August 15, 2017

Docket: PROJ0769

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. 8870 (eRAI No. 8870) on the NuScale Topical Report, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," TR-0516-49417, Revision 0

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 8870 (eRAI No. 8870)," dated June 30, 2017
2. NuScale Topical Report, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," TR-0516-49417, Revision 0, dated July 2016

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. 8870:

- 01-14

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
Director, Regulatory Affairs
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. 8870

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8870

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

NuScale Power

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8870
Date of RAI Issue: 06/30/2017

NRC Question No.: 01-14

In accordance with 10 CFR 50 Appendix A GDC 10, “Reactor design,” the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The SRP 15.0.2 acceptance criteria with respect to evaluation models specifies that the chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view.

Bullet 2 of Section 5.2, “Assumptions and Limitations,” of the topical report (TR), TR-0516-49417-P, states that the core bypass flow cannot be modeled. This section correctly states that this will impact the predicted rate of subcooled boiling, and hence void fraction in the riser section. The TR indicates a parametric study could be performed to account for the impact of the neglected core bypass flow on riser void fraction. However details of such a proposed parametric sensitivity study and how the core bypass effect is captured by such a study are not clear.

In order to make an affirmative finding associated with the above regulatory requirement important to safety, NRC staff requests NuScale to describe how the core bypass flow effect is captured by a parametric sensitivity study, and clarify if it intends to carry out such a study as part of the stability analysis methodology.

NuScale Response:

The statement in Section 5.2, bullet 2 of the topical report describes the consequences of the one-dimensional modeling of the reactor coolant system flow, including through the core. The core section includes all possible flow paths (e.g. through fuel bundles and instrument tubes). The statement in Section 5.2, bullet 2 dispositions local effects of the one-dimensional flow in regards to simulating stability. One of these effects is the planar distribution of power and flow which are spread by virtue of the one-dimensional flow approximation. Since planar distribution affects the inception of subcooled boiling, the PIM code allows for a means to check this by allowing the user to control subcooled boiling through input. The topical report states that any such impact is insignificant because the magnitude of subcooled voids and the vertical length they occupy are too small (relative to the riser) to impact the density head. This was confirmed by sensitivity studies that were performed during model development. The default subcooled boiling coefficient is consistently used for all the reported calculations.

Impact on Topical Report:
There are no impacts to the Topical Report TR-0516-49417, Evaluation Methodology for Stability Analysis of the NuScale Power Module, as a result of this response.