October 11, 1995

50-327

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 1 (TAC NO. M92961) (TS 95-15)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 214 to Facility Operating License No. DPR-77 for the Sequoyah Nuclear Plant, Unit 1. The amendment is in response to your application dated July 19, 1995, which was superseded by your letter dated September 7, 1995, and supplemented by letters dated September 15 and 26, 1995.

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 similar to guidance given in Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

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. 50-327

Enclosures: 1. Amendment No. 214 to License No. DPR-77
2. Safety Evaluation

cc w/enclosures: See next page

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 214
License No. DPR-77

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 19, 1995, which was superseded by letter dated September 7, 1995, and supplemented by letters dated September 15 and 26, 1995, complies with the standards and requirements of the Atomic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-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 214, 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: October 11, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 214

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

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 4-7	3/4 4-7
3/4 4-9	3/4 4-9
---	3/4 4-9a
---	3/4 4-9b
3/4 4-10	3/4 4-10
3/4 4-14	3/4 4-14
B3/4 4-3	B3/4 4-3
B3/4 4-4	B3/4 4-4
---	B3/4 4-4a
---	B3/4 4-4b

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

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

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

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

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

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

d. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length. *

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

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

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

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

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 occuring or either inside or outside of a tube.

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

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

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

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

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

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

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

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 *

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

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

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

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

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

d. Not applicable to SQN.

e. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.10.a, 4,4.5.4.a.10.b, and 4.4.5.4.a.10.c.

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

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

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

*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 95-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle.

*

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

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

4.4.5.5 Reports

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

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

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported pursuant to Specification 6.6.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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

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

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

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

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

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

*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. *

BASES

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

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

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

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

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

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

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

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

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

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

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

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

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. *

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 214 TO FACILITY OPERATING LICENSE NO. DPR-77

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT UNIT 1

DOCKET NO. 50-327

1.0 INTRODUCTION

By application dated July 19, 1995, the Tennessee Valley Authority (TVA or the licensee) proposed an amendment to the Technical Specifications (TS) for Sequoyah Nuclear Plant (SQN) Units 1 and 2. The requested changes would revise TS surveillance requirements and bases to incorporate alternate steam generator (SG) tube plugging criteria at tube support plate (TSP) intersections. The approach taken is similar to guidance given in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995.

By letter dated September 7, 1995, TVA superseded the July 19, 1995 application. The new application withdrew the proposed changes to Unit 2, supplied revised data applicable to Unit 1, and indicated that implementation of the inspection/reinspection requirements would be applicable for the next operating cycle on Unit 1 (Cycle 8) only. In addition, letters dated September 15 and 26, 1995, supplied supplemental information. None of these subsequent letters changed the no significant hazards consideration that was originally published for this amendment request. This safety evaluation addresses the proposed TS changes for Unit 1 only. The Notice of Withdrawal for Unit 2 has been handled separately.

During a telephone call held on September 11, 1995, the licensee agreed to the addition by the staff of a note to each TS page affected by this amendment that would state, "The indicated changes to this page are applicable to Cycle 8 operation only."

2.0 EVALUATION

2.1 Assessment of Radiological Consequences

2.1.1 Background

In support of the amendment request to apply a voltage-based repair limit for the SQN Unit 1 SG tube support plate intersections experiencing outside diameter stress corrosion cracking, the licensee stated that their assessment

ENCLOSURE

of the radiological dose consequences of a main steam line break (SLB) accident was based upon a 3.7 gpm primary to secondary leak initiated by the accident in the faulted SG and the TS allowable value for primary to secondary leakage from each intact SGs of 150 gpd per SG. The licensee's conclusion concerning the acceptability of the radiological doses also assumed an allowable activity level of dose equivalent ^{131}I of 1.0 $\mu Ci/g$ in the primary coolant and 0.1 $\mu Ci/g$ in the secondary coolant.

