ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-92-16)

LIST OF AFFECTED PAGES

Unit 1

3/4 4-27

Unit 2

3/4 4-32

3/4.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 AND 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10. The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10. In addition to the requirements of Specification 4.0.5,

a. Each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C 4.b of Regulatory Guide 1.14, Revision 1, August 1975. FP

~~b. The reactor vessel nozzles shall be inspected at the end of each 10 year inspection interval, using techniques at least as sensitive as those used to conduct the supplemental examination performed prior to fuel loading. The results of this examination will be reported to the Commission.~~

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APPLICABILITY: All MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

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ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-92-16)

DESCRIPTION AND JUSTIFICATION FOR

DELETION OF SURVEILLANCE REQUIREMENT 4.4.10.b

Description of Change

TVA proposes to modify the Sequoyah Nuclear Plant (SQN) Units 1 and 2 technical specifications (TSs) to delete Surveillance Requirement (SR) 4.4.10.b. This SR supplements the requirements of Specification 4.0.5 (Section XI of the American Society of Mechanical Engineers [ASME] Boiler and Pressure Vessel Code) for maintaining structural integrity of ASME Code Class 1, 2, and 3 components in the reactor coolant system. SR 4.4.10.b currently requires, by reference to supplemental examination performed before fuel loading, that SQN's reactor pressure vessel (RPV) nozzles be inspected for underclad "cold" cracks at the end of each 10-year inspection interval using techniques at least as sensitive as those used to conduct the supplemental examination performed before fuel loading.

Reason for Change

TVA is proposing to delete SR 4.4.10.b to eliminate the need for supplemental inspection of SQN's reactor vessel nozzles for underclad cold cracking. Evaluation indicates that there is no value or technical basis for performing a supplemental 10-year nozzle inspection.

The current TS requirement for performing a supplemental 10-year inspection of SQN's RPV nozzles was added to the SQN TSs in 1980 (the SQN Unit 1 full-power license was issued in September 1980). At that time, ultrasonic testing (UT) examinations were performed on Units 1 and 2 RPV nozzles to establish whether underclad cold cracking was present. The results of the UT examinations are documented in Section 5.2.6 of the Sequoyah Safety Evaluation Report, NUREG-0011, Supplement 1. It was recognized at that time that the SQN nozzles met the ASME Section XI code with sufficient margin against flaw-induced fracture. A 10-year supplemental examination requirement was however added to the SQN TSs for added assurance. The SQN TSs are unique with regard to this supplemental requirement.

SQN Units 1 and 2 are both approaching their Cycle 6 refueling outages. During these outages TVA will be conducting the standard 10-year ASME Section XI in-service inspections (ISIs). TVA has determined that minor expansion of these standard ASME Section XI nozzle inspections would provide sufficient examination coverage for establishing whether indications recorded during the 1980 UT examinations satisfy the ASME Section XI code criteria. The minor expansion to the standard ASME examination would be applicable to SQN Unit 1 and would encompass indications that were found in the safe-end region of the Loop 1 outlet nozzle (where the wall thickness is relatively thin). Expansion of the examination coverage would be achieved by increasing the safe-end weld scan limit two inches into the nozzle bore. The expanded examination of the safe-end region, plus the standard ASME Section XI inspections, would encompass 55 percent of the underclad cold crack indications reported in 1980 for Unit 1. TVA's proposed examinations would thereby include coverage of the most structurally significant indications and would encompass over one-half the total number of underclad cold crack indications.

TVA has evaluated the impact of performing the special supplemental RPV nozzle examinations to comply with the current SR 4.4.10.b. TVA

estimates the cost for setup, UT examination, and recovery to be approximately $60,000. This does not include replacement power costs associated with the additional two days of critical path outage time required to perform these supplemental examinations. Current estimates of the replacement power cost during the spring months (SQN's Unit 1 Cycle 6 refueling outage is scheduled from April to June 1993) would be approximately $700,000. These costs would again be duplicated during the Unit 2 outage and each unit's subsequent 10-year inspection interval. Based on the technical information that is now available on the underclad cold cracking issue, the upcoming performance of required nozzle inspections for ASME Section XI (including the expanded UT examination coverage of the Loop 1 outlet nozzle safe-end region), and the monetary impact associated with performing the supplemental nozzle inspections, TVA finds that compliance with the current SR 4.4.10.b results in an unusual hardship without a compensating increase in the level of quality and safety.

Justification for Change

The following discussion provides a historical summary of the reactor vessel nozzle underclad cracking issue. The reactor vessel nozzle crack issue arises from a fabrication problem discovered in France by the French Nuclear Steam Supplier, Framatone. Framatone engineers in June-July 1979 discovered, using then new UT techniques, that certain reactor vessels had "underclad" cracks in the reactor vessel nozzle area. This type of cracking is referred to as underclad cold cracking or hydrogen-induced cracking. Westinghouse Electric Corporation learned of this issue and began an extensive investigation and evaluation program.

