ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-94-14)

LIST OF AFFECTED PAGES

<u>Unit 1</u>

3/4 4-7
3/4 4-9
B 3/4 4-3

<u>Unit 2</u>

3/4 4-11
3/4 4-13
B 3/4 4-3

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.

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:

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.

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.

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. FP

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).

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.

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 = 500 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. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

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 plugging limit of 40% of the tube nominal wall thickness. 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.

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. R4(

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.

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.

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:

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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

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

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

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

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

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

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness.

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.

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).

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 = 500 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. Operating plants have demonstrated that primary-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

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 plugging limit of 40% of the tube nominal wall thickness. 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.

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 6.9.1 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

The 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.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-94-14)

DESCRIPTION AND JUSTIFICATION FOR

THE TS REVISION REGARDING STEAM GENERATOR

TUBE INDICATIONS BELOW THE TUBESHEET

Description of Change

TVA proposes to modify the Sequoyah Nuclear Plant (SQN) Units 1 and 2 Technical Specifications (TSs) to revise the surveillance requirements (SRs) to clarify that the portion of the steam generator (S/G) tubes from the end of the tube up to the start of the tube-to-tubesheet weld is exempt from the result and action required portions of Table 4.4-2 and from the plugging limit since it is not part of the reactor coolant system (RCS) pressure boundary. Specifically a note will be added in TS SR 4.4.5.2.c.2 immediately preceding the sentence beginning with, "The results of each sample . . .," which states, "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." In addition, a sentence will be added immediately after the existing sentence in TS SR 4.4.5.4.a.6 stating, "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)."

The bases will also be changed to insert new text in Section 3/4.4.5 in the paragraph, which begins, "Wastage-type defects . . ." The following text will be inserted after the third sentence of the existing paragraph, "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."

Reason for Change

The portion of the S/G tubing, which extends beyond the tubesheet (i.e., the portion from the end of the tube to the start of the tube-to-tubesheet weld), is not part of the RCS pressure boundary. Tube degradation in this area should not fall under the same criteria for repair and sample inspection as degradation in areas that are part of the pressure boundary. Degradation confined to this region of the tube would not affect the structural integrity of the RCS pressure boundary. TVA discovered indications within this region of the tubes on S/G No. 1 on Unit 2 during eddy current testing using the motorized rotating pancake coil (RPC).

During the Unit 2 Cycle 6 S/G maintenance work, Westinghouse Electric Corporation was performing removal of Combustion Engineering roll plugs from the Row 1 tubes in all four S/Gs. The plugs were being removed to return these tubes to service following a U-bend heat treatment operation. The Westinghouse roll plug removal process consisted of a helical scan with a tungsten inert gas torch on the inside diameter of the plug with subsequent plug removal with a pulling tool.

The acceptance criteria for roll plug removal is a visual examination with a remote TV camera. The results of the visual inspections on all tubes from which plugs were removed were acceptable. Upon inspection of the removed roll plugs from the first two S/Gs (Nos. 1 and 4), significant burn through (approximately 180 degrees and 0.125-inch wide) was identified on some roll plugs. An unscheduled RPC eddy current exam was performed on the bottom 6 inches of the hot leg tubes in S/G Nos. 1 and 4 to verify no tube damage resulted from the burn through on the roll plugs, before continuing the roll plug removal process on S/G Nos. 2 and 3.

The RPC inspection resulted in the detection of single and multiple axial indications in the protruding tube ends in 39 tubes on S/G No. 1. Thirty three of 39 tubes will be returned to service, 6 will be plugged for other reasons. These indications were very short and appear to be both inside diameter and outside diameter in nature. No indications were detected at the tube ends in S/G No. 4. Bobbin coil inspection of the Row 1 tube ends did not detect any degradation on S/G Nos. 1 and 4. Bobbin coil inspection did not detect the indications because the response from the tube ends greatly exceeds the amplitude of the indications, thus masking their presence.

The location of the axial indications observed by RPC are below the tube-to-tubesheet weld and the hardroll and Westinghouse explosive tube expansion (WEXTEX) regions of the Row 1 tubes in S/G No. 1. It is concluded that the indications do not affect the structural and leakage integrity of the pressure boundary.

Based on these conclusions, this change is being proposed to the TSs to prevent the criteria for repairs and sample expansion from being inappropriately applied to indications within this region.

Justification for Changes

Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet. Thus, the Regulatory Guide 1.121 criteria are satisfied.

