RAIO-0718-61158

August 02, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 490 (eRAI No. 9556) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 490 (eRAI No. 9556)," dated June 15, 2018

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 Questions from NRC eRAI No. 9556:

- 16-45
- 16-46
- 16-47
- 16-48
- 16-49

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,

Jennie Wike
Manager, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9556

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NuScale Response to NRC Request for Additional Information eRAI No. 9556

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

eRAI No.: 9556
Date of RAI Issue: 06/15/2018

NRC Question No.: 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).

NuScale Response:

As described in the response to RAI question 16-38, no AIL performance criterion in addition to the operational leakage criterion is required for the NuScale SG design based on structural integrity performance evaluations.

NuScale FSAR 15.0.3.8.2 describes the assumptions used in the radiological analysis of a postulated steam generator tube failure (SGTF). As described there, the scenario assumes leaks into the secondary side of the intact SG at the maximum leak rate of 150 gallons per day (gpd). This limit is consistent with the total combined primary to secondary leak rate proposed in NuScale GTS 3.4.5, "RCS Operational LEAKAGE." As described in the associated Bases, the NuScale design does not support the ability to determine which one of the two SGs has the primary to secondary leakage. Therefore total primary to secondary leakage is conservatively attributed to one SG.

The Main Steam Line Break (MSLB) accident analyses assume 300 gallons per day primary-to-secondary leakage and are described in FSAR 15.1.5, "Steam Piping Failures Inside and Outside of Containment." The MSLB evaluation assumes two intact SG lines and conservatively assumes a cumulative 300 gpd primary-to-secondary leakage as the combined leakage from both SGs and steam lines. This limit is conservative with respect to the GTS limit in LCO 3.4.5.

The SGTF accident analysis assumes 150 gallons per day primary-to-secondary leakage and is described in FSAR 15.6.3, "Steam Generator Tube Failure (Thermal Hydraulic)." The SGTF evaluation assumes only one intact SG eligible for leakage and thus assumes only 150 gpd leakage. This limit is equal to or more conservative than the GTS limit in LCO 3.4.5.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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

eRAI No.: 9556
Date of RAI Issue: 06/15/2018

NRC Question No.: 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 <u>and</u> 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 <u>or</u> 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."

NuScale Response:

a. TR-1116-52011-NP was submitted to describe the development process of the NuScale Power Plant TS. A revision is planned for submission in the future, however it is not maintained as a 'living' document. The proposed TS and the associated Bases, in combination with the balance of the DCA and RAI responses and other docketed interactions with the staff describe the basis for review of the TS as submitted. As noted in the TSTF traveler description of the changes between Revision 2 and Revision 1, the change to the program description consisted of changing "periods" to "intervals". The NuScale proposed TS include this term, consistent with Revision 2 of TSTF traveler 510.

1. The NuScale TS were developed considering NUREG-1431, Revision 4 Section 5.5.9, as the basis for the proposed TS. The proposed NuScale SG program described in TS 5.5.4 with changes implemented to reflect the NuScale design. The adequacy of the design, operation, and maintenance of the SG are addressed in the FSAR.

   NuScale is not a member of the TSTF industry group, and therefore did not participate in the developmental history of traveler 510, or Revisions 1, or 2. NuScale is not aware of the NuScale or similar SG designs being considered in the development of TSTF-510.

   The only information available to NuScale is that which has been released to the public such as via the publicly available NRC docketed files. The staff has historically participated in the development of TSTF travelers, including informal interactions such as meeting discussions that may not be fully reflected in the publicly available record. The undocumented assumptions and interactions are often key to understanding the intent or appropriate interpretation of traveler content. An example of this is the use of "and" rather than "or" as noted in item b. below. Common usage would indicate that the "or" is appropriate and the "and" would require both conditions to apply - however this does not appear to be the interpretation taken by the staff. Based on this, NuScale does not believe it appropriate to perform an evaluation of traveler incorporation by specific exception comparison. Rather NUREG-1431, Revision 4 and the best available TSTF-510 information were considered to develop a program consistent with the NuScale design and operation.

   A markup identifying the differences between the NuScale SG Program in 5.5.4 and the content of TSTF-510, Revision 2 available to NuScale is provided in the attached file.

