February 23, 2006

Mr. Karl W. Singer
Chief Nuclear Officer and
Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 1 - ISSUANCE OF AMENDMENT REGARDING STEAM GENERATOR TUBE INTEGRITY (TAC NO. MC8243)

Dear Mr. Singer:

The Commission has issued the enclosed Amendment No. 306 to Facility Operating License No. DPR-77 for the Sequoyah Nuclear Plant, Unit 1. This amendment is in response to your application dated August 31, 2005 (TS-05-07).

The amendment revises the Technical Specifications (TSs) associated with steam generator tube integrity consistent with Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." A notice of availability for this TS improvement using the consolidated line item improvement process was published in the *Federal Register* on May 6, 2005 (70 FR 24126).

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Douglas V. Pickett, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-327

Enclosures: 1. Amendment No. 306 to License No. DPR-77
2. Safety Evaluation

cc w/enclosures: See next page

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Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 306
License No. DPR-77

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated August 31, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (I) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 306, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented no later than 60 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael L. Marshall, Jr., Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: February 23, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 306

FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1-4	1-4
1-5	1-5
3/4 4-6	3/4 4-6
3/4 4-7	3/4 4-7
3/4 4-8	-
3/4 4-9	-
3/4 4-10	-
3/4 4-11	-
3/4 4-12	-
3/4 4-14	3/4 4-14
6-11	6-11
-	6-11a
-	6-11b
6-13a	6-13a
6-14	6-14

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 306 TO FACILITY OPERATING LICENSE NO. DPR-77

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNIT 1

DOCKET NO. 50-327

1.0 INTRODUCTION

By application dated August 31, 2005 (ADAMS Accession No. ML052500560), Tennessee Valley Authority (the licensee) proposed an amendment to the Technical Specifications (TSs) for Sequoyah Nuclear Plant (SQN) Unit 1. This amendment request is the culmination of Nuclear Regulatory Commission (NRC) and industry efforts since the mid-1990s to develop a programmatic, largely performance-based regulatory framework for ensuring steam generator (SG) tube integrity. In letters dated March 14 and September 9, 2003, October 7, 2004, and January 14 and April 14, 2005, the Technical Specification Task Force (TSTF) proposed requirements for SG tube integrity and changes to the SG program in the standard technical specifications (STS) (NUREGs 1430 - 1432) on behalf of the industry. This proposed change is designated TSTF-449, Revision 4.

The scope of the TS amendment request includes:

a. Revised Table of Contents
b. Revised TS definitions of “identified leakage” and “pressure boundary leakage”
c. Existing TS 3/4.4.5, “Steam Generators,” removed and new TS 3/4.4.5, “Steam Generator (SG) Tube Integrity” inserted
d. Revised TS 3/4.6.2, “Operational Leakage”
e. Addition of TS 6.8.4.k, “Steam Generator (SG) Program”
f. Addition of TS 6.9.1.16, “Steam Generator Tube Inspection Report”
g. Revised TS Bases sections

The proposed new TS 3/4.4.5, “Steam Generator (SG) Tube Integrity,” in conjunction with the proposed addition of administrative TS 6.8.4.k, “Steam Generator (SG) Program,” would establish a new programmatic, largely performance-based framework for ensuring SG tube integrity. Proposed TS Bases B 3/4.4.5 documents the licensee’s bases for this framework. Proposed TS 3/4.4.5 would establish new limiting conditions for operation (LCOs) related to SG tube integrity; namely, (1) SG tube integrity shall be maintained, and (2) all SG tubes satisfying the tube repair criteria (i.e., tubes with measured flaw sizes exceeding the tube repair criteria) shall be plugged in accordance with the SG Program. TS 3/4.4.5 would include surveillance requirements (SRs) to verify that the above LCOs are met in accordance with the SG Program.

Proposed administrative TS 6.8.4.k, “Steam Generator (SG) Program,” is added to the TSs to complement the LCO. This added TS would require establishing and implementing a program that ensures that SG tube integrity is maintained. Tube integrity is defined in the proposed TS in terms of specified performance criteria for structural and leakage integrity. TS 6.8.4.k would also provide for monitoring the condition of the tubes relative to these performance criteria during each SG tube inspection and for ensuring that tube integrity is maintained between scheduled inspections of the SG tubes. TS 6.8.4.k would retain the currently specified tube repair limit(s).

The proposed addition of TS 6.9.1.16, “Steam Generator (SG) Tube Inspection Report,” revises the existing requirements for, and the contents of, the SG tube inspection report consistent with the proposed revisions to TS 6.8.4.k. The current requirement for a 12-month report would be changed to a 180-day report.

The proposed amendment revises the TS definition of identified leakage and pressure boundary leakage. Currently, the TS definitions refer to “steam generator tube leakage.” The more appropriate term, “primary to secondary leakage,” is used in the new TS definitions of identified leakage and pressure boundary leakage.

The proposed amendment includes proposed revisions to TS 3/4.6.2, “[Reactor Coolant System (RCS)] Operational Leakage,” and its bases to clarify the requirements related to primary-to-secondary leakage.

Finally, the TS Bases would be revised to reflect the proposed TS changes.

2.0 REGULATORY EVALUATION

2.1 Current Licensing Basis / SG Tube Integrity

The SG tubes in pressurized-water reactors (PWRs) have a number of important safety functions. These tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system and are relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damage severe accidents.

Title 10 of the Code of Federal Regulations (10 CFR) establishes the fundamental regulatory requirements with respect to the integrity of the SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 states that the RCPB shall have “an extremely low probability of abnormal leakage . . . and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards possible" (GDC 30), and shall be designed to permit "periodic inspection and testing . . . to assess . . . structural and leak tight integrity" (GDC 32). To this end, 10 CFR 50.55a specifies that components that are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). Section 50.55a further requires, in part, that throughout the service life of a PWR facility, ASME

Code Class 1 components meet the requirements, except design and access provisions and preservice examination requirements, in Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code.

In the 1970s, Section XI requirements pertaining to ISI of SG tubing were augmented by additional SG tube SRs in the TSs. Paragraph (b)(2)(iii) of 10 CFR, 50.55a, states that where TS SRs for SGs differ from those in Article IWB-2000 of Section XI of the ASME Code, the ISI program shall be governed by the TSs.

The existing plant TSs include LCOs and accompanying SRs and action statements pertaining to the integrity of the SG tubing. SG operability in accordance with the SG tube surveillance program is necessary to satisfy the LCOs governing RCS loop operability, as stated in the accompanying TS Bases. The LCO governing RCS Operational LEAKAGE includes limits on allowable primary-to-secondary LEAKAGE through the SG tubing. Accompanying SRs require verification that RCS operational LEAKAGE is within limits every 72 hours by an RCS water inventory balance and that SG tube integrity is in accordance with the SG tube surveillance program. The SG tube surveillance program requirements are contained in the administrative TSs. These administrative TSs state that the SGs are to be determined OPERABLE after the actions required by the surveillance program are completed. Under the plant TS SG surveillance program requirements, licensees are required to monitor the condition of the SG tubing and to perform repairs, as necessary. Specifically, licensees are required by the plant TSs to perform periodic ISIs and to remove from service, by plugging, all tubes found to contain flaws with sizes exceeding the acceptance limit, termed "plugging limit" (old terminology) or "tube repair criteria" (new terminology). The frequency and scope of the inspection and the tube repair limits are specified in the plant TSs.