Westinghouse reported to TVA in October 1979 that vessels manufactured by Rotterdam Shipyard were potentially susceptible to the underclad cold crack problem; therefore, the vessels at TVA's Watts Bar Nuclear Plant (WBN) and SQN should be evaluated. TVA conducted an extensive evaluation of the WBN Unit 2 vessel (all eight nozzles) to ensure there were no safety issues. The results of that investigation were reported by Westinghouse to NRC on January 31, 1980, in a letter from T. Anderson to D. Eisenhut. TVA reviewed the Westinghouse report, and as a matter of added technical assurance, insisted that one nozzle at SQN be examined. The examination was performed on Unit 1 on February 2, 1980. This examination revealed no underclad cold cracks; however, it did show underclad reheat cracks. These underclad reheat cracks are common in cladding heat-affected zones where certain base materials and welding techniques are combined (i.e., deposition process). The potential for reheat cracks from the deposition process has been recognized, previously evaluated, and found to be acceptable by NRC.

NRC looked at the Westinghouse report on February 20, 1980. During a telephone conversation the same day between NRC and TVA, NRC requested that the issue be resolved before the licensing of SQN.

Consequently, on February 22, 1980, NRC requested that TVA conduct UT inspections of the remaining seven nozzles on SQN Unit 1 for detection of underclad cracks. The inspection was conducted by personnel from Sonic Systems International and supervised by Westinghouse personnel. The NRC inspector witnessed portions of the UT inspection on six nozzles for conformance to procedure requirements. The inspector also reviewed the qualification records and equipment certifications. Indications were found in the seven remaining nozzles. The UT inspection results from all eight Unit 1 nozzles are summarized below:

Loop 1 Inlet	3 reportable indications (underclad cracking)
Loop 2 Inlet	21 reportable indications (underclad cracking)
Loop 3 Inlet	2 reportable indications (underclad cracking)
Loop 4 Inlet	5 reportable indications (underclad cracking)
Loop 1 Outlet	5 reportable indications (underclad cracking)
Loop 2 Outlet	1 reportable indication (innocuous)
Loop 3 Outlet	Indications of reheat cracking phenomena
Loop 4 Outlet	1 reportable indication (innocuous)

The results of the UT determined that one outlet nozzle contained a cracking pattern characteristic of underclad reheat cracking; two outlet nozzles contained no indications of either reheat or cold cracking; and the remaining five nozzles (one outlet, four inlets) contained cracking characteristic of underclad cold cracking confined to the nozzle bore. In the Sequoyah Safety Evaluation Report, NUREG-0011, Supplement 1, NRC conservatively considered that one nozzle contained underclad reheat cracking, and the remaining seven nozzles contained crack indications. All SQN Unit 1 indications were within the acceptance standards established by Section XI of the ASME code. A complete report was filed by TVA with NRC's Director of Nuclear Reactor Regulation on February 26, 1980. A revised report dated February 28, 1980, addressed additional questions raised by NRC and superseded the February 26, 1980, report.

At a meeting with NRC in Bethesda, Maryland, on February 22, 1980, TVA committed to perform UT inspections of the remaining SQN Unit 2 and WBN Unit 1 reactor vessel nozzles to determine the extent of the Framatone underclad cracking. The SQN Unit 2 inspection revealed indications in six of the eight RPV nozzles that were typical of underclad reheat cracking. Indications in the seventh nozzle were small but did not fall in bands commonly associated with underclad reheat cracking. A case of underclad cold cracking, as experienced in the Framatone nozzles, could not be substantiated since the SQN Unit 2 indications fall in an area clad with only one layer. The eighth nozzle contained only one small indication. Accordingly, the SQN Unit 2 nozzle indications did not reveal evidence of any underclad cold cracking since the cladding technique precluded the formation of underclad cold cracks. All 16 SQN Unit 2 and WBN Unit 1 nozzles met ASME Section XI preservice inspection acceptance standards with the exception of one indication on WBN Unit 1 that was removed. Both vessels were subsequently determined to be acceptable for use. The results of the SQN Unit 2 and WBN Unit 1 inspections were submitted to NRC in a March 20, 1981, letter.

In March 1980, NRC issued Supplement 1 to their 1979 Safety Evaluation Report (NUREG-0011) regarding TVA's application for SQN's operating license. In Section 5.2.6 entitled "In-Service Inspection Report," NRC recognized that compliance with ASME Section XI code provided adequate assurance that SQN's RPV nozzles had sufficient margin against flaw-induced fracture. In order to provide added assurance that adequate margins continue to be maintained during service, NRC required TVA to inspect the RPV nozzles at the end of each 10-year inspection interval using techniques at least as sensitive as those used to conduct the supplemental UT examination performed before the fuel loading. This requirement (SR 4.4.10.b) was included in SQN's original (full power for Unit 1) TSs for both units. It should again be noted the existence of underclad cold cracking on Unit 2 is precluded by the fact that the Unit 2 nozzles were clad with only one layer. In addition, all indications (both reheat and cold cracks) from Units 1 and 2 were found to be within the acceptance criteria of ASME Section XI.

