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By letter dated September 7, 1995, TVA superseded the July 19, 1995 application.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 μCi/g in the primary coolant and 0.1 μCi/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 μCi/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 μCi/g of dose equivalent $^{131}$I. The accident initiated an increase in the release rate of iodine from the fuel by a factor of 500 over the release rate to maintain an activity level of 1.0 μCi/g of dose equivalent $^{131}$I in 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 $\mu$Ci/g of dose equivalent $^{131}$I.

   Pre-existing Spike Value ($\mu$Ci/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 ($\mu$Ci/g) | 1.0 |
| Secondary Coolant DE $^{131}$I concentration ($\mu$Ci/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 $\mu$Ci/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) | $1.64 \times 10^{-3}$ |
| LPZ (0-8 hours) | $1.96 \times 10^{-4}$ |
| Control Room (0-2 hours) | $3.18 \times 10^{-3}$ |
| Control Room (2-8 hours) | $1.01 \times 10^{-3}$ |

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 $1 \times 10^{-2}$, 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 de[illegible]bed in the September 7, 1995 submittal to meet the probe wear criterion [illegible] 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 $1x10^{-2}$ 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.

Principal Contributors: John J. Hayes, Phillip J. Rush, Jai Raj Rajan,

Dated: October 11, 1995

Mr. Oliver D. Kingsley, Jr.
Tennessee Valley Authority

**SEQUOYAH NUCLEAR PLANT**

cc:

Mr. O. J. Zeringue, Sr. Vice President
Nuclear Operations
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Dr. Mark O. Medford, Vice President
Engineering & Technical Services
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. D. E. Nunn, Vice President
New Plant Completion
Tennessee Valley Authority
3B Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. R. J. Adney, Site Vice President
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

General Counsel
Tennessee Valley Authority
ET 11H
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. P. P. Carier, Manager
Corporate Licensing
Tennessee Valley Authority
4G Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Ralph H. Shell
Site Licensing Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

TVA Representative
Tennessee Valley Authority
11921 Rockville Pike
Suite 402
Rockville, MD 20852

Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW., Suite 2900
Atlanta, GA 30323

Mr. William E. Holland
Senior Resident Inspector
Sequoyah Nuclear Plant
U.S. Nuclear Regulatory Commission
2600 Igou Ferry Road
Soddy Daisy, TN 37379

Mr. Michael H. Mobley, Director
Division of Radiological Health
3rd Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

County Judge
Hamilton County Courthouse
Chattanooga, TN 37402-2801