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July 15, 2011

10 CFR 50.90

ATTN: Document Control Desk  
U.S. Nuclear Regulatory Commission  
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Unit 2  
Facility Operating License No. DPR-79  
NRC Docket No. 50-328

**Subject: Application to Modify Technical Specifications for Replacement Steam Generators (TS-SQN-2011-01)**

**Reference:** Letter from NRC to TVA, "Sequoyah Nuclear Plant, Unit 2 - Issuance of Amendment Regarding Steam Generator Tube Integrity (TAC No. MD0145)," dated May 22, 2007

In accordance with the provisions of 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," the Tennessee Valley Authority (TVA) is submitting a request for an amendment to Facility Operating License DPR-79 for Sequoyah Nuclear Plant (SQN), Unit 2.

The license amendment request proposes to revise the SQN, Unit 2, Technical Specifications (TS) requirements for steam generator tube inspections to reflect the replacement steam generators to be installed during SQN, Unit 2, refueling outage 18 (U2R18) presently scheduled for the fall of 2012. Previous changes to the SQN, Unit 2, TS to reflect Technical Specification Task Force (TSTF) Standard Technical Specification Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4 were approved as Amendment No. 305 by the reference letter dated May 22, 2007. The changes proposed in this license amendment request reflect the inspection requirements of TSTF-449, Revision 4.

The enclosure to this letter provides a description of the proposed changes and confirmation of applicability. Attachments 1 and 2 to the enclosure provide the existing TS and Bases pages marked-up to show the proposed changes. Attachments 3 and 4 to the enclosure provide the existing TS and Bases pages retyped to show the proposed changes.

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The change is proposed for implementation following SQN, Unit 2 refueling outage U2R18, when the replacement steam generators will have been installed. Accordingly, TVA requests NRC approval on a schedule to allow implementation of this TS change to coincide with the SQN, Unit 2, refueling outage U2R18.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS change qualifies for categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9).

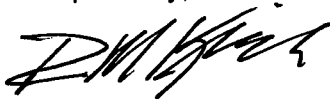
The SQN Plant Operations Review Committee and the SQN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of SQN in accordance with the proposed change will not endanger the health and safety of the public.

Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosure to the Tennessee Department of Environment and Conservation.

There are no new regulatory commitments associated with this submittal. Please address any questions regarding this request to Dan Green at 423-751-8423.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 15th day of July 2011.

Respectfully,



R. M. Krich

Enclosure: Evaluation of the Proposed Change

cc (Enclosure):

NRC Regional Administrator - Region II  
NRC Senior Resident Inspector - Sequoyah Nuclear Plant  
Director, Division of Radiological Health - Tennessee Department of  
Environment and Conservation

**ENCLOSURE**

**TENNESSEE VALLEY AUTHORITY  
SEQUOYAH NUCLEAR PLANT  
UNIT 2**

**EVALUATION OF THE PROPOSED CHANGE**

**Subject: Request for Change to Technical Specification Section 6 for  
Replacement Steam Generators (TS-SQN-2011-01)**

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4. Proposed TS Bases Changes (Final Typed)

## 1.0 SUMMARY DESCRIPTION

By letter dated May 22, 2007 (Reference 1), License Amendment No. 305 was approved for the Sequoyah Nuclear Plant (SQN), Unit 2. That amendment implemented Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler 449 (TSTF-449) for the existing SQN, Unit 2, steam generators (SGs). This License Amendment Request (LAR) seeks implementation of the TSTF-449 inspection requirements for the replacement steam generators (RSGs), that are being installed during the SQN, Unit 2, refueling outage 18 (U2R18) scheduled to start in the fall of 2012. The RSGs differ from the existing SGs in that the tube material in the RSGs is Alloy 690 Thermally Treated (TT) versus Alloy 600 in the existing SGs. The inspection requirements for Alloy 690 TT material permit longer periods between 100 percent tube population inspections and between individual SG inspections. Additionally, this LAR removes inspection requirements that are designated for specific damage conditions in the existing SGs, removes tube repair techniques approved by previous license amendments for the existing SGs, and removes inspection and reporting requirements specific to those repair techniques.

The elements of TSTF-449 regarding the Technical Specifications (TS) definition of Identified Leakage, TS definition of Pressure Boundary Leakage, and TS 3.4.6.2, Reactor Coolant System Operational Leakage, were incorporated into the SQN, Unit 2, TS by License Amendment No. 305. Those elements are consistent with this proposed change. Since those elements are integral to the overall maintenance of RSG tube integrity, reference to them in the *Federal Register Notice* (Reference 2) discussions applicable to Sections 3.0 through 5.0 below continues to remain appropriate.

## 2.0 DETAILED DESCRIPTION

The proposed TS changes are as follows.

- Revised TS Section 6.8.4.k, "Steam Generator (SG) Program"
- Revised TS Sections 6.9.1.16.2, 6.9.1.16.3, 6.9.1.16.4, and 6.9.1.16.5, "Steam Generator (SG) Tube Inspection Report"

The revisions are necessary because of two factors. The inspection frequency for Alloy 690 TT tube material, as defined in TSTF-449, differs from the inspection frequency for Alloy 600, and the tube repair processes and products in the existing TS are not applicable to the RSGs.

