

## NuScaleDCRaisPEm Resource

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**From:** Cranston, Gregory  
**Sent:** Friday, June 15, 2018 1:53 PM  
**To:** Request for Additional Information  
**Cc:** Lee, Samuel; Mitchell, Matthew; Makar, Gregory; Tesfaye, Getachew; Chowdhury, Prosanta; NuScaleDCRaisPEm Resource  
**Subject:** Request for Additional Information No. 490 eRAI No. 9556 (16)  
**Attachments:** Request for Additional Information No. 490 (eRAI No. 9556).pdf

Attached please find NRC staff's request for additional information (RAI) concerning review of the NuScale Design Certification Application. Password will be sent separately.

Please submit your technically correct and complete response within 60 days of the date of this RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

**Hearing Identifier:** NuScale\_SMR\_DC\_RAI\_Public  
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**From:** Cranston, Gregory

**Created By:** Gregory.Cranston@nrc.gov

**Recipients:**

"Lee, Samuel" <Samuel.Lee@nrc.gov>  
Tracking Status: None  
"Mitchell, Matthew" <Matthew.Mitchell@nrc.gov>  
Tracking Status: None  
"Makar, Gregory" <Gregory.Makar@nrc.gov>  
Tracking Status: None  
"Tsfaye, Getachew" <Getachew.Tsfaye@nrc.gov>  
Tracking Status: None  
"Chowdhury, Prosanta" <Prosanta.Chowdhury@nrc.gov>  
Tracking Status: None  
"NuScaleDCRaisPEM Resource" <NuScaleDCRaisPEM.Resource@nrc.gov>  
Tracking Status: None  
"Request for Additional Information" <RAI@nuscalepower.com>  
Tracking Status: None

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## Request for Additional Information No. 490 (eRAI No. 9556)

Issue Date: 06/15/2018

Application Title: NuScale Standard Design Certification - 52-048

Operating Company: NuScale Power, LLC

Docket No. 52-048

Review Section: 16 - Technical Specifications

Application Section: 3.4.5, "RCS Operational LEAKAGE;" 3.4.9, "Steam Generator (SG) Tube Integrity;" and 5.5.4, "Steam Generator (SG) Program"

### QUESTIONS

16-45

With respect to tube integrity, plant technical specifications (TS) meet the requirements of § 50.36 of Title 10 of the *Code of Federal Regulations* (10 CFR), in part, by having an operational leakage limit and an accident-induced leakage (AIL) limit. The NuScale Generic Technical Specifications (GTS) include a statement in Subsection 5.5.4.b.2 that the primary-to-secondary AIL rate for any design basis accident (other than a tube rupture) shall not exceed the leakage rate assumed in the accident analysis.

In Question 16-38 of Request for Additional Information (RAI) (Accession No. ML17353A373 in the U.S. Nuclear Regulatory Commissions (NRC) Agencywide Documents Access and Management System (ADAMS)), the NRC staff requested additional information about the AIL performance criterion in the NuScale GTS Subsection 5.5.4.b.2. Specifically, the NRC staff asked why the AIL in the GTS (150 gallons per day through the steam generators (SGs)) was not greater than the operational leakage limit in order to account for a potential increase in operational leakage due to accident conditions. The response (ADAMS Accession No. ML18032A391) indicated that an AIL performance criterion greater than the operational leakage criterion is not required for the NuScale design based on structural integrity performance evaluations.

In a clarification phone call on April 4, 2018 (ADAMS Accession No. ML18109A537), NuScale indicated that accident analyses were performed at an assumed leakage value of up to 300 gallons per day through the SGs. In order to understand how the proposed GTS meet the requirements of 10 CFR 50.36 with respect to AIL, the NRC staff requests a description of the accident analysis that assumed 300 gallons per day, and how that analysis compares to the accident analysis described in Section 15.0.3 in Tier 2 of the Final Safety Analysis Report (FSAR) that assumed a maximum leak rate of 150 gallons per day (other than a SG tube failure).