The tube repair limits in the TSs were developed with the intent of ensuring that degraded tubes (1) maintain factors of safety against gross rupture consistent with the plant design basis (i.e., consistent with the stress limits of the ASME Code, Section III) and (2) maintain leakage integrity consistent with the plant licensing basis while, at the same time, allowing for potential flaw size measurement error and flaw growth between SG inspections.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design basis accidents (DBAs) such as a SG tube rupture (SGTR) and main steam line break (MSLB). These analyses consider the primary-to-secondary leakage through the tubing which may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of 10 CFR Part 100 for offsite doses, GDC-19 criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits).

2.2 10 CFR 50.36

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TSs. In doing so, the Commission emphasized those matters related to the preventing of accidents and mitigating their consequences. As recorded in the Statements of Consideration, Technical Specifications for Facility Licenses: Safety Analysis Reports (33 FR 18610, December 17, 1968), the Commission noted that applicants are expected to incorporate into their TSs those items that are directly related to maintaining the integrity of the physical barriers

designed to contain radioactivity. Pursuant to 10 CFR 50.36, TSs are required to include items in five specific categories related to station operation. Specifically, those categories include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCO); (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS. The licensee's application contains proposed LCOs, SRs and administrative controls involving SG integrity, an important element of the physical barriers designed to contain radioactivity.

Additionally, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether an LCO is required to be included in the TS for a certain item. These criteria are as follows:

1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
2. A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that assumes either the failure of or presents a challenge to the integrity of a fission product barrier.
3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The NRC staff has reviewed the proposed changes to ensure that these changes conform with 10 CFR 50.36 as discussed herein.

2.3 Background - Technical Specification Amendment Request

The current TS requirements for inspection and repair of SG tubing date to the mid-1970s and define a prescriptive approach for ensuring tube integrity. This prescriptive approach involves inspection of the tubing at specified intervals, implementation of specified tube inspection sampling plans, and repair or removal from service by plugging all tubes found by inspection to contain flaws in excess of specified flaw repair criteria. However, as evidenced by operating experience, the prescriptive approach defined in the TSs is not sufficient in-and-of-itself to ensure that tube integrity is maintained. For example, in cases of low to moderate levels of degradation, the TSs require that only 3 to 21 percent of the tubes be inspected, irrespective of whether the inspection results indicate that additional tubes may need to be inspected to reasonably ensure that tubes with flaws that may exceed the tube repair criteria, or that may impair tube integrity, are detected. In addition, the TSs (and ASME Code, Section XI) do not explicitly address the inspection methods to be employed for different tube degradation mechanisms or tube locations, nor are the specific objectives to be fulfilled by the selected methods explicitly defined. Also, incremental flaw growth between inspections can, in many instances, exceed what is allowed in the specified tube repair criteria. In such cases, the specified inspection frequencies may not ensure reinspection of a tube before its integrity is impaired. In short, the current TS SRs do not require licensees to actively manage their SG surveillance programs so as to provide reasonable assurance that tube integrity is maintained.

In view of the shortcomings of the current TS requirements, licensees experiencing significant degradation problems have frequently found it necessary to implement measures beyond minimum TS requirements to ensure that adequate tube integrity is being maintained. Until the 1990s, these measures tended to be ad hoc. By letter dated December 16, 1997 (Reference 1), the Nuclear Energy Institute (NEI) provided NRC with a copy of NEI 97-06 (Original), "Steam Generator Program Guidelines," and informed the NRC of the following formal industry position.

> Each licensee will evaluate its existing steam generator program and, where necessary, revise and strengthen program attributes to meet the intent of the guidance provided in NEI 97-06, "Steam Generator Program Guidelines," no later than the first refueling outage starting after January 1, 1999.

The stated objectives of this initiative were to have a clear commitment from utility executives to follow industry SG related guidelines developed through Electric Power Research Institute (EPRI) to assure a unified industry approach to emerging SG issues and to apply tube integrity performance criteria in conjunction with the performance-based philosophy of the maintenance rule, 10 CFR 50.65. Reference 2 is the most recent update to NEI 97-06 available to the NRC staff. NEI 97-06 provides general, high-level guidelines for a programmatic, performance-based approach to ensuring SG tube integrity. NEI 97-06 references a number of detailed EPRI guideline documents for programmatic details. Subsequently, the NRC staff had extensive interaction with the industry to resolve NRC staff concerns with this industry initiative and to identify needed changes to the plant TSs to ensure that tube integrity is maintained (Reference 3).

Ultimately, in consideration of the performance-based objective of this initiative, the NRC staff determined it was not necessary for the NRC staff to formally review or endorse the NEI 97-06 guidelines or the EPRI guideline documents referenced by NEI 97-06. The subject application for changes to the TS is programmatically consistent with the industry's NEI 97-06 initiative. As discussed in this safety evaluation, these changes will ensure that a SG program that provides reasonable assurance that SG tube integrity will be maintained will be implemented.

3.0 EVALUATION

3.1 TS 3/4.4.5, "Steam Generator (SG) Tube Integrity"

The current TSs established an operability requirement for the SG tubing; namely, the tubes shall be determined OPERABLE after completion of the actions defined in the SG tube surveillance program (TS 6.8.4.k). In addition, this surveillance program (and SG operability) is directly invoked by TS 3/4.6.2, which contains the LCO relating to RCS leakage. However, these specifications do not directly require that tube integrity be maintained. Instead, they require implementation of an SG tube surveillance program, which is assumed to ensure tube integrity, but, as discussed above, may not depending on the circumstances of degradation at a plant.

To address this shortcoming, the Sequoyah, Unit 1 TS amendment package includes a proposed new specification, TS 3/4.4.5, "Steam Generator (SG) Tube Integrity," which includes a new LCO requirement and accompanying conditions, required actions, completion times, and SRs. The new LCO is applicable in MODES 1, 2, 3, and 4 and requires: 1) SG tube integrity shall be maintained, AND 2) all SG tubes satisfying the tube repair criteria shall be plugged in

accordance with the Steam Generator Program (specified in the proposed TS 6.8.4.k). This LCO supplements the LCO in TS 3/4.6.2 to directly make tube integrity an operating restriction. This is consistent with Criterion 2 of 10 CFR 50.36(c)(2)(ii), since the assumption of tube integrity as an initial condition is implicit in DBA analyses (with the exception of analysis of a design-basis SGTR where one tube is assumed not to have structural integrity) and is acceptable to the NRC staff.

Proposed SR 3/4.4.5.0 would require that SG tube integrity be verified in accordance with the Steam Generator Program, which is described in proposed revisions to TS 6.8.4.k. The required frequency for this surveillance would also be in accordance with the SG Program, thus meeting the requirements of 10 CFR 50.36(c)(3). The revised TS 6.8.4.k would define tube integrity in terms of satisfying tube integrity performance criteria for tube structural integrity and leakage integrity as specified therein. SR 3/4.4.5.0 would replace the existing surveillance requirement (SR 3/4.6.2.2) in the RCS operational leakage specification (TS 3/4.6.2), which provides that tube integrity is verified in accordance with the SG surveillance program as provided in the current TS 6.8.4.k. The proposed SR improves upon the current SR in that it refers to a program that is directly focused on maintaining tube integrity rather than on implementing a prescriptive surveillance program which, as discussed above, may not be sufficient to ensure tube integrity is maintained. Proposed SR 3/4.4.5.1 would require verification that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the SG Program. The tube repair criteria are contained in the SG Program. The required frequency for SR 3/4.4.5.1 is prior to entering MODE 4 following a SG tube inspection. The NRC staff concludes that SR 3/4.4.5.0 and SR 3/4.4.5.1 are sufficient to determine whether the proposed LCO is met, meet the requirements of 10 CFR 50.36(c)(3), and are acceptable.