The detailed technical basis for deleting the supplemental UT examination of the SQN Units 1 and 2 RPV nozzles for underclad cold cracking has been included in Enclosure 4. To summarize the Enclosure 4 information, Westinghouse, Framatome, and the Electricite de France (EdF) (owner of nuclear power plants in France) have performed fracture-mechanics analyses to assess the pressure vessel margin against fracture. These analyses indicate no detrimental impact from underclad reheat cracks and underclad cold cracks. Based on these fracture-mechanics analyses, the original design margins will be maintained for continued operation. Since 1980, EdF has performed periodic ISIs of RPV nozzles with known underclad cold cracks similar to SQN Unit 1. A total of 26 EdF plants were involved in the RPV nozzle bore underclad cold cracking. For plants not in service, the examination plan required preservice examinations and subsequent periodic in-service examinations during the first-, fifth-, and tenth-year refueling outages. For plants already in service, examinations were required at the earliest possible refueling outage and repeated five and ten years later.

Thus far, ten EdF plants have been through the entire ten-year augmented inspection program cycle. There has been no evidence of crack growth. In the case of 11 EdF plants, results of the first- and fifth-year inspections showed no cracks greater than 5 millimeters deep. By agreement between EdF and the regulatory authority, the requirement to perform the tenth-year examination was not imposed for these plants, and EdF does not anticipate performing additional nozzle examinations until the twentieth refueling outage. The remaining five plants are in various stages of their respective augmented examination program cycles, and no crack extension has been reported.

In addition to the fracture-mechanics analyses and the EdF inspection results, the ASME code Section XI and Regulatory Guide (RG) 1.150 require the examination of a sizeable length of the nozzle bore, using a demonstrated technique capable of detecting a near-surface two percent notch. During the Unit 1 Cycle 6 refueling outage, TVA will be performing the standard 10-year ASME Section XI and RG 1.150 inspection of the Unit 1 RPV nozzles. In addition to the ASME Section XI and RG 1.150 examinations, TVA will examine an additional length of nozzle

bore known to incorporate the area where four UT indications are located. These four indications are located toward the safe-end region on the Unit 1, Loop 1 outlet nozzle. This examination coverage, plus the required ASME Section XI examination coverage, will include 55 percent of the total underclad crack indications identified during SQN's 1980 UT examinations. This coverage provides a representative sample of the 1980 indications to ensure that these indications satisfy the ASME Section XI code criteria. Based on the above information and TVA's review of SQN's 1980 UT reports, it is concluded that there is no technical basis for further supplemental ISI of prior indications.

Environmental Impact Evaluation

By performance of an expanded UT examination of the Unit 1, Loop 1 outlet nozzle and the standard ASME Section XI and Regulatory Guide 1.150 UT examinations, TVA has bounded the structurally significant underclad cold crack indications. The proposed change request 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 UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN TS-92-16)

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.

   TVA proposes to modify the SQN Units 1 and 2 TSs to delete an existing surveillance requirement (SR) (SR 4.4.10.b) that currently requires that the reactor vessel nozzles be inspected for underclad cold cracks at the end of each 10-year inspection interval using techniques at least as sensitive as those used to conduct the supplemental examination performed before fuel loading. Ultrasonic examinations were performed on SQN's reactor vessel nozzles in 1980. The Unit 1 nozzles were potentially susceptible to the underclad cold cracking because of the two-layer cladding process. Examination of the Unit 1 nozzles revealed one outlet nozzle with a cracking pattern characteristic of underclad reheat cracking, two outlet nozzles with no indications, and five nozzles (one outlet, four inlets) with a cracking pattern characteristic of underclad cold cracking confined to the nozzle bore. The results of the Unit 2 examinations revealed no evidence of any underclad cold cracking since the single-layer cladding process used on Unit 2 precluded the formation of cold cracks. All indications from the Unit 1 and Unit 2 nozzles were within the acceptance criteria of the American Society of Mechanical Engineers (ASME) code.

   Westinghouse Electric Corporation has performed fracture-mechanics analyses that conclude that the minimum critical flaw depth is 1.5 inches at the governing region of the nozzle (the nozzle corner region) and that fatigue crack growth would not exceed 0.06 inch over a 10-year service period. For a 40-year service life, these flaws would not exceed 0.21 inch. These analyses indicate that there would be no detrimental impact from underclad indications and that the original design margins would be maintained for the life of the plant. In addition, nuclear power plants in France that exhibited known underclad cold cracking similar to SQN Unit 1 have performed preservice and periodic ISIs and have reported no evidence of crack growth. Examinations performed in accordance with Section XI of the ASME code as supplemented by Regulatory Guide 1.150, plus expansion of the examination coverage to the safe-end regions on the Unit 1, Loop 1 outlet nozzle, provide a sufficient basis upon which to determine whether indications previously recorded during the 1980 SQN examinations satisfy the ASME Section XI code criteria. The expanded ASME Section XI examination of the Unit 1 nozzles will include over one-half the total number of underclad cold crack indications reported in 1980.

