September 21, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 371 (eRAI No. 9373) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 371 (eRAI No. 9373)," dated February 27, 2018

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9373 :

- 15.06.06-2

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9373

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NuScale Response to NRC Request for Additional Information eRAI No. 9373

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9373

Date of RAI Issue: 02/27/2018

NRC Question No.: 15.06.06-2

In accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 35, “Emergency Core Cooling,” a system that provides abundant emergency core cooling shall be provided and 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 ensure 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. The staff notes that the applicant requests an exemption in Part 7 of the design certification application from the electric power provisions of this GDC.

To meet the requirements mentioned above, as they relate to the emergency core cooling system (ECCS) providing abundant core cooling during the inadvertent ECCS actuation event (a design basis event described in FSAR Tier 2, Section 15.6.6), the accident analysis should be performed using an acceptable analytical method as discussed in the NuScale Design Specific Review Standard, Section 15.6.6.

In Final Safety Analysis Report (FSAR) Tier 2, Section 15.6.6.3.1, “Evaluation Model,” the applicant states that the loss of coolant accident (LOCA) evaluation model (EM), detailed in TR-0516-49422, "LOCA Evaluation Model," is conservatively used to analyze the spurious opening of a reactor pressure vessel (RPV) valve.

Furthermore, the applicant states in the same FSAR Tier 2 section that changes were applied to the LOCA EM to be consistent with modeling the event as an anticipate operational occurrence

(AOO). The applicant then lists the changes made to the LOCA EM as part of this FSAR Tier 2 analysis. The NRC staff recognizes that these changes to the LOCA EM combined with the LOCA EM itself constitute a single methodology; however, the staff notes that the applicant does not clearly define this as a methodology in this FSAR Tier 2 section. Furthermore, the NRC staff notes that the changes made to the LOCA EM, which are a part of the FSAR Tier 2, Section 15.6.6 methodology, are not described in enough detail in the FSAR for the staff to make a safety finding. For example, the applicant states that valve models for the reactor safety valves (RSVs), reactor vent valves (RVVs), and reactor recirculation valves (RRVs) are revised from normal operation to reflect a transient event initiation. The NRC staff does not understand what this means and cannot make a safety finding related to the DSRS acceptance criteria, which indicates that the applicant's analysis should use an acceptable analytical model. Lastly, the staff notes that the applicant does not justify applicability of the LOCA EM, with its modifications, to this FSAR Tier 2 event.

The staff requests the applicant to:

1) Clearly define in FSAR Tier 2, Section 15.6.6.3.1, the methodology used to analyze this event. The staff expects to see language clearly defining what constitutes the methodology used to anlayze this event, e.g. "The methodology used to analyze this event is the LOCA EM (TR-0516-49422) plus the following changes..."

2) Clearly describe in FSAR Tier 2, Section 15.6.6.3.1, in more detail than what currently exists, the changes made to the LOCA EM. For the NRC staff to complete its review of the methodology used to analyze this event, the docketed information should contain sufficient information that the NRC staff can use as basis for its safety findings. The level of detail should be consistent with that provided for other methodology reviews (e.g. LOCA EM topical report). The staff is not requesting the applicant to repeat the LOCA EM topical report information in the FSAR; however, the staff is requesting the applicant to provide sufficient detail on the changes made to the LOCA EM. For example, describe the RSV, RVV, and RRV model revisions in a level of detail consistent with the level of detail that would be presented in the LOCA EM topical report for valve models.

3) Clearly justify in FSAR Tier 2, Section 15.6.6.3.1, the applicability of the LOCA EM with its modifications to analyze this event. For example, justify why the RSV, RVV, and RRV valve models are appropriate for analyzing this event. As another example, justify why the proposed 95/95 limit is appropriate for this event.

NuScale Response:

FSAR Section 15.6.6.3.1 states, "Due to the phenomenological similarities to the LOCA pipe break events described in Section 15.6.5, the LOCA evaluation model is conservatively used in this analysis to evaluate the spurious opening of an RPV event." This statement has been revised to include a reference to Appendix B of the LOCA EM topical report (TR-0516-49422). The LOCA EM is described in the LOCA EM topical report, and Appendix B was added to provide the methodology used to analyze the inadvertent operation of the ECCS. Appendix B of the LOCA EM topical report was provided with RAI 15.06.06-2 of eRAI 9536 (Letter RAIO-0918-61859). The scope of LOCA EM topical report Appendix B is as follows:

- Regulatory requirements and classification of the inadvertent opening of an RPV valve (IORV)
- NPM design features important to the IORV event scenarios
- Development of the 95/95 MCHFR limit for IORV Analysis
- Description of the IORV NRELAP5 analytical model, and changes from the LOCA EM.
- Applicability for IORV analysis of NRELAP5 assessments against separate effects tests (SETs) and integral effects tests (IETs)
- Applicability evaluation determining the adequacy of NRELAP5 for NPM IORV analyses
- IORV analysis results and sensitivity studies

FSAR Section 15.6.6.3.1 was updated to add references to the LOCA EM topical report Appendix B and the details of the methodology were removed from the FSAR as shown in the page revisions provided with this response.

As part of the development of the LOCA topical report appendix material, the IORV analysis was revised to address modeling errors and inconsistencies between the IORV calculation and the LOCA EM. As a result of these corrections, slower depressurization rates were calculated for the inadvertent RVV transient cases, which resulted in increased margin to CHF. The updated analysis also demonstrated similar MCHFR results for the inadvertent RVV and RRV cases when considering the uncertainty in predicted CHF margin. Consequently, the limiting event analysis provided in FSAR Section 15.6.6 has been updated as shown in the revised FSAR pages provided with this response.

Impact on DCA:

FSAR Section 15.6.6, Table 15.6-15 through Table 15.6-17, and Figure 15.6-55 through 15.6-68 have been revised as described in the response above and as shown in the markup provided in this response.

2) Maximum cladding oxidation - The calculated maximum total oxidation of cladding shall not exceed 17 percent of the total cladding thickness before oxidation.

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.

4) Coolable geometry - Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5) Long-term cooling - After any calculated 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 and Table 15.6-14 demonstrate that the collapsed level remains above the top of the active fuel (Figure 15.6-45), MCHFR remains greater than the safety limit, and containment pressure and temperature remain within design limits. Therefore, acceptance criteria 1 through 4 are met.

The long-term cooling acceptance criterion 5 is also met. As discussed in Section 15.6.5.3.3, 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

RAI 15-2S1, RAI 15.06.06-2

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 NuScale design of ECCS is described in Section 6.3. The ECCS consists of three RVVs exiting the top of the RPV and two RRVs creating an opening to the RPV in the downcomer region above the core. Each ECCS valve includes an inadvertent actuation block (IAB) feature to reduce the frequency of inadvertent opening of the valve during power operation. Section 3.9.1 provides a description of possible failures of the ECCS valves, and concludes that the inadvertent opening of more than one ECCS valve is considered a beyond design basis event due to the ECCS valve IAB ~~device~~feature. Thus, the inadvertent operation of ECCS consists of the ~~spurious~~inadvertent opening of one RVV or one RRV. The failure of an ECCS valve to a partially open position was evaluated and determined not to be a credible initiating event.

RAI 15-2S1, RAI 15.06.06-2

~~An inadvertent ECCS signal coincident with a single failure of one of the IAB devices on an ECCS valve could result in opening of that valve at RCS operating pressure. A mechanical failure of an ECCS valve could also result in the opening of a single ECCS valve, where the valve opening is considered the initiating event. The mechanical failure of an ECCS valve bounds the inadvertent ECCS actuation signal coincident with a single failure of an IAB device because it takes two failures for an ECCS valve to open due to an inadvertent ECCS actuation signal. Therefore, the event analyzed for an inadvertent operation of ECCS is a mechanical failure of an RVV or RRV.~~An ECCS valve will open when the force from the pressure in the valve control chamber is less than the opening force of the main valve spring plus the pressure force on the underside of the disc. Depressurization of the control chamber occurs when coolant is lost from the control chamber at a rate greater than is made up through the control chamber orifice that connects to the RCS. The principle method to depressurize the control chamber is by opening the associated ECCS trip valve, which drains the RCS fluid in the chamber to the containment unless it is blocked by the IAB function. The control chamber fluid can also be drained as a result of a mechanical failure of the valve assembly.

RAI 15-2S1

If an ECCS trip valve opens, the IAB feature will stop the loss of fluid from the control chamber by blocking the trip line flow path if the differential pressure between the RCS and containment is greater than the IAB threshold. The threshold is determined by the opening force of a spring internal to the IAB device. The flowpath from the control chamber through the trip line is blocked by a rod in the IAB arming valve moving into its seat. The IAB is actuated by the differential pressure between the RCS on one side of the rod and the pressure in the trip line. When a trip valve opens, fluid drains into containment and the pressure in the trip line decreases, which creates a large differential pressure across the rod. When the force from the differential pressure across the rod is greater than the IAB spring force, the rod moves into its seat and blocks the control chamber fluid from exiting through the trip line. The pressure in the control chamber is maintained by fluid entering through the orifice from the RCS, which prevents the ECCS valve from opening.

RAI 15-2S1

The IAB function is a passive sub-component feature of an ECCS valve as discussed in Section 15.0.0.5. A failure of one of the IAB features on an ECCS valve could result in the opening of a single ECCS valve if an ECCS actuation signal is present or DC power (EDSS) is not available (causes trip valve to fail open). Since the IAB is a passive component, failure of this device is an initiating event. Depressurization of the valve control chamber by a mechanical failure of the valve assembly is a similar initiating event. The mechanical failure results in an ECCS valve opening independent of the status of an ECCS signal or DC power availability. Single active failures, discussed in 15.6.6.2, were considered in each of these events but did not result in more limiting results for the acceptance criteria. The limiting event analyzed is a mechanical failure of the valve that depressurizes the control chamber at operating pressure.

