June 15, 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. 24 (eRAI No. 8746) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 24 (eRAI No. 8746)," dated May 16, 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. 8746:

- 15.06.02-1

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. 8746

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Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8746

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

eRAI No.: 8746
Date of RAI Issue: 05/16/2017

NRC Question No.: 15.06.02-1

In accordance with 10 CFR 52.47(a)(2)(iv)(A) and 10 CFR 52.47(a)(2)(iv)(B), the failure of a small line break carrying primary coolant outside containment shall not result in 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 in excess of 25 roentgen equivalent man (rem) total effective dose equivalent (TEDE), and shall not result in a dose to an individual at any point on the outer boundary of the low population zone (LPZ) in excess of 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 failure of small lines carrying primary coolant outside of containment, 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 and by maximizing reactor coolant system (RCS) pressure.

In the Final Safety Analysis Report (FSAR), Tier 2, Section 15.6.2.3.2, "Input Parameters and Initial Conditions," the applicant states that both loss of power conditions and no loss of power conditions are examined at the start of the event to assess conditions which maximize the severity of the consequences of the event. The applicant then goes on to qualitatively discuss the loss of normal alternating current (AC) power input assumption, the loss of normal direct current (DC) and normal AC power input assumption, and the loss of the highly reliable DC power system, normal DC and normal AC power input assumption, but does not provide any information related to the event in which no loss of power occurs. Based on the docketed information, the staff is unable to determine which power input assumption is the most limiting. The staff requests the applicant to provide additional information in the FSAR discussing the no-loss-of-power input assumption and justify whether or not the applicant's current assumed loss of normal AC power input assumption is the most limiting in terms of maximizing mass and energy release and maximizing RCS pressure.

NuScale Response:

FSAR Section 15.6.2.3.2 provides a description of the loss of power assumptions for the failure of small lines carrying primary coolant outside containment. The loss of power scenarios outlined in FSAR Section 15.6.2.3.2 are consistent with the loss of power scenarios outlined in FSAR Section 15.0.0.6.5, Availability of Offsite Power. In addition, consistent with Section 15.0.0.6.2, power is assumed available for the event if the consequences of the event are more limiting.

The loss of normal AC scenario outlined in FSAR Section 15.6.2.3.2 describes why a loss of normal AC is the limiting event by describing what happens when the turbine is tripped and how closing the turbine stop valves leads to a decreased capacity of the steam generator to remove heat from the reactor pressure vessel (RPV). The reduced heat removal then causes a decrease in water density resulting in a pressurizer (PZR) insurge, which increases PZR level and pressure and maximizes mass release, iodine spiking time, and reactor coolant system (RCS) pressure. By not having the turbine trip, the steam generator continues to remove heat from the RPV, thus the resultant effects of the decreased capacity for removing heat by the steam generator does not occur. FSAR Section 15.6.2.3.2 has been revised to include this additional detail.

The NuScale Power Module design does not have automatic reactor coolant makeup capability. FSAR Section 9.3.4.2.3 describes the system operation for the chemical and volume control system, which states that automatic makeup is not provided to avoid masking leaks. The small line failure outside of containment is detected by low PZR pressure or low PZR level, therefore a loss of normal AC power is conservative because it increases PZR pressure and level in order to delay the event detection and maximize the mass release. By having the turbine available, as in the case with power available, the mass release and RCS pressure are lower than if a loss of normal AC occurs, as described above and in FSAR Section 15.6.2.3.2.

FSAR Section 15.6.2.3.2 has been revised to include the additional detail outlined above on why the loss of normal AC power is more limiting than no loss of power for the failure of small lines carrying primary coolant outside containment.

Impact on DCA:

FSAR Section 15.6.2.3 has been revised as described in the response above and as shown in the markups provided in this response.

model is based on the design features of an NPM. The non-loss-of-coolant accident (non-LOCA) NRELAP5 model is discussed in Section 15.0.2.

15.6.2.3.2 Input Parameters and Initial Conditions

This evaluation considers the rupture of the CVCS makeup line, CVCS letdown line, or pressurizer spray line located outside the containment boundary. The assumptions and initial conditions of the evaluations are selected to maximize the severity of the accident by maximizing the mass and energy release out of the break, maximize the duration of the resultant iodine spike, and maximize RCS pressure. Unless specified below, the analyses assume the control systems and engineered safety features perform as designed, with allowances for instrument inaccuracy. No operator action is credited to mitigate the effects of a CVCS line break outside containment.