TS Section 6.8.4.k is revised to change the tube inspection frequency, as specified in TSTF-449, from that applicable to Alloy 600 to that applicable to Alloy 690TT tube material.

TS Section 6.8.4.k is revised to delete information on repair techniques and inspection requirements specific to tube repairs.

TS Section 6.8.4.k is revised to delete inspection requirements that are designated for specific damage conditions in the existing SGs.

TS Sections 6.9.1.16.2, 6.9.1.16.3, 6.9.1.16.4, and 6.9.1.16.5 are revised to remove reporting requirements associated with the deleted repair techniques.

Proposed revisions to the TS Bases are also included in this application for information only. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4, is an integral part of implementing this TS change. The changes to the affected TS Bases pages will be incorporated in accordance with the SQN, Unit 2, TS Bases Control Program.

### **3.0 TECHNICAL EVALUATION**

The Tennessee Valley Authority (TVA) has reviewed the safety evaluation published on March 2, 2005, (70 FR 10298) (Reference 2), as part of the Consolidated Line Item Improvement Program (CLIIP) Notice for Comment. This included the NRC's Safety Evaluation (SE), the supporting information provided in TSTF-449, and the changes associated with Revision 4 to TSTF-449. TVA has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC are applicable to SQN, Unit 2, and justify this amendment for the incorporation of the changes to the SQN, Unit 2, TS.

Calculations performed by the manufacturer of the RSGs have confirmed the acceptability of the 40 percent tube plugging limit included in the existing TS. Those calculations were performed in accordance with the guidance and recommendations of Regulatory Guide 1.121 (Reference 4).

The proposed 100 percent inspection frequency and maximum interval for inspecting a RSG are in accordance with Revision 4 of TSTF-449 for steam generators with Alloy 690 TT tube material.

There are no repair processes currently approved for the RSGs tube material; therefore, the reference to repair, and the application of repair processes, in TS Section 6.8.4.k must be removed. Additionally, inspection requirements that are designated for specific damage conditions in the existing SGs must be removed from TS Section 6.8.4.k.

### **4.0 REGULATORY EVALUATION**

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on May 6, 2005 (70 FR 24126) (Reference 3), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

#### **4.1 Applicable Regulatory Requirements /Criteria**

The applicable regulatory requirements/criteria associated with this application are adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

The following information concerning the RSGs is provided to support the NRC review of this LAR.

Plant Name, Unit Number:	Sequoyah Nuclear Plant (SQN), Unit 2			
Steam Generator (G) Model(s):	Westinghouse Electric Company Model 57AG <sup>+</sup>			
Effective Full Power Years (EFPY) of service for currently installed SGs:	New at startup from SQN, Unit 2, Refueling Outage in Fall 2012.			
Tubing Material (e.g., 600M, 600TT, 690TT):	690TT (Thermally Treated)			
Number of tubes per SG:	4,983			
Number and percentage of tubes plugged in each SG*:	<u>SG 1</u>	<u>SG 2</u>	<u>SG 3</u>	<u>SG 4</u>
	0	0	0	0
	0	0	0	0
Number of tubes repaired in each SG:	0			
Degradation mechanism(s):	None			
Current primary-to-secondary leakage limits:	Per SG: 150 gallons per day through any one steam generator per TS 3.4.6.2			
Approved Alternate Repair Criteria (ARC):	None.			
Approved SG Tube Repair Methods:	None.			
Performance criteria for accident leakage:	Primary-to-secondary leak rate values assumed in the limiting SQN, Unit 2, licensing basis accident analysis are 0.1 gallons per minute (gpm) for each of the non-faulted SGs and 1.0 gpm for the faulted SG.			

\* There were no tubes plugged during fabrication. The tubes will be pre-service eddy current tested beginning in January 2012. Based on this testing, if tube plugging is required, TVA will provide the number and percentage of tubes plugged in each steam generator within 90 days following completion of eddy current testing.

## 4.2 Precedent

This application is made in accordance with TSTF-449. TVA is not proposing any variations or deviations from the TS changes described in TSTF-449, Revision 4, or the NRC's model Safety Evaluation published on March 2, 2005, (70 FR 10298).

The proposed changes for the SQN, Unit 2, RSGs are similar to proposed changes included in the following license amendment requests.