The spurious opening of a single ECCS valve 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 opening of an RPV valve 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

RAI 15.06.06-2

Sensitivity analyses are performed to identify the limiting event for the spurious operation of an ECCS valve. The limiting initiating event for this transient is the inadvertent opening of one R~~V~~RV. However, it is of note that the resulting MCHFR is similar to that of the inadvertent opening of an RVV. The sequence of events is provided in Table 15.6-15. Unless otherwise specified, the analysis of an inadvertent opening of an R~~V~~RV 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 Core and System Performance

15.6.6.3.1 Evaluation Model

The thermal hydraulic response to an inadvertent opening of an ECCS valve event exhibits unique transient progression relative to other AOO events analyzed for the NPM. This progression is divided into two phases:

RAI 15.06.06-2

- The first phase is initiated with a spurious opening of an RPV valve (RSV, RVV, or RRV) that results in a blowdown of the RCS into the containment vessel. This breach can be characterized as a steam region breach (i.e., opening of an RSV or RVV) or a liquid region breach (i.e., opening of an RRV). For the limiting event of an inadvertent opening of an R~~V~~RV, this phase ends when the remaining ECCS valves are actuated as designed by the MPS.
- The second phase begins with ECCS actuation through designed MPS operation and ends when the NPM reaches a safe, stable condition and transitions to long-term ECCS cooling.

These two phases align with the two phases of the LOCA transient progression for the NPM. The LOCA evaluation model and Reference 15.6-1 have:

- identified and ranked important phenomena which occur during these transient phases for the NPM,
- assessed NRELPA5 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 (See Section 15.6.5).

RAI 15.06.06-2

Due to the phenomenological similarities to the LOCA pipe break events described in Section 15.6.5, the LOCA evaluation model, with modifications, is conservatively used in this analysis to evaluate the spurious opening of an RPV valve event, consistent with Appendix B of Reference 15.6-1. ~~Following are examples of some modeling assumptions of the LOCA evaluation model that provide additional conservatism when applied to this AOO event:~~

RAI 15.06.06-2

- ~~DHRS actuation is not credited in the assessment to conservatively reduce the heat removed from the RPV.~~
- ~~Cross-flow between the hot and average core assemblies is conservatively not modeled to prevent hot and cool fluid streams from mixing.~~
- ~~Core power profiles are designed to bound potential core designs as well as core power transients such as Xenon.~~
- ~~A CHF correlation is implemented that conservatively models reverse flow conditions where stagnant flow values are applied when significant downward mass fluxes exist.~~
- ~~The 1973 ANS decay heat standard with a 1.2 multiplier for actinide decay heat is applied, which is more conservative than the decay heat models used in other non-LOCA analyses, as shown in Section 15.0.0.6.2.~~
- ~~Conservative fuel properties are applied that maximize fuel pellet stored energy and energy release.~~
- ~~Core bypass flow is set to be 8.5 percent of the total system flow to be consistent with the subchannel analysis methodology (Reference 15.6-3).~~

RAI 15.06.06-2

~~The following changes are applied to the LOCA EM to be consistent with modeling this event as an AOO.~~

RAI 15.06.06-2

- ~~Valve models for the RSVs, RVVs, and RRVs are revised from normal operation to reflect a transient event initiation.~~
- ~~Initial conditions are biased within the minimum and maximum ranges to determine the limiting set of initial conditions for the parameter of interest.~~
- ~~The LOCA EM adds an additional bias to UO2 thermal conductivity and heat capacity (Reference 15.6-1). This additional bias is removed, which is consistent with AOO events.~~
- ~~Bounding axial power shapes used in subchannel analysis methodology are applied, which is consistent with other AOO events.~~

RAI 15.06.06-2

- ~~The CHF correlation applied in the LOCA evaluation model discussed in Reference 15.6-1 is modified to evaluate CHF against the 95/95 CHFR acceptance criterion of an AOO. The Extended Hench-Levy CHF correlation for high mass fluxes (≥135.6 kg/m^2-s) and the Griffith-Zuber model for low mass fluxes (≤ 67.8 kg/m^2-s), with interpolation used between the two correlations in the intermediate mass flux range is applied. KATHY NuFuel HTP2TM CHF data from Reference 15.6-2 is applied for high flow conditions and results in a 95/95 design limit of 1.122 CHFR with uncertainty. It is noted that the sensitivity cases for this analysis show MCHFR occurring while in the high flow correlation range shortly after event initiation as reactor power is still elevated at this time. Therefore, the analysis acceptance criterion is reported against a 95/95 CHFR design limit of 1.122.~~

RAI 15.06.06-2

~~No KATHY NuFuel HTP2TM CHF data exists for the low flow conditions, therefore STERN CHF data is applied for low flow conditions and results in a bounding design limit of 1.37 CHFR, with uncertainty. It is noted that the sensitivity cases evaluated in this analysis show significant margin to the 1.37 CHFR design limit as reactor power has reduced to decay heat by the time the low flow condition is reached.~~

- ~~The end-of-cycle (EOC) gap conductance (maximum bounding value) is applied because it is more limiting for MCHFR due to increasing fuel pin heat flux during the initial blowdown phase of the transient.~~
- ~~A radial peaking analytical limit of 1.504 is applied to the hot channel.~~

15.6.6.3.2 Input Parameters and Initial Conditions

RAI 15.06.06-2

The input parameters and initial conditions used in the evaluation of a spurious opening of an R~~V~~RV are selected to provide a conservative calculation and to minimize~~aximize~~ 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 a spurious opening of an R~~V~~RV.

Table 15.6-16 provides inputs and assumptions. 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.

RAI 15.06.06-2

- ~~The nominal RCS average temperature is biased high such that the initial hot leg temperature is 595 F.~~The RCS average temperature is biased high to 555F, per Table 15.0-6, to maximize initial RCS energy.~~This places the RCS closer to saturation at the time of the event initiation, leading to more void generation in the core during the initial blowdown.~~

RAI 15.06.06-2

- RCS flow is biased lowhigh to minimize MCHFR conditions in the core.~~maximize core inlet temperature when core outlet temperature is fixed.~~

RAI 15.06.06-2

- Pressurizer pressure ~~is biased~~ was analyzed for two conditions: biased high and biased low. The limiting case was found with pressure biased high.~~low to place the RCS closer to saturation at the time of event initiation, leading to more void generation in the core during the initial blowdown.~~

RAI 15.06.06-2

- The initial pressurizer level is biased high to 68 percent. This pressurizer level resulted in a slightly more limiting MCHFR than an initial pressurizer level biased low.~~set at 64 percent. The pressurizer level is increased by half of the level measurement uncertainty of eight percent because sensitivity cases found that a pressurizer level at 64 percent maximizes the initial RCS depressurization, leading to a lower minimum CHFR, and delaying the onset of two-phase choked flow at the lifted RVV.~~
- Beginning-of-cycle core parameters for Doppler temperature coefficient and moderator temperature coefficient are used for calculating scram worth.
- The following conservative scram characteristics are assumed.
  - The maximum time delay from the MPS signal to control rod movement (scram) is applied.
  - The most reactive control rod is assumed to be stuck in the fully withdrawn position.
  - The bounding control rod drop rate, shown in Figure 15.0-2, is applied.
- Beginning-of-cycle kinetic parameters with an additional 6 percent biasing are used in order to prolong the fission power transient, consistent with Reference 15.6-1.
- Minimal reactivity feedback coefficents are conservatively applied in order to minimize negative feedback, consistent with Reference 15.6-1.

RAI 15.06.06-2

- A bounding ~~bottom~~middle peaked axial power shape is applied to maximize the highest axial peaking factor~~the average volumetric coolant temperature in the core, yielding more void generation during the initial transient blowdown~~. Sensitivity studies confirm that ~~the bottom peak~~this shape is limiting.
- An energy deposition factor of 1.0 is implemented such that all the core power is conservatively deposited in the fuel, consistent with Reference 15.6-1.
- The following loss of power scenarios are considered.

RAI 15.06.06-2

  - No loss of power - In this scenario, all MPS and ESFs actuate as designed. The ECCS valve opening is dependent on both the MPS ECCS actuation setpoints on high CNV water level and low RPV riser level, and the IAB release pressure setpoint.~~Sensitivity studies show that this scenario results in the limiting MCHFR. The all power available scenario is shown to yield the largest decrease in hot channel inlet flow at the time MCHFR occurs.~~

~~Reactor power also remains elevated during the time of MCHFR as a reactor trip has not yet occurred.~~

– 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 and DHRS actuation occur after a 60-second delay following a loss of normal AC power. ECCS actuation occurs after a 24-hour delay following a loss of normal AC power. However, because DC power is still available, the MPS can still actuate a reactor trip containment isolation, and DHRS, earlier if a separate actuation limit is reached. However, DHRS is not credited in this analysis. 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. This scenario is non-limiting for the reasons described above.

– Loss of the normal DC power system (EDNS) and normal AC - Power to the reactor trip breakers is provided via the EDNS, so the primary difference to a loss of normal AC power is that the reactor trip will occur sooner. This scenario is non-limiting for the reasons described above.

RAI 15-2S1, RAI 15.06.06-2

– Loss of the highly reliable DC power system (EDSS), EDNS, and normal AC - This scenario results in an immediate actuation of the reactor trip system, DHRS (although not credited in the analysis), ~~the 24-hour timer for the ECCS valves,~~ and containment isolation. As power to the MPS is lost, the ECCS valve opening is dependent only on the IAB pressure release ~~setpoint~~threshold. This scenario is ~~non-limiting for the reasons described above~~identified as limiting although it is similar to the power available scenarios due to the rapid nature of the transient.

