April 9, 1997

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 AMENDMENTS FOR THE SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 (TAC NOS. M96998 AND M96999) (TS 96-05)

Dear Mr. Kingsley:

The Commission has issued the enclosed Amendment No. 222 to Facility Operating License No. DPR-77 and Amendment No. 213 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Units 1 and 2, respectively. These amendments are in response to your application dated October 18, 1996 as supplemented March 12, March 17, April 4, and April 9, 1997.

The amendments revise Technical Specifications (TS) 3.4.6.2, 4.4.5.4, and 4.4.5.5 and associated Bases to permanently incorporate requirements associated with steam generator tube inspections and repair in the Sequoyah Nuclear Plant, Units 1 and 2 TS. The amendments make permanent the alternate steam generator tube plugging criteria (APC) at the tube support plate intersections incorporated into the TS by previous amendments to the operating licenses for Operating Cycle 8. These revised criteria are based on NRC Generic Letter 95-05. The proposed amendments remove the reference to Cycle 8, thereby making the requirements applicable to all future operating cycles. The amendments also impose a new license condition for each license to incorporate the commitments made by TVA in a letter dated March 12, 1997.

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

Ronald W. Hernan, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 222 to License No. DPR-77
2. Amendment No. 213 to License No. DPR-79
3. Safety Evaluation

140022

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. 222
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 October 18, 1996 as supplemented March 12, March 17, April 4 and April 9, 1997, 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. Paragraphs 2.C.(2) and 2.C.(9) of Facility Operating License No. DPR-77 are hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 222, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

A new condition 2.C.(9)(d) is added as follows:*

(9) <u>Steam Generator Inspection (Section 5.3.1)</u>

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(d) By May 20, 1997, TVA shall establish a steam generator inspection program that is in accordance with the commitments listed in Enclosure 2 to the TVA letter to the Commission on this subject dated March 12, 1997.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days of its issuance.

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: 1. Page 5 of License
2. Changes to the Technical Specifications

Date of Issuance: April 9, 1997

* Page 5 of the composite license is attached to reflect this change

(9) Steam Generator Inspection (Section 5.3.1)

(a) Prior to March 1, 1981, TVA shall provide to the NRC the results of its tests to determine the feasibiity of using a steam generator camera device.

(b) Prior to start-up after the first refueling, TVA must install inspection ports in each steam generator if the results of the camera device inspection are not satisfactory to the NRC.

(c) Prior to start-up after the first refueling, TVA will plug Row 1 of the steam generator tubes, if required by NRC.

(d) By May 20, 1997, TVA shall establish a steam generator inspection program that is in accordance with the commitments listed in Enclosure 2 to the TVA letter to the Commission on this subject dated March 12, 1997, as modified by TVA letter dated March 17, 1997.

(10) Water Chemistry Control Program (Section 5.3.2)

This requirement has been deleted.

(11) Negative Pressure in the Auxiliary Building Secondary Containment Enclosure (ABSCE) (Section 6.2.3)

After the final ABSCE configuration is determined, TVA must demonstate to the satisfaction of the NRC that a negative pressure of 0.25 inches of water gauge can be maintained in the spent fuel storage area and in the ESF pump room.

(12) Environmental Qualification (Section 7.2.2)

(a) No later than November 1, 1980, TVA shall submit information to show compliance with the requirements of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Elecctrical Equipment," for safety-related equipment exposed to a harsh environment. Implementation shall be in accordance with NUREG-0588 by June 30, 1982.

(b) By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified to document complete compliance by June 30, 1982.

ATTACHMENT TO LICENSE AMENDMENT NO. 222

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 4-9a
3/4 4-9b	3/4 4-9b
3/4 4-10	3/4 4-10
3/4 4-14	3/4 4-14
B 3/4 4-3	B 3/4 4-3
B 3/4 4-4	B 3/4 4-4
B 3/4 4-4a	B 3/4 4-4a

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. R193

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion 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. R193

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

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:

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

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

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

Note 1: The lower voltage repair limit is 1.0 volt for 3/4-inch diameter tubing or 2.0 volts for 7/8-inch diameter tubing.