- Pacific Gas and Electric Company for the Diablo Canyon, Units 1 and 2, "License Amendment Request 07-01 Revision to Technical Specifications to Support Steam Generator Replacement," dated January 11, 2007 (Reference 5); Approved by NRC letter to Pacific Gas and Electric Company, dated January 8, 2008 (Reference 6)
- Florida Power Corporation for the Crystal River, Unit 3, "Crystal River Unit 3 - License Amendment Request #301, Revision 1: Application to Modify Improved Technical Specifications for Replacement Steam Generators and Response to Request for Additional Information (TAC NO. MD9547)," dated

January 19, 2009 (Reference 7); Approved by NRC letter to Progress Energy, dated May 29, 2009 (Reference 8)

#### **4.3 Significant Hazards Consideration**

The proposed License Amendment Request (LAR) revises the Sequoyah Nuclear Plant (SQN), Unit 2, Technical Specifications (TS) Section 6.8.4.k, "Steam Generator (SG) Program," and TS Sections 6.9.1.16.2, 6.9.1.16.3, 6.9.1.16.4, and 6.9.1.16.5, "Steam Generator (SG) Tube Inspection Report." The proposed changes are necessary to revise the current SQN, Unit 2, TS for Replacement Steam Generators (RSGs) to be installed in the fall of 2012 refueling outage.

TVA has evaluated the proposed LAR against the criteria of 10 CFR 50.92(c) to determine if any significant hazards consideration is involved. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change for RSGs continues to implement the current SG Program that includes performance criteria which provide reasonable assurance that the RSG tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specifications). This change removes repair criteria from the SG Program that were approved by previous License Amendments for the existing SGs which are not applicable to the RSGs. It removes references to use of repairs and reporting of repair results in other TS sections. This change removes inspection requirements that are designated for specific damage conditions in the existing SGs. The change also revises the inspection interval for 100 percent inspections of SG tubes and the maximum interval for inspection of a single SG consistent with Technical Specification Task Force (TSTF) Standard Technical Specification Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4 for the Alloy 690 tube material in the RSGs. The revised inspection requirements are based on properties and experience with the improved Alloy 690 tube material. The revised inspection requirements will result in the same outcome that SG tube integrity will continue to be maintained.

This change continues to implement SG performance criteria for tube structural integrity, accident induced leakage, and operational leakage for the RSGs. Meeting the performance criteria provides reasonable assurance that the RSG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident (DBA). The performance criteria are only a part of the SG Program required by the

existing TS. The program, defined by NEI 97-06, "Steam Generator Program Guidelines," includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, and leakage monitoring. These features will continue to be implemented as they are currently approved. The proposed changes do not, therefore, significantly increase the probability of an accident previously evaluated.

The consequences of DBAs are, in part, functions of the Dose Equivalent I-131 in the primary coolant and the primary to secondary leakage rates resulting from an accident. Therefore, limits are included in the TS for Operational Leakage and for Dose Equivalent I-131 in the primary coolant to ensure the plant is operated within its analyzed condition. The analysis of the limiting DBA assumes that the primary to secondary leak rate, after the accident, is 1 gallon per minute with no more than 150 gallons per day in any one SG, and that the reactor coolant activity levels of Dose Equivalent I-131 are at the TS values before the accident. The proposed change to the SG inspection program does not affect the design of the SGs, their method of operation, operational leakage limits, or primary coolant chemistry controls. The proposed change does not adversely impact any other previously evaluated DBA. In addition, the proposed changes do not affect the consequences of a main steam line break, rod ejection, a reactor coolant pump locked rotor event, or other previously evaluated accident. Therefore, the proposed change does not affect the consequences of a SG tube rupture accident and the probability of such an accident is unchanged.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed license amendment does not affect the method of operation of the SGs, or the primary or secondary coolant chemistry controls. In addition, the proposed amendment does not impact any other plant system or component. The change modifies existing SG inspection requirements based on the RSG design and the properties and experience associated with their improved materials. The revised inspection requirements will result in the same outcome that SG tube integrity will continue to be maintained. Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the



SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes. SG tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change to the SG inspection program does not affect tube design or operating environment. The existing SG Program is maintained in this change. The repair criteria that are being removed are specific to the existing SGs and are not applicable to the RSGs. If tube defects are detected that exceed limits in the RSGs, then the tube will be removed from service. The effective tube plugging percentage will continue to be tracked for all plugging in each SG in accordance with TS Section 6.9.1.16.1 to ensure the heat transfer function of the SGs is not adversely affected. For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

Based on the above, TVA concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

#### **4.4 Conclusion**

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

#### **5.0 ENVIRONMENTAL CONSIDERATION**

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared in connection with the proposed amendment.

#### **6.0 REFERENCES**

1. Letter from NRC to TVA, "Sequoyah Nuclear Plant, Unit 2 - Issuance of Amendment Regarding Steam Generator Tube Integrity (TAC No. MD0145)," dated May 22, 2007
2. *Federal Register* Notice for Comment published on March 2, 2005 (70 FR 10298)
3. *Federal Register* Notice of Availability published on May 6, 2005 (70 FR 24126)

4. U.S. Nuclear Regulatory Commission Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976
5. Letter from Pacific Gas and Electric Company to NRC, "License Amendment Request 07-01 Revision to Technical Specifications to Support Steam Generator Replacement," dated January 11, 2007
6. Letter from NRC to Pacific Gas and Electric Company, "Diablo Canyon Power Plant, Unit Nos. 1 and 2 - Issuance of Amendments Re: Revise Technical Specifications to Support Steam Generator Replacement (TAC Nos. MD3992 and MD3993)," dated January 8, 2008
7. Letter from Progress Energy to NRC, "Crystal River Unit 3 - License Amendment Request #301, Revision 1: Application to Modify Improved Technical Specifications for Replacement Steam Generators and Response to Request for Additional Information (TAC NO. MD9547)," dated January 19, 2009
8. Letter from NRC to Progress Energy, "Crystal River Unit 3 - Issuance of Amendment Regarding the Revision of the Steam Generator Portion of the Technical Specifications to Reflect the Replacement of the Steam Generators (TAC No. MD9547)," dated May 29, 2009