RAI 15-2S1, RAI 15.06.06-2

- ~~Single failure evaluation of a single RVV to open, a single RRV to open, and failure of one ECCS division (one RVV and one RRV) to open was performed to determine the most conservative scenario. The evaluation showed that the single failure cases have no impact on MCHFR or other acceptance criteria evaluated in this analysis. Therefore, the scenario of no single failure is applied in this analysis.~~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 limiting MCHFR occurs within the first one second of the RRV opening. No failures occur in a timeframe that affect this result. The evaluation showed that the single failure cases have no adverse impact on the limiting MCHFR or other acceptance criteria evaluated in this analysis. Therefore, the scenario with no single failure is limiting for this analysis.

RAI 15-2S1

- The failure modes that could lead to a partial opening of an ECCS valve were characterized as having a remote probability of occurrence or were 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 Regulatory Guide 1.105.
- No operator action is credited.

15.6.6.3.3 Results

RAI 15.06.06-1, RAI 15.06.06-2

Figure 15.6-55 to Figure 15.6-68 show the system response to an inadvertent RVV opening event. Table 15.6-17 contains the results of the event. The limiting case is initiated by a spurious opening of an R~~V~~RV. Sensitivity analysis show that the limiting scenario has ~~all power available and~~a loss of normal AC and DC power and no single failure occurs.

RAI 15.06.06-2

Upon the spurious R~~V~~RV opening, the large blowdown of the RCS into the containment causes rapid depressurization of the RCS and rapid pressurization of the containment. Spurious R~~V~~RV flow is shown in Figure 15.6-55 and Figure 15.6-56. The assumed loss of AC and DC power at time zero result in an immediate reactor scram, secondary system isolation, and DHRS actuation. However, DHRS operation is conservatively not credited in this analysis. The RCS and containment pressures are shown in Figure 15.6-57. The high containment pressure analytical limit is reached shortly after event initiation.~~The MPS signal actuates control rod insertion, secondary system isolation, and DHRS actuation. However, DHRS operation is conservatively not credited in this analysis.~~

RAI 15.06.06-1

The rapid RCS depressurization causes voiding in the core and a momentary decrease in RCS flow (Figure 15.6-58 and Figure 15.6-59), leading to a reduction in CHFR (Figure 15.6-67 and Figure 15.6-68). Reactor power decreases during this time due to control rod insertion and negative void feedback, as seen in Figure 15.6-60. Following the occurrence of transient MCHFR (Figure 15.6-67), a temporary increase in RCS flow is observed due to the increased density gradient from voiding in the riser (Figure 15.6-58).

RAI 15.06.06-2

The isolation of the secondary system following the ~~high containment pressure signal~~loss of normal AC and DC power at transient initiation causes an increase in steam generator pressure, as seen in Figure 15.6-61. 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 a spurious opening of an RPV valve event.

RAI 15.06.06-2

As primary coolant is released to the containment through the open R~~V~~RV, the inventory level inside the containment increases (Figure 15.6-62). ~~until the level~~

~~reaches the high containment level ECCS actuation limit of 221 inches. This generates the MPS ECCS actuation signal. Because~~As the RPV continues to depressurize, the differential pressure between the RPV and containment ~~has already~~ dropsped below the IAB threshold pressure, allowing the ECCS valves ~~immediately~~ to open. Pressure and temperature inside the RPV continue a gradual downward trend, as shown in Figure 15.6-57, Figure 15.6-63, and Figure 15.6-64.

After the remaining ECCS valves open and pressure equalizes across the RRVs, liquid coolant from the containment begins to flow into the RPV downcomer region. This establishes a two phase natural circulation loop 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 has been established and RCS pressure and temperature continues to decrease. The module remains in a safe condition with liquid level maintained above the top of the core through the entire transient. The fuel volume average temperature is shown in Figure 15.6-65 and fuel cladding temperature is shown in Figure 15.6-66.

The MPS is credited to protect the module in the event of an inadvertent opening of an RVV by the following MPS signals:

- high containment pressure, and
- high containment water level

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

Section 15.0.3 provides the radiological consequences for the NuScale infrequent events and postulated accidents. Radiological consequence analyses are not required for AOOs. Section 15.0.3 also presents the design basis source term methodology and the radiological consequences of the Category 2 maximum hypothetical accident. The inadvertent opening of an RPV valve does not result in fuel failure, therefore the design basis source term 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 the NuScale Power Plant design meets them, are listed below. Table 15.6-17 provides the results of the limiting scenario of a spurious opening of an RVV.

1) Fuel cladding integrity shall be maintained by ensuring that minimum DNBR remains above the 95/95 DNBR limit. Minimum critical heat flux ratio (MCHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

RAI 15.06.06-1, RAI 15.06.06-2

The fuel integrity is not challenged by the ~~spurious~~inadvertent opening of the ~~RVV~~RRV. The fuel temperatures decrease upon the reactor trip, as shown in Figure 15.6-65 and Figure 15.6-66, and the water level remains above the top of the active fuel, as shown in Figure 15.6-62. The MCHFR ~~is 1.240,~~ is above the acceptance criterion as shown in Table 15.6-17 and as shown in Figure 15.6-67 and Figure 15.6-68~~, which is greater than the 95/95 limit of 1.122~~. As noted in Section 15.6.6.3.1, MCHFR occurs in the high flow correlation range shortly after event initiation as reactor power is still elevated at this time.

2) RCS pressure should be maintained below 110 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2310 psia.

   The RCS pressure is below the acceptance criterion, as shown in Table 15.6-17.

3) The main steam pressure should be maintained below 110 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2310 psia.

   The main steam pressure, presented as steam generator pressure, is below the acceptance criterion, as shown in Table 15.6-17.

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 opening of an RPV valve is similar to the response of the NPM following a LOCA. The long-term cooling analysis, results and conclusions are discussed in Section 15.6.5.

15.6.7 References

15.6-1 NuScale Power, LLC, Topical Report, "LOCA Evaluation Model," TR-0516-49422, Rev. 0.

15.6-2 NuScale Power, LLC, Topical Report, "NuScale Power Critical Heat Flux Correlations," TR-0116-21012, Rev. 1.

15.6-3 NuScale Power, LLC, Topical Report, "Subchannel Analysis Methodology," TR-0915-17564, Rev. 1.

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Table 15.6-15: Inadvertent Operation of an Emergency Core Cooling System Valve - Sequence of Events

Event	Time (s)*
R~~V~~RV opens	0
Loss of normal AC and DC power~~High containment pressure signal (9.5 psia)~~	0~~.5~~
Control Rods begin to fall~~Low pressurizer pressure signal (1600 psia)~~	0~~.5~~
Minimum CHFR occurs	0.5~~7~~
Control rods fully inserted into core~~Reactor trip signal~~	2.3~~5~~
Remaining ECCS valves open~~Containment isolation signal~~	50~~2.5~~
Natural circulation from containment to reactor pressure vessel is established~~Reactor trip~~	483~~4.5~~
Peak steam generator pressure is reached~~Containment isolation~~	490~~4.5~~
Minimum collapsed liquid level above the core~~Remaining ECCS valves open~~	630~~759.0~~
~~Natural circulation from containment to reactor pressure vessel is established~~	~~1369.0~~

*Time rounded to the nearest tenth of a second.

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Table 15.6-16: Inadvertent Operation of an Emergency Core Cooling System Valve - Inputs

Description	Units	Nominal	Analyzed Value
Core power	MWt	160	163.2 (102%)
Pressurizer pressure	psia	1850	1920~~1780~~
Pressurizer level	%	60	68~~4~~
Reactor Coolant System Flow	lbm/sec	See Table 15.0-6	1179~~1477~~

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Table 15.6-17: Inadvertent Operation of an Emergency Core Cooling System Valve - Results*

Parameter	Acceptance Criteria	Value
Peak reactor pressure	≤2310 psia	1936~~796~~ psia
Peak steam generator pressure	≤2310 psia	588~~1106~~ psia
MCHFR	1.13	1.41
Fuel cladding integrity maintained	Yes	Yes
Generate more serious plant condition?	No	No

*Values increased to the whole number.

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Figure 15.6-55: Inadvertent Operation of an Emergency Core Cooling System Valve – Spurious Reactor Vent Valve Flow