2. As noted above, the technical report was a description of the development of the NuScale TS. The basis for the TS and associated Bases is provided in the balance of the DCA,

primarily Part 2, and other docketed interactions with the staff. NuScale understands that TSTF-510, Revision 2 is one method that the staff has found acceptable to ensure the SG tube integrity is maintained. The NuScale TS were developed considering NUREG-1431, Revision 4 Section 5.5.9, with changes implemented to reflect the NuScale design. The adequacy of the design, operation, and maintenance of the SG are addressed in the FSAR.

b. NuScale has replaced "or" with "and" in the phrase "affected or (now and) potentially affected."

Impact on DCA:

The Technical Specifications have been been revised as described in the response above and as shown in the markup provided in this response.

5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected ~~or~~and potentially affected unit SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). 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.

5.5.5 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation and low pressure turbine disc stress corrosion cracking. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

NuScale SG Program Markup Comparison to TSTF 510 Revision 2

Summary of changes identified in the body of the text below:

- Grammatical changes ('or' relocated appropriately, etc.)
- Design specific usage consistent with balance of TS (in-service versus inservice.)
- MODE definitions to reflect NuScale design and TS.
- Modified reference to appropriate RCS Operational LEAKAGE LCO.
- Design-specific failure mode ('rupture' versus 'failure') terminology used.
- Design-specific leakage rate and locations used.
- Removal of reviewer's notes that are not applicable to design.
- Removal of discussion of SG tube repairs throughout.
- Clarifications because program description applies to new design and not an existing plant or design as the TSTF was directed towards.

5.5.4 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

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 ~~[or repair]~~ of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected~~,~~ or plugged~~, [or repaired]~~ 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 ~~in-service~~inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, ~~hot standby,~~ and 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 and a safety factor of 1.4 against burst 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. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor ~~of 1~~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 ~~rupture~~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 ~~[1 gpm] per SG [, except for specific types of degradation at specific locations as described in paragraph c of the Steam Generator Program].~~150 gallons per day.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.~~13~~5, "RCS Operational LEAKAGE~~."~~."

c. Provisions for SG tube plugging ~~[or repair]~~ 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 ~~[or repaired].~~.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~

~~Alternate tube plugging [or repair] criteria currently permitted by plant technical specifications are listed here. The description of these alternate tube plugging [or repair] criteria should be equivalent to the descriptions in current technical specifications and should also include any allowed accident induced leakage rates for specific types of degradation at specific locations associated with tube plugging [or repair] criteria.~~

~~---~~

~~[The following alternate tube plugging [or repair] criteria may be applied as an alternative to the 40% depth based criteria:~~

~~1. . . .]~~

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-tube sheet weld at the tube outlet, and that may satisfy the applicable tube plugging ~~[or repair]~~ 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.

~~-----------------------------------REVIEWER'S NOTE-----------------------------------~~

~~Plants are to include the appropriate Frequency (e.g., select the appropriate Item 2.) for their SG design. The first Item 2 is applicable to SGs with Alloy 600 mill annealed tubing. The second Item 2 is applicable to SGs with Alloy 600 thermally treated tubing. The third Item 2 is applicable to SGs with Alloy 690 thermally treated tubing.~~

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1. Inspect 100% of the tubes in each SG during the first refueling outage following ~~SG installation.~~

~~[2. After the first refueling outage following SG installation, inspect each steam generator at least every 24 effective full power months or at least every refueling outage (whichever results in more frequent inspections). In addition, inspect 100% of the tubes at sequential periods of 60 effective full power months beginning after the first refueling outage inspection following SG installation. Each 60 effective full power month inspection period may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period~~initial startup and SG replacement.

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube ~~the subsequent inspection period begins at the conclusion of the included SG inspection outage. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair~~plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. ~~The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period.~~

~~[2. After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated.~~ The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period

defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

~~----------------------------- Reviewer's Note ----------------------------------~~

~~A licensee may elect to retain historical and existing inspection period lengths in order to not revise those inspection periods.~~

~~--~~

~~a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;~~

~~b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and~~

~~c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.~~

~~[2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.~~

~~----------------------------- Reviewer's Note ----------------------------------~~

~~A licensee may elect to retain historical and existing inspection period lengths in order to not revise those inspection periods.~~

~~--~~

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

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 not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). 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.

