June 21, 2017

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. 03 (eRAI No. 8744) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 03 (eRAI No. 8744)," dated April 25, 2017

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 Questions from NRC eRAI No. 8744:

- 15.02.08-1
- 15.02.08-2
- 15.02.08-3
- 15.02.08-4

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 Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, TWFN-6E55
Samuel Lee, NRC, TWFN-6C20
Rani Franovich, NRC, TWFN-6E55

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8744

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8744

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

eRAI No.: 8744
Date of RAI Issue: 04/25/2017

NRC Question No.: 15.02.08-1

In accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 31, “Fracture prevention of reactor coolant pressure boundary,” the reactor coolant pressure boundary (RCPB) shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties; (2) the effects of irradiation on material properties; (3) residual, steady state and transient stresses; and (4) size of flaws.

To meet the requirements of GDC 31, as it relates to the feedwater line break (FWLB) accident resulting in a limiting RCS pressure, the accident analysis should consider appropriate uncertainties for determining conservative temperatures and pressures at the RCPB to show that the probability of rapidly propagating fracture is minimized for this transient.

In Final Safety Analysis Report (FSAR) Tier 2, Section 15.2.8.4, “Input Parameters and Initial Conditions,” the applicant states that a 30% uncertainty is added to the steam generator heat transfer. However, in FSAR Tier 2, Table 15.2-28, “Biases and Uncertainties – Feedwater Line Break,” the applicant states that a -30% uncertainty is used for the steam generator heat transfer. The staff notes that these statements in the FSAR are inconsistent. Based on the docketed information, the staff is unable to determine what uncertainty value is used for the steam generator heat transfer and the adequacy of the reported 30% uncertainty addition to the steam generator heat transfer. The staff requests the applicant to clearly state in the FSAR which uncertainty value is used and provide justification in the FSAR as to why the applicant adds to (or subtracts from) the steam generator heat transfer.

NuScale Response:

The limiting reactor coolant system (RCS) pressure transient results from the double-ended guillotine (DEG) feedwater line break (FWLB). For cases evaluated to maximize primary pressure, a negative 30% bias to the primary-to-secondary heat transfer rate was applied to maximize the heat retained in the RCS. In addition, the design limit steam generator (SG) fouling factor (10^{-4} hr-ft^2 - $^{\circ}$F/BTU) and SG tube plugging (10%) were applied to maximize RCS heatup and, therefore, RCS pressure. The limiting decay heat removal system (DHRS) case was also a DEG FWLB using the same heat transfer bias, fouling and tube plugging values as the limiting RCS pressure case.

For the DEG FWLB evaluated for the limiting SG pressure, the highest pressure was obtained by using a positive 30% heat transfer bias, no fouling and no SG tube plugging. However the DEG FWLB was not the limiting break for SG pressure. Smaller breaks of 8% to 10% of feedwater flow with a negative 30% primary to secondary heat transfer bias, a large fouling factor and 10% tube plugging, resulted in slightly higher SG pressure. The 10% FWLB with failure of the feedwater isolation valve (FWIV) backflow prevention device was selected as the representative limiting SG case for the sequence of events FSAR Table 15.2-25 and SG pressure FSAR Figures 15.2-42 and 15.2-43. Similar FWLBs, 4% to 8% break size, with the same heat transfer biases, resulted in the minimum critical heat flux ratio (MCHFR). The 8% FWLB was selected as the representative limiting MCHFR case for the sequence of events in FSAR Table 15.2-26 and MCHFR Figure 15.2-44. Note that the title of FSAR Figure 15.2-44 was mislabeled as the MCHFR for the “Maximum RCS Pressure Case” and has been corrected to be the “Limiting MCHFR Case” as shown in the attached FSAR markup.

Therefore, the limiting cases for RCS pressure, DHRS performance, SG pressure and MCHFR all used the heat transfer bias of -30% stated in FSAR Table 15.2-28. The text in FSAR Section 15.2.8.4 was intended to reflect the information in the table. The wording in the attached FSAR markup has been clarified to eliminate the discrepancy. It is noteworthy that varying parameters for heat transfer, SG tube plugging and fouling had less than a 10 psi effect on peak primary or secondary pressure. The limiting calculated RCS pressure (2164 psia) and SG pressure (1328 psia) were well below the acceptance criteria (2310 psia) such that the variation of these parameters was not a significant contributor to determining the acceptability of the results. Additional detail on the basis for sensitivity studies is presented in the Non-Loss-of-Coolant Accident Analysis Methodology, TR-0516-49416-P, Revision 0, Section 4.2 and Section 7.2.12.

During separate discussions with NRC staff, questions were also raised regarding the wording in FSAR Section 15.2.8.4 relating to the assumptions on closure time for the FWIVs. When credited, the FWIVs are assumed to close in the maximum design closure time (7 seconds), not at the design limit closure rate. For the majority of events, the safety-related check valve

(backflow prevention device) will seat such that the closure time of the FWIV is not required to prevent backflow though the FW system. For these events, the FWIV closure time is not a consideration because the check valve closes very quickly (approximately 1 second). Therefore, the FWIV closure is removed from the description where the safety-related check valve is credited. The description of these input parameters is clarified in the attached FSAR markup. The description of the sequence of events in FSAR Section 15.2.8.2 is also modified to reflect the limiting cases discussed in this response.

In the course of preparing this response, NuScale has included some editorial corrections and clarifications in the attached FSAR markup. In FSAR Section 15.2.8.2, an incorrect characterization of the limiting MCHFR case was eliminated. In FSAR Section 15.2.8.4, the term “Case A” was deleted because it refers to an internal calculation reference. Additionally, wording was added to clarify that the FWIV backflow prevention device is the safety-related FW check valve. The description of the limiting SG pressure event was also corrected to indicate that AC power is lost at the time of the break. In FSAR Section 15.2.8.5, a statement was added to clarify that the high pressurizer pressure reactor trip is delayed by biasing pressurizer pressure and level low.

Impact on DCA:

FSAR Sections 15.2.8.2, 15.2.8.4, 15.2.8.5, and FSAR Figure 15.2-44 have been revised as described in the response above and as shown in the markups provided in this response.

non-impacted steam generator system and DHRS loop will continue to provide cooling to the RCS.

Feedwater breaks outside of containment will cause a loss of feedwater flow to the steam generators and a heatup and subsequent pressure increase in the RCS. Larger breaks will cause a rapid heatup and will trip the reactor and actuate DHRS on high pressurizer pressure, whereas smaller breaks will cause a gradual heatup and loss of pressure in the main steam system resulting in high RCS temperature signals or a high steam superheat signal that will actuate the RTS and DHRS. The actuation of DHRS isolates the secondary piping and provides cooling and depressurization to the RCS via the intact loop.

