RAIO-0418-59389

April 03, 2018

Docket No. 52-048

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. 294 (eRAI No. 9273) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 294 (eRAI No. 9273)," dated December 06, 2017

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. 9273:

- 05.04.02.01-1

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,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Samuel Lee, NRC, OWFN-8G9A
Prosanta Chowdhury NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9273

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9273

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

eRAI No.: 9273
Date of RAI Issue: 12/06/2017

NRC Question No.: 05.04.02.01-1

As part of the RCPB, SG tubes must be designed to meet the requirements of 10 CFR Part 50, Appendix A, GDC 32, as it relates to maintaining the integrity of the RCPB. As described in DSRS Section 5.4.2.2, the standard technical specifications (STS) provide for the establishment and implementation of a SG program for maintaining tube integrity. The STSs define structural integrity performance criteria and include a plant-specific tube plugging criterion based on a minimum acceptable tube wall thickness. As stated in DSRS Section 5.4.2.2, RG 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” is an acceptable methodology for determining the minimum wall thickness that ensures adequate safety margins against both burst and collapse.

In Section 5.4.1.6.1 in Tier 2 of the FSAR and Technical Specification (TS) 5.5.4.c, the NuScale design proposes to plug SG tubes containing flaws with a depth of 40 percent or more of the nominal tube wall thickness. According to Table 1.9-2 in Tier 2 of the FSAR, the value was determined according to RG 1.121. However, the NRC staff was not able to determine that the proposed plugging criterion satisfies the structural integrity performance criterion due to the SG tubes in this design being more sensitive to buckling/collapse from external pressure than to burst from internal pressure. Therefore, the NRC staff requests the following information:

a. In order to clarify that the tube plugging criterion is determined according to RG 1.121, revise Section 5.4.1.6 in Tier 2 of the FSAR to include a statement to that effect.
b. An explanation for the statement in Section 5.4.1.2 in Tier 2 of the FSAR that tube integrity is maintained at 50 percent tube thickness degradation under all loading conditions, given that the plugging criterion in TS 5.5.4.c is 40 percent.
c. The 40 percent plugging criterion is based on wear flaws caused by tube support tabs and having a limited circumferential and axial extent. As proposed, the 40 percent through-wall plugging criterion does not limit the axial or circumferential length of the flaws despite being determined for flaws of limited length. Discuss how you determined the axial length that satisfies the plugging criterion and whether that axial length was for both burst and collapse pressure. In addition, discuss your basis for using the von Mises strain in the flat-bottomed flaw under compression at external differential pressure to determine margins rather than the safety factor of 3 as proposed in your TS 5.5.4.b.1 and required for conformance to NEI 97-06 and RG 1.121.
d. The NuScale design has higher pressure on the outside of the SG tubes which creates a

compressive hoop stress on the SG tubes. However, cold-bending the SG tubes could introduce residual tensile stresses. Describe how you evaluated the residual stress in the cold-worked tubes and the potential for creating axial or circumferential inside diameter (ID) surface cracks. Describe how you determined that all potential cracks on the ID surface would not contribute to failure from external pressure. If the bend radius thermal stress relief specification developed for large light-water reactors was used to limit residual stress, how was that specification found to be adequate for the loads applied to NuScale SG tubes? In addition, failure pressure may be sensitive to the ovalization in helically fabricated tubes, therefore, did your analyses with respect to ovalization include the tube bend radius for helical and transition regions where the tube is thinned from bending?

e. Combined License (COL) Item 5.4-1 on Page 5.4-13 of Tier 2 of the FSAR states that a COL applicant referencing the NuScale Power Plant design certification will develop and implement a SG program that includes tube plugging. To meet the requirements of 10 CFR 50.36, the SG program is described in the TSs. While "40%" in your proposed TS 5.5.4.c is not bracketed (bracketed would mean the "40%" is preliminary), it is the NRC staff's understanding from a September 12, 2017, audit status teleconference, that NuScale intends to bracket "40%" in the TSs. In order for the design to fully describe the basis for your proposed tube plugging criterion, add a description to the FSAR of how plant-specific factors were included in the tube plugging criterion and may, therefore, result in COL applicants proposing a criteria other than 40 percent.

