April 3, 1996

Mr. Oliver D. Kingsley, Jr.
President, TVA Nuclear and
Chief Nuclear Officer
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: ISSUANCE OF TECHNICAL SPECIFICATION AMENDMENT FOR THE SEQUOYAH NUCLEAR PLANT UNIT 2 (TAC NO. M94235) (TS 95-23)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 211 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Unit 2. The amendment is in response to your application dated December 12, 1995, which was supplemented by letter dated March 4, 1996.

The amendment revises the technical specification surveillance requirements and bases to incorporate alternate steam generator tube plugging criteria at tube support plate intersections. The approach taken is based on guidance given in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking." The amendment is approved for Cycle 8 operation only.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

David E. LaBarge, Sr. Project Manager
Project Directorate II-3
Division of Reactor Projects - I/I
Office of Nuclear Reactor Regulation

Docket No. 50-328

Enclosures: 1. Amendment No. 211 to License No. DPR-79
2. Safety Evaluation

cc w/enclosures: See next page

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Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 211
License No. DPR-79

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated December 12, 1995, which was supplemented by letter dated March 4, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 211, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented within 45 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Frederick J. Hebdon

Frederick J. Hebdon, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: April 3, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 211

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
3/4 4-11	3/4 4-11
---	3/4 4-11a
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
---	3/4 4-14a
---	3/4 4-14b
3/4 4-18	3/4 4-18
B3/4 4-3	B3/4 4-3
---	B3/4 4-3a
B3/4 4-4	B3/4 4-4

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

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

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

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

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

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

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

Note: Tube degradation identified in the portion of the tube that is not a reactor coolant pressure boundary (tube end up to the start of the tube-to-tubesheet weld) is excluded from the Result and Action Required in Table 4.4-2. R181

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. *

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.

*The indicated changes to this page are applicable to Cycle 8 operation only. *

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)

4.4.5.4 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 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. *

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

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

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

10. Tube Support Plate Plugging Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominately axially oriented outside diameter stress corrosion cracking confined within the thickness of the *

*The indicated changes to this page are applicable to Cycle 8 operation only.

tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:

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

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

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

d. Not applicable to SQN.

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

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

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

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

*The indicated changes to this page are applicable to Cycle 8 operation only.

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

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 <u>Reports</u>

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.

*The indicated changes to this page are applicable to Cycle 8 operation only. *

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

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

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

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

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

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

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

*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,

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

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

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:

*Replacement of c. with cc. is applicable for Cycle 8 operation only. *

3/4.4.5 STEAM GENERATORS

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

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

The voltage-based repair limits of SR 4.4.5 implement the guidance in * GL 95-05 and are applicable only to Westinghouse-designed steam generators (S/Gs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of S/G tube degradation nor are they applicable to ODSCC that occurs at other locations within the S/G. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of SR 4.4.5 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

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

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

*The indicated changes to this page are applicable to Cycle 8 operation only.

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

The mid-cycle equation of SR 4.4.5.4.a.10.e should only be used during unplanned inspection in which eddy current data is acquired for indications at the tube support plates.

SR 4.4.5.5 implements several reporting requirements recommended by GL 95-05 for situations which NRC wants to be notified prior to returning the S/Gs to service. For SR 4.4.5.5.d., Items 3 and 4, indications are applicable only where alternate plugging criteria is being applied. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the S/Gs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing GL Sections 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per GL Section 6.b(c) criteria.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the repair limit defined in Surveillance Requirement 4.4.5.4.a. The portion of the tube that the plugging limit does not apply to is the portion of the tube that is not within the RCS pressure boundary (tube end up to the start of the tube-to-tubesheet weld). The tube end to tube-to-tubesheet weld portion of the tube does not affect structural integrity of the steam generator tubes and therefore indications found in this portion of the tube will be excluded from the Result and Action Required for tube inspections. It is expected that any indications that extend from this region will be detected during the scheduled tube inspections. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. * R181

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. *

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

*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.