The licensee has proposed conditions, required actions, and completion times for the new LCO 3/4.4.5 as shown in Table 1. The proposed TS 3/4.4.5 allows separate condition entry for each SG tube.

Table 1 - TS 3/4.4.5 ACTIONS[1]

Condition	Required Action	Completion Time
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next inspection. AND A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection.
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5	6 hours 36 hours

Should SG tube integrity be found by the SG Program not to be maintained, Required Actions B.1 and B.2 would require that the plant be in MODE 3 within 6 hours and MODE 5 within 36 hours, respectively. These required actions and completion times are consistent with (1) the general requirements in TS 3.0.3 for failing to meet an LCO and (2) the requirements of TS 3/4.6.2 when the LCO on primary to secondary leakage rate is not met. The NRC staff concludes that these required actions and completion times provide adequate remedial measures should SG tube integrity be found not to be maintained and are acceptable to the NRC staff.

Condition A of proposed TS 3/4.4.5 addresses the condition where one or more tubes satisfying the tube repair criteria are inadvertently not plugged in accordance with the SG Program. Under Required Action A.1, the licensee would be required to verify within 7 days that tube integrity of the affected tubes is maintained until the next inspection. The accompanying Bases state that the tube integrity determination would be based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next inspection. The NRC staff notes that details of how this assessment would be performed are not included in proposed TS 3/4.4.5 or 6.8.4.k. The NRC staff finds this to be consistent with having performance-based requirements, finds that the performance criteria (i.e., performance objectives) for assessing tube integrity are clearly defined (in TS 6.8.4.k), and finds that it is appropriate that the licensee has the flexibility to determine how best to perform this assessment based on what information is and is not available concerning the circumstances of the subject flaw. The proposed 7 days allowed to complete the assessment ensures that the risk increment associated with operating with tubes in this condition will be very small. Should

[1] Table provided in format of Standard TSs (NUREG-1431). TSs for Sequoyah, Unit 1 have not be converted to this format but proposed changes are consistent with those in TSTF-449. This Safety Evaluation (SE) refers to the Conditions described in the table.

the assessment reveal that tube integrity cannot be maintained until the next scheduled inspection or if the assessment is not completed in 7 days, Condition B applies, leading to Required Actions B.1 and B.2, which are evaluated above. Finally, if Required Action A.1 successfully verifies that tube integrity is being maintained until the next inspection, Required Action A.2 would require that the subject tube be plugged in accordance with the SG Program prior to entering MODE 4 after the next refueling outage or SG inspection. Based on the above, the NRC staff concludes that the proposed LCO and accompanying ACTIONS related to failure to plug a tube that satisfies the tube repair criteria to be acceptable.

The licensee has proposed additional editorial changes to the TS (to reflect plant-specific format) and TS Bases supporting the proposed new TS 3/4.4.5. Although the TS Bases are controlled under the auspices of 10 CFR 50.59 and TS 6.8.4.j, TS Bases Control Program, the NRC staff finds the proposed changes to the proposed TS 3/4.4.5 Bases to be acceptable.

3.2 Steam Generator Operability

Changes described in TSTF-449 for the Bases section for TS 3/4.4.1, “Reactor Coolant Loops and Coolant Circulation,” are not required for Sequoyah, Unit 1. Although not directly related to a proposed change for Sequoyah, Unit 1, the NRC staff reiterates that the operability of a SG will be defined under the definition of OPERABLE - OPERABILITY defined in TS 1.19 and stated below:

> A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, a normal and an emergency electrical power source, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

3.3 Proposed Administrative TS 6.8.4.k, ”Steam Generator Program”

The proposed Administrative TS 6.8.4.k, ”Steam Generator Program” replaces the requirements in the current TS 3/4.4.5, “Steam Generators.” The current TS 3/4.4.5 defines a prescriptive strategy for ensuring tube integrity consisting of tube inspections performed at specified intervals, with specified inspection scope (tube inspection sample sizes), and with a specified tube acceptance limit for degraded tubing, termed “tube repair criterion,” beyond which the affected tubes must be plugged. The proposed TS 6.8.4.k incorporates a largely performance-based strategy for ensuring tube integrity, requiring that a SG Program be established and implemented to ensure tube integrity is maintained. The proposed specification contains only a few details concerning how this is to be accomplished, the intent being that the licensee will have the flexibility to determine the specific strategy to be employed to satisfy the required objective of maintaining tube integrity. However, as evaluated below, the NRC staff concludes that proposed TS 6.8.4.k provides reasonable assurance that the SG Program will maintain tube integrity.

The proposed BASES for TS 3/4.4.5, “Steam Generator (SG) Tube Integrity,” state that NEI 97-06 and its referenced EPRI guideline documents will be used to establish the content of the SG Program. The guidelines are industry-controlled documents and licensee SG programs

may deviate from these guidelines. Except as may be specifically invoked by the TSs, the NRC staff's evaluation herein takes no credit for any of the specifics in the guidelines.

3.3.1 Performance Criteria for SG Tube Integrity

Proposed TS 6.8.4.k would require that SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage as specified therein.

The NRC staff's criteria for evaluating the acceptability of these performance criteria are that meeting these criteria are sufficient to ensure that tube integrity is within the plant licensing basis and that meeting these criteria, in conjunction with implementation of the SG Program, ensure no significant increase in risk. These performance criteria must also be evaluated in the context of the overall SG Program such that if the performance criteria are inadvertently exceeded, the consequences will be tolerable before the situation is identified and corrected. In addition, the performance criteria must be expressed in terms of parameters that are measurable, directly or indirectly.

3.3.1.1 Structural Integrity Criterion

The proposed structural integrity criterion is as follows:

> All inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification) and design basis accidents. This includes maintaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to differential pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

The NRC staff has evaluated this proposed criterion for consistency with the safety factors embodied in the current licensing basis, specifically, the safety factors embodied in the TS tube repair criterion. The tube repair criterion typically specified in plant TSs is 40 percent of the initial tube wall thickness. This criterion is typically applicable to all tubing flaws found by inspection, except for certain flaw types at certain locations for which less restrictive repair criterion may be applicable (as specified in the TSs) and for certain sleeve repairs for which a more restrictive tube repair criterion may be specified. For Sequoyah, Unit 1, the 40 percent tube repair criterion is the only such criterion and is applicable to all flaw types at all tube locations.

In 1976 the NRC staff prepared RG 1.121 (Draft), “Basis for Plugging Degraded PWR Steam Generator Tubes,” (Reference 4) describing a technical basis for the development of tube repair criteria. This draft RG was issued for public comment, but was never finalized. Although not finalized, the RG is generally cited in licensee and industry documentation as the bases for the TS tube repair criterion in plant TSs. The draft RG includes the following with respect to safety factors:

a. Degraded tubing should retain a factor of safety against burst of not less than three under normal operating conditions.

b. Degraded tubing should not be stressed beyond the elastic range of the tube material during the full range of normal reactor operation. The draft RG also states that loadings associated with normal plant conditions, including startup, operation in the power range, hot standby, and cooldown, as well as all anticipated transients (e.g., loss of electrical load, loss of off-site power) that are included in the design specifications for the plant, should not produce a primary membrane stress in excess of the yield stress of the tube material at operating temperature.

c. Degraded tubes should maintain a margin of safety against tube failure under postulated accidents consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the ASME Code. Note, NB-3225 specifies that the rules in Appendix F of Section III may be used for evaluating these loadings.