2.1.2 Analysis

The staff has independently calculated the doses resulting from a main steam line break accident using the methodology associated with Standard Review Plan (SRP) 15.1.5, Appendix A. Two assessments were performed. One was based upon a pre-existing iodine spike activity level of 60 $\mu Ci/g$ of dose equivalent ^{131}I and the other was based upon an accident initiated iodine spike. For the accident initiated spike, the staff assumed that the primary coolant activity level was 1.0 $\mu Ci/g$ 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 $\mu Ci/g$ of dose equivalent ^{131}I in primary coolant. For these two cases, the staff calculated doses for individuals located at the Exclusion Area Boundary (EAB) and at the Low-Population Zone (LPZ).

The control room operator's thyroid dose was also calculated. The parameters that were used in the staff's assessment are shown in Table 1. The doses calculated by the staff are shown 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. The control room operator thyroid dose would be less than the guidelines of SRP 6.4 of NUREG-0800.

Therefore, the staff concluded that, based upon an acceptance criterion of 300 rem thyroid at the EAB for the pre-existing spike case and an acceptance criterion of 30 rem thyroid dose for the accident initiated spike case and for the control room operator dose assessments, a leak rate of 3.7 gpm is an acceptable limit for the maximum primary to secondary leakage initiated in the faulted SG by the main steam line break accident. Consequently, the results of the radiological analysis are acceptable.

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)	3,546
Secondary Coolant Liquid Volume (ft^3)	2,322
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
Intact SG	0.1
Primary to Secondary Leakage	1.0

7. Steam Released to the environment

Faulted SG (lbs/2 hours)	87,000 plus primary to secondary leakage
Intact Sgs (lbs/2 hours)	479,000 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

EAB (0-2 hours)	1.64×10^{-3}
LPZ (0-8 hours)	1.96×10^{-4}
Control Room (0-2 hours)	3.18×10^{-3}
Control Room (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)	2,600
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)

	DOSE	
LOCATION	**Pre-Existing Spike**	**Accident-Initiated Spike****
EAB	47.0*	25.6
LPZ	7.7*	8.2
Control Room **	10.0	15.0

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

2.2 Accident Analysis

2.2.1 Background and Analysis

Accident analyses for Model 51 SGs have been documented in Westinghouse topical report WCAP-12871, Revision 2, February 1992 for the J. M. Farley Units 1 and 2 SGs. These analyses relate to SG tube integrity during postulated accidents involving breaks in the primary coolant loop (loss of coolant accident- LOCA), and main SLB and feedwater line break (FLB), in combination with a safe shutdown earthquake (SSE). The staff has previously reviewed and accepted these analyses for the J. M. Farley nuclear plant SGs. The licensee has provided the requisite information in its submittals of July 19 and September 7, 1995 to establish the applicability of the analysis results for Farley to SQN.

The seismic loads for the Farley analysis were obtained from a generic seismic analysis for Series 51 SGs. The generic analysis was performed using an umbrella spectra that was generated from the plant-specific spectra for a number of plants with Series 51 SGs. The plant-specific spectra for SQN were included in the generation of the umbrella spectra. Thus, the tube support plate (TSP) loads from the umbrella analysis, which were used for the Farley evaluation, are also applicable to the SQN units.

The tube deformation calculations for Farley were performed using TSP loads for the most limiting large break LOCA event. A transient dynamic analysis for Farley for both primary piping and branch line breaks shows the primary breaks to result in TSP loads that are three to four times higher than the branch line breaks. It has subsequently been determined that the induced pressure loadings from a large piping break at Farley bound the loadings from a branch line break for SQN. Thus, using the large pipe break loads for Farley to calculate tube deformation provides a conservative basis for the SQN branch line breaks.

The TSP deformation characteristics 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 flow area. The TSP geometry and wedge configuration (load transfer locations) are the same for both Farley and SQN Unit 1. Thus, the TSP deformation characteristics are the same for both plants. Since the loads used to calculate flow area reduction for Farley are a conservative basis for SQN Unit 1, the flow area reduction calculations will be conservative.

Combined SSE plus SLB/FLB loads were evaluated for Farley relative to the potential for SSE-induced bending stress to reduce the burst pressure for the tubes. 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 Unit 1 spectra, the SSE stresses used in the Farley analysis also apply to SQN. Therefore, the discussion in Reference 1, relative to the effect on burst strength of the combined SSE plus SLB/FLB stresses for Farley also applies to SQN.

