ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2

DOCKET NO. 50-328

(TVA-SQN-TS-95-23)

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SURVEILLANCE REQUIREMENTS (Continued)

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.

INSERT A

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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. R18

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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.

* THE INDICATED CHANGES TO THIS PAGE ARE APPLICABLE TO CYCLE 8 OPERATION ONLY.

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 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). R181

INSERT C

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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.

* THE INDICATED CHANGES TO THIS PAGE ARE APPLICABLE TO CYCLE 8 OPERATION ONLY.

SURVEILLANCE REQUIREMENTS (Continued)

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.

INSERT D

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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 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. R28

INSERT E

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* THE INDICATED CHANGES TO THIS PAGE ARE APPLICABLE TO CYCLE 8 OPERATION ONLY.

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,

INSERT F

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator,

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

e. 40 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. 1 GPM leakage at a React - Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3 and 4

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, and leakage from Reactor Coolant System Pressure Isolation Valves, 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.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

* REPLACEMENT OF C. WITH CC. IS APPLICABLE FOR CYCLE 8 OPERATION ONLY.

BASES

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 = ~~500~~ 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. ~~Operating plants have~~ SEQUOYAH HAS demonstrated that primary-to-secondary leakage of ~~500~~ 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. *

INSERT G *

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 ~~plugging limit of 40% of the tube nominal wall thickness~~ 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. R181

INSERT H *

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 6.9.1 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.

* THE INDICATED CHANGES TO THIS PAGE ARE APPLICABLE TO CYCLE 8 OPERATION ONLY.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurances of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

INSERT I

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~~The total steam generator tube leakage limit of 1 GPM for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.~~

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

* THE INDICATED CHANGES TO THIS PAGE ARE APPLICABLE TO CYCLE 8 OPERATION ONLY.

Insert A

4.4.5.2.b.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.

Insert B

4.4.5.2.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.

Insert C

4.4.5.4.a.6. 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.

Insert D

4.4.5.4.a.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 iimit (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:

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

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

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
Δt	=	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.

Insert E

4.4.5.5.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 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

Insert F

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

Insert G

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.

Insert H

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.

Insert I

The total steam generator tube leakage limit of 600 gallons per day for all steam generators and 150 gallons per day for any one steam generator will minimize the potential for a significant leakage event during steam line break. Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 3.7 gpm in the faulted loop, which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. If the projected end of cycle distribution of crack indications results in primary-to-secondary leakage greater than 3.7 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 3.7 gpm.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2

DOCKET NO. 50-328

(TVA-SQN-TS-95-23)

DESCRIPTION AND JUSTIFICATION FOR

TS AMENDMENT

Description of Change

TVA proposes to modify the SQN Unit 2 technical specifications (TSs) to incorporate new requirements associated with steam generator (S/G) tube inspection and repair. The new requirements establish alternate S/G tube plugging criteria at tube support plate (TSP) intersections. Note, these changes are identified as Cycle 8 only based on open industry issues identified during the Unit 1 review and approval of a similar change. The changes are identified with an * and explanation at the bottom of each affected page. The proposed changes are as follows:

1. Add Surveillance Requirement (SR) 4.4.5.2.b.4

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

2. Add SR 4.4.5.2.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."

3. Add requirements to SR 4.4.5.4.a.6.

   "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."

4. Add SR 4.4.5.4.a.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, b, & 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
Δt	=	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 TSs 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.

5. Add SR 4.4.5.5.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 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence."

6. Add TS 3.4.6.2.cc

   "Primary-to-secondary leakage shall be limited to 150 gallons per day through any one steam generator." *

   Additionally, the following phrase should be added "* Replacement of c. with cc. is applicable for Cycle 8 operation only."

7. Change Bases 3/4.4.5, "Steam Generator," to reflect the new primary-to-secondary leakage limit (150 gallons per day per S/G) and include a reference to the tube repair limit as defined in Specification 4.4.5.4.a. In addition, Bases Section 3/4.4.6.2, "Operational Leakage," is revised to reflect the S/G operational leakage limits.

