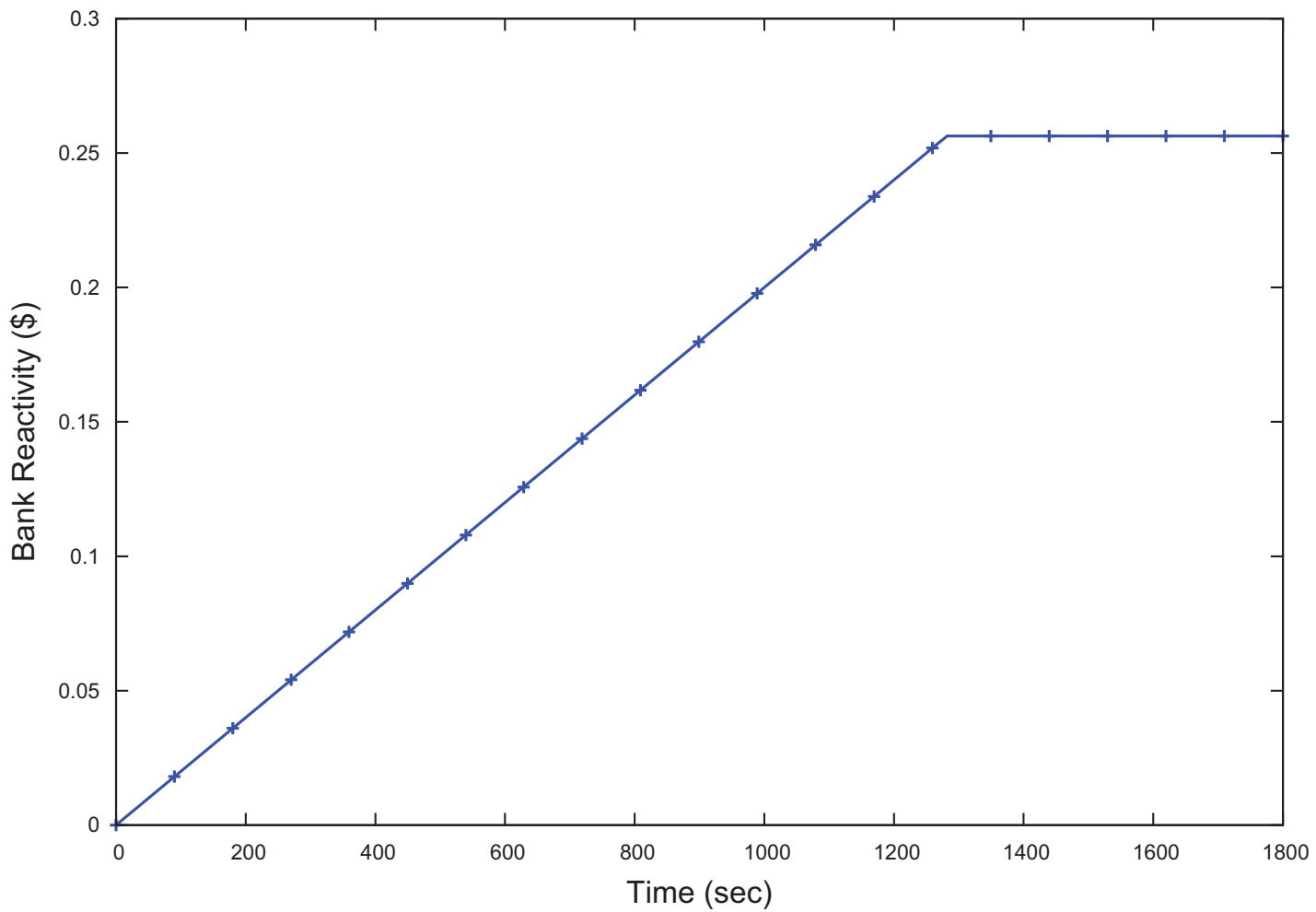


Figure 15.4-6: Reactor Power
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)

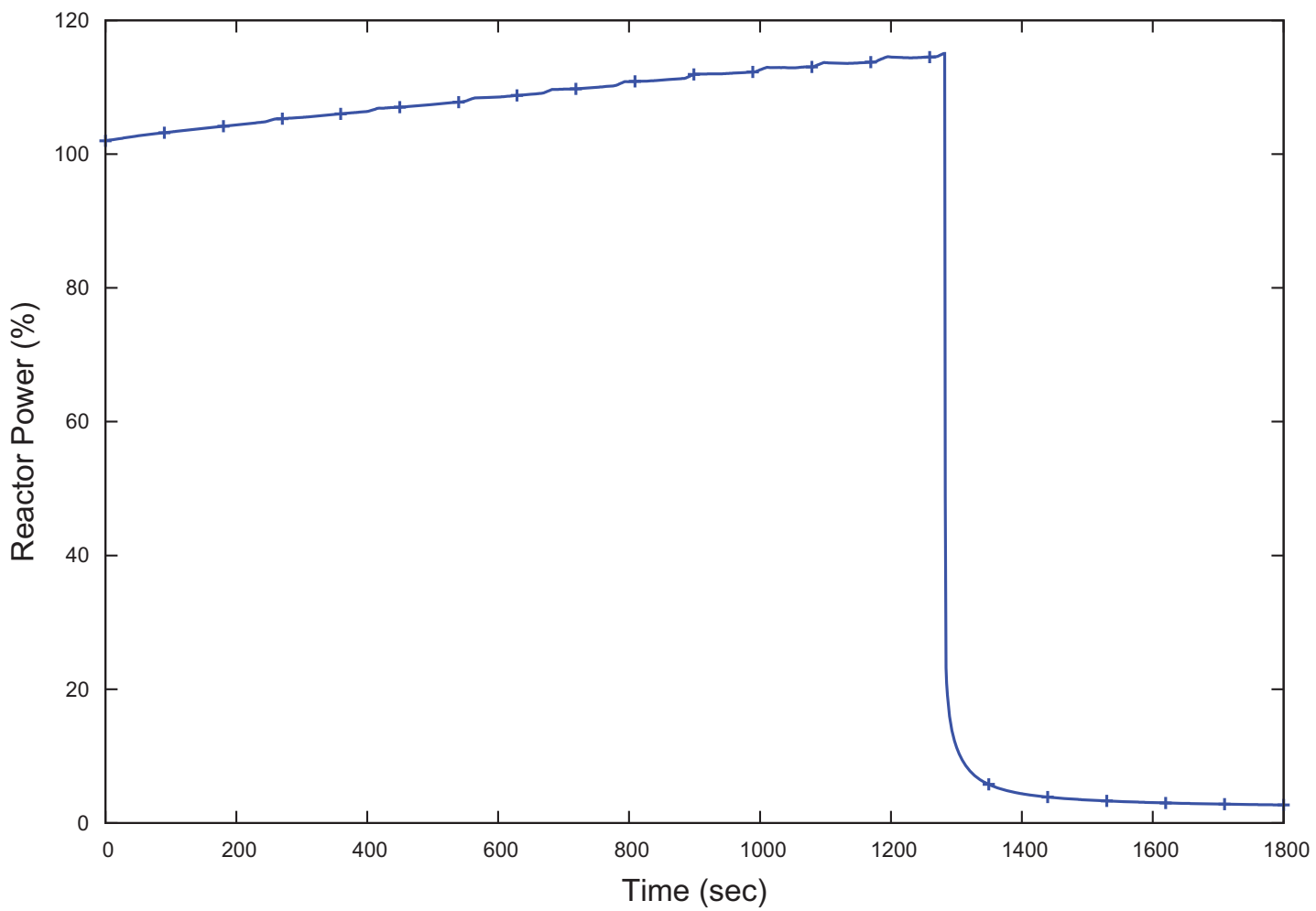


**Figure 15.4-7: Reactor Coolant System Pressure for Limiting Minimum Critical Heat Flux Ratio Case
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)**

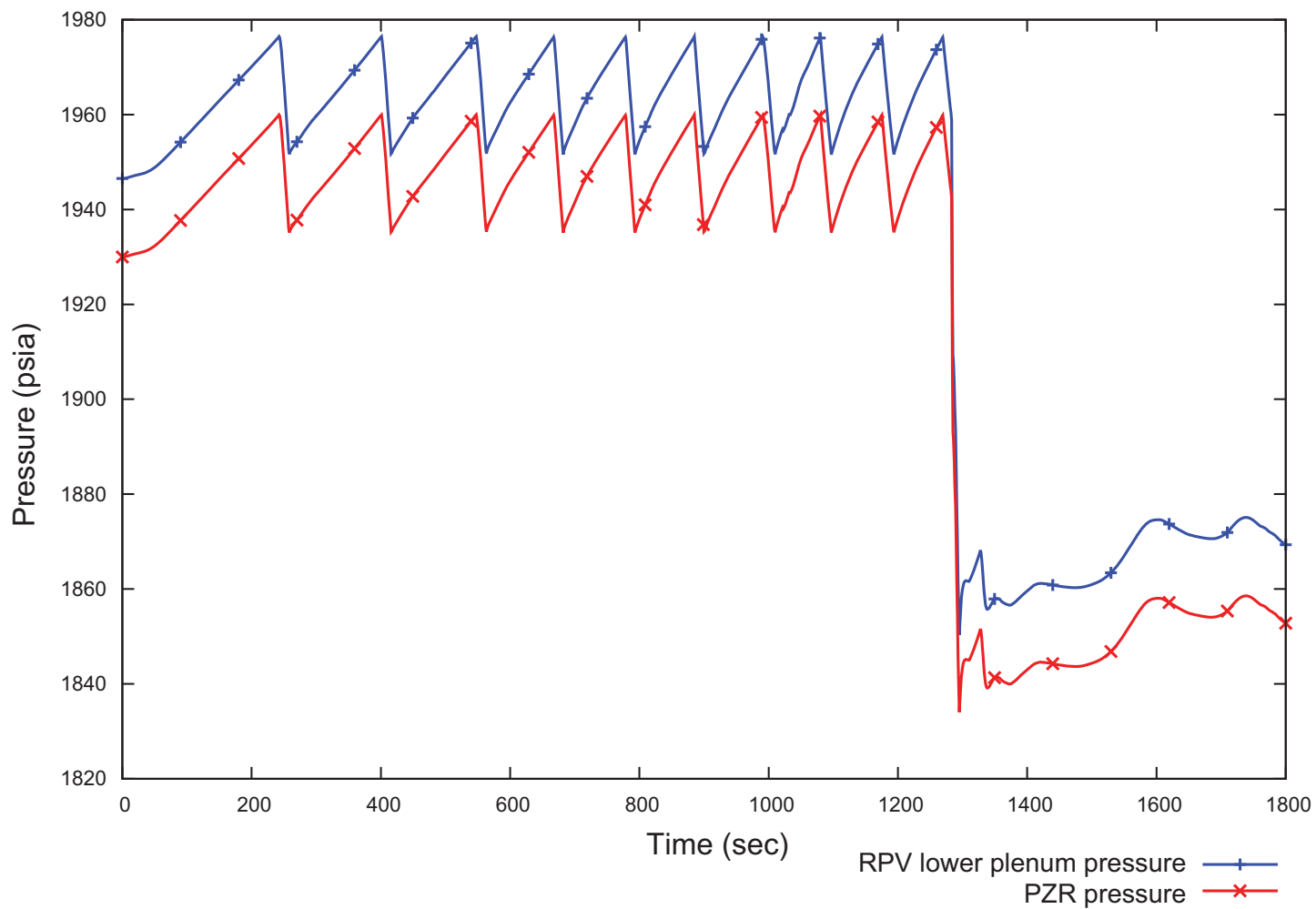


**Figure 15.4-8: Reactor Coolant System Temperatures
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)**

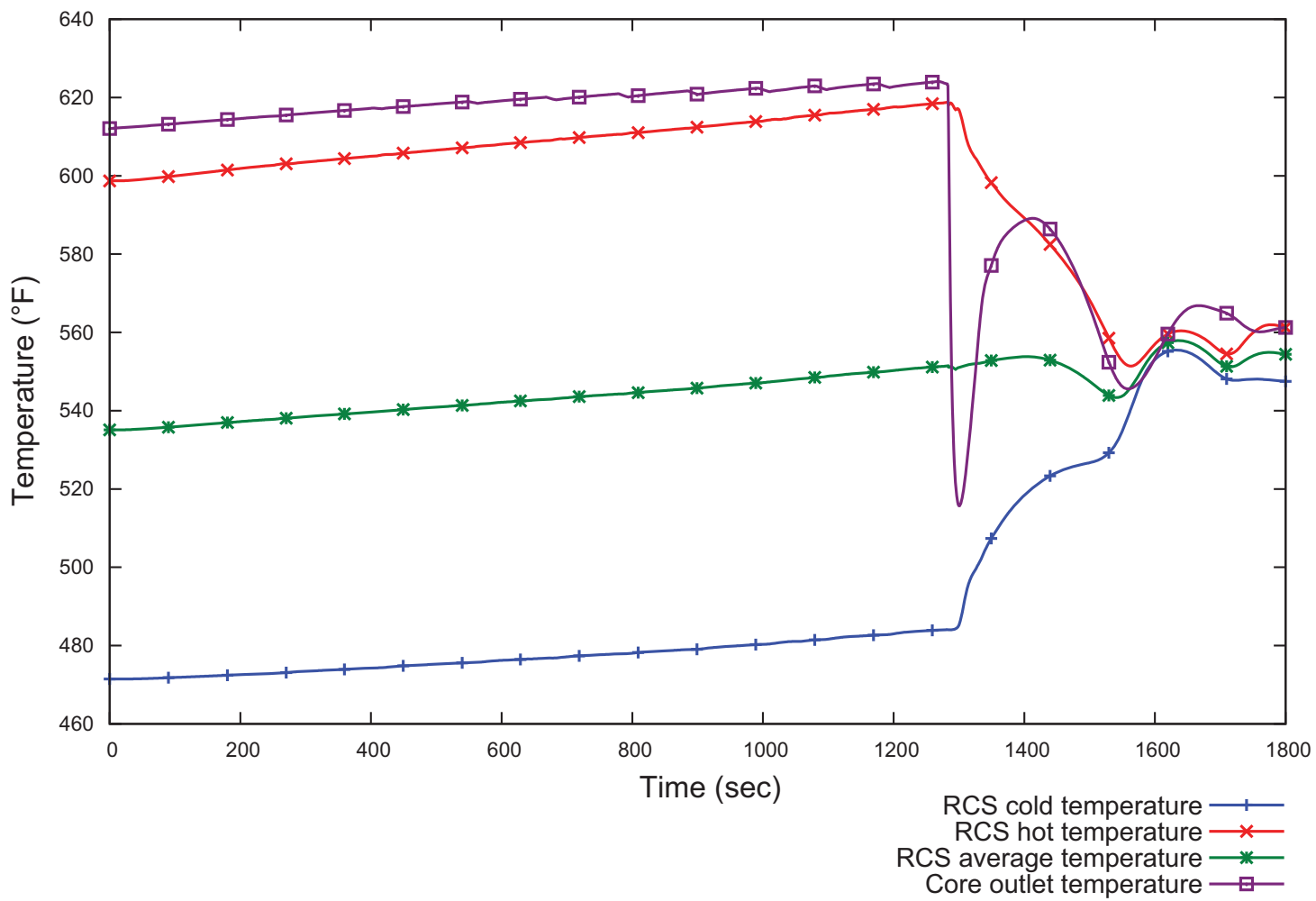


**Figure 15.4-9: Reactor Coolant System Flow
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)**

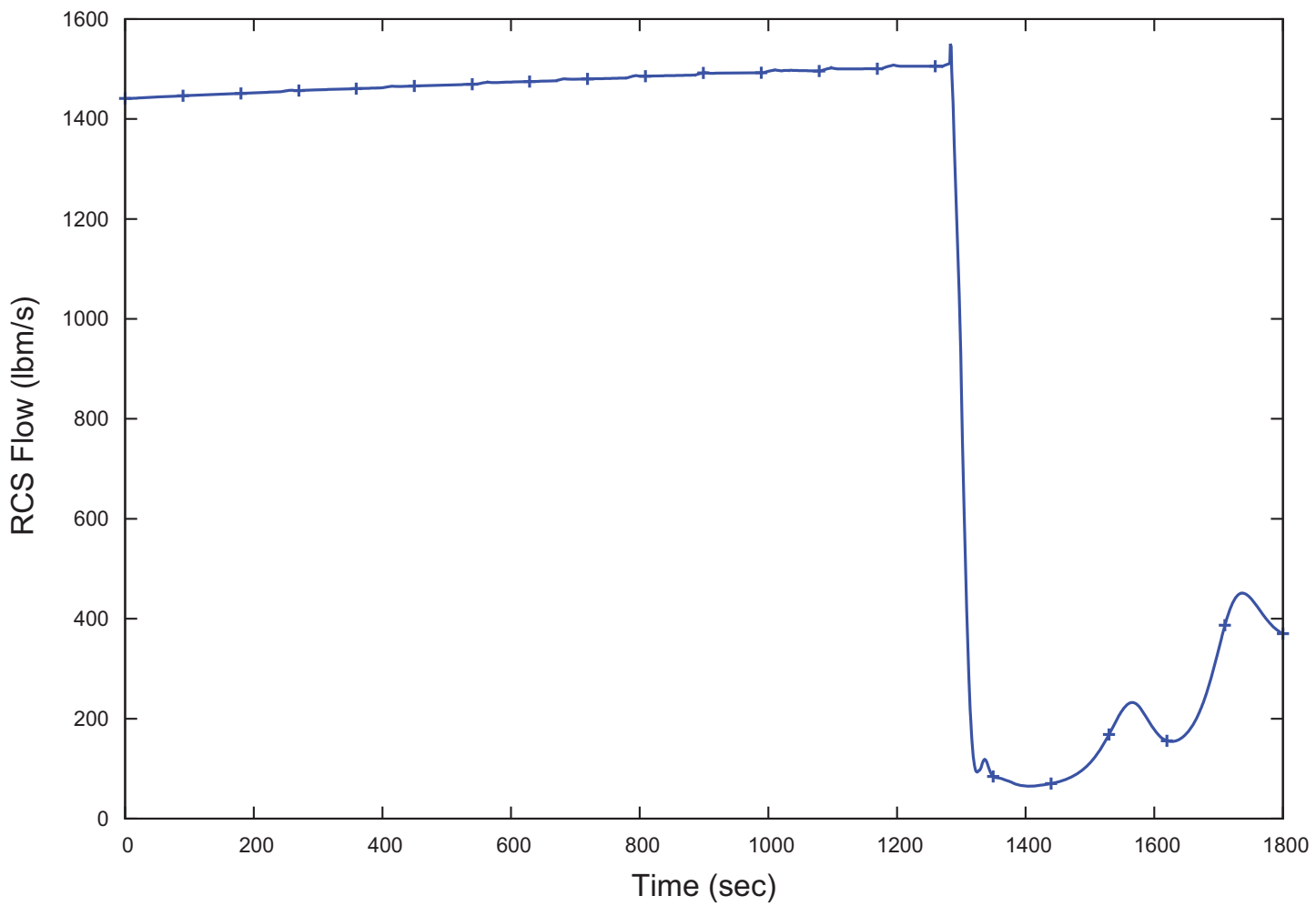


**Figure 15.4-10: Critical Heat Flux Ratio
(15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power)**

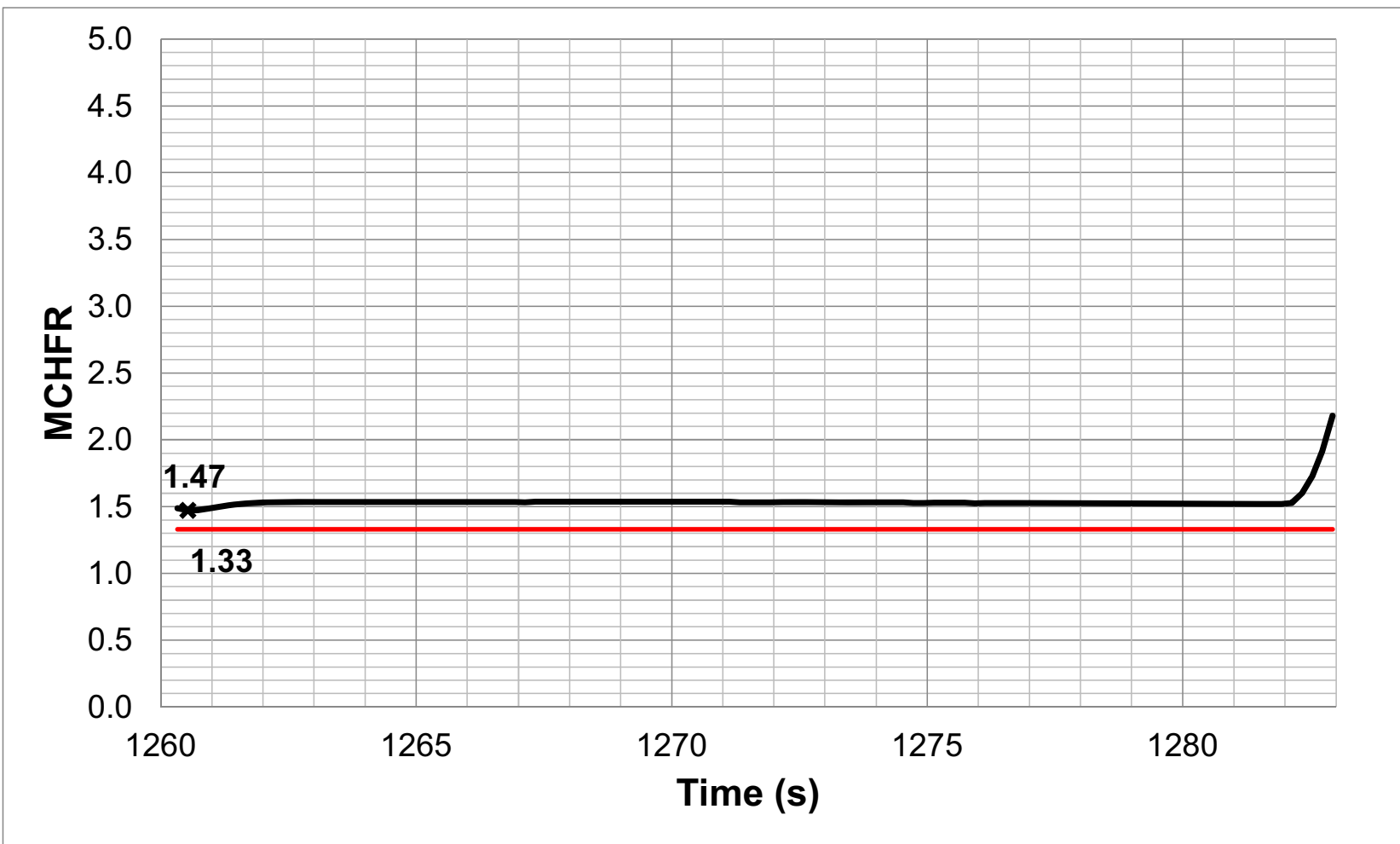


Figure 15.4-11: Reactivity
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

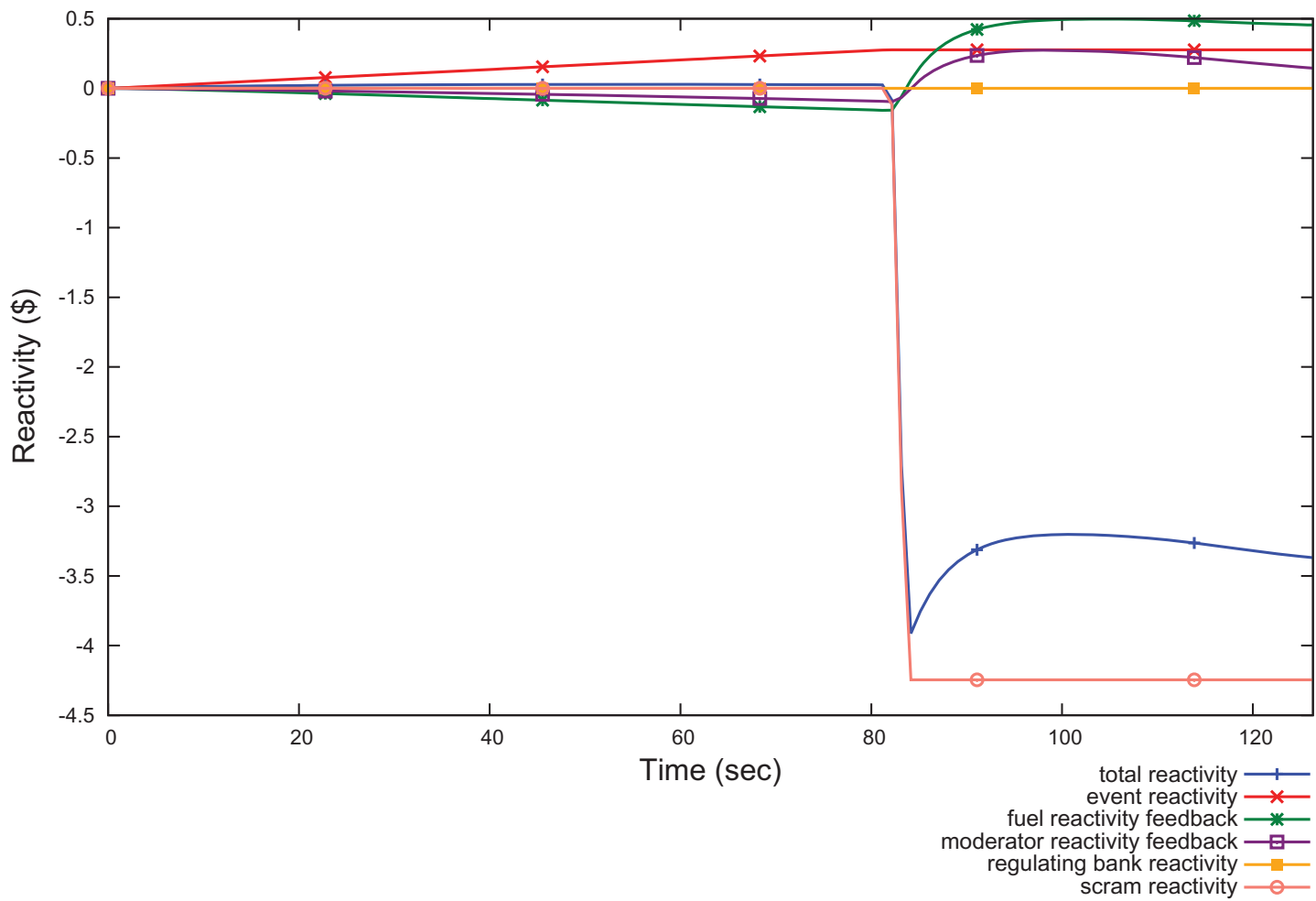


Figure 15.4-12: Reactor Power
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

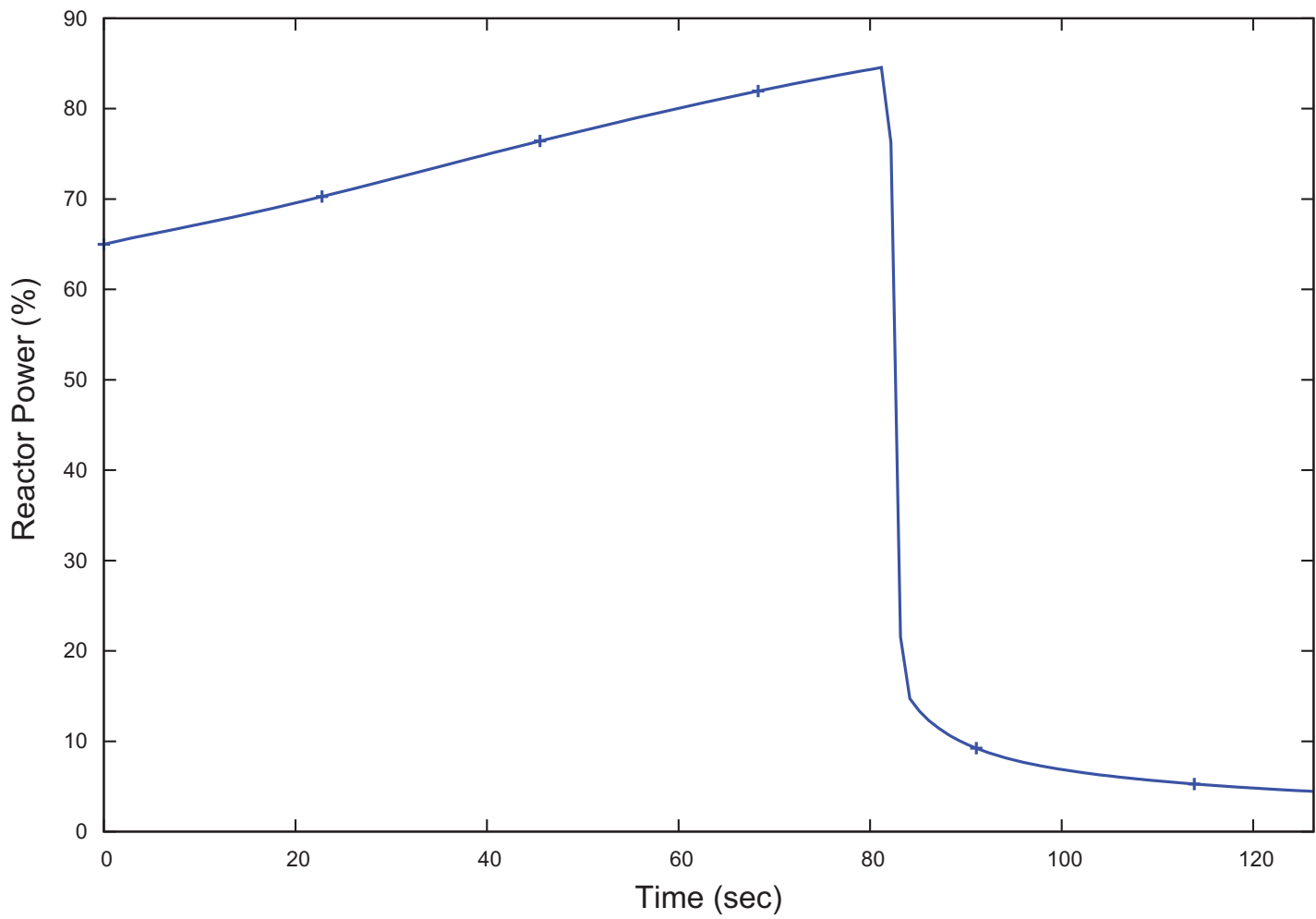


Figure 15.4-13: Reactor Coolant System Pressure for Limiting Minimum Critical Heat Flux Ratio Case (15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

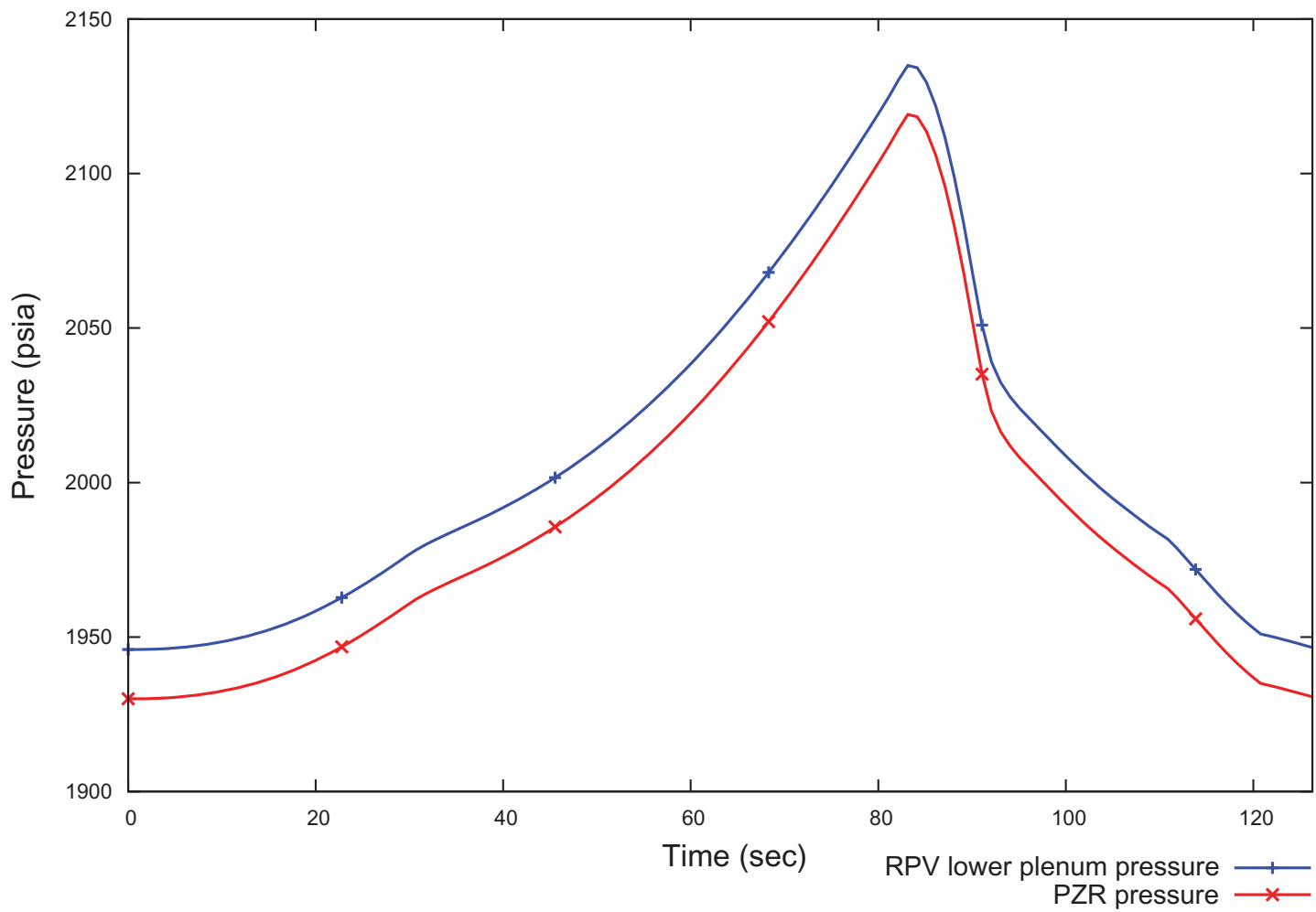


**Figure 15.4-14: Reactor Coolant System Temperatures
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)**

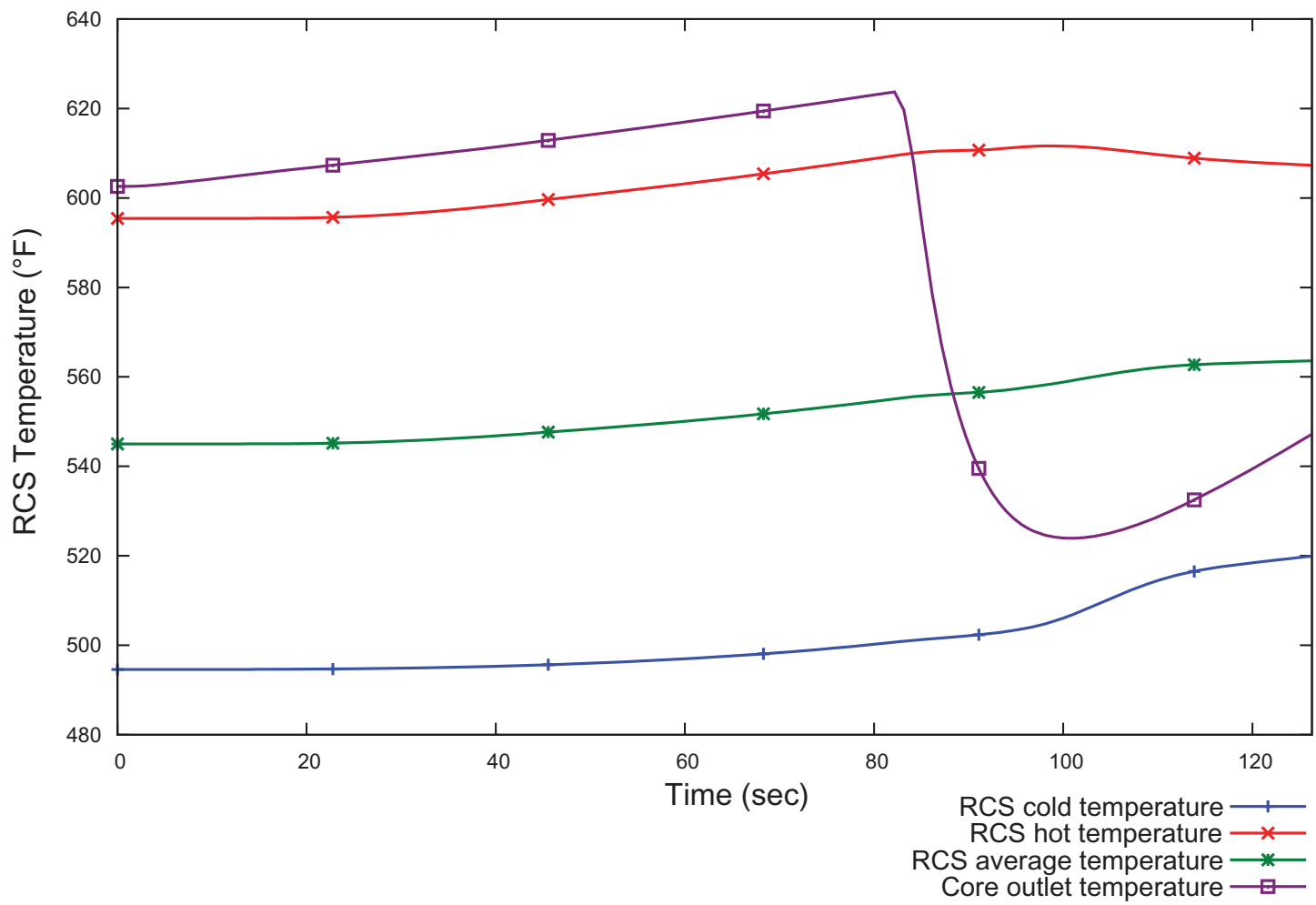


Figure 15.4-15: Reactor Coolant System Flow
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)

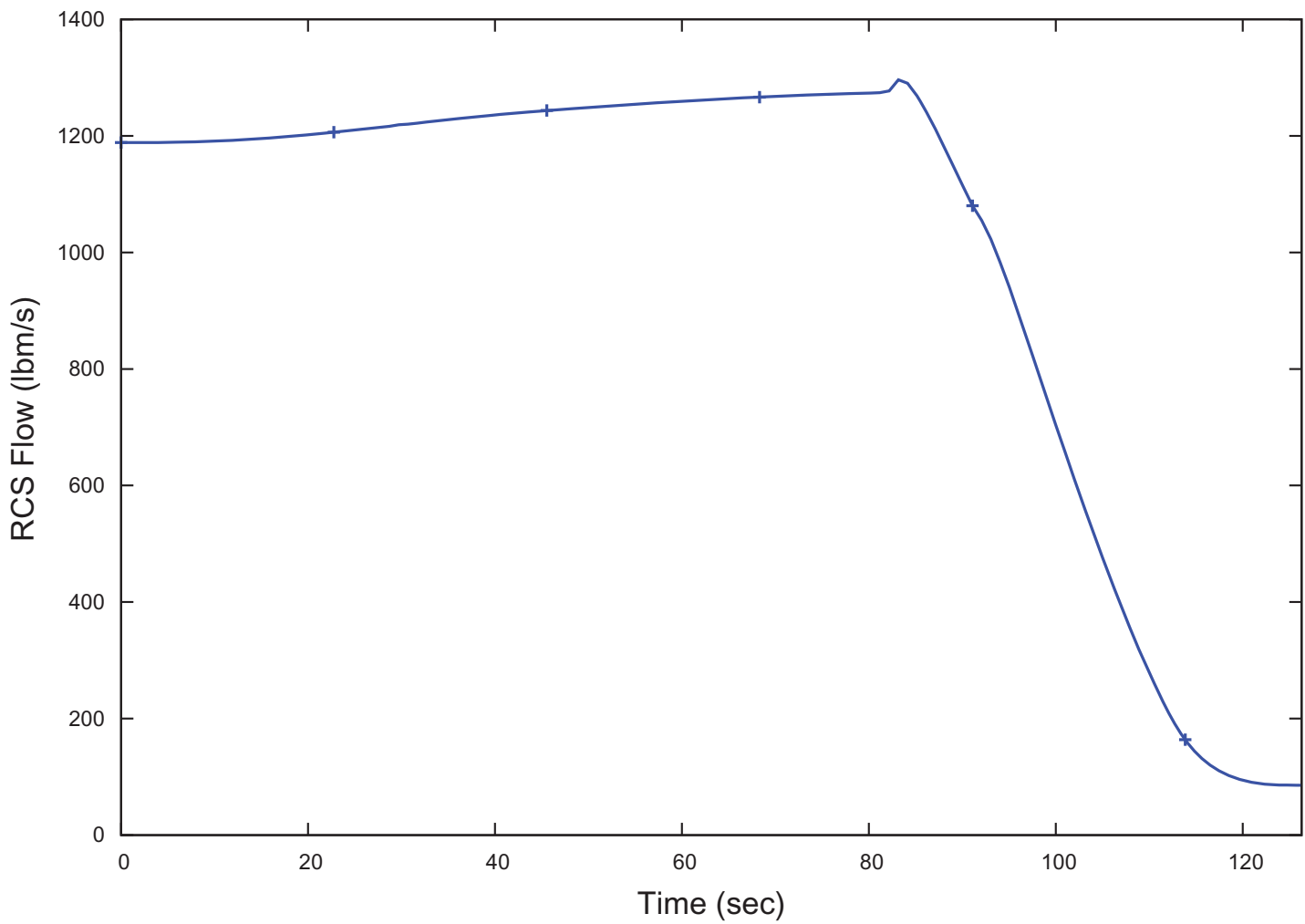


**Figure 15.4-16: Critical Heat Flux Ratio
(15.4.3 Control Rod Misoperation, Single Control Rod Assembly Withdrawal)**

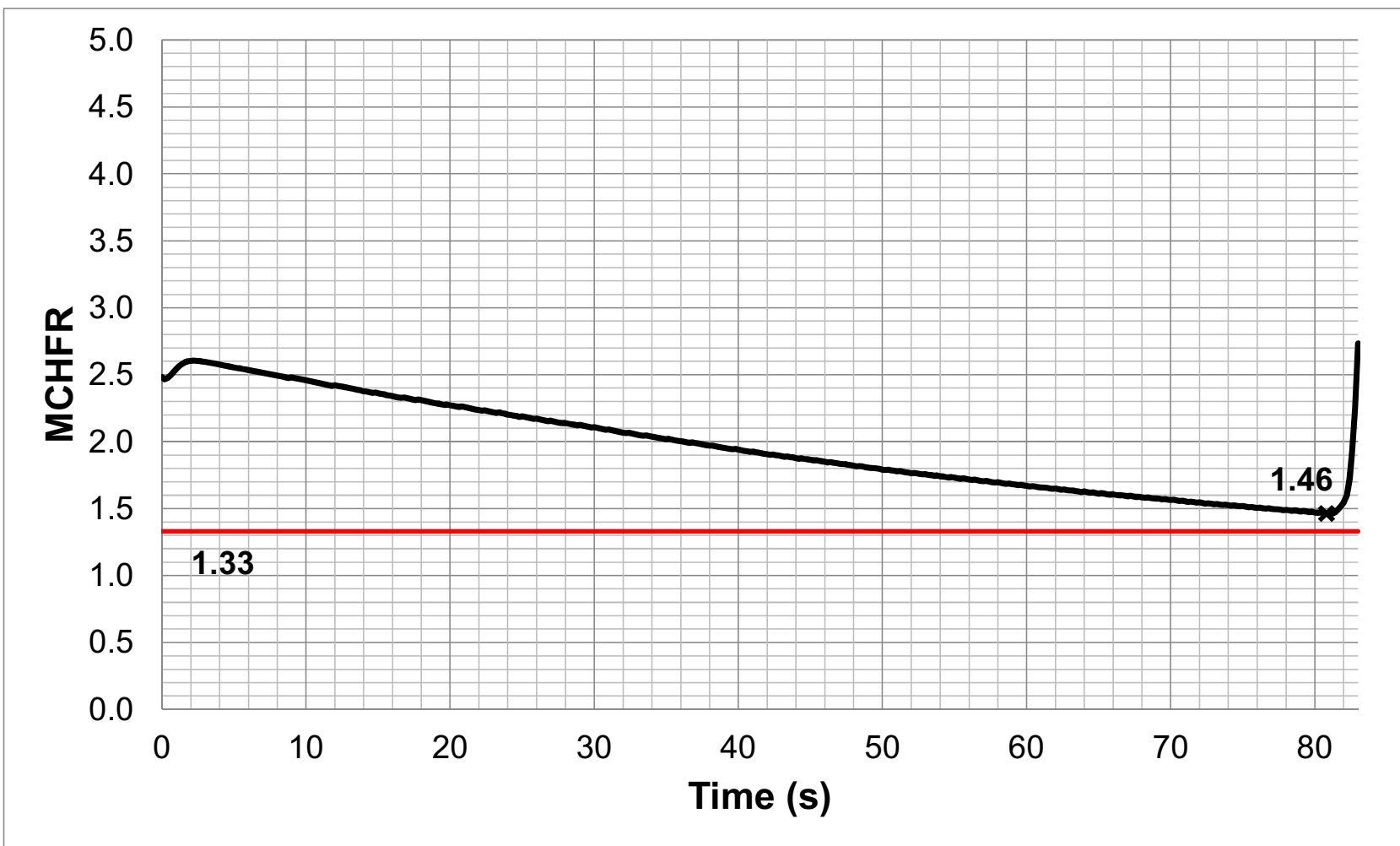


Figure 15.4-17: Spectrum of Fuel Misload Arrangements
(15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position)

Quarter Core						
			1	2	3	4
			5	6	7	8
			9	10	11	
			12	13		

Half Core						
		10	11			
		12	13	14		
		15	16	17	18	
		1	2	3	4	
		5	6	7		
		8	9			

Cross Core						
		13				
	14	15				
16	17	18				
				1	2	3
			4	5	6	7
			8	9	10	
			11	12		

**Figure 15.4-18: Peak Power Pulse at 0% Initial Power, End of Cycle
(15.4.8 Rod Ejection Accident)**

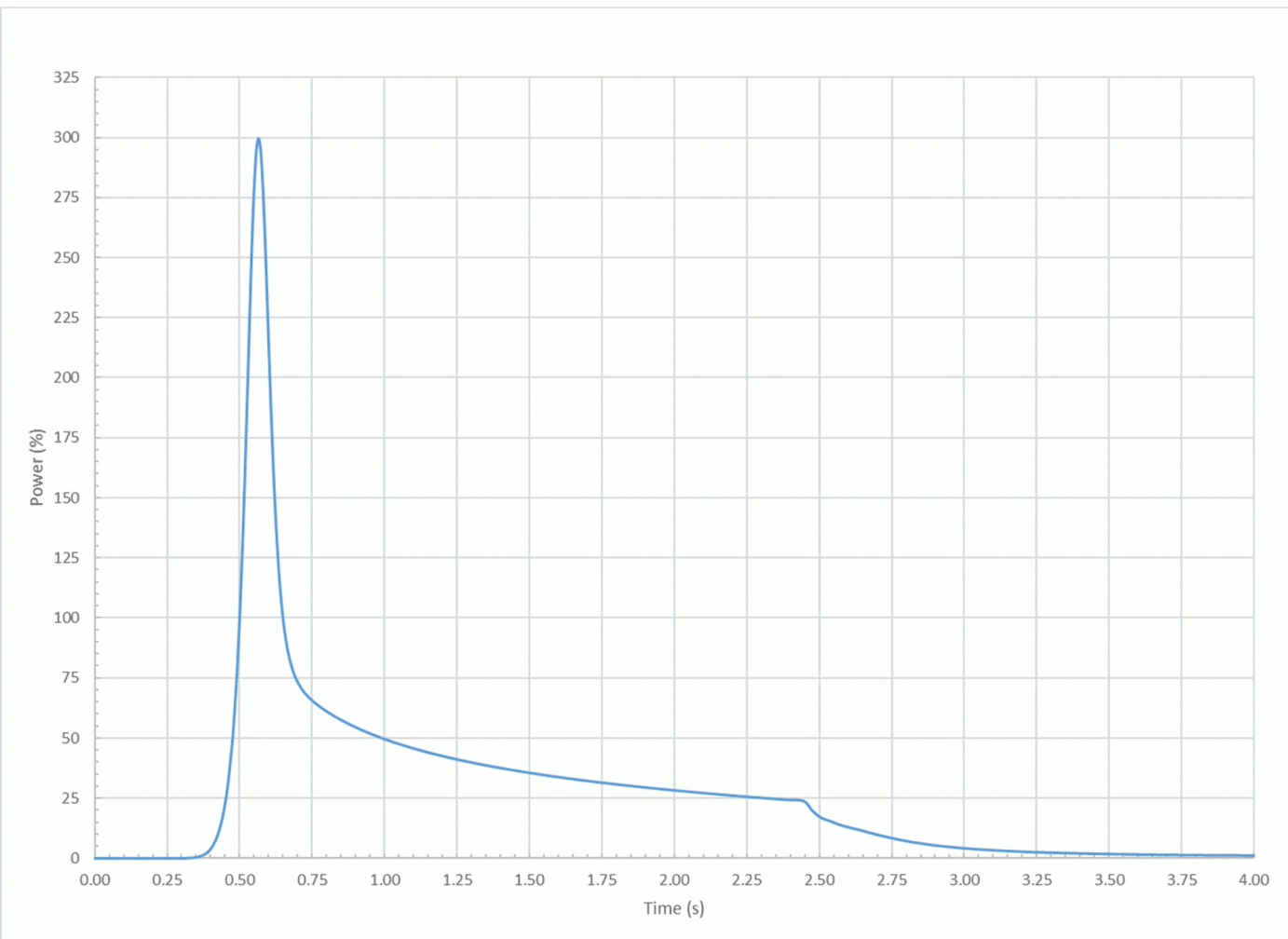
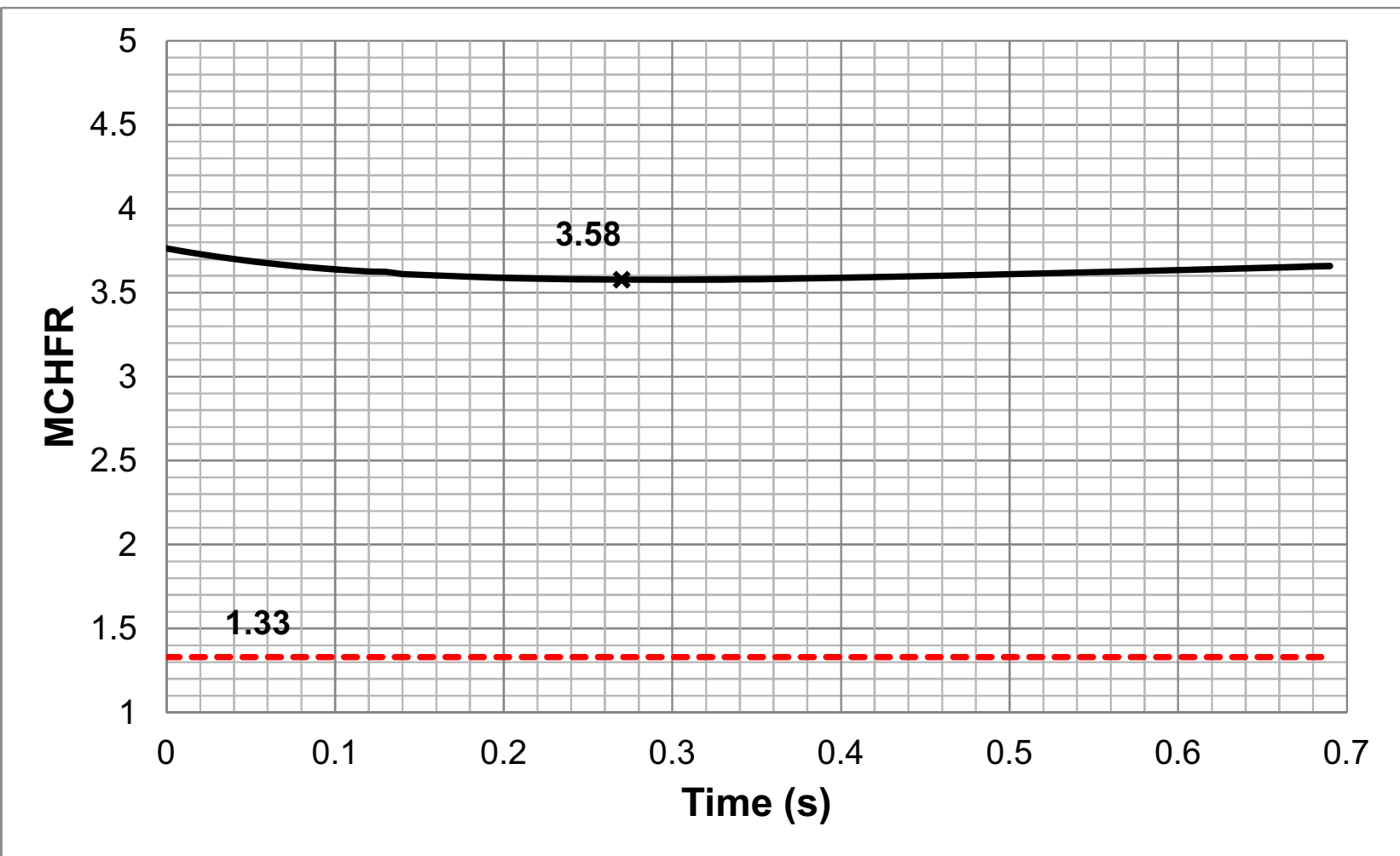


Figure 15.4-19: Critical Heat Flux Ratio
(15.4.8 Rod Ejection Accident)



15.5 Increase in Reactor Coolant Inventory

This section addresses design-basis events associated with a potential unplanned increase in reactor coolant system (RCS) inventory. In the NuScale Power Plant US460 standard design, such an event could occur due to unplanned chemical and volume control system (CVCS) operation. Depending on the boron concentration, temperature of injected water, and automatic control system operation, such events have the potential for causing a positive reactivity addition.

15.5.1 Chemical and Volume Control System Malfunction

15.5.1.1 Identification of Causes and Accident Description

An increase in RCS inventory could be caused by a malfunction of the CVCS makeup pumps resulting in an excessive increase in RCS inventory. Borated water, at the same concentration as the RCS, generates a reactor trip on high pressurizer pressure or level.

The increase in RCS inventory event is classified as an anticipated operational occurrence, as indicated in Table 15.0-1.

Adding water to the RCS at a lower boron concentration is a boron dilution event, as addressed in Section 15.4.6.

15.5.1.2 Sequence of Events and Systems Operation

The CVCS is a nonsafety-related system that allows flow to be recirculated from the RCS through regenerative and non-regenerative heat exchangers and demineralizers. The CVCS discharge line takes suction from the RCS downcomer region below the steam generator. The CVCS injection line discharges inside the riser above the core. Some or all of the discharge flow from the RCS can be diverted to the liquid radioactive waste system through the CVCS discharge line. The remaining flow is returned to the RCS by the recirculation pumps through the injection line and pressurizer spray. In addition, borated water, demineralized water, or both can be added to the injection line from two 20 gpm makeup pumps. The system is maintained pressurized at approximately RCS pressure throughout the recirculation cycle. During normal recirculation operation, the volume of RCS withdrawn from the RCS through the CVCS discharge line is equal to the amount pumped through the CVCS injection line and pressurizer spray lines, unless water from the makeup pumps is added.

The increase in RCS inventory event is terminated by the module protection system (MPS) actuation of a reactor trip and CVCS isolation. The MPS signal is initiated by a high pressurizer level or high pressurizer pressure condition. The CVCS is isolated from the RCS by redundant safety-related containment isolation valves located on both the exit and entry containment penetrations. These valves are isolated on a containment isolation signal or a loss of augmented DC power (EDAS).

For the purposes of this analysis, the amount of water directed to letdown from the discharge line is assumed to be zero gpm and the maximum makeup of 40 gpm, with the same boron concentration as the RCS, is added to the injection line at the beginning of the event.

All scenarios assume the availability of AC power and DC power (EDNS and EDAS). The CVCS cannot function without AC power or EDNS and the CVCS flow pathways are isolated on a loss of EDAS. Therefore, the loss of any of these power supplies would terminate the CVCS flow addition event.

The isolation valve closure isolates the secondary side, resulting in a loss of secondary side heat sink until the decay heat removal system (DHRS) actuates and begins cooling.

For cases where pressurizer spray is enabled, the reactor trip occurs on high pressurizer level with no significant power or pressure transient. Overpressurization occurs after the reactor is tripped, due to the large inventory. The peak pressure is mitigated by isolation of the secondary system and DHRS actuation. The RPV peak pressure remains below the opening setpoint of a reactor safety valve (RSV). For cases where pressurizer spray is disabled, pressurizer pressure gradually increases until reactor trip occurs on high pressurizer pressure. Following reactor trip, pressure increases in several cycles until an RSV lifts and pressure decreases. The sequence of events for the increase in RCS inventory for the limiting RCS pressure case is provided in Table 15.5-1.

There are no single failures that would result in a more serious outcome for the increase in RCS inventory events. The diversity, redundancy, and independence of the MPS ensure the system performs its intended function despite a single failure. The redundancy and passive nature of the DHRS ensure that the system performs its intended function despite a single failure. The RSVs and containment isolation valves also have redundancy to accommodate a single failure. No cases involving loss of AC or DC power are evaluated as the loss of either AC or DC power terminates this event. No operator action is credited to mitigate this event.

15.5.1.3 Thermal Hydraulic and Subchannel Analyses

15.5.1.3.1 Evaluation Models

The thermal hydraulic analysis of the Nuscale Power Module response to an increase in RCS inventory event is performed using NRELAP5. The NRELAP5 model is based on the design features of the NuScale Power Module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel critical heat flux (CHF) analysis.

The subchannel CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal-hydraulic analysis under normal operating conditions, operational transients, and events

of moderate severity. Section 15.0.2 provides a discussion of the VIPRE-01 code and evaluation model.

15.5.1.3.2 Input Parameters and Initial Conditions

The initiating event for the increase in RCS inventory is the malfunction of the CVCS that results in a maximum CVCS makeup flow of 40 gpm with no letdown.

The following initial conditions are applied to analyzed cases to provide the most limiting initial conditions:

- Reactor power is biased high (102 percent).
- Fuel temperature is biased low.
- End of cycle exposure is applied.
- Decay heat is biased high.
- Pressurizer pressure is biased high.
- RCS flow is biased low.
- Initial reactor pool temperature is biased high (120 degrees F).

The following initial conditions are varied to determine a limiting case for each acceptance criterion:

- RCS temperature
- pressurizer level
- makeup flow rate
- makeup temperature
- pressurizer spray availability

Table 15.5-2 summarizes the input parameters and biases.

15.5.1.3.3 Results

Sensitivity studies on an increase in reactor coolant inventory are performed with biased conditions to determine a limiting case for each acceptance criterion.

The limiting results for the CVCS malfunction are shown in Table 15.5-3. Figure 15.5-1 through Figure 15.5-7 show the pressurizer level, reactor power, total reactivity, RCS average temperature, RCS mass addition, RCS flow, and DHRS flow, respectively, for the maximum RCS pressure case. The limiting RCS pressure meets the acceptance criteria and is shown in Figure 15.5-8.

Figure 15.5-9 shows the maximum steam generator pressure for the limiting RCS pressure case. The most limiting steam generator pressure case results in a secondary pressure that is higher than shown in Figure 15.5-9 but still

meets the acceptance criteria as shown in Table 15.5-3. The limiting minimum critical heat flux ratio (MCHFR) meets the acceptance criteria and is shown on Figure 15.5-10 for the limiting MCHFR case. This event is not limiting for any of the three anticipated operational occurrence acceptance criteria. This event does not lead to a more serious fault condition.

As the predicted MCHFR is greater than the acceptance criteria for this event, no fuel failure is predicted to occur. As a result, the radiological consequences for this event are acceptable as no fuel failure or release of primary coolant outside containment is predicted to occur.

15.5.1.4 Radiological Consequences

The radiological consequences of an increase in RCS inventory event are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.5.1.5 Conclusions

The acceptance criteria are met for the inadvertent increase in RCS inventory event. A comparison of the limiting pressure values and the associated acceptance criteria is shown in Table 15.5-3.

Acceptance criteria for the increase in reactor coolant inventory event are as follows:

- 1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
 - The limiting RCS pressure for this event, as shown in Table 15.5-3, is below 110 percent of the design value for the reactor coolant system.
 - The limiting steam generator pressure, as shown in Table 15.5-3, is below 110 percent of the design value for the main steam system up to the main steam isolation valves.
- 2) Fuel cladding integrity should be maintained by ensuring the MCHFR remains above the CHF analysis limit based on acceptable correlations.
 - The MCHFR for this event, as shown in Table 15.5-3, is above the CHF analysis limit.
- 3) An incident of moderate frequency should not generate a more serious plant fault.
 - The analyses presented for this event shows that a safe stabilized condition is reached, and the acceptance criteria are met.
- 4) The guidance provided in Regulatory Guide (RG) 1.105, "Instrument Spans and Setpoints," can be used to analyze the impact of the instrument spans and setpoints on the plant response to the type of transient addressed in this DSRS section, in order to meet the requirements of GDCs 10 and 15.

- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.
- 5) The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A to 10 CFR Part 50, shall be identified as assumed in the analysis and shall satisfy the positions of RG 1.53.
- No limiting single failure is identified that provided more limiting RCS pressure, steam generator pressure, or MCHFR. Results from a single failure scenario do not challenge the identified limits.
- 6) The guidance provided in SECY-77-439 and SECY-94-084 with respect to the consideration of the performance of nonsafety-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.
- The inputs and assumptions for the operation of nonsafety-related systems and single failures as discussed in Section 15.5.1.2 and Section 15.5.1.3 ensure the guidance provided is met.

**Table 15.5-1: Sequence of Events - Chemical and Volume Control System
Malfunction - Limiting Reactor Coolant System Pressure Case**

Event	Time [s]
Makeup initiation	0
RTS and DHRS actuation (following high pressurizer pressure signal)	144
Secondary system isolation	144
Maximum secondary pressure	223
Maximum primary pressure	2534
CVCS isolation (following high pressurizer level signal)	2539

Table 15.5-2: Initial Conditions - Chemical and Volume Control System Malfunction

Case	PZR Level [%] ¹	RCS Temperature [°F] ¹	Makeup Temperature [°F] ¹	Makeup Flow [gpm]	Pressurizer Spray
Limiting RCS pressure	52 (-8)	535 (-5)	150 (+90)	40	Off
Limiting SG pressure	52 (-8)	545 (+5)	40 (-20)	25	Off
Limiting MCHFR	52 (-8)	545 (+5)	40 (-20)	40	On

¹ Biases are in parentheses.

