NuScale Power

RAIO-0118-58473

January 31, 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. 306 (eRAI No. 9234) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 306 (eRAI No. 9234)," dated December 19, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 9234:

- 16-38
- 16-39
- 16-40
- 16-41

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 Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com.

Sincerely,

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

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9234

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9234

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

eRAI No.: 9234
Date of RAI Issue: 12/19/2017

NRC Question No.: 16-38

With respect to steam generator (SG) tube integrity, the Standard Technical Specifications (STS) 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 accident-induced leakage limit.

According to NuScale Limiting Condition for Operation (LCO) 3.4.5.d and Technical Specification (TS) 5.5.4.b.2, both the operational and accident-induced leakage limits are 150 gallons per day (other than a SG 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 SG 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 TSs and Bases to identify an accident- induced leakage limit higher than the operational leakage limit.

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

a. 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 [reactor coolant system] Specific Activity." (underline added)

b. Section 15.0.3.8.3 in Tier 2 of the FSAR, "Main Steam Line Break Outside Containment Accident," identifies the leakage limit as 150 gallons per minute. (underline added)

NuScale Response:

FSAR Section 5.4.1.3.1 describes the evaluation of combined design basis pipe break and safe shutdown earthquake loads. With the SG tube wall thickness uniformly reduced by the design degradation allowance, a safety margin (sufficient wall thickness) remains such that the maximum stress is less than the allowable stress limit as defined by the most limiting, ASME Code Level D Service (faulted conditions). Based on this design, no additional leakage will result from an accident.

In addition, tube structural integrity performance evaluations, which considered accident primary-to-secondary pressure differentials plus other significant (seismic) loads demonstrate tube integrity is maintained under accident conditions for characteristic wear defects and cracks up to the tube plugging limit. These evaluations provide further basis that no additional leakage will result from an accident. This position is consistent with the limit in LCO 3.4.5 and the SG Program in section 5.5.4.b.2, as well as the safety analyses. The Bases of 3.4.9 have been revised to clarify this information.

The typographical errors identified have been corrected as shown on the attached pages.

Impact on DCA:

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

As described in Section 15.6.3, a single failure of the main steam isolation valve (MSIV) for the faulted SG delays isolating the steamline, resulting in a larger release. A loss of normal AC power causes the steamline to isolate earlier, limiting the release. Therefore, a loss of normal AC power is not assumed to occur for the portion of the SGTF analysis used to determine the radiological releases. However, a loss of normal AC power is assumed to occur for the thermal-hydraulic portion of the SGTF analysis used to determine peak pressures presented in Section 15.6.3.

Doses are determined at the EAB, LPZ, and for personnel in the control room and TSC. The control room model is described in Section 15.0.3.7.1. The dose results for the SGTF event are presented in Table 15.0-12.

15.0.3.8.3 Main Steam Line Break Outside Containment Accident

Radiological consequences of the MSLB outside containment accident are calculated based on the guidance provided in Appendix E of RG 1.183. Section 15.1.5 describes the sequence of events and thermal-hydraulic response to a MSLB outside containment.

The radiological dose consequence analysis considers the MSLB event with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. A description of the scenario evaluated is summarized as follows

1) An MSLB occurs in one of the two main steam lines.

2) The iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design basis limits for each of the iodine spiking scenarios. The primary coolant contains a concentration of 0.2 μCi/gm DE I-131 for the coincident Iodine spike scenario and 12 μCi/gm DE I-131 for the pre-incident Iodine spike scenario. For both iodine spiking scenarios, the primary coolant contains 60 μCi/gm DE Xe-133.

RAI 15.00.03-4

3) Primary coolant leaks into the secondary side of the intact SGs at the maximum leak rate of 150 gallons per day~~minute~~ allowed by design basis limits. The leakage continues until the primary system pressure is less than the secondary system pressure.

4) A time-dependent release is modeled that effectively releases the activity directly to the environment through the break.

5) The non-faulted steam line continues to release a small quantity of radiation through valve leakage.

The assumptions used from Appendix E of RG 1.183 are:

- coincident iodine spiking factor- 500
- duration of coincident iodine spike- 8 hr
- density for leak rate conversion- 62.4 lbm/ft^3

BASES

LCO (continued)

~~Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).~~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.

