TVA

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

March 29, 2002

TVA-SQN-TS-02-02 10 CFR 50.90

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

Gentlemen:

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

**SEQUOYAH NUCLEAR PLANT (SQN) - UNIT 1 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 02-02, STEAM GENERATOR (SG) ALTERNATE REPAIR CRITERIA (ARC) DELETION AND SG INSPECTION INTERVAL REVISION**

In accordance with the provisions of 10 CFR 50.90, TVA is submitting a request for an amendment to SQN's License DPR-77 to change the TSs for Unit 1. The proposed change revises the SQN SG Specification 3/4.4.5 to eliminate surveillance requirements (SRs) associated with two ARCs. The associated License Condition 2.C.9.d is also deleted.

In addition, the proposed change revises SR 3/4.4.5.3.a to allow a one-time, 40-month SG inspection interval after the first (post-Unit 1 SG replacement) inservice inspection resulting in a C-1 category. The proposed change is in lieu of the current TS criteria that requires two consecutive category C-1 inspections for application of the 40-month SG inspection interval. TVA's request for a one-time, 40-month inspection interval is similar to the NRC licensing amendment for the Braidwood Station that was approved by NRC letter dated August 9, 2001. It may be noted that in contrast to the Braidwood amendment, SQN will adhere to the Electric Power Research Institute pressurized water reactor SG guidelines without exception.

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TVA has determined that there are no significant hazards considerations associated with the proposed change and that the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22 (c) (9). The SQN Plant Operations Review Committee and the SQN Nuclear Safety Review Board have reviewed the proposed change and determined that operation of SQN Unit 1, in accordance with the proposed change, will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Tennessee State Department of Public Health.

Enclosure 1 to this letter provides the description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains copies of the appropriate TS pages from Unit 1 marked-up to show the proposed change. Enclosure 3 forwards the revised TS pages for Unit 1 that incorporates the proposed change.

TVA requests NRC review and approval prior to the SQN Unit 1 Cycle 12 refueling outage (scheduled to begin March 16, 2003) to support TVA's schedule needs for this outage. TVA requests that the revised TS be made effective during the Unit 1 Cycle 12 refueling outage. It should be noted that TVA will continue to follow the ongoing industry/NRC efforts associated with NEI 97-06, "Steam Generator Program Guidelines."

This letter is being sent in accordance with NRC RIS 2001-05. No commitments are contained in this letter. If you have any questions about this change, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 29th day of March

Peggy M Billingsley
Notary Public

My Commission Expires October 9, 2002

Enclosures

**ENCLOSURE 1**

**TENNESSEE VALLEY AUTHORITY**
**SEQUOYAH NUCLEAR PLANT (SQN)**
**UNIT 1**
**DOCKET NO. 327**

**PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE NO. 02-02**
**DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE**

## I. DESCRIPTION OF THE PROPOSED CHANGE

TVA is proposing a change to the SQN Unit 1 TS to delete the requirements associated with voltage-based alternate repair criteria (ARC). SQN Unit 1 TS contains two steam generator (SG) ARC that apply during eddy current inservice inspections. The first ARC applies to axial outside diameter stress corrosion cracking (ODSCC) at non-dented tube support plates and the second ARC applies to axial primary water stress corrosion cracking (PWSCC) at dented tube support plates.

The affected TS section is TS 3/4.4.5 "Steam Generators." The following surveillance requirements (SRs) from this TS are applicable to the two ARCs and are being deleted:

| SR | ARC Type |
|---|---|
| 4.4.5.2.b.4 | ODSCC ARC |
| 4.4.5.2.d | ODSCC ARC |
| 4.4.5.2.e | PWSCC ARC |
| A portion of 4.4.5.4.a.6 | ODSCC ARC and PWSCC ARC |
| 4.4.5.4.a.10 | ODSCC ARC |
| 4.4.5.4.a.11 | PWSCC ARC |
| 4.4.5.5.d | ODSCC ARC |
| 4.4.5.5.e | PWSCC ARC |

In conjunction with the above TS SRs, TVA is proposing to delete License Condition 2.C.9.d that references TVA commitment letters associated with SG inspection. The TVA commitments will no longer apply to the SQN Unit 1 SGs following their replacement during the upcoming Unit 1 Cycle 12 refueling outage. Accordingly, License Condition 2.C.9.d is deleted as part of TVA's proposed TS change.