The potential effect of the tube crack propagation on the tube-to-tubesheet welds, is addressed by examining the limiting case where the weld would no longer be effective because of the presence of cracking. It is shown (WCAP-13532, Revision 1, "Sequoyah Units 1 and 2 W* Tube Plugging Criteria for S/G Tubesheet Region of WEXTEX Expansions," November 1992) that even if a tube is circumferentially degraded within the tubesheet to the extent that tube separation could occur, a length of engagement of tube can be defined that would prevent pullout of the tube and would result in negligible leakage during all plant conditions. This length of engagement has been calculated for SQN Unit 2 and is less than 5.6 inches from the bottom of the WEXTEX expansion transition or approximately 15 inches above the S/G tube end. Pressure boundary integrity can only be challenged if these indications propagate beyond the tube-to-tubesheet weld. If these indications were to propagate beyond the tube-to-tubesheet weld region of the tube, bobbin coil would be able to detect indications of concern. Therefore, eddy current bobbin inspection of the indications on a cycle-to-cycle basis is sufficient to monitor S/G tube integrity.

Plant operating experience shows that primary water stress corrosion cracking axial crack propagation rate for WEXTEX plants, which is a function of crack length, is less than 0.315-inch per effective full power year in the WEXTEX zone. Indications in the portion of the S/G tube between the end of the tube and the start of the tube-to-tubesheet weld do not affect the structural integrity of the RCS pressure boundary. However, to ensure that the indications are properly monitored, TVA will change the appropriate inspection procedures by January 30, 1995, to include a bobbin coil inspection of the protruding tube ends in the 33 tubes in Unit 2 S/G No. 1 in subsequent refueling outages, or until the indications are removed or the tubes are plugged. If indications are detected with the bobbin coil inspection in this region, TVA will expand the inspection based on the number and nature of the indications in the nonplugged hot leg Row 1 tubes in Unit 2. This will provide the means to monitor any possible propagation into the RCS pressure boundary area that could affect structural integrity. Indications that are not detectable by bobbin coil in this region do not compromise S/G tube integrity. Additional justification is provided in the Westinghouse Safety Evaluation Checklist provided in Enclosure 4.

Environmental Impact Evaluation

The proposed change does not involve an unreviewed environmental question because operation of SQN Units 1 and 2 in accordance with this change would not:

1. Result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by NRC's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or decisions of the Atomic Safety and Licensing Board.

2. Result in a significant change in effluents or power levels.

3. Result in matters not previously reviewed in the licensing basis for SQN that may have a significant environmental impact.

ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-94-14)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Significant Hazards Evaluation

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will clarify the requirements for indications found in the region of the steam generator (S/G) tube, which protrudes below the tubesheet. This region of the tube does not affect the structural integrity of the reactor coolant system (RCS) pressure boundary, since it is not part of the pressure boundary. This revision will exempt this portion of the tube from being considered under the result and action required sections of Table 4.4-2 in SQN's TS. Therefore, indications in this region will not require repairs and will not be used for the purpose of expanding the sample of tubes to be inspected under the requirements of the TS.

   The condition described in this evaluation results in tube integrity considerations commensurate with Regulatory Guide 1.121 criteria both analytically and empirically. If the indications are hypothetically considered as cracks, the Row 1 tube end indications neither adversely affect S/G tube integrity or any other component, nor does the presence of the indications alter the function of the S/G or any other component. Continuing the hypothetical scenario, even if the crack propagated beyond the weld, the only consequence of an accident that could be caused by plant operation or by the occurrence of a faulted condition event with the tube end indications, would be negligible leakage from the primary to secondary system. Such leakage is expected to be insignificant at both normal and faulted conditions. Therefore, plant operation with the tube end indications present in the Row 1 tubes does not increase the probability of an analyzed accident such as a S/G tube rupture event, nor does it increase the consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

   Any hypothetical accident as a result of plant operation with the Row 1 tube end indications would be bounded by the consequences of a postulated S/G tube rupture. Therefore, this change does not create the possibility of a new or different kind of accident from any previously analyzed.

3. Involve a significant reduction in a margin of safety.

   The locations of the axial indications observed are below the tube-to-tubesheet weld. Consequently, it is concluded that the axial indications do not affect the structural and leakage integrity of the primary pressure boundary. Should the indications be single or multiple axial cracks on the tube ends, the effect of crack propagation was evaluated. Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet. Therefore, crack lengths do not need to be limited by burst considerations and operating leakage limits are not required to detect crack lengths associated with tube burst. However, primary to secondary leakage must be shown to remain within acceptable limits during all plant conditions. Leak-rate testing shows that such leakage would be negligible during all plant conditions. Since the pressure boundary integrity, acceptable leak rate, and function of the S/G are not affected by the presence of the tube end indications, the margin of safety is not reduced.