16-46

In Question 16-39 of RAI 9234, the NRC staff requested that Technical Report (TR)-1116-52011-NP, Rev. 0, "Technical Specifications Regulatory Conformance and Development" (ADAMS Accession No. ML17005A136), be revised to clarify how the NuScale GTS incorporated TSTF-510, Rev. 2, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection," and explain any departures from TSTF-510. The NRC staff also identified specific departures from TSTF-510 in the NuScale GTS, and requested that the applicant revise the GTS and Bases to be consistent with TSTF-510 or explain the departures. In order to understand incorporation of TSTF-510 and the proposed changes to the GTS and Bases to be consistent with TSTF-510, the NRC requests the following:

a. In the response to Question 16-39 of RAI 9234, NuScale stated, "TSTF-510 was considered during preparation of the NuScale GTS, however, because of the substantial design differences, any list of exceptions is not appropriate for comparison." While design differences may be an acceptable justification for a departure from the current standard technical specifications (STS) and TSTF-510, the NRC staff needs an adequate technical justification to determine whether the GTS incorporate TSTF-510 as intended, consistent with the NuScale design, and conclude that GTS Subsections 3.4.5, 3.4.9, 5.5.4, and 5.6.5 together satisfy 10 CFR 50.36(c), Subsections (2), (3), and (5), and that Subsections B 3.4.5 and B 3.4.9 satisfy 10 CFR 50.36(a) and are therefore acceptable. Therefore, the NRC staff requests the following:

1. A justification for NuScale's position that "any list of exceptions is not appropriate for comparison."
2. A discussion of NuScale's plans to revise TR-1116-52011-NP, Rev. 0, to address how TSTF-510 was incorporated into the NuScale GTS and to include a technical justification for each departure from TSTF-510.

b. In the response to RAI 9234, Question 16-39.i., NuScale stated that the omitted terminology ("affected and potentially affected") in GTS Subsection 5.5.4, paragraph d.3 has been incorporated. However, the NRC staff notes that in the provided markup of GTS Subsection 5.5.4, paragraph d.3, NuScale proposed to incorporate "affected or potentially affected." Please provide a technical justification for using "or" instead of "and" or alternatively revise GTS Subsection 5.5.4, paragraph d.3 to say "affected and potentially affected."

16-47

In Question 16-40 of RAI 9234, the NRC requested justification for the use of the longest SG tube inspection intervals despite not having operating experience. In the response to Question 16-40 of RAI 9234, NuScale stated that the inspection intervals proposed in the GTS are appropriate because the SG tubes are thermally-treated Alloy 690, use of industry primary and secondary chemistry controls, and verification of acceptable flow induced vibration design of the SG tube supports. The NRC staff does not clearly understand NuScale's justification for the longest SG tube inspection intervals for the reasons described below.

The inspection requirements in the STS are based on the well-established behavior of the predominant tube materials in SGs with longstanding designs. The modes of degradation and examination techniques have been established, along with the ability to detect and manage service degradation and flaws from other sources (such as manufacturing).

New examination techniques may have to be developed for NuScale steam generators for preservice and inservice inspection of the tubes. Since there is uncertainty about what tubing conditions need to be detected and characterized after operation, there may be a higher degree of uncertainty in the inspection results for the initial NuScale steam generators, at least until the tube behavior is understood and the detection and characterization capabilities of the examination techniques are determined. This could result in tube flaws being missed, or mischaracterized, during the initial inspection, with no subsequent inspection planned for that nuclear power module for 72 effective full-power months. The consequences on tube integrity can be difficult to predict due to uncertainty in the probability of detection and because

the growth rate of flaws (e.g., wear) in NuScale steam generators may be different than existing designs. Therefore, it is unclear to the NRC staff that the GTS require enough inspection to account for the lack of inspection experience and operating experience.

Describe how the GTS (unlike the STS for traditional light water reactors with extensive operational experience) are sufficient to ensure tube integrity can be maintained without inspection, degradation, and flaw growth experience. Alternatively, revise GTS Subsection 5.5.4, paragraph d.2 to require more inspection in the initial operating cycles of the initial NuScale nuclear power modules.