Table 15.5-3: Summary of Results - Chemical and Volume Control System Malfunction

Acceptance Criteria	Limit	Analysis Value
Maximum RCS Pressure	2420 psia	2286 psia
Maximum SG Pressure	2420 psia	1357 psia
MCHFR	1.33	2.10

Figure 15.5-1: Pressurizer Level - Chemical and Volume Control System Malfunction - Limiting Reactor Coolant System Pressure Case

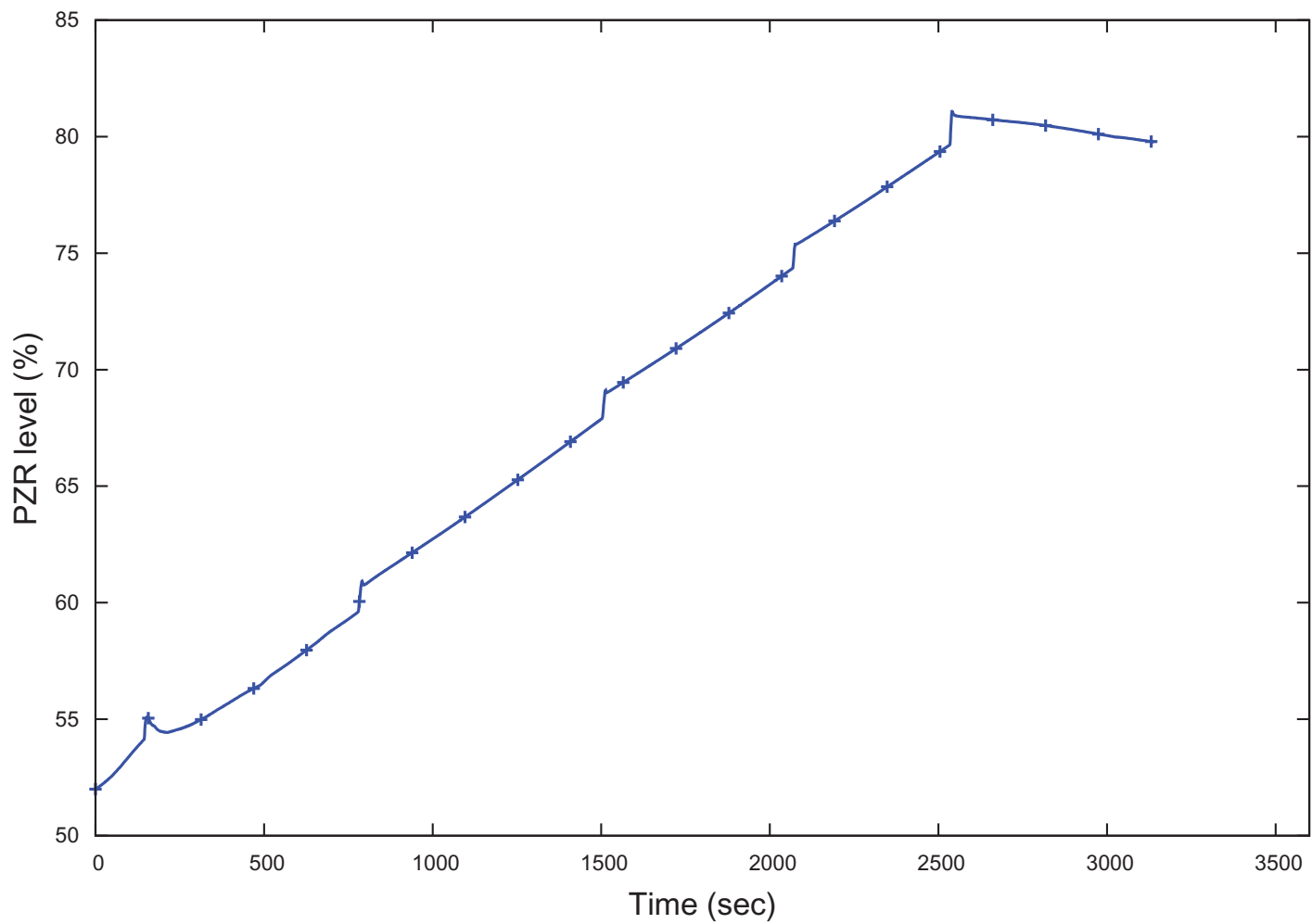
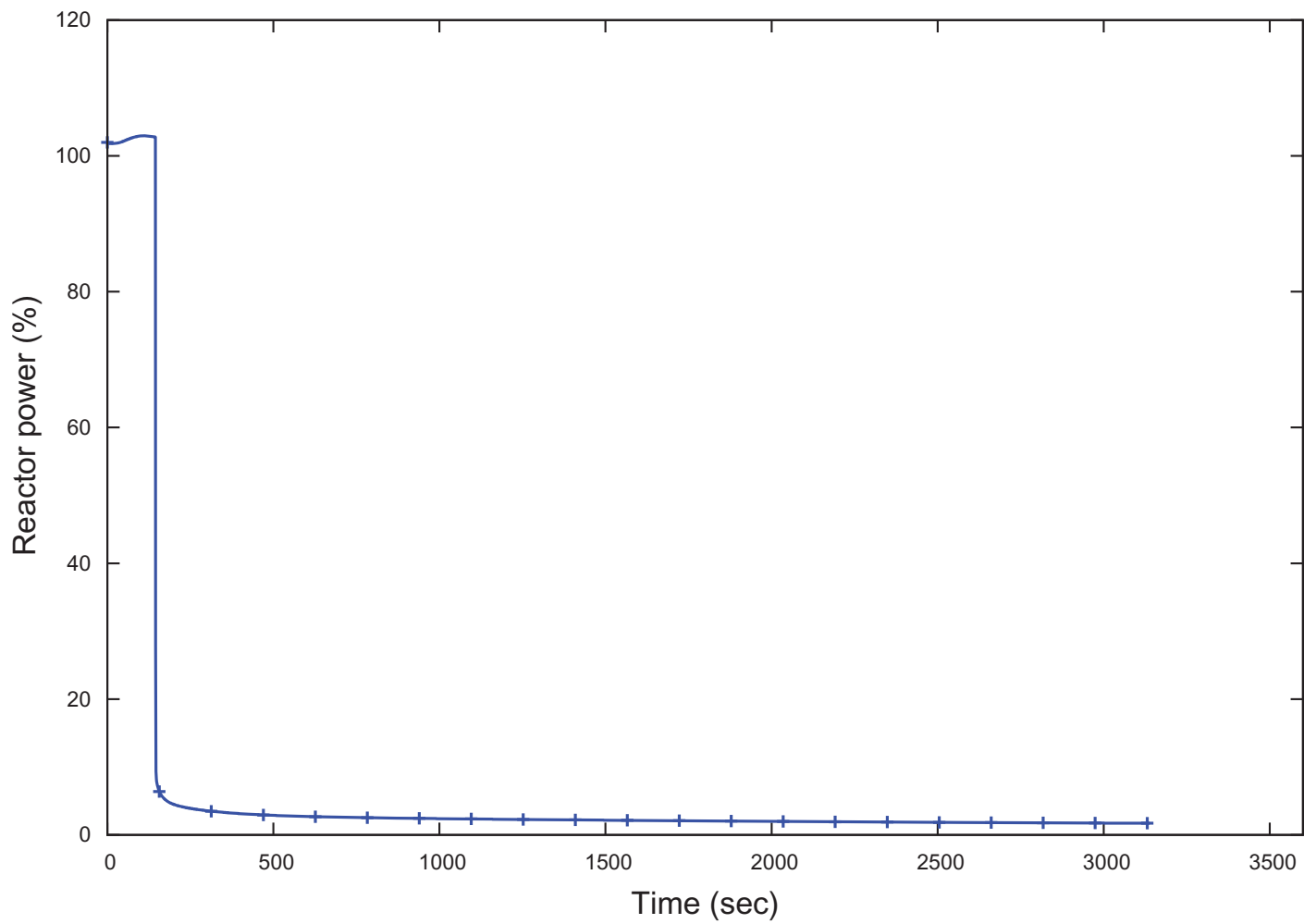


Figure 15.5-2: Reactor Power - Chemical and Volume Control System Malfunction - Limiting Reactor Coolant System Pressure Case



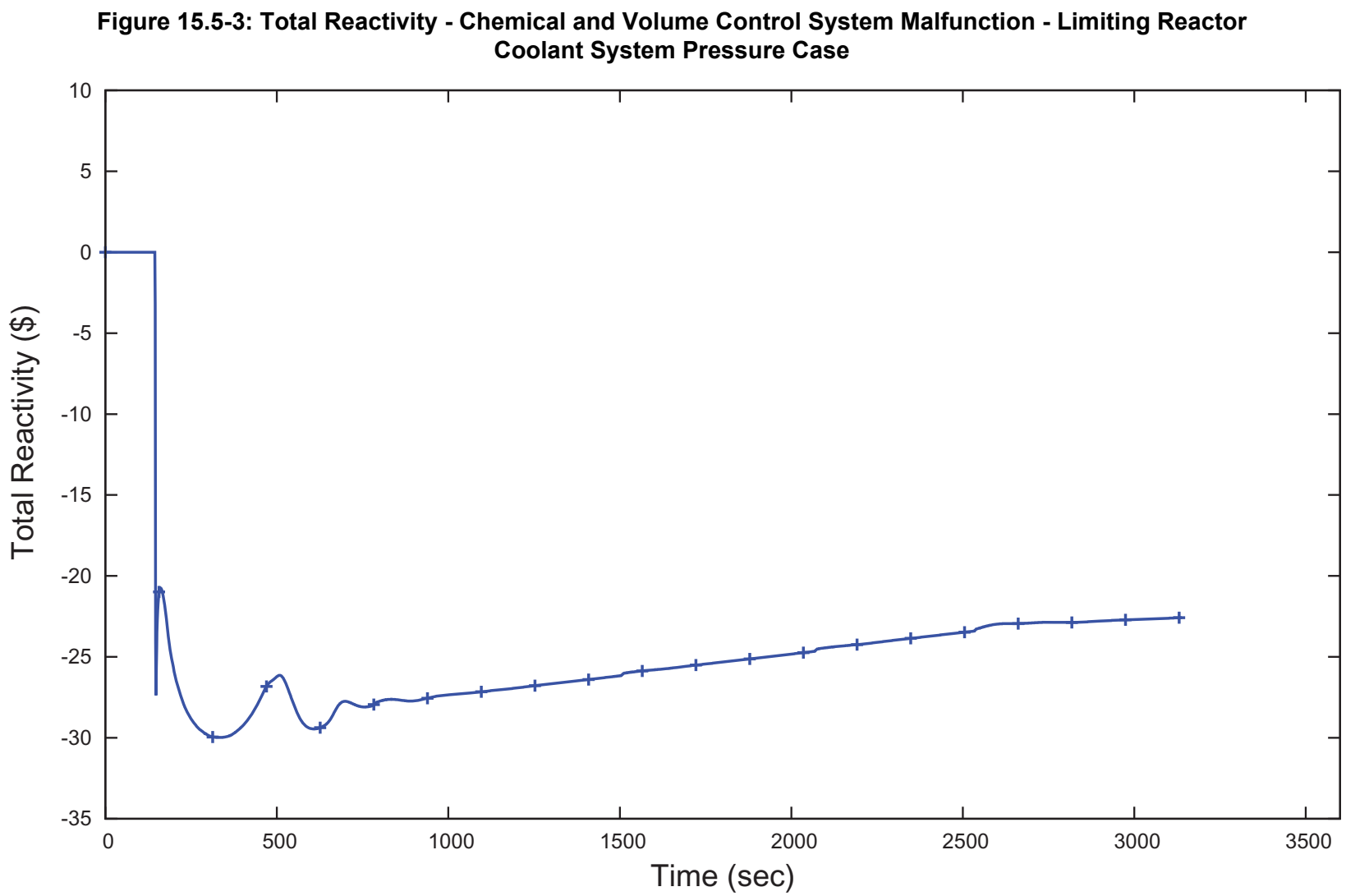


Figure 15.5-4: Reactor Coolant System Average Temperature - Chemical and Volume Control System Malfunction - Limiting Reactor Coolant System Pressure Case

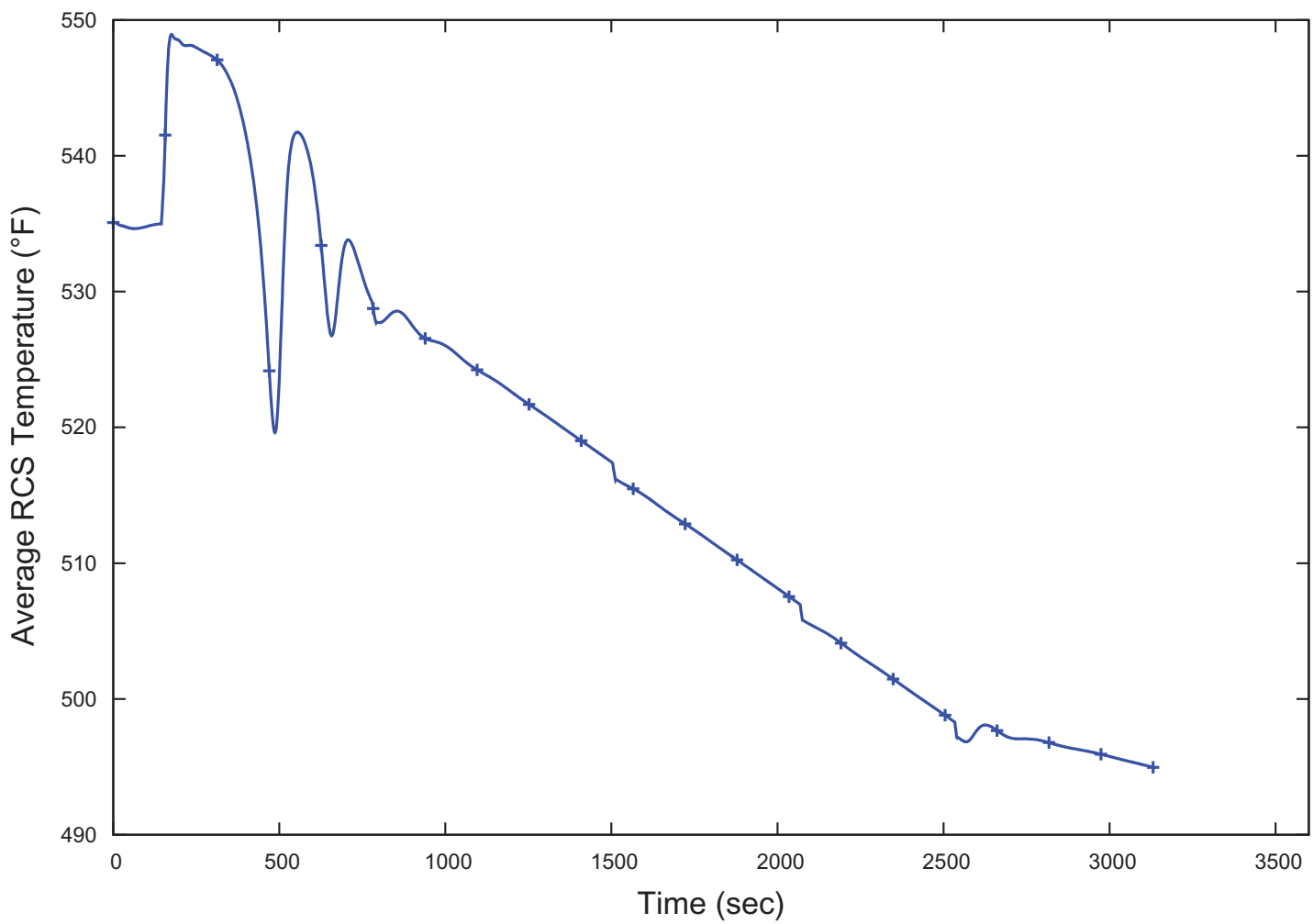


Figure 15.5-5: Makeup Flow Rate - Chemical and Volume Control System Malfunction - Limiting Reactor Coolant System Pressure Case

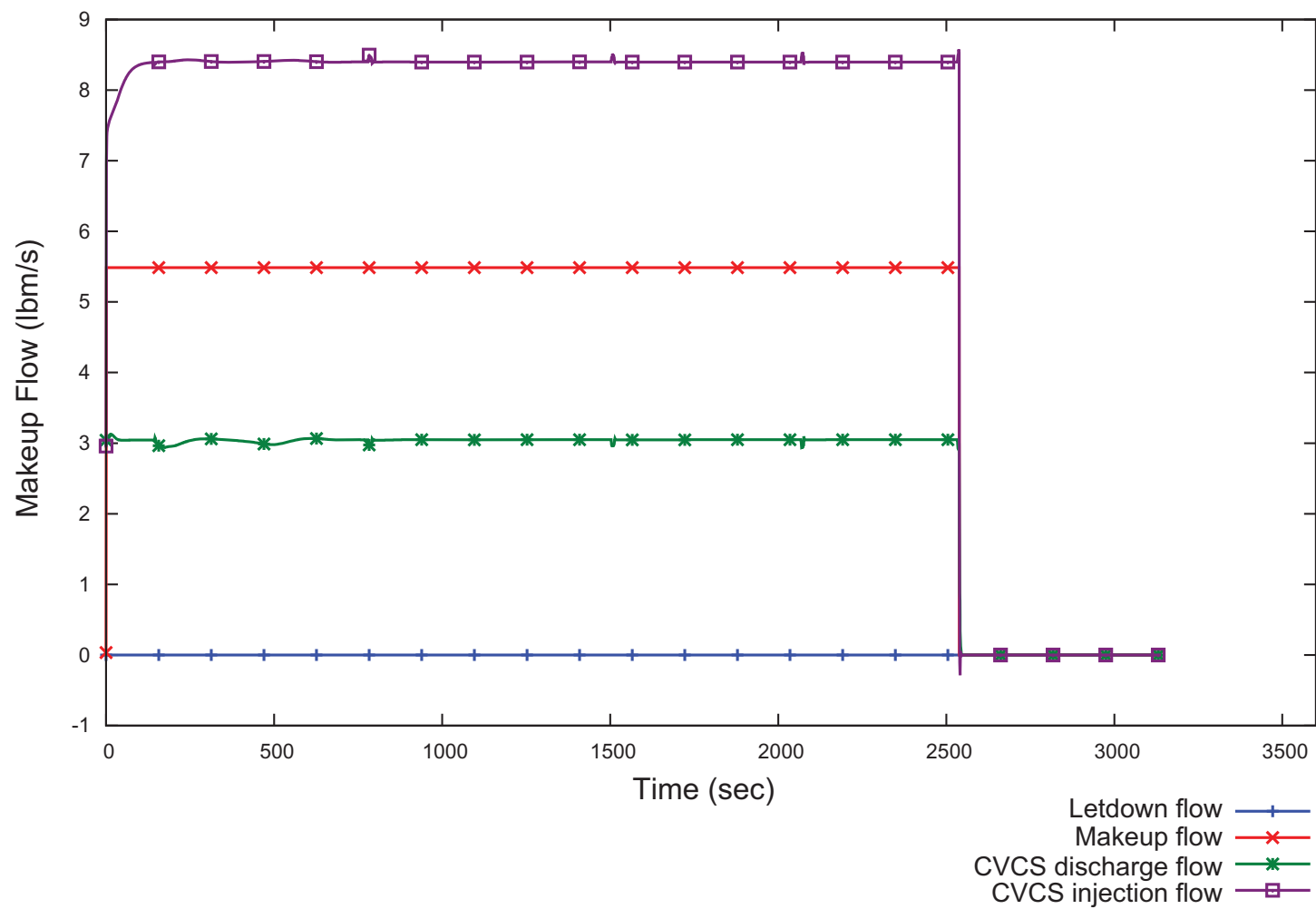


Figure 15.5-6: Reactor Coolant System Flow - Chemical and Volume Control System Malfunction - Limiting Reactor Coolant System Pressure Case

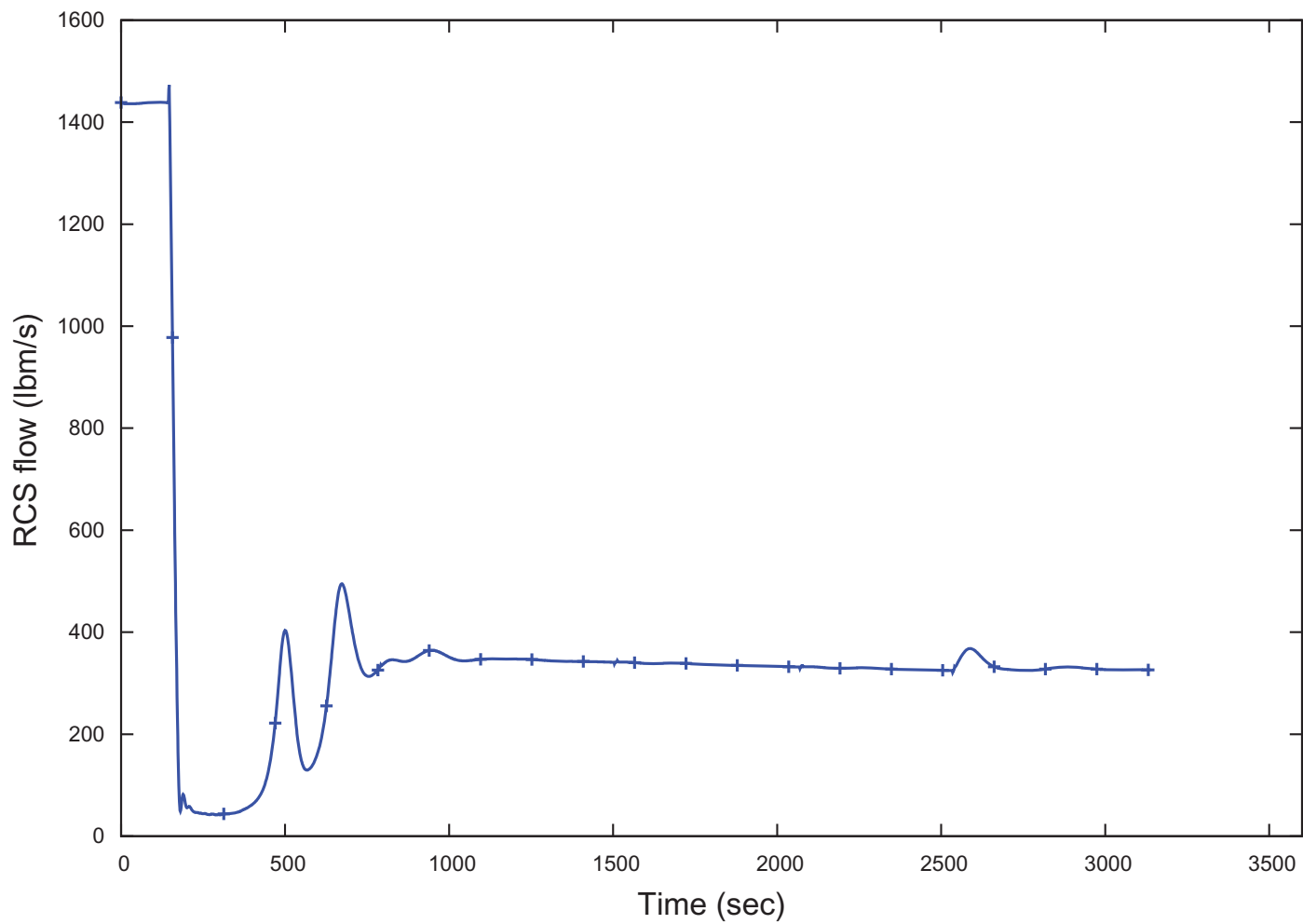


Figure 15.5-7: Decay Heat Removal System Flow - Chemical and Volume Control System Malfunction - Limiting Reactor Coolant System Pressure Case

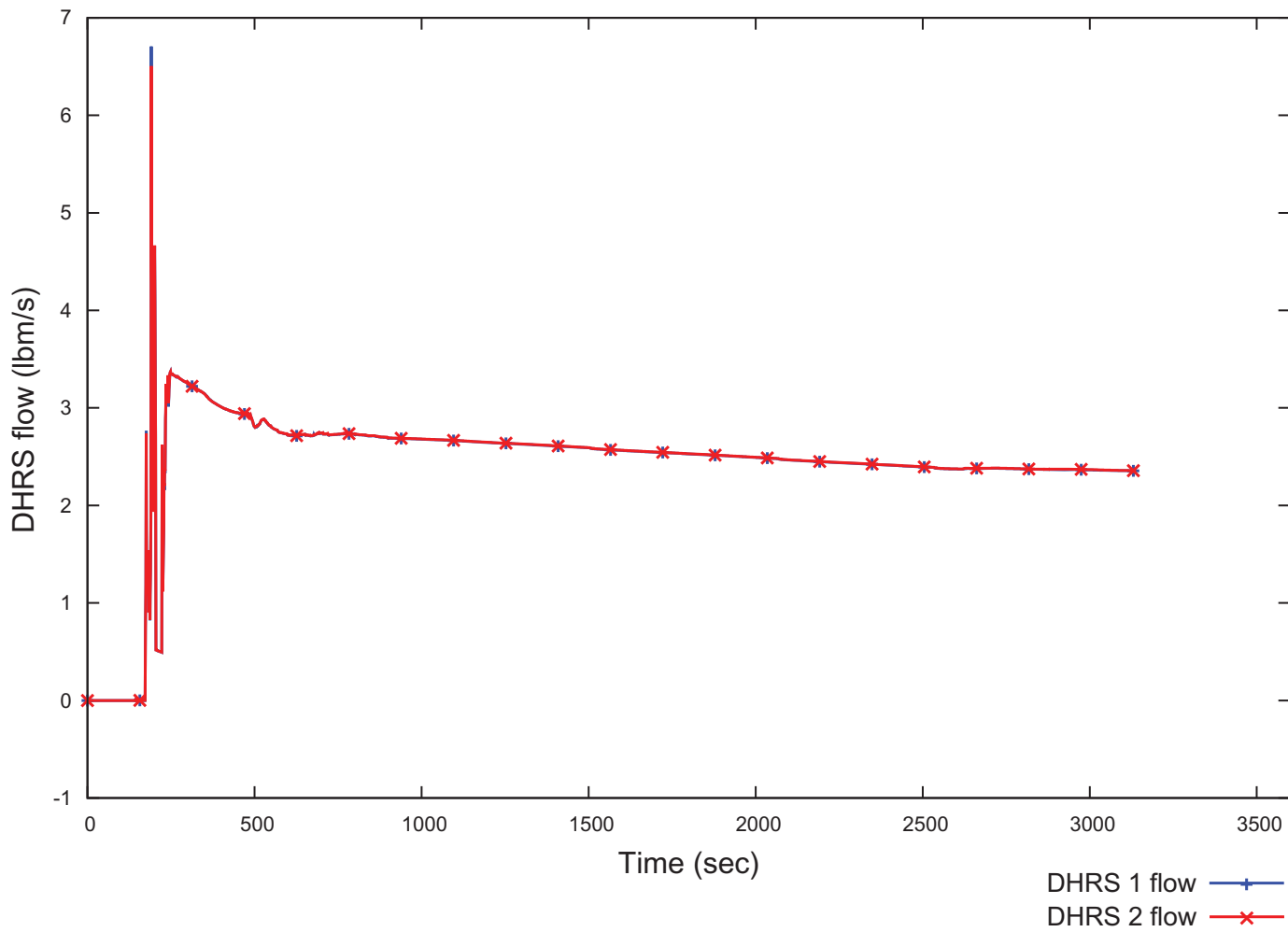


Figure 15.5-8: Reactor Pressure Vessel Pressure - Chemical and Volume Control System Malfunction - Limiting Reactor Coolant System Pressure Case

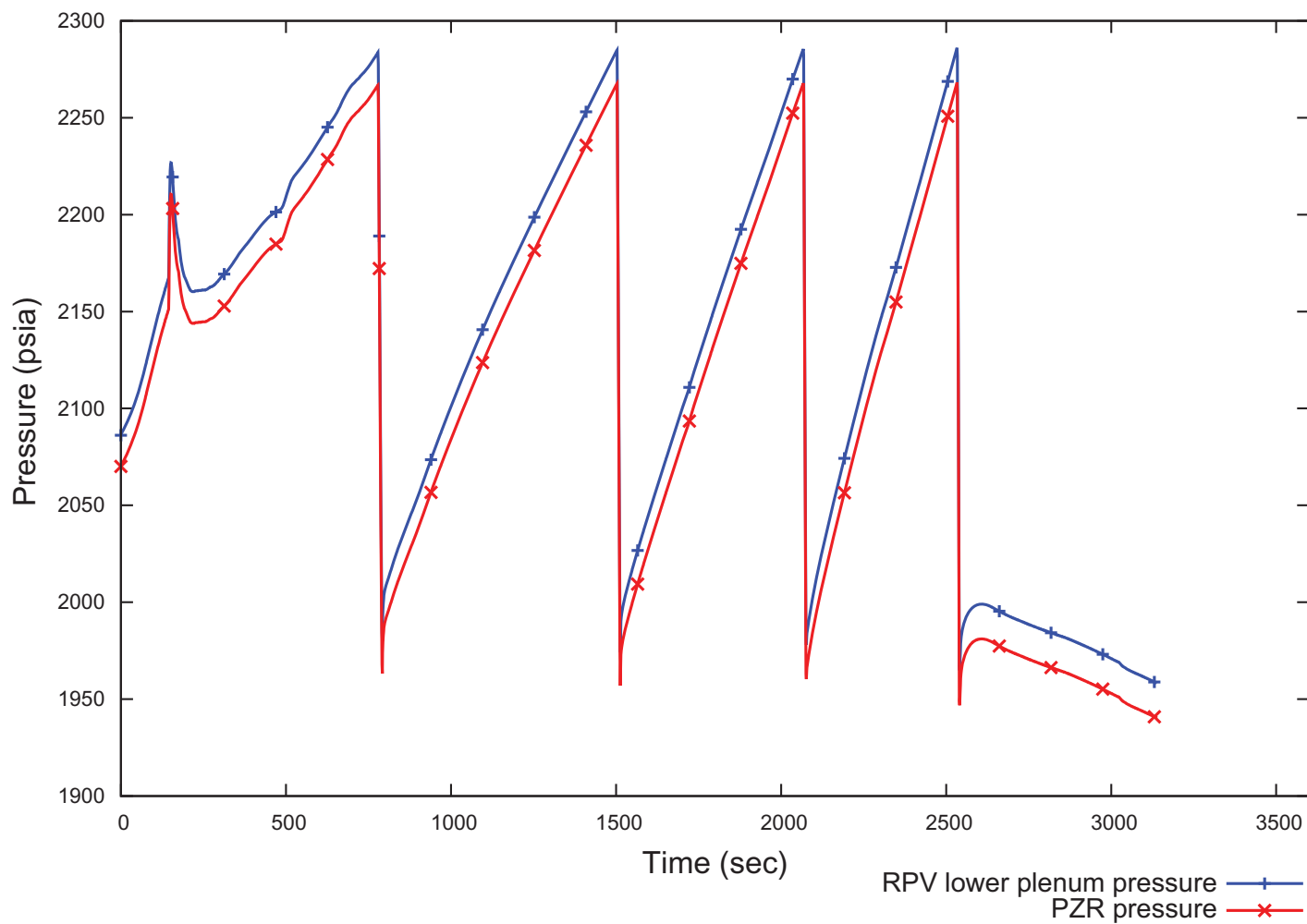


Figure 15.5-9: Steam Generator Pressure - Chemical and Volume Control System Malfunction - Limiting Reactor Coolant System Pressure Case

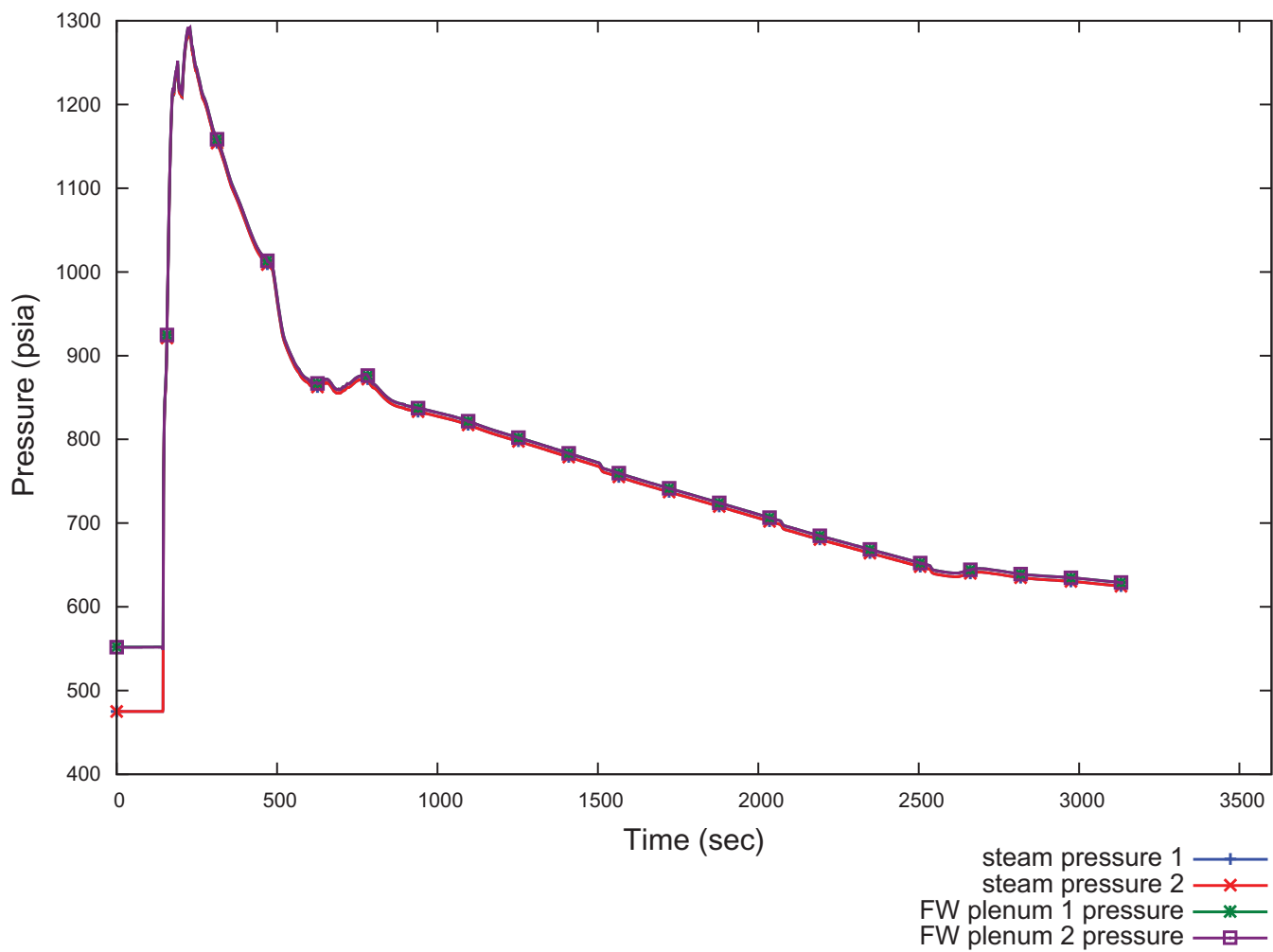
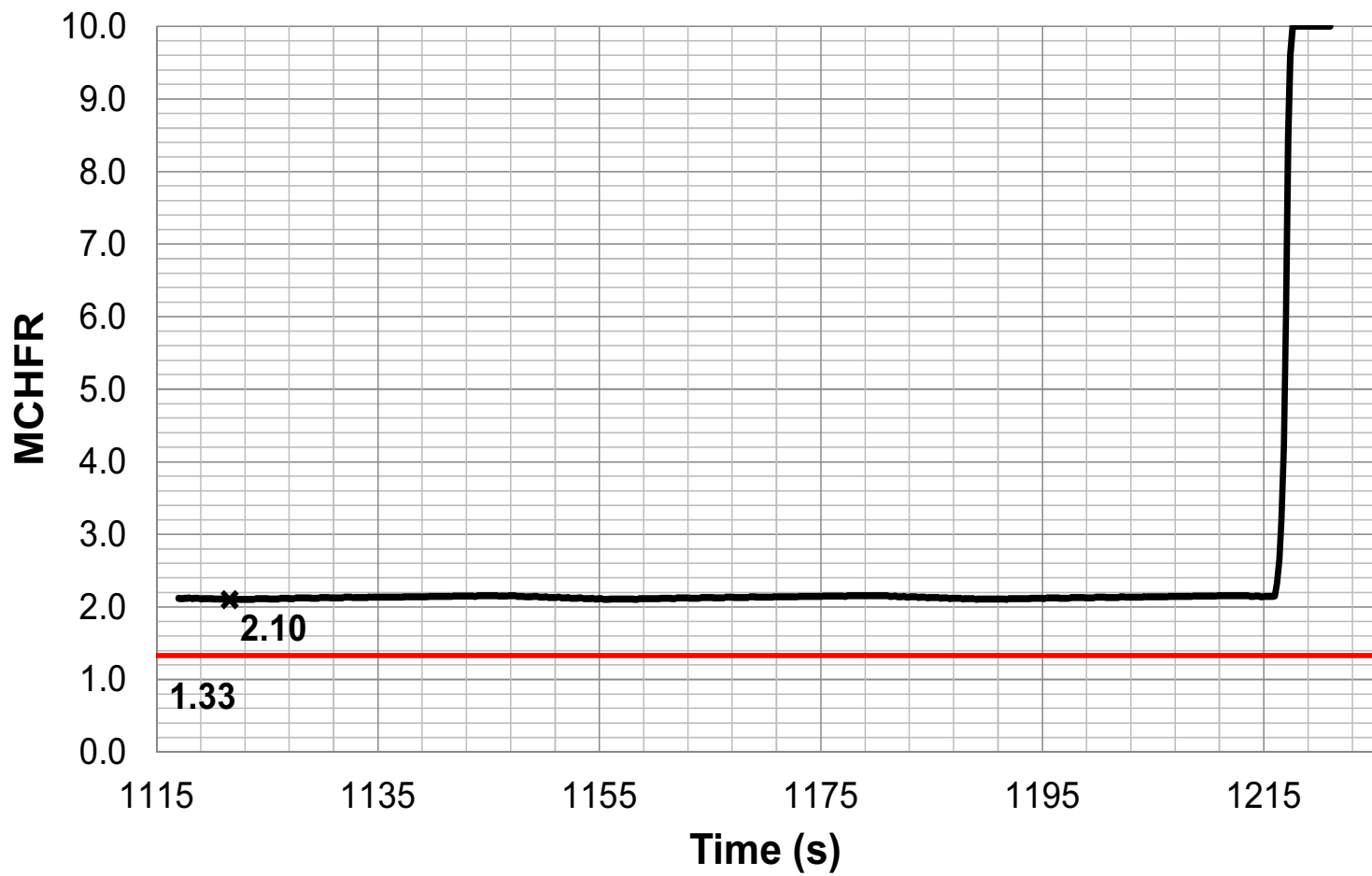


Figure 15.5-10: Minimum Critical Heat Flux Ratio - Chemical and Volume Control System Malfunction - Limiting Minimum Critical Heat Flux Ratio Case



15.6 Decrease in Reactor Coolant Inventory

This section addresses design basis events associated with a potential unplanned decrease in reactor coolant system (RCS) inventory.

15.6.1 Inadvertent Opening of a Reactor Safety Valve

Reactor safety valves (RSVs) provide over-pressure protection of the NuScale Power Module (NPM). The inadvertent opening of an RSV event can be caused by a mechanical valve failure and is classified as an anticipated operational occurrence (AOO) in Table 15.0-1.

The inadvertent RSV actuation event is bounded by the inadvertent operation of the emergency core cooling system (ECCS), which opens both reactor vent valves (RVVs). Inadvertent opening of an RSV with a loss of the augmented direct current (DC) power system (EDAS) is evaluated, which also opens both RVVs. This event has similar results to inadvertent ECCS operation but is less limiting for MCHFR due to the loss of EDAS causing immediate insertion of the control rods.

The inadvertent ECCS operation event, including an evaluation of the inadvertent opening of an RSV, is presented in Section 15.6.6.

15.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment

Lines that carry primary coolant outside containment are the chemical and volume control system (CVCS) injection and discharge lines, pressurizer spray lines, and reactor pressure vessel (RPV) high point vent line. The CVCS lines extend from the RPV and exit the containment vessel (CNV) through double containment isolation valves (CIVs). Failure of lines carrying primary coolant outside containment is analyzed for thermal hydraulic effects and radiological consequences. This event is classified as an infrequent event, as shown in Table 15.0-1.

15.6.2.1 Identification of Causes and Accident Description

Failure of lines carrying primary coolant outside containment is a non-mechanistic break in the CVCS injection line, CVCS discharge line, RPV high point vent line, or pressurizer spray line. The pressurizer spray line and high point vent line are the same size, are located at the top of the pressurizer, and exit through the containment head. A break in either of these lines is bounded by the CVCS injection and discharge lines.

To determine the most severe consequences of the failure of lines carrying primary coolant outside containment, a spectrum of break sizes and locations is evaluated. Primary coolant is released from the break into the Reactor Building (RXB) until CVCS CIVs close. The piping carrying primary coolant outside containment are not expected to fail during the life of the plant, so this event is classified as an infrequent event, as indicated in Table 15.0-1.

15.6.2.2 Sequence of Events and Systems Operation

The analysis considers the rupture of the CVCS injection and discharge lines outside containment. Primary coolant is discharged from the line break into the RXB until a CVCS isolation signal occurs. The CVCS can be isolated from the RCS by two redundant safety-related CIVs located outside containment on the exit and entry containment penetrations. All four CVCS lines isolate on a containment isolation signal. In addition, the pressurizer spray line and the RPV high point vent line isolate on low pressurizer pressure.

A spectrum of break sizes and break locations are analyzed to determine the most severe consequences. The analyses show that the reactor trips, and the decay heat removal system (DHRS) actuates to remove decay heat, but ECCS actuation setpoints are not reached. Since ECCS does not actuate, once the CVCS CIVs close, the reactor coolant remains in the RPV, as opposed to discharging into containment, and shutdown of the NPM proceeds using DHRS. Since these events are terminated by containment isolation and do not rely on ECCS for mitigation, the transients are treated as non-loss-of-coolant accident (non-LOCA) events.

Dose considerations for breaks in CVCS lines outside containment are addressed in Section 15.0.3. Therefore, the analysis of CVCS line breaks in this section focuses on maximizing the RCS pressure for addressing acceptance criteria.

Table 15.6-1 provides the sequence of events for the maximum RCS pressure scenario.

15.6.2.3 Thermal Hydraulic and Subchannel Analyses

15.6.2.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to the failure of lines carrying primary coolant outside containment is performed using NRELAP5. The NRELAP5 model is based on the design features of an NPM. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.

15.6.2.3.2 Input Parameters and Initial Conditions

This evaluation considers the rupture of the CVCS injection line or CVCS discharge line located outside the containment boundary. The assumptions and initial conditions of the evaluations are selected to maximize the severity of the accident by maximizing the RCS pressure. Unless specified below, the analyses assume the control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a CVCS line break outside containment.

Table 15.6-2 provides inputs and assumptions. The maximum RCS pressure scenario is an equivalent five percent cross-sectional area CVCS injection line break. The following are key input parameters:

- core power (102 percent) - The enthalpy in the riser, where the makeup line is located, and the density in the downcomer, where the letdown line is located, is maximized at the maximum power of 102 percent.
- pressurizer pressure - In order to delay the low pressurizer pressure signal, which initiates CVCS isolation and terminates the break flow from the NPM, the nominal steady state pressure of 2000 psia is increased by the pressure uncertainty of 70 psia.
- pressurizer level (68 percent) - The pressurizer level is increased by the level measurement uncertainty of eight percent in order to delay the low pressurizer level reactor trip and CNV isolation.
- A combination of core parameters is used to provide a limiting power response. Sensitivity cases show the beginning-of cycle (BOC) core parameters maximize RCS pressure.
- Loss of normal alternating current (AC) power conditions and no loss of normal AC power conditions are examined at the start of the event and concurrent with a reactor trip. On a loss of normal AC power, the turbine is tripped and feedwater is lost. The module protection system (MPS) remains powered so safety systems are not automatically actuated. The small line failure outside of containment is detected by the MPS on low pressurizer pressure or low pressurizer level. When the turbine is tripped, the turbine stop valves close, leading to a decreased capacity of the steam generators to remove heat from the RPV. The heatup causes the pressurizer pressure to increase, the water density to decrease, and the pressurizer level to increase, which delays the event termination and maximizes RCS pressure. Therefore, a loss of AC power at the start of the event is conservative, as confirmed in sensitivity studies.
- No single failure is assumed.

15.6.2.3.3 Results

Figure 15.6-1 to Figure 15.6-10 show the system response to the failure of lines carrying primary coolant outside containment. Table 15.6-3 contains the results of the event.

The maximum RPV pressure scenario starts with an equivalent five percent cross-sectional area break of the CVCS makeup line with a coincident loss of normal AC power. The turbine stop valves close as a result of the loss of normal AC power, increasing the steam line pressure and RPV pressure (Figure 15.6-1 and Figure 15.6-2). A high pressurizer pressure signal occurs, initiating a reactor trip, SSI and DHRS actuation. The reactor trip is evident in the reactor power decrease depicted in Figure 15.6-3 and DHRS flow shown in Figure 15.6-4. The main steam isolation valve (MSIV) closure as a result of the SSI actuation signal causes a high steam line pressure signal. As the system cools due to DHRS, as shown in the pressure, temperature, and level

response of Figure 15.6-1 and Figure 15.6-2, and Figure 15.6-5 through Figure 15.6-8, a low pressurizer level signal initiates the CVCS containment isolation valves to close, isolating the break flow from the RCS. The system continues to cool using DHRS (Figure 15.6-4), RCS flow stabilizes (Figure 15.6-9), RCS temperature (Figure 15.6-5) and fuel temperature (Figure 15.6-8) stabilize and continue to decline, the core remains subcritical (Figure 15.6-10), and the water level is above the top of the active fuel (Figure 15.6-7). The system response shows that the event terminates and the NPM reaches a safe, stabilized condition.

15.6.2.4 Radiological Consequences

Section 15.0.3 provides the radiological consequences of a failure in small lines carrying primary coolant outside containment.

15.6.2.5 Conclusions

The acceptance criteria for an infrequent event are listed in Table 15.0-2. These acceptance criteria, followed by how the NuScale Power Plant US460 standard design meets them are listed below.

- 1) Potential core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit. Minimum critical heat flux ratio (CHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

The fuel integrity is not challenged by a break of a CVCS line outside containment. The fuel temperatures decrease upon the reactor trip and DHRS actuation, as shown in Figure 15.6-8, and the water level remains above the top of the active fuel, as shown in Figure 15.6-7. In addition, the event is bounded by the rapid depressurization predicted during the inadvertent ECCS actuation, which is analyzed for critical heat flux as presented in Section 15.6.6.

- 2) RCS pressure should be maintained below 120 percent of the design value.

Table 15.6-3 presents the results. The RCS pressure for the limiting RCS pressure scenario is below the acceptance criterion.

- 3) The main steam pressure should be maintained below 120 percent of the design value.

Table 15.6-3 presents the results. The main steam pressure, presented as steam generator pressure, is below the acceptance criterion.

- 4) The containment pressure should be maintained below the design pressure.

The failure of small lines carrying primary coolant outside containment is not an event that challenges containment pressure. Events that discharge RCS fluid directly inside containment bound this event. The peak containment pressure for design basis events is evaluated in Section 6.2.

- 5) A postulated accident, including an IE, shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

The analysis presented for this event shows that the reactor is tripped, the CNV is isolated, and that stable DHRS cooling is reached. Therefore, the acceptance criterion is satisfied.

15.6.3 Steam Generator Tube Failure (Thermal Hydraulic)

15.6.3.1 Identification of Cause and Accident Description

A steam generator tube failure (SGTF) could be caused by a rapid propagation of a circumferential crack that leads to a double-ended rupture of the tube. Reactor coolant passes from the primary side of the SG to the secondary side where steam is produced and travels through the main steam lines to the turbine and condenser. Radionuclides contained in the primary coolant are discharged through the failed tube into the atmosphere via the condenser until the faulted SG is isolated by automatic closure of the MSIVs and main steam isolation bypass valves (MSIBVs). The design of the helical coil steam generators, described in Section 5.4, is different from the design of SGs in conventional pressurized water reactors because primary coolant is located on the outside of the tubes, or shell side of the SG. Following a tube failure, the primary coolant travels from the shell side of the SG into the tube through the break.

An SGTF is classified as a postulated accident as described in Section 15.0.

The SGTF analysis evaluates the primary and secondary system response to the transient to verify that the event meets the acceptance criteria specified in Table 15.0-2. The radiological consequences of the SGTF event are addressed in Section 15.0.3.

15.6.3.2 Sequence of Events and Systems Operation

Sensitivity analyses are performed to identify limiting scenarios for the areas of interest for the SGTF event. The scenarios are limiting RCS pressure and limiting SG pressure.

The sequence of events for the limiting SGTF scenarios are provided in Table 15.6-4 and Table 15.6-5. Unless otherwise specified, the analysis of an SGTF event assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of a SGTF.

15.6.3.3 Thermal Hydraulic Analyses

15.6.3.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to a SGTF is performed using NRELAP5. The NRELAP5 model is based on the design features of an NPM. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.

15.6.3.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the evaluation of the SGTF event are selected to provide a conservative calculation and to maximize the RCS and SG pressure.

Table 15.6-6 provides inputs and assumptions for the scenarios. The following are key input parameters:

- core power (102 percent) - A high biased power is conservative with respect to primary and secondary side pressures.
- pressurizer pressure - A higher initial pressure leads to a higher secondary pressure. Thus, the nominal steady state pressure of 2000 psia is increased by the pressure uncertainty of 70 psia.
- pressurizer level (68 percent) - The pressurizer level is increased by the level measurement uncertainty of eight percent.
- feedwater temperature - Sensitivity analyses show that applying a low bias to the feedwater temperature maximizes RCS and SG pressure. Thus, the nominal feedwater temperature of 250 degrees F is reduced to 230 degrees F.
- fuel kinetics - A combination of core parameters is used to provide a limiting power response. Sensitivity cases show that the beginning of cycle (BOC) core parameters maximize RCS and SG pressure.
- RCS average temperature - The nominal RCS average temperature of 540 degrees F is biased low by 5 degrees F. A lower RCS average temperature results in higher RCS and SG pressures.
- SG pressure - Maximum RCS and SG pressure is achieved with a high SG pressure bias.
- SG tube plugging - Sensitivity is performed on the impact of 10 percent SG tube plugging. Results show that 10 percent tube plugging leads to higher RCS and SG pressures.
- Loss of power - No loss of power and a loss of normal AC power at the start of the event and concurrent with a reactor trip are examined. Loss of normal AC power at the start of the event is limiting for RCS pressure. The turbine is tripped and feedwater is lost. The module protection system (MPS) remains powered so none of the safety systems are automatically actuated. When the turbine is tripped, the turbine stop valves close, leading to a decreased capacity of the steam generators to remove heat

from the RPV. The heatup causes the pressure to increase and the water density to decrease, which maximizes RCS pressure. Having AC power available delays reactor trip, which is more limiting for SG pressure.

- single failure - Sensitivity analyses show that steam line pressure and RPV pressure is greater when the MSIV closes. Therefore, assuming no single failure is more conservative than assuming the MSIV fails to close for these cases.
- SG tube failure location - Sensitivity analyses show that a tube failure at the top of the SG results in higher RCS and SG pressure.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide (RG) 1.105.
- The highest worth control rod is assumed to be stuck at the fully withdrawn position.

15.6.3.3.3 Results

Figure 15.6-11 through Figure 15.6-26 show the system response to the SGTF. Table 15.6-7 contains the results of the event. The scenarios begin with a failure of an SG tube. The system response from the tube failure is similar in the analyzed scenarios, only the timing of the MPS signals vary as a result of the inputs and assumptions used to maximize the parameter of interest.

The limiting RPV pressure scenario begins with a partial tube failure (20 percent of tube area) at the top of the SG with a coincident loss of normal AC power. The subsequent loss of feedwater causes heatup of the RCS fluid. In addition, the SG heat removal capability is degraded after the SGTF. The water in the RPV expands causing an increase in RCS pressure and pressurizer level (Figure 15.6-14) until the MPS actuates a reactor trip, SSI and DHRS actuation on a high pressurizer pressure signal. The RCS and SG pressure response is shown in Figure 15.6-12. The SSI actuation includes closure of the MSIVs, isolating the SG and stopping the break flow (Figure 15.6-11) through the tube to the environment. Figure 15.6-13 shows the SG levels, which are below 50 percent at the time of the MSIV closure, thus the valve closes in a steam environment and SG overfill occurs after MSIV closure. RCS flow and average temperature are shown in Figure 15.6-15 and Figure 15.6-16, respectively. DHRS flow is shown in Figure 15.6-17 and total reactivity for the event is shown in Figure 15.6-18.

The limiting SG pressure scenario begins with a partial tube failure (70 percent tube area) at the top of the steam generator with no loss of normal AC power. The timing of the MPS signals, reactor trip, SSI actuation, DHRS actuation and MSIV closure changes due to power availability and the parameters and assumptions that maximize the SG pressure. With normal cooling available, the break flow (Figure 15.6-19) drops pressurizer level (Figure 15.6-22) until the MPS reactor trip setpoint is reached on low pressurizer level. The RCS and SG pressure response is shown in

Figure 15.6-20. The SG levels are presented in Figure 15.6-21. The RCS flow and average temperature response are shown in Figure 15.6-23 and Figure 15.6-24, respectively. DHRS flow is presented in Figure 15.6-25 and total reactivity for the event is shown in Figure 15.6-26.

The MPS is credited to protect the plant in the event of SGTF. The following MPS signals provide the plant with protection during an SGTF:

- high pressurizer pressure,
- low pressurizer pressure, and
- low pressurizer level.

The MSIVs, MSIBVs, secondary MSIVs, and secondary MSIBVs are credited for isolating the faulted SG, depending on the scenario. All of these valves are qualified for the conditions analyzed and are designed to close under design basis conditions.

15.6.3.4 Radiological Consequences

Section 15.0.3 provides the radiological consequences of the SGTF.

15.6.3.5 Conclusions

The acceptance criteria for a postulated accident are listed in Table 15.0-2. These acceptance criteria, followed by how the NuScale Power Plant US460 standard design meets them, are listed below.

- 1) Potential core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit. MCHFR is used instead of minimum DNBR, as described in Section 4.4.2.

The fuel integrity is not challenged by a SGTF. The water level remains in the pressurizer level region, above the top of the active fuel, as shown in Figure 15.6-13 and Figure 15.6-22. In addition, the event is bounded by the rapid depressurization predicted during the inadvertent ECCS operation event, which is analyzed for critical heat flux and presented in Section 15.6.6.

- 2) RCS pressure should be maintained below 120 percent of the design value.

Table 15.6-7 presents the results of the limiting scenarios. The RCS pressure is below the acceptance criterion.

- 3) The main steam pressure should be maintained below 120 percent of the design value.

Table 15.6-7 presents the results of the limiting scenarios. The main steam pressure, presented as steam generator pressure, is below the acceptance criterion.

- 4) The containment pressure should be maintained below the design pressure.

An SGTF is not an event that challenges containment pressure. Events that discharge RCS fluid directly inside containment bound this event. The peak containment pressure for design basis events is evaluated in Section 6.2.

- 5) A postulated accident, including an IE, shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

The analysis presented for this event shows that stable DHRS cooling is reached (Figure 15.6-17 and Figure 15.6-25) and therefore the acceptance criterion is satisfied.

15.6.4 Main Steam Line Failure Outside Containment (BWR)

This event is not applicable to the NuScale Power Plant US460 standard design.

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

A LOCA is an event that compromises the reactor coolant pressure boundary (RCPB), resulting in RCS inventory loss at a rate that exceeds the capacity of normal makeup flow. A spectrum of break sizes and locations of the RCS pressure boundary piping are assessed. Table 15.6-11 provides the spectrum of break sizes and locations. The ECCS valve flange connections are subject to break exclusion as described in Section 3.6.2 and are not evaluated in this section.

A LOCA for the NPM design is unique compared to traditional large light water reactors because the diameters of the RCS piping are small so there is no distinction between "large break LOCA" and "small break LOCA" scenarios. In addition, RCS inventory is preserved within containment and available for recirculation soon after event initiation. A LOCA is classified as a potential accident, as shown in Table 15.0-1. Inadvertent opening of an ECCS valve is not considered a LOCA and is addressed in Section 15.6.6.

15.6.5.1 Identification of Causes and Accident Description

A LOCA is a postulated accident that is initiated by a non-mechanistic break in a pipe inside containment connected to the RPV. The break location and break size determines the rate of RCS inventory loss and depressurization rate. Thus a spectrum of break sizes is postulated to occur at various locations in the pipelines penetrating the RCPB, as shown in Table 15.6-11. The postulated break sites are the rupture of the RCS injection and discharge lines, high point vent line, and pressurizer spray supply lines inside of containment. Breaks in lines outside of containment are addressed in Section 15.6.2.

15.6.5.2 Sequence of Events and Systems Operation

The initiating event for this transient is a rupture of the RCS injection or discharge line, RPV high point vent line, or pressurizer spray supply line inside of containment. The LOCA break spectrum is separated into two categories: liquid space breaks in either the RCS injection line or the discharge line; and steam space breaks in either the high point vent line or the pressurizer spray supply line.

A steam space break initiates a blowdown of the RCS inventory into the CNV from the top of the RPV. A liquid space break causes blowdown of the RCS inventory into the CNV from the liquid filled region of the RPV. The progression of the steam and liquid space LOCA events are similar, but result in different timing of the key events and the liquid and steam composition of the break flow.

The RCS injection line connects from the CVCS system through the CNV and RPV into the riser above the core. The discharge line connects from the RPV downcomer and exits through the RPV and CNV. Breaks in either of these lines discharges RCS fluid into the CNV. The MPS initiates a reactor trip on high CNV pressure and actuates CNV isolation, DHRS, and SSI. ECCS actuates when RPV riser level drops below the MPS setpoint. Upon an ECCS actuation signal, both RVVs open immediately and the RRVs open when the differential pressure between the RPV and CNV drops below the inadvertent actuation block (IAB) release pressure.

Both the pressurizer spray line and the high point vent are in the vapor space of the pressurizer. A break in either of these lines into the CNV starts to depressurize the RCS and pressurize the CNV. The RCS depressurizes faster in a steam space break than a liquid space break. The MPS initiates a reactor trip on high CNV pressure and actuates CNV isolation, DHRS and SSI. ECCS actuates when RPV riser level drops below the MPS setpoint. Upon an ECCS actuation signal, both RVVs open immediately and the RRVs open when the differential pressure between the RPV and CNV drops below the IAB release pressure. Steam space breaks take longer to reach the ECCS setpoint since less mass is released into the CNV than from liquid space breaks.

The limiting MCHFR case for LOCA is the 100 percent discharge line break, with a loss of AC power and EDAS at event initiation, at 102 percent rated thermal power. Due to the loss of EDAS, control rods insert and both RVVs open. The RPV depressurizes through the break and the RVVs. Depressurization causes voiding in the core region resulting in decreased power due to moderator density feedback. As voids are generated, some expansion occurs, which temporarily reduces core inlet flow. The location of the discharge line in the downcomer reduces the core flow more than steam space breaks or injection line breaks due to the proximity of the break location to the core inlet. While flow to the core remains positive during the discharge line break, the decrease in flow results in the minimum CHFR.

Table 15.6-8 shows the sequence of events for the limiting MCHFR case, a 100 percent discharge line break from 102 percent power. The MPS is credited to

initiate the reactor trip, isolate containment, initiate DHRS, SSI and ECCS. No operator action is credited in this event analysis.

The transition from the LOCA analysis to the post-LOCA long-term core cooling phase occurs when natural circulation between the RPV and the containment through the RVVs and RRVs reaches a stable state and decay and residual heat is being removed.

15.6.5.3 Thermal Hydraulic Analyses

15.6.5.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to a LOCA uses NRELAP5. Section 15.0.2 provides details on the modeling requirements and code modifications needed to appropriately capture the phenomena and features of the LOCA evaluation model. Section 15.0.2 discusses the LOCA Evaluation Model Development and Assessment Process (EMDAP). Details of the LOCA methodology are provided in the LOCA Evaluation Model topical report (Reference 15.6-1). Utilizing the results of the break spectrum, the methodology demonstrates that the design and operating conditions analyzed results in a safe condition of the NPM for postulated design basis LOCAs.

The post-LOCA long-term core cooling analysis is performed using the NRELAP5 model to support the ECCS long-term cooling methodology. A spectrum of cases is performed to encompass minimum and maximum cooldown scenarios. The results of the long-term cooling analysis are then compared to the acceptance criteria developed for evaluating the margin to boron precipitation to show that boron precipitation is avoided during the long-term cooling phase. The details of the long-term cooling phase and boron transport are described in the Extended Passive Cooling and Reactivity Control Methodology topical report (Reference 15.6-4).

15.6.5.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the LOCA analysis are selected to provide a conservative calculation. The parameter of interest for the LOCA is MCHFR. Thus, inputs and assumptions are chosen to determine the minimum CHFR.

Unless otherwise specified, the LOCA analysis assumes that the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of a LOCA for the duration of the event, including the post-LOCA long-term cooling phase.

Table 15.6-9 provides inputs and assumptions for the limiting-break LOCA analysis. The following are key input parameters common to the spectrum of breaks analyzed in Table 15.6-11:

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent instrumentation uncertainty.
- RCS average temperature is initialized at a temperature of 545 degrees F to maximize RCS energy.
- Pressurizer pressure is biased high to maximize the RCS energy and containment peak pressure cases and biased low for MCHFR.
- Pressurizer level is nominal for MCHFR and is biased low to minimize inventory for other analyses.
- Containment pressure is biased high to maximize containment peak pressure.
- Main steam pressure is biased high to maximize overall system energy.
- Feedwater temperature is biased high to maximize the overall system energy and limit heat transfer to the steam generators.
- RCS flow is biased low to maximize RCS energy.
- Bypass flow through the reflector and guide tubes is maximized to 8.0 percent of the total core flow to minimize flow through the hot assembly. This value is consistent with the subchannel analysis methodology (Reference 15.6-2 and Reference 15.6-3) discussed in Section 15.0.2.
- Reactor pool temperature is assumed to be 140 degrees F to reduce heat sink potential. Sensitivities demonstrate that the initial reactor pool temperature has negligible impact on the LOCA acceptance criteria.
- A minimum reactor pool level is applied to reduce heat sink potential. Sensitivities demonstrate that the initial reactor pool level has negligible impact on the LOCA acceptance criteria.
- Maximum radial peaking factors are applied to the hot channel and hot pin models, consistent with Reference 15.6-1.
- Minimal reactivity feedback coefficients are conservatively applied, consistent with Reference 15.6-1.
- Conservative reactor kinetics parameters biasing are used in order to prolong the fission power transient, consistent with Reference 15.6-1.
- The decay heat standard for the LOCA evaluation model is described in Section 15.0.2.
- The reactor trip reactivity insertion is calculated to account for the moderator and Doppler temperature defects. The reactivity analysis includes the assumption of the highest worth rod stuck at the fully withdrawn position.

- The following loss of power scenarios are considered:
 - No loss of power - In this scenario, MPS actuations occur as designed.
 - Loss of normal AC power - When normal AC power is lost, the feedwater pumps coast down and a turbine trip is initiated, thus limiting RCS cooling via the secondary system. Reactor trip, containment isolation, and DHRS actuation occur after a 60-second delay following a loss of normal AC power. ECCS actuation occurs either by an 8-hour timer after reactor trip (if not bypassed by the operators) or by a 24-hour timer following the loss of normal AC power. However, because DC power is still available, the MPS can still actuate these safety functions, including reactor trip, earlier if a separate actuation limit is reached. The event sequence for a loss of normal AC power is similar to that when no power is assumed lost. The primary difference is an earlier termination of secondary cooling. Thus, a loss of normal AC power conservatively maximizes the RCS thermal conditions after event initiation.
 - Loss of EDAS, normal DC power system (EDNS), and normal AC at event initiation - Power to the MPS is provided via the EDAS, so this scenario results in an immediate actuation of the reactor trip system, DHRS, SSI, containment isolation, opening of the RVVs, and opening of the RRVs when differential pressure between the RCS and the CNV drops below the IAB release pressure.
- Single failure evaluations of the failure of a single RVV to open, a failure of a single RRV to open, and failure of one ECCS division (one RVV and one RRV) to open were performed to determine the most conservative scenario. The evaluations show that the limiting single failure for some cases is the failure of one RVV and one RRV to open. The limiting MCHFR case is not affected by single failures because the minimum CHFR occurs before the single failures can occur.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.

The input parameters and initial conditions used in the limiting case for the post-LOCA long-term cooling analysis are also selected to provide a conservative calculation. The limiting criterion is minimum collapsed liquid level above the core during the long-term cooling period and occurs for a 100-percent RCS injection line break with the following inputs and assumptions.

- A 1.2 multiplier to decay heat.
- ECCS valve flow capacity biased low.
- A minimum reactor pool temperature of 65 degrees F.
- The reactor pool level of 54 feet.
- A single failure of one RVV.

- EDAS, EDNS, and normal AC power available.
- Zero non-condensable gases are modeled.

15.6.5.3.3 Results

The LOCA analysis is performed for a spectrum of break sizes and break locations (Table 15.6-11) to determine the location and size of the break that is limiting for MCHFR.

Results for the 100 percent discharge line break at 102 percent RTP with electric power (AC and DC) unavailable are given in Table 15.6-10 and Figure 15.6-27 through 15.6-41. After break initiation, blowdown flow from the RPV to containment depressurizes the RPV. The break flow from the RPV to the CNV is shown in Figure 15.6-27 and the flow from the CVCS line to the CNV is shown in Figure 15.6-28. Since EDAS power is assumed to be unavailable, the RVVs open immediately. The initial flow from the RVVs is shown in Figure 15.6-29 and the longer term flow is shown in Figure 15.6-30. Once the differential pressure between the RPV and the CNV drops below the IAB release pressure, the RRVs opens. The RRV flow is shown in Figure 15.6-31. Loss of fluid through the break and operation of ECCS causes RPV pressure to drop as shown in Figure 15.6-32. Similarly, the steam and water entering the CNV causes pressure to increase as shown in Figure 15.6-33. Loss of EDAS also actuates SSI, which isolates the SG, initially increasing SG pressure as shown in Figure 15.6-34. As the event progresses, RPV riser level continues to decrease, shown in Figure 15.6-35, and CNV level continues to rise, shown in Figure 15.6-36.

The reactor trip for this event is immediate due to the loss of EDAS power. Control rods insert and power decreases as shown in Figure 15.6-37. Reactivity for the event is shown in Figure 15.6-38. RCS flow decreases to almost zero as RPV level drops below the riser (Figure 15.6-39). The lowest MCHFR is reached within one second of the event and recovers within two seconds (Figure 15.6-40). Fuel temperature decreases throughout the event (Figure 15.6-41) and the cladding is not challenged.

The discharge line break MCHFR event is less limiting if power is available. A sensitivity study for the spectrum of discharge line breaks is provided in Table 15.6-12 and shows the MCHFR values for each size break with and without AC and EDAS available.

Other LOCA events have less limiting MCHFR values. Table 15.6-13 provides the limiting sensitivity parameters for each break type. In addition to MCHFR, the limiting sensitivities for collapsed liquid level (CLL), CNV pressure, and CNV temperature are provided. Note that these values consider loss of power, single failures, and the size of the break. However, the initial conditions are established to determine the lowest MCHFR. Section 6.2 contains the limiting CNV temperature and pressure analysis. The CLL is not challenged during the early phases of a LOCA. The limiting CLL is evaluated in the Extended

Passive Cooling and Reactivity Control Methodology topical report (Reference 15.6-4) and the results are presented in Section 15.0.5.

During the post-LOCA long-term cooling phase, the containment and RPV temperatures and pressures continue to decrease, indicating that the decay and residual heat are being removed from the RPV and containment.

The MPS is credited to protect the NPM in the event of a LOCA. The following MPS signals provide the plant with protection during a LOCA:

- high pressurizer pressure
- high containment pressure
- low pressurizer level
- low pressurizer pressure
- low-low pressurizer pressure
- low-low pressurizer level
- low RPV riser level
- low-low RPV riser level

15.6.5.4 Radiological Consequences

Section 15.0.3 presents the iodine spike design basis source term (DBST) methodology and the radiological consequences of the iodine spike DBST. The LOCA does not result in fuel failure so the iodine spike DBST bounds the source term, and thus the dose consequences, of a LOCA.

15.6.5.5 Conclusions

The acceptance criteria for a LOCA are listed below. NuScale-specific acceptance criteria are in Table 15.0-4.

- 1) Peak cladding temperature - The calculated maximum fuel element cladding temperature shall not exceed 2200 degrees F. Because MCHFR remains above the CHF analysis limit for LOCA events, peak cladding temperature does not increase during these events.
- 2) Maximum cladding oxidation - The calculated maximum total oxidation of cladding shall not exceed 17 percent of the total cladding thickness before oxidation. Because MCHFR remains above the CHF analysis limit for LOCA events, oxidation of fuel cladding does not occur during these events.
- 3) Maximum hydrogen generation - The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the fuel rod plenum volume, were to react. Because MCHFR

remains above the CHF analysis limit for LOCA events, hydrogen generation does not occur during these events.

- 4) Coolable geometry - Calculated changes in core geometry shall be such that the core remains amenable to cooling. Because MCHFR remains above the CHF analysis limit for LOCA events, the fuel maintains its geometry throughout the events.
- 5) Long-term cooling - After a successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

As discussed in Section 15.0.2, acceptance criteria 1 through 4 are met and no fuel failure occurs by demonstrating that the collapsed liquid level remains above the top of the active fuel, MCHFR remains greater than the safety limit, and containment pressure and temperature remains within design limits.

Section 15.6.5.3.3, Section 15.0.5 and Table 15.6-10 demonstrate that the collapsed level remains above the top of the active fuel (Figure 15.6-35), MCHFR remains greater than the safety limit, and containment pressure and temperature remain within design limits.

The long-term cooling acceptance criterion 5 is also met. As discussed in Section 15.6.5.3.3 and Section 15.0.5, the core temperature is maintained at an acceptably low value and decay heat is removed for 72 hours after the initiation of the event. The resultant core temperature and inventory is sufficient to preclude boron precipitation.

15.6.6 Inadvertent Operation of Emergency Core Cooling System

15.6.6.1 Identification of Causes and Accident Description

An inadvertent operation of emergency core cooling system (ECCS) is defined as an accidental reactor vessel depressurization and decrease of reactor vessel coolant inventory that could be caused by a spurious electrical signal, hardware malfunction, or operator error. The design of ECCS is described in Section 6.3. The inadvertent opening of more than one RRV is considered a beyond design basis event due to the RRV IAB feature. Because the RVVs do not have the IAB feature, inadvertent opening of more than one RVV is considered. The failure of an ECCS valve to a partially open position is evaluated and determined not to be a credible initiating event.

The following combinations of inadvertent ECCS valve openings are evaluated:

- one RVV
- two RVVs
- one RRV

- one RVV and one RRV
- two RVVs and one RRV

The opening of one RSV, with and without EDAS power, is also evaluated.

The limiting scenario is the inadvertent opening of two RVVs with AC and EDAS power available. With power available, the reactor trip does not occur until an MPS actuation limit is reached. Both RVVs opening with power available and a delayed reactor trip is more limiting than the larger break area of the inadvertent opening of an RRV or RSV with both RVVs opening (on a loss of EDAS power) due to the immediate reactor trip associated with the loss of EDAS power. However, the difference between the inadvertent opening of two RVVs with power available and inadvertent opening of an RRV or RSV with loss of EDAS power is small. Values for MCHFR for the inadvertent opening of the spectrum of RPV valves with and without AC or EDAS power are given in Table 15.6-17.

The inadvertent operation of ECCS is not expected to occur during the lifetime of a module. However the event is conservatively categorized as an AOO, as indicated in Table 15.0-1.

The inadvertent operation of ECCS analysis evaluates the primary system response to the transient to verify that the event meets the acceptance criteria specified in Table 15.0-2.

15.6.6.2 Sequence of Events and Systems Operation

Sensitivity analyses are performed to identify the limiting event for the inadvertent operation of ECCS. The initiating event that results in the limiting MCHFR for this transient is an inadvertent ECCS signal that opens both RVVs with AC and EDAS power available. The RRVs remain closed until the differential pressure between the RPV and CNV drops below the release pressure.

The sequence of events for the limiting MCHFR case is provided in Table 15.6-14. Unless otherwise specified, the analysis of an inadvertent operation of ECCS assumes the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of the event.

15.6.6.3 Thermal Hydraulic Analyses

15.6.6.3.1 Evaluation Model

The thermal hydraulic response to an inadvertent operation of ECCS event exhibits unique transient progression relative to other AOO events analyzed for the NPM.

The inadvertent ECCS actuation event is analyzed in phases that align with the phases of the LOCA transient progression for the NPM. The LOCA evaluation model and Reference 15.6-1 have:

- identified and ranked important phenomena that occur during these transient phases for the NPM,
- assessed NRELAP5 against separate effects tests and integral effects tests related to these phenomena,
- determined NRELAP5 to be applicable for evaluating these phenomena, and
- developed a conservative NRELAP5 input model for transient analyses, which involve an un-isolatable decrease in the RCS inventory event (Section 15.6.5).

Due to the phenomenological similarities to the LOCA pipe break events described in Section 15.6.5, the LOCA evaluation model is used in this analysis to evaluate the inadvertent operation of ECCS event, consistent with Reference 15.6-1.

15.6.6.3.2 Input Parameters and Initial Conditions

The input parameters and initial conditions used in the evaluation of an inadvertent operation of ECCS are selected to provide a conservative calculation and to minimize the MCHFR. Unless otherwise specified, the analysis assumes that the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of an inadvertent operation of ECCS.

Table 15.6-15 provides inputs and assumptions for the limiting MCHFR case. The following are key input parameters:

- Initial power level is assumed 102 percent of nominal power. A high biased power is conservative with respect to MCHFR.
- The RCS average temperature is biased high to 545 degrees F to maximize initial RCS energy.
- RCS flow is biased low to minimize MCHFR conditions in the core.
- Pressurizer pressure is biased low.
- The initial pressurizer level is nominal (60 percent).
- Core feedback, kinetics, and fuel properties are biased to maximize the fission power transient and initial fuel stored energy.
- Minimal (least negative) reactivity feedback coefficients are conservatively applied in order to minimize negative feedback.
- A bounding bottom peaked axial power shape is applied to maximize the highest axial peaking factor. Sensitivity studies confirm that this shape is limiting for MCHFR.

- Gap conductance is biased to its maximum value for evaluation of MCHFR and to its minimum value for other analyses.
- The following loss of power scenarios are considered.
 - No loss of power - In this scenario, MPS actuations occur as designed. The ECCS valve openings, other than those that inadvertently open to initiate the event, are dependent on the MPS ECCS actuation setpoints.
 - Loss of normal AC - When normal AC power is lost, the feedwater pumps coast down and a turbine trip is initiated, thus limiting RCS cooling via the secondary system. Reactor trip, containment isolation, secondary system isolation, and DHRS actuation occur after a 60-second delay following a loss of normal AC power. ECCS actuation occurs either by an 8-hour delay timer after reactor trip (if not bypassed by the operators) or by a 24-hour timer following the loss of normal AC power. However, because DC power is still available, the MPS can still actuate a reactor trip, secondary system isolation, containment isolation, and DHRS, earlier if a separate actuation limit is reached. The event sequence for a loss of normal AC power is similar to that when no power is assumed lost. The primary difference is an earlier termination of secondary cooling.
 - Loss of EDAS, EDNS, and normal AC at event initiation - This scenario results in an immediate actuation of the reactor trip system, secondary system isolation, DHRS, and containment isolation. As power to the MPS is lost, both RVVs open immediately and the RRVs open when differential pressure between the RCS and the CNV drops below the IAB release pressure.
- The single failure evaluation considered one RVV failing to open, one RRV failing to open, or failure of one ECCS division causing one RVV and one RRV to fail to open. The evaluation compared the results to a scenario with no single failure. The evaluation showed that the single failure cases have no adverse impact on the limiting MCHFR. Therefore, the scenario with no single failure is limiting for the MCHFR analysis.
- The failure modes that could lead to a partial opening of an ECCS valve are characterized as having a remote probability of occurrence or are determined to not be credible. None of the credible component failure mechanisms that could prevent a full stroke of the ECCS valve have the potential to cause an ECCS valve to open. Therefore, a partial opening of an ECCS valve is not a credible initiating event.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in RG 1.105.
- No operator action is credited.

15.6.6.3.3 Results

Figure 15.6-42 to Figure 15.6-52 show the system response to an inadvertent ECCS operation event. Table 15.6-16 provides the results of the event. The limiting MCHFR case is initiated by an inadvertent opening of both RVVs.

Upon the inadvertent opening of two RVVs, the large blowdown of steam into the containment causes rapid depressurization of the RCS and rapid pressurization of the containment. The RVV flow is shown in Figure 15.6-42. The increase in containment pressure initiates a reactor trip, containment isolation and secondary system isolation. The RPV and containment pressures are shown in Figure 15.6-43 and Figure 15.6-44, respectively.

The rapid RCS depressurization causes voiding in the core and a decrease in RCS flow (Figure 15.6-45), leading to a reduction in CHFR (Figure 15.6-46). Reactor power decreases during this time due to control rod insertion and negative void feedback, as seen in Figure 15.6-47. Following the occurrence of transient MCHFR (Figure 15.6-46), a temporary increase in RCS flow is observed due to the increased density gradient from voiding in the riser (Figure 15.6-45).

The isolation of the secondary system from high containment pressure causes an increase in steam generator pressure, as seen in Figure 15.6-48. DHRS actuates on high CNV pressure, however, DHRS does not actuate in time to have an impact on the MCHFR analysis. Heat transfer from the RCS to the secondary coolant isolated in the steam generator region is limited due to the decreasing RCS temperatures associated with decreasing pressure and saturation temperature. Steam generator pressure is not limiting for an inadvertent operation of ECCS event.

As primary coolant is released to the containment through the open RVVs, the inventory level inside the containment increases. Pressure and temperature inside the RPV continue a gradual downward trend, as shown in Figure 15.6-43 and Figure 15.6-49.

After the RRVs open and pressure equalizes across the RRVs, liquid coolant from the containment begins to flow into the RPV downcomer region. A two phase natural circulation loop is established through the ECCS valves with steam exiting the pressurizer area into containment through the RVVs and liquid returning from the containment to the RPV through the RRVs. Decay heat and residual heat is transferred from the containment to the reactor pool resulting in the pressure and the temperature inside the RPV and containment continuing to decrease.

The transient continues until stable ECCS cooling is established with RCS pressure and temperature continuing to decrease. The module remains in a safe condition with liquid level maintained above the top of the core through the entire transient. The minimum collapsed liquid level (Figure 15.6-50) occurs after equilibrium conditions are established between the RPV and containment and is approximately 10 feet above the top of the active fuel. The

fuel volume average temperature is shown in Figure 15.6-51, and the total reactivity for the event is shown in Figure 15.6-52.

The MPS is credited to protect the module in the event of an inadvertent operation of ECCS by the following MPS signals:

- high containment pressure
- low RPV riser level
- low-low RPV riser level
- low pressurizer pressure

No operator actions are credited for this event.

The event transitions to long-term cooling, similar to that described in Section 15.6.5.

15.6.6.4 Radiological Consequences

Radiological consequence analyses are not required for AOOs. Section 15.0.3 presents the iodine spike DBST methodology and the radiological consequences of the iodine spike DBST. The inadvertent operation of ECCS does not result in fuel failure so the iodine spike DBST bounds the source term, and thus the dose consequences, of this event.

15.6.6.5 Conclusions

The acceptance criteria for an AOO are listed in Table 15.0-2. These acceptance criteria, followed, by how they are met, are listed below. Table 15.6-16 provides the results of the limiting MCHFR scenario of an inadvertent operation of ECCS.

- 1) Fuel cladding integrity shall be maintained by ensuring that minimum DNBR remains above the 95/95 DNBR limit. MCHFR is used instead of minimum DNBR, as described in Section 4.4.2.

The fuel integrity is not challenged by the inadvertent opening of both RVVs. The fuel temperatures decrease upon reactor trip, as shown in Figure 15.6-51, and the water level remains above the top of the active fuel, as shown in Figure 15.6-50. The MCHFR is above the CHF analysis limit as shown in Table 15.6-16 and Figure 15.6-46. The MCHFR occurs shortly after event initiation with reactor power still elevated at that time.

- 2) RCS pressure should be maintained below 110 percent of the design value.

The RCS pressure is below the acceptance criterion as shown in Table 15.6-16.

- 3) The main steam pressure should be maintained below 110 percent of the design value.

The inadvertent ECCS and RPV valve opening events are cooldown events, which do not challenge secondary pressure design limits. Limiting secondary pressure events are addressed in Section 15.2.

- 4) The event should not generate a more serious plant condition without other faults occurring independently.

The analysis presented for this event shows that the NPM continues to be cooled with natural circulation through the ECCS valves and the event terminates in a safe, stabilized condition.

The response of the NPM during the long-term cooling phase following the inadvertent operation of ECCS is similar to the response of the NPM following a LOCA. The long-term cooling analysis, results, and conclusions are discussed in Section 15.0.5 and Section 15.6.5.

15.6.7 References

- 15.6-1 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422-P, Revision 3.
- 15.6-2 NuScale Power, LLC, "Subchannel Analysis Methodology," TR-0915-17564-P-A, Revision 2.
- 15.6-3 NuScale Power LLC, "Statistical Subchannel Analysis Methodology, Supplement 1 to TR-0915-17564-P-A, Revision 2," TR-108601-P, Revision 2.
- 15.6-4 NuScale Power LLC, "Extended Passive Cooling and Reactivity Control Methodology," TR-124587-P, Revision 0.