**ATTACHMENT 1**

**TENNESSEE VALLEY AUTHORITY  
SEQUOYAH NUCLEAR PLANT  
UNIT 2**

**PROPOSED TS CHANGES (Mark-Ups)**

## ADMINISTRATIVE CONTROLS

- d. Proposed changes that meet the criteria of Specification 6.8.4.j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

k. 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:

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. Provisions for 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 SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents (DBAs). This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and ~~except for flaws addressed through application of the alternate repair criteria discussed in TS 6.8.4.k.c.1,~~ a safety factor of 1.4 against burst applied to the DBA primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the DBAs, 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.

~~For predominantly axially oriented outside diameter stress corrosion cracking (ODSCC) at the tube support plate elevations, (refer to 6.8.4.k.c.1) the probability of burst (POB) of one or more indications given a steam line break shall be less than  $1 \times 10^{-2}$ .~~

2. Accident induced leakage performance criterion: The accident-induced leakage ~~from all sources, excluding the leakage attributed to the degradation described in 6.8.4.k.c.1 and 2,~~ is not to exceed 1.0 gpm for the faulted SG and 0.1 gpm for each of the non-faulted SGs. The primary-to-secondary accident induced leakage rate for any DBA, 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.

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3. The operational leakage performance criterion is specified in Limiting Condition for Operation (LCO) 3.4.6.2, "Reactor Coolant System, Operational Leakage."

c. Provisions for SG Tube 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.

The following alternate tube repair criteria (ARC) may be applied as an alternative to the 40% depth based criteria:

1. NRC Generic Letter (GL) 95-05 Voltage-Based ARC (Tube Support Plate [TSP])

A voltage-based TSP repair criteria is used for the disposition of an alloy 600 SG tube for continued service that is experiencing predominately axially oriented ODSCC confined within the thickness of the tube support plates (TSPs). At TSP intersections, the repair criteria is described below:

- a) SG tubes, whose degradation is attributed to ODSCC within the bounds of the TSP with bobbin voltages less than or equal to 2.0 volts, will be allowed to remain in service.
- b) SG tubes, whose degradation is attributed to ODSCC within the bounds of the TSP with a bobbin voltage greater than 2.0 volts will be plugged, except as noted in Item 6.8.4.k.c.1.c) below.
- c) SG tubes, with indications of potential degradation attributed to ODSCC within the bounds of the TSP with a bobbin voltage greater than 2.0 volts, but less than or equal to the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented), may remain in service if a rotating pancake coil inspection or comparable technology does not detect degradation.
- d) SG tubes with indications of ODSCC degradation with a bobbin coil voltage greater than the upper voltage repair limit (calculated according to the methodology in GL 95-05 as supplemented) will be plugged.
- e) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in Items 6.8.4.k.c.1.a), .b), .c) and .d).

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr \frac{(CL - \Delta t)}{CL}}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL}) \frac{(CL - \Delta t)}{CL}$$

where:

$V_{URL}$  = upper voltage repair limit

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$V_{LRL}$	=	lower voltage repair limit
$V_{MURL}$	=	mid-cycle upper voltage repair limit based on time into cycle
$V_{MLRL}$	=	mid-cycle lower voltage repair limit based on $V_{MURL}$ and time into cycle
$\Delta t$	=	length of time since last scheduled inspection during which $V_{URL}$ and $V_{LRL}$ were implemented
CL	=	cycle length (the time between two scheduled SG inspections)
$V_{SL}$	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS items 6.8.4.k.c.1.a), .b), .c) and .d).

2. W\* Methodology

The following terms/definitions apply to the W\*.

- Bottom of WEXTEx Transition (BWT) is the highest point of contact between the tube and tubesheet at, or below the top of tubesheet (TTS), as determined by eddy current testing.
- W\* Distance for the hot-leg tubesheet is the larger of the following two distances as measured from the TTS: (a) 8 inches below the TTS or (b) 7 inches below the bottom of the WEXTEx transition plus the uncertainty associated with determining the distance below the bottom of the WEXTEx transition as defined by WCAP-14797, Revision 2.
- W\* distance for the cold-leg tubesheet is 10.5 inches below TTS.

Service induced flaws identified in the W\* distance shall be plugged on detection. Flaws located below the W\* distance may remain in service regardless of size.

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

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at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, d.3, d.4, and d.5 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 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 SG replacement.

144, 108, 72 and thereafter,

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SGs shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.

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3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever 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.

