NUSCALE POWER™

RAIO-0718-60801

July 10, 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. 473 (eRAI No. 9483) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 473 (eRAI No. 9483)," dated May 11, 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 Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 9483:

- 15.01.01-2
- 15.01.01-3
- 15.01.01-4
- 15.01.01-5
- 15.01.01-6
- 15.01.01-7
- 15.01.01-8
- 15.01.01-9
- 15.01.01-10

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 473 (eRAI No. 9483). NuScale requests that the proprietary version 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 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses 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. 9483, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9483, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0718-60802

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9483, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9483, nonproprietary

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

eRAI No.: 9483
Date of RAI Issue: 05/11/2018

NRC Question No.: 15.01.01-2

General Design Criterion (GDC) 10, "Reactor design," in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). Design-Specific Review Standard for NuScale small modular reactor (SMR) Design (DSRS) Section 15.1.1-15.1.4, "Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of the Turbine Bypass System or Inadvertent Operation of the Decay Heat Removal System," provides guidance for meeting GDC 10 and states that mitigating systems should be assumed to be actuated in the analyses at setpoints with allowance for instrument inaccuracy in accordance with the guidance in Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation."

Final Safety Analysis Report (FSAR) Tier 2, Sections 15.1.1, “Decrease in Feedwater Temperature,” 15.1.2, “Increase in Feedwater Flow,” 15.1.3, “Increase in Steam Flow,” and 15.1.5, “Steam Piping Failures Inside and Outside of Containment,” state that a 5 percent uncertainty is added to the high core power trip to account for a decalibration effect in the excore detectors caused by increased coolant density in the downcomer. The staff audited engineering calculation {{
}}[2(a),(c)] (the calculation supporting FSAR Tier 2, Section 15.1.1) and notes that the 5 percent is based on {{

}}[2(a),(c)].

To ensure that the analyses use conservative setpoints that will adequately protect SAFDLs, provide the basis for the 5 percent uncertainty in the high core power trip. In particular, provide {{
}}[2(a),(c)] .

NuScale Response:

The feedwater temperature analysis defines the bounding decalibration factor at 0.5 percent / degree F. Final sensor uncertainties are not determined at the time of the DCA submittal or approval as described in the NuScale Instrument Setpoint Methodology Technical Report, TR-0616-49121 Revision 0. However, FSAR Table 15.0-7 has been modified to include a footnote to document this analysis limit, consistent with the treatment of other sensor and module protection system characteristics that will be determined in the detailed design process.

The limiting minimum critical heat flux ratio case presented in the FSAR results in a total core inlet temperature drop of approximately 8 degrees F (Figure 15.1-2). Therefore, assuming the over power trip at 125 percent is additional conservatism applied in the analysis.

Impact on DCA:

FSAR Table 15.0-7 has been revised as described in the response above and as shown in the markup provided with this response.

RAI 15.00.02-20, RAI 15.01.01-2, RAI 15.04.01-4

Table 15.0-7: Analytical Limits and Time Delays

Signal	Analytical Limit	Basis and Event Type	Actuation Delay
High Power	~~25%, 120% rated thermal power (RTP)~~120%[5] rated thermal power (RTP) (≥ 15% RTP) 25% RTP (<15% RTP)	This signal is designed to protect against exceeding critical heat flux (CHF) limits for reactivity and overcooling events.	2.0 sec
Source and Intermediate Range Log Power Rate	3 decades/min[6]	This signal is designed to protect against exceeding CHF and energy deposition limits during startup power excursions	Variable
High Power Rate	±15% RTP/min	This signal is designed to protect against exceeding CHF limits for reactivity and overcooling events.	2.0 sec
High Startup Range Count Rate	5.0 E+05 counts per second	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 Reactor Coolant System (RCS) Hot Temperature	610°F	This signal is designed to protect against exceeding CHF limits for reactivity and heatup 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 emergency core cooling system (ECCS) function during these events.	2.0 sec
High Pressurizer Pressure	2000 psia	This signal is designed to protect against exceeding reactor pressure vessel (RPV) pressure limits for reactivity and heatup events.	2.0 sec
High Pressurizer Level	80%	This signal is designed to detect and mitigate chemical and volume control system (CVCS) malfunctions to protect against overfilling the pressurizer.	3.0 sec
Low Pressurizer Pressure	1720 psia[1]	This signal is designed to detect and mitigate primary high energy line break (HELB) outside containment and protect RCS subcooled margin for protection against instability events.	2.0 sec
Low Low Pressurizer Pressure	1600 psia[2]	This signal is designed to detect and mitigate primary HELB outside containment and protect RCS subcooled margin for protection against instability events.	2.0 sec
Low Pressurizer Level	35%	This signal is designed to protect the pressurizer heaters from uncovering and overheating during decrease in RCS inventory events.	3.0 sec
Low Low Pressurizer Level	20%	This signal is designed to detect and mitigate loss-of-coolant accidents (LOCAs) to protect RCS inventory and ECCS functionality during LOCA and primary HELB outside containment events.	3.0 sec

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

Signal	Analytical Limit	Basis and Event Type	Actuation Delay
High Under-the-Bioshield Temperature	250°F	This signal is designed to detect high energy leaks or breaks at the top of the NuScale Power Module under the bioshield to reduce the consequences of high energy line breaks on the safety related equipment located on top of the module.	8.0 sec

Notes:

1. If RCS hot temperature is above 600°F. See Figure 15.0-9.
2. If RCS hot temperature is below 600°F. See Figure 15.0-9.
3. RPV riser level and CNV water level are presented in terms of elevation where reference zero is the bottom of the reactor pool. The ranges allow ±20" from the nominal ECCS level setpoint of 370" and 240", respectively.
4. Normal AC voltage is monitored at the bus(es) suppling the battery chargers for the highly reliable DC power system. The analytical limit is based on Loss of Normal AC Power to plant buses (0 volts) but the actual bus voltage is based upon the voltage ride-thru characteristics of the EDSS battery chargers.
5. The overcooling event analyses account for a cooldown event specific process error analytical limit of 0.5%/°F.
6. The high count rate trip is treated as a source range over power trip that occurs at a core power analytical limit of 500kW which functionally equates neutron monitoring system counts per second to core power. This trip is bypassed once the intermediate range signal has been established.

Response to Request for Additional Information
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eRAI No.: 9483
Date of RAI Issue: 05/11/2018

NRC Question No.: 15.01.01-3

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. To ensure compliance with GDC 10, the transient scenario having the most severe consequences for SAFDLs should be analyzed and presented in the FSAR.

FSAR Tier 2, Figure 15.1-1, "Feedwater Temperature (15.1.1 Decrease in Feedwater Temperature)," shows that the feedwater (FW) temperature at the top of the containment vessel starts at 310°F and decreases linearly to about 140°F, at which point it remains constant. However, the event description in FSAR Section 15.1.1.3.3 states that the limiting event initiates with a linear decrease in FW temperature to the minimum possible temperature of 100°F over 160 seconds. The staff acknowledges that the reason the temperature in Figure 15.1-1 does not decrease to 100°F may be because the FW isolation valves (FWIVs) close before 160 seconds. However, given the initial FW temperature of 310°F, the rate of temperature decrease becomes about 1.3°F/s. Therefore, the FW temperature should decrease by about 184°F over the 140 seconds (per FSAR Tier 2, Table 15.1.1, "Sequence of Events (15.1.1 Decrease in Feedwater Temperature)") prior to FWIV closure, leading to a final temperature of 126°F. To demonstrate that the results for the most limiting decrease in FW temperature scenario have been presented, explain why Figure 15.1-1 does not show a minimum value of 126°F. Update the FSAR as necessary.

NuScale Response:

The decrease in feedwater event is analyzed by changing the secondary side source boundary condition temperature modeled through an NRELAP5 time-dependent volume. The event analysis also contains pipe components between the secondary side source and containment vessel which are representative of feedwater system piping. While this piping otherwise has no influence on the event analysis, the length of piping causes a time delay in temperature change between the secondary side source and the top of the containment vessel as the colder feedwater flows towards containment. For this reason, temperature at the top of the containment vessel is approximately 140 degrees F at 140 seconds (FSAR Figure 15.1-1) while the secondary side source temperature is approximately 126 degrees F at 140 seconds (note source temperature is not plotted in the FSAR results). As identified in this request for additional information, feedwater temperature remains relatively constant after 140 seconds because the feedwater isolation valves are closed.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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eRAI No.: 9483
Date of RAI Issue: 05/11/2018

NRC Question No.: 15.01.01-4

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 13 requires the provision of instrumentation to monitor variables and systems over their anticipated ranges of normal operation, and of appropriate controls to maintain listed variables and systems within prescribed operating ranges. DSRS Section 15.1.1-15.1.4 provides guidance for meeting GDC 10 and 13 and guides the reviewer to review the sequence of events from initiation until a stabilized condition is reached. The sequence of events should be justified based upon the expected values of the relevant monitored parameters and instrument indications.

FSAR Tier 2, Table 15.1-1, lists most of the key events for the decrease in FW temperature event. However, it does not list the time of minimum critical heat flux ratio (MCHFR), despite the fact that MCHFR is the key acceptance criterion for the event. The staff also notes that Table 15.1-1 shows the reactor trips at 133 seconds, which is well beyond the time scale of FSAR Figure 15.1-9, "Critical Heat Flux Ratio (15.1.1 Decrease in Feedwater Temperature)." In order for the staff to assess the event response, add the timing of MCHFR to FSAR Table 15.1-1. In addition, add the timing of MCHFR to the sequence of events tables for other FSAR Chapter 15 events that seek to identify the limiting MCHFR, if not already present.

NuScale Response:

FSAR Section 15.1 provides the timing for the limiting minimum critical heat flux ratio (MCHFR) in Table 15.1-1 and the correct timescale of Figure 15.1-9 consistent with the timing. The correct Table 15.1-1 and Figure 15.1-9 are presented in Revision 1 of the NuScale FSAR.

The limiting MCHFR timing was added to Table 15.1-1 and provided to the NRC with changes due to the updated NSP4 critical heat flux correlation on December 8, 2017. The timescale of Figure 15.1-9 as well as other limiting MCHFR plots in FSAR Chapter 15 were corrected after the inconsistencies were identified in eRAI 8764, Question 15.04.02-1 and an internal corrective action resolution.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
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eRAI No.: 9483
Date of RAI Issue: 05/11/2018

NRC Question No.: 15.01.01-5

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. One of the specific acceptance criteria in DSRS Section 15.1.1-15.1.4 to meet this requirement is that the values of the parameters used in the analytical model should be suitably conservative.