~~The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTF, is within the accident analysis assumptions. 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." 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.~~

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during unit~~plant~~ operation. The limit on operational LEAKAGE is contained in LCO 3.4.8, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG 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.

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

eRAI No.: 9234
Date of RAI Issue: 12/19/2017

NRC Question No.: 16-39

According to Section 4.2 of Technical Report (TR)-1116-52011-NP, Rev. 0, “Technical Specifications Regulatory Conformance and Development” (Accession No. ML17005A136 in the NRC’s Agencywide Documents Access and Management System (ADAMS)), 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, “Revision 2, “Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection,” which became available on October 27, 2011, and explain any exceptions.

Revision 4 of the STS 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 NRC staff notes the following exceptions in the affected GTS subsections. Please provide the justification for these exceptions, or revise the GTS and Bases for consistency with TSTF-510.

a. The NRC staff notes the following differences in GTS Subsection 3.4.5, “RCS Operational LEAKAGE,” when compared to NUREG-1431, Revision 4, “Standard Technical Specifications,” for Westinghouse plants (ADAMS Accession No. ML12100A222), Subsection 3.4.13, “RCS Operational LEAKAGE.”
    1. Condition A in the STS states, “RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary leakage.” Corresponding Condition A in the GTS omits “operational” and “or primary to secondary leakage.”
    2. Condition B in the STS states, “Required Action and associated Completion Time of Condition A not met.” Corresponding Condition B in the GTS omits the phrase “of Condition A,” which the NRC staff believes should be included for clarity, since Condition B has two other condition statements.
    3. The Completion Time for Required Action B.2 to “Be in MODE 5” (RCS average temperature ≤ 200°F) in the STS is “36 hours.” The Completion Time for corresponding Required Action B.2 to “Be in MODE 5 with RCS temperature hot < 200 °F” in the GTS is “48 hours.” The NRC staff notes that this difference is also addressed from the perspective of the associated Bases for the completion time by Sub-question 8 of Question 16-32 of RAI 228-9034 (ADAMS Accession No.

ML17257A227).

4. The Surveillance Requirement (SR) 3.4.13.1 in the STS uses “NOTES” while corresponding SR 3.4.5.1 in the GTS uses “NOTE.” Since there are two surveillance column Notes, SR 3.4.5.1 should use “NOTES.”
5. The Frequencies for SR 3.4.13.1 and SR 3.4.13.2 in the STS are “[72 hours OR In accordance with the Surveillance Frequency Control Program].” The Frequencies for corresponding SR 3.4.5.1 and SR 3.4.5.2 in the GTS omit “72 hours OR” and the associated brackets. The associated GTS Bases for these SRs also omit the basis for the 72 hour Frequency, which is included in the STS Bases for these SRs. The NRC staff notes that this difference is also addressed from the perspective of TSTF-425, Revision 3, “Relocate Surveillance Frequencies to Licensee Control - Risk Informed Technical Specifications Task Force (RITSTF) Initiative 5b,” in Question 16-30 of RAI 228-9034 (ADAMS Accession No. ML17257A227); specifically in Sub-questions d and e.

b. Generic TS Subsections 3.4.9 and B 3.4.9, “Steam Generator (SG) Tube Integrity,” use the phrase, “tube repair criteria.” The NRC staff notes that bracketed information about SG tube repair methods and associated repair criteria, including “reviewer’s notes,” which is included in TSTF-510, for both TSs and Bases, applies to plants with tube repair methods (i.e., sleeve installation) previously approved by the NRC staff. This bracketed information may be included in the GTS and Bases as combined license (COL) information items, provided it is consistent with such information in TSTF-510. This is done to facilitate development of plant-specific TSs by COL applicants who have received NRC staff approval of SG tube repair methods and repair criteria proposed in their combined license applications that reference the NuScale design certification. If NuScale does not include this bracketed information in the GTS and Bases, then phrases such as “tube repair criteria” or “SG repair criteria” (see Bases for Action A) should be changed to “tube plugging criteria” or “SG tube plugging criteria;” and all bracketed information (and reviewer’s notes) related to tube repair criteria and methods, included in TSTF-510, including alternate tube plugging criteria, as potentially applicable to NuScale SGs, should be omitted in the GTS and Bases.
c. The comment above about the use of “repair criteria” also applies to GTS Subsection 5.5.4, “Steam Generator (SG) Program.”
d. The first paragraph of GTS Subsection 5.5.4 includes the word, “provisions,” which was deleted in TSTF-510.
e. Generic TS 5.5.4.a includes an unnecessary comma after the term “plugged” in the last sentence (editorial).
f. The first sentence of GTS 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. The NRC staff believes the intent was for the first sentence to read, “Structural integrity performance criterion: All inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, and cool down and all anticipated transients included in the design specification) and design