Reason for Change

TVA is proposing to change SQN Unit 2 TSs to reduce the need for repairing or plugging S/G tubes having indications that exceed the current TS depth-based plugging limit. TVA proposes to add alternate tube plugging criteria at TSP intersections that are based on maintaining structural and leakage integrity of tubes with indications of ODSCC within the confines of the TSP regions. Westinghouse Electric Corporation has performed analyses to: (1) show that indications within the TSP region meet Regulatory Guide (RG) 1.121 criteria for tube structural integrity, and (2) leakage in a faulted condition remains below that assumed in calculating the allowable offsite radiation dose limits. The guidance of Generic Letter (GL) 95-05 was utilized.

The proposed change would preserve the reactor coolant flow margin and reduce the radiation exposure incurred in the process of plugging or repairing the S/G tubes (approximately 0.060 man-rem per tube of exposure would be saved for a plugging operation). Other benefits of not plugging TSP indications that meet the alternate plugging criteria (APC) would be a reduction in man-hours and potential impact to critical path time during refueling outages.

TVA's goal is to prolong S/G life over the expected remaining plant life. This goal is best achieved by proactive measures that defer or eliminate the need to replace S/Gs. S/G replacement results from the loss-of-tube plugging margin.

Accordingly, the proposed TS change would prolong S/G life and reduce personnel exposure while maintaining the SQN S/G plugging margin.

Justification for Changes

The proposed APC for SQN can be summarized as follows:

Tube Support Plate APC

Tubes with bobbin indications exceeding the 2.0-volt APC voltage repair limit and less than or equal to 5.4 volts are plugged or repaired if confirmed as flaw indications by a rotating pancake coil (RPC) inspection. Bobbin indications greater than 5.4 volts attributable to ODSCC are repaired or plugged independent of RPC confirmation.

Operating Leakage Limits

Plant shutdown will be implemented if normal operating leakage exceeds 150 gallons per day per S/G.

Steam Line Break (SLB) Leakage Criterion

Projected end-of-cycle SLB leak rate from tubes left in service, including a probability of detection adjustment and allowances for nondestructive examination (NDE) uncertainties and ODSCC growth rates, must be less than 3.7 gallons per minute for the S/G in the faulted loop. If necessary to satisfy the allowable leakage limit, additional indications less than the repair limit shall be plugged or repaired.

Tube Burst Conditional Probability

The projected end-of-cycle SLB tube burst conditional probability shall be calculated and compared with the value 1 X 10^{-2} as defined in GL 95-05.

Exclusion from Tube Plugging Criteria

The APC does not apply to TSP intersections having:

1. Dent signals greater than 5 volts as measured with the bobbin probe.
2. Mixed residuals of sufficient magnitude to cause a 1-volt ODSCC indication (as measured with a bobbin probe) to be missed or evaluated incorrectly.
3. Circumferential indications.

These indications shall be evaluated to the TS limit of 40 percent depth.

SQN's current TS plugging limit of 40 percent throughwall applies throughout the tube length and is based on the tube structural integrity for general area wall loss such as pitting or wear. Tube plugging criteria are based upon the conservative assumptions that the tube to TSP crevices are open (negligible crevice deposits or TSP corrosion) and that the TSPs are displaced under accident conditions. The ODSCC existing within the TSPs is thus assumed to be free-span degradation under accident conditions and the principal requirement for tube plugging considerations is to provide margins against tube burst in accordance with RG 1.121. The open crevice assumption leads to maximum leak rates compared with packed crevices and also maximizes the potential for TSP displacements under accident conditions.

One pulled tube with two TSP intersections from SQN Unit 1 support ODSCC as the dominant corrosion mechanism consistent with the Electric Power Research Institute (EPRI) database of pulled tubes. The EPRI database, which includes the SQN pulled tube data, is more conservative for SLB leak rate analyses and will be used for all SLB analyses.

RG 1.121 guidelines establish the structural limit as the more limiting of three times normal operating pressure differential ($3\Delta P_{NO}$) or 1.43 times the SLB pressure differential ($1.43\Delta P_{SLB}$) at accident conditions. At normal operating conditions, the tube constraint provided by the TSP assures that $3\Delta P_{NO}$ burst capability is satisfied. At SLB conditions, the EPRI alternate repair criteria (ARC) are based on free-span indications under the conservative assumption that SLB TSP displacements uncover the ODSCC indications formed within the TSPs at normal operation. From Figure 6-1 of WCAP-13990, the bobbin voltage corresponding to $1.43\Delta P_{SLB}$ (3,657 pounds per square inch) is 8.82 volts.

