October 3, 2019 Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Changes to “NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report," TR-0918-60894, Revision 1

REFERENCES: Letter from NuScale Power, LLC to Nuclear Regulatory Commission, “NuScale Power, LLC Submittal of ‘NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report,’ TR-0918-60894, Revision 1,” dated August 2, 2019 (ML19214A253)

During a September 30, 2019 public teleconference with project manager Marieliz Vera, and reviewers Timothy Lupold, Yuken Wong and Steven Hambric (consultant) of the NRC Staff, NuScale Power, LLC (NuScale) discussed potential updates to the "NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report," TR-0918-60894, Revision 1. As a result of this discussion, NuScale has changed the "NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report." The Enclosure to this letter provides a mark-up of the report pages incorporating revisions to the report in redline/strikeout format. NuScale will include these changes as part of a future revision to the "NuScale Comprehensive Vibration Assessment Program Measurement and Inspection Plan Technical Report.”

This letter makes no regulatory commitments or revisions to any existing regulatory commitments.

If you have any questions, please feel free to contact Marty Bryan at 541-457-7172 or mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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6.1.5.3 Upper/Lower Riser Slip Joint

The upper and lower riser sections meet at the slip joint where the bellows is designed to handle the thermal expansion of the riser to ensure positive force at the interface. The slip joint is designed to be maintained in a closed condition and has a small pressure difference with a convergent flow passage. LFI is not expected at this annular passage. This location, however, is monitored to confirm this conclusion.

The upper riser in the area of the transition is shown in Figure 6-3. A bellows allows for thermal expansion in the upper riser and is located just above the slip joint. The slip joint movement could occur in either the vertical or lateral directions.

6.2 FIV Detection Methodology

Characterization of FIV mechanisms for the SG are provided during the TF-3 testing. Validation testing is also planned for the SG inlet flow restrictor and for AR in the CNTS steam piping. The purpose of the startup testing measurements discussed here is not primarily for structural validation but to detect any unexpected FIV responses. In order to do so, the sensors and acquisition platform used for startup testing needs to be engineered to characterize expected FIV to distinguish it from unexpected. The only expected response is turbulent driven FIV.

Acceptance criteria can be defined in two ways 1) in terms of an unexpected response and 2) sustained vibration stress per the design life (such as an endurance limit based on ASME fatigue curves and/or ASME OM guidance (Reference 9.1.15)). Using the location/magnitude of the highest peak stress intensity, the modal strains and displacements at sensor locations can be determined relative to the peak stress intensity on a normalized basis.

Although the measurements are not intended to be used for structural evaluation, acceptance criteria in terms of stresses is most readily related to measured strain or displacement amplitudes as these are directly proportional. During any unexpected events, it is also important to characterize the motion such that likely sources can be identified. For startup testing, this includes being able to distinguish modes of vibration related to FIV (generally first few modes) by both frequency and predominate direction of motion. To make this determination, test and analytical modes need to correlate with respect to the predominant modal direction and corresponding modal frequencies.

Response amplitudes are best trended in both RMS and peak units to capture the overall vibration energy trends and departures from expected trends (peaks) as flow increases. Unexpected wear caused by impacting; however remote of a possibility, should also be able to be characterized by the monitoring system. Impacting differs in characterization as it is not readily detectable through RMS or peak amplitudes alone. Rather, impacting is typically best characterized by the crest factor (ratio of peak to RMS), and is recommended to be measured over a frequency range (2,000+ Hz) considerably higher than is necessary to detect typical FIV phenomenon. Measuring to the higher range ensures that the full peak effect of any impulses are fully characterized (e.g. not attenuated by filtering). For components with strongly-coupled, single-mode responses, crest factors of 2 to 3 are typically observed. For piping and other components exposed to more variable excitation,

9.0 References

9.1 Referenced Documents

9.1.1 American Society of Mechanical Engineers, *Operation and Maintenance of Nuclear Power Plants, Division ~~3~~2: OM Standards~~Guides~~*, ASME OM-2012, Part 3~~11~~, Vibration Testing of Piping Systems~~and Assessment of Heat Exchangers~~, New York, NY.

9.1.2 American Society of Mechanical Engineers, Standard for Verification and Validation in Computational Fluid Dynamics and Heat Transfer, ASME V&V 20-2009, New York, NY.

9.1.3 Au-Yang, M.K., Flow-Induced Vibration of Power and Process Plant Components, A Practical Workbook, ASME Press, New York, NY, 2001.

9.1.4 NuScale Comprehensive Vibration Assessment Program Analysis Technical Report, TR-0716-50439.

9.1.5 U.S. Nuclear Regulatory Commission, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," Regulatory Guide 1.20, Revision 3, March 2007.

9.1.6 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section III, "Rules for Construction of Nuclear Facility Components," New York, NY.

9.1.7 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section V, Nondestructive Examination, New York, NY.

9.1.8 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York, NY.

9.1.9 Ziada, S. and Shine, S., Shrouhal Number of Flow-Excited Acoustic Resonance of Closed Side Branches, Journal of Fluids and Structures, Volume 13 (1999), pg. 127-142.

9.1.10 Schardt, J.F., "Flow-Induced Vibration Characteristics of BWR/6-238 Jet Pumps," GEAP-22201, UC-78, General Electric Company, September 1982.

9.1.11 Au-Yang, M.K. and Jordan, K.B., Dynamic Pressure Inside a PWR – A Study Based on Laboratory and Field Test Data. Nuclear Engineering and Design 58, pg 113-125, 1980.

9.1.12 Chen, Shoei-Sheng. Flow-Induced Vibration of Circular Cylindrical Structures. ANL-85-51, June 1985.

9.1.13 Giraudeau, M., et al. Two-Phase Flow-Induced Forces on Piping in Vertical Upward Flow: Excitation Mechanisms and Correlation Models. Journal of Pressure Vessel Technology, Vol. 135, p. 030907, 2013.

9.1.14 ASME Boiler and Pressure Vessel Code, Section III, Division 1 – Appendix N, Dynamic Analysis Methods, 2013 Edition, no Addenda. New York, NY.

9.1.15 American Society of Mechanical Engineers, *Operation and Maintenance of Nuclear Power Plants, Division 3: OM Guides*, ASME OM-2012, Part 11, Vibration Testing and Assessment of Heat Exchangers, New York, NY.