In addition, TVA is proposing a change to TS SR 4.4.5.3.a to allow application of the extended 40-month inspection interval after one SG inspection that is categorized as C-1. The TS SR currently states:

"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."

TVA's proposed change adds the following provision to the end of the current SR:

"SQN Unit 1 may take a one-time exception to this extension criteria for the inspection scheduled after SQN Unit 1 Cycle 13 refueling outage inspection if the Unit 1 Cycle 13 inspection results in a category of C-1. The one-time exception would allow a 40-month inspection interval based on one inspection resulting in a C-1 category."

The applicable TS Bases sections are revised to reflect the above changes.

## II. REASON FOR THE PROPOSED CHANGE

TVA is planning to replace the SQN Unit 1 SGs during the Unit 1 Cycle 12 refueling outage. The replacement SGs will contain tubing of thermally treated Alloy 690. Industry experience indicates thermally treated Alloy 690 is more resistant to stress corrosion cracking. In addition, differences in SG design is expected to greatly decrease the occurrence of stress corrosion cracking in the tubing material. Due to the increased resistance to tube degradation mechanisms, TVA is proposing removal of the ARC from Unit 1 TSs following replacement of the Unit 1 SGs.

TVA is also proposing to optimize SG inspections to allow application of the maximum 40-month inspection interval after one SG inspection resulting in a category C-1. Current TS criteria requires two consecutive inspection results in the C-1 category before the inspection interval can be extended from a 24 calendar month interval to a maximum of once per 40 month interval. This one-time change is proposed to reduce the number of SG inspections for the purpose of reducing personnel dose, inspection time, and cost to TVA.

## III. SAFETY ANALYSIS

<u>Background</u>

SG tubes are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the reactor coolant system (primary system) pressure and inventory. SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. The SG tubes are also relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity, as necessary to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damage severe accidents. To ensure SG tubing performs its intended function, SG tubing must maintain structural integrity (defined as 3 times normal operating pressure difference or 1.43 times accident pressure difference) and maintain leakage integrity (limit the primary-to-secondary leakage such that doses are a fraction of 10 CFR 100 limits and less than GDC-19 limits). The results of the inspection of SG tubing are used in condition monitoring and operational assessments to ensure structural integrity and leakage integrity is maintained during plant operation and during accident conditions.

Justification for Proposed Removal of ARC from TSs

The current SQN Unit 1 TSs allow TVA to inspect SGs with two voltage-based ARC. The first ARC is applicable to ODSCC at non-dented tube support plate (TSP) intersections. This ARC was implemented at SQN by TVA TS Change 95-15 and was based on the guidance of Generic Letter 95-05. TVA's proposed removal of this ARC and returning to the Standard TS 40 percent through-wall tube plugging limit is inherently more conservative.

The SQN Unit 1 TSs also allow TVA to inspect SGs and apply a second voltage-based ARC for PWSCC at TSP intersections. Application of this ARC is limited to operation of SQN Unit 1 through fuel Cycles 11 and 12. The ARC for PWSCC would not be applicable beyond the Unit 1 Cycle 12 refueling outage. This ARC was implemented at SQN by TVA TS Change 99-12 and was based on repair criteria documented in Westinghouse Electric Company WCAP-15128. TVA's proposed removal of the PWSCC ARC and returning to the Standard TS 40 percent through-wall plugging limit is inherently more conservative.

Justification for Extending SG Inspection Interval

The current SQN Unit 1 TSs allow TVA to extend the SGs inspection interval from 24 calendar months to a maximum of once per 40 months after two SG inspections resulting in a category of C-1. TVA is proposing to change this criteria to allow a one-time, 40-month inspection interval after one SG inspection that results in a category of C-1. This change is intended for application following the replacement of SQN Unit 1 SGs during the Cycle 12 refueling outage (Spring 2003).