basis accidents."

g. Generic TS 5.5.4.d uses the phrase, "An assessment of degradation," while TSTF-510 changed this phrase to, "A degradation assessment" to be consistent with industry program documents.
h. Generic TS 5.5.4.d.1 uses the phrase, "initial startup and SG replacement." The TSTF-510 uses "SG installation" to allow the SG program to apply to existing and new plants.
i. Generic TS 5.5.4.d.3 omits the phrase, "affected and potentially affected" which was added in TSTF- 510 to clarify the term "each SG." Generic TS 5.5.4.d.3 also 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."

NuScale Response:

The incorporation of TSTF-510 is reflected in the NuScale GTS. The NuScale design is different from existing large PWRs as described in the FSAR. Chapter 5 describes the RCS and connecting systems including the integral steam generator (SG) design. 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. Issues identified specifically are addressed individually below.

With Respect to a.1, the word 'operational' and 'or primary to secondary leakage' have been added to the Condition A description.

With respect to a.2, the phrase 'of Condition A' has been added to the description of Condition B.

With respect to a.3, see the response to RAI 9034, question 16-32.

With respect to a.4, an 'S' was appended to the word NOTE.

With respect to a.5, see the response to RAI -9034, question 16-30.

With respect to b. and c., the term 'repair criteria' has been modified to 'plugging criteria' throughout the identified sections.

With respect to d. the word "provisions" has been removed.

With respect to e. the comma has been removed.

With respect to f. the hyphen has been removed and the additional comma inserted.

With respect to g. the phrase 'A degradation assessment' has replaced the previously existing phrase.

With respect to h. the TSTF-510 terminology was revised to reflect the NuScale design and fabrication which does not 'install' a steam generator because it is fabricated integral to the upper module.

With respect to i. the omitted terminology has been incorporated. The word 'unit' was inserted to clarify that the scope of expanded inspections is applicable to the individual NuScale unit and not all 12 units comprising a NuScale plant.

Impact on DCA:

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Operational LEAKAGE

LCO 3.4.5 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 0.5 gpm unidentified LEAKAGE,

c. 2 gpm identified LEAKAGE from the RCS, and

d. 150 gallons per day primary to secondary LEAKAGE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS hot temperature ~~hot~~ ≥ 200 °F.

--NOTE--
This LCO is not applicable if one or more ECCS valves is open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 2. AND B.2 Be in MODE 3 with RCS hot temperature ~~hot~~ < 200 °F.	6 hours 48 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	-------------------------------NOTES---------------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	In accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Steam Generator (SG) Tube Integrity

LCO 3.4.9 SG tube integrity shall be maintained.

<u>AND</u>

All SG tubes satisfying the tube ~~repair~~plugging criteria shall be plugged in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1 and 2,
MODE 3 and not PASSIVELY COOLED.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging~~repair~~ criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	<u>AND</u>	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 3 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 2.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 3 and PASSIVELY COOLED.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.9.2	Verify that each inspected SG tube that satisfies the tube ~~repair~~plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 3 following a SG tube inspection

5.5 Programs and Manuals

5.5.2 Radioactive Effluent Control Program (continued)

7. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary shall be in accordance with the following:

   i. For noble gases: a dose rate ≤ 500 mrem/yr to the whole body and a dose rate ≤ 3000 mrem/yr to the skin and

   ii. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: a dose rate ≤ 1500 mrem/yr to any organ;

8. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit~~MODULE~~ to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days in gaseous effluents released from each unit~~MODULE~~ to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

10. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

b. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.3 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Section 3.9, cyclic and transient occurrences to ensure that components are maintained within the design limits.