~~[f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.~~

~~----------------------------------REVIEWER'S NOTE----------------------------------~~

~~Tube repair methods currently permitted by plant technical specifications are to be listed here. The description of these tube repair methods should be equivalent to the descriptions in current technical specifications. If there are no approved tube repair methods, this section should not be used.~~

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~~1. . . .]~~

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

eRAI No.: 9556
Date of RAI Issue: 06/15/2018

NRC Question No.: 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.

NuScale Response:

The NuScale steam generator (SG) tube inspection intervals are based on assuring “an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture” as described in GDC 14 of Appendix A to 10 CFR 50.

Based on a combination of design attributes and consideration of the primary mechanisms that contribute to SG tube degradation, NuScale considers the industry-standard SG inspection frequencies adequate and appropriate to satisfy the criteria of GDC 14. While the SG design has some differences from traditional light water reactor designs, industry experience has been appropriately applied to support a conclusion that inspection frequencies will be appropriately established and managed.

SG tube degradation occurs because of two primary mechanisms; corrosion and wear. The SG designs in the existing PWR fleet were intended to address and eliminate SG tube corrosion problems on the secondary side on the outside of the SG tubes where boiling occurs, with wear being a lesser historic concern at the design stage.

Although the primary coolant side of the SG will not be subjected to the challenges of concentrated chemistry issues where steam is formed, the NuScale design considered industry experience in the support plate and tube interface design. The design also considered the relatively low primary system flow velocities compared to traditional plant designs.

The industry-prevalent broached hourglass designs of PWR tube support plates was originally based on limiting the tube-to-support crevice area and maintaining flow near this interface. This was to prevent concentrated chemistry resulting from boiling outside of the tubes, which contributed to rapid tube wastage. However, this design results in a more limited interface surface between the tube and the tube support. Fretting wear effects are incurred in a more localized area with this design geometry and can lead to more rapid penetration of tube walls, as compared to the NuScale design. The NuScale SG tube support design provides more generous contact area between tubes and support plates to reduce fretting wear effects. The NuScale design also eliminates the potential for concentrated chemistry to occur due to boiling on the tube and support plate side of the SG. A detailed description of these and other design details is provided in FSAR section 5.4.1.2. As noted in Section 5.4.1.2, the design of the SG tube supports and the tubesheets ensures that only minimal corrosion products will be present on the outside of the SG tubes.

The NuScale SGs have no secondary-side crevices or low-flow regions that could concentrate corrosion products or impurities accumulating during the steam generation process. Secondary coolant impurities and corrosion products may deposit directly on the interior tube surfaces as a scale or film, or be removed from the SG by carryover. The concentration of corrosion products and impurities is low based on selection of materials for the condensate system and chemistry control requirements. Any unacceptable buildup of corrosion product films on the secondary surfaces of the SG tubes will be removed through periodic cleaning performed during outage

periods. The cleaning methods and techniques will be based on proven chemical or mechanical methods already employed in traditional pressurized water reactors.

The NuScale SG design incorporates industry-proven, thermally-treated Alloy 690 (A690) tubing. As noted by the staff in “The Review of Lessons Learned from the San Onofre Steam Generator Tube Degradation Event,” observation of 46 commercial nuclear SGs in the US over nearly 30 years indicates that “to date there have been <u>no reported instances of corrosion related degradation</u> of Alloy 690 thermally-treated tubing” [emphasis added].

NuScale has committed to operational practices consistent with industry experience to manage secondary water chemistry to inhibit SG tube degradation as described in technical specification 5.5.5, “Secondary Water Chemistry Program.”

Based on this and the industry data, corrosion of A690 SG tubing to the point of failure during any reasonable operating period (20-30 years) is an extremely low probability event. Although the NuScale design has operational differences (e.g., boiling inside the tubes), extrapolation of the existing data assures that the likelihood of severe and undetectable corrosion is extremely unlikely.