ENCLOSURE 4

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-94-14)

WESTINGHOUSE ELECTRIC CORPORATION

SAFETY EVALUATION CHECK LIST

SECL-94-133

SECL-94-133
Customer Reference No(s).

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Westinghouse Reference No(s).

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WESTINGHOUSE
SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S): Sequoyah Unit 2

2) CHECK LIST APPLICABLE TO: Steam Generator Tube End Indications

3) The written safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2. Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A

3.1) Yes__ No X A change to the plant as described in the FSAR?
3.2) Yes__ No X A change to procedures as described in the FSAR?
3.3) Yes__ No X A test or experiment not described in the FSAR?
3.4) Yes X No__ A change to the plant technical specifications (Appendix A to the Operating License)?

4) CHECK LIST - PART B (Justification for Part B answers must be included on page 2.)

4.1) Yes__ No X Will the probability of an accident previously evaluated in the FSAR be increased?
4.2) Yes__ No X Will the consequences of an accident previously evaluated in the FSAR be increased?
4.3) Yes__ No X May the possibility of an accident which is different than any already evaluated in the FSAR be created?
4.4) Yes__ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.5) Yes__ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
4.6) Yes__ No X May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
4.7) Yes__ No X Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answer to any of the above questions in Part (3.4) or Part B cannot be answered in the negative, the change review requires an application for license amendment in accordance with 10 CFR 50.59 (c) and submitted to the NRC pursuant to 10 CFR 50.90.

5) REMARKS:

The answers given in Section 3, Part A, and Section 4, Part B, of the Safety Evaluation Checklist, are based on the attached Safety Evaluation.

Reference document(s): ______________________

FOR FSAR UPDATE

Section: ________ Pages: ________ Tables: ________ Figures: ________

Reason for / Description of Change:

SIGNATURES

Prepared By: Gary W. White Date: 8/24/94

Verified By: [signature] Crane Date: 8/24/94

Reviewed By: [signature] Date: 8-24-94

1.0 INTRODUCTION

The roll plugs from the row 1 tubes in all four steam generators at Sequoyah Unit 2 were removed so that the tubes can be returned to service following u-bend heat treatment. The roll plug removal process consisted of a helical scan with a TIG torch on the inside diameter of the plug with subsequent plug removal with a pulling tool. The acceptance criteria for roll plug removal is a visual inspection with a remote camera. The visual inspection on all tubes from which plugs were removed was acceptable. Following plug removal, inspection of the roll plugs from the first two steam generators (Steam Generators 1 and 4) resulted in the identification of significant burn through on some plugs. Consequently, an unscheduled Motorized Rotating Pancake Coil (RPC) eddy current examination was performed on the bottom 6 inches of the hot leg tubes in Steam Generators 1 and 4 to verify no damage from the burn through on the roll plugs, before continuing the roll plug removal process on Steam Generators 2 and 3. Bobbin coil inspection of the row 1 tube ends did not detect any degradation in the tube ends in Steam Generators 1 and 4. Bobbin coil inspection did not detect the indications because the response from the tube ends greatly exceeds the amplitude of the indications, thus masking their presence.

The RPC inspection results show that 39 Row 1 tubes in Steam Generator 1 (33 of which are planned to be returned to service following u-bend heat treatment) have both single and multiple axial indications near the tube ends at the bottom of the tubesheet (Reference 1). All indications are on the hot leg tube ends. Rotating pancake coil (RPC) probe data for the tubes show the indications to be within 0.04 to 0.23 inch from the bottom of the tube end. The indications appear to be very short and the indications have phase angles which suggest both tube inner and outer diameter origin. Visual and eddy current bobbin inspection of the row 1 tubes in Steam Generators 2 and 3 following plug removal revealed no detectable degradation at the tube ends.

Each Sequoyah steam generator tube has undergone a partial depth hardroll along with Westinghouse Explosive Tube Expansion (WEXTEX) expansion over the full depth of the tubesheet. From the design drawings, the tube ends extend a minimum of 0.37 inches below the bottom of the tubesheet and 0.22 inches below the tubesheet cladding. The locations of the axial indications observed by RPC (remaining in service) are below the tube to tubesheet weld and the hardroll and WEXTEX expansion regions (and therefore, below the primary pressure boundary). The root cause of the indications is not known. Tennessee Valley Authority concludes that the most probable cause was the cold working of the tube ends in the roll plug installation and removal processes. Visual inspection of the tube ends show no apparent damage due to a loose part in the primary system.