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~~ provisions~~:

5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

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

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

1. Structural integrity performance criterion: All in~~-~~service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, and cool down and all anticipated transients included in the design specification) and design basis accidents. 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 1.2 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, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 150 gallons per day.

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

5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

c. Provisions for SG tube plugging~~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.

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

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

5.5 Programs and Manuals

5.5.4 Steam Generator (SG) Program (continued)

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.

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 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~~is less~~). 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;

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

eRAI No.: 9234
Date of RAI Issue: 12/19/2017

NRC Question No.: 16-40

In GTS 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 Response:

The NuScale steam generator (SG) design is described in FSAR section 5.4.1. The SG subcomponents that form portions of the reactor coolant pressure boundary are classified Quality Group A and designed as Class 1 in accordance with Section III of the ASME Boiler and Pressure Vessel Code (BPVC), Section III. The SGs and connected components up to the feedwater isolation valves and main steam isolation valves are Seismic Category I components.

Transient conditions applicable to the SGs are discussed in FSAR section 3.9.1 and design limits, loads, and load combinations are addressed in FSAR section 3.9.3. The NuScale SGs are fabricated from A690 thermally treated tubing, with a 0.050 inch nominal wall thickness. The NuScale design uses industry standard primary and secondary chemistry controls. This combination of materials and chemistry has a demonstrated record of reliable service over the past 25 years and is the basis for the recommended inspection intervals which NuScale has adopted. The NuScale SGs will be subject to a 100% pre-service eddy current inspection, which will verify that tubing is free of unacceptable defects introduced during tube manufacturing or SG assembly. It is also noted that the 0.050 inch wall thickness of the NuScale SG tubes exceeds other designs (such as the AP1000) by approximately 10%.

The inspection intervals described in the SG Program are based on conformance with approved industry standards in design, fabrication, and testing. The only significant difference in the NuScale SG design as compared to the existing commercial fleet which is considered to be strongly correlated to SG tube integrity is the SG tube support design. The NuScale design is inherently different based on the use of helically bent seamless tubing, however the existing history of SG damage related to tube support configurations is well understood and has been considered in the development of the NuScale SG tube support design.

Local fluid velocities in combination with tube support geometry are the factors that contribute to SG tube wear. The local (gap) flow velocities on the tube support side of the NuScale SG tubes are less than 2.0 ft/s, which is approximately 5-10 times lower than is observed in the shell side of existing LWR SGs. Based on single phase SG shell side flow, the prediction of shell side flow velocities within the NuScale SG is significantly simplified as compared to recirculating SGs. This makes it unlikely that large errors could occur in predicting the shell side flow velocities or that unexpected (and unknown) high flow velocities could exist on the shell side of a NuScale SG. It is understood that one contributing factor to the SONGs SG tube failure was errors in prediction of some SG shell side local flow velocities.

NuScale has undertaken an extensive effort to verify acceptable flow induced vibration design of the SG tube supports as discussed in Tier 2, Section 1.5.1.9. This testing includes flow testing of a helical tube bundle assembled with NuScale prototypic SG tube supports and will provide a basis to validate the analysis that currently shows that NuScale SG tube integrity will be maintained for a minimum of 60 years. This testing addresses the only significant unknown with respect to factors that could cause unacceptable SG tube degradation within a short period of time and provide a basis for not performing the initial SG tube inservice inspection prior to the first refueling outage. No other potential degradation (corrosion) has the potential to cause unacceptable tube degradation within a two year period under the chemistry and operating conditions for the NuScale SGs.

It is noted that the unexpected SG tubing wear observed in the SONGS SGs was readily detectable at the two year inspection point. If such unexpected degradation were to occur in the NuScale SGs during the initial operating interval, the SG program (through the degradation assessment process) would mandate implementation of more frequent inspections. Based on the adaption inherent to the industry accepted SG inspection programs, it is not necessary to mandate more restrictive intervals for subsequent inspections a priori.

The intent of the SG program is to monitor SG performance in the future and adjust inspection frequencies if future results require adjustment. As described in 5.5.4.d.1, 100% of tubes in each SG will be inspected during the first refueling outage. Results of that, and subsequent inspections will be used to evaluate subsequent SG inspection intervals.