The replacement SGs for Unit 1 incorporate significant SG design improvements and include thermally treated Alloy 690 tubing. TVA is performing a 100% full-length eddy current inspection (hot tube end to cold tube end, including U-bends) on all four SGs during the Cycle 13 refueling outage (first inspection following SG replacement). Application of the proposed one-time, 40 month interval would begin from the Cycle 13 SG inspection.

Design Improvements Associated with the Unit 1 Replacement SGs

The design improvements of the SQN Unit 1 replacement SGs include the following:

- Tube to tubesheet joint
- Advanced tube support grids (ATSG)
- U-bend supports
- Designed to minimize flow induced vibration
- Designed to preclude the introduction of loose parts
- Designed for increased circulation ratio
- Thermally treated Alloy 690 tubing

A summary of these improvements is discussed below:

Tube to Tubesheet Joint

The SQN Unit 1 replacement SG tubes are flush welded to the primary face of the tubesheet and hydraulically expanded the full thickness of the tubesheet to maximize mechanical strength and to minimize the tube-to-tubesheet crevice. The tubes are installed into the tubesheet after the SG lower shell and tubesheet have been welded and received post weld heat treatment (PWHT). This precludes any possibility of tube sensitization and avoids subjecting the tube-to-tubesheet joint to thermal stresses from these operations. Also, this eliminates concerns over loosening of tubes or the creation of crevices as a result of relaxation of the expanded region. A temperature limit imposed on the tubes during PWHT of the final primary head assembly to tubesheet weld further ensures the tubes are adequately protected during fabrication.

The replacement SG tubes are hydraulically expanded through essentially the entire thickness of the tubesheet. The hydraulic seals are of elastomeric material and designed so that no metal parts are impressed upon the inside surface of the tube when the hydraulic pressure is applied. The position of the seal at the secondary face of the tubesheet is controlled to ensure that expansion of the tube is as close as possible to the secondary face of the tubesheet without going past the face. After expansion, the inside profile of each tube is measured through the entire expanded area of the tubesheet (including the transition) using an eddy current method and recorded. The measurement indicates both the position and condition of the tube expansion and becomes a baseline for subsequent in-service inspections. Tests on hydraulically-expanded joints made with thermally treated Alloy 690 tubes, in triangular-pitched holes have shown that residual stresses exist in the transition region between the expanded and un-expanded tube. The hydraulic expansion process has been designed and qualified to minimize residual stresses while maintaining joint integrity.

Advanced Tube Support Grids (ATSG)

The replacement SG design uses a Type 409 stainless steel grid tube support. The ATSG provides:

- Higher circulation ratio (through lower flow resistance) than the original SG
- Minimum tube-to-tube-support contact
- Superior vibration restraint and fretting resistance
- Lower tendency to accumulate deposits than a broached or drilled tube support and standard "eggcrate"
- Elimination of denting potential due to selection of stainless steel

The ATSG is an improved version of the "eggcrate" design employed typically in such designs as Palo Verde Units 1, 2, and 3 and the recent series of plants in operation or under construction in South Korea. The size and pitch selected for the SQN Unit 1 replacement SG design include an optimized grid design to minimize the live contact between tubes and supports. This configuration has been tested in a prototypical arrangement of tubes and support spacing to verify acceptable vibration characteristics.

U-bend Supports

The upper bundle support system is designed to support the U-bends against harmful wear and vibration to minimize the potential for sludge deposition and to maximize circulation. The upper bundle support system features diagonal and vertical strip assemblies which provides support to the U-bends against flow-induced vibration. These assemblies are fabricated from perforated strips. As the tube bundle is assembled, these assemblies are positioned and interlocked by the center vertical strip. The upper ends of the vertical strips are captured by "crescent plates" above the tube bundle. The outer ends of the diagonals are also linked together. The lower ends of the assembly, where the vertical and diagonal bars intersect, are supported and spaced by "slotted-tees" attached to the top most ATSG. A combination of increased vertical pitch and the perforation of the diagonal/vertical strips, creates a low resistance flow through the upper bundle. The perforated strips promote local washing of tube surfaces and thereby minimize the potential for local sludge deposition. The upper bundle support system is integral with the U-bends of the tube bundle and generally moves with the tube bundle during heat-up and cooldown.