The structural limit is reduced by allowance for NDE uncertainties and crack growth. The EPRI ARC supplies the NDE uncertainty (WCAP-13990, Section 7.3) at 95 percent uncertainty to obtain an allowance of 20.5 percent of the repair limit. For SQN, there is insufficient ODSCC data to define the voltage growth rates. In EPRI Report TR-100407, Draft Revision 1, "PWR Steam Generator Tube Repair Limit - Technical Support Document for Outside Diameter Stress Corrosion Crack at Tube Support Plates," the EPRI criteria provides a growth allowance of 35 percent per effective full power years (EFPY) when plant specific growth data is not available. For SQN, the near-term cycle lengths are bounded by 1.23 EFPY. The growth allowance for SQN is then 43.1 percent. The full APC repair limit is obtained by dividing the structural limit of 8.82 volts by 1.64 (1.0 + 20.5 percent for NDE uncertainties and 43 percent for crack voltage growth). Thus, the full EPRI ARC defined repair limit is obtained as 5.4 volts. This repair limit conservatively bounds the limit obtained by applying either the EPRI database, as described above, or the NRC database additions described in WCAP-13990, Section 5.1.

In addressing the combined effects of loss-of-coolant accident (LOCA), plus safe shutdown earthquake (SSE) on the S/G component (as required by General Design Criteria 2), it has been determined that tube collapse may occur in the S/Gs at some plants. This is the case as the TSP may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate because of the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with S/G tube collapse. First, the collapse of S/G tubing reduces the reactor coolant system flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA, which in turn, may potentially increase peak clad temperature. Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Consequently, since the leak-before-break methodology is applicable to the SQN reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. LOCA loads for the primary pipe breaks were used to bound the conditions at SQN for smaller breaks. The results of the analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in S/G tube collapse or significant deformation. The LOCA, plus SSE tube collapse evaluation performed for another plant with Series 51 S/Gs using bounding input conditions (large-break loadings), is applicable to SQN. Additional supporting information relative to NRC review of J. M. Farley Nuclear Plant was provided in Enclosure 5, Item 3 of TVA's submittal dated September 7, 1995 (TAC No. M92961).

Environmental Impact Evaluation

The proposed change does not involve an unreviewed environmental question because operation of SQN Unit 2 in accordance with this change would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by NRC's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or decisions of the Atomic Safety and Licensing Board.

2. Result in a significant change in effluents or power levels.

3. Result in matters not previously reviewed in the licensing basis for SQN that may have a significant environmental impact.

ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) 2

DOCKET NO. 50-328

(TVA-SQN-TS-95-23)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Significant Hazards Evaluation

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) Unit 2 in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Testing of model boiler specimens for free-span tubing (no tube support plate restraint) at room temperature conditions shows burst pressures in excess of 5,000 pounds per square inch (psi) for indications of outer diameter stress corrosion cracking with voltage measurements as high as 19 volts. Burst testing performed on intersections pulled from SQN with up to a 1.9-volt indication shows measured burst pressure in excess of 6,600 psi at room temperature. Burst testing performed on pulled tubes from other plants with up to 7.5-volt indications shows burst pressures in excess of 5,200 psi at room temperatures. Correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the safety-factor requirements of NRC Regulatory Guide (RG) 1.121.

   Tube burst criteria are inherently satisfied during normal operating conditions because of the proximity of the tube support plate (TSP). Since tube-to-tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics that maintain a margin of safety of 1.43 times the bounding faulted condition steam line break (SLB) pressure differential. During a postulated SLB, the TSP has the potential to deflect during blowdown following a main SLB, thereby uncovering the TSP intersections.

