April 13, 2018

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. 375 (eRAI No. 9201) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 375 (eRAI No. 9201)," dated February 28, 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. 9201:

- 05.02.05-7

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 Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com.

Sincerely,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9201

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NuScale Response to NRC Request for Additional Information eRAI No. 9201

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

eRAI No.: 9201
Date of RAI Issue: 02/28/2018

NRC Question No.: 05.02.05-7

If an item meets any of the criteria specified in 10 CFR 50.36(c)(2)(ii), then a technical specification (TS) limiting condition for operation (LCO) must be established for that item.

In its response to RAI 8843, NuScale stated that the leak-before-break (LBB) leakage limit on leakage from the main steam (MSS) or feedwater (FWS) lines does not satisfy 10 CFR 50.36(c)(2)(ii) Criterion (2) for TS LCO because these lines are not part of the reactor coolant pressure boundary and the limit is not a “process variable, design feature, or operating restriction” for an initial condition of the analyses. NuScale further stated that the LBB leakage limit is solely an indicator for the need to take further action to investigate the source of leakage and evaluate the potential consequences of that leakage. Therefore, NuScale did not propose a TS LCO for the LBB leakage limit.

If a DC applicant asserts that no LCO is needed, it must show that none of the four criteria of 10 CFR 50.36(C)(2)(ii) are satisfied for that item. NuScale has not addressed whether Criterion 4 is satisfied for LBB leakage limit in its RAI response. Therefore, NuScale’s RAI response is insufficient to support its position that no LCO is needed.

After reviewing the applicant’s RAI response, the staff view is that NuScale’s characterization of the LBB leakage limit as solely an indicator for the need to take further action is not fully correct. It should be noted that the LBB leakage limit is related to the critical crack size in the LBB analyses described in FSAR Section 3.6.3. Beyond the critical crack size, the crack growth becomes unstable, and the success of LBB to prevent gross pipe failures (i.e., high-energy pipe breaks) cannot be assured with the technical information currently available to the staff in the DC application. Accordingly, the dynamic effects resulting from the potential pipe breaks should be evaluated to meet the GDC 4 requirement such that nearby SSCs important to safety are protected from the dynamic effects resulting from the postulated high-energy pipe breaks. The NuScale FSAR Tier 2, Section 3.6.2 states that the dynamic effects of MSS or FWS pipe breaks are not analyzed based on the success of LBB to prevent such high energy line breaks. As discussed above, the staff view is that the SSCs important to safety inside the NuScale containment are not protected from the dynamic effects of jet impingement and pipe whip from possible MSS and FWS high energy line breaks when the LBB leakage limit is exceeded.

In its response to RAI 8843, NuScale proposed to use the procedures being used in RG 1.45 for prolonged low-level RCS leakage to also monitor the LBB leakage. However, leakage with no upper limit, as proposed by the applicant, is not related to or determined by the LBB critical crack size. As discussed above, the consequences of exceeding the LBB limit compounded with the design of unprotected instrumentation and unprotected SSCs to mitigate a design basis accident are serious. Even though MSS and FWS lines are not part of the reactor coolant pressure boundary, the failure of LBB could result in the break of these high energy lines, and the dynamical effects could lead to:

- the failure of the instrumentation used to detect/indicate a significant abnormal degradation of the reactor coolant pressure boundary (as indicated in Criterion 1 of 10 CFR 50.36(c)(2)(ii), or

- the failure of or a challenge to the integrity of a fission product barrier due to jet impingement and pipe whip and an initial condition (critical crack size) for the LBB analyses (as indicated in Criterion 2 of 10 CFR 50.36(c)(2)(ii).

In addition, the risks associated with the failure of LBB compounded with unprotected SSCs inside containment have not been analyzed by the applicant in the RAI response. In the past, all design certifications (such as AP1000 and USEPR) that proposed to credit LBB for RCS, and MSS lines have TS LCOs for the LBB leakage limit. AP1000 TS LCO 3.7.8 for the main steam line is a good example.

Therefore, the NRC staff determined that 10 CFR 50.36(c)(2)(ii) Criteria 1, 2, and possibly Criterion 4 apply to the LBB leakage limit of any high energy line break including MSS and FWL. The applicant is requested to propose such a TS LCO.

NuScale Response:

The following response was based on, and includes consideration of, the clarification provided during the February 28, 2018 call between NuScale Power and the NRC staff.

DCA section 3.6.3 describes leak-before-break (LBB) evaluation procedures applicable to ASME Class 2 main steam and feedwater piping systems inside the containment. Section 3.6.3.1 evaluates potential degradation mechanisms for this piping, and section 3.6.3.2 evaluates the materials used and fabrication of the piping. DCA section 3.6.3.3 describes the analysis used to demonstrate that the plant design includes adequate assurance and margin to ensure leak detection before such a break propagates. Section 3.6.3.4 provides an analysis of the piping inside the containment to demonstrate that it meets the criteria for LBB detection and treatment. Section 3.6.3.5 describes the available leak detection instrumentation by reference to DCA section 5.2.5. In addition, DCA Section 3.6.3.5 indicates that limits and controls over availability of secondary system leakage detection will be included in the owner-controlled requirements manual.

