RAIO-0717-54899

July 17, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 35 (eRAI No. 8786) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 35 (eRAI No. 8786)," dated May 26, 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 Question from NRC eRAI No. 8786:

- 15.06.05-2

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

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8786

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8786

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

eRAI No.: 8786
Date of RAI Issue: 05/26/2017

NRC Question No.: 15.06.05-2

10 CFR Part 50 Appendix K, I.C.1 - *Break Characteristics and Flow,* requires that a spectrum of possible pipe breaks be considered in the analyses of loss-of-coolant accidents (LOCAs). Section 15.6.5 of the NuScale Design Specific Review Standard states that, "a spectrum of LOCA break sized is to be evaluated and the limiting break identified through sufficient analyses …" The applicant indicates in Section 15.6.5.1 of the Final Safety Analysis Report (FSAR) that a spectrum of break sizes were analyzed. However, the input parameters, initial conditions, and results from the analyses for the spectrum of breaks are not presented in the FSAR. NRC staff relies upon the docketed input parameters, initial conditions, and results from the spectrum of break sizes to establish a finding that the limiting pipe break has been identified and evaluated. Accordingly, NRC staff requests that NuScale update Section 15.6.5 of the FSAR to include a summary of the input parameters, initial conditions, and results from the spectrum of pipe breaks considered in the analyses of LOCA.

NuScale Response:

FSAR Section 15.6.5 has been revised to include a table of the spectrum of breaks assumed in the loss-of-coolant (LOCA) analysis. To identify the limiting LOCA scenario, each of the break locations and sizes were analyzed. The limiting break with respect to the collapsed liquid level above the top of active fuel identified from the sensitivity of break sizes and break locations was then further analyzed for three power scenarios: no loss of power, loss of AC power, and loss of AC and DC power. The limiting event was analyzed for the following single failure scenarios: no single failure, failure of a single reactor vent valve (RVV) to open, failure of a single reactor recirculation valve (RRV) to open, and failure of one emergency core cooling system (ECCS) division (i.e., one RVV and one RRV) to open. From the sensitivity study results, the limiting event with respect to the collapse liquid level above the top of active fuel was identified as a 10-percent injection line break with a loss of normal AC power and no single failure. FSAR Section 15.6.5 has also been revised to include tables providing the results of the sensitivity analyses performed to support the selection of the limiting LOCA event.

Impact on DCA:

FSAR Section 15.6.5 and Table 15.6-18 through Table 15.6-24 have been revised as described in the response above and as shown in the markup provided in this response.

The analysis presented for this event shows that stable DHRS cooling is reached and therefore the acceptance criterion is satisfied.

15.6.4 Main Steam Line Failure Outside Containment (BWR)

This event is a BWR-specific event and not applicable to the NuScale Design.

15.6.5 Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary

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A LOCA is an event that compromises the reactor coolant pressure boundary (RCPB), resulting in RCS inventory loss at a rate that exceeds the capacity of normal makeup flow. A spectrum of break sizes and locations of the RCS pressure boundary piping are assessed. Table 15.6-18 provides the spectrum of break sizes and locations.

A LOCA for the NuScale design is unique compared to traditional large light water reactors because the diameters of the RCS piping are small so there is no distinction between "large break LOCA" and "small break LOCA" scenarios. In addition, RCS inventory is preserved within containment and available for recirculation soon after event initiation. A LOCA is analyzed for thermal hydraulic effects and is classified as a potential accident, as shown in Table 15.0-1. Inadvertent opening of an ECCS valve is not considered a LOCA and is addressed in Section 15.6.6.

15.6.5.1 Identification of Causes and Accident Description

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A LOCA is a postulated accident that is initiated by a non-mechanistic break in a pipe inside containment connected to the RPV. The break location and break size determines the rate of RCS inventory loss and depressurization rate. Thus a spectrum of break sizes is postulated to occur at various locations in the pipelines penetrating the RCPB, as shown in Table 15.6-18. The postulated break sites are ~~focused on~~ the rupture of the RCS injection and discharge lines, high point vent line, and pressurizer spray supply lines inside of containment.

15.6.5.2 Sequence of Events and Systems Operation

The initiating event for this transient is a rupture of the RCS injection or discharge line, RPV high point vent line, or pressurizer spray supply line inside of containment. The LOCA break spectrum is separated into two categories: a liquid space break consisting of the RCS injection line and discharge line; and a steam space break consisting of the high point vent line and pressurizer spray supply line.

A steam space break initiates a blowdown of the RCS inventory into the CNV from the top of the RPV. A liquid space break causes blowdown of the RCS inventory into the CNV from the liquid filled region of the RPV. The progression of the event, including the actuation of the engineered safety features, is similar to a steam space break with the exception of different timing of the key events and the liquid/steam composition of the break flow.