A break in the feedwater line is not expected to occur during the life of the NPM, so it has been classified as an accident as shown in Table 15.0-1.

15.2.8.2 Sequence of Events and Systems Operation

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

For some FWLBs with AC power available, the loss of FW flow causes steam pressure to decrease resulting in an RTS and DHRS actuation on low steam pressure. Depending on the size of the break and initial conditions, the high steam superheat trip is reached in some events before SG pressure drops below the low pressure setpoint.

For FWLB events where AC power is lost, the turbine and feedwater pumps trip resulting in a loss of RCS cooling and rapid pressurization. In these cases, the high PZR pressure signal will cause a reactor trip and DHRS actuation.

For cases where DC power (EDSS) is lost, a reactor trip and CNV isolation occur at the time of the FWLB, which results in less limiting break scenarios as mitigating actions are accomplished sooner. Therefore, DC power is assumed to be available for all cases presented in this section.

Breaks inside containment cause a rapid pressurization of the CNV resulting in a reactor trip and CNV isolation with a DHRS actuation. These break locations are not limiting for RCS pressure, SG pressure or MCFHR, but are the most challenging to the DHRS as one heat exchanger is disabled with the inventory released inside the CNV. The CNV response is addressed in Section 6.2.1.4.

RAI 15.02.08-1, RAI 15.02.08-2

The limiting FWLB for RCS pressure is a double ended guillotine (DEG) break outside containment with a coincident loss of AC power. The combination of turbine trip and the feedwater pipe break result in a loss of cooling to the primary and a heatup and pressurization of the RCS. Reactor trip and DHRS~~,~~ actuation occur from the high pressurizer pressure MPS signal followed by the lifting of an RSV that quickly reduces primary pressure below the RSV reset pressure. Secondary pressure increases during

the event until stable DHRS cooling is established, at which point the temperature and pressure in both the primary and secondary decrease. Table 15.2-24 provides the sequence of events for the limiting RCS pressure case.

RAI 15.02.08-1

The limiting MCHFR FWLB event was ~~essentially the same as the limiting RCS pressure event, with a slightly lower MCHFR~~ observed for small feedwater breaks ~~resulting in a decrease of 4% to 8% in FW flow~~. The decrease in heat removal from the loss of AC power at the time of the break had a more significant impact on MCHFR than the size of the break. The limiting MCHFR sequence of events is provided in Table 15.2-26.

RAI 15.02.08-2

The limiting SG pressure case occurs from a small FW line break ~~decreasing flow by approximately 8% with AC power available during the transient. In this case, the decreased flow results in reactor trip and DHRS actuation on the high steam super heat MPS signal. A heatup and subsequent pressure increase occur in the RCS, but the pressure increase peaks below the high pressurizer pressure MPS actuation setpoint~~with a loss of AC power at the time of the break. Similar peak SG pressure was reached by the 8% break with no additional failures and the 10% break with failure of a FWIV backflow prevention device to seat. The peak secondary pressurization is a function of the DHRS heat removal capacity. Table 15.2-25 provides the sequence of events for the limiting secondary pressure case resulting from the 10% FW line break with the failure of the FWIV backflow prevention device.

The limiting DHRS function case involves a DEG break in the feedwater piping inside of containment. Unlike breaks outside of containment, this break results in the complete loss of one train of DHRS. Upon break initiation, pressure inside of the CNV rapidly increases, reaching the high CNV pressure analytical limit and actuating reactor trip, containment isolation, and DHRS. The remaining DHRS loop provides cooling to the module and is sufficient to remove 100% of decay heat and drive flow through the core. This event is not limiting for any of the acceptance criteria. The sequence of events for this case is provided in Table 15.2-27.

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

- Low steam pressure
- High pressurizer pressure
- High steam superheat
- High CNV pressure

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

RAI 15.02.08-1

No single failures have an impact on the limiting primary pressure~~, SG pressure~~ or MCHFR results. ~~One failure of note is the potential~~The failure of the safety-related check valve (FWIV backflow prevention device) to close on the intact SG did result in a

limiting value for SG pressure. For FWLB inside containment, in the event of the failure of the safety-related check valve, ~~. In the event of this failure,~~ the second nonsafety-related check valve is credited to ensure that adequate inventory is maintained in the intact steam generator and DHRS condenser. Therefore, there are no single active failures that cause the FWLB event to have unacceptable results.

15.2.8.3 Thermal Hydraulic and Subchannel Analyses

15.2.8.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to an FWLB is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.2.8.4 Input Parameters and Initial conditions

The FWLB inside containment scenarios are bounded by the consequences of the FWLB outside of containment from a primary and secondary system response perspective. The initial conditions and assumptions used in the evaluation of a FWLB event result in a conservative calculation.

RAI 15.02.08-1

The DEG break size outside of containment and loss of power assumptions for the FWLB ~~Case A~~ cause a loss of cooling effect that heats up and pressurizes the RCS, resulting in the limiting RCS pressure. The following initial conditions are assumed in the analysis of the FWLB ~~Case A~~ to ensure that the transient results in the limiting RCS pressure.

- Initial power level is assumed to be 102% of nominal to account for a 2% 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 limiting beginning of cycle (BOC) core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/°F) and DTC (-1.4 pcm/°F) are used to minimize the power response for this event.
- A loss of offsite power is assumed to occur at the time of the break, causing an immediate turbine and feedwater pump trip. The normal and highly reliable DC power systems are assumed to be available, which delays the reactor trip and MPS actuations that would occur immediately on a loss of DC power.

RAI 15.02.08-1

- ~~The FWIVs are assumed to close at the maximum design limit closure rate while~~The FWIV backflow prevention device is assumed to close on the faulted FW line to minimize fluid loss, which maximizes the RCS heatup. MSIVs are assumed to close rapidly to maximize the heatup affect.
- System biases include: high RCS temperature, high fuel temperature, high pressurizer pressure, high pressurizer level, and minimum RCS flow.
- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30% uncertainty ~~added to~~ for the primary to secondary~~the~~ heat transfer.

RAI 15.02.08-1

The limiting MCHFR event is the ~~4%~~8% split break size outside of containment with a loss of AC power. This FWLB involves a loss of cooling that heats up and pressurizes the RCS, resulting in the limiting MCHFR. The following initial conditions are assumed in the analysis to ensure that the transient results in the limiting CHF ratio.