Designed to Minimize Flow Induced Vibration

Prevention of excessive flow induced vibration and fretting wear is maintained by a combination of design, analysis, and testing. The upper bundle support system is arranged to meet the design limits established for fluid elastic instability and for response to turbulence. Diagonal and vertical strip width is sized to optimize the amount of live contact between tube and strip while taking into account the presence of the ventilating perforation. The potential for fretting has been assessed by performing a flow induced vibration sensitivity analysis. The replacement SG design achieves a minimum resistance to riser flow with an open flow configuration and support bar orientation that is compatible with the flow direction.

Designed to Preclude the Introduction of Loose Parts

The replacement SG feed-ring includes spray pipes with small holes which will preclude the introduction of significant loose parts from feedwater.

Designed for Increased Circulation Ratio

Circulation ratio is defined as the ratio of riser mass flow rate to steam outlet mass flow rate. By maximizing circulation ratio of the SG secondary side fluid, concerns regarding heat transfer performance, generator sludge management, corrosion product transfer, tube dry-out, etc., can be minimized. The replacement SG design has a circulation ratio approximately double that of the original SGs.

Thermally Treated Alloy 690 Tubing

The SQN Unit 1 replacement SGs contain thermally treated Alloy 690 tubing. Thermally treated Alloy 690 tubing is similar to low temperature mill annealed Alloy 600 tubing but contains 13 percent more chromium and a corresponding decrease in nickel content. The higher chromium content reduces the degree of sensitization (i.e., the amount of chromium depleted in areas adjacent to the metal grain boundaries), thus increasing resistance to corrosion attack at the metal grain boundaries. Heat treatment of Alloy 690 for optimum stress corrosion cracking resistance involves mill annealing at temperatures sufficient to put all the carbon into solution, followed by a thermal treatment to precipitate carbides on the metal grain boundaries into the tube metal microstructure. Resistance to stress corrosion cracking is greatest when the metal grain boundaries are fully populated with carbides.

Extensive testing has been performed which demonstrates thermally treated Alloy 690 tubing is superior to low temperature mill annealed Alloy 600 tubing in its resistance to both primary and secondary system stress corrosion cracking, pitting, and general corrosion. Examples of this data are given in proceedings from the 1986 Electric Power Research Institute (EPRI) Workshop on Thermally Treated Alloy 690 Tubes for Nuclear Steam Generator (Reference 1). Testing was performed with statically loaded reversed U-bend (RUB) specimens, where cracking was observed within approximately 300 hours for low temperature mill annealed Alloy 600 tubing and 800 hours for thermally treated Alloy 600 tubing, while cracking was not observed for the thermally treated Alloy 690 tubing even after 12,000 hours. Testing was also performed on statically loaded tensile specimens tested in 680 degrees Fahrenheit (°F) primary water. While low temperature mill annealed Alloy 600 tubing exhibited cracking within 2,900 hours, thermally treated Alloy 690 did not exhibit cracking even after 7,000 hours of testing. Thermally treated Alloy 690 was also compared to low temperature mill annealed Alloy 600 tubing in steam tests to produce accelerated PWSCC. Steam tests have been performed in 760°F steam produced from hydrogenated pure water. These test results showed low temperature mill annealed Alloy 600 tubing exhibited cracking within 1,000 hours, while thermally treated Alloy 690 did not exhibit any signs of cracking after 6,000 hours (References 2 and 3). Industry experience with replacement SG tubing of thermally treated Alloy 690 supports laboratory test results and demonstrates the superior performance of thermally treated Alloy 690 as compared to low temperature mill annealed Alloy 600 tubing.

