RAIO-0618-60708

June 29, 2018

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. 456 (eRAI No. 9478) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 456 (eRAI No. 9478)," dated May 01, 2018

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

The Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 9478:

- 15.01.05-2
- 15.01.05-3

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

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

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9478

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NuScale Response to NRC Request for Additional Information eRAI No. 9478

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

eRAI No.: 9478
Date of RAI Issue: 05/01/2018

NRC Question No.: 15.01.05-2

In accordance with 10 CFR 52.47(a)(2)(iv)(A) and 10 CFR 52.47(a)(2)(iv)(B), an evaluation and analysis of the postulated fission product release events must demonstrate that a dose to an individual at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release will not exceed 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE), and that a dose to an individual at any point on the outer boundary of the low population zone (LPZ) shall not exceed 25 rem TEDE from exposure to the radioactive cloud resulting from the postulated fission product release.

To meet the requirements mentioned above, as they relate to the steam system piping failure events, the accident analysis should assume initial conditions and input parameters that maximize the severity of the accident by maximizing the mass and energy release out of the break.

In Final Safety Analysis Report (FSAR) Tier 2, Section 15.1.5.3.2, "Input Parameters and Initial Conditions," under the "[steam line break (SLB)] Cases Resulting in Limiting Radiological Consequences" heading, the applicant states that, for the two limiting radiological consequences cases it analyzed, nominal intial conditions are used. The staff understands that biasing certain initial conditions, within their uncertainty ranges and tolerances, can exacerbate the severity of the event. Based on the docketed information, the staff cannot understand why the applicant's current assumptions (i.e. nominal initial conditions) make the event limiting with respect to maximizing the mass and energy release through the break. The staff requests the applicant to justify in the FSAR its current assumptions of using nominal initial conditions as opposed to using conservatively biased initial conditions for the limiting radiological consequences SLB case. If justification cannot be provided, the staff requests the applicant to provide in the FSAR the results of a limiting radiological consequences SLB case that utilizes conservatively biased initial conditions and input parameters. The staff requests the applicant to make changes to the FSAR as necessary.

NuScale Response:

FSAR Section 15.1.5.3.2 identifies two limiting radiological cases. However, more cases were evaluated as described in Table 7-25 of Reference 1. As described in FSAR Section 15.1.5.3.2, the two limiting cases are characterized as the largest undetected break and the transient sequence that maximizes the delay between reactor trip and secondary isolation. In order to maximize the size of the undetected break, the core power must sufficiently respond to the decrease in RCS temperature to avoid pressurizer level shrinkage and the resultant low pressurizer pressure or level trip while still avoiding the high reactor power trip. For this evaluation, nominal initial conditions were compared to high biased reactor coolant system (RCS) temperature, pressurizer pressure and level, high biased steam generator heat transfer and time in cycle (BOC, EOC) conditions. Startup zero power conditions were also evaluated. Of these sensitivities, the biased initial conditions exaggerated the short term transient response resulting in reactor trip for the same break size where the nominal case did not trip.

The maximum delay between reactor trip and secondary isolation is also a smaller break size that continues the over cooling event after the high reactor power trip is reached. For this scenario the nominal initial conditions were also compared to high biased RCS temperature, pressurizer pressure and level, high biased steam generator heat transfer and EOC vs BOC conditions. For the nominal conditions, low pressurizer level generated the secondary isolation signal while the biased conditions resulted in the generation of either a high or low steam super heat trip causing a more rapid secondary isolation.

In summary, the steam line break radiological analysis was performed in accordance with the methodology described in Reference 1 including a sufficient number of sensitivity cases to determine the most limiting radiological release scenarios. The consequences for both scenarios are maximized by detection avoidance which is exacerbated with a nominal initialization that starts the transient furthest from the trip condition.

References:

1. TR-0516-49416-P. "Non-Loss-of-Coolant Accident Analysis Methodology," Revision 1.

Impact on DCA:

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

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Response to Request for Additional Information
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eRAI No.: 9478
Date of RAI Issue: 05/01/2018

NRC Question No.: 15.01.05-3

10 CFR 50, Appendix A, General Design Criterion (GDC) 28, "Reactivity limits," requires that the reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents will neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

To meet GDC 28, as it relates to a steam line break that minimizes the critical heat flux ratio in the core, the applicant should demonstrate that the potential amount and rate of reactivity increase due to the steam line break cooldown does not produce a minimum critical heat flux ratio (MCHFR) that sufficiently disturbs the core and impairs significantly the capability to cool it.

In FSAR Tier 2, Section 15.1.5.3.2, "Input Parameters and Initial Conditions," the applicant states that initial conditions and assumptions used in the evaluation of a SLB result in a conservative calculation. The staff, as part of the Chapter 15 audit, reviewed calculation files that support this statement. For the limiting MCHFR case, the staff could not verify some initial condition assumptions used that supposedly would result in conservative calculations. For example, the staff could not verify the SG heat transfer and tube plugging/fouling assumption used for the limiting MCHFR case. The staff understands that a high steam generator (SG) heat transfer bias along with no tube plugging/fouling will exacerbate the cooldown resulting in a more pronounced increase in power and thus a more limiting MCHFR. Because the staff cannot verify the conservative nature of some input parameters/assumptions, the staff cannot determine if the applicant's analysis meets GDC 28. The staff requests the applicant to provide in the FSAR the assumptions used for important parameters directly affecting the outcome of the case, e.g. in the case of the limiting MCHFR the applicant should provide the SG heat transfer uncertainty and tube plugging/fouling assumptions used.

