**Tennessee Valley Authority**, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

February 15, 2006

TVA-SQN-TS-05-09 10 CFR 50.90

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

Gentlemen:

| | | |
|---|---|---|
| In the Matter of | ) | Docket No. 50-328 |
| Tennessee Valley Authority | ) | |

**SEQUOYAH NUCLEAR PLANT (SQN) - UNIT 2 - TECHNICAL SPECIFICATIONS (TS) CHANGE 05-09 - APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY, AND DELETION OF LICENSE CONDITION**

Pursuant to 10 CFR 50.90, Tennessee Valley Authority (TVA) is submitting a request for a TS change (TS-05-09) to License DPR-79 for SQN Unit 2.

The proposed TS change revises the Unit 2 TS requirements related to steam generator tube integrity and removes a Unit 2 Operating License Condition that is associated with steam generator inspection. The TS change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." The availability of this TS improvement was announced in the *Federal Register* on May 6, 2005 (70 FR 24126), as part of the consolidated line item improvement process (CLIIP).

Enclosure 1 provides a description of the proposed change and confirmation of applicability. Enclosure 2 provides the existing TS pages marked-up to show the proposed change. Enclosure 3 provides the applicable TS Bases pages associated with the TS change. Enclosure 4 provides copies of TVA commitment letters.

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TVA proposed change is best implemented during a refueling outage. The next refueling outage for SQN Unit 2 is scheduled for November 2006. Accordingly, TVA requests NRC approval on a schedule to allow implementation of this TS to coincide with the Unit 2 outage. TVA requests that the implementation of the revised TS be during the Unit 2 Cycle 14 refueling outage.

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).

Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee State Department of Public Health.

There are no commitments contained in this submittal.

If you have any questions about this change, please contact me at 843-7170 or Jim Smith at 843-6672.

I declare under penalty of perjury that the foregoing is true and correct. Executed on this 15th day of February, 2006.

Sincerely,

for

P. L. Pace
Manager, Site Licensing
and Industry Affairs

Enclosures:
1. TVA Evaluation of the Proposed Changes
2. Proposed Technical Specifications Changes (mark-up)
3. Changes to Technical Specifications Bases Pages
4. TVA commitment Letters

cc: See page 3

Enclosures
cc (Enclosures):

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY (TVA)
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 2

**Description and Assessment**

## 1.0 INTRODUCTION

The proposed license amendment revises the requirements in the technical specification (TS) related to steam generator (SG) tube integrity. The proposed amendment is for Operating License DPR-79 for SQN Unit 2. The changes are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4. The availability of this TS improvement was announced in the *Federal Register* on May 2, 2005, as part of the consolidated line item improvement process (CLIIP).

In addition, TVA is proposing deletion of Unit 2 License Condition 2.C.8.b that is associated with SG inspection.

## 2.0 DESCRIPTION OF PROPOSED AMENDMENT

Consistent with NRC-approved Revision 4 of TSTF-449, the proposed TS changes include:

- Revised TS definition of "LEAKAGE"
- Revised TS 3.4.6.2, "Operational Leakage"
- Revised TS 3.4.5, "Steam Generator (SG) Tube Integrity"
- New TS Administrative Controls Section 6.8.4.k, "Steam Generator (SG) Program"
- New TS Administrative Controls Section 6.9.1.16, "Steam Generator Tube Inspection Report"

Proposed revisions to the TS Bases are also included in this application. 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 improvement. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program.

In addition to the above, TVA is deleting a Unit 2 license condition. The license condition is incorporated by the proposed TS change and is no longer carried as a stand alone requirement for SG inspection.

## 3.0 BACKGROUND

The background for this application is 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 background for TVA's proposed deletion of the SQN Unit 2 License Condition (2.C.8.b) is as follows:

Unit 2 License Condition (2.C.8.b)

The background for Unit 2 License Condition 2.C.8, Item b is contained in an NRC letter to TVA dated April 9, 1997. The April 1997 letter provides staff acceptance of commitments made by TVA for application of the voltage-based alternate repair criteria to the Unit 2 SGs. The basis for acceptance was compliance with NRC Generic Letter 95-05. The commitments made by TVA are described in TVA letters dated March 12, 1997, and March 17, 1997. A copy of these letters is provided in Enclosure 4. A comparison review of these commitments to the enclosed proposed TS for Unit 2 indicates that all aspects of the commitments are incorporated by the proposed TS. Accordingly, since the commitments of TVA's March 12, 1997, and March 17, 1997 letters are bounded by the proposed TS, TVA considers the proposed deletion of License Condition 2.C.8.b to be acceptable.

## 4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The applicable regulatory requirements and guidance 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.

## 5.0 TECHNICAL ANALYSIS

TVA has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR 10298), as part of the CLIIP Notice for Comment. This included the NRC staff's SE, the supporting information provided to support 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 staff are applicable to SQN Unit 2 and justify this amendment for the incorporation of the changes to the SQN TSs.

TVA has reviewed SQN Unit 2 License Condition 2.C.8, Item b that references TVA letters from 1997 that contain commitments associated with NRC Generic Letter 95-05 and the application of voltage-based alternate repair criteria to SQN Unit 2 steam generators. Based on TVA's review, the provisions and requirements of the enclosed TS change bound the TVA commitments. Therefore, TVA is proposing deletion of the SQN Unit 2 License Condition 2.C.8, Item b. This proposed change is administrative in nature and does not involve technical analysis.

## 6.0 REGULATORY ANALYSIS

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); the NRC Notice for Comment published on March 2, 2005 (70 FR 10298); and TSTF-449, Revision 4.

### 6.1 Verification and Commitments

The following information is provided to support the NRC staff's review of this amendment application:

| **Plant Name, Unit No.** | **Sequoyah, Unit 2** |
|---|---|
| SG Model(s): | Westinghouse Electric Company Model 51 |
| Effective Full Power Years (EFPY) of service for currently installed SGs | 16.4 EFPY (As of October 2005) |
| Tubing Material (e.g., 600M, 600TT, 660TT) | 600M |
| Number of tubes per SG | 3388 |
| Number and percentage of tubes plugged in each SG | SG 1: 77, 2.3%; SG 2: 171, 5.0%; SG 3: 126, 3.7%; SG 4: 123, 3.6% |
| Number of tubes repaired in each SG | No tubes - tube repair methods are not applicable |
| Degradation mechanism(s)identified | Primary water stress corrosion crack (PWSCC), outside diameter stress corrosion cracking (ODSCC), anti-vibration bar(AVB) wear, loose parts wear and cold-leg thinning |
| Current primary-to-secondary leakage limits | 150 gallons per day per SG and (600 gallons per day total leakage from 4 SGs) leak rates are evaluated at 70 degrees Fahrenheit. |
| Approved alternate tube repair criteria (ARC) | Voltage based ARC for ODSCC in the tube support plate as approved by NRC letter to TVA dated April 9, 1997, "Issuance of Technical Specification Amendments for the Sequoyah |