The SQN Unit 1 replacement SG tubing was procured from an approved supplier who demonstrated, by the manufacture and the examination of pre-production tube samples, the ability to manufacture tubing to applicable requirements. These requirements were delineated in detail in the tubing specification to which the tube manufacturer had to comply. This specification described all technical requirements which must be met for the tubing, including the bent regions. In addition to the material requirements criteria were specified for dimension, properties, cleanliness, heat treatment, defects and surface finish, as well as inspection, testing, quality assurance, non-destructive examination, and packing/shipping.

The SQN Unit 1 replacement SG tubes received a 100 percent volumetric ultrasonic inspection designed to detect indications equal to or greater than 0.003 inch deep. Tubes are rejected if they have one or more flaws in excess of 0.004 inch deep. The replacement SG tubing production tubes also received as a minimum, 100 percent full length eddy current inspection after the replacement SG were completely fabricated. In addition, an eddy current examination for signal to noise ratio was performed prior to bending on 100 percent of the tube length with a criteria for acceptance of a minimum signal-to-noise ratio of 30:1 in any straight fixed 1/6 meter lengths of any tube. This provides for the detection of extremely small flaws during in-service inspections and the monitoring of flaw growth over time. Tube bend geometry restricts use of the signal-to-noise criteria to straight runs of tubes. An EPRI Report, "Guidelines for Procurement of Alloy 690 Steam Generator Tubing," Report NP-6743-L, Volume 2, Revision 1 was used as a basis to develop procurement requirements for the replacement SG tubes.

In addition to the thermal treatment process that was performed on all tubing, additional stress relief was performed on all U-bends up through row 16. The minimum centerline U-bend radius in the SQN Unit 1 replacement SGs (3.1875 inches) is larger than in the original SGs (2.1875 inches). The larger radius reduces residual stress in the low row U-bend region. The additional stress relief and larger minimum U-bend radius design provides added assurance that this region will not develop stress corrosion cracking.

Regulatory Precedence

By letter dated February 9, 2001, Exelon Nuclear Corporation submitted TS Change request for its Unit 1 Braidwood Station. The request proposed a one-time change to revise the SG inspection frequency requirements and allow a 40-month inspection interval after one SG inspection rather than after two consecutive inspections resulting in C-1 classification. The request was subsequently approved by NRC in a Safety Evaluation Report (SER) letter dated August 9, 2001. The proposed change for TVA's SQN Unit 1 SGs is similar to the change approved for the Braidwood Station. It may be noted that in contrast to the Braidwood amendment, SQN will adhere to the EPRI pressurized water reactor SG guidelines without exception.

## IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of Sequoyah Nuclear Plant (SQN) Unit 1, in accordance with the proposed change to the technical specifications (TS) and License Condition, does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

TVA is proposing to modify SQN Unit 1 TS 3/4.4.5, "Steam Generators" to delete surveillance requirements (SRs) that describe steam generator (SG) tube plugging limits for two alternate repair criteria (ARC). The first ARC is for axial outside diameter stress corrosion cracking (ODSCC) at non-dented tube support plates and the second ARC is for axial primary water stress corrosion cracking (PWSCC) at dented tube support plates. TVA's proposed amendment removes both ARCs through the deletion of the following SRs: SR 4.4.5.2.b.4, 4.4.5.2.d, 4.4.5.2.e, a portion of 4.4.5.4.a.6, 4.4.5.4.a.10, 4.4.5.4.a.11, 4.4.5.5.d, and 4.4.5.5.e. TVA's proposed removal of these SRs for ARC reestablishes standard tube plugging criteria within the TS for SQN Unit 1. Returning to the standard TS 40 percent through-wall tube plugging limit is inherently more conservative.

Included with the above change is deletion of License Condition 2.C.9.d that references prior TVA commitment letters for SG inspection. The TVA letters and their commitments will no longer apply following replacement of the Unit 1 SGs.

