RAIO-0419-65203

April 11, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 143 (eRAI No. 8957) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 143 (eRAI No. 8957)," dated August 05, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 143 (eRAI No.8957)," dated October 04, 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. 8957:

- 03.09.06-11

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 Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8957

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8957

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

eRAI No.: 8957
Date of RAI Issue: 08/05/2017

NRC Question No.: 03.09.06-11

F. NuScale FSAR Tier 2, Section 5.2.2.10, “Testing and Inspection,” states that testing and inspection of overpressure protection equipment is conducted in accordance with accepted industry standards including Sections III and XI of the ASME BPV Code, Mandatory Appendix I of the ASME OM Code, and the requirements of 10 CFR 50.34(f)(2)(x) promulgating Three Mile Island action plan recommendation item II.D.1. Section 5.2.2.10 also provides a summary of planned testing of each RSV to be performed by the supplier. The summary of planned RSV testing in NuScale FSAR Tier 2, Section 5.2.2.10 does not address ASME Standard QME- 1-2007, or include all ASME BPV Code testing requirements to demonstrate the functional capability of the RSVs for their full range of operating conditions and their required capacity. In addition, some of the planned RSV testing provisions do not appear consistent with accepted practice for safety valve qualification testing and production testing and seat tightness.

Describe the basis for the summary of the RSV testing provided in the NuScale FSAR. Describe the plans for the RSVs to satisfy 10 CFR 50.34(f)(2)(x) that requires, in part, an application to provide a test program and associated model development and conduct tests to qualify the reactor coolant system relief and safety valves for all fluid conditions expected under operating conditions, transients, and accidents, including consideration of anticipated transients without scram (ATWS) conditions.

NuScale Response:

By letter dated October 4, 2017, NuScale provided a response to NRC RAI 8957, Question 03.09.06-11 (ML17277B826). In this RAI response, NuScale stated that testing to satisfy 10 CFR 50.34(f)(2)(x) for the reactor safety valves (RSVs) would include consideration of anticipated transients without scram (ATWS) conditions. By this letter, this statement is being superseded, as decribed below.

NuScale has provided an exemption request (see Design Certification Application (DCA), Part 7, Chapter 3, *"10 CFR 50.62(c)(1) Reduction of Risk from Anticipated Transients Without Scram"* for further details) that removes the requirement for a system that is diverse and independent from the reactor trip system to automatically initiate a turbine trip under conditions of an ATWS. In addition, NuScale also requested exemption from the requirement to provide a diverse auxiliary feedwater system (AFWS) initiation. The NuScale design does not rely on diverse and independent turbine trip functionality to mitigate the consequences of ATWS events. Additionally, the NuScale Power Plant design does not include an AFWS (or rely on the decay heat removal system to mitigate the consequences of ATWS events); therefore, the portion of 10 CFR 50.62(c)(1) that requires diverse capability to initiate AFWS is not applicable to the NuScale design. Therefore, the full ATWS requirement is not applicable to the NuScale design.

Based on the requested exemption, statements provided in response to RAI 8957, Question 03.09.06-11 were erroneous and require correction.

NuScale is hereby correcting the response to RAI 8957, Question 03.09.06-11 as follows:

The original response is below. The context of this discussion regards the testing program for the RSVs.

> *"Tier 2, FSAR Table 1.9-5 states that 10 CFR 50.34(f)(2)(x) is applicable except for application to PORV block valves (which are not included in the NuScale design). The testing will include flow testing for all fluid conditions expected under operating conditions, transients and accidents, as required in 10 CFR 50.34(f)(2)(x). Consideration of anticipated transients without scram (ATWS) conditions will be included in the test program. The proposed testing is consistent with accepted practice for safety valve qualification testing and production testing and seat tightness."*

Revised response (emphasis added to show modifications):

> *"Tier 2, FSAR Table 1.9-5 states that 10 CFR 50.34(f)(2)(x) is applicable except for application to PORV block valves (which are not included in the NuScale design) <u>and consideration of ATWS conditions in the testing program (due to the exemption request provided in the NuScale DCA, Part 7, Chapter 3)</u>. The testing will include flow testing for all fluid conditions expected under operating conditions, transients and accidents, as required in*

10 CFR 50.34(f)(2)(x). The proposed testing is consistent with accepted practice for safety valve qualification testing and production testing and seat tightness."