| Plant Name, Unit No. | Sequoyah, Unit 2 |
|---|---|
| | Nuclear Plant, Units 1 and 2 (TAC Nos. M96998 and M96999) (TS 96-05)"<br><br>W* criteria for SG Tubesheet Region WEXTEX Expansions as approved by NRC letter to TVA dated May 3, 2005, "Sequoyah Nuclear Plant, Unit 2 - Issuance of Amendment Regarding Changes to the Inspection Scope for the SG Tubes (TAC No. MC5212) (TS-03-06)"<br><br>The accident leakage limit approved for ODSCC ARC and for W* calculated leakage is 3.7 gallons per minute in the faulted SG. The structural performance criteria approved for ODSCC ARC is 1 x $10^{-2}$ probability of burst. No exceptions or clarifications to the structural performance criteria are applicable to W*.<br><br>No exceptions or clarifications to the structural criteria that apply to the ARC. |
| Approved SG tube repair methods | Not Applicable |
| Performance criteria for accident leakage | Primary-to-secondary leak rate values assumed in SQN's licensing basis accident analysis is 0.1 gallon per minute (gpm) for the non-faulted SGs and 3.7 gpm for the faulted SG (assumed at 70 degrees Fahrenheit temperature condition). The 3.7 gpm leakage limit is the approved plant analyses that is used as the leakage limit for ODSCC ARC and for W* calculated leakage. The accident-induced leakage for non-ARC application is conservatively limited by TVA to 1 gpm for the faulted SG. |

## 7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

TVA has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298), as part of the consolidated line item improvement process (CLIIP). TVA has concluded that the proposed determination presented in the notice is applicable to SQN and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

TVA's proposed deletion of the SQN Unit 2 License Condition is an administrative change that does not affect plant analysis or operation and does not: 1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or, 2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or 3) Involve a significant reduction in a margin of safety.

## 8.0 ENVIRONMENTAL EVALUATION

TVA has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298), as part of the CLIIP. TVA has concluded that the staff's findings presented in that evaluation are applicable to SQN and the evaluation is hereby incorporated by reference for this application.

## 9.0 PRECEDENT

This application is being made in accordance with the CLIIP. With the exception of the administrative deletion of SQN's Unit 2 License Condition, TVA is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4, or the NRC staff's model SE published on March 2, 2005 (70 FR 10298).

## 10.0 REFERENCES


1. *Federal Register* Notice - Notice for Comment published on March 2, 2005 (70 CFR 10298)

2. *Federal Register* Notice - Notice of Availability published on May 6, 2005 (70 FR 24126)

3. 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)

4. NRC letter to TVA dated May 3, 2005, Sequoyah Nuclear Plant, Unit 2 - Issuance of Amendment Regarding Changes to the Inspection Scope for the SG Tubes (TAC No. MC5212) (TS-03-06)

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 2

Proposed Technical Specification Changes (mark-up)

d. Failure to complete any tests included in the described program (planned or scheduled) for power levels up to the authorized power level.

(4) Monitoring Settlement Markers (SER/SSER Section 2.6.3)

TVA shall continue to monitor the settlement markers along the ERCW conduit for the new ERCW intake structure for a period not less than three years from the date of this license. Any settlement greater than 0.5 inches that occurs during this period will be evaluated by TVA and a report on this matter will be submitted to the NRC.

(5) Tornado Missiles (Section 3.5)

Prior to startup after the first refueling of the facility, TVA shall reconfirm to the satisfaction of the NRC that adequate tornado protection is provided for the 480 V transformer ventilation systems.

(6) Design of Seismic Category Structures (Section 3.8)

Prior to startup following the first refueling, TVA shall evaluate all seismic Category I masonry walls to final NRC criteria and implement NRC required modifications that are indicated by that evaluation.

(7) Low Temperature Overpressure Protection (Section 5.2.2)

Prior to startup after the first refueling, TVA shall install an overpressure mitigation system which meets NRC requirements.

(8) Steam Generator Inspection (Section 5.3.1)

(a) Prior to start-up after the first refueling, TVA shall install inspection ports in each steam generator or have an alternative for inspection that is acceptable to the NRC.

(b) By May 20, 1997, TVA shall establish a steam generator inspection program that is in accordance with the commitments listed in Enclosure 2 to the TVA letter to the Commission on this subject dated March 12, 1997, as modified by TVA letter dated March 17, 1997.

(9) Containment Isolation Systems (Section 6.2.4)

Prior to startup after the first refueling, TVA shall modify to the satisfaction of the NRC the one-inch chemical feed lines to the main and auxiliary feedwater lines for compliance with GDC 57.

(10) Environmental Qualification (Section 7.2.2)

a. No later than June 30, 1982, TVA shall be in compliance with the requirements of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," for safety-related equipment exposed to a harsh environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage, such as that from pump seals or valve packing (except reactor coolant pump seal injection or leakoff) that is captured and conducted to collection systems or a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

(primary to secondary leakage)

MEMBER(S) OF THE PUBLIC

1.17 DELETED

OFFSITE DOSE CALCULATION MANUAL

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, or component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

primary to secondary

PRESSURE BOUNDARY LEAKAGE

1.22 PRESSURE BOUNDARY LEAKAGE shall be leakage (except ~~steam generator tube~~ leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.23 The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the LTOP arming temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.1.15.

PROCESS CONTROL PROGRAM (PCP)

1.24 DELETED

PURGE - PURGING

1.25 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.26 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, which-ever is greater.

Remove Pages 3/4 4-10 through -16 and replace with INSERT A.



## 3/4.4.5 STEAM GENERATORS

### LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing $T_{avg}$ above 200°F.

### SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.5.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

4. Indications left in service as a result of application of the tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.

2. The inspections include those portions of the tubes where imperfections were previously found.

Note: Tube degradation identified in the portion of the tube that is not a reactor coolant pressure boundary (tube end up to the start of the tube-to-tubesheet weld) is excluded from the Result and Action Required in Table 4.4-2.

d. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate 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.

e. Implementation of the steam generator WEXTEX expanded region inspection methodology (W*) requires a 100 percent rotating coil probe inspection of the hot leg tubesheet W* distance.

The results of each sample inspection shall be classified into one of the following three categories:

| Category | Inspection Results |
|---|---|
| C-1 | Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective. |

C-2 One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.