The staff also notes that in FSAR Tier 2, Section 15.1.5.3.2, under the heading "Steam Line Break Cases Resulting in the Limiting Minimum Critical Heat Flux Ratio," the applicant states:

"the most negative [moderator temperature coefficient (MTC)] and least negative doppler temperature coefficient (DTC)] are used to ***minimize*** the power response for this event." The staff believes "minimize" should actually read "maximize," since a cooldown with a strongly negative MTC will accentuate the power increase. The staff requests the applicant to consider this potential error and make any necessary changes to ensure the information in the FSAR is clear and consistent if it is not already so.

NuScale Response:

FSAR Section 15.1.5.3.2 describes the initial conditions for the steam line break (SLB) cases that result in limiting conditions for reactor coolant system (RCS) pressure, minimum critical heat flux ratio (MCHFR), and radiological consequences. The steam generator tube plugging and fouling conditions as well as the decay heat removal system heat transfer uncertainty assumed in the limiting RCS pressure and MCHFR cases were added to FSAR Section 15.1.5.3.2 as shown in the markup provided with this response. The steam generator heat transfer uncertainty assumed in the limiting MCHFR case was added to FSAR Section 15.1.5.3.2 as shown in the markup provided with this response. Additional important parameter biases can be found in FSAR Table 15.1-13 for the limiting RCS pressure and MCHFR cases. As noted in FSAR Section 15.1.5.3.2, the limiting radiological consequence cases are initialized from nominal conditions.

FSAR Section 15.1.5.3.2 incorrectly characterizes the assumed reactivity coefficients as minimizing the power response for the event. The text has been corrected to state that the coefficients maximize the power response for the event as shown in the markup provided with this response.

Impact on DCA:

FSAR Section 15.1 has been revised as described in the response above and as shown in the markup provided in this response.

small SLB event result in a conservative calculation. The SLB that results in limiting RCS and main steam pressures is a 12.5 percent split break in the main steam piping just outside of containment. Small SLBs result in limiting MCHFR conditions. A representative 3.3 percent split break in the main steam piping just outside of containment is provided in this section to demonstrate a case with a limiting MCHFR.

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

The break size and loss of power assumptions for this SLB case cause an overheating effect that pressurizes the RCS, resulting in the limiting RCS pressure. Table 15.1-13 provides key inputs for the limiting SLB cases. The following initial conditions are assumed in the analysis of the SLB to ensure that the transient results in the limiting RCS pressure.

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

RAI 15.01.05-3

- The limiting beginning-of-cycle core parameters are used to provide a limiting power response. The most positive MTC (0.0 pcm/degrees F) and DTC (-1.40 pcm/degrees F) are used to ~~minimize~~ maximize the power response for this event.

RAI 15.01.05-3

- The beginning-of-life SG is assumed, which includes no SG tube plugging and no SG fouling. A ~~a~~ 30 percent uncertainty is added to the SG and DHRS heat transfer to maximize the cooldown event.
- The single failure identified for this event is the failure of an MSIV to close.

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

A thermal hydraulic analysis is performed to provide the limiting boundary conditions for the downstream subchannel analysis, which evaluates the final CHF value. The following initial conditions are assumed in the analysis of a range of small SLB cases to ensure that the boundary conditions calculated for the subchannel analysis will result in the limiting MCHFR.

- 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.

RAI 15.01.05-3

- The most limiting combination of core cycle parameters is used to provide a limiting power response. The most negative MTC (-43.0 pcm/degrees F) and least negative DTC (-1.40 pcm/degrees F) are used to ~~minimize~~maximize the power response for this event.

RAI 15.01.05-3

- The beginning-of-life SG is assumed, which includes no SG tube plugging and no SG fouling. A 30 percent uncertainty is added to both the SG and DHRS heat transfer to maximize the cooldown event.
- The single failure identified for this event is the failure of an MSIV to close, which results in more severe plant conditions. However, this single failure does not affect MCHFR because it occurs after CHFR reaches a minimum.

The results from the thermal hydraulic evaluations are used as input to the subchannel analysis to determine the limiting MCHFR for this event. The subchannel evaluation model is discussed in Section 15.0.2.3.

SLB Cases Resulting in Limiting Radiological Consequences

There are two different SLB cases that are limiting from a radiological release perspective. The first limiting radiological case is a 5 percent break from nominal initial conditions. The other limiting radiological case is a 7.5 percent break that maximizes the time between the reactor trip and secondary isolation. This SLB case is also initialized at nominal conditions with an EOC core exposure. A single failure of the MSIV on the affected train is modeled to maximize the mass and energy release after the isolation signal.

15.1.5.3.3 Results

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

The sequence of events for the limiting RCS pressure SLB case event is provided in Table 15.1-10. Figure 15.1-32 through Figure 15.1-37 show the transient behavior of key parameters for this SLB case. The transient initiates with a 12.5 percent split break of the main steam piping outside containment and a coincident loss of AC power. The effects of the postulated SLB on other systems are considered in Section 3.6 consistent with Branch Technical Position (BTP) 3-3 and BTP 3-4. The loss of AC power trips the turbine and feedwater pumps, and there is an immediate loss of the PZR heaters, PZR spray and CVCS recirculation flow. EDNS and EDSS power is still available from the batteries, which prevents an immediate reactor trip in response to the power loss. Most SLB events have a cooldown effect on the RCS, where there is little or no pressurization of the RCS. However, this SLB case represents a small break in the steam line with a coincident loss of power that causes the RCS to heat up and pressurize, providing a limiting RCS pressure for steam piping failure events.

After the break occurs, the steam flow through the break increases to a peak just under 90 lbm/s, which is insufficient to overcome the steam pressurization caused by the turbine and feedwater pump trips. However, the break flow is sufficient to