4. ~~GL 95-05 Voltage-Based ARC for TSP~~

~~Indications left in service as a result of application of the TSP voltage-based repair criteria shall be inspected by bobbin coil probe every 24 effective full power months or every refueling outage, whichever is less.~~

~~Implementation of the SG tube TSP repair criteria requires a 100 percent bobbin coil inspection for hot-leg and cold-leg TSP intersections down to the lowest cold-leg TSP with known ODSCC indications. The determination of the lowest cold-leg TSP intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.~~

5. ~~W\* Inspection~~

~~When the W\* methodology has been implemented, inspect 100 percent of the inservice tubes in the hot-leg tubesheet and 20 percent of the inservice tubes in the cold-leg tubesheet regions with the objective of detecting flaws that may satisfy the applicable tube repair criteria of TS 6.8.4.k.c.2.~~

e. Provisions for Monitoring Operational Primary-to-Secondary Leakage.

i. Component Cyclic and Transient Limit

This program provides controls to track the FSAR, Section 5.2.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

## ADMINISTRATIVE CONTROLS

### STEAM GENERATOR (SG) TUBE INSPECTION REPORT (continued)

- a. The scope of inspections performed on each SG,
- b. Active degradation mechanisms found,
- c. Nondestructive examination techniques utilized for each degradation mechanism,
- d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
- e. Number of tubes plugged during the inspection outage for each active degradation mechanism,
- f. Total number and percentage of tubes plugged to date,
- g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and
- h. The effective plugging percentage for all plugging in each SG.

6.9.1.16.2 A report shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the steam generator program (6.8.4.k) when voltage based alternate repair criteria have been applied. The report shall include information described in Section 6.b of Attachment 1 to NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

6.9.1.16.3 For implementation of the voltage-based repair criteria for tube support plate (TSP) intersections, notify the staff prior to initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.k, "Steam Generator (SG) Program," should any of the following conditions arise:

- 1) If circumferential crack-like indications are detected at the TSP intersections.
- 2) If indications are identified that extend beyond the confines of the TSP.
- 3) If indications are identified at the TSP elevations that are attributable to primary water stress corrosion cracking.

6.9.1.16.4 For implementation of  $W^*$ , the calculated steam line break leakage from the application of TSP alternate repair criteria and  $W^*$  inspection methodology shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.k, "Steam Generator (SG) Program." The report will include the number of indications within the tubesheet region, the location of the indications (relative to the bottom of the WEXTEx transition [BWT] and TTS), the orientation (axial, circumferential, skewed, volumetric), the severity of each indication (e.g., near through-wall or not through-wall), the side of the tube from which the indication initiated (inside or



## ADMINISTRATIVE CONTROLS

6.9.1.16.5 For implementation of the probability of prior cycle detection (POPCD) method, for the voltage-based repair criteria at tube support plate intersections, if the end-of-cycle conditional tube rupture probability for a postulated main steam line break, the projected primary to secondary leak rate during a postulated main steam line break, or the number of indications are under predicted by the previous cycle operational assessment, the following shall be reported to the Commission within 90 days after initial entry into MODE 4 following completion of inspection performed in accordance with specification 6.8.4.k, "Steam Generator Program."

1. The assessment of the probable causes for the under prediction, proposed corrective actions, and any recommended changes to probability of detection or growth methodology indicated by potential methods assessments.
2. An assessment of the potential need to revise the alternate repair criteria analysis methods if: the burst probability is under predicted by more than 0.001 (i.e., 10 percent of the performance criteria) or an order of magnitude; or the leak rate is under predicted by more than 0.5 gallon per minute (gpm) or an order of magnitude.
3. An assessment of the potential need to increase the number of predicted low voltage indications at the beginning of cycle if the total number of as-found indications in any SG are underestimated by greater than 15 percent or by greater than 150 indications.

## SPECIAL REPORTS

6.9.2.1 Special reports shall be submitted within the time period specified for each report, in accordance with 10 CFR 50.4.

6.9.2.2 This specification has been deleted.

## 6.10 RECORD RETENTION (DELETED)

**TS-SQN-2011-01**  
**TS INSERT**

Insert A:

In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period.

**ATTACHMENT 2**

**TENNESSEE VALLEY AUTHORITY  
SEQUOYAH NUCLEAR PLANT  
UNIT 2**

**PROPOSED TS BASES CHANGES (Mark-Ups)**

REACTOR COOLANT SYSTEM

BASES

APPLICABLE  
SAFETY  
ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this specification. The analysis of an SGTR event assumes a bounding primary to secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.6.2 "Operational Leakage," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves. The main condenser isolates based on an assumed concurrent loss of off-site power.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere depends on the accident and whether there are faulted SGs associated with the accident. For a steamline break (SLB), the maximum primary to secondary leakage under accident conditions is limited to 3.7 gpm from the faulted SG and 0.1 gpm from each of the non-faulted SGs.