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 and cycle distribution of crack indications results in primary-to-secondary leakage greater than 3.7 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 3.7 gpm. *

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. *

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 211 TO FACILITY OPERATING LICENSE NO. DPR-78

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT UNIT 2

DOCKET NO. 50-328

1.0 INTRODUCTION

By letter dated December 12, 1995, and a supplement dated March 4, 1996, the Tennessee Valley Authority (the licensee) submitted an application to amend License DPR-79 to change the technical specifications (TS) for the Sequoyah Nuclear Plant, Unit 2 (SQN-2). The licensee proposed to incorporate voltage-based repair criteria for steam generator tubing into TS Sections 4.4.5.2, 4.4.5.4, 4.4.5.5, 3.4.6.2, Bases 3/4.4.5, and Bases 3/4.4.6.2. The proposed TS changes were requested for Cycle 8 operation only. The tube repair criteria will be implemented in the upcoming steam generator inspection during Cycle 7 refueling outage scheduled for April 1996.

The supplement supplied additional information that did not affect the previous no significant hazards consideration.

Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," provides guidance on the voltage-based (alternate) repair criteria for steam generator tubing. The voltage-based repair criteria allow tubes having axial cracking, resulting from outside diameter stress corrosion cracking (ODSCC) that is confined within the thickness of the tube support plate (TSP) intersections, to remain in service on the basis of eddy current inspection and acceptable structural integrity analyses. Attachment 1 to GL 95-05 provides technical guidance regarding implementation of the alternate repair criteria. Attachment 2 to GL 95-05 is a model TS containing specific acceptance criteria.

Sequoyah Unit 2 has four Westinghouse Model 51 steam generators, which use Alloy 600 mill-annealed tubing. These steam generators use drilled-hole TSP and no flow distribution baffle plates. The outside diameter and wall thickness of each tube are 0.875 inch and 0.050 inch, respectively.

2.0 EVALUATION

2.1 Assessment of Radiological Consequences

The licensee performed an assessment of the radiological dose consequences of a main steam line break accident in support of their amendment request to apply a voltage-based repair criteria for the Sequoyah Unit 2 steam generator tube support plate intersections experiencing outside diameter stress corrosion cracking. In performing this assessment, the licensee assumed a 3.7 gpm primary to secondary leak in the faulted steam generator initiated by the accident and the TS-allowable primary to secondary leakage value from each intact steam generator of 150 gpd. The licensee also assumed a primary coolant activity level of 1.0 μCi/g dose equivalent ^{131}I and a secondary coolant activity level of 0.1 μCi/g dose equivalent ^{131}I. On the basis of this assessment, the licensee determined that the radiological consequences of a main steam line break accident (assuming the above leakages) would be within 10 percent of the 10 CFR 100 guidelines.

The staff has independently calculated the doses resulting from a main steam line break accident using the methodology associated with SRP 15.1.5, Appendix A of NUREG-0800. The staff performed two separate assessments. The first assessment was based upon a pre-existing iodine spike activity level of 60 μCi/g of dose equivalent ^{131}I in the primary coolant. The other assessment was based upon an accident initiated iodine spike. For the accident initiated spike case, the staff assumed that the initial primary coolant activity level was 1.0 μCi/g of dose equivalent ^{131}I. The accident initiated an increase in the release rate of iodine from the fuel by a factor of 500 over the release rate to maintain an activity level of 1.0 μCi/g of dose equivalent ^{131}I in the primary coolant. For each of these two cases, the staff calculated doses for individuals located at the Exclusion Area Boundary (EAB) and at the Low-Population Zone (LPZ). For each case, the staff also calculated the thyroid dose to the control room operator. The parameters which were utilized in the staff's assessment are contained in Table 1. The doses calculated by the staff for each of the two cases are contained in Table 2.

The staff's calculations showed that the thyroid doses for the EAB and LPZ would be less than the guidelines established by SRP 15.1.5, Appendix A of NUREG-0800. The calculated thyroid dose to the control room operator would be less than the guidelines of SRP 6.4 of NUREG-0800. Therefore, the staff concludes that a leak rate design limit of 3.7 gpm is acceptable for the maximum primary to secondary leakage initiated in the faulted steam generator by the main steam line break accident.