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Table 15.6-12 shows the sequence of events for ~~a~~the limiting LOCA. The MPS is credited to initiate the reactor trip, isolate containment, and initiate DHRS and ECCS. DHRS is not credited for cooling following a LOCA. No operator action is credited in this event analysis.

The transition from the LOCA analysis to the post-LOCA long-term core cooling phase occurs when natural circulation between the RPV and the containment through the RVVs and RRVs has reached a stable state and decay and residual heat is being removed. The purpose of the post-LOCA long term cooling evaluation is to show that continued cooling occurs without boron precipitation for at least 72-hours after the initiation of a LOCA.

15.6.5.3 Core and System Performance

15.6.5.3.1 Evaluation Model

The thermal hydraulic analysis of the plant response to a LOCA uses NRELAP5. Section 15.0.2 provides details on the modeling requirements and code modifications needed to appropriately capture the phenomena and features of the LOCA evaluation model. Section 15.0.2 discusses the LOCA Evaluation Model Development and Assessment Process (EMDAP). Utilizing the results of the break spectrum, the methodology demonstrates that the design and operating conditions analyzed will result in a safe condition of a NPM for postulated design basis LOCAs.

The post-LOCA long-term core cooling analysis is performed using the NRELAP5 model to support the ECCS long term cooling methodology. A spectrum of cases is performed to encompass minimum and maximum cooldown scenarios. The results of the long-term core cooling analysis is then compared to the acceptance criteria developed for evaluating the margin to boron precipitation to show that boron precipitation is avoided during the long-term core cooling phase.

For the boron precipitation portion of the analysis, the following methodology is used. The determination of the boron precipitation temperature for a given mixing volume starts with the calculation of the entire mass of boron in the RCS. A corresponding concentration is calculated for the mixing volume assumption. Finally, the precipitation temperature is obtained for the mixing volume concentration using the boron precipitation curve. These calculations are performed for various mixing volumes corresponding to various elevation of liquid levels above the core. Temperature and level results from the long-term core cooling calculation are compared to the boron precipitation results to determine if boron precipitation could occur.

15.6.5.3.2 Input Parameters and Initial Conditions

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The input parameters and initial conditions used in the LOCA analysis are selected to provide a conservative calculation. The parameter of interest for the LOCA is the collapsed liquid level above the core. Thus, inputs and assumptions are chosen to determine the minimum collapsed liquid level above the core. ~~T~~As shown in Table 15.6-19 through Table 15.6-24, the 10-percent RCS injection line break is limiting for maintaining the collapsed liquid level above the core.

Unless otherwise specified, the LOCA analysis assumes that the plant control systems and engineered safety features perform as designed, with allowances for instrument uncertainty. No operator action is credited to mitigate the effects of a LOCA for the duration of the event, including the post-LOCA long-term core cooling phase.

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Table 15.6-13 provides inputs and assumptions for the limiting-break LOCA analysis. The following are key input parameters common to the spectrum of breaks analyzed in Table 15.6-18:

- Initial power level is assumed to be 102 percent of nominal to account for a 2 percent instrumentation uncertainty.
- RCS average temperature is initialized to yield a maximum riser operation temperature of 595 degrees F.
- Pressurizer pressure is biased high to maximize the RCS energy and containment peak pressure.
- Pressurizer level is biased low to minimize inventory availability.
- Containment pressure is biased high to maximize containment peak pressure.
- Main steam pressure is biased high to maximize overall system energy.
- Feedwater temperature is biased high to maximize the overall system energy and limit heat transfer to the steam generators.
- RCS flow is dependent on other initial conditions and no additional biases are applied.
- The ECCS IAB release pressure is assumed to be at the lowest value of 1000 psid. Using the minimum value of the release pressure results in the lowest minimum collapsed water level.
- Bypass flow through the reflector and guide tubes is maximized to 8.5 percent of the total core flow to minimize flow through the hot assembly. This value is consistent with the subchannel analysis methodology discussed in Section 15.0.2.
- Reactor pool temperature is assumed to be 140 degrees F to reduce heat sink potential. Sensitivities demonstrate that the initial reactor pool temperature has negligible impact on the LOCA acceptance criteria.

- Single failure evaluations of a single RVV to open, a single RRV to open, and failure of one ECCS division (one RVV and one RRV) to open were performed to determine the most conservative scenario. The evaluations show that the maximum rate of depressurization yields the minimum collapsed liquid level. The maximum rate of depressurization occurs when all ECCS valves open. Therefore, the LOCA analysis is performed with no single failure. Assuming a single failure in the analysis yields non-conservative results.
- Allowances for instrument inaccuracy are accounted for in the analytical limits of mitigating systems in accordance with the guidance provided in Regulatory Guide 1.105.