Table 15.6-1: Failure of Lines Carrying Primary Coolant Outside Containment - Sequence of Events - Maximum Reactor Pressure Vessel Pressure Scenario

Event	Time (s)
CVCS Injection Line Break (5 percent)	0
Loss of Normal AC	0
Turbine Trip	0
Loss of Feedwater	0
High Pressurizer Pressure Analytical Limit	5
Secondary System Isolation Signal	7
DHRS Actuation Signal	7
RTS Actuation (high pressurizer pressure)	7
Peak RCS Pressure	9
Low Pressurizer Pressure Analytical Limit	286
CVCS and CNV Isolation Signal (low pressurizer level)	662

Table 15.6-2: Failure of Lines Carrying Primary Coolant Outside Containment - Inputs - Maximum Reactor Pressure Vessel Pressure Scenario

Parameter	Nominal	Maximum RCS Pressure Bias
Core power	100% (250 MWt)	+2% (255 MWt)
RCS average temperature	540 °F	+5 °F
Pressurizer pressure	2000 psia	+70 psia
Pressurizer level	60%	+8%
SG pressure	475 psia	+35 psia
Feedwater temperature	250 °F	-20 °F
RCS flow	647 kg/s - 821 kg/s	low (647 kg/s)

**Table 15.6-3: Failure of Lines Carrying Primary Coolant Outside Containment - Results
Maximum Reactor Pressure Vessel Pressure Scenario**

Parameter	Acceptance Criteria	Value
Maximum RCS Pressure	≤ 2640 psia	2303 psia
Maximum SG Pressure	≤ 2640 psia	1467 psia

Table 15.6-4: Steam Generator Tube Failure - Sequence of Events - Limiting Reactor Pressure Vessel Pressure

Event	Time (s)
SGTF (20% tube area split break) at top of SG	0
Loss of AC power	0
Reactor trip actuation (high pressurizer pressure)	7
DHRS actuation	7
Pressurizer heater isolation	7
Secondary system isolation signal	7
Maximum RCS pressure	9
MSIV fully closed (SG isolated)	17
Containment isolation actuation signal	61

Table 15.6-5: Steam Generator Tube Failure - Sequence of Events - Limiting Steam Generator Pressure

Event	Time (s)
SGTF (70 percent tube area split break) at top of SG	0
Reactor trip actuation (low pressurizer level)	654
DHRS actuation	654
Secondary system isolation	654
CNV isolation	654
CVCS isolation	654
PZR heater isolation	654
MSIV fully closed (SG isolated)	664
Maximum SG pressure reached	5999

Table 15.6-6: Steam Generator Tube Failure - Inputs

Description	Nominal	Limiting RCS Pressure Scenario Bias	Limiting Steam Generator Pressure Scenario Bias
Core Power	250 MWt	255 (102%)	255 (102%)
Pressurizer Pressure	2000 psia	+70 psia	+70 psia
Pressurizer Level	60%	+8%	+8%
Feedwater Temperature	250 °F	-20 °F	-20 °F
Fuel Kinetics	N/A	BOC	BOC
RCS Average Temperature	540 °F	-5 °F	-5 °F
Steam Generator Pressure	475 psia	+35 psia	+35 psia
Steam Generator Tube Plugging	N/A	10%	10%

Table 15.6-7: Steam Generator Tube Failure - Results

Parameter	Acceptance Criteria	Value	Limiting Case
Maximum RCS Pressure	≤ 2640 psia	2304 psia	Limiting Reactor Pressure
Maximum SG Pressure	≤ 2640 psia	2276 psia	Limiting Steam Generator Pressure

Table 15.6-8: Loss-of-Coolant Accident - Sequence of Events - Limiting Minimum Critical Heat Flux Ratio Case

Event	Time (sec)
100 percent discharge line break	0
Loss of normal AC and EDAS	0
Containment isolation, SSI, reactor trip, and DHRS actuated	0
Both RVVs open	0.1
Minimum RCS flow prior to rod insertion	0.7
MCHFR	0.7
Control rods fully inserted	2.3
Containment isolation complete	10

Table 15.6-9: Loss-of-Coolant Accident - Inputs

Description	Units	Nominal	Analyzed Value
Core power - beginning of cycle	MWt	250	102%
RCS average temperature	°F	540	545
Pressurizer pressure	psia	2000	2070
Pressurizer level	%	60	60
Containment pressure	psia	<1 psia	3
Main steam pressure	psia	475	510
Feedwater temperature	°F	250	270
Bypass flow (reflector and guide tubes)	%	n/a	8.0
Reactor pool temperature	°F	100	140
Reactor pool level	ft	53	52
RCS flow	kg/s	647 - 821	647

Table 15.6-10: Loss-of-Coolant Accident - Results - Limiting Minimum Critical Heat Flux Ratio Case

Parameter	Acceptance Criteria	Value
Minimum collapsed liquid level	> 0 ft above top of core	9.5 ft
Minimum critical heat flux ratio	> 1.20	1.32
Containment pressure ⁽¹⁾	< 1200 psia	780 psia
Containment temperature ⁽¹⁾	< 600 °F	475 °F

Notes:

(1) Section 6.2 contains the limiting containment analysis. The containment pressure and temperature reported here is from the limiting MCHFR scenario.

Table 15.6-11: Loss-of-Coolant Analysis - Summary of Break Spectrum

Break Area (%)	Break Area (in ²)	Equivalent Diameter (in)	Break Area Analyzed			
			Discharge Line	Injection Line	High Point Vent	Pressurizer Spray Supply
100	2.23	1.69	X	X	X	-
75	1.67	1.46	X	X	X	-
50	1.12	1.19	X	X	X	-
35	0.78	1.00	X	X	X	X
20	0.45	0.75	X	X	X	-
10	0.22	0.53	X	X	X	-
5	0.11	0.38	X	X	X	-
2.2	0.05	0.25	X	X	X	-

Table 15.6-12: Loss-of-Coolant Analysis - Discharge Line Break Spectrum - Minimum Critical Heat Flux Ratio Sensitivity to Loss of Power

Case Break Area (%)	AC & DC Power Available		Loss of AC Power		Loss of EDAS Power	
	MCHFR	Time of MCHFR (s)	MCHFR	Time of MCHFR (s)	MCHFR	Time of MCHFR (s)
100	1.42	1.8	1.44	1.6	1.32	0.7
75	1.46	1.9	1.48	1.6	1.33	0.6
50	1.51	1.9	1.53	1.5	1.35	0.6
35	1.54	2.0	1.56	1.4	1.36	0.6
20	1.57	2.1	1.59	1.1	1.37	0.6
10	1.58	33.0	1.41	13.0	1.37	0.6
5	1.58	125.0	1.43	12.5	1.37	0.6
2.2	1.60	136.0	1.44	12.0	1.37	0.6

Table 15.6-13: Loss-of-Coolant Analysis - Limiting Sensitivity Cases for Key Parameters⁽¹⁾

Event	Percent Break	Single Failure	Power Lost	MCHFR	Collapsed Liquid Level (ft above top of core)	CNV Pressure (psia)	CNV Temperature (°F)
Injection Line	5	none	AC/DC	1.38	X	X	X
Injection Line	100	RVV, RRV	none	X	8.4	X	X
Injection Line	75	none	AC	X	X	834	X
Injection Line	35	none	AC	X	X	X	490
Discharge line	100	none	AC/DC	1.32	X	X	X
Discharge line	100	RVV, RRV	none	X	8.6	X	X
Discharge line	100	none	AC	X	X	849	X
Discharge line	50	none	none	X	X	X	492
High Point Vent	100	none	AC/DC	1.36	X	X	X
High Point Vent	2	RVV, RRV	AC	X	8.5	X	X
High Point Vent	100	none	AC/DC	X	X	770	X
High Point Vent	100	none	AC	X	X	X	479
Pressurizer Spray	35	none	AC/DC	1.37	X	X	X
Pressurizer Spray	35	RVV, RRV	none	X	8.7	X	X
Pressurizer Spray	35	none	AC/DC	X	X	768	X
Pressurizer Spray	35	none	none	X	X	X	478

Note:

- (1) Although the sensitivity to parameters other than MCHFR is shown, other initial conditions are selected based on the limiting MCHFR case.

Table 15.6-14: Inadvertent Operation of the Emergency Core Cooling System - Sequence of Events - Limiting Minimum Critical Heat Flux Ratio Case

Event	Time (s)
Inadvertent ECCS signal	0
Both RVVs open	0.1
Low pressurizer pressure analytical limit	0.2
MCHFR	0.8
Reactor trip, DHRS, SSI, CNV isolation signals	2.2
Control rods fully inserted	4.5
CNV fully isolated	12

Table 15.6-15: Inadvertent Operation of the Emergency Core Cooling System - Inputs - Limiting Minimum Critical Heat Flux Ratio Case

Description	Units	Nominal	Analyzed Value
Core power	MWt	250	255
RCS average temperature	°F	540	545
RCS flow	kg/sec	647 - 821	647
Pressurizer pressure	psia	2000	1930
Pressurizer level	%	60	60
Containment pressure	psia	<1	3
Core bypass flow	%	n/a	8

Table 15.6-16: Inadvertent Operation of the Emergency Core Cooling System - Results - Limiting Minimum Critical Heat Flux Ratio Case

Parameter	Acceptance Criteria	Value
Minimum collapsed liquid level	> 0 ft above top of core	9.5 ft
Minimum critical heat flux ratio	> 1.20	1.32
Peak reactor vessel pressure	< 2420 psia	2070
Containment temperature	< 600 °F	473 °F
Containment pressure	< 1200 psia	741 psia

Table 15.6-17: Inadvertent Operation of the Emergency Core Cooling System - Minimum Critical Heat Flux Ratio Results

Event	AC & DC Power Available		Loss of AC Power		Loss of EDAS Power	
	MCHFR	Time of MCHFR (s)	MCHFR	Time of MCHFR (s)	MCHFR	Time of MCHFR (s)
RVV	1.41	1.0	1.42	1.0	1.38	0.6
Inadvertent ECCS (2 RVVs)	1.32	0.8	1.33	0.9		
RRV	1.51	2.0	1.53	1.5	1.35	0.6
RSV	1.52	1.2	1.52	1.0	1.36	0.6

Figure 15.6-1: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Pressure Vessel Pressure

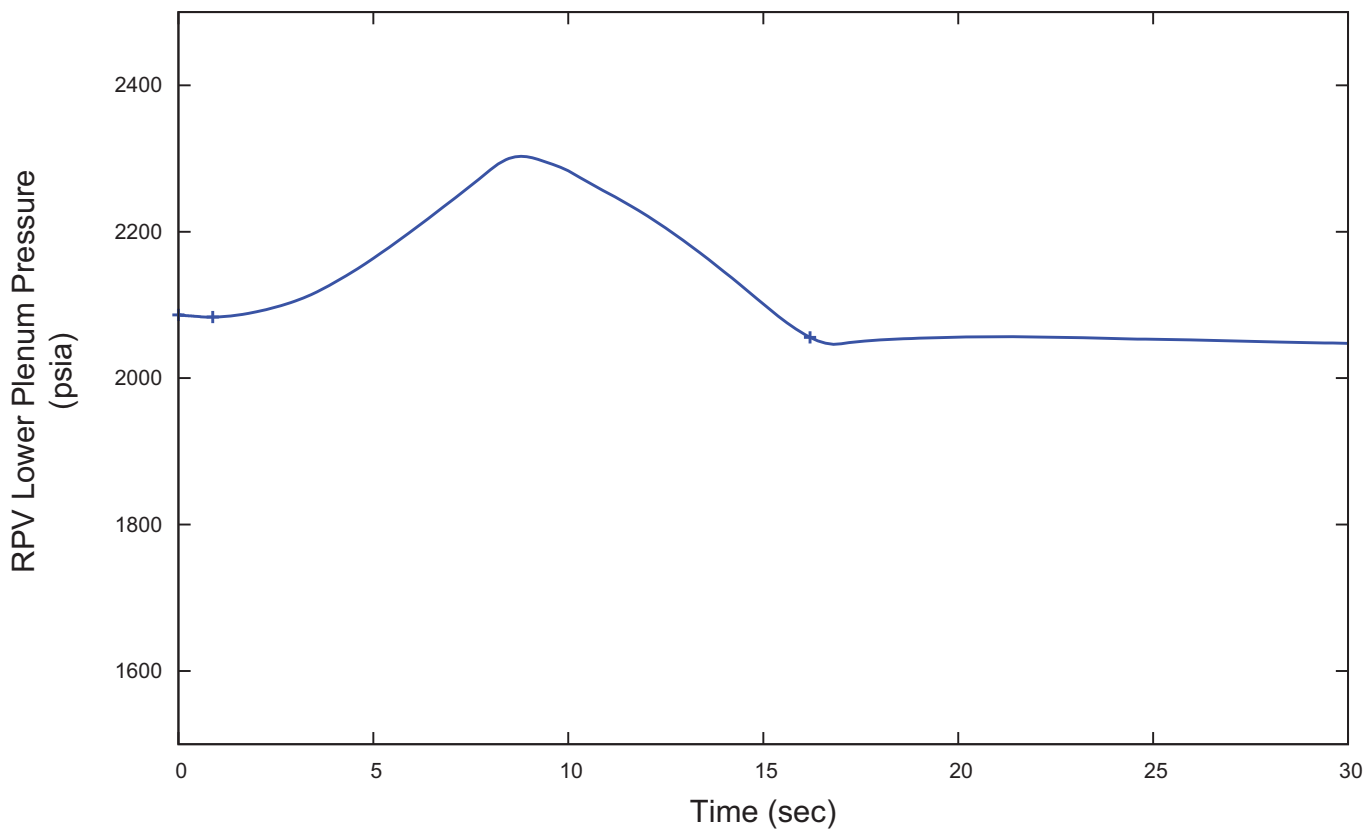
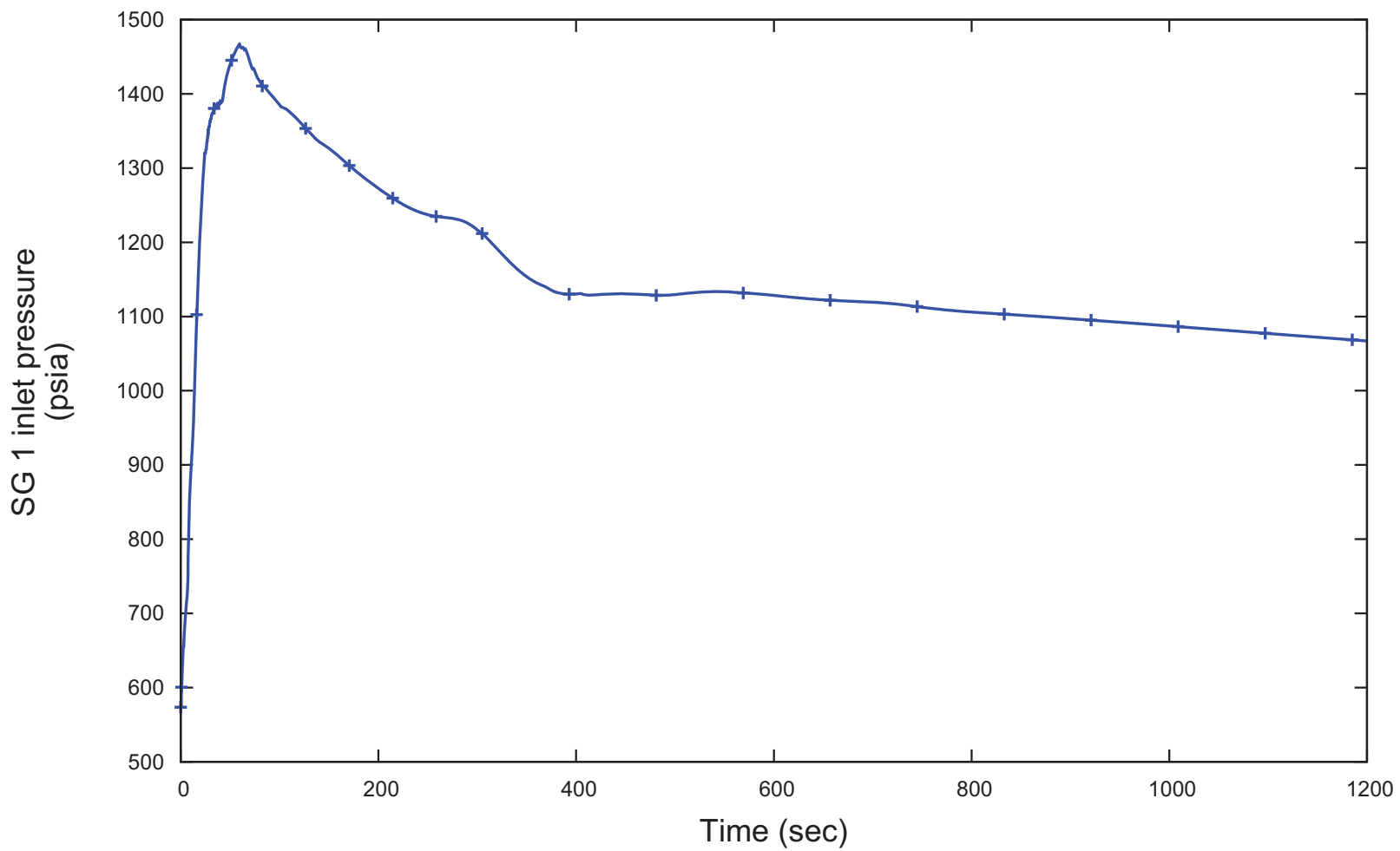


Figure 15.6-2: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Steam Generator Pressure (Typical of One Steam Generator)



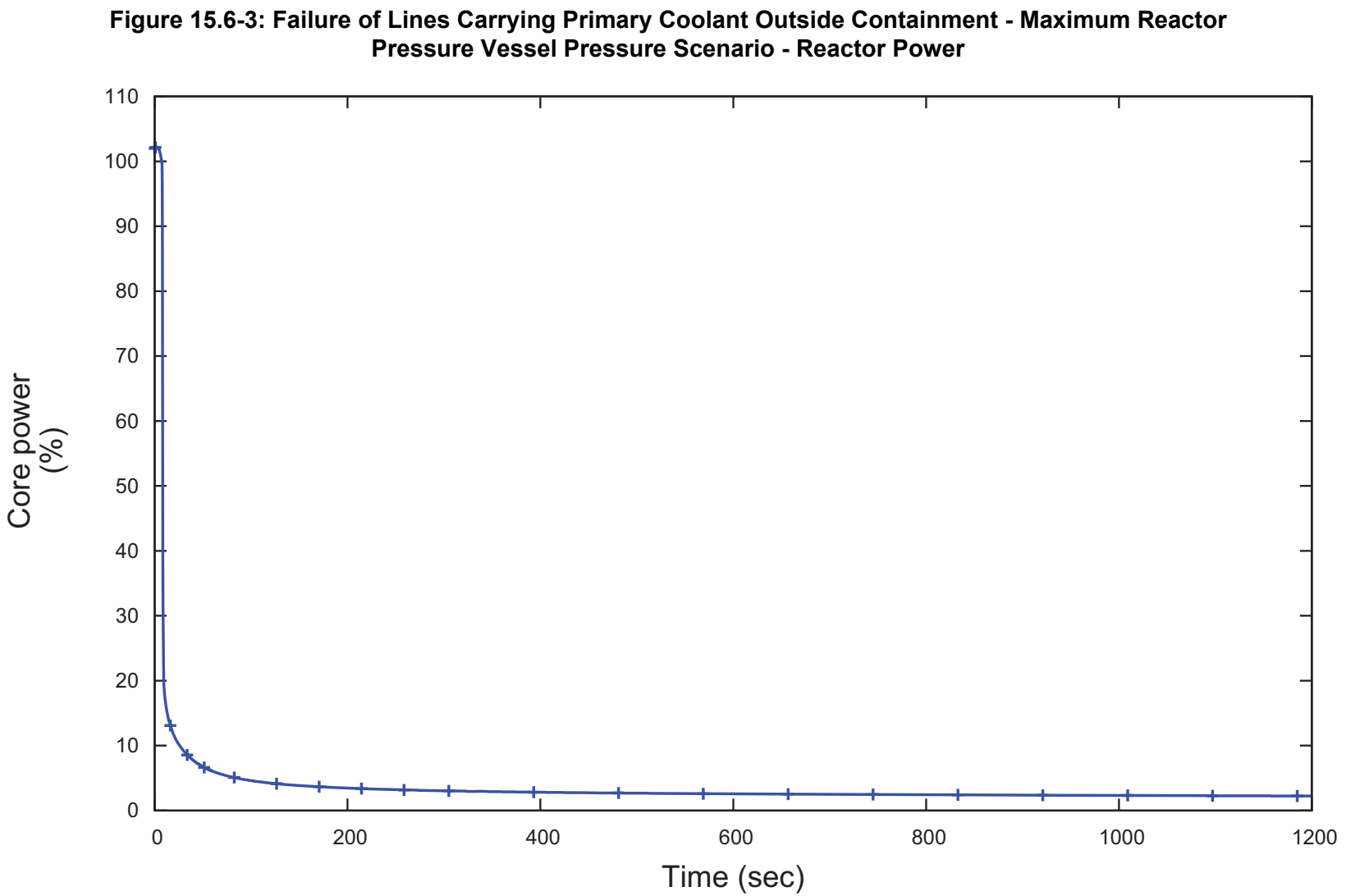


Figure 15.6-4: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Decay Heat Removal System Flow

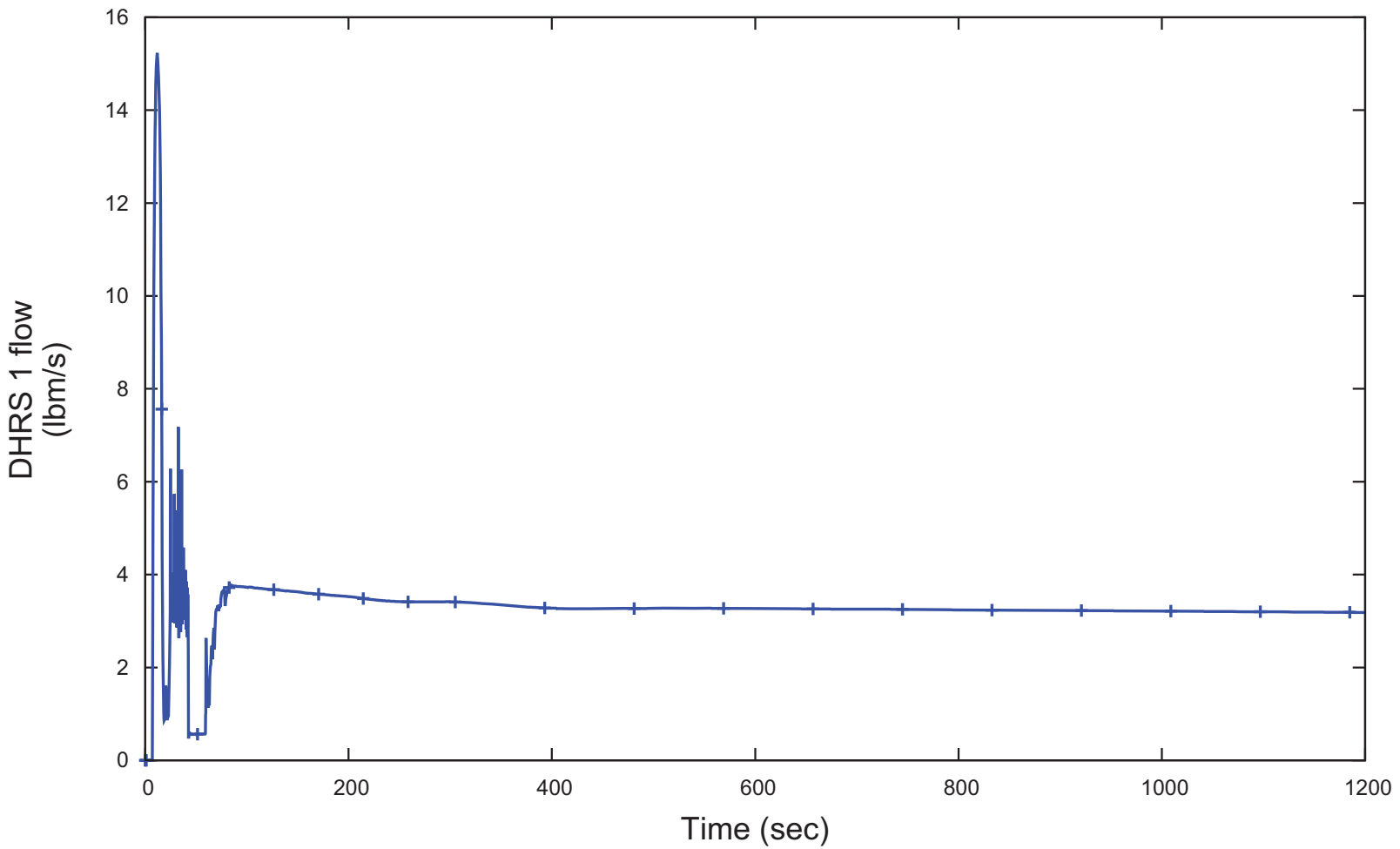


Figure 15.6-5: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Coolant System Average Temperature

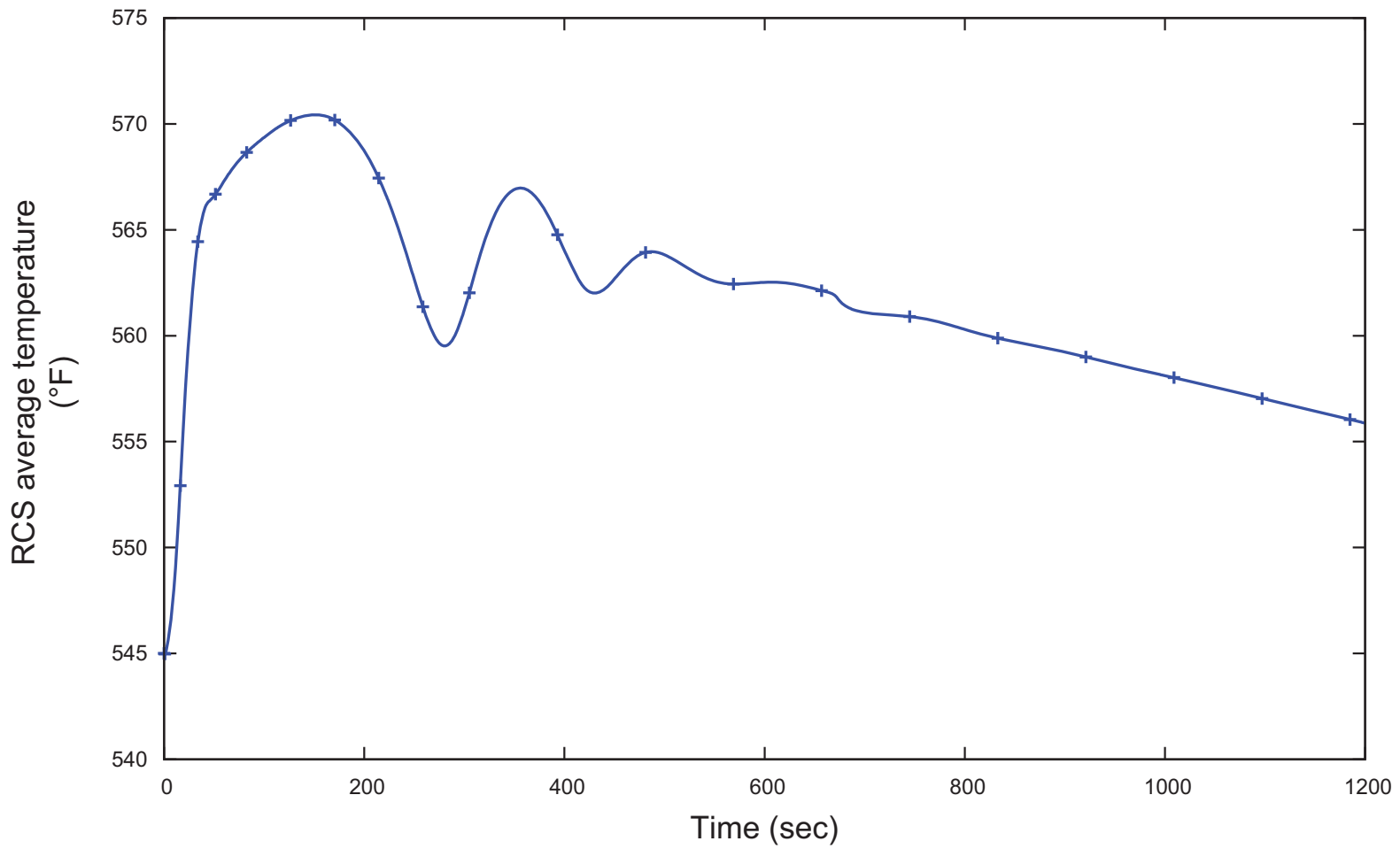


Figure 15.6-6: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Pressurizer Level

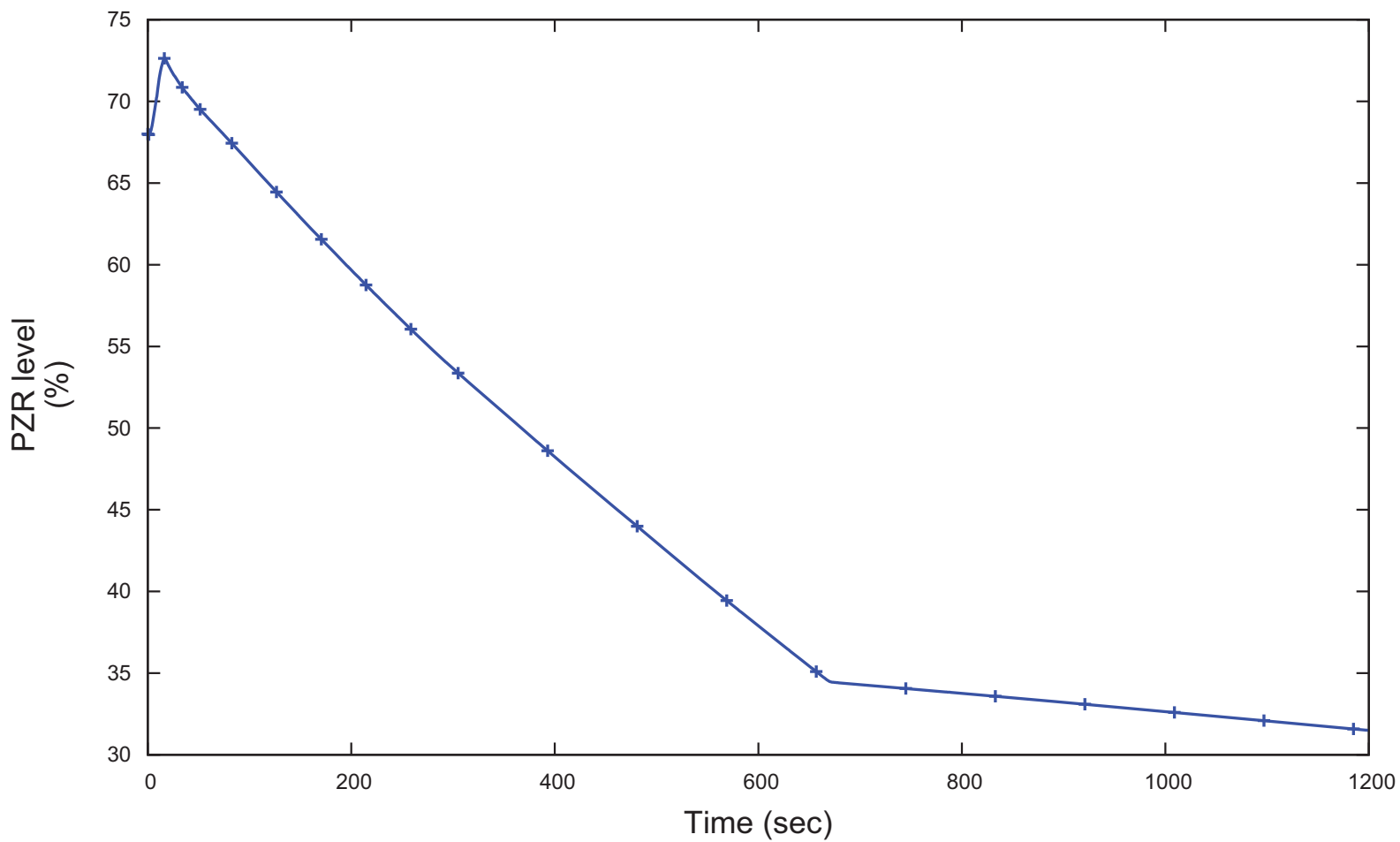


Figure 15.6-7: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Levels Relative to the Top of the Core

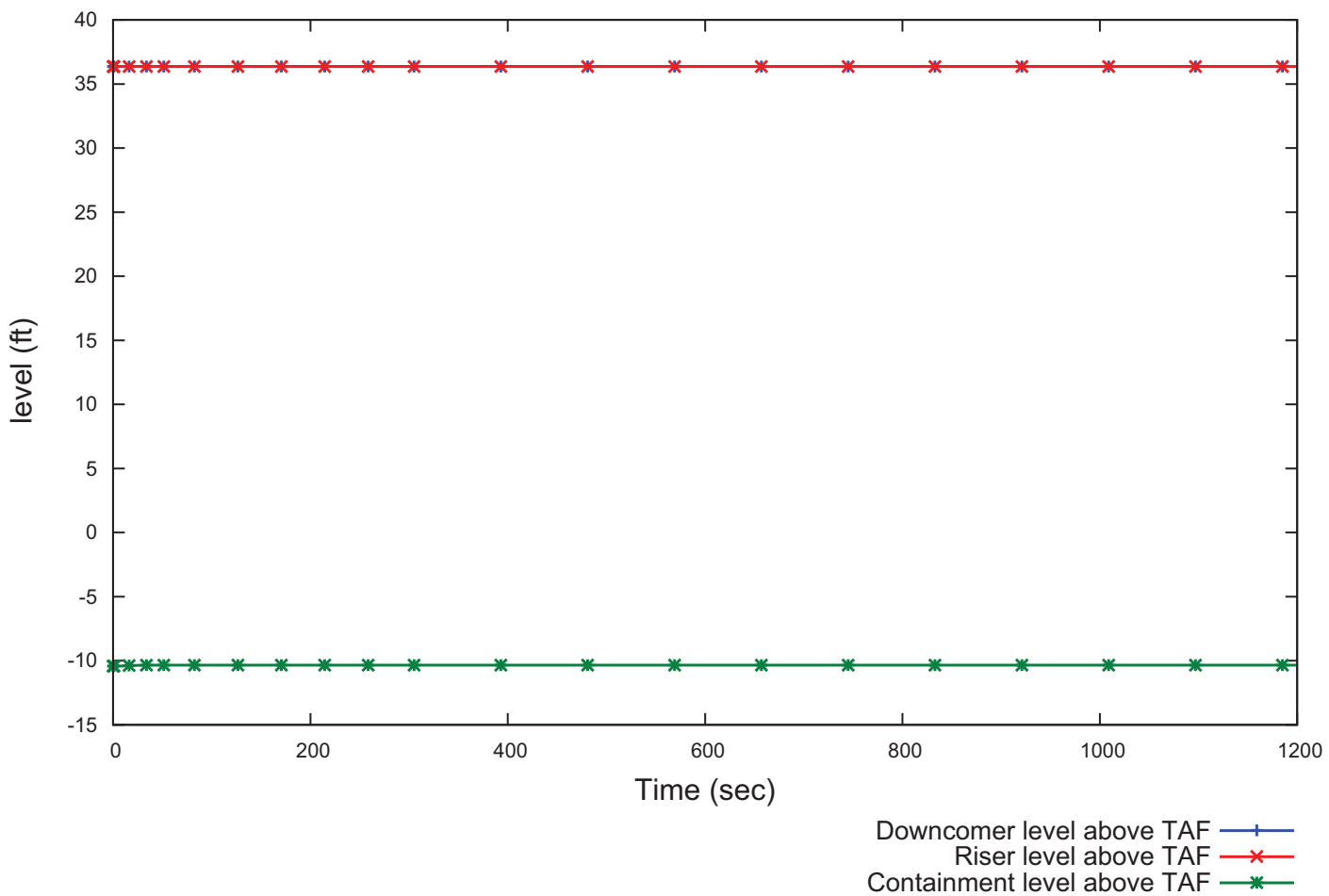


Figure 15.6-8: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Fuel Volume Average Temperature

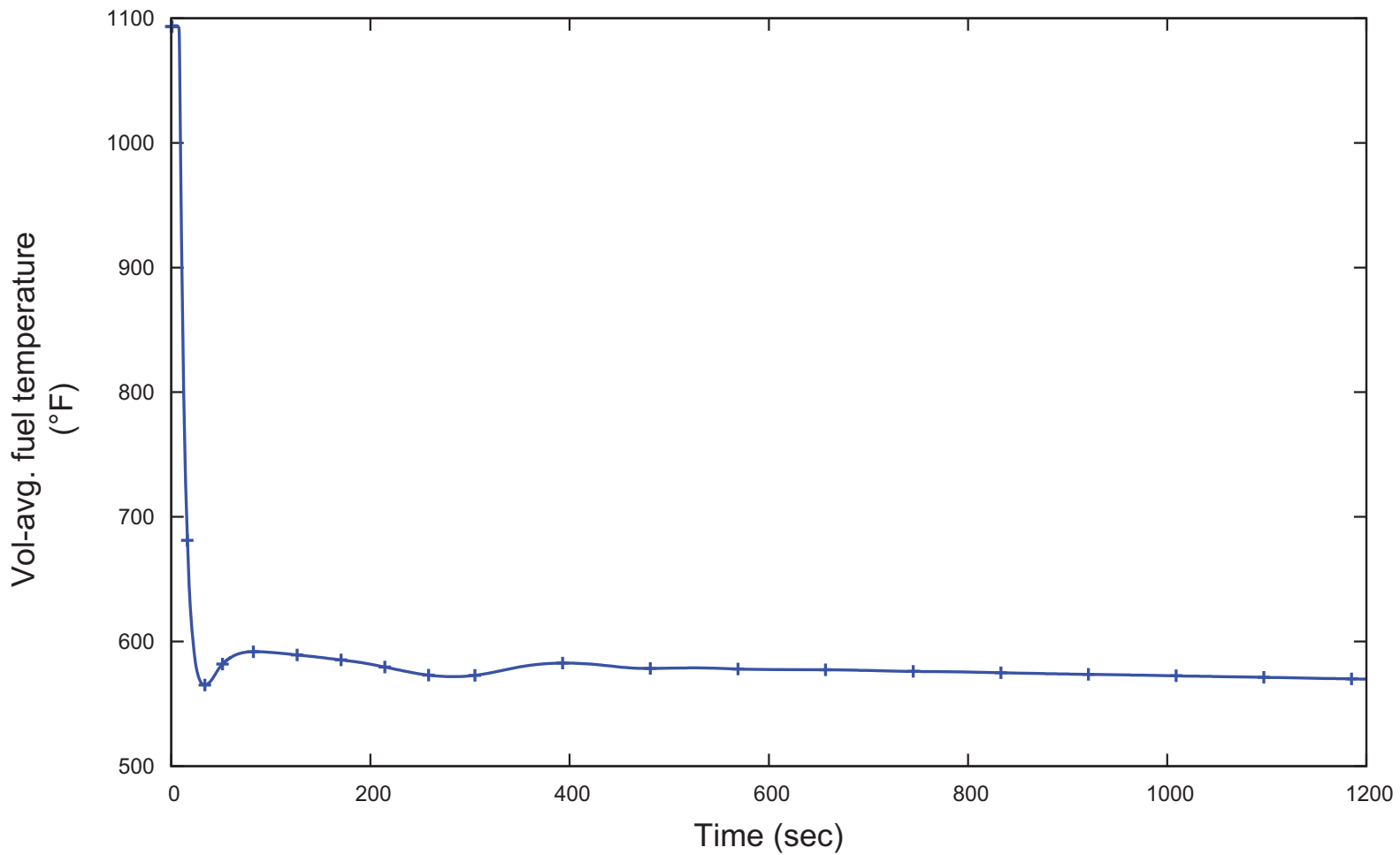
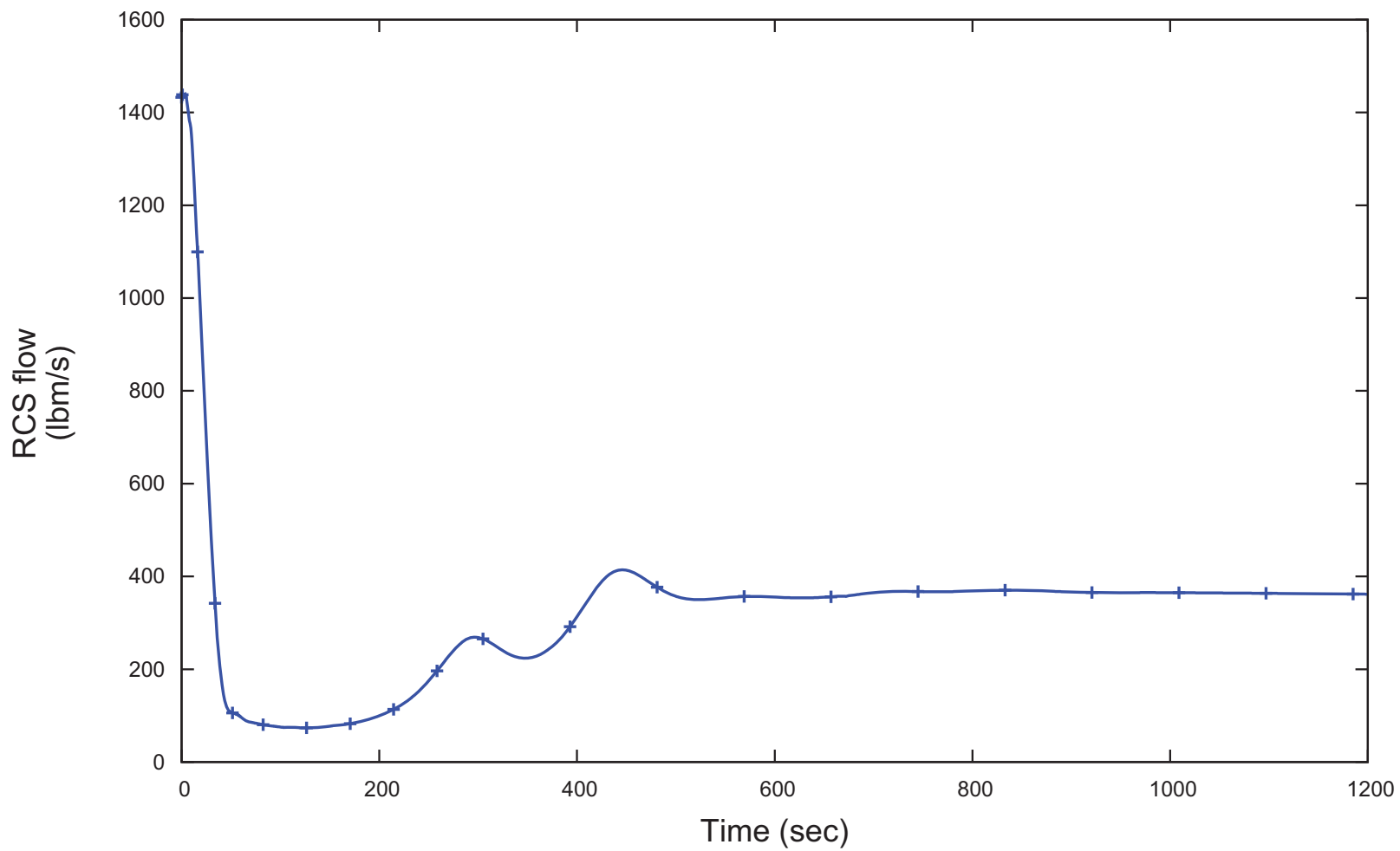


Figure 15.6-9: Failure of Lines Carrying Primary Coolant Outside Containment - Maximum Reactor Pressure Vessel Pressure Scenario - Reactor Coolant System Mass Flow



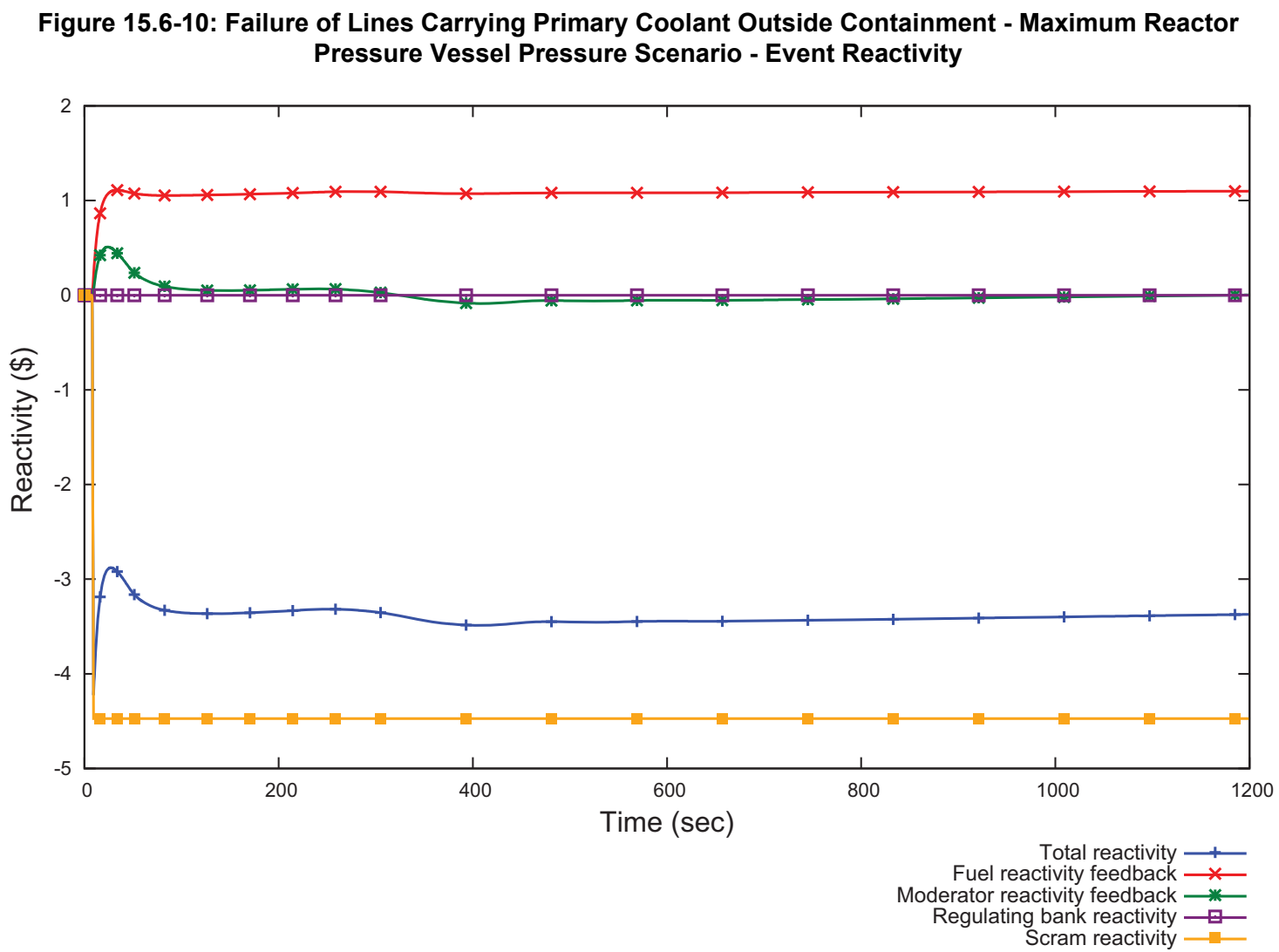


Figure 15.6-11: Steam Generator Tube Failure - Primary to Secondary Mass Flow - Limiting Reactor Coolant System Pressure Case

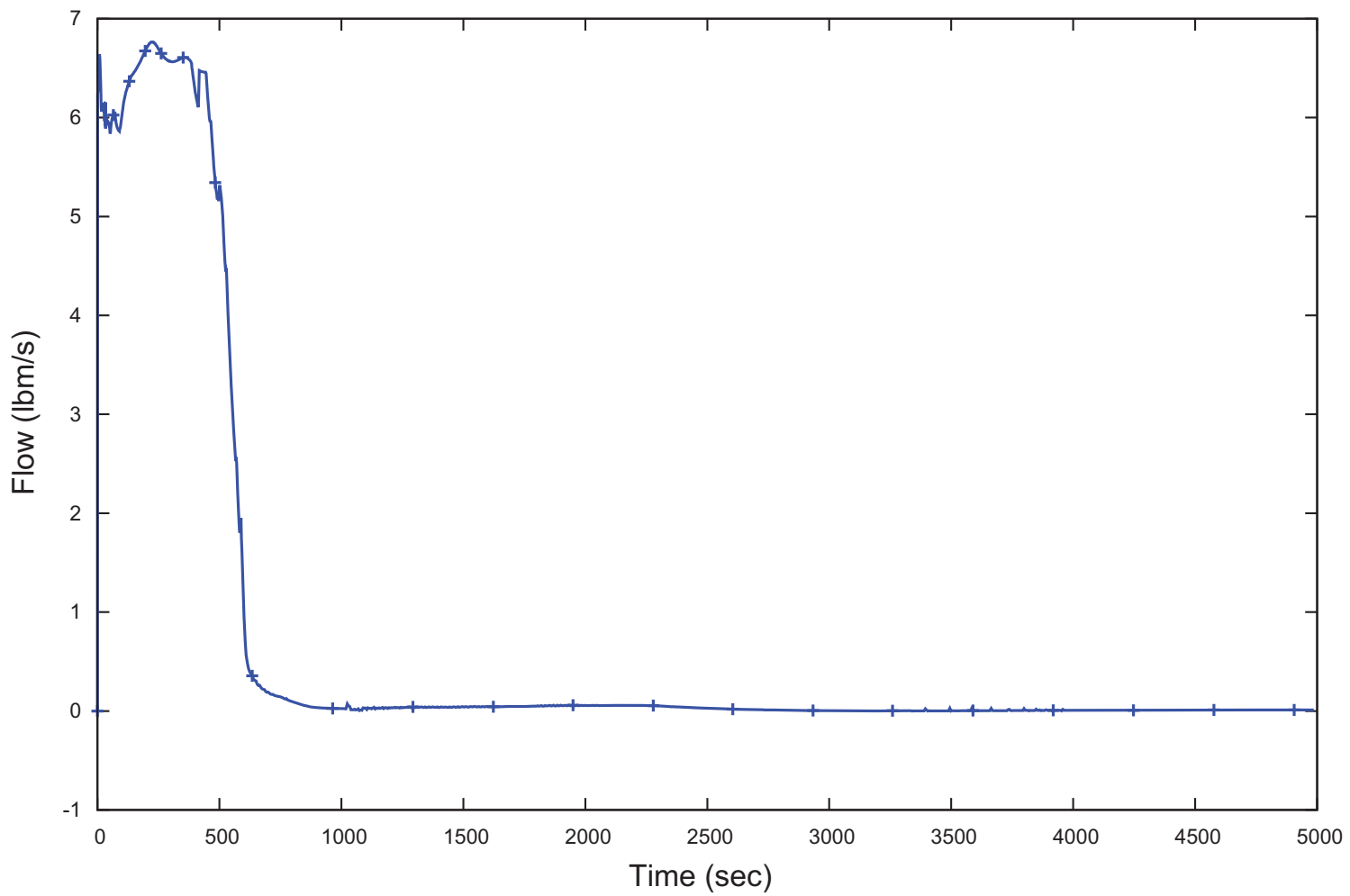


Figure 15.6-12: Steam Generator Tube Failure - Primary and Secondary Pressures - Limiting Reactor Coolant System Pressure Case

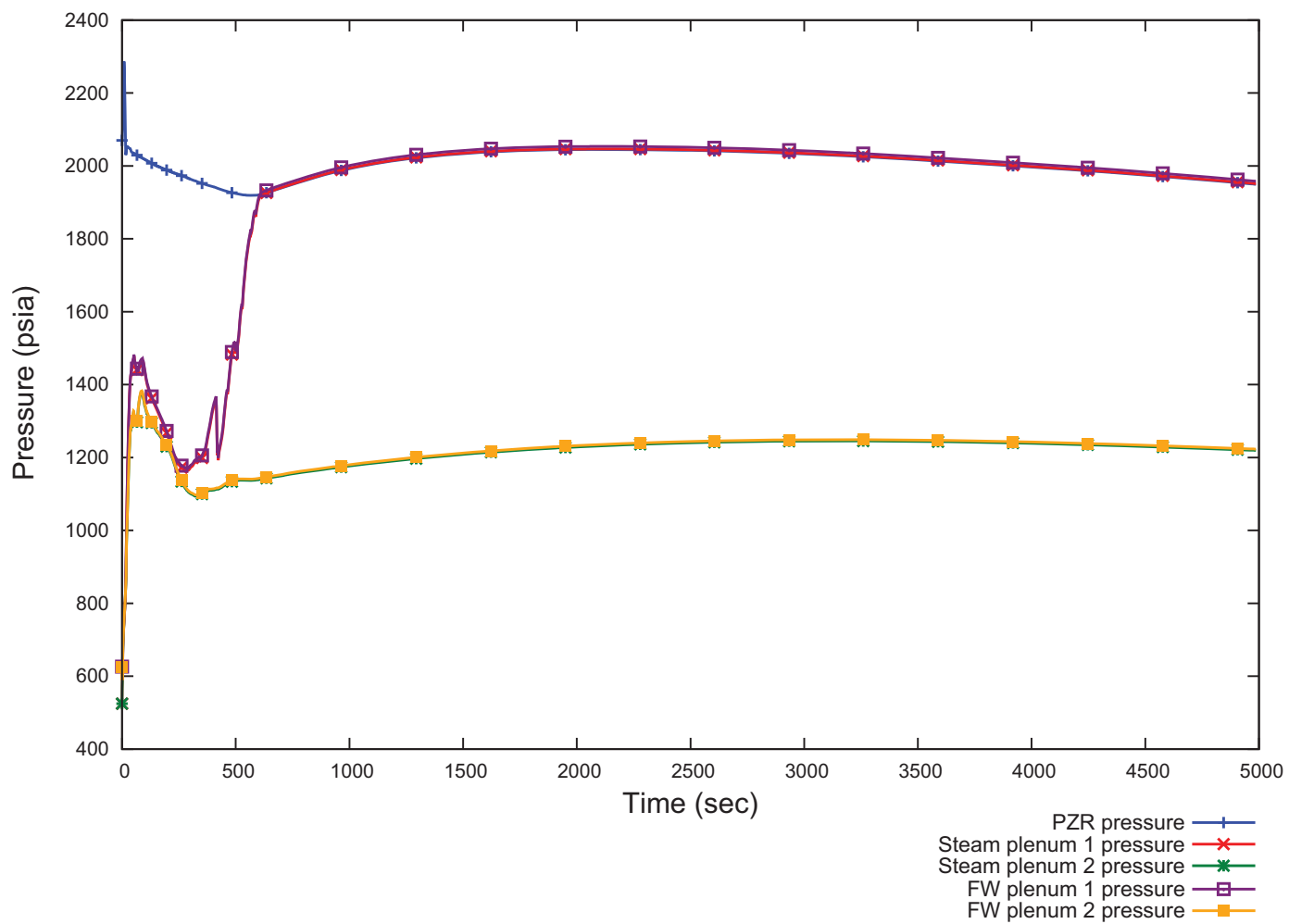


Figure 15.6-13: Steam Generator Tube Failure - Steam Generator Levels - Limiting Reactor Coolant System Pressure Case

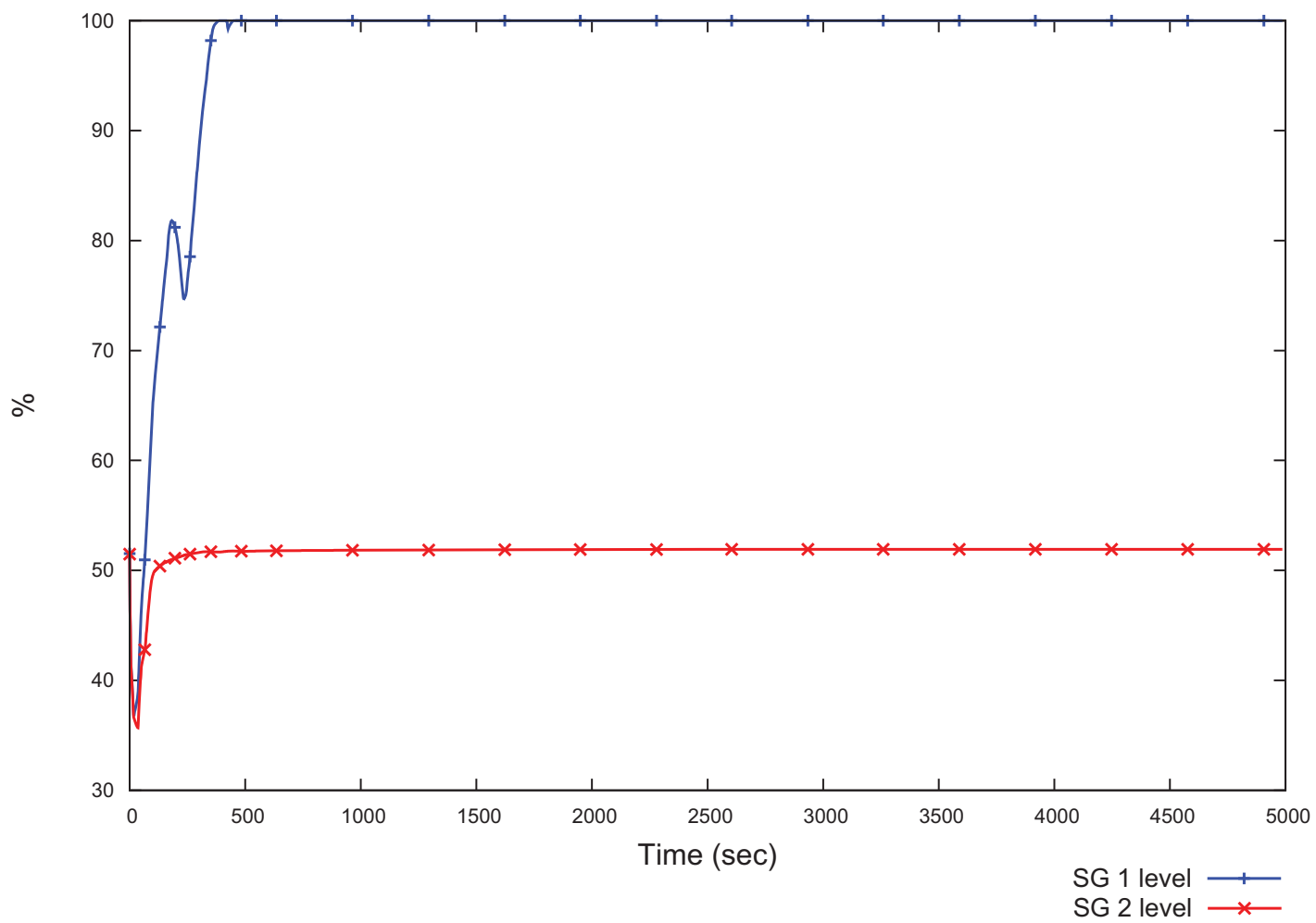


Figure 15.6-14: Steam Generator Tube Failure - Pressurizer Level - Limiting Reactor Coolant System Pressure Case

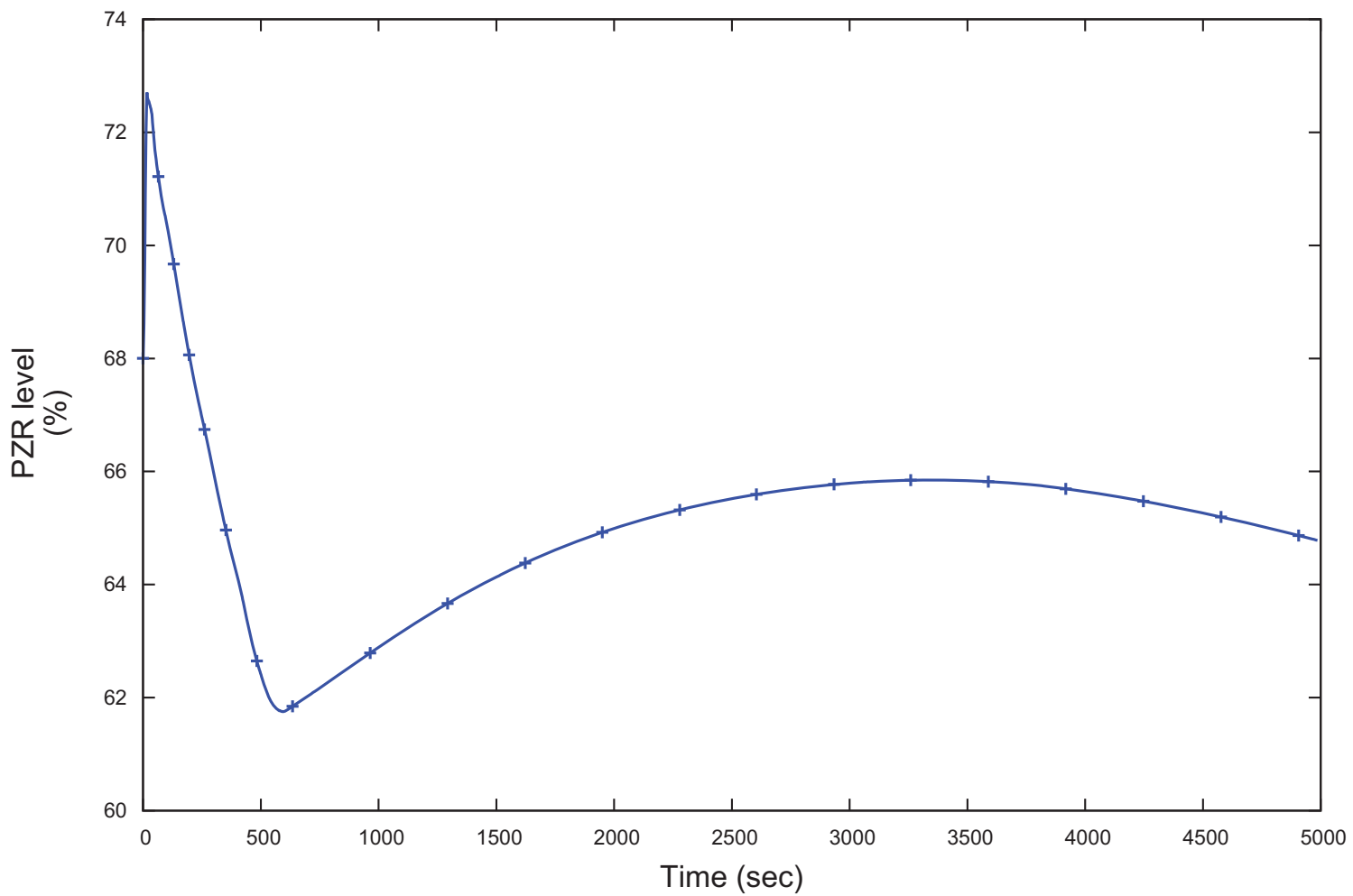


Figure 15.6-15: Steam Generator Tube Failure - Reactor Coolant System Flow - Limiting Reactor Coolant System Pressure Case

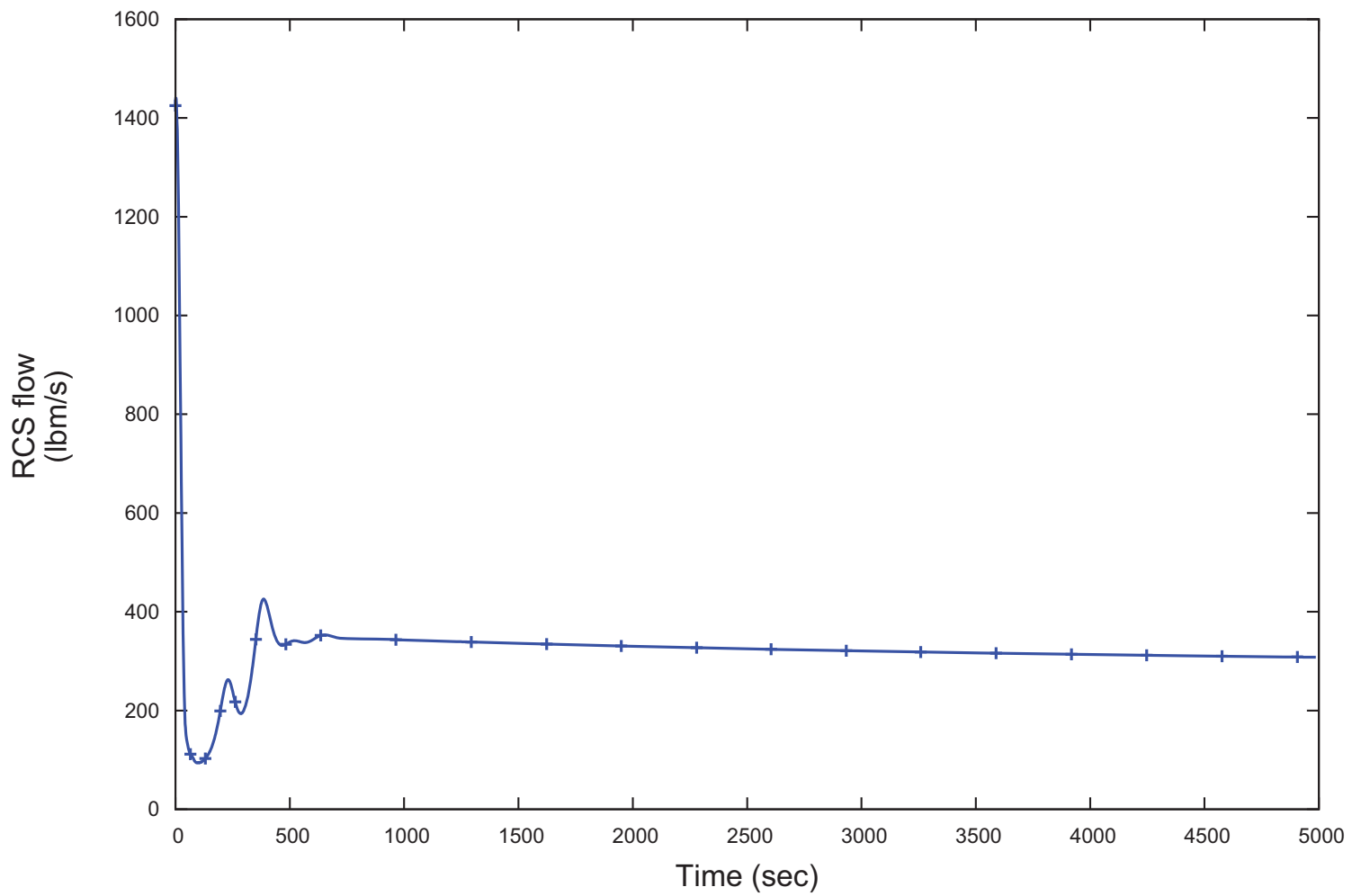


Figure 15.6-16: Steam Generator Tube Failure - Reactor Coolant System Average Temperature - Limiting Reactor Coolant System Pressure Case

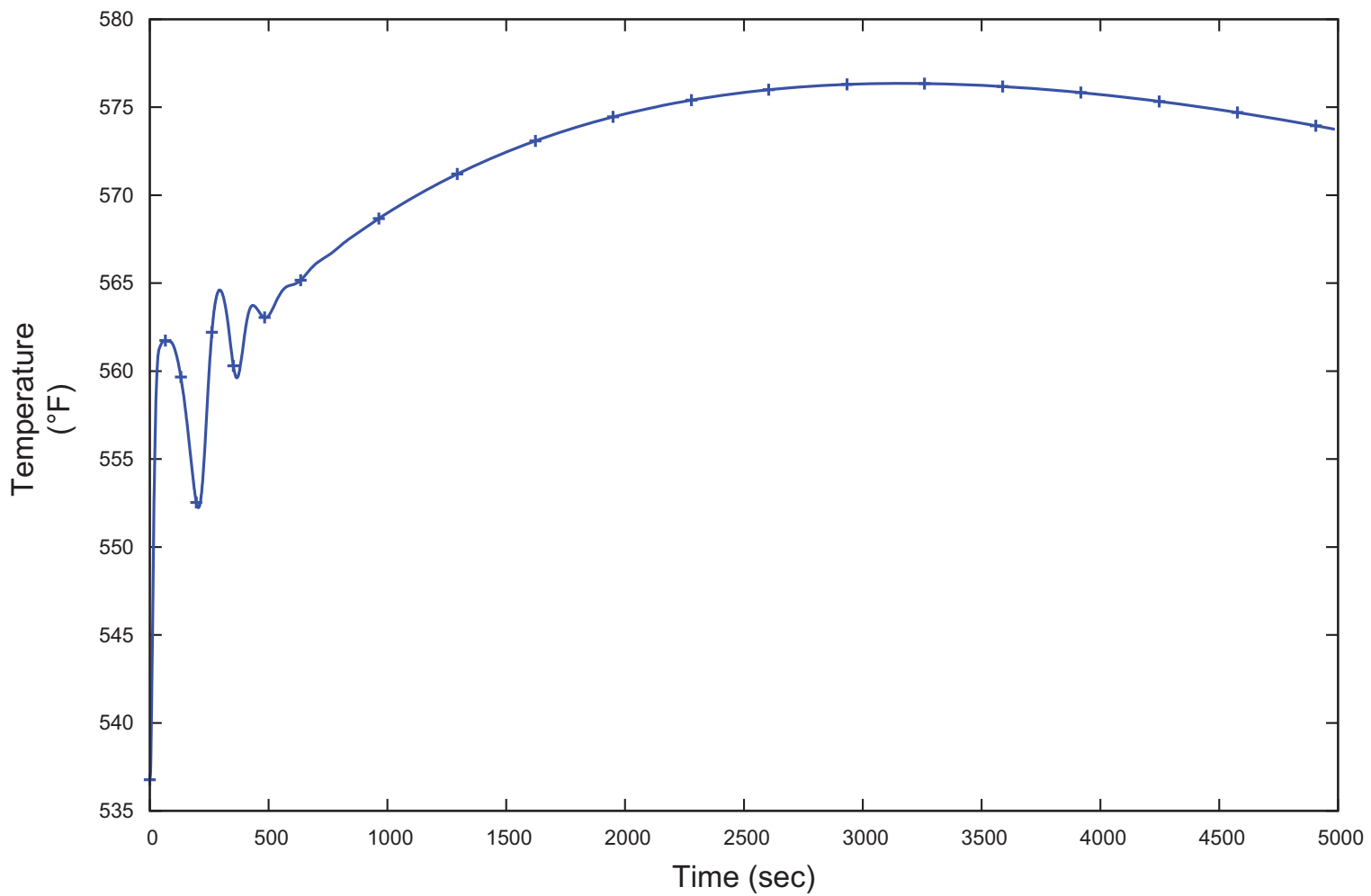


Figure 15.6-17: Steam Generator Tube Failure - Decay Heat Removal System Flow - Limiting Reactor Coolant System Pressure Case

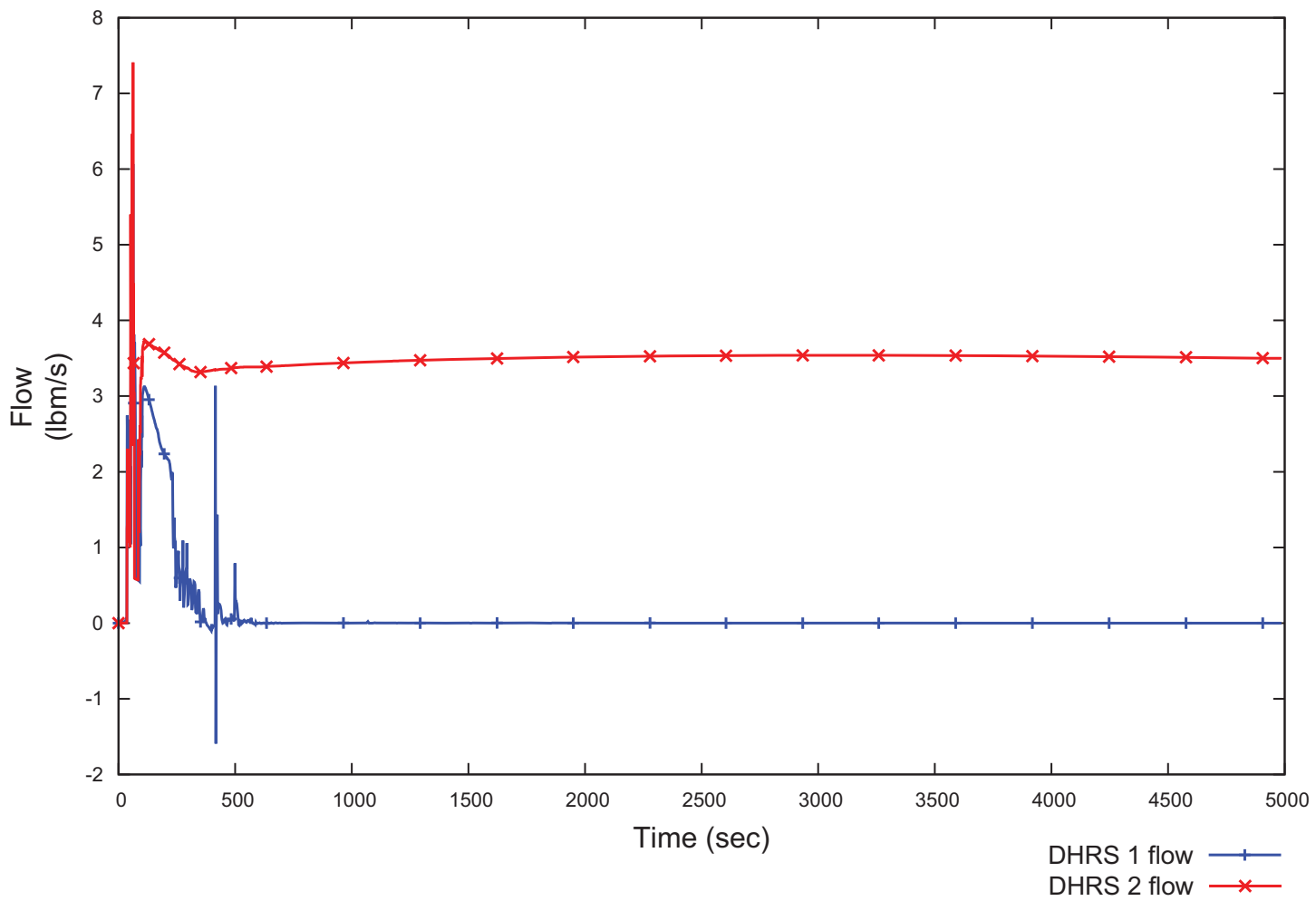


Figure 15.6-18: Steam Generator Tube Failure - Total Reactivity - Limiting Reactor Coolant System Pressure Case