Note 2: The upper voltage repair limit is calculated according to the methodology in GL 90-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle.

SURVEILLANCE REQUIREMENTS (Continued)

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. |R40

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

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

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

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

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

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

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. 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. R16

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: R16

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 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 to 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 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. R193

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. R40

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 100 CFR 100 guidelines. If the projected and cycle distribution of crack indications results in primary-to-secondary leakage greater than 3.7 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 3.7 gpm.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 213
License No. DPR-79

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 18, 1996 as supplemented March 12, March 17, April 4, and April 9, 1997, 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. Paragraphs 2.C.(2) and 2.C.(9) of Facility Operating License No. DPR-79 are hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 213 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   A new condition 2.C.(8)(b) is added as follows:*

   (9) Steam Generator Inspection (Section 5.3.1)

   .

   .

   (b) By May 20, 1997, TVA shall establish a steam generator inspection program that is in accordance with the commitments listed in Enclosure 2 to the TVA letter to the Commission on this subject dated March 12, 1997, as modified by TVA letter dated March 17, 1997.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 45 days of its issuance.

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: 1. Page 4 of License
2. Changes to the Technical Specifications

Date of Issuance: April 9, 1997

* Page 4 of the composite license is attached to reflect this change

d. Failure to complete any tests included in the described program (planned or scheduled) for power levels up to the authorized power level.

(4) Monitoring Settlement Markers (SER/SSER Section 2.6.3)

TVA shall continue to monitor the settlement markers along the ERCW conduit for the new ERCW intake structure for a period not less than three years from the date of this license. Any settlement greater than 0.5 inches that occurs during this period will be evaluated by TVA and a report on this matter will be submitted to the NRC.

(5) Tornado Missiles (Section 3.5)

Prior to startup after the first refueling of the facility, TVA shall reconfirm to the satisfaction of the NRC that adequate tornado protection is provided for the 480 V transformer ventilation systems.

(6) Design of Seismic Category Structures (Section 3.8)

Prior to startup following the first refueling, TVA shall evaluate all seismic Category I masonry walls to final NRC criteria and implement NRC required modifications that are indicated by that evaluation.

(7) Low Temperature Overpressure Protection (Section 5.2.2)

Prior to startup after the first refueling, TVA shall install an overpressure mitigation system which meets NRC requirements.

(8) Steam Generator Inspection (Section 5.3.1)

(a) Prior to start-up after the first refueling, TVA shall install inspection ports in each steam generator or have an alternative for inspection that is acceptable to the NRC.

(b) By May 20, 1997, TVA shall establish a steam generator inspection program that is in accordance with the commitments listed in Enclosure 2 to the TVA letter to the Commission on this subject dated March 12, 1997, as modified by TVA letter dated March 17, 1997.

(9) Containment Isolation Systems (Section 6.2.4)

Prior to startup after the first refueling, TVA shall modify to the satisfaction of the NRC the one-inch chemical feed lines to the main and auxiliary feedwater lines for compliance with GDC 57.

(10) Environmental Qualification (Section 7.2.2)

a. No later than June 30, 1982, TVA shall be in compliance with the requirements of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," for safety-related equipment exposed to a harsh environment.

ATTACHMENT TO LICENSE AMENDMENT NO. 213

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

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

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

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

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

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

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

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

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

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

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

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

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

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

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

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

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

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness. Plugging limit does not apply to that portion of the tube that is not within the pressure boundary of the reactor coolant system (tube end up to the start of the tube-to-tubesheet weld). 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. R181

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

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

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

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

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:

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

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

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC)

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.

Note 1: The lower voltage repair limit is 1.0 volt for 3/4-inch diameter tubing or 2.0 volts for 7/8-inch diameter tubing.

Note 2: The upper voltage repair limit is calculated according to the methodology in GL 90-05 as supplemented. V_{URL} may differ at the TSPs and flow distribution baffle.