- Initial power level is assumed to be 102% of nominal to account for a 2% 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 limiting BOC core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/°F) and DTC (-1.4 pcm/°F) are used to minimize the power dampening response for this heatup event.
- A loss of offsite power is assumed to occur at the time of the break, causing an immediate turbine and feedwater pump trip. The normal and highly reliable DC power systems are assumed to be available, which delays a reactor trip from mitigating the power excursion.

RAI 15.02.08-1

- ~~The FWIVs are assumed to close at the maximum design limit closure rate while~~The FWIV backflow device is assumed to close on the faulted FW line to minimize fluid loss, which maximizes the RCS heatup. MSIVs are assumed to close rapidly to maximize the heatup affect.
- System biases include: high RCS temperature, high fuel temperature, low pressurizer pressure, low pressurizer level, and minimum RCS flow.
- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30% uncertainty ~~added to the~~for the primary to secondary heat transfer.

RAI 15.02.08-1

The limiting event for SG pressure is the ~~4%~~10% split break size outside of containment that causes a loss of cooling effect that heats up and increases pressure in the primary and increases the superheat in the steam generator. The resulting effect in conjunction with actuation of DHRS results in the limiting SG pressure. The following initial

conditions are assumed in the analysis to ensure that the transient results in the limiting secondary pressure.

- Initial power level is assumed to be 102% of nominal to account for a 2% 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 limiting BOC core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/°F) and DTC (-1.4 pcm/°F) are used to minimize the power response for this event.
- ~~AC power is assumed to be available with the turbine trip occurring at the time of reactor trip.~~A loss of offsite power is assumed to occur at the time of the break, causing an immediate turbine and feedwater pump trip. The normal and highly reliable DC power systems are assumed to be available, which delays the reactor trip and MPS actuations that would occur immediately on a loss of DC power.
- The FWIVs are assumed to close at the ~~design limit closure rate while~~maximum design closure time to maximize fluid loss. MSIVs are assumed to close rapidly to maximize the heatup affect.
- The single failure assumed in this case was the failure of a FWIV backflow device to seat, reducing the inventory in the second SG train.
- System biases include: high RCS temperature, high fuel temperature, high pressurizer pressure and pressurizer level, and minimum RCS flow.

RAI 15.02.08-1

- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30% uncertainty ~~added to the~~for the primary to secondary heat transfer.

The DEG split break size inside of containment releases the inventory of the FW and DHRS to the CNV. The following initial conditions are assumed in the analysis to ensure that the transient results are conservative in the assessment of DHRS functionality.

- Initial power level is assumed to be 102% of nominal to account for a 2% 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 limiting BOC core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/°F) and DTC (-1.4 pcm/°F) are used to minimize the power response for this event.

RAI 15.02.08-1

- The FWIVs are assumed to close at the ~~design limit closure rate~~maximum design closure time to maximize fluid loss, while MSIVs are assumed to close rapidly to maximize the heatup affect.

- System biases include: high RCS temperature, high fuel temperature, low pressurizer pressure and level, and minimum RCS flow.
- DHRS performance is assumed to be low (-30%) with a high biased pool temperature.
- AC power is assumed to be available with the turbine trip occurring at the time of reactor trip.
- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30% uncertainty ~~added to the~~for the primary to secondary heat transfer.

The results from the thermal hydraulic evaluations are used as input to the subchannel analysis to determine the limiting MCHFR for this event. Other key inputs and assumptions used in the analysis are included in Table 15.2-28.

15.2.8.5 Results

For the limiting RCS pressure case, the FW line is assumed to have a double-ended guillotine break just outside containment. AC power is assumed to be lost, causing the turbine and feedwater pumps to trip, resulting in a rapid loss of cooling and large RCS pressurization. In this scenario, the reactor is tripped and DHRS is actuated by the high pressurizer pressure MPS signal. Pressure in the RCS continues to increase following reactor trip until the reactor safety valve lift limit is reached. Once the RSV lifts, system pressure quickly returns to below the RSV reset pressure. The limiting peak RCS pressure from FWLB is provided in Figure 15.2-35. DHRS cooling is sufficient to begin to depressurize the system such that a second RSV lift does not occur. The reactor power drops sharply following the reactor trip and follows the typical decay heat curve as shown on Figure 15.2-36. SG pressure rises equally in both SG as shown in Figure 15.2-37 and Figure 15.2-38. RCS temperature decreases and RCS flow stabilizes as shown in Figure 15.2-39 and Figure 15.2-40, respectively. The DHRS actuation closes the FWIVs and stops the loss of secondary fluid. The integrated break flow for the event is shown in Figure 15.2-41.

The peak SG pressure case results from a small split line break in the FW line just outside containment. AC power is assumed to be lost at the time of the break. The FWIV backflow check valve is assumed to fail to seat upon reverse flow. This condition results in asymmetrical pressure in the SGs as shown in Figure 15.2-42 and Figure 15.2-43. The intact SG has the peak SG pressure (Figure 15.2-42).