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.: 9234
Date of RAI Issue: 12/19/2017

NRC Question No.: 16-41

The NRC staff notes that the Bases in GTS Subsection B 3.4.5 are significantly different than the Bases in STS Subsection B 3.4.13. Please provide the justification for the exceptions noted below, or revise the GTS Bases for consistency with the STS Bases.

a. GTS Subsection B 3.4.5, "RCS Operational LEAKAGE"
   1. The fourth paragraph of the Background section changes the STS phrase "into the containment area" to "outside of the reactor coolant pressure boundary." In addition, this paragraph changes the STS phrase "Quickly separating the identified LEAKAGE …." to "When possible, separating the identified LEAKAGE …."
   2. The Background section omits the STS paragraph which says, "A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight."
   3. The fifth paragraph of the Background section omits the STS phrase "and the core from inadequate cooling."
   4. The first paragraph of the Applicable Safety Analyses section changes the STS phrase "other operational LEAKAGE" to "other forms of RCS Operational LEAKAGE."
   5. The discussion in the Applicable Safety Analyses section implies that the operational leakage limit is the same as the accident-induced leakage limit (150 gpd). The STS compares 150 gpd to the accident-induced leakage limit. In addition, STS paragraphs related to accident-induced leakage are changed or missing.
   6. Paragraph "a." in the LCO section, under the heading "Pressure Boundary LEAKAGE," adds the phrase, "defined as LEAKAGE (except primary to secondary LEAKAGE)…defined in 10CFR50.2" and other information not in the STS. Editorial change, "10CFR50.2" should read "10 CFR 50.2."
   7. The unidentified leakage value of 0.5 gpm in Paragraph "b." in the LCO section, under the heading "Unidentified LEAKAGE," is less than the STS value of 1 gpm.
   8. The identified leakage value of 2 gpm in Paragraph "c." in the LCO section, under the heading "Identified LEAKAGE," is less than the STS value of 10 gpm.
   9. In the Actions section, the discussion of Required Action A.1 changes the STS phrase as indicated by markup: "or reduce RCS Operational LEAKAGE to within limits…" (Underlined part added to STS). This is an addition seen throughout the Bases.
   10. In the Actions section, the discussion of Action B differs from the corresponding STS

discussion. In particular, Required Action B.2 allows 48 hours to be in MODE 3 with RCS temperature hot < 200°F rather than the 36 hours allowed by the STS to be in MODE 5 (RCS average temperature ≤ 200°F). The NRC staff notes that this difference is also addressed from the perspective of the associated Bases for the completion time by Sub-question 8 of Question 16-32 of RAI 228-9034 (ADAMS Accession No. ML17257A227).

11. In the SRs section, the discussion of SR 3.4.5.1, the second, third, fourth, and fifth paragraphs are significantly different than the STS.

b. GTS Subsection B 3.4.9, "Steam Generator (SG) Tube Integrity"

1. The end of the first paragraph of the Background section omits the sentence from corresponding STS Subsection B 3.4.20 that lists the LCOs governing the requirements for the SG heat removal function. TR-1116-52011-NP, "Technical Specifications Regulatory Conformance and Development," Revision 0, Table B-1, "Comparison of standard technical specifications with NuScale generic technical specifications," indicates that the GTS do not include equivalent LCOs (STS LCOs 3.4.4, 3.4.5, 3.4.6, and 3.4.7) because they are "not applicable to NuScale design" and because there are "no corresponding credited features." However, it appears that the decay heat removal system, which utilizes the SGs, does address the SG heat removal function. The applicant is requested to explain why the GTS include no LCO addressing the SG heat removal function and why no design-specific replacement sentence is proposed, or revise Subsection B 3.4.9 to include a sentence referencing the LCO that governs the SG heat removal function. The NRC staff notes that conforming changes may also need to be made to TR-1116-52011-NP to describe the heat removal function of the SGs.
2. The second paragraph in the Applicable Safety Analyses section omits an accident-induced leakage value, which is included in the corresponding paragraph in STS Subsection B 3.4.20.
3. The LCO section omits the second paragraph of the LCO section in STS Subsection B 3.4.20, regarding plugging tubes during inspections.
4. The sixth paragraph in the LCO section is not a separate paragraph in the LCO section of STS Subsection B 3.4.20, where it is the last sentence of the sixth paragraph.
5. The eighth paragraph in the LCO section omits an accident-induced leakage value, which is included in the corresponding ninth paragraph of the LCO section of STS Subsection B 3.4.20. The eighth paragraph points to accident-induced leakage not exceeding the limit in LCO 3.4.8, "RCS Specific Activity." In addition, this paragraph uses "SGTF" instead of "SGTR" and there appears to be a typo in the phrase, "does not exceed the limit specified in equal to the LCO."
6. The ninth (and last) paragraph in the LCO section, refers to LCO 3.4.8, "RCS Operational LEAKAGE." This should be changed to LCO 3.4.5. In addition, this paragraph states that LCO 3.4.5 "limits primary to secondary LEAKAGE through any one SG to 150 gallons per day." However, LCO 3.4.5 does not include the phrase, "through any one SG."
7. In the Actions section, the first paragraph in the discussion of Required Actions A.1