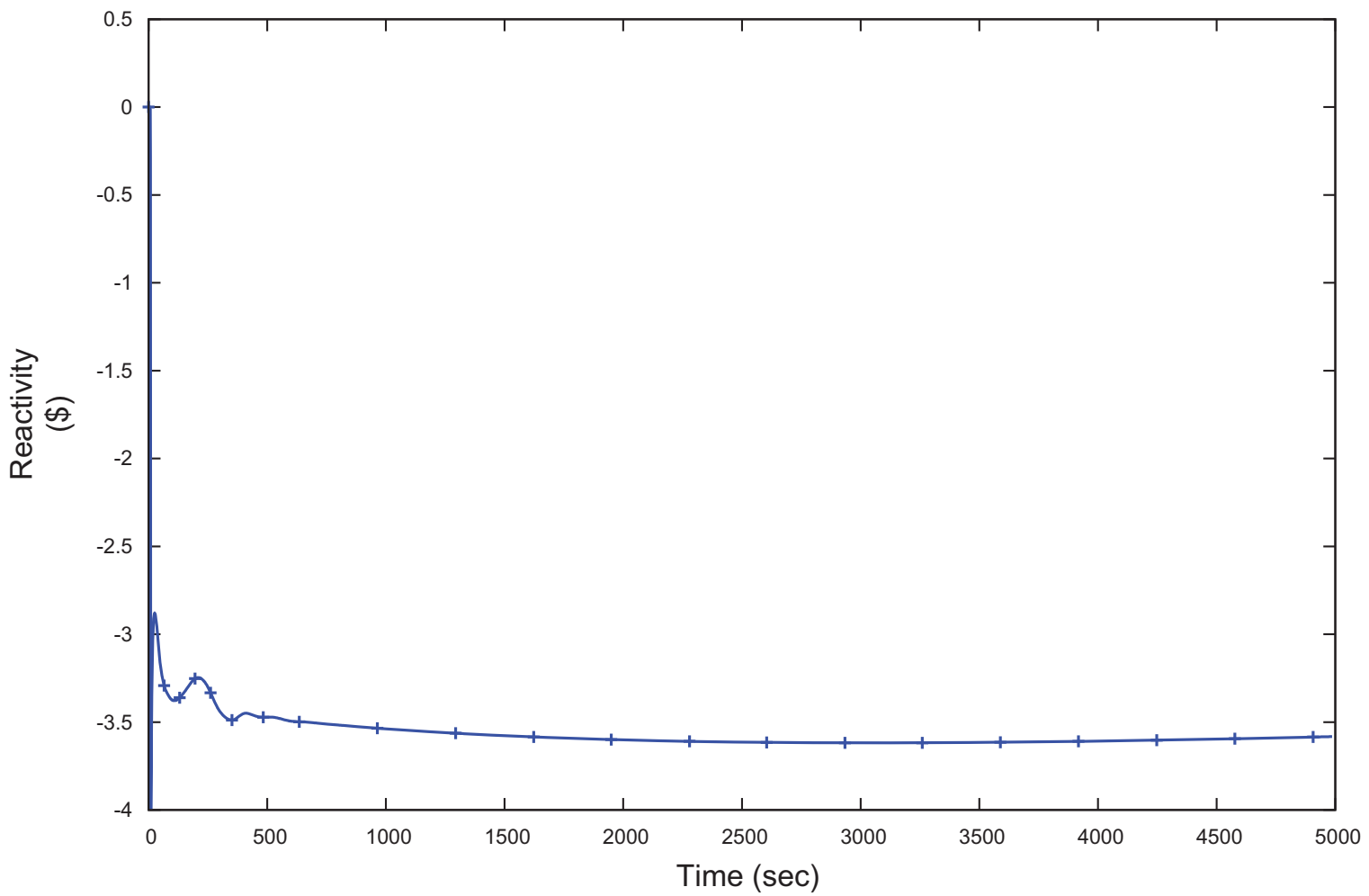


Figure 15.6-19: Steam Generator Tube Failure - Primary to Secondary Mass Flow - Limiting Steam Generator Pressure Case

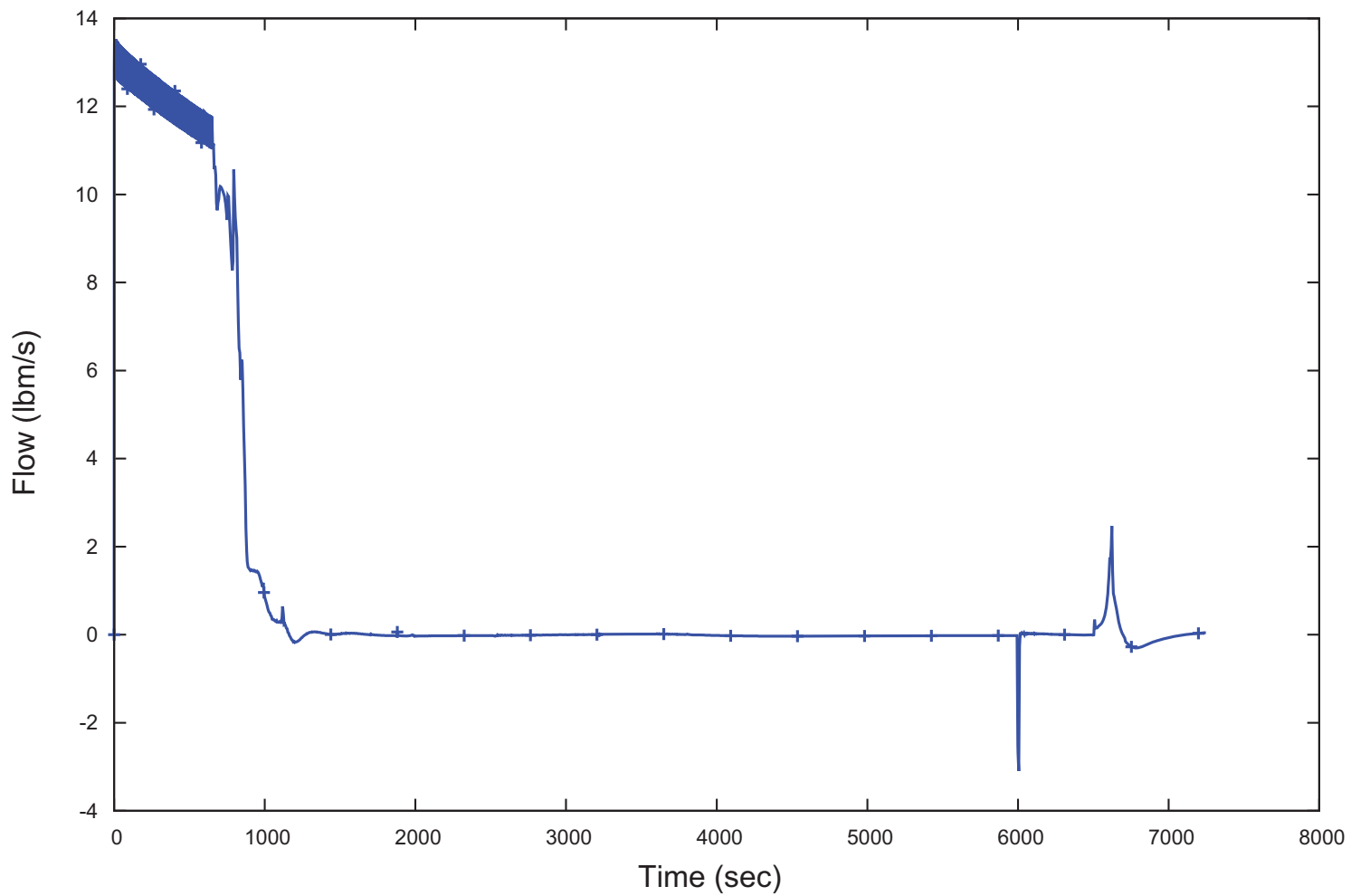


Figure 15.6-20: Steam Generator Tube Failure - Primary and Secondary Pressures - Limiting Steam Generator Pressure Case

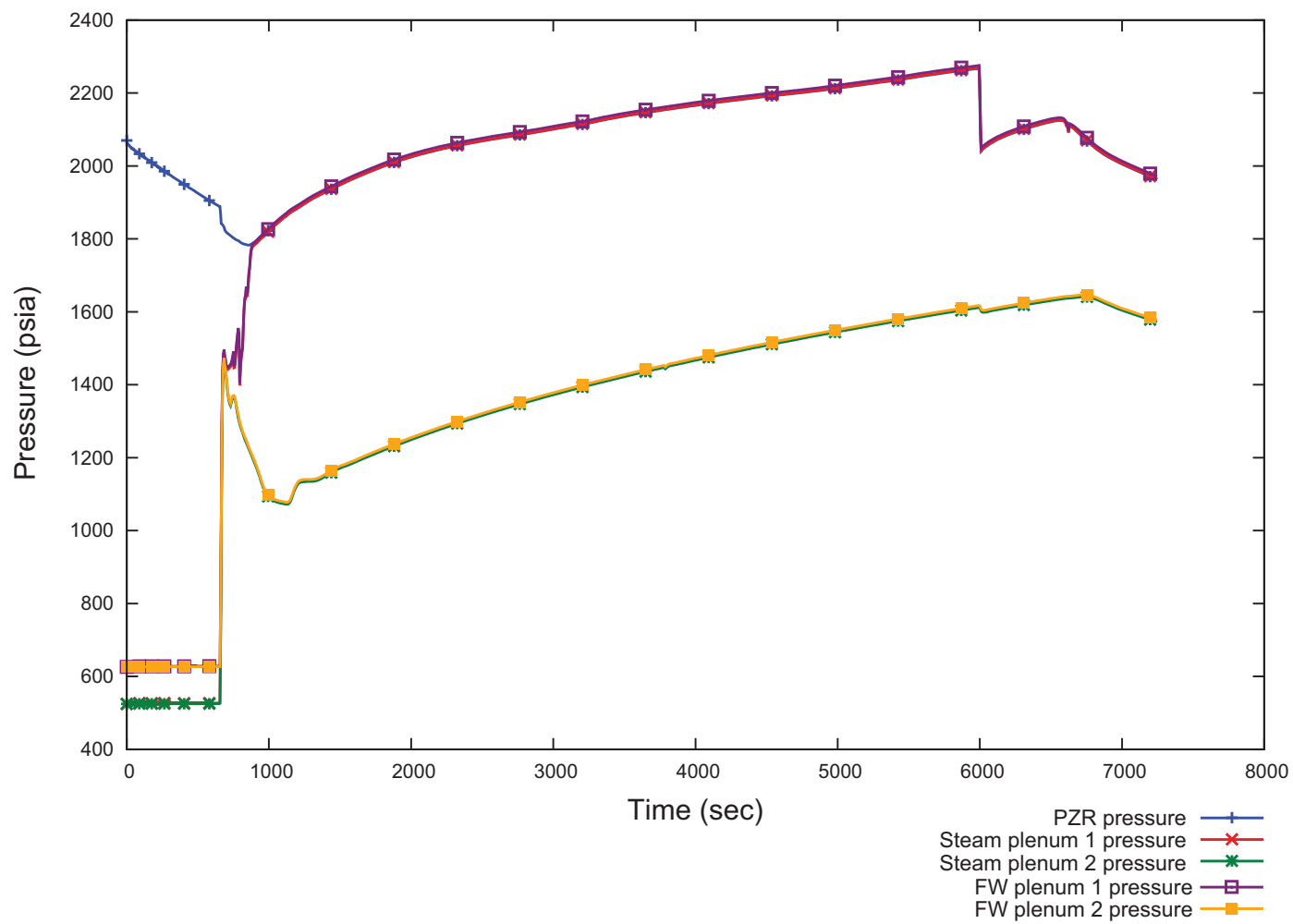


Figure 15.6-21: Steam Generator Tube Failure - Steam Generator Levels - Limiting Steam Generator Pressure Case

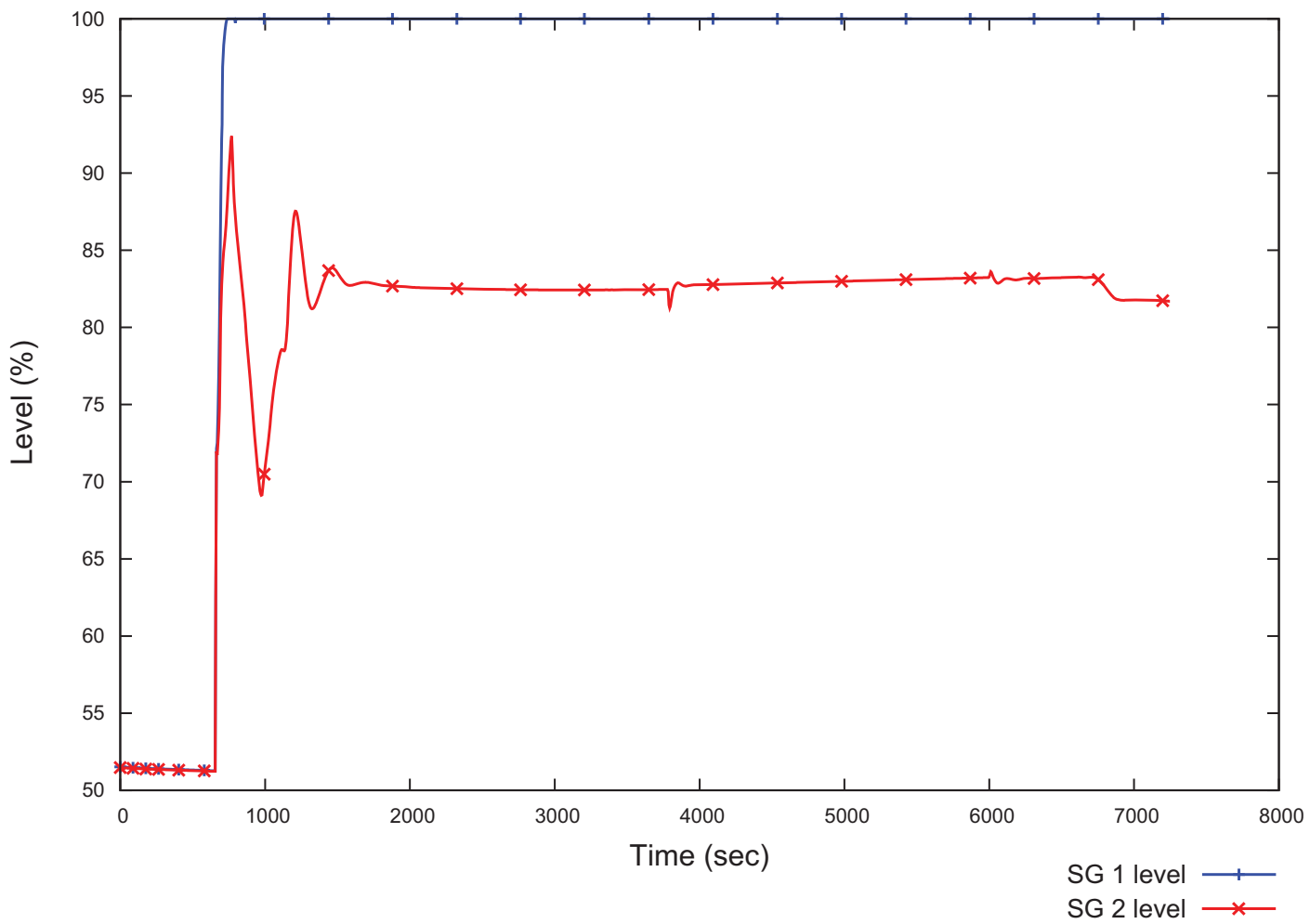


Figure 15.6-22: Steam Generator Tube Failure Pressurizer Level - Limiting Steam Generator Pressure Case

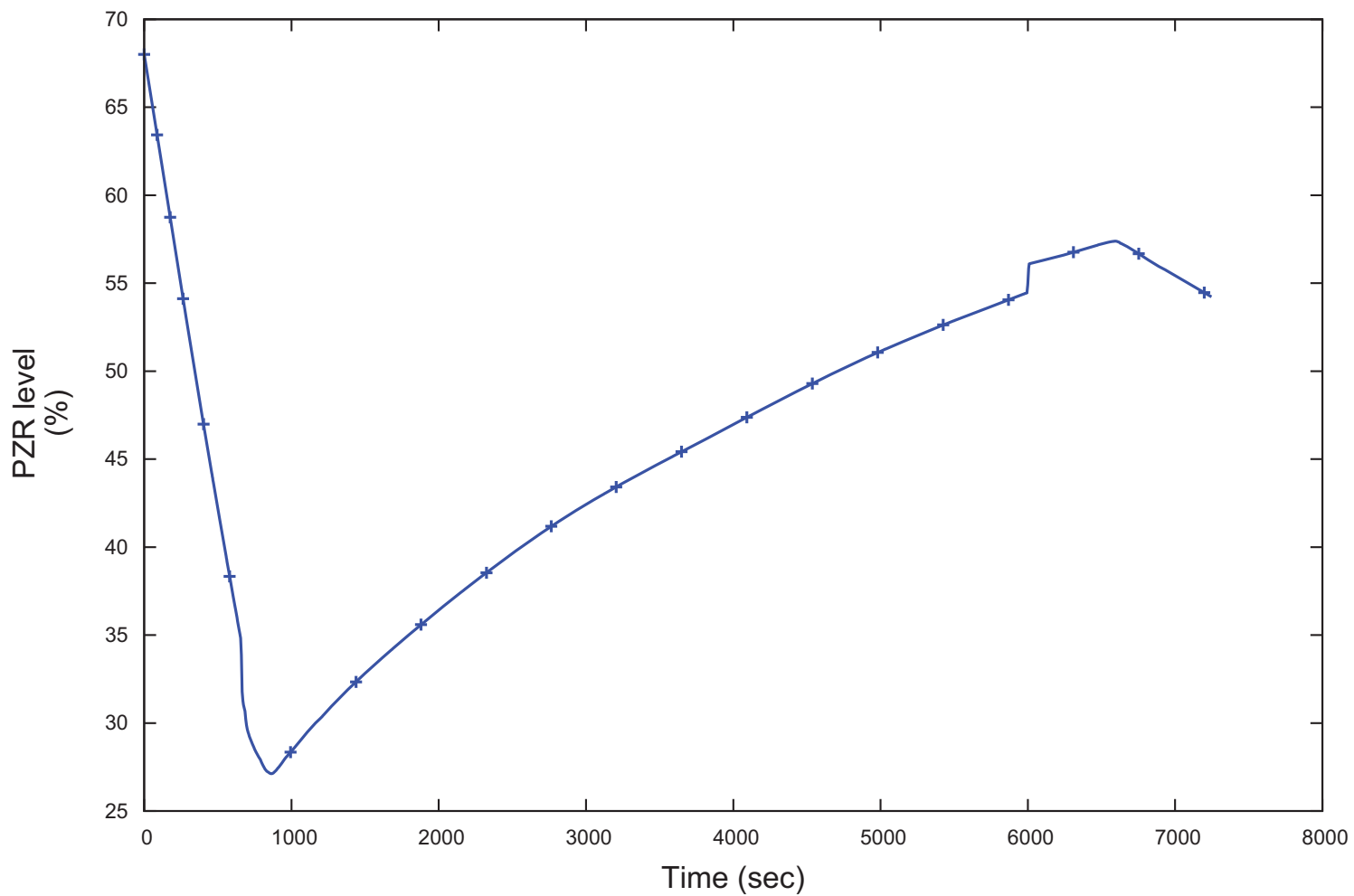


Figure 15.6-23: Steam Generator Tube Failure - Reactor Coolant System Flow - Limiting Steam Generator Pressure Case

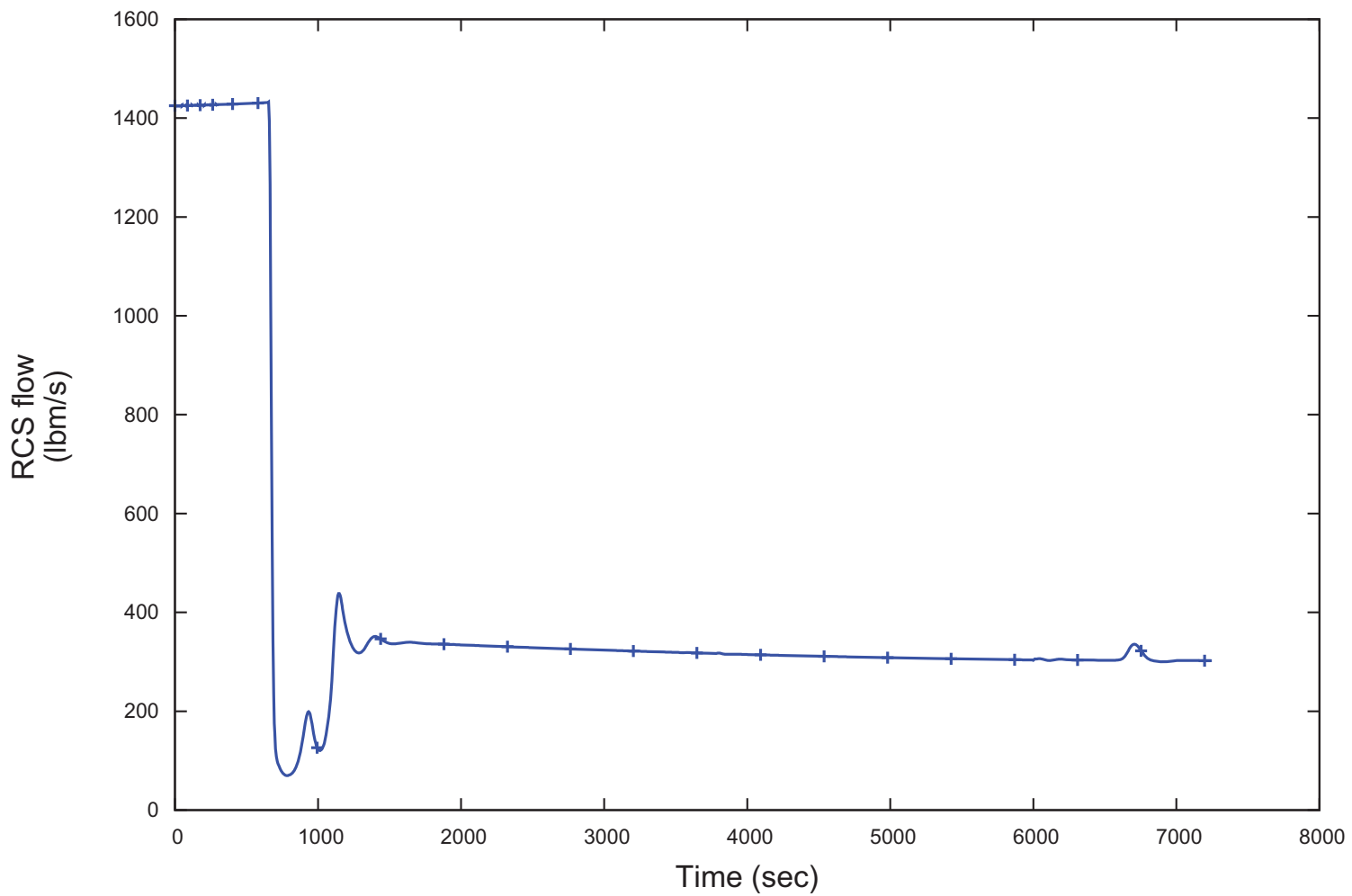


Figure 15.6-24: Steam Generator Tube Failure - Reactor Coolant System Average Temperature - Limiting Steam Generator Pressure Case

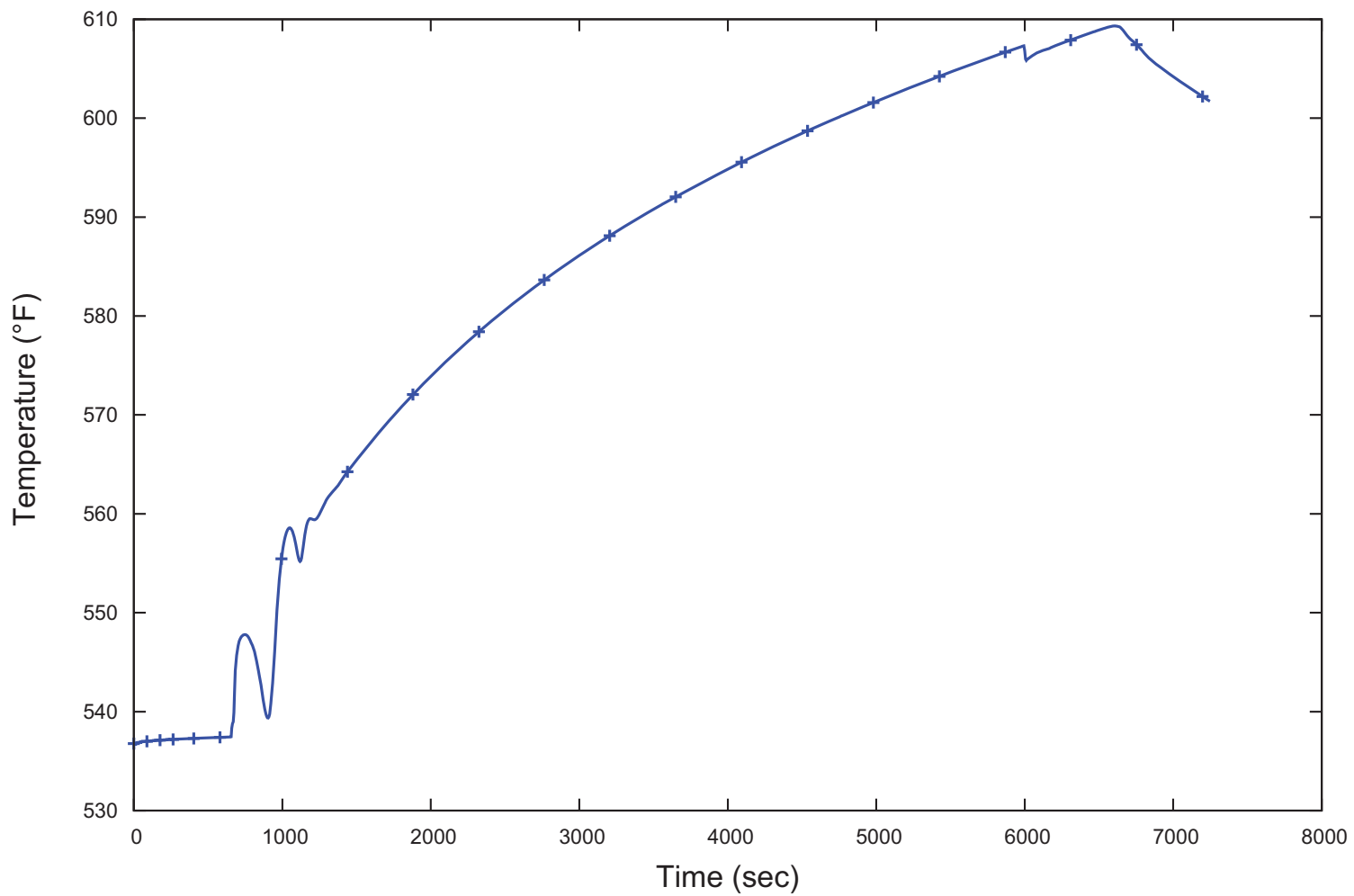


Figure 15.6-25: Steam Generator Tube Failure - Decay Heat Removal System Flow - Limiting Steam Generator Pressure Case

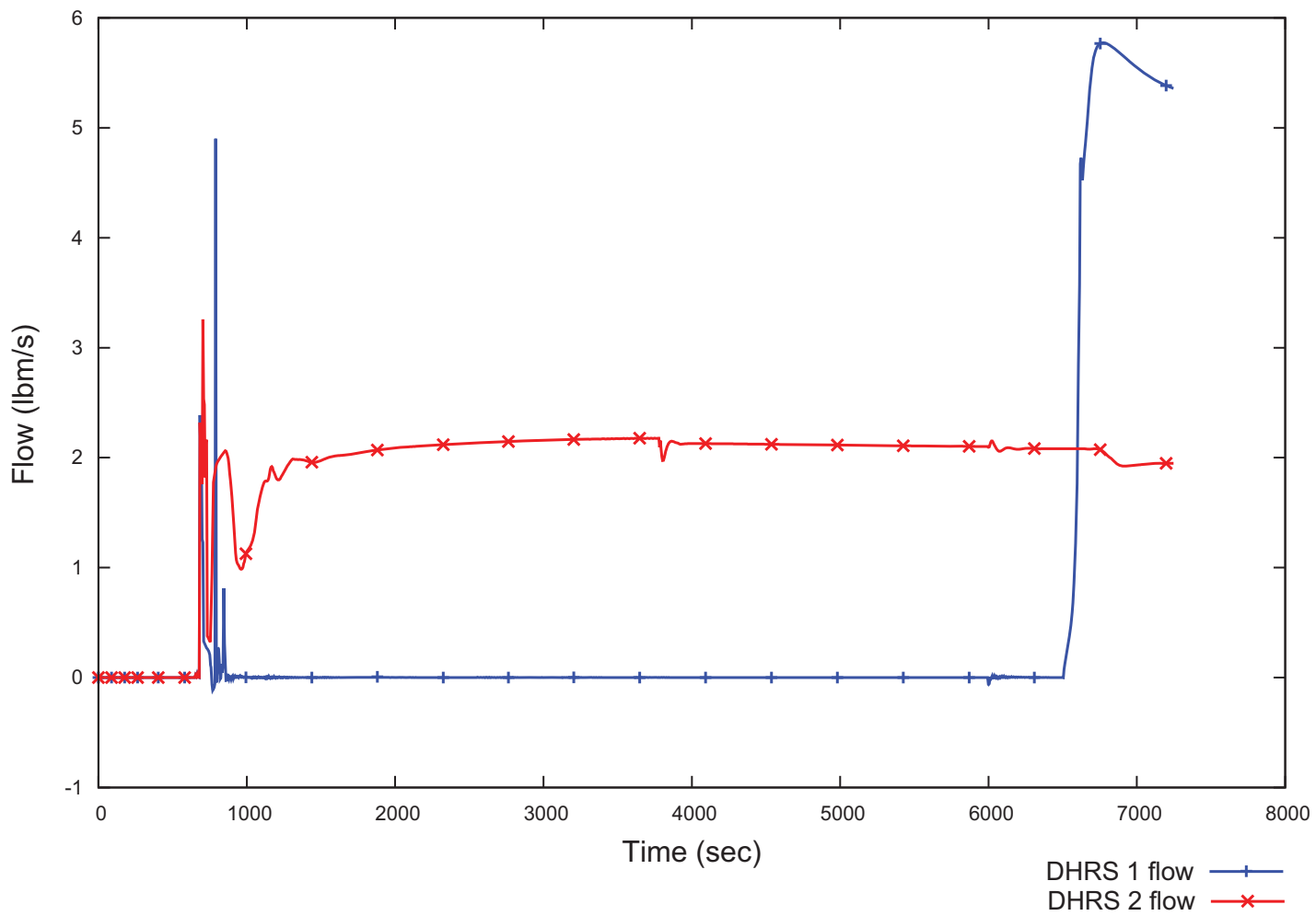


Figure 15.6-26: Steam Generator Tube Failure Total Reactivity - Limiting Steam Generator Pressure Case

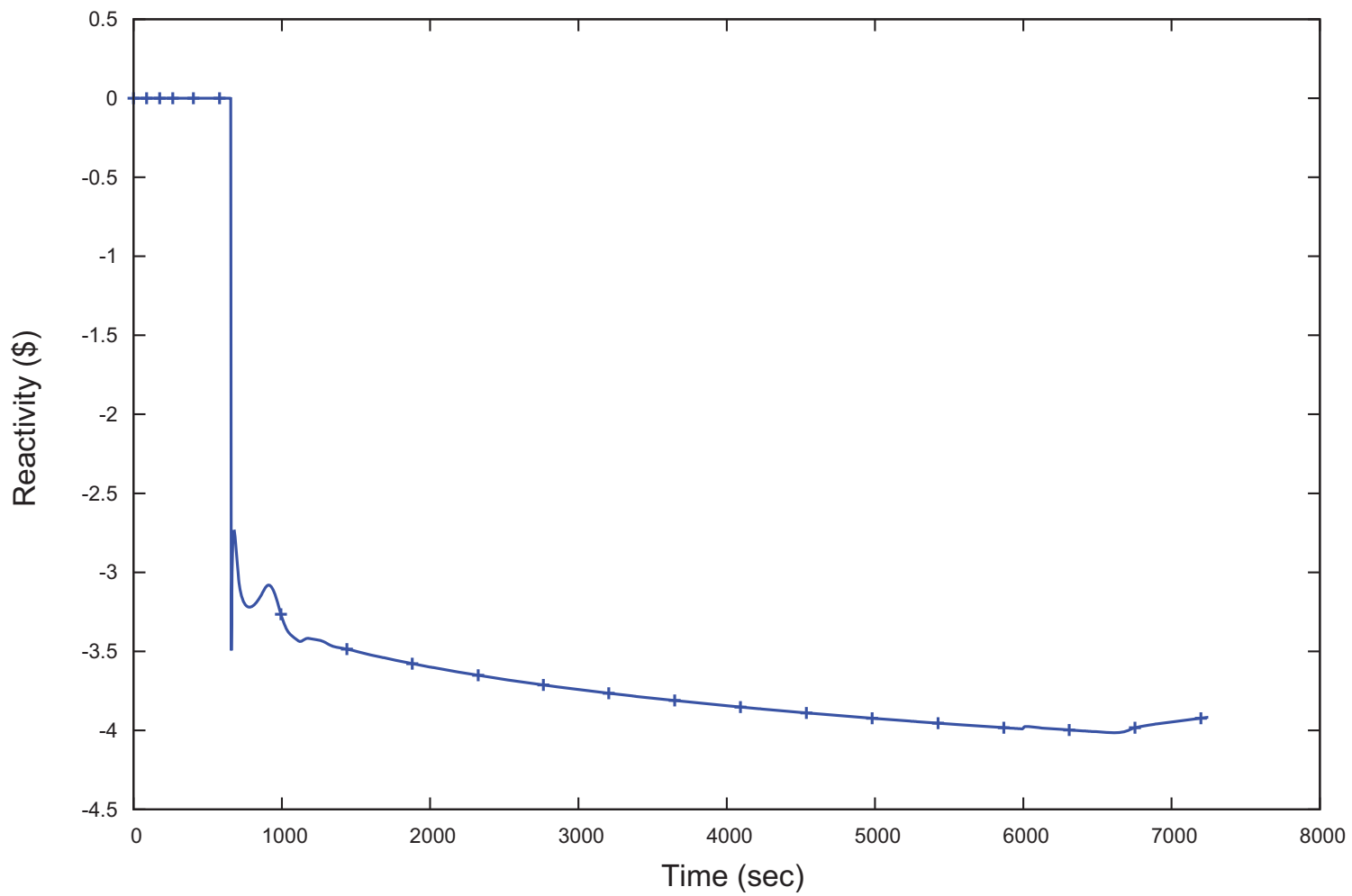


Figure 15.6-27: Loss of Coolant Accident - Discharge Line Break Flow (RPV to CNV) - Limiting Minimum Critical Heat Flux Ratio Case

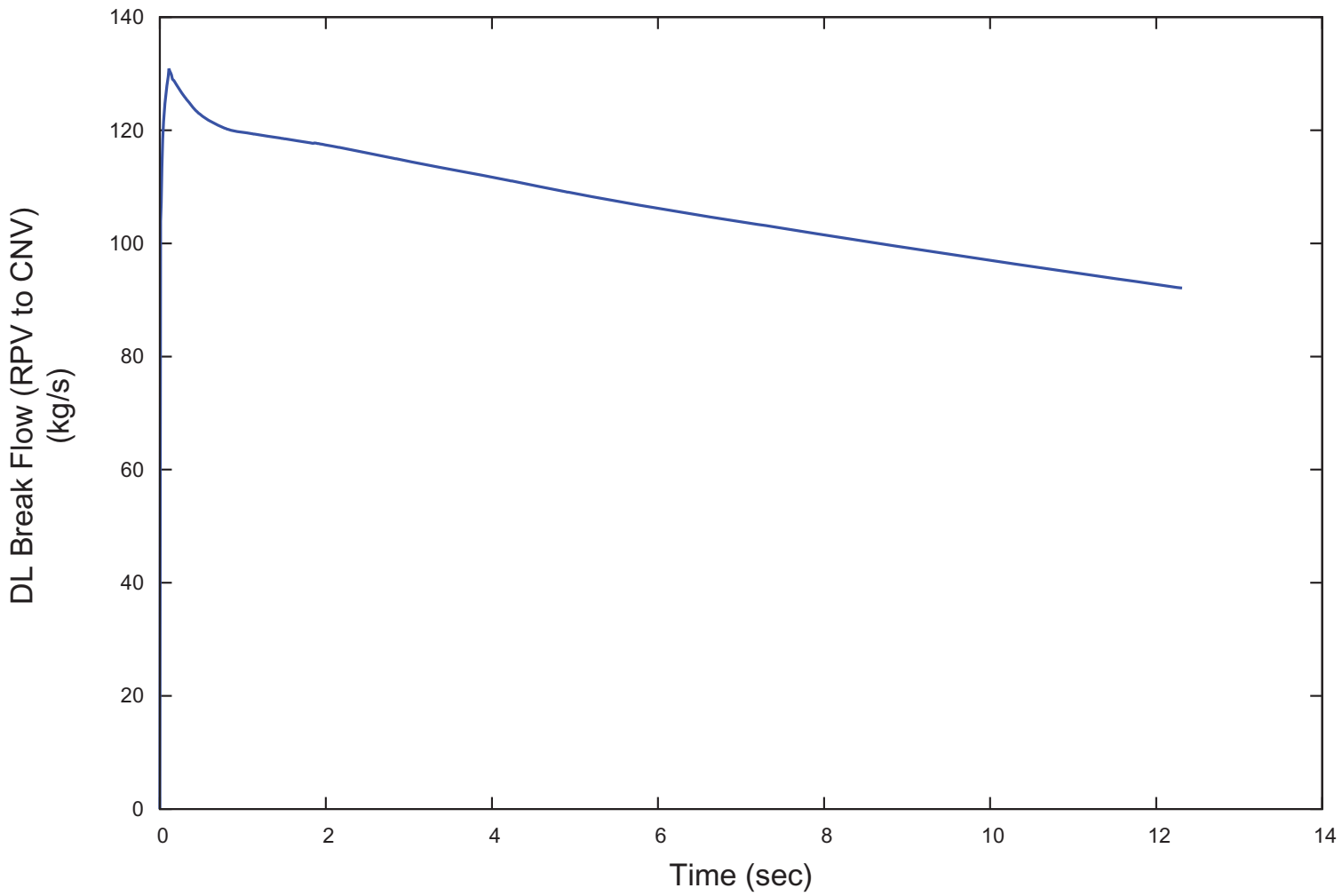


Figure 15.6-28: Loss of Coolant Accident - Discharge Line Break Flow (CVCS to CNV) - Limiting Minimum Critical Heat Flux Ratio Case

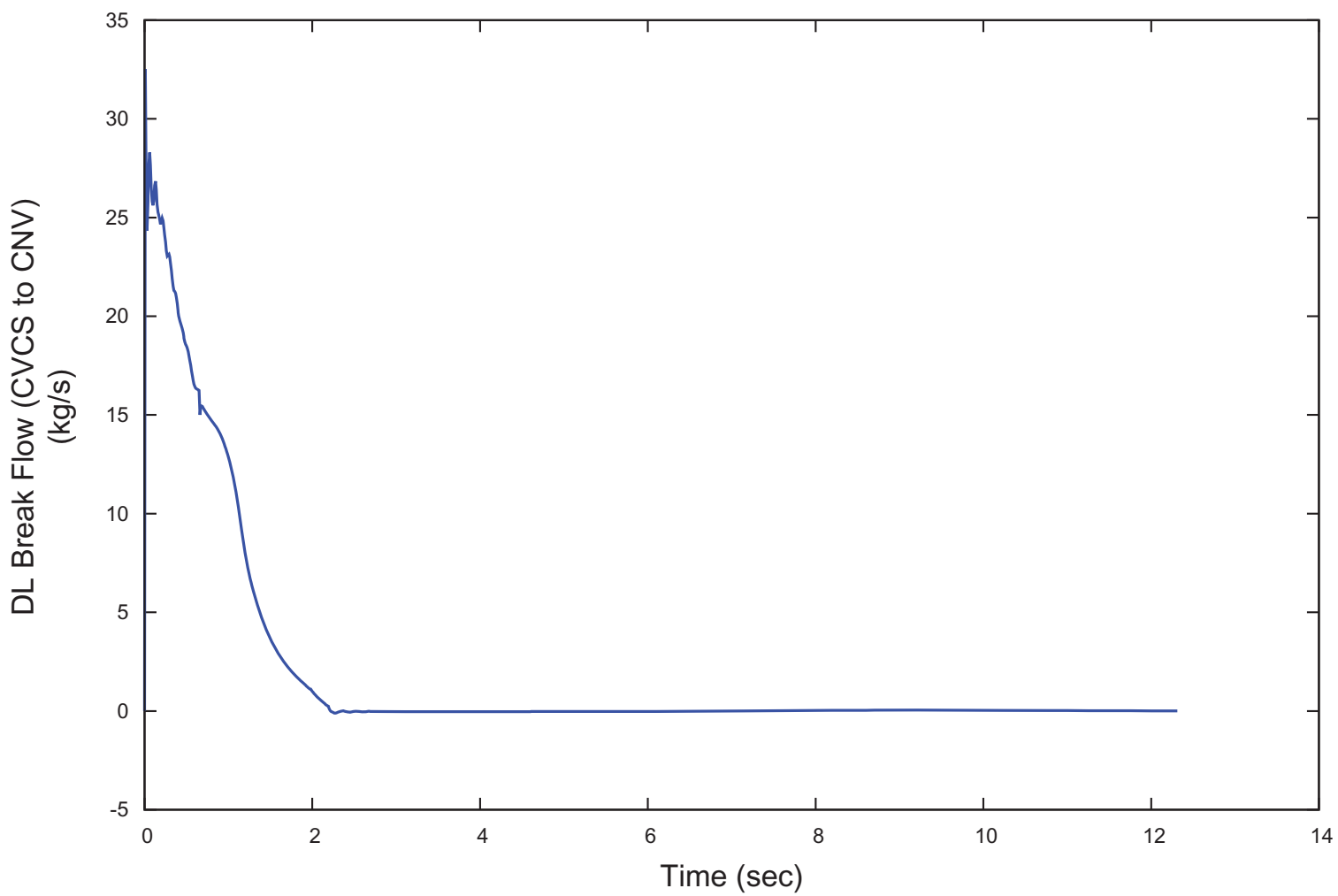


**Figure 15.6-29: Loss of Coolant Accident - Discharge Line Break Reactor Vent Valve Initial Flow
(Typical of One RVV) - Limiting Minimum Critical Heat Flux Ratio Case**

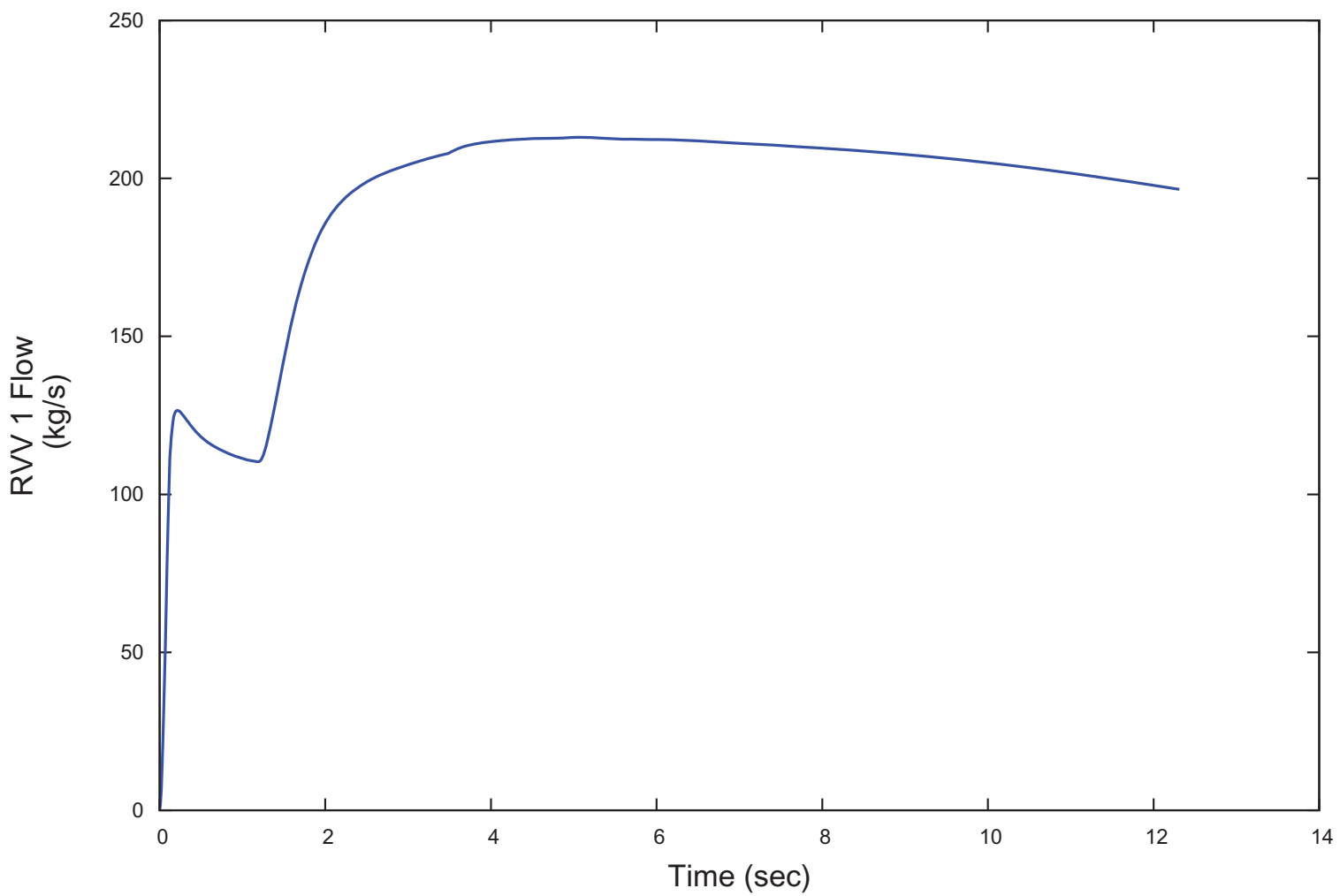


Figure 15.6-30: Loss of Coolant Accident - Discharge Line Break Reactor Vent Valve Flow (Typical of One RVV) - Limiting Minimum Critical Heat Flux Ratio Case

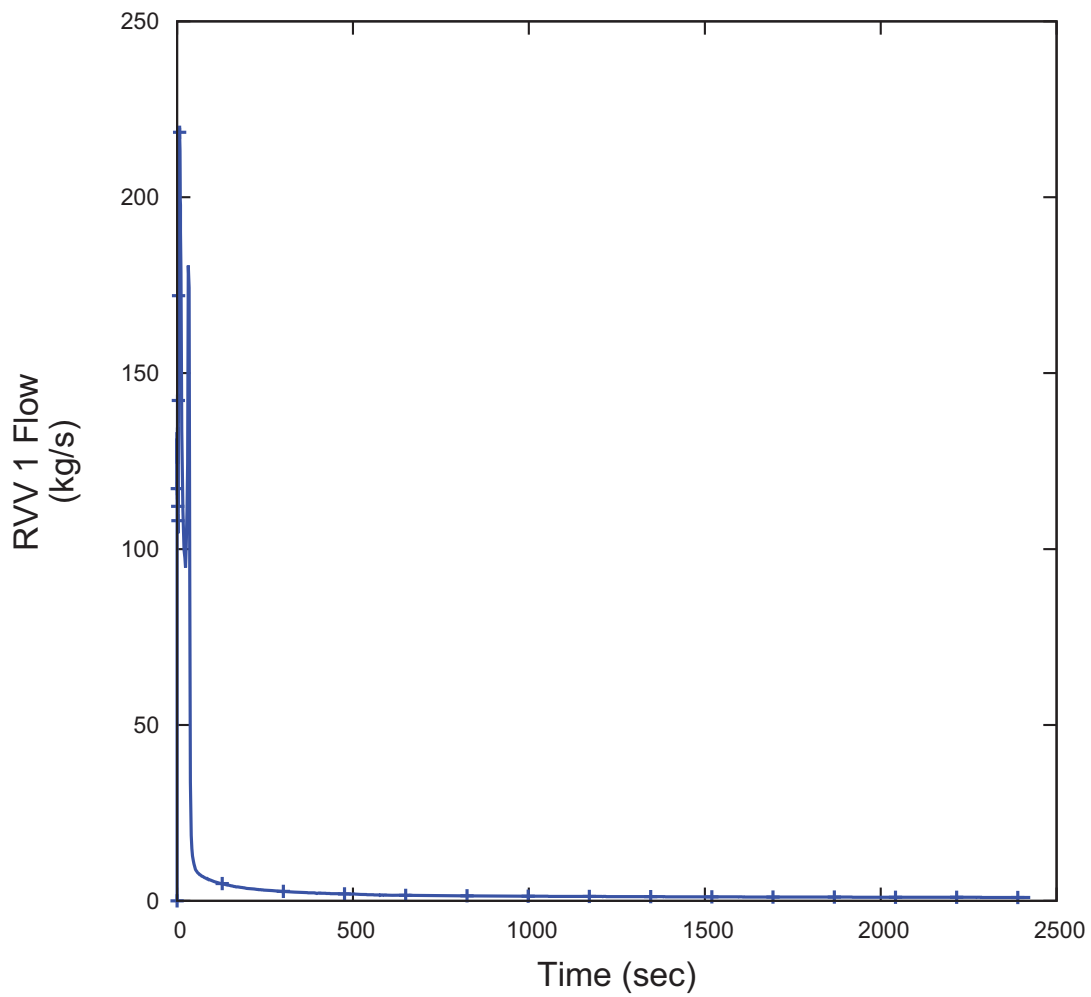


Figure 15.6-31: Loss of Coolant Accident - Discharge Line Break Reactor Recirculation Valve Flow (Typical of One RRV) - Limiting Minimum Critical Heat Flux Ratio Case

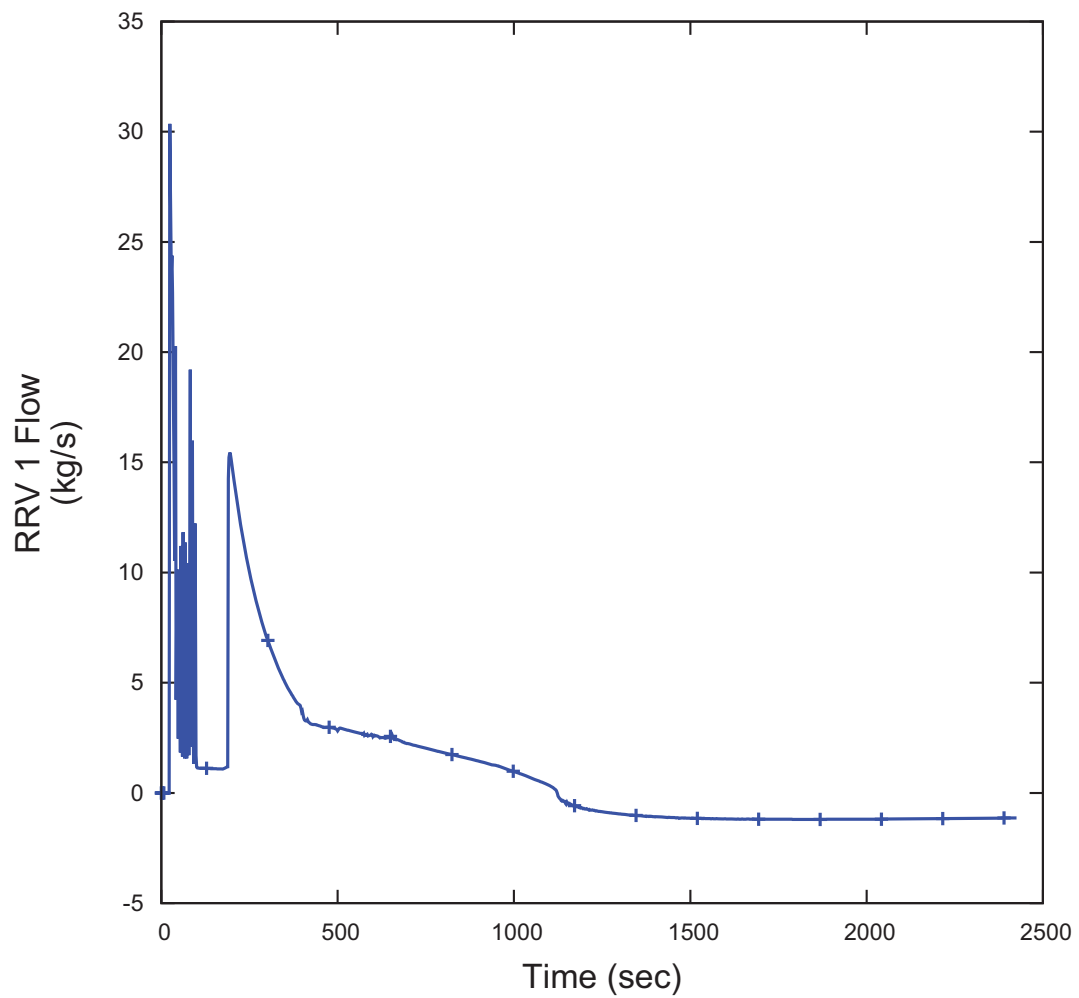


Figure 15.6-32: Loss of Coolant Accident - Discharge Line Break Reactor Pressure Vessel Pressure - Limiting Minimum Critical Heat Flux Ratio Case

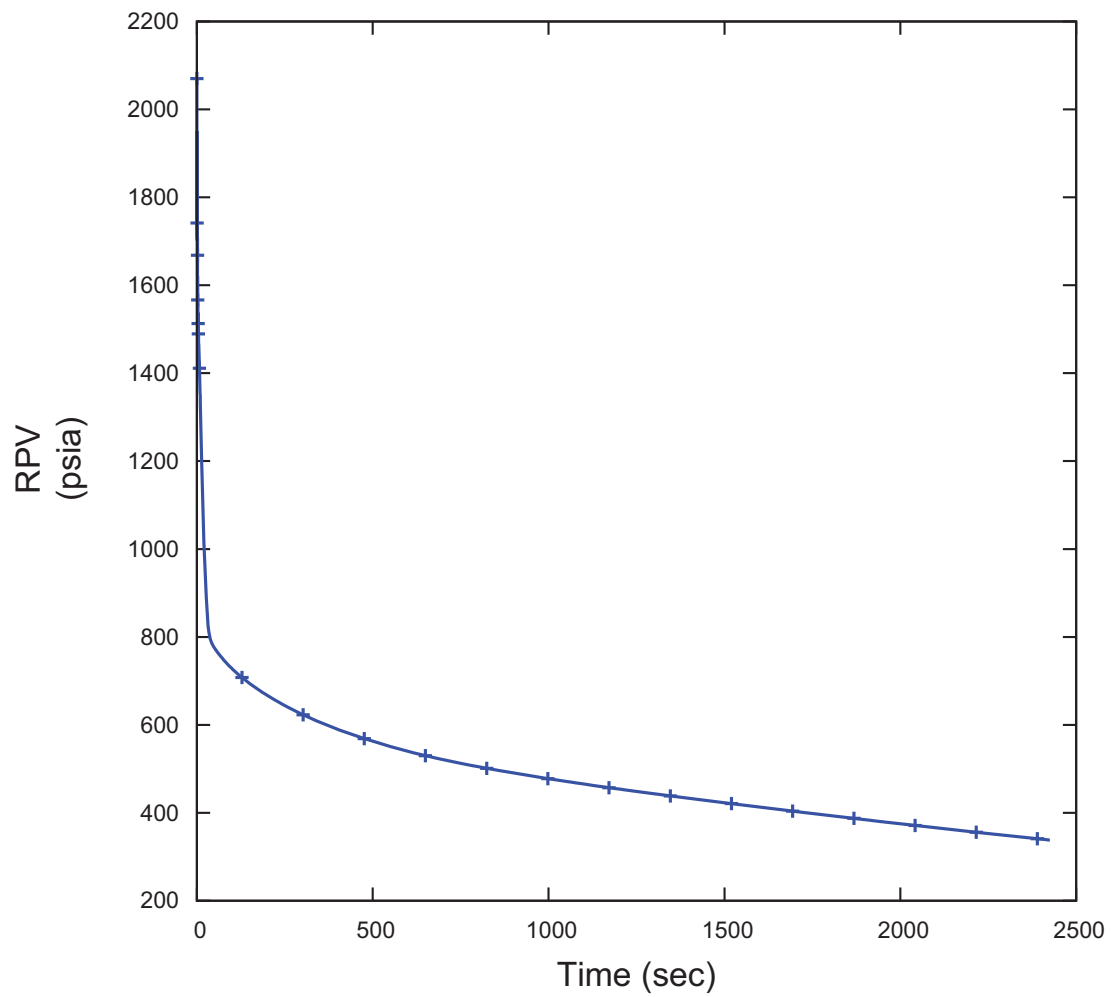


Figure 15.6-33: Loss of Coolant Accident - Discharge Line Break Containment Pressure - Limiting Minimum Critical Heat Flux Ratio Case

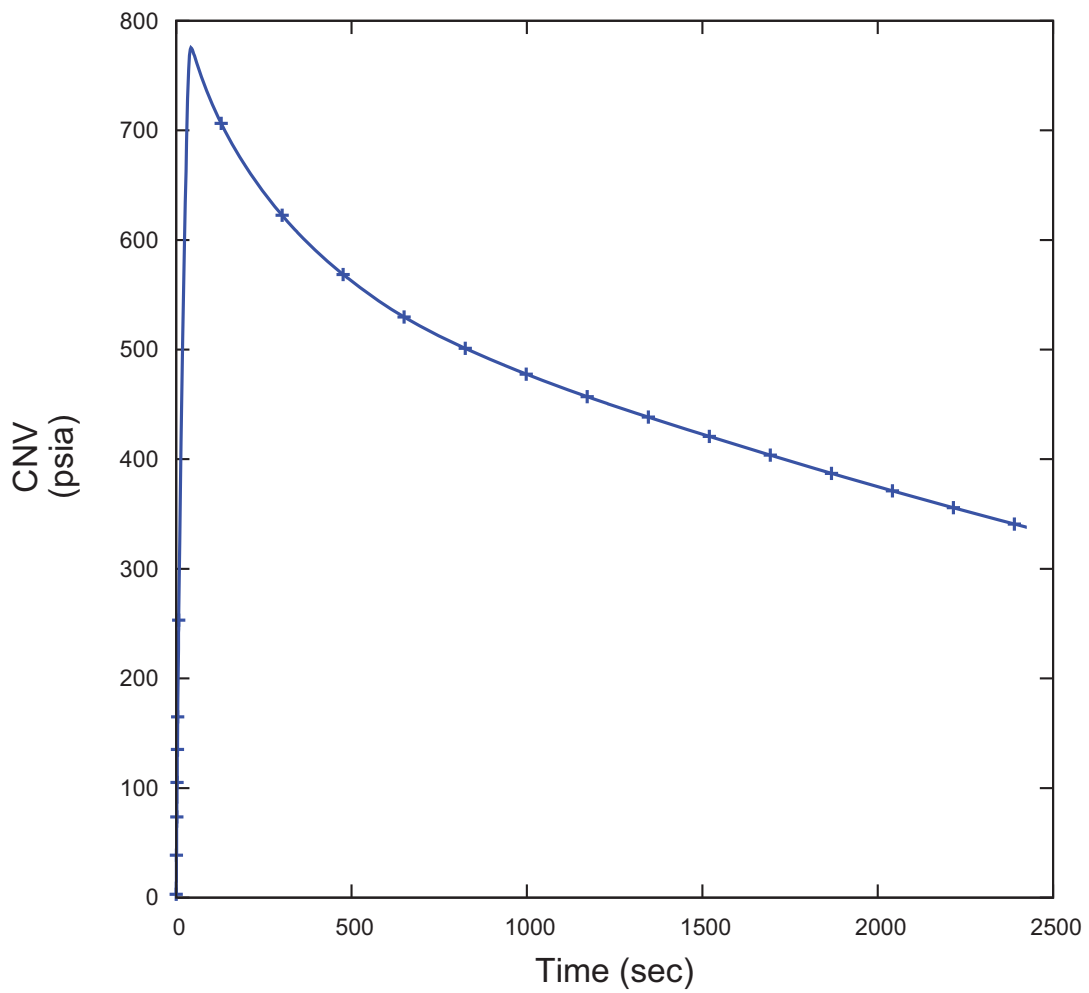


Figure 15.6-34: Loss of Coolant Accident - Discharge Line Break Steam Generator Pressure (Typical of One SG) - Limiting Minimum Critical Heat Flux Ratio Case

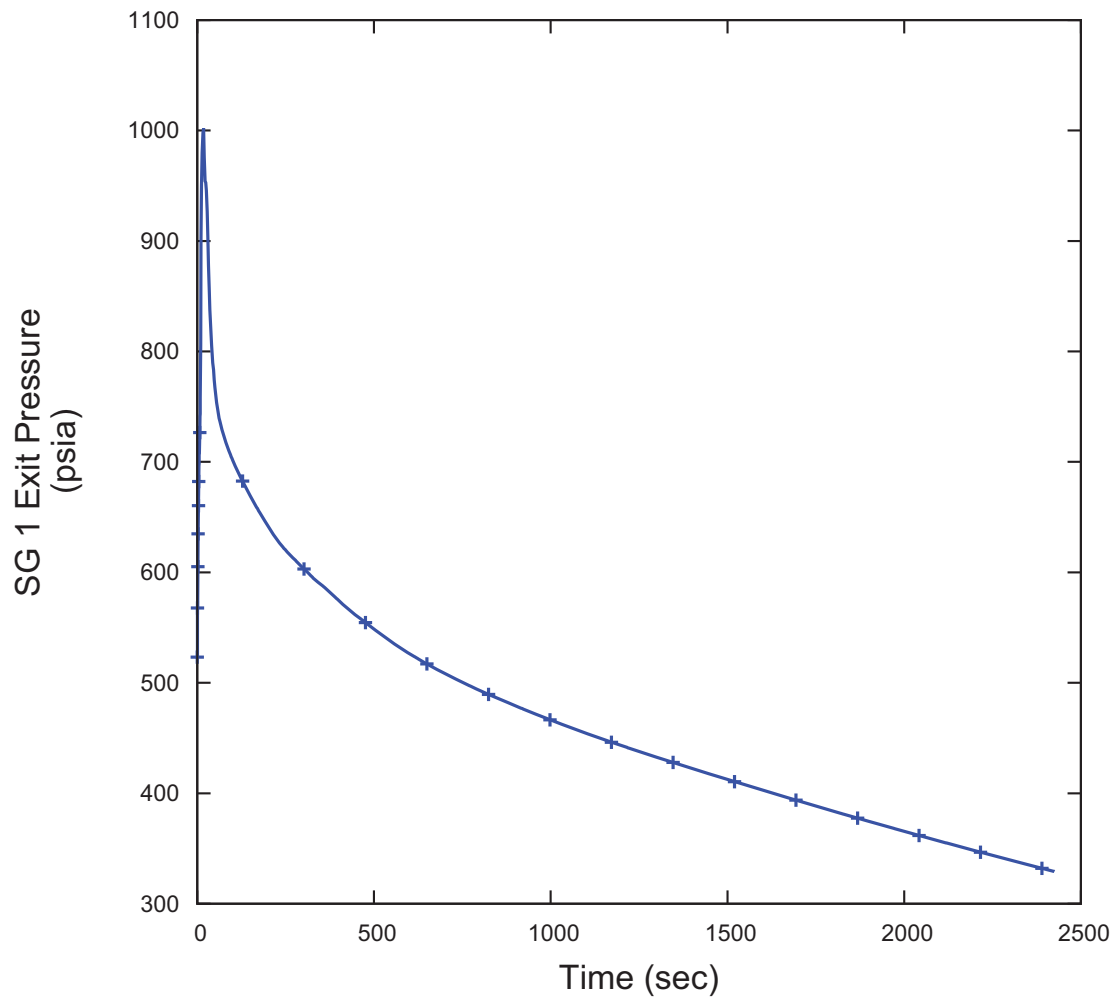


Figure 15.6-35: Loss of Coolant Accident - Discharge Line Break Reactor Pressure Vessel Riser Level - Limiting Minimum Critical Heat Flux Ratio Case

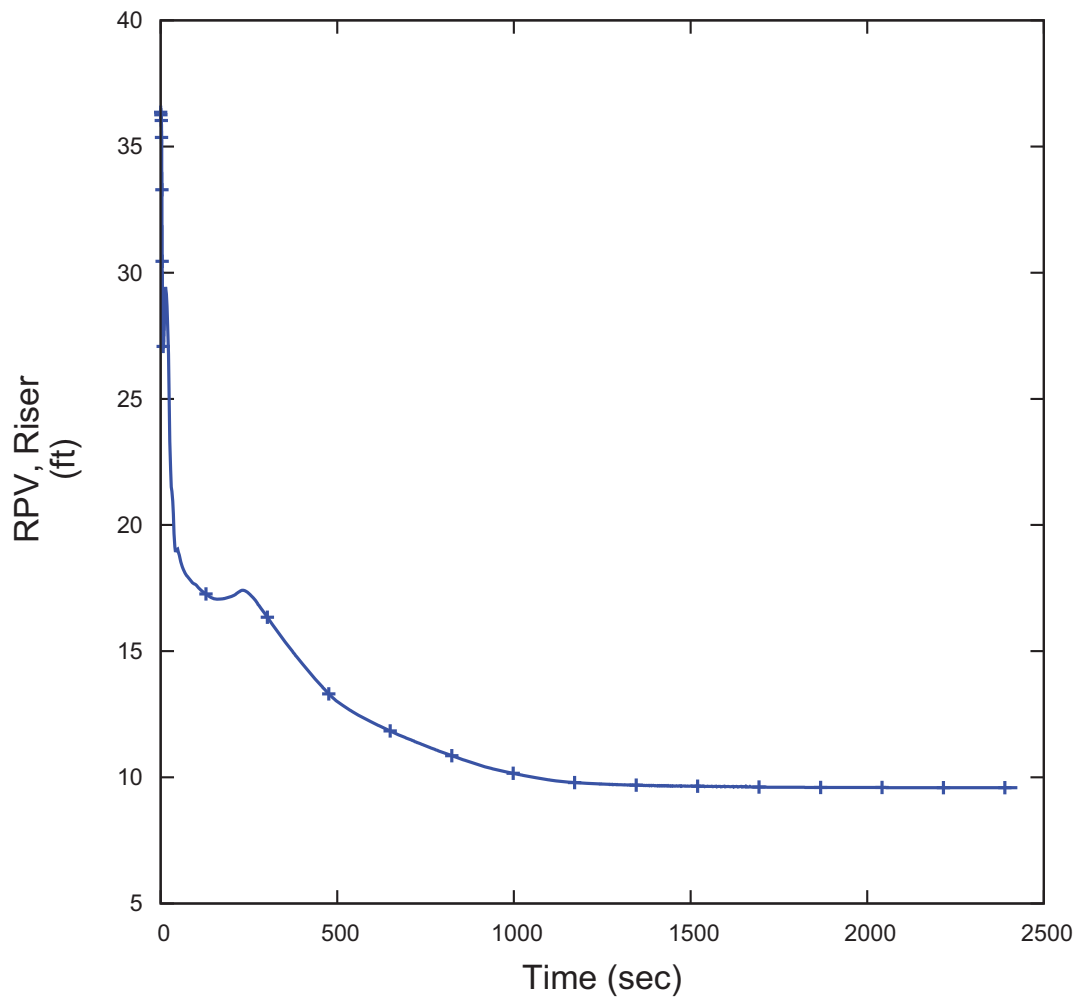


Figure 15.6-36: Loss of Coolant Accident - Discharge Line Break Containment Level - Limiting Minimum Critical Heat Flux Ratio Case

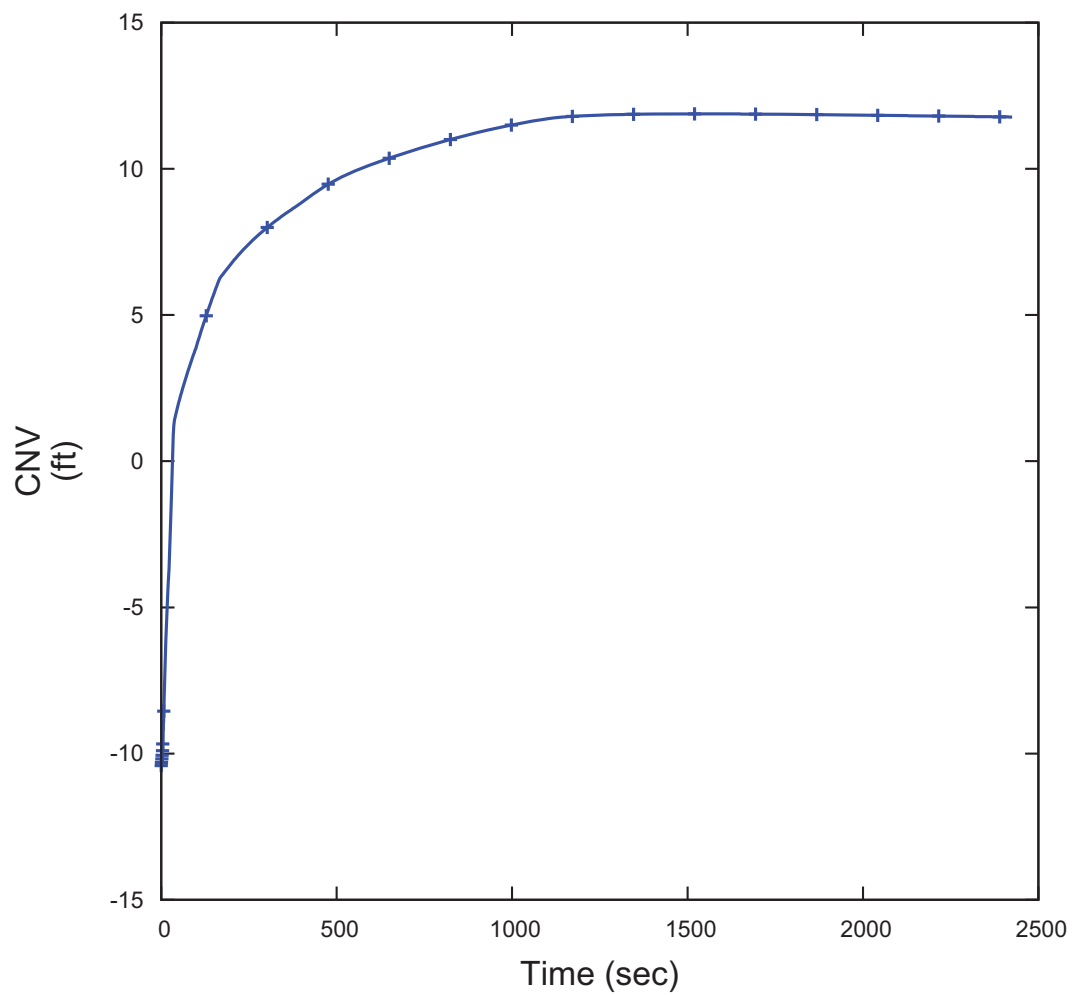


Figure 15.6-37: Loss of Coolant Accident - Discharge Line Break Reactor Power - Limiting Minimum Critical Heat Flux Ratio Case