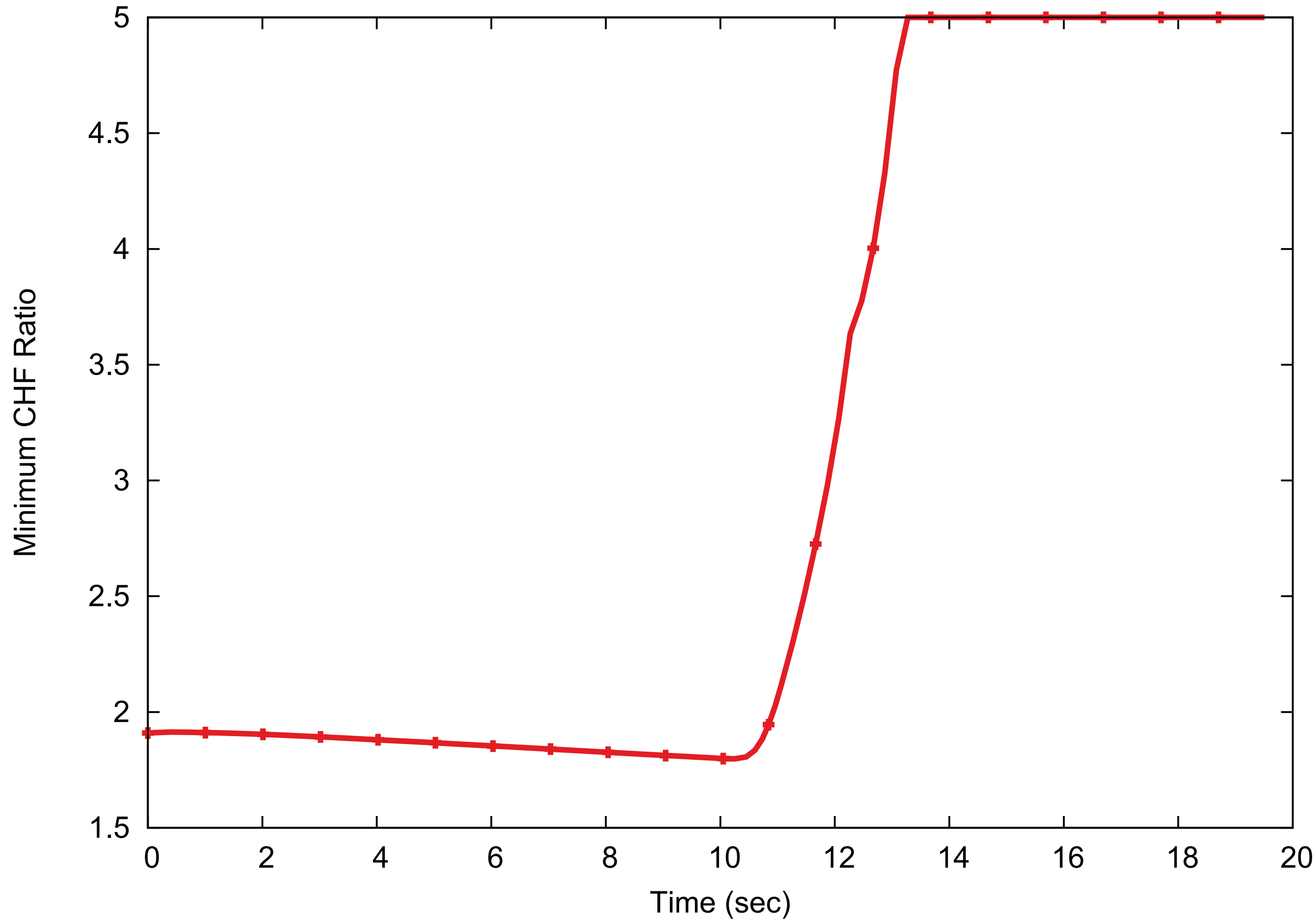
RAI 15.02.08-1

In the limiting MCHFR case, the high pressurizer pressure trip is delayed, by biasing pressurizer pressure and level low, resulting in higher temperatures and the MCHFR occurs right before the reactor trips. The MCHFR versus time is presented on Figure 15.2-44.

When the break is inside containment, the rapid pressurization of the CNV results in the reactor trip, containment isolation, and DHRS actuation and therefore is not limiting for pressure or CHF criteria. Such a break results in the limiting function of the DHRS. The break results in the loss of an entire DHRS loop as the fluid from the feedwater line

RAI 15.02.08-1

Figure 15.2-44: Hot Channel Node MCHFR - ~~Peak RCS Pressure~~Limiting MCHFR Case (15.2.8 Feedwater Line Break)



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

eRAI No.: 8744
Date of RAI Issue: 04/25/2017

NRC Question No.: 15.02.08-2

In accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 31, "Fracture prevention of reactor coolant pressure boundary," the reactor coolant pressure boundary shall be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties; (2) the effects of irradiation on material properties; (3) residual, steady state and transient stresses; and (4) size of flaws.

To meet the requirements of GDC 31, as it relates to the feedwater line break (FWLB) accident resulting in a limiting reactor coolant system (RCS) pressure, the accident analysis should show the maximum RCS pressure.

In Final Safety Analysis Report (FSAR) Tier 2, Section 15.2.8.2, "Sequence of Events and Systems Operation," the applicant states that for the limiting steam generator pressure case, the RCS pressure peaks below the high pressurizer pressure module protection system (MPS) actuation setpoint. In FSAR Tier 2, Table 15.2-5, "Feedwater Line Break Sequence of Events – Peak Steam Generator Pressure Case," the applicant states that the high pressurizer pressure analytical limit of 2000 psia is reached at 5 seconds. The staff notes that these two statements in the FSAR are inconsistent. Based on the docketed information, the staff is unable to determine which results are correct and should be reviewed. The staff requests the applicant to update the FSAR with consistent information regarding this FWLB case.

NuScale Response:

As discussed in the response to RAI Question 15.02.08-1, the representative limiting steam generator (SG) pressure case was the 10% feedwater line break (FWLB) with a single failure of the feedwater isolation valve (FWIV) backflow prevention device (safety-related check valve). The information in FSAR Table 15.2-25 and Figures 15.2-42 and 15.2-43 present the

results for this case. The text in FSAR Section 15.2.8.2 described an earlier version of the FWLB case and has been revised in the attached FSAR markups to reflect the selected limiting case and resolve the inconsistency.

Impact on DCA:

FSAR Section 15.2.8.2 been revised as described in the response above and as shown in the markups provided in this response.

non-impacted steam generator system and DHRS loop will continue to provide cooling to the RCS.

Feedwater breaks outside of containment will cause a loss of feedwater flow to the steam generators and a heatup and subsequent pressure increase in the RCS. Larger breaks will cause a rapid heatup and will trip the reactor and actuate DHRS on high pressurizer pressure, whereas smaller breaks will cause a gradual heatup and loss of pressure in the main steam system resulting in high RCS temperature signals or a high steam superheat signal that will actuate the RTS and DHRS. The actuation of DHRS isolates the secondary piping and provides cooling and depressurization to the RCS via the intact loop.

A break in the feedwater line is not expected to occur during the life of the NPM, so it has been classified as an accident as shown in Table 15.0-1.

15.2.8.2 Sequence of Events and Systems Operation

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

For some FWLBs with AC power available, the loss of FW flow causes steam pressure to decrease resulting in an RTS and DHRS actuation on low steam pressure. Depending on the size of the break and initial conditions, the high steam superheat trip is reached in some events before SG pressure drops below the low pressure setpoint.

For FWLB events where AC power is lost, the turbine and feedwater pumps trip resulting in a loss of RCS cooling and rapid pressurization. In these cases, the high PZR pressure signal will cause a reactor trip and DHRS actuation.

For cases where DC power (EDSS) is lost, a reactor trip and CNV isolation occur at the time of the FWLB, which results in less limiting break scenarios as mitigating actions are accomplished sooner. Therefore, DC power is assumed to be available for all cases presented in this section.

