RAIO-0319-64749

March 05, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 375 (eRAI No. 9201) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 375 (eRAI No. 9201)," dated February 28, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 375 (eRAI No.9201)," dated April 13, 2018
3. NuScale Power, LLC Supplemental Response to NRC "Request for Additional Information No. 375 (eRAI No. 9201)," dated December 20, 2018

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

Sincerely,

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

Enclosure 1:

NuScale Supplemental 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 staff provided additional comments on the original responses to this RAI:

Suggested bases from staff:

> *"The LBB leakage limit on main steam system and feedwater system piping inside containment satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) because it prevents an inside containment steam line break and feedwater line break from occurring, which protects the initial condition assumption in the analyses of these postulated accidents, that "the engineered safety features perform as designed (Refs. 2 and 3)." Were either of these postulated accidents to occur, the validity of this initial condition assumption and the associated safety analysis would not be assured because of the potential for adverse interaction between affected secondary system piping within the containment and other credited safety related equipment located within the containment."*

Backup suggestion:

> *This specification has been included in Technical Specifications due to the potential for adverse interaction between in containment secondary system piping and other safety related equipment located inside the containment if a postulated failure occurred."*

The initial TS surveillance frequency of SR 3.7.3.1 needs to be provided so that the staff and licensee will know the value/starting point for use with the Surveillance Frequency Control Program.

The Bases for LCO 3.7.3 were modified to address the staff concerns and align with the staff's provided option. The base frequency for SR 3.7.3.1 was added to FSAR Table 16.1-1.

Impact on DCA:

The Technical Specifications have been been revised as described in the response above and as shown in the markup provided in this response.

RAI 05.02.05-7S2, RAI 16-30, RAI 16-1S1, RAI 16-60

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies

Surveillance Requirement	Base Frequency	Basis
3.1.1.1	24 hours	The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required shutdown margin (SDM). This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.
3.1.2.1	31 effective full-power days (EFPDs)	The required subsequent Frequency of 31 EFPDs, following the initial 60 EFPDs after exceeding 5% rated thermal power (RTP), is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g. axial offset (AO)) monitored by the core monitoring system for prompt indication of an anomaly.
3.1.4.1	12 hours	Verification that individual control rod assembly (CRA) positions are within alignment limits at a 12 hour Frequency provides a history that allows the operator to detect a CRA that is beginning to deviate from its expected position. The specified Frequency takes into account other CRA position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.
3.1.4.2	92 days	The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the CRAs.
3.1.5.1	12 hours	Since the shutdown CRAs are not moved during routine operation, except as part of planned Surveillances, verification of shutdown CRA position at a Frequency of 12 hours is adequate to ensure that the shutdown CRAs are within their insertion limits. Also, the Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.
3.1.6.1	12 hours	Verification of the regulating group insertion limits at a Frequency of 12 hours is sufficient to detect CRA that may be approaching the insertion limits since, normally, very little rod motion is expected to occur in 12 hours.
3.1.8.1	30 minutes	Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.
3.1.8.2	24 hours	The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.
3.1.9.1	31 days	A 31 day Frequency is considered reasonable in view of other administrative controls that will ensure a misconfiguration of the chemical and volume control system (CVCS) makeup pump demineralized water flow path is unlikely. Also, the Frequency takes into account other information available in the control room for the purpose of monitoring the status of CVCS makeup pump demineralized water flow path configuration.
3.1.9.2	24 months	The 24 month Frequency is based on the potential for unplanned plant transients if the Surveillances were performed with the unit at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment. The actuation logic is tested as part of Engineered Safety Features Actuation System (ESFAS) Actuation and Logic testing, and valve performance is monitored as part of the Inservice Testing Program.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.6.2.1	12 hours	The Frequency of 12 hours is based on the similarity of the test to a CHANNEL CHECK as performed throughout existing large plant designs. The test verifies the accumulator pressure and thereby assures the OPERABILITY of the valves, as well as the status of the automatically monitored pressure alarms.
3.6.2.2	31 days	Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions.
3.6.2.4	24 months	The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Industry operating experience has shown that these components usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
3.7.1.1	12 hours	The Frequency of 12 hours is based on the similarity of the test to a CHANNEL CHECK as performed throughout existing large plant designs. The test verifies the accumulator pressure and thereby assures the OPERABILITY of the valves, as well as the status of the automatically monitored pressure alarms.
3.7.2.1	12 hours	The Frequency of 12 hours is based on the similarity of the test to a CHANNEL CHECK as performed throughout existing large plant designs. The test verifies the accumulator pressure and thereby assures the OPERABILITY of the valves, as well as the status of the automatically monitored pressure alarms.
3.7.3.1	72 hours	The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in assuring detection of a condition that may be indicative of not meeting the leak-before-break criteria applicable to the in-containment secondary piping.
3.8.1.1	12 hours	The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified for similar neutron detector instruments in LCO 3.3.1.
3.8.1.2	24 months	Industry operating experience has shown that similar components usually pass this Surveillance when performed at the 24 month Frequency.

BASES

APPLICABLE SAFETY ANALYSES

The safety significance of plant leakage inside containment varies depending on its source, rate, and duration. Therefore, detection and monitoring of plant leakage inside containment are necessary. This is accomplished via the instrumentation required by LCO 3.4.7, “RCS Leakage Detection Instrumentation,” and the Reactor Coolant System (RCS) water inventory balance (SR 3.4.5.1). Subtracting identified leakage into the containment area from the total detected leakage inside containment provides qualitative information to the operators regarding possible main steam or feedwater line leakage. This allows the operators to take action should leakage occur which would be detrimental to the safety of the facility if a seismic event occurred.

This specification has been included in Technical Specifications due to the potential for adverse interaction between in-containment secondary system piping and other safety related equipment located inside the containment.

LCO

In-containment secondary piping leakage is defined as leakage inside containment in any portion of the main steam line or feedwater pipe walls. Up to 1.5 gallons per hour (gph) of leakage is allowable because it is below the leak rate for LBB analyzed cases of a secondary line crack twice as long as a crack leaking at the detectable leak rate under normal operating conditions including the stress imposed by postulated seismic events. Violation of this LCO could result in continued degradation of the main steam line.

APPLICABILITY

Because of elevated secondary system temperatures and pressures, the potential for in-containment secondary system piping leakage is greatest in MODES 1, 2, and MODE 3 when not PASSIVELY COOLED.

In MODE 3 when PASSIVELY COOLED, and in MODES 4 and 5 an in-containment secondary system piping leakage limit is not provided. In MODE 3 when PASSIVELY COOLED, the secondary system temperatures and pressures are rapidly reducing, resulting in lower stresses and reduced potential for leakage or adverse effects from a postulated secondary system pipe rupture. In MODES 4 and 5 the secondary system piping is depressurized.