TABLE 1

INPUT PARAMETERS FOR SEQUOYAH EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT

1. Primary coolant concentration of 60 μCi/g of dose equivalent ^{131}I.

Pre-existing Spike Value (μCi/g)

^{131}I	=	46.3
^{132}I	=	16.7
^{133}I	=	74.2
^{134}I	=	10.4
^{135}I	=	40.8

2. Volume of primary coolant and secondary coolant.

Primary Coolant Volume (ft^3)	12,600
Primary Coolant Temperature (°F)	590
Secondary Coolant Steam Volume (ft^3)	3546
Secondary Coolant Liquid Volume (ft^3)	2322
Secondary Coolant Steam Temperature (°F)	526.2
Secondary Coolant Feedwater Temperature (°F)	434.6

3. TS limits for DE ^{131}I in the primary and secondary coolant.

Primary Coolant DE ^{131}I concentration (μCi/g)	1.0
Secondary Coolant DE ^{131}I concentration (μCi/g)	0.1

4. TS value for the primary to secondary leak rate.

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, total all SGs (gpd)	600

5. Maximum primary to secondary leak rate to the faulted and intact SGs.

Faulted SG (gpm)	3.7
Intact SGs (gpm/SG)	0.1

6. Iodine Partition Factor

Faulted SG	1.0
Intact SG	0.1
Primary to Secondary Leakage	1.0

7. Steam Released to the environment

Faulted SG (lbs/(0-10) minutes)	87,000 plus primary to secondary leakage
Intact SGs (lbs/(0-2) hours)	479,000 plus primary to secondary leakage
(lbs/(2-8) hours)	1,030,000 plus primary to secondary leakage

8. Letdown Flow Rate (gpm) 75

9. Release Rate for 1.0 μCi/g of Dose Equivalent ^{131}I

	Ci/hr
^{131}I =	9.75
^{132}I =	23.9
^{133}I =	24.6
^{134}I =	35.2
^{135}I =	25.2

10. Atmospheric Dispersion Factors (sec/m^3)

EAB (0-2 hours)	1.64×10^{-3}
LPZ (0-2 hours)	1.96×10^{-4}
(2-8 hours)	2.64×10^{-5}
Control Room (0-2 hours)	3.18×10^{-3}
(2-8 hours)	1.01×10^{-3}

11. Control Room Parameters

Filter Efficiency (%)	95
Volume (ft^3)	260,000
Makeup flow (cfm)	1,000
Recirculation Flow (cfm)	3,000
Unfiltered Inleakage (cfm)	51
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

TABLE 2

THYROID DOSES FROM SEQUOYAH MAIN STEAM LINE BREAK ACCIDENT (REM)

LOCATION	DOSE (REM)	
	Pre-Existing Spike	Accident-Initiated Spike
EAB	46.5*	27.1**
LPZ	7.5*	8.2**
Control Room**	4.4	6.7

* Acceptance Criterion = 300 rem thyroid
** Acceptance Criterion = 30 rem thyroid

2.2 Technical Specification Changes

In accordance with Attachment 2 to GL 95-05, the licensee proposed to incorporate the following specifications into TS Sections 3/4.4.5, Reactor Coolant System and 3/4.4.6, Reactor Coolant System Leakage, for SQN-2 for Cycle 8 operation only:

> 4.4.5.2.6.4. Indications left in service as a result of application of TSP voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.
>
> 4.4.5.2.d Implementation of the steam generator tube/TSP repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg TSP intersections down to the lowest cold-leg TSP with known ODSCC indications. The determination of the cold-leg TSP intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.
>
> 4.4.5.4.a.6 This [the existing 40-percent through-wall degradation plugging limit] definition does not apply to TSP 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.
>
> 4.4.5.4.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 predominantly axially oriented ODSCC confined within the thickness of the TSPs. At TSP intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described in Items a through e in this section.

a. Steam generator tubes, whose degradation is attributed to ODSCC within the bounds of the TSP with bobbin voltages less than or equal to the lower voltage repair limit (Note 1) will be allowed to remain in service.

b. Steam generator tubes, whose degradation is attributed to ODSCC within the bounds of the TSP with 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.