C-3 More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

   2. A seismic occurrence greater than the Operating Basis Earthquake.

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

   4. A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness. Plugging limit does not apply to that portion of the tube that is not within the pressure boundary of the reactor coolant system (tube end up to the start of the tube-to-tubesheet weld). This definition does not apply to tube support plate intersections if the voltage-based repair criteria are being applied. Refer to 4.4.5.4.a.10 for the repair limit applicable to these intersections. This definition does not apply to service induced degradation identified in the W* distance. Service induced degradation identified in the W* distance below the top-of-tube sheet (TTS), shall be plugged on detection.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg excluding the portion of the tube within the tubesheet below the W* distance, the tube to tubesheet weld and the tube end extension.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

10. Tube Support Plate Plugging Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

    a. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to the lower voltage repair limit (Note 1), will be allowed to remain in service.

    b. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1), will be repaired or plugged, except as noted in 4.4.5.4.a.10.c below.

    c. Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion-cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (Note 1), but less than or equal to upper voltage repair limit (Note 2), may remain in service if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion-cracking degradation with a bobbin coil voltage greater than the upper voltage repair limit (Note 2) will be plugged or repaired.

    d. Not applicable to SQN.

    e. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

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

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

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

where:

| | | |
|---|---|---|
| $V_{URL}$ | = | upper voltage repair limit |
| $V_{LRL}$ | = | 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 steam generator 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 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

Note 1: The lower voltage repair limit is 1.0 volt for 3/4-inch diameter tubing or 2.0 volts for 7/8-inch diameter tubing.

Note 2: The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented. $V_{URL}$ may differ at the TSPs and flow distribution baffle.

11. a) Bottom of WEXTEX Transition (BWT) is the highest point of contact between the tube and tubesheet at, or below the top-of-tubesheet, as determined by eddy current testing.

    b) The W* distance is the larger of the following two distances as measured from the top-of-the-tubesheet (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.

    c) W* Length is the length of tubing below the bottom of the WEXTEX transition (BWT), which must be demonstrated to be non-degraded in order for the tube to maintain structural and leakage integrity. For the hot leg, the W* length is 7.0 inches which represents the most conservative hot-leg length defined in WCAP-14797, Revision 2.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2.

4.4.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported as a degraded condition pursuant to 10 CFR 50.73 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the staff prior to returning the steam generators to service should any of the following conditions arise:

1. Leakage is estimated based on the projected end-of-cycle (or if not practical using the actual measured end-of-cycle) voltage distribution. This leakage shall be combined with the postulated leakage resulting from the implementation of the W* criteria to tubesheet inspection depth. If the total projected end-of-cycle accident induced leakage from all sources exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle, the staff shall be notified.

2. If circumferential crack-like indications are detected at the tube support plate intersections.

3. If indications are identified that extend beyond the confines of the tube support plate.

4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1 X $10^{-2}$, notify the NRC and provide an assessment of the safety significance of the occurrence.

e. The calculated steam line break leakage from the application of tube support plate alternate repair criteria and W* inspection methodology shall be submitted in a Special Report in accordance with 10 CFR 50.4 within 90 days following return of the steam generators to service (MODE 4). 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 outside diameter), and an assessment of whether the results were consistent with expectations with respect to the number of flaws and flaw severity (and if not consistent, a description of the proposed corrective action).

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

| Preservice Inspection | No | | | Yes | | |
|---|---|---|---|---|---|---|
| No. of Steam Generators per Unit | Two | Three | Four | Two | Three | Four |
| First Inservice Inspection | All | | | One | Two | Two |
| Second & Subsequent Inservice Inspections | One[1] | | | One[1] | One[2] | One[3] |

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.

3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

## TABLE 4.4-2

## STEAM GENERATOR TUBE INSPECTION

| 1ST SAMPLE INSPECTION | | | 2ND SAMPLE INSPECTION | | 3RD SAMPLE INSPECTION | |
|---|---|---|---|---|---|---|
| Sample Size | Result | Action Required | Result | Action Required | Result | Action Required |
| A minimum of S Tubes per S.G. | C-1 | None | N/A | N/A | N/A | N/A |
| | C-2 | Plug defective tubes and inspect additional 2S tubes in this S.G. | C-1 | None | N/A | N/A |
| | | | C-2 | Plug defective tubes and inspect additional 4S tubes in this S.G. | C-1 | None |
| | | | | | C-2 | Plug defective tubes |
| | | | | | C-3 | Perform action for C-3 result of first sample |
| | | | C-3 | Perform action for C-3 result of first sample | N/A | N/A |
| | C-3 | Inspect all tubes in this S.G. plug defective tubes and inspect 2S tubes in each other S.G. | All other S.G are C-1 | None | N/A | N/A |
| | | | Some S/Gs C-2 but no additional S.G. are C-3 | Perform action for C-2 result of second sample | N/A | N/A |
| | | | Additional S/G is C-3 | Inspect all tubes in each S.G. and plug defective tubes. | N/A | N/A |

$S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

## INSERT A

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained.

AND

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

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS*:

a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program, within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

AND

b. Plug the affected tube(s) in accordance with the Steam Generator Program prior to startup following the next refueling outage or SG tube inspection.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Verify steam generator tube integrity in accordance with the Steam Generator Program.

4.4.5.1 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to startup following a SG tube inspection.

* Separate Action entry is allowed for each SG tube.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day of primary-to-secondary leakage through any one steam generator, and

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4

or with primary-to-secondary leakage not within limits,

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

or primary-to-secondary leakage

Verify

is within

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages ~~shall be verified to be within each of the above~~ limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours.*

The provision of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

4.4.6.2.2 ~~Verify steam generator tube integrity is in accordance with the requirements of Technical Specification 3/4.4.5, "Steam Generators."~~

Verify primary-to-secondary leakage is ≤150 gallons per day through any one steam generator at least once per 72 hours.*

The above surveillance requirement is not applicable to primary-to-secondary leakage.

* Not required to be performed until 12 hours after establishment of steady state operation.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq 6$ psig for at least two minutes.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

i. Configuration Risk Management Program (DELETED)

j. Technical Specification (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these TSs.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

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).

INSERT B

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted in accordance with 10 CFR 50.4.

STARTUP REPORT

6.9.1.1 DELETED

6.9.1.2 DELETED

6.9.1.3 DELETED

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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 and/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 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, except for outside diameter stress corrosion crack (ODSCC) and W* indications that have an approved limit of 3.7 gallons per minute (gpm). 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 of 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:

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GL 95-05 Voltage-Based ARC (Tube Support Plate [TSP])

A voltage-based TSP plugging limit 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 plugging (repair) limit is based on maintaining SG tube serviceability as 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 the lower voltage repair limit (Note 1), 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 the lower voltage repair limit (Note 1), will be repaired or plugged, except as noted in Item 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 the lower voltage repair limit (Note 1), but less than or equal to upper voltage repair limit (Note 2), may remain in service if a rotating pancake coil inspection does not detect degradation. SG tubes with indications of ODSCC degradation with a bobbin coil voltage greater than the upper voltage repair limit (Note 2) will be plugged or repaired.

d) Not applicable to SQN.

e) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in Items a, b, and c.