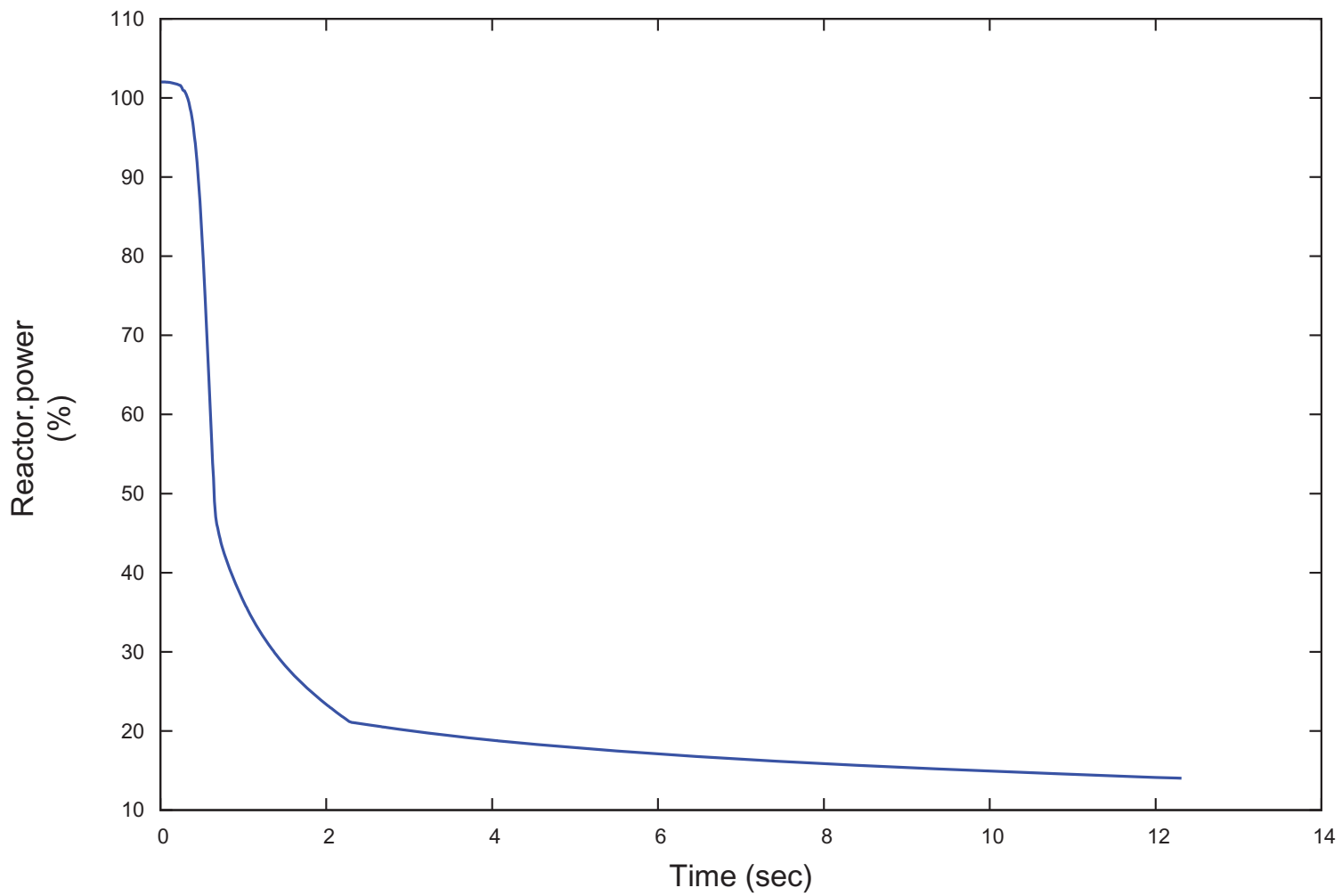


Figure 15.6-38: Loss of Coolant Accident - Discharge Line Break Event Reactivity - Limiting Minimum Critical Heat Flux Ratio Case

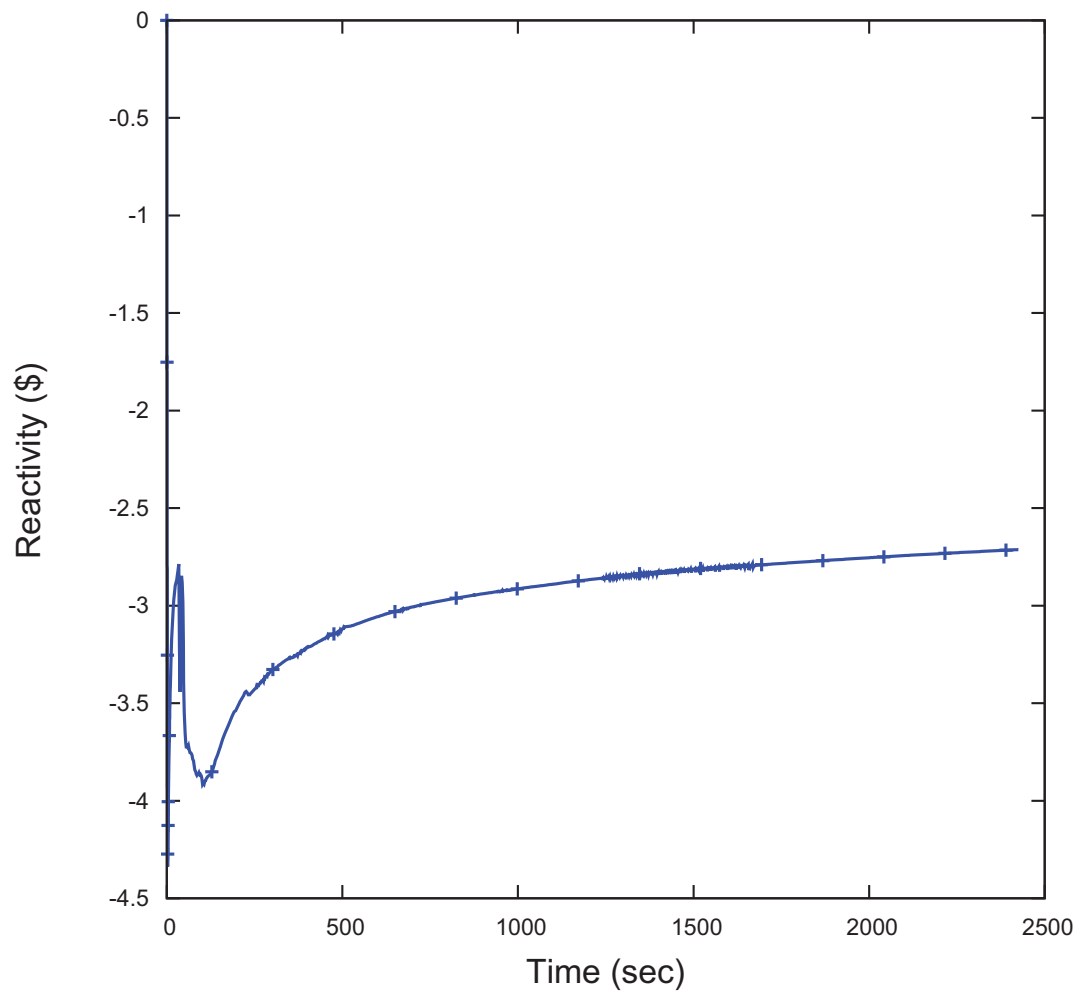


Figure 15.6-39: Loss of Coolant Accident - Discharge Line Break Reactor Coolant System Flow - Limiting Minimum Critical Heat Flux Ratio Case

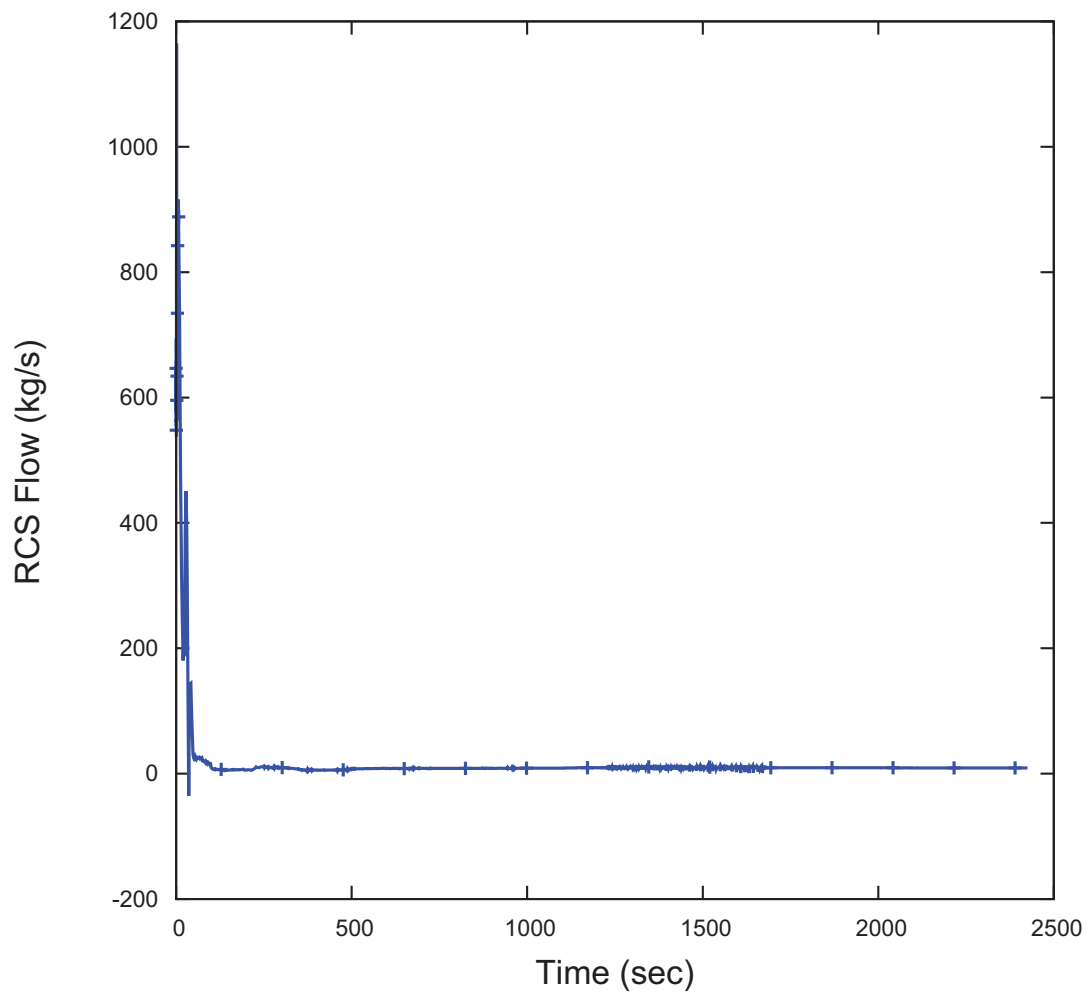


Figure 15.6-40: Loss of Coolant Accident – Discharge Line Break Minimum Critical Heat Flux Ratio - Limiting Minimum Critical Heat Flux Ratio Case

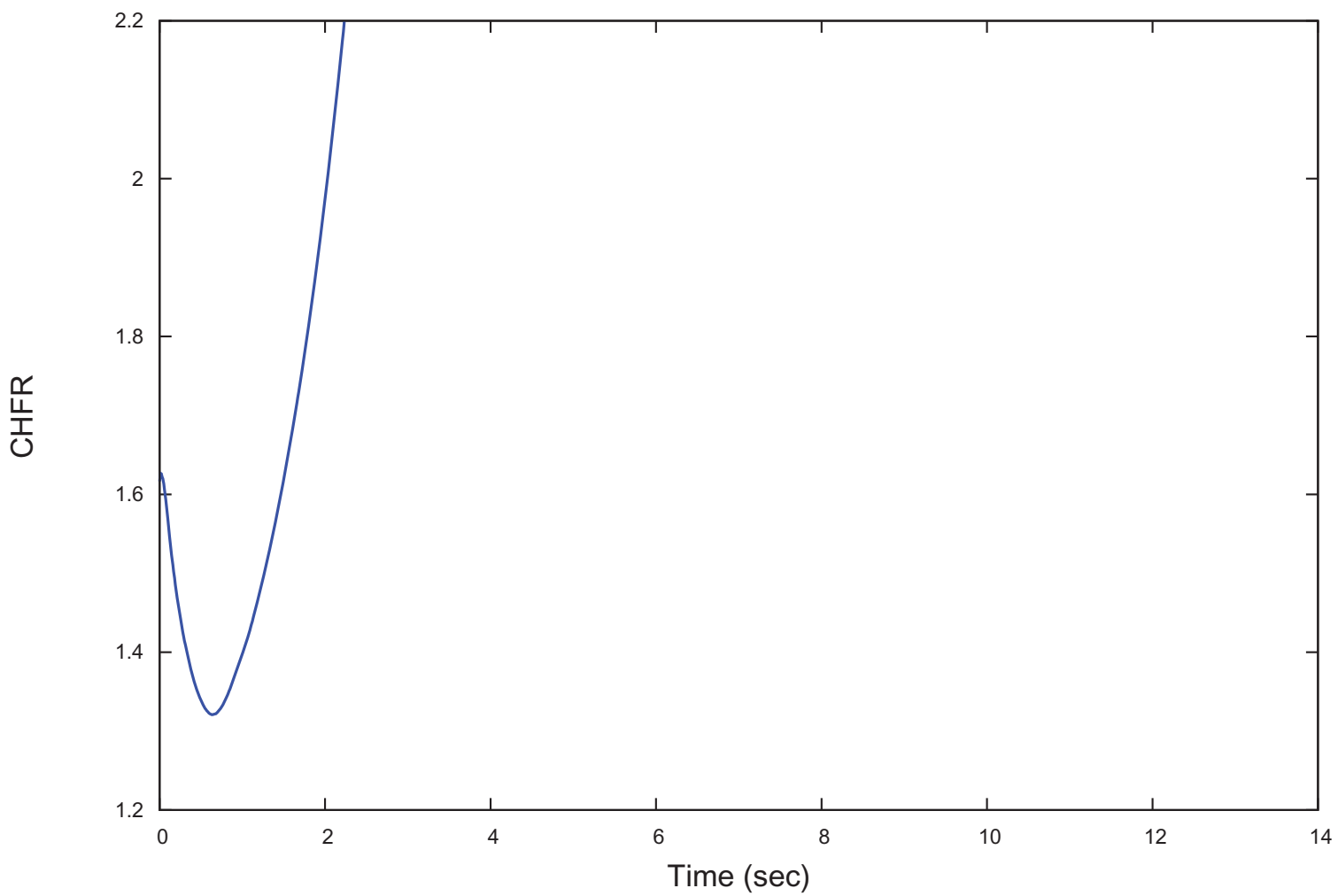


Figure 15.6-41: Loss of Coolant Accident - Discharge Line Break Fuel Temperature - Limiting Minimum Critical Heat Flux Ratio Case

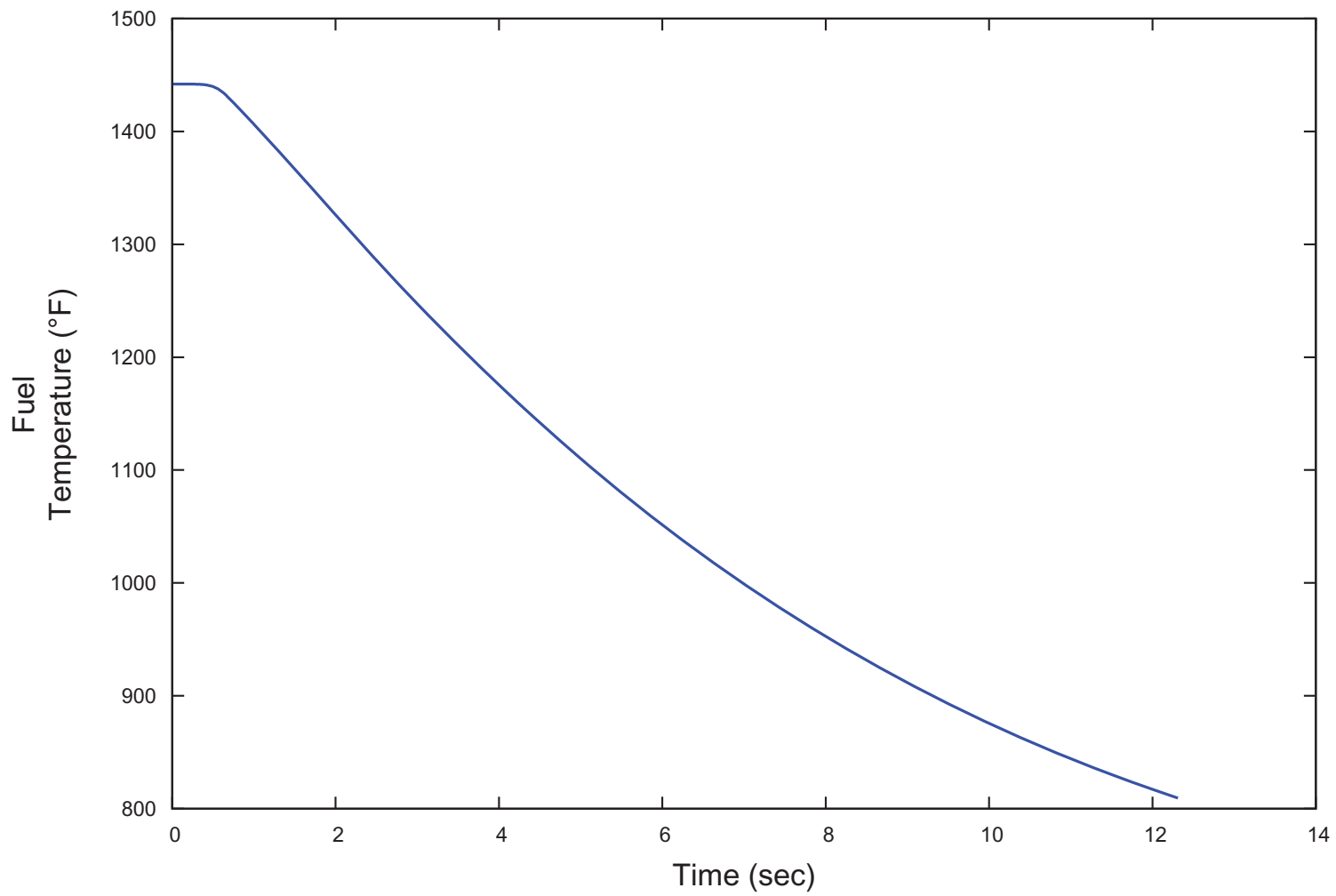


Figure 15.6-42: Inadvertent Operation of the Emergency Core Cooling System – Reactor Vent Valve Flow (Typical of One RVV) - Limiting Minimum Critical Heat Flux Ratio Case

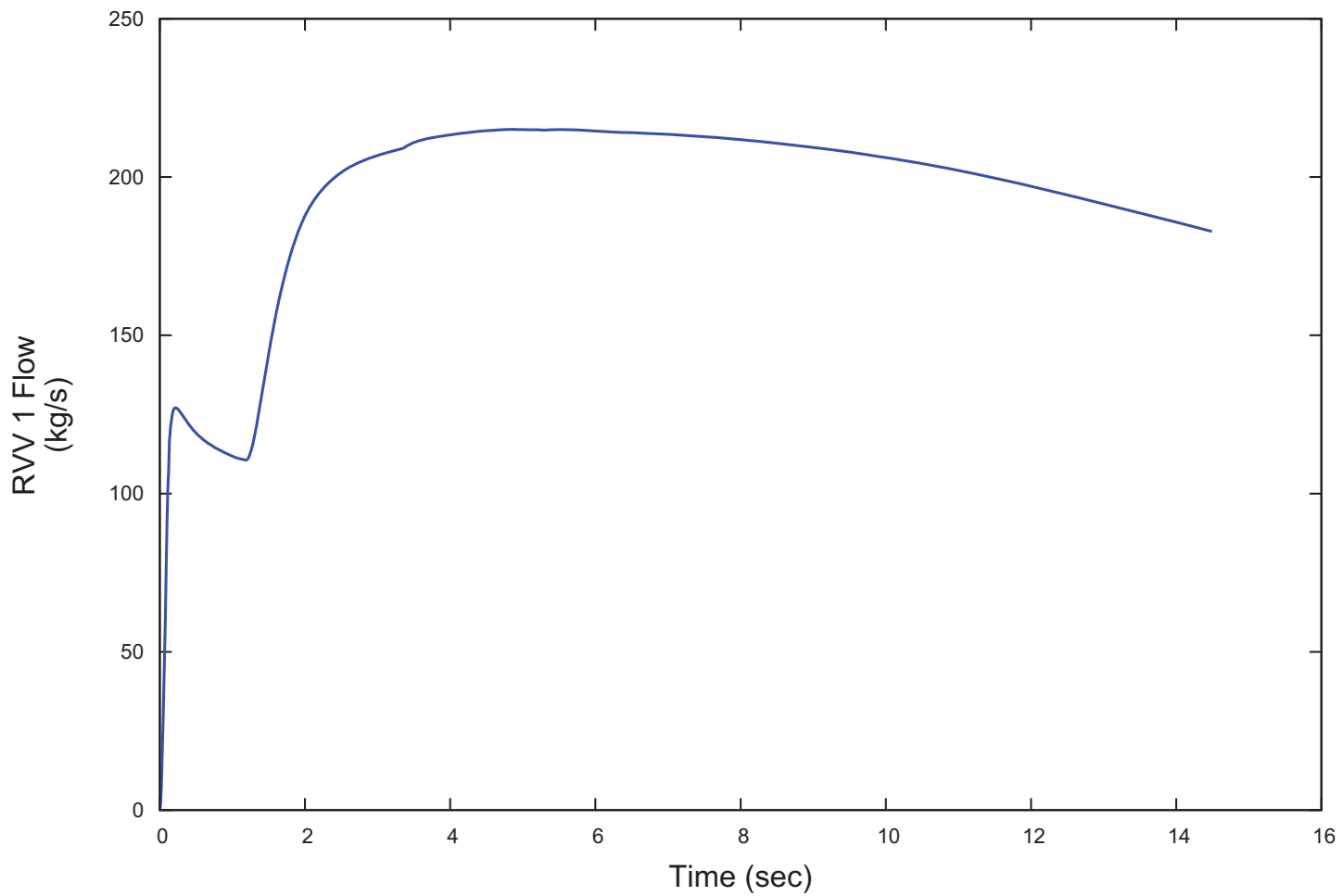


Figure 15.6-43: Inadvertent Operation of the Emergency Core Cooling System – Reactor Pressure Vessel Pressure - Limiting Minimum Critical Heat Flux Ratio Case

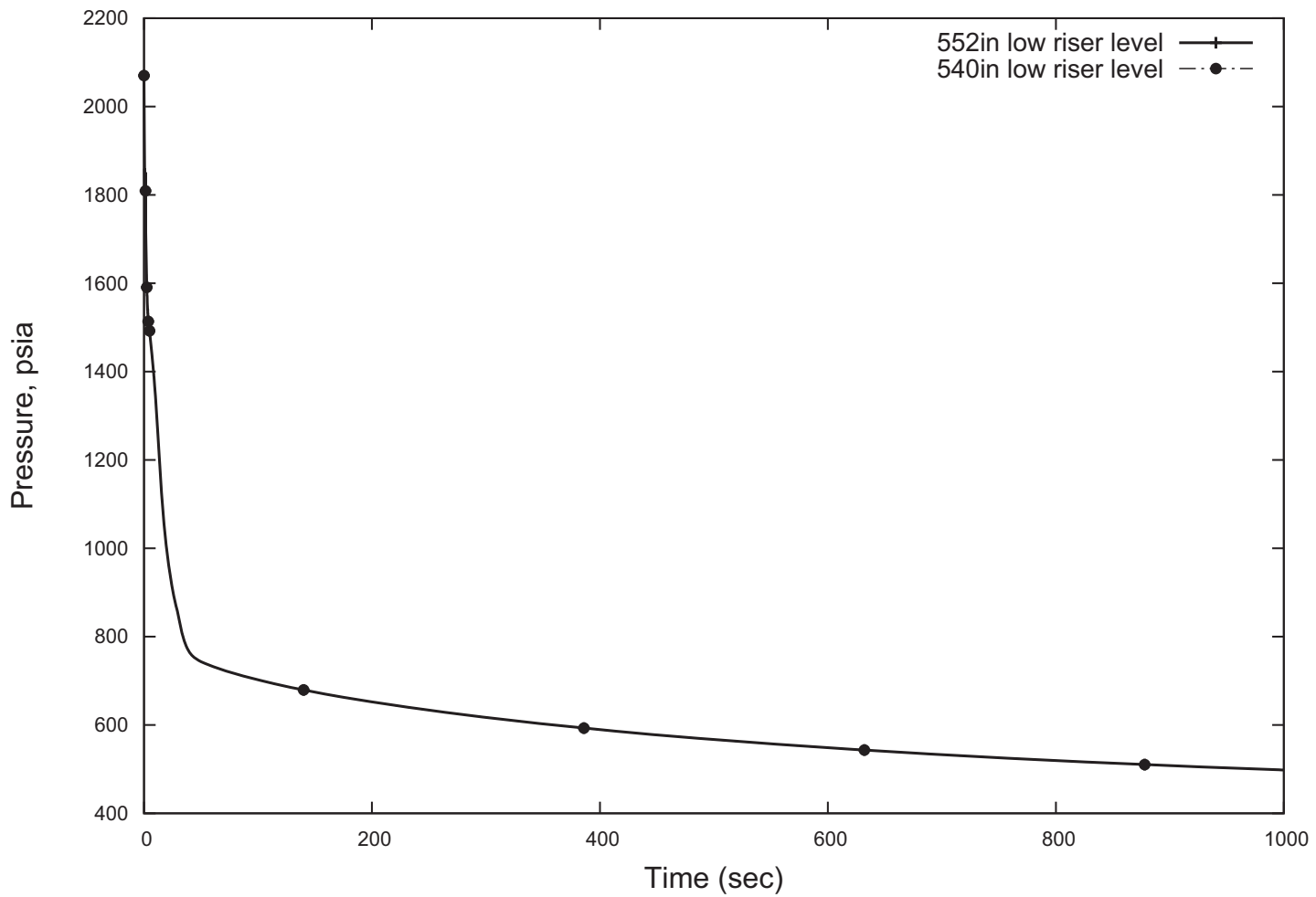


**Figure 15.6-44: Inadvertent Operation of the Emergency Core Cooling System – Containment Pressure
- Limiting Minimum Critical Heat Flux Ratio Case**

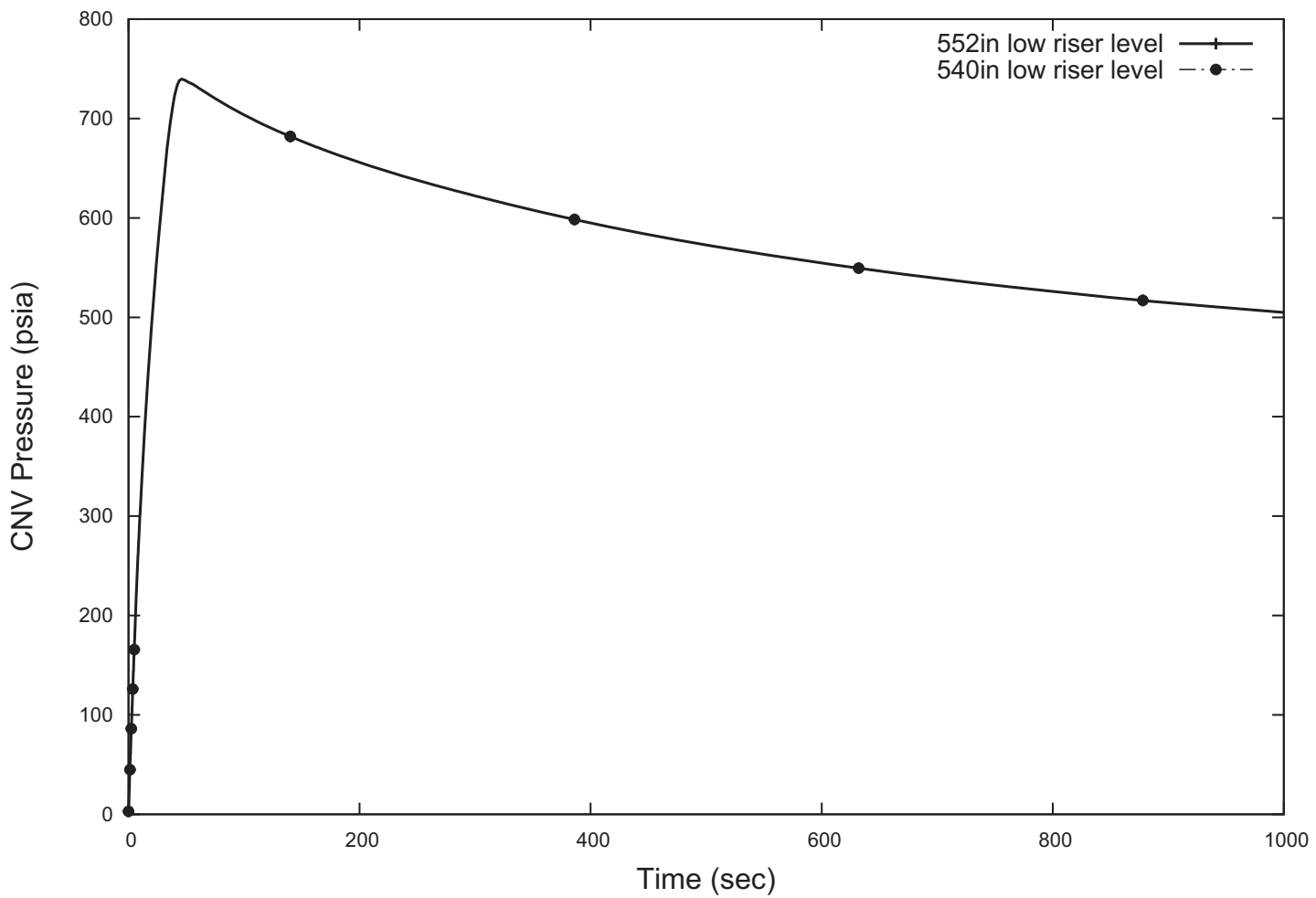


Figure 15.6-45: Inadvertent Operation of the Emergency Core Cooling System – Reactor Coolant System Flow - Limiting Minimum Critical Heat Flux Ratio Case

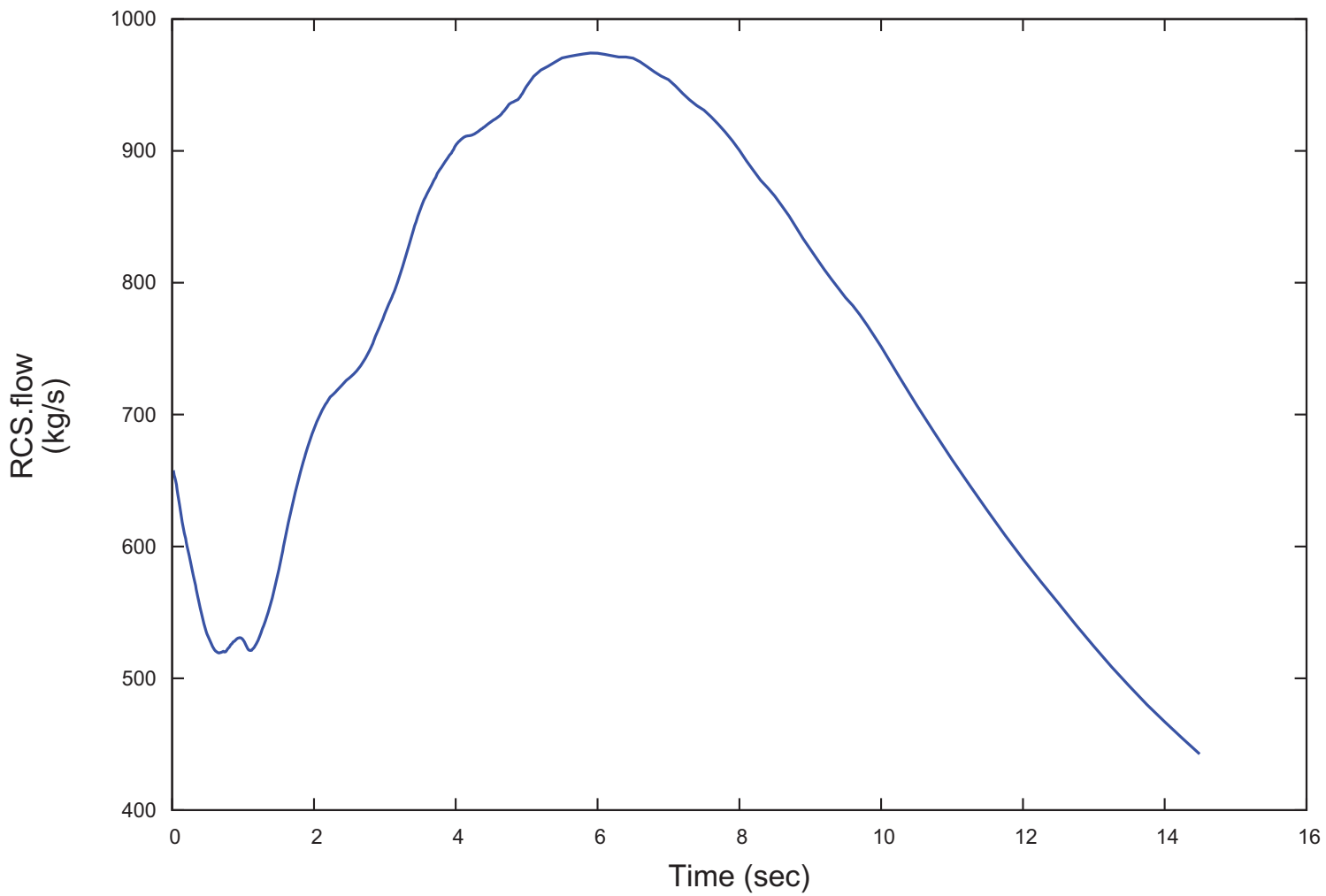


**Figure 15.6-46: Inadvertent Operation of the Emergency Core Cooling System – Critical Heat Flux Ratio
- Limiting Minimum Critical Heat Flux Ratio Case**

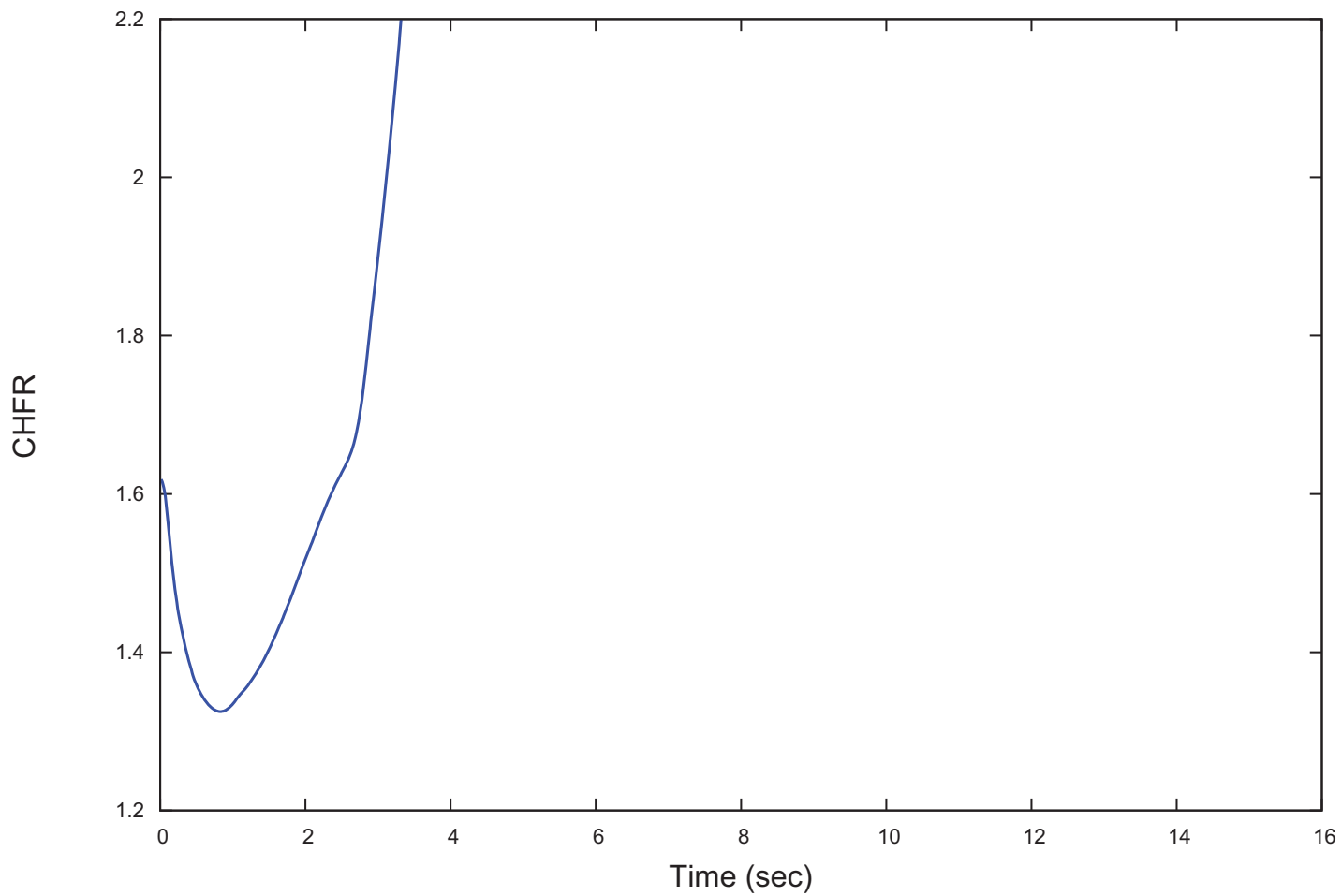


Figure 15.6-47: Inadvertent Operation of the Emergency Core Cooling System – Reactor Power - Limiting Minimum Critical Heat Flux Ratio Case

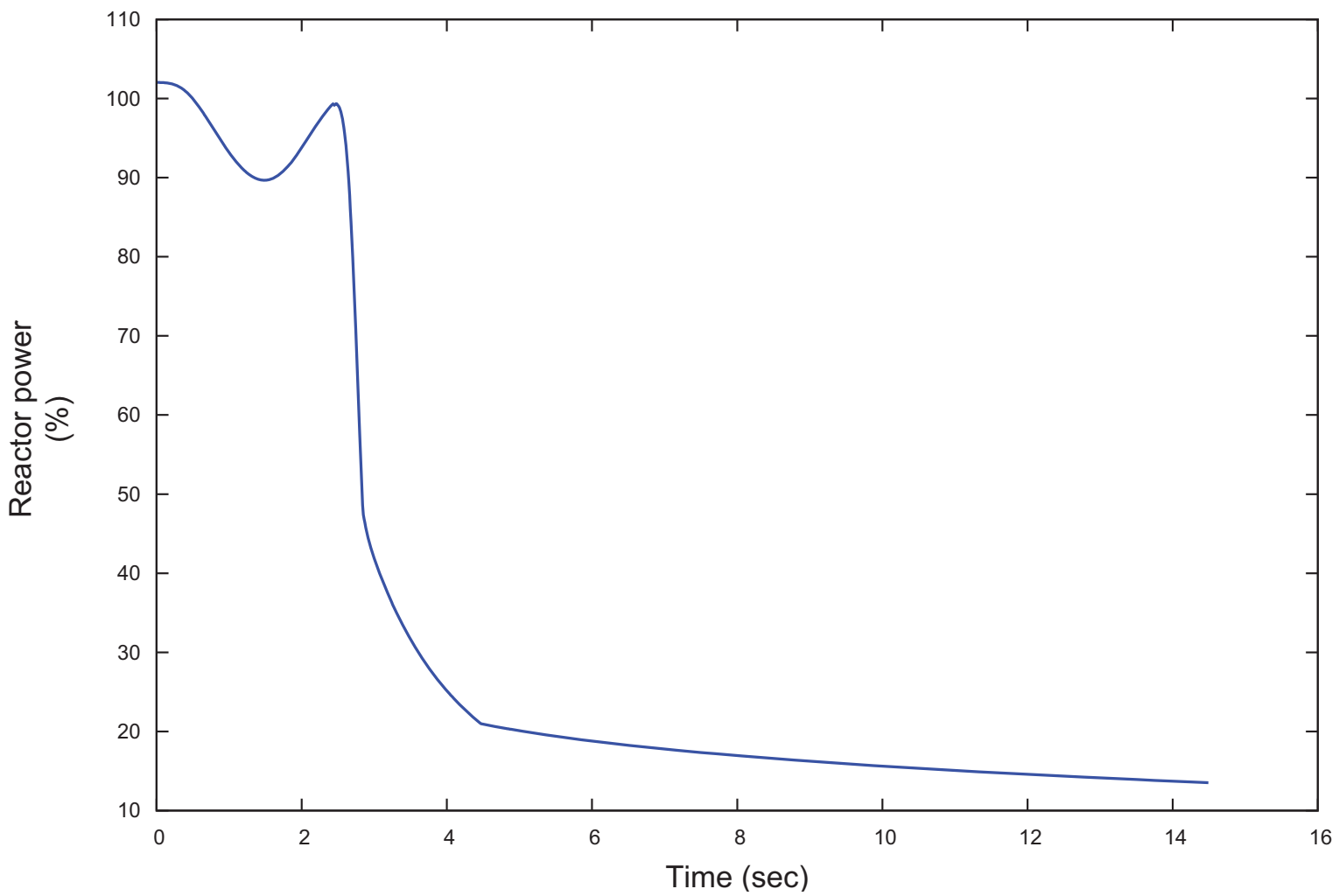


Figure 15.6-48: Inadvertent Operation of the Emergency Core Cooling System – Steam Generator Pressure (Typical of One SG) - Limiting Minimum Critical Heat Flux Ratio Case

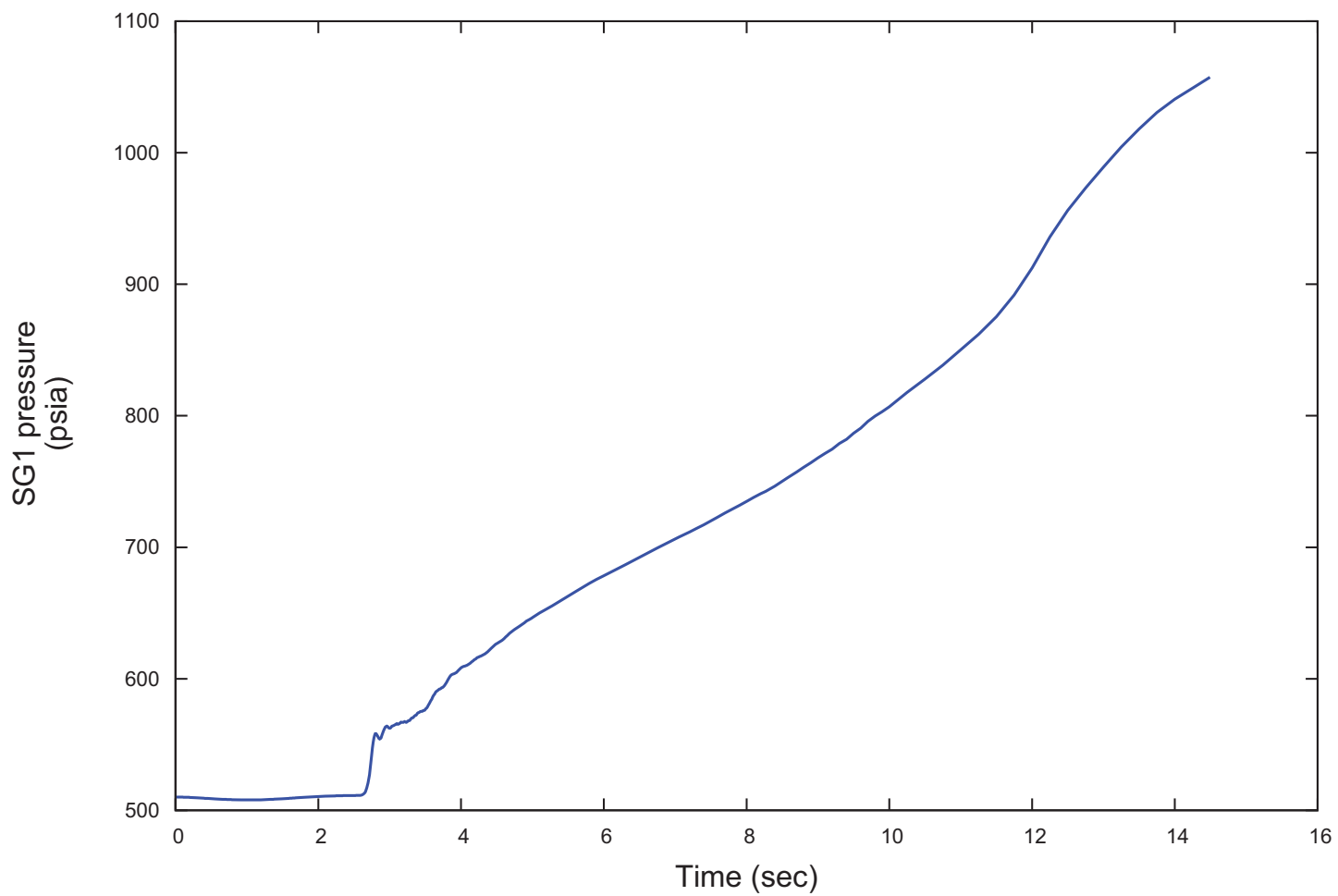


Figure 15.6-49: Inadvertent Operation of the Emergency Core Cooling System – Reactor Coolant System Temperature - Limiting Minimum Critical Heat Flux Ratio Case

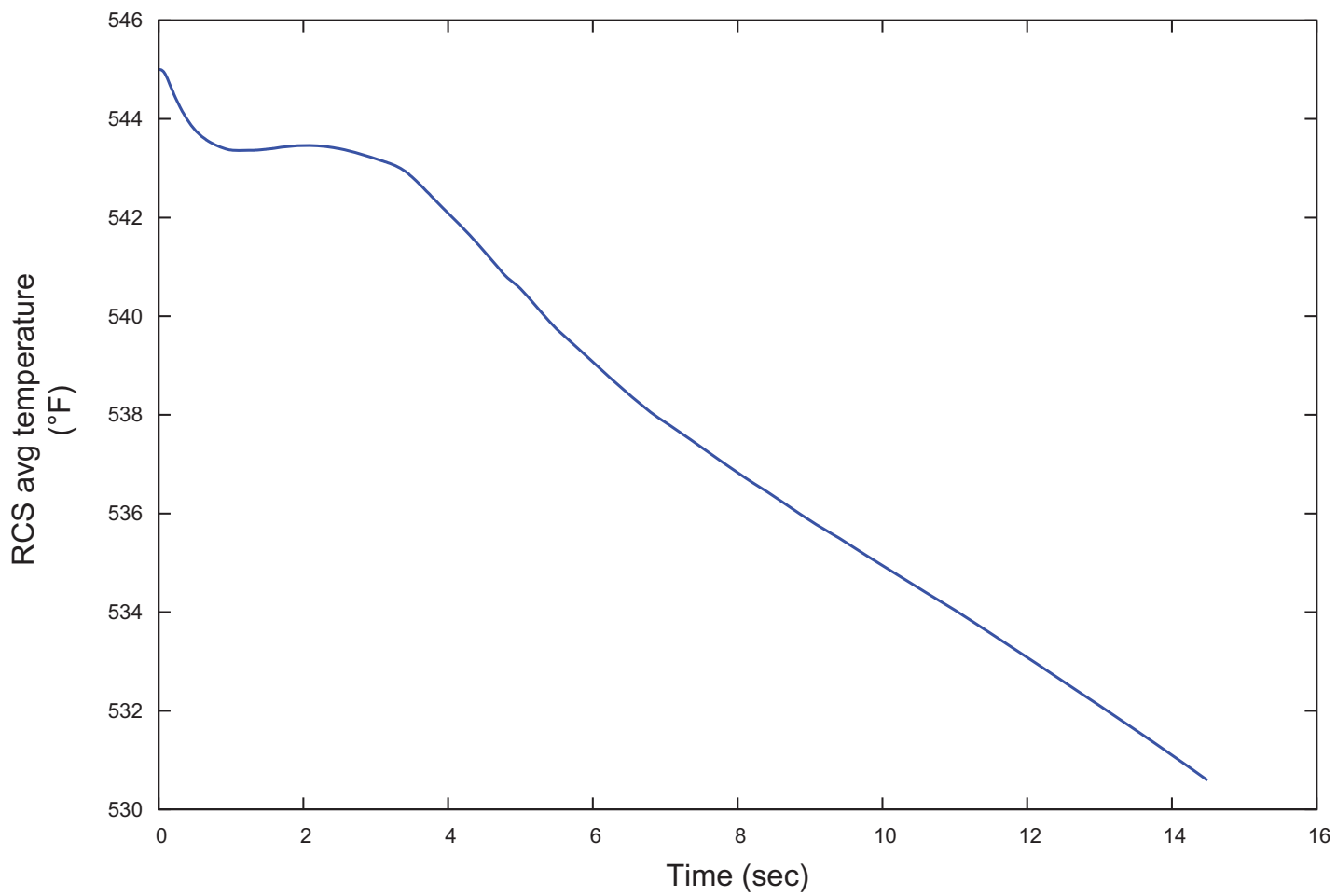


Figure 15.6-50: Inadvertent Operation of the Emergency Core Cooling System Valve – Riser Collapsed Level - Limiting Minimum Critical Heat Flux Ratio Case

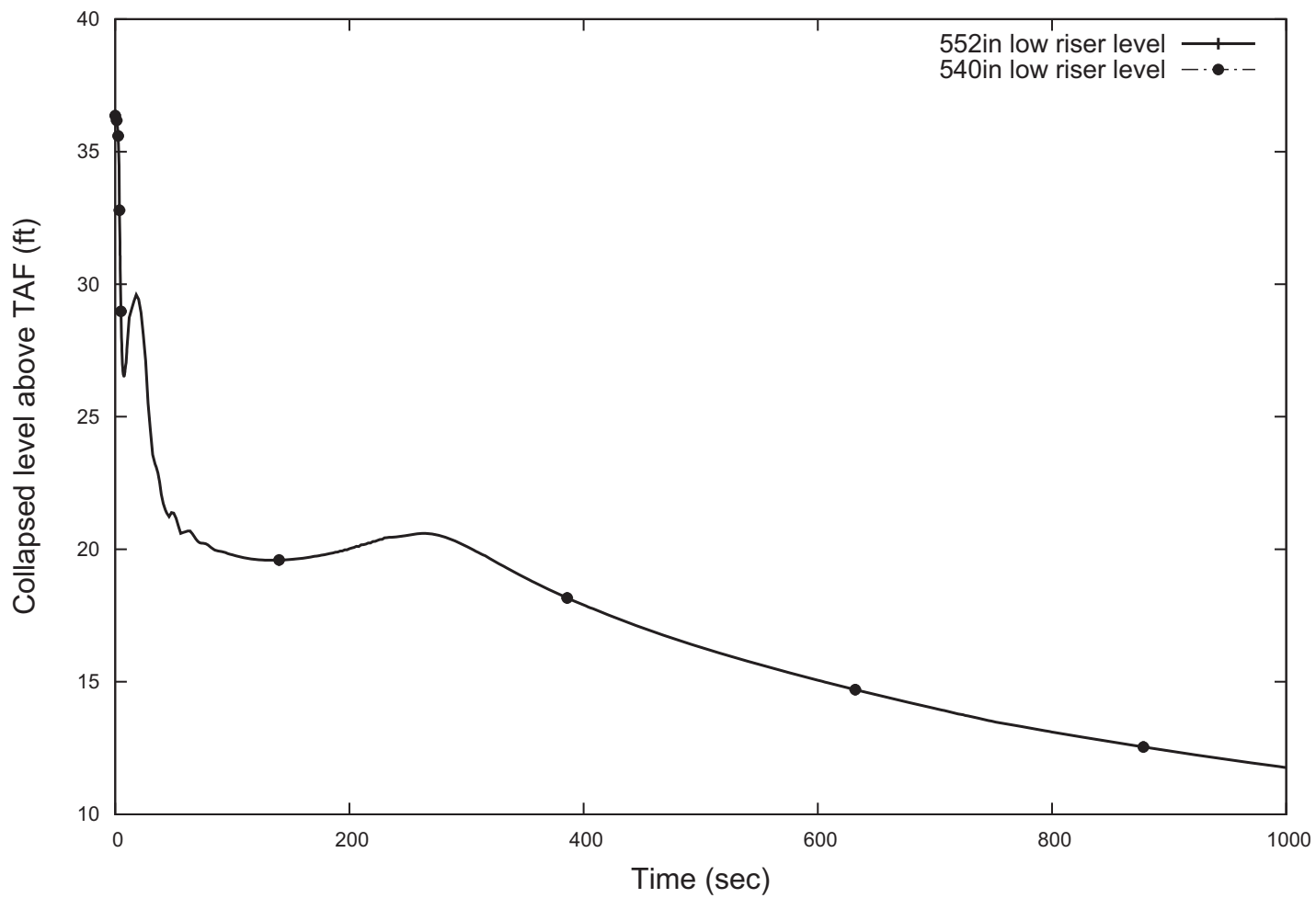


Figure 15.6-51: Inadvertent Operation of the Emergency Core Cooling System – Fuel Temperature - Limiting Minimum Critical Heat Flux Ratio Case

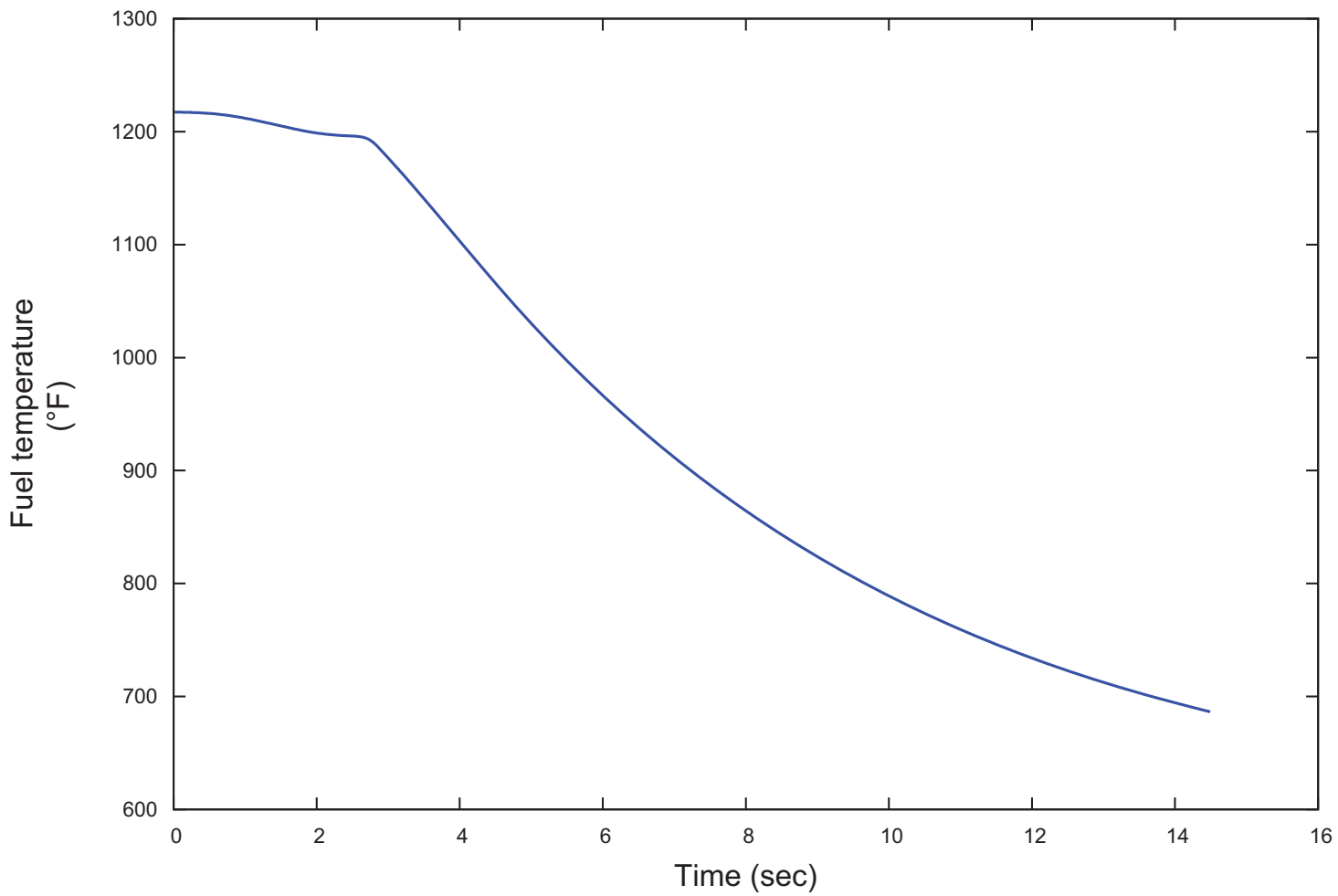
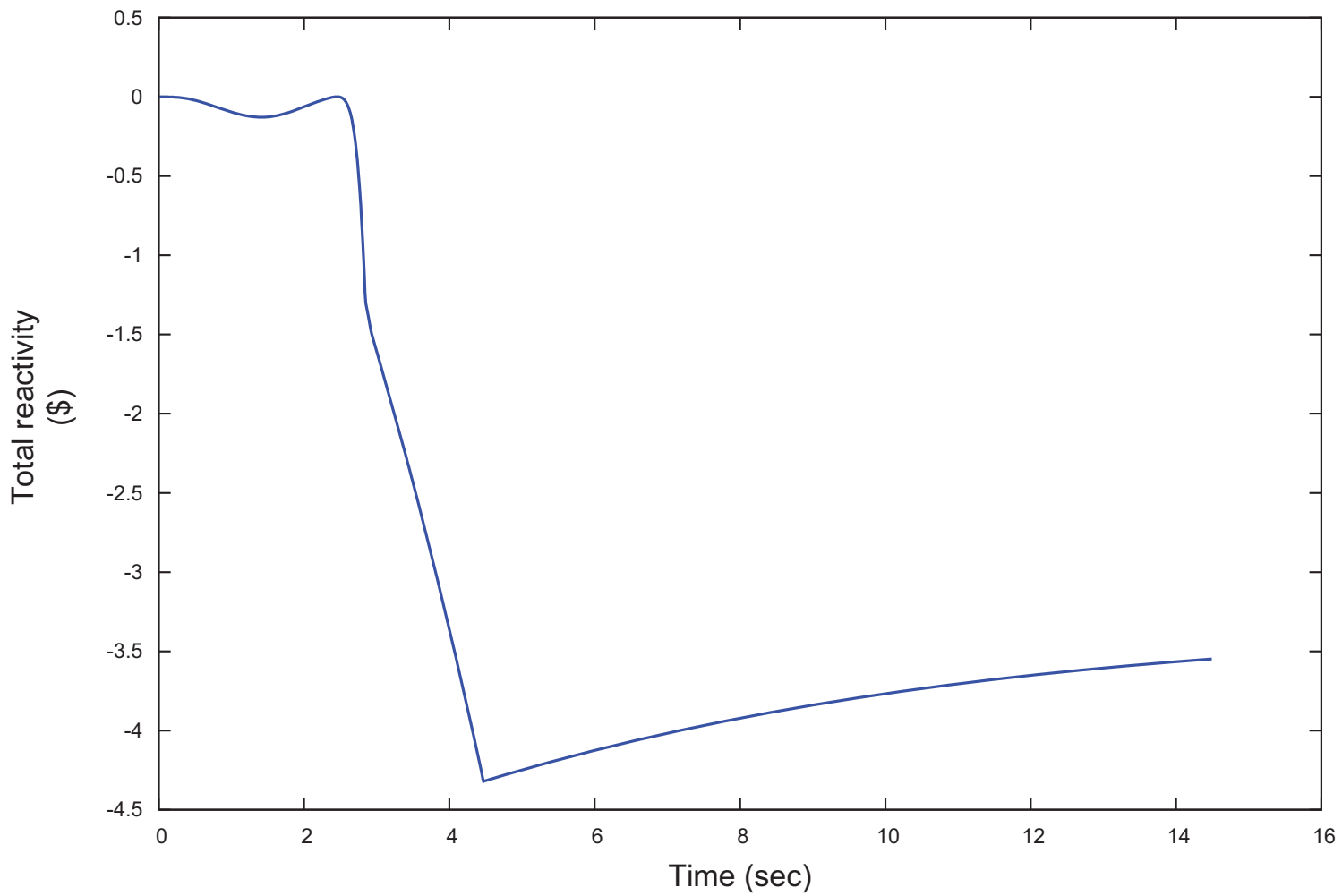


Figure 15.6-52: Inadvertent Operation of the Emergency Core Cooling System – Total Reactivity - Limiting Minimum Critical Heat Flux Ratio Case



15.7 Radioactive Release from a Subsystem or Component

This section addresses events that could result in a radioactive release from a component or system other than the reactor coolant system. The sources of such releases are waste processing systems and fuel handling systems. The NuScale Power Plant US460 standard design is similar to current generation pressurized water reactors with respect to waste processing systems and fuel handling systems. The design is unique in that the entire NuScale Power Module (NPM) is moved for refueling.

15.7.1 Gaseous Waste Management System Leak or Failure

Releases from waste processing systems are addressed in Chapter 11 and are not treated as design-basis events in Chapter 15.

15.7.2 Liquid Waste Management System Leak or Failure

Releases from waste processing systems are addressed in Chapter 11 and are not treated as design-basis events in Chapter 15.

15.7.3 Postulated Radioactive Releases Due to Liquid-Containing Tank Failures

Releases from waste processing systems are addressed in Chapter 11 and are not treated as design-basis events in Chapter 15.

15.7.4 Fuel Handling Accidents

A fuel handling accident may occur during movement of fuel. The failure of one entire assembly is assumed to occur when an assembly is dropped in the reactor pool above the spent fuel racks or weir wall, dropped in the reactor core during refueling, or when an assembly impacts a spent fuel cask during loading. Activity is assumed to be instantaneously released into the pool water from all irradiated fuel rods in the assembly. Specific isotopes are assumed to remain in the pool water or enter the Reactor Building atmosphere instantaneously, either fully or partially, and are assumed to be directly released into the environment over a two-hour period.

Fuel handling accidents are classified as postulated accidents as indicated in Table 15.0-1. The radiological consequence analysis for the fuel handling accident is provided in Section 15.0.3.

15.7.5 Spent Fuel Cask and NuScale Power Module Drop Accidents

The Reactor Building crane (RBC) is used to move the spent fuel cask and NPMs in the Reactor Building refueling area. The RBC system design conforms to the American Society of Mechanical Engineers standards specified in Table 9.1.5-1, so that a credible failure of a single component does not result in the loss of capability to stop and hold a critical load.

The use of the RBC precludes the need to perform load drop evaluations. As a result, no design-basis accident analysis is performed to assess the radiological consequences of a spent fuel cask or NPM drop accident.

Section 9.1.5 provides additional information regarding the RBC system design and capabilities, including a description of system interlocks, safe load paths, and load exclusion zones.

Chapter 19 provides a description of the low power and shutdown (LPSD) probabilistic risk assessment. All stages of a nominal refueling outage are included in the LPSD probabilistic risk assessment, including movement and disassembly of an NPM during refueling. An NPM drop event is also evaluated in the LPSD probabilistic risk assessment in Section 19.1.6.

15.8 Anticipated Transients Without Scram

An anticipated transient without scram (ATWS) is characterized as a failure of the module protection system (MPS) to initiate a reactor trip in response to an anticipated operational occurrence. The probability of an anticipated operational occurrence, in coincidence with a failure to scram, is much lower than the probability of other events evaluated in Chapter 15. Therefore, an ATWS event is classified as a beyond-design-basis event.

The ATWS contribution to core damage frequency for the NuScale Power Plant US460 standard design is significantly below the SECY-83-293 safety goal of $1.0\text{E-}5/\text{year}$, as demonstrated in Section 19.1. This low contribution is based on the reliability of the reactor trip function of the MPS. The MPS, described in Section 7.1 and Section 7.2, includes a robust reactor protection system with internal diversity that avoids common cause failures and reduces the probability of a failure to scram. The MPS uses the highly integrated protection system (HIPS) platform. NuScale TR-1015-18653 (Reference 15.8-1) describes integration of fundamental instrumentation and control design principles into the HIPS design. The HIPS platform encompasses the principles of independence, redundancy, predictability and repeatability, and diversity and defense-in-depth. The redundancy and diversity of the MPS design ensures an ATWS occurrence is a very low probability event, which meets the intent of the Standard Review Plan Section 15.8 acceptance criterion 3.A.i for evolutionary plants.

The design supports an exemption from the portion of 10 CFR 50.62(c)(1) requiring diverse turbine trip capabilities because the design does not rely on diverse turbine trip functionality to reduce the risk associated with ATWS events. Additionally, the design does not include an auxiliary (or emergency) feedwater system. Therefore, the portion of 10 CFR 50.62(c)(1) that requires diverse capability to initiate an auxiliary (or emergency) feedwater system is not applicable to the NuScale Power Plant US460 standard design. An analysis of the beyond-design-basis ATWS event is described in Section 19.2.

15.8.1 References

- 15.8-1 NuScale Power, LLC, "Design of the Highly Integrated Protection System Platform," TR-1015-18653-P-A, Revision 2.

15.9 Stability

Events that could result in thermal-hydraulic instability within the reactor pressure vessel are considered significant only for boiling water reactors (BWRs), where individual fuel assemblies with high power-to-flow ratios may undergo instabilities or the neutronic conditions may lead to power oscillation. Pressurized water reactors use pumps for forced circulation, which keeps core flow essentially constant with power level. The NuScale Power Module (NPM) employs natural circulation. With this design feature, flow through the core is not held constant by pumps providing forced circulation. Thus, variations in flow may result in changes in power level and vice versa. The identification and evaluation of the significance of these mechanisms is addressed in Section 4.4.7. Perturbations and bounding flow instability are evaluated in this section. The evaluation includes reactivity coefficients that span beginning of cycle (BOC) to end of cycle (EOC), and demonstrates that the NPM is protected from unstable flow oscillations provided that operation is limited by a defined pressure-temperature exclusion zone such that no boiling in the riser area above the core is allowed. The negative moderator reactivity coefficient may stabilize flow even if riser boiling occurs, but this is conservatively not credited in the stability methodology.

15.9.1 Consideration of Thermal-Hydraulic Stability

The NuScale Stability Evaluation Methodology Topical Report (Reference 15.9-1) presents an analysis of the thermal-hydraulic stability of a representative NPM and demonstrates compliance with 10 CFR 50, Appendix A, General Design Criteria (GDC) 10 and GDC 12. The topical report considers potential power and hydraulic stability mechanisms during anticipated operational occurrences (AOOs) and normal operating conditions. Thermal-hydraulic instability during infrequent events or accidents is not considered because the acceptance criteria for such events allow for conditions beyond the specified acceptable fuel design limits imposed on AOO events experiencing instabilities. The topical report considers flow stability without making assumptions based on comparisons to other nuclear systems. The topical report describes computational methods developed for the analysis of the limiting instability modes for a representative NPM design during steady state, normal operation, and anticipated transients.

The region exclusion stability protection solution is shown in Reference 15.9-1 to be an acceptable approach for preventing the occurrence of instabilities in an NPM.

The limiting instability mode is a natural circulation instability as described in Reference 15.9-1.

The Reference 15.9-1 methodology and conclusions, as well as the conditions and limitations of the associated Safety Evaluation Report, have been reviewed and verified to be applicable to the NPM for the NuScale Power Plant US460 standard design. Updated stability analyses using inputs in a manner consistent with Reference 15.9-1, but with NPM parameters reflecting the US460 standard design, are presented in this section.

15.9.2 Stability Analyses

Several cases are analyzed over a wide range of power and primary system flow operating conditions and possible scenarios to demonstrate that stability is maintained during routine power operations in the NPM.

15.9.2.1 Stability Analysis for Power Operations

This section analyzes power operations over a range of power level and flow conditions in the presence of a small perturbation in operating conditions as described in Section 8.1 of Reference 15.9-1.

Only the limiting case is presented in detail in Section 15.9.2.1.1 through Section 15.9.2.1.4.

The analyses show that the NPM is stable at power levels above 5 percent, with EOC conditions providing more damping (i.e. more stability) than BOC conditions.

The flow stability condition that is the least stable occurs at 12.5 MW core power (5 percent of rated) with a BOC reactivity condition. The primary coolant flow response shows damped oscillations with a period of over a minute at this power level. The flow is less stable than the higher power cases, but with the low power level, there is no challenge to fuel limits.

15.9.2.1.1 Identification of Causes and Event Description

The 12.5 MW power case at BOC is the least stable of the power operations considered.

15.9.2.1.2 Sequence of Events and Systems Operation

No systems operations occur in response to the event.

15.9.2.1.3 Input Parameters and Initial Conditions

Input parameters and initial conditions for the limiting case (5 percent power at BOC) are presented in Table 15.9-1.

15.9.2.1.4 Results

Table 15.9-2, Figure 15.9-1, and Figure 15.9-2 provide the analysis case that produced the limiting results. The figures show the transient response for the time period over which the decay ratio is calculated.

15.9.3 Stability Analysis for Operational Occurrences

The nature of the natural circulation system performance narrows the analysis down to examining transients that are credible in the NPM. Several operational events are investigated with externally imposed boundary conditions applied to influence the system response as described in Section 8.2 of Reference 15.9-1.

The analyses demonstrate an acceptable operating region for the NPM where instability does not occur.

The NPM system response is obtained by the computer code, PIM, which is used in demonstrating system stability at initially steady-state operation. The PIM code is described in Section 4.4.7.

15.9.3.1 Increase in Heat Removal by the Secondary System

15.9.3.1.1 Identification of Causes and Event Description

Stability perturbations can occur from a rapid increase of feedwater flow. The flow increase can be caused by feedwater pump speed increase, valve alignment changes, or other causes. However, the analyzed change is sufficiently small that the module protection system (MPS) does not actuate, and control systems, such as those for steam pressure, maintain other parameters at the original value.

Other causes of increased heat removal, such as decreasing feedwater temperature or decreasing steam pressure (that causes increased boiling in the SGs), are bounded by changes in feedwater flow. The change in feedwater flow is bounding because the potential for change in feedwater temperature is more gradual when considering the entire feedwater system train (e.g., preheaters, piping lengths) and large rapid changes in steam pressure are expected to cause either compensating control actions or MPS trips.

15.9.3.1.2 Sequence of Events and Systems Operation

A disturbance results in feedwater flow being increased by 30 percent in 0.1 seconds. This change is chosen because, while it would normally cause a reactor trip, this trip is not simulated and, thus, it bounds smaller changes to feedwater flow that would not result in a reactor trip. No systems operations occur in response to the event, so no sequence of events table is generated.

15.9.3.1.3 Input Parameters and Initial Conditions

The event is analyzed for both the reactor at 100 percent power and 20 percent power. The partial power condition evaluates stability behavior at the expected lowest power level for electricity production. Input parameters and initial conditions for the 100 percent and 20 percent power cases are presented in Table 15.9-3 and Table 15.9-4, respectively. Both BOC and EOC reactivity conditions are considered in each analysis, but only EOC results are presented as they are the limiting results.

15.9.3.1.4 Results

Table 15.9-5 and Table 15.9-6 present the results for 100 percent of rated power at EOC and for 20 percent of rated power at EOC, respectively. Figure 15.9-3 and Figure 15.9-4 show transient responses for the limiting case

of 20 percent of rated power at EOC. These results indicate that the NPM is stable during a postulated increase in heat removal by the secondary system.

15.9.3.2 Decrease in Heat Removal by the Secondary System

15.9.3.2.1 Identification of Causes and Event Description

Stability following reduction of feedwater flow is addressed in this section. A hypothetical rapid decrease in feedwater flow occurs due to feedwater pump speed change, valve alignment changes, or other causes. However, complete loss of feedwater is not considered because it would result in actuation of the MPS and a trip.

Other causes of decreased heat removal, such as increasing feedwater temperature or increasing steam pressure are bounded by changes in feedwater flow because larger changes would result in a trip on high reactor power.

15.9.3.2.2 Sequence of Events and Systems Operation

Feedwater flow is decreased by 30 percent in 0.1 seconds while maintaining feedwater temperature and steam pressure. This magnitude of change is chosen to determine the acceptability of a partial loss of feedwater. While this magnitude of change would normally cause a reactor trip, this trip is not simulated and, thus, it conservatively bounds smaller changes to feedwater flow that would not result in a reactor trip.

The resulting reduction in the heat removal from the primary coolant flow initiates a transient in which primary coolant temperature starts to rise and negative moderator feedback reduces the fission power. The combined reduction of the heat sink and core power restore the primary coolant temperature to a value above its initial value. The Doppler reactivity compensates for the difference, and the net average reactivity is restored to zero. The density head driving the primary coolant flow is reduced, and the flow is reduced from its initial value.

No systems operations occur in response to the event.

15.9.3.2.3 Input Parameters and Initial Conditions

The event is analyzed for both the reactor at 100 percent power and 20 percent power. The partial power condition evaluates stability behavior at the expected lowest power level for electricity production. Input parameters and initial conditions for the 100 percent and 20 percent power cases are presented in Table 15.9-3 and Table 15.9-4, respectively. Both BOC and EOC reactivity conditions are considered in each analysis, but only EOC results are presented as they are the limiting results.

15.9.3.2.4 Results

Table 15.9-7 and Table 15.9-8 present the results for 100 percent of rated power at EOC and for 20 percent of rated power at EOC, respectively. Figure 15.9-5 and Figure 15.9-6 show transient responses for the limiting case of 20 percent of rated power at EOC. These results indicate that the NPM is stable during a postulated decrease in heat removal by the secondary system.

15.9.3.3 Decrease in Reactor Coolant System Flow Rate

The effect of a decrease in primary system flow rate (in isolation from other effects) is not considered a credible event for stability analysis because there is no source for changing the primary system flow without other influences, and because there are no primary system pumps in the NPM to directly influence primary system flow.