FSAR Tier 2, Table 15.1-2, "Decrease in Feedwater Temperature - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)," lists inputs and initial conditions for the decrease in FW temperature event. However, the table does not include values for initial pressurizer level or steam generator (SG) heat transfer. Tables 15.1-4, "Increase in Feedwater Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)," and 15.1-6, "Increase in Steam Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)," are also missing these values, as well as the initial feedwater temperature. The staff notes that the values chosen for these parameters and their biases influence MCHFR. To demonstrate the use of suitably conservative values for these parameters, add the parameter values and biases for initial pressurizer level and SG heat transfer to Tables 15.1-2, 15.1-4, and 15.1-6, and add the value and bias for initial FW temperature to Tables 15.1-4 and 15.1-6.

NuScale Response:

The initial values and biases for pressurizer level and steam generator heat transfer were added to FSAR Table 15.1-2, Table 15.1-5, and Table 15.1-8 as shown in the markup provided with this response. The initial value and bias for the feedwater temperature were added to FSAR Table 15.1-5 and Table 15.1-8 as shown in the markup provided with this response.

Impact on DCA:

FSAR Table 15.1-2, Table 15.1-5, and Table 15.1-8 have been revised as described in the response above and as shown in the markup provided with this response.

RAI 15.01.01-5, RAI 15.01.01-8

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

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1174 lbm/s
RCS average temperature	545 °F	+10 °F
SG pressure	500 psia	+35psia
FW temperature	300 °F	+10 °F
Pressurizer level	60%	+8%
SG heat transfer bias	Nominal	N/A
Reactor pool temperature	40 °F - 200 °F	Minimum (40 °F)
DHRS heat transfer	Nominal	+30%
MTC	EOC	Most Negative
DTC	BOC	Least Negative

RAI 15.01.01-5, RAI 15.01.01-8

Table 15.1-5: Increase in Feedwater Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1173 lbm/s
RCS average temperature	545 °F	+10 °F
SG pressure	500 psia	+35psia
Pressurizer level	60%	+8%
SG heat transfer	Nominal	-30%
Feedwater temperature	300 °F	-10 °F
Reactor pool temperature	40 °F - 200 °F	Minimum (40 °F)
DHRS heat transfer	Nominal	+30%
MTC	EOC	Most Negative
DTC	BOC	Least Negative

RAI 15.01.01-5, RAI 15.01.01-8

Table 15.1-8: Increase in Steam Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1172 lbm/s
RCS average temperature	545 °F	+10 °F
SG temperature	500 psia	+35psia
Pressurizer level	60%	+8%
SG heat transfer	Nominal	-30%
FW temperature	300 °F	-10 °F
Reactor pool temperature	40 °F - 200 °F	Minimum (40 °F)
DHRS heat transfer	Nominal	+30%
MTC	EOC	Most Negative
DTC	BOC	Least Negative

NuScale Power

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eRAI No.: 9483
Date of RAI Issue: 05/11/2018

NRC Question No.: 15.01.01-6

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 13 requires the provision of instrumentation to monitor variables and systems over their anticipated ranges of normal operation, and of appropriate controls to maintain listed variables and systems within prescribed operating ranges. DSRS Section 15.1.1-15.1.4 provides guidance for meeting GDC 10 and 13 and guides the reviewer to review the extent to which plant and reactor protection systems are required to function.

FSAR Section 15.1.2.2, "Sequence of Events and Systems Operation," states that, for the increase in feedwater flow event, the high RCS pressure trip provides protection for cases that do not reach the high power setpoint. However, the staff notes that the initial part of the transient depressurizes the RCS, so it is unclear why the low steam superheat or high steam pressure reactor trips would not occur before a high RCS pressure trip should a trip not occur on high power. To further illustrate this point, FSAR Figure 15.1-16, "Reactor Coolant System Pressure (15.1.2 Increase in Feedwater Flow)," shows that RCS pressure is about 1920 psia, well under the 2000 psia high RCS pressure trip setpoint, by the time of reactor trip. However, FSAR Table 15.1-3, "Sequence of Events (15.1.2 Increase in Feedwater Flow)," shows that the low steam superheat limit is reached one second before the high reactor power limit, and FSAR Figure 15.1-18, "Main Steam System Pressure (15.1.2 Increase in Feedwater Flow)," shows a rapid increase in steam pressure at the beginning of the transient. Furthermore, the list of credited MPS signals in FSAR Section 15.1.2.2 does not include the high RCS pressure trip. Explain the statement in the FSAR regarding the high RCS pressure trip, and update the FSAR as necessary.

NuScale Response:

FSAR Section 15.1.2.2 incorrectly stated that the high RCS pressure trip provides protection during an increase in feedwater flow event. This statement was removed from the FSAR as shown in the markup provided with this response. The MPS trip signals that provide protection during an increase in feedwater flow are provided in FSAR Section 15.1.2.2.

Impact on DCA:

FSAR Section 15.1.2 has been revised as described in the response above and as shown in the markup provided with this response.

15.1.2.2 Sequence of Events and Systems Operation

The sequence of events for an increase in feedwater flow event is provided in Table 15.1-4.

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

The FWS could malfunction and increase feedwater flow by turning on a feedwater pump, opening a feedwater regulator valve, 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. The feedwater pump system includes two pumps with a third for backup. If the backup pump were to spuriously turn on at its maximum flow rate, the total increase in feedwater flow would be 50 percent. In order to conservatively bound all possible feedwater flow increase scenarios, a spectrum of feedwater flow increases up to 100 percent normal flow are analyzed. The steam outlet is modeled as a constant mass flow boundary to allow the secondary system pressure to change in response to the increase in feedwater flow.

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

RAI 15.01.01-6

The MPS is credited to protect the plant in the event of an increase in feedwater flow. If the feedwater 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 where the acceptance criteria could be challenged. ~~For cases that do not reach the high power setpoint, the high RCS pressure trip provides protection.~~ The following MPS signals provide the plant with protection during an increase in feedwater flow:

- high core power (5 percent uncertainty added)
- high core power rate (not credited in the safety analysis of this event)
- high steam superheat (not credited in the safety analysis of this event)
- high steam pressure
- low steam superheat

Due to the cooling of the RCS during an increase in feedwater 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 core power rate trip is not credited in the analyses, and a 5 percent uncertainty is added to the high core power trip. The high steam superheat trip is also not credited in this analysis.

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eRAI No.: 9483
Date of RAI Issue: 05/11/2018

NRC Question No.: 15.01.01-7

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 15, "Reactor coolant system design," requires that the RCS 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 (RCPB) are not exceeded during any condition of normal operation, including AOOs. One of the specific acceptance criteria in DSRS Section 15.1.1-15.1.4 necessary to meet the requirements of GDC 10 and 15 for incidents of moderate frequency is that the event should not generate a more serious plant condition without other faults occurring independently.

FSAR Section 15.1.2, "Increase in Feedwater Flow," evaluates the potential overfilling of a steam generator (SG) due to the increasing feedwater supply. Overfilling of a SG could result in loss of a decay heat removal system (DHRS) train, degrading decay heat removal capability. FSAR Section 15.1.2 concludes that the SG will not overfill, and DHRS capability is maintained. However, the staff audited the SG overfill case in Attachment 5 to engineering calculation (EC)-0000-2016, "Increase in Feedwater Flow Analysis," which supports FSAR Section 15.1.2, and noted two potential issues.

First, the peak pressures and levels for SG trains 1 and 2 are identical, despite the fact that a single failure of a FWIV is assumed. The staff would expect peak pressure and level in the train with the failed FWIV to be higher than in the train with the fully functional FWIV.

Second, a {{ }}2(a),(c) is used for the overfilling case. However, the staff notes that the maximum transient SG level is higher for the baseline cases, which were not biased to achieve {{ }}2(a),(c) cases. Thus, it appears that the approximately 80-percent level in both SG trains could potentially increase if a different initialization were used.

Furthermore, as stated in FSAR Section 15.0.0.6.6, the non-safety-related FW regulating valve (FWRV) is relied upon for backup to the failed FWIV. The FWRVs must be capable of closing under the conditions expected during the increase in feedwater flow event. The staff notes that the supplemental response to RAI 8888, Question 06.02.04-6, dated August 16, 2017, states that the FWRV meets the same flow requirements as the FWIVs; however, neither the FSAR nor the attached FSAR markup contains such information regarding the capability of the FWRVs.

Considering these observations, the staff is concerned about degraded heat removal capability via the DHRS should the level in both SG trains grow due to a more limiting initialization and/or a FWRV that may not close under the particular event conditions. Provide additional justification, such as sensitivity studies, that the SGs do not overfill and impede DHRS heat removal capability under the worst case conditions, and update the FSAR as necessary. In addition, add a statement to the FSAR such as the one provided in the supplemental response to RAI 8888, Question 06.02.04-6, mentioned above, that confirms the flow requirements for the FWRVs are equivalent to requirements for the FWIVs.

NuScale Response:

First Observation

For the first observation, the peak pressures and levels for steam generator trains 1 and 2 were identical because the feedwater pump deadheaded before the isolation valves were fully closed. Since the feedwater flow rate reduced to zero before the feedwater isolation valve (FWIV) in Line 2 was fully closed (FWIV in Line 1 failed open), the feedwater delivered to each steam generator was nearly identical. As a result, the steam generator level and pressure responses were identical.

However, the transient response indicates that the steam generator pressure rapidly increases upon the closure of the main steam isolation valves followed by a decrease in pressure upon the opening of the decay heat removal system (DHRS) actuation valves while the feedwater regulating valves (FWRVs) are still closing. The analysis conservatively deadheads the pumps 100 percent above normal system operating pressure to sufficiently bound the expected feedwater system transient response.

It is noted that the increase in heat removal events are primarily evaluated for MCHFR as described in the Non-Loss of Coolant Accident Analysis Methodology Topical Report (TR-0516-49416 Rev. 1). Sensitivity results for peak secondary pressure have shown a strong influence by primary temperature due to the condensing and boiling nature of the DHRS operation. Higher primary temperature will result in high steam temperatures and pressure on the secondary side. The cooldown events are therefore less limiting for peak secondary pressure than the heatup or reactivity events.

Second Observation

Steam generator level is not explicitly identified as a figure of merit for the NuScale module safety analysis discussed in the Non-Loss of Coolant Accident Analysis Methodology Topical Report (TR-0516-49416, Rev. 1). As described in FSAR Section 5.4.3.1, the DHRS consists of two independent trains, with each capable of performing the system safety function in the event the other train fails. Therefore, a degraded or failed single DHRS loop will not prevent the other loop from performing as required. Therefore, the conclusions in the FSAR Section 15.1.2 are valid.

Third Observation

For the third observation, FSAR Section 10.4.7 states that the FWRV is designed to perform a feedwater isolation function as a backup to the FWIV. A statement that the FWRV meets the same flow requirements as the FWIV has been added to this section as shown in the markup provided with this response. In the event that the FWRVs are required to provide backup feedwater isolation, the heat removal capability of the DHRS is not degraded because closure of the other train's FWIV ensures at least one DHRS train remains operable.

Impact on DCA:

FSAR Section 10.4 has been revised as described in the response above and as shown in the markup provided with this response.