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

$V_{LRL}$ = 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

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| | | |
|---|---|---|
| $\Delta t$ | = | 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 length | = | average growth rate per cycle |
| 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 a, b, and c.

Note 1: The lower voltage repair limit is 1.0 volt for 3/4 inch diameter tubing or 2.0 volts for 7/8 inch diameter tubing.

Note 2: The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented. $V_{URL}$ may differ at the TSPs and flow distribution baffle.

The accident leakage limit approved for ODSCC ARC and for W* calculated leakage is 3.7 gallons per minute in the faulted SG.

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 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.
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

## INSERT B

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.

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 during all future refueling outages.

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.

W* Methodology

Implementation of the SG WEXTEX expanded region inspection methodology (W*) requires a 100 percent rotating coil probe inspection of the hot-leg tubesheet W* distance. The implementation of W* does not apply to service induced degradation identified in the W* distance. Service induced degradation identified in the W* distance below the top-of-tubesheet (TTS) shall be plugged on detection. The inspection of SG tubes is from the point of entry (hot-leg side) completely around the U-bend to the top support of the cold leg excluding the portion of the tube within the tubesheet below the W* distance, the tube-to-tubesheet weld and the tube end extension.

The following terms/definitions apply to the W*.

a) Bottom of WEXTEX Transition (BWT) is the highest point of contact between the tube and tubesheet at, or below the TTS, as determined by eddy current testing.

b) W* Distance 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.

c) W* Length is the length of tubing below the bottom of the BWT which must be demonstrated to be non-degraded in order for the tube to maintain structural and leakage integrity. For the hot leg, the W* length is 7.0 inches which represents the most conservative hot leg length defined in WCAP-14797, Revision 2.

The postulated leakage resulting from the implementation of the voltage-based repair criteria to TSP intersections shall be combined with the postulated leakage resulting from the implementation of W* criteria to tubesheet inspection depth.

e. Provisions for monitoring operational primary-to-secondary leakage.

CORE OPERATING LIMITS REPORT (continued)

6. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985, (W Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

7. WCAP-10266-P-A, Rev. 2, "THE 1981 REVISION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).
   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

8. BAW-10227P-A, "Evaluation of Advance Cladding and Structural Material (M5) in PWR Reactor Fuel," February 2000, (FCF Proprietary)
   (Methodology for Specification 3/4.2.2 - Heat Flux Hot Channel Factor)

6.9.1.14.b The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.14.c THE CORE OPERATING LIMITS REPORT shall be provided within 30 days after cycle start-up (Mode 2) for each reload cycle or within 30 days of issuance of any midcycle revision of the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS (PTLR) REPORT

6.9.1.15 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, LTOP arming, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

Specification 3.4.9.1, "RCS Pressure and Temperature (P/T) Limits"

Specification 3.4.12, "Low Temperature Over Pressure Protection (LTOP) System"

6.9.1.15.a The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. Westinghouse Topical Report WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

2. Westinghouse Topical Report WCAP-15321, "Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation."

3. Westinghouse Topical Report WCAP-15984, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2."

6.9.1.15.b The PTLR shall be provided to the NRC within 30 days of issuance of any revision or supplement thereto.

SPECIAL REPORTS

INSERT C

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.

## INSERT C

STEAM GENERATOR (SG) TUBE INSPECTION REPORT

6.9.1.16 A report shall be submitted within 180 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 shall include:

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.

i. For implementation of the voltage-based repair criteria to tube support plate (TSP) intersections, notify the staff prior to returning the SGs to service should any of the following conditions arise:

   1) Leakage is estimated based on the projected end-of-cycle (or if not practical using the actual measured end-of-cycle) voltage distribution. This leakage shall be combined with the postulated leakage resulting from the implementation of the W* criteria to tubesheet inspection depth. If the total projected end-of-cycle accident induced leakage from all sources exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle, the staff shall be notified.

   2) If circumferential crack-like indications are detected at the TSP intersections.

   3) If indications are identified that extend beyond the confines of the TSP.

   4) If indications are identified at the TSP elevations that are attributable to primary water stress corrosion cracking.

   5) If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1 X $10^{-2}$, notify the NRC and provide an assessment of the safety significance of the occurrence.

j. 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 in a Special Report in accordance with 10 CFR 50.4 within 90 days following return of the SGs to service (MODE 4). 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,

**INSERT C**

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 outside diameter), and an assessment of whether the results were consistent with expectations with respect to the number of flaws and flaw severity (and if not consistent, a description of the proposed corrective action).

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 2

Changes to Technical Specifications Bases Pages

INSERT D



## 3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Sequoyah has demonstrated that primary-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown or condenser off-gas. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

The voltage-based repair limits of SR 4.4.5 implement the guidance in GL 95-05 and are applicable only to Westinghouse-designed steam generators (S/Gs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of S/G tube degradation nor are they applicable to ODSCC that occurs at other locations within the S/G. 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 SR 4.4.5 requires 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 LTL 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 SR 4.4.5.4.a.10.e should only be used during unplanned inspection in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements recommended by GL 95-05 for situations which NRC wants to be notified prior to returning the S/Gs to service. For SR 4.4.5.5.d, Items 3 and 4, indications 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 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 S/Gs 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.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the repair limit defined in Surveillance Requirement 4.4.5.4.a. The portion of the tube that the plugging limit does not apply to is the portion of the tube that is not within the RCS pressure boundary (tube end up to the start of the tube-to-tubesheet weld). The tube end to tube-to-tubesheet weld portion of the tube does not affect structural integrity of the steam generator tubes and therefore indications found in this portion of the tube will be excluded from the Result and Action Required for tube inspections. It is expected that any indications that extend from this region will be detected during the scheduled tube inspections. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Tubes experiencing outside diameter stress corrosion cracking within the thickness of the tube support plate are plugged or repaired by the criteria of 4.4.5.4.a.10.

The W* criteria incorporate 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 the tubesheet to the bottom of the W* length including the distance from the top of the tubesheet to the BWT and measurement uncertainties.

Indications detected within the W* distance below the top-of-tube sheet (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, including a complete circumferential separation of the tube, is acceptable. As applied at Sequoyah Nuclear Plant Unit 2, the W* methodology is used to define the required tube inspection depth into the hot-leg tubesheet, and is not used to permit degradation in the W* distance to remain in service. Thus while primary to secondary leakage in the W* distance need not be postulated, primary to secondary leakage from potential degradation below the W* distance will be assumed for every inservice tube in the bounding steam generator.

The postulated leakage during a steam line break shall be equal to the following equation:

Postulated SLB Leakage = $\text{ARC}_{\text{GL 95-05}}$ + Assumed Leakage $_{0''\text{-}8'' < \text{TTS}}$ + Assumed Leakage $_{8''\text{-}12'' < \text{TTS}}$ + Assumed Leakage $_{>12'' < \text{TTS}}$

Where: $\text{ARC}_{\text{GL 95-05}}$ is the normal SLB leakage derived from alternate repair criteria methods and the steam generator tube inspections.

