## NuScaleDCDocsPEm Resource

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**Attachments:** NuScale 5.4.1 Issue List.docx; NuScale 5.4.1.2 Issue List.docx

06/28/2017 NOTICE OF PUBLIC TELECONFERENCE TO DISCUSS NUSCALES DESIGN CERTIFICATION APPLICATION (DCA) CHAPTER 5, SECTION 5.4.1., STEAM GENERATORS

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## MCB Issue List Regarding NuScale FSAR Section 5.4.1 (Steam Generators)

### Issue MCB-5.4.1- #1

The NuScale Final Safety Analysis Report (FSAR) Tier 2, Section 5.4.1.2, "System Design," describes the attachment of the steam generator tube ends to the tubesheets using an expansion fit and a weld at the tube end.
For compliance with Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 1, GDC 30, and 10 CFR 50.55a, the reactor coolant pressure boundary must be designed to the highest quality standards practical, including the design requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code). These requirements apply to the part of the reactor coolant pressure boundary formed by the tube-to-tubesheet joint for each steam generator tube.
In order to clarify the requirements for the tube-to-tubesheet joint in the NuScale design, please provide the following:

a. Describe the method for expanding the tubes into the tubesheet, and whether the expansion fit is designed to be part of the pressure boundary.
b. Describe the requirements for the tube-end welds and whether they are designed as pressure boundary (structural) welds in accordance with NB-3000 of the ASME Code.
c. If the expansion fit is designed to be part of the pressure boundary, describe the applicable design requirements and how the design complies with 10 CFR 50.55a (given that the ASME Code does not have design requirements that apply specifically to an expansion joint).
d. Provide your plans for identifying in the FSAR how the tube ends form the pressure boundary, and the associated design requirements.
e. Describe the provisions for preventing and detecting degradation associated with the crevice and expansion transition at the primary face of the tubesheet.

### Issue MCB-5.4.1- #2

The design of the steam generators must meet the requirements of 10 CFR Part 50, Appendix A, GDC 14 and GDC 15 as they relate to accessibility to the steam generator tubes. Accessibility for inspection can be affected by corrosion products and other contaminants on the tube internal or external surfaces. FSAR Tier 2 Section 5.4.1.2 states that unacceptable buildup of corrosion products will be prevented through materials selection and periodic cleaning. To enable the staff to understand how this was accomplished for the NuScale design, please provide the following:

a. Describe the evaluations performed to characterize the corrosion and deposition expected on the steam generator tube internal surfaces and how corrosion and deposition affect the ability to inspect the tubes.
b. Describe the periodic cleaning that will be performed on the secondary surfaces and how it has been shown to enable effective tube inspections.
c. Describe to what extent the design permits access to the steam generator tube external surfaces, and how that has been demonstrated.
d. Describe the status of qualifying tooling and procedures for foreign object search and retrieval.

**Issue MCB-5.4.1- #3**

FSAR Tier 2 Section 5.4.1.2, page 5.4-2 states that there is not a bulk reservoir of water at the inlet plena where deposits could accumulate and impurities could concentrate. However, in FSAR Tier 2, Figure 5.4-5, the feedwater plenum design appears to have space below the tube inlet elevation. Describe the analyses or testing that have been performed to evaluate the potential for sludge accumulation in the plena and possible adverse effects (such as blockage of flow through the tubes or concentration of corrosion products). This information is needed for the staff to evaluate how the design of the steam generator meets 10 CFR Part 50, Appendix A, GDC 4, 14, 15, and 31, as they relate to maintaining the integrity of the reactor coolant pressure boundary through design and compatibility of materials with the environmental conditions.

