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 $\mu Ci/g$ dose equivalent ^{131}I and a secondary coolant activity level of 0.1 $\mu Ci/g$ 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 $\mu Ci/g$ 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 $\mu Ci/g$ 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 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

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(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)

	DOSE (REM)	
LOCATION	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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Dated: April 3, 1996