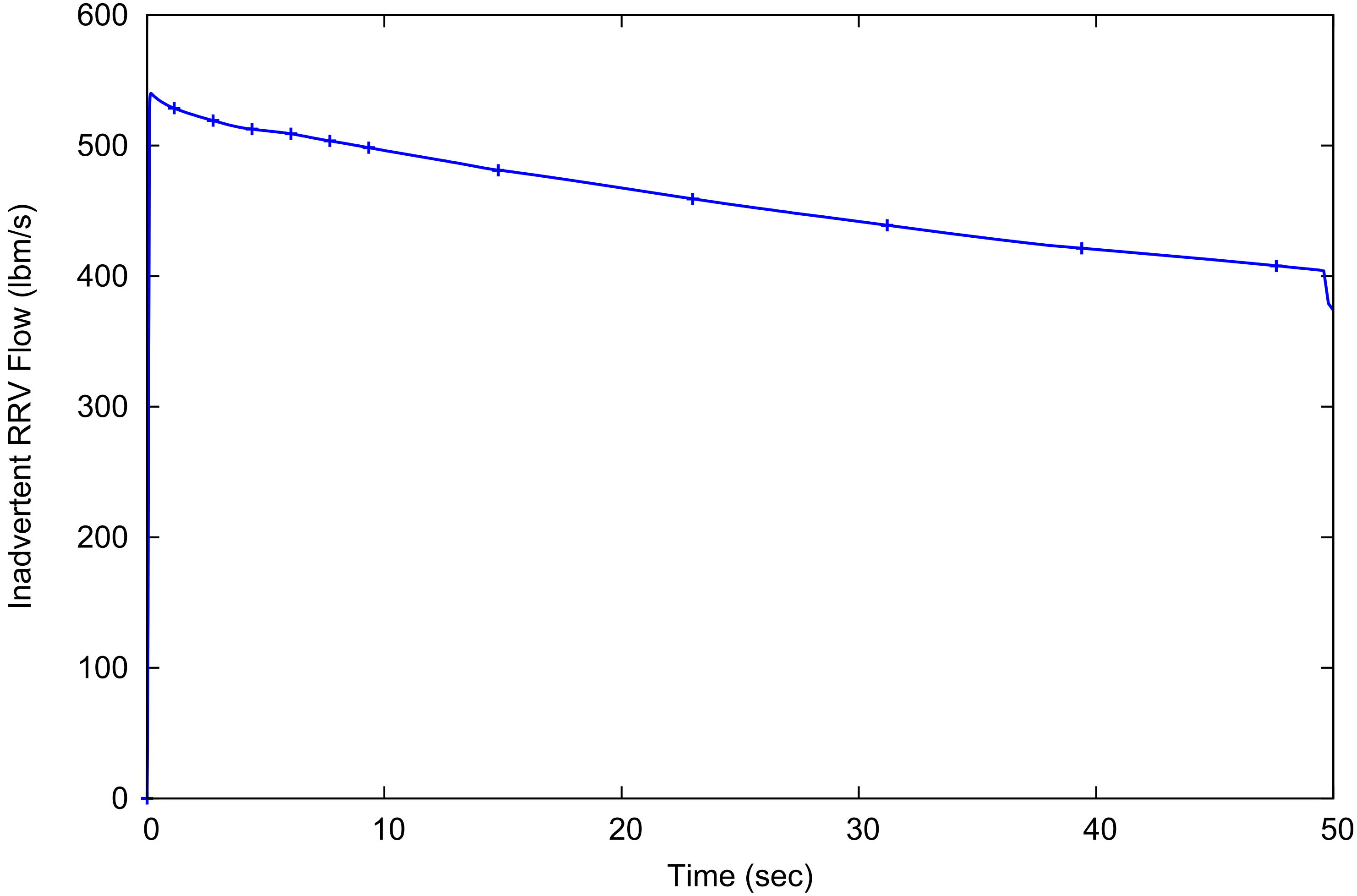

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Figure 15.6-56: Inadvertent Operation of an Emergency Core Cooling System Valve – Spurious Reactor Vent Valve Flow

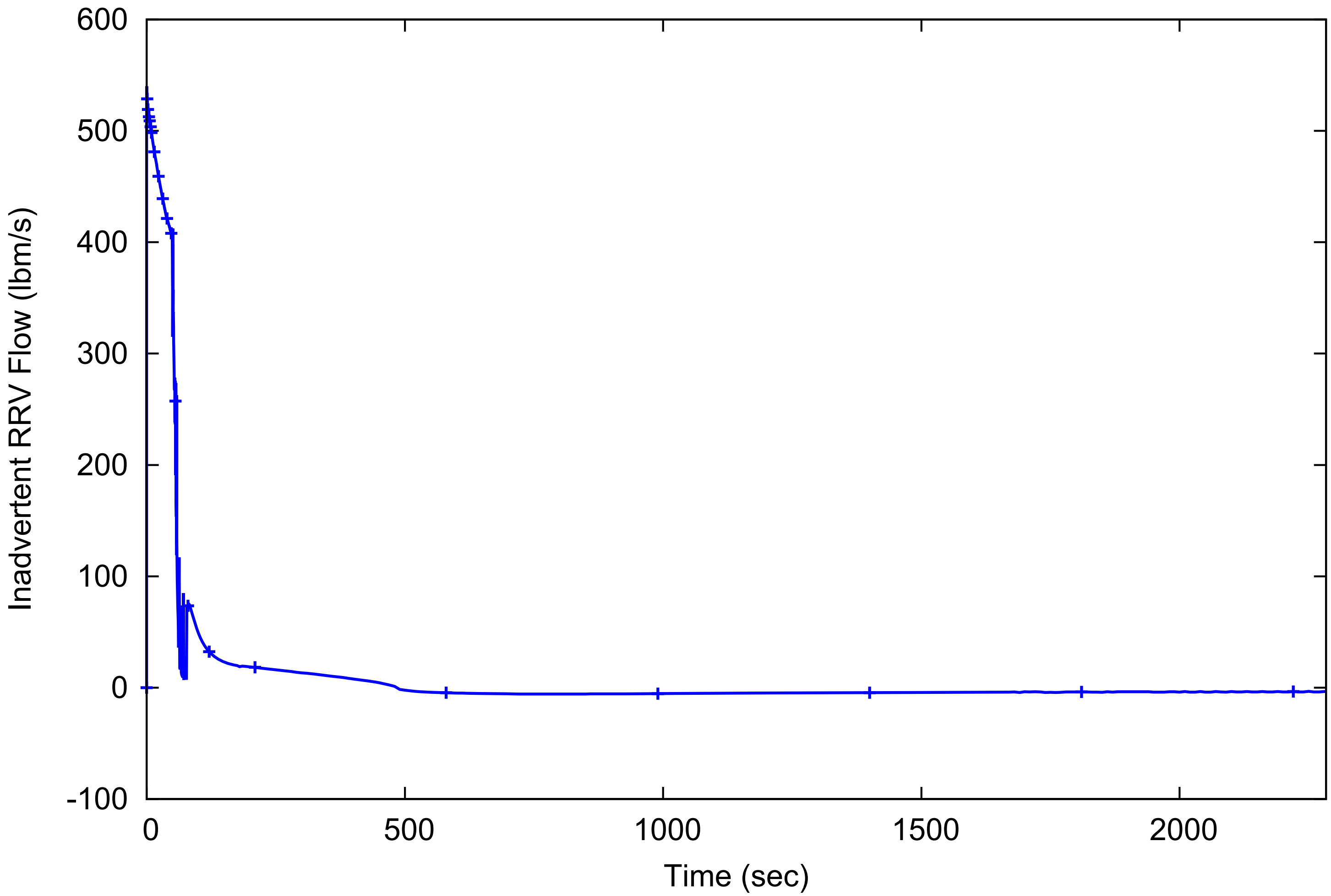

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Figure 15.6-57: Inadvertent Operation of an Emergency Core Cooling System Valve – Pressures

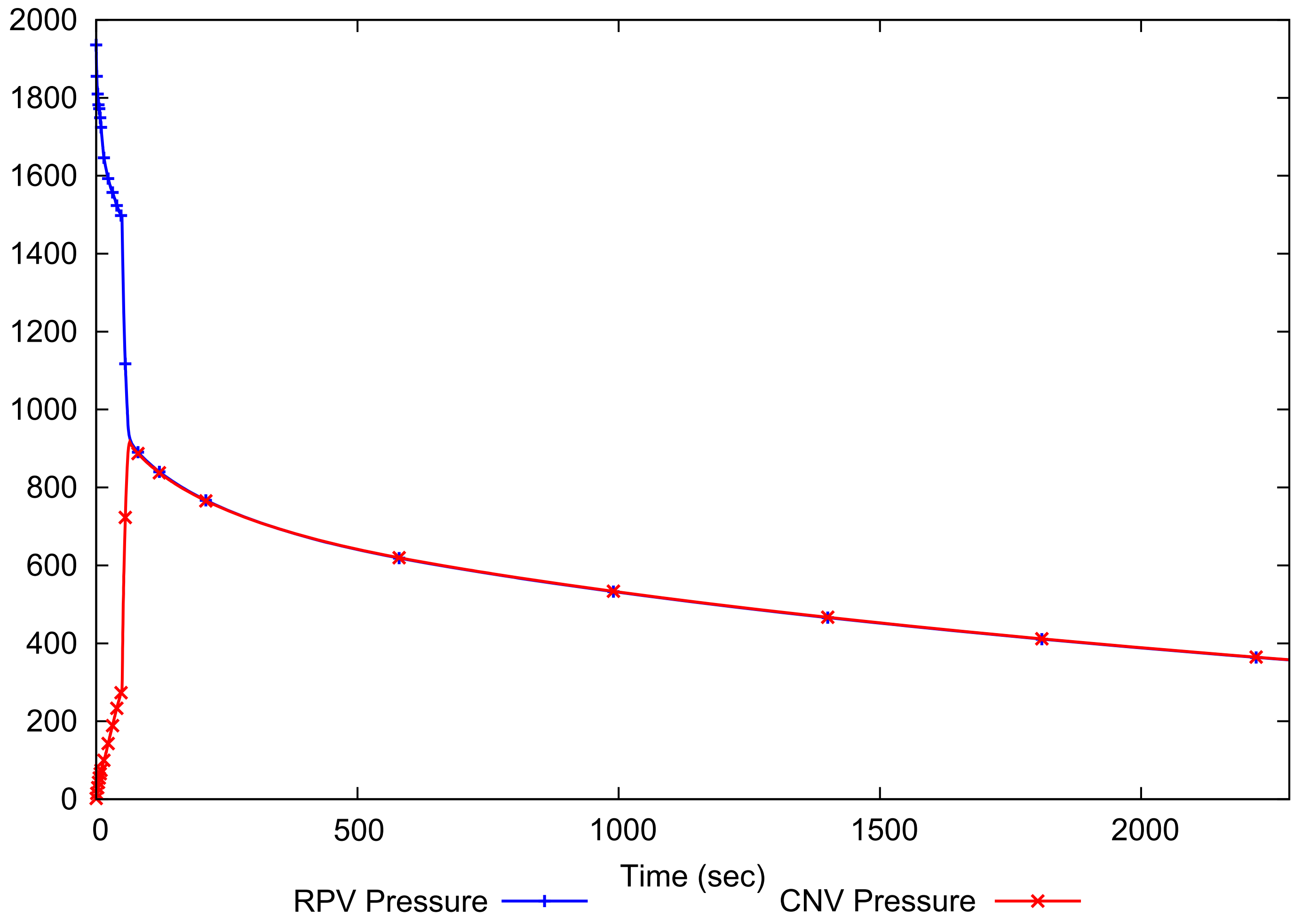

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Figure 15.6-58: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Flow