15.9.3.4 Increase in Reactor Coolant Inventory

The effects of increasing reactor coolant system (RCS) inventory are not important in the stability assessment because subcooled margin in the riser increases with increasing primary system pressure, and overall stability behavior is not sensitive to pressure changes for a single-phase system.

The effect of adding cold water with the chemical and volume control system (CVCS) during an increasing RCS inventory event is bounded by analyses of increased heat removal by the secondary system. The potential for minor reduction in primary system flow can occur from adding cooler water to the riser. However, the relatively small cooldown that may occur at high power conditions and the very long time for coolant to transit from the CVCS return line located in the riser, around the primary system, and into the core in low power operations make this a secondary consideration in comparison to secondary side cooldowns.

15.9.3.5 Reactivity and Power Distribution Anomalies

15.9.3.5.1 Identification of Causes and Event Description

The effect on NPM stability from a reactivity anomaly can be caused by changes in boron concentration, by an uncontrolled control rod assembly withdrawal, or similar events that result in reactivity insertion.

15.9.3.5.2 Sequence of Events and Systems Operation

Reactivity is added over 60 seconds to reach the high power trip or is added over 5 seconds to reach the high power rate trip while other reactivity components perform normally. The trips are not simulated to bound smaller reactivity insertions that would not initiate these trips.

15.9.3.5.3 Input Parameters and Initial Conditions

The event is analyzed for both the reactor at 100 percent power and 20 percent power. Input parameters and initial conditions for the 100 percent and 20 percent power cases are presented in Table 15.9-3 and Table 15.9-4, respectively. Both BOC and EOC reactivity conditions are considered, but only EOC conditions are presented as they are limiting. A maximum of 0.90 dollars of reactivity is added.

15.9.3.5.4 Results

The effect of a change in boron concentration is slow to develop and is bounded by the applied variations in reactivity conditions.

Reactivity increases that do not result in reactor trip on high flux or high flux rate develop slowly and are bounded by effects of increasing heat removal from the secondary side. These events cause pressurizer insurges that maintain or increase subcooling in the riser.

The results are presented in Table 15.9-9 and Table 15.9-10 for 100 percent of rated power at EOC and for 20 percent of rated power at EOC, respectively. Figure 15.9-7 and Figure 15.9-8 show transient responses for the limiting case of 20 percent of rated power at EOC. These results indicate that the NPM is stable during a postulated addition of reactivity event.

15.9.3.6 Decrease in Reactor Coolant Inventory

Decreasing RCS inventory without changes in primary pressure is not important in the stability assessment. Riser subcooling is maintained and the protection system trips the NPM on low pressurizer level before any appreciable effect can be seen regarding stability.

Decreasing reactor coolant inventory that results in decreasing pressure but without a level trip is expected to produce no significant effect on stability as long as the primary coolant in the riser remains subcooled. However, further depressurization beyond the trip setpoint that results in riser voiding can destabilize the system. This section provides analysis results using the PIM code that show the effects of depressurization and ability of the MPS to mitigate the event.

15.9.3.6.1 Identification of Causes and Event Description

Stability following a rapid depressurization is addressed in this section. A decrease in RCS inventory without changes in primary pressure is not analyzed, as riser subcooling is maintained and the protection system trips the NPM on low pressurizer level before any appreciable effect can be seen regarding stability.

15.9.3.6.2 Sequence of Events and Systems Operation

Reactor power is 250 MW (100 percent power) when two different rapid depressurizations are assumed. The first is from the initial pressure to just above the low pressurizer pressure trip setpoint over a period of 5 seconds such that no system operations occur in response to the event. The second is from the initial pressure to well below the low pressurizer pressure trip setpoint over a period of 10 seconds such that the trip occurs. The second depressurization rate is approximately consistent with the opening of a single reactor vent valve. The high containment pressure trip, which is expected to occur earlier, is neglected.

15.9.3.6.3 Input Parameters and Initial Conditions

The event is analyzed for the reactor at 100 percent power because the depressurization transients are related to minimum subcooling margin. Input parameters and initial conditions are presented in Table 15.9-3. Both BOC and EOC reactivity conditions are considered for each depressurization case, but only the results of the limiting reactivity condition for each depressurization case are presented.

15.9.3.6.4 Results

Table 15.9-11, Figure 15.9-9, and Figure 15.9-10 present the results of the limiting depressurization case that remains above the low pressurizer pressure trip. Subcooling in the riser is maintained throughout the transient. The results for this case indicate that the NPM is stable for depressurizations that do not result in a reactor trip.

Table 15.9-12 presents the sequence of events for the limiting depressurization case that results in a low pressurizer pressure trip. Although subcooling in the riser is lost, the reactor is safely shut down before the development of oscillations due to loss of subcooling. A decay ratio is not determined. The results for this case indicate that the MPS is effective at shutting down the NPM before instability occurs.

These results indicate that the NPM is stable during a postulated decrease in reactor coolant inventory.

15.9.4 Demonstration of Module Protection System Functions to Preclude Instability

At rated power, the minimum loop time for the NPM is more than 60 seconds. The response delay for the MPS is no more than 8.0 seconds for setpoints that are pertinent to stability analysis and the reactor trip time is less than 2.5 seconds. The time from the first reactor trip setpoint being reached to the control rods being fully inserted is less than 11 seconds, which is significantly less than the minimum loop time for the NPM. Therefore, the MPS shuts down the reactor before any potential instability manifests itself as a divergent primary flow oscillation.

15.9.5 Conclusions

There are two main aspects of the stability methodology. The first is the use of a regional exclusion as the stability solution type and the rationale for its selection. The second aspect is the demonstration that the NPM maintains stability within the region of operation allowed by the MPS. The NPM returns to the original oscillation-free condition after steady state conditions are perturbed.

Operational events do not result in unstable NPM behavior.

The radiological consequences of events in Section 15.9.3 are bounded by the design-basis accident analyses presented in Section 15.0.3.

15.9.6 References

- 15.9-1 NuScale Power, LLC, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," TR-0516-49417-P-A, Revision 1.

Table 15.9-1: Initial Conditions (5 Percent of Rated Power Cases)

Parameter		Analysis Value ¹
Initial reactor power		12.5 MWt
Core inlet temperature		386.5 °F
Core inlet mass flow rate		221.1 kg/s
Pressurizer pressure		2000 psia
Feedwater temperature		120 °F
Feedwater flow		37,320 lbm/hr
Steam generator pressure		182.1 psia
Moderator density coefficients, BOC ²	a0	0.8328
	a1	0.4946
	a2	-0.5580
	a3	0.3442
	a4	-0.1177
Doppler reactivity coefficient, BOC ³		7.91x10 ⁻⁴

1. With the exception of the reactivity coefficients, analysis values are nominal values without bias or uncertainty applied as described in Reference 15.9-1.
2. Moderator density coefficients are implemented as described in Section 5.6.1.2 of Reference 15.9-1, but with the coefficients shown in this table instead of the coefficients shown in Reference 15.9-1.
3. Doppler reactivity coefficient is implemented as described in Section 5.6.1.1 of Reference 15.9-1, but with the coefficient shown in this table instead of the coefficients shown in Reference 15.9-1.

**Table 15.9-2: Normal Power Operation - Limiting Analysis Results
(5 Percent of Rated Power Case, Beginning of Cycle)**

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤ 0.8	0.58

Table 15.9-3: Initial Conditions (100 Percent of Rated Power Cases)

Parameter		Analysis Value ¹
Initial reactor power		250 MWt
Core inlet temperature		482.0 °F
Core inlet mass flow rate		730.9 kg/s
Pressurizer pressure		2000 psia
Feedwater temperature		250.0 °F
Feedwater flow		812,900 lbm/hr
Steam generator pressure		475.0 psia
Moderator density coefficients, EOC ²	b0	0.2382
	b1	2.4281
	b2	-3.1346
	b3	2.0925
	b4	-0.5676
Doppler reactivity coefficient, EOC ³		8.56x10 ⁻⁴

1. With the exception of the reactivity coefficients, analysis values are nominal values without bias or uncertainty applied as described in Reference 15.9-1. The nominal values are consistent with the values or ranges in Table 15.0-6.
2. Moderator density coefficients are implemented as described in Section 5.6.1.2 of Reference 15.9-1, but with the coefficients shown in this table instead of the coefficients shown in Reference 15.9-1.
3. Doppler reactivity coefficient is implemented as described in Section 5.6.1.1 of Reference 15.9-1, but with the coefficient shown in this table instead of the coefficients shown in Reference 15.9-1.

Table 15.9-4: Initial Conditions (20 Percent of Rated Power Cases)

Parameter	Analysis Value ¹
Initial reactor power	50 MWt
Core inlet temperature	517.7 °F
Core inlet mass flow rate	383.6 kg/s
Pressurizer pressure	2000 psia
Feedwater temperature	177.7 °F
Feedwater flow	153,500 lbm/hr
Steam generator pressure	700.0 psia
Moderator density coefficients, EOC	Table 15.9-3
Doppler reactivity coefficient, EOC	Table 15.9-3

1. With the exception of the reactivity coefficients, analysis values are nominal values without bias or uncertainty applied as described in Reference 15.9-1.

Table 15.9-5: Increase in Heat Removal by the Secondary System - Limiting Analysis Results (Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤ 0.8	0.01

Table 15.9-6: Increase in Heat Removal by the Secondary System - Limiting Analysis Results (20 Percent of Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤ 0.8	0.08

Table 15.9-7: Decrease in Heat Removal by the Secondary System - Limiting Analysis Results (Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤ 0.8	0.05

Table 15.9-8: Decrease in Heat Removal by the Secondary System - Limiting Analysis Results (20 Percent of Rated Power Case)

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤ 0.8	0.16

**Table 15.9-9: Reactivity and Power Distribution Anomalies - Limiting Analysis Results
(Rated Power Case)**

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤ 0.8	0.03

**Table 15.9-10: Reactivity and Power Distribution Anomalies - Limiting Analysis Results
(20 Percent of Rated Power Case)**

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤ 0.8	0.13

**Table 15.9-11: Decrease in Reactor Coolant Inventory - Limiting Analysis Results
(Rated Power Case without Trip)**

Acceptance Criteria	Limit	Analysis Value
Decay ratio	≤ 0.8	0.02

**Table 15.9-12: Decrease in Reactor Coolant Inventory - Sequence of Events
(Rated Power Case with Trip)**

Event	Time (seconds)
Depressurization begins ¹	10.0
Control rods fully inserted ²	18.1
Loss of subcooling in the riser	19.2

1. The case is initialized at steady state conditions for 10 seconds before the transient is initiated.

2. A bounding 5 second delay is assumed from the time the low pressurizer pressure setpoint is reached until control rods are fully inserted to account for the trip delay and rod insertion.

Figure 15.9-1: Primary Coolant Flow Response to a Perturbation at 5 Percent of Rated Power and Beginning of Cycle Reactivity

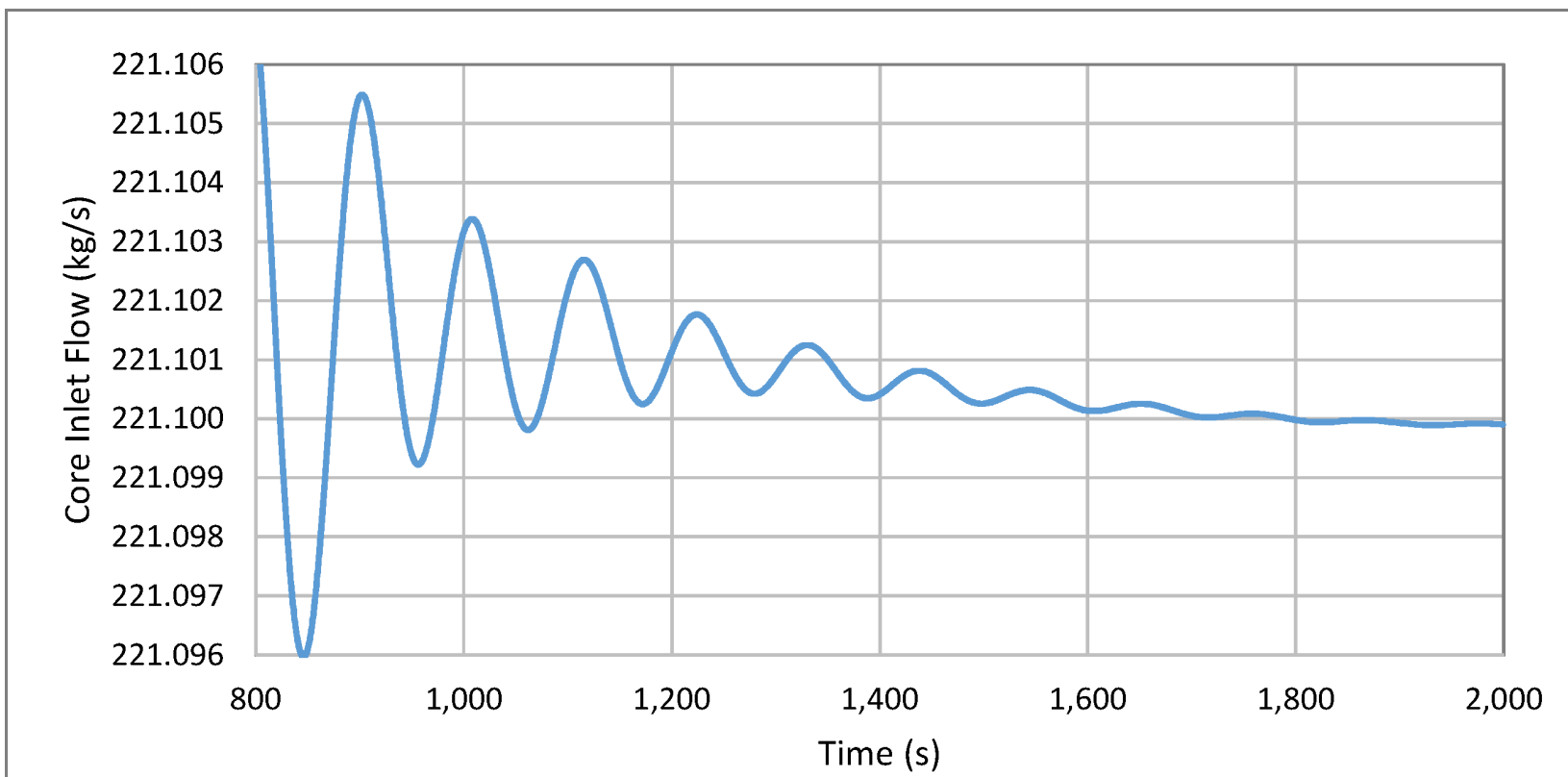


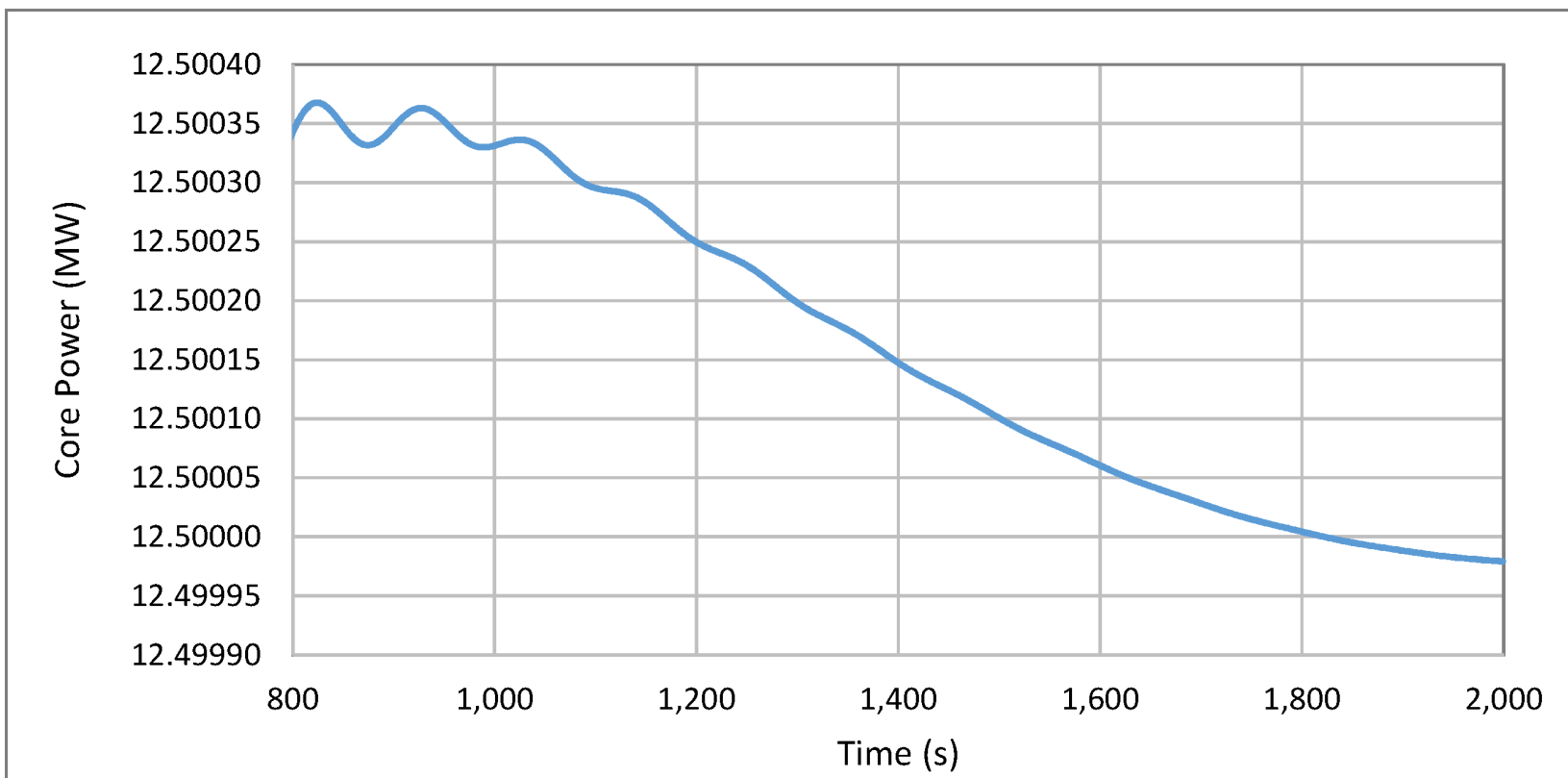
Figure 15.9-2: Power Response to a Perturbation at 5 Percent of Rated Power and Beginning of Cycle Reactivity

Figure 15.9-3: Primary Coolant Flow Response to an Increase in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity

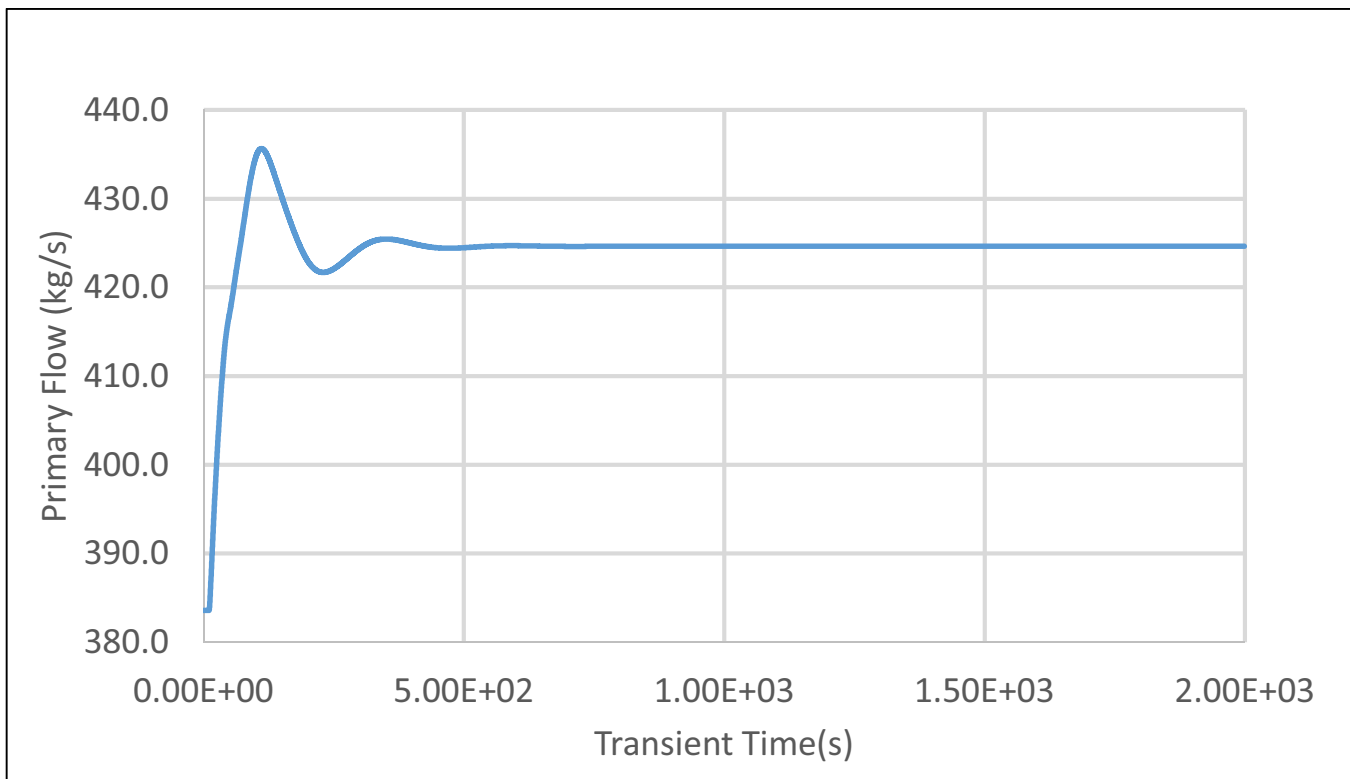


Figure 15.9-4: Power Response to an Increase in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity

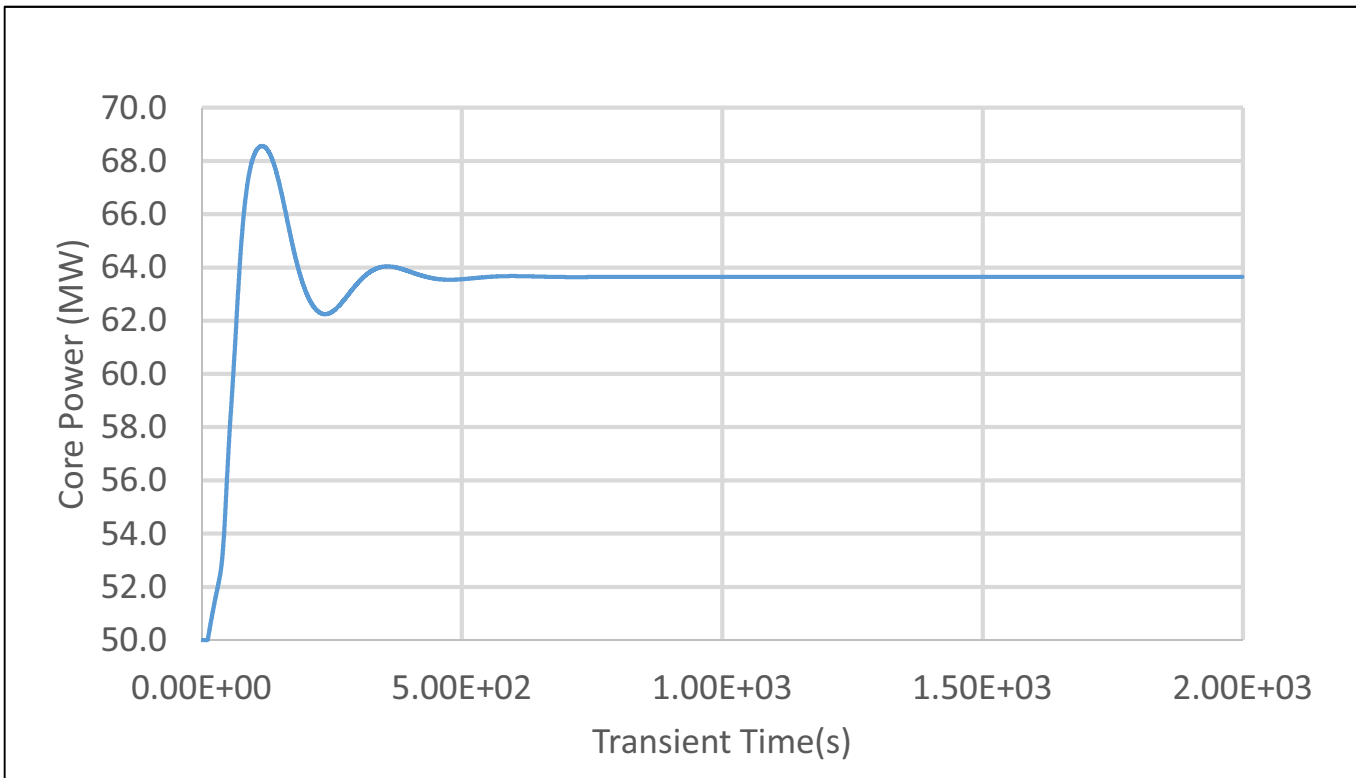


Figure 15.9-5: Primary Coolant Flow Response to a Decrease in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity

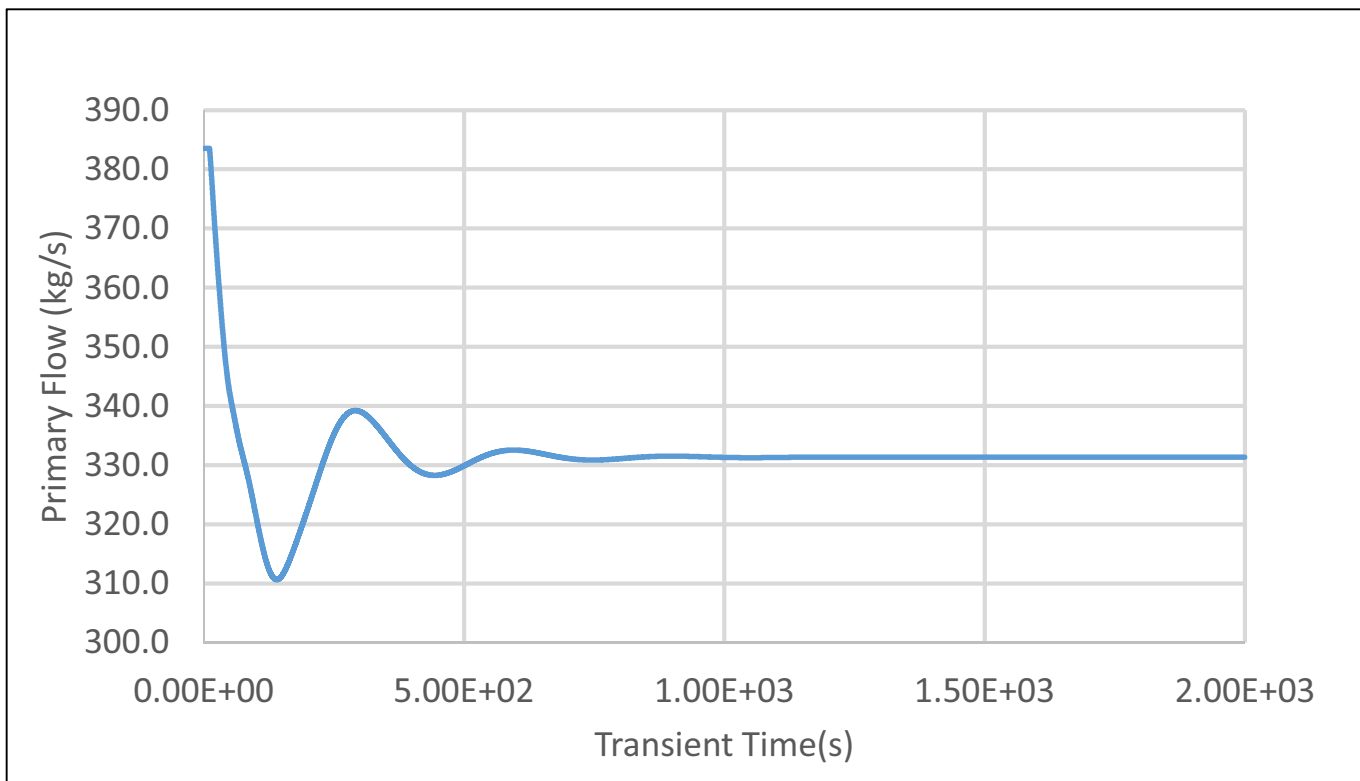


Figure 15.9-6: Power Response to a Decrease in Heat Removal by the Secondary System at 20 Percent of Rated Power and End of Cycle Reactivity

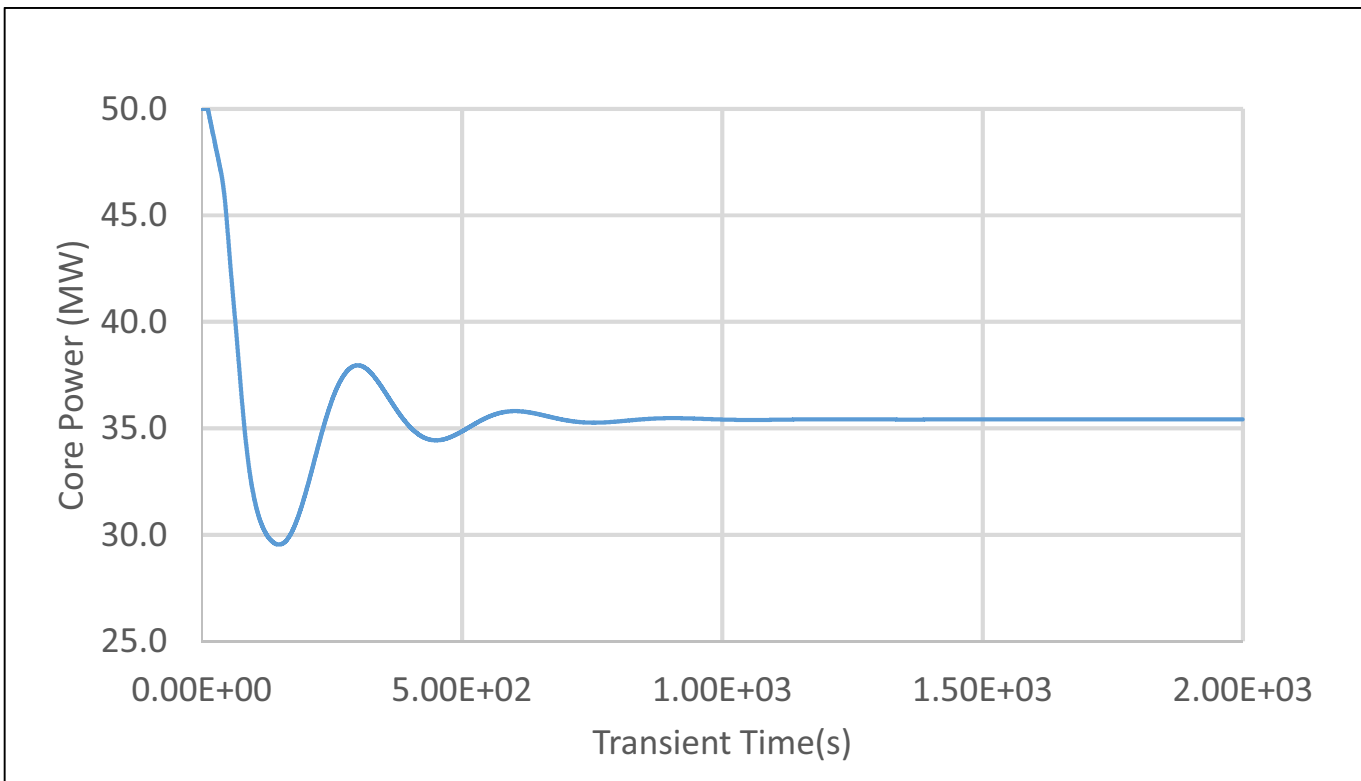


Figure 15.9-7: Primary Coolant Flow Response to Reactivity and Power Distribution Anomalies at 20 Percent of Rated Power and End of Cycle Reactivity

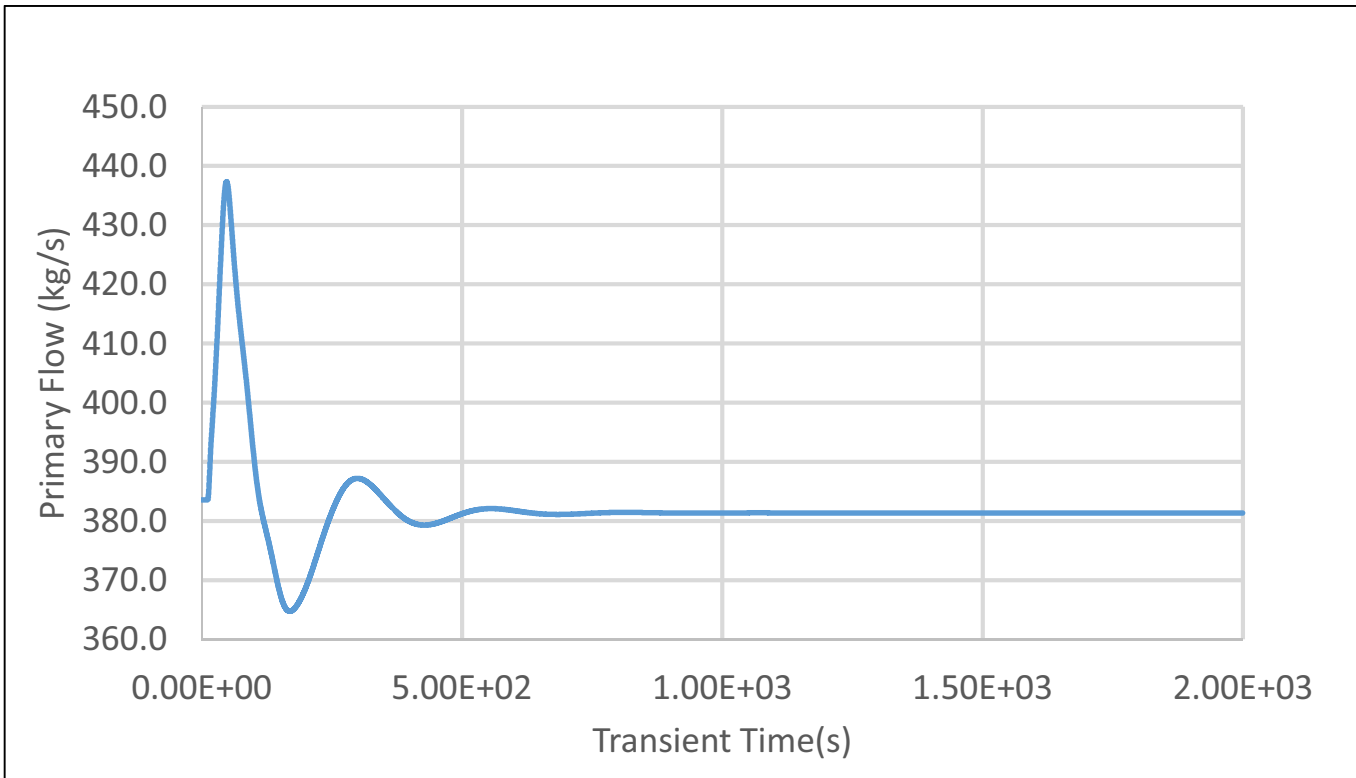


Figure 15.9-8: Power Response to Reactivity and Power Distribution Anomalies at 20 Percent of Rated Power and End of Cycle Reactivity

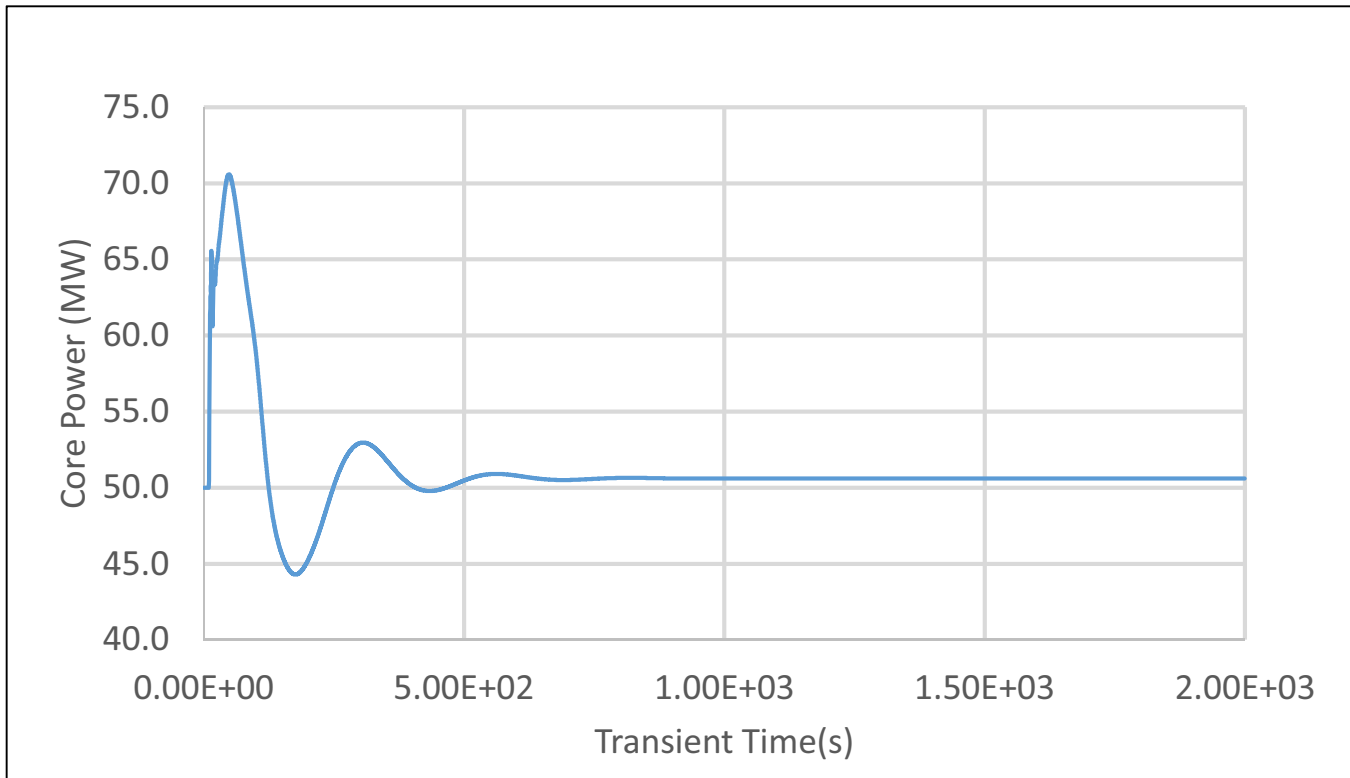


Figure 15.9-9: Primary Coolant Flow Response to a Decrease in Reactor Coolant Inventory at 100 Percent of Rated Power and End of Cycle Reactivity

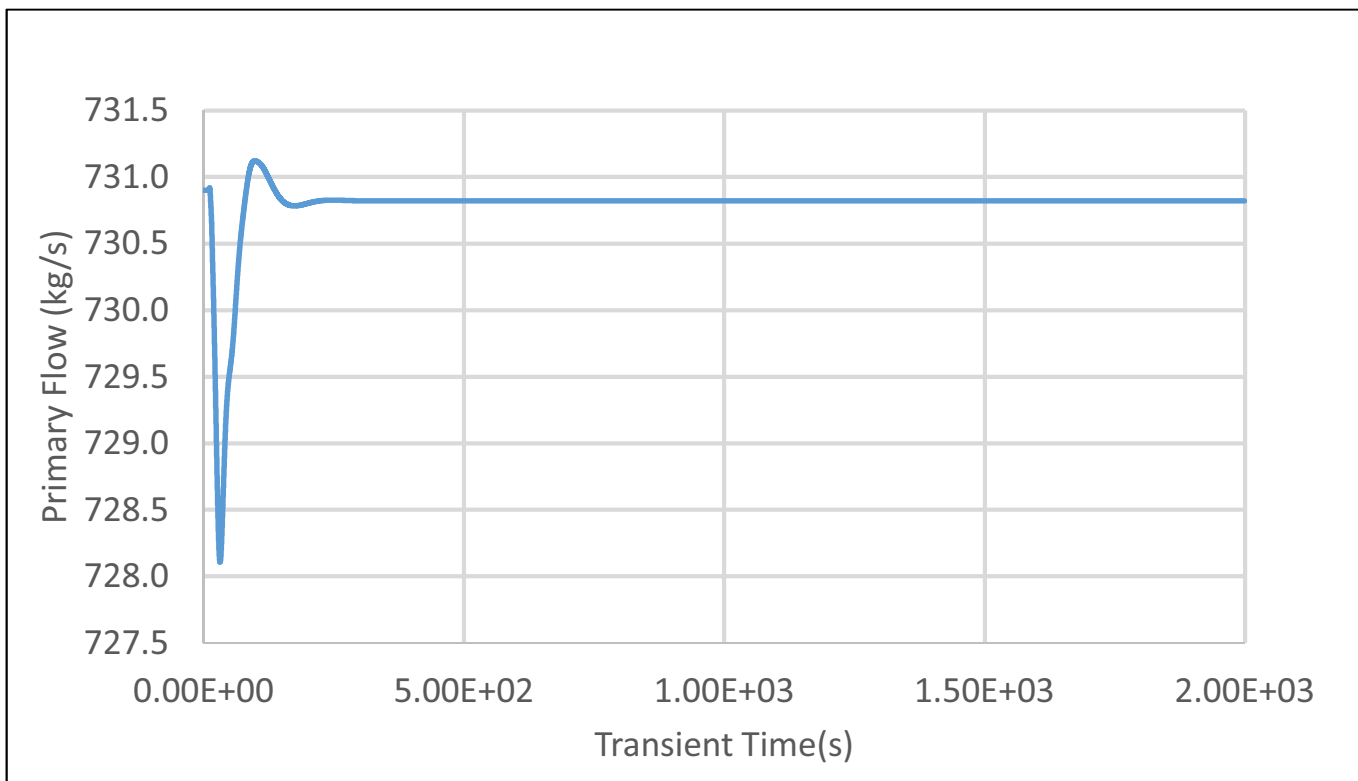
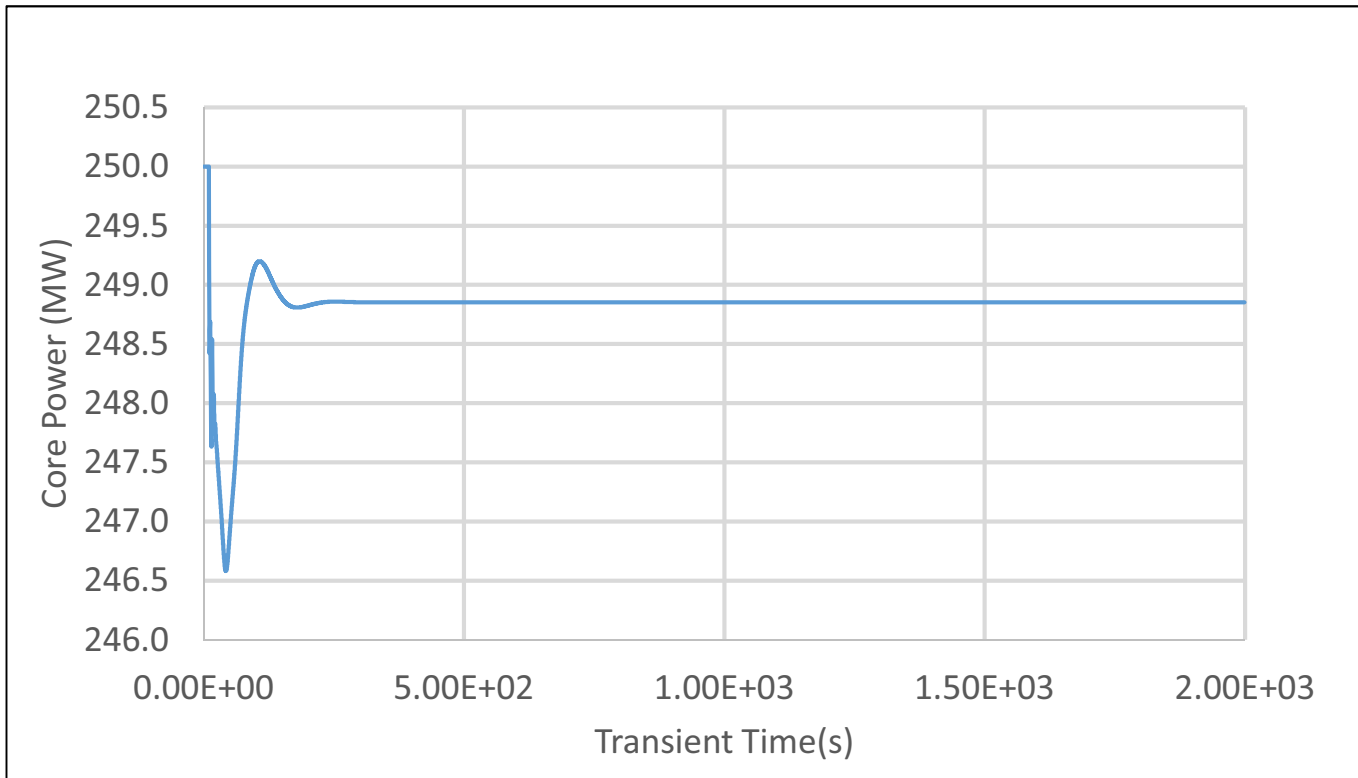


Figure 15.9-10: Power Response to a Decrease in Reactor Coolant Inventory at 100 Percent of Rated Power and End of Cycle Reactivity



15.10 Core Damage Event

A beyond-design-basis core damage event (CDE), with an associated core damage source term (CDST) consisting of a set of key parameters derived from a spectrum of surrogate accident scenarios, is postulated. The analysis of the beyond-design-basis CDST against the acceptance criteria of 10 CFR 52.137(a)(2)(iv) provides reasonable assurance that, even in the extremely unlikely event of a severe accident, the NuScale Power Plant US460 standard design features and site characteristics provide adequate protection of the public.

15.10.1 Inputs, Methods, and Assumptions

15.10.1.1 Core Radionuclide Inventory

The core radionuclide inventory described in Section 15.0.3 and shown in Table 11.1-1 is assumed for the CDST.

15.10.1.2 Primary Coolant Radionuclide Inventory

The primary coolant radionuclide inventory is assumed to be zero and is not considered as a contributor to dose in the CDE radiological consequence analysis, in accordance with the Accident Source Term Methodology Topical Report (Reference 15.10-1).

15.10.1.3 Secondary Coolant Activity

The secondary coolant activity is assumed to be zero and is not considered as a contributor to dose in the CDE radiological consequence analysis, in accordance with the methodology of Reference 15.10-1.

15.10.1.4 Source Term Release Timing and Magnitude

The CDST associated with the CDE consists of a set of key parameters, such as fuel release fractions and timing, derived from a spectrum of surrogate accident scenarios. A surrogate accident scenario is a postulated event that results in core damage with subsequent release of appreciable quantities of fission products into an intact containment, that serves as a surrogate to the large break loss-of-coolant accident with a substantial meltdown of the core typically evaluated for light water reactors. Surrogate accident scenarios derived from intact-containment internal events in the Level 1 probabilistic risk assessment were used to establish the CDST in accordance with the methodology of Reference 15.10-1.

Each of the surrogate accident sequence cases involves various failure of the emergency core cooling system (ECCS) (i.e., all ECCS valves failing to open, the reactor vent valves (RVVs) failing to open, or the reactor recirculation valves (RRVs) failing to open). In each case, the decay heat removal system is assumed available to remove heat.

Case 1: chemical and volume control system (CVCS) injection line break with all ECCS valves failing to open.

Case 2: CVCS injection line break with RVVs failing to open.

Case 3: CVCS injection line break with RRVs failing to open.

Case 4: loss of direct current power with the RVVs failing to open.

Case 5: loss of direct current power with the RRVs failing to open.

Each surrogate accident scenario case is modeled using MELCOR to provide a representative range of release timing and fractions for the development of the CDST. The methodology used to identify the release magnitude and timing of the CDST is provided in Reference 15.10-1. Release timing and core inventory release fractions for the CDST are listed in Table 15.10-1 and Table 15.10-2, respectively.

The radioactive source term is calculated by multiplying the core inventory from Section 15.10.1.1 by the release fractions in Table 15.10-2. The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment.

15.10.1.5 Aerosol and Elemental Iodine Removal

Natural deposition phenomena, including sedimentation, diffusiophoresis, thermophoresis, and hygroscopicity, result in aerosol removal. The NuScale aerosol removal methodology uses the aerosol removal code STARNAUA to track these various deposition phenomena in calculating time-dependent airborne aerosol mass and removal rates, as described in Reference 15.10-1. Aerosol removal rates used for the CDE radiological consequence analysis are in Table 15.10-3. A key assumption of the aerosol transport methodology is that there is no maximum limit on iodine decontamination factor because removal is facilitated by natural processes, as opposed to an active spray system. The removal rate calculation methodology is based on the calculated time-dependent airborne aerosol mass in accordance with Appendix A, Section 3.3 of Regulatory Guide 1.183. The methodology conservatively does not take credit for elemental iodine removal; only aerosol removal is credited. A summary of the aerosol transport and removal calculation process is described in Reference 15.10-1. Treatment of aerosol resuspension and revaporization is discussed in Reference 15.10-1.

15.10.1.6 Chemical Form of Iodine

Reference 15.10-1 provides the methodology for the radiological consequences of the CDE, based on the guidance provided in Appendix A of Regulatory Guide 1.183. The chemical form of radioiodine released to the containment atmosphere is 95 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide. Note that the methodology considers cesium iodide as an aerosol.

15.10.1.7 RADTRAD Modeling

The RADTRAD modeling techniques described in Reference 15.10-1 are used to analyze the CDST.

15.10.1.8 pH_T and Iodine Re-Evolution

The CDE radiological consequence methodology calculates the post-accident pH_T. The pH_T code is used to calculate the extent of iodine re-evolution inside containment. During the postulated CDE, additional acids and bases may enter the coolant and cause a change in pH_T. The expected overall pH_T of the coolant is modeled over a period of 30 days. Reference 15.10-1 provides a discussion of the NuScale pH_T program used to calculate post-accident pH_T.

Details about the methodologies used for evaluating post-accident pH_T in coolant water following an event that results in significant core damage are presented in Reference 15.10-1. The results of implementing the methodology show the post-accident pH_T inside containment is between 6.0 and 7.0.

The CDE radiological consequence methodology assumes a negligible amount of iodine re-evolution occurs between pH_T values of 6.0 and 7.0, and iodine re-evolution does not need to be explicitly included in the dose analysis calculation. This assumption simplifies the analysis without an impact to the conservatism of the calculated dose results. The treatment of iodine re-evolution is described in Reference 15.10-1.

15.10.1.9 Atmospheric Dispersion Factors (χ/Q), Breathing Rates, and Occupancy Factors

The atmospheric dispersion factor (χ/Q), breathing rate, and occupancy factor inputs to RADTRAD described in Section 15.0.3 are assumed in the CDE radiological consequence analysis.

15.10.1.10 Dose Conversion Factors

The dose conversion factors described in Section 15.0.3 are assumed in the CDE radiological consequence analysis.

15.10.1.11 Containment Leakage

The containment leakage assumptions described in Section 15.0.3 are assumed in the CDE radiological consequence analysis. Activity is released to the atmosphere from the containment at the design-basis containment leak rate provided in Table 6.5-1 for 24 hours, and at 50 percent of the design-basis containment leak rate after 24 hours.

15.10.1.12 Secondary-Side Decontamination

The secondary-side decontamination assumptions described in Section 15.0.3 are assumed in the CDE radiological consequence analysis.

15.10.1.13 Reactor Building Decontamination Factors

The Reactor Building decontamination assumptions described in Section 15.0.3 are assumed in the CDE radiological consequence analysis.

15.10.1.14 Receptor Location Considerations

The receptor location, Main Control Room, and Technical Support Center assumptions described in Section 15.0.3 are assumed in the CDE radiological consequence analysis.

15.10.2 Radiological Consequences of the Core Damage Source Term

Using the inputs, methods, and assumptions described in Section 15.10.1, the potential radiological consequences of the CDE are calculated and presented in Table 15.0-10. As shown in Table 15.0-10, the design meets the radiological acceptance criteria for the CDE.

15.10.3 References

- 15.10-1 NuScale Power, LLC, "Accident Source Term Methodology," TR-0915-17565-P-A, Revision 4.

Table 15.10-1: Core Damage Source Term Release Timing

Parameter	Value (hours)
Delay of radionuclide release into containment	2.6
Duration of radionuclide release into containment	1.0

Table 15.10-2: Core Inventory Release Fractions

Radionuclide Group	Release Fraction into Containment Vessel
Noble gases	7.1E-01
Halogens	2.0E-01
Alkali metals	2.5E-01
Alkaline earths	5.7E-03
Tellurium group	2.0E-01
Molybdenum group	5.9E-02
Noble metals	1.3E-03
Lanthanides	2.7E-08
Cerium group	2.7E-08

Table 15.10-3: Containment Aerosol Removal Rates

Time (hours)	Removal Rate (hours ⁻¹)
0.00E+00	0.00E+00
2.60E+00	1.44E+01
2.81E+00	6.86E+00
3.01E+00	2.51E+00
3.28E+00	1.74E+00
3.52E+00	1.61E+00
3.74E+00	1.55E+00
4.10E+00	1.54E+00
5.49E+00	7.35E-01
5.10E+01	0.00E+00

Section B

The table below provides the NuScale responses to each of the Nuclear Regulatory Commission readiness assessment observations on draft Chapter 15, "Transient and Accident Analyses" of the Standard Design Approval Application.

Section	Observation	Response
15.0.0.6.5	The SDAA and additional electronic reading room information is missing a discussion and sufficient justification of the safety classification determination for the EDAS as nonsafety-related. The EDAS does not appear to comply with the 10 CFR 50.2 definition of safety-related considering that the function of the EDAS is relied on to keep the ECCS valves closed during certain design basis events with initiating event frequencies of an AOO. The current analysis appears to treat the EDAS as if it is classified safety-related or Class 1E by assuming it is available at event initiation then it is presumed to always successfully perform its safety function. Given the EDAS is classified as nonsafety-related the SDAA is missing a discussion and design basis analysis using approved codes and methods of the consequences of the EDAS not performing the function of keeping the ECCS valves closed at any time after event initiation. The SDAA should include a determination on whether AOO acceptance criteria would continue to be met, the impact on PCT, the impact on the containment, the impact on fuel damage and impact on offsite dose consequences. The SDAA does contain some discussion of the EDAS FMEA and expected frequency of loss, which would be pertinent if in lieu of the design basis analysis described in the above paragraph, an exemption from applicable regulations were being sought for the EDAS. However, no exemption is mentioned. If regulatory compliance is not intended, then appropriate technical justification for a(n) exemption(s) must be provided.	Section 15.0.0.6.5 has been revised to provide additional information regarding the availability of the augmented DC power system (EDAS) during the Chapter 15 event analyses.
15.0.3	NuScale indicated in the entrance presentation for this phase that shine radiation exposure had not been updated in the Chapter 15 radiological analysis. Section 15.0.3.6.4, also indicates that the methodology for calculating doses from shine following a radiological release will be different than the methodology in TR-0915-17565-P-A, Revision 4 (which was what was used in the DCA). Therefore, the new methodology also needs to be described in the SDA application.	Section 15.0.3.6.4 has been revised to describe the methodology for calculating shine doses. The shine dose results have also been provided in Section 15.0.3.6.4 and Table 15.0-10.

Section	Observation	Response
15.0.5	This section does not contain a discussion and evaluation of the buildup and transport of radiolytic gases (H₂ and O₂) during the long-term phases when (1) decay and residual heat removal while RCS inventory is retained inside the RPV or (2) decay and residual heat removal while RCS inventory has been redistributed between the RPV and the CNV after the RRVs and RRVs are opened. Discussion and evaluations for buildup and transport of radiolytic gases on heat transfer and combustion limits should be included. (ex. Section 15.6.5.3.2 states that zero non-condensable gases are modeled.)	The Extended Passive Cooling and Reactivity Control Methodology topical report (TR-108553) documents that the potential mass of non-condensable gas in the steam generator (SG)/ decay heat removal system (DHRS) loop, including effects of radiolysis has negligible impact on heat transfer capacity. The LOCA topical report assumes conservative values for reactor pressure vessel (RPV) and containment vessel (CNV) non-condensable gases, which includes radiolytic gases. The loss of coolant accident (LOCA) evaluation model is used to calculate CNV peak pressure. Final Safety Analysis Report (FSAR) Section 6.2.5.3 addresses combustible gases. No information about radiolytic or combustible gases is included in Section 15.0.5.
15.0.5	Page 15.0-29, discussion concerning “the following are decay and heat removal scenarios” lacks sufficient context. Is discussion a comprehensive treatment of all possible scenarios, or of bounding scenarios? If not, then what is the significance of these particular scenarios? Additional context is needed to document basis for acceptability.	Additional information was added to the four scenarios described in Section 15.0.5. Also, sections were added to 15.0.5 to describe the analysis done for DHRS and emergency core cooling system (ECCS) extended passive cooling.
15.0.5	Page 15.0-30, for scenarios 1-3, information concerning the flow hole locations has been reduces relative to DCA. There is no information concerning which sets of flow holes are operative or necessary for these scenarios (i.e., both sets, upper, lower), nor why flow hole placement is adequate. While some details may be relocated to a topical report, key points necessary to understand the design, its safety basis, and significant differences in plant response for the specific scenarios being contrasted in the discussion, must be included in the SDAA.	Additional information was added to the four scenarios described in Section 15.0.5. Also, sections were added to 15.0.5 to describe the analysis done for DHRS and ECCS extended passive cooling.
15.0.5	Page 15.0-30, for scenarios 2 and 3, discussion of operator actions should be clarified. Existing description could be construed as credit for operator action to bypass the ECCS timer. However, given passive plant design, not clear such is the intent. Discussion should clearly distinguish	Additional information was added to Section 15.0.5 to describe how the operator action for blocking the ECCS 8-hour signal is addressed in the analysis. The action is not

Section	Observation	Response
	credited actions and plant responses within the design basis from expected actions and system responses that are not credited but which could occur.	credited. The analysis addresses the scenarios with and without the action since the action may be taken by operators.
15.0.5	Section 15.0.5 does not appear to contain sufficient information supporting the evaluation of flow hole adequacy. DCA information on this topic has been substantially removed. While referencing a topical report (bottom of page 15.0-29) for detailed information is appropriate, SDAA should still summarize technical basis for flow hole adequacy.	Additional information was added to the four scenarios described in Section 15.0.5. Also, sections were added to 15.0.5 to describe the analysis done for DHRS and ECCS extended passive cooling.
15.4.6	The analysis description of reactor pool dilution during Mode 5 operation does not provide a basis for relying on the maximum reactor pool water level technical specification to terminate the dilution transient. This appears to credit operator action, please clarify.	The analysis of a reactor pool dilution in Mode 5 determines the volume of pure water required to dilute the RCS (by dilution of the pool) to result in a loss of shutdown margin, independent of the potential source and flow rate of such a volume. The results show this dilution volume is much larger than the volume that would result in exceeding technical specification limits. Compliance with the plant technical specifications, including all required actions and their timing, is a license requirement at all times. Therefore, the technical specifications for pool level preclude the possibility of a loss of shutdown margin in Mode 5. No specific operator actions, or timing of actions, are credited in the analysis.
15.6.5	Chapter 15 is missing a discussion and evaluation of the LOCA break spectrum related to the impact of the venturis in the ECCS valve to vessel connections. Specifically, Chapter 15 does not include an analysis, nor includes a surrogate or other similar location yielding representative results, for the hypothetical loss of coolant at the reactor vessel to ECCS valve flange (i.e., vessel to piping system boundary, but upstream of the venturi). A break at this location will produce greater consequences than what is currently analyzed. LOCAs are defined in 10 CFR 50.46(c) and 10 CFR 50, Appendix A, and analyzed in FSAR Chapter 15 as nonmechanistic hypothetical breaks to establish the design basis of the emergency core cooling system. The LOCA break spectrum is not associated with, or within the scope of GDC 4. GDC 4 only addresses protection against the dynamic effects associated with pipe breaks and while consideration of certain dynamic effects may be excluded the functional design for the emergency core	Section 15.6.5 has been revised to state that the ECCS valve flange connections are not a break location based on Section 3.6.2 and are not evaluated.

Section	Observation	Response
	cooling system is separate and must be based on the 10 CFR 50.46 nonmechanistic losses of coolant. The SDAA must include an analysis of the full LOCA break spectrum. If the regulations are not intended to be complied with then appropriate justification for a(n) exemption(s) must be provided.	
15.9	SDA Section 15.9 references the topical report TR-0516-49417-P-A, Revision 1 for stability analysis methodology. This section is missing a discussion and evaluation of the applicability of the topical report to the SDA. The analysis presented in the referenced topical report contains analysis for DCA conditions instead of the requested SDA conditions.	A discussion of the applicability of topical report TR-0516-49417-P-A to the US460 design was added to FSAR Section 15.9.
15.9	SDA Table 15.9-3 presents initial conditions for an updated stability analysis. This section is missing a discussion in the SDA section that discusses the related analyses or how the initial conditions were determined. For example, the table indicates the pressurizer pressure assumed is 2000 psia. However, NuScale's SDA design pressure is 2070 psia.	A discussion of the source of the FSAR Table 15.9-3 values has been added to the table. Note that the US460 design pressure is not 2070 psia. The design pressure is 2200 psia per FSAR Table 5.3-1. The nominal operating pressure is 2000 psia, with an uncertainty of 70 psia, per FSAR Table 15.0-6.
15.9	Significant technical details have been removed from this section without justification.	The technical details removed from Section 15.9 (relative to the corresponding design certification application (DCA) Section 15.9) were discussions that were duplicative of content in the topical report TR-0516-49417-P-A, Revision 1, "Evaluation Methodology for Stability Analysis of the NuScale Power Module." The methodology referenced in Section 15.9 is an NRC approved evaluation methodology for stability analysis. Changes to the stability analysis evaluation methodology would require NRC review and approval.
15.10	Descriptions of the provisions to ensure operability of any credited non-safety systems in any of the dose consequence analyses in chapter 15 should be included. All dose consequence analyses in chapter 15 (design basis and beyond design basis) should use conservative 95% atmospheric dispersion coefficients. In addition, the information in the application should be consistent with what was used in the analysis.	Section 15.0.0.6.6 discusses treatment of nonsafety-related systems including the dose consequence analyses. Section 15.10.2 has been revised to eliminate discussion of possible alternate assumptions, such as for atmospheric dispersion.

Section C

TR Number	TR Title
TR-102621-NP Revision 0	Treatment of DC Power in Safety Analysis

Licensing Technical Report

Treatment of DC Power in Safety Analyses

December 2022

Revision 0

Docket: 52-050

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Abstract

This technical report provides a single-source document of the technical and regulatory bases for the safety analysis treatment of the augmented DC power system for the NuScale Power Module in the US460 Standard Design. The document provides a high-level overview of the following:

- The US460 design, including the NuScale Power Module, the emergency core-cooling systems, the decay heat removal system, and the DC electrical power systems.
- The NuScale design process to determine the safety and risk classification of systems, structures, and components, including the application of this process to the DC power electrical systems.
- Event identification and classification for safety analyses
- Identification and quantification of initiating events for safety analysis
- Deterministic technical basis for the loss of power considerations in safety analysis
- An evaluation of the inadvertent operation of the emergency core cooling system with the loss of augmented DC power
- A summary of the probabilistic risk assessment for the US460

Executive Summary

The US460 NuScale Power Module is a unique integral PWR design that relies on natural circulation and a limited number of safety-related system to mitigate the consequences of postulated accidents.

NuScale's methodology for classifying and categorizing SSCs is a risk-informed, performance-based design process consistent with guidance in ANSI/ANS 30.3 and ANSI/ANS 58.14. The NRC approved NuScale's risk significance determination methodology to determine system-level categorizations in TR-0515-13952-NP-A, Revision 0. Design system functions are assigned to plant-level safety functions and categorization of those system functions based on risk-significance and defense-in-depth. A systematic method is applied to ensure that specific plant SSC are designed, manufactured, procured, installed, and operated with established processes to ensure that their quality and reliability are commensurate with their influence on safety.

SSC are classified based on the functions provided in the licensing basis. These classifications are (1) safety-related, (2) non-safety-related, and (3) nonsafety-related with augmented requirements (i.e., special treatment). Safety-related SSCs are relied upon to remain functional during and following design-basis events to meet the requirements of 10 CFR 50.2. SSCs not classified as safety-related are classified as nonsafety-related (with or without augmented requirements).

Nonsafety-related system functions are screened against a set of generic functions needed to meet the General Design Criteria or other regulatory requirements to determine if the nonsafety-related functions are supplemented with augmented requirements. When this NRC-approved process is applied to the augmented DC power system (EDAS), the EDAS is classified as not risk-significant, non-safety related with augmented design requirements.

The design of the EDAS prevents the failure of any single power channel from causing a loss of DC power to the module. An initiating event analysis has determined that probability of a random failure (at any time) of the EDAS is $2.6\text{E-}04$ mcyr (per module critical year), and the probability of an EDAS "smart failure" during a reactivity addition accident is on the order of $1.5\text{E-}8$ mcyr.

In the unlikely event of an EDAS "smart failure" during a reactivity insertion accident, analyses show that the CHFR remains above the MCHFR limit for the majority of state-point conditions.

}}2(a),(c) well below the peak clad temperature limit of 2200°F for 10 CFR 50.46, preserving a coolable geometry.

Finally, the NuScale probabilistic risk assessment quantifies the risk impact of all identified and credible design-basis and beyond design-basis events. The risk metrics that NuScale quantifies are core damage frequency and large release frequency, which the NRC have identified as surrogates for cancer fatality risk and early fatality risk, respectively. The results of the PRA show that the risk metrics satisfy the NRC safety goals by multiple orders of magnitude, thereby preserving public health and safety. Additionally, when compared to the PRA results from the US600, approved by the NRC in the US600 DCA, the large release frequency is reduced by two orders of magnitude.

Probabilistic risk assessment results for NuScale designs

Design - Model	CDF	LRF
US460 - Base model	5.4E-09	3.4E-13
US600 - Base model	2.7E-10	1.7E-11
NRC Safety Goal	1.0E-04	1.0E-06

1.0 Introduction

1.1 Purpose

The purpose of this report is to provide a technical and regulatory justification regarding the availability of the nonsafety-related augmented direct current (DC) electrical system (EDAS) during design-basis events considered in the safety analysis. This report demonstrates that the approach used in the Final Safety Analysis Report (FSAR) Chapter 15 safety analysis is consistent with regulatory requirements for safety analysis and consistent with the deterministically evaluated mechanistic, causal system/component failures of the electrical system design for the NuScale Power Plant US460 standard design.

1.2 Scope

Section 2.0 provides a description of the NuScale Power Module and its operation. Section 3.0 provides an overview of the NuScale safety and risk classification for systems, structures, and components (SSC). Section 4.0 discusses the event initiation frequencies for the safety analysis. Section 5.0 summarizes the event classification process for the safety analyses based on the event frequencies and design and operation of the NPM. Section 6.0 provides the application of the NuScale design process to the EDAS. Section 7.0 provides analysis of the inadvertent operation of a release valve (IORV) event assuming the coincident loss of AC and DC power in conjunction with the worst stuck control rod. Finally, Section 8.0 shows that the probabilistic risk analysis results for the US460 are more favorable than the NRC-approved results from the US600 and exceed NRC minimums by more than an order of magnitude.

Additional details are in the appendices. Appendix A provides a listing of the applicable regulatory requirements. Appendix B provides further detail of the US460 design and operation, including the ECCS, DHRS and DC power system. Appendix C provides the technical and regulatory basis for the loss of power considerations for safety analyses

1.3 Abbreviations

Table 1-1 Abbreviations

Term	Definition
AC	alternating current
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
BWR	boiling water reactor
CCF	common cause failure
CDF	core damage frequency
CES	containment evacuation system
CFR	Code of Federal Regulations
CHF	critical heat flux
CHFR	critical heat flux ratio
CHW	chilled water
CNV	containment vessel
CNTS	containment system
CRDS	control rod drive system
CRHS	control room habitability system
CVCS	containment volume control system
DBE	design basis event
DC	direct current
DHRHX	decay heat removal heat exchanger
DHRS	decay heat removal system
DID	Defense-in-depth
D-RAP	design reliability assurance program
DSRS	design specific review standard
ECCS	emergency core cooling system
EDAS	augmented direct current power system
EDNS	normal direct current power system
EHVS	high voltage AC electrical system
ELVS	low voltage AC electrical system
EMVS	medium voltage AC electrical system
EPRI	Electric Power Research Institute
ESF	engineered safety feature
FMEA	failure modes and effects analysis
FSAR	final safety analysis report
GDC	general design criteria
GQA	graded quality assurance
IAB	inadvertent actuation block
IEEE	Institute of Electrical and Electronics Engineers
INL	Idaho National Laboratory
IORV	inadvertent operation of a relief valve
LBE	licensing basis events
LOCA	loss of coolant accident
LRF	large release frequency

Table 1-1 Abbreviations (Continued)

Term	Definition
LWR	light water reactor
MCHFR	minimum critical heat flux ratio
MCR	main control room
MCS	module control system
mcyr	module critical year
MFIV	main feed isolation valve
MLD	master logic diagram
MOV	Motor operated valve
MPS	module protection system
MSIV	main steam isolation valve
NPM	NuScale power module
NSSS	nuclear steam supply system
PA	postulated accident
PAM	post-accident monitoring
PCS	plant control system
PCT	peak cladding temperature
PRA	probabilistic risk assessment
PWR	pressurized water reactor
QA	quality assurance
QAPD	quality assurance program description
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	regulatory guide
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RSV	reactor safety valve
RVV	reactor vent valve
RTNSS	regulatory treatment of nonsafety systems
SBO	station blackout
SE	special event
SFP	spent fuel pool
SG	steam generator
SRP	standard review plan
SSC	structures, systems, and components
UHS	ultimate heat sink
VRLA	valve-regulated lead acid

2.0 NuScale Power Module Description and Operation

2.1 General Plant Design

The US460 design consists of up to six NPMs, each is a small, integral PWR with passive safety systems. The NPM consists of the nuclear steam supply system (NSSS), which includes the nuclear core, the helical coil SGs and the pressurizer, within a single pressure vessel and the compact steel containment vessel (CNV) that houses the NSSS.

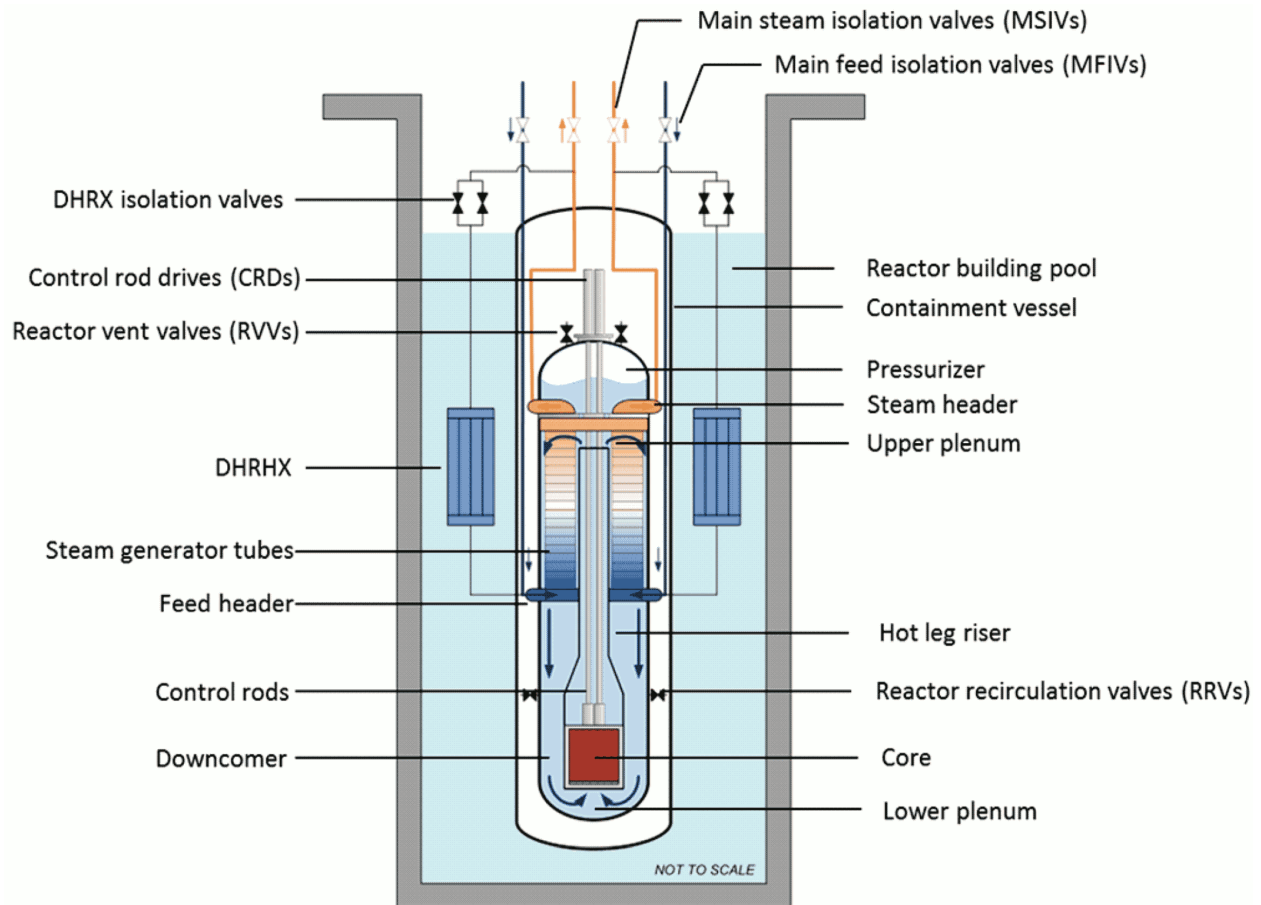
Unique features of the plant design include the following:

- reduced core size
- natural circulation reactor coolant flow (i.e., no reactor coolant pumps)
- integrated SG and a pressurizer inside the reactor pressure vessel (RPV)
- simplified passive safety-related systems
- high-pressure steel containment
- containment partially immersed in a water-filled pool providing an effective passive heat sink for emergency cooling

The NPM is designed to operate efficiently at full-power conditions using natural circulation as the means of providing core coolant flow, eliminating the need for reactor coolant pumps. As shown in Figure 2-1, the reactor core is located inside a shroud connected to the hot leg riser. The reactor core heats reactor coolant, decreasing its density, causing the coolant to flow upward through the riser. When the heated reactor coolant exits the riser, it passes across the tubes of the helical coil SG, which acts as a heat sink. As the reactor coolant passes over the SG tubes, it cools, increases in density, and naturally circulates through the downcomer to the reactor core.