The “safety factor of three” criterion stems from Section III of the ASME Code which, in part, limits primary membrane stress under design conditions to one third of ultimate strength. The proposed structural integrity criterion would limit application of the “safety factor of three” criterion to those pressure loadings existing during normal full power, steady state operating conditions. Differential pressures under this condition are plant specific, ranging from 1250 psi to 1500 psi (Reference 5). However, differential pressure loadings can be considerably higher during normal operating transients, ranging to between 1600 psi and 2150 psi during plant heatup and cooldown (Reference 5). Given a factor of safety equal to three under normal full-power conditions, the factor of safety during heatups and cooldowns can be as low as about two. The industry stated in a white paper (Reference 5) that it was not the intent of the 40 percent depth-based tube repair criterion to ensure a factor of safety of three for operating transients such as heatups and cooldowns. The industry stated that maintaining a safety factor of three for such transients would lead to a tube repair criterion less than the standard 40 percent criterion for many plants. The NRC staff has independently performed calculations that support the industry’s contention that applying the “safety factor of three” criterion to the full range of normal operating conditions would lead to a tube repair criterion more restrictive than the 40 percent criterion that the NRC staff has accepted since the 1970s. The NRC staff concludes that the “safety factor of three” criterion for application to normal full power, steady state pressure differentials, as proposed by the licensee and the industry, is consistent with the safety margins implicit in existing TS tube repair criteria and, thus, is consistent with the current licensing basis.

Item b above from draft RG 1.121 is often referred to as the “no yield” criterion. The purpose of this criterion is to prevent permanent deformation of the tube to assure that degradation of the

tube will not occur due to mechanical effects of the service condition. This is consistent with the ASME Code, Section III, stress limits, which serve to limit primary membrane stress to less than yield. The proposed structural integrity criteria do not include this “no yield” criterion. The industry states in its white paper (Reference 5) that, if a tube satisfies the “safety factor of three” criterion at full power operating pressure differentials, the tube will generally satisfy the “no yield” criterion for the operating transient (e.g., heatup and cooldown) pressure differentials. The white paper acknowledges that this may not be true for all plant-specific conditions and material properties. For this reason, NEI 97-06, Rev. 1, and the EPRI Steam Generator Integrity Assessment Guidelines state that, in addition to meeting the safety factor of three for normal steady state operation, the integrity evaluation shall verify that the primary pressure stresses do not exceed the yield strength for the full range of normal operating conditions. The white paper, which has been incorporated as part of the EPRI Steam Generator Integrity Assessment Guidelines, recommends that this be demonstrated for each plant using plant specific conditions and material properties.

The NRC staff concurs that the “no yield” criterion need not be specifically spelled out in the TS definition of the structural integrity criterion. The NRC staff finds that the appropriate focus of the TS criteria should be on preventing burst. The NRC staff calculations confirm that the proposed “safety factor of three” criterion bounds or comes close to bounding the “no yield” criterion for most of the cases investigated. This is not absolute, however. For once-through SGs (OTSGs), the NRC staff noted a case where elastic hoop stress in a uniformly thinned tube could exceed the yield strength by 20-percent under heatup and cooldown conditions and still satisfy the “safety factor of three” criterion against burst under normal steady state, full power operating conditions. Such a tube would still retain a factor of safety of two against burst under heatup and cooldown conditions. The amount of plastic strain induced would be limited to between 1 and 2 percent based on typical strain hardening characteristics of the material. This is quite small compared to cold working associated with fabrication of tube u-bends and tube expansions. Operating experience shows that this level of plastic strain (i.e., permanent strain caused by exceeding the yield stress) has not adversely affected the stress corrosion cracking resistance of OTSG tubing relative to that expected for nonplastically strained tubing. Thus, the NRC staff concludes that the “safety factor of three” criterion is sufficient to limit plastic strains to values that will not contribute significantly to degradation of the tubing and that the “no yield’ criterion need not be specifically spelled out in the structural integrity performance criterion.

The proposed safety factor of 1.4 against burst applied to design basis primary-to-secondary pressure differentials derives from the 0.7 times ultimate strength limit for primary membrane stress in the ASME Code, Appendix F, F-1331.1(a). This criterion is consistent with the stress limit criterion used to develop the standard 40 percent tube repair criterion in the TSs and with the safety factor criteria used in the derivation of alternate tube repair criteria in plant TSs, such as the voltage-based criterion for outer-diameter stress corrosion cracking. Thus, the criterion is consistent with the current licensing basis and is acceptable.

Apart from differential pressure loadings, other types of loads may also contribute to burst. Examples of such loads include bending moments on the tubes due to flow induced vibration, earthquake, and loss-of-coolant accident (LOCA) rarefaction waves. For OTSGs, axial loads are induced in the tubes due to pressure loadings acting on the SG shell and tube sheets and due to differential thermal expansion between the tubes and the SG shell. Such nonpressure loads generally produce negligible primary stress during normal operating conditions from the standpoint of influencing burst pressure. In general, such nonpressure loads may be more

significant under certain accident loadings depending on SG design, flaw location, and flaw orientation. Such nonpressure sources of primary stress under accident conditions were explicitly considered in the development of the 40 percent tube repair criterion relative to ASME Code, Appendix F, stress limits.

The proposed structural criterion requires that, apart from the safety-factor requirements applying to pressure loads, additional loads associated with DBAs, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine whether these loads contribute significantly to burst or collapse. The NRC staff notes that examples of such additional loads include bending moments during LOCA, MSLB, or safe shutdown earthquake (SSE) and axial, differential thermal loads. “Combination of accidents” refers to the fact that the design and licensing basis for many plants are that DBAs, such as LOCA and MSLB, are assumed to occur concurrently with SSE. Whereas “burst” is the failure mode of interest where primary-to-secondary pressure loads are dominant, “collapse” is a potential limiting failure mode (although an unlikely one, according to industry, based on a recent study (Reference 6)) for loads other than pressure loads. “Collapse” refers to the condition where the tube is not capable of resisting further applied loading without unlimited displacement. Although the occurrence of a collapsed tube or tubes would not necessarily lead to perforation of the tube wall, the consequences of tube collapse have not been analyzed and, thus, the NRC staff finds it both appropriate and conservative to ensure there is margin relative to such a condition.

Where nonpressure loads are determined to significantly contribute to burst or collapse, the proposed structural criterion requires that such loads be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 safety factor on axial secondary loads. The 1.2 safety factor for combined primary loads was derived from the ratio of burst or collapse load divided by allowable load from ASME Code for faulted conditions. Burst or collapse load was assumed to be equal to the material flow stress, assuming Code minimum yield and ultimate strength values and a flow stress coefficient of 0.5. Allowable load was determined from ASME Code, Section III, Appendix F, F-1331.3.a, which defines an allowable primary membrane plus bending load for service level d (faulted) conditions. The NRC staff finds this 1.2 safety factor acceptable. The proposed 1.0 safety factor for axial secondary loads goes beyond what is required by the design basis in Section III of the ASME Code, since Section III assumes that a one-time application of such a load cannot lead to burst or collapse. However, this is not necessarily the case for tubes with circumferential cracks. The proposed safety factor criterion of 1.0 is conservative for loads that behave as secondary since it ignores the load relaxation effect associated with axial yielding before tube severance (burst) occurs.