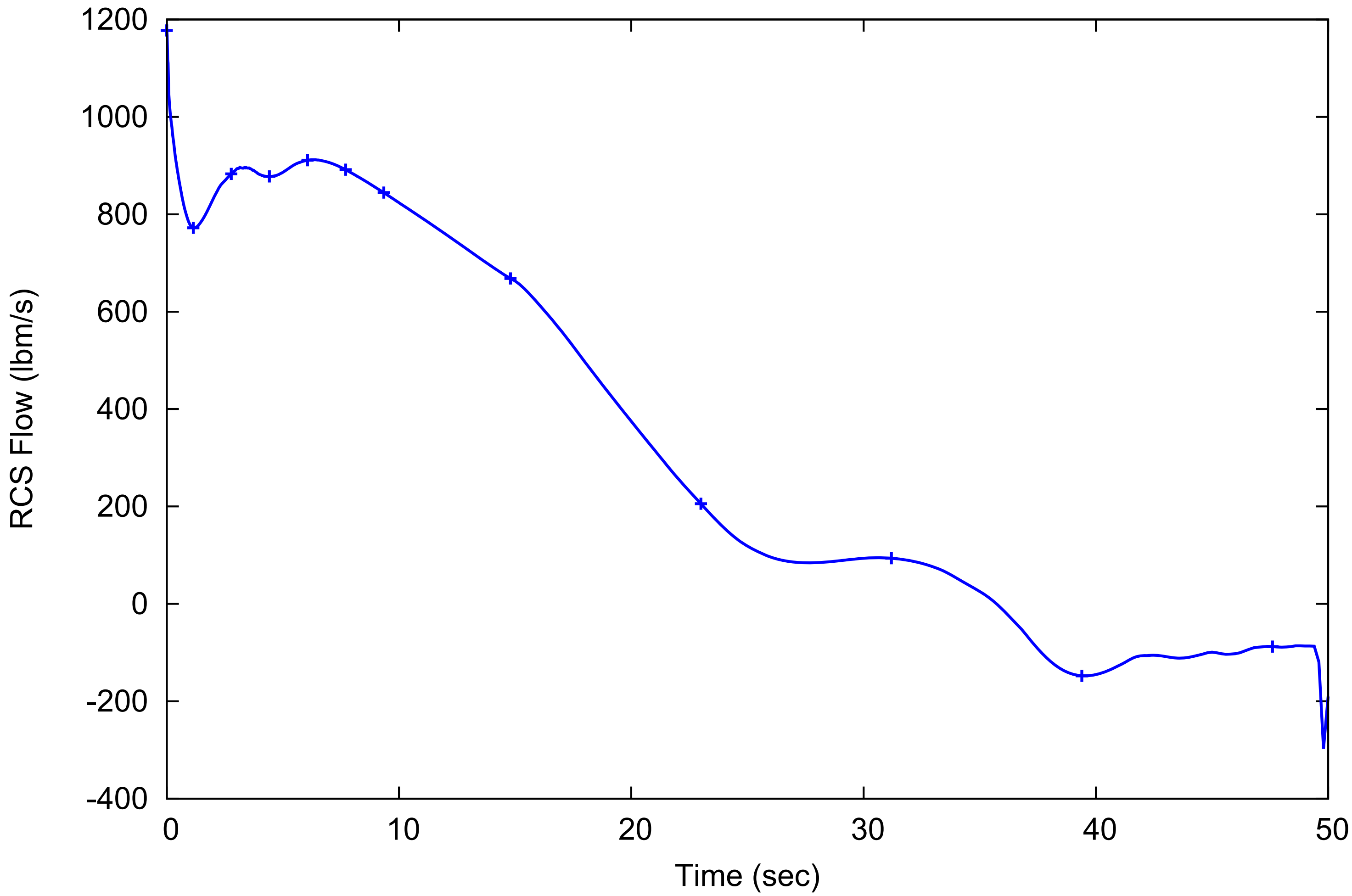

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Figure 15.6-59: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Flow

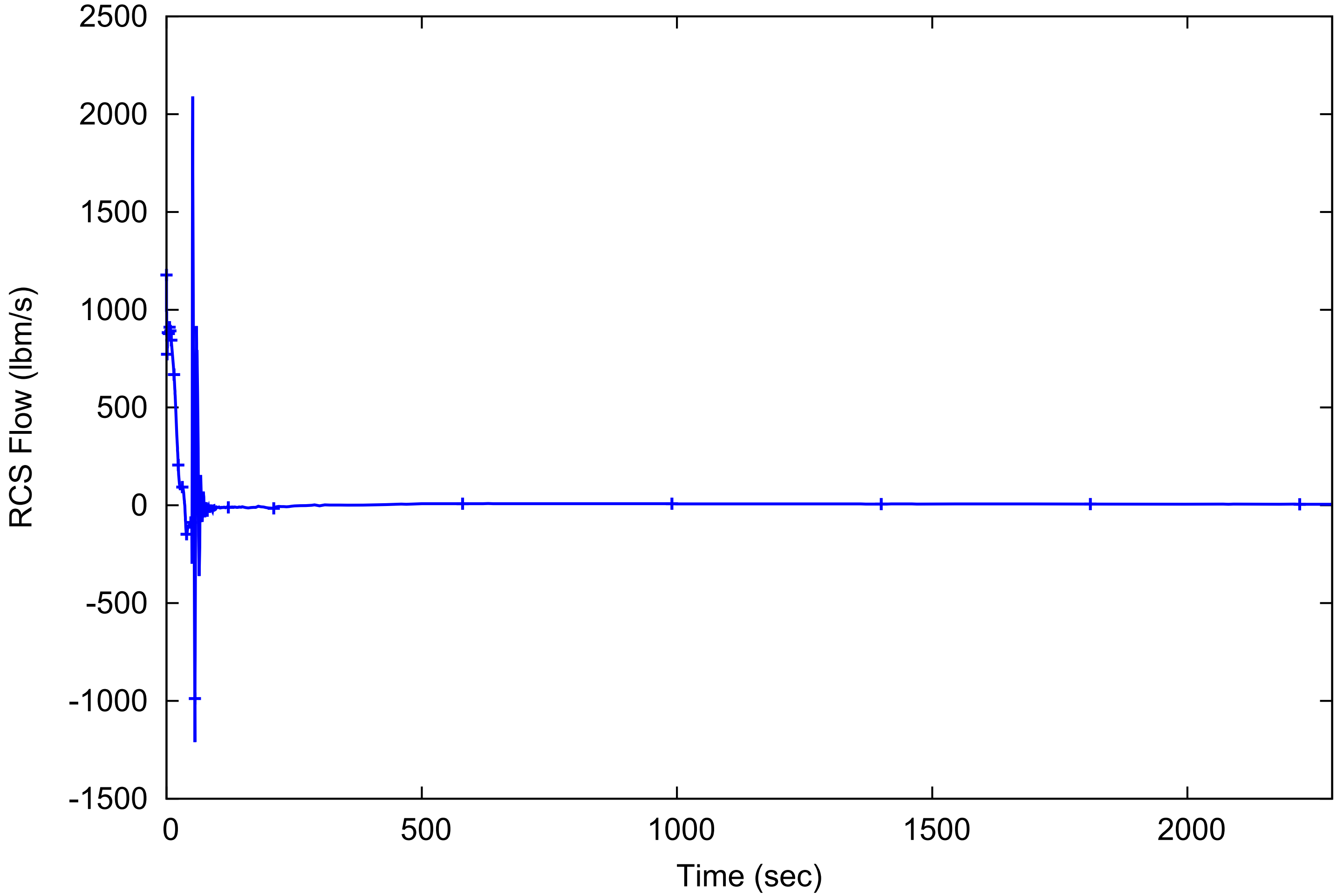

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Figure 15.6-60: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Power