Assumed Leakage $_{0''\text{-}8'' < \text{TTS}}$ is the postulated leakage for undetected indications in steam generator tubes left in service between 0 and 8 inches below the top of the tubesheet.

Assumed Leakage $_{8\text{-}12'' < \text{TTS}}$ is the conservatively assumed leakage from the total of identified and postulated unidentified indications in steam generator tubes left in service between 8 and 12 inches below the top of the tubesheet. This is 0.0045 gpm multiplied by the number of indications. Postulated unidentified indications will be conservatively assumed to be in one steam generator. The highest number of identified indications left in service between 8 and 12 inches below TTS in any one steam generator will be included in this term.

Assumed Leakage $_{>12'' < \text{TTS}}$ is the conservatively assumed leakage for the bounding steam generator tubes left in service below 12 inches below the top of the tubesheet. This is 0.00009 gpm multiplied by the number of tubes left in service in the least plugged steam generator.

The aggregate calculated SLB leakage from the application of all alternate repair criteria and the above assumed leakage shall be reported to the NRC in accordance with applicable Technical Specifications. The combined calculated leak rate from all alternate repair criteria must be less than the maximum allowable steam line break leak rate limit in any one steam generator in order to maintain doses within 10 CFR 100 guideline values and within GDC-19 values during a postulated steam line break event.

B 3.4 REACTOR COOLANT SYSTEM

B 3/4.4.5 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary 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 Limiting Condition of Operation (LCO) 3.4.1.1, "Startup and Power Operation," LCO 3.4.1.2, "Hot Standby," LCO 3.4.1.3, "Shutdown," and LCO 3.4.1.4, "Cold Shutdown."

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 wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and 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 6.8.4.k, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.k, 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 6.8.4.k. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

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

BASES

APPLICABLE SAFETY ANALYSES

The steam generator tube 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 is based on a primary to secondary leakage of 0.1 gallons per minute (gpm) for the non-faulted SGs and 3.7 gpm for the faulted SG. This limit is approved for use for alternate repair criteria (ARC) and W* leakage calculations. For non-ARC applications, the accident induced leakage in the faulted SG is limited to 1.0 gpm, which is bounded by the maximum leakage established by the plant safety analysis. 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).

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.

INSERT D

BASES

LCO (continued)

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 (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).

BASES

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. In the main steam line break (MSLB) analysis for ARC, SG leakage is assumed to be 3.7 gpm for the faulted SG and 0.1 gpm for the non-faulted SGs. Limiting the allowable leakage in the faulted SG to 1.0 gpm for non-ARC applications ensures that the MSLB analysis remains conservative and bounding. 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 3.7 gpm is approved for use in ARC applications where the cracks are limited to locations within the tubesheet or within a drilled tube support plate.

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 MSLB. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in 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.

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.

INSERT D



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 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. 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 following the next refueling outage or SG inspection.

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 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.

BASES

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.

INSERT E

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."

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Application of 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 (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.c 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 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.i, Items 3 and 4, indications 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)

## INSERT E

criteria.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice SG tube examinations. Plugging will be required for all tubes with imperfections exceeding the repair limit defined in Specification 6.8.4.k.c. The portion of the tube that the plugging limit does not apply to is the portion of the tube that is not within the RCS pressure boundary (tube end up to the start of the tube-to-tubesheet weld). The tube end tube-to-tubesheet weld portion of the tube does not affect structural integrity of the SG tubes and therefore indications found in this portion of the tube will be excluded from the "Result and Action Required" for tube inspections. It is expected that any indications that extend from this region will be detected during the scheduled tube inspections. SG tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

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

b) <u>W* Methodology</u>

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, including a complete circumferential separation of the tube, is acceptable. As applied at Sequoyah Nuclear Plant Unit 2, the W* methodology is used to define the required tube inspection depth into the hot-leg tubesheet, and is not used to permit degradation in the W* distance to remain in service. Thus while primary to secondary leakage in the W* distance need not be postulated, primary to secondary leakage from potential degradation below the W* distance will be assumed for every inservice tube in the bounding SG.

c) <u>Calculation of Accident Leakage</u>

The postulated leakage during a steam line break (SLB) shall be equal to the following equation:

Postulated SLB Leakage = ARC $_{\text{GL 95-05}}$ + Assumed Leakage $_{\text{0"-8" <TTS}}$ + Assumed Leakage $_{\text{8"-12" <TTS}}$ + Assumed Leakage $_{\text{>12" <TTS}}$

## INSERT E

Where: $ARC_{GL\ 95\text{-}05}$ is the normal SLB leakage derived from ARC methods and the SG tube inspections.

Assumed Leakage $_{0''\text{-}8''<TTS}$ is the postulated leakage for undetected indications in SG tubes left in service between 0 and 8 inches below the TTS.

Assumed Leakage $_{8''\text{-}12''<TTS}$ is the conservatively assumed leakage from the total of identified and postulated unidentified indications in SG tubes left in service between 8 and 12 inches below the TTS. This is 0.0045 gpm multiplied by the number of indications. Postulated unidentified indications will be conservatively assumed to be in one SG. The highest number of identified indications left in service between 8 and 12 inches below TTS in any one SG will be included in this term.

Assumed Leakage $_{>12''<TTS}$ is the conservatively assumed leakage for the bounding SG tubes left in service below 12 inches below the TTS. This is 0.00009 gpm multiplied by the number of tubes left in service in the least plugged SG.

The aggregate calculated SLB leakage from the application of all ARC and the above assumed leakage shall be reported to the NRC in accordance with applicable technical specifications. The combined calculated leak rate from all ARC must be less than the maximum allowable SLB leak rate limit in any one SG in order to maintain doses within 10 CFR 100 guideline values and within GDC-19 values during a postulated SLB event.

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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)

## 3/4.4.6.2 OPERATIONAL LEAKAGE

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified leakage is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for ~~an event~~ events resulting in steam discharge to the atmosphere ~~assumes a 1 gpm primary to secondary leakage as the initial condition.~~ account for a maximum normal operational leakage of 0.4 gpm (0.1 gpm per steam generator).

steam generator tube rupture or a

also

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 rupture (SGTR)~~. The leakage contaminates the secondary fluid.

from all four SGs

0.4 gpm operational

The FSAR (Ref. 3) analysis for 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 ~~1 gpm~~ primary to secondary leakage is relatively inconsequential.