   Based on the existing database, the RG 1.121 criterion requiring maintenance of a safety factor of 1.43 times the SLB pressure differential on tube burst is satisfied by 7/8-inch-diameter tubing with bobbin coil indications with signal amplitudes less than 8.82 volts (WCAP-13990), regardless of the indicated depth measurement. A 2.0-volt plugging criterion (resulting in a projected end-of-cycle [EOC] voltage) compares favorably with the 8.82-volt structural limit considering the extremely slow apparent voltage growth rates and few numbers of indications at SQN. Using the established methodology of RG 1.121, the structural limit is reduced by allowances for uncertainty and growth to develop a beginning of cycle (BOC) repair limit that would preclude indications at EOC conditions that exceed the structural limit. The nondestructive examination (NDE) uncertainty component is 20.5 percent, and is based on the Electric Power Research Institute (EPRI) alternate repair criteria (ARC).

Test data indicates that tube burst cannot occur within the TSP, even for tubes that have 100 percent throughwall electro-discharge machining notches, 0.75 inch long, provided that the TSP is adjacent to the notched area. Because of the few number of indications at SQN, the EPRI methodology of applying a growth component of 35 percent per effective full power year (EFPY) will be used. Near-term operating cycles at SQN are expected to be bounded by 1.23 years, therefore, a 43 percent growth component is appropriate. When these allowances are added to the BOC alternate plugging criteria (APC) of 2.0 volts in a deterministic bounding EOC voltage of approximately 3.26 volts for Cycle 7, operation can be established. A 5.56-volt deterministic safety margin exists (8.82 structural limit - 3.26-volt EOC equal 5.56-volt margin).

For the voltage/burst correlation, the EOC structural limit is supported by a voltage of 8.82 volts. Using this structural limit of 8.82 volts, a BOC maximum allowable repair limit can be established using the guidance of RG 1.121. The BOC maximum allowable repair limit should not permit the existence of EOC indications that exceed the 8.82-volt structural limit. By adding NDE uncertainty allowances and an allowance for crack growth to the repair limit, the structural limit can be validated. Therefore, the maximum allowable BOC repair limit (RL) based on the structural limit of 8.82 volts can be represented by the expressions:

$$RL + (0.205 \times RL) + (0.43 \times RL) = 8.82 \text{ volts, or,}$$

the maximum allowable BOC repair limit can be expressed as,

$$RL = 8.82\text{-volt structural limit}/1.64 = 5.4 \text{ volts.}$$

This RL (5.4 volts) is the appropriate limit for APC implementation to repair bobbin indications greater than 2.0 volts independent of rotating pancake coil (RPC) confirmation of the indication. This 5.4-volt upper limit for nonconfirmed RPC calls is consistent with other recently approved APC programs (Farley Nuclear Plant, Unit 2).

The conservatism of the growth allowance used to develop the repair limit is shown by the most recent SQN eddy current data. Only seven tubes in Unit 2 required repair because of outside diameter stress corrosion cracking (ODSCC) at the TSP intersections.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated main SLB outside of containment, but upstream of the main steam isolation valve (MSIV), represents the most limiting radiological condition relative to the APC. Implementation of the APC will determine whether the distribution of cracking indications at the TSP intersections is projected to be such that primary-to-secondary leakage would result in site boundary doses within a small fraction of the 10 CFR 100 guidelines. A separate analysis has determined this allowable SLB leakage limit to be 3.7 gallons per minute (gpm) in the faulted loop. This limit uses the TS reactor coolant system (RCS) Iodine-131 activity level of 1.0 microcuries per gram dose equivalent Iodine-131 and the recommended Iodine-131 transient spiking values consistent with NUREG-0800. The analysis method is WCAP-14277, which is consistent with the guidance of the NRC generic letter (GL) and will be used to calculate EOC leakage. Because of the relatively low number of indications at SQN, it is expected that the actual leakage values will be far less than this limit. Additionally, the current Iodine-131 levels at SQN range from about 25 to 100 times less than the TS limit.