Apart from being consistent with the current licensing basis, NRC risk studies have indicated that maintaining the performance criteria safety factors is important to avoiding undue risk, particularly risk associated with severe accident scenarios involving a fully pressurized primary system and depressurized secondary system and where the tubes may heat to temperatures well above design basis values, significantly reducing the strength of the tubes (Reference 7).

Based on the above, the NRC staff finds that the proposed structural performance criterion is consistent with the margins of safety embodied in existing plant licensing bases. Exceeding this criterion is not likely to lead to consequences that are intolerable provided that such a condition is infrequent and that, if it occurs, it is promptly detected and corrected so as to ensure that risk is limited. Even if a tube should degrade to the point of rupture under normal operating

conditions, such an occurrence is an analyzed condition with reasonable assurance that the radiological consequences will be acceptable. Finally, the structural performance criterion is expressed in terms of parameters that are measurable. Specifically, structural margins can be directly demonstrated through in situ pressure testing or can be calculated from burst prediction models using as input flaw size measurements obtained by inspection. Thus, the NRC staff finds the proposed structural performance criterion to be acceptable.

3.3.1.2 Accident Induced Leakage Criterion

The proposed accident induced leak rate criterion is as follows:

> The primary-to-secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

This performance criterion for accident induced leak rate is consistent with leak rates assumed in the licensing basis accident analyses for purposes of demonstrating that the consequences of DBAs meet the limits in 10 CFR Part 100 for offsite doses, GDC 19 for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits). This criterion does not apply to design basis SGTR accidents for which leakage corresponding to a postulated double ended rupture of a tube is assumed in the analysis. The proposed criterion ensures that from the standpoint of accident-induced leakage the plant will be operated within its analyzed condition and is acceptable.

For certain severe accident sequences involving high primary side pressure and a depressurized secondary system (“high-dry” condition), primary-to-secondary leakage may lead to more heating of the leaking tube than would be the case were it not leaking, thus increasing the potential for failure of that tube and a consequent large early release. The proposed 1.0 gpm limit on total leakage from each SG during DBAs (other than an SGTR) ensures that the potential for induced leakage during severe accidents will be maintained at a level that will not increase risk.

It is not likely that exceeding this criterion will lead to intolerable consequences provided that such an occurrence is infrequent and that such an occurrence, if it occurs, is promptly detected and corrected so as to ensure that risk is minimized. It should be noted that the criterion applies to leakage that could be induced by an accident in the unlikely event that such an accident occurs. Finally, the accident leakage performance criterion is expressed in terms of parameters that are measurable, both directly and indirectly. Specifically, structural margins can be directly demonstrated through in situ pressure testing or can be calculated using leakage prediction models using flaw size measurements obtained by ISI as input.

Based on the foregoing, the NRC staff finds the proposed accident leakage performance criterion to be acceptable.

3.3.1.3 Operational Leakage Criterion

Proposed TS 6.8.4.k states that the operational leakage performance criterion is specified in LCO 3/4.6.2, “RCS operational leakage.” Given the TS LCO limit, a separate performance criterion for operational leakage is unnecessary for ensuring prompt shutdown should the limit be exceeded. However, operational leakage is an indicator of tube integrity performance, though not a direct indicator. It is the only indicator that can be monitored while the plant is operating. Maintaining leakage to within the limit provides added assurance that the structural and accident leakage performance criteria are being met. Thus, the NRC staff believes that inclusion of the TS leakage limit among the set of tube integrity performance criteria is appropriate from the standpoint of completeness and is, therefore, acceptable.

3.3.2 Condition Monitoring Assessment

Proposed TS 6.8.4.k would require that the SG Program include provisions for condition monitoring assessments as follows:

> Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

The NRC staff finds that the proposed requirement for condition monitoring assessments addresses an essential element of any performance-based strategy, namely, the need to monitor performance relative to the performance criteria. Confirmation that the tube integrity criteria are met would confirm that the overall programmatic goal of maintaining tube integrity has been met to that point in time. However, failure to meet the tube integrity criteria would be indicative of potential shortcomings in the effectiveness of the licensee’s SG Program and the need for corrective actions relative to the program to ensure that tube integrity is maintained in the future. Failure to meet either the structural or accident induced leakage performance criterion would be reportable pursuant to 10 CFR 50.72 and 50.73 in accordance with guidelines in Reference 8. In addition, the NRC Regional Office would followup on such an occurrence as appropriate, consistent with the NRC Reactor Oversight Program (ROP) (Reference 10) and the risk significance of the occurrence.

TS 6.8.4.k would require that condition monitoring be performed at each ISI of the tubing. The NRC staff’s evaluation of the proposed frequency of ISI is addressed in section 3.3.3 of this safety evaluation.

3.3.3 Inservice Inspection

The proposed TS 6.8.4.k would require that the SG Program include periodic tube inspections. This proposal includes a new performance-based requirement that the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next inspection. This is a performance-based requirement that complements the requirement for condition monitoring from the standpoint of ensuring tube integrity is maintained. The requirement for condition monitoring is backward looking, in that it is intended to confirm that tube integrity has been maintained up to the time the assessment is performed. The ISI requirement, by contrast, is forward looking. It is intended to ensure that tube inspections in conjunction with plugging of tubes are performed such as to ensure that the performance criteria will continue to be met at the next SG inspection. This would be followed again by condition monitoring at the next SG inspection to confirm that the performance criteria were in fact met.

With respect to scope and methods of inspection, the proposed specification would also require that the number and portions of tubes inspected and method of inspection be performed with the objective of detecting flaws of any type (for example, volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criterion. Furthermore, an assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

The NRC staff finds that this proposal concerning the scope and methods of inspection includes a number of improvements relative to the current specification. The current specification requires tube inspections to be conducted from the point of entry on the hot leg side completely around the u-bend to the top support plate on the cold leg side. Thus, the current TS does not require inspection of tubing on the cold leg side up to the uppermost support plate elevation. Operating experience demonstrates that the entire length of tubing is subject to various forms of degradation. The proposed specification addresses this issue by requiring cold leg as well as hot leg inspections. Also, the proposed requirement clarifies the licensee's obligation under existing TSs and 10 CFR Part 50, Appendix B, to employ inspection methods capable of detecting flaws of any type that the licensee believes may potentially be present anywhere along the length of the tube based on a degradation assessment.

The proposed specification specifically excludes the tubesheet welds and the tube ends beyond the welds from the inspection requirements therein. The NRC staff finds this to be consistent with current actual practice and to be acceptable. The tube ends beyond the tube-to-tubesheet welds are not part of the primary pressure boundary.

The proposed specification would replace current specific requirements pertaining to the number of tubes to be inspected at each inspection, in part, with a requirement that is performance-based; that is, the number and portions of tubes inspected (in conjunction with other elements of inspection) shall be such as to ensure that tube integrity is maintained until the next inspection. The current minimum tube sampling requirement for an SG inspection is 3 percent of the SG tubing at the plant. The purpose of this initial sample is to determine whether active degradation is present and whether there is a need to perform additional inspection

sampling. Actual industry practice, consistent with NEI 97-06 and the EPRI Examination Guidelines, Rev. 6, typically involves initial inspection samples of at least 20 percent. If moderate numbers of tubes (i.e., category C-2 as defined in the current TSs) are found to contain flaws, the current TSs require that an additional 6 to 18 percent of the tubes be inspected. In many cases this requirement is very nonconservative since no consideration is given to whether uninspected tubes may contain flaws that could challenge the tube integrity performance criteria prior to the next inspection. Current industry practice and the industry guidelines involve substantially higher levels of sampling under these circumstances. This practice has been motivated by a desire to minimize forced outages as well as to ensure tube integrity. The NRC staff finds, therefore, that current TS sampling requirements do not drive actual sampling programs in the field for plants with low to moderate levels of tube degradation, and that for moderate levels of tube degradation the current TS requirements do not ensure adequate levels of sampling to ensure tube integrity will be maintained. The proposed specification addresses this shortcoming by requiring that inspection scope be consistent with the overall performance objective that tube integrity be maintained until the next SG inspection.