ARC

,with ARC applied leakage,

through the affected

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes ~~1~~ gpm primary to secondary leakage ~~in one~~ generator 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 ~~8.21~~ 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 ~~8.21~~ 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 ~~8.21~~ gpm.

a maximum 3.7

3.7

and 0.3 gpm through the non-affected generators

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

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 sump level monitoring equipment can collectively detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

c. Primary to Secondary Leakage through Any One Steam Generator (SG)

INSERT G

~~The 150 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.~~

~~The 150-gallons per day limit incorporated into Surveillance 4.4.6.2.1 is more restrictive than the standard operating leakage limit and is intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that, should a significant leak be experienced, it will be detected, and the plant shut down in a timely manner.~~

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the RCS Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for reactor coolant PRESSURE BOUNDARY LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

LCO 3/4.4.6.3, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

ACTIONS

Action a:

or with primary to secondary leakage not within limits,

If any PRESSURE BOUNDARY LEAKAGE exists, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within the following 30 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

or

Action b:

UNIDENTIFIED LEAKAGE~~,~~ IDENTIFIED LEAKAGE, ~~or primary-to-secondary leakage~~ in excess of the LCO limits must be reduced to within limits within 4 hours. This completion time allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB. If UNIDENTIFIED LEAKAGE~~,~~ IDENTIFIED LEAKAGE~~, or primary-to-secondary leakage~~ cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within the following 30 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

Surveillance 4.4.6.2.1

Verifying RCS leakage to be within the LCO limits ensures the integrity of the RCPB is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of an RCS water inventory balance. ~~Primary-to-secondary leakage is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

The

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup, letdown, and RCP seal injection and return flows). ~~Therefore,~~ a footnote ~~is added allowing~~ that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. Performance of this surveillance within the 12-hour allowance is required to maintain compliance with the provisions of Specification 4.0.3.

The surveillance is modified by a footnote.

states

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment pocket sump level. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3/4.4.6.1, "Leakage Detection Instrumentation."

INSERT H

The 72 hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

Surveillance 4.4.6.2.2

INSERT I

~~This surveillance provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this surveillance cannot be performed at normal operating conditions.~~

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.4.3.
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

## INSERT G

The limit of 150 gallons per day per SG is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion, in conjunction with the implementation of the Steam Generator Program, is an effective measure for minimizing the frequency of SG tube ruptures.

## INSERT H

Notation associated with this SR states that this SR is not applicable to primary to secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

## INSERT I

This SR verifies that primary to secondary leakage is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.5, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at 70 degrees Fahrenheit (Reference 5). The operational leakage rate limit applies to leakage through any one SG. If it is not practical to assign the leakage to an individual SG, all the primary-to-secondary leakage should be conservatively assumed to be from one SG.

The surveillance is modified by a note which states that the surveillance is not required to be performed until 12 hours after establishment of steady state operation. For RCS primary-to-secondary leakage determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The surveillance frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

ENCLOSURE 4

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNIT 2

TVA Commitment Letters Dated March 12, 1997 and March 17, 1997

TVA

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379-2000

March 12, 1997

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Gentlemen:

| | | |
|---|---|---|
| In the Matter of | ) | Docket No. 50-327 |
| Tennessee Valley Authority | ) | 50-328 |

SEQUOYAH NUCLEAR PLANT (SQN) - NRC REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF TECHNICAL SPECIFICATION CHANGE 96-05 REGARDING VOLTAGE-BASED ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBES SEQUOYAH UNITS 1 AND 2

Reference:

1. NRC letter to TVA dated February 19, 1997, "Request for Additional Information - Technical Specification change Request TS 96-05 for Sequoyah Nuclear Plant Units 1 and 2 (TAC NOS. M96998 and M96999)
2. TVA letter to NRC dated October 18, 1996, "Sequoyah Nuclear Plant (SQN) - Technical Specification (TS) Change 96-05, 'Elimination of Cycle 8 Limitation For Steam Generator (S/G) Alternate Plugging Criteria (APC)'"

Enclosed is TVA's response to NRC's request for additional information (reference 1) on the above subject. The response is associated with SQN's proposed TS Change 96-05 (reference 2) that implements steam generator alternate plugging criteria (APC).

Enclosure 1 provides the requested information. Enclosure 2 provides the TVA commitments.

Please direct questions concerning this issue to Don Goodin at (423) 843-7734.

Sincerely,

R. H. Shell

R. H. Shell
Site Licensing and Industry Affairs Manager

cc: R. W. Hernan, Senior Project Manager
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ENCLOSURE 1

TVA Responses to NRC Request for Information

Item 1: NRC Request: TVA referenced the Westinghouse report, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," WCAP-14277, as a source for performing calculation for burst probability, end of cycle voltage distribution, and leak rate. Westinghouse published revision 1 of WCAP-14277 in December 1996 because the staff did not find the original report to be acceptable for referencing in safety evaluations. TVA needs to update its commitment to use Revision 1 of the report.

TVA Response: TVA will utilize Revision 1 of WCAP-14277 for performing calculations for burst probability, end of cycle voltage distribution, and leak rate.

Item 2: NRC Request: TVA needs to update its commitment to the latest database it intends to use in performing the calculations specified in Generic Letter (GL) 95-05 for the upcoming Unit 1 steam generator inspection. On a permanent basis, TVA needs to commit to the protocols for the use of the NRC-approved steam generator database.

TVA Response: TVA is committing to use the latest NRC approved database for the Cycle 8 and all future steam generator inspections. TVA is cognizant of the Request For Additional Information regarding NP 7480-L, Addendum 1, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking At Tube Support Plates, Database For Alternate Repair Limits," 1996 Database Update, November 1996, in Stewart L. Magruder's letter to David Modeen, Nuclear Energy Institute dated January 24, 1997. If this database is approved by NRC, before the start of the Unit 1 Cycle 8 Steam Generator inspection, TVA will utilize it for burst and leakage calculations. If this database is not approved, TVA will utilize the one previously approved by the staff in the April 6, 1996 Safety Evaluation Report for SQN TS change 95-23 including data from any additional pulled tubes in accordance with exclusion criteria protocol in GL 95-05. TVA will follow the industry database protocol when agreement is reached with the staff.

Item 3: NRC Request: TVA submitted plans for inspecting dents at tube support plates for both units in its previous TS amendment requests. However, for the current amendment request, TVA needs to clarify its dent inspection criteria in light of revisions to the inspection criteria that may be needed due to the results of inspecting dents less than 5 volts, either in the past or in the future.

TVA Response: TVA has reviewed Unit 1 Cycle 7 refueling outage data and recent industry information on dented tube support plate inspections in dented TSP intersections less than 5 volts. TVA's inspection plans for Unit 1 are identified in Attachment 1. TVA's dent inspection plan for Unit 2 will continue to follow the guidance of Section 3.b.3 of Attachment 1 to GL 95-05.