NuScale Response:

a. As indicated in Tier 2, Table 1.9-2, the NuScale SG design as discussed in Tier 2 Section 5.4 (including SG Program discussed in section 5.4.1.6) complies with Regulatory Guide (RG) 1.121. FSAR Section 5.4.1.1 has been updated to reflect NuScale compliance with RG 1.121.

b. Analytically, tube integrity is indicated for wear defects of up to 50% of the tube wall thickness under the accident and normal operating loading conditions specified by NEI 97-06. The 40% tube plugging criteria implemented is based on considering uncertainty in the measured flaw depth, as well as uncertainty in the eddy-current inspection method and interpretation of the eddy-current results.

c. The axial length of the wear flaw is postulated based on the prototypic length of an individual case of contact between a SG tube and a SG tube support, which is 0.500" for the NuScale design. The postulated (prototypic) wear flaw (0.500" length) was implemented in a finite element model to demonstrate acceptable performance under the required loading conditions. Both a "flat" and a "tapered" wear flaw were implemented in the FEA analysis, with 50% and 60% through wall depths, respectively. The loading conditions considered for the wear flaw included both three times the normal operating pressure differential (3*NOPD), which for NuScale is an external pressure loading condition (i.e. "collapse" pressure) as well as accident conditions which included both internal (i.e. "burst" pressure) and external pressure differential loads combined with accident (seismic) loads, including applicable safety factors. These loading conditions bounded all other loading conditions (e.g. 1.4 times Limiting Accident Differential Pressure) specified by NEI 97-06.

Per section 2.1 (Structural Integrity Performance Criterion) of NEI 97-06:

"All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential."

As discussed, the prototypic wear flaws were evaluated using 3*NOPD as a loading condition, therefore the factor of three proposed in NEI 97-06 and discussed in NuScale TS 5.5.4.b.1 is considered. The equivalent Von-mises strain observed under the applied loading condition of 3*NOPD was used to demonstrate that tube wall integrity was maintained. The FEA model included a kinematic strain hardening material modeling (plastic behavior), therefore the calculated equivalent Von-Mises stress was compared against the ultimate tensile stress to demonstrate that structural integrity was maintained. Equivalent Von-mises stress was less than 40% of the UTS taken at a bounding design temperature of 650 degrees F and large displacements indicative of buckling were not observed. Therefore, it was concluded that tube wall integrity was maintained under the 3*NOPD loading condition stipulated in NEI 97-06.

d. Residual stresses were not evaluated directly, however utilizing the existing industry guidance on post-bend stress relief (discussed further below) provides adequate protection from excessive residual stress due to any cold work induced during the tube bending process. The effect of residual stresses does not impact the axial and circumferential crack sizes used to establish the tube plugging criteria; residual stresses would only impact the likelihood and/or growth rates of cracks. Existing industry guidelines (i.e.EPRI SG Integrity Assessment Guidelines) provide conservative upper bound crack growth rates for A690 tubing. Based on these growth rates, it has been demonstrated (with margin) for the NuScale design that maximum crack growth rates will not result in any detectable cracks growing to a through-wall depth within a two year period. Therefore, the NuScale SG design adequately addresses any potential impacts to SG tube integrity based on residual stresses due to tube bending.

   The analysis supporting the tube plugging criteria included both axial and circumferential through wall cracks; therefore cracks evaluated were neither ID or OD, but "through-wall". It is considered that full through-wall cracks bound any ID surface partial, through-wall cracks with respect to structural integrity of the tube.

The design of NuScale SG tubes has been shown to provide the same margins (e.g. 3*NOPD, etc.) for all applicable SG tube loading conditions as demonstrated for other existing Light Water Reactors (LWR) designs. Therefore, NuScale uses the same thermal stress relief specification for A690 thermally treated tubing as required for existing designs; only bends with radii less than 10 times the tube OD require stress relief. The NuScale design has no bends with a radius less than 10 times the tube OD. The existing specification for light water reactors is considered adequate for the NuScale design. This conclusion is based on the fact that operating conditions for the SG tubes in the NuScale plant bound the surface stress conditions which occur in the operating LWR fleet because of the operating compressive stresses induced by primary pressure applied to the tube OD.