Framatone, a Westinghouse licensee; and Electricite de France (EdF), owner of nuclear power plants in France, have also performed fracture-mechanics analyses that conclude that the critical flaw size is greater than one inch and that fatigue crack growth of underclad cold cracks is insignificant. Based on the above information, TVA concludes that the elimination of the supplemental 10-year inspection requirement from TSs does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The ASME Section XI examinations, plus the expanded examination of the safe-end region on the Unit 1, Loop 1 outlet nozzle, will provide sufficient information to determine if nozzle indications that were recorded during the SQN 1980 preservice examinations satisfy the ASME Section XI code criteria. Accordingly, the elimination of the current TS requirement would 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.

   Fracture-mechanics analyses performed by Westinghouse, Framatone, and EdF indicate that reheat cracking and underclad cold cracking in reactor vessel nozzles do not reduce the original design margins over 40-year plant life. In addition, periodic ISIs of reactor vessel nozzles with known underclad cracks have demonstrated no growth. The ASME Section XI examinations, plus the expanded examination of the safe-end region on the Unit 1, Loop 1 outlet nozzle, will provide sufficient information to determine if nozzle indications that were recorded during the SQN 1980 preservice examinations satisfy the ASME sectional code criteria. Consequently, the removal of the supplemental examination requirement from SQN TSs does not involve a significant reduction in the margin of safety.

ENCLOSURE 4

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-92-16)

WESTINGHOUSE SAFETY EVALUATION

CHECKLIST (SECL-92-375)

B38 930107 801

Customer No(s).
N/A
Westinghouse No(s).
N/A

WESTINGHOUSE
NUCLEAR SAFETY EVALUATION CHECK LIST

1.0 NUCLEAR PLANT(S): Sequoyah Units 1 and 2

2.0 SUBJECT (TITLE): Deletion of Augmented 10-Year Inservice Inspection for Reactor vessel Nozzles for Underclad Cracking

3.0 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)?

* Elimination of the requirement for supplemental inspection of reactor vessel nozzles requires modifications to the Sequoyah Unit 1 and Unit 2 Technical Specifications, Surveillance Requirement 4.4.10.b.

4.0 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 Section 5.0 REMARKS and explain below.

If the answer to any of the above questions in Section 4 cannot be answered in the negative, based on written safety evaluation, the change cannot be approved without an application for license amendment submitted to the NRC pursuant to 10CFR50.90.

5.0 REMARKS

The following summarizes the justification upon the written safety evaluation, (1) for answers given in Part B of the Safety Evaluation Check List:

FSAR UPDATE

Section: N/A Page(s): N/A Table(s): N/A Figure(s): N/A

6.0 APPROVAL LADDER

Nuclear Safety Preparer:	DL Cecchett D. L. Cecchett	Date:	01-06-93
Nuclear Safety Reviewer:	L. V. Tomasic	Date:	1.6.93
Coordinated with Engineers:	W. M. Schivley (Signature on file)	Date:	

1.0 BACKGROUND INFORMATION AND SUMMARY

This safety evaluation addresses the deletion of the Sequoyah Units 1 and 2 Technical Specifications requirement, Surveillance Requirement 4.4.10.b, to perform augmented 10-year inspections of the reactor vessel nozzles.

Revision 2 is made to reflect the fact that the determination of indication growth cannot be realized using standard ASME Section XI/Regulatory Guide 1.150 techniques. Therefore, statements made in Revision 1 referring to this had to be modified. Instead these techniques will be used to determine if the indications are within the acceptance standards of ASME Section XI, IWB-3500.

In 1980, the Sequoyah Units 1 and 2 reactor vessel nozzle bores were subject to ultrasonic examinations to establish whether underclad cold cracking was present. Although indications were observed during these examinations, the majority were characteristic of underclad reheat cracking. Analyses were performed assuming the more conservative case - underclad cold cracks - which demonstrated no impact on design margin. These conclusions were accepted by the US Nuclear Regulatory Commission. However, a requirement for an augmented inservice examination was imposed.

Westinghouse has reviewed the NRC imposed requirement for augmented inspection of reactor vessel nozzles for underclad cracking at the end of each 10-year inspection interval outlined in the Sequoyah Safety Evaluation Report NUREG-0011, Supplement 1 (Reference 1), and performed a technical evaluation of that report. Based upon the available technical information pertaining to underclad cracking it was concluded: 1) that no technical basis exists for the supplemental requirement; and 2) that examinations performed according to ASME Section XI (Reference 2) with a modification to the scan plan on one nozzle to safe end weld and Regulatory Guide 1.150 (Reference 3) will provide a sufficient basis upon which to establish whether the indications recorded during the 1980 examinations are within the acceptance standards of ASME Section XI, IWB-3500.