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

4.4.5.5 Reports

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

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

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported pursuant to Specification 6.6.1 prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. R28

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.

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. 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:

3/4.4.5 STEAM GENERATORS

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

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

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

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

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

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

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

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

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

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

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

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurances of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

The total steam generator tube leakage limit of 600 gallons per day for all steam generators and 150 gallons per day for any one steam generator will minimize the potential for a significant leakage event during steam line break. Based on the NDE uncertainties, bobbin coil voltage distribution and crack growth rate from the previous inspection, the expected leak rate following a steam line rupture is limited to below 3.7 gpm in the faulted loop, which will limit the calculated offsite doses to within 10 percent of the 10 CFR 100 guidelines. If the projected and cycle distribution of crack indications results in primary-to-secondary leakage greater than 3.7 gpm in the faulted loop during a postulated steam line break event, additional tubes must be removed from service in order to reduce the postulated primary-to-secondary steam line break leakage to below 3.7 gpm.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

The Tennessee Valley Authority (TVA, the licensee) requested amendments to Operating Licenses DPR-77 and DPR-79 for Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively, in a letter dated October 18, 1996 as supplemented on March 12, March 17, April 4, and April 9, 1997. The amendments would revise Technical Specifications (TS) 3.4.6.2, 4.4.5.4, and 4.4.5.5 and their associated Bases by permanently incorporating the requirements associated with steam generator tube inspections and repair at SQN. These requirements establish alternate steam generator tube plugging criteria (APC) at the tube support plate intersections. These revised criteria had been incorporated into the TS by previous amendments to each the operating licenses but only for Operating Cycle 8. The proposed amendments would remove the reference to Cycle 8, thereby making the requirements applicable to all future operating cycles. No new or different requirements are imposed by these amendments. The alternate repair criteria, previously approved for only one operating cycle, allow steam generator tubes having outside diameter stress corrosion cracking (ODSCC) that is predominately axially oriented and confined within the tube support plates to remain in service on the basis of bobbin coil voltage response. The U. S. Nuclear Regulatory Commission (NRC) guidance on the alternate repair criteria is specified in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

The March 12, March 17, April 4, and April 9, 1997, letters provided clarifying informaton that did not change the scope of the October 18, 1996, application and the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The acceptance criteria (i.e., plugging limits) for steam generator tubes are specified in the plant TS. The traditional strategy for achieving adequate structural and leakage integrity of the tubes has been to establish a minimum wall thickness requirement in accordance with NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR [pressurized water reactor] Steam Generator Tubes." The minimum wall thickness requirement was developed with the assumption of a uniform thinning of the tube wall. This assumed degradation mechanism is inherently conservative for certain forms of tube degradation. Conservative repair limits may lead to removing degraded tubes from service that may otherwise have adequate structural and leakage integrity for further service.

To reduce unnecessary conservatism in the minimum wall thickness requirement for certain degradation, the industry proposed voltage-based repair criteria for ODSCC confined within the thickness of the tube support plates. The staff 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 GL 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 GL 95-05 that took into consideration public comments on the draft GL cited above, domestic operating experience under the voltage-based repair criteria, and additional data made available from European nuclear power plants.

The guidance of GL 95-05 does not set depth-based limits on predominantly axially oriented ODSCC at tube support plate locations; rather, it relies on empirically derived correlations between a nondestructive inspection parameter, the bobbin coil voltage, and tube burst pressure and leak rate. The staff recognizes that although the total tube integrity margins may be reduced following application of a voltage-based repair criteria, the guidance in GL 95-05 ensures that structural and leakage integrity continue to be maintained at acceptable levels consistent with the requirements of 10 CFR Part 50 and the guideline values in 10 CFR Part 100. Since the voltage-based repair criteria do require minimum tube wall thickness, there is the possibility for tubes with through-wall cracks to remain in service. Because of the increased likelihood of such flaws, the staff included provisions for augmented steam generator tube inspections and restrictive operational leakage limits.