In addition, TVA is proposing a revision to TS 3/4.4.5.3.a to allow application of the 40-month inspection interval after one SG inspection resulting in a C-1 category. The proposed change replaces the current TS requirement that invokes the extended 40-month inspection interval after two consecutive inspections resulting in a category of C-1. TVA's proposed change provides a relaxation of the SG inspection requirements and schedule. The relaxation in the inspection schedule is intended to coincide with replacement of SQN Unit 1 SGs during the Cycle 12 refueling outage (Spring 2003). The replacement of the SQN Unit 1 SGs incorporate significant design improvements that include thermally treated Alloy 690 SG tubing. The improvements in SG design and tube material properties increase the resistance to SG tube degradation mechanisms and allow optimization of SG inspection schedules. The proposed optimization of SG inspections reduce the cumulative number of SG inspections over the life of the plant and result in significant dose, schedule, and cost savings to TVA.

**A. <u>The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.</u>**

TVA's proposed TS amendment does not compromise limits associated with SG tube integrity. TVA's proposed change removes existing SG tube plugging criteria (i.e., ARC) from the TS and reestablishes the standard TS criteria (40 percent through-wall criteria). This change is inherently more conservative. The proposed allowance for an extended inspection interval is a conservative inspection strategy that is based on improved SG design features and SG tube materials that have been shown to resist degradation and preserve SG tube integrity.

The proposed revision does not alter plant equipment, test methods or operating practices. The proposed change continues to provide controls for safe operation of SQN SGs within the required limits. The proposed change does not contribute to events or assumptions associated with postulated design basis accidents (i.e., SG tube rupture). The proposed change does not affect operator indicators or actions required to diagnose or mitigate a SG tube rupture accident. The proposed revisions continue to maintain the required safety functions. Accordingly, the probability of an accident or the consequences of an accident previously evaluated is not increased.

B. **<u>The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.</u>**

TVA's proposed amendment removes existing repair criteria and incorporates the more conservative TS limit for SG tube plugging (i.e., plug tubes with degradation depths equal to or greater than 40 percent through-wall). This change will not give rise to new failure modes. The failure of a SG tube to maintain leakage integrity during operation is an analyzed event in the SQN Updated Final Safety Analysis Report. TVA's proposed change to the SG inspection interval will not introduce a new or different kind of accident scenario. Accordingly, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

C. **<u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>**

TVA's proposed TS amendment is conservative with respect to the margin of safety. The margin of safety is preserved through ensuring structural integrity and leakage integrity of the SG tubes.

TVA's proposed change that to remove ARC from the TS does not compromise structural integrity or leakage integrity of SG tubes. The proposed change invokes the standard TS tube plugging criteria limit (40 percent through-wall criteria) which is inherently conservative.

TVA's proposed change to include a one-time extension to the SQN Unit 1 SG inspection interval retains conservative inspection strategy that maintains the structural and leakage integrity of the SGs. TVA intends to replace SQN Unit 1 SGs during the Cycle 12 refueling outage and perform a 100 percent full length inspection of SG tubes during the Cycle 13 refueling outage to verify that damage mechanisms do not exist. Twelve years of SG operation history indicate that corrosion damage mechanisms do not appear in replacement SGs that contain thermally treated Alloy 690 tubing. The replacement SG design also contains design improvements that provide reasonable assurance that tube degradation is not likely to occur over the proposed 40-month operating period (Cycle 13 refueling outage to Cycle 15 refueling outage). The corrosion resistant properties of the thermally treated Alloy 690 tubing and the improved design will limit the initiation of damage mechanisms and limit growth rate such that tube structural and leakage integrity will be maintained over two operating cycles.

TVA's proposed change to extend the SG inspection interval does not result in a change to system design features. The proposed change does not affect the plant conditions, setpoints, or safety limits that could result in precursors to accidents or degrade accident mitigation systems. Accordingly, plant system safety functions are not altered by the proposed change.

The effect of this change is to extend allowable SG inspection intervals while retaining conservative margins to maintain the structural and leakage integrity of the SGs. Consequently, the proposed TS revisions does not reduce the margin of safety.

