RAIO-0718-60964

July 20, 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 Response to NRC Request for Additional Information No. 491 (eRAI No. 9557) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 491 (eRAI No. 9557)," dated June 15, 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 Question from NRC eRAI No. 9557:

- 05.02.02-2

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 Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Jennie Wike
Manager, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9557

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NuScale Response to NRC Request for Additional Information eRAI No. 9557

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

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

eRAI No.: 9557
Date of RAI Issue: 06/15/2018

NRC Question No.: 05.02.02-2
Supplemental RAI to RAI 8957 Q 03.09.06-8

Background

In RAI 8957, Question 03.09.06-8, the staff noted that NuScale FSAR Tier 2, Section 5.2.2.6, "Applicable Codes and Classification," states that the RSVs are designed in accordance with ASME BPV Code, Section III, Subarticle NB-3500; and function to satisfy the overpressure protection criteria described in ASME BPV Code, Section III, Article NB-7000. As such, the staff submitted a request to NuScale to provide a detail design description including specific design aspects to address the capacity certification for the various fluid conditions over their full range of operating conditions up to and including design-basis accident conditions.

However, the NuScale response did not include sufficient detail to address the sizing of the RSVs. The response stated that analyses of bounding overpressurization transients showed that reactor coolant pressure boundary pressure remained below 110% of design pressure. NuScale FSAR Tier 2, Section 5.2.2.2.1, "Overpressure Protection During Power Operations," states that a turbine trip at full power without bypass capability is the most severe AOO and is the bounding event used in the determination of RSV capacity and the RPV overpressure analyses. The response did not provide the assumptions and reactor input parameters used in the analysis as cited in NUREG-0800, Section 5.2.2, "Overpressure Protection," SRP Acceptance Criteria 3.B.i through v; for example, the staff could not conclude that the analysis had credited the second safety-grade signal from the reactor protection system as initiating the reactor scram as specified in Acceptance Criteria 3.B.iii.

Basis

General Design Criterion (GDC) 15, as it relates to designing the Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems with sufficient margin to assure that the design conditions of the reactor coolant pressure boundaries (RCPB) are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs).

NUREG-0800, Section 5.2.2, "Overpressure Protection," SRP Acceptance Criteria 3.B states

that to assure sufficient overpressure protection, the designs of the safety valves should have sufficient capacity to limit the pressure to less than 110 percent of the RCPB design pressure during the most severe AOO with reactor scram, as specified by ASME Code Article NB-7000.

Request

Please provide a detailed discussion of the RSV sizing capacity and the RPV overpressure protection analyses including the assumptions and initial conditions relative to the guidance in NUREG-0800, Section 5.2.2 Acceptance Criteria 3.B.

NuScale Response:

NuScale is providing below a detailed discussion of the Reactor Safety Valve (RSV) sizing capacity and the Reactor Pressure Vessel (RPV) overpressure protection analyses including how the analysis relates to the guidance in NUREG-0800, Section 5.2.2, *Overpressure Protection*, Acceptance Criteria 3.B.

The sizing calculation for the RSVs uses a methodology based on identifying the maximum volumetric surge rate into the pressurizer, which occurs during the turbine trip transient. There are three general methods to add conservatism to the RSV capacity: increase the heat addition to the Reactor Coolant System (RCS), decrease the heat lost from the RCS, and increase the coefficient of thermal expansion of the coolant (i.e., initiate the transient with coolant at a higher temperature).

This calculation uses a reactor power of 163.2 MW. This is the maximum reactor power for steady state operation assuming a 2% instrument uncertainty. A higher reactor power generates a higher volumetric surge rate into the pressurizer, so this produces the most severe transient in accordance with NUREG-0800, Section 5.2.2, Acceptance Criteria 3.B.i.

In the analysis, reactor trip is artificially delayed until after the RSV lifts, which is after the second trip signal (both high steam pressure and high pressurizer pressure occur before the RSV setpoint is reached). Only pressurizer surge rate data prior to the RSV lift is used in the sizing calculation for the RSVs, so the reactor trip is not credited in this analysis for rapid response. This satisfies the intent of NUREG-0800, Section 5.2.2, Acceptance Criteria 3.B.iii.

The steam generator pressure is set to 600 psia, which is higher than the planned control value. The feedwater temperature is set to 302.5 degrees F, which is the highest expected operating temperature. The elevated temperature and pressure reduce the ability for the secondary side to remove heat, which increases the volumetric surge rate into the pressurizer.

The hot leg temperature is set to 610 degrees F, which is the maximum value allowed by the analytical limits. Hotter RCS coolant has an increased coefficient of thermal expansion, which

increases the surge rate into the pressurizer.

The heat up and pressurization during the selected transient will tend to reduce core power due to reactivity feedback associated with the moderator temperature. Reactivity feedback is neglected to ensure that core power is not reduced, which will result in a greater rate of coolant expansion.

These biased initial system conditions and core properties satisfy NUREG-0800, Section 5.2.2, Acceptance Criteria 3.B.ii.

NUREG-0800, Section 5.2.2, Acceptance Criteria 3.B.iv is not applicable to the NuScale design. The NuScale RSV sizing methodology is based on the maximum volumetric pressurizer surge rate. The minimum required rated capacity for RSVs used in the NuScale Power Module (NPM) design is sized to ensure that that volumetric flow from a single RSV exceeds the maximum volumetric pressurizer surge rate. Therefore, the RCS cannot continue to pressurize once one RSV is open, which occurs at a minimum pressure of 103% of design pressure as specified by American Society of Mechanical Engineers (ASME) Code subparagraph NB-7512.1.

Therefore, NuScale has concluded that the RSV design has sufficient capacity to limit the pressure to less than 110% of the Reactor Coolant Pressure Boundary design pressure during the most severe Anticipated Operational Occurence with reactor scram, as specified by ASME Code Article NB-7000.

Impact on DCA:

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