GL 95-05 specifies, in part, that: (1) the repair criteria is only applicable to predominantly axially oriented ODSCC located within the bounds of the tube support plates; (2) licensees perform an evaluation to confirm that the steam generator tubes will retain adequate structural and leakage integrity from cycle to cycle; (3) licensees adhere to specific inspection criteria to ensure consistency in methods between inspections; (4) tubes must be periodically removed from the steam generators, examined, and destructively tested to verify the morphology of the degradation and provide additional data for structural and leakage integrity evaluations; (5) the operational leakage limit be reduced; (6) licensees implement an operational leakage monitoring program; and (7) specific reporting requirements be incorporated into the plant technical specifications.

The staff approved the licensee's interim repair criteria (or APC) for SQN Unit 1 as documented in license amendment No. 214, issued on October 11, 1995, and for Unit 2 as documented in license amendment No. 211, issued on April 3, 1996. The APC were considered interim because the amendments was approved for only one operating cycle while resolution of a number of technical issues was in the resolution process with the Nuclear Energy Institute. The APC are made permanent with these amendments and will eliminate the need for periodic license amendment applications in the future.

Each Sequoyah unit has four Westinghouse Model 51 steam generators, which use mill-annealed alloy 600 tubing. These steam generators use drilled-hole tube support plates and do not have flow distribution baffle plates. The outside diameter and wall thickness of each tube is 7/8 inch and 0.050 inch, respectively.

3.0 EVALUATION

The licensee stated that it will comply with the guidance in GL 95-05 for its APC. In addition, the licensee proposed to incorporate verbatim the model technical specifications in GL 95-05 into the SQN TS. The major issues related to the licensee's permanent implementation of the APC are discussed below.

3.1 Tube Repair Limits

The APC approved in these amendments (1) permit tubes having indications confined to within the thickness of the tube support plates with bobbin voltages less than or equal to 2.0 volts to remain in service; (2) permit tubes having indications confined to within the thickness of the tube support plates with bobbin voltages greater than 2.0 volts but less than or equal to the upper voltage limit to remain in service if a motorized rotating pancake coil probe or acceptable alternative inspection does not detect degradation; and (3) require tubes having indications confined to within the thickness of the tube support plates with bobbin voltages greater than the upper voltage limit be plugged or repaired.

The proposed lower voltage limit of 2.0 volts is based on the use of a correlation between the burst pressure and the bobbin coil voltage of pulled tube and model boiler data and is consistent with the recommended value specified in GL 95-05 for 7/8 inch steam generator tubing. The upper voltage limit is based on the lower 95 percent prediction interval of the burst pressure versus bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95 percent confidence level. The upper voltage limit is further reduced to account for uncertainty in the nondestructive examination technique and flaw growth over the next operating cycle. The industry periodically updates the database for burst pressure and bobbin voltage when the destructive test data from pulled tubes are available; therefore, the upper voltage limit may vary as additional data are incorporated into the correlation.

3.2 Inspection Issues

Section 3.c.3 of Attachment 1 to GL 95-05 specifies guidance for probe wear. The licensee proposed to use an alternative to section 3.c.3. The alternative approach, developed through the Nuclear Energy Institute, specifies that if the probe does not satisfy the voltage variability criterion for wear of ± 15 percent limit before its replacement, all tubes which exhibited flaw signals with voltage responses measured at 75 percent or greater of the lower repair limit must be reinspected with a bobbin probe satisfying the ± 15 percent wear standard criterion. The voltages from the reinspection should be used as the basis for tube repair. The NRC staff completed a review of the Nuclear Energy Institute proposed alternative method and concluded that the approach is acceptable as discussed in a letter from Brian Sheron of the NRC to Alex Marion of the Nuclear Energy Institute dated March 18, 1996. The licensee's proposal to follow the industry approach to address probe wear is acceptable.