Breaks inside containment cause a rapid pressurization of the CNV resulting in a reactor trip and CNV isolation with a DHRS actuation. These break locations are not limiting for RCS pressure, SG pressure or MCFHR, but are the most challenging to the DHRS as one heat exchanger is disabled with the inventory released inside the CNV. The CNV response is addressed in Section 6.2.1.4.

RAI 15.02.08-1, RAI 15.02.08-2

The limiting FWLB for RCS pressure is a double ended guillotine (DEG) break outside containment with a coincident loss of AC power. The combination of turbine trip and the feedwater pipe break result in a loss of cooling to the primary and a heatup and pressurization of the RCS. Reactor trip and DHRS~~,~~ actuation occur from the high pressurizer pressure MPS signal followed by the lifting of an RSV that quickly reduces primary pressure below the RSV reset pressure. Secondary pressure increases during

the event until stable DHRS cooling is established, at which point the temperature and pressure in both the primary and secondary decrease. Table 15.2-24 provides the sequence of events for the limiting RCS pressure case.

RAI 15.02.08-1

The limiting MCHFR FWLB event was ~~essentially the same as the limiting RCS pressure event, with a slightly lower MCHFR~~ observed for small feedwater breaks ~~resulting in a decrease of 4% to 8% in FW flow~~. The decrease in heat removal from the loss of AC power at the time of the break had a more significant impact on MCHFR than the size of the break. The limiting MCHFR sequence of events is provided in Table 15.2-26.

RAI 15.02.08-2

The limiting SG pressure case occurs from a small FW line break ~~decreasing flow by approximately 8% with AC power available during the transient. In this case, the decreased flow results in reactor trip and DHRS actuation on the high steam super heat MPS signal. A heatup and subsequent pressure increase occur in the RCS, but the pressure increase peaks below the high pressurizer pressure MPS actuation setpoint~~with a loss of AC power at the time of the break. Similar peak SG pressure was reached by the 8% break with no additional failures and the 10% break with failure of a FWIV backflow prevention device to seat. The peak secondary pressurization is a function of the DHRS heat removal capacity. Table 15.2-25 provides the sequence of events for the limiting secondary pressure case resulting from the 10% FW line break with the failure of the FWIV backflow prevention device.

The limiting DHRS function case involves a DEG break in the feedwater piping inside of containment. Unlike breaks outside of containment, this break results in the complete loss of one train of DHRS. Upon break initiation, pressure inside of the CNV rapidly increases, reaching the high CNV pressure analytical limit and actuating reactor trip, containment isolation, and DHRS. The remaining DHRS loop provides cooling to the module and is sufficient to remove 100% of decay heat and drive flow through the core. This event is not limiting for any of the acceptance criteria. The sequence of events for this case is provided in Table 15.2-27.

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

- Low steam pressure
- High pressurizer pressure
- High steam superheat
- High CNV pressure

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

RAI 15.02.08-1

No single failures have an impact on the limiting primary pressure~~, SG pressure~~ or MCHFR results. ~~One failure of note is the potential~~The failure of the safety-related check valve (FWIV backflow prevention device) to close on the intact SG did result in a

limiting value for SG pressure. For FWLB inside containment, in the event of the failure of the safety-related check valve,~~. In the event of this failure,~~ the second nonsafety-related check valve is credited to ensure that adequate inventory is maintained in the intact steam generator and DHRS condenser. Therefore, there are no single active failures that cause the FWLB event to have unacceptable results.

15.2.8.3 Thermal Hydraulic and Subchannel Analyses

15.2.8.3.1 Evaluation Model

The thermal hydraulic analysis of the NPM response to an FWLB is performed using NRELAP5. The NRELAP5 model is based on the design features of a NuScale module. The non-LOCA NRELAP5 model is discussed in Section 15.0.2.2.2. The relevant boundary conditions from the NRELAP5 analyses are provided to the downstream subchannel Critical Heat Flux (CHF) analysis.

The subchannel core CHF analysis is performed using VIPRE-01. VIPRE-01 is a subchannel analysis tool designed for general-purpose thermal hydraulic analysis under normal operating conditions, operational transients, and events of moderate severity. See Section 15.0.2.3 for a discussion of the VIPRE-01 code and evaluation model.

15.2.8.4 Input Parameters and Initial conditions

The FWLB inside containment scenarios are bounded by the consequences of the FWLB outside of containment from a primary and secondary system response perspective. The initial conditions and assumptions used in the evaluation of a FWLB event result in a conservative calculation.

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The DEG break size outside of containment and loss of power assumptions for the FWLB ~~Case A~~ cause a loss of cooling effect that heats up and pressurizes the RCS, resulting in the limiting RCS pressure. The following initial conditions are assumed in the analysis of the FWLB ~~Case A~~ to ensure that the transient results in the limiting RCS pressure.

- Initial power level is assumed to be 102% of nominal to account for a 2% 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 limiting beginning of cycle (BOC) core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/°F) and DTC (-1.4 pcm/°F) are used to minimize the power response for this event.
- A loss of offsite power is assumed to occur at the time of the break, causing an immediate turbine and feedwater pump trip. The normal and highly reliable DC power systems are assumed to be available, which delays the reactor trip and MPS actuations that would occur immediately on a loss of DC power.

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- ~~The FWIVs are assumed to close at the maximum design limit closure rate while~~The FWIV backflow prevention device is assumed to close on the faulted FW line to minimize fluid loss, which maximizes the RCS heatup. MSIVs are assumed to close rapidly to maximize the heatup affect.
- System biases include: high RCS temperature, high fuel temperature, high pressurizer pressure, high pressurizer level, and minimum RCS flow.
- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30% uncertainty ~~added to~~ for the primary to secondary~~the~~ heat transfer.

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The limiting MCHFR event is the ~~4%~~8% split break size outside of containment with a loss of AC power. This FWLB involves a loss of cooling that heats up and pressurizes the RCS, resulting in the limiting MCHFR. The following initial conditions are assumed in the analysis to ensure that the transient results in the limiting CHF ratio.

- Initial power level is assumed to be 102% of nominal to account for a 2% 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 limiting BOC core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/°F) and DTC (-1.4 pcm/°F) are used to minimize the power dampening response for this heatup event.
- A loss of offsite power is assumed to occur at the time of the break, causing an immediate turbine and feedwater pump trip. The normal and highly reliable DC power systems are assumed to be available, which delays a reactor trip from mitigating the power excursion.