~~Of the 3.7 gpm primary-to-secondary leak rate assumed during the SLB, no more than 1.0 gpm can come from sources that have not been specifically exempted from the 1.0 gpm limit by the NRC. The leakage attributed to the flaws left in service as a result of implementing TS 6.8.4.k.c.1 and .2 have been exempted from the 1.0 gpm limit by the NRC staff.~~ For other accidents that assume a faulted SG (e.g., feedwater line break), the maximum primary to secondary leakage under accident conditions is limited to 1.0 gpm from the faulted SG and 0.1 gpm from each of the non-faulted SGs. For accidents in which there are no faulted SGs, the primary to secondary leakage is limited to 0.1 gpm from each SG. 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, "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), and 10 CFR 100 (Ref. 3), or the NRC approved licensing basis.

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

~~Voltage-Based Alternate Repair Criteria (ARC) and W\* Methodology~~

~~a) Voltage-Based ARC~~

~~The voltage-based repair limits implement the guidance in Generic Letter (GL) 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking~~

REACTOR COOLANT SYSTEM

BASES

APPLICABLE  
SAFETY ANALYSIS (continued)

(ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of voltage-based repair limits require a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95 percent prediction interval curve reduced to account for the lower 95/95 percent tolerance bound for tubing material properties at 650°F (i.e., the 95 percent lower tolerance limit curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit;  $V_{URL}$ , is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{GR} - V_{NDE}$$

where  $V_{GR}$  represents the allowance for flaw growth between inspections and  $V_{NDE}$  represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

The mid-cycle equation of TS 6.8.4.k.c.1.e should only be used during unplanned inspection in which eddy current data is acquired for indications at the tube support plates.

Specification 6.9.1.16.3 implements several reporting requirements recommended by GL 95-05 for situations which NRC wants to be notified prior to returning the SGs to service. For 6.9.1.16.3 Items 2 and 3,

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SAFETY ANALYSIS (continued)

are applicable only where alternate plugging criteria is being applied. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle (EOC) voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing GL Sections 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per GL Section 6.b(c) criteria.

For the operational assessment, the Probability of Prior Cycle Detection (POPCD) voltage based probability of detection (POD) method, as approved by NRC letter dated March 24, 2008, is used to determine the beginning of cycle voltage distributions. The POPCD method is an exception to the GL 95-05 guidance that requires the application of a POD of 0.6 to all previous bobbin indications.

Tubes experiencing ODSCC within the thickness of the tube support plate are plugged by the criteria of 6.8.4.k.c.1.

b) W\* Methodology

The W\* criteria incorporates the guidance provided in WCAP-14797, Revision 2, "Generic W\* Tube Plugging Criteria for 51 Series Steam Generator Tubesheet Region WEXTEx Expansions." W\* length is the length of tubing into the tubesheet below the bottom of the WEXTEx transition (BWT) that precludes tube pullout in the event of a complete circumferential separation of the tube below the W\* length. W\* distance is the distance from the top-of-tube sheet (TTS) to the bottom of the W\* length including the distance from the TTS to the BWT and measurement uncertainties.

Indications detected within the W\* distance below the TTS, will be plugged upon detection. Tubes to which WCAP-14797 is applied can experience through-wall degradation up to the limits defined in Revision 2 without increasing the probability of a tube rupture or large leakage event. Tube degradation of any type or extent below W\* distance,



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PWSCC tube indications are identified in the cold-leg tubesheet region the cold-leg OA leakage is 0.0 gpm.

All other sources of accident induced primary to secondary leakage is the primary to secondary accident induced OA leakage from all other degradation mechanisms other than the voltage based axial ODSCC at tube support plates repair criteria and  $W^*$  leakage calculations as determined by the Operational Assessment.

d) Calculation of Condition Monitoring (CM) Accident Induced Leakage

The postulated leakage during a SLB shall be equal to the following equation and is performed for each steam generator:

Postulated SLB CM Leakage =  $ARC_{GL\ 95-05} + \text{Assumed Leakage}_{0"-8" < TTS} + \text{Assumed Leakage}_{8"-12" < TTS} + \text{Assumed Leakage}_{>12" < TTS} + \text{All other sources of accident induced primary to secondary leakage.}$

Where:  $ARC_{GL\ 95-05}$  is the SLB CM leakage for predominantly axially oriented outside diameter stress corrosion cracking indications as determined from the methodology described in GL 95-05 as revised by Technical Specification Change 06-06.

Assumed Leakage  $0"-8" < TTS$  is the postulated CM leakage for indications detected in SG tubes between 0 and 8 inches below the TTS for both the hot-leg and cold-leg tubesheet.

Assumed Leakage  $8"-12" < TTS$  is the conservatively assumed CM leakage from the total of identified and postulated unidentified indications in SG tubes left in service between 8 and 12 inches below the TTS for both the hot-leg and cold-leg tubesheet. This is 0.0045 gpm multiplied by the number of indications.

Assumed Leakage  $>12" < TTS$  is the conservatively assumed CM leakage for the bounding SG tubes in service 12 inches below the TTS for both the hot-leg and cold-leg tubesheet. This is 0.00009 gpm multiplied by the number of tubes left in service in the SG. When no PWSCC tube indications are identified in the cold-leg tubesheet region the cold-leg CM leakage is 0.0 gpm.

