ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2

DOCKET NO. 50-328

(TVA-SQN-TS-94-09)

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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.

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.

INSERT A HERE

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

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

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

    4. A main steam line or feedwater line break.

INSERT B HERE

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.

INSERT C HERE

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.

INSERT A

Implementation of the steam generator tube/tube support plate interim plugging criteria for one fuel cycle (Cycle 7) requires a 100% bobbin probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications.

INSERT B

d. Tubes left in service as a result of application of the tube support plate interim plugging criteria shall be inspected by bobbin coil probe during all future refueling outages.

INSERT C

This definition does not apply for tubes experiencing outer diameter stress corrosion cracking confirmed by bobbin probe inspection to be within the thickness of the tube support plates. See 4.4.5.4.a.10 for plugging limit for use within the thickness of the tube support plate.

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 HERE

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 HERE

INSERT D

10. The Tube Support Plate Interim Plugging Criteria is used for disposition of a steam generator tube for continued service that is experiencing outer diameter initiated stress corrosion cracking confined within the thickness of the tube support plates. For application of the tube support plate interim plugging limit, the tube's disposition for continued service will be based upon standard bobbin probe signal amplitude. The plant-specific guidelines used for all inspections shall be amended as appropriate to accommodate the additional information needed to evaluate tube support plate signals with respect to the voltage/depth parameters. Pending incorporation of the voltage verification requirement in ASME standard verifications, an ASME standard calibrated against the laboratory standard (See WCAP-13990, Rev. 0) will be utilized in the Sequoyah Nuclear Plant Unit 2 steam generator inspections for consistent voltage normalization.

   1. A tube can remain in service if the signal amplitude of a crack indication is less than or equal to 2.0 volts, regardless of the depth of tube wall penetration, if, as a result, the projected end-of-cycle distribution of crack indications is verified to result in primary-to-secondary leakage less than 4.3 gpm in the faulted loop during a postulated steam line break event. The methodology for calculating expected leak rates from the projected crack distribution must be consistent with WCAP-13990, Rev. 0, and as prescribed in Draft NUREG-1477.

   2. A tube should be plugged or repaired if the signal amplitude is greater than 2.0 volts except as noted in Specification 4.4.5.4.a.10.3 below.

   3. A tube can remain in service with a bobbin coil signal amplitude greater than 2.0 volts but less than or equal to 3.6 volts if a rotating pancake probe inspection does not detect degradation. Indications of degradation with a bobbin coil signal amplitude greater than 3.6 volts will be plugged or repaired.

INSERT E

d. The results of inspections performed under 4.4.5.2 for all tubes in which the tube support plate elevations interim plugging limit has been applied and where not plugged shall be reported to the Commission within 15 days following the inspection. The report shall include:

   1. Listing of applicable tubes.

   2. Location (applicable intersections per tube) and extent of degradation (voltage).

e. The result of steam line break leakage analysis performed under T/S 4.4.5.4.a.10 will be reported to the Commission prior to restart for Cycle 7.

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,

600 gpd

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

150

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 Reactor 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:

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 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 demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected. ~~by radiation monitors of steam generator blowdown.~~ Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

150

repair limit defined in Specification 4.4.5.4.a

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.~~ 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.

INSERT F HERE

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.

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.

~~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.~~

INSERT G HERE

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.

INSERT F

Tubes experiencing outer 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 G

Maintaining an operating leakage limit of 150 gpd per steam generator (600 gpd total) for Cycle 7 will minimize the potential for a large 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 4.3 gpm in the faulted loop, which will limit the calculated offsite doses to within 10% of the 10 CFR 100 guidelines. Leakage in the intact loops is limited to 150 gpd. If the projected end of cycle distribution of crack indications results in primary-to-secondary leakage greater than 4.3 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 4.3 gpm.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2

DOCKET NO. 50-328

(TVA-SQN-TS-94-09)

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. The proposed changes are as follows:

1. Add requirements to Surveillance Requirement (SR) 4.4.5.2.c.2

   "Implementation of the steam generator tube/tube support plate interim plugging criteria for one fuel cycle (Cycle 7) requires a 100% bobbin probe inspection for all hot leg tube support plate intersections and all cold leg intersections down to the lowest cold leg tube support plate with outer diameter stress corrosion cracking (ODSCC) indications."

