Response to SDAA Audit Question

Question Number: DWO-SC-26

Receipt Date: 09/27/2024

Question:

Supporting Analyses
When responding to the RAI on the SG tube plugging criterion (RAI-10189, Question 5.4.1.6.1-1), NuScale should describe how the DWO effects are considered in determining the form and rate of degradation, including how DWO effects were considered as part of the SG tube plugging criterion.

Response:

Engineering evaluation EE-172738, Revision 0, “Steam Generator Tube Plugging Criterion Evaluation,” {{

}}2(a),(c), ECI

Engineering evaluation EE-172738, Revision 0, is in the electronic reading room (eRR) for Standard Design Approval Application (SDAA) Chapter 5. {{
}}2(a),(c), ECI supports the 40 percent steam generator tube plugging criterion in the SDAA Technical Specifications 5.5.4.

Engineering calculation EC-155351, Revision 1, “Steam Generator Tube Sliding and Wear Evaluation for the Low Power Density Wave Oscillation Transient,” {{

}}2(a),(c), ECI Engineering calculation EC-155351, Revision 1, is in the Chapter 3 eRR. The tube plugging criterion evaluation in EE-172738, Revision 0, is consistent with the DWO sliding and wear calculation in EC-155351, Revision 1, {{
}}2(a),(c), ECI

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}}2(a),(c), ECI

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}}2(a),(c), ECI

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}}2(a),(c), ECI

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}}2(a),(c), ECI

Other details supporting the NPM-20, 40 percent tube plugging criterion are in Request For Additional Information (RAI) 5.4.1.6.1-1.

Markups of the changes associated with RAI 5.4.1.6.1-1 are provided below.

Audit Question A-3.5.1.3-2, Audit Question A-3.7.3-3, Audit Question A-3.11.2.3-1, Audit Question A-5.2.3.4.2-1, Audit Question A-6.1.1-2, Audit Question A-6.1.1-8, Audit Question A-6.2.5-1, Audit Question A-8.1-4, Audit Question DWO-SC-26, Audit Question EDAS Deep Dive Action Item 1, Audit Question EDAS Deep Dive Action Item 3, Audit Question EDAS Deep Dive Action Item 4, Audit Question EDAS Deep Dive Action Item 5, Audit Question EDAS Deep Dive Action Item 6, Audit Question EDAS Deep Dive Action Item 9, Audit Question EDAS Deep Dive Action Item 11, Audit Question EDAS Deep Dive Action Item 14
RAI 5.4.1.6.1-1, RAI 19.2-1, RAI 19.2-3, RAI 19.2-4

Table 1.9-2: Conformance with Regulatory Guides

RG	Title	Rev.	Conformance Status	Comments	Section
1.6	Safety Guide 6 - Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	0	Not Applicable	The onsite electrical AC power systems do not contain Class 1E distribution systems.	Not Applicable
1.7	Control of Combustible Gas Concentrations in Containment	3	Partially Conforms	The design complies with the intent of RG 1.7 regulatory positions that address atmosphere mixing, hydrogen gas production, and containment structural integrity. However, the design deviates from the positions on hydrogen and oxygen monitors. The design includes a passive autocatalytic recombiner (PAR) that is sized to limit oxygen concentrations to a level that does not support combustion (i.e., less than four percent), ~~this results in~~maintaining an inert containment atmosphere. The design and quality standards applied to the PAR are commensurate with its safety-related, non-risk-significant function in the NuScale design, rather than the non-safety-related, risk-significant function underlying regulatory position C.1. The NuScale design does not include combustible gas monitoring~~supports an exemption to 10 CFR 50.44(c)(4)~~.	6.2.5
1.8	Qualification and Training of Personnel for Nuclear Power Plants	4	Not Applicable	This guidance governs site-specific programmatic and operational activities that are the responsibility of the applicant or licensee.	Not Applicable
1.9	Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants	4	Not Applicable	The NuScale design does not require or include safety-related emergency diesel generators.	Not Applicable
1.11	Instrument Lines Penetrating the Primary Reactor Containment	1	Not Applicable	No instrument lines penetrate the NuScale Power Module (NPM) containment.	Not Applicable