## V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

**ENCLOSURE 2**

**TENNESSEE VALLEY AUTHORITY**
**SEQUOYAH PLANT (SQN)**
**UNIT 1**

**PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE**
**MARKED PAGES**

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## I. AFFECTED PAGE LIST

Page 5 of SQN Operating License (License Condition 2.C.9.d)

3/4.4-7
3/4.4-8
3/4.4-9
3/4.4-9a
3/4.4-9b
3/4.4-10
3/4.4-10a

B3/4.4-3
B3/4.4-4
B3/4.4-4a

## II. MARKED PAGES

See attached.

(9) Steam Generator Inspection (Section 5.3.1)

(a) Prior to March 1, 1981, TVA shall provide to the NRC the results of its tests to determine the feasibility of using a steam generator camera device.

(b) Prior to start-up after the frist refueling, TVA must install inspection ports in each steam generator if the results of the camera device inspection are not satisfactory to the NRC;

(c) Prior to start-up after the first refueling, TVA will plug Row 1 of the steam generator tubes, if required by NRC.

~~(d) 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 letters dated March 17, 1997 and May 14, 2001.~~

(10) Water Chemistry Control Program (Section 5.3.2)

This requirement has been deleted.

(11) Negative Pressure in the Auxiliary Building Secondary Containment Enclosure (ABSCE) (Section 6.2.3)

After the final ABSCE configuration is determined, TVA must demonstate to the satisfaction of the NRC that a negative pressure of 0.25 inches of water gauge can be maintained in the spent fuel storage area and in the esf pump room.

(12) Environmental Qualification (Section 7.2.2)

(a) No late than November 1, 1980, TVA shall submit information to show compliance with the requirement of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Elecctrical Equipment," for safety-related equipment exposed to a harsh environment. Implementation shall be in accordance with NUREG-0588 by June 30, 1982.

(b) By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified to document complete compliance by June 30, 1982.

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.

Deleted

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.

Deleted

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. Inspection of dented tube support plate intersections will be performed in accordance with WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. This alternate repair criteria is applicable to Cycle 11 and 12 operation.

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.

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.

INSERT 1

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.

**INSERT 1**

SQN Unit 1 may take a one-time exception to this extension criteria for the inspection scheduled after SQN Unit 1 Cycle 13 refueling outage inspection if the Unit 1 Cycle 13 inspection results in a category of C-1. The one-time exception would allow a 40-month inspection interval based on one inspection resulting in a C-1 category.

SURVEILLANCE REQUIREMENTS (Continued)

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 or 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. For Cycle 11 and 12 operation, this definition does not apply for axial PWSCC indications, or portions thereof, which are located within the thickness of dented tube support plates which exhibit a maximum depth greater than or equal to 40 percent of the initial tube wall thickness. Refer to 4.4.5.4.a.11 for the repair limits applicable to these intersections.
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.
9. Preservice Inspection means a tube inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service 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.

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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:

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

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

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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 90-05 as supplemented. $V_{URL}$ may differ at the TSPs and flow distribution baffle.

11. Primary Water Stress Corrosion Cracking (PWSCC) Tube Support Plate Plugging Limit is used for the disposition of an Alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented PWSCC at dented tube support plate intersections as described in WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. This alternate repair criteria is applicable to Cycle 11 and 12 operation.

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 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 pursuant to Specification 6.6.1 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.

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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. If estimated leakage based on the projected end-of-cycle (or if not practical using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

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.

Deleted

e. For implementation of the depth-based repair criteria for axial PWSCC at dented TSPs, the results of the condition monitoring and operational assessments will be reported to the NRC within 120 days following completion of the inspection. The report will include tabulations of indications found in the inspection, tabulations of tubes repaired and left in service under the ARC, and growth rate distributions for indications found in the inspection as well as the growth distributions used to establish the tube repair limits. Any corrective actions found necessary in the event that condition monitoring requirements are not met will be identified in the report.

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 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 to 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.

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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 steam generator tube repair limits for primary water stress corrosion cracking (PWSCC) of SR 4.4.5 represents a steam generator tube alternate repair criteria for greater than or equal to 40 percent deep PWSCC indications which are located within the thickness of tube support plates. The repair bases for PWSCC are not applicable to other types of localized tube wall degradation located at the tube-to-tube support plate intersections.