Based upon this evaluation, it is concluded that elimination of the requirement for a 10-year augmented inspection of the reactor vessel nozzles does not result in an unreviewed safety question as defined in 10 CFR 50.59. This safety evaluation provides the basis for the modifications to the Sequoyah Units 1 and 2 Technical Specifications.

2.0 LICENSING BASIS

Sequoyah Units 1 and 2 FSAR Chapter 5.4, "Reactor Vessel and Appurtenances," provides the design bases, description, evaluations and tests, and inspections for the reactor vessel and its appurtenances. This includes the requirements for ultrasonic examinations based on ASME Code Section XI criteria.

NRC Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," describes the requirements and procedures with regard to implementation of preservice and inservice examinations of reactor vessel welds by ultrasonic testing.

General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A of 10

CFR Part 50, requires, in part, that components which are part of the reactor coolant pressure boundary (RCPB) be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The design of the ASME Code Class A components of the RCPB in Sequoyah Units 1 and 2 incorporates provisions for access for inservice inspection in accordance with Section XI of the ASME Code.

Reference 1, Supplement 1 to the Sequoyah Units 1 and 2 NRC Safety Evaluation Report, states that, "Compliance with Section XI of the ASME Code provides adequate assurance that the reactor vessel has sufficient margin against flaw induced fracture." However, to provide added assurance that adequate margins continue to be maintained during service, the NRC required TVA to inspect the reactor vessel nozzles at the end of each 10-year interval. The intent of that requirement was met by adding Surveillance Requirement 4.4.10.b to LCO 3.4.10, "ASME Code Class 1, 2 and 3 Components," in the Sequoyah Units 1 and 2 Technical Specifications.

The technical basis for deleting a supplemental inservice inspection of the Sequoyah Units 1 and 2 reactor pressure vessel nozzles for underclad cracking is provided below.

3.0 EVALUATIONS

Reference 4 provides the technical basis (with the appropriate references) for the Safety Evaluation Checklist. A portion of that document is repeated here.

In 1980, the Sequoyah Units 1 and 2 reactor vessel nozzle bores were ultrasonically examined to establish whether underclad cold cracking was present. In Unit 1, the ultrasonic tests determined that one outlet nozzle contained a cracking pattern characteristic of underclad reheat cracking, two outlet nozzles contained no indications of cracking, and the remaining five nozzles (one outlet, four inlets) contained cracking characteristic of underclad cold cracking confined to the nozzle bore[1]. In Unit 2, the ultrasonic tests determined that six nozzles (three inlet, three outlet) contained a cracking pattern characteristic of underclad reheat cracking, and the remaining inlet and outlet nozzle had small reflectors not characteristic of either underclad reheat or cold cracking.

A review of the welding processes for Sequoyah Units 1 and 2 reactor vessel nozzle bores has resulted in the following:

- Sequoyah Unit 1 (TVA)

  For multi-layer strip cladding a preheat of 194°F min. was maintained on the 1st layer followed by low temperature 392°F min. heat treatment for 2 hrs. No preheat was applied to the subsequent layers. For multi-layer shielded metal arc (SMA) cladding a preheat of 212°F min.

1: In the Sequoyah Safety Evaluation Report, NUREG-0011, Supplement 1 Reference 1), the NRC conservatively considered that one nozzle contained underclad reheat cracking and the remaining seven nozzles contained crack indications.

was maintained on the 1st layer followed by a low temperature (392°F min.) heat treatment for 2 hrs. A preheat of 212°F min. was applied to the subsequent layers. This information has been verified by statements from Rotterdam Dockyard Company (RDM) personnel.

Westinghouse's records indicate that the Unit 1 inlet and outlet nozzles cladding techniques were identical with the exception that on the vessel side of the outlet nozzles at the nozzle protrusion more layers have been welded (layer thickness > 0.67 in./17mm). The nozzle bore was covered with multi-layer (2 layers) strip cladding (30 x 0.5 mm/309-308). The regions near the vessel side and the safe-ends were cladded with two layers (309-308) utilizing a manual shielded metal arc (MSMA) technique.

The nozzle bore region was clad using welding parameters consistent with nozzles confirmed to have underclad cold cracking. Therefore, Sequoyah Unit 1 may be susceptible to such cracking.

- Sequoyah Unit 2 (TEN)

  RDM maintained a preheat of 194°F min. on all clad layers regardless of cladding technique. A low temperature (392°F min.) heat treatment for 2 hrs. was applied after the first layer and again after the final layer. This information has been verified by statements from RDM personnel.