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- ~~The FWIVs are assumed to close at the maximum design limit closure rate while~~The FWIV backflow device is assumed to close on the faulted FW line to minimize fluid loss, which maximizes the RCS heatup. MSIVs are assumed to close rapidly to maximize the heatup affect.
- System biases include: high RCS temperature, high fuel temperature, low pressurizer pressure, low pressurizer level, and minimum RCS flow.
- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30% uncertainty ~~added to the~~for the primary to secondary heat transfer.

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The limiting event for SG pressure is the ~~4%~~10% split break size outside of containment that causes a loss of cooling effect that heats up and increases pressure in the primary and increases the superheat in the steam generator. The resulting effect in conjunction with actuation of DHRS results in the limiting SG pressure. The following initial

conditions are assumed in the analysis to ensure that the transient results in the limiting secondary pressure.

- Initial power level is assumed to be 102% of nominal to account for a 2% 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 limiting BOC core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/°F) and DTC (-1.4 pcm/°F) are used to minimize the power response for this event.
- ~~AC power is assumed to be available with the turbine trip occurring at the time of reactor trip.~~A loss of offsite power is assumed to occur at the time of the break, causing an immediate turbine and feedwater pump trip. The normal and highly reliable DC power systems are assumed to be available, which delays the reactor trip and MPS actuations that would occur immediately on a loss of DC power.
- The FWIVs are assumed to close at the ~~design limit closure rate while~~maximum design closure time to maximize fluid loss. MSIVs are assumed to close rapidly to maximize the heatup affect.
- The single failure assumed in this case was the failure of a FWIV backflow device to seat, reducing the inventory in the second SG train.
- System biases include: high RCS temperature, high fuel temperature, high pressurizer pressure and pressurizer level, and minimum RCS flow.

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- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30% uncertainty ~~added to the~~for the primary to secondary heat transfer.

The DEG split break size inside of containment releases the inventory of the FW and DHRS to the CNV. The following initial conditions are assumed in the analysis to ensure that the transient results are conservative in the assessment of DHRS functionality.

- Initial power level is assumed to be 102% of nominal to account for a 2% 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 limiting BOC core parameters are used to provide a limiting power response. The least negative MTC (0.0 pcm/°F) and DTC (-1.4 pcm/°F) are used to minimize the power response for this event.

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- The FWIVs are assumed to close at the ~~design limit closure rate~~maximum design closure time to maximize fluid loss, while MSIVs are assumed to close rapidly to maximize the heatup affect.

- System biases include: high RCS temperature, high fuel temperature, low pressurizer pressure and level, and minimum RCS flow.
- DHRS performance is assumed to be low (-30%) with a high biased pool temperature.
- AC power is assumed to be available with the turbine trip occurring at the time of reactor trip.
- The end of life steam generator was assumed which includes the design limit tube plugging and fouling in addition to a negative 30% uncertainty ~~added to the~~for the primary to secondary heat transfer.

The results from the thermal hydraulic evaluations are used as input to the subchannel analysis to determine the limiting MCHFR for this event. Other key inputs and assumptions used in the analysis are included in Table 15.2-28.

15.2.8.5 Results

For the limiting RCS pressure case, the FW line is assumed to have a double-ended guillotine break just outside containment. AC power is assumed to be lost, causing the turbine and feedwater pumps to trip, resulting in a rapid loss of cooling and large RCS pressurization. In this scenario, the reactor is tripped and DHRS is actuated by the high pressurizer pressure MPS signal. Pressure in the RCS continues to increase following reactor trip until the reactor safety valve lift limit is reached. Once the RSV lifts, system pressure quickly returns to below the RSV reset pressure. The limiting peak RCS pressure from FWLB is provided in Figure 15.2-35. DHRS cooling is sufficient to begin to depressurize the system such that a second RSV lift does not occur. The reactor power drops sharply following the reactor trip and follows the typical decay heat curve as shown on Figure 15.2-36. SG pressure rises equally in both SG as shown in Figure 15.2-37 and Figure 15.2-38. RCS temperature decreases and RCS flow stabilizes as shown in Figure 15.2-39 and Figure 15.2-40, respectively. The DHRS actuation closes the FWIVs and stops the loss of secondary fluid. The integrated break flow for the event is shown in Figure 15.2-41.

The peak SG pressure case results from a small split line break in the FW line just outside containment. AC power is assumed to be lost at the time of the break. The FWIV backflow check valve is assumed to fail to seat upon reverse flow. This condition results in asymmetrical pressure in the SGs as shown in Figure 15.2-42 and Figure 15.2-43. The intact SG has the peak SG pressure (Figure 15.2-42).

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In the limiting MCHFR case, the high pressurizer pressure trip is delayed, by biasing pressurizer pressure and level low, resulting in higher temperatures and the MCHFR occurs right before the reactor trips. The MCHFR versus time is presented on Figure 15.2-44.

When the break is inside containment, the rapid pressurization of the CNV results in the reactor trip, containment isolation, and DHRS actuation and therefore is not limiting for pressure or CHF criteria. Such a break results in the limiting function of the DHRS. The break results in the loss of an entire DHRS loop as the fluid from the feedwater line