**Issue MCB-5.4.1- #4**

Steam generator tube supports must be designed to meet the requirements of 10 CFR Part 50, Appendix A, GDC 4, 14, 15, and 31 as they relate to maintaining the integrity of the reactor coolant pressure boundary. Given that this is a first-of-a-kind design, please provide the following information to address how the support structure meets these requirements:

a. Describe the testing and analyses performed to evaluate the design of the tube support structures with respect to providing adequate support, limiting accumulation of corrosive or particulate materials, and resisting degradation.

b. FSAR Tier 2 Table 5.2-4 identifies Type 304/304L stainless steel as the support material for the Alloy 690 steam generator tubes. Given that there is a significant difference in the coefficients of thermal expansion for Alloy 690 and Type 304 stainless steel, describe how this thermal expansion difference has been considered in the evaluation of steam generator performance and tube integrity, and what effect it is expected to have. (According to product data sheets from alloy producers, the average coefficient of thermal expansion between room temperature and 600°F is approximately $8.1 \times 10^{-6}$ in/in/°F for Alloy 690, and $9.6 \times 10^{-6}$ in/in/°F for Type 304 stainless steel.)

c. Clarify the design requirements for the tube support structures and how they are addressed in the Tier 1 and Tier 2 material in the FSAR. Tier 2 Section 5.4.1.5 states that the tube supports conform to the requirements of Subsection NG of the ASME Code. This appears to contradict Tier 2 Table 3.2-1, which identifies the supports as Quality Group A, corresponding to ASME Code Class 1. In addition, it is not clear to the staff whether or not ITAAC #2 in Tier 1 Table 2.1-4 regarding ASME Code Class 1 and 2 components applies to the SG tube support structures (Class NG).

**Issue MCB-5.4.1- #5**

The flow restrictors in the steam generator tube ends must enable the full range of design flow rates and resist degradation that could lead to damage of the steam generator tubes and other downstream components in the steam and feedwater system. This is necessary for the flow restrictors to satisfy the requirements of 10 CFR Part 50, Appendix A, GDC 1, 4, 15, 30, and 31, as they relate to ensuring the integrity of the reactor coolant pressure boundary. Please provide the following information about the flow restrictors:

a. Identify the design, fabrication, and inspection requirements for the flow restrictors. FSAR Tier 2 Table 3.2-1 indicates that while the flow restrictors are safety-related, Quality Group and Safety Classifications are not applicable. This appears to be inconsistent with the statement on FSAR Tier 2 Page 5.4-11 that the flow restrictors are Quality Group B, but designed, fabricated, constructed, tested, and inspected as non-ASME Code components.

b. Describe where the flow restrictor mounting plate is attached to the inlet plenum.
c. FSAR Tier 2 Table 5.4-3 includes components called “flow restrictor bolts,” “flow restrictor stud bolts,” and “flow restrictor mounting plate spacer.” Since these items are not identified in DCD Tier 2 Figure 5.4-8, please clarify their location and functions.
d. Discuss whether the NuScale flow restrictor is a first-of-a kind design feature or a design with operating experience. If there is operating experience, describe that experience and how it relates to the NuScale design.
e. Describe how the flow restrictor design was qualified for use in the NuScale reactor.
f. Describe the evaluations performed for potential degradation related to the flow restrictors and fasteners, and the provisions for managing that degradation.

**Issue MCB-5.4.1- #6**

A part of meeting the requirements of 10 CFR Part 50, Appendix A, GDC 14, 15, and 31, is designing the reactor coolant pressure boundary with an appropriate allowance for deterioration of the steam generator materials. This can be accomplished through compliance with ASME Code Section III (NB-2160 and NB-3121), which requires an appropriate allowance for corrosion and other forms of degradation.

FSAR Tier 2 Section 5.4.1.2 states that the steam generator tubes are designed with a corrosion allowance of 0.01 inch to account for degradation on the internal surface. Please describe the information used to specify this allowance and conclude that it is adequate for the potential degradation mechanisms.

**Issue MCB-5.4.1- #7**

The staff requests clarification of the steam generator tube inspection requirements in FSAR Tier 2 Section 5.4.1.4, “Tests and Inspections.” The second paragraph of Section 5.4.1.4 states that inspection requirements are “provided” in NEI 97-06 and include the examination requirements from the ASME Code. The paragraph ends with a statement that the SG program follows the guidance of NEI 97-06 and is described in the technical specifications. The language is unclear with respect to the program requirements. For example, it would be clearer for the FSAR to state at the beginning of the paragraph that the program is based on the guidance in NEI 97-06 and Section 5.5.4 of the technical specifications. Following the guidance in NEI 97-06 and adopting the “Steam Generator Program” from the latest standard technical specifications satisfies the requirements of GDC 32 and 10 CFR 50.55a as they relate to SG tube inspection. If SG tube inspection is based on NEI 97-06 and the technical specifications, the less specific ASME Code requirements are also met.