2. Add SR 4.4.5.3.d.

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

3. Add clarification statements to SR 4.4.5.4.a.6.

   "This definition does not apply for tubes experiencing outer diameter stress corrosion cracking confirmed by bobbin probe inspection to be within the thickness of the tube support plates. See 4.4.5.4.a.10 for plugging limit for use within the thickness of the tube support plate."

4. Add SR 4.4.5.4.a.10.

   "The Tube Support Plate Interim Plugging Criteria is used for disposition of a steam generator tube for continued service that is experiencing outer diameter initiated stress corrosion cracking confined within the thickness of the tube support plates. For application of the tube support plate interim plugging limit, the tube's disposition for continued service will be based upon standard bobbin probe signal amplitude. The plant-specific guidelines used for all inspections shall be amended as appropriate to accommodate the additional information needed to evaluate tube support plate signals with respect to the voltage/depth parameters. Pending incorporation of the voltage verification requirement in ASME standard verifications, an ASME standard calibrated against the laboratory standard (see WCAP-13990, Rev. 0) will be utilized in the Sequoyah Nuclear Plant Unit 2 steam generator inspections for consistent voltage normalization."

   "1. A tube can remain in service if the signal amplitude of a crack indication is less than or equal to 2.0 volts, regardless of the depth of tube wall penetration, if, as a result, the projected end-of-cycle distribution of crack indications is verified to result in primary-to-secondary leakage less than 4.3 gpm in the faulted loop during a postulated steam line break event. The methodology for calculating expected leak rates from the projected crack distribution must be consistent with WCAP-13990, Rev. 0, and as prescribed in Draft NUREG-1477.

"2. A tube should be plugged or repaired if the signal amplitude is greater than 2.0 volts except as noted in Specification 4.4.5.4.a.10.3 below.

"3. A tube can remain in service with a bobbin coil signal amplitude greater than 2.0 volts but less than or equal to 3.6 volts if a rotating pancake probe inspection does not detect degradation. Indications of degradation with a bobbin coil signal amplitude greater than 3.6 volts will be plugged or repaired."

5. Add SRs 4.4.5.5.d and 4.4.5.5.e.

"d. The results of inspections performed under 4.4.5.2 for all tubes in which the tube support plate elevations interim plugging limit has been applied and where not plugged shall be reported to the Commission within 15 days following the inspection. The report shall include:

"1. Listing of applicable tubes.

"2. Location (applicable intersections per tube) and extent of degradation (voltage).

"e. The result of steam line break leakage analysis performed under T/S 4.4.5.4.a.10 will be reported to the Commission prior to restart for Cycle 7."

6. Change Limiting Condition for Operation (LCO) 3.4.6.2.c to state:

"600 gpd total primary-to-secondary leakage through all steam generators and 150 gallons per day through any one steam generator."

7. Change Bases 3/4.4.5, "Steam Generators," 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 the 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 tube support plate (TSP) intersections that are based on maintaining structural and leakage integrity of tubes with indications within the TSP regions. The alternate plugging criteria (APC) are an interim criteria that are applicable to Unit 2 Cycle 7 operation. 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 proposed change would preserve the reactor coolant flow margin and reduce the radiation exposure incurred in the process of plugging or repairing 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 interim plugging criteria (IPC) would be a reduction in man-hours and potential impact to critical path time during the Unit 2 Cycle 6 refueling outage.

TVA's goal is to prolong S/G life over the expected plant life of 40 years. 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 IPC for SQN Unit 2 can be summarized as follows:

Tube Plugging Criteria

Tubes with bobbin flaw indications exceeding the 2.0-volt IPC voltage repair limit and less than or equal to 3.6 volts are plugged or repaired if confirmed as flaw indications by RPC inspection. Bobbin flaw indications greater than 3.6 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 (POD) adjustment and allowances for nondestructive examination (NDE) uncertainties and ODSCC growth rates, must be less than 4.3 gallons per minute (gpm) 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 Probability

The projected end-of-cycle SLB tube burst probability shall be calculated and compared with the value of 2.5×10^{-2} found acceptable in NUREG-0844.

Exclusions from Tube Plugging Criteria

Indications excluded from application of the IPC repair limits include:

1. indications found by inspection (bobbin or RPC) to extend outside the TSP,
2. indications not attributable to ODSCC, and
3. circumferential indications.