For SGs with high levels of degradation (i.e., category C-3 as defined in current TS), the current TS requires that the inspections be expanded to include 100 percent of the tubes in the affected SG. This requirement is conservative in cases where the active degradation is confined to specific groups of tubes in the SG. This requirement does drive actual sampling programs in the field since industry guidelines would permit 100 percent sampling to be confined to those portions of the SG bounding the region where the degradation has been found to be active. The proposed specification would give licensees the flexibility to implement less than 100-percent inspection of the SG in these cases provided it is consistent with the performance-based objective of ensuring that tube integrity is maintained until the next SG inspection.

Overall, the NRC staff concludes that the proposed specification ensures that the licensee will implement inspection scopes consistent with the overall objective that tube integrity is maintained. To meet this requirement, it will be necessary to inspect tubes that may contain flaws that may challenge the tube integrity performance criteria prior to the next inspection. The proposed specification gives the licensee the flexibility to define an inspection scope that ensures that this objective is met while avoiding any unnecessary inspections.

With respect to frequency of inspection, the current specification requires that SG inspections be performed every 24 calendar months. This frequency may be extended to once every 40 calendar months if the previous two inspections revealed only low-level degradation (i.e., category C-1 results as defined in the TS). The inspection frequency is required to revert from the 40 calendar months to 20 calendar months if an extensive level of degradation (i.e., category C-3 results as defined in the TS) is observed during the most recent inspection. Except in cases where extensive degradation (i.e., category C-3) is found in any SG, SGs may be inspected on a rotating basis at each inspection. Thus, for 4-loop plants performing SG inspections at 24-month intervals, intervals for individual SGs may range to 96 months. Similarly, for 4-loop plants performing SG inspections at 40-month intervals, intervals for individual SGs may range to 160 months. However, these prescriptive requirements bear no direct relationship to the overall objective of ensuring tube integrity is maintained. These requirements apply irrespective of the flaw detection and sizing performance of the inspection methods utilized and the rate at which flaws may be growing in the subject SGs. These requirements do not ensure that flawed tubing remaining in service following a SG tube

inspection and the incremental flaw growth that may take place prior to the next inspection are with within the allowances provided for by the TS tube repair limit or that tube integrity will be maintained prior to the next inspection.

Plants operating with their originally installed SGs have typically inspected each SG at each refueling outage, which typically occur at intervals of less than 24 calendar months. The vast majority of these SGs contained alloy 600 mill annealed (MA) tubing, which quickly became moderately to extensively degraded (i.e., category C-2 or C-3 as defined in the TS) such that the TS would not allow longer intervals. The 24-month inspection interval requirement usually proved sufficient in maintaining tube integrity. Nonetheless, there have been instances where licensees have performed mid-cycle inspections to ensure tube integrity would be maintained.

However, many SGs with alloy 600 MA tubing have been replaced with SGs with alloy 600 TT or alloy 690 TT tubing, which have proven to be much more resistant to stress corrosion cracking (SCC) than alloy 600 MA tubing. In addition, a few plants are operating with originally installed SGs with alloy 600 TT tubing. Based on early low levels of degradation, some of the plants with SGs with alloy 600 TT or 690 TT tubing are taking advantage of the longer inspection intervals permitted by the TS.

Under the proposed specification (TS 6.8.4.k), the required frequency of inspection in conjunction with inspection scope and inspection methods shall be such as to ensure that tube integrity is maintained until the next SG inspection. This addresses existing shortcomings in the current requirements in that it requires that inspection frequency be part of a management strategy aimed at ensuring tube integrity. The proposed TS 3/4.4.5 Bases states that inspection frequency will be determined, in part, by operational assessments that utilize additional information on existing degradation and flaw growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next SG inspection.

The NRC staff also notes, however, that any assessment or projection of the future condition of the SG tubing based on the existing condition of the tubing and anticipated flaw growth rates can involve significant uncertainty that may be difficult to conservatively and reliably bound. For this reason, the proposed specification (TS 6.8.4.k) supplements the performance-based requirement concerning inspection frequencies with a set of prescriptive requirements that provide added assurance that tube integrity will be maintained.

The proposed prescriptive requirements include a requirement that 100 percent of the tubes in each SG be inspected at the first refueling outage following SG replacement. The NRC staff notes that this requirement is a moot point for Sequoyah, Unit 1 since the first ISI of the replacement SGs has already been performed. The required scope of this inspection is substantially more restrictive than the current requirement, which requires a 3 percent sample of the total SG tube population and requires inspection of only two of the four SGs.

For Sequoyah, Unit 1, which has alloy 690 TT tubing, the proposed specification would require that 100 percent of the tubes be inspected at sequential periods of 144, 108, 72, and, thereafter, 60 effective full-power months (EFPM), with the first sequential period being considered to begin at the time of the first ISI of the SGs following SG replacement. This sliding scale is intended to address the increased potential for the initiation of stress corrosion cracking over time. In addition, the licensee would be required to inspect 50 percent of the tubes by the refueling

outage nearest the mid-point of the period and the remaining 50 percent by the refueling outage nearest the end of the period. However, no SG shall operate for more than 72 EFPM or three refueling outages (whichever is less) without being inspected.

Regardless of the type of tubing, if crack indications are found in any tube, the proposed specification requires that the next inspection for each SG for the degradation mechanism causing the crack indication shall not exceed 24 EFPM or one refueling outage (whichever is less). As a point of clarification, the proposed requirements stipulate that if definitive information, such as from examination of a pulled tube, diagnostic nondestructive testing, or engineering evaluation, indicates that a crack-like indication is not a crack, then the indication need not be treated as such.

These proposed prescriptive requirements, in total, cannot be described simplistically as more restrictive or less restrictive than current requirements. They are quite a different set of requirements, being generally more restrictive for SGs with low-to-moderate levels of degradation (i.e., categories C-1 to C-2 as defined in current TS) to somewhat less restrictive for plants with extensive levels of degradation other than cracks. As previously noted, management of SCC mechanisms relative to the performance criteria poses a particular challenge compared to other degradation mechanisms. The proposed requirement to limit inspection intervals to one refueling outage to address any cracking mechanism found to be present in the SGs is a substantially more restrictive requirement than current TS requirements that apply for plants with low-to-moderate levels of cracked tubes and, for practical purposes, leads to the same inspection frequency (every refueling outage) as would be required under current TS requirements for plants with moderate to extensive levels of cracked tubes.

The proposed prescriptive requirements relating to inspection frequency have been developed based on qualitative engineering considerations and experience, reflecting the improved SCC resistance of alloy 690 TT tubing relative to alloy 600 TT and particularly relative to alloy 600 MA tubing, that the potential for cracking increases with increasing time in service, and the particular challenges associated with the management of SCC with respect to satisfying the tube integrity performance criteria. The proposed prescriptive requirements are intended primarily to supplement the performance-based requirement that inspection frequency in conjunction with inspection scope and methods be such as to ensure tube integrity is maintained. This performance-based requirement must be satisfied in addition to the prescriptive requirements. The NRC staff concludes that the proposed performance-based requirement, in conjunction with the proposed prescriptive requirements, represents a significantly more effective strategy for ensuring tube integrity than that provided by current TS requirements and will serve to ensure that tube integrity is maintained between SG inspections.