Feedwater heaters preheat the feedwater before returning to the steam generators. This improves the thermodynamic efficiency of the system, reduces plant operating costs, and helps reduce thermal stress on the steam generators.

Condensate is progressively warmed in the tube side of successive FWHs by turbine extraction steam. The FWHs can be isolated and have a full-flow bypass.

Level in each of the FWHs is automatically controlled using a modulating drain control valve on the downstream heater.

Feedwater Heater Vents and Drains

The heater vents and drains subsystem manages the condensing extraction steam flow through the shell side of the FWHs. Cascading drains flow by gravity to the condenser. Drain coolers are used to remove excess heat. Each FWH is individually vented to the condenser.

Feedwater Pumps

Three feedwater pumps are located downstream of the low-pressure feedwater heater (LP-FWH) and the intermediate-pressure FWH (IP-FWH), and upstream of the HP-FWH. Feedwater pump flow is monitored for each pump with minimum flow protection provided through a dedicated recirculation line sized for the pump required minimum flow. The feedwater pumps and pump control system are designed so that the trip of one feedwater pump does not result in a turbine generator trip or reactor trip. Standby feedwater pumps are provided with autostart capability on low pressure or pump trip.

Feedwater Regulating Valves

The FWRVs are used during normal and transient operation to control and equalize feedwater flow to the steam generators. The FWRVs are located in the RXB and are upstream of the FWIVs.

RAI 15.01.01-7

Normal control of the FWRVs is through the MCS. In off-normal conditions the MPS overrides normal control of the valves and can force closure. Each FWRV is designed to fail closed on loss of power or control signal, regardless of the operating mode, and performs a feedwater isolation function as a backup to the FWIV. As such, the FWRVs meet the same flow requirements as the FWIVs.

Feedwater Check Valves

Two check valves are installed in each feedwater line. Both feedwater check valves prevent reverse flow from the steam generators whenever the feedwater system is not in operation and are designed to withstand the forces of closing after a CFWS line rupture.

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eRAI No.: 9483
Date of RAI Issue: 05/11/2018

NRC Question No.: 15.01.01-8

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 13 requires the provision of instrumentation to monitor variables and systems over their anticipated ranges of normal operation, and of appropriate controls to maintain listed variables and systems within prescribed operating ranges. GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. DSRS Section 15.1.1-15.1.4 provides the guidance for meeting GDC 10, 13, and 15, and directs the reviewer to review the sequence of events to determine the extent to which plant and reactor protection systems are required to function and the operation of engineered safety systems that is required, including time delays for actuation. In addition, the DSRS directs the reviewer to ensure a stabilized condition is reached.

The events in FSAR Sections 15.1.1-15.1.3 all result in DHRS actuation for post-trip core cooling, but some key assumptions related to DHRS cooling are not included in the FSAR. First, it is unclear when the DHRS valves open for these events relative to the actuation signal. In addition, the assumed reactor pool temperature and DHRS heat transfer bias (if any) are not provided. The staff notes that conditions that maximize heat removal by the DHRS would be the most consequential in terms of reaching a stabilized condition following reactor trip and DHRS actuation. Clarify the timing of the DHRS valve opening for FSAR Sections 15.1.1- 15.1.3, and provide the reactor pool temperature and DHRS heat transfer bias assumed for the events. Update the FSAR to include these values.

NuScale Response:

FSAR Section 15.1 has been updated to include the assumed reactor pool temperature and decay heat removal system (DHRS) heat transfer and the associated biases in Table 15.1-2, Table 15.1-5, and Table 15.4-8.

The events in FSAR Section 15.1.1 through Section 15.1.3 assume that the DHRS valves begin to open at the "DHRS actuation" time provided in the sequence of events table for each event. The 30 seconds delay assumed in the heatup events is not used in the overcooling events to maximize heat removal through the DHRS.

Impact on DCA:

FSAR Table 15.1-2, Table 15.1-5, and Table 15.1-8 have been revised as described in the response above and as shown in the markup provided with this response.

RAI 15.01.01-5, RAI 15.01.01-8

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

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1174 lbm/s
RCS average temperature	545 °F	+10 °F
SG pressure	500 psia	+35psia
FW temperature	300 °F	+10 °F
Pressurizer level	60%	+8%
SG heat transfer bias	Nominal	N/A
Reactor pool temperature	40 °F - 200 °F	Minimum (40 °F)
DHRS heat transfer	Nominal	+30%
MTC	EOC	Most Negative
DTC	BOC	Least Negative

RAI 15.01.01-5, RAI 15.01.01-8

Table 15.1-5: Increase in Feedwater Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1173 lbm/s
RCS average temperature	545 °F	+10 °F
SG pressure	500 psia	+35psia
Pressurizer level	60%	+8%
SG heat transfer	Nominal	-30%
Feedwater temperature	300 °F	-10 °F
Reactor pool temperature	40 °F - 200 °F	Minimum (40 °F)
DHRS heat transfer	Nominal	+30%
MTC	EOC	Most Negative
DTC	BOC	Least Negative

RAI 15.01.01-5, RAI 15.01.01-8

Table 15.1-8: Increase in Steam Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1172 lbm/s
RCS average temperature	545 °F	+10 °F
SG temperature	500 psia	+35psia
Pressurizer level	60%	+8%
SG heat transfer	Nominal	-30%
FW temperature	300 °F	-10 °F
Reactor pool temperature	40 °F - 200 °F	Minimum (40 °F)
DHRS heat transfer	Nominal	+30%
MTC	EOC	Most Negative
DTC	BOC	Least Negative

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

eRAI No.: 9483
Date of RAI Issue: 05/11/2018

NRC Question No.: 15.01.01-9

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 13 requires the provision of instrumentation to monitor variables and systems over their anticipated ranges of normal operation, and of appropriate controls to maintain listed variables and systems within prescribed operating ranges. GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. DSRS Section 15.1.1-15.1.4 provides the guidance for meeting GDC 10, 13, and 15, and directs the reviewer to review the sequence of events from initiation until a stabilized condition is reached as well as time-related variations of parameters such as coolant conditions and RCS pressure.

The staff notes that the figures of core inlet and core outlet temperatures provided in FSAR Sections 15.1.1-15.1.3 show coolant conditions increasing rather than stabilizing because of the time scale of the figures. Furthermore, FSAR Section 15.0.4 states, "Safety analyses of design basis events 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." To allow the staff to make a finding with regard to conditions stabilizing for the events in FSAR Sections 15.1.1-15.1.3, provide the following figures on time scales sufficient to show the design reaches and maintains a stable condition, including any return to power from this specific event, and update the FSAR to include them:

- All core outlet temperature figures for FSAR Sections 15.1.1-15.1.3 (Figures 15.1-7, 15.1-17, and 15.1-28); alternatively, a figure of core average temperature for each event
- All total core reactivity figures for FSAR Sections 15.1.1-15.1.3 (Figures 15.1-4, 15.1-14, and 15.1-25)
- Reactor coolant system pressure for FSAR Section 15.1.3 (Figure 15.1-27)

In addition, the staff notes that FSAR Sections 15.1.2 and 15.1.3 refer to FSAR Section 15.0.6, "Evaluation of a Return to Power," for discussion on possible return to power scenarios. Because the return to power scenario is presented for a bounding cooldown event in 15.0.6, including ECCS actuation, and could result from various scenarios of power availability for the cooldown events in FSAR Section 15.1, add a reference in FSAR Section 15.1.1 to FSAR Section 15.0.6, similar to the ones in FSAR Sections 15.1.2 and 15.1.3.

NuScale Response:

Core outlet temperature and core reactivity plots that span the entire transient have been added to FSAR Section 15.1 for the following events: decrease in feedwater temperature, increase in feedwater flow, and increase in steam flow. The reactor coolant system pressure plot was added to FSAR Section 15.1 for the increase in steam flow event.

A reference to FSAR Section 15.0.6 was added to Section 15.1.1 for return to power scenarios.

Impact on DCA:

FSAR Section 15.1 andFSAR Figure 15.1-54 through Figure 15.1-60 have been revised as described in the response above and as shown in the markup provided with this response.

temperature reductions reveal that the limiting case with respect to MCHFR is a reduction to 100 degrees F in 160 seconds.

The initial conditions used in the evaluation of the limiting decrease in feedwater temperature event result in a conservative calculation. Table 15.1-2 provides key inputs for the limiting decrease in feedwater temperature case. 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 2 percent instrumentation uncertainty.
- Conservative scram 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 end-of-cycle core parameters are used to provide a limiting power response. The most negative moderator temperature coefficient (MTC) of -43.0 pcm/degrees F and the least negative doppler temperature coefficient of -1.40 pcm/degrees F are used to provide the largest power response for this event.
- 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 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 15.0.2.3.

15.1.1.3.3 Results

RAI 15.01.01-9

The sequence of events for the limiting decrease in feedwater temperature event is provided in Table 15.1-1. Figure 15.1-1 through Figure 15.1-10 and Figure 15.1-54 through Figure 15.1-55 show the transient behavior of key parameters for the limiting decrease in feedwater temperature event. A spectrum of feedwater temperature cases were analyzed, and the limiting temperature decrease is a case in which the cooldown rate yields simultaneous reactor trips on high power and high hot leg temperature. The limiting event initiates with a linear decrease in feedwater temperature to the minimum possible temperature of 100 degrees F over 160 seconds. The RCS response to the overcooling event begins once the cold feedwater front propagates through the secondary system piping and reaches the SG. This decrease in feedwater temperature leads to an increase in the heat removal rate from the RCS via the SG. During the over-cooling 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 were disabled, a similar power response would be 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 hot leg temperature rises in response to the increase in reactor power. The high hot leg temperature reactor limit is reached at about 125 seconds into the transient. During the assumed reactor trip signal delay, the rise in reactor power initiates a high power reactor trip at approximately 131 seconds into the transient. The peak RCS pressure occurs just prior to the reactor trip. The high hot leg temperature trip also actuates the DHRS valves to open. The feedwater isolation valves (FWIVs) and MSIVs close, isolating the SG from the rest of the secondary system.

Steam generator pressure does not change significantly during the initial phase of the transient. However, after DHRS actuation at 133 seconds, the closure of the FWIVs and MSIVs causes 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 feedwater temperature event itself. The maximum secondary pressure is reached just after main steam and feedwater isolation.

The CHFR decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the design limit. The MCHFR for the limiting decrease in feedwater temperature does not violate the CHF limit.

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During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. At approximately 2000 seconds, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of DHRS, and stabilization of RCS flow demonstrate the plant response to a decrease in feedwater temperature, and a return to a stable condition with no operator actions. For a discussion on possible return to power scenarios, see Section 15.0.6.