Finally, changes have been made in the Final Safety Analysis Report (FSAR), Table 1.9-5 and FSAR, Sections 5.2.2.1 and 5.2.2.2 to further clarify the NuScale design with respect to the requested exemption from 10 CFR 50.62(c)(1). Changes in FSAR, Table 1.9-5 indicate that testing for ATWS requirements is not technically relevant since NuScale has taken an exemption to 10 CFR 50.62(c)(1). Changes to FSAR, Sections 5.2.2.1 and 5.2.2.2 removed information that indicated that ATWS is a design basis requirement for overpressure protection and associated RSV functions during ATWS conditions.

Impact on DCA:

FSAR Table 1.9-5 and Sections 5.2.2.1 and 5.2.2.2 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 03.09.06-11S1, RAI 06.02.04-4S1, RAI 06.02.04-4S2, RAI 06.02.04-7S1, RAI 06.02.04-9, RAI 06.02.04-9S1, RAI 08.01-1, RAI 08.02-4, RAI 08.02-6, RAI 08.03.02-1, RAI 09.02.06-1, RAI 12.03-31, RAI 12.03-32, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35, RAI 12.03-36, RAI 12.03-37, RAI 12.03-39, RAI 12.03-40

Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)

Item	Regulation Description / Title	Conformance Status	Comments	Section
50.34(f)(1)(i)	Perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant (II.B.8)	Partially Conforms	Design certification will address reliability of core and containment heat removal systems, with an update required by COL applicant to reflect site-specific conditions.	19.0 19.1 19.2
50.34(f)(1)(ii)	Perform an evaluation of the proposed auxiliary feedwater system (II.E.1.1)	Not Applicable	This rule requires an evaluation of proposed PWR auxiliary feedwater (AFW) systems. The NuScale plant design does have an AFW system like a typical LWR. Neither the literal language nor the intent of this rule applies to the NuScale design.	Not Applicable
50.34(f)(1)(iii)	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA (II.K.2.16 and II.K.3.25)	Not Applicable	The NuScale reactor design differs from large PWRs because the NuScale design does not require or include reactor coolant pumps. Rather, the NuScale design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable
50.34(f)(1)(iv)	Perform an analysis of the probability of a small-break LOCA caused by a stuck-open power-operated relief valve (PORV) (II.K.3.2)	Not Applicable	This guidance is applicable only to PWRs that are designed with power-operated pressurizer relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable
50.34(f)(1)(v)	Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection and reactor core isolation cooling system initiation levels (II.K.3.13)	Not Applicable	This requirement applies only to BWRs.	Not Applicable
50.34(f)(1)(vi)	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves (II.K.3.16)	Not Applicable	This requirement applies only to BWRs. Regardless, the issue contemplated by this requirement was related to power-operated relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable
50.34(f)(1)(vii)	Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system design modifications that would eliminate the need for manual activation (II.K.3.18)	Not Applicable	This requirement applies only to BWRs.	Not Applicable

Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)

Item	Regulation Description / Title	Conformance Status	Comments	Section
50.34(f)(2)(viii)	Provide capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials (II.B.3)	Departure~~Partially Conforms~~	The NuScale design does not rely on primary coolant or containment samples to assess the extent of potential core damage. The NuScale design relies upon radiation monitors under the bioshield and core exit temperature indications for this assessment. Per DSRS 9.3.2, post-accident sampling is a contingency plan to be developed by a COL applicant (COL Item 9.3-2). The NuScale design supports an exemption from the portions of 10 CFR 50.34(f)(2)(viii) related to demonstrating the personnel radiation exposures.~~As described by SRP 9.3.2, I.6, and RG 1.206, C.I.9.3.2, a post-accident sampling system is not required provided that the guidance provided in SRP 9.3.2 for utilizing the normal process sampling system (post-accident) has been satisfied.~~	9.3.2 11.5 12.4
50.34(f)(2)(ix)	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction (II.B.8)	Not Applicable	Pursuant to 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), Paragraph (f)(2)(ix) is excluded from the information required to be included in an application for a design certification.	Not Applicable
50.34(f)(2)(x)	Provide a test program and associated model development, and conduct tests to qualify reactor coolant system relief and safety valves and, for PWRs, PORV block valves (II.D.1)	Partially Conforms	This requirement is applicable to the DCA except for aspects specifying PORV block valve testing and consideration of ATWS conditions in the testing program. The NuScale design does not use power-operated relief valves. The ATWS provision is not technically relevant to the NuScale design. This aspect of the regulation relates to reactor designs that rely on the relief and safety valves to mitigate the consequences of an ATWS event. The NuScale design supports an exemption from 10 CFR 50.62(c)(1) because the NuScale design relies on protection system diversity to prevent an ATWS, rather than design features to mitigate the condition. As a result, the module response to an ATWS is not analyzed in FSAR Section 15.8, such that the performance of the relief and safety valves is not relied upon to meet the ATWS safety criteria. Therefore, consideration of ATWS conditions in the relief and safety valve test program is not necessary to ensure acceptable performance.	5.2.2

5.2.2 Overpressure Protection

Each NPM is provided with overpressure protection features to protect the RCPB, including the primary side of auxiliary systems connected to the RCS, and the secondary side of the SGs from overpressurization.