Application of the criteria requires the projection of postulated SLB leakage, based on the projected EOC voltage distribution for Cycle 8 operation. Projected EOC voltage distribution is developed using the most recent EOC eddy current results and a voltage measurement uncertainty. Data indicates that a threshold voltage of 2.8 volts would result in throughwall cracks long enough to leak at SLB condition. The GL requires that all indications to which the APC are applied must be included in the leakage projection. Tube pull results from another plant with 7/8-inch tubing with a substantial voltage growth database have shown that tube wall degradation of greater than 40 percent throughwall was readily detectable either by the bobbin or RPC probe. The tube with maximum throughwall penetration of 56 percent (42 average) had a voltage of 2.02 volts. The SQN Unit 1 pulled tube had a 1.93-volt indication with a maximum depth of 91 percent and did not leak at SLB condition. Based on the SQN pulled tube and industry pulled tube data supporting a lower threshold for SLB leakage of 2.8 volts, inclusion of all APC intersections in the leakage model is quite conservative. The ODSCC occurring at SQN is in its earliest stages of development. The conservative bounding growth estimations to be applied to the expected small number of indications for the upcoming inspection should result in very small levels of predicted SLB leakage. Historically, SQN has not identified ODSCC as a contributor to operational leakage.

In order to assess the sensitivity of an indication's BOC voltage to EOC leakage potential, a Monte Carlo simulation was performed for a 2.0-volt BOC indication. The maximum EOC voltage (at 99.8 percent cumulative probability) was found to be 4.8 volts. The leakage component from an indication of this magnitude, using the EPRI leakage model, is 0.028 gpm.

Therefore, as implementation of the 2.0-volt APC does not adversely affect steam generator (S/G) tube integrity and implementation will be shown to result in acceptable dose consequences, the proposed amendment does not result in significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

   Implementation of the proposed S/G tube APC does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism that could result in an accident outside of the region of the TSP elevations; no ODSCC is occurring outside the thickness of the TSP. Neither a single or multiple tube rupture event would be expected in a S/G in which the plugging criteria is applied (during all plant conditions).

   TVA will implement a maximum leakage rate limit of 150 gallon per day per S/G to help preclude the potential for excessive leakage during all plant conditions. The SQN TS limits on primary-to-secondary leakage at operating conditions include a maximum of 0.42 gpm (600 gallons per day [gpd]) for all S/Gs, or, a maximum of 150 gpd for any one S/G. The RG 1.121 criterion for establishing operational leakage rate limits that require plant shutdown is based upon leak-before-break considerations to detect a free-span crack before potential tube rupture during faulted plant conditions. The 150-gpd limit should provide for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. RG 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 1.43 against bursting at faulted conditions maximum pressure differential. A voltage amplitude of 8.82 volts for typical ODSCC corresponds to meeting this tube burst requirement at a lower 95 percent prediction limit on the burst correlation coupled with 95/95 lower tolerance limit material properties. Alternate crack morphologies can correspond to 8.82 volts so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, typical burst pressure versus through-wall crack length correlations are used below to define the "longest permissible crack" for evaluating operating leakage limits.

   The single through-wall crack lengths that result in tube burst at 1.43 times the SLB pressure differential and the SLB pressure differential alone are approximately 0.57 inch and 0.84 inch, respectively. A leak rate of 150 gpd will provide for detection of 0.4-inch-long cracks at nominal leak rates and 0.6-inch-long cracks at the lower 95 percent confidence level leak rates. Since tube burst is precluded during normal operation because of the proximity of the TSP to the tube and the potential exists for the crevice to become uncovered during SLB conditions, the leakage from the maximum permissible crack must preclude tube burst at SLB conditions. Thus, the 150-gpd limit provides for plant shutdown before reaching critical crack lengths for SLB conditions. Additionally, this leak-before-break evaluation assumes that the entire crevice area is uncovered during blowdown. Partial uncover will provide benefit to the burst capacity of the intersection.

As S/G tube integrity upon implementation of the 2.0-volt APC continues to be maintained through in-service inspection and primary-to-secondary leakage monitoring, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

3. Involve a significant reduction in a margin of safety.

The use of the voltage based APC at SQN is demonstrated to maintain S/G tube integrity commensurate with the criteria of RG 1.121. RG 1.121 describes a method acceptable to the NRC Staff for meeting General Design Criteria (GDC) 14, 15, 31, and 32 by reducing the probability or the consequences of S/G tube rupture. This is accomplished by determining the limiting conditions of degradation of S/G tubing, as established by in-service inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst-case conditions, the occurrence of ODSCC at the TSP elevations is not expected to lead to a S/G tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the TSP elevations will be confirmed to result in acceptable primary-to-secondary leakage during all plant conditions and radiological consequences are not adversely impacted.