In the laboratory and field studies supporting the alternative probe wear criteria, the correlation of voltages measured by worn probes and new probes shows that for all significant voltage levels, the worn probe voltages are never less than 75 percent of the new probe voltage as discussed in the letter from Alex Marion of the Nuclear Energy Institute to Brian Sheron of the NRC dated January 23, 1996. However, in the 90-Day inspection report for Byron Unit 1 dated September 9, 1996, Commonwealth Edison, the licensee for Byron, compared the worn probe voltage to the new probe voltage and found that the worn probe voltage was substantially less than 75 percent of the new probe voltage for a few indications. Commonwealth Edison evaluated these indications and concluded that the criteria to retest tubes with worn probe voltages above 75 percent of the repair limit is adequate and generally conservative due to the trend for worn probe voltages to exceed new probe voltages. Comparison of the actual and projected end-of-cycle voltages did not show anything unusual attributable to the alternate probe wear criteria. The staff concludes that the aforementioned probe wear results do not indicate an immediate need to modify the probe wear criteria developed by the industry. However, the staff will continue to monitor probe wear in the 90-Day inspection reports.

Section 3.b.3 of Attachment 1 to GL 95-05 specifies that all tube support plate intersections with dent signals greater than 5 volts should be inspected with a rotating pancake coil (RPC). Any tube with indications found at such intersections by RPC will be repaired. If indications are circumferentially oriented or caused by primary water stress corrosion cracking (PWSCC), it may be necessary to expand the RPC sampling plan to include tubes with dents showing signals less than 5.0 volts.

Steam generators 3 and 4 in SQN Unit 1 have experienced many dent signals that are greater than 5 volts such that inspecting all affected dents at the tube support plate intersections, as specified under section 3.b.3, would not be practical. The licensee requested an exception to section 3.b.3. The licensee proposed to initially sample 20 percent of the total dented intersections in the hot leg side of steam generators 3 and 4 with an RPC. The sample will begin at the lowest tube support plate elevations, which has the highest probability of dented intersections, and continue to higher elevations. If the RPC identified circumferential ODSCC or PWSCC at the dented intersections that the bobbin probe had missed, the sample will be expanded in accordance with a prescribed expansion plan. For steam generators 1 and 2 in SQN Unit 1, the initial sample will be 100 percent of the total dented tube support plate intersections in the hot leg side.

For Unit 1 steam generators, the licensee stated that, using RPC, it will inspect all dent signals less than 5 volts at all tube support plate elevations (and lower tube support plates) where, based on past inspections, degradation has occurred (defining a critical area) and perform a 20 percent sample of the buffer zone to bound the affected area. The licensee defined the buffer zone as the next higher tube support plate elevation where no degradation has been observed. The buffer zone is created to ensure that the critical area is bounded. The degradation (circumferential ODSCC or PWSCC not detected by bobbin coil) identified from the past inspection of the dented intersections would determine the initial sample. Each initial sample will be determined independently. If degradation was not identified in the past inspection, then a minimum 20 percent sample of the dents at the first tube support plate intersection will be examined. During future outages, a different 20 percent sample would be inspected, such that over five outages 100 percent of the dents at this elevation would be inspected. If indications are identified in the buffer zone, the sample will be expanded in accordance with the prescribed plan. The classification in TS section 4.4.5.2 will be used to classify the inspection results in the buffer zone, except that, when a sample size is less than 200, only category C-2 results apply

The licensee also proposed an alternative to the aforementioned inspection plan for dent signals greater than 5 volts for steam generators 3 and 4 in Unit 1. The alternative plan would follow the proposed inspection plan for dents less than 5 volts. In addition, if a tube support plate elevation has less than 250 dented intersections when selecting a buffer zone, additional dented intersections at the next higher elevation will be inspected to make the total number of dented intersections to be inspected equal to 50.