c. Steam generator tubes, with indications of potential degradation attributed to ODSCC within the bounds of the TSP with a bobbin voltage greater than the lower voltage repair limit (Note 1), but less than or equal to the upper voltage repair limit (Note 2), may remain in service if the degradation is not detected during inspection using the rotating pancake coil. Steam generator tubes having indications of ODSCC degradation with a bobbin voltage greater than the upper voltage repair limit (Note 2) will be plugged or repaired.

d. Not applicable to Sequoyah Unit 2

e. If an unscheduled midcycle inspection is performed, the midcycle repair limits as specified in Attachment 2 to GL 95-05 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.

Note 1: The lower voltage repair limit is 2.0 volts for 7/8-inch diameter tubing in the SQN-2 steam generators.

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

4.4.5.5.d For implementation of the voltage-based repair criteria to TSP intersections, notify the [NRC] staff before 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, based on 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 TSP intersections.

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

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

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

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

In addition to the above TS changes, the licensee also proposed to implement the guidelines in Section 3, Inspection Criteria, and Section 4, Tube Removal and Examination/Testing, of Attachment 1 to GL 95-05. To satisfy the reporting guidelines in Section 6 of Attachment 1 to GL 95-05, and as stated in its letter of December 12, 1995, the licensee will submit to the NRC (1) inspection results including metallurgical examinations, (2) voltage and associated uncertainty distributions, and (3) structural integrity evaluations within 90 days of unit restart.

2.3 Tube Inspection Program

The licensee's proposed tube inspection program is consistent with Section 3 of Attachment 1 to GL 95-05. For Sections 3.b.3; 3.c.2; and 3.c.3, the licensees provided either additional information or alternatives to GL 95-05.

Section 3.b.3 of Attachment 1 to GL 95-05 specifies that a rotating pancake coil (RPC) inspection should be performed of all dents with a bobbin coil voltage response greater than five volts because primary water stress corrosion cracking (PWSCC) and circumferential cracking may develop at dented TSP intersections. If circumferential cracking or PWSCC indications are detected, licensees may need to expand RPC inspections to include a sample of dents with bobbin coil voltages less than five volts. Inspecting with an RPC probe improves the ability to detect the onset of PWSCC and circumferential cracking at dented TSP intersections.

The licensee committed to selecting a 20-percent initial sample of dents less than five volts from the total population if circumferential cracking is detected in the tubes. The 20-percent sample would be concentrated at the affected TSP elevation and at all lower TSP elevations. The initial sample for each steam generator will be selected independently with the sample weighted toward the lower TSPs. If a circumferential crack is identified in a less than 5-volt sample, an additional 20-percent sample of the original population will be examined and again weighted toward the lower TSPs. If the initial sample or an expanded sample has no circumferential crack indications, no additional samples will be examined.

The licensee stated that axial indications in dents less than five volts that are structurally significant will be detected with bobbin coil examinations; therefore, the only RPC sample expansions that are planned are those described above. This is consistent with the Unit 1 experience and will be confirmed during the Unit 2 inspection.

The licensee's commitment to inspect intersections with dent signals less than five volts provides added assurance that voltage-based repair criteria are not being applied to tubes containing either PWSCC or circumferential cracks at the TSP intersections because the repair criteria in GL 95-05 apply only to predominately axially oriented cracks caused by ODSCC. The staff has determined that the licensee's proposed sampling plan for inspecting dents is in accordance with Section 3.b.3 of Attachment 1 to GL 95-05.

Section 3.c.2 of Attachment 1 to GL 95-05 specifies criteria for probe variability. GL 95-05 states that once the probe has been calibrated on the 20-percent through-wall holes, the voltage response of new bobbin coil probes for the 40-percent to 100-percent through-wall holes should not differ from the nominal voltage by more than ±10 percent.