and A.2 uses the phrase “tube repair criteria,” which is consistent with the corresponding STS paragraph, but according to TSTF-510, this should be changed to “tube plugging criteria.”

8. In the Actions section, the second paragraph in the discussion of Required Actions B.1 and B.2 states that the “allowed Completion Times are reasonable, based on operating experience,” even though there is no operating experience for the NuScale design. The NRC staff notes that the 36 hour completion time to be in MODE 5 is also addressed from the perspective of the associated Bases for the completion time by Sub-question 8 of Question 16-32 of RAI 228-9034 (ADAMS Accession No. ML17257A227).
9. In the SR section, the third paragraph of the discussion of SR 3.4.9.1 uses the phrase “tube repair criteria,” which is consistent with the corresponding STS paragraph. But according to TSTF-510, this should be changed to “tube plugging criteria.”
10. In the SR section, the fourth paragraph of the discussion of SR 3.4.9.1 omits the closing sentence of the markup of STS from TSTF-510 about crack indications.
11. In the SR section, both paragraphs of the discussion of SR 3.4.9.2 use the phrase “tube repair criteria.” But according to TSTF-510, this should be changed to “tube plugging criteria.”

NuScale Response:

The NuScale design is significantly different from existing PWR designs. Much of the descriptive language used in the STS was inappropriate for the NuScale design and the Bases were modified to reflect the NuScale design, system names, descriptive usage, and facility operations.

For example, the reactor coolant system is contained within a pressure vessel within the containment vessel and the steam generators are located within the RCS pressure vessel. The reactor vessel occupies a significant fraction of the containment vessel volume as shown in FSAR figure 6.2-1. Typical STS descriptions of 'containment' implies large buildings or structures unlike the NuScale design. The majority of the differences between the NuScale Bases and the STS Bases are simply the result of the significant differences from existing PWR designs, functions, and operations. The FSAR provides additional information regarding design details with emphasis on Chapters 4 through 9 applicable to these Bases.

The responses below address the questions identified regarding both B 3.4.5 and B 3.4.9. Individual responses are identified by number corresponding to the question listed; paragraphs numbered A.# are related to the Bases of 3.4.5, and those numbered B.# are related to the Bases of 3.4.9.

With Respect to A.1, FSAR section 6.2, Containment Systems provides details of the NuScale containment design. The Bases were modified to reflect the containment design that does not generally refer to containment as 'areas' due to its size relative to the RCS. The discussion of separating identified LEAKAGE was modified to be consistent with, and accurately reflect the NuScale LEAKAGE capabilities as described in FSAR 5.2.5.

With Respect to A.2, the paragraph is inappropriate for the NuScale design as described in FSAR 5.2.5. The containment operates at a very low absolute pressure and systems within containment will be maintained leaktight, especially when compared to the operation of existing PWRs as described in the STS.

With Respect to A.3, RCS LEAKAGE that is not primary-to-secondary does not affect the adequacy and availability of core cooling because LEAKAGE would flow into the containment. RCS inventory in the containment vessel would be available to support emergency core cooling if required, and as described in FSAR Section 6.3. FSAR Section 15.6 addresses decreases in reactor coolant inventory events.