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 five Design Specific Review Standard (DSRS) acceptance criteria for this AOO are met for the limiting decrease in feedwater temperature case. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below:

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 AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this

15.1.2.3.2 Input Parameters and Initial Conditions

The initiating amount of feedwater flow increase is a 100 percent increase from normal flow. Feedwater flow is assumed to linearly increase from its initial steady state value to its final value over a time span of 0.1 seconds. This conservatively bounds the possible rates of feedwater flow increases.

The initial conditions used in the evaluation of the limiting increase in feedwater flow event result in a conservative calculation. Table 15.1-5 provides key inputs for the limiting increase in feedwater flow case. 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 2 percent measurement uncertainty.
- Conservative scram 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 end-of-cycle core parameters is used to provide a limiting power response. The most negative MTC of -43.0 pcm/degrees F and the least negative DTC of -1.40 pcm/degrees F are used to provide the largest power response for this 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.

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 15.0.2.3.

15.1.2.3.3 Results

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The sequence of events for the limiting increase in feedwater flow event is provided in Table 15.1-4. Figure 15.1-11 through Figure 15.1-21 and Figure 15.1-56 through Figure 15.1-57 show the transient behavior of key parameters for an increase in feedwater flow. A spectrum of feedwater flow increase cases are analyzed, and the limiting feedwater flow increase is a near instantaneous increase of 100 percent of normal feedwater flow. The steam boundary is allowed to increase with feedwater flow to maximize the cooling effect. This increase in feedwater flow leads to an increase in the heat removal rate from the RCS via the SG. During the over-cooling 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 were disabled, a similar power response would be driven by moderator feedback. The reactor power exceeds the high core power limit for the reactor trip system, initiating a scram.

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. See Section 15.0.2 for 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 14 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. This conservatively bounds the possible rates of steam flow increases.

The initial conditions used in the evaluation of the limiting increase in steam flow event result in a conservative calculation. Table 15.1-8 provides key inputs for the limiting increase in steam flow case. 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 2 percent measurement uncertainty.
- Conservative scram 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 end-of-cycle core parameters is used to provide a limiting power response. The most negative MTC of -43.0 pcm/degrees F and the least negative DTC of -1.40 pcm/degrees F are used to provide the largest power response for this 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.

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 15.0.2.3.

15.1.3.3.3 Results

RAI 15.01.01-9

The sequence of events for the limiting increase in steam flow event is provided in Table 15.1-7. Figure 15.1-22 through Figure 15.1-31 and Figure 15.1-58 through Figure 15.1-60 show the transient behavior of key parameters for an increase in steam flow event. A spectrum of steam flow increase cases were analyzed, and the limiting steam increase is a case in which the cooldown rate produces a power response that peaks just below the high power trip. The limiting event initiates with an increase in steam flow by approximately 14 percent of full power steam flow in 0.1 seconds. The feedwater pump 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 SG begins to increase. 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. Reactor power reaches a peak at approximately 58 seconds. The peak reactor power is just below the high core power limit of the RTS. If the regulating control rod bank were disabled, a similar power response would be driven by moderator feedback.

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 high hot leg temperature limit is reached at 60 seconds into the transient, tripping the reactor. The peak RCS pressure occurs around the time of the reactor scram. The high hot leg temperature signal also actuates the DHRS valves to open. The FWIVs and MSIVs close, isolating the SG from the rest of the secondary system.

Steam generator pressure does not change significantly during the initial phase of the transient. However, after DHRS actuation at 68 seconds, the closure of the FWIVs and MSIVs 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 feedwater isolation.

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

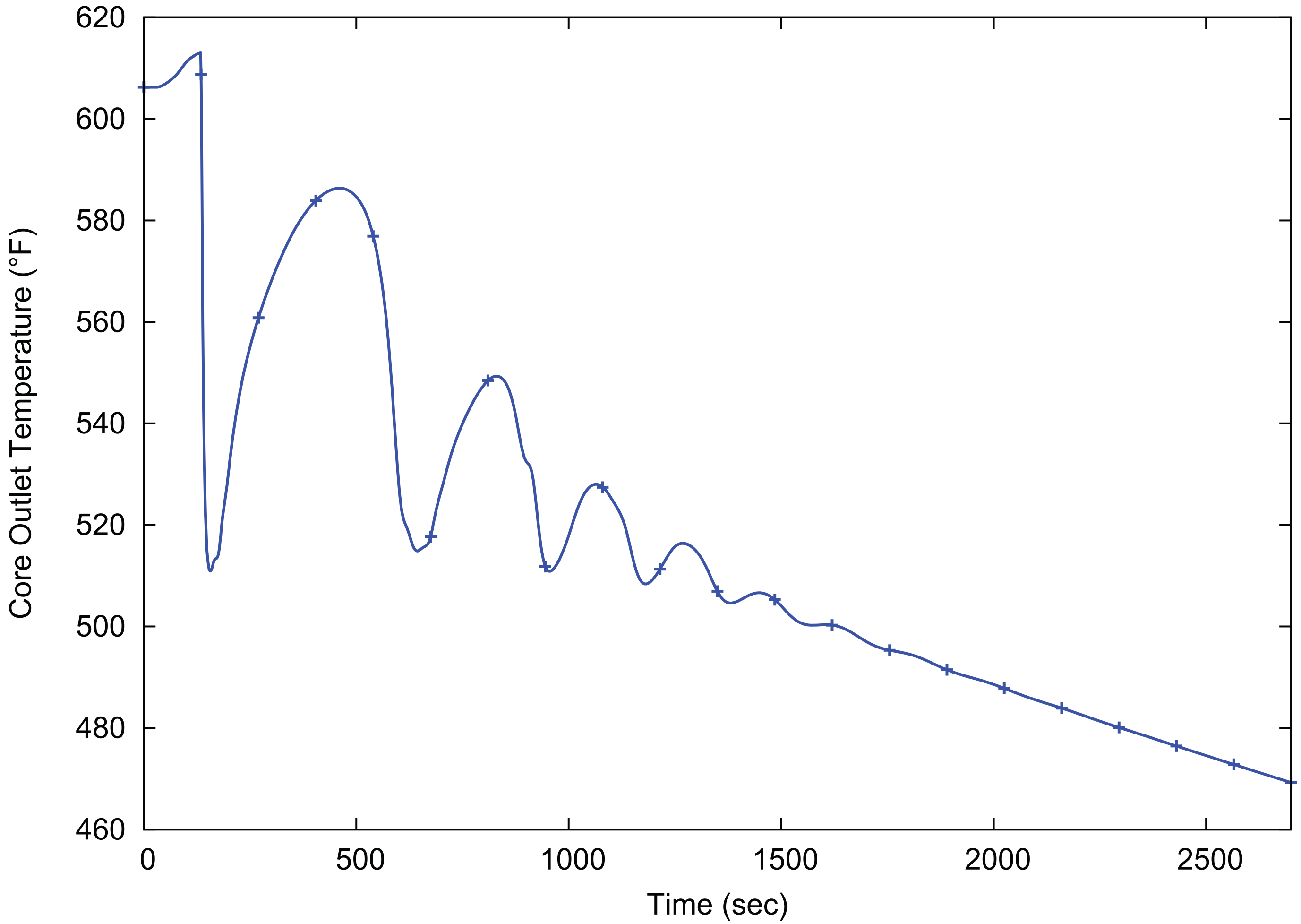
During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. After 2000 seconds, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of 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. For a discussion on possible return to power scenarios, see Section 15.0.6.

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.

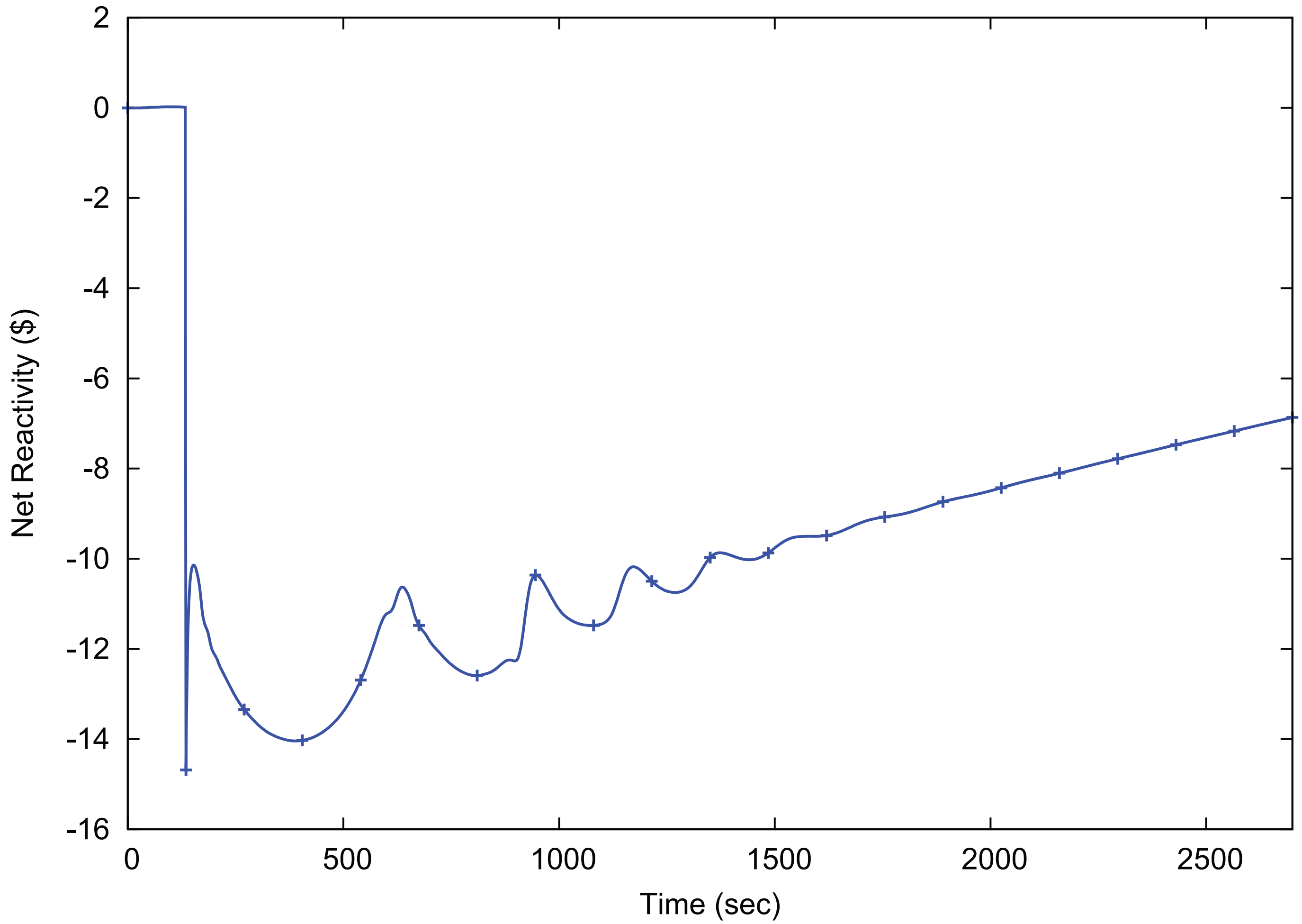
RAI 15.01.01-9

Figure 15.1-54: Core Outlet Temperature (15.1.1 Decrease in Feedwater Temperature)