RAI 10.03-3

Integrated overpressure protection is provided for the RCPB by two ASME BPVC Section III safety valves during normal operations and anticipated operational occurrences (AOOs). Integrated overpressure protection is provided for the secondary side components with the same design pressure as the RCPB by system design that does not exceed the ASME BPVC service limits during normal operations and AOOs. The low temperature overpressure protection (LTOP) system consists of the RVVs and provides overpressure protection during low temperature conditions.

Two pilot-operated RSVs are installed above the pressurizer volume on the top of the RPV head to provide overpressure relief for the RCS. These valves relieve the RCS pressure directly into containment. Structural design and valve qualification information related to the RSVs is provided in Section 5.2.2.4.

During NuScale Power Module startup and shutdown conditions with the module at low temperature conditions and the RPV not vented, LTOP is provided by the RVVs to prevent exceeding the maximum allowable pressure. Three RVVs are connected to the RPV above the PZR volume and discharge steam and water directly into containment. A temperature dependent pressure setpoint is provided when RPV temperature is below the LTOP enabling temperature. Upon LTOP actuation by the module protection system logic, the RVVs open to limit RCS pressure below the maximum allowable pressure.

Further description of the qualification, design, and operation of the RVVs, including the actuators, is provided in Section 6.3.

5.2.2.1 Design Bases

RAI 03.09.06-11S1

Overpressure protection for the RCPB is provided to ensure design pressure conditions are not exceeded during the normal range of operations, including AOOs in accordance with the requirements of 10 CFR 50, Appendix A, GDC 15. The overpressure protection system is designed with sufficient capacity to prevent RCPB pressure from exceeding 110 percent of design pressure during normal operations and AOOs~~ and ensures that design limits are not exceeded during an anticipated transient without scram (ATWS)~~. The overpressure protection system is able to perform its function assuming a single active failure and a concurrent loss of normal AC power.

The overpressure protection system for the secondary system ensures the ASME BPVC service limits are not exceeded during specified service conditions.

RAI 10.03-3

- loss of condenser vacuum
- inadvertent MSIV closure
- steam pressure regulator failure closed
- loss of normal AC power
- loss of feedwater
- inadvertent operation of the DHRS

These AOOs are further described in DCD Chapter 15 and include plant initial conditions and system parameters, assumptions used in the analysis, and a list of systems and equipment assumed to operate, reactor trip signals, and sensitivity of the system's performance to variations in the event conditions, parameters, and characteristics.

RAI 03.09.06-11S1

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 RCS and the PZR steam space are sized to avoid an RSV lift during normal operational transients at full power conditions, with system and core parameters within normal operating range, that produce the highest RPV pressure. In the event of a safety valve lift, the size of the PZR steam space is sufficient to preclude liquid discharge~~, except for an ATWS condition, where multiple lifts with liquid discharge are predicted~~.

RAI 03.09.06-11S1

The RSVs are designed with sufficient capacity to prevent RCPB pressure from exceeding 110 percent of design pressure under normal and abnormal conditions~~ and to ensure that design limits are not exceeded during an ATWS~~. The analytical model used for the analysis of the overpressure protection system and the basis for its validity is provided in NuScale Topical Reports TR-0516-49416, "Non-LOCA Transient Analysis Methodology," (Reference 5.2-1) and TR-0516-49422, "LOCA Evaluation Model" (Reference 5.2-2). The analysis maximizes the net heat input into the RCS and maximizes the coefficient of thermal expansion of the RCS coolant to determine the volumetric capacity of a single RSV. The RSV volumetric capacity considers the largest surge rate into the PZR.

5.2.2.2.2 Low Temperature Overpressure Protection System

Overpressure protection during low temperature conditions is provided by the RVVs, which are part of the RCPB and are designed in accordance with ASME BPVC, Section III, Subsection NB.

An RCS overpressurization during low temperature conditions could occur due to equipment malfunctions or operator error that results in excessive heat or inventory being added to the RCS, including inadvertent energization of the PZR heaters, inadvertent operation of the module heatup system, or excessive CVCS makeup. Increased RCS inventory events and inadvertent operation of the module heatup system are terminated by isolation of RCS injection line on high PZR water