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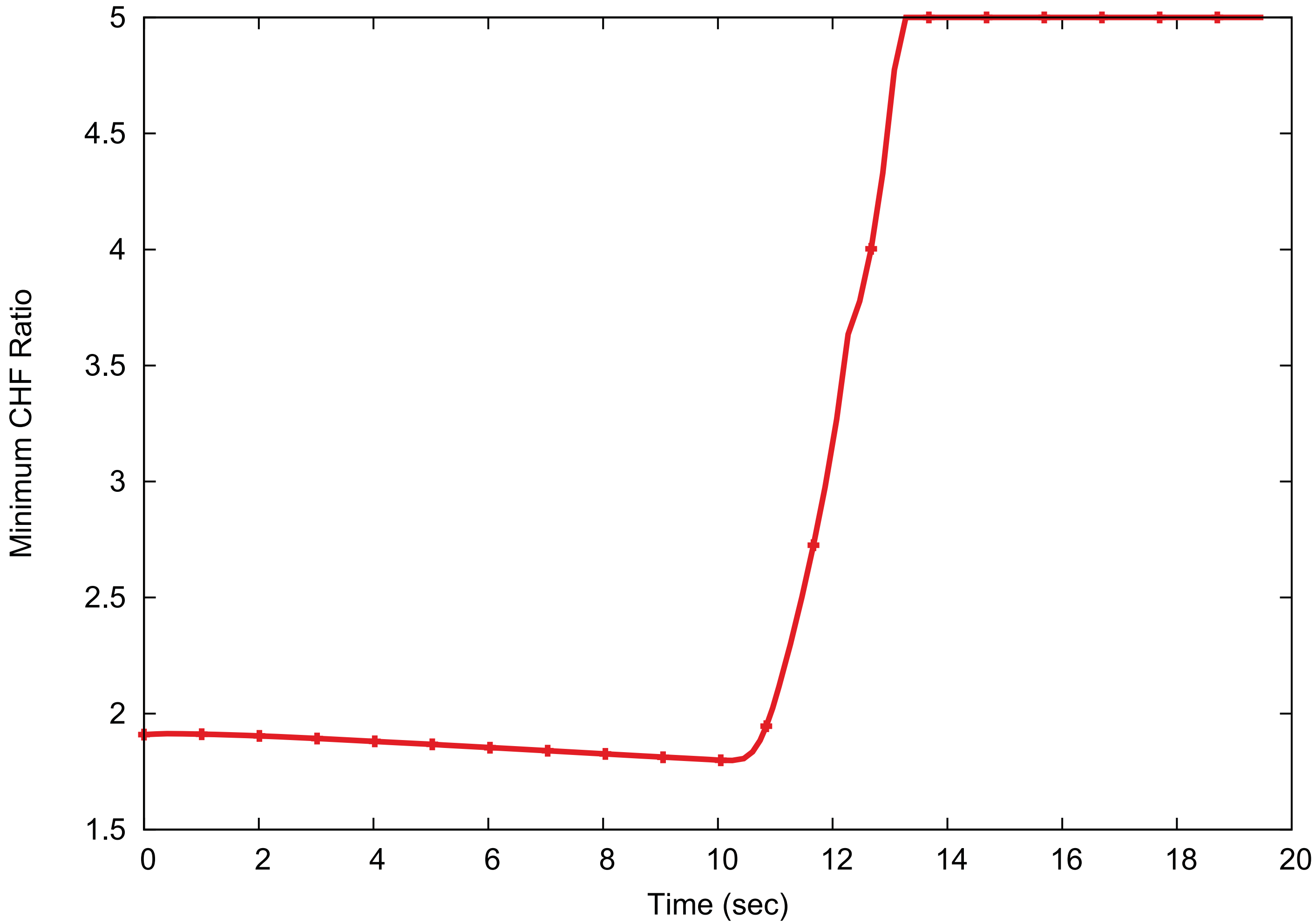
Figure 15.2-44: Hot Channel Node MCHFR - ~~Peak RCS Pressure~~Limiting MCHFR Case (15.2.8 Feedwater Line Break)



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

eRAI No.: 8744
Date of RAI Issue: 04/25/2017

NRC Question No.: 15.02.08-3

In accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 27, "Combined reactivity control systems capability," the reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes to ensure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. The applicant has requested an exemption from 10 CFR 50, Appendix A, GDC 27 as presented in Part 7 of the Design Certification Application (DCA) and has proposed a principle design criterion (PDC 27) to justify the departure from the regulation. Thus, if the staff approves the applicant's use of PDC 27, then the reactivity control systems shall be designed in accordance with PDC 27.

To meet the requirements mentioned above, as it relates to the feedwater line break (FWLB) accident, the accident analysis should show that the capability to cool the core is maintained even after ECCS initiates (either on a valid ECCS signal or after a loss of power to the ECCS valves).

In Final Safety Analysis Report (FSAR) Tier 2, Section 15.2.8.2, "Sequence of Events and Systems Operations," the applicant states that for a double-ended guillotine break in the feedwater piping inside containment, a complete loss of one train of decay heat removal system (DHRS) occurs. For this break location, the staff notes that the faulted steam generator, its associated DHRS train, and associated piping would completely drain into containment. Furthermore, in the event power is lost to the ECCS valves (either concurrent with the initiating event or some time thereafter) or a valid ECCS signal is generated, applying the single failure assumption to an inadvertent actuation block (IAB) valve would cause the ECCS valves to actuate and continued decay heat removal will be a function of the intact steam generator and its associated DHRS train as well as the ECCS recirculation loop. Considering this accident progression, the secondary system water that drained into containment from the faulted steam generator and all of its associated piping may now enter and dilute the ECCS recirculation loop and potentially affect criticality in the core. Based on the docketed information, the staff is unable to determine if such a design basis accident could challenge the capability to cool the core if a re-criticality were to occur. The staff

requests the applicant to provide additional information in the FSAR justifying how the capability to cool the core is maintained considering the amount of un-borated water from the secondary system that returns to the core after ECCS actuation or clarify the progression of such an event if the staff's understanding of the system interactions described above is incorrect.

NuScale Response:

The following provides a clarification of the progression of the subject events. For a feedwater line break (FWLB) or a steam line break (SLB) inside containment, inventory from the secondary side, including from the associated decay heat removal system (DHRS) train, drains into containment. A pipe break in the reactor component cooling water system (RCCWS) inside containment would also cause unborated inventory to drain into the containment vessel (FSAR Section 15.1.6, Containment Flooding). The level increase in containment (CNV) due to the FWLB, SLB or CNV flooding event is less than that required to actuate ECCS on high CNV level due to isolation of feedwater, containment isolation and the limited volume of RCCWS. However, for each of these cases, if a loss of AC and DC power is assumed, the ECCS actuation valves will de-energize, resulting in venting the ECCS trip line to the CNV. The inadvertent actuation block (IAB) will then close due to the differential pressure between the reactor coolant system (RCS) and CNV, holding the ECCS valves closed. The IAB is excluded from single failure consideration (this position will be addressed in the response to RAI 15-2, eRAI 8815), so the ECCS valves do not open until RCS pressure is reduced below the IAB release pressure. The DHRS is also actuated by the loss of DC power, or other MPS signal, which will cool and depressurize the RCS to the point of IAB release. When the IAB release pressure is reached, the ECCS valves will open and RCS fluid will blowdown into containment through the reactor recirculation valves (RRVs) and the reactor vent valves (RVVs). RCS fluid released through the RRVs will combine with any secondary water or RCCWS fluid in the CNV. After the RCS and containment pressures equalize, recirculation flow from containment through the RRVs will become established.