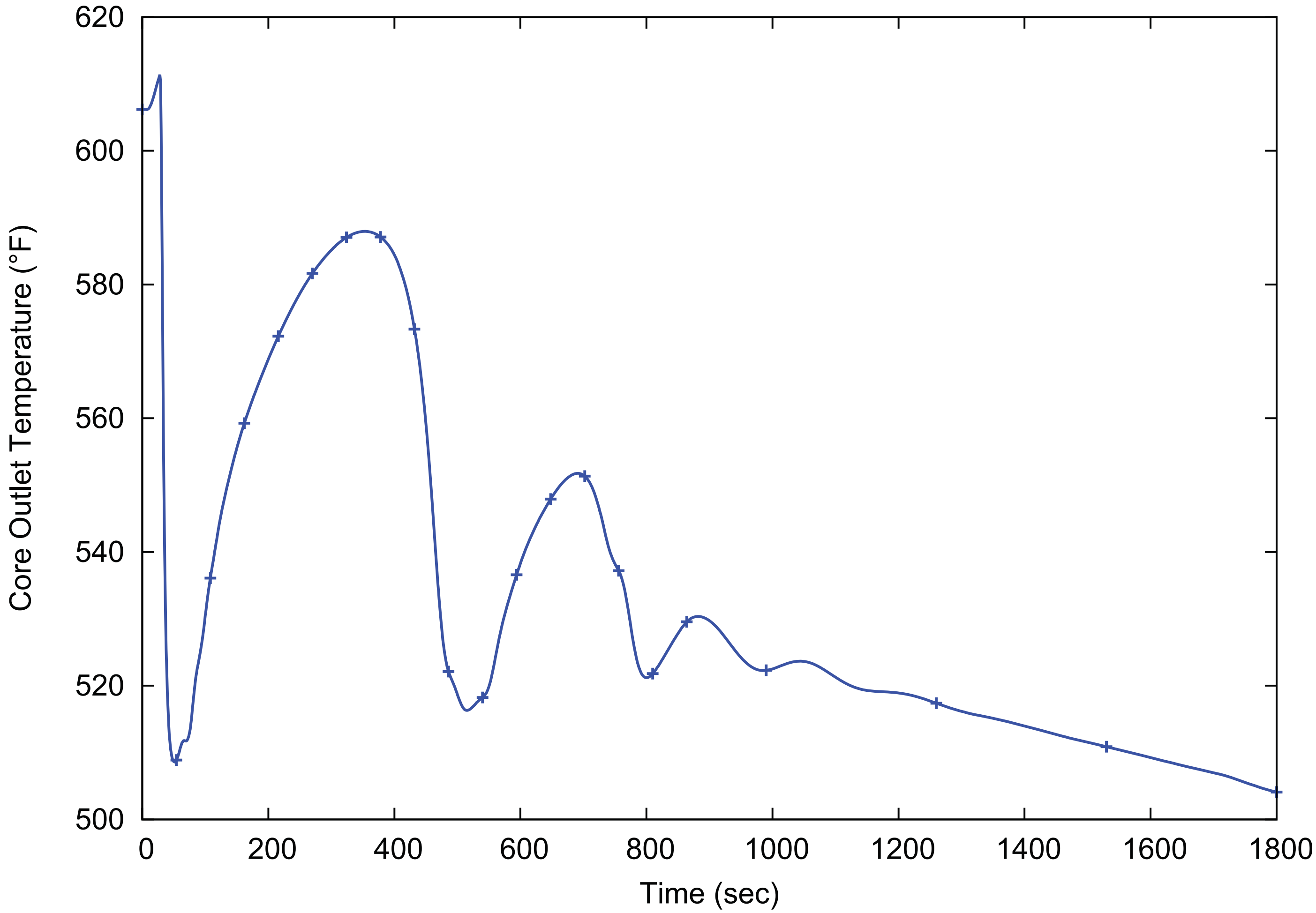
RAI 15.01.01-9

Figure 15.1-55: Core Reactivity – Entire Transient (15.1.1 Decrease in Feedwater Temperature)



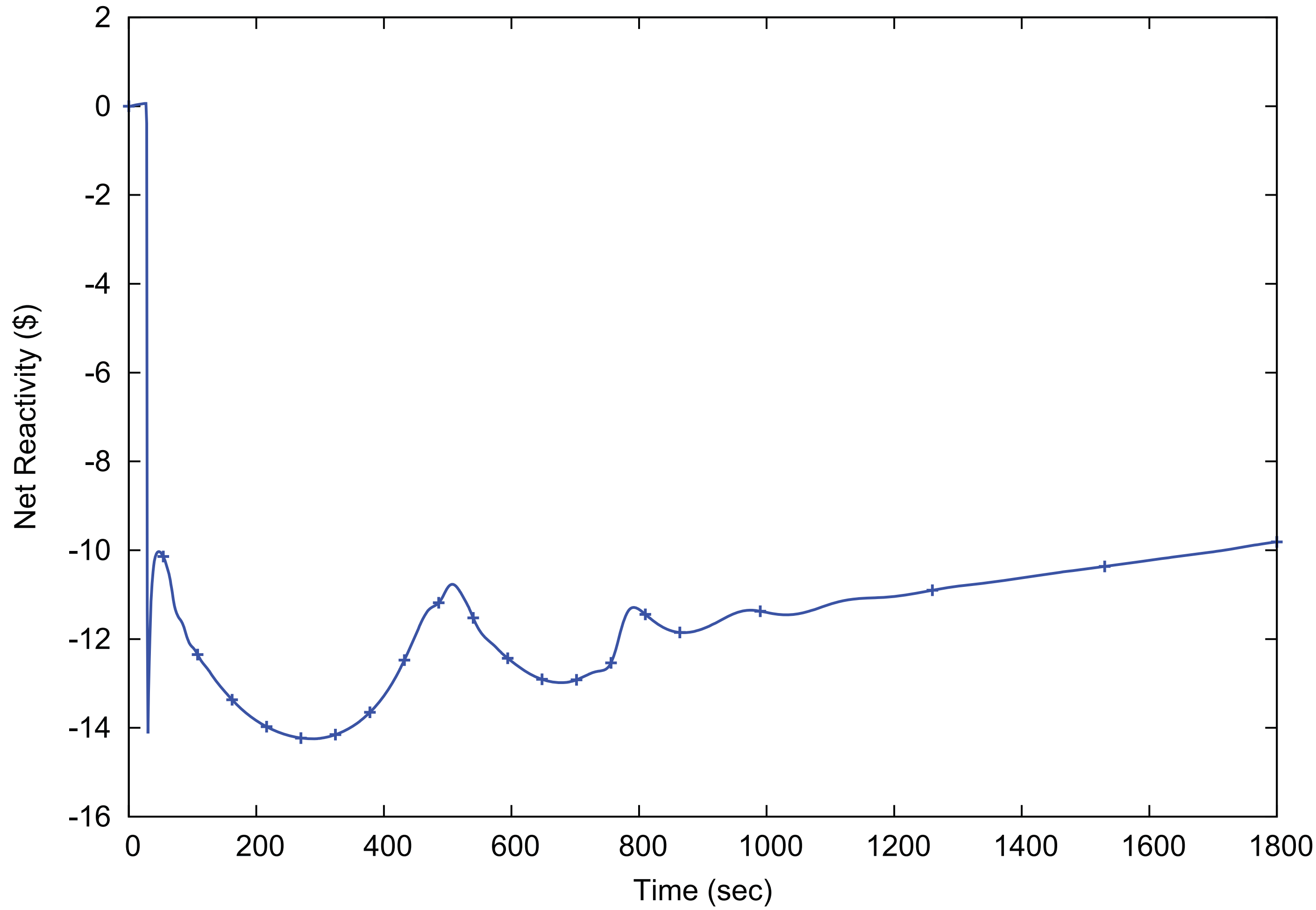
RAI 15.01.01-9

Figure 15.1-56: Core Outlet Temperature (15.1.2 Increase in Feedwater Flow)



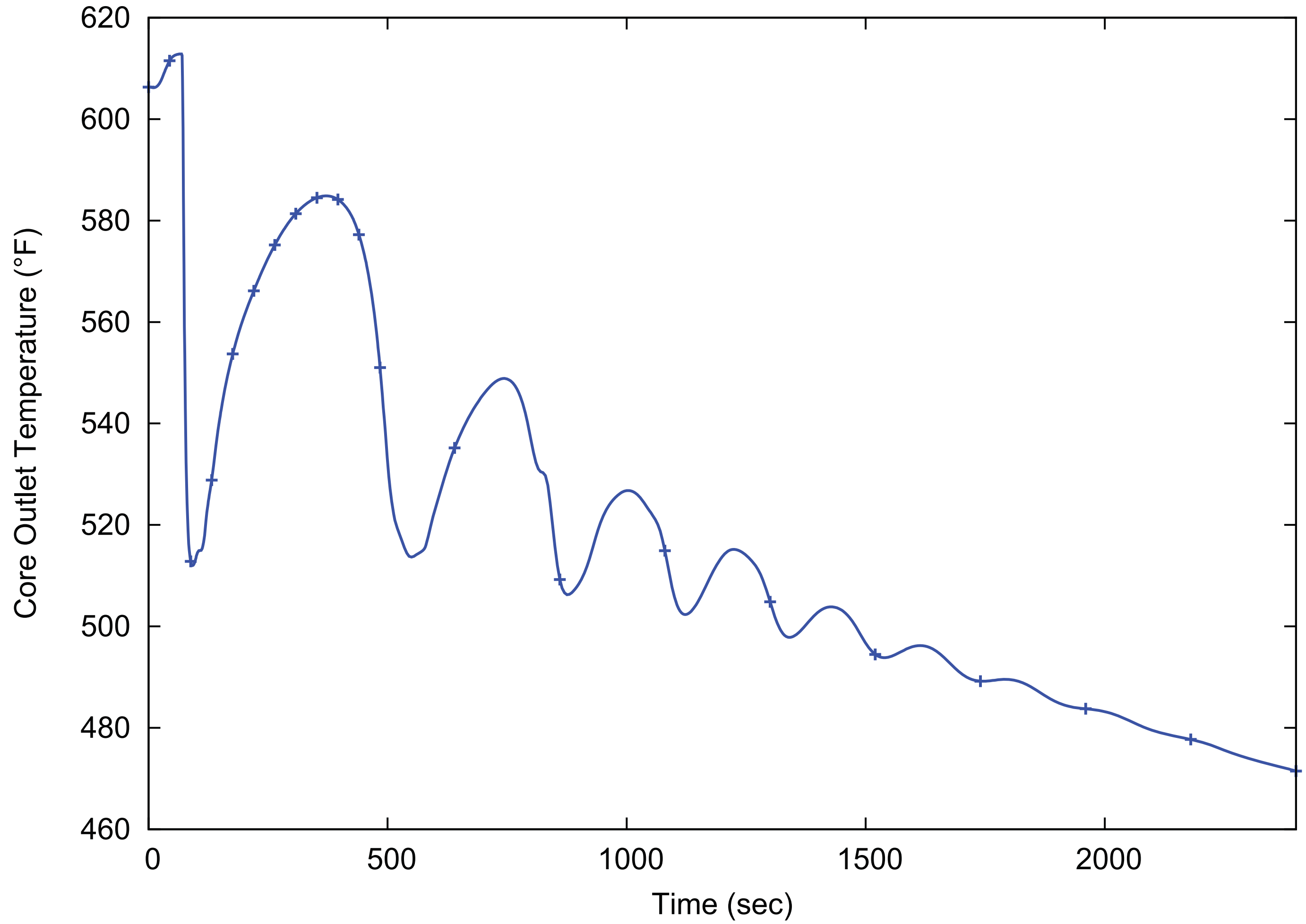
RAI 15.01.01-9

Figure 15.1-57: Core Reactivity – Entire Transient (15.1.2 Increase in Feedwater Flow)