In addressing the combined effects of loss-of-coolant accident (LOCA), plus safe shutdown earthquake (SSE) on the S/G component (as required by GDC 2), it has been determined that tube collapse may occur in the S/Gs at some plants. This is the case as the TSP may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate because of the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with S/G tube collapse. First, the collapse of S/G tubing reduces the RCS flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA, which in turn, may potentially increase peak clad temperature (PCT). Second, there is a potential that partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse.

Consequently, since the leak-before-break methodology is applicable to the SQN reactor coolant loop piping, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. The limiting LOCA event becomes either the accumulator line break or the pressurizer surge line break. LOCA loads for the primary pipe breaks were used to bound the conditions at SQN for smaller breaks. The results of the analysis using the larger break inputs show that the LOCA loads were found to be of insufficient magnitude to result in S/G tube collapse or significant deformation. The LOCA, plus SSE tube collapse evaluation performed for another plant with Series 51 S/Gs using bounding input conditions (large-break loadings), is applicable to SQN. Therefore, at SQN, no tubes will be excluded from using the voltage repair criteria due to deformation of collapse of S/G tubes following a LOCA plus an SSE. Additional supporting information relative to NRC review of J. M. Farley Nuclear Plant was provided in Enclosure 5, Item 3 of TVA's submittal dated September 7, 1995 (TAC No. M92961).

Addressing RG 1.83 considerations, implementation of the bobbin probe voltage based interim tube plugging criteria of 2.0 volt is supplemented by: (1) enhanced eddy current inspection guidelines to provide consistency in voltage normalization, (2) a 100 percent eddy current inspection sample size at the TSP elevations, and (3) RPC inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

As noted previously, implementation of the TSP elevation plugging criteria will decrease the number of tubes that must be repaired. The installation of S/G tube plugs reduces the RCS flow margin. Thus, implementation of the alternate plugging criteria will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin of safety.

ENCLOSURE 4

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2

DOCKET NO. 50-328

(TVA-SQN-TS-95-23)

TVA COMMITMENT

TVA COMMITMENT

TVA will revise the SQN steam generator program for Unit 2 by April 19, 1996, to include the following requirements:

A. "If alternate plugging criteria (APC) is implemented, the following results, distributions, and evaluations will be submitted to the NRC staff within 90 days of unit restart:

1. The results of metallurgical examinations of tube intersections removed from the unit.

2. End-of-cycle (EOC) voltage distribution - all indications found during the inspection regardless of a rotating pancake coil (RPC) confirmation.

3. Cycle voltage growth rate distribution (i.e., from beginning of cycles to EOC).

4. Voltage distribution for EOC repaired indications - distribution of indications presented in Item 2 that were repaired (i.e., plugged or sleeved).

5. Voltage distribution for indications left in service at the beginning of the next operating cycle regardless of RPC confirmation - obtained from Items 2 and 4 above.

6. Voltage distribution for indications left in service at the beginning of the next operating cycle that were confirmed by RPC to be crack-like or not RPC inspected.

7. Nondestructive examination uncertainty distribution used in predicting of the EOC (for the next cycle of operation) voltage distribution.

8. Conditional probability of burst during main steam line break (MSLB) evaluation.

9. Total leak rate during MSLB evaluation.

B. If the APC is implemented, the actions and administrative controls provided in Enclosure 5 become effective.

C. TVA will implement the probe wear inspection/reinspection requirements of Enclosure 5 for one cycle of operation (Cycle 8)."

ENCLOSURE 5

SEQUOYAH NUCLEAR PLANTS (SQN'S)

STEAM GENERATOR (S/G) PROGRAM PLAN FOR THE

IMPLEMENTATION OF GENERIC LETTER (GL) 95-05

1. The inspection guidance discussed in Section 3 of Attachment 1 of GL 95-05 will be implemented.

3.b Rotating pancake coil (RPC) for the purposes of the technical specification (TS) change, also includes the use of comparable or improved nondestructive examination techniques.