The NuScale SG design has been developed with full knowledge of the contemporary design problems related to SG lifetime management, which is overwhelmingly SG tube wear. Wear damage to A690 tubes has been observed and managed in commercial nuclear SGs. The need for design validation was demonstrated by industry experience at San Onofre Nuclear Generating Station Unit 3, rapid through-wall tube wear was observed, resulting in wear greatly in excess of tube plugging criteria before the first refueling outage.

NuScale believes that design validation is the appropriate means to prevent wear damage, especially during the first inspection interval. NuScale developed a SG flow induced vibration (FIV) testing program as part of the Comprehensive Vibration Assessment Program to address this concern. The testing program utilizes prototypic tube support designs and prototypic tube geometries. It includes testing over an extensive range of flowrates, adequate to justify conclusion that sufficient margins will exist to FIV wear degradation at NuScale design flow rates. It will validate vibration analysis results and therefore provide a test basis to conclude the NuScale SG tubes have an extremely low probability of leakage due to FIV tube wear during the first inspection interval.

As noted in the RAI, and as exists across the traditional reactor designs, new examination techniques may be developed for pre-service and inservice inspection of the tubes. The industry has generically addressed the concern with new or different detection capabilities in the EPRI SG Examination Guidelines. The detection characterization capabilities for existing or new examination techniques are required to be determined before the initial pre-service inspection as described in Appendix H, “Performance Demonstration for Current Eddy Examination” of the EPRI SG Examination Guidelines. A different technique could result in greater uncertainty in the inspection results, however, Appendix I of the EPRI SG Examination Guidelines include

quantification of NDE measurement uncertainties. NuScale implementation of the NEI 97-06 guidelines is described in FSAR section 5.4, and addressed in COL Item 5.4-1.

The capability and quantified uncertainties are included in the SG Integrity Assessments used to apply the inspection results to the tube plugging criteria and to projection of degradation rates used to determine inspection intervals. Based on this any limits on characterization capability or uncertainty in examination technique will be addressed in integrity assessments.

The inspection intervals for A690 tubing are variable and are confirmed and adjusted as necessary prior to each outage. Following the initial refueling outage 100% inspection, defect growth rates will be evaluated and updated as necessary, then used to determine the next inspection interval. The SG tube inspection system for the NuScale design is not yet developed or qualified, and therefore no specific detection threshold has been established, and uncertainties for characteristic defects are unknown.

All subsequent inspection intervals are contingent on the results of the initial inspection, including quantitative consideration of the detection and sizing uncertainties associated with the qualified inspection system.

With a NuScale plant design of 12 modules, each of which will require 100% inspection at the first refueling, 24 SGs are likely to have been subjected to pre-service and inservice inspection before the first individual module could operate for 72 months following the first 100% inspection. The individual modules will be operated in a similar manner consistent with similar controls. It is reasonable to assume that all modules will exhibit comparable corrosion and wear behavior based on similarity in design, fabrication and commonality of operational practices. The SG Program requires that if any unexpected results were observed for one SG, then the inspection interval be evaluated and if appropriate adjusted based on the observed conditions.

The 72-month limit is a limiting maximum duration for the second inspection interval that will only be achieved if numerous other criteria are satisfied. Even if no degradation indications are observed in the initial inspection, it may not be possible to realize a 72 month inspection interval based on the characteristics of the as of yet undetermined inspection method detection threshold and uncertainty.

The RAI requested consideration of potential for missed or mischaracterized indications. As discussed, the available data (as incorporated into the existing degradation assessment for the NuScale SGs) indicate wear defects are the most probable degradation mechanism. The characteristics of wear defects due to tube-to-support wear or tube-to-tube contact are a result of the physical contact interface, which is based on fully known design geometry. As such, characteristics of wear defects for the NuScale design do not represent an unknown. They can be determined with extremely low uncertainty, and fully assessed during the inspection technique qualification process. Therefore, the detectability threshold (including uncertainties) of characteristic wear defects for the NuScale SG can be understood and quantified high certainty,

without direct operational experience. It is effectively impossible to qualify an inspection system which cannot detect the characteristic wear defects which are used to evaluate the system.

