UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 6, 2018

Mr. Thomas Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC
1100 NE Circle Boulevard, Suite 200
Corvallis, OR 97330

SUBJECT: COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR THE NUSCALE POWER, LLC REACTOR INTERNALS – ANALYSIS METHODS

Dear Mr. Bergman:

The purpose of this letter is to communicate the U.S. Nuclear Regulatory Commission (NRC) staff's concerns related to the NuScale Power, LLC (NuScale) reactor internals comprehensive vibration assessment program (CVAP) described in Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," of the NuScale Final Safety Analysis Report provided with the NuScale reactor design certification application (DCA) (see NRC Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A272). To date, the NRC staff has been unable to reach a finding of reasonable assurance of adequate protection for the NuScale CVAP.

General Design Criterion (GDC) 4, "Environmental and dynamic effects design bases," of Title 10 of the *Code of Federal Regulations* (CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," requires structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) Section 3.9.2, "Dynamic Testing and Analysis of Systems, Structures, and Components," addresses the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports under vibratory loadings, including those due to fluid flow and postulated seismic events. Regulatory Guide 1.20, "CVAP for Reactor Internals During Preoperational and Initial Startup Testing," Section 2.1(1)(b) states that the method for determining pressure fluctuations, vibration, and resultant cyclic stress in plant systems would need to be justified. RG 1.20 further states that scale testing can be applied for the frequency-dependent acoustic pressure loading and for verifying the pressure loading results from computational fluid dynamics analyses and the supplemental analyses, where the bias error and random uncertainties are properly addressed. Although the SRP and RG are not regulatory requirements, adherence to the guidance would be deemed as meeting GDC 4 requirements.

The NuScale Power Module (NPM) design contains many features that are not present in the current fleet of operating reactors. Examples include helical coil steam generators (SG) within the reactor pressure vessel with primary-side flow over the tubes, and secondary-side flow inside the tubes (including a phase change from liquid to steam inside the tubes), and a SG

tube support structure that differs from features that serve similar safety functions in the operating fleet. Because some features of the NPM design are not present in the current fleet of operating reactors, there are no flow-induced vibration test data from other nuclear power plants available to provide benchmarking for the NuScale analyses.

During interactions with the NRC staff, NuScale has emphasized that the design of the NPM is based on equations and data from open literature using conservative assumptions and that large safety margins exist. However, the NRC staff found during an audit that some of the vibration analyses contained apparently nonconservative assumptions or values, as discussed in the audit report (ML18023A093). The staff is concerned about the potential impact of these apparently nonconservative assumptions or values on the margins in the analytical results pertaining to SG tube margin against fluid-elastic instability, SG tube margin against vortex shedding, CRDS support margin against vortex shedding, in-core instrument guide tube (ICIGT) margin against vortex shedding, decay heat removal system (DHRS) piping margin against acoustic resonance, and control rod assembly guide tube (CRAGT) wear and tube support margin against turbulence buffeting. The staff also found that non-conservatisms in some of the flow-induced vibration (FIV) mechanisms such as fluid-elastic instability, vortex shedding, and the turbulent buffeting wear analyses may out-weigh the conservatisms in these analyses. For example, SG tube damping is assumed to be 1.5 percent instead of 1 percent. If damping is higher than 1 percent, RG 1.20 states that damping coefficients should be strongly substantiated with measurements. Another example is the use of averaged flow velocity instead of maximum velocity for reactor internals analyses. Another example is that there are no mesh convergence studies, which could indicate possible bias to results. Also, there were no computational results provided for the SG inlet flow restrictor.

A meeting was held with NuScale on February 23, in the NuScale office in Rockville, MD. At this meeting, the NRC staff discussed the observations made regarding the apparent non-conservatisms and areas were analyses have not been conducted. These observations were documented in the audit summary report and made available to NuScale prior to the meeting. The NRC requests that NuScale evaluate the information provided at and prior to the meeting and the RAIs that the staff has submitted as follow-up to RAI 8884 and those that are forthcoming from the CVAP audit and develop a schedule for addressing the staff concerns that will fit within the staff's published review scheduled for the NuScale Power Reactor. If a schedule within the established review schedule cannot be achieved, propose an alternate schedule, requesting new revised milestones for the review. The staff requests the schedule be developed and provided to the NRC by April 19, 2018. If you have any questions, please contact Omid Tabatabai at (301) 415-6616 or via e-mail at omid.tabatabai@nrc.gov.

Respectfully,

/RA/

Frank Akstulewicz, Director
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission

Docket No. 52-048
cc: NuScale listserv

SUBJECT: COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR THE NUSCALE POWER, LLC REACTOR INTERNALS – ANALYSIS METHODS
DATED: **3/06/2018**

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