In addition, the fifth and final paragraph of FSAR Tier 2 Section 5.4.1.4 appears to be a summary of the previous descriptions of the inspection of the steam generators (5.2.4) and steam generator tubes (5.4.1.6). If true, please clarify this in the last paragraph. For example, add the phrase “In summary,” at the beginning of the paragraph. (Note that this paragraph also refers to the SG program “described in Section 5.4.2.2.” This appears to be an error, since “Steam Generator Program” is Section 5.4.1.6.)

**Issue MCB-5.4.1- #8**

The description of the thermal relief valves for the secondary side of the steam generators (pages 5.2-6 and 5.4-5) is not adequate for the staff to evaluate them. The FSAR describes the purpose of the valves but does not identify the valve numbers or drawings that show the valves in relation to other piping and components.  This information is needed for the staff to be able to consider the effect of the valves on the secondary side of the steam generator.

**<u>Issue MCB-5.4.1- #9</u>**

For both the steam and feed plenums, please provide drawings that show the plenum, access ports, and port covers, identifying each component and the material.  FSAR Tier 2, Table 5.4-3 lists materials for the "integral steam and feed plenum access ports," but the distinction between the plenums and access ports is not clear to the staff, particularly for the feed plenums.  This information is necessary for the staff to determine if the design satisfies the requirements of 10 CFR 50, Appendix A, GDC 1, 4, 15, 30, and 31, and 10 CFR 50.55a, as they relate to the materials and design for the steam generator section of the reactor vessel.

**<u>Issues MCB-5.4.1-#10</u>**

The staff has identified additional editorial comments which appear to need to be corrected.

(**Editorial**) The last paragraph of FSAR Tier 2, Section 5.4.1.4, refers to the "SG Program described in Section 5.4.2.2."  The program is described in Section 5.4.1.6.

(**Editorial**) FSAR Tier 1, Section 2.1.1, pages 2.1-1 to 2.1-3, has two lists of "non-safety-related, risk-significant functions" performed by the Nuclear Power Module and verified by Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC).  Based on the functions listed, such as "maintaining the pressure boundary of the RPV," the first list appears to address safety-related functions.  Please revise the section as needed to correctly identify the safety-related and non-safety-related functions.

(**Editorial**) In addition, the third item on the first list of "non-safety-related" functions has a typographical error (highlighted), "The SG supports the RCS by suppling part of the RCPB."

# MCB Issue List Regarding NuScale FSAR Section 5.4.1.2

## Issue MCB-5.4.1.2- #1

Inservice inspection (ISI) of the reactor coolant pressure boundary is required under Title 10 of the Code of Federal Regulations (10 CFR) Part 50.55a, "Codes and standards." For steam generator (SG) tubing, the ISI program is described in the Steam Generator Program in the Technical Specifications (TSs) and applies to the full length of each tube. Some of the requirements for SG tubing inspection are defined on a SG basis. For example, TS 5.5.4.d.2 states, "the number of times the SG is scheduled to be inspected in the inspection period ..."

The first paragraph of NuScale Final Safety Analysis Report (FSAR) Tier 2, Section 5.4, states that each NuScale Power Module contains two SGs. With respect to meeting the requirements for inspecting all SG tubes, please clarify in FSAR Section 5.4 whether implementation of the Steam Generator Program will treat each unit as having two SGs or one SG.

## Issue MCB-5.4.1.2- #2

Maintaining steam generator tube integrity in accordance with the TSs requires the ability to remove each tube from service. Please provide the following information about how this requirement is met for the NuScale SG design:

- Describe in the FSAR whether plugging and stabilization of the NuScale SG tubes will be accomplished with existing methods or with design-specific methods. In the FSAR description, identify the source of the preservice and inservice inspection requirements for tube plugs.
- Describe how plugging and stabilization methods have been qualified for SG tubes in the NuScale design.

## Issue MCB-5.4.1.2- #3

Provide additional description of the SG tube preservice inspection (PSI) requirements in FSAR Tier 2 Section 5.4.1.4 to identify the scope, method, capabilities, timing, and acceptance criteria.

