NuScale Power

RAIO-0917-55908

September 11, 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. 138 (eRAI No. 8794) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 138 (eRAI No. 8794)," dated August 05, 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. 8794:

- 15.06.03-2

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,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8794

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NuScale Response to NRC Request for Additional Information eRAI No. 8794

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

eRAI No.: 8794
Date of RAI Issue: 08/05/2017

NRC Question No.: 15.06.03-2

Title 10 of the Code of Federal Regulations (10 CFR) 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report (FSAR) that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 for control room radiological habitability. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in Standard Review Plan (SRP) Section 15.0.3. NRC staff needs to ensure that a suitably conservative estimate is determined for the radiological release associated with the steam generator tube rupture (SGTR) event. In addition, 10 CFR Part 50, Appendix A, GDC 54, "Piping systems penetrating containment," requires piping systems penetrating primary reactor containment to be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems.

As indicated by the applicant in FSAR Tier 2, Section 15.6.3.1, "[t]he design of the helical coil steam generators, described in Section 5.4, is different from the design of SGs in conventional pressurized water reactors [PWRs] because primary coolant is located on the outside, or shell side, of the tubes." In addition, the staff notes that the inventory of the SGs is also very small, so the radiological consequences of a SGTR could be more severe than for conventional PWRs. The mitigation of the SGTR event is totally dependent upon closure of the main steam isolation valve (MSIV) or the secondary MSIV, depending on the single active failure assumed. FSAR Figure 15.6- 19 indicates the affected SG level is rapidly increasing as the isolation valve is closing, and Figures 15.6-20 and 15.6-21 confirm that liquid fraction is quickly increasing as the valve is closing. The staff is concerned about the ability of the isolation valve to close under potentially water-solid conditions since, as noted, the volume of NuScale SG secondary is quite small compared to that of conventional PWRs and may be prone to filling during the SGTR event.

For these reasons, the NRC staff requests that the applicant confirm that the case for "Limiting Mass Release" per FSAR Section 15.6.3.2 is the limiting case for highest potential to refill the SGs or provide an evaluation of the limiting case of SG filling. In addition, the NRC staff requests that the applicant confirm that the isolation valves are qualified to close under the

worst predicted steam quality conditions.

NuScale Response:

As stated in FSAR Section 15.6.3.2, the following three limiting steam generator tube failure (SGTF) event scenarios are presented to demonstrate meeting the acceptance criteria of a postulated accident: 1) limiting mass release and iodine spiking time for assessing the dose consequences, 2) limiting reactor coolant system (RCS) pressure for meeting a postulated accident acceptance criterion, and 3) limiting steam generator pressure for meeting a postulated accident acceptance criterion.

The main steam isolation valves (MSIVs) and secondary MSIVs are credited for isolating the faulted steam generator, depending on the scenario. The MSIVs and secondary MSIVs are designed to close in a liquid environment. However, the scenarios analyzed for meeting the acceptance criteria for a postulated accident are mitigated with the MSIVs and secondary MSIVs closing in a steam environment. FSAR Section 15.6.3.3.3 states that the MSIVs and secondary MSIVs close in a steam environment and steam generator overfill occurs well after secondary MSIV isolation.

For the limiting mass release case, the MSIV on the faulted steam generator is assumed to fail open, extending the RCS mass flow from the faulted steam generator until the secondary MSIV closes approximately 30 seconds after the decay heat removal system (DHRS) actuation signal. The secondary MSIV is closed at approximately 300 seconds following the initiation of the event. FSAR Figure 15.6-19 shows the steam generator level, which is below 15 percent at the time of the secondary MSIV closure. Thus the secondary MSIV closes in a steam environment and steam generator overfill occurs well after secondary MSIV isolation.

For the limiting RCS pressure and steam generator pressure cases, it is more conservative to assume that the MSIVs function as designed because the MSIVs close before the secondary MSIVs, maximizing system pressure. FSAR Figure 15.6-28 shows the steam generator level, which demonstrates that the steam generator is less than 5 percent filled at the time of the MSIV closure. Thus the MSIVs close in a steam environment and steam generator overfill occurs well after MSIV closure. FSAR Figure 15.6-29 and FSAR Figure 15.6-30 show the void fraction and static quality of the steam at the MSIV, also indicating a steam environment.