Section 3.c.3 of Attachment 1 to GL 95-05 addresses probe wear criteria. GL 95-05 states that probe wear should be controlled by either an inline measurement device or through the use of a periodic wear measurement. When using the periodic wear measurement approach, if a probe is found to be out of specification, all tubes inspected since the last successful calibration should be reinspected with the new calibrated probe. Alternatives to this approach may be permitted subject to the staff approval. The staff believes that alternative methods may be used if an assessment is performed demonstrating that (1) they provide equivalent detection and sizing capability on a statistically significant basis when compared to the guidance in GL 95-05 and (2) they are consistent with current methods for determining the end-of-cycle voltage distributions that are used in the tube integrity analyses.

Since issuance of GL 95-05, the staff has been working with industry through the Nuclear Energy Institute (NEI) on the issues of probe wear and new probe variability as they relate to Sections 3.c.2 and 3.c.3 of Attachment 1 to GL 95-05. By two letters dated January 23, 1996, NEI proposed an alternative to the probe wear criteria in GL 95-05 and a methodology for implementing the ±10 percent probe variability criteria. The staff approved the NEI's proposals subject to certain observations and restrictions in a letter from Brian Sheron of NRC to Alex Marion of NEI, dated February 9, 1996. By a letter dated February 23, 1996, NEI addressed the staff's observations and restrictions and agreed to supply certain confirmatory information. The licensee, in its letter dated March 4, 1996, committed to implement the alternative criteria on probe wear and the industry methodology for limiting new probe variability as defined in the NEI's letters dated January 23 and February 23, 1996. The licensee will ensure that the confirmatory information related to new probe variability has been provided to the staff prior to requesting permanent alternate plugging criteria for Sequoyah Units 1 and 2. The staff finds that the licensee's commitment is acceptable.

2.4 Structural Integrity

2.4.1 Deterministic Structural Integrity Assessment

The licensee's tube repair limits are based on the burst pressure and the bobbin coil voltage correlation, which is derived from data of pulled tube and experimental (model boiler) tests. In accordance with GL 95-05, the licensee will use the burst pressure versus bobbin voltage correlation that contains all applicable data consistent with the latest industry database (including the tube pull data). The staff finds the licensee's proposed voltage limits acceptable given the current burst pressure/bobbin voltage database, the licensee's assumed growth rates, and the nondestructive examination uncertainty estimates.

To confirm the nature of the degradation occurring at the TSPs, Section 4 of Attachment 1 to GL 95-05 recommends that tubes be periodically pulled from the steam generators for destructive analysis. The removed tubes can confirm that (1) axial ODSCC is the dominant degradation mechanism; (2) monitor the degradation mechanism over time; (3) provide additional data to enhance the correlations between voltage and burst pressure, probability of leakage, and leak rate; and (4) assess inspection capability. GL 95-05 states that licensees should pull at least two tubes with the objective of retrieving as many TSP intersections as practical (a minimum of four TSP intersections) during the plant steam generator inspection outage that implements the voltage-based repair criteria or during an inspection outage preceding initial application of these criteria.

The licensee committed to pulling a minimum of two tubes and four TSP intersections during the Unit 2 Cycle 7 refueling outage and implementing a tube pull program consistent with GL 95-05.

2.4.2 Probabilistic Structural Integrity Assessment

GL 95-05 states that a structural integrity assessment be submitted to the NRC within 90 days of each restart following a steam generator inspection. The licensee committed to performing a probabilistic analysis to quantify the potential for steam generator tube ruptures, given a main steam line break, at the end of cycle. The results of the probabilistic analysis will be compared to a threshold value of $1x10^{-2}$ in accordance with GL 95-05. This threshold value will provide assurance that the probability of burst is acceptable considering the assumptions of the deterministic calculation and the results of the staff's generic risk assessment for steam generators contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates that ODSCC confined to within the thickness of the TSP could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation that was assumed and evaluated as acceptable in NUREG-0844.

The licensee intends to calculate the conditional probability of burst in accordance with GL 95-05. The licensee referenced WCAP-14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated January 1995, as a document containing the details of the methodology for calculating the conditional probability of burst given a main steam line break. The staff finds that the licensee's proposal to perform the burst probability calculation is in accordance with GL 95-05 and is acceptable for this outage-specific application.