Item 4: NRC Request: TVA committed to various industry criteria for probe wear and variability in previous amendment requests; however, there have been new probe wear and variability criteria developed since 1995. Therefore, TVA needs to update its commitment to comply with the criteria proposed, finalized, and agreed upon in the following letters:

(1) Nuclear Energy Institute (NEI) letter to NRC, subject: "Eddy current probe replacement Criteria for Use in ODSCC Alternate Repair Criteria," January 23, 1996;

(2) NEI letter to NRC, subject: "New Probe Variability for Use in the SCC Alternate Repair Criteria," January 23, 1996;

(3) NEI letter to NRC subject: "Eddy Current Probe Replacement Criteria for Use in ODSCC Alternate Repair Criteria (Project No. 689), " February 23, 1996;

(4) NRC letter from B. Sheron to A. Marion of NEI dated February 9, 1996; and

(5) NRC letter from B. Sheron of NRC to A. Marion of NEI dated March 18, 1996.

TVA Response: TVA will comply with the criteria proposed, finalized and agreed upon in the aforementioned letters. In addition, TVA will comply with the probe variability criteria in the NEI letter to NRC dated October 15, 1996, "Response to NRC letter Dated February 9, 1996, Regarding New Probe Variability Criteria (Project 689)."

Item 5: NRC Request: TVA has incorporated the model TSs specified in GL 95-05 into the existing Units 1 and 2 technical specifications and has committed to certain sections in Attachment 1 of GL 95-05. To clarify, TVA needs to commit to comply with GL 95-05 in its entirety. Alternatively, TVA needs to provide exceptions to GL 95-05, should there be any.

TVA Response: TVA commits to comply with the sections in Attachment 1 of GL 95-05 with the following exceptions;

2.a.3 SQN steam generators do not contain flow distribution baffle plates.

3.b.3 SQN Unit 1 takes exception to inspecting all dented TSP intersections and proposes Attachment 1 as an alternative. SQN Unit 2 will comply with the requirements of this section.

3.c.2 TVA will comply with probe variability as defined in letters referenced in item 3 of this response.

3.c.3 TVA will comply with probe wear as defined in the letters referenced in item 3 of this response.

Unit 1 Dent Sampling Plan for dents greater than or equal to 5 volts

The initial sample in S/Gs 1 and 2 shall be 100 percent of the total hot-leg (HL) dented tube support plate (TSP) population in S/Gs 1 and 2.

The initial sample in S/Gs 3 and 4 will be 20 percent of the total HL dented TSP population in S/Gs 3 and 4.

The dent examinations will be performed with a technique qualified to Appendix H of the Electric Power Research Institute (EPRI) Steam Generator Examination Guidelines. An RPC inspection will be performed. Alternate probes, that have demonstrated detection capability for axial and circumferential indications comparable to or better than the RPC probes, can be used for these inspections. RPC is used as a general term to reflect an acceptable technique.

The dented TSP intersections selection for S/Gs 3 and 4 will begin at the lowest HL TSP elevations, which has the highest probability that stress corrosion cracking will occur. The initial sample will be 20 percent of the total HL dents in the respective S/G and systematically distributed at the first HL TSP.

If the RPC inspection of dented intersections identifies circumferential ODSCC or PWSCC indications not detected by bobbin, the RPC inspection shall be expanded consistent with Table 1. Any indications identified that exceed the plugging limit shall be repaired. The result classification as defined in TS Section 4.4.5.2 shall be utilized.

Expansion samples would be selected from the lowest HL dented TSP intersections and continue to higher TSP elevations.

The dent inspection frequency shall be performed coinciding with the S/G surveillance requirements. If an unscheduled mid-cycle S/G surveillance is required, the dented TSP inspection shall be performed.

Table 1 : SQN Unit 1 SGs 3 and 4 Expansion of the greater than or equal to 5 volt HL dented TSP Sample

| Initial Sample | | First Expansion | | Second Expansion | |
|---|---|---|---|---|---|
| **Result** | **Action Required** | **Result** | **Action Required** | **Result** | **Action Required** |
| C-1 | None | N/A | N/A | N/A | N/A |
| C-2 | Inspect an additional 20% sample of TSP intersections in this SG | C-1 | None | N/A | N/A |
| | | C-2 | Inspect an additional 20% sample of TSP intersections in this SG | C-1 | None |
| | | | | C-2 | Inspect all remaining TSP intersections in this SG |
| | | | | C-3 | Inspect all remaining TSP intersections in this SG and a 20% sample in other SGs |
| | | C-3 | Inspect all remaining TSP intersections in this SG and a 20% sample in other SGs | N/A | N/A |
| C-3 | Inspect all remaining TSP intersections in this SG and a 20% sample in other SGs | C-1 in other SG | None | N/A | N/A |
| | | C-2 but not C-3 in other SG | Inspect an additional 20% sample of TSP intersections in other SG | N/A | N/A |
| | | C-3 in other SG | Inspect all remaining TSP intersections in other SGs | N/A | N/A |

TSP = dented hot-leg tube support plate

Unit 1 Dent Sampling Plan for dents less than 5 volts;

TVA will sample with RPC in a SG all dents less than 5 volts at all TSP elevations (and lower TSPs) where, based on past inspections, degradation has occurred (defining a critical area) and perform a 20% sample of the next higher TSP elevation (a buffer zone) to bound the affected area. The buffer zone, in this application, is the next higher tube support plate elevation where no degradation has been observed. This buffer zone area is to ensure that the critical area is bounded. The degradation (circumferential ODSCC or PWSCC not detected by bobbin coil) identified from the past dented TSP inspection would determine the initial sample.

Each SG initial sample will be determined independently. If no degradation was identified in the past inspection, a minimum 20% sample of the dents (less than 5 volts) at the first TSP will be examined. During future outages a different 20% sample would be inspected, such that over five outages 100% of the dents at this elevation would be inspected.

If indications are identified in the buffer zone, this sample will be expanded in accordance with Table 2. Any indication identified that exceeds the plugging limit shall be repaired. The buffer zone result classification as defined in TS Section 4.4.5.2 shall be utilized, except when a sample size is less than 200, then only C-2 results apply.

Alternative Dented TSP Inspection Program (greater than or equal to 5 volts);

TVA proposes an alternative inspection program for SGs 3 and 4, for the greater than 5 volt dents which is the same methodology as the proposed program for less than 5 volt dented tube support plate inspection with one additional requirement. If a TSP elevation has less than 50 dented intersections when selecting a buffer zone, then additional intersections at the next higher elevation shall be inspected to make the total number of intersections to be inspected equal to 50. TVA would like the option to employ either method to the greater than or equal to 5 volt dent population.