RAI 15.01.01-9

Figure 15.1-58: Core Outlet Temperature (15.1.3 Increase in Steam Flow)



RAI 15.01.01-9

Figure 15.1-59: Core Reactivity – Entire Transient (15.1.3 Increase in Steam Flow)

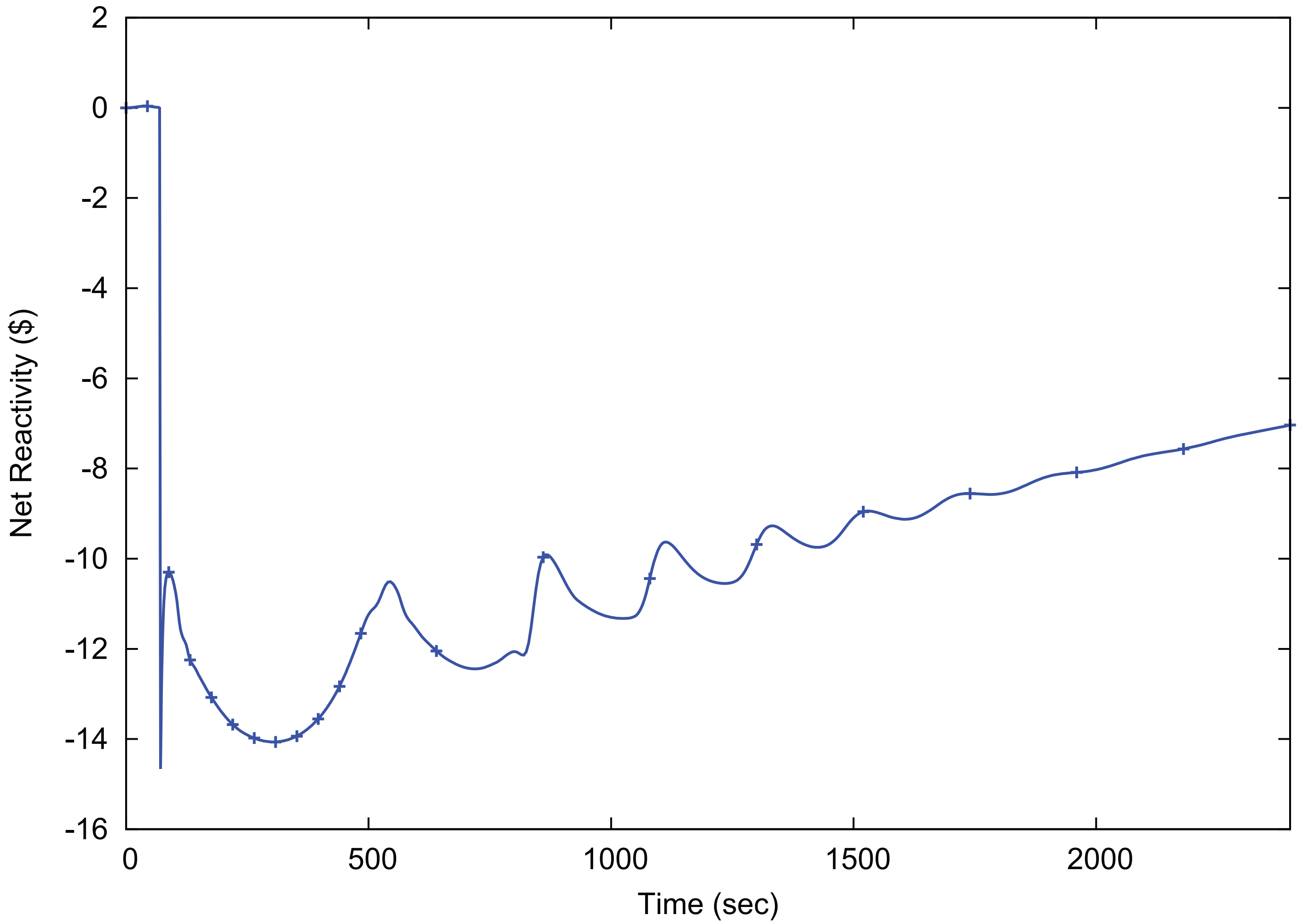


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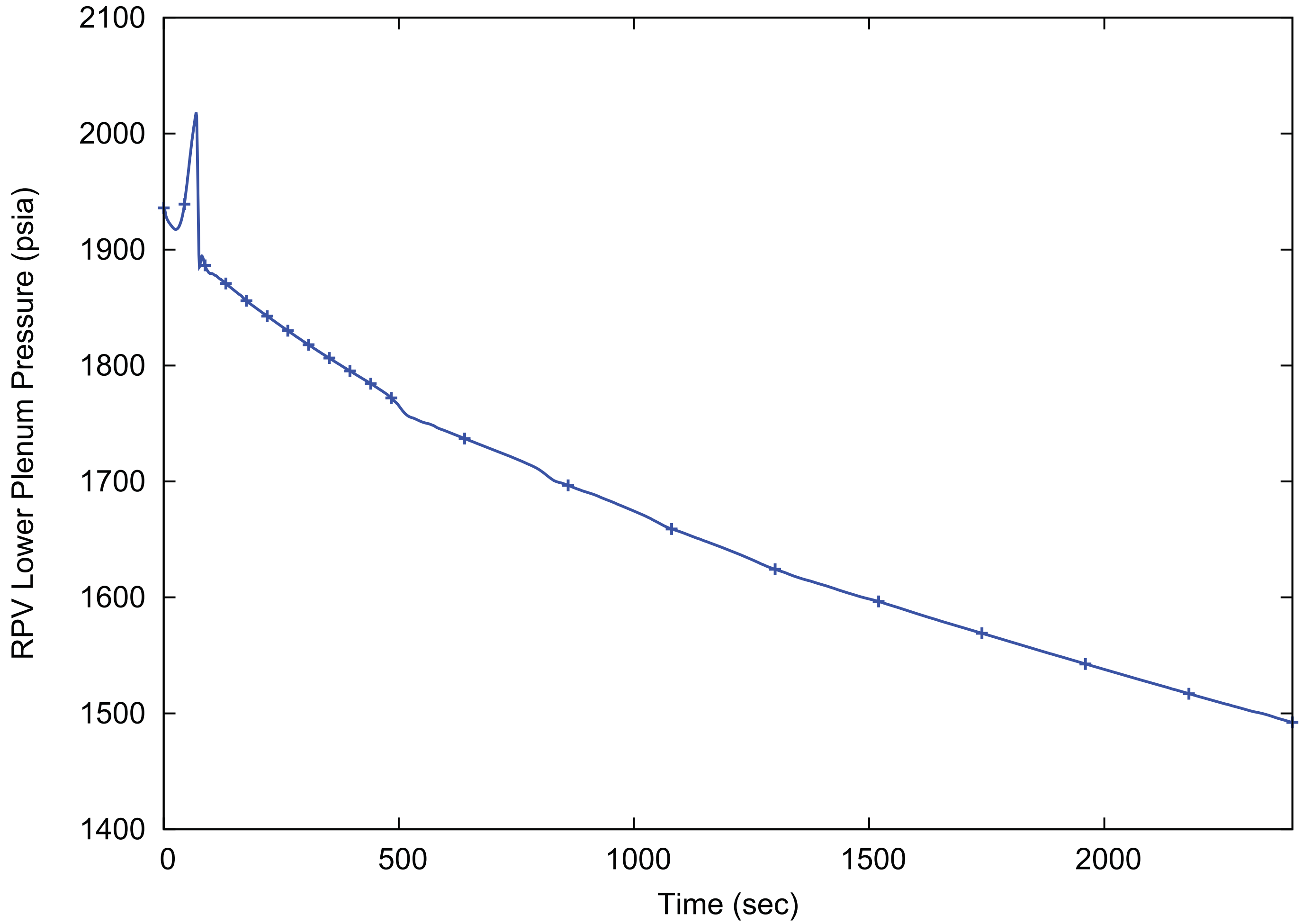
Figure 15.1-60: Reactor Coolant Pressure – Entire Transient (15.1.3 Increase in Steam Flow)



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

eRAI No.: 9483
Date of RAI Issue: 05/11/2018

NRC Question No.: 15.01.01-10

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 13 requires the provision of instrumentation to monitor variables and systems over their anticipated ranges of normal operation, and of appropriate controls to maintain listed variables and systems within prescribed operating ranges. GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

The staff notes that FSAR Tier 2, Sections 15.1.1-15.1.3 contain several apparent typographical errors that affect technical meaning or details. These errors are listed below:

- FSAR Tier 2, Section 15.1.1.2 states that the steam outlet boundary is modeled as a constant mass flow. However, based on the staff's audit of engineering calculation (EC)- 0000-2017, "Decrease in Feedwater Temperature Analysis" (the analysis supporting FSAR Section 15.1.1), the limiting case uses a constant steam pressure boundary.
- FSAR Tier 2, Section 15.1.1.2 discusses a decrease in FW event in the first sentence of the fourth full paragraph on Page 15.1-2 instead of a decrease in FW temperature event.
- FSAR Tier 2, Section 15.1.1.2 refers to steam flow events rather than a decrease in FW temperature event on Page 15.1-3.
- FSAR Tier 2, Section 15.1.1.3.3 states that the maximum secondary pressure is reached "just after" main steam and FW isolation, which seems inconsistent with FSAR Tier 2, Table 15.1-1, which shows that the maximum secondary pressure is actually reached 41 seconds after main steam and FW isolation.
- FSAR Tier 2, Section 15.1.2.3.3 states that SG pressure does not change significantly during the initial phase of the transient. However, based on FSAR Tier 2, Figure 15.1-18, the SG pressure increases rapidly prior to DHRS actuation.
- FSAR Tier 2, Section 15.1.3.3.3 states that reactor power reaches a peak at approximately 58 seconds for the increase in steam flow event. However, FSAR Tier 2, Table 15.1-5 shows that the peak reactor power is reached at 55 seconds.