2.5 Leakage Integrity

2.5.1 Normal Operational Leakage

In accordance with GL 95-05, the licensee will limit the amount of operating leakage through any one steam generator to 150 gallons per day as proposed in TS Section 3.4.6.2.cc. The staff finds this acceptable.

2.5.2 Accident Leakage Analysis

For the 90-day reporting requirement, the licensee will use the methodology in WCAP-14277 to calculate the total leak rate during a main steam line break. The model for calculating the steam generator tube leakage from the faulted steam generator during a postulated main steam line break event consists of (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage model) and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak-rate model). The licensee proposed methodology for determining the amount of primary-to-secondary leakage under postulated accident conditions has been previously approved by the staff as stated in Section 2.3.2 of this Safety Evaluation. The staff finds this methodology for calculating accident leakage is consistent with the guidance of GL 95-05 and is acceptable for use in this outage-specific application.

2.6 Accident Analysis

General Design Criterion 2 of Appendix A to 10 CFR 50 requires structures, systems, and components important to safety be designed to withstand effects of normal and accident conditions with the effects of natural phenomena such as earthquakes. Section 1.b.1 of Attachment 1 to GL 95-05 specifies that the alternate plugging criteria do not apply to tube-to-TSP intersections where the tubes with degradation may collapse or deform under loss-of-coolant accident (LOCA) and safe shutdown earthquake (SSE) loadings. Licensees should perform or reference an analysis that identifies which intersections are to be excluded.

The staff was concerned about a scenario that the TSP may deform as a result of lateral loads at the wedge supports under the combined LOCA and SSE loadings. The pressure differential on the deformed tubes may cause some of the deformed tubes to collapse. There are two concerns associated with tube collapse. First, the collapse of steam generator tubes reduces the flow area of the reactor coolant system. The reduction in flow area restricts the steam flow from the core during a LOCA, which in turn, may increase peak clad temperature of the reactor core. Second, existing shallow cracks in tubes may propagate to through-wall cracks when tubes are deformed or collapsed.

The licensee stated that the accident analysis submitted by Southern Nuclear Company for Farley Units 1 and 2, which also have Westinghouse Series 51 steam generators, is applicable to SQN-2. The Farley analysis is documented in Westinghouse topical report WCAP-12871, "J. M. Farley Units 1 and 2 Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates," February 1992, Revision 2. Westinghouse analyzed steam generator tube integrity for Farley using LOCA loadings of the primary coolant loop, and main steam and feedwater line breaks (SLB/FLB) in combination with an SSE. The staff approved the Farley analysis as documented in a letter from S. Hoffman of NRC to W. Hairston of Southern Nuclear Operating Company, subject: "Issuance of Amendment No. 95 to Facility Operating License No. NPF-2 Regarding Steam Generator Tube Interim Plugging Criteria for Joseph M. Farley Nuclear Plant, Unit 1 (TAC NO. M84343)," dated October 8, 1992. The licensee has provided the requisite information in its submittals of September 7, 1995, and December 12, 1995 to establish the applicability of the analysis results for Farley to SQN-2.

The staff has approved the licensee's application of the leak-before-break approach to the reactor coolant loop piping. The leak-before-break analysis showed that breaks in the primary loop piping is sufficiently remote that they need not be considered in the design basis. The limiting LOCA load may, therefore, be derived from either the accumulator line break or the pressurizer surge line break. The licensee, however, opted to conservatively use the LOCA loads from the primary pipe breaks to bound the conditions at SQN-2 for breaks of smaller size piping.

Westinghouse evaluated tube deformation for Farley steam generators using TSP loads derived from the limiting large break LOCA event. A transient dynamic analysis for Farley for both primary piping and branch line breaks shows that the TSP loads resulting from the primary piping breaks are three to four times higher than that of the branch line breaks. Westinghouse determined that the induced pressure loadings from a large piping break at Farley bound the loadings from a branch line break for SQN Unit 2. Thus, using the primary pipe break loads for Farley to calculate tube deformation for SQN-2 is conservative for the SQN-2 branch line breaks.