Certain commitments were made for SQN Unit 1 in the licensee's letter dated March 12, 1997. The licensee stated that the inspection frequency for dented intersections will be performed coinciding with steam generator surveillance requirements. If an unscheduled mid-cycle steam generator surveillance is required, the dented intersections will be inspected. The licensee will inspect the dented intersections with a technique qualified to Appendix H of the Steam Generator Examination Guidelines published by the Electric Power Research Institute. The licensee stated that any indications identified that exceed the plugging limit will be repaired. Until a technique is qualified for sizing and validated for site specific applicability, tubes with PWSCC or ODSCC circumferential indications at dented intersections will be plugged on detection.

The licensee will follow the guidelines in GL 95-05 to inspect all dent signals in Unit 2 steam generators.

The proposed inspection plan for dent signals greater than 5 volts in Unit 1 steam generators is the same as the one that the staff approved in licensee amendment number 214. For dent signals less than 5 volts in Unit 1 steam generators, the staff finds the proposed inspection plan acceptable because it is conservative. The staff has no objection to the proposed alternative inspection plan for dent signals greater than 5 volts because the sample would result in an adequate number of dent intersections being inspected to identify defected tubes and to expand the inspection as appropriate. The inspection plan for all dented intersections in Unit 2 steam generators follows the GL and, therefore, is acceptable.

3.3 Structural and Leakage Integrity Assessments

The staff guidance for the implementation of the voltage-based repair criteria focuses on maintaining tube structural integrity during the full range of normal, transient and postulated accident conditions with adequate allowance for eddy current test uncertainty and flaw growth projected to occur during the next operating cycle. Tube structural limits based on RG 1.121 criteria require maintaining a margin of safety of 1.43 against tube failure under postulated accident conditions and maintaining a margin of safety of 3 against burst during normal operation. Because GL 95-05 addresses tubes affected with ODSCC confined to within the thickness of the tube support plate during normal operation, the staff concluded that the structural constraint provided by the tube support plate ensures all tubes to which the voltage-based criteria applies will retain a margin of 3 with respect to burst under normal operating conditions. For a postulated main steam line break accident, however, the tube support plate may displace axially during steam generator blowdown such that the ODSCC affected portion of the tubing may no longer be fully constrained by the tube support plate. Accordingly, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under postulated main steam line break conditions.

In order to confirm the structural and leakage integrity of the tube until the next scheduled inspection, GL 95-05 specifies a methodology to determine the conditional burst probability and the total primary-to-secondary leak rate from an affected steam generator during a postulated main steam line break event. To complete GL 95-05 prescribed assessments, the licensee proposes to follow the methodology described in WCAP-14277, Revision 1, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated December 1996. The staff finds the methodology in WCAP-14277, Revision 1, acceptable.

GL 95-05 specifies that the structural and leakage integrity assessments should use the latest available data from destructive examinations of tubes removed from Westinghouse-designed steam generators. The licensee stated that it will use NRC approved database. For the upcoming cycle 8 inspection at Unit 1, the licensee will use the database previously approved by the staff in the licensee amendment No. 211 for Unit 2 dated April 3, 1996, including data from any additional pull tubes in accordance with exclusion criteria protocol in GL 95-05. For the long term, the staff is reviewing the industry data which is documented in the EPRI report, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates--Database for Alternate Repair Limits, 1996 Database Update, Addendum 1," NP-7480-L, November 1996. The staff is also working with the Nuclear Energy Institute to finalize the protocol for the periodical update of the database. The licensee stated that Sequoyah intends to use the database and to follow the protocols for future outages once they are approved. The staff finds that the licensee's intentions to use NRC-approved database to perform structural and leakage assessments will enhance its steam generator inspection program.

3.3.1 Conditional Probability of Burst

The licensee will use the methodology described in Revision 1 of WCAP-14277 for performing a probabilistic analysis to quantify the potential for steam generator tube ruptures given an main steam line break event. The results of the probabilistic analysis will be compared to a threshold value of $1x10^{-2}$ per cycle in accordance with GL 95-05. This threshold value provides 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 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 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 assumed and evaluated as acceptable in NUREG-0844. The NRC staff concludes the licensee's proposed methodology for calculating the conditional burst probability is consistent with the guidance in GL 95-05 and is acceptable.