FSAR Section 5.4.1.4 states that the PSI of the steam generator tubing is performed using eddy current examinations according to the EPRI Steam Generator Management Program guidelines and American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI. The additional description is necessary because the PSI is necessary to meet the requirements of 10 CFR 50.55a, but the PSI requirements for SG tubing are not clear in the ASME Code or in the TSs. In addition, the PSI description should not specify an examination technique, such as eddy current testing. The standard TSs do not specify the examination technique to avoid unnecessary restriction. An adequate description would be a statement such as the following:

> A volumetric, full-length preservice inspection of 100% of the tubing in each steam generator shall be 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 shall be performed after tube installation and shop or field primary side hydrostatic testing, and prior to initial power operation to provide a definitive

baseline record against which future inservice inspections can be compared. Any tubes with flaws that exceed [40%] of the nominal tube wall thickness shall be plugged. Any tubes with flaws that could potentially compromise tube integrity prior to the performance of the first inservice inspection and any tubes with indications that could affect future inspectability of the tube shall also be plugged. The volumetric technique used for the PSI shall be capable of detecting the types of preservice flaws that may be present in the tubes and shall permit comparisons to the results of the inservice inspections expected to be performed to satisfy the steam generator tube inspection requirements in the plant Technical Specifications.

Issue MCB-5.4.1.2- #4

Explain the statement in FSAR Tier 2 Section 5.4.1.2 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. This appears to be an inconsistency because there is no discussion of the 40 percent plugging criterion in FSAR Tier 2 Section 5.4.

Issue MCB-5.4.1.2- #5

Please revise the first paragraph of FSAR Section 5.4.1.3.1 to clarify the following items:

- What is meant, quantitatively, by “minimum wall thickness SG tubes?”
- Explain the “worst-case loading conditions.”
- What is the “most critical” location in the SG, and how was that determined?

Issue MCB-5.4.1.2- #6

10 CFR Part 50.55a(g) requires that SG tubing, part of the reactor coolant pressure boundary, must meet the applicable inspection requirements of ASME Code Section XI. SG ISI and PSI programs follow the industry guidelines established in NEI 97-06, “Steam Generator Program Guidelines,” which is referenced in FSAR Tier 2 Section 5.4 and the TS Bases for the Steam Generator Program (TS 5.5.4). Please address the following about the Steam Generator Program requirements:

- The end of FSAR Tier 2 Section 5.4.1.1 identifies ASME Code Sections III and XI, respectively, for PSI and ISI. There is also a mention of an “SG program” but not the TSs. In FSAR Tier 2 Section 5.4.1.1, please clarify that the SG Program is in the TSs and is the program for implementing ASME Code Section XI. Similarly, the PSI is defined in the FSAR and conforms to NEI 97-06, and this should be clarified in Tier 2 Section 5.4.1.1.

- The third paragraph of FSAR Section 5.4.1.6 (on page 5.4-12) includes a statement that successful degradation mitigation strategies applied to the NuScale design “do not provide a basis for NuScale to deviate from the established NEI 97-06 SG Program requirements that have led to high levels of SG reliability and integrity in the operating commercial fleet.” This appears to be an affirmation that the SG Program will follow NEI 97-06. Since this was already stated in the second paragraph of Section 5.4.1.6, please clarify or delete the statement in the third paragraph.

Issue MCB-5.4.1.2- #7

The end of the fifth paragraph of FSAR Section 5.4.1.6 (bottom of page 5.4-12) states that "prototypic testing of the SG supports is performed to validate acceptable performance." The use of word "is" makes the timing of this testing unclear. This statement is not clear about whether the tests were already performed. If the testing has been completed, revise the FSAR to clarify this and provide a reference for the "evaluation of the design of the SG tube supports" mentioned in the subsequent sentence.

Issue MCB-5.4.1.2- #8

Combined License (COL) Item 5.4-1, on page 5.4-13 of FSAR Tier 2, describes implementation of the Steam Generator Program and states that the program is "based on" NEI 97-06 and applicable EPRI guidelines. For consistency with other statements in the FSAR, the COL item should be revised to make it clear that the SG Program follows or conforms to NEI 97-06. The term "based on" implies that there are exceptions, in which case they should be identified and justified.