All other sources of accident induced primary to secondary leakage is the primary to secondary accident induced CM leakage from all other degradation mechanisms other than the voltage based axial ODSCC at tube support plates repair criteria and  $W^*$  leakage calculations as determined by Condition Monitoring.



## REACTOR COOLANT SYSTEM

### BASES

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#### APPLICABLE SAFETY ANALYSIS (continued)

~~The aggregate calculated accident induced primary to secondary SLB leakage from the application of all approved ARC (W\* and voltage-based axial ODSCC at TSP) shall be reported to the NRC in accordance with Technical Specification 6.9.1.16.4. The combined calculated leak rate from all ARC and all other sources of accident induced leakage must be less than the accident induced primary to secondary leakage rate assumed in the SLB accident analyses.~~

#### 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 6.8.4.k "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 burst 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 burst 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

## REACTOR COOLANT SYSTEM

### BASES

#### LCO (continued)

(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 burst/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 requires that the primary membrane stress intensity in a tube not exceed the yield strength for all American Society of Mechanical Engineers (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).

The accident induced leakage performance criterion ensures that the primary to secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analyses assumptions are discussed in the ~~Applicability~~ Safety Analyses section. 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.

Applicable

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2, “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 SGTR under the stress conditions of a loss-of-coolant accident (LOCA) or a ~~M~~SLB. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

## REACTOR COOLANT SYSTEM

### BASES

#### ACTIONS (continued)

If the evaluation determines that the affected tube(s) have tube integrity, Action (a) allows plant 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 startup following the next refueling outage or SG inspection. This allowed time is acceptable since operation until the next inspection is supported by the operational assessment.

If ~~at any time, evaluation determines~~ SG tube integrity is not being maintained ~~(applies to any SG tube; either inadvertently not plugged or left in service in accordance with the approved repair criteria)~~, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours and the affected tube(s) plugged prior to restart ~~(Mode 4)~~.

The action times are reasonable, based on operating experience, to reach the desired plant condition from full power in an orderly manner and without challenging plant systems.

#### SURVEILLANCE REQUIREMENTS

##### SR 4.4.5.0

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 "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 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

## REACTOR COOLANT SYSTEM

### BASES

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#### SURVEILLANCE REQUIREMENTS (continued)

specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.0. 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 6.8.4.k contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

#### SR 4.4.5.1

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.k 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 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.

The frequency of this surveillance ensures that the surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential (i.e., prior to HOT SHUTDOWN following a SG tube inspection).

REACTOR COOLANT SYSTEM

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REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 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."
7. ~~NRC Generic Letter 95-05, "Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."~~
8. ~~NRC letter to TVA dated April 9, 1997, "Issuance of Technical Specification Amendments for the Sequoyah Nuclear Plant, Units 1 and 2 (TAC Nos. M96998 and M96999) (TS 96-05)."~~
9. ~~NRC letter to TVA dated May 3, 2005, "Sequoyah Nuclear Plant, Unit 2 – Issuance of Amendment Regarding Changes to the Inspection Scope for the Steam Generator Tubes (TAC No. MC5212) (TS-03-06)."~~

## REACTOR COOLANT SYSTEM

### BASES

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#### APPLICABLE SAFETY ANALYSES (continued)

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

The FSAR (Ref. 3) analysis for steam generator tube rupture (SGTR) assumes the contaminated secondary fluid is released via safety valves for up to 30 minutes. Operator action is taken to isolate the affected steam generator within this time period. The 0.4 gpm operational primary to secondary leakage safety analysis assumption is relatively inconsequential.

The SLB ~~with ARC applied leakage~~ is more limiting for site radiation releases. The safety analysis for the SLB accident assumes a ~~maximum~~ 3.7 gpm primary to secondary leakage through the affected generator and 0.3 gpm through the non-affected generators as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits). ~~Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 3.7 gpm at atmospheric conditions and 70°F in the faulted loop, which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. If the projected and cycle distribution of crack indications results in primary-to-secondary leakage greater than 3.7 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 3.7 gpm.~~

The RCS operational leakage satisfies Criterion 2 of the NRC Policy Statement.

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#### LCO

RCS operational leakage shall be limited to:

a. 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. UNIDENTIFIED LEAKAGE

One gpm of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment pocket

**ATTACHMENT 3**

**TENNESSEE VALLEY AUTHORITY  
SEQUOYAH NUCLEAR PLANT  
UNIT 2**

**PROPOSED TS CHANGES (Final Typed)**

## ADMINISTRATIVE CONTROLS

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- d. Proposed changes that meet the criteria of Specification 6.8.4.j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

- k. 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:

- 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. Provisions for 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 SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents (DBAs). 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 DBA primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the DBAs, 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 accident-induced leakage is not to exceed 1.0 gpm for the faulted SG. The primary-to-secondary accident induced leakage rate for any DBA, 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.
3. The operational leakage performance criterion is specified in Limiting Condition for Operation (LCO) 3.4.6.2, "Reactor Coolant System, Operational Leakage."