  Westinghouse's records indicate that the Unit 2 inlet and outlet nozzles cladding techniques were identical with the exception that on the vessel side of the outlet nozzles at the nozzle protrusion more layers have been welded (layer thickness > 0.67 in./17mm). However, there was a difference in cladding techniques between Units 1 and 2 namely the nozzle bores were cladded with only one layer of 60.0 x 0.5 mm (309) strip in Unit 2 in lieu of two layers of 30.0 x 0.5 mm (309-308) strip used on the bores of Unit 1. The regions near the vessel side and the safe-ends were cladded with two layers (309-308) utilizing a manual shielded metal arc (MSMA) technique.

  The welding parameters of the Sequoyah Unit 2 nozzles were such that underclad cold cracking should not exist.

The technical basis for deleting a supplemental inservice inspection of the Sequoyah Unit 1 and 2 reactor pressure vessel nozzles for underclad cracking is as follows:

Underclad Reheat Cracking

Reheat cracking, typical of that detected in the Sequoyah Units 1 and 2 reactor vessel nozzles was the subject of an exhaustive international research effort between 1970 and 1974. The characteristics of reheat cracking were well defined and agreed upon by the metallurgical and welding communities involved. The results of these programs were documented in Welding Research Council Bulletin 197, dated August 1974.

Metallurgical investigations identified reheat cracking as grain boundary separations in the base metal, restricted in extent to the grain coarsened part of the weld heat-affected zone directly beneath the weld bead overlap. It appears that the anomaly is associated with a post weld heat treatment of a specific grade of material (SA508 Class 2 composition) clad by specific welding processes. Welding processes associated with grain boundary separations (reheat cracking) include strip cladding performed by European fabricators and six wire cladding performed by Babcock and Wilcox. Cladding processes that produce excessive heat in process leading to grain coarsening of the weld heat-affected zone in SA508 Class 2 forging material will result in reheat cracking. Thus many, if not all plants operating in the United States today which were fabricated using SA508 Class 2 forging material will exhibit reheat cracking. Both Sequoyah Unit 1 and 2 reactor pressure vessel nozzles were fabricated from SA508 Class 2 forging grade steel. Thus both Sequoyah Units are susceptible to underclad reheat cracking.

Westinghouse, as well as others, performed investigations on nozzle drop-outs which represented all manufacturing parameters and all weld overlays. The observed separations (reheat cracks) depths ranged from 40 mils to a maximum of 99 mils. The observed lengths ranged from 59 mils to a maximum of 394 mils. The aspect ratio (length/depth) for the maximum length separation was 5 to 2.

Westinghouse performed a fracture mechanics analysis, using ASME Code Section XI, IWB-3600 and Appendix A methods, to determine the fatigue crack growth of reheat cracks and reported that the maximum crack growth in 40 years of plant life is 4 mils. The estimated fatigue crack growth of 4 mils is somewhat conservative because Westinghouse assumed a maximum flaw depth equal to the maximum depth of grain coarsened region at the overlay area, i.e. 125 mils, and that the flaws were continuous.

Based upon these evaluations performed by Westinghouse and the exhaustive international research efforts, the Nuclear Regulatory Commission concluded that reheat cracking did not adversely affect reactor pressure vessel integrity. NUREG-0011, Supplement 1 reiterates these earlier conclusions. Therefore there is no technical basis for augmented inservice inspections of prior indications classified as reheat cracking.

Underclad Cold Cracking

In late 1979 Framatome, a Westinghouse licensee, reported detection of cracking in reactor pressure vessel nozzles beneath the cladding. The cracking which was detected during shop inspections existed in a broad area along the nozzle bores but was more prevalent in the thicker section of the nozzles. This type of cracking is referred to as underclad cold cracking or hydrogen-induced cracking as opposed to underclad reheat cracking. This type of cracking is associated with a multiple layer, strip electrode cladding process in which the second layer is deposited without preheat. This cladding process is similar to that used in the nozzle bores of Sequoyah Unit 1. Destructive

metallurgical examinations of boat samples revealed that these underclad cold cracks have the following primary characteristics:

- The cracks are confined to the heat-affected zone produced by the second layer of cladding and are located in the low alloy steel base material. The destructive investigations found that the heat-affected zone produced by the second layer of cladding was relatively hard compared to the heat-affected zone produced by the first weld pass. This hard heat-affected zone microstructure combined with the hydrogen content and residual stresses were the major contributing factors of the cracking process.

- The cracks are oriented perpendicular to the cladding direction.

- The crack depth ranges from 100 mils to 325 mils.