Issue MCB-5.4.1.2- #9

With respect to SG tube integrity, the Standard Technical Specifications (STSs) meet the requirements of 10 CFR Part 50.36, in part, by having an operational leakage limit and accident-induced leakage limit.

According to NuScale Limiting Condition for Operation (LCO) 3.4.5.d and TS 5.5.4.b.2, both the operational and accident-induced leakage limits are 150 gallons per day (other than a steam generator tube failure). As stated in the NuScale TS Bases (Page B 3.4.9-3), "the accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident." Describe how conditions during an accident (other than a steam generator tube failure) remain within the bounds of the accident analyses (150 gpd) if operational leakage is 150 gpd and there appears to be no allowance for leakage induced during the accident. If appropriate, revise the TS and Bases to identify an accident-induced leakage limit higher than the operational leakage limit.

The staff also notes the following apparent typographical errors related to this question:

- TS Bases Page B 3.4.9-3, last paragraph, states, "The accident analysis assumes that accident induced leakage does not exceed the limit specified in equal to the LCO 3.4.8, "RCS Specific Activity." (underline added)
- FSAR Tier 2 Section 15.0.3.8.3 on page 15.0-30 identifies the leakage limit as 150 gallons per minute. (underline added)

Issue MCB-5.4.1.2- #10

According to Section 4.2 of Technical Report TR-1116-52011-NP, Rev. 0, the Technical Specification Task Force (TSTF) travelers and revisions available to NuScale and issued before November 1, 2016, were considered during preparation of the NuScale Generic Technical Specifications (GTS). Please clarify in TR-1116-52011-NP how the NuScale GTS incorporate TSTF-510, which became available on October 27, 2011, and explain any exceptions. Revision

4 of the STSs was also completed in October 2011 and does not incorporate the language from TSTF-510. The NuScale GTS appear to generally adopt TSTF-510; however, the staff notes the following exceptions in the TSs. Please provide the justification for these exceptions, or revise the TS and Bases for consistency with TSTF-510.

- NuScale TS 3.4.9 (Steam Generator (SG) Tube Integrity) uses the phrase, “tube repair criteria.” In TSTF-510, the corresponding wording was changed to “tube plugging [or repair] criteria.” The brackets indicate that “repair criteria” applies only to plants approved to perform tube repairs (i.e., sleeve installation). For all other plants, conditions requiring tubes to be plugged was changed by TSTF-510 to the “plugging criteria.” This difference also applies to the associated TS Bases.
- The comment above about the use of “repair criteria” rather than “plugging criteria” also applies to TS 5.5.4.
- The first paragraph of NuScale TS 5.5.4 includes the word, “provisions,” which was changed to “indications” in TSTF-510.
- The first sentence of NuScale TS 5.5.4.b.1, the structural integrity performance criterion, does not match the wording and punctuation in the STS as modified by TSTF-510. While differences may be necessary due to the non-standard MODE definitions in the NuScale design, the sentence is unclear due to the location of parentheses and lack of commas.
- In TS 5.5.4.d.2, justify the use of the longest inspection intervals despite not having operating experience. Why was it considered unnecessary to have more frequent inspection of NuScale nuclear power modules (NPMs), or at least the initial NPMs in service, in order to obtain operating experience for early identification of degradation?
- NuScale TS 5.5.4.d.3 uses the phrase, “whichever is less” (referring to 24 effective full power months or one refueling outage). This phrase was changed in TSTF-510 for clarification to “whichever results in more frequent inspections.”

<u>Issue MCB-5.4.1.2- #11</u>

The staff notes that the TS Bases are significantly different than the STS Bases. The staff proposes discussing these differences before requesting any written responses.

| **Bases Section B 3.4.5, “RCS Operational Leakage”** |
|---|
| BACKGROUND:<br>• 4th paragraph, “into the containment area” changed to “outside of the reactor coolant pressure boundary”<br>• 4th paragraph, “Quickly separating the identified LEAKAGE ….” changed to “When possible, separating the identified LEAKAGE ….”<br>• Missing paragraph, “A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight.<br>• 5th paragraph, “protection of the reactor coolant pressure boundary (RCPB) from degradation ~~and the core from inadequate cooling~~,” (omission by NuScale)<br>APPLICABLE SAFETY ANALYSES<br>• 1st paragraph, “other operational LEAKAGE” changed to “other forms of RCS Operational LEAKAGE”<br>• The discussion in this section implies that the operational leakage limit is the same as the accident-induced leakage limit (150 gpd). STS compares 150 gpd |

to the accident-induced leakage limit. Paragraphs related to AIL are changed or missing.