- FSAR Tier 2, Table 15.1-5, "Sequence of Events (15.1.3 Increase in Steam Flow)," shows that the reactor trips on high hot leg temperature instantaneously instead of after the 8-second actuation delay specified in FSAR Tier 2, Table 15.0-7.
- FSAR Tier 2, Table 15.1-6, "Increase in Steam Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)," lists SG temperature as a parameter, when it appears it should be SG pressure.
- Figure 15.1-22, "Steam Flow (15.1.2 Increase in Steam Flow)," appears to be mistitled since the increase in steam flow event is described in FSAR Section 15.1.3.
- FSAR Tier 2, Figure 15.1-26 appears to be mistitled, as it shows reactor power but is titled "Total Core Reactivity."

The information in the design certification application that supports meeting the regulations (as cited above) needs to be precise and consistent so the staff is able to make a reasonable assurance finding. Please address the above items by either (1) updating the FSAR to correct them or (2) justifying why the information is correct.

NuScale Response:

FSAR Section 15.1.1.2 incorrectly stated that the steam outlet boundary was modeled as a constant mass flow. The steam outlet boundary is modeled at a constant steam pressure. However, both boundary conditions were analyzed and no sensitivity was observed. So, the statement has been removed from the FSAR as shown in the markup with this response.

The remaining errors identified in this request for additional information have been corrected as shown in the markup with this response.

Impact on DCA:

FSAR Section 15.1.1, Section 15.1.2, Section 15.1.3, Table 15.1-7, Table 15.1-8,Figure 15.1-22, and Figure 15.1-26 have been revised as described in the response above and as shown in the markup provided with this response.

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 design-specific event that causes an overcooling of the primary system due to a loss of the 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 transient response of the plant to these events is described in the following sections:

- Section 15.1.1 - Decrease in Feedwater Temperature
- Section 15.1.2 - Increase in Feedwater Flow
- Section 15.1.3 - Increase in Steam Flow
- Section 15.1.4 - Inadvertent Opening of Steam Generator Relief or Safety Valve
- Section 15.1.5 - Steam Piping Failures Inside and Outside of Containment
- Section 15.1.6 - Loss of Containment Vacuum/Containment Flooding

15.1.1 Decrease in Feedwater Temperature

15.1.1.1 Identification of Causes and Accident Description

A decrease in feedwater temperature could be caused by a failure in the feedwater system. A lower feedwater temperature would increase the heat removal from the primary system, leading to a higher moderator density. As the reactor coolant system (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 an increase in core power, and a decrease in the minimum critical heat flux ratio (MCHFR).

A decrease in feedwater temperature is expected to occur one or more times in the life of the reactor, so it is classified as an AOO. The categorization of the NuScale Power Plant design basis events (DBEs) is discussed in Section 15.0.

15.1.1.2 Sequence of Events and Systems Operation

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

Unless specified below, the analysis of a decrease in feedwater temperature assumes the plant control systems (PCSs) 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 feedwater temperature event.

RAI 15.01.01-10

Normally, as the colder feedwater enters the steam generator (SG), the feedwater controls would reduce feedwater flow, helping to mitigate the overcooling event. This event is conservatively analyzed with the feedwater controls disabled, forcing a constant feedwater flow rate, and is not credited to mitigate the event. Similarly, the turbine throttle and stop valves are not credited with allowing pressurization of the steam line. ~~The steam outlet boundary is conservatively modeled as a constant mass flow to maximize the overcooling event.~~

The reactivity control system is assumed to be in normal automatic mode. Operator action is not credited for regulating control rod movement or increasing boron concentration. This ensures that 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 feedwater temperature. If the feedwater temperature were to drop to a level that causes a high enough power excursion, the MPS high power signal would trip the reactor, preventing the reactor from reaching a power level where the acceptance criteria could be challenged. The following MPS signals provide the plant with protection during a decrease in feedwater temperature:

- high core power (5 percent uncertainty added)
- high core power rate (not credited in the safety analysis of this event)
- high hot leg temperature
- high steam superheat
- low pressurizer (PZR) level
- low PZR pressure

Due to the cooling of the RCS during a decrease in feedwater 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 core power rate trip is not credited in the analyses, and a 5 percent uncertainty is added to the high core power trip.

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In a decrease in feedwater temperature event that results in a reactor trip, the subsequent actuation of the decay heat removal system (DHRS) is credited with maintaining reactor cooling. The MPS signals credited for DHRS actuation are low PZR pressure, high steam superheat, low PZR level, high steam pressure, or high hot leg temperature.

There are no single failures that could make the limiting decrease in feedwater temperature MCHFR case more severe. A single failure of one of the main steam isolation valves (MSIVs) could allow a brief continuation of RCS overcooling. If the non-safety related turbine trip is not credited with mitigating this single failure, there is a short period of time between the failure of the MSIV and the closure of the backup isolation valve that would allow this continuation of RCS overcooling. However, this single failure does not affect the MCHFR because the reactor has already tripped when the failure occurs. Therefore, this single failure is not modeled in the case presented in this section, which demonstrates the limiting decrease in feedwater temperature with respect to MCHFR.

Normal alternating current (AC) power is assumed to be available for this event. A loss of AC power is not a conservative condition for a decrease in feedwater temperature event. The loss of normal power scenarios are listed below:

- Loss of normal AC – In this scenario, MPS remains powered so none of the safety systems are automatically actuated, but feedwater is lost and the turbine is tripped.

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  - Loss of normal AC at the time of the ~~steam flow increase~~decrease in feedwater temperature is non-limiting because feedwater is lost which reduces the overcooling event.
  - Loss of normal AC at the time of reactor trip is non-limiting because feedwater is lost which reduces the overcooling event.

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- Loss of the normal DC power system (EDNS) and normal AC – Power to the reactor trip breakers is provided via the EDNS, so this scenario is the same as discussed above with addition of reactor trip at the time at which power is lost. For the ~~increase in steam flow~~decrease in feedwater temperature events, this scenario is non-limiting for the reasons listed above and from the immediate reactor trip.
- Loss of the highly reliable DC power system (EDSS), EDNS and normal AC – Power to the MPS is provided via the EDSS so this scenario results in an actuation of reactor trip and all of the ESFs. In terms of the overcooling event, this scenario is non-limiting for the reasons discussed above.

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 feedwater temperature is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale Power Module (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. See Section 15.0.2. for a discussion of the VIPRE-01 code and evaluation model.

15.1.1.3.2 Input Parameters and Initial Conditions

The lowest feedwater temperature that could result from a failure in the feedwater system (FWS) is 100 degrees F. This is the temperature of the feedwater before it passes through the feedwater heaters. Sensitivities on a spectrum of feedwater

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 high hot leg temperature reactor limit is reached at about 125 seconds into the transient. During the assumed reactor trip signal delay, the rise in reactor power initiates a high power reactor trip at approximately 131 seconds into the transient. The peak RCS pressure occurs just prior to the reactor trip. The high hot leg temperature trip also actuates the DHRS valves to open. The feedwater isolation valves (FWIVs) and MSIVs close, isolating the SG from the rest of the secondary system.

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Steam generator pressure does not change significantly during the initial phase of the transient. However, after DHRS actuation at 133 seconds, the closure of the FWIVs and MSIVs causes 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 feedwater temperature event itself. The maximum secondary pressure is reached ~~just~~ after main steam and feedwater isolation.

The CHFR decreases as reactor power and RCS pressure increase. The automatic protection systems terminate this transient before the CHFR reaches the design limit. The MCHFR for the limiting decrease in feedwater temperature does not violate the CHF limit.

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During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. At approximately 2000 seconds, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of DHRS, and stabilization of RCS flow demonstrate the plant response to a decrease in feedwater temperature, and a return to a stable condition with no operator actions. For a discussion on possible return to power scenarios, see Section 15.0.6.

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 five Design Specific Review Standard (DSRS) acceptance criteria for this AOO are met for the limiting decrease in feedwater temperature case. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below:

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

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 peak RCS pressure occurs around the time of the reactor scram. The FWIVs and MSIVs close, isolating the SG from the rest of the secondary system.

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Steam generator pressure ~~does not change significantly~~increases rapidly during the initial phase of the transient. ~~However, after~~Pressure in the steam generator continues to increase following DHRS actuation, the closure of the FWIVs and MSIVs causes pressurization of the SG. This second ~~S~~steam generator pressure increase resulting from main steam isolation is expected and is not a direct consequence of the increase in feedwater flow event itself. The maximum secondary pressure is well below the design pressure of the steam system piping. The steam generator level calculated during an increase in feedwater flow demonstrates that the steam generator will not overfill.

The MCHFR for the limiting increase in feedwater flow event does not fall below the 95/95 acceptance criterion discussed in Section 4.4.4. The MCHFR for the limiting increase in feedwater flow case does not violate the design limit.

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. Eventually, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of DHRS, and stabilization of RCS flow demonstrate the plant response to increase in feedwater flow, and a return to a stable condition with no operator actions. For a discussion on possible return to power scenarios, see Section 15.0.6.

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 15.0.4.

15.1.2.5 Conclusions

The five DSRS acceptance criteria for this AOO are met for the limiting increase in feedwater flow case. These acceptance criteria, followed by how the NuScale Power Plant design meets them are listed below:

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 AOOs presented in Section 15.2, decrease in heat removal by the secondary side. Therefore, this acceptance criterion is met for increase in steam flow event. The maximum pressure values for the cases analyzed are shown in Table 15.1-6.

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. See Section 15.0.2 for 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 14 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. This conservatively bounds the possible rates of steam flow increases.

The initial conditions used in the evaluation of the limiting increase in steam flow event result in a conservative calculation. Table 15.1-8 provides key inputs for the limiting increase in steam flow case. 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 2 percent measurement uncertainty.
- Conservative scram 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 end-of-cycle core parameters is used to provide a limiting power response. The most negative MTC of -43.0 pcm/degrees F and the least negative DTC of -1.40 pcm/degrees F are used to provide the largest power response for this 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.

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 15.0.2.3.

15.1.3.3.3 Results

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The sequence of events for the limiting increase in steam flow event is provided in Table 15.1-7. Figure 15.1-22 through Figure 15.1-31 and Figure 15.1-58 through Figure 15.1-60 show the transient behavior of key parameters for an increase in steam flow event. A spectrum of steam flow increase cases were analyzed, and the limiting steam increase is a case in which the cooldown rate produces a power response that peaks just below the high power trip. The limiting event initiates with an increase in steam flow by approximately 14 percent of full power steam flow in 0.1 seconds. The feedwater pump 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 SG begins to increase. 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. ~~Reactor power reaches a peak at approximately 58 seconds.~~ The peak reactor power is just below the high core power limit of the RTS. If the regulating control rod bank were disabled, a similar power response would be driven by moderator feedback.