The input parameters and initial conditions used in the limiting case for the post-LOCA long-term core cooling analysis are also selected to provide a conservative calculation. Sensitivity studies show that the minimum collapsed liquid level above the core occurs in a maximum cooldown scenario, which is a letdown line break with the following inputs and assumptions.

- A 1.0 multiplier to decay heat.
- A fouling factor of 10.0 at the RPV shell heat structures to increase heat transfer.
- A fouling factor of 10.0 at the containment shell heat structures to increase heat transfer.
- A minimum reactor pool temperature of 40 degrees F to increase heat transfer.
- The nominal reactor pool level with an increased reactor pool volume.
- A single failure of ECCS division (one RRV and one RVV).
- A loss of EDSS, EDNS, and normal AC.

15.6.5.3.3 Results

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The LOCA analysis is performed for a spectrum of break sizes and break locations (Table 15.6-18) to determine the location and size of the break that is limiting for maintaining the collapsed liquid level above the core. Table 15.6-19 through Table 15.6-22 provide the results for each of the analyzed breaks. From these results, the 10 percent cross-sectional area break of the RCS injection line has the minimum collapsed level above the top of active fuel.

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The 10-percent injection line break was then analyzed for different power scenarios, as shown in Table 15.6-23. The limiting power scenario is the loss of normal AC power, which is then analyzed for different single failure scenarios, as shown in Table 15.6-24.

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Thus, ~~T~~the limiting scenario begins with an equivalent 10 percent cross-sectional area break of the RCS injection line with a loss of normal AC and no single failure.

The results presented for the LOCA are for the limiting event. Figure 15.6-41 to Figure 15.6-54 show the system response to a LOCA. Table 15.6-14 contains the results of the limiting LOCA event.

Upon initiation of a LOCA, the RCS inventory flows out of the break into the containment (Figure 15.6-41). A coincident loss of normal AC power is assumed at time zero. A loss of normal AC power stops feedwater flow, thus terminating RCS cooling via the secondary system. The reactor trip does not occur until 60 seconds after the loss of normal AC power or until a separate MPS analytical limit is reached. Given the small break size and the loss of secondary cooling, the RCS undergoes a short-term pressurization while the reactor is still at power (Figure 15.6-42). The increasing RCS pressure reaches the MPS high pressurizer pressure setpoint causing the reactor trip, as evident in Figure 15.6-43 and Figure 15.6-44. A high pressurizer pressure signal also initiates the isolation of the SGs by closing the MSIVs and the feedwater isolation valves. This also opens the valves in the DHRS system, which completes the recirculation loop between the steam generators in the RPV and the DHRS heat exchangers in the reactor pool. However, to provide a bounding LOCA analysis, DHRS cooling is not credited.

As primary coolant is discharged into the containment from the break (Figure 15.6-41), the inventory level inside the RPV continues to decrease and the inventory level and pressure inside the containment continues to increase (Figure 15.6-45) until the high containment level limit generates the MPS ECCS actuation signal. However, as the differential pressure between the RPV and containment exceeds the IAB threshold pressure, the IAB feature prevents the ECCS valves from opening. Pressure and temperature inside the RPV continues a gradual downward trend as primary inventory continues to flow into the containment through the break, as shown in Figure 15.6-42, Figure 15.6-46 and Figure 15.6-47.

The RVVs and RRVs open once the differential pressure between the RPV and containment decreases below the IAB pressure release setpoint, as shown in Figure 15.6-48 and Figure 15.6-49. With all ECCS valves open, the RPV pressure (Figure 15.6-42) and RCS temperature (Figure 15.6-47) rapidly drop, causing voiding in the core and a temporary reduction in the collapsed liquid level above the top of the core (Figure 15.6-45). Core CHFR remains above the safety limit, ensuring that fuel damage due to local dry out conditions does not occur. Inventory released to the containment is allowed to flow back into the RPV downcomer through the RRVs, increasing the collapsed level in the RPV. RCS flowrate is shown in Figure 15.6-50. Steam generator pressure is shown in Figure 15.6-51.

Pressure and temperature inside the containment also experience a rapid increase, as shown in Figure 15.6-42 and Figure 15.6-52. Containment pressure and temperature reaches a maximum value after the ECCS valves open and then decreases as thermal energy is transferred to the reactor pool through the containment wall. The peak containment pressure for design basis events is evaluated in Section 6.2.