As flow recirculates from containment into the RCS through the RRVs, and vapor vents from the RCS to containment through the RVVs, ECCS core cooling is provided primarily by boiling in the core and condensation in containment. Soluble boron present in the RCS liquid prior to the event does not transport with the vapor through the RVVs, such that boron accumulates in the liquid in the core and riser region. The concentration of boron in liquid in the core and riser region is assessed to demonstrate that coolable geometry is maintained with respect to assuring the boron concentration remains below the solubility limit during long term cooling. Boron precipitation is addressed in the Technical Report, Long-Term Cooling (LTC) Methodology, TR-0916-51299-P, Revision 0, Section 6.0. After the initial blowdown and pressure equalization, the recirculation flow rate decreases to approximately 5 pounds mass per second (lbm/s) and continues to decreases as decay heat is reduced. The NuScale power

module is designed to maintain sufficient liquid in the reactor pressure vessel to maintain liquid level above the top of active fuel after ECCS valves open and throughout the cooldown process. Relatively low recirculation flow rates limit the rate that boron is concentrated in the reactor vessel and the rate at which coolant with lower boron concentration recirculates from the containment into the downcomer and lower plenum, and then into the core region.

During ECCS cooldown, both temperature and void reactivity feedback must be considered. The temperature decrease will add positive reactivity, while void generation provides negative reactivity. Even small concentrations of boron will tend to suppress the temperature-driven positive reactivity insertion during cooldown. As the borated water cools, the increased density of boron counters the positive reactivity from the increased moderation of the water. Therefore, the limiting condition for the ECCS mode overcooling return to power is end of cycle with low initial boron concentration, where the boron concentrating effects of vapor generation in the core are negligible. Since the limiting condition for the return to power event has low boron concentration, the impact from potential boron dilution during these events is minimal. The ECCS mode return to power event, described in FSAR Section 15.0.6, shows that the temperature reactivity feedback is balanced by the void feedback due to the boiling in the core. Events with high decay heat have much greater void formation and do not have a return to criticality due the relatively large negative reactivity from void feedback. For the analysis described in the FSAR 15.0.6, the equilibrium critical power level, including the available decay heat, is much less than decay heat removal capability. The LTC analysis presented in Section 15.6.5 is evaluated at conservatively high decay heat conditions that are more limiting from a heat load perspective than the ECCS overcooling return to power event. The high heat load results in relatively high void formation such that any return to power event would provide negligible additional heat. Therefore, the LTC analyses confirm adequate capability of the ECCS to cool the core, even if a return to power were to occur following an ECCS actuation.

The evaluation of the return to power event described in the FSAR 15.0.6 demonstrates that void feedback during ECCS cooling limits the heat load from a potential recriticality. The LTC analysis demonstrated the capability to remove the greatest post accident heat load. These analyses provide bounding conditions for the heat load that would occur from a FWLB or SLB, including the potential heat generated from a return to criticality.

Impact on DCA:

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

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

eRAI No.: 8744
Date of RAI Issue: 04/25/2017

NRC Question No.: 15.02.08-4

In accordance with 10 CFR 50, Appendix A, General Design Criterion (GDC) 34, "Residual Heat Removal," the decay heat removal system (DHRS) safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to 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 departs from GDC 34 by adopting Principle Design Criterion (PDC) 34 presented in Final Safety Analysis Report (FSAR) Tier 2, Section 3.1.

To meet the requirements mentioned above, as they relate to the feedwater line break (FWLB) inside containment challenging the proper functioning of the DHRS, the accident analysis should show that despite the worst single failure, the DHRS is still capable of adequately removing decay heat.

In FSAR Tier 2, Section 15.2.8.2, "Sequence of Events and Systems Operations," the applicant states that one potential single failure to note is of the safety-related check valve to close on the intact SG. In the event of this failure, the second, non-safety-related check valve is credited to ensure adequate inventory is maintained in the intact SG and DHRS condenser. The applicant then states that there are no single active failures that cause the FWLB event to have unacceptable results. The staff notes that for the FWLB inside containment, the faulted SG and its associated DHRS are empty and rendered non-functional. When applying the single failure assumption to the safety-related check valve in the intact feed line, it appears the non-safety-related check valve upstream of the safety-related check valve is credited to ensure proper functioning of the intact SG and its associated DHRS. However, the applicant does not provide justification for why it is acceptable to credit a non-safety-related component for design basis accident mitigation. Based on the docketed information, if the non-safety-related components are not relied upon for this design basis accident mitigation, it is unclear to the staff if the intact DHRS train would be able to perform its safety related

function of removing decay heat and preserving the specified acceptable fuel design limits (SAFDLs). The staff requests the applicant to provide additional information justifying the credit of non-safety-related components for design basis accident (DBA) mitigation or explain how the SAFDLs are met without crediting the non-safety-related check valve on the intact DHRS train.

In addition, the staff notes that in FSAR Tier 2, Section 15.0.0.5, "Limiting Single Failures," the applicant states that "design basis event mitigation credits valves that are classified as safety-related," but goes on to discuss how both the safety-related and non-safety-related feedwater check valves are credited for mitigating the consequences of the FWLB accident by retaining DHRS inventory in the short term until the feedwater isolation valves can close. The staff seeks clarification regarding these two statements in Section 15.0.0.5 as they appear to contradict each other.

NuScale Response:

As stated in FSAR Section 15.0.0.5, one safety-related check valve per feedwater (FW) line is credited to mitigate consequences of design basis events. If both of these check valves perform as designed, no credit is required for nonsafety-related check valves. With respect to the single failure considerations for feedwater line breaks (FWLBs) inside containment, the backup nonsafety-related check valve is credited to maintain DHRS inventory if the safety-related check valve fails. The nonsafety-related check valve is credited to close in one second and maintain DHRS inventory until the associated feedwater isolation valve (FWIV) closes. The FWIVs close in less than approximately 10 seconds, based on a maximum valve stroke time plus signal processing time for the module protection system to isolate FW.

Crediting a nonsafety-related component in the event of a single failure of the associated safety-related component is considered acceptable and is based upon guidance from NUREG-0138. In NUREG-0138, Issue 1, the NRC staff concluded that non-safety grade components in the steam and FW systems could be used as backup to a single failure in safety grade components when their design and performance are compatible with the accident conditions for which they are called upon to function. The non-safety grade components discussed in NUREG-0138 were not required to be designed to Seismic Category I or have augmented testing requirements. For the NuScale design, the nonsafety-related FW supply check valves are listed in FSAR Section 3.2, Table 3.2-1 with requirements for augmented quality, designed to Seismic Category I and included in the in-service testing program. Although these valves are classified as nonsafety-related, the augmented design specifications and testing requirements exceed those found acceptable for similar applications. Therefore, NuScale considers that the augmented quality nonsafety-related FW check valves meet the NRC guidance to be credited in the safety analysis as a backup in the event of a failure of the safety-related check valves.

Impact on DCA:

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