Table 15.6-4 provides inputs and assumptions for the three scenarios. The maximum mass release scenario is a double ended CVCS letdown line break and the maximum iodine spiking time scenario is an equivalent 100 percent cross-sectional area makeup line break. The maximum RCS pressure scenario is an equivalent 100 percent cross-sectional area makeup line break. The following are key input parameters:

- core power (102 percent) - The enthalpy in the riser, where the makeup line is located, and the density in the downcomer, where the letdown line is located, is maximized at the maximum power of 102 percent.
- pressurizer pressure (1920 psia) - In order to delay the low pressurizer pressure signal, which initiates CVCS isolation and terminates the break flow from the NPM, the nominal steady state pressure of 1850 psia is increased by the pressure uncertainty of 70 psia.
- pressurizer level (68 percent) - The pressurizer level is increased by the level measurement uncertainty of eight percent in order to delay the low pressurizer level trip, which could cause the reactor trip to occur. When the reactor trip occurs, the resulting cooling of the primary side water increases the rate of depressurization, which leads to a low pressurizer pressure trip and CVCS isolation, ending the transient. So, delaying the reactor trip delays the CVCS isolation resulting in more primary coolant flow from the break.
- A combination of core parameters is used to provide a limiting power response. Sensitivity cases show that the end-of-cycle core (EOC) parameters maximize iodine spiking time, while the beginning-of cycle (BOC) core parameters maximize mass release and RCS pressure. Table 15.0-8 provides the EOC and BOC moderator temperature and Doppler coefficients.
- Loss of power - Loss of power conditions, as described below, and no loss of power conditions are examined at the start of the event and concurrent with a reactor trip.

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  - Loss of normal AC - The turbine is tripped and feedwater is lost. The module protection system (MPS) remains powered so safety systems are not automatically actuated. The small line failure outside of containment is detected by the MPS on low pressurizer pressure or low pressurizer level. When the turbine is tripped, the turbine stop valves close, leading to a decreased capacity of the steam generators to remove heat from the RPV. This causes the pressurizer pressure to increase, ~~and~~ the water density to decrease, and the pressurizer level to increase which delays the event detection and maximizes the mass release, iodine spiking time, and RCS pressure. It also leads to a reactor trip followed by MPS signals to initiate containment isolation. By having the turbine available, as in the case with power available, the mass release and RCS pressure are lower than if a loss of normal AC occurs. Therefore, a loss of AC power at the start of the event is conservative, as confirmed in sensitivity studies. ~~Sensitivity studies show that a loss of AC power at the start of the event is conservative for each of the scenarios.~~

- Loss of the normal DC power system (EDNS) and normal AC - Power to the reactor trip breakers is provided via the EDNS, so, in addition to the above, a reactor trip occurs. Having the reactor trip closer to the time of event initiation leads to quicker containment isolation and reduced mass release. Thus, it is conservative to extend the reactor trip.
- Loss of the highly reliable DC power system (EDSS), EDNS, and normal AC - Power to the module protection system (MPS) is provided via the EDSS, so this scenario results in an actuation of DHRS, the 24 hr timer for ECCS, and containment isolation. This scenario is non-conservative for the reasons outlined above.

• A single failure of the main steam isolation value (MSIV) on one steam generator to close is included as a sensitivity case. The sensitivity shows that, because the MSIV closes faster than the secondary MSIV, steam line pressure and RPV pressure is greater when the MSIV closes. Therefore, assuming no single failure is more conservative.

15.6.2.3.3 Results

Figure 15.6-1 to Figure 15.6-16 show the system response to the failure of lines carrying primary coolant outside containment. Table 15.6-5 contains the results of the event. The three limiting scenarios begin with a break of a CVCS line outside containment with a coincident loss of normal AC power. The system response from the breaks is similar, only the timing of the MPS signals varies as a result of the inputs and assumptions used to maximize the parameter of interest.

The maximum mass release scenario starts with a double-ended CVCS letdown line break outside containment with a coincident loss of normal AC power. The turbine stop valves close as a result of the loss of normal AC power increasing steam line pressure. A high steam line pressure signal initiates the reactor trip and actuates DHRS. The DHRS actuation signal causes the MSIVs and feedwater isolation valves (FWIVs) to close, and the DHRS actuation valves to open. To maximize the release for radiological purposes, a break in the makeup line is modeled to increase the flow from the RCS to simulate the double-ended break. A low pressurizer level signal occurs, which trips the pressurizer heaters. A low pressurizer pressure signal initiates closure of the containment isolation valves on the CVCS lines, isolating the