Electricite de France (EdF), owner of the nuclear power plants in France that exhibited underclad cold cracking in the nozzle bores, performed preservice and inservice inspections as a result of this discovery. Indications of underclad cold cracking (similar to that seen in Sequoyah Unit 1) were detected during these inspections. The plants were permitted to go into service and/or continue operating based upon the results of the fracture mechanics performed by Framatome/EdF which showed that the critical flaw size was greater than 1 inch and that the fatigue crack growth extension of the underclad cold cracks would be insignificant. A total of twenty-six plants operated by Electricite de France (EdF) were involved in the reactor vessel nozzle bore underclad cold cracking inspection program which was established in the early-1980's.

For plants not in service, the examination plan required preservice examinations and subsequent periodic inservice examinations during the first, fifth and tenth-year refueling outages. For plants already in service, examinations were required at the earliest possible refueling outage and repeated five and ten years later.

Thus far, ten EdF plants have been through the entire ten-year augmented inspection program cycle. There has been no evidence of crack growth, thus they have been permitted to return to the normal inspection frequency of once every ten years. In the case of eleven EdF plants, results of the first and fifth-year inspections showed no cracks greater than 5 mm deep. By agreement between EdF and the regulatory authority, the requirement to perform the tenth-year examination was not imposed for these plants and EdF does not anticipate performing additional nozzle examinations until the twentieth refueling outage. The remaining five plants are in various stages of their respective augmented examination program cycles and no crack extension has been reported.

Westinghouse also performed a fracture mechanics analysis which concluded that the minimum critical flaw depth is 1.5 inches at the governing region in the nozzle, the nozzle corner region. The minimum critical flaw depths at all other locations are substantially larger. Fatigue crack growth of these flaws are small at all locations, not exceeding 0.06" in the worst case (nozzle corner) over a ten year period and does not exceed 0.21" over the full service life.

As with underclad reheat cracking the Safety Evaluation Report NUREG-0011, Supplement 1 states that the NRC evaluations determined that the flaws found in the Sequoyah nozzles are acceptable and are within the acceptance standards established by Section XI of the ASME Code which provides

adequate assurance that the reactor vessel has sufficient margin against fracture. In addition the ASME Code Section XI and Regulatory Guide 1.150 requires the examination of a sizeable length of the nozzle bore using a demonstrated technique capable of detecting a near surface 2% of the nozzle thickness notch. In fact the ultrasonic technique to be applied to the nozzle bores of Sequoyah Units 1 and 2 was specifically developed in response to the need for a better examination of the base metal-to-cladding interface and near surface regions of the reactor vessel. This technique has been demonstrated to provide effective coverage extending down to 1 inch below the surface thereby encompassing the ASME Section XI, IWB-3500 acceptable flaw indication criteria and the fracture mechanics-determined critical flaw depth region. The ASME Code Section XI also requires the examination of the nozzle to safe end weld. A similar near surface ultrasonic technique as being applied to the nozzle bores is to be performed. With an extension of the examination volume for one nozzle to safe end weld by approximately 1 inch and the normal Section XI requirements over 50% of the total number of nozzle indications reported during the 1980 examination are encompassed. This sampling of indications includes all of the previous indications within the thinner section of the nozzles, two of the three previous indications having the maximum ultrasonic test responses, and the highest stress areas. This sampling will provide sufficient evidence as to whether the flaw indications found during the augmented 1980 examinations are within the acceptance standards of ASME Section XI, IWB-3500. Therefore there is no technical basis for augmented inservice inspection of prior indications classified as underclad cold cracking if the standard Section XI examinations supplemented with one scan extension and Regulatory Guide 1.150 requirements are implemented.

4.0 DETERMINATION OF UNREVIEWED SAFETY QUESTION

The information presented above, describing the proposed elimination of the current Technical Specification requirement, Surveillance Requirement 4.4.10.b, from the Sequoyah Units 1 and 2, forms the basis upon which the responses to the questions presented in Section 4 of the Checklist can be discussed.

4.1 Will the probability of an accident previously evaluated in the FSAR be increased?

Elimination of the current Technical Specification requirement (SR 4.4.10.b) to perform a 10-year supplemental inspection of the reactor vessel nozzles does not result in a design change to safety-related components in the reactor coolant pressure boundary (RCPB). Existing indications have been well-characterized and have been shown to have no effect on the structural integrity of the reactor vessel nozzle. Fracture mechanics evaluations (ASME Section XI, IWB-3600 and Appendix A) have demonstrated that even the worst case crack growth over the full plant life would remain well within the calculated critical flaw size. In addition, the Section XI/ R.G. 1.150 examination requirements would include over half of the previously found indications; thus, there will be sufficient basis to determine whether the indications are within the acceptance standards of ASME Section XI, IWB-3500. The structural integrity of the reactor vessel will be maintained. Therefore, the probability of an accident previously evaluated in the FSAR will not be increased.

4.2 Will the consequences of an accident previously evaluated in the FSAR be increased?

The integrity of the reactor vessel nozzles, as assumed during accident conditions (higher pressures, lower temperatures, etc...), will be maintained. Implementation of this change does not modify or degrade any assumptions made in the safety analyses, and therefore, the consequences of an accident previously evaluated in the FSAR will not be increased.