## ADMINISTRATIVE CONTROLS

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c. Provisions for SG Tube 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 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 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 SG replacement.
2. Inspect 100% of the tubes at sequential periods of 144, 108, 72 and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SGs shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.
3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever 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.

I. Component Cyclic and Transient Limit

This program provides controls to track the FSAR, Section 5.2.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

## ADMINISTRATIVE CONTROLS

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### STEAM GENERATOR (SG) TUBE INSPECTION REPORT (continued)

- a. The scope of inspections performed on each SG,
- b. Active degradation mechanisms found,
- c. Nondestructive examination techniques utilized for each degradation mechanism,
- d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
- e. Number of tubes plugged during the inspection outage for each active degradation mechanism,
- f. Total number and percentage of tubes plugged to date,
- g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and
- h. The effective plugging percentage for all plugging in each SG.

## ADMINISTRATIVE CONTROLS

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### SPECIAL REPORTS

6.9.2.1 Special reports shall be submitted within the time period specified for each report, in accordance with 10 CFR 50.4.

6.9.2.2 This specification has been deleted.

### 6.10 RECORD RETENTION (DELETED)

SEQUOYAH - UNIT 2

6-15

Amendment No. 28, 44, 50, 64, 66, 107,  
134, 146, 153, 165, 169, 206, 214, 223, 231,  
249, 284, 309,

**ATTACHMENT 4**

**TENNESSEE VALLEY AUTHORITY  
SEQUOYAH NUCLEAR PLANT  
UNIT 2**

**PROPOSED BASES CHANGES (Final Typed)**

REACTOR COOLANT SYSTEM

BASES

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APPLICABLE  
SAFETY  
ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this specification. The analysis of an SGTR event assumes a bounding primary to secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.6.2 "Operational Leakage," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves. The main condenser isolates based on an assumed concurrent loss of off-site power.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere depends on the accident and whether there are faulted SGs associated with the accident. For a steamline break (SLB), the maximum primary to secondary leakage under accident conditions is limited to 3.7 gpm from the faulted SG and 0.1 gpm from each of the non-faulted SGs. For other accidents that assume a faulted SG (e.g., feedwater line break), the maximum primary to secondary leakage under accident conditions is limited to 1.0 gpm from the faulted SG and 0.1 gpm from each of the non-faulted SGs. For accidents in which there are no faulted SGs, the primary to secondary leakage is limited to 0.1 gpm from each SG. 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, "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) and 10 CFR 100 (Ref. 3).

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

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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 6.8.4.k "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.

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#### 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 burst 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 burst 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 burst/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 requires that the primary membrane stress intensity in a tube not exceed the yield strength for all American Society of Mechanical Engineers (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).

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#### LCO (continued)

The accident induced leakage performance criterion ensures that the primary to secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analyses assumptions are discussed in the Applicable Safety Analyses section. 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 plant operation. The limit on operational leakage is contained in LCO 3.4.6.2, "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 SGTR under the stress conditions of a loss-of-coolant accident (LOCA) or a SLB. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

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#### 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 MODES 1, 2, 3, or 4.

Reactor coolant system (RCS) conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

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#### ACTIONS

The ACTIONS are modified by a clarifying footnote that Action (a) may be entered independently for each SG tube. This is acceptable because the actions provide appropriate compensatory measures for each affected SG tube. Complying with the actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent action entry, and application of associated actions.

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#### ACTIONS (continued)

##### Actions (a) and (b)

Action (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 4.4.5.1. 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 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 refueling outage or SG tube inspection. If it is determined that tube integrity is not being maintained until the next SG inspection, Action (a) requires unit shutdown and Action (b) requires the affected tube(s) be plugged.

An allowed time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Action (a) allows plant 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 startup following the next refueling outage or SG inspection. This allowed time is acceptable since operation until the next inspection is supported by the operational assessment.

If SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours and the affected tube(s) plugged prior to restart.

The action times are reasonable, based on operating experience, to reach the desired plant condition from full power in an orderly manner and without challenging plant systems.



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#### SURVEILLANCE REQUIREMENTS

##### SR 4.4.5.0

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 "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 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, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.0. 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 6.8.4.k contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

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#### SURVEILLANCE REQUIREMENTS (continued)

##### SR 4.4.5.1

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.k 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 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.

The frequency of this surveillance ensures that the surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

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#### REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 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."

REACTOR COOLANT SYSTEM

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APPLICABLE  
SAFETY ANALYSES (continued)

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

The FSAR (Ref. 3) analysis for steam generator tube rupture (SGTR) assumes the contaminated secondary fluid is released via safety valves for up to 30 minutes. Operator action is taken to isolate the affected steam generator within this time period. The 0.4 gpm operational primary to secondary leakage safety analysis assumption is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes a 3.7 gpm primary to secondary leakage through the affected generator and 0.3 gpm through the non-affected generators as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

The RCS operational leakage satisfies Criterion 2 of the NRC Policy Statement.

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LCO

RCS operational leakage shall be limited to:

a. 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. UNIDENTIFIED LEAKAGE

One gpm of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment pocket