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

Inspection requirements to be fulfilled if the proposed IPC is implemented at SQN are as follows:

- Eddy current analysis guidelines and voltage normalization must be consistent with that of Westinghouse Commercial Atomic Power (WCAP) 13990 Appendix A.

- Eddy current analysts shall be trained specifically to voltage sizing in accordance with the WCAP-13990 Appendix A analysis guidelines, and at least lead analysts shall be qualified to the industry standard Qualified Data Analysis program of the Electric Power Research Institute (EPRI) Pressurized Water Reactor S/G Examination Guidelines.

- Use of American Society of Mechanical Engineers (ASME) standards cross-calibrated to the reference laboratory standard and use of a probe wear standard requiring probe replacement at a voltage change of 15 percent from that found for the new probe shall be implemented in accordance with WCAP-13990 Appendix A.

- 100 percent bobbin coil examination of all hot leg TSPs and all cold leg TSPs at which ODSCC has been detected will be primarily performed with a 0.740-inch-diameter bobbin probe (0.720 bobbin coil probes may also be utilized).

- RPC inspection of all bobbin indications greater than the 1.5 volts shall be inspected to confirm axial ODSCC as the dominant mechanism for indications at the TSPs.

- RPC sample inspection of at least 100 TSP intersections with dents or artifact/residual signals that could potentially mask a 2.0-volt bobbin coil signal. The RPC sample shall emphasize dented TSP intersections, but include artifact signals that the analysts judge could mask a repairable indication. Any RPC detected flaw indication in this sample will be plugged or repaired.

- The projected end-of-cycle SLB tube burst probability shall be calculated and compared with the value of 2.5×10^{-2} found acceptable in NUREG-0844.

- NRC will be informed before plant restart from the Unit 2 Cycle 6 refueling outage of any unexpected inspection findings relative to the assumed characteristics of the flaws at the TSP intersections. This includes any detectable circumferential indications or detected ODSCC indications extending outside the thickness of the TSP.

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. The proposed IPC for SQN Unit 2 applies the criteria approved for Farley Unit 1 and D. C. Cook Unit 1 nuclear plants. This IPC includes inspection requirements and the 2.0-volt repair criteria. SLB analyses utilize methods and data described in the Farley Unit 1 and D.C. Cook Unit 1 safety evaluation reports.

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.

Two pulled TSPs from SQN Unit 1 support ODSCC as the dominant corrosion mechanism consistent with the EPRI database of pulled tubes. The EPRI database, which includes the SQN pulled tube data, is more conservative for SLB leak rate analyses using draft NUREG-1477 methodology than the data obtained from the SQN pulled tubes. Therefore, the more conservative EPRI database is 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 (Enclosure 4), the bobbin voltage corresponding to $1.43\Delta P_{SLB}$ (3,657 pounds per square inch {psi}) is 8.82 volts.

The structural limit must be reduced by allowances for NDE uncertainties and crack growth during the Cycle 7 fuel cycle. The EPRI ARC apply 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 Sequoyah, there is insufficient prior 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 provide 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 alternate plugging criteria repair limit is obtained by dividing the structural limit of 8.82 volts by 1 plus 64 percent (20.5 percent for NDE uncertainties and 43 percent for crack voltage growth). Thus, the full EPRI APC defined repair limit is obtained as 5.4 volts. The Farley Unit 1 and D.C. Cook Unit 1 IPC conservatively applied 3.6 volts for the full repair limit. Since the proposed SQN IPC follows this precedence, the 3.6-volt limit is also applied for the SQN IPC repair limit. This repair limit conservatively bounds the IPC limit obtained by applying either the EPRI database, as described above, or the NRC database additions described in WCAP-13990, Section 5.1.

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 effluent 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 (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2

DOCKET NO. 50-328

(TVA-SQN-TS-94-09)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Significant Hazards Evaluation

TVA has evaluated the proposed 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) 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 500 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 6,300 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). Test data indicates that tube burst cannot occur within the TSP, even for tubes that have 100 percent throughwall electrodischarge machining notches, 0.75-inch long, provided that the TSP is adjacent to the notched area. 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, 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).

