RAIO-0618-60721

June 29, 2018

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 Supplemental Response to NRC Request for Additional Information No. 138 (eRAI No. 8794) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 138 (eRAI No. 8794)," dated August 05, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 138 (eRAI No.8794)," dated September 11, 2017

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

The Enclosure to this letter contains NuScale's supplemental 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 Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A

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

Enclosure 1:

NuScale Supplemental 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:

This response supplements the docketed response originally provided to Request for Additional Information (RAI) Set No. 138 (eRAI No. 8794) by NuScale letter RAIO-0917-55908 dated September 11, 2017. The NuScale corrective action process identified inconsistencies in internal documentation on the closure requirements for Main Steam Isolation Valves (MSIVs). The original response stated several times that the MSIVs and secondary MSIVs are capable of closing in a liquid environment. Changes to the FSAR in Sections 10.3.2 and 15.6.3 were provided to explicitly state that the MSIVs close in liquid conditions. This supplement response revises those statements to state that the MSIVs and secondary MSIVs close in design basis conditions. This includes the maximum break flow conditions for the limiting steam line break scenarios which generates the worst case loading requirements for the design and qualification of the main steam isolation valve actuators. Lower quality steam flow conditions where the MSIVs are also required to close, like the increase in feedwater flow event or the steam generator tube failure event, generate significantly lower isolation loads on the valve actuator at the time of MSIV closure. This change is therefore still consistent with the original intent of the response to eRAI No. 8794. However, the break flows that were identified as the limiting conditions for the MSIV isolation as part of the response to eRAI No. 8888 were generated at high quality steam conditions.

FSAR Section 15.6.3 was revised to state that the MSIVs and secondary MSIVs close in design basis conditions as shown in the markup provided with this response. FSAR Section 6.2 and Section 10.3 were updated with NuScale's supplemental response to RAI Set No. 70 (eRAI No. 8888) by letter RAIO-1217-57872 dated August 16, 2017.

Impact on DCA:

FSAR Section 15.6.3 has been revised as described in the response above and as shown in the markup provided in this response.

a steam environment and SG overfill occurs well after secondary MSIV isolation. Figure 15.6-20 and Figure 15.6-21 show the void fraction and static quality of the steam at the secondary MSIV. The system response continues as described for the limiting pressure cases below.

The limiting RPV pressure scenario begins with a tube failure at the top of the SG with a coincident loss of normal AC power, closing the turbine stop valves. The SG heat removal capability is degraded after the SGTF. The water in the RPV expands causing a pressure increase until the MPS actuates a reactor trip and DHRS actuation on a high pressurizer pressure signal (Figure 15.6-22 and Figure 15.6-23). The reactor trip is evident in the core power decrease depicted in Figure 15.6-24 and Figure 15.6-25 and successful DHRS actuation is evident in DHRS flow shown in Figure 15.6-26 and Figure 15.6-27. The DHRS actuation includes closure of the MSIVs, isolating the break flow through the tube to the environment. The MSIVs close five seconds following DHRS actuation at approximately 15 seconds following the initiation of the event. Because MSIVs close before secondary MSIVs resulting in maximizing the system pressure, it is more conservative to assume that the MSIVs function as designed. Figure 15.6-28 shows the SG level, which is below 5 percent at the time of the MSIV closure, thus the valve closes in a steam environment and SG overfill occurs well after MSIV closure. Figure 15.6-29 and Figure 15.6-30 show the void fraction and static quality of the steam at the MSIV.

As the NPM cools with DHRS, as shown in the pressure, temperature, and level response depicted in Figure 15.6-22 and Figure 15.6-23, and Figure 15.6-31 through Figure 15.6-37, the pressurizer level decreases and the pressurizer heater is tripped by the low pressurizer level setpoint signal. The system continues to cool using DHRS (Figure 15.6-27), RCS flow stabilizes (Figure 15.6-38 and Figure 15.6-39), RCS temperature (Figure 15.6-31 and Figure 15.6-32) and fuel temperature (Figure 15.6-36 and Figure 15.6-37) stabilize and continue to decline, the core remains subcritical (Figure 15.6-40), and the water level is well above the top of the active fuel (Figure 15.6-35). The system response shows that the event has terminated and the NPM reaches a safe, stabilized condition.

The limiting SG pressure scenario begins with a tube failure at the bottom of the steam generator with a concurrent loss of normal AC power. The progression of the scenario is similar to the limiting RPV pressure scenario described above. The timing of the MPS signals, reactor trip, DHRS actuation and MSIV closure changes due to the parameters and assumptions that maximize the SG pressure.

The MPS is credited to protect the plant in the event of SGTF. The following MPS signals provide the plant with protection during an SGTF:

- high pressurizer pressure,
- low pressurizer pressure, and
- low pressurizer level.

RAI 15.06.03-2, RAI 15.06.03-2S1

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~~design basis 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.

5) The event should not generate a more serious plant condition without other faults occurring independently.