RAIO-0817-55206

August 02, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

**SUBJECT:** NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 12 (eRAI No. 8773) on the NuScale Design Certification Application

**REFERENCES:**

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 12 (eRAI No. 8773)," dated April 25, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 12 (eRAI No. 8773)," dated June 22, 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. 8773:

- 04.04-2

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 Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8773

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**Enclosure 1:**

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 8773

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

**eRAI No.:** 8773
**Date of RAI Issue:** 04/25/2017

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**NRC Question No.:** 04.04-2

10 CFR 50.36(c)(2)(ii)(B) requires that a technical specification limiting condition for operation (LCO) be established for a “process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.” The initial condition ranges assumed in the evaluation of design basis events is provided in Table 15.0-6 of the Final Safety Analysis Report (FSAR). This table includes the range of reactor coolant system (RCS) flowrates assumed to bound the minimum and maximum RCS flowrates. However, the NRC staff has not identified an LCO that would limit operation within the bounds of the RCS flowrates assumed in the transient and accident analyses. The NRC staff relies upon such an LCO to establish a finding that each NuScale Power MODULE will be operated within the bounds of the safety analyses. Accordingly, the NRC staff requests that NuScale provide sufficient justification that the RCS flowrate during normal operation will be maintained within the bounds of the transient and accident analyses.

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**NuScale Response:**

This response supplements the original RAI 04.04-2 (eRAI 8773) response. NuScale has modified Final Safety Analysis Report (FSAR) Section 4.4.5.2 to clarify that Reactor Coolant System flow measurement will be performed during power ascension following each refueling outage. This flow measurement will confirm that the flow used in the steady state and transient analyses remains bounding.

**Impact on DCA:**

FSAR Section 4.4.5.2 has been revised as described in the response above and as shown in the markup provided in this response.

#### 4.4.5.2 Initial Power and During Operation

RAI 04.04-2S1

During power ascension, core power distribution measurements and inlet and outlet thermocouple measurements are taken to confirm that the peaking factors used in the thermal-hydraulic design are conservative. In addition, RCS flow measurement is performed during power ascension following refueling outages. This flow measurement provides confirmation that the RCS loop resistance used in the thermal-hydraulic design and Chapter 15 transient and accident analyses remains bounding.

#### 4.4.5.3 Component Inspections

Fuel assembly component surveillance is performed during refueling outages as described in Section 4.2.4.6.

### 4.4.6 Instrumentations Requirements

#### 4.4.6.1 Incore Instrumentation System

The in-core instrumentation system (ICIS) uses neutron flux measurements in twelve (12) fuel assemblies to determine a three-dimensional power distribution in the core (see Section 4.3). During startup testing (Section 14.2), this power distribution is compared to the power distribution assumed in the thermal-hydraulic analysis to ensure that the peaking factor used in the analysis is conservative.

In addition, temperature is continuously monitored at the inlet and outlet of the 12 fuel assemblies and this information is used to verify that proper flow rates are being used in the thermal-hydraulic analysis. The location of the 12 assemblies that contain the incore flux and inlet and outlet temperature detectors is shown in Figure 4.3-18

The conservatism of the VIPRE-01 subchannel code is established by comparison against experimental data and other computer code analyses as described in Reference 4.4-3.

#### 4.4.6.2 Module Protection System

The following MPS reactor trips provide automatic protection of the reactor core safety limits:

- RCS high pressure
- RCS low pressure (above 600 degrees F) and low-low pressure
- nuclear high power trip

These protective trips ensure that MCHFR limits are not exceeded and that fuel centerline temperature stays below the melting point. These trips also ensure that average enthalpy in the riser is less than the enthalpy of saturated liquid and that core exit quality is within the limits defined by the CHF correlation.