LCO

- Paragraph a, "Pressure Boundary Leakage," adds, "defined as LEAKAGE (except primary ……defined in 10CFR50.2" and other information not in the STS
- Paragraph b, "Unidentified LEAKAGE," Value of 0.5 gpm less than STS. Add "Containment Evacuation System."
- Paragraph c, "Identified LEAKAGE," identified leakage reduced from 10 gpm (STS) to 2 gpm. Omits statement about RCP seal leakoff
- Paragraph d, "Primary to Secondary LEAKAGE," adds statement about not being able to determine which one of the two steam generators is leaking.

APPLICABILITY

- This section longer and significantly different than STS based on NuScale design

ACTIONS

- Action A.1, "or reduce RCS Operational LEAKAGE to within limits…" (underlined part added to STS)
- Actions B.1, B.2, Second half of paragraph different than STS, including 48 hours for MODE 3 rather than 36 for MODE 5.

SURVEILLANCE REQUIREMENTS

- SR 3.4.5.1 discussion in paragraphs 2-5 significantly different than STS
- SR 3.4.5.2 first two paragraphs have NuScale-specific differences

**Bases Section B 3.4.9, "SG Tube Integrity"**

BACKGROUND:

- End of first paragraph missing sentence from STS about loops and modes for SG heat removal function. No design-specific replacement.
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APPLICABLE SAFETY ANALYSES

- 2nd paragraph, "assumed not to failure." (Typo?)
- 2nd paragraph, different than STS – no AIL value identified

LCO

- Missing 2nd paragraph about plugging tubes during inspections
- 5th paragraph uses "tube failure" instead of "tube burst"
- 8th paragraph different than STS because no AIL value provided. Points to AIL not exceeding the limit in LCO 3.4.8, "RCS Specific Activity."
- 8th paragraph, "does not exceed the limit specified in equal to the LCO" (typo?)
- 9th paragraph, refers to LCO 3.4.8, "RCS Operational LEAKAGE." The section number doesn't match the title. The intent is probably 3.4.5.

APPLICABILITY

- 1st paragraph, ends with, "MODE 1, 2, or 3 and not PASSIVELY COOLED," instead of "MODE 1, 2, 3, or 4."
- 2nd paragraph, same idea using the NuScale MODE definitions

ACTIONS

- Actions A.1 and A.2, 1st paragraph, uses "repair" twice when it should be "plugging" (TSTF-510)
- Actions A.1 and A.2, 4th paragraph, MODE 3 instead of MODE 4
- Actions B.1 and B.2, 1st paragraph, uses the different MODE definitions as well as the term "PASSIVELY COOLED"

SURVEILLANCE REQUIREMENTS

- SR 3.4.9.1, 3rd paragraph uses "repair" where it should use "plugging"

- SR 3.4.9.1, 4th paragraph, omits the closing sentence from TSTF-510 about crack indications.
- SR 3.4.9.2, both paragraphs use “repair” where they should use “plugging”
- SR 3.4.9.2, 2nd paragraph uses MODE 3 instead of MODE 4

Issue MCB-5.4.1.2- #12 (Editorial)

In the fourth paragraph of FSAR Section 5.4.1.6 (on page 5.4-12), there is a statement that Table 5.4-3 shows the tube wall degradation allowance. This appears to be an error, since Table 5.4-3 identifies materials specifications. If this statement was intended to reference Table 5.4-2, the staff notes that Table 5.4-2 provides the tube thickness, but it does not provide the amount of degradation allowance as the text implies. Please correct and clarify this statement about the degradation allowance.

Issue MCB-5.4.1.2- #13 (Editorial)

There is a quotation mark at the end of Technical Specification 5.5.4.e that should be deleted.