The radial loads due to combined LOCA and SSE could potentially 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 during a LOCA. The reduction in flow area increases the resistance to flow of steam from the core which in turn may potentially increase core peak clad temperature (PCT). In addition, there is a potential concern that tubes with partial through-wall cracks could progress to through-wall cracks during tube deformation. The resulting in-leakage is a potential concern since the cumulative leakage may cause an increase in the core PCT.

Utilizing results from previous tests and analysis programs, it has previously been shown for the Farley plant that tubes will undergo permanent deformation 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, these results would be bounding for SQN Unit 1.

The effect of SSE bending stresses on the burst strength of tubes with axial cracks was assessed for the Farley plant in Reference 1. 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, it was 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, it is concluded that the burst strength of tubes with through-wall cracking is not affected by an SSE event at SQN.

Based on a review of the information provided by the licensee, the staff concluded that the accident analyses performed for the Farley nuclear plant SGs are applicable to the SQN Unit 1 SGs. It is further concluded that no significant tube leakage is likely to occur during an SSE plus LOCA event that has been identified as the most limiting condition from tube deformation considerations for the SGs at SQN Unit 1. Therefore, the results of the accident analysis are acceptable.

2.3 Tube Inspection Program

2.3.1 Background

The staff has developed generic criteria for voltage-based limits for outside diameter stress corrosion cracking (ODSCC) confined within the thickness of the tube support plates. The staff has published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes" and in a draft generic letter titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the

Federal Register on August 12, 1994 (59 FR 41520). On August 3, 1995, the staff issued Generic Letter (GL) 95-05 that took into consideration public comments on the draft generic letter cited above, domestic operating experience under the voltage-based repair criteria, and additional data which have been made available from European nuclear power plants.

The licensee's initial amendment application dated July 19, 1995, was submitted to the NRC before the NRC issued GL 95-05. This submittal requested a permanent amendment to the SQN Unit 1 TS and contained exceptions to the guidance in GL 95-05. These exceptions are generic to the industry and will not be resolved prior to the date when the licensee would implement the proposed voltage-based repair criteria in the Cycle 7 refueling outage. The NRC staff discussed the areas of the amendment that deviated from the guidance in the GL during a phone call with the licensee on August 28, 1995, and indicated that the staff would not approve a permanent TS amendment due to several of the exceptions to GL 95-05 that were proposed. These exceptions relate to issues that are generic to the entire industry and will be fully resolved by the staff at a later date. The licensee resubmitted the TS change on September 7, 1995, and indicated that its proposed implementation of inspection/reinspection guidance of the GL would apply only to the next operating cycle (Cycle 8). In a phone call on September 11, 1995, the licensee agreed to the inclusion of a footnote on each affected TS page to clearly indicate that the changes are applicable to Cycle 8 only. Following further discussions with the staff, the licensee supplied additional information regarding the proposed TS change in submittals dated September 7 and 15, 1995.

The licensee's proposal is applicable to Cycle 8 operation and is consistent with GL 95-05 except as noted below.

Proposed changes to TS 4.4.5.2, 4.4.5.4, 4.4.5.5, and 3.4.6.2 and Bases 3/4.4.5 and 3/4.4.6.2 would specify the voltage-based tube repair criteria for ODSCC confined to within the thickness of the tube support plates. The changes are similar to those included in Attachment 2 to GL 95-05. The proposed changes for Cycle 8 implementation of the voltage-based tube repair criteria include, in part:

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

b. Specifying that the implementation of the SG 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 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.

c. Specifying that the 40-percent through wall degradation plugging limit definition of TS 4.4.5.4.a.6 does not apply to tube support plate intersections if the voltage-based repair criteria are being applied.

d. Including a tube support plate plugging limit used for the disposition of an alloy 600 SG tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining SG tube serviceability as described below:

   1. SG tubes, whose degradation is attributed to ODSCC within the bounds of the tube support plate with bobbin voltage less than or equal to the lower voltage repair limit (2.0 volts), will be allowed to remain in service.

   2. SG tubes, whose degradation is attributed to ODSCC within the bounds of the tube support plate with bobbin voltage greater than the lower voltage repair limit (2.0 volts), will be repaired or plugged, except as noted in d.3 below.