16-48

As stated in Section 5.4.2.2 of the design-specific review standard (DSRS), General Design Criteria 32 applies to the SG tubing to ensure this part of the reactor coolant pressure boundary is designed to permit periodic inspection to assess structural and leak tight integrity. In addition, DSRS Section 5.4.2.2 explains that the STS provides for the establishment of the SG program, and that 10 CFR 50.36(c), Subsections (3) and (5) apply to the SG program in the TS. For these reasons the NRC staff is reviewing NuScale GTS Subsection 3.4.9, "Steam Generator (SG) Tube Integrity," and the associated Bases in Subsection B 3.4.9.

In Subsection B 3.4.9, the Applicable Safety Analyses section identifies 10 CFR Part 100 as a source of the limits for dose consequences, consistent with the STS. However, since 10 CFR Part 100 does not contain the dose criteria within the text of the regulation for reactor site applications after 1997, this reference is not applicable. Instead, 10 CFR 100.21, "Non-seismic siting criteria," refers to 10 CFR 50.34(a)(1) as the location of the criteria related to the allowed radiological dose consequences of postulated accidents. The NRC staff requests that NuScale revise this statement in the Bases to replace the citation to 10 CFR Part 100 with the applicable regulation for the allowed dose criteria, which is 10 CFR 50.34(a)(1).

16-49

NuScale's response to Question 16-41 of RAI 9234 did not adequately address some of the specific departures from TSTF-510 identified by the NRC staff. The NRC staff needs an adequate justification for each departure to determine whether the GTS incorporates TSTF-510 as intended, consistent with the NuScale design, and conclude that GTS Subsections 3.4.5, 3.4.9, 5.5.4, and 5.6.5 together satisfy 10 CFR 50.36(c), Subsections (2), (3), and (5), and that Subsections B 3.4.5 and B 3.4.9 satisfy 10 CFR 50.36(a) and are therefore acceptable. Therefore, the applicant is requested to correct the following deficiencies in its response to Question 16-41:

a. NuScale stated that the omitted second paragraph of the Limiting Conditions for Operation (LCO) section of Subsection B 3.4.9 was inserted. The NRC staff notes that the inserted paragraph in the provided markup of this LCO section, includes the term "repair criteria," which does not apply to NuScale. Therefore, revise the inserted second paragraph of the LCO section of Subsection B 3.4.9 by replacing "repair criteria" with "plugging criteria."

b. The NRC staff notes that, in the LCO section of Subsection B 3.4.9, the reference to the "RCS Operational LEAKAGE" LCO was correctly changed from 3.4.8 to 3.4.5. However, the reference to the "RCS Operational LEAKAGE" LCO was not corrected in the markup provided with NuScale's reponse to Question 16-38 of RAI 9234.

c. NuScale stated that the term "repair criteria" was changed to "plugging criteria" in the first paragraph of the Bases for Required Actions A.1 and A.2 in the Actions section of Subsection B 3.4.9. However, the provided markup of this first paragraph does not show the term "repair criteria" modified to "plugging criteria." Therefore, revise the markup of the first paragraph by replacing "repair criteria" with "plugging criteria."

d. NuScale stated that the phrase "tube repair criteria" was replaced with "tube plugging criteria" in the third paragraph of the discussion of Surveillance Requirement (SR) 3.4.9.1 and both paragraphs of SR 3.4.9.2 in the SRs section of Subsection B 3.4.9. However, the NRC staff notes that the response provided no markup of the pages to show these changes. In order to evaluate how the stated changes were incorporated, please provide a markup of the pages showing the changes to the Bases for SR 3.4.9.1 and SR 3.4.9.2.

e. NuScale stated that the phrase "tube repair criteria" was replaced with "tube plugging criteria" in response to Question 16-41.B.10 of RAI 9234. However, the NRC staff notes that Question 16-41.B.10 was related to an omitted closing sentence about crack indications from the fourth paragraph of the Bases discussion of SR 3.4.9.1 in the SRs section of Subsection B 3.4.9. Therefore, please provide a technical justification for omitting this closing sentence or alternatively revise the fourth paragraph of the Bases discussion of SR 3.4.9.1 to add the closing sentence to be consistent with TSTF-510. Please see Question 2.b of this RAI regarding use of the phrase "affected and potentially affected" as it applies to the omitted sentence.