4.3 May the possibility of an accident which is different than any already evaluated in the FSAR be created?

The elimination of the requirement to perform supplemental 10-year inspections of the reactor vessel nozzles will not create an accident different than previously evaluated in the FSAR. This change does not involve the introduction of, or removal of, safety-related components. Fracture mechanics analyses and periodic inservice inspections[2] of reactor vessel nozzles, performed since the initial identification of the reheat cracking and underclad cold cracking phenomena, have shown that original design margins of safety will be maintained and that propagation of known underclad cold cracks will be minimal (within 4 mils over forty years of plant life), well within minimum critical flaw depths of one to one and a half inches. Therefore, the possibility of an accident which is different than any already evaluated in the FSAR will not be created.

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

The proposed change does not result in a modification to any safety-related components, or systems/components important to safety. Therefore, the probability of a malfunction of safety-related equipment previously evaluated in the FSAR will not be increased.

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

Fracture mechanics analyses have shown that crack growth will be minimal. The planned Section XI/R.G. 1.150 examinations are sufficient to verify that the previously recorded indications are within the acceptance standards of ASME Section XI, IWB-3500. Thus, the change in the examination requirements will not affect systems or components important to safety, and therefore, the consequences of such a malfunction of equipment will not be increased.

2: Twenty-six plants operated by EdF have been involved in the reactor nozzle bore underclad cold cracking inspection program. For operating plants, inspections take place during the fifth- and tenth-year refueling outages (for plants being placed in service for the first time, the periodic examinations also include the first refueling outage). Ten EdF plants have participated in the entire ten year augmented inspection program cycle, and there has been no evidence of crack growth. In eleven plants, results of the first and fifth-year inspections have shown no cracks greater that 5 mm deep. The regulatory authority agreed to remove the requirement for the ten-year examination on these plants and EdF does not anticipate additional examinations until the twentieth refueling outage.

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

The proposed change to the reactor vessel nozzle testing procedure does not add to existing equipment or modify any components of the RCPB. In addition, implementation of the change does not result in any new operational procedures or requirements for existing plant safety-related equipment. Therefore, the possibility of a malfunction of equipment important to safety different than already evaluated in the FSAR will not be created.

4.7 Will the margin of safety as defined in the BASES to any technical specifications be reduced?

The margin of safety of the reactor vessel is defined, in part, through adherence to the provisions if the ASME Code. The fracture mechanics analysis results, as will be adequately verified by the Section XI/R.G. 1.150 examinations, demonstrate that the reactor vessel integrity is maintained. Therefore, the margin of safety as defined in the Bases to any Technical Specification will not be reduced.

5.0 CONCLUSIONS

Available technical data pertaining to reheat cracking and underclad cold cracking in reactor vessel nozzles have been reviewed. Based on this review, Westinghouse has concluded that the augmented examinations of the Sequoyah Units 1 and 2 reactor vessel nozzles should not be required based on the following:

- Westinghouse, Framatome, and the Electricite de France have performed fracture mechanics analyses to assess the pressure vessel margin against fracture, and these analyses indicate no detrimental impact of these indications. Therefore, the original design margins will be maintained for continued operation.

- Since 1980 Electricite de France has performed periodic inservice inspections of reactor vessel nozzles with known underclad cold cracks, similar to Sequoyah Unit 1, and have concluded the cracks have not grown.

- Examinations performed in accordance with Section XI of the ASME Code as supplemented by Regulatory Guide 1.150 will provide a sufficient basis upon which to establish whether indications recorded during the 1980 examinations are within the acceptance standards of ASME Section XI, IWB-3500. Review of the 1980 results indicates that the examination volumes specified by Section XI with an extension of the scan volume for one nozzle to safe end weld include over one-half the total number of nozzle indications classified as underclad cold cracks and reported during the 1980 examination program.

In summary, it is concluded that the proposed deletion of the requirement to perform 10-year augmented inspections of the reactor vessel nozzles at Sequoyah Units 1 and 2 does not involve an unreviewed safety question as defined in 10 CFR 50.59

6.0 REFERENCES

1. NUREG-0011, Supp. No. 1, "Supplement No. 1 to the Safety Evaluation Report by the Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission in the Matter of Tennessee Valley Authority Sequoyah Nuclear Plant, Units 1 and 2, Docket Nos. 50-327 and 50-328".

2. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1977 Edition with 1978 Summer Addenda.

3. U.S. Nuclear Regulatory Commission Regulatory Guide 1.150, Revision 1, Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations, February 1983.

4. MT-MNA-001(93), "Technical Basis for Deletion of Augmented 10-Year Inservice Inspections of Sequoyah Unit 1 and 2 Reactor Pressure Vessel Nozzles for Underclad Cracking," Revision 2, January 4, 1993.