   3. SG tubes, with indications of potential degradation attributed to ODSCC within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit (2.0 volts), but less than or equal to the upper voltage repair limit (5.4 volts), may remain in service if a rotating pancake coil inspection does not detect degradation. SG tubes, with indications of ODSCC degradation with a bobbin voltage greater than the upper voltage repair limit (5.4 volts) will be plugged or repaired.

   4. If an unscheduled mid-cycle inspection is performed, the mid-cycle repair limits as specified in Attachment 2 of GL 95-05 apply instead of the limits identified in d.1, d.2 and d.3 above.

e. Adding the following reporting requirements:

   For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the Sgs 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 SLB) for the next operating cycle.

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

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

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

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

f. Specify a limit on primary-to-secondary leakage of 150 gallons per day through any one SG.

In addition to the above technical specification changes, the licensee has also made the following commitments:

1. The requested actions of GL 95-05 will be followed with the following exceptions: (1) the use of a probe wear standard, (2) the use of bobbin coil probes with the voltage response tolerance specified in Section 3.c.2 of GL 95-05, and (3) the inspection of all dents greater than 5 volts. These exceptions are discussed below (Section 2.3.2.1).

2. Calculation of the conditional probability of burst and total leak rate during a main steam line break (MSLB) will follow the methodology described in WCAP-14277, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections." As discussed in WCAP-14277, these methods are intended to be in accordance with the NRC's generic letter on voltage-based SG tube repair criteria (GL 95-05).

3. All intersections where copper signals interfere with the detection of flaws will be inspected with a motorized rotating pancake coil probe.

4. All intersections with large mixed residuals will be inspected with a rotating pancake coil probe.

5. All bobbin flaw indications with voltages greater than 2.0 volts will be inspected with a rotating pancake coil probe.

6. The licensee will perform an inspection of all dented tube support plate (TSP) intersections with bobbin coil voltages greater than 5 volts in the lower two hot-leg (HL) support plates of Sgs 3 and 4. The inspection will utilize a rotating pancake coil probe, or equivalent. In addition, the licensee will perform an inspection of 20-percent of the dented intersections over 5 volts at the third HL TSP in Sgs 3 and 4. All dents signals larger than 5 volts at the HL TSP intersections in Sgs 1 and 2 will be inspected with a rotating pancake coil.

7. The licensee will complete a sample inspection of dented TSP intersections less than 5 volts in accordance with the criteria in the licensee's letter to the NRC dated September 15, 1995.

2.3.2 Analysis

2.3.2.1 Inspection Issues

The licensee's inspection program is consistent with the guidance of GL 95-05 with the exception of the probe wear re-inspection requirements, the use of bobbin coil probes with the voltage response characteristics specified in Section 3.c.2 of the generic letter, and the guidance specifying an inspection of all dent signals greater than 5 volts with a rotating pancake coil (RPC). For the probe wear re-inspection requirements, the licensee proposes to use the same practices currently proposed by the industry. The industry (i.e., Nuclear Energy Institute (NEI)) approach is such that if any of the probe wear standard signal amplitudes prior to probe replacement exceed the ±15 percent limit, by a value of "X%", then any indications measured since the last acceptable probe wear measurement that are within "X%" of the plugging limit will be reinspected with the new probe. Alternatively, the voltage criterion may be lowered to compensate for the excess variation.

Regarding the proposed alternate procedures for re-inspecting tubes that fail to meet the probe wear criterion, the staff has concluded that alternate methods may be used provided an assessment is performed demonstrating the alternate methods (a) provide equivalent detection and sizing capability on a statistically significant basis when compared to the guidance in GL 95-05, and (b) are consistent with current methods for determining the end-of-cycle voltage distributions which are used in the tube integrity analyses. These assessments, along with the statistical criteria for demonstrating that the techniques are equivalent, should be provided to the NRC for review and approval. With respect to this cycle-specific application for SQN Unit 1 Cycle 8 operation, the NRC staff has concluded that the proposed alternate methods described in the September 7, 1995 submittal to meet the probe wear criterion are acceptable.