The NPMs are partially immersed in a reactor pool and protected by passive safety-related systems. Each NPM has a dedicated emergency core-cooling system (ECCS), chemical and volume control system (CVCS), and decay heat removal system (DHRS).

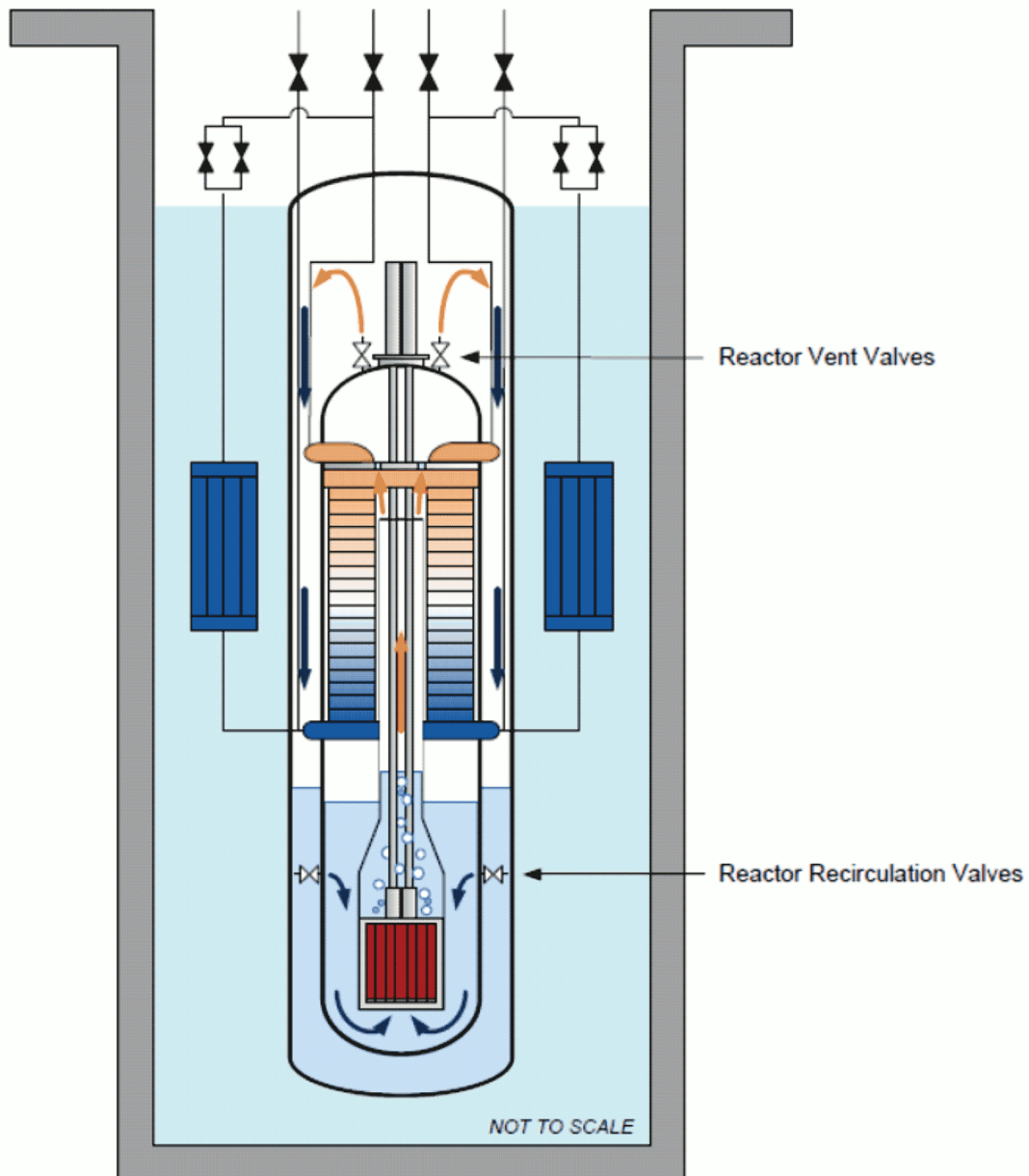
In the US460 design, primary components are integral to the RPV, eliminating external coolant loops and pressurizer piping, which significantly reduces the size and number of possible LOCA scenarios.

Figure 2-1 A Single NuScale Power Module During Normal Operation (Representative)

The NPM is designed with the intent of reducing the impact of a LOCA event. Both the DHRs and the ECCS are designed to actuate in response to a LOCA event. As shown in Figure 2-2, the ECCS consists of two independent reactor vent valves (RVV) and two independent reactor recirculation valves (RRV).

The ECCS is initiated by opening the RVVs exiting the top of the RPV and the RRVs entering the RPV in the downcomer region (above the core elevation). Opening the valves allows the RPV and the CNV pressure to equalize, creating a natural circulation path to remove decay heat from the core. Water that is vaporized in the core leaves as steam through the RVVs, is condensed and collected in the CNV, and is then returned to the downcomer region inside the RPV through the RRVs by natural circulation.

Figure 2-2 Schematic of NuScale Power Module with Decay Heat Removal System and Emergency Core Cooling System in Operation



2.2 Plant Operation

Pressurizer heaters and a pressurizer spray system are used to maintain nominal operating pressure. At full-power conditions, the flow rate is dependent on the fluid density differences through the loop, the losses incurred along the loop, and the elevation difference between the core and the SG.

During nominal full-power conditions, the control rods are retracted up to or above their insertion limits. Borated water is used as the primary coolant and the CVCS regulates the boron concentration to maintain criticality. The CVCS provides reactor inventory make-up through the RCS injection line in the riser and inventory let-down through a separate RCS discharge line in the downcomer region.

The secondary side is operated such that the SGs remove the heat generated by the reactor core. The DHRS heat exchangers are isolated from the steam line and do not remove heat during normal operation.

The containment is evacuated during normal operation to provide an insulated barrier between the reactor and containment.

2.3 Engineered Safety Features Operation

The Module Protection System (MPS) is composed primarily of the reactor trip system and the engineered safety features actuation system. The MPS protection functions are limited to automated safety responses to off-normal conditions. The MPS functional response to an initiating event is a reactor trip; isolation of main feedwater, main steam, CVCS, demineralized water system, and containment; followed by an integrated safety actuation of one or more of the passive safety-related systems (DHRS and ECCS). Containment isolation is achieved by closing the containment isolation valves, which also fail closed on a loss of power.

The reactor trip system consists of four independent separation groups with independent measurement channels to monitor plant parameters that can generate a reactor trip. Each measurement channel trips when the parameter exceeds a predetermined setpoint.

The engineered safety features actuation system also consists of four independent separation groups with independent measurement channels that monitor plant parameters that activate the operation of the engineered safety features.

ECCS is actuated by the MPS on appropriate signals indicating a LOCA event. Interlocks are designed to prevent ECCS actuations for expected operational conditions or non-LOCA transients.

2.3.1 Emergency Core Cooling System

The ECCS is a two-phase natural circulation system that maintains a liquid water supply to the core during its operation in a LOCA scenario, which results in a collapsed liquid level in the RPV that is above the top of the core.

The ECCS consists of two independent RVVs and two independent RRVs. The ECCS is initiated by simultaneously actuating the RVVs on the top of the RPV in the pressurizer region and the RRVs on the side of the RPV in the downcomer region. The RRVs are designed to provide a low-resistance flow path for coolant to flow from the CNV into the RPV. The RVVs are designed to equalize pressure between the two vessels allowing steam from the reactor to vent to the containment and to provide hydrostatic equalization that allows coolant flow through the RRVs back into the reactor downcomer region.

During ECCS actuation, the RPV depressurizes due to liquid and steam exiting the RPV through the RVVs. Steam entering containment is condensed on the containment wall, which in turn is cooled by the reactor pool. Initially, the containment pressure increases to a peak, and then decrease as flow from the RPV decreases and heat is transferred from the CNV to the reactor pool. The RPV water inventory decreases while the containment level increases due to inventory transferred from the RPV.

As the pressure between the two vessels reach a near-equilibrium condition, the collapsed liquid level in the containment rises to a level higher than the RRV elevation, creating enough static head to overcome the pressure difference between the RPV and CNV. At this point, the condensed liquid in containment enters the RPV through the RRVs while steam continues to exit the RPV through the RVVs. This boiling/condensing circulation process continues, maintaining a collapsed water level above the top of the active fuel.

The RRVs are equipped with an inadvertent actuation block (IAB) feature that prevents their opening at full operating pressure. The IAB allows the RRV main valve to open only after the differential pressure between the RPV and CNV has decreased below the IAB release pressure setpoint. Meanwhile, the RVV main valves open immediately after actuation, resulting in rapid depressurization of the RPV into containment that allows ECCS recirculation cooling to be established after the RRVs open.

The ECCS valves open on low differential pressure between the RPV and CNV, independent of an ECCS actuation signal. This action is a function of the mechanical design of the valves, where the valve spring causes the valves to open if the pressure difference across the main chamber drops below approximately 15 psid. The ECCS valves open by this function during some rapid depressurization events as the RPV and CNV pressures equalize. This function opens the ECCS valves earlier in some depressurization scenarios but does not significantly impact accident progression or results.

2.3.2 Decay Heat Removal System

The DHRS is a passive safety-related system that relies on natural circulation to remove heat from the RCS through the SG and reject heat to the reactor pool through the DHRS condenser. The DHRS is composed of two DHRS trains; one train is associated with each of the two NPM SGs. Each DHRS train is capable of independently removing 100 percent of decay heat. The DHRS piping connects to the main steam and feedwater lines specific to the associated SG. During normal operation, the DHRS condenser and piping are isolated by valves on the steam side of the SG. The condensate side of the DHRS is open to the feedwater piping supplying the associated SG.

Upon actuation of the DHRS, the SG feedwater and steam isolation valves, as well as their nonsafety-related backup valves, close and the DHRS isolation valves open, creating a closed loop between the SG and DHRS condenser. Both liquid and vapor are contained in the SG/DHRS loop on system actuation. Because the DHRS is a closed system, the total water mass remains constant during the system operation.

For successful operation, liquid water enters the SG through the feedwater line and is boiled by heat from the RCS. The vapor exits the SG through the steam line and is directed to the DHRS condenser where it condenses back to liquid before return to the SG. Thus, the loop transfers heat from the RCS to the DHRS fluid and then from the DHRS to the reactor pool water.

The bottom of the DHRS condenser is located above the bottom of the SG providing the static head to drive natural circulation.

2.4 Augmented DC Power System

The EDAS consists of two DC subsystems: the module-specific subsystem (EDAS-MS) and the common plant subsystem (EDAS-C).

2.4.1 Module Specific Subsystem

The EDAS-MS contains four power channels for each module. Power channels A and C are a part of EDAS Division I. Power channels B and D are a part of EDAS Division II. Each power channel contains one battery, one battery charger, and one DC distribution panel assembly. Each distribution panel assembly consists of a bus, fused disconnect switch, tie breaker, breakers, relays, metering, associated interconnections, and supporting structure.

Each EDAS-MS power channel charger is sized to carry 100 percent of the divisional DC bus loading during normal plant operation. In the event of a loss of a charger for maintenance or equipment failure, the divisional power channels can be connected together with the functional battery charger providing power to the divisional loads while maintaining connected batteries on float charge.

Each EDAS-MS battery is sized with sufficient capacity to provide power to ECCS Hold Mode loads for 24 hours. The EDAS-MS channel B and channel C are designed with additional capacity to provide battery power to PAM Only Mode loads for 72 hours.

2.4.2 Common Plant Subsystem

The EDAS-C contains two divisions, EDAS Division I and EDAS Division II. Each common plant subsystem division contains one battery, two identical battery chargers, and one DC distribution panel assembly. Each distribution panel assembly consists of a bus, disconnect switch, breakers, relays, metering, associated interconnections, and supporting structure.

Each EDAS-C charger is sized to carry 100 percent of the respective divisional DC bus loading during normal plant operation while maintaining connected batteries on float charge.

Each EDAS-C battery is sized with sufficient capacity to provide power to required loads for 72 hours.

2.4.3 Functions

EDAS functions operationally to:

- Provide electrical power for the prevention of unintended ECCS actuation.
- Provide backup electrical power for Post-Accident Monitoring (PAM) instrumentation for type B, C, D, and F variables.
- Provide module-specific and common DC electrical power with adequate voltage and current to EDAS loads when AC power is either available or not available to EDAS.

Module-specific loads include:

- Module protection system
- Neutron monitoring system
- Radiation monitoring system

Common loads include:

- Safety display indication
- Plant protection system

- Provide module-specific and common DC electrical power with adequate voltage, current, and capacity to specified EDAS module-specific and common loads when AC power is either available or not available to EDAS.

Loads include:

- EDAS battery monitoring system (module-specific, common)
- Main control room lighting (common)
- Provide EDAS instrumentation and control, capability for EDAS battery testing, and EDAS battery monitoring.

2.5 Normal DC Power System

The normal DC power system (EDNS) is an electrical DC and AC distribution system consisting of batteries, battery chargers, direct current (DC) switchboards, inverters, voltage regulating transformers, maintenance bypass switches, alternating current (AC) panel boards, fused transfer switch boxes, battery monitors, associated electrical protective devices, and instrumentation, and interconnecting system cabling.

2.5.1 Functions

The functions supported by the EDNS include:

- Provide AC and/or DC electrical power with adequate voltage and current to EDNS loads.

Loads include:

- The EDNS provides DC control power to the normal high voltage, medium voltage, and low voltage distribution systems (EHVS, EMVS, and ELVS, respectively).
- The EDNS provides electrical power to the module control system (MCS) and plant control system (PCS).
- Provide battery backup AC and/or DC electrical power with adequate voltage and current to EDNS loads when AC power is not available.
 - The EDNS backup battery capacity is 40 minutes.

3.0 Overview of the Risk-Informed Performance-Based Design Process

The NuScale risk-informed, performance-based design process is shown in Figure 3-1 and is consistent with guidance in ANSI/ANS 30.3 (Reference 10.1) and ANSI/ANS 58.14 (Reference 10.2).

The first step in this process is to define plant safety functions and requirements. These need to take into consideration fundamental safety functions and high-level safety goals from the regulator and the designer. Design requirements are the set of functional, nonfunctional, and performance requirements that bound plant design and help ensure a satisfactory final design. The design requirements are dependent on a range of site, technology, and stakeholder-specific needs. These external inputs are formally documented, reviewed, and evaluated together to ensure that a wide variety of design constraints can be appropriately translated into a multilevel design requirements document. The requirements analysis process is the process for examining, evaluating, and translating applicable inputs into design requirements that can be objectively evaluated and verified in later phases of design. Performance-based safety objectives are used to support decision-making related to design and evaluation of plant systems and components.

In supporting the safety requirements definition, the plant response to postulated initiating events including coincident equipment failures and malfunctions that could challenge plant safety are analyzed. The licensing basis events (LBEs) include the collection of design-basis events (DBEs) and special events (SEs) for the plant design. At a minimum, DBEs consist of anticipated operational occurrences (AOOs) and postulated accidents (PAs). Initiating events for DBEs are identified by developing a list of functional transient and accident types that could challenge plant-level safety functions and associated mitigating systems depending on the operating mode of the plant at the time of the challenge. Initiating event selection begins with the listing of potential DBE scenarios using both traditional deterministic processes and the design-specific PRA. Initiating events are categorized according to expected frequency of occurrence and by functional event type. Categorization by frequency of occurrence provides a basis for selection of the applicable analysis acceptance criteria for each initiating event. Categorization of initiating events by functional event type provides a basis for comparison between initiating events, which makes it possible to identify and evaluate the limiting cases. Both traditional deterministic processes and the design-specific PRA should use hazard analysis methods (e.g., failure modes and effects analysis, hazard and operability analysis, and fault tree analysis), combined with malfunctions of individual plant systems and historical experience, as available, for defining an initial list of individual initiators leading to different DBE types.

DBEs are those AOOs and PAs that form the basis for the design of the plant including SSC relied upon to protect design-specific fission product barriers during those transients and accidents. Using the DBEs as input to formal design of the plant results in a robust plant that, using conservative analyses of plant response, can be shown to meet limits established in applicable codes, standards, and regulations with design safety margin, thereby demonstrating adequate protection of the health and safety of the public.

Once the conceptual design has progressed to a high enough level of maturity or detail, additional quantitative evaluations of risk and margins are conducted. From a safety perspective, these evaluations are risk-informed such that they consider both probabilistic and deterministic information. At this early stage in the design process, margins may be small or exceeded in limited cases depending on the state of the design. Design iterations may continue to occur within the conceptual design phase to address these issues, but the design may also progress to the next phase of design if the exceedances are known and plans to address them are agreed upon. In other words, as long as the design is feasible, it may progress to the next phase of design even with some potential exceedance of safety limits or limited margins.

The conceptual design, identification of LBEs, PRA, and design-basis safety analysis activities in practice occur in parallel versus the idealized sequential order shown in Figure 3-1.

Binning of the initiating events into functional event types, the deterministic approach considers the potential for additional malfunctions or plant operating conditions that may occur coincident with the initiating events under evaluation. The design-specific PRA is then reviewed to identify other failures that may be worth consideration as a part of the selected DBEs or accidents beyond those identified deterministically. This review focuses on the consequences of the event or event sequence with a consideration of the probability of the event.

In deterministic safety analyses, potential for single active failures are evaluated following a postulated transient or accident initiating event. These single active failures are postulated within the collection of plant systems relied upon in response to the transient and accident initiating event.

A review of the accident sequences of the PRA is performed to identify any risk-significant mitigating system equipment malfunctions that should be included in the design-basis analyses. Potential equipment malfunctions for inclusion in the design-basis analyses include those single failures and common-cause failure that, when combined with a given initiating event, can challenge regulatory requirements.

To establish whether there are common-cause or multiple random failures that may be risk-significant with respect to the plant design, a review of fault tree analyses for the mitigating systems relied upon during a DBE is performed. Selected dominant contributors to mitigating system failure probability in the form of common-cause or multiple random failures are reviewed and an engineering rationale developed as to whether or not these common-cause or multiple random failures should be included as a part of the DBEs. Such a rationale considers whether the initiating event for the DBE, combined with the common-cause or multiple random failures, would result in challenges to the fission product barriers that exceeded those for existing DBEs or if the probability of the common cause failure is significantly less than that for single failures or coincident occurrences identified

At the end of the conceptual design phase, design system functions are assigned to plant-level safety functions and categorization of those system functions based on risk-significance and defense-in-depth is conducted. As the safety of a nuclear plant is dependent on the reliability and integrity of its individual SSC, a systematic method is necessary to ensure that specific plant SSC are designed, manufactured, procured, installed, and operated with established processes to ensure that their quality and reliability are commensurate with their influence on safety. Selection of these specific SSC is achieved through a classification and categorization process. NuScale utilizes an NRC-approved risk significance determination methodology in TR-0515-13952-NP-A, Revision 0 (Reference 10.3) and the SSC classification process to first determine system-level categorizations. Once this classification is achieved, the preliminary design phase can begin.

The system functions that are relied upon in the safety analyses of DBEs to provide or support the plant-level safety-related functions are determined. These systems and their corresponding safety-related plant-level functions are classified as safety-related. Only those systems necessary and sufficient to accomplish any of the basic safety-related functions need to be classified as safety-related. Not all systems capable of performing a given plant-level safety function may need to be classified as safety-related and that for some DBEs, not all plant-level safety-related functions are required.

Nonsafety-related system functions are screened against a set of generic functions needed to meet the General Design Criteria or regulatory requirements to determine if the nonsafety-related functions are supplemented with augmented requirements. For each system function that is classified as nonsafety-related with augmented requirements functionality, the basis is documented and the associated augmented design requirements are identified. Multiple nonsafety-related with augmented requirements functions may be applied to a single system function.

The system function categorizations are reviewed by the design reliability assurance program (D-RAP) expert panel. The expert panel is a select team of experts with collective experience in safety analysis, licensing, PRA, design engineering, and operations and maintenance processes. The expert panel reviews the recommendation for categorization of system functions by the presenting subject matter expert. The licensing representative is responsible for ensuring licensing commitments are verified as part of panel deliberations. The expert panel reviews and provides input on the creation of and modification to the nonsafety-related with augmented requirements functions.

In the preliminary design phase, system design is progressed to component design or selection. Failure modes and effects analysis (FMEAs) are conducted on select systems to evaluate system reliability and search for failure modes not previously considered in the conceptual design phase. Similar to the conceptual design phase, preliminary design is iterative, recursive, and many design activities progress in parallel that are not shown in Figure 3-1.

At the end of the preliminary design phase, SSC are classified regarding the functions provided in the licensing basis. These classifications are (1) safety-related, (2) non-safety-related, and (3) nonsafety-related with augmented requirements (i.e., special treatment).

As defined in 10 CFR 50.2, safety-related SSC are those that are relied upon to remain functional during and following design-basis events to assure:

1. The integrity of the reactor coolant pressure boundary
2. The capability to shut down the reactor and maintain it in a safe shutdown condition; or
3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11, as applicable.

Safety classification of SSC is based on the functions that the items are credited to perform in the safety analyses of design-basis events and not on design criteria or operations requirements. Some items providing only nonsafety-related functions might be subject to operations requirements such as technical specifications, emergency operating procedures, or surveillance procedures. Operations requirements, however, may not be the basis for functionally classifying an item as safety-related.

The components (including component-level structures) and parts (including consumables) are classified by considering the function of the components or parts in the safety-related systems and the safety-related functions that have been identified in the systems.

General criteria for the classification of components and parts are listed below:

- The classification of a component or part shall be based upon its function. If the component or part is relied upon for the satisfactory performance of a safety-related function, the component or part shall be classified as safety-related.
- A component or part that is relied upon in an analysis of a design-basis event to provide a safety-related function shall be classified as safety-related.
- A component or part that is relied upon to satisfy a safety-related interface requirement shall be classified as safety-related.
- A component or part that is relied upon to initiate, control, or maintain a safety-related function shall be classified as safety-related.
- A component or part that is relied upon during or following a design-basis event to maintain an environment to ensure that a safety-related item can perform its safety-related function shall be classified as safety-related.

- A component or part that is relied upon to prevent the release of radioactive material that could result in potential off-site exposures comparable to 10 CFR 100, Subparts A and B guideline exposures, during or following a design-basis event, shall be classified as safety-related.
- If the function of a component or part is not safety-related, and its failure could not prevent the satisfactory performance of a safety-related function, the component or part shall be classified as nonsafety-related.

The above classification criteria applies to all plant components and parts.

Items that are not classified as safety-related per the criteria above are classified as nonsafety-related.

When evaluating failures of a component or part that does not perform or support a safety-related function, any potential failure of the component or part is deemed credible unless justified otherwise in accordance with an established basis (Reference 10.3). Such justification may be based upon the following:

1. The failure is not physically possible for the functional or environmental conditions that would exist during or following a DBE;
2. The component or part is qualified to resist the failure. For example, for a component or part that is Seismic Category I or II, failures due to a safe shutdown earthquake do not need to be assessed. For a component or part that is environmentally qualified, failures due to environmental conditions that it is qualified for do not need to be assessed;
3. A probability evaluation shows that the frequency of occurrence for the DBE sequence including the failure is less than 10⁻⁷/year or less than 10⁻⁶/year if when combined with reasonable qualitative arguments, the realistic frequency can be shown to be lower.; A documented basis exists for declaring the failure incredible (e.g., catastrophic reactor vessel failure).

The expert panel reviews the list of SSC that are risk-significant, nonsafety-related with augmented requirements, or require regulatory treatment of non-safety systems (RTNSS), once the equipment lists are compiled into the D-RAP summary report and D-RAP list. The chair of the expert panel is responsible for the complete and accurate documentation of the expert panel findings, with support from the D-RAP coordinator. The chair approves the D-RAP summary report that documents the expert panel results.

During the review of the D-RAP summary report, defense-in-depth (DID) adequacy is evaluated. The DID approach contributes specific requirements to reactor design and operations, as determined by the subject matter expert and reviewed by the D-RAP expert panel. These requirements are established based on considerations including PRA information, selection of LBEs, SSC safety classifications, and special treatments for SSC. The SSC are identified, documented, and classified into appropriate levels (or bins), and DID attributes assigned accordingly.

For each of the SSC presented to the expert panel in the D-RAP list, the functional and performance criteria are clearly identified.

The expert panel performs a DID adequacy evaluation towards the end of the SSC classification process that is documented in the D-RAP summary report. This evaluation ensures the D-RAP list reflects the concept that multiple layers of DID have been evaluated in response to internal and external initiating events; and, that the reliability and capability of SSC to perform their safety functions for the life of the plant for the specific events that an SSC participates in is assured through specific programmatic actions that result in application of conservative safety margins against performance targets and limits.

The expert panel considers the following DID fundamental objectives when conducting DID adequacy evaluations:

- prevent accidents or lessen the effects of damage if an accident or malfunction occurs
- provide multiple barriers (including functional layers of defense) against fission product releases
- provide redundancy in safety function performance
- ensure that safety is not wholly dependent on any single feature of the design or operation
- select engineered features preferentially over administrative controls
- ensure that the public is adequately protected on the intended site and that well-conceived, workable emergency plans surround the nuclear facility
- use risk analysis to improve engineering and operational decisions by identifying and taking advantage of opportunities to reduce risk or reduce unnecessary burdens in cost beneficial ways.

In meeting its objectives, the D-RAP expert panel:

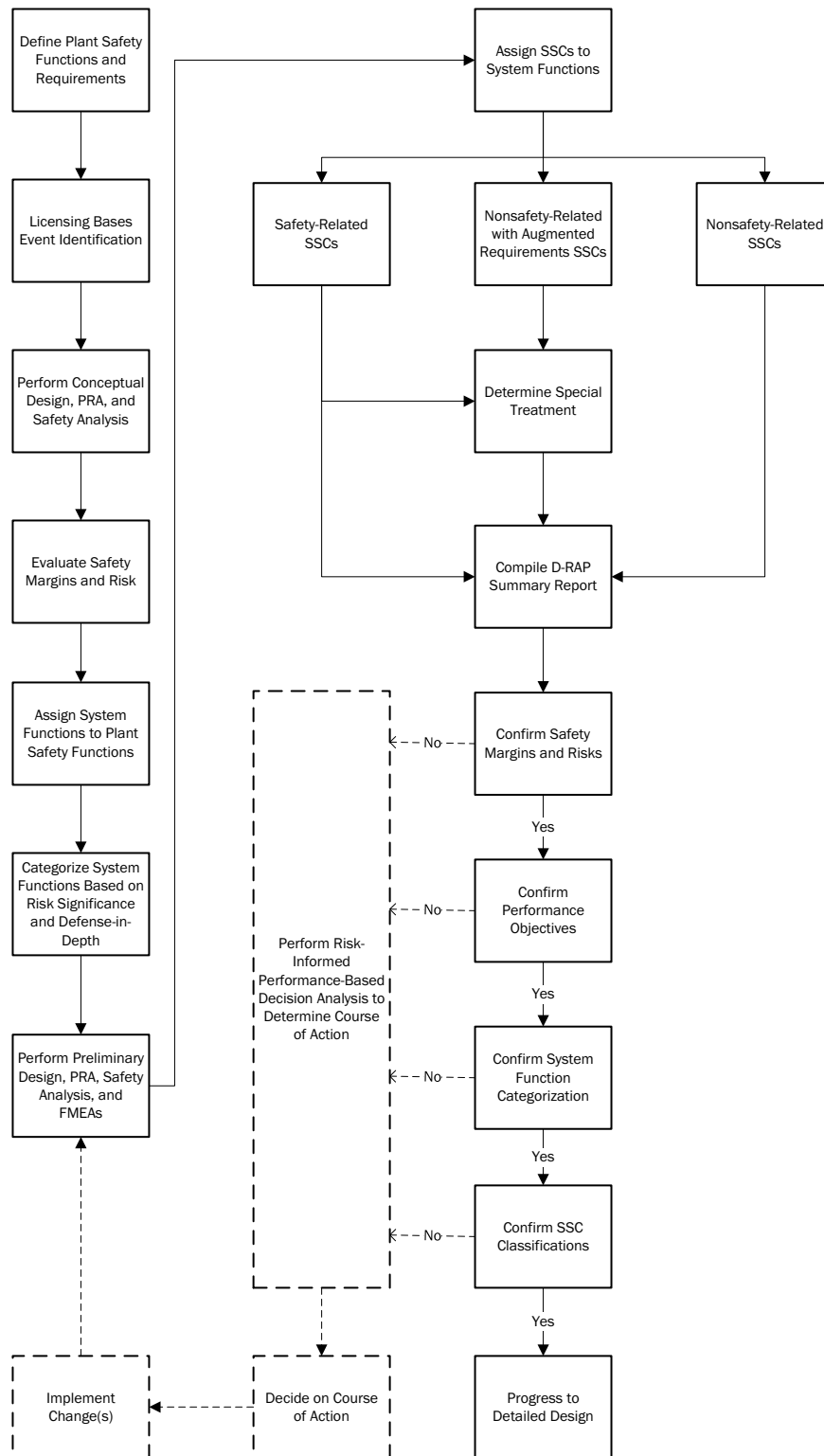
- confirms that a balance between event prevention and mitigation is reflected in the layers of defense for risk-significant LBEs.
- confirms whether any single feature is excessively relied on to achieve public safety objectives, and, if so, identify options to reduce or eliminate such dependency.
- confirms that adequate technical bases for classifying SSC exist and their capabilities to execute the safety functions are defined.
- confirms that the effectiveness of physical and functional barriers to retain radionuclides in preventing or limiting release is established.
- evaluates sources of uncertainty that need to be addressed via programmatic and plant capability DID measures have been adequately addressed.

The results of the expert panel DID adequacy evaluation are documented in the D-RAP summary report.

In some cases, DID adequacy may be challenged (e.g., limited margin) and as such, a decision analysis is conducted to evaluate the best course of action.

Decision making during expert panel deliberations is based on criteria specified at the highest level that is feasible to meet functional objectives. This process enables flexibility for realizing successful system performance.

Expert panel recommendations are provided to the NuScale engineering change control board to support effective and timely decisions that may include design, analysis, and/or methodology changes. If necessary, the design process is repeated until DID adequacy objectives are met.

Figure 3-1 System Design and SSC Classification Process

4.0 Event Identification and Classification for Safety Analyses

A review of NPM systems that could cause a reactor trip is used to determine the subset of possible events is within the scope of Chapter 15 design basis analyses based on regulatory guidance from the SRP, NuScale DSRS, and the specifics of the NuScale US460 standard design.

This evaluation leverages the experience gained from the NuScale US600 and changes to the NuScale design for US460 to document the scope of events relevant to the safety design-basis.

4.1 Basis for Event Classification

The SRP and DSRS discuss categorization of events according to frequency of occurrence.

For new applications, the SRP endorses two distinct categories:

- Anticipated Operational Occurrences (AOOs) are events that are expected to occur one or more times during the design life of the nuclear power plant. As identified in the SRP, among other acceptance criteria, fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 limit for PWRs.
- Postulated accidents are events that are not expected to occur during the design life of the plant as defined in the DSRS. As identified in the SRP, among other acceptance criteria for accidents, fuel is assumed to have failed if the minimum DNBR limit is not met but the release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100.

The DSRS Section 15.0 discusses AOOs and postulated accidents, and also recognizes 'infrequent events' as a subcategory of postulated accidents with more restrictive radiological acceptance criteria.

The SRP discussion of postulated accidents points to 10 CFR Part 100, but dose criteria used for new reactors are from 10 CFR 52.137.

Considering this evolution of the design-basis event categorization and consequential acceptance criteria, and the DSRS Section 15.0, the NPM design-basis events are categorized into one of three categories:

- AOOs - These are events that are expected to occur one or more times in the design life of the plant. For a 60-year plant design life, events with a frequency greater than $1/60$ or 1.67×10^{-2} per module-year could be considered AOOs. For conservatism, events that are expected to occur more than once every 100 years, 1×10^{-2} per module year, are considered AOOs.

- Infrequent Events - Infrequent events include events that are not expected to occur within the lifetime of a plant but have stricter radiological acceptance criteria than postulated accidents. Due to aspects of the NPM design, the expected frequency of occurrence is less than 1×10^{-2} per module year, and therefore these events are not expected to occur during the design life of the NPM operation. However, considering the historical treatment of these events as potentially occurring during plant operation, it is not considered appropriate to apply the radiological acceptance criteria appropriate for accidents.' Therefore, for infrequent events, some fuel damage may occur but the radiological acceptance criteria imposed are stricter than those imposed for accidents. Classification of these types of events as infrequent events would require specific evaluation of the event frequency for the NPM design.

Infrequent events also include some of the typical LWR events for which the NPM design is similar, and the specified radiological acceptance are stricter than imposed for accidents.

- Accidents - These are design-basis events with a probability of occurrence lower than 1×10^{-2} per module-year. For an event that is not expected to occur, classification as an AOO or an infrequent event is conservative because more restrictive acceptance criteria are applied.

Defining a lower bound frequency for accidents is considered. For defining the scope of design-basis events, historically, engineering judgment was used to determine what was insufficiently credible to be considered within the design-basis and was approved through the regulatory process. Therefore, events such as large secondary side system breaks, that may have a low probability of occurrence, generally remain part of operating plant design-bases. Considering that the NuScale plant systems are similar to operating PWR systems in many ways, NuScale has not attempted to define a lower bound frequency for accidents to define the limit between design-basis events and beyond-design-basis events. The Chapter 15 safety analysis is intended to identify a sufficiently broad spectrum of transients and accidents, or initiating events.

With respect to analyses presented in Chapter 15 of the FSAR, there are two additional categories of analysis presented in the US460 safety analysis, that are classified as 'special events':

- Some events such as Anticipated Transients without Scram (ATWS) are beyond-design-basis events but are required to be addressed per NRC regulations (i.e., assessment of a specific condition is required by the code of federal regulations).
- Typically, Chapter 15 events are analyzed from a starting point of steady-state hot conditions at the power that minimizes margin to the acceptance criterion of interest for the particular transient. With limited exceptions (for example, analysis of fuel handling accidents), the plant response to required operational occurrences such as startup and shutdown, power changes for load follow, or events that occur during the process of refueling are not typically presented in Chapter 15 of the FSAR. However, given the nature of the NuScale design as a natural circulation reactor and the refueling process, there are some design-specific aspects of normal operation such as reactor stability that is presented in the FSAR.

By exclusion, events that are not anticipated to occur during the course of normal operation bounded by identified design-basis events, are beyond-design-basis events.

4.2 Basis for Event Frequency

For classification of DBEs as AOOs, infrequent events, or accidents, the following approach is used:

- For events that are initiated by abnormal system conditions where the NPM system design is similar to that of operating plants, the event classification as AOO, infrequent event, or accident for the NPM is specified consistent with historical precedent. Radiological acceptance implicitly apply to infrequent events, which is supported by DSRS.
- For events that are specific to the NPM design, or where the NPM design is such that the event frequency is expected to be different than those of operating plants or other design certification applications, then the PRA event frequency for the NPM is considered. If it is available, the PRA event frequency is considered in classifying the event as an AOO, infrequent event or accident. If the event is conservatively classified as an AOO, the specific event frequency may not be determined.

This approach simplifies the event classification by using historical precedent where appropriate, and using NuScale PRA data to inform event classification where the NPM design is unique.

4.3 Event Identification by Chapter 15 Subsection

Table 4-1 provides a summary of the Chapter 15 DBEs and event classification applicable to the NPM.

Table 4-1 Listing of Design Basis Events by FSAR Chapter 15 Section

Event Class ⁽¹⁾	Event ⁽¹⁾	FSAR Section ⁽¹⁾	Classification
Increase in heat removal by secondary system	Decrease in feedwater temperature	15.1.1	AOO
	Increase in feedwater flow	15.1.2	AOO
	Increase in steam flow	15.1.3	AOO
	Inadvertent opening of steam generator relief or safety valve	15.1.4	AOO
	Steam system piping failures	15.1.5	Accident
	Containment flooding/loss of containment vacuum	15.1.6	AOO
Decrease in heat removal by secondary system	Loss of external load	15.2.1	AOO
	Turbine trip	15.2.2	AOO
	Loss of condenser vacuum	15.2.3	AOO
	Main steam isolation valve closure	15.2.4	AOO
	Loss of non-emergency AC power to station auxiliaries	15.2.6	AOO

Table 4-1 Listing of Design Basis Events by FSAR Chapter 15 Section (Continued)

Event Class⁽¹⁾	Event⁽¹⁾	FSAR Section⁽¹⁾	Classification
Decrease in heat removal by secondary system (continued)	Loss of normal feedwater flow (partial or complete)	15.2.7	AOO
	Feedwater system pipe breaks	15.2.8	Accident
	Inadvertent DHRS Operation	15.2.9	AOO
Reactivity and power distribution anomalies	Uncontrolled control rod assembly bank withdrawal from a subcritical or low power startup condition	15.4.1	AOO
	Uncontrolled control rod assembly bank withdrawal at power	15.4.2	AOO
	Control rod misoperation (system malfunction or operator error)	15.4.3	AOO
	Addition of cooler water to the reactor system	15.4.4 - 15.4.5	n/a
	Inadvertent decrease in boron concentration in the reactor coolant	15.4.6	AOO
	Inadvertent loading and operation of a fuel assembly in an improper position	15.4.7	Infrequent Event
	Spectrum of rod ejection accidents	15.4.8	Accident
Increase in RCS Inventory	Chemical and volume control system malfunction that increase RCS inventory	15.5.1	AOO
Decrease in RCS Inventory ⁽⁸⁾	Inadvertent opening of a reactor safety valve	15.6.1	AOO
	Failure of small lines carrying primary coolant outside containment	15.6.2	Infrequent Event
	Steam generator tube failure	15.6.3	Accident
	Loss of coolant accident resulting from spectrum of postulated piping breaks within reactor coolant pressure boundary	15.6.5	Accident
	Inadvertent operation of Emergency Core Cooling System	15.6.6	AOO
Other	Radiological consequences of fuel handling accidents	15.7.4	Accident
Other	Spent fuel cask and NPM drop accidents	15.7.5	Accident
Other - BDB	ATWS	15.8	Special event
Other	Demonstration of Stability	15.9	Special event

Notes:

- (1) The event class, event, and FSAR section are outlined based on NUREG-0800 and RG 1.206 Section C.I.15, and the NuScale US600 design certification application. Some event classes/events from the SRP are not listed because they are BWR-specific (do not apply to NuScale), or do not apply by nature of the NuScale design (for example, Section 15.3 loss of flow transients). Sections 15.0.3, 15.4.8A, and 15.6.5A are specific to addressing radiological acceptance criteria and are not listed above because these do not represent different initiating events.

4.3.1 Special Consideration for Decrease in RCS Inventory Events

For consideration of LOCA events, the NuScale US460 design differs significantly from operating PWRs in three areas:

1. There are no large pipes connected to the reactor pressure vessel and therefore the traditional large break LOCA is precluded by the design.
2. There are no safety-related systems to add coolant to the reactor vessel. The only means of adding additional water to the RPV is through nonsafety-related CVCS normal makeup.
3. ECCS actuation for the NuScale design results in rapid release of coolant from the primary side, creating a response similar to a pipe break LOCA inside of containment.

Given these significant design differences, the treatment of decrease of RCS inventory events are discussed in Table 4-2.

Table 4-2 Events Involving Decrease of RCS Inventory

Event	Event Category	Basis/Discussion
RCS Leakage outside of containment within the capacity of normal (non-safety) makeup system	n/a	These leakage events are not LOCAs by the definition from 10 CFR 50.46 and 10 CFR 50 Appendix A. 10 CFR Appendix A includes GDC for emergency cooling separate from those regarding makeup for anticipated leakage. In the NuScale design, normal makeup is provided by CVCS; however normal makeup is not automatically initiated and must be authorized by operators. Since makeup is not automatic, various event progressions are considered. For leakage from a pipe outside of containment, whether or not makeup is authorized by the operator affects the volume of primary system fluid that leaks from the RCS prior to containment isolation, which terminates the leakage. This scenario is addressed by the radiological assessment of small pipe breaks outside of containment. If makeup is not authorized and no other action is taken to shut down the module, eventually a low or low-low pressurizer level signal is expected, that isolates containment.

Table 4-2 Events Involving Decrease of RCS Inventory (Continued)

Event	Event Category	Basis/Discussion
RCS Leakage inside of containment within the capacity of normal (non-safety) makeup system	Special Event	These leakage events are not LOCAs by the definition from 10 CFR 50 Appendix A. In the NuScale design, normal makeup is provided by CVCS; however normal makeup is not automatically initiated and must be authorized by operators. Since makeup is not automatic, various event progressions are considered. For leakage inside of containment, if the leakage is within the capacity of the containment evacuation system or the leakage condenses quick enough, the containment pressure does not increase. Then, similar to a leak outside of containment, if makeup is not authorized and no other action is taken to shut down the module, eventually a low or low-low pressurizer level signal is expected, which terminates the leakage. A low pressurizer level signal actuates DHRS. If the leakage exceeds the capacity of the containment evacuation system, then a high containment pressure signal is expected that terminates the leakage and actuate DHRS, consistent with the progression expected in other initiating events resulting in a decrease in inventory from the RCS into containment.
Spurious ECCS signal	AOO	A spurious ECCS signal could signal for opening of all four ECCS valves, or one division of ECCS (1 RRV and 1 RVV). The RRV design includes an inadvertent actuation block feature to prevent opening until the RCS is depressurized to below a set pressure. The RVV does not include an IAB in the US460 design. An inadvertent ECCS signal for 1 ECCS division would result in the full opening of 1 RVV, while a failure causing a signal for all valves would result in simultaneous opening of both RVVs. The IABs on the RRVs would initially block the RRV opening, delaying the RRV opening until after the threshold differential pressure is reached due to RPV discharge into containment through the opened vent valve(s). Although a spurious ECCS signal leading to ECCS valve opening while the module is at power operation is not expected to occur in the life of the plant, the event is conservatively categorized as an AOO and analyzed against AOO acceptance criteria.

Table 4-2 Events Involving Decrease of RCS Inventory (Continued)

Event	Event Category	Basis/Discussion
Spurious opening of single ECCS valve	AOO	A mechanical failure or failure of both solenoids on one valve could result in the opening of a single ECCS valve. The valve opening is considered the initiating event (i.e. the valve opening is not the result of an initiating event plus a coincident single failure). Although an ECCS valve is not expected to spuriously open during the life of a module, the event is conservatively categorized as an AOO and analyzed against the AOO acceptance criteria.
Spurious opening of multiple ECCS valves due to mechanical failure	Beyond Design Basis	Simultaneous mechanical failures on multiple ECCS valves is beyond-design-basis with respect to identifying initiating events. A spurious ECCS signal that could result in the opening of multiple RVVs is categorized as a separate event, per discussion of spurious ECCS signal.
Spurious opening of one reactor safety valve	AOO	The reactor safety valves provide over-pressure protection of the reactor pressure vessel. Mechanical failure associated with the valve internals could result in spurious opening of a reactor safety valve. Consistent with the categorization of the spurious opening of one ECCS valve, this event is conservatively categorized as an AOO and analyzed against AOO acceptance criteria.
Break of small pipes connected to reactor vessel inside containment	Accident	A pipe break is a mechanical failure that is not expected to occur during the design life of a reactor module. This event is classified as an accident.
Break of small pipes connected to reactor vessel outside containment (isolatable)	Accident	A pipe break is a mechanical failure that is not expected to occur during the design life of a reactor module. Consistent with the classification of the break of small pipes connected to the reactor vessel inside of containment, this event is categorized as an accident. It is noted that the radiological consequences for breaks of small pipes outside of containment are of particular relevance and the rate of inventory loss relative to any supplied makeup is important with respect to when, or if, reactor trip and containment isolation are actuated.

Table 4-2 Events Involving Decrease of RCS Inventory (Continued)

Event	Event Category	Basis/Discussion
Break of small pipes connected to reactor vessel outside containment (unisolable)	N/A	Consistent with current plant designs, in the NuScale design this event is mitigated by isolating the break via redundant isolation valves in series for piping connected to the RCS. Therefore, any unisolable break of RCS piping outside of containment is beyond-design-basis (because multiple failures of safety-related isolation valves are necessary following the initiating event). With respect to small pipes or lines connected to the reactor vessel and running outside containment, the following are specifically considered: • In the current design there are no RCS fluid-sensing instrument lines that carry RCS fluid outside of containment pressure boundary • The ECCS valve design includes a hydraulic line inside containment from the main valve on the reactor coolant pressure boundary to a trip valve and reset valve. The trip valve and reset valve are mounted on the exterior of the CNV and are part of the containment pressure boundary. As part of the containment pressure boundary, a gross failure of the trip valve or reset valve that results in an unisolable bypass of RCS fluid to outside the containment pressure boundary is beyond design basis.
Steam generator tube failure	Accident	A steam generator tube failure is classified as an accident. Steam generator tube ruptures are not typically considered AOOs in operating plants (it is noted that the NuScale DSRS identifies that steam generator tube leaks are an example of an AOO, but steam generator tube ruptures are not listed as an example postulated accident). NuScale DSRS Section 15.0.3 lists steam generator tube failure in Table 1 “SMR Accident Dose Criteria”. Therefore, it is appropriate to classify the event as an accident, although it is recognized that the radiological analysis of this event considers various assumptions and different acceptance criteria are applicable depending on those assumptions.

Table 4-2 Events Involving Decrease of RCS Inventory (Continued)

Event	Event Category	Basis/Discussion
Potential LOCA resulting from control rod ejection	N/A	A rod ejection accident (REA) has historically been postulated as the failure of a control rod drive mechanism pressure housing that causes the ejection of a control rod from the reactor. Some plant safety analysis reports identify that the loss of coolant effects for a control rod ejection accident are bounded by the spectrum of LOCA break locations. For the NuScale design the control rod ejection accident is analyzed as a limiting reactivity insertion event. The inventory decrease and associated RCS pressure decrease from a postulated control rod housing failure are insignificant with respect to: • demonstrating that acceptance criteria are met (for the fuel enthalpy increase criterion), • conservatively bounded with respect to demonstrating that acceptance criteria are met (for the radiological acceptance criterion), or • it is conservative to neglect the depressurization effect (for determining maximum RCS pressure and assessing the maximum power response). It is noted that there is precedent in design certification applications that the REA is a rapid positive reactivity insertion analyzed independent of determining a specific failure mechanism.
Gross rupture of reactor vessel	Beyond design basis	Consistent with operating plants, gross rupture of the reactor pressure vessel is beyond design basis.

5.0 Identification and Quantification of Initiating Events

5.1 Introduction

The initiating event analysis identifies plant upsets originating within the plant boundary that disrupt the at-power operations of an NPM. These initiators include random equipment malfunctions and operator errors that either cause an automatic reactor trip or demand a manual trip. Hazards such as floods, fires, earthquakes, and high winds are addressed in separate analyses; upsets unique to low power and shutdown operations are also assessed separately. Accidents arising from acts of sabotage or security threats are beyond the scope of this evaluation.

The quantification of initiating event frequencies typically makes use of Bayesian estimation methods. This statistical inference methodology employs generic prior data and plant-specific data to produce a posterior distribution of an event frequency using Bayes' Theorem. However, since NuScale does not have plant-specific operating experience to draw from, initiating event frequencies are estimated based solely on the generic prior data collected by the NRC through licensee event reports from the U.S. nuclear industry.

In accordance with the ASME/ANS PRA Standard (Reference 10.4) and Regulatory Guide 1.200, the technical characteristics of the initiating event analysis include:

- sufficiently detailed identification and characterization of initiating events
- grouping of events according to plant response and mitigating requirements
- proper screening of individual or grouped initiating events

5.2 Methodology

The key steps in the initiating event analysis are:

1. Identification of Plant Faults

A listing of plant faults that have the potential for interrupting normal NPM operation is populated through a variety of sources including a literature search of industry data, a master logic diagram (MLD), and failure modes and effects analyses (FMEAs) of plant systems.

2. Screening and Categorization of Plant Faults

The plant faults are evaluated for their applicability to the NuScale design. The basis for screening out plant faults is described in Section 6.2.2. The remaining plant faults are categorized and grouped.

3. Grouping and Quantification of Initiating Events

The categorized plant faults are grouped into initiating events according to plant response. An initiating event frequency is estimated by applying industry data or performing NuScale-specific analyses.

5.2.1 Identification of Plant Faults

The initial identification of plant faults for the at-power, internal events PRA includes application of FMEAs and development of an MLD.

Additional data are also reviewed in order to develop a comprehensive list of initiating events for the NuScale PRA:

- NuScale design-basis events
- industry (generic) data
- PRA studies from operating plants
- advanced reactor designs

5.2.2 Screening and Categorization of Plant Faults

The plant faults identified and tabulated are reviewed for applicability to the NuScale design and screened out from further consideration as appropriate. The plant faults that have been screened out use the following criteria:

- Fires and floods as potential initiators are not within the scope of the at-power, internal events PRA because they are addressed in separate analyses.
- Certain plant faults that are relevant to traditional LWR designs that are not applicable to the NuScale design. For example, reactor coolant pump seal loss of coolant accidents (LOCAs) are not applicable to a design without reactor coolant pumps. Similarly, pressurizer power operated relief valve (PORV) opening is excluded because the NuScale design does not include pressurizer PORVs. In addition, interfacing systems loss of coolant accident (ISLOCAs) is screened, as the NuScale design is not susceptible to traditional ISLOCAs.

The plant faults that remain following the screening process are categorized based on plant response, success criteria, timing, potential for radionuclide release, and the effects on the operability and performance of mitigating systems and plant operators. The categories defined for the initiating events are:

- pipe breaks and LOCAs
- steam generator tube failures
- secondary side line breaks
- losses of electric power
- transients

5.2.3 Grouping and Quantification of Initiating Events

Each initiating event category is represented by one or more initiating events. The initiating events themselves are groupings of events that demand a reactor trip or controlled shutdown and lead to a common plant response. The events comprising an initiating event are grouped in order to facilitate the delineation of accident sequences in the PRA event trees. For example, the secondary side line break initiating event is a grouping of pipe breaks/leaks in the main steam, feedwater, and decay heat removal lines because the response of the plant to these occurrences can be assessed similarly in the accident sequence analysis. The events comprising an initiating event are also bounded by the worst case impact within the group. For example, the general reactor trip includes several initiators and the accident progression considers a reactor safety valve (RSV) demand because some initiators result in immediate isolation of the feedwater system.

Quantification of the initiating events is performed by a combination of the following techniques:

- applying relevant operating experience from U.S. nuclear power plants
- use of NuScale-specific studies

Typical practice for PRAs across the nuclear industry uses Bayesian analysis methods to estimate occurrence frequencies. The Bayesian methodology consists of using industry-wide generic data and subjective experience to assign a prior distribution to a parameter, such as failure rate. The second part of Bayesian estimation is in applying plant-specific data to transform the prior distribution into a posterior distribution according to Bayes' Theorem. Bayes' Theorem provides a statistical framework for processing new information as it becomes available over time. In the NuScale design, plant-specific data do not currently exist. Therefore, Bayes' Theorem, in its entirety, cannot be used. The initiating event frequencies have been quantified based solely on generic prior information.

Initiating event frequencies in the NuScale PRA are defined in terms of occurrences per module critical year (mcy). As described in supporting requirement IE-C5 and note 1 to Table 2-2.1.4(c) in ASME/ANS RA-Sa-2009 (Reference 10.5), initiating event frequencies are to be calculated on a reactor year basis. Included in the initiating event analysis is the plant availability, which is assumed to be 100 percent.

The calculated initiating event frequencies for the NuScale power module are summarized in Table 5-1.

Table 5-1 Initiating Event Frequencies

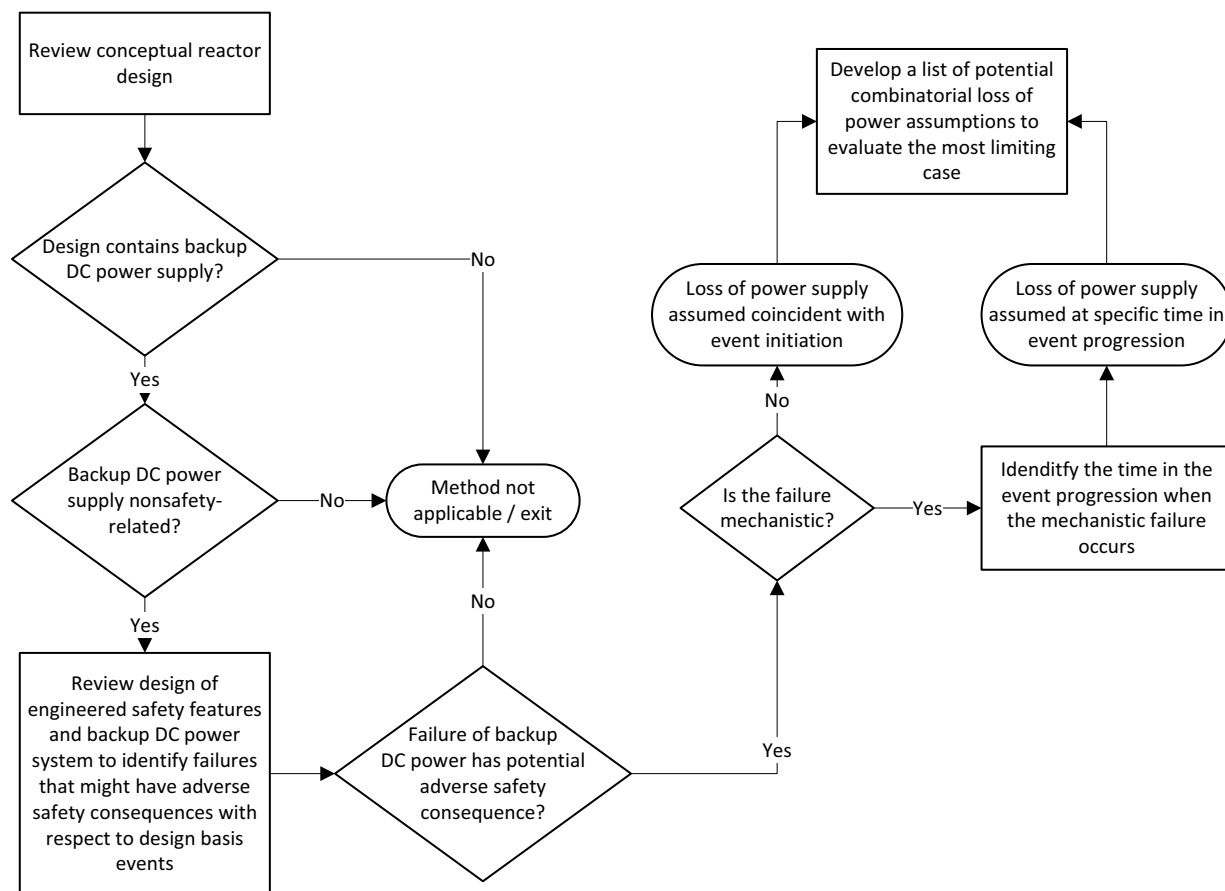
Category	Initiator	Mean (per mcyr)
Pipe breaks and loss of coolant accidents	CVCS injection line pipe break outside containment	1.7E-05
	CVCS discharge line pipe break outside containment	2.5E-06
	CVCS injection line LOCA inside containment	4.1E-04
	LOCA inside containment	1.3E-03
	Spurious opening of an ECCS valve	7.2E-04
Steam generator tube failures	Steam generator tube failure	4.6E-05
Secondary side line breaks	Secondary side line break	4.4E-05
Loss of electric power	Loss of offsite power - EHVS	2.5E-02
	Loss of DC power - EDAS	2.6E-04
Transients	General reactor trip	5.8E-01
	Loss of support system	5.2E-03

6.0 Implementation of the NuScale Design Process for DC Power Systems

The purpose of this section is to provide the application of the design evaluation process described in Section 3.0. The evaluation is provided for the NuScale Power Module (NPM) US460 as described in Section 2.0. Additional system design details are described as needed to provide context for the evaluation and conclusion.

The DC power systems are first classified based on their functional design purpose to provide nonsafety-related back-up power and then evaluated against the failure credibility criteria and safety-related classification guidance listed in Section 3.0.

Figure 6-1 Treatment of Nonsafety-Related DC Power System Failure Methodology



6.1 Identification of Nonsafety-Related DC Power Systems

As described in Section 2.4 and Section 2.5, the NPM design has two nonsafety-related DC power supply systems, the EDAS with augmented requirements and the EDNS. Each of these systems is considered in the design evaluation.

6.2 Review of Nonsafety-Related DC Power System Functions and Loads

The focus of the review is to identify system functions and loads in order to characterize the interface of each system with the range of design-basis event progressions. The interface is characterized in terms of failures of the system functions and impact on the reactor module thermal-hydraulic response.

6.2.1 Normal DC Power System (EDNS)

As described in Section 2.5, a principal EDNS system function is to supply power to required loads. Therefore, during a design-basis event, if the EDNS is operating as designed and performing this function, this supports:

- AC power systems operating as designed
- Module control system operating as designed
- Plant control system operating as designed

Failures of the EDNS to perform the function of supplying power to each of these loads is evaluated to characterize the interface with the design-basis event progression.

6.2.1.1 Failure to Provide DC Control Power to AC Power Distribution Systems

The principal function of each AC power distribution system is to provide electrical power at the required voltage or current to its specified loads. Therefore, should the EDNS fail to provide control power to AC power distribution systems, the AC power distribution systems may not provide power at the required voltage or current to their specified loads. Ultimately, this failure affects power supply to components that directly affect normal reactor module operation, such as the feedwater pumps, chemical volume control pumps, and pressurizer heaters.

Failure of the normal AC power supply is a safety analysis design-basis initiating event; loss of normal AC power supply is considered in conjunction with design-basis initiating events and due to postulated mechanistic failures during the event progression. Therefore, failure of the EDNS to provide control power to AC power distribution systems is encompassed within the scope of existing design-basis safety analyses.

6.2.1.2 Failure to Provide Power to Module Control System

In the NPM design, the module control system (MCS) provides module-specific normal control functions during power operation, including:

- Pressurizer pressure control, by combination of pressurizer spray and heater control
- Pressurizer water level control, by operation of CVCS makeup pumps and letdown control valve
- Core average coolant temperature control, by combination of secondary side steam demand and control rod movement
- Secondary steam pressure control, by combination of turbine throttle and bypass valves
- Turbine load control, by combination of steam pressure control and feedwater flow rate
- Containment pressure control, by operation of the containment evacuation system

Should the EDNS fail to provide power to the MCS, the module control system does not perform as designed during the design-basis event progression. For example, if pressurizer pressure increases, the MCS would be unable to increase the pressurizer spray flow to attempt to control pressure; the pressurizer heater power could remain constant rather than decreasing, or the pressurizer heater power could cease due to loss of control signal and/or power supply.

Normal plant control system response is evaluated as part of the design-basis safety analyses. If as-designed operation of the normal, nonsafety-related plant control system(s) improves the design-basis event progression, then normal operation is not credited, by assuming that the MCS does not respond to changes in the sensed parameter control input (e.g., pressure, temperature, flow rate, control rod position). Therefore, this type of failure is encompassed within the scope of existing design-basis safety analyses.

Failure of the EDNS to supply power to the MCS could also result in failure of the controlled component output (e.g., pressurizer heaters cease to emit energy, or containment evacuation pumps cease to operate). This type of failure could also be caused by failure of AC power supply. As discussed in Section 6.2.1.1, failure of normal AC power supply is a safety analysis design-basis initiating event and loss of normal AC power supply is considered in conjunction with design-basis initiating events. Therefore, this type of failure is encompassed within the scope of existing design-basis safety analyses.

Based on this evaluation, failure of the EDNS to provide power to the module control system is encompassed within the scope of existing design-basis safety analyses.

6.2.1.3 Failure to Provide Power to Plant Control System

In the NPM design, the plant control system (PCS) provides monitoring and control function for systems with equipment and instrumentation common to all modules, such as:

- Auxiliary boiler system
- Air cooled condenser systems
- Backup power supply system
- Reactor component cooling water system
- Fixed area radiation monitoring systems
- Safety display and indication system
- Neutron monitoring system for refueling
- Module assembly equipment bolting
- Remote recovery platform
- Reactor building crane, turbine building crane
- Boron addition system
- Containment flood and drain system
- Module heatup system
- Integrated control system (for refueling and remote handling)
- Control room habitability system
- Normal control room, reactor building, turbine building, radioactive waste building HVAC systems
- Gaseous, liquid, solid radioactive waste systems
- Radioactive waste drain system, balance of plant drains system
- Demineralized water system
- Chilled water, utility water, site cooling water, potable water systems
- Nitrogen distribution system
- Process sampling system
- Ultimate heat sink
- Turbine lube oil storage
- Instrument and control / service air system
- Seismic monitoring system
- Plant-wide video monitoring system
- Pool cooling and cleanup system

- Meteorological environmental monitoring system
- Fire protection system
- Diesel fuel transfer system
- Reactor building components
- Grounding and lighting protection system
- Equipment that is common to all modules, although it belongs to a system whose components and equipment are predominantly module-specific, such as:
 - Chemical volume control system isolation valves at the interface with the module heatup system
 - Equipment and instrumentation for the condensate polishing resin regeneration in the condensate and feedwater system
 - Common equipment in augmented DC power system, normal DC power system
 - Common equipment in low, medium, high voltage AC electrical distribution systems

Should the EDNS fail to provide power to the PCS, the plant control system, does not perform as designed during the design-basis event progression.

The PCS provides monitoring and control functions of many shared plant systems that do not affect the ability to control reactivity, reactor coolant inventory, reactor coolant pressure or temperature during normal power operation. For example, the radioactive waste systems, seismic monitoring system, or equipment associated with refueling operations do not affect design-basis event progressions.

The PCS provides monitoring and control functions of the reactor building HVAC system, which supports a range of initial conditions for component environments that are integrated with appropriate equipment qualification programs. The safety analyses rely on the equipment qualification program to demonstrate that the equipment within the program is capable of performing its required functions in the specified environmental conditions for the required post-accident durations. Therefore, failures associated with the power supply to the PCS are not evaluated in further detail with respect to the equipment environmental conditions.

The PCS provides monitoring and control functions of shared systems which are used to establish a range of plant initial conditions that are assumed in the design-basis safety analyses. For example, the pool cooling and cleanup system maintains the ultimate heat sink temperature initial condition within bounds analyzed in the safety analyses. The safety analyses subsequently address a range of ultimate heat sink temperatures appropriate to the design-basis event duration, where PCS may be operating and maintaining the ultimate heat sink

temperature within normal bounds or PCS may not be operating due to loss of power supply and the pool temperature may rise over the long-term event progression.

Failures of PCS control of other systems are encompassed in the design-basis safety analyses as initiating events. For example, failures affecting the boron addition system or the demineralized water system could result in an inadvertent boron dilution event and reactivity insertion.

Based on this evaluation, failure of the EDNS to provide power to the plant control system is encompassed within the scope of existing design-basis safety analyses in terms of the initiating events, and the range of initial and boundary conditions analyzed.

6.2.1.4 Summary - EDNS

The system-level functions of the EDNS and the loads associated with the functions are reviewed. The failure of the EDNS to perform these functions, essentially failure to provide power to the identified loads, is considered. Failure of the EDNS to perform the identified functions is encompassed within the scope of the existing design-basis safety analyses, considering the range of initiating events and event progressions (i.e., AC power availability and normal control system operation). Therefore, failures of the EDNS do not introduce a new or different challenge to be addressed in the scope of the design-basis safety analyses.

6.2.2 Augmented DC Power System (EDAS)

As described in Section 2.4, EDAS functions operationally to:

- Provide electrical power for the prevention of unintended ECCS actuation.
- Provide backup electrical power for Post-Accident Monitoring (PAM) instrumentation for type B, C, D, and F variables.
- Provide module-specific and common DC electrical power with adequate voltage and current to EDAS loads when AC power is available to EDAS.
- Provide module-specific and common DC electrical power with adequate voltage, current, and capacity to specified EDAS module-specific and common loads not requiring augmented requirements DC power when AC power is not available to EDAS.

Failures of the EDAS to perform the function of supplying power to required loads is evaluated to characterize the interface with the design-basis event progression. These functions are evaluated separately, which is appropriate for evaluation at the function level and does not require knowledge of whether components and equipment serve different or multiple functions (e.g., equipment with augmented requirements for required loads vs. equipment without augmented requirements for other loads). EDAS

is not relied upon in the safety analyses of DBEs to provide or support plant-level safety-related functions.

6.2.2.1 Failure to Provide Power to Maintain ECCS Valves Closed

The EDAS provides electrical power through the module protection system (MPS) DC-DC converter Class 1E isolation device to prevent unintended ECCS actuation. Power is then routed through the MPS to the ECCS solenoid trip valves.

Failure of the EDAS power supply to the MPS results in reactor trip, containment isolation, DHRS actuation, and ECCS actuation. As described in Section 2.3.1, ECCS actuation results in the RVVs opening and rapid depressurization of the RPV into containment. If the RPV and CNV are initially at normal pressure conditions, opening of the RRVs is initially blocked by the IABs; the RRVs subsequently open after the IAB releases.

ECCS actuation results in a rapid depressurization of the RPV, pressurization of the containment, and redistribution of reactor coolant between the RPV and the containment vessels. The design-basis safety analyses address inadvertent ECCS operation as an initiating event:

- One ECCS valve inadvertently opens (postulated due to mechanical failure)
- Inadvertent ECCS actuation signal, resulting in both vent valves opening simultaneously

The effect of loss of EDAS power to the MPS and resulting ECCS actuation must also be considered in the context of other design-basis initiating events. For example, a postulated pipe break LOCA concurrent with loss of EDAS power supply would result in break flow rates from the RPV to CNV in excess of flow rates resulting from either the LOCA or inadvertent ECCS actuation alone. A design-basis event that causes reactivity insertion can bring the RPV power, pressure, and temperature near the limits for MPS actuations, outside the range of initial conditions assumed for the design-basis event analyses. Therefore, failure of the EDAS function to provide power to maintain the ECCS valves closed is considered in more detail to define the appropriate treatment in the safety analysis scope.

The frequency of ECCS actuation following an AOO is determined by first identifying those AOOs that require ECCS actuation for mitigation and summing their frequencies.

The AOOs that require ECCS are those that result in the relocation of reactor coolant from the reactor pressure vessel to the containment vessel, or otherwise actuate ECCS as part of the mitigation strategy. Of the events classified as AOOs, only the inadvertent opening of an RSV or an ECCS valve relocates coolant to containment.

Then, other causes of ECCS actuation are evaluated. Of the ECCS actuation causes considered, the loss of DC power probabilistic risk assessment (PRA) initiating event is included in this frequency analysis. This initiating event involves the common cause failure of two or more DC buses and is therefore not an AOO. However, it is included as it would be a direct cause of ECCS actuation. Considering this frequency is analogous to safety analyses of AOOs that deterministically assume DC power is not available.

The inadvertent RSV opening is a contributor to the loss of coolant accident (LOCA) inside containment, and only that portion of the frequency from the PRA initiating event is used; the spurious ECCS valve opening and loss of DC power are stand-alone initiating events and their frequencies are used directly.

The total frequency of ECCS actuation following an AOO is shown in Table 6-1.

Table 6-1 AOOs resulting in ECCS actuations

Event	Frequency (mcyr)
Spurious RSV opening	1.1E-03
Spurious ECCS valve opening	7.2E-04
Loss of DC power	2.6E-04
Total	2.1E-03

The results in Table 6-1 show that the frequency of ECCS actuation following such events is approximately 2.1E-03/mcyr (module critical year), or once in 476 years. It can therefore be reasonably concluded that ECCS actuation following an AOO is not expected in the 60-year design lifetime of an NPM.

6.2.2.2 Failure to Provide Power for Post-Accident Monitoring

The EDAS provides electrical power for post-accident monitoring (PAM) by providing electrical power through the module protection system (MPS) DC-DC converter Class 1E isolation device. Power is then routed through the MPS to other equipment and components to provide information for PAM variables to the safety display and indication system displays in the main control room. After a design-basis event, the NPM provides automated actuations that place and maintain the NPM passively cooled and subcritical for at least 72 hours. There are no Type A PAM variables in the NPM design. No operators actions, based on PAM indications for Type A variables, are required in response to a design-basis event to accomplish safety-related functions.

Therefore, postulated EDAS failure to provide electrical power for PAM does not affect the ability to control reactivity, reactor coolant inventory, reactor coolant pressure, or temperature during normal power operation.

6.2.2.3 Failure to Provide Power for Other Loads not Requiring Augmented DC Power Supply

The EDAS provides power for other module-specific and common loads that do not require augmented DC power supply. Loads powered from the EDAS differ depending on whether AC power supply is available or not available to the EDAS.

When AC power supply is available to the EDAS, module-specific loads include:

- Module protection system
- Neutron monitoring system
- Radiation monitoring systems

When AC power supply is available to the EDAS, common loads include:

- Safety display indication system
- Plant protection system
- Plant lighting system for control room normal and emergency lighting

When AC power supply is not available to the EDAS, module-specific and common loads include:

- EDAS battery monitoring system
- Main control room lighting

The neutron monitoring system provides analog neutron flux data to the MPS and the MPS continuously monitors for fault signals from the NMS. If EDAS power supply for the neutron monitoring system fails, a high voltage power supply fault is generated, detected by MPS, and then reactor trip is actuated. This scenario is encompassed within the safety analysis.

The radiation monitoring does not affect the control of reactivity, reactor coolant inventory, reactor coolant pressure or temperature. Therefore failures of EDAS to perform this function are not considered further.

The EDAS supplies power to common loads including the safety display indication system and plant lighting system for control room normal and emergency lighting. These systems do not affect the control of reactivity, reactor coolant inventory, reactor coolant pressure or temperature. Therefore, failures of EDAS to perform these functions are not considered further.

The EDAS supplies power to the plant protection system (PPS). The PPS does not perform safety-related functions. Plant protection system functions include providing control room habitability actuation signals; providing plant information for PAM variables, providing data to operators for control and indication as part of power generation; providing plant information for spent fuel pool level; providing power to sensors as required. These functions do not affect the reactor module.

Therefore, EDAS failure to provide power to the PPS does not affect the control of reactivity, reactor coolant inventory, reactor coolant pressure or temperature. Therefore failures of EDAS to perform this function are not considered further.

When AC power supply is not available to EDAS, EDAS loads that do not require augmented DC power supply include power for the EDAS battery monitoring system and main control room lighting. The EDAS failure to provide power to these loads does not affect the control of reactivity, reactor coolant inventory, reactor coolant pressure or temperature.

6.2.2.4 Summary - EDAS

The system-level functions of the EDAS and the loads associated with the functions are reviewed. The failure of the EDAS to perform these functions, essentially failure to provide power to the identified loads, is considered. In some cases, such as failure to provide power for PAM, the control of reactivity, reactor coolant inventory, reactor coolant pressure or temperature are unaffected. Failure of the EDAS to provide power to the neutron monitoring system is encompassed within the scope of the design-basis safety analyses. Failure of the EDAS to provide power to the MPS results in reactor trip, DHRS actuation, containment isolation, and ECCS actuation. Spurious reactor trip, DHRS actuation, containment isolation are encompassed within the scope of design-basis safety analysis.

Failure of the EDAS function to provide power through MPS to maintain the ECCS valves closed is considered in more detail in Section 6.3 to define the appropriate treatment in the safety analysis scope.

6.3 Review of EDAS Design to Identify Mechanistic Failures

The EDAS system design description and system failure modes and effects analysis (FMEA) are the next layers of design detail considered to identify failures that could result in failure to supply power through MPS to maintain ECCS valves closed when not actuated during a design-basis event progression.