Steam generator overfill is not an issue in the NuScale design. In addition, to provide assurance that the MSIVs and secondary MSIVs are designed for the limiting environment in which they are required to perform, the MSIVs and secondary MSIVs are designed to close in a liquid environment. FSAR Section 6.2.4, FSAR Section 10.3, and FSAR Section 15.6.3 have been revised to clarify that the MSIVs and secondary MSIVs are capable of closing in a liquid environment. The changes to FSAR Section 6.2.4 are shown in the previously submitted NuScale response to RAI 06.02.04-6 of eRAI 8888.

Impact on DCA:

FSAR Sections 10.3.2 and 15.6.3 have been revised as described in the response above and as shown in the markup provided in this response.

accordance with the provisions of ASME Power Piping Code Section B31.1. Additional detail of the safety, quality, and seismic classification of the MSS components is provided in Section 3.2.

Main Steam Piping

Figure 10.1-1 depicts the MSS boundaries, including interconnections with other systems.

The two steam lines combine to mix and equalize the output of the two SG coils. Flanges immediately downstream of the MSIVs are provided to enable disconnection of the piping from the NPM in preparation for moving the module for refueling or maintenance. Immediately downstream of the flanges, the MSS lines pass through the secondary MSIV and secondary MSIBVs. Ball-joint type flanges are used downstream of the secondary MSIVs to reduce containment vessel nozzle stress.

The steam lines from six NPMs are then routed inside the RXB toward the center of the building and then exit the building above ground. They are supported on a pipe rack between the RXB and the TGB.

In the TGB, the MSS lines are each routed to their separate turbine generator set.

Secondary Main Steam Isolation Valves

Design parameters and associated values for the secondary MSIVs are provided in Table 10.3-1.

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Each secondary MSIV is provided with two independent actuator control systems to ensure successful performance of the secondary MSIV function, assuming a single failure. In response to a main steam isolation signal, the secondary MSIVs automatically close. The secondary MSIVs are capable of closing in steam and liquid conditions.

Each secondary MSIV is designed with the capability to periodically test the operability of the valve and associated apparatus, and to determine if valve leakage is within acceptable limits. Each secondary MSIV is seat leakage tested in the forward and reverse flow directions by the valve supplier. Periodic leak testing of each secondary MSIV is performed as described in Section 3.9.

Secondary Main Steam Isolation Bypass Valves

Each of the two secondary MSIVs has a bypass valve that may be used for pressure equalization and warming during NPM startup. The secondary MSIBVs are normally closed and are Seismic Category I, quality group D, ASME B31.1 components. An isolation valve is provided to allow secondary MSIV maintenance, and a safety valve is provided on the bypass line for overpressurization protection.

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The MSIVs and the secondary MSIVs are credited for isolating the faulted SG, depending on the scenario. The MSIVs and secondary MSIVs are designed for the conditions analyzed. The MSIVs and secondary MSIVs are designed to close in steam and liquid conditions. Classification information for the MSIVs and secondary MSIVs are listed in Section 3.2, Table 3.2-1.

15.6.3.4 Radiological Consequences

Table 15.6-11 provides the inputs to the SGTF radiological consequence analysis presented in Section 15.0.3.

15.6.3.5 Conclusions

The acceptance criteria for a potential accident are listed in Table 15.0-2. These acceptance criteria, followed, by how the NuScale Power Plant design meets them, are listed below.

1) Potential core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit. Minimum critical heat flux ratio (CHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

   The fuel integrity is not challenged by a SGTF. The fuel temperatures decrease upon the reactor trip and DHRS actuation, as shown in Figure 15.6-36 and Figure 15.6-37, and the water level remains above the top of the active fuel, as shown in Figure 15.6-35. In addition, the event is bounded by the rapid depressurization predicted during the inadvertent RVV opening event, which is analyzed for critical heat flux and presented in Section 15.6.6.

2) RCS pressure should be maintained below 120 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2520 psia.

   Table 15.6-10 presents the results of the three limiting scenarios. The RCS pressure is below the acceptance criterion.

3) The main steam pressure should be maintained below 120 percent of the design value. The design pressure for the reactor vessel is 2100 psia, thus the acceptance criterion is 2520 psia.

   Table 15.6-10 presents the results of the three limiting scenarios. The main steam pressure, presented as steam generator pressure, is below the acceptance criterion.

4) The containment pressure should be maintained below the design pressure of 1000 psia.

   An SGTF is not an event that challenges containment pressure. Events that discharge RCS fluid directly inside containment bound this event. The peak containment pressure for design basis events is evaluated in Section 6.2.