Section 3.c.2 of GL 95-05 specifies that the voltage response for the 40 percent to 100 percent through-wall holes of new bobbin coils calibrated on the 20 percent through-wall holes should not differ from the nominal voltage by more than ±10 percent. The industry previously presented limited details for resolving the issue of new probe variability in a meeting with the NRC staff on November 3, 1994. Since the NRC/NEI meeting, the NRC staff specifically mentioned several key areas to be addressed related to probe variability as documented in its review of public comments to the draft generic letter (memorandum dated May 30, 1995, from Mr. Frank J. Miraglia to Mr. Edward L. Jordan). During a phone call with the NRC staff on August 28, 1995, the licensee described its plan for meeting the guidance related to the probe variability issue. The proposed method addresses the areas outlined in the NRC memorandum dated May 30, 1995, and is incorporated into the licensee's submittal dated September 7, 1995. The staff has reviewed the licensee's plan for demonstrating acceptable probe variability and concluded that it is acceptable for this cycle-specific amendment.

There is the potential for the development of primary water stress corrosion cracking (PWSCC) and circumferential cracking at dented TSP intersections. GL 95-05 specifies that licensees should perform an RPC inspection of all dents with a bobbin coil voltage response greater than 5 volts. If circumferential cracking or PWSCC indications are identified, then it may be necessary to expand RPC inspections to include a sample of dents with bobbin coil voltages less than 5 volts. Inspecting with an RPC probe improves the ability to detect the onset of PWSCC and circumferential cracking at dented TSP intersections where bobbin coil signals may be difficult to interpret.

The licensee has identified considerable tube denting at the TSP elevations in the SQN Unit 1 SGs. The majority of these dents are in SGs 3 and 4 and are located at the lower HL TSP elevations. Due to the large number of dents present in the SQN Unit 1 SGs, the licensee has proposed to inspect a limited sample of dented TSP intersections with an RPC probe. An initial baseline inspection of dents during the current refueling outage would be followed by a reduced scope inspection in future outages.

The initial RPC inspection scope includes all of the dented TSP intersections with bobbin coil voltages greater than 5 volts in the lower two HL support plates of SGs 3 and 4. In addition, the licensee will perform an inspection of 20 percent of the dented intersections over 5 volts at the third HL TSP elevation in SGs 3 and 4. All dents larger than 5 volts at the HL TSP intersections in SGs 1 and 2 will be inspected with a rotating pancake coil. Any indications found at these intersections with RPC will be repaired since intersections with dent signals over 5 volts are specifically excluded from the voltage-based repair criteria per GL 95-05.

During the SQN Unit 1 outage in 1993, the licensee completed a 100 percent RPC inspection of all dents over 5 volts at the first through the fourth HL TSP elevations. The RPC inspections identified several PWSCC indications, and all but one indication was detectable with the bobbin coil probe. No circumferential cracking was apparent at any of the TSP intersections inspected. The RPC inspection plan related to the proposed TS change also includes criteria for expanding the scope of the inspections based on the results from the sample inspection. The expansion criteria were developed to address the possibility that PWSCC and circumferential cracking could occur at TSP dented intersections. These criteria include measures to assess the extent of PWSCC or circumferential cracking at dented intersections over 5 volts as well as a plan to inspect for the occurrence of these cracks at locations with dent signals less than 5 volts.

The NRC staff has reviewed the licensee's proposed sampling plan for the inspection of dented TSP intersections with an RPC probe. The large number of tubes to be inspected in the current refueling outage is sufficient to reveal the extent of PWSCC and the onset of circumferential cracking if such cracks are present. In addition, the licensee's commitment to expand RPC inspections to include dented intersections above and below 5 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. The NRC staff has concluded that the licensee's proposed RPC dent inspection sampling plan is acceptable for Cycle 8 implementation.

2.3.2.2 Structural Integrity - Deterministic Structural Integrity Assessment

The licensee's tube repair limits are based on a correlation between the burst pressure and the bobbin coil voltage of pulled tube and model boiler data. In accordance with GL 95-05, the licensee will use the burst pressure versus bobbin voltage correlation containing all applicable data consistent with the latest revision of the industry database as approved by NRC with the latest 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 non-destructive examination uncertainty estimates.