The maximum limiting inspection intervals remain appropriate based on the description above, the information provided in the FSAR, and the use of industry standards and practices as applicable. This information was used in developing the NuScale SG design, operating plans, inspection program, and other relevant programmatic controls. Based on these factors, including validation of the SG design, more limiting inspection intervals are not necessary to assure an extremely low probability of SG tube failure.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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

eRAI No.: 9556
Date of RAI Issue: 06/15/2018

NRC Question No.: 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).

NuScale Response:

The Technical Specification Bases have been modified to refer to 10 CFR 50.34 as described in the RAI. Additional conforming changes were made in the Bases references to 3.1.1, "Shutdown Margin (SDM)," and 3.3.1, "Module Protection System (MPS)."

Impact on DCA:

The Technical Specifications have been been revised as described in the response above and as shown in the markup provided in this response.

BASES

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR Chapter 15, "Transient and Accident Analyses."
3. ~~10 CFR 100.~~

BASES

BACKGROUND (continued)

affected. The channel as-found condition will be entered into the Corrective Action Program for further evaluation and to determine the required maintenance to return the channel to OPERABLE.

During AOOs, which are those events expected to occur one or more times during the plant life, the acceptable limits are:

- The critical heat flux ratio (CHFR) shall be maintained above the SL value to prevent critical heat flux (CHF);
- Fuel centerline melting shall not occur; and
- Pressurizer pressure SL of 2285 psia shall not be exceeded.

Maintaining the variables within the above values ensures that the offsite dose will be within the 10 CFR 50 (Ref. 2) and 10 CFR 50.34~~100~~ (Ref. 3) criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the plant life. The acceptable limit during accidents is that the offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 (Ref. 3) limits. Different accident categories allow a different fraction of these limits based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

The MPS includes devices and circuitry that generate the following signals when monitored variables reach levels that are indicative of conditions requiring protective action:

1. Reactor Trip System (RTS) actuation;
2. Emergency Core Cooling System (ECCS) actuation;
3. Decay Heat Removal System (DHRS) actuation;
4. Containment Isolation System (CIS) actuation;
5. Chemical and Volume Control System Isolation (CVCSI) actuation;
6. Demineralized Water Supply Isolation (DWSI) actuation;
7. Pressurizer Heater Trip (PHT) actuation; and
8. Low Temperature Overpressure Protection (LTOP) actuation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

When an interlock or permissive is not supporting the associated Function's OPERABILITY at the existing plant conditions, the affected Function's channels must be declared inoperable and appropriate ACTIONS taken.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

SR 3.3.1.5

Class 1E isolation devices ensure that electrical power to the associated MPS circuitry and logic will not adversely affect the ability of the system to perform its safety functions. The devices de-energize and isolate the MPS components if such a condition is detected. This surveillance verifies the setpoints and functions of the isolation devices including associated alarms and indications.

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

REFERENCES

1. Regulatory Guide 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation.”
2. 10 CFR 50, Appendix A, GDC 21.
3. 10 CFR 50.34~~100~~.
4. FSAR, Chapter 7, "Instrumentation and Controls."
5. FSAR, Chapter 14, "Initial Test Program and ITAAC."
6. 10 CFR 50.49.
7. TR-0606-49121, Rev. 0, “NuScale Instrument Setpoint Methodology.”
8. IEEE Standard 603-1991.
9. FSAR, Chapter 15, "Transient and Accident Analyses."

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube failure (SGTF) accident is the limiting design basis event for SG tubes and avoiding an SGTF is the basis for this Specification. The analysis of a SGTF event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.5, “RCS Operational LEAKAGE,” plus the leakage rate associated with a double-ended failure of a single tube. The accident analysis for a SGTF assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than a SGTF assume the SG tubes retain their structural integrity (i.e., they are assumed not to fail.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs or is assumed to increase as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, “RCS Specific Activity,” limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 50.34~~100~~ (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.4, “Steam Generator Program,” and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

BASES

REFERENCES

1. NEI 97-06, Rev. [3], “Steam Generator Program Guidelines.”

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.34~~100~~.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.

6. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines,” Rev. [4].

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

eRAI No.: 9556
Date of RAI Issue: 06/15/2018

NRC Question No.: 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.