The EDAS system design description clarifies that there are different equipment and components for module-specific loads compared to common loads. The failure of the EDAS to supply power to a module is the appropriate failure mode to consider for the design-basis safety analyses. In other words, failure to supply power to maintain the ECCS valves closed is considered as part of failure to supply power to the MPS rather than considering these separately.

Key observations from review of the EDAS FMEA:

- There are no single failures that could cause loss of power supply from EDAS
 - Loss of a single module-specific power channel does not result in loss of power supply, reactor trip, inadvertent ECCS actuation, or loss of PAM monitoring capacity.
 - Failure of any two module-specific power channels results in reactor trip and ECCS actuation.
- EDAS is designed with independent and diverse monitoring devices such that no single monitoring device failure goes undetected and prevents detection of the failure mechanisms evaluated in the FMEA.
- Failures that could affect power supply from EDAS:
 - Fire in the battery rooms is assumed to result in an EDAS module-specific subsystem failure that results in reactor trip and ECCS valve opening.

The observations from the EDAS FMEA are considered in the context of design-basis events. Fire is outside the scope of, and therefore is not postulated to occur, as part of the design-basis safety analyses. As described in the EDAS system design description, the EDAS battery room post-accident conditions and conditions during a 72-hour loss of AC power event are limited to a mild environment by design. The safety analyses rely on the equipment qualification program to demonstrate that the equipment within the program is capable of performing its required functions in the specified environmental conditions for the required post-accident durations. Therefore, it is concluded that design-basis event progressions does not result in a mechanistic or consequential failure of the EDAS to provide power to maintain the ECCS valves closed.

6.4 Summary of EDAS Failures Addressed in Design-Basis Safety Analysis

Based on the reviews performed in Section 6.3, the following EDAS scenarios to supply power are defined for the safety analysis.

- A non-mechanistic failure is assumed where EDAS fails to provide power to the MPS, coincident with the design-basis event initiation. Analysis of this failure demonstrates that the EDAS is not relied upon to perform its functions during the design-basis event, consistent with a classification of the system and its functions as nonsafety-related.
- No mechanistic failures are identified to occur over the course of any design-basis event progression.

The EDAS failure to supply power results in reactor trip, containment isolation, DHRS actuation and ECCS actuation, design-basis events that require operation of the normal feedwater system, main steam system, chemical volume control system, or the control rod drive system to initiate the event essentially collapse to a single initiating event. The reactor trip and containment isolation prevent evolution of those unique initiating events coincident with the EDAS failure. However, other initiating events such as primary or

secondary side break events, inadvertent ECCS actuation, or inadvertent reactor valve opening events would have a different progression when EDAS failure is assumed coincident with the event initiation.

No mechanistic failures of the EDAS due to the progression of a design-basis event are identified to be addressed in the safety analyses. This conclusion is logically consistent with the EDAS system functions and design. A key function of the EDAS system is to provide backup power supply to specified loads when AC power supply is not available. In particular, EDAS provides power for post-accident monitoring, and provides power to prevent inadvertent actuation of the ECCS valves when they are not required to provide cooling. Through the design reliability assurance program (D-RAP), augmented requirements have been applied to the design to support performance of these functions. For design-basis event progressions where EDAS is initially available to supply power, EDAS continues to perform its functions to supply power as designed. Therefore, for event progression analyses where EDAS is assumed to be initially available to supply power, EDAS is assumed to continue to supply power as designed.

6.5 Results

The evaluation concludes:

- Failure of the EDAS to supply power to the MPS results in opening of ECCS valves, which is addressed in the design-basis safety analyses.
- No mechanistic failures of the EDAS are identified to occur over the course of any design-basis event progression.
- Failure of the EDAS is appropriately addressed in the design-basis safety analyses by assuming a non-mechanistic failure where EDAS fails to provide power to the MPS, coincident with the design-basis event initiation. Analysis of this failure demonstrates that the safety analyses do not rely on the EDAS to perform its functions during a design-basis event.

Based on the conclusions of the evaluation, design-basis safety analyses is performed assuming either:

- EDAS performs as designed and power is available throughout the event progression, or
- EDAS power fails coincident with the event initiation.

Because of the extremely low probability¹ of a failure of the EDAS during an event (also known as a "smart failure"), a "smart failure" of EDAS is, at most, a beyond design basis event. This analytical method is consistent with:

- ANS-30.2-2022 (Reference 10.1), Section 5.2.1.1 (frequency below 10^{-7} /plant-yr is beyond design basis event) citing NUREG 75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (reissued as NUREG-0800) and WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Power Reactors," (1973).
- ANS-58.14-2022, (Reference 10.2), Section 4.5.2 ("low probability events [such as "smart failure" of EDAS] considered not credible"); ANS 58.14 is cited extensively in DG-1371 (ML20168A883).
- ANS-51.1-1973 (Reference 10.6), Section 2.1 ("the full spectra of plant conditions in accordance with their anticipated frequency of occurrence and consequences").
- ANS-51.1-1983 (Reference 10.8), Section 3.2.3 ("If the frequency of occurrence of an event is shown to be $< 10^{-6}$ /reactor year on a best estimate basis, this event shall not be considered for design").
- NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development,"² at the safety classification flowchart on page 32 (results in "smart failure" of EDAS being, at most, a beyond design basis event (BDBE)).
- Memorandum to the Commissioners, "Alternative Risk Metrics for New Light-Water Reactor Risk-Informed Applications," and enclosed "White Paper on Options for Risk Metrics for New Reactors," (ML090150636, ML090160004) (Feb. 12, 2009) ($1E-6$ threshold)

For the US460 design, because the EDAS failure to supply power results in reactor trip, containment isolation, DHRS actuation and ECCS actuation, design-basis events that require operation of the normal feedwater system, main steam system, chemical volume control system, or the control rod drive system to initiate the event essentially collapse to a single initiating event. The reactor trip and containment isolation prevent evolution of those unique initiating events coincident with the EDAS failure. However, other initiating events such as primary or secondary side break events, inadvertent ECCS actuation, or inadvertent reactor valve opening events would have a different progression when EDAS failure is assumed coincident with the event initiation.

1. That extremely low probability is $1.5E-8$ /yr. An AOO, by definition, is expected to occur one or more times during the life of a plant. The AOO of interest is a reactivity insertion event that results in a trip on high power. Although generic data on such an event is not readily available, the frequency of a PWR general transient was conservatively used (frequency = $5E-1$ /yr, INL/EXT-21-65055, November 2021, C-2.1.2). Based on system design and redundancy, the subsequent loss of DC power considered failure of 2 DC buses. Using the Loss of DC power initiating event frequency in Table 5-1 and an exposure time of 1 hour, the probability of a subsequent loss of DC power was calculated to be $\sim 3.0E-8$ ($2.6E-4$ /mcy / 8760 hr/yr * 1 hr = $2.97E-8$). Therefore, an AOO followed by a loss of DC power is approximately $1.5E-8$ /yr (0.5 /yr * $3E-8$).

2. The NRC approved NEI 18-04 in RG 1.233 at 13 (ML20091L698) (June 2020), with clarifications not affecting this analysis. NEI 18-04 is technology-neutral, and while the title says "non-LWR," nothing in the body of NEI 18-04 nor its endorsing document, RG 1.233, forbid its use with LWRs. Technology-neutral, by its own definition, must include LWRs.

7.0 Evaluation of Plant Defense-in-Depth with Loss of DC Power

This section evaluates the ECCS blowdown consequences following a loss of EDAS power during a design-basis event progression with specific consideration of MCHFR margin. This is not considered to be a credible failure and is conducted to evaluate plant defense-in-depth. A loss of EDAS is evaluated beginning from a spectrum of initial conditions with parameters, which range from normal operating conditions to the analytical trip limits for cooldown, heatup, reactivity, and loss of inventory transient events. Additionally, an evaluation of post-CHF heat transfer and clad heatup is performed. This spectrum of calculation results is performed to quantify conditions that could challenge MCHFR margin if EDAS power supply is randomly lost during a design-basis event progression up to the time of reactor trip.

7.1 Range of Evaluated Conditions

The evaluated reactor power ranges from nominal 100 percent to the high power limit of 115 percent, pressurizer pressure ranges from the low pressure limit of 1850 psia to the high pressure limit of 2100 psia, and the hot leg temperature ranges from an approximate nominal value of 600°F to the high hot leg temperature limit of 620°F. These ranges are representative of the limiting module conditions during design-basis events before reactor trip occurs and provide an evaluation of the limiting MCHFR following a loss of EDAS after event initiation, which is not credible but conducted to evaluate plant defense-in-depth.

7.2 MCHFR State-Point Spectrum

It is expected that the conditions that result in the most limiting MCHFR occurs just prior to reactor trip. Evaluation of the MCHFR in loss-of-coolant accident (LOCA) and inadvertent ECCS actuation events show a rapid increase in CHF margin once the control rods are inserted. Therefore, a loss of EDAS transient is evaluated beginning from a spectrum of initial conditions with parameters that align with the analytical trip limits for cooldown, heatup, reactivity, and loss of inventory events. These state-points represent the expected worst possible conditions before reactor trip occurs. The spectrum of initial power levels, hot leg temperatures, and pressures are shown in Table 7-1. Except for these parameters, other model inputs and biases are conservatively implemented to reduce MCHFR, consistent with the methodology for Phase 0 (early blowdown) analysis of LOCA or inadvertent ECCS actuation events (Reference 10.6).

- Case 1 - Transients that increase power, pressure, and hot leg temperature (reactivity events, cooldown events).
- Case 2 - Transients that increase power and hot leg temperature (reactivity events, cooldown events).
- Case 3 - Transients that increase pressure and temperature (heatup events).
- Case 4 - Transients that increase power (reactivity events).
- Case 5 - Transients that increase temperature (heatup events).
- Case 6 - Transients that decrease pressure (loss of inventory events).

The MCHFR spectrum results are shown in Table 7-1 and can be compared to the Phase 0 event analysis limit of 1.20. {{

}}2(a),(c)

Table 7-1 Minimum Critical Heat Flux Ratio Initial Condition Spectrum Cases and Results

{{

}}2(a),(c)

7.3 Estimate of Post-CHF Clad Temperature

The MCHFR spectrum results in Table 7-1 show that {{
}}2(a),(c) To gauge the potential cladding temperature response should CHF be reached, post-CHF fuel cladding temperatures are estimated using boundary conditions from case 2-7. Conditions from case 2-7 are applied since these yield the most limiting MCHFR.

In general, the methodology to evaluate post-CHF cladding temperature is an extension of the NuScale LOCA evaluation model, which itself conforms with 10 CFR Part 50, Appendix K. This method defines model inputs and biases that are conservative for core temperatures and already maximizes peak cladding temperature. However, because loss of inventory events and other design-basis accidents do not violate MCHFR margin limits, NuScale has not developed an NRC approved methodology for evaluating post-CHF heat

transfer. NuScale is not seeking NRC approval of a method to evaluate post-CHF heat transfer. These results are provided to illustrate the magnitude of cladding temperature increase during a postulated event with random loss of EDAS after event initiation but up to the time of reactor trip. This evaluation applies the default post-CHF heat transfer correlations and transition logic built into NRELAP5.

A multiplier of $\{ \{ \}^{2(a),(c)}$ is applied to the CHF to ensure NRELAP5 transitions to post-CHF heat transfer. Results from the unmodified case 2-7 are also plotted to compare the core response when CHF is not reached.

Figure 7-1 presents peak clad temperature as a function of time. $\{ \{$

$\}^{2(a),(c)}$ The increase in flow allows the clad surface to re-wet and transition back to the nucleate boiling heat transfer regime. The clad temperature excursion lasts for approximately 5 seconds before returning to temperatures less than the initial value. After this excursion, the transient behaves similarly to the longer-term transient. Decreased core power and continuous liquid coverage ensure margin to CHF is maintained over the long-term. Additionally, no loss of coolable geometry is anticipated due to low PCT compared to the 2200°F limit of 10 CFR 50.46(b)(1).

Figure 7-1 Peak Clad Temperature.

{{

}}2(a),(c)

8.0 Probabilistic Risk Assessment Insights

NuScale probabilistic risk assessment (PRA) evaluates the risk of beyond-design-basis events. The risk metrics that NuScale quantifies are core damage frequency (CDF) and large release frequency (LRF), which the NRC has identified as surrogates for cancer fatality risk and early fatality risk, respectively. Although the NRC has established safety goals for the CDF and LRF risk metrics, NuScale has shown that the design is far safer than that required by these safety goals by multiple orders of magnitude.

As part of the assessment performed for determining SSC required to be included in the regulatory treatment of nonsafety-related systems (RTNSS), a "focused" PRA is quantified that only takes credit for safety-related SSC (i.e., all nonsafety-related SSC fail). The objective of this assessment is to identify those nonsafety-related SSC that need to be added to the quantification in order to meet the NRC's safety goals. Those nonsafety-related SSC that need to be added into the focused PRA in order to meet the safety goals are then classified as RTNSS. For the NuScale design, no nonsafety-related SSC (including EDAS) need to be credited; the NuScale design satisfies the NRC safety goals crediting only safety-related SSC.

In the context of assessing the NuScale plant response to various upset conditions, the primary function of EDAS is to maintain the ECCS in a standby configuration; that is, EDAS prevents the opening of the ECCS valves when they do not need to open. (Note that as with all safety-related actuations, a loss of EDAS results in the actuation of ECCS.)

In the analysis performed to support the PRA, the ECCS functions to preserve RCS inventory: a single train is sufficient to allow core cooling without RCS makeup from external source. ECCS reliability is high because there is no dependence on any support system (i.e., AC or DC power, or service water) or operator action, no reliance on external sources of inventory addition, redundancy in the design, and because of the fail-safe nature of ECCS operation.

The only other post-accident function of DC power is post-accident monitoring (PAM). However, in the context of mitigating severe accidents, there are no plant responses or operator actions that require or depend on PAM. Therefore, the absence of PAM (due to the potential unavailability of EDAS) does not affect the public health and safety in any quantifiable manner.

Table 8-1 provides the point estimate results for both the CDF and LRF per module critical year (mcy) for the US460 design. The results from the US600 design are also provided for comparison purposes.

Table 8-1 Probabilistic risk assessment results for NuScale designs

Design - Model	CDF	LRF
US460 - Base model	5.4E-09	3.4E-13
US600 - Base model	2.7E-10	1.7E-11
NRC Safety Goal	1.0E-04	1.0E-06

9.0 Summary and Conclusions

The US460 NuScale Power Module is a unique integral PWR design that relies on natural circulation, and a limited number of safety-related system to mitigate the consequences of postulated accidents. NuScale applies a risk-informed, performance-based design process, consistent with ANSI/ANS 30.3, ANSI/ANS 58.14, and other guidance and policy to determine the risk and safety classification of individual system functions. SSCs are classified as (1) safety-related, (2) nonsafety-related, or (3) nonsafety-related with augmented requirements (i.e., special treatment). This design process is applied to the DC power systems to conclude that the EDAS is appropriately classified as not risk-significant, non-safety related with augmented design requirements.

The design of the EDAS prevents the failure of any single power channel from causing a loss of DC power to the module. An initiating event analysis has determined that probability of a random failure (at any time) of the EDAS is $2.6\text{E-}04$ mcy (per module critical year), and the probability of an EDAS "smart failure" during a reactivity addition accident is on the order of $1.5\text{E-}8$ mcy.

In addition to being highly unlikely, a "smart failure" of EDAS during a reactivity addition accident has low to no safety consequences. The CHFR remains above the LOCA/IORV MCHFR limit for the majority of state-point conditions. The CHFR decreases below the analysis limit for cases that combine high power with high hot leg temperature. The cladding temperature response as a function of time is determined from the most limiting conditions to determine the significance of this result. {{

}}2(a).(c) well

below the peak clad temperature limit of 2200°F for 10 CFR 50.46, preserving a coolable geometry.

Finally, the NuScale probabilistic risk assessment (PRA) quantifies the risk impact of all identified and credible design-basis and beyond design-basis events. The risk metrics that NuScale quantifies are core damage frequency (CDF) and large release frequency (LRF), which the NRC have identified as surrogates for cancer fatality risk and early fatality risk, respectively. The results of the PRA show that the risk metrics satisfy the NRC safety goals by multiple orders of magnitude, thereby preserving public health and safety. Additionally, when compared to the PRA results from the US600, approved by the NRC in the US600 DCA, the LRF is reduced by an order of magnitude.

10.0 References

- 10.1 American National Standards Institute, "Light Water Reactor Risk-Informed, Performance Based Design," ANSI/ANS-30.3-2022, La Grange Park, IL.
- 10.2 American National Standards Institute, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors," ANSI/ANS-58.14-2022, La Grange Park, IL.
- 10.3 NuScale Power, LLC, "Risk Significance Determination," TR-0515-13952-NP-A, Revision 0.
- 10.4 American Society of Mechanical Engineers/American Nuclear Society, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-Sa-2009, ASME/ANS, New York, NY, February 2009.
- 10.5 NuScale Power, LLC, "Non-Loss-of-Coolant Accident Methodology." TR-0516-49416-P, Revision 4.
- 10.6 NuScale Power, LLC, "Loss-of-Coolant Accident Methodology." TR-0516-49422-P, Revision 3.
- 10.7 American National Standards Institute, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Water Reactor Plants." ANSI/ANS 51.1-1973, La Grange Park, IL.
- 10.8 American National Standards Institute, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Water Reactor Plants." ANSI/ANS 51.1-1983, La Grange Park, IL.

Appendix A Background

The NPM design is unique as a passive, natural circulation reactor, with reliance on a limited number of safety-related SSC.

A.1 Regulatory Framework

This section identifies the regulatory requirements relevant to safety classification of EDAS and the treatment of EDAS in safety analysis for the NPM. Several of the GDCs are subject to exemptions for the US460 standard design; they are listed here because the rules inform NuScale's approach to safety analysis and demonstration that the design meets the underlying purpose of the GDCs.

10 CFR 50.2

Safety-related structures, systems and components means those structures, systems and components that are relied upon to remain functional during and following design-basis events to assure:

- (1) The integrity of the reactor coolant pressure boundary
- (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
or
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 50.34(a)(1) or 100.11 of this chapter, as applicable.

10 CFR 50 Appendix A GDCs

GDC 10 - Reactor design.

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC 15 - Reactor coolant system design.

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

GDC 17 - Electric power systems.

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

GDC 18 - Inspection and testing of electric power systems.

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

GDC 20 - Protection system functions.

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC 26 - Reactivity control system redundancy and capability.

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC 29 - Protection against anticipated operational occurrences.

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

GDC 33 - Reactor coolant makeup.

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

GDC 34 - Residual heat removal.

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC 35 - Emergency core cooling.

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC 38 - Containment heat removal.

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC 44 - Cooling water.

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Appendix B Design Attributes

Table B-1 includes reactor plant design and operational attributes that support a determination that plant electrical systems do not fulfill functions that, per the regulatory definitions of “safety-related” and “Class 1E,” justify a Class 1E classification. These attributes include augmented design, qualification, and quality assurance (QA) requirements that would be prescribed as minimum requirements on electrical systems that, although determined to be nonsafety-related, are essential to the post-accident monitoring of Type B and Type C variables. Table B-1 also specifies a reliability comparison between the nonsafety-related DC electrical system(s) and typical Class 1E DC electrical system(s). The augmented requirements to be applied to such electrical system(s) are detailed in Table B-2.

The process NuScale used to develop the attributes of Table B-1 is the same as that used historically by the nuclear industry in establishing whether the appropriate safety classification of an electrical system is Class 1E or non-Class 1E. The foundation of this methodology is the 10 CFR 50.2 definition of “safety-related” and the associated definition of “Class 1E” in NRC-endorsed industry standards. As safety-related and Class 1E are functional terms, the assessment of an electrical system for appropriate safety classification focuses on determining whether electrical power is necessary to fulfill any of the functions included in the definitions of “safety-related” and “Class 1E.”

Thus, the safety classification assessment involves first identifying those safety functions contemplated within these regulatory definitions, and then a methodical “function-by-function” evaluation of the role electrical power plays in the fulfillment of each function. If electrical power provided by one or more electrical systems is necessary to achieve and maintain the function (i.e., is an auxiliary supporting feature), then the electrical system(s) or portions thereof that are essential to the performance of the function would be appropriately classified as Class 1E.

Conversely, if electrical power is not necessary to achieve and maintain the function, then the electrical power would not need to be provided from a Class 1E electrical supply. Specifically, the application of augmented requirements is consistent with the process established in the NRC regulatory framework for “special treatment” of nonsafety-related SSC that are determined to have risk-significance. This well-established framework includes previous Commission policy statements in SECY-94-084 and SECY-95-132 and their associated Staff Requirements Memorandums as implemented further in NUREG-0800, Section 19.3, “Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors.” The application of the augmented requirements described in Table B-2 via Table B-1 is consistent with the approach of these regulatory policy and guidance documents.

Table B-1 EDAS Safety Classification Basis

Condition No.	Design Attributes and Conditions Satisfied	Justification
1.	For a design-basis event, electrical power is not necessary to achieve safe shutdown, core cooling, containment isolation and integrity, and RCPB integrity, and to maintain them for a minimum of 72 hours.	Design-basis Event Assumptions • 72 hour stabilized condition DBE end state without operator actions required • DBE analysis includes loss of electrical power • SBO Loss of AC and DC power for 72 hour duration
a.	Reactor trip is assured.	• CRDS does not rely on electrical power • SBO reactor trip
b.	Sufficient reactor coolant inventory and negative reactivity are assured during and following a design-basis event to achieve and maintain safe shutdown.	• Shutdown capability does not rely on electrical power • Decrease in inventory event analyses do not rely on electrical power or credit active injection sources • SBO does not rely on electrical power for shutdown or inventory control
c.	Core cooling systems achieve and maintain their safety-related functions.	• DHRS function does not rely on electrical power • ECCS function does not rely on electrical power • DBE analysis does not credit electrical power for DHRS or ECCS functions • Fuel and core acceptance criteria confirm core cooling • SBO core cooling relies on DHRS and ECCS
d.	Containment isolation is achieved and maintained.	• CNV isolation function does not rely on electrical power • SBO containment integrity does not rely on electrical power
e.	Containment integrity is achieved and maintained with no reliance on active ESF heat removal and combustible gas control systems.	• Passive CNTS and UHS design does not include active ESF heat removal and combustible gas control systems • DBE thermal hydraulic acceptance criteria confirm containment peak pressure margin • No credit for active ESF heat removal for containment integrity in SBO analysis

Table B-1 EDAS Safety Classification Basis (Continued)

Condition No.	Design Attributes and Conditions Satisfied	Justification
f.	Active fission product control and removal systems (e.g., ESF atmospheric clean-up systems, containment spray systems, etc.) are not needed to maintain doses less than the applicable guideline exposures set forth in 10 CFR 100.21, 10 CFR 50.34(a)(1)(ii)(D), and 10 CFR 52.137(a)(2)(iv)	• Active fission product removal systems are not required • DBE radiological consequences show guidelines maintained
g.	RCPB overpressure protection is achieved and maintained.	• Overpressure protection system does not rely on electrical power • SBO RPV pressure margin • DBE thermal hydraulic analyses confirm margin to RCS pressure acceptance criteria
2.	Post-accident monitoring instrumentation is not necessary to fulfill the functions contemplated in the IEEE Std. 603 and IEEE Std. 308 definition of Class1E in response to a design-basis event.	
a.	Operator action is not required to achieve safe shutdown, core cooling, containment isolation and integrity, and RCPB integrity, and to maintain them for a minimum of 72 hours.	• PAM design • No credit for manual actions in SBO analysis
b.	The design does not include Type A variables as defined in IEEE Std. 497-2002 as modified in Position C.4 of Regulatory Guide 1.97, Revision 4.	• PAM design
c.	The design does not include Type B and Type C variables that are necessary for the performance of operator actions in response to a design-basis event.	• PAM design
3.	Adequate cooling and shielding of spent fuel assemblies (i.e., water level greater than 10 feet above the top of spent fuel assemblies in the spent fuel pool [SFP]) is assured for a minimum of 72 hours with no reliance on electrical power or operator action.	• UHS and SFP integrated passive cooling function does not rely on electrical power • SFP design maintains 10 feet in SFP above racks without active systems

Table B-1 EDAS Safety Classification Basis (Continued)

Condition No.	Design Attributes and Conditions Satisfied	Justification
4.	Adequate cooling and shielding of reactor fuel assemblies in the process of core unloading and loading, including movement between the spent fuel pool and the reactor core, is assured for a minimum of 72 hours with no reliance on active systems that require electrical power or operator action.	• UHS and SFP integrated passive cooling function does not rely on electrical power • SFP design maintains 10 feet in SFP above racks without active systems • UHS provides a minimum of 10 feet for shielding available during fuel handling
5.	Electrical power is not necessary to actuate control room habitability system (CRHS) functions and to maintain them for a minimum of 72 hours.	• CRHS does not rely on electrical power • Main control room habitable during SBO without relying on electrical power
6.	Temperatures in building areas housing equipment relied upon to operate during station blackout (SBO) conditions are maintained within required equipment qualification limits for a minimum of 72 hours, with no reliance on electrical power and assuming the worst-case heat loads for the areas of concern.	• PAM environmental qualification • 72 hour loss of ventilation • EDAS environment • SBO mitigation equipment for 10 CFR 50.63 (DHRS and ECCS) is operable in SBO environment
7.	Active ventilation or fission product removal systems are not necessary in response to a postulated design-basis event - including a fuel handling accident - to maintain doses less than the applicable guideline exposures set forth in 10 CFR 100.21, 10 CFR 50.34(a)(1)(ii)(D), and 10 CFR 52.137(a)(2)(iv).	Active fission product removal systems • Active fission product removal systems are not required to meet regulatory requirements Active ventilation systems not required to mitigate the consequences of design-basis events • Reactor building ventilation system • No engineered safety function ventilation in design Dose analyses • No active ventilation systems are required to maintain offsite doses within applicable guidelines
8.	The reliability of the EDAS is comparable to that of a typical Class 1E DC power system	• EDAS reliability evaluation
9.	The plant emergency lighting capability satisfies the guidance of SRP/DSRS Section 9.5.3; Regulatory Guide 1.189; and NUREG-0700, with appropriate portions powered from EDAS.	• SBO emergency lighting • Fire protection • Emergency lighting design
10.	Compliance with 10 CFR 50.63	• SBO

Table B-1 EDAS Safety Classification Basis (Continued)

Condition No.	Design Attributes and Conditions Satisfied	Justification
11.	EDAS SSC that provide backup power are seismic category I	• EDAS seismic design
12.	Operator actions are not necessary to ensure the performance of safety-related functions for any postulated DBE (i.e., the design does not include Type A variables as defined in IEEE Std. 497-2002, as modified in RG 1.97, Regulatory Position C.4)	• PAM design does not include type A variables • Operator actions not needed to support DBE analysis
13.	The frequency for which a combination of an AOO and an actuation of the NuScale ECCS is not expected to occur during the lifetime of the module	• Section B.1
14.	The reactor can be brought to a safe shutdown using only safety-related equipment in the absence of electrical power following a DBE, with margin for stuck rods.	• Shutdown capability • CVCS safety evaluation • No return to power

Table B-2 US460 Augmented Design, Qualification and Quality Assurance Requirements to Support EDAS Safety Classification

Topic	Provision
Quality Assurance (QA)	Graded QA (GQA) Program GQA as described in QA Program Description (QAPD) meets or exceeds augmented QA provisions specified in RG 1.155, Appendix A
Environmental Qualification (EQ)	Batteries providing backup DC power are environmentally qualified per IEEE 323-2003 and located in a mild environment
Batteries	Commercial grade valve-regulated lead acid (VRLA) batteries - Design and installation per IEEE Std. 1187-2013 - Maintenance and testing shall be performed in accordance with IEEE Std. 1188-2005(R2010) with 2014 amendment - Sizing per IEEE Std. 485-2020 - Instrumentation, indication, and alarms per IEEE Std. 946-2020, IEEE Std. 1491-2012, IEEE Std. 1187-2013, and IEEE 1188-2005
Onsite Standby Power Sources	Nonsafety-related backup generators - Per SRP Section 19.3 - Per EPRI Utility Requirements Document, Rev. 13 - Per SECY-94-084, SECY-95-132, and associated SRMs
Identification	Per IEEE Std. 384-1992 as endorsed with modification by RG 1.75
Independence	Independence via physical separation and electrical isolation per IEEE Std. 384-1992 as endorsed with modification by RG 1.75
Single Failure Criterion	The single-failure criterion is applied to EDAS SSC that provide electrical power to prevent unintended ECCS valve actuation per IEEE 379-2020 as endorsed by RG 1.53.
Common-Cause Failure (CCF)	CCF probability minimized to the extent practicable - redundant divisions and channels - protection from environmental and dynamic effects of internal equipment failures via augmented seismic design/qualification, environmental qualification, and quality assurance provisions, and protection from natural phenomena via augmented seismic design/qualification and quality assurance provisions and location of the EDAS within Seismic Category I structures - Each battery supply shall be immediately available during both normal operations and following the loss of power from the AC system. - EDAS batteries are connected to the DC distribution panel and maintained on a float charge. Upon a loss of AC power the batteries immediately supply all required EDAS loads without interruption.

Table B-2 US460 Augmented Design, Qualification and Quality Assurance Requirements to Support EDAS Safety Classification (Continued)

Topic	Provision
Protection	Equipment protection and coordination studies are performed in accordance with IEEE 242, IEEE 946, and IEEE 1375.
Power Quality	Safety systems protected from variations in voltage, frequency, and waveform Class 1E isolation equipment at each interface between the non-Class 1E electrical system and downstream Class 1E circuits.
Location of Indicators and Controls	Controls and indication provided inside and outside the MCR: • EDAS instrumentation, indication, and alarming features are consistent with guidance contained in the IEEE 946, IEEE 1491, IEEE 1187, and IEEE 1188
Surveillance and Testing	Surveillance and testing: • Periodic inspection and testing is performed on the EDAS for operational, commercial, and plant investment protection purposes.
Multi-Module Considerations	No sharing of DC power between modules that results in potential adverse interactions.

B.1 Frequency of ECCS Actuation after AOOs

The frequency of ECCS actuation following an AOO is determined by first identifying those AOOs that require ECCS actuation for mitigation and summing their frequencies.

The AOOs that require ECCS are those that result in the relocation of reactor coolant from the reactor pressure vessel to the containment vessel, or otherwise actuate ECCS as part of the mitigation strategy. Of the events classified as AOOs, only the inadvertent opening of an RSV or the ECCS relocates coolant to containment.

Then, other causes of ECCS actuation are evaluated. Of the ECCS actuation causes considered, the loss of DC power (i.e., EDAS) probabilistic risk assessment (PRA) initiating event is included in this frequency analysis. This initiating event involves the common cause failure of two or more DC buses and is therefore not an AOO. However, it is included as it would be a direct cause of ECCS actuation. Considering this frequency is analogous to safety analyses of AOOs that deterministically assume DC power is not available.

The inadvertent RSV opening is a contributor to the loss of coolant accident (LOCA) inside containment, and only that portion of the frequency from the PRA initiating event is used; the spurious ECCS valve opening and loss of DC power are stand-alone initiating events and their frequencies are used directly.

The total frequency of ECCS actuation following an AOO is shown in Table B-3.

Table B-3 AOOs resulting in ECCS actuations

Event	Frequency (mcyr)
Spurious RSV opening	1.1E-03
Spurious ECCS valve opening	7.2E-04
Loss of DC power	2.6E-04
Total	2.1E-03

The results in Table B-3 show that the frequency of ECCS actuation following such events is approximately 2.1E-03/mcyr (module critical year), or once in 476 years. It can therefore be reasonably concluded that ECCS actuation following an AOO is not expected in the 60-year design lifetime of an NPM.

Appendix C Loss of Power Considerations

In typical LWR safety analyses, 'loss of offsite power' is considered for many transients coincident with the initiating event and the limiting single failure of systems relied on to mitigate the transient. However, the role of 'offsite power' is not required for NuScale plants. The reliance on natural circulation for normal operation and safety system performance reduces the impact to the core from a loss of electric power relative to a design with forced circulation. The US460 product has no safety-related electrical power supply. The EDAS employs multiple channels with battery backups to provide DC power for module protection system (MPS) operation for at least 72 hours, and maintaining ECCS valves closed for at least 24 hours following a loss of AC power. With a loss of AC power to the EDAS battery charger, the EDAS batteries supply DC power to keep the ECCS valves closed. After 24 hours, the ECCS valves open.

From a non-LOCA transient perspective, details of the power systems are used to define what constitutes a loss of power. Specifically, affected systems and equipment are identified to help categorize the system responses.

The following areas are discussed:

- Whether to consider a loss of power for an event.
- Which electrical systems to consider.
 - Systems and components lost.
- Time that a source of power is lost.
 - Loss of power scenarios and associated system responses.
 - EDAS failure is not a unique initiating event.
 - Generic evaluations of selected loss of power scenarios.
- Summary of loss of power scenarios for FSAR Chapter 15.

The effect of a specific power availability scenario should be assessed as part of the transient simulation. This assessment may be qualitative, quantitative, or a combination of both. An example is any combination of:

- Reference to a higher-level methodology that documents why a specific power availability scenario is not limiting for that transient.
- Qualitative engineering assessment to document the limiting or non-limiting nature of a specific power availability scenario for that transient.
- Quantitative assessment to document sensitivity results for that specific power availability scenario.

Assessments are needed for each applicable acceptance criterion because the power availability scenarios generally differ. The applicable acceptance criteria are those defined in the SRP and the NuScale DSRs. For non-LOCA transients, the time that the ECCS valves open depends on EDAS availability (Section C.2). If EDAS is lost, ECCS actuates and the event is no longer addressed under the non-LOCA methodology. ECCS actuation is addressed in FSAR Sections 15.6.5 and 15.6.6 using LOCA methodology or IORV methodology.

C.1 Basis for Considering Loss of Power for Non-LOCA Transients

The events from Table C-1 are reviewed to identify events that explicitly consider loss of power. This review is specific to the relevant SRP or NuScale DSRS subsection. The list of events is then reduced to those events that employ the non-LOCA methodology (Reference 10.6). The non-LOCA transients that explicitly consider loss of power are listed in Table C-1. The comments provide a summary of regulator expectations.

Review Procedure 7 to SRP 15.4.1 and Review Procedure 5 to SRP 15.4.2 direct that a loss of offsite power in conjunction with the limiting single active failure must be considered when evaluating the results of these anticipated operational occurrences for new applications. This position is based on interpretation of GDC 17, as documented in the Final Safety Evaluation Report for the ABB-CE System 80+ design certification.

A review of the event classifications from Table 4-1 indicates that the non-LOCA transients of Table C-1 correspond to AOOs and accidents. Those non-LOCA transients classified as accidents are shown with *red italics* in Table C-1. For non-LOCA transients not identified in Table C-1, power must generally be available for the system or component to malfunction.

Each non-LOCA transient considers the effect of loss of power when evaluating the acceptance criterion of interest.

Table C-1 Loss of Power for non-LOCA Transients

FSAR Section	Event	Comment
15.1.5	Steam system piping failures	Note 1.
15.2.1	Loss of external load	Note 2.
15.2.2	Turbine trip	Note 2.
15.2.3	Loss of condenser vacuum	Note 2.
15.2.4	Main steam isolation valve closure	Note 2.
15.2.7	Loss of normal feedwater flow	Note 3.
15.2.8	Feedwater system pipe breaks	Note 4.
15.4.1	Uncontrolled control rod assembly bank withdrawal from a subcritical or low power startup condition	Note 5.
15.4.2	Uncontrolled control rod assembly bank withdrawal at power	Note 6.

1. Assumption 2 to DSRS 15.1.5: Assumptions as to the loss of offsite power and the time of loss should be made to study their effects on the consequences of the accident. A loss of offsite power could occur simultaneously with the pipe break or during the accident, or offsite power might not be lost.
2. Technical Rationale 4 of DSRS 15.2.1 - 15.2.5: GDC 17 requirements provide assurance that SAFDLs and reactor coolant pressure boundary design conditions are not exceeded in initiating events that decrease heat removal by the secondary system, concurrent with a loss of offsite power.
3. Item 2 of Review Procedures to DSRS 15.2.7: Loss of feedwater should be analyzed with and without a loss of offsite power in combination with a single active failure.
4. Assumption 5B to DSRS Acceptance Criteria for DSRS 15.2.8: Assumptions as to the loss of offsite power and the time of loss should be conservative. Offsite power may be lost simultaneously with the pipe break, the loss may occur during the accident, or offsite power may not be lost.
5. Technical Rationale 3 of DSRS 15.4.1: Meeting GDC 17 ensures that fuel cladding integrity is not challenged during an uncontrolled control rod assembly withdrawal in conjunction with a loss of onsite or of offsite power.
6. Technical Rationale 3 of DSRS 15.4.2: Meeting the requirements of GDC 17 provides reasonable assurance that an uncontrolled control rod assembly withdrawal at power, in combination with loss of offsite power, does not result in a reactor transient that could cause the reactor coolant pressure boundary design conditions or the fuel design limits to be exceeded.

C.2 NuScale Electrical Systems and Loss of Normal Power for Safety Analyses

The following electrical systems are considered to identify the loads they supply that have a direct impact on the safety analysis:

- EHVS - High voltage (13.8kV) AC electrical system and switchyard
- EMVS - Medium voltage (4.16 kV) AC electrical distribution system
- ELVS - Low voltage (480V and 120V) AC electrical distribution system
- EDAS - Augmented DC power system
- EDNS - Normal DC power system

Safety analysis does not credit the backup power supply system to provide power following a loss of normal AC power and therefore this system is not considered further.

The plant lighting, grounding and lightning protection system and the security power system are not relevant to the design basis safety analyses and are not considered further.

Table C-2 summarizes the electrical system loads and whether they are of direct impact to the safety analysis calculations.

Table C-2 Review of Electrical Systems for Loads Important to Safety Analysis

System	Relevant Loads	Impact
EHVS	AC power supply to EMVS	Direct ⁽¹⁾
EMVS	Site cooling water pumps ⁽²⁾	Other ⁽¹⁾
	Chiller packages ⁽²⁾	Other
	AC power supply to ELVS	Direct
ELVS	EDAS Battery Chargers (module-specific)	Direct
	Feedwater pump motors	Direct
	Condensate pump motors	Direct
	Pressurizer heaters	Direct
	Containment evacuation system vacuum pumps	Direct
	CRDS power and control power	Direct
	Motor control center in RXB ⁽³⁾	Direct
	Motor control center in TGB ⁽³⁾	Direct
	EDNS Battery Chargers	Direct
	Air cooled condenser fans, vacuum pumps, & recirculation pump motors ⁽²⁾	Other
	EDAS Battery Chargers (common loads)	Other
EDNS	MCS modules, cabinets	Direct
EDAS	MPS (via Class 1E isolation devices)	Direct
	NMS (via Class 1E isolation devices)	Other

Table C-2 Notes:

- (1) A 'direct' impact to safety analysis is one that, if lost, has direct impact on the reactor module primary or secondary side conditions. An 'other' impact is one that may affect a system that connects to systems directly impacting the module, or may affect the ability to monitor the system.
- (2) Safety analysis does not directly model these systems. Loss of power to these loads might trigger a reactor shutdown or downstream effect on systems modeled by safety analysis (such as increase in feedwater temperature due to loss of condenser cooling).
- (3) The loads relevant to safety analysis include:
- Chemical and volume control RCS injection pump motors
 - Chemical and volume control recirculation pump motors
 - Containment evacuation system vacuum pump motors
 - Turbine stop valve control oil pump motor

Motor operated valves (MOVs) are identified as part of the electrical loads because these valves use an electric motor to adjust position. Since a loss of power may affect their operation, which systems utilize MOVs is important. MOVs are used in the chemical and volume control (CVCS), chilled water, containment evacuation (CES), nitrogen distribution, pool cooling and cleanup, reactor component cooling water, and site cooling water. The MOVs in CVCS, containment evacuation system, and main steam directly impact the primary or secondary side of the reactor module, therefore, an assessment for each system is provided.

If ELVS is not available, the motors for the CVCS RCS injection pumps and recirculation pumps do not have power. CVCS flow stops regardless of the impact of the loss of power

on any MOVs in the flow path. Therefore, the specific impact of loss of ELVS on MOVs in the CVCS is not further investigated.

If ELVS is not available, the motors for the CES vacuum pumps do not have power. Flow from the vacuum pumps stops regardless of the impact of the loss of power on MOVs in the flow path. Therefore, the specific impact of loss of ELVS on MOVs in the CES is not further investigated. If ELVS is not available, the motor for the turbine generator system control oil pump does not have power. Consequently, 1) the hydraulically operated turbine stop valve uses the high pressure fire-resistant hydraulic fluid from the control oil system for control and protection; and, 2) the vendor supplied main turbine control system monitors and controls all turbine and generator subsystems, including the turbine control oil. If the main turbine control system determines there is a problem that warrants immediate action, the controller initiates a turbine trip to close the turbine stop valve and turbine control valve. The module control system would act to maintain secondary pressure by opening the turbine bypass valve. However, with ELVS not available, turbine bypass is not operable because the motors for the feedwater and condensate pumps do not have power. Impacts to the main steam system valves due to loss of control signal are bounded by the turbine stop valve closure without turbine bypass. Therefore, the main steam system is not further investigated.

Based on Table C-2, the electrical systems and main loads important to safety analysis are summarized below and are illustrated in Figure C-1:

- EHVS - 13.8 kV high voltage electrical system (AC power) No main loads important to safety analysis
 - Powers EMVS, and from there to ELVS, EDNS chargers, EDAS chargers. No specific loads to components or other systems that are relevant to safety analysis.
- EMVS - 4.16 kV medium voltage electrical system (AC power) No main loads important to safety analysis
 - Powers ELVS, and from there to EDNS chargers, EDAS chargers. No specific loads to components or other systems that are relevant to safety analysis.
- ELVS - 480 V, 120 V low voltage electrical system (AC power) Loads relevant to safety analysis include:
 - Feedwater pump motors, condensate pump motors, pressurizer heaters
 - Control rod drive mechanisms
 - Power to the motor control center in the reactor building and turbine generator building. The loads relevant to safety analysis are:
 - Chemical and volume control RCS injection pump motors
 - Chemical and volume control recirculation pump motors
 - Containment evacuation system vacuum pump motors
 - Turbine stop valve control oil pump motor
 - Power to EDNS battery chargers, EDAS battery chargers

- EDNS - 250 V DC, 120/208 V AC - Normal DC power system Loads relevant to safety analysis include:
 - Module control system
 - Only the EDNS provides power to the MCS.
 - The turbine generator automatically trips on several set points, including reactor scram. Therefore, the turbine generator is expected to trip as a result of reactor scram.
- EDAS - 125V DC - Augmented DC power system - Loads relevant to safety analysis include:
 - Module protection system
 - Only the EDAS provides power to the MPS safety-related loads.

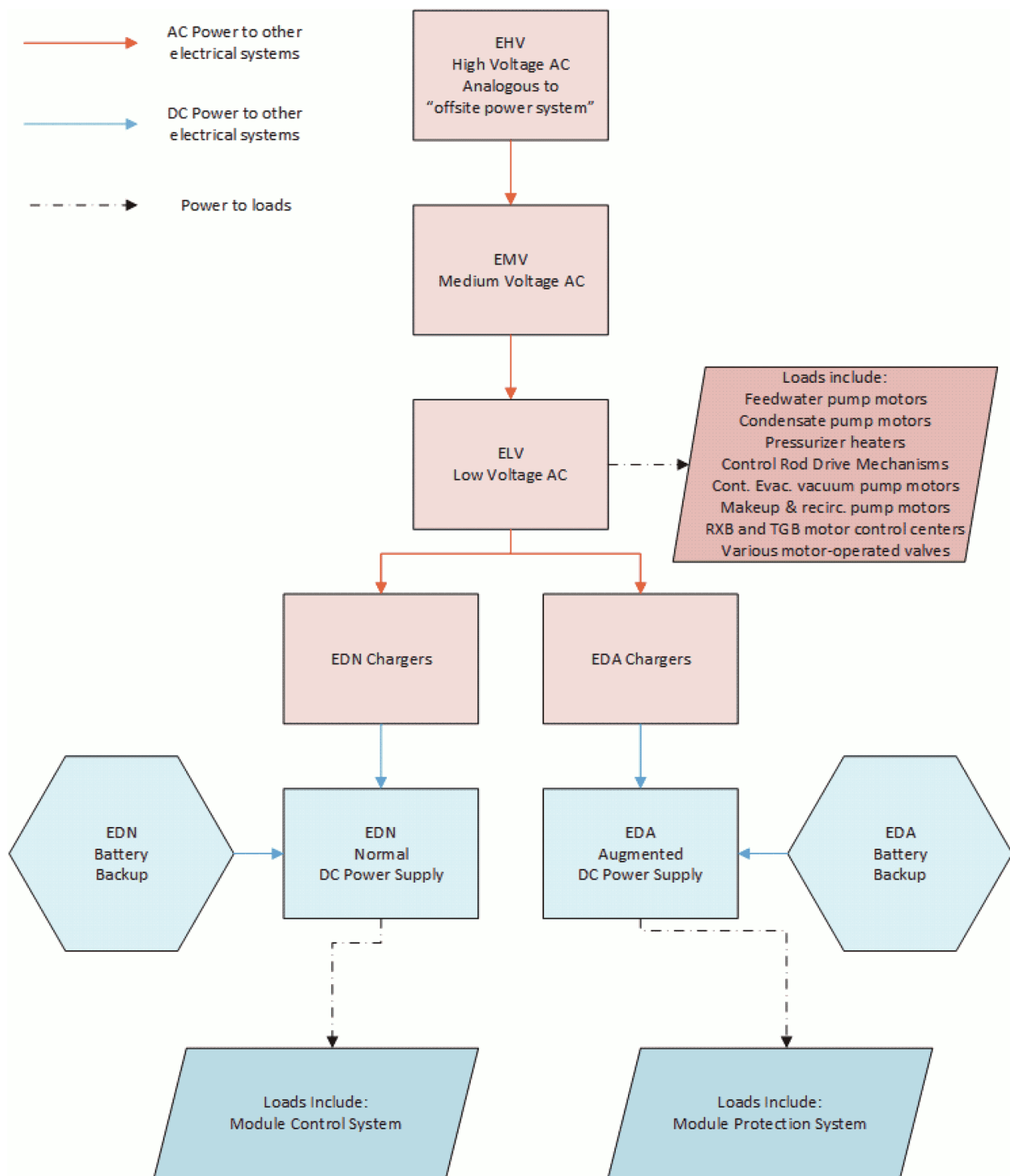
Both the EDNS and the EDAS are designed with battery backup for continued DC power in the event that normal AC power supply to the chargers is not available. The EDNS batteries are sized to supply loads for 40 minutes.

If normal AC power to the EDAS chargers is lost, the MPS initiates reactor trip, actuate DHRS, and close containment isolation valves. If the EDAS battery backup is available and the operators prevent ECCS actuation 8 hours after reactor trip, the EDAS continues to provide power to the MPS and to hold the ECCS valves closed for 24 hours, after that point the ECCS valves are actuated by the MPS. Upon loss of EDAS, ECCS actuates. The RVVs open immediately and the RRVs open when RCS pressure drops to below the IAB release pressure.

Based on this review of the electrical system design, for FSAR Chapter 15 safety analyses, 'loss of normal AC power' means loss of the ELVS, either due to failures within the ELVS or due to loss of power provided through EMVS and EHVS. Failures of EMVS or EHVS are not separately considered.

Since EDNS and EDAS are not safety-related systems, and have battery backup, the availability or unavailability of these DC power supply systems is also be considered.

The timing for considering the loss of power supply is discussed in Section C.3.

Figure C-1 Simplified Schematic of Electrical Systems and Loads Important to Safety Analysis

C.3 Loss of Power Timing and NuScale Event Progression

C.3.1 Timing for Assumption of Loss of Power

SRP Section 15 and associated sub-sections and DSRS Section 15 and associated sub-sections are not prescriptive regarding when to consider a loss of power. For example, DSRS Section 15.1.5 states:

A loss of offsite power may occur simultaneously with the pipe break or during the accident, or offsite power may not be lost.

For non-LOCA transients, loss of normal AC power is considered at two different points in the event progression:

- Coincident with the event.
- Coincident with turbine trip.

The basis for considering loss of power coincident with the event is that an external event could cause the initiating event as well as cause a loss of normal AC power. For example, a seismic event that might cause a pipe rupture could result in loss of normal AC power supply to the plant.

During an event progression, nonsafety-related controls are expected to trip the turbine following reactor trip. The basis for considering loss of normal AC power coincident with turbine trip during the event progression is that typically, large nuclear plants consider loss of offsite AC power a consequence of the event due to a disruption of the electrical grid following a turbine trip and loss of the ~1000 MWe power supply from the nuclear plant. The NuScale design is much smaller and loss of power from a single NPM is not expected to significantly disrupt the grid power supply, however this event progression is assessed as part of the safety analyses. Delays between reactor trip and turbine trip are conservatively treated in the safety analysis.

Considering loss of power coincident with the event and coincident with turbine trip is consistent with other advanced reactor designs and SRP Section 8.3.1. Specifically, Technical Rationale 4 of SRP Section 8.3.1 states:

The trip of a nuclear power unit ... can result in reduced switchyard voltage, potentially actuating the plant's degraded voltage protection and separating the plant's safety buses from offsite power. It can also result in grid instability, potential grid collapse, inadequate switchyard voltages, and a subsequent [loss of offsite power] due to loss of the real and/or reactive power support supplied to the grid from the nuclear unit.

With respect to EDNS and EDAS availability, these nonsafety-related systems could be deterministically postulated to randomly fail during the event. However, it is not typical or generally required to consider the random loss of a non-safety system part-way through an event. In particular, the redundancy and augmented

requirements of the EDAS, supports not assuming a loss of EDAS after event initiation, if it is initially available.

Facts supporting this conclusion are discussed below:

The EDAS battery backup has an independent connection to the DC bus that provides power to MPS, therefore, a loss of AC power does not have a direct causal link to failure of the DC bus supported by the EDAS batteries. Failure of the battery charger due to an electrical fault is electrically separated from the DC bus. Although the EDAS is not safety-related, many of the augmented requirements of a safety-related system are part of the design.

The Failure Modes and Effects Analysis (FMEA) for the EDAS shows that there are no single failures that could cause loss of power supply from EDAS. Consequently, loss of a single module-specific power channel does not result in loss of power supply, reactor trip, inadvertent ECCS actuation, or loss of post-accident monitoring capacity. Failures to multiple module-specific power channels must occur to initiate reactor trip or ECCS actuation.

The EDAS is designed with independent and diverse monitoring devices such that no single monitoring device failure goes undetected and prevent detection of the failure mechanisms evaluated in the FMEA. The EDAS batteries are designed to Seismic Category I requirements.

The EDAS FMEA conclusions are further summarized below:

Transition from normal EDAS and/or BPSS AC power supply available (Condition A) to loss of all AC (Condition B)

There is no adverse causal relationship during the transition from EDAS operation with AC power available (condition A) and AC power unavailable (condition B). There are no failure modes while EDAS is operating in condition A that causes EDAS operating in condition B to be unable to meet the required functions and duty cycle.

DC System Monitoring

The use of DC system condition monitoring equipment (battery, charger/switchgear) combined with periodic testing provides assurance that undetectable VRLA cell failures and associated EDAS subsystem failure does not occur. Proper consideration of monitoring system and test data allows for proactive corrective maintenance prior to an unacceptable VRLA battery cell condition.

All Equipment Available

When considering all EDAS equipment available, the functional EDAS is not susceptible to subsystem failure consequences, following the failure of any single component.

Fire Considerations

Fire is considered for EDAS due to the arrangement of the power channel components in the reactor building and the consequences on interfacing systems. The following circumstances could cause an EDAS subsystem to fail:

1. With AC power available: fire in a switchgear room causes reactor trip and ECCS actuation.
2. With AC power unavailable: fire in a battery string or switchgear room causes ECCS actuation.

In both instances, EDAS power is maintained by the other power channel components and no safety functions are impacted.

Digital Common Cause Failure

1. With AC power available: digital common cause failures of the battery chargers during normal plant operation do not prevent the batteries from fulfilling the EDAS mission. Should a digital control issue result in the battery chargers supplying abnormal or erratic voltage, the DC-DC converters function to ensure safety systems are either powered within a specific input voltage span or isolated to preclude degradation of the safety system(s).
2. With AC power unavailable: all battery chargers are not working. Therefore, digital common-cause failures need not be considered.
3. With AC power available or unavailable: common cause failures of equipment and monitoring during normal plant operation or following a loss of AC power does not affect the ability of EDAS to fulfill its mission. Because the battery monitoring devices only provide indication, the batteries supplying backup power are not susceptible to this failure mechanism.

The EDNS serves different functions than EDAS and accordingly does not require augmented design requirements. Non-mechanistic failure of the EDNS power supply is postulated, but has limited effects on transient event progression as summarized in Section C.3.2.

Considering the regulatory guidance and EDAS design features, the following scenarios are considered:

- EDNS and/or EDAS power is available.
- EDNS and/or EDAS power is lost coincident with the event either due to the conditions that result in loss of normal AC coincident with the event or as a non-mechanistic assumption.
- EDNS power is lost coincident with the turbine trip when the EDNS battery backup to the buses fail to provide power after normal AC power is lost.

C.3.2 Loss of Power Scenarios for Chapter 15 Design Basis Event Analysis and Transient Progression

Based on the NuScale electrical power system design and the timing of when AC or DC power is assumed to be available or lost, the loss of power scenarios for FSAR Chapter 15 design basis event analysis are defined. First, for completeness, the scenarios are identified in a parametric manner:

1. AC power is available
 - a. EDNS available, EDAS available
 - b. EDNS fails (at event initiation), EDAS available
 - c. EDNS available, EDAS fails (at event initiation)
 - d. EDNS fails (at event initiation), EDAS fails (at event initiation)
2. AC power is lost coincident with the event
 - a. EDNS available, EDAS available
 - b. EDNS fails (at event initiation), EDAS available
 - c. EDNS available, EDAS fails (at event initiation)
 - d. EDNS fails (at event initiation), EDAS fails (at event initiation)
3. AC power is lost at turbine trip
 - a. EDNS available, EDAS available
 - b. EDNS fails (at turbine trip), EDAS available

The plant response in each of these loss of power scenarios is described in Table C-3. As shown in Table C-3, the plant response to several of the loss of power scenarios is equivalent.

Some of the power scenarios need to be considered individually for each initiating event. However, other power scenarios are considered generically to demonstrate why they are non-limiting or are not considered specific initiating events. Two scenarios are considered generically in the following subsections:

- Loss of EDAS but AC power and EDNS batteries available.
- Loss of all AC and DC power at event initiation.

Table C-3 NuScale Plant Response in Various Power Scenarios

Scenario	AC Power	EDNS	EDAS	Description of Plant Response	Equivalent Scenario
1a	Available	Available	Available	All power sources available	n/a
1b	Available	Not available through transient	Available	Loss of EDNS causes MCS to lose power. Therefore: • Turbine trip at event initiation due to loss of control power. • Feedwater pumps stop at event initiation due to loss of control power. • CVCS pumps stop at event initiation due to loss of control power. • Pressurizer heaters turn off at event initiation due to loss of control power.	n/a
1c	Available	Available	Not available through transient	Loss of EDAS removes power from the MPS. Therefore, power to the reactor trip breakers is removed and all safety-related valves are actuated to go to their respective safety positions on loss of power to the valve actuators: • Reactor trip at event initiation due to loss of EDAS. • DHRS actuation valves open due to loss of EDAS. • Containment isolation valves and secondary side backup isolation valves close due to loss of EDAS. • ECCS valves are actuated; both RRVs open immediately, the inadvertent actuation block holds the RRVs closed until the RCS pressure decreases below the IAB release pressure. • Feedwater and CVCS pumps have power but the isolation valves close and block the flow path.	discussion in Section C.3.3.
1d	Available	Not available through transient	Not available through transient	Loss of EDNS causes the MCS to lose power. Loss of EDAS removes power from the MPS. The plant response is equivalent to scenario 2d (loss of AC, EDNS and EDAS at event initiation) because although AC power is available, the loss of control signal stops the feedwater pumps and CVCS pumps, and power to the pressurizer heaters is removed.	2d

Table C-3 NuScale Plant Response in Various Power Scenarios

Scenario	AC Power	EDNS	EDAS	Description of Plant Response	Equivalent Scenario
2a	Lost at event initiation	Available	Available	AC power is lost at the event initiation. Therefore: - Control rods begin to fall after loss of power to the CRDMs (before MPS actuation). - The MPS actuates DHRS and containment isolation after 60seconds due to low AC voltage to the EDAS battery chargers, if another signal is not generated. - Turbine trip at event initiation due to loss of AC power. - The feedwater pumps and CVCS pumps stop at event initiation due to loss of AC power to the pump motors. - Pressurizer heaters turn off at event initiation because power to the heaters is lost. The EDNS and EDAS battery backups are available (40 min for EDNS and 24/72 hours for EDAS). At 40 min, the battery backup to EDNS is depleted, which removes power from the MCS. Since the plant already tripped, there are no additional changes to the plant status or equipment. 8 hours after reactor trip: If the operators do not bypass the ECCS actuation signal, ECCS actuates. All ECCS valves are expected to open if RCS pressure is below the IAB threshold; otherwise the RVVs open immediately and the RRVs open at the IAB release pressure. If the operators bypassed ECCS actuation at 8 hours, then at 24 hours, the MPS reduces the load on the EDAS batteries by removing power to the ECCS valves. The RCS is depressurized by the DHRS by this time and all ECCS valves are expected to open. If RCS pressure remains above the IAB threshold, the RVVs still open due to removal of EDAS power. This transition occurs well beyond the short-term transient progression and is addressed in long term cooling analysis.	n/a
2b	Lost at event initiation	Lost at event initiation	Available	AC power is lost at event initiation and the EDNS battery backups do not pick up. The plant response is equivalent to scenario 2a because the effects of the loss of the EDNS batteries (removes power to the MCS) are already caused by the loss of AC power.	2a

Table C-3 NuScale Plant Response in Various Power Scenarios

Scenario	AC Power	EDNS	EDAS	Description of Plant Response	Equivalent Scenario
2c	Lost at event initiation	Available	Lost at event initiation	AC power is lost at event initiation and the EDAS battery backups do not pickup. The plant response is equivalent to scenario 2d (loss of AC, EDNS and EDAS at event initiation). The loss of AC power and EDAS causes reactor trip, DHRS actuation, containment isolation, ECCS valve actuation (RVVs open immediately, RRVs open at IAB release pressure), turbine trip, loss of power to the feedwater pumps, loss of power to the CVCS pumps, loss of power to the pressurizer heaters. The availability of the EDNS battery backup to the MCS does not affect the plant response.	2d
2d	Lost at event initiation	Lost at event initiation	Lost at event initiation	AC power is lost at event initiation. The EDNS and EDAS batteries fail to provide power. All AC and DC power is lost. Therefore: • Reactor trip at event initiation due to loss of AC power or EDAS. • DHRS actuation at event initiation due to loss of EDAS. • Containment isolation valves and secondary side backup isolation valves close at event initiation due to loss of EDAS. • Turbine trip at event initiation due to loss of AC power. • The feedwater pumps and CVCS pumps stop due to loss of AC power to the pump motors. • Pressurizer heaters turn off because power to the heaters is lost. • The ECCS valves are actuated due to loss of power from the EDAS to the valve actuators. The RVVs open immediately and the RRVs open when RCS pressure drops below the IAB release pressure.	n/a

Table C-3 NuScale Plant Response in Various Power Scenarios

Scenario	AC Power	EDNS	EDAS	Description of Plant Response	Equivalent Scenario
3a	Lost at turbine trip	Available	Available	The initiating event progresses until the MPS actuates reactor trip to protect the module. Turbine trip is initiated due to reactor trip. Due to grid disruption from the turbine trip, AC power is lost. EDNS and EDAS batteries are available. The loss of AC power at the turbine trip results in: • The MPS actuates DHRS and containment isolation at the turbine trip after 60 seconds due to loss of AC power to the EDAS battery chargers (if not already actuated by the MPS; reactor trip is already initiated by the MPS due to module conditions). • The feedwater pumps and CVCS pumps stop due to loss of AC power to the pump motors. • Pressurizer heaters turn off because power to the heaters is lost. The EDNS and EDAS battery backups are available (40 min for EDNS and 24/72 hours for EDAS). At 40 min, the battery backup to EDNS is depleted, which removes power from the MCS. Since the plant already tripped, there are no additional changes to the plant status or equipment. 8 hours after reactor trip: If the operators do not bypass the ECCS actuation signal, ECCS actuates. All ECCS valves are expected to open if RCS pressure is below the IAB threshold; otherwise the RVVs open immediately and the RRVs open at the IAB release pressure. If the operators bypassed ECCS actuation at 8 hours, then at 24 hours, the MPS reduces the load on the EDAS batteries by removing power to the ECCS valves. The RCS is depressurized by the DHRS by this time and all ECCS valves are expected to open. If RCS pressure remains above the IAB threshold, the RVVs still opens due to removal of EDAS power. This transition occurs well beyond the short-term transient progression and is addressed in long term cooling analysis.	n/a

Table C-3 NuScale Plant Response in Various Power Scenarios

Scenario	AC Power	EDNS	EDAS	Description of Plant Response	Equivalent Scenario
3b	Lost at turbine trip	Lost at turbine trip	Available	The initiating event progresses until the MPS actuates reactor trip to protect the module. Turbine trip is initiated due to reactor trip. Due to grid disruption from the turbine trip, AC power is lost. EDNS batteries fail to provide power. EDAS batteries are available. The plant response is equivalent to scenario 3a because the effects of the loss of the EDNS batteries (removes power to the MCS) are already caused by the loss of AC power.	3a

C.3.3 Generic Evaluation of Loss of EDAS

The EDAS is a non-safety related system with augmented requirements. Typically in safety analysis, non-safety systems are assumed to function or to not function in order to determine if the condition is conservative with respect to the transient results (CVCS function for pressurizer heaters or spray, for example). Loss of a non-safety system or malfunction in a non-safety system can also be an event initiator (for example, malfunctions in the condensate and feedwater system, which result in a loss of feedwater). First, the basis for not considering loss of EDAS as a unique initiating event is discussed.

Loss of EDAS is not considered as a unique event initiator because a complete loss of EDAS is unlikely due to the level of redundancy in the design and the augmented requirements imposed; and, because loss of EDAS is considered in conjunction with each of the other design-basis events. Although the effects of a loss of EDAS are considered as part of the power availability scenarios, the plant response is not evaluated with non-LOCA methodology. Instead, the loss of EDAS is evaluated with the inadvertent opening of reactor valve(s) methodology.

The EDAS provides power to the MPS. If the EDAS is lost, but normal AC power is available (Table C-3, Scenario 1c), the following response is expected:

- Reactor trips at event initiation due to loss of EDAS;
- DHRS actuates due to loss of EDAS, to open DHRS actuation valves and close feedwater and main steam isolation valves, feedwater regulating valves, and secondary main steam isolation valves;
- Containment isolation valves actuate (to close) due to loss of EDAS; and,
- ECCS valves actuate - RVVs open immediately.
 - The inadvertent actuation block delays opening of the RRVs until the RCS pressure decreases below the IAB release pressure.

Although a loss of EDAS coincident with the event actuates the safety systems, the overall plant response and transient analysis acceptance criteria for non-LOCA transients are expected to be bounded by other events evaluated using the inadvertent opening of reactor valve(s) methodology.

Variations in the system response for effects of scram time, valve closure times, etc. are bounded by other events. For example, if loss of EDAS is postulated as an initiating event as part of DHRS actuation, the feedwater and main steam valves close while the feedwater pumps are assumed to continue running. If the main steam isolation valves close quickly while the main feedwater isolation valves close slowly or experience a single failure; this could result in higher pressure in the steam generator secondary side. However, this scenario is bounded by considering the same combination of valve closure times and single failure in the loss of load event.

Considering the redundancy and augmented requirements imposed on the EDAS, loss of the EDAS independent of any other type of initiating event is not expected to occur in the life of the module (demonstrated quantitatively in Section 5.0). Since loss of the EDAS actuates the safety systems, loss of EDAS as an event initiator does not introduce a different or more limiting plant transient progression relative to other scenarios that are analyzed. Therefore, as a nonsafety-related system, the EDAS available/not available is considered as part of the power availability scenarios for all design basis events. Loss of EDAS is not treated as a unique initiating event.

Although the loss of EDAS is bounded by other events, the power availability Scenario 1c is generically evaluated to assist with understanding the expected response.

In Table C-3, Scenario 1c postulates loss of EDAS while AC power and EDNS are available.

The main analysis acceptance criteria of concern are maximum RCS pressure, maximum secondary side pressure, minimum critical heat flux ratio, maximum fuel centerline temperature, and maximum containment pressure. Each of these acceptance criteria are considered below. Loss of EDAS causes both RRVs to open immediately and RRVs to open when RCS pressure drops below the IAB release pressure.

- RCS pressure: Actuation of reactor trip coincident with the event reduces core power to decay heat levels, and DHRS is actuated coincident with the event to provide cooling. Therefore the RCS pressure response is non-limiting compared to scenarios where the safety system response is delayed until MPS analytical limits are reached, or compared to normal operating conditions. Different event types are considered below.
 - Heatup events - RCS pressure expected to increase, so earlier MPS actuation is a benefit.
 - Cooldown events - RCS pressure tends to decrease; this acceptance criterion is not challenged.
 - Reactivity events - Events driven by rod movement are mitigated by immediate reactor trip and containment isolation, except control rod ejection. Control rod ejection response is mitigated by the fuel design, followed by reactor trip. Opening the RRVs limits the pressure increase. Boron dilution is mitigated by immediate reactor trip and demineralized water system isolation.
 - Increase inventory event - Halted by containment isolation valve closures, so non-limiting.

- Secondary side pressure: The secondary side inventory and the primary side conditions at the time of DHRS actuation affect the peak secondary side pressure. Higher secondary side inventory and lower loop volume are expected to result in increased peak pressure. Therefore, actuation of reactor trip and DHRS coincident with the event is bounded by considering the range of conditions that increase secondary side inventory prior to DHRS actuation or containment isolation, and by considering the effects of single failures of the main steam isolation valve and feedwater isolation valve on the secondary loop volume. Different event types are considered below:
 - Heatup events - The turbine trip event or the main steam isolation valve closure stops steam flow out of the steam generator; as the feedwater pumps continue to operate, the secondary side inventory increases until a DHRS actuation signal is generated. This event bounds considering a DHRS actuation signal generated coincident with the event, and other heatup events where feedwater flow is not available.
 - Cooldown events - Some cooldown events, such as increase in feedwater flow, increase the steam generator inventory until a DHRS actuation signal is generated. This event bounds considering a DHRS actuation signal generated coincident with the event.
 - Reactivity events - Events driven by rod movement are mitigated by an immediate reactor trip and containment isolation, except for control rod ejection. Control rod ejection response is mitigated by the fuel design, followed by reactor trip. Boron dilution is mitigated by immediate reactor trip and demineralized water system isolation. Secondary side pressure is not expected to be challenged for these events, compared to heatup or cooldown events.
 - Increased inventory event - Mitigated by containment isolation and therefore containment isolation coincident with the event is non-limiting.
- Minimum critical heat flux ratio and maximum fuel centerline temperature: The immediate reactor trip coincident with the event reduces core power and heat fluxes. Actuation of reactor trip coincident with the event generally increases the margin compared to scenarios where the safety system response is delayed until MPS analytical limits are reached. For rod ejection, loss of EDAS causes an immediate scram that reduces the total reactivity inserted and the resulting peak power, which increases margin regardless of the RCS pressure response.
- Maximum containment pressure: For the non-LOCA events that do not have initial mass and energy release into the containment, opening the ECCS valves later in the event progression (on the 8 hour or 24 hour timer) is bounded by the inadvertent opening of a valve in conjunction with the immediate opening of both RVVs from full power conditions (inadvertent ECCS actuation is analyzed as an AOO). For secondary side high energy line breaks inside containment, the secondary side mass and energy release followed by ECCS valve opening with an assumed loss of EDAS could challenge containment pressure. These scenarios and the impacts of single failures are considered in the bounding containment pressure analysis for FSAR Chapter 6.

With respect to radiological dose consequences for infrequent events and accidents, loss of EDAS results in immediate containment isolation, and ECCS RVV valve opening. Similar to an inadvertent ECCS actuation, containment isolation reduces the potential release paths to containment leakage. Fuel failures are not expected to occur for the main steam line break, main feedwater line break, small line break outside containment, or the steam generator tube failure events. For these events, loss of EDAS and opening of the ECCS RVVs results in a higher containment pressure response compared to the event-specific scenario evaluations for loss of AC power at event initiation (where the ECCS valves remain closed until 8 hours after reactor trip or 24 hours after event initiation). In the radiological dose analyses, the containment leak rate is determined assuming that the containment leaks at the design basis leak rate for 24 hours and at half the design basis leak rate thereafter. Since the design basis leak rate is not a function of the peak pressure, this scenario is non-limiting with respect to releases prior to containment isolation and does not need to be specifically considered for these events to provide separate containment pressure input for the radiological dose analyses.

Fuel failure in the control rod ejection accident is determined by the physics of the design, such as the core design peaking and control rod worth. Similar to the other infrequent events and accidents, the radiological dose analysis bounds the plant response using conservative assumptions for transport of radionuclides into containment and containment leak rate. The power scenario 1c is not specifically considered for the control rod ejection accident to provide input for the radiological dose analyses. Since a loss of EDAS would open the RVVs concurrent with ejecting the rod, this event scenario is evaluated to demonstrate that fuel failure for the rod ejection accident is comparable to the inadvertent ECCS actuation.

As discussed above, scenario 1c (loss of EDAS) does not produce limiting consequences for maximum RCS pressure, maximum secondary side pressure, minimum critical heat flux rate, maximum fuel centerline temperature, or radiological dose criteria. Further consideration of loss of EDAS is not required for these acceptance criteria. The bounding containment pressure analysis for FSAR Chapter 6 considers the response to a secondary side break followed by ECCS valve opening due to loss of EDAS.

Loss of EDAS is not considered a separate, unique initiating event because of the redundancy and augmented requirements imposed on the design. Therefore, for the design basis events, if AC power is available, EDAS is assumed to be available.

C.3.4 Generic Evaluation of Loss of all Power at Event Initiation

In Table C-3, power scenario 2d is loss of AC power, loss of EDNS batteries, and loss of EDAS batteries at event initiation. All AC and DC power is lost. Therefore, the following plant response is expected:

- Reactor trip at event initiation due to loss of power;
- DHRS actuation due to loss of power, to open DHRS valves and close main feedwater and main steam isolation valves and backup isolation valves;
- Containment isolation valves actuate (to close) due to loss of power;
- Turbine trip at event initiation due to loss of power;
- Feedwater pumps and CVCS pumps stop due to loss of AC power to the pump motors;
- Pressurizer heaters turn off due to loss of power; and,
- ECCS valves actuate due to loss of EDAS - RVVs open immediately.
 - The inadvertent actuation block delays opening of the RRVs until the RCS pressure decreases to below the IAB release pressure.

The plant response to this loss of power scenario is the same as that in scenario 2a (loss of AC power) except that the ECCS valves actuate and the turbine trips at event initiation. Although the ECCS design changes produce a significantly different system response, the turbine trip has little impact on the margin for secondary pressure because the isolation valves close on loss of EDAS. With respect to ECCS actuation, the RVVs open immediately while the RRVs are closed until RCS pressure decreases to below the IAB release pressure. The availability of AC power does not change the timing for safety system actuation, ECCS operation, or margin for secondary pressure, so the system response is the same as scenario 1c (loss of EDAS at time zero). Therefore, the discussion in Section C.3.3 is applicable.

C.4 Summary of Loss of Power Cases for FSAR Chapter 15 Design Basis Events

Based on Table C-3 and the generic evaluations in Section C.3.3 and Section C.3.4, the following loss of power scenarios are considered for the FSAR Chapter 15 design basis events.

Scenarios considered on an event-specific basis:

1. AC power, EDNS and EDAS available - in this scenario all normal power sources are available.
 - a. Table C-3 Scenario 1a
2. AC power and EDAS available, EDNS not available during the transient.
 - a. Table C-3 Scenario 1b
3. Loss of AC power at event initiation, EDAS available; EDNS available
 - a. Table C-3 Scenario 2a
4. Loss of AC power at turbine trip, EDAS available; EDNS available
 - a. Table C-3 Scenario 3a

Enclosure 3:

Affidavit of Carrie Fosaaen, AF-132467

NuScale Power, LLC

AFFIDAVIT of Carrie Fosaaen

I, Carrie Fosaaen, state as follows:

- (1) I am the Senior Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.
- (2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:
 - (a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.
 - (b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.
 - (c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.
 - (d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.
 - (e) The information requested to be withheld consists of patentable ideas.
- (3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the method by which NuScale develops its Transient and Accident Analyses.

NuScale has performed significant research and evaluation to develop a basis for this method and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

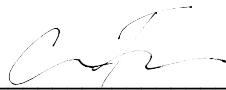
If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

- (4) The information sought to be withheld is in the enclosed report entitled Chapter 15, "Transient and Accident Analyses," Revision 0. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.
- (5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC §

552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

- (6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:
- (a) The information sought to be withheld is owned and has been held in confidence by NuScale.
 - (b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.
 - (c) The information is being transmitted to and received by the NRC in confidence.
 - (d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.
 - (e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 12/31/2022.



Carrie Fosaaen