The ARC includes completion of a condition monitoring assessment to determine the end-of-cycle (EOC) condition of the tube bundle. An operational assessment is completed to determine the need for tube repair on a forward-fit basis. The ARC is based on the use of crack depth profiles obtained from Plus Point analyses. Burst pressures and leak rates are calculated from depth profiles by searching the total crack length for the partial length that results in the lowest burst pressure and the longest length that would tear through-wall at steam-line break conditions. The repair bases for PWSCC at dented TSP intersections is obtained by projecting the crack profile to the end of the next operating cycle and determining if the projected profile meets the requirements of WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. The following provides the limits and bases for repair established in the WCAP analyses:

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Freespan Indication Repair Limits

The tube will be repaired if the crack length outside the dented TSP is ≥40% maximum depth.

Crack Length Limit for ≥40% Maximum Depth

The crack length limit for ≥40% maximum depth indications is defined as 0.375 inch from the centerline of the TSP. This limit defines the edges of the TSP thickness of 0.75 inch for Model 51 S/Gs. It is acceptable for the crack to extend to both edges of the TSP as long as the maximum depth of the crack outside the TSP is <40% maximum depth and the requirements for EOC conditions are acceptable.

Operational Assessment Repair Bases

If the indication satisfies the above maximum depth and length requirements, the repair bases is then obtained by projecting the crack profile to the end of the next operating cycle and determining the burst pressure and leakage for the projected profile. The burst pressure and leakage is compared to the requirements in WCAP 15128, Revision 2, dated February 2000 as supplemented by TVA's letter to NRC dated March 2, 2000. Separate analyses are required for the total crack length and the length outside the TSP due to differences in requirements. If the projected EOC requirements are satisfied, the tube will be left in service.

The results of the condition monitoring and operational assessments will be reported to the NRC within 120 days following completion of the inspection.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.6.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

**ENCLOSURE 3**

**TENNESSEE VALLEY AUTHORITY**
**SEQUOYAH PLANT (SQN)**
**UNITS 1 AND 2**

**PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE 02-02**
**REVISED PAGES**

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## I. AFFECTED PAGE LIST

Page 5 of SQN Operating License (License Condition 2.C.9.d)
3/4.4-7
3/4.4-8
3/4.4-9
3/4.4-10
B3/4.4-3
B3/4.4-4
B3/4.4-4a

## II. REVISED PAGES

(see attached)

(9) Steam Generator Inspection (Section 5.3.1)

(a) Prior to March 1, 1981, TVA shall provide to the NRC the results of its tests to determine the feasibility of using a steam generator camera device.

(b) Prior to start-up after the frist refueling, TVA must install inspection ports in each steam generator if the results of the camera device inspection are not satisfactory to the NRC;

(c) Prior to start-up after the first refueling, TVA will plug Row 1 of the steam generator tubes, if required by NRC.

(10) Water Chemistry Control Program (Section 5.3.2)

This requirement has been deleted.

(11) Negative Pressure in the Auxiliary Building Secondary Containment Enclosure (ABSCE) (Section 6.2.3)

After the final ABSCE configuration is determined, TVA must demonstrate to the satisfaction of the NRC that a negative pressure of 0.25 inches of water gauge can be maintained in the spent fuel storage area and in the esf pump room.

(12) Environmental Qualification (Section 7.2.2)

(a) No late than November 1, 1980, TVA shall submit information to show compliance with the requirement of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," for safety-related equipment exposed to a harsh environment. Implementation shall be in accordance with NUREG-0588 by June 30, 1982.

(b) By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified to document complete compliance by June 30, 1982.

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.

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.

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.

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. SQN Unit 1 may take a one-time exception to this extension criteria for the inspection scheduled after SQN Unit 1 Cycle 13 refueling outage inspection if the Unit 1 Cycle 13 inspection results in a category of C-1. The one-time exception would allow a 40-month inspection interval based on one inspection resulting in a C-1 category.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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 or 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).
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.
9. Preservice Inspection means a tube inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service 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.

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 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 pursuant to Specification 6.6.1 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.

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 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.

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