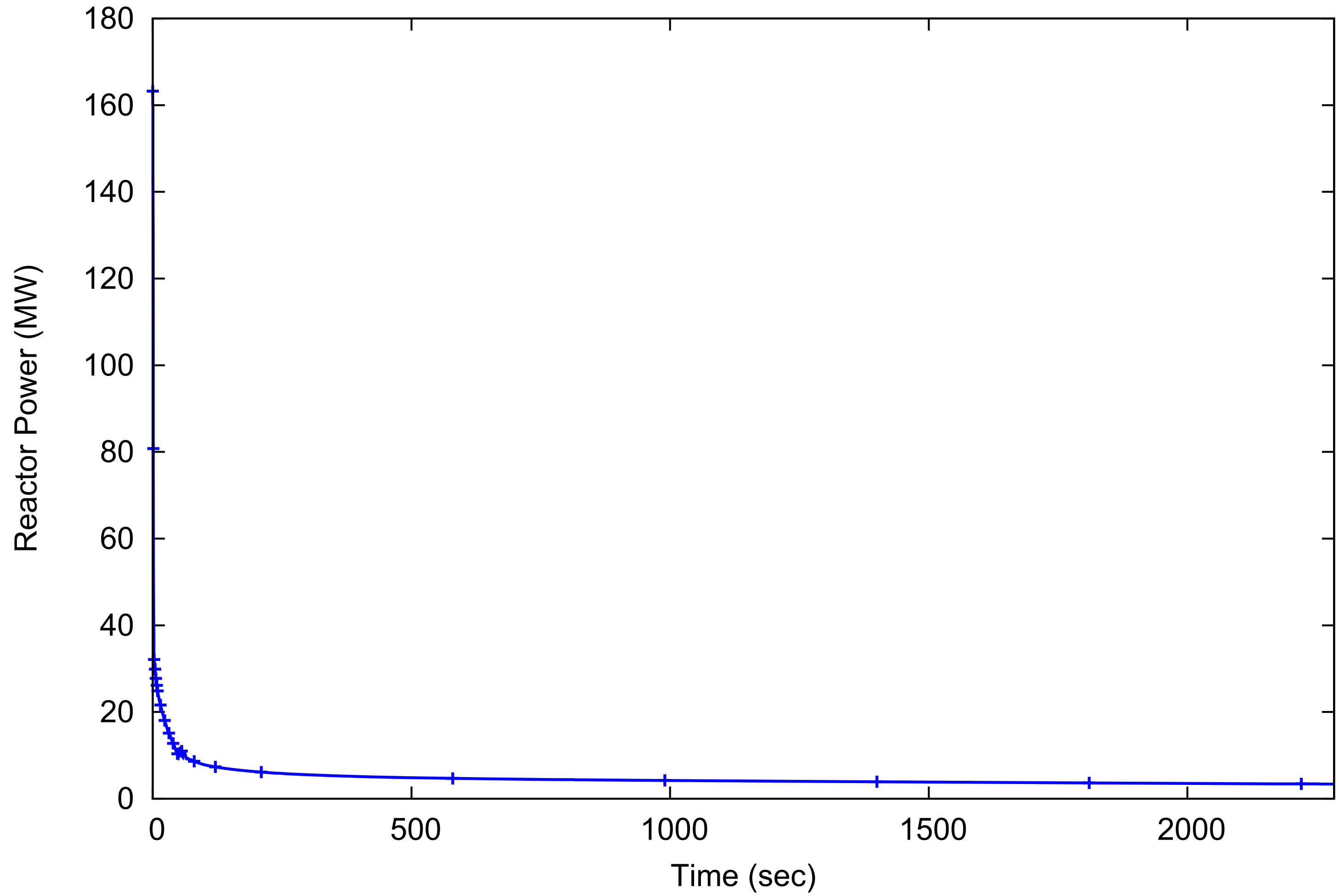

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Figure 15.6-61: Inadvertent Operation of an Emergency Core Cooling System Valve – Steam Generator Pressure

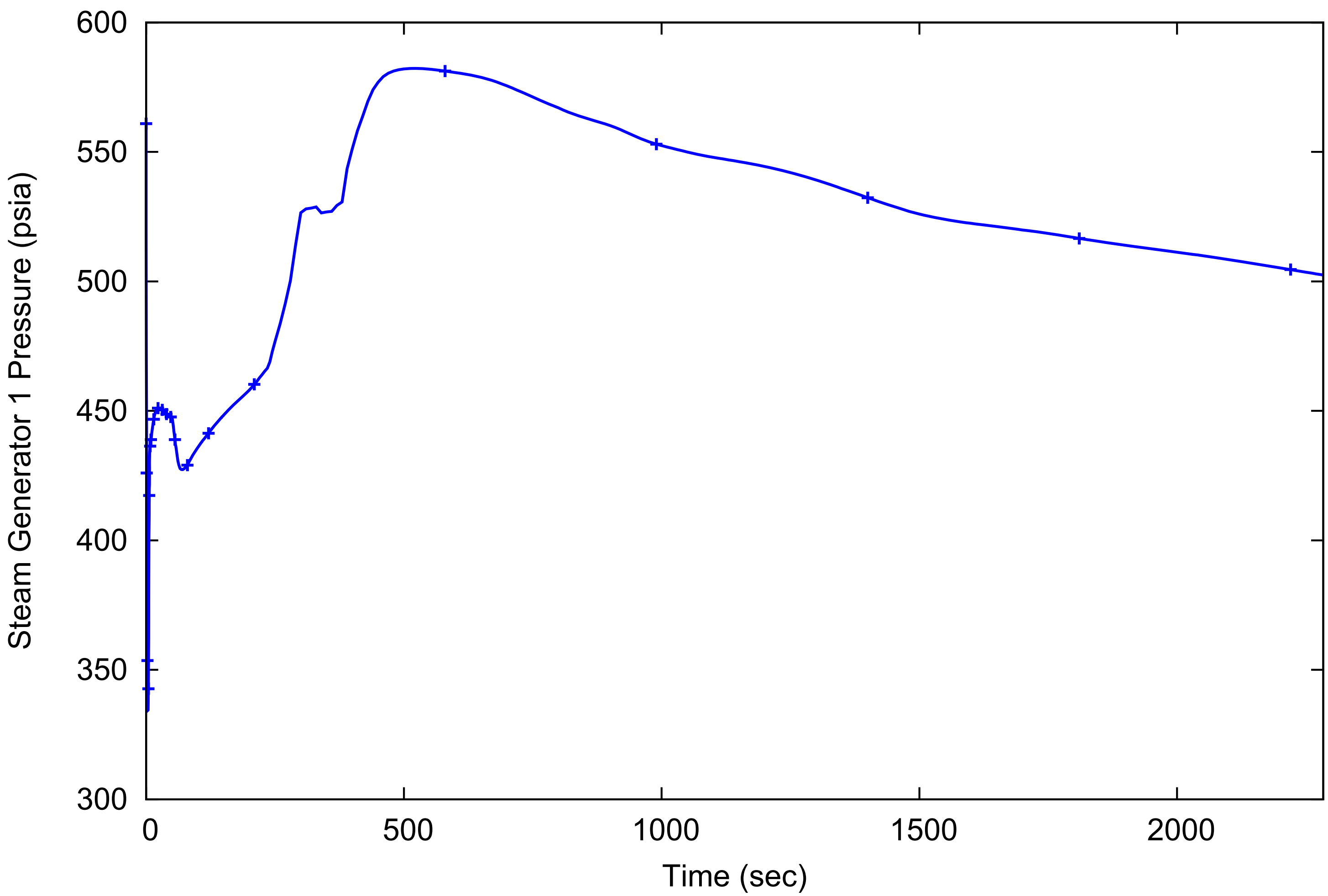

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Figure 15.6-62: Inadvertent Operation of an Emergency Core Cooling System Valve – Collapsed Liquid Level Above Active Core

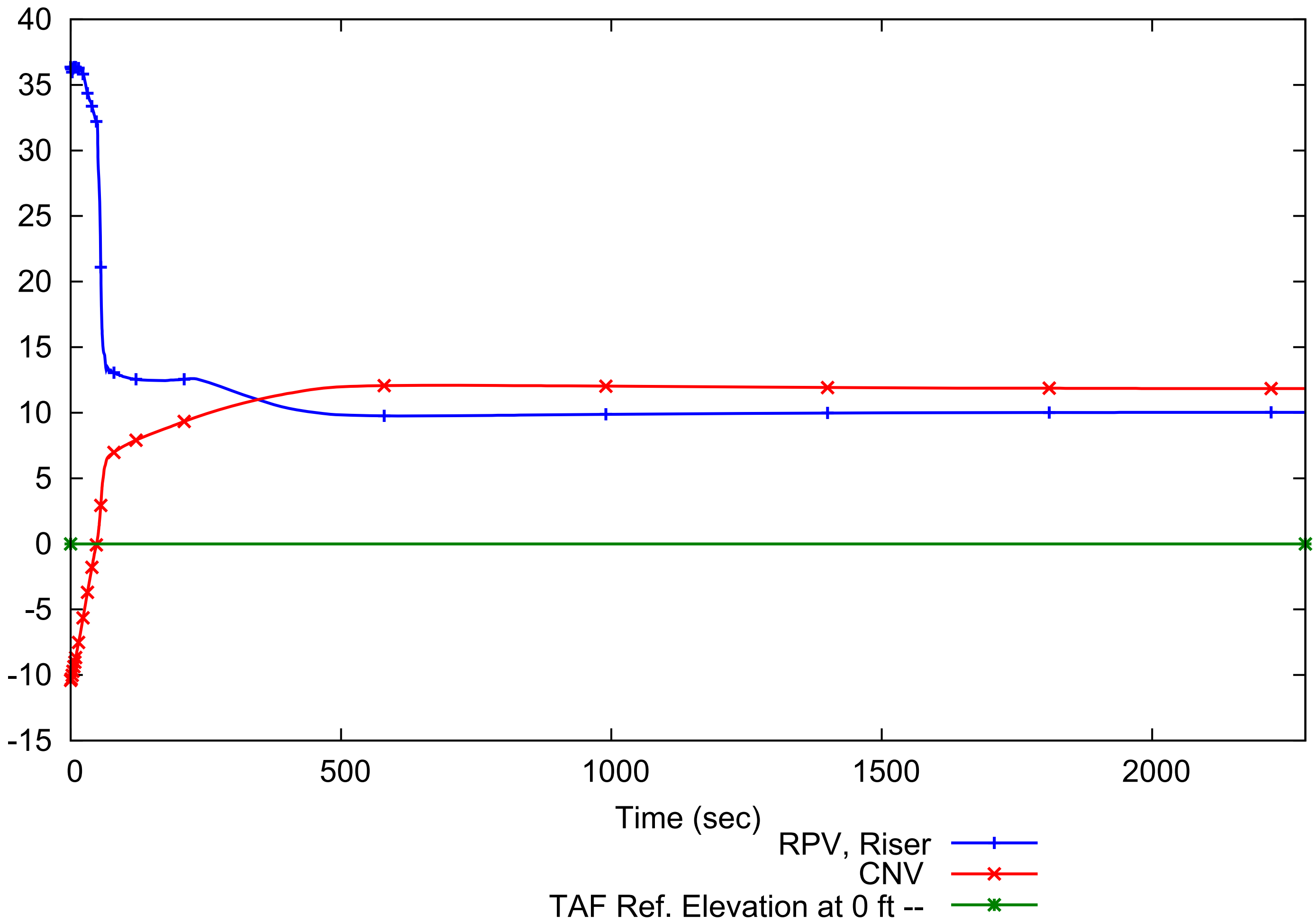

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Figure 15.6-63: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Temperature