The purpose of this evaluation is to address the safety significance of subsequent plant operation with the identified steam generator Row 1 tubes remaining in service with the tube end indications and the resulting technical specification changes related to this issue. This evaluation demonstrates that: 1) the integrity of the tube bundle will be maintained with the single and multiple axial indications present on the tube ends in Steam Generator 1, and 2) operation of Sequoyah Unit 2 following implementation of a technical specification change which excludes removing from service tubes with

degradation occurring in that portion of the tube from the tube end to the start of the tube-to-tubesheet weld does not represent an unreviewed safety question.

2.0 REGULATORY BASIS

The thin-walled tubing of the four steam generators constitutes more than one-half of the reactor coolant pressure boundary (RCPB). Maintenance of the structural and leakage integrity of the RCPB is a requirement under Title 10 of the Code of Federal Regulations Part 50, Appendix A.

Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes", issued for comment, is used as a standard for assessing the operability of degraded steam generator tubing within the industry.

Specific requirements governing the maintenance of steam generator tube integrity are contained in the plant technical specifications and Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). These include requirements for periodic inservice inspection of the tubing, flaw acceptance criteria, and primary to secondary leakage limits.

These requirements are addressed below in assuring adequate steam generator tube integrity during subsequent plant operation of Sequoyah Unit 2.

3.0 EVALUATION

Tube Structural Integrity Assessment

The locations of the axial indications observed by RPC are currently below the tube-to-tubesheet weld and the hardroll and WEXTEX expansion regions of the Row 1 tubes in Steam Generator 1. Consequently, it is concluded that the axial indications do not affect the structural and leakage integrity of the pressure boundary.

However, should the indications represent single or multiple axial cracks on the tube ends, the potential for crack propagation into the primary pressure boundary needs to be assessed.

Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet. Thus the RG 1.121 criteria are satisfied by tubesheet constraint. Crack lengths do not need to be limited by burst considerations and operating leakage limits are not required to detect crack lengths associated with tube burst.

Addressing the potential effect of tube crack propagation on the tube-to-tubesheet welds, in the limiting case where the weld would no longer be effective due to the presence of cracking, it is shown in Reference 2 that even if a tube is circumferentially degraded within the tubesheet to the extent that tube separation could occur, a length of engagement of tube can be defined that would prevent pullout of the tube and would result in negligible leakage during all plant conditions. This length of engagement has been calculated for Sequoyah Unit 2 and is less than 5.6 inches from the bottom of

the WEXTEX expansion transition or approximately 15 inches above the steam generator tube end. Should the tube end indications be stress corrosion cracking, eddy current bobbin inspection of the indications on a cycle to cycle basis is more than sufficient to monitor steam generator tube integrity. Any tube end indications not detected by the bobbin coil probe do not compromise steam generator tube integrity. No future inspections other than typical technical specification inspections are necessary. Plant operating experience shows that PWSCC axial crack propagation rate for WEXTEX plants, which is a function of crack length, is less than 0.315 inches per effective full power year in the WEXTEX zone.

Operating Leakage Considerations

Plant operating experience shows that primary water stress corrosion cracking (PWSCC) can occur in both the hardroll and WEXTEX expansion regions in the tubesheet of a steam generator. Extensive European operating experience has been obtained with axial PWSCC cracks left in service. This operating experience has demonstrated negligible normal operating leakage from PWSCC cracks even under free span conditions in roll transitions. PWSCC cracks (if they were to occur in WEXTEX expansions) in the tubesheet region would be even further limited by the tight tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint.

Consequently, negligible operating leakage would be expected from cracks in the tubesheet region of Steam Generator 1 should the Row 1 tube end indications be representative of cracks and should the cracks propagate into the hardroll and WEXTEX expansion region of the tubes.

Accident Leakage Considerations

The accident leakage must be limited to acceptable limits established by the FSAR evaluations. The postulated steam line break (SLB) leakage remains limiting for the Sequoyah Unit 2 steam generators. Reference 2 shows that the SLB leak rate for a long crack extending to an elevation approximately 7 inches below the WEXTEX expansion transition is about 7 X 10^{-4} gpm. This leakage rate further decreases by about 40% per inch of additional depth within the tubesheet region and as the tubesheet radius increases. Therefore, should the tube end indications represent cracks and should the cracking propagate into the pressure boundary, total SLB leakage as a result of any crack propagation would be expected to be negligible.

Loose Parts Assessment

Visual inspection of the Steam Generator Row 1 tube ends did not show any damage due to loose part impacting. Based on this observation and because the degradation is axially oriented, no tube end loose part fragments are expected to be generated during operation.