3.3.4 Tube Repair Criteria

Revised TS 6.8.4.k would retain the current TS tube repair criteria (termed plugging limit[s] in current TSs) requirements. Specifically, the proposed specification would require that tubes found by ISI to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube wall thickness be plugged. This criterion is consistent with the tube integrity performance criteria in that flaws not exceeding the tube repair criterion satisfy the performance criteria with allowances for flaw size measurement error and incremental crack growth between inspections. The TS tube repair criteria provide added assurance that tube integrity will be maintained, given the performance-based strategy that is also to be followed under the proposed specification.

The inclusion of tube repair criteria as part of the proposed specification also ensures that the NRC staff has the opportunity to review any risk implications should the licensee propose a license amendment for alternate tube repair criteria, in conjunction with alternate tube integrity performance criteria, at some time in the future.

3.3.5 Monitoring of Operational Primary to Secondary Leakage

Proposed TS 6.8.4.k would require that the SG Program include provisions for monitoring primary-to-secondary leakage. The NRC staff's evaluation of this proposal is included as part of the NRC staff's evaluation of the proposed change to TS 3/4.6.2, "RCS Operational Leakage," in Section 3.5 of this safety evaluation.

3.4 TS 6.9.1.16, "Steam Generator (SG) Tube Inspection Report"

The proposed administrative TS 6.9.1.16 would revise the reporting requirements of existing TS 4.4.5.5, "Reporting." Currently, this specification requires that the complete results of the SG Tube Surveillance Program (i.e., the ISI results) be reported within 12 months following completion of the program and include (1) the number and extent of the tubes inspected, (2) the location and percent of wall thickness penetration for each indication, and (3) identification of tubes plugged. Under the revised requirement, a report shall be submitted within 180 days of entry into MODE 4 following a SG inspection. The report shall include:

- the scope of the inspections performed in each SG,
- active degradation mechanisms found,
- nondestructive examination techniques used for each degradation mechanism,
- location, orientation (if linear), and measured sizes (if available) of service induced indications,
- number of tubes plugged during the inspection outage for each active degradation mechanism,
- total number and percentage of tubes plugged to date,
- the results of condition monitoring, including the results of tube pulls and in-situ testing, and
- the effective plugging percentage for all plugging in each SG.

This revised reporting requirement is a more comprehensive requirement than the current 12-month report and will enhance the NRC staff's ability to monitor the kinds of inspections being performed, the extent and severity of each active degradation mechanism, degradation trends (stable or getting worse), and the degree of challenge faced by the licensee in maintaining tube integrity. The 180-day reporting requirement is adequate given that the failure of the SG program to maintain tube integrity as indicated by condition monitoring would be promptly reportable in accordance with 10 CFR 50.72 and Reference 8, allowing the NRC staff to engage in any follow-up activities that it determines to be necessary.

The specification currently requires that the number of tubes plugged in each SG be reported to the NRC within 15 days following completion of the program. In addition, the specification currently requires that inspection results falling into Category C-3 shall be reported to the NRC pursuant to 10 CFR 50.73 prior to the resumption of plant operation and that the report include a description of the tube degradation and corrective measures taken to prevent recurrence. The proposed administrative TS 6.9.1.16 deletes both of these requirements. The NRC staff finds deletion of these requirements to be acceptable. Neither the number of tubes plugged nor the finding of Category C-3 results (i.e., 10 percent of the tubes inspected contain degradation or 1 percent of the tubes inspected satisfy the tube repair criterion) have any real bearing on whether tube integrity is being maintained. The NRC staff also notes that the proposed TS 6.8.4.k would delete the definition of inspection results categories in the current TSs. If the SG program is effectively maintaining tube integrity, tubes found to be degraded or to be pluggable will also satisfy the tube integrity performance criteria. The regulation 10 CFR 50.72, in conjunction with Reference 8, requires that the NRC staff be promptly notified in the event that the tube integrity performance criteria are not met. The NRC staff would have the opportunity under the NRC ROP to follow up on such an occurrence as warranted. The regulation at 10 CFR 50.73 requires that a Licensee Event Report be issued within 60 days of the finding which addresses, in part, the degraded condition of the tube(s) and corrective measures being taken.

Based on the foregoing, the NRC staff finds the proposed revisions to the reporting requirements to be acceptable.

3.5 Definition of Leakage[2]

Technical Specification 1.1 currently defines leakage as a) Identified leakage, b) Unidentified leakage, and c) Pressure Boundary leakage. The third definition under Identified leakage is: “Reactor Coolant System (RCS) leakage through a steam generator (SG) to the Secondary System.” Pressure Boundary leakage is defined as “leakage (except SG Leakage) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.” The licensee has proposed to replace the term “SG leakage” with “primary to secondary leakage” because “SG leakage” is not used in the TSs or TS Bases. Therefore, the third definition of Identified leakage will state: “Reactor Coolant System (RCS) leakage through a steam generator to the Secondary System (primary to secondary leakage),” and the definition of Pressure Boundary leakage will state: “leakage (except primary to secondary leakage) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.” The proposed changes are editorial in nature and adequately reflect the terminology used throughout the TS and Bases. Therefore, the NRC staff finds the proposed revisions to the definition of leakage to be acceptable.

3.6 TS 3/4.6.2, RCS Operational Leakage

The licensee proposed several changes to the LCO, required actions, and SRs for TS 3/4.6.2, RCS Operational Leakage. These changes include administrative changes to the LCO, required action statements, and SRs. The format and language of Sequoyah, Unit 1 TSs are

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[2] The Sequoyah, Unit 1 TSs include individual definitions of the terms instead of defining them under the broader definition of leakage. The change is equivalent to that described in TSTF-449, Revision 4.

slightly different than the STSs and, therefore, the changes needed to reflect TSTF-449 are different. The changes proposed by the licensee incorporate the necessary parts of TSTF-449 into the plant-specific TSs.

The NRC staff has reviewed these administrative changes and finds them acceptable. In particular, the addition of "or primary to secondary leakage" to Condition A and the Note to SR 3/4.6.2.1 are considered to be administrative changes because these changes support the more restrictive addition of primary to secondary leakage to Condition B and SR 3/4.6.2.2. The need for the Note with respect to SR 3/4.6.2.1 (i.e., not applicable to primary to secondary leakage) and for the proposed new SR 3/4.6.2.2, which deals with primary to secondary leakage, is discussed in the proposed revision to the BASES in B3/4.6.2.2. The revised BASES states that SR 3/4.6.2.1 is not applicable to primary to secondary leakage because leakage rates of 150 gpd or less cannot be accurately measured by an RCS water inventory balance.

3.7 TS 3/4.6.2 Condition B Primary to Secondary Leakage

The primary to secondary leakage limit, together with the allowable accident induced leakage limit, helps to ensure that the dose contribution from tube leakage will be limited to less than the 10 CFR Part 100 and General Design Criterion (GDC) 19 dose limits or other NRC approved licensing basis for postulated accidents. The licensee proposed to add an additional OR statement to Condition B with regards to primary to secondary leakage. As proposed, Condition B would state:

> "Required Action and associated Completion Time of Condition A not met.
> OR
> Pressure boundary leakage exists.
> OR
> Primary to secondary leakage not within limit."