The design, construction, operation, and maintenance of a NuScale facility will be conducted in accordance with the Licensing Basis, including the design basis, as described in the DCA and other regulatory correspondence. The design basis described in the DCA for the main steam and feedwater pipe is subject to LBB. If a leak were detected in excess of the LBB limit, the licensing basis for the facility would no longer be met because the feedwater and main steam line piping would no longer satisfy its design basis. Plant procedures will require the facility to address and resolve such a failure in a timely manner consistent with Appendix B of 10 CFR 50. Indication of a leak from the main steam or feedwater piping inside containment would require resolution by placing the unit in a safe condition - in this case depressurizing the piping by shutdown of the unit. No additional control to assure conformance with the licensing and design basis of the plant, nor to correct a situation in which the LBB limit is challenged is required.

Specifically with respect to Criterion 4 and as clarified in the February 28, 2018 call, 10 CFR 50.36 requires that a technical specification limiting condition for operation LCO be established for

> [a] structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Main steam line and feedwater line leak detection instrumentation and limits do not satisfy this criterion. The Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors 58 FR 39132, clarifies the intent of this criterion. The NuScale design is unique and the structures, systems, and components (SSC) explicitly listed in the clarification do not include similarly designed SSC in the NuScale design.

The clarification continues with a discussion of applicability of the criterion to SSC not otherwise listed explicitly. The policy states

> It is the intent of this criterion that those requirements that [probabilistic safety analysis (PSA)] or operating experience exposes as significant to public health and safety, consistent with the Commission's Safety Goal and Severe Accident Policies, be retained or included in Technical Specifications.
>
> The Commission expects that licensees, in preparing their Technical Specification related submittals, will utilize any plant-specific PSA or risk survey and any available literature on risk insights and PSAs. This material should be employed...to verify that none of the requirements...contain constraints of prime importance in limiting the likelihood or severity of accident sequences that are commonly found to dominate risk.

Chapter 19 of the DCA describes the probabilistic risk assessment and severe accident evaluation for the NuScale design. The chapter demonstrates how the risk associated with the design compares against the NRC goals of less than $1x10^{-4}$/year for core damage frequency and less than $1x10^{-6}$/year for large release frequency. The evaluation provided in the DCA includes consideration of secondary side line break as described in DCA section 19.1.4.1.1.2. The accident sequence modeled in the PRA is described, including the event tree for secondary side line break, is in DCA section 19.1.4.1.1.4. The DCA PRA demonstrates that the design exceeds the NRC goals by orders of magnitude.

Additionally, risk significance is discussed in FSAR Section 19.1.4.1.1.9. The PRA evaluates risk significance for systems that mitigate initiating events. There is also a metric that identifies risk significant initiating events and human actions. The PRA does not specifically model systems designed to anticipate a feedwater or main steam line break inside containment. However the PRA does model secondary line breaks, and considers breaks inside and outside of containment. Such breaks do not pose a significant challenge to the NuScale design as they are easily mitigated as described in DCA Chapter 15, "Transient and Accident Analyses." Secondary line break initiating events are not measurable contributors to risk (defined as 20% of CDF or LRF). And a sensitivity study using generic data for the frequency of a secondary line break had a negligible impact on risk (less than a factor of 2 increase in CDF).

There are no human actions that meet the risk significance thresholds based on the Level 1 PRA. The analysis done to calculate the frequency of a secondary line break in the PRA does not consider leak before break actions. The analysis is based on generic pipe failure data and the screening of failures expected to be prevented or mitigated by the NuScale piping system designs (using expert judgment coupled with insights of underlying causes of pipe degradation).

The NuScale design is new and therefore no design-specific operating experience exists. However the plant systems are designed to similar codes and standards as existing plants, and operations and maintenance will be similarly managed and implemented as described in the DCA. Based on operating experience at commercial nuclear power plants, inclusion of secondary system leak before break detection instrumentation and limits on detection does not satisfy criterion 4 of 10 CFR 50.36(c)(2)(ii). This conclusion is consistent with NRC Staff's evaluation of the AP1000 TS LCO 3.7.8 as described in the NUREG-1793, Final Safety Evaluation Report for the AP1000, Section 16.2.10, which states "the main steam line leakage limit ... does not satisfy any of the criteria in 10 CFR 50.36(c)(2)(ii)...."

Based on this analysis, the secondary system leak before break detection instrumentation and limits on detection does not satisfy criterion 4 with regard to operating experience or probabilistic risk assessment as established in 10 CFR 50.36(c)(2)(ii).

Impact on DCA:

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