NuScale Response:

Items a., c., and d.

The term 'repair' has been replaced with 'plugging' in each location in the Bases for LCO 3.4.9.

Item b.

The reference to the "RCS Operational LEAKAGE" LCO was corrected however it did not appear in the markup provided in response to 16-38, eRAI 9234 because of the timing of the changes being prepared and reviewed for submission. The corrected reference is shown in the Applicable Safety Analyses section provided in the attached pages.

Item e.

The sentence has been added to the Bases discussion of SR 3.4.9.1.

Impact on DCA:

The Technical Specifications have been been revised as described in the response above and as shown in the markup provided in this response.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry secondary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.5.2, "Decay Heat Removal System (DHRS)."

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 5.5.4, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.4, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.4. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube failure (SGTF) accident is the limiting design basis event for SG tubes and avoiding an SGTF is the basis for this Specification. The analysis of a SGTF event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.5, “RCS Operational LEAKAGE,” plus the leakage rate associated with a double-ended failure of a single tube. The accident analysis for a SGTF assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser.

The analysis for design basis accidents and transients other than a SGTF assume the SG tubes retain their structural integrity (i.e., they are assumed not to fail.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs or is assumed to increase as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, “RCS Specific Activity,” limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 50.34~~100~~ (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the ~~repair~~plugging criteria be plugged in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~plugging criteria is removed from service by plugging. If a tube was determined to satisfy the ~~repair~~plugging criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.4, “Steam Generator Program,” and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

BASES

LCO (continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube failure or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube failure is defined as, “The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation.” Tube collapse is defined as, “For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero.” The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term “significant” is defined as “An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting failure/collapse condition to be established.” For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity and the accident induced leakage performance criteria ensures that calculated stress intensity in a SG tube not exceed ASME Code, Section III (Ref. 4) limits for Design and all Service Level A, B, C and D Conditions included in the design specification. SG tube Service Level D represents limiting accident loading conditions. Additionally, NEI 97-06 Tube Structural Integrity Performance Criterion establishes safety factors for tubes with characteristic defects (axial and longitudinal cracks and wear defects), including normal operating pressure differential and accident pressure differential, in addition to other associated accident loads consistent with guidance in Draft Regulatory Guide 1.121 (Ref. 5). Therefore in addition to meeting the structural integrity criteria, no additional accident induced primary-to-secondary LEAKAGE is assumed to occur as the result of a postulated design basis accident other than a SGTF.

BASES

LCO (continued)

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during unit operation. The limit on operational LEAKAGE is contained in LCO 3.4.5, “RCS Operational LEAKAGE,” and limits primary to secondary LEAKAGE to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTF under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, or 3 and not PASSIVELY COOLED.

RCS conditions are far less challenging in MODE 3 and PASSIVELY COOLED, MODES 4 and 5 than during MODES 1, 2, and 3 and not PASSIVELY COOLED. In MODE 3 and PASSIVELY COOLED, MODES 4 and 5, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube ~~repair~~plugging criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.9.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG ~~repair~~plugging criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG

BASES

ACTIONS (continued)

performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of unit operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows unit operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 3 following the next unit refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 2 within 6 hours and MODE 3 and PASSIVELY COOLED within 36 hours.

The allowed Completion Times are reasonable, based on operating requirements, to reach the desired unit conditions from full power conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the

BASES

SURVEILLANCE REQUIREMENTS (continued)

“as found” condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube ~~repair~~plugging criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.9.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.4 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

If crack indications are found in any SG tube, the maximum inspection interval for all affected and potentially affected unit SGs is restricted by Specification 5.5.4 until subsequent inspections support extending the inspection interval.

SR 3.4.9.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program ~~repair~~plugging criteria is removed from service by plugging. The tube ~~repair~~plugging criteria delineated in Specification 5.5.4 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube ~~repair~~plugging criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of prior to entering MODE 3 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the ~~repair~~plugging criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, Rev. [3], “Steam Generator Program Guidelines.”
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 50.34~~100~~.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.
6. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines,” Rev. [4].