With Respect to A.4, the description was modified to reflect the NuScale design including specifically the RCS and ECCS designs as described in FSAR Chapter 5, Section 6.3, as well as the analyses in FSAR Chapter 15.

With Respect to A.5, see the response to RAI 16-38 and the accompanying changes. The 150 gpd limit is consistent with the NuScale design such that no additional accident-induced leakage will occur. The changes to the Bases were made to reflect the design.

With Respect to A.6, paragraph "a." was modified to align with the corresponding paragraph from NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 4.0.

With Respect to A.7, the Bases are consistent with the LCO limit, and the NuScale design and analyses.

With Respect to A.8, the Bases are consistent with the LCO limit, the NuScale design and analyses.

With Respect to A.9, the Bases terminology and usage is consistent with the LCO.

With Respect to A.10, the NuScale response to the referenced RAI Question 16-32 addresses the differences between STS completion times applicable to operations of existing PWRs and the completion times in the NuScale proposed TS. That response was submitted to the NRC on November 13, 2017 and is available in ADAMS at ML17317B552.

With Respect to A.11, the Bases are consistent with the SR and the NuScale design.

With Respect to B.1, the sentence was restored and refers to LCO 3.5.2, "Decay Heat Removal System (DHRS)." TR-1116-52011-NP will be clarified when it is revised.

With Respect to B.2, see the response provided to RAI 16-38.

With Respect to B.3, the omitted paragraph has been inserted.

With Respect to B.4, this formatting difference was corrected.

With Respect to B.5, see the response to RAI 16-38 regarding the accident-induced leakage criteria. The reference to LCO 3.4.8 was corrected to read LCO 3.4.5, "RCS Operational LEAKAGE." NuScale uses the phrase 'steam generator tube failure' rather than the industry standard 'steam generator tube rupture' because of the helical coil SG design in which the higher pressure is outside the tubes. The term SGTF is defined at the beginning of the ASA section of the Bases for LCO 3.4.9.

With Respect to B.6, the reference to LCO 3.4.8 was corrected to read LCO 3.4.5, "RCS Operational LEAKAGE." The "through any one SG" was removed, consistent with the limits in LCO 3.4.5.

With Respect to B.7, the term 'repair criteria' has been modified to 'plugging criteria' throughout the identified sections.

With Respect to B.8, this discussion was previously modified in response to RAI 16-32 and as shown in the response to that question.

With Respect to B.9, B.10, and B.11, the phrase 'tube repair criteria' was replaced with 'tube plugging criteria.'

Impact on DCA:

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube failure (SGTF). The leak contaminates the secondary fluid.

The FSAR Chapter 15 (Ref. 3) analyses for the accidents involving secondary side releases assume 150 gpd primary to secondary LEAKAGE as an initial condition. The design basis radiological consequences resulting from a postulated SLB accident and SGTF are provided in Sections 15.1.5 and 15.6.3 of FSAR Chapter 15, respectively.

The RCS Operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

~~No pressure boundary LEAKAGE, defined as LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall within the reactor coolant boundary defined in 10CFR50.2 is allowed. LEAKAGE of this type is unacceptable as the leak itself could cause further material deterioration, resulting in higher LEAKAGE. If LEAKAGE through a fault in an RCS component body, pipe wall, or vessel wall is isolated, it is no longer considered pressure boundary LEAKAGE. A fault in an RCS component body, pipe wall, or vessel wall is isolated if LEAKAGE through the isolation device is ≤ 0.5 gpm per nominal inch of valve size up to a maximum limit of 5 gpm. This will prevent further material degradation. LEAKAGE past seals, gaskets, valve seats, and mechanical or threaded connections is not pressure boundary LEAKAGE.~~No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

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

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.

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.

BASES

LCO (continued)

~~Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).~~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.

~~The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTF, is within the accident analysis assumptions. 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." 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.~~

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during unit~~plant~~ operation. The limit on operational LEAKAGE is contained in LCO 3.4.5~~8~~, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE ~~through any one SG~~ 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.

BASES

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.

<u>A.1 and A.2</u>

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair 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 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