The seismic loads for the Farley analysis were obtained from a generic seismic analysis for Series 51 steam generators. The generic analysis was performed using an umbrella spectra that was generated from the plant-specific spectra for a number of plants having Series 51 steam generators. The plant-specific spectra for SQN-2 was included in the generation of the umbrella spectra. Thus, the TSP loads from the umbrella analysis, which were used for the Farley evaluation, are also applicable to SQN-2.

Westinghouse assessed the effect of SSE bending stresses on the burst strength of tubes with axial cracks for Farley. The tensile stress in the tube wall would tend to close the cracks while compressive stress would tend to open the cracks. On the basis of previously-performed tests, Westinghouse determined that bending stress on the order of yield stress of the tube material is necessary before the burst strength of the tube is affected to any significant degree. The maximum bending stress on the tube wall calculated to occur during a seismic event at Farley was determined to be substantially less than the yield stress of the tube material. Since the seismic loads at Farley bound those at SQN-2, the licensee determined that the burst strength of tubes with through-wall cracking is not affected by an SSE event at SQN-2.

For Farley, Westinghouse evaluated the potential for SSE-induced bending stress in reducing the burst pressure of the tubes under combined SSE and SLB/FLB loads. The effect on burst strength is a function of the SSE bending stresses at TSP locations. Since the seismically-induced tube stresses are the result of a generic analysis that bounds the SQN-2 spectra, the SSE stresses used in the Farley analysis also apply to SQN-2. Therefore, the effect on burst strength of the combined SSE plus SLB/FLB stresses for Farley also applies to SQN-2.

The characteristics of TSP deformation used in the Farley analysis are based on crush tests performed for Series 51 SGs. The TSP loads were used to calculate tube deformation and consequent reduction in the flow area of the tubes. The TSP geometry and wedge configuration (load transfer locations) are the same for both Farley and SQN-2. Thus, the TSP deformation characteristics are the same for both plants. Because the loads used to calculate flow area reduction for Farley are conservative for SQN-2, the flow area reduction calculations will be conservative for SQN-2.

The radial loads from the LOCA and SSE events could result in yielding in the TSP at the wedge support. Some tubes in the vicinity of the wedge supports could partially deform and subsequently collapse. The resulting in-leakage is a potential concern since the cumulative leakage may cause an increase in the core peak clad temperature.

Utilizing results from previous tests and analysis programs, Westinghouse showed for the Farley plant that tubes will deform permanently if the change in diameter exceeds 0.025 inch. This threshold for tube deformation is related to the concern for tubes with pre-existing tight cracks that could potentially open during a combined LOCA plus SSE event. For the Farley plant, the LOCA plus SSE loads were determined to be of such magnitude that none of the tubes are predicted to exceed this deformation limit and, therefore, will not lead to significant tube leakage. Based on the applicability of the analyses for the Farley plant to SQN-2 established earlier, these results would be bounding for SQN-2.

From the above evaluation, the staff determines that (1) the accident analyses for the steam generators at Farley are applicable to the SQN-2 steam generators and (2) significant tube leakage is not likely to occur during an SSE plus LOCA event, which has been identified as the limiting condition from tube deformation considerations for the steam generators at SQN-2. Therefore, at SQN-2, no tubes will be excluded from using the voltage repair criteria due to tube deformation or collapse following an LOCA and SSE event.

3.0 Summary

Based on its review of the information provided by the licensee, the staff has determined that (1) the licensee's proposed plan to implement the voltage-based repair criteria for SQN-2 steam generator tubing comply with GL 95-05;

(2) the licensee's method in calculating structural and leakage integrity of steam generator tubing is in accordance with GL 95-05; (3) the licensee's proposed changes to SQN-2 TS to incorporate the voltage-based repair criteria, limited to Cycle 8 operation, comply with GL 95-05; and (4) a leak rate of 3.7 gpm is an acceptable limit for the maximum primary to secondary leakage initiated in the faulted steam generator by the main steam line break accident.

Therefore, the staff concludes that the licensee may incorporate the proposed voltage-based repair criteria for steam generator tubing in the TS for SQN-2, for Cycle 8 operation. In addition, based on this radiological, accident, and tube inspection program analyses, the staff has determined that the proposed amendment is acceptable for SQN-2, Cycle 8 operation.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 183). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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