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 high hot leg temperature limit is reached at 60 seconds into the transient, tripping the reactor. The peak RCS pressure occurs around the time of the reactor scram. The high hot leg temperature signal also actuates the DHRS valves to open. The FWIVs and MSIVs close, isolating the SG from the rest of the secondary system.

Steam generator pressure does not change significantly during the initial phase of the transient. However, after DHRS actuation at 68 seconds, the closure of the FWIVs and MSIVs 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 feedwater isolation.

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

During the overcooling phase, RCS flow steadily increases in response to rising reactor power. The reactor scram causes a rapid decrease in flow as the heat source driving natural circulation is reduced. The flow oscillates until RCS temperatures re-equilibrate. After 2000 seconds, RCS flow stabilizes and passive DHRS cooling dominates. The reactor trip, subsequent actuation of 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. For a discussion on possible return to power scenarios, see Section 15.0.6.

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.

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Table 15.1-7: Sequence of Events (15.1.3 Increase in Steam Flow)

Event	Time [s]
Steam flow begins to increase	0
Peak reactor power is reached	55
High hot leg temperature limit is reached	60
Reactor trips on high hot leg temperature signal	68~~0~~
DHRS actuation	68
Peak RPV pressure is reached	69
FWIVs and MSIVs fully close	75
Peak MSS pressure is reached	123
CVCS isolation on low pressurizer pressure	1353

RAI 15.01.01-5, RAI 15.01.01-8, RAI 15.01.01-10

Table 15.1-8: Increase in Steam Flow - Inputs (Limiting Minimum Critical Heat Flux Ratio Case)

Parameter	Nominal	Bias
Core power	160 MWt	+2%
Pressurizer pressure	1850 psia	+70psia
RCS flow rate	See Table 15.0-6 for range	1172 lbm/s
RCS average temperature	545 °F	+10 °F
SG ~~temperature~~pressure	500 psia	+35psia
Pressurizer level	60%	+8%
SG heat transfer	Nominal	-30%
FW temperature	300 °F	-10 °F
Reactor pool temperature	40 °F - 200 °F	Minimum (40 °F)
DHRS heat transfer	Nominal	+30%
MTC	EOC	Most Negative
DTC	BOC	Least Negative

RAI 15.01.01-10

Figure 15.1-22: Steam Flow (15.1.3~~2~~ Increase in Steam Flow)

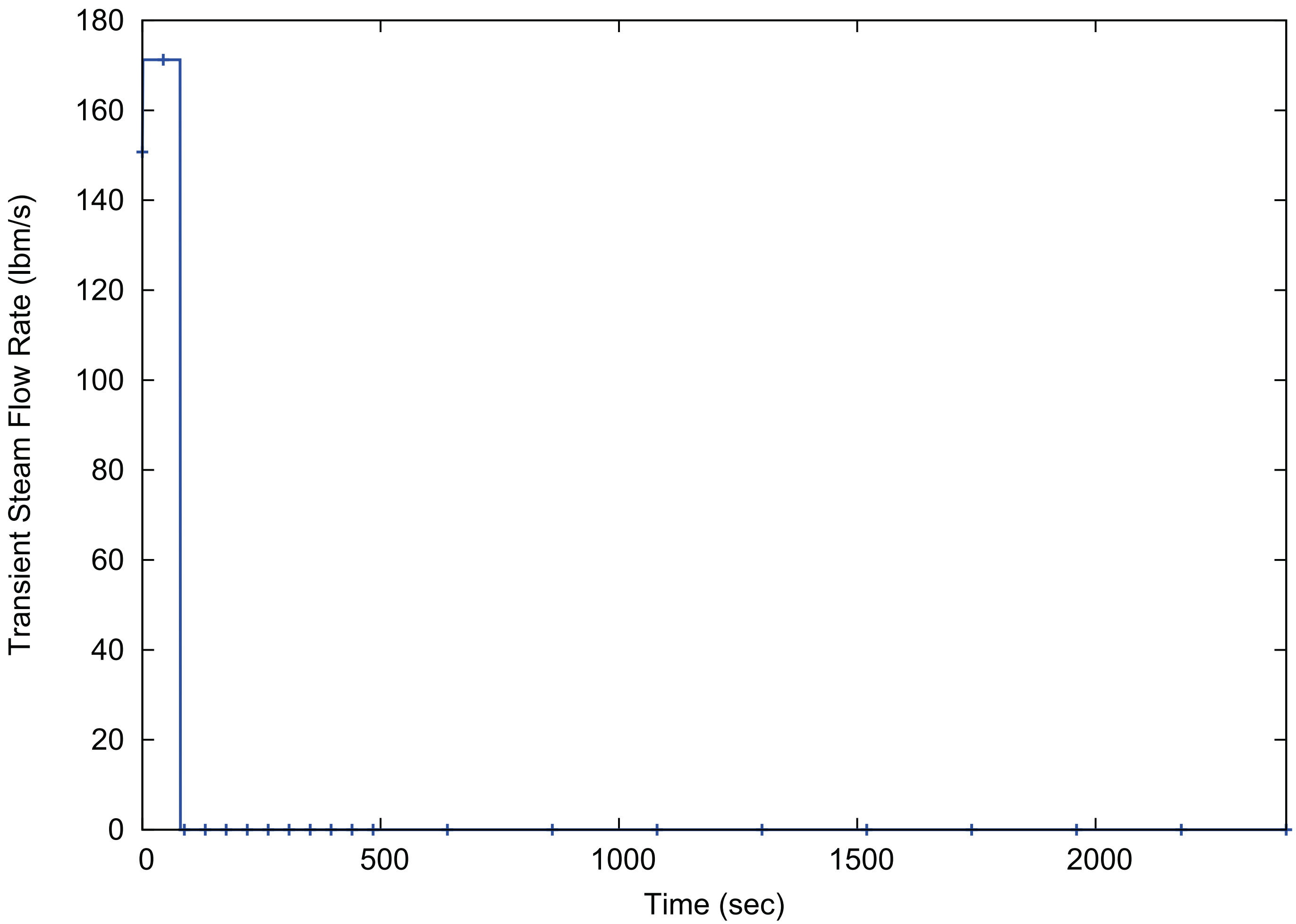

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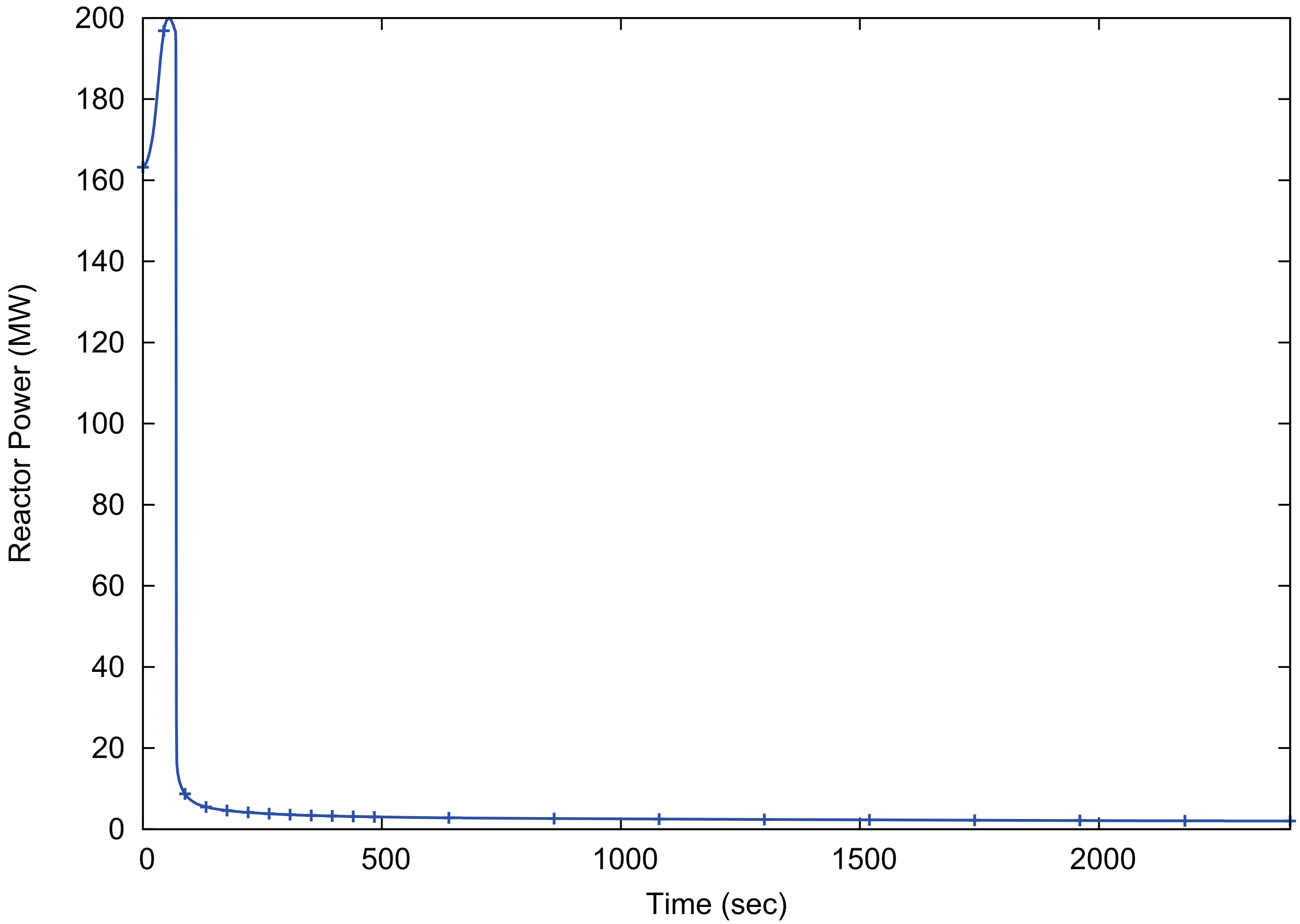
Figure 15.1-26: ~~Total Core Reactivity~~Reactor Power (15.1.3 Increase in Steam Flow)



Enclosure 3:

Affidavit of Zackary W. Rad, AF-0718-60802

NuScale Power, LLC
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NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, 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 Request for Additional Information response reveals distinguishing aspects about the process and method by which NuScale develops its safety analysis of the Nuscale Power Module.

    NuScale has performed significant research and evaluation to develop a basis for this process and 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 response to NRC Request for Additional Information No. 473, eRAI No. 9483. 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 July 10, 2018.

Zackary W. Rad