At this point, the LOCA event transitions to the post-LOCA long term core cooling phase. A gradual cool down and depressurization of the containment and RPV is

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Table 15.6-18: Loss-of-Coolant Analysis - Summary of Break Spectrum

Break Area %	Break Area in^2	Equivalent Diameter in	Break Area Analyzed			
			Discharge Line	Injection Line	High Point Vent	Pressurizer Spray Supply
100	2.23	1.69	X	X	X	-
75	1.67	1.46	X	X	X	-
50	1.12	1.19	X	X	X	-
35	0.78	1.00	X	X	X	X
20	0.45	0.75	X	X	X	-
10	0.22	0.53	X	X	X	-
5	0.11	0.38	X	X	X	-
2.2	0.05	0.25	X	X	-	-

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Table 15.6-19: Loss-of-Coolant Analysis - Discharge Line Break Spectrum with All Electric Power Available

Break Size (%)	Time of RTS (s)	Time of ECCS (s)	MCHFR	Time of MCHFR(1) (s)	Min Collapsed Level above TAF (ft)	Time of Min Collapsed Level above TAF (s)
100	5	202	1.772	2	10.193	708
75	6	271	1.795	2	10.205	800
50	7	413	1.818	2	10.220	970
35	8	616	1.831	2	10.236	1190
20	12	2006	1.844	2	5.256	2016
10	25	5692	1.853	2	2.261	5702
5	96	11837	1.857	2	1.919	11848
2.2	661	25005	1.859	2	3.661	25020

(1) A small decrease in CHFR from its initial value is observed. Many cases report MCHFR occurring at 2 seconds due to calculation edit frequency; the actual time of MCHFR occurs earlier.

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Table 15.6-20: Loss-of-Coolant Analysis - Injection Line Break Spectrum with All Electric Power Available

Break Size (%)	Time of RTS (s)	Time of ECCS (s)	MCHFR	Time of MCHFR(1) (s)	Min Collapsed Level above TAF (ft)	Time of Min Collapsed Level above TAF (s)
100	8	492	1.857	2	10.117	1102
75	8	528	1.857	2	10.129	1128
50	9	663	1.858	2	10.141	1234
35	10	809	1.859	2	9.948	824
20	13	2653	1.86	2	1.682	2662
10	23	6140	1.86	2	0.426	6156
5	76	11759	1.861	2	0.522	11776
2.2	497	22859	1.86	496	1.998	22875

(1) A small decrease in CHFR from its initial value is observed. Many cases report MCHFR occurring at 2 seconds due to calculation edit frequency; the actual time of MCHFR occurs earlier.

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Table 15.6-21: Loss-of-Coolant Analysis - High Point Vent Line Break Spectrum with All Electric Power Available

Break Size (%)	Time of RTS (s)	Time of ECCS (s)	MCHFR	Time of MCHFR(1) (s)	Min Collapsed Level above TAF (ft)	Time of Min Collapsed Level above TAF (s)
100	6	2516	1.819	2	10.013	5894
75	7	2879	1.828	2	10.02	6418
50	8	3421	1.839	2	9.978	6958
35	11	4273	1.845	2	10.003	7785
20	17	6265	1.852	2	10.055	9688
10	48	9641	1.857	2	10.248	12192
5	255	14941	1.859	2	10.184	18218

(1) A small decrease in CHFR from its initial value is observed. Many cases report MCHFR occurring at 2 seconds due to calculation edit frequency; the actual time of MCHFR occurs earlier.

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Table 15.6-22: Loss-of-Coolant Analysis - Pressurizer Spray Supply Line Break Spectrum with All Electric Power Available

Break Size (%)	Time of RTS (s)	Time of ECCS (s)	MCHFR	Time of MCHFR(1) (s)	Min Collapsed Level above TAF (ft)	Time of Min Collapsed Level above TAF (s)
35	7	2955	1.83	2	10.02	6494

(1) A small decrease in CHFR from its initial value is observed. Many cases report MCHFR occurring at 2 seconds due to calculation edit frequency; the actual time of MCHFR occurs earlier.

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Table 15.6-23: Loss-of-Coolant Analysis - Ten-percent Injection Line Break with Evaluation of Electric Power Available

Case	Time of RTS (s)	Time of ECCS (s)	MCHFR	Time of MCHFR (s)[1]	Min Collapsed Level Above TAF (ft)	Time of Min Collapsed Level Above TAF (s)
All power available	23	6140	1.86	2	0.426	6156
Loss of Normal AC Power	10	6181	1.796	12	0.137	6196
Loss of normal AC and DC power	0	6129	1.859	2	0.441	6146

(1) Many cases report MCHFR occurring at 2 seconds due to calculation edit frequency; the actual time of MCHFR occurs earlier.

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Table 15.6-24: Loss-of-Coolant Analysis - Ten-percent Injection Line Break with Loss of Normal AC Power Evaluation of Single Failure

Single Failure	Time of RTS (s)	Time of ECCS (s)	MCHFR	Time of MCHFR (s)	Min Collapsed Level Above TAF (ft)	Time of Min Collapsed Level Above TAF (s)
No failure	10	6181	1.796	12	0.137	6196
Failure of one RVV to open	10	6181	1.796	12	0.232	6198
Failure of one RRV to open	10	6181	1.796	12	0.584	6198
Failure of one RVV and RRV to open	10	6181	1.796	12	0.715	6198