Table 2 : SQN Unit 1 Expansion of the greater than or equal to and less than 5 volt HL dented TSP Sample

| Initial Sample | | First Expansion | | Second Expansion | |
|---|---|---|---|---|---|
| **Result** | **Action Required** | **Result** | **Action Required** | **Result** | **Action Required** |
| C-1 Buffer Zone | None | N/A | N/A | N/A | N/A |
| C-2 Buffer Zone | Inspect all remaining TSP intersections at this elevation and a 20% Buffer Zone of the next elevation | C-1 Buffer Zone | None | N/A | N/A |
| | | C-2 Buffer Zone | Inspect all remaining TSP intersections at this elevation and a 20% of the next elevation | C-1 Buffer Zone | None |
| | | | | C-2 Buffer Zone | Inspect all remaining TSP intersections in this SG |
| | | | | C-3 Buffer Zone | Inspect all remaining TSP intersections in this SG and an additional 20% sample of the lowest TSP not yet 100% inspected in other SGs |
| | | C-3 Buffer Zone | Inspect all remaining TSP intersections at this elevation and a 100% of the next elevation | N/A | N/A |
| C-3 Buffer Zone | Inspect all remaining TSP intersections at this elevation and a 100% Buffer Zone of the next elevation | C-1 Buffer Zone | None | C-1 Buffer Zone | None |
| | | C-2 Buffer Zone | Inspect a 20% Buffer Zone of the next elevation | C-2 Buffer Zone | Inspect all remaining TSP intersections in this SG |
| | | C-3 Buffer Zone | Inspect all remaining TSP intersections in this SG and an additional 20% sample of the lowest TSP not yet 100% inspected in other SGs | C-3 Buffer Zone | Inspect all remaining TSP intersections in this SG and an additional 20% sample of the lowest TSP not yet 100% inspected in other SGs |

TSP = dented hot-leg tube support plate

Enclosure 2

TVA Commitments

TVA will revise SQN's steam generator inspection program (0-SI-SXI-068-114.2) prior to unit restart from the Unit 1 Cycle 8 Refueling outage. The program will be revised to:

1) utilize revision 1 of WCAP-14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections."

2) utilize the database previously approved by the staff in April 6, 1996 Safety Evaluation Report for SQN TS Change 95-23 including data from any additional pulled tubes in accordance with exclusion criteria protocol in GL 95-05. TVA will follow the industry protocol when agreement is reached with the staff.

3) for Unit 1, adopt the inspection plans contained in Attachment 1 of Enclosure 1 of this letter for dents less than 5 volts and greater than or equal to 5 volts.

4) comply with the probe wear and probe variability criteria contained in the following letters:

   (a) Nuclear Energy Institute (NEI) letter to NRC dated January 23, 1996, "Eddy current probe replacement Criteria for Use in ODSCC Alternate Repair Criteria."

   (b) NEI letter to NRC dated January 23, 1996, "New Probe Variability for Use in the SCC Alternate Repair Criteria."

   (c) NEI letter to NRC dated February 23, 1996, "Eddy Current Probe Replacement Criteria for Use in ODSCC Alternate Repair Criteria (Project No. 689)."

   (d) NRC letter dated February 9, 1996, from B. Sheron of NRC to A. Marion of NEI.

   (e) NRC letter dated March 18, 1996, from B. Sheron of NRC to A. Marion of NEI.

   (f) NEI letter to NRC dated October 15, 1996, entitled, "Response to NRC letter dated February 9, 1996, Regarding New Probe Variability Criteria (Project 689)."

Enclosure 2

TVA commitments (continued)

5) comply with the sections in Attachment 1 of GL 95-05 with the following exceptions:

2.a.3 SQN steam generators do not contain flow distribution baffle plates.

3.b.3 SQN Unit 1 takes exception to inspecting all dented TSP intersections and proposes Attachment 1 of Enclosure 1 of this letter as an alternative.

3.c.2 TVA will comply with probe variability as defined in letters referenced in item 3 of this response.

3.c.3 TVA will comply with probe wear as defined in the letters referenced in item 3 of this response.

Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37379-2000

March 17, 1997

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Gentlemen:

| | | |
|---|---|---|
| In the Matter of | ) | Docket No. 50-327 |
| Tennessee Valley Authority | ) | 50-328 |

SEQUOYAH NUCLEAR PLANT (SQN) - NRC REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF TECHNICAL SPECIFICATION CHANGE 96-05 REGARDING VOLTAGE-BASED ALTERNATE REPAIR CRITERIA FOR STEAM GENERATOR TUBES SEQUOYAH UNITS 1 AND 2

Reference: TVA letter to NRC dated March 12, 1997, "Sequoyah Nuclear Plant (SQN) - NRC Request for Additional Information-Review of Technical Specification Change 96-05 Regarding Voltage-Based Alternate Repair Criteria for Steam Generator Tubes Sequoyah Units 1 and 2."

In response to NRC questions from a teleconference on March 17, 1997, TVA is providing a clarification to page 6 of enclosure 1 from the referenced letter. The clarification revises the language from "If a TSP elevation has less than 50 dented intersections" to "If a TSP elevation has less than 250 dented intersections." This change ensures that a 20 percent expansion of the buffer zone would equate to a sample size of 50 dented TSP intersections.

Enclosed is the revised page 6. This page supersedes the page 6 previously provided in the referenced letter.

Please direct questions concerning this issue to Don Goodin at (423) 843-7734.

Sincerely,

R. H. Shell

R. H. Shell
Site Licensing and Industry Affairs Manager

cc: R. W. Hernan, Senior Project Manager
Nuclear Regulatory Commission
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11555 Rockville Pike
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101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323-2711

Enclosure

Sequoyah Nuclear Plant

Revised Page 6

to TVA letter dated March 12, 1997.

Unit 1 Dent Sampling Plan for dents less than 5 volts;

TVA will sample with RPC in a SG all dents less than 5 volts at all TSP elevations (and lower TSPs) where, based on past inspections, degradation has occurred (defining a critical area) and perform a 20% sample of the next higher TSP elevation (a buffer zone) to bound the affected area. The buffer zone, in this application, is the next higher tube support plate elevation where no degradation has been observed. This buffer zone area is to ensure that the critical area is bounded. The degradation (circumferential ODSCC or PWSCC not detected by bobbin coil) identified from the past dented TSP inspection would determine the initial sample.

Each SG initial sample will be determined independently. If no degradation was identified in the past inspection, a minimum 20% sample of the dents (less than 5 volts) at the first TSP will be examined. During future outages a different 20% sample would be inspected, such that over five outages 100% of the dents at this elevation would be inspected.

If indications are identified in the buffer zone, this sample will be expanded in accordance with Table 2. Any indication identified that exceeds the plugging limit shall be repaired. The buffer zone result classification as defined in TS Section 4.4.5.2 shall be utilized, except when a sample size is less than 200, then only C-2 results apply.

Alternative Dented TSP Inspection Program (greater than or equal to 5 volts);

TVA proposes an alternative inspection program for SGs 3 and 4, for the greater than 5 volt dents which is the same methodology as the proposed program for less than 5 volt dented tube support plate inspection with one additional requirement. If a TSP elevation has less than 250 dented intersections when selecting a buffer zone, then additional intersections at the next higher elevation shall be inspected to make the total number of intersections to be inspected equal to 50. TVA would like the option to employ either method to the greater than or equal to 5 volt dent population.