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 interim plugging criteria 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,

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

It is reasonable that this RL (5.4 volts) could be applied for interim plugging criteria (IPC) implementation to repair bobbin indications greater than 2.0 volts independent of rotating pancake coil (RPC) confirmation of the indication. Conservatively, an upper limit of 3.6 volts will be used to assess tube integrity for those bobbin indications that are above 2.0 volts but do not have confirming RPC calls. This 3.6-volt upper limit for nonconfirmed RPC calls is consistent with other recently approved IPC programs.

The conservatism of the growth allowance used to develop the repair limit is shown by the most recent SQN eddy current data. Two tubes plugged in Unit 1 during the last outage had less than one volt of growth over the past five operating cycles. Only one tube in Unit 2 required repair because of outer-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 IPC. In support of implementation of the IPC, it will determine whether the distribution of cracking indications at the TSP intersections at the end of Cycle 7 for Unit 2 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 4.3 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 projected SLB leakage rate calculation methodology prescribed in Section 3.3 of draft NUREG-1477 is 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 of 1.0.

Application of the criteria requires the projection of postulated SLB leakage, based on the projected EOC voltage distribution for Cycle 7. 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. Draft NUREG-1477 requires that all indications to which the IPC 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. This indication also was the largest recorded bobbin voltage from the EOC eddy current data. 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 IPC 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. The current leakage level at SQN is less than 1.0 gallon per day (gpd).

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. Using NUREG-1477 and EPRI leakage models, the leakage component from an indication of this magnitude is 0.12 and 0.028 gpm, respectively.

Therefore, as implementation of the 2.0-volt IPC criterion during Cycle 7 in Unit 2 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 any 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 IPC criteria 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 have been applied (during all plant conditions).

   TVA will implement a maximum leakage rate limit of 150 gpd 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 are to be a maximum of 0.42 gpm (600 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 IPC 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 bobbin probe interim TSP elevation plugging criteria 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 that 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 considered applicable to SQN.

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

TVA COMMITMENTS

COMMITMENTS

1. Eddy current analysts shall be trained specifically to voltage sizing in accordance with the WCAP-13990 Appendix A analysis guidelines, and at least lead analysts shall be qualified to the industry standard Qualified Data Analysis program of the Electric Power Research Institute (EPRI) Pressurized Water Reactor S/G Examination Guidelines. Training shall be completed prior to S/G inspection during the Unit 2 Cycle 6 refueling outage.

2. Use of ASME standards cross-calibrated to the reference laboratory standard and use of a probe wear standard requiring probe replacement at a voltage change of 15 percent from that found for the new probe shall be implemented per WCAP-13990 Appendix A. Implementation shall be completed prior to restart from the Unit 2 Cycle 6 refueling outage.

3. RPC inspection of all bobbin indications greater than the 1.5 volts shall be inspected to confirm axial ODSCC as the dominant mechanism for indications at the TSPs. Inspection, if needed, shall be completed prior to restart from the Unit 2 Cycle 6 refueling outage.

4. RPC sample inspection of at least 100 TSP intersections with dents or artifact/residual signals that could potentially mask a 2.0-volt bobbin coil signal. The RPC sample shall emphasize dented TSP intersections but include artifact signals that the analysts judge could mask a repairable indication. Any RPC detected flaw indication in this sample will be plugged or repaired. Repair or plugging shall be completed, as applicable, prior to restart from the Unit 2 Cycle 6 refueling outage.

5. The projected end of cycle SLB tube burst probability shall be calculated and compared with the value of 2.5×10^{-2} found acceptable in NUREG-0844. Calculations and comparisons shall be completed prior to restart from the Unit 2 Cycle 6 refueling outage.

6. NRC will be informed before plant restart from the Unit 2 Cycle 6 refueling outage of any unexpected inspection findings relative to the assumed characteristics of the flaws at the TSP intersections. This includes any detectable circumferential indications or detected ODSCC indications extending outside the thickness of the TSP.

ENCLOSURE 5

SEQUOYAH UNIT 2

WESTINGHOUSE COMMERCIAL ATOMIC POWER 13990

STEAM GENERATOR TUBE

PLUGGING CRITERIA FOR INDICATIONS

AT TUBE SUPPORT PLATES

(PROPRIETARY VERSION)

ENCLOSURE 7

WESTINGHOUSE ELECTRIC CORPORATION

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