To confirm the nature of the degradation occurring at the tube support plate elevations, tubes are periodically removed from the SGs for destructive analysis. Tube pulls can confirm that the nature of the degradation being observed at the tube support plate elevations is predominantly axially oriented ODSCC, provide data for assessing the reliability of the inspection methods, and supplement the existing databases (e.g., burst pressure, probability of leakage, and leak rate). GL 95-05 contains guidance that states utilities should remove at least two pulled tube specimens with the objective of retrieving as many intersections as practical (a minimum of four intersections) during the plant SG inspection outage preceding initial application of the voltage-based repair criteria.

In 1992, the licensee removed a single tube with two TSP for metallographic examination, burst testing and leak rate testing from the Unit 1 SGs. A metallurgical examination performed on the tubes concluded that the dominant degradation mechanism for the indications at the support plate elevations in the pulled tubes was axially oriented ODSCC. In accordance with GL 95-05, the licensee will remove two pulled tube specimens with the objective of retrieving as many intersections as practical (a minimum of four intersections) during the Cycle 7 refueling outage.

2.3.2.3 Structural Integrity - Probabilistic Structural Integrity Assessment

The licensee will complete a probabilistic analysis to quantify the potential for SG tube ruptures, given an MSLB. The results of the probabilistic analysis will be compared to a threshold value of 1×10^{-2} per reactor-year in accordance with GL 95-05. This threshold value will provide assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the staff's generic risk assessment for SGs 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 tube support plate 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 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 an MSLB. The NRC staff has previously approved the use of methodology in WCAP-14277 for other one-cycle applications of voltage-based repair criteria as documented in the Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 106 to Facility Operating License NPF-8, Southern Nuclear Operating Company Incorporated, Joseph M. Farley Nuclear Plant, Unit 2, dated April 7, 1995. The staff notes that the acceptable method for calculating the distribution of bobbin indications as a function of voltage at the beginning of cycle is outlined in Section 2.b.1 of GL 95-05. The staff concludes that the licensee's proposed methodology for calculating the conditional burst probability is consistent with the guidance in GL 95-05 and is acceptable for use in this outage-specific application.

2.3.2.4 Structural Integrity - Normal Operational Leakage

In accordance with the guidance in GL 95-05, the licensee will limit the amount of operating leakage through any one SG to 150 gallons per day. This requirement was submitted in the proposed TS change, to be in effect for operation during the next operating cycle.

2.3.2.5 Accident Leakage

The licensee indicated that they will calculate the leakage and MSLB tube burst probability following the guidance of GL 95-05. In order to complete these calculations, the licensee will follow the methodology outlined in WCAP-14277. The model for calculating the SG tube leakage from the faulted SG during a postulated MSLB consists of two major components: (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 calculational methodology being proposed by the licensee for Unit 1 to determine the amount of primary-to-secondary leakage under postulated accident conditions has previously been reviewed and approved by the staff as stated above. The staff finds this methodology acceptable for an assessment of the Unit 1 SGs for operation in Cycle 8.

2.4 SUMMARY

The licensee submitted an application for a one cycle amendment to the SQN Unit 1, TS that would permit the use of voltage-based SG tube repair criteria. The licensee's submittal follows the guidelines provided in GL 95-05. The staff reviewed the proposed one-cycle amendment to the Unit 1 TS and concluded that the methods proposed by the licensee are consistent with the guidance in GL 95-05 except as noted above. The staff concludes that adequate structural and leakage integrity can be ensured, consistent with applicable regulatory requirements, for indications to which the voltage-based repair criteria will be applied for Cycle 8 operation.

Therefore, based on this radiological, accident and tube inspection program analyses, the staff has determined that the proposed amendment is acceptable.

The staff's approval of the voltage-based repair criteria is based, in part, on the licensee being able to demonstrate, in accordance with GL 95-05, that the projected end-of-cycle conditional probability of burst and the primary-to-secondary leakage during a postulated MSLB will be acceptable.

3.0 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.

4.0 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. 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 (60 FR 39189). 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.

5.0 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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