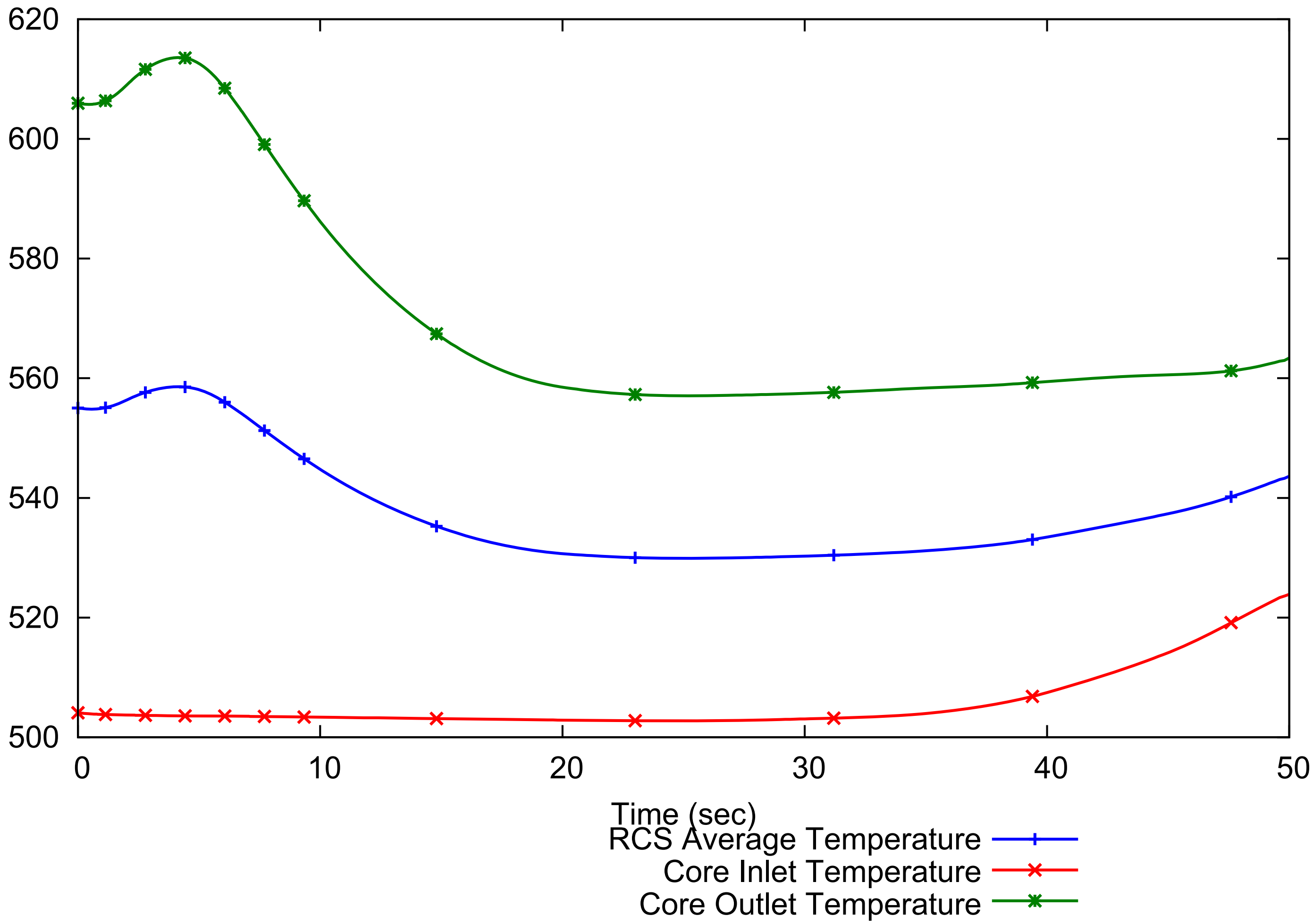

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Figure 15.6-64: Inadvertent Operation of an Emergency Core Cooling System Valve – Reactor Coolant System Temperature

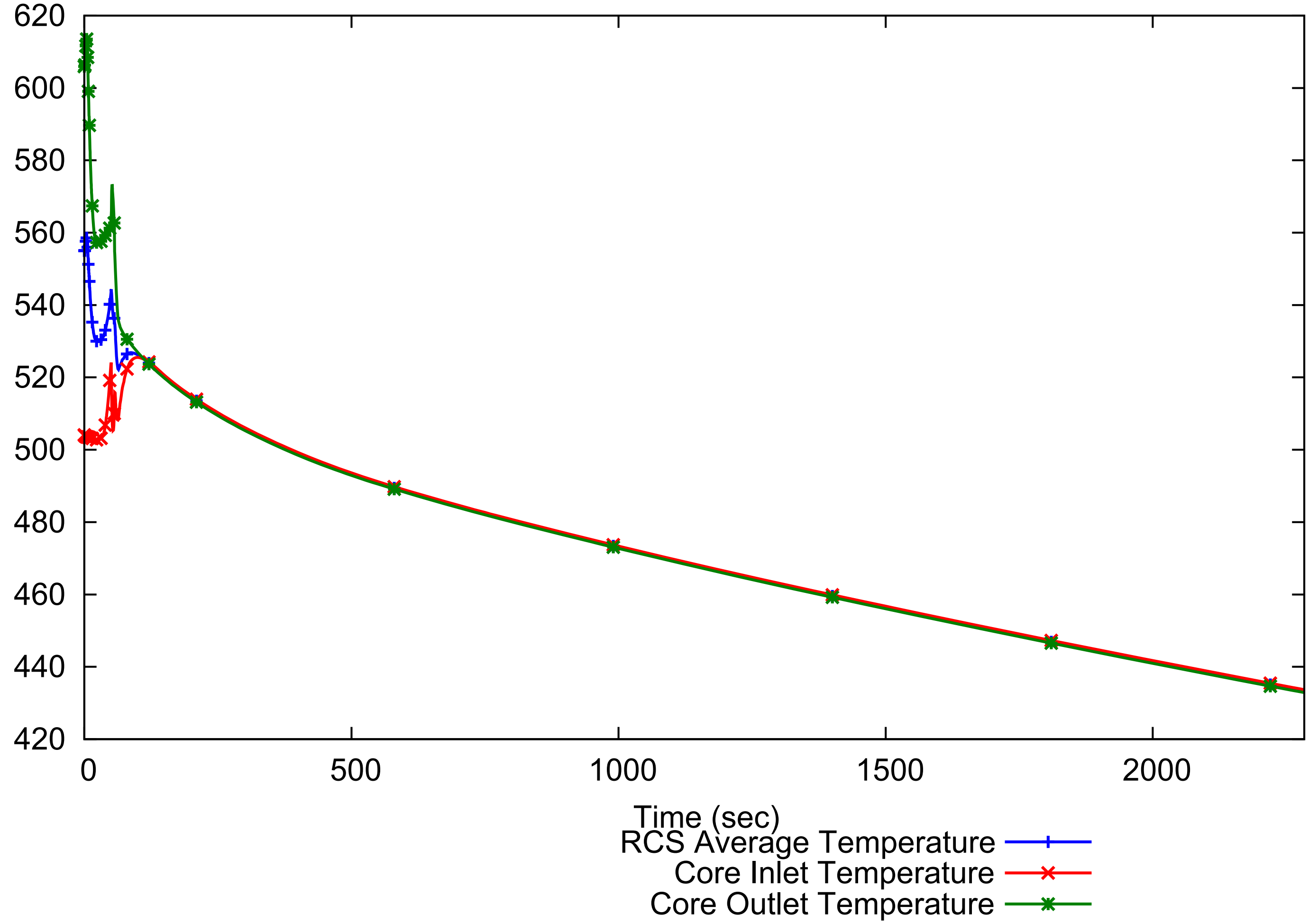

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Figure 15.6-65: Inadvertent Operation of an Emergency Core Cooling System Valve – Fuel Volume Average Temperature