4.0 UNREVIEWED SAFETY QUESTION ASSESSMENT

Plant operation with the tube end indications in the Row 1 tubes of Steam Generator 1 of Sequoyah Unit 2 has been evaluated and determined not to involve an unreviewed safety question on the basis of the following justification.

1. Will the probability of occurrence of an accident previously evaluated in the FSAR be increased?

   No. The condition described in this evaluation results in tube integrity considerations commensurate with Reg. Guide 1.121 criteria both analytically and empirically. If the indications are hypothetically considered as cracks, the Row 1 tube end indications neither adversely affect steam generator tube integrity or any other component, nor does the presence of the indications alter the function of the steam generator or any other component. Therefore, plant operation with the tube end indications present in the Row 1 tubes does not increase the probability of an analyzed accident such as a steam generator tube rupture event.

2. Will the consequence of an accident previously evaluated in the FSAR be increased?

   No. The accidents of interest are steam generator tube rupture, large break LOCA and steam line break. The only consequence which could be caused by plant operation or by the occurrence of a faulted condition event with the tube end indications would be negligible leakage between the primary to secondary systems. Such leakage is expected to be insignificant at both normal and faulted conditions. Therefore, plant operation with the tube end indications present in Steam Generator 1 does not result in an increase in the consequences of a previously analyzed accident.

3. May the possibility of an accident of a different type than already evaluated in the FSAR be created?

   No. Any hypothetical accident as a result of plant operation with the Row 1 tube end indications would be bounded by the consequences of a postulated steam generator tube rupture. Therefore, operation of the steam generator in this condition does not result in a previously unanalyzed accident.

4. Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

   No. Plant operation with the Row 1 tube end indications present in Steam Generator 1 will not adversely affect the integrity of the steam generator or other components. Plant operation with the tube end indications does not adversely effect the function of the steam generator or any other component.

5. Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

   No. The only consequence that could be caused by subsequent plant operation of Sequoyah Unit 2 would be negligible primary to secondary leakage during all plant conditions. The installed systems assumed operable to mitigate the radiological consequences of an accident per plant Technical Specifications and analysis assumptions are not adversely affected by the presence of the tube end indications.

6. May the possibility of a malfunction of equipment important to safety different than that already evaluated in the FSAR be created?

No. Plant operation with the tube end indications present in Steam Generator 1 does not introduce new failure scenarios or create new failure modes. The continued operability of systems required by plant Technical Specifications or assumed in accident analyses to assure plant safety are not adversely impacted.

7. Will the margin of safety as defined in the basis of any Technical Specification be reduced?

No. The locations of the axial indications observed are currently below the tube-to-tubesheet weld. Consequently, it is concluded that the axial indications do not affect the structural and leakage integrity of the primary presure boundary. Should the cracks be single or multiple axial cracks on the tube ends, the effect of crack propagation was evaluated. Tube burst is precluded for cracks within the tubesheet by the constraint provided by the tubesheet. Therefore, crack lengths do not need to be limited by burst considerations and operating leakage limits are not required to detect crack lengths associated with tube burst. However, primary to secondary leakage must be shown to remain within acceptable limits during all plant conditions. Leak rate testing shows that such leakage would be negligible during all plant conditions. Since the pressure boundary integrity, acceptable leak rate, and function of the steam generator are not affected by the presence of the tube end indications, the margin to safety is not reduced.

5.0 CONCLUSION

Based on the above, plant operation of Sequoyah Unit 2 with the tube end indications in the Row 1 tubes in Steam Generator 1 does not represent an unreviewed safety question as defined in 10 CFR 50.59 (a)(2).

6.0 REFERENCES

1. Telecopy - David Hughes to P.J. Prabhu, "TVA S2C6 EC Row 1 Roll Plug Removal SG -1 Hot Leg RPC HTE + 6 inches for Deplugged Tube Ends Indications", 8/3/94

2. WCAP-13532, Rev. 1 ,"Sequoyah Units 1 and 2 W* Tube Plugging Criteria for SG Tubesheet Region of WEXTEX Expansions", November, 1992

ENCLOSURE 5

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE

SEQUOYAH NUCLEAR PLANT (SQN) UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-94-14)

COMMITMENT LIST

COMMITMENT LIST

TVA will change the appropriate inspection procedures by January 30, 1995, to include a bobbin coil inspection of the protruding tube ends in the 33 tubes in Unit 2 Steam Generator No.1 in subsequent refueling outages, or until the indications are removed or the tubes are plugged. These procedure changes will include contingencies for indications that are detected with the bobbin coil inspection in this region. In this case, TVA will expand the inspection based on the number and nature of the indications in the nonplugged hot leg Row 1 tubes in Unit 2.