3.b.1 TVA will inspect all bobbin flaw indications with voltages greater than lower voltage repair limit volts utilizing a RPC probe.

3.b.2 TVA will inspect all intersections where copper signals interfere with the detection of flaws utilizing a RPC probe. Any indications found at such intersections with RPC should cause the tube to be repaired.

3.b.3 TVA will inspect all intersections with dent signals greater than 5 volts with a RPC. Any indications found at such intersections with RPC should cause the tube to be repaired. If circumferential cracking or primary water stress corrosion cracking indications are detected, it may be necessary to expand the RPC sampling plan to include dents less than 5.0 volts.

If circumferential cracking is detected at the tube support plate (TSP) locations dented in the 5- to 6-volt range, then the following criteria will be applied.

1. A 20 percent initial sample of dents less than 5 volts will be selected from the total population at TSP elevations (and TSPs below) where circumferential cracking has been detected. The initial sample for each S/G will be selected independently with the sample weighted toward the lower TSPs.

2. In a particular S/G, if a circumferential crack is identified in a less than 5-volt sample, then an additional 20 percent sample shall be examined and again weighted toward the lower TSPs.

3. In a particular S/G, if the initial sample or an expanded sample has zero circumferential crack indications, no additional expanded samples are required.

4. Axial indications in dents less than 5 volts that are structurally significant will be detected with bobbin coil exams; therefore, no additional RPC examinations are required. For axial flaws that are determined to be structurally insignificant no expansion is required in the less than 5 volt dent sample.

5. Repair criteria as defined in the appropriate technical specifications shall apply.

3.b.4 TVA will inspect all intersections with large mixed residuals utilizing a RPC probe. Any indications found at such intersections with RPC should cause the tube to be repaired.

3.c.1 TVA will use a bobbin coil calibrated against a reference standard used in the laboratory as part of the development of the voltage-based approach, through the use of a transfer standard.

3.c.2 TVA will comply with a ±10 percent probe variability. TVA will increase the number of probe samples to 20. Testing will be performed at the mix frequency. TVA plans to follow the industry approach to probe variability that was presented to NRC in November 1994.

Probe wear inspection/reinspections will be governed by the following:

If the last probe-wear-standard signal amplitudes prior to probe replacement exceed the ±15 percent limit by a value of X percent, then any indications measured since the last acceptable probe wear measurement that are within X percent of the plugging limit must be reinspected with the new probe. For example, if any of the last probe wear signal amplitudes prior to probe replacement were 17 percent above or below the initial amplitude, then indications that are within 2 percent (17 - 15 percent) of the plugging limit must be reinspected with the new probe. Alternatively, the voltage criterion may be lowered to compensate for the excess variation, for the case above, amplitudes ≥ 0.98 times the voltage criterion would be subject to repair.

3.c.4 TVA data analysts will be trained and qualified in the use of analysis guidelines and procedures.

3.c.5 Data analysts will use quantitative noise criteria guidelines in the evaluation of the data. However, it is expected that these criteria will be evolving over the inspection and as a result, are subject to change. Data failing to meet these criteria should be rejected, and the tube will be reinspected.

2. If the alternate plugging criteria is implemented, SQN will pull a minimum of two tubes and four intersections during the Unit 2 Cycle 7 outage and implement a tube pull program consistent with the GL.

3. In support of Section 4.4.5.4.a.10 of the proposed TS change, the methodology used for calculating the (1) conditional probability of burst, (2) methods for projecting EOC voltage distributions, (3) upper voltage repair limit, and (4) total leak rate during main steam line break, will be in accordance with Westinghouse Electric Corporation, WCAP-14277, "Steam Line Break Leak Rate and Tube Burst Probability Analysis Method for ODSCC at TSP Intersections, January 1995."

   The data sets for burst pressure verses bobbin voltage will contain all applicable data consistent with the latest revision of the industry data base as approved by NRC with the latest tube pull data.

4. SQN Abnormal Operating Instruction 24 provides instructions on the trending and response to rapidly increasing leaks. This instruction is a defense-in-depth measure that provides a means for identifying leaks during operation to enable repair before such leaks result in tube failure.