Maximum allowed tube ovality (based on use of ASME Code Case N-759-2, Tier 2 Table 5.2-1) was included in the FEA models used to demonstrate structural integrity criteria was satisfied for characteristic wear defects. No additional potential wall thinning due to tube bending was considered. For evaluation of axial and circumferential cracks, neither tube ovalization or potential wall thinning due to tube bending was considered. Localized wall thinning due to bending in the helical portion of the tube would be approximately 2%, however greater thinning (up to approximately 5%) could occur in the tighter radius transition bends. The overall margins (e.g. factor of 3 for NOPD) used to develop the tube plugging criteria are substantial, therefore explicit evaluation of all other possible factors (such as wall thinning due to bending) that could impact results is not performed.

e. NuScale has implemented a bracketed value for the tube plugging criteria. However, plant specific factors were not directly considered in developing the NuScale tube plugging criteria. The specific factors that were considered in developing the tube plugging criteria were tube material, tube wall thickness, defect geometry, significant accident loads (i.e. seismic), normal and accident tube differential pressures and eddy current inspection technique and defect sizing uncertainty. These factors are in accordance with NEI 97-06 and EPRI guidance.

   These "design" factors for the NuScale tube plugging criteria are intended to be bounding, however until specific plant sites are identified (COL initiation) this cannot be confirmed. As an example, assumed eddy current uncertainties used to develop the 40% tube plugging criteria considered general EPRI guidance and could be different for the eddy current system which is ultimately qualified for inspection of the NuScale SGs at a specific plant location. Therefore a bracketed value is appropriate to ensure any plant specific factors are considered when establishing the tube plugging criteria for a specific plant.

Impact on DCA:

FSAR Section 5.4.1.1 has been revised as described in the response above and as shown in the markup provided in this response.

The SGs also provide two primary safety-related functions: they form a portion of the RCPB and they transfer decay heat to the DHRS described in Section 5.4.3.

The portions of the SGs that form a part of the RCPB provide one of the fission product barriers. In the event of fuel cladding failure, the barrier isolates radioactive material in the reactor coolant preventing release to the environment.

The SGs perform an integral part of the reactor residual and decay heat removal process when the DHRS is in operation. They transfer heat from the primary coolant to the naturally circulating closed loops that transfer decay heat to the reactor pool.

RAI 05.04.02.01-1

10 CFR 50.55a(g) requires the inservice inspection (ISI) program to meet the applicable inspection requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC). The steam generator system (SGS) components are designed such that the ISI requirements of ASME BPVC, Section XI can be performed, including the preservice inspections of ASME Section III. A SG program, based on NEI 97-06 (Reference 5.4-1) and Regulatory Guide 1.121 is described in Section 5.5.4 of the technical specifications, ~~is used for~~ and implementss~~ing~~ ASME Code Section III and XI for the SG tubes. The primary and secondary sides of the SGs are designed to permit implementation of a SG program that provides reasonable assurance the structural and leakage integrity of the SG tubes is maintained. Integrity of SGs, integral steam plenum, and feedwater plena that make up portions of the RCPB is discussed in Section 5.2.

5.4.1.2 System Design

RAI 05.04.02.01-5

Each SG, located inside the RPV, is comprised of interlacing helical tube columns connecting to two feed and two steam plena. The feed and steam plena comprising a single SG are configured 180 degrees apart. As shown in Figure 5.4-1 and Figure 5.4-2, the configuration of the helical tube columns of the two SGs form an intertwined bundle of tubes around the upper riser assembly with a total of four feed and four steam plena located 90 degrees apart around the RPV. Figure 5.4-3 shows the cross-sectional arrangement of the integral steam plena and feed plena, while Figure 5.4-4 and Figure 5.4-5 show individual cross-sectional views of an individual steam and feed plenum. The main steam supply nozzles and the feedwater supply nozzles are also part of the SGS. Each SG has a pair of feedwater and main steam supply nozzles. The main steam supply nozzles are integral to the steam plenum access ports and the feedwater supply nozzles are integral to the feed plenum access ports as shown in Figure 5.4-4 and Figure 5.4-5, respectively. The primary reactor coolant circulates outside the SG tubes with steam formation occurring inside the SG tubes.

Each SG tube is comprised of a helix with bends at each end that transition from the helix to a straight configuration at the entry to the tubes sheets as shown in Figure 5.4-1. The helical tubes are seamless with no intermediate welds. The helical tubes terminate at the feed and steam plenum tubesheets, where the tubes are secured to the tubesheet by expansion fit and are welded to the tubesheet on the