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Title	Rev.	Conformance Status	Comments	Section
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	1	Not Applicable	This guidance governs analysis of the aquatic dispersion of radioactive liquid effluents from component failures, in accordance with BTP 11-6. Because the NuScale facility provides an approved design mitigative feature (metal-lined concrete dike around the pool surge control subsystem storage tank), such an analysis is not required.	Not Applicable
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit	3	Partially Conforms	This guidance is applicable except for site-specific guidance that is the responsibility of the applicant or licensee. Consistent with the discussion in RG 1.114, Section B.1, the ability of the applicant to meet this guidance is facilitated by the control room design and layout (including the designated surveillance area described in Position C.1.3). Portions of this guidance that implement operator staffing requirements of 10 CFR 50.54(m)(2)(i) and (iii) (e.g., Position C.1.5) are not applicable to applicants.	18.5
1.115	Protection Against Turbine Missiles	2	Conforms	None.	3.5
1.117	Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants	2	Conforms	Confirmation of site-specific design-basis automobile missile parameters and site proximity missiles~~that nearby structures exposed to extreme wind loads will not adversely affect the RXBor the Seismic Category I portion of the Control Building~~ is the responsibility of the applicant or licensee. This guidance is not applicable to the CRB, however, the Seismic Category I portions of the CRB conform to relevant guidance of RG 1.117.	3.5 9.1.2
1.118	Periodic Testing of Electric Power and Protection Systems	3	Partially Conforms	This guidance is applicable except for site-specific guidance that is the responsibility of the applicant or licensee.	7.2 14.2
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	0	Partially Conforms	~~None.~~This guidance assumes primary coolant pressure inside the steam generator tubes, while the NuScale design has primary coolant pressure outside the tubes. The design complies with the intent of the guidance but uses an alternate loading condition.	5.4

current test meets Electric Power Research Institute (EPRI) 1013706 (Reference 5.4-2).

Preservice examinations performed in accordance with the ASME BPVC, Section III, Subsubarticle NB-5280 and Section XI, Subarticle IWB-2200 (Reference 5.4-5) use examination methods of ASME BPVC Section V, except as modified by Section III, Paragraph NB-5111. These preservice examinations include essentially 100 percent of the pressure boundary welds.

Audit Question A-5.4.1.4-1, Audit Question A-5.4.1.6.1-1, Audit Question DWO-SC-26
RAI 5.4.1.6.1-1

A preservice volumetric, full-length preservice inspection of ~~essentially~~ 100 percent of the tubing in each SG is performed. The length of the tube extends from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet welds are not part of the tube. The preservice inspection is performed after tube installation and shop or field primary-side hydrostatic testing and before initial power operation to provide a definitive baseline record, against which future ISI can be compared. Technical Specifications Section 5, Administrative Controls, defines the tube plugging criterion as the maximum allowable flaw in the tube wall. Tubes with flaws that are equal to or exceed ~~40 percent of the nominal tube wall thickness~~the tube plugging criterion are plugged. Tubes with flaws that could potentially compromise tube integrity before the performance of the first ISI, and tubes with indications that could affect future inspectability of the tube, are also plugged. The volumetric technique used for the preservice examination is capable of detecting the types of preservice flaws that may be present in the tubes and permits comparisons to the results of the ISI expected to be performed to satisfy the SG tube inspection requirements in accordance with the plant technical specifications.

Audit Question A-3.9.2-26-F, Audit Question A-3.9.2-28, Audit Question DWO-SC-23, Audit Question DWO-SC-24, Audit Question DWO-SC-36, Audit Question DWO-SC-37

As discussed above, the operational inservice testing and inspection programs described in Section 5.2.4, RCPB ISI and Testing, and the SG program described in Section 5.4.1.6, Steam Generator Program, provide testing and inspection requirements following initial plant startup. The SG inlet flow restrictors are examined by VT-3 in accordance with IWA-2213 when removed for SG tube examinations. Inservice inspection and testing of the SGS steam and feedwater piping is described in Section 6.6.