The current requirements, Condition A, have a completion time of four hours to reduce leakage (other than pressure boundary leakage) to within limits after which Condition B (plant shutdown) must be entered. The TS limit is more restrictive than the current requirements in that if primary to secondary leakage exceeds 150 gpd, then a plant shutdown must be commenced without an allowance to reduce leakage, as provided in Condition A. The revised Condition B would require the reactor to be in MODE 3 in 6 hours and MODE 5 in 36 hours if primary to secondary leakage is not within limits. As discussed in Section 3.6 above, the licensee has excluded primary to secondary leakage from Condition A. The NRC staff has reviewed the proposed change to Condition B. These changes are additional restrictions on plant operations that enhance safety; therefore, the NRC staff has concluded that the addition of the primary to secondary leakage OR statement to Condition B is acceptable.

3.8 TS 4.4.5.2 Surveillance Requirements - Primary to Secondary Leakage

SR 3/4.6.2.1 currently requires verification that RCS operational leakage is within limits by performance of RCS water inventory balance. The accompanying BASES state that primary to secondary leakage is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems. The BASES further state that the RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. As previously discussed in

Section 3.6 of this SE, the licensee has proposed adding a note to SR 3/4.6.2.1 stating that this particular surveillance requirement is not applicable to primary to secondary leakage. The licensee would revise the accompanying BASES justifying this change, namely, leakage of 150 gpd cannot be measured accurately by an RCS water inventory balance. The licensee has proposed a new surveillance requirement, SR 3/4.6.2.2, which would verify with a frequency of 72 hours that primary to secondary leakage does not exceed the 150 gpd LCO limit. The NRC staff believes this to be acceptable and in accordance with 10 CFR 50.36(c)(3). The revised requirement would not specify the specific method to be employed; however, it would require that the SG Program include provisions for monitoring primary to secondary leakage. There are a variety of methods that can be used and the NRC staff concludes there is no need to tie this surveillance to a specific method in order to ensure that the plant is operated safely and within its LCO limits. The licensee would state in the accompanying BASES that the primary to secondary leakage measurement uses continuous process radiation monitors or radio chemical grab sampling. The NRC staff notes that the EPRI PWR Primary-to-Secondary Leak Guidelines provide extensive guidance to this effect.

The accompanying BASES would also state that primary to secondary leakage is measured against the 150 gpd limit under room temperature conditions as described in the EPRI PWR Primary-to-Secondary Leak Guidelines. The BASES state that steam line break (SLB) is the most limiting accident or transient from the standpoint of dose releases from primary to secondary leakage. The Sequoyah, Unit 1 safety analysis for SLB assumes 3.7 gpm and 0.1 gpm primary to secondary leakage (for room temperature conditions) in the faulted and intact SGs respectively as an initial condition. Thus, the assumed total primary to secondary leakage from all SGs is 1440 gpd (1 gpm). The NRC staff concludes that measurement of operational primary to secondary leakage under room temperature conditions relative to the 150 gpd operational limit is acceptable since it ensures that leakage under hot operational conditions will be less than assumed in the Sequoyah, Unit 1 safety analysis and, thus, is in accordance with 10 CFR 50.36(c)(2)(ii).

The new SR, SR 3/4.6.2.2, with respect to primary to secondary leakage replaces the current SR which involved verifying SG tube integrity in accordance with the SG Tube Surveillance Program. As discussed earlier in this SE, much of the current TS 3/4.4.5, “Steam Generators,” would be moved to TS 6.8.4.k, “Steam Generator (SG) Program.” The SR to verify tube integrity would be addressed in the proposed new TS 3/4.4.5, “Steam Generator (SG) Tube Integrity,” SRs.

Based on the above, the NRC staff concludes that the proposed revisions are in accordance with 10 CFR 50.36(c)(3) and 10 CFR 50.36(c)(2)(ii) and are acceptable.

3.9 Technical Evaluation - Summary and Conclusions

The proposed Sequoyah, Unit 1 TS changes establish a programmatic, largely performance-based regulatory framework for ensuring SG tube integrity is maintained. The NRC staff finds that it addresses key shortcomings of the current framework by ensuring that SG programs are focused on accomplishing the overall objective of maintaining tube integrity. It incorporates performance criteria for evaluating tube integrity that the NRC staff finds consistent with the structural margins and the degree of leak tightness assumed in the current plant licensing basis. The NRC staff finds that maintaining these performance criteria provide reasonable assurance that the SGs can be operated safely without increase in risk.

The revised TSs would contain limited details concerning how the SG Program is to achieve the required objective of maintaining tube integrity, the intent being that the licensee will have the flexibility to determine the specific strategy for meeting this objective. However, the NRC staff finds that the revised TSs include sufficient regulatory constraints on the establishment and implementation of the SG Program such as to provide reasonable assurance that tube integrity will be maintained.

Failure to meet the performance criteria will be reportable pursuant to 10 CFR 50.72 and 50.73. The NRC ROP provides a process by which the NRC staff can verify that the licensee has identified any SG Program deficiencies that may have contributed to such an occurrence and that appropriate corrective actions have been implemented.

In conclusion, the NRC staff finds that the Sequoyah, Unit 1 TS amendment request conforms to the requirements of 10 CFR 50.36 and establishes a TS framework that will provide reasonable assurance that tube integrity is maintained without undue risk to public health and safety.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 70643; November 22, 2005). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1) Letter, R.E. Beedle, NEI, to L. J. Callan, NRC, December 16, 1997, transmitting NEI 97-06 (Original), “Steam Generator Program Guidelines.”
2) NEI 97-06, Revision 1, “Steam Generator Program Guidelines,” January 2001. ADAMS Accession No. ML010430054.
3) SECY-00-0078, “Status and Plans for Revising the Steam Generator Tube Integrity Regulatory Framework,” March 30, 2000.
4) Draft Regulatory Guide 1.121, “Bases for Plugging Degraded PWR Steam Generator tubes,” August 1976.
5) Memorandum dated September 8, 1999, to W. H. Bateman, Chief, EMCB, NRR, NRC from J. W. Anderson, EMCB, NRR, NRC, “Summary of August 27, 1999, Senior Management Meeting with NEI/EPRI/Industry to Discuss Issues Involving Implementation of NEI 97-06.” This memorandum encloses Industry White Paper entitled, “Deterministic Structural Performance Criterion Pressure Loading Definition.”
6) Memorandum dated May 19, 2004, from J. L. Birmingham, Project Manager, NRR, NRC to Cathy Haney, Program Director, Policy and Rulemaking Program, Division of Regulatory Improvement Programs, NRR, NRC, “Summary of May 14, 2004 Meeting with Nuclear Energy Institute (NEI) on Status of Steam Generator Structural Integrity Performance Criteria.” ADAMS Accession No. ML041540500.
7) NUREG-1570, “Risk Assessment of Severe Accident -Induced Steam Generator Tube Rupture,” March 1998.
8) NUREG-1022, Rev 2, “Event Reporting Guidelines 10 CFR 50.72 and 50.73,” October 31, 2000.[3]
9) NUREG-1649, Rev 3, “Reactor Oversight Process,” July 2000.

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[3] On September 24, 2004, a Federal Register notice (69 FR 57367) was published noticing the issuance of an errata to Revision 2 of NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73." The errata indicates that steam generator tube degradation is considered serious if either of the two criteria specified in Section 3.2.4(A)(3) of NUREG-1022 (i.e., the structural and accident leakage performance criteria), Revision 2, are not satisfied.