3.3.2 Accident Leakage

The licensee will use the methodology described in Revision 1 of WCAP-14277 for calculating the steam generator tube leakage from the faulted steam generator during a postulated main steam line break event. The model 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 staff concludes that the licensee's proposed methodology for calculating the tube leakage is consistent with the guidance in GL 95-05 and is acceptable.

3.3.3 Primary-to-Secondary Leakage During Normal Operation

Because the voltage-based repair criteria would allow degraded tubes to remain in service, the degraded tubes may develop through-wall cracks during an operational cycle, thus creating the potential for primary-to-secondary leakage during normal operation, transients, or postulated accidents. Therefore, as a defense-in-depth measure, GL 95-05 specifies that the operational leakage limits of the plant TS be limited to 150 gallons per day from any one steam generator. The staff concludes that adequate leakage integrity during normal operation is reasonably assured by the TS limits on allowable primary-to-secondary leakage. Sequoyah Units 1 and 2 TS limit the primary-to-secondary leakage through one steam generator to 150 gallons per day. The staff finds that the leakage requirement in the Sequoyah TS is consistent with the guidance in GL 95-05 and is, therefore, acceptable.

3.4 Degradation Monitoring

To confirm the nature of the degradation at the tube support plate elevations, tubes are periodically removed from the steam generators for destructive tests. The test data from removed tubes can confirm that the nature of the degradation observed at these locations 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 specifies that at least two tubes be removed from steam generators with the objective of retrieving as many intersections as practical (minimum of four intersections) during the plant steam generator inspection outage preceding initial application of the voltage-based repair criteria. On an ongoing bases, additional tube specimens (minimum of two intersections) should be removed at the first refueling outage following 34 effective full power months of operation or at the maximum interval of three refueling outages after the previous tube pull. Alternatively, the licensee may participate in an industry-sponsored tube pull program endorsed by the staff as described in GL 95-05.

The licensee removed two tubes during cycle 6 outage and three tubes during cycle 7 outage from the Unit 1 steam generators and two tubes during cycle 7 outage from the Unit 2 steam generators for burst and leak rate testing and metallographic examination. The metallurgical examination confirmed that the degradation mechanism for the indications at the tube support plates was predominantly axially oriented ODSCC. The licensee stated that it will comply with GL 95-05 for future tube removal. The staff concludes that the licensee satisfies the tube removal guidance of GL 95-05.

4.0 SUMMARY

The licensee submitted an application for a license amendment to permit the use of the voltage-based repair criteria for steam generator tubes at Sequoyah Units 1 and 2 on a permanent basis. This would be accomplished by removing the footnote on each affected TS page that resticted the changes to Cycle 8 only (e.g., "The indicated changes to this page are applicable to Cycle 8 operation only").

The staff has reviewed the proposed amendment and concludes that the proposed alternate repair criteria are consistent with GL 95-05 and, therefore, the proposed TS changes are acceptable. The staff also concludes that adequate structural and leakage integrity can be assured, consistent with applicable regulatory requirements, for indications to which the voltage-based repair criteria will be applied. The staff approves the proposed voltage-based repair criteria based in part on the licensee being able to successfully demonstrate after each inspection outage that the conditional probability of burst and the primary-to-secondary leakage during a postulated main steam line break will be acceptable per the guidance in GL 95-05. The staff's approval of this amendment request relies upon information provided by TVA in their letters related to this request dated October 18, 1996, March 12 and 17, 1997, and their concurrence with a proposed license condition on April 4, 1997. Specifically, the commitments made by TVA in their March 12, 1997, letter have been incorporated into the "Steam Generator Inspection" license condition section (Condition 2.C.9 for Unit 1 and 2.C.8 for Unit 2) as discussed in the license amendment. This change is acceptable.

5.0 STATE CONSULTATION

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

6.0 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 changes 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 (62 FR 6276 dated February 11, 1997). Accordingly, the amendments meet 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 amendments.

7.0 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 Contributor: John C. Tsao

Dated: April 9, 1997