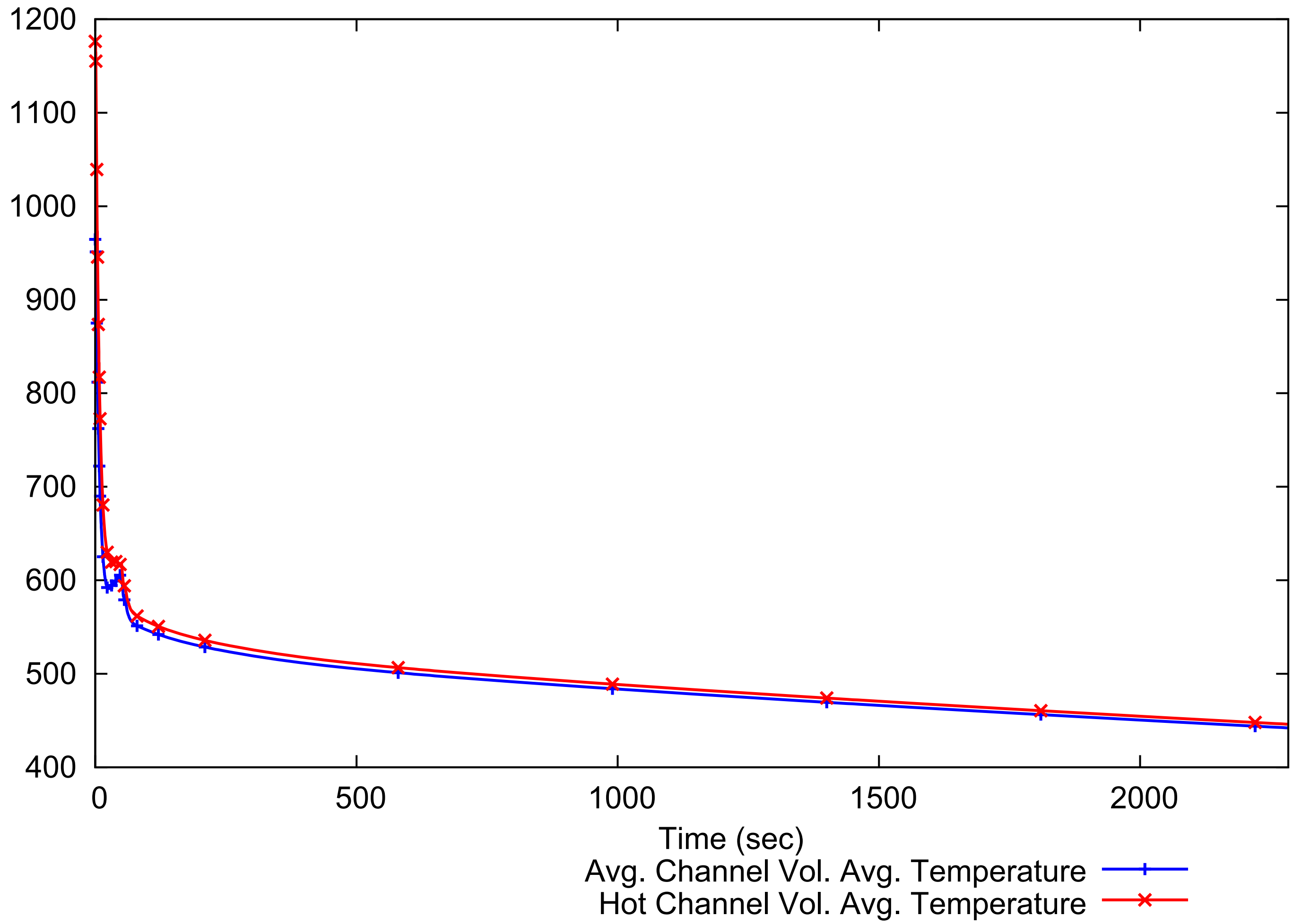

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Figure 15.6-66: Inadvertent Operation of an Emergency Core Cooling System Valve – Fuel Cladding Temperature

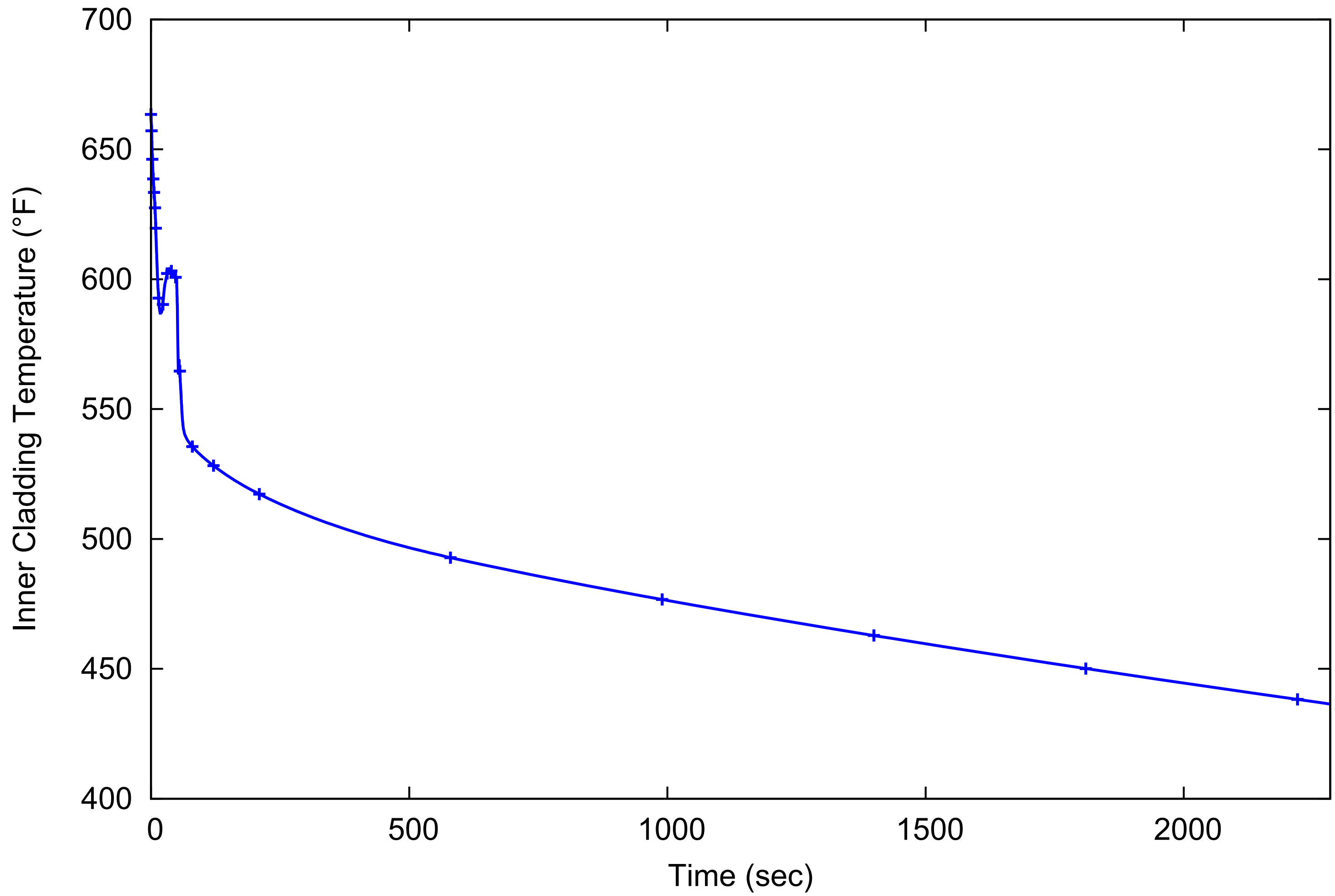

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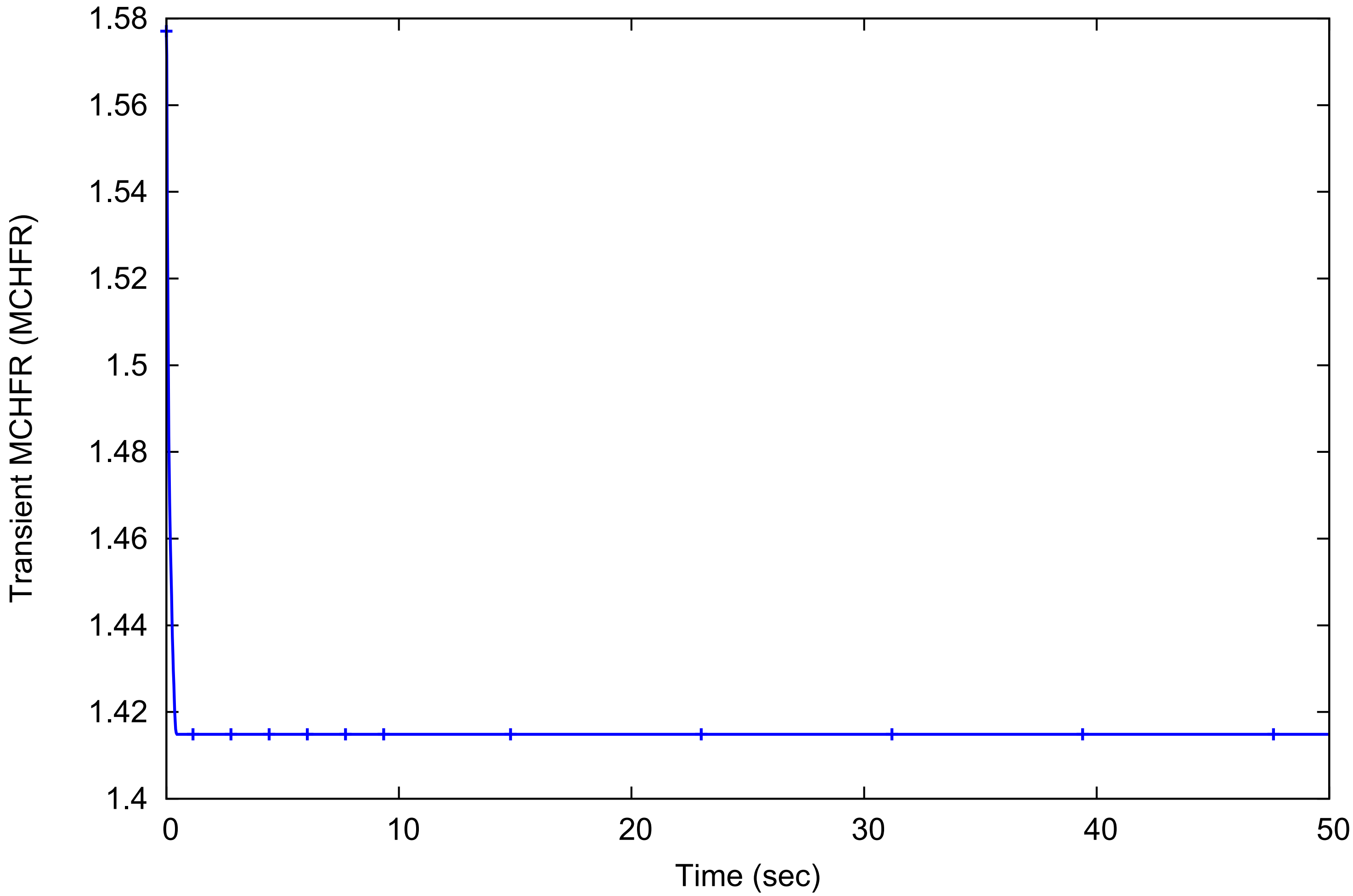
Figure 15.6-67: Inadvertent Operation of an Emergency Core Cooling System Valve – Critical Heat Flux Ratio



RAI 15.06.06-1, RAI 15.06.06-2

Figure 15.6-68: Inadvertent Operation of an Emergency Core Cooling System Valve – Critical Heat Flux Ratio