5.4.1.5 Steam Generator Materials

Selection and fabrication of pressure boundary materials used in the SGs and associated components are in accordance with the requirements of ASME BPVC Section III and Section II as described in Section 5.2.3, RCPB Materials, and the materials used in the fabrication of the SGs are in Table 5.2-3.

The RCPB materials used in the SGS are Quality Group A and their design, fabrication, construction, tests, and inspections conform to Class 1 in accordance with the ASME BPVC and the applicable conditions promulgated in 10 CFR 50.55a(b). The SGS materials forming the RCPB, including weld materials, conform to fabrication, construction, and testing requirements of ASME degradation allowance (additional tube wall thickness above minimum required for design) as discussed in Section 5.4.1.2, System Design. The NPM reactor coolant flowrates are also lower than the flowrates across the SG tubes in PWR recirculating SGs as discussed in Section 5.1, RCS and Connecting Systems. This low flow rate reduces the flow energy available to cause FIV wear degradation of SG tubes. Based on the additional tube wall margin and the additional margin against FIV turbulent buffeting wear (the most likely SG tube degradation mechanism), application of the existing PWR SG Program requirements to the design is appropriate.

For SGs in the PWR fleet with SB-163 UNS N06690 SG tubing, the only observed degradation has been wear as a result of FIV (tube-to-tube or tube-to-support plate) or wear due to foreign objects. With respect to the risk of introduction of foreign objects, the NPM is at no greater risk than existing designs; therefore, the design does not warrant deviations from existing SG program guidelines. From the standpoint of SG tube design, the two significant differences between the SG design and current large PWR designs is the helical shape of the SG tubing and the SG tube support structure. The helical shape of the SG tubing itself does not represent risk of degradation based on the minimum bend radius of the helical tubing being within the historical experience base of PWR SG designs. Prototypic testing of the SG tube supports validates acceptable performance (including wear) of the SG tube support design. Implementation of a typical SG program is appropriate based on evaluation of the design of the SG tube supports.

5.4.1.6.1 Degradation Assessment

Audit Question A-5.4.1.6.1-1, Audit Question DWO-SC-26
RAI 5.4.1.6.1-1

A degradation assessment of the NPM SG identifies several potential degradation mechanisms. Wear is the most likely degradation mechanism, and there is the potential for several secondary side corrosion mechanisms, including under deposit pitting and intergranular attack based on the once-through design with secondary boiling occurring inside the tubes. ~~The estimated growth rates for these potential defects is sufficiently low that the SG tube plugging criterion for the SG is a 40 percent through wall defect.~~ Operational SG tube integrity is ensured by implementing tube plugging criteria, implementing elements of the SG program, and implementing the SG inspections.

Audit Question A-3.9.2-26-F, Audit Question A-3.9.2-28, Audit Question DWO-SC-23, Audit Question DWO-SC-24, Audit Question DWO-SC-36, Audit Question DWO-SC-37

A 100 percent SG tube inspection is completed during the first refueling outage following initial startup or SG replacement. After the first refueling outage, a 100 percent SG inspection is completed on a staggered basis over the next 72 effective full power months in order to evaluate ongoing SG tube degradation.

COL Item 5.4-1: An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Steam Generator Program for periodic monitoring of the degradation of steam generator components to ensure that

5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, and cool down and all anticipated transients included in the design specification) and design basis accidents. Structural integrity is defined as no tube failure through gross structural deformation in burst, collapse, or buckling. This includes retaining a safety factor of greater than 2.0~~3.0~~ against ~~burst~~collapse or buckling under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against ~~burst~~failure applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to ~~burst or collapse~~failure. In the assessment of tube integrity, those loads that do significantly affect ~~burst or collapse~~failure shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube failure, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gallons per day.

5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.5, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding [40%] of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following initial startup or SG replacement.

2. After the first refueling outage following SG installation, inspect 100% of the tubes in each SG at least every 72~~96~~ effective full power months, which defines the inspection period.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected unit SG for the degradation mechanism that caused the crack indication shall be at the next refueling outage. If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.