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

July 10, 2012

Mr. Joseph W. Shea
Manager, Corporate Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT REGARDING THE PROPOSED TECHNICAL SPECIFICATION CHANGES FOR REPLACEMENT STEAM GENERATOR TUBE INSPECTION REQUIREMENTS (TAC NO. ME6707) (TS-SQN-2011-01)

Dear Mr. Shea:

The Commission has issued the enclosed Amendment No. 323 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant, Unit 2. This amendment is in response to your application dated July 15, 2011, as supplemented by a letter dated October 20, 2011.

The license amendment revises the Sequoyah Nuclear Plant, Unit 2 Technical Specifications requirements for steam generator tube inspections to reflect the replacement steam generators to be installed during fall 2012 refueling outage. The changes made in this license amendment reflect the inspection requirements of Technical Specification Task Force (TSTF) Standard Technical Specification Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4.

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

Sincerely,

Siva P. Lingam

Siva P. Lingam, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-328

Enclosures:
1. Amendment No. 323 to License No. DPR-79
2. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 323
License No. DPR-79

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated July 15, 2011, as supplemented by a letter dated October 20, 2011, 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-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 323, 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 upon startup from fall 2012 refueling outage after completing the installation of new steam generators.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Eva Brown for/

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: July 10, 2012

ATTACHMENT TO LICENSE AMENDMENT NO. 323

FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace page 3 of Operating License DPR-79 with the attached page 3. 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
6-10a	6-10a
6-10b	6-10b
6-10c	-------
6-10d	-------
6-14a	6-14a
6-15	6-15

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 323 are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

c. Performance of any test at power level different from there described; and

d. Proposed changes that meet the criteria of Specification 6.8.4.j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

k. Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for Condition Monitoring Assessments.

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 an 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.

b. Provisions for Performance Criteria for SG Tube Integrity.

SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

1. Structural integrity performance criterion: All in-service SG 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 (DBAs). This includes retaining 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 the DBA primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the DBAs, 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 pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The accident-induced leakage is not to exceed 1.0 gpm for the faulted SG and 0.1 gpm for each of the non-faulted SGs. The primary-to-secondary accident induced leakage rate for any DBA, 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.

3. The operational leakage performance criterion is specified in Limiting Condition for Operation (LCO) 3.4.6.2, "Reactor Coolant System, Operational Leakage."

c. Provisions for SG Tube Repair Criteria.

Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG Tube Inspections.

Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., 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 criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. 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.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72 and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SGs shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for Monitoring Operational Primary-to-Secondary Leakage.

l. <u>Component Cyclic and Transient Limit</u>

This program provides controls to track the FSAR, Section 5.2.1, cyclic and transient occurrences to ensure that components are maintained within the design limits.

STEAM GENERATOR (SG) TUBE INSPECTION REPORT (continued)

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing, and

h. The effective plugging percentage for all plugging in each SG.

SPECIAL REPORTS

6.9.2.1 Special reports shall be submitted within the time period specified for each report, in accordance with 10 CFR 50.4.

6.9.2.2 This specification has been deleted.

6.10 RECORD RETENTION (DELETED)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 323 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-328

1.0 INTRODUCTION

By application dated July 15, 2011 (Agencywide Document Access and Management System (ADAMS) Accession No. ML11199A212), as supplemented by a letter dated October 20, 2011 (ADAMS Accession No. ML11298A081), Tennessee Valley Authority (the licensee) proposed an amendment to revise the Technical Specifications (TSs) requirements for steam generator (SG) tube inspections to be replaced during the fall 2012 refueling outage (RFO) for Sequoyah Nuclear Plant (SQN), Unit 2. Previous changes to the SQN, Unit 2, TSs to reflect the Technical Specification Task Force (TSTF) Standard Technical Specification Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4, were approved by U.S. Nuclear Regulatory Commission (NRC) on May 22, 2007 (ADAMS Accession No. ML071210013). The changes proposed in the current license amendment request reflect the inspection requirements of TSTF-449, Revision 4. The replacement steam generator (RSG) tubes will be made of Alloy 690 thermally treated (TT) material, and the existing SGs have Alloy 600 tubes. The revisions to TSs are required because the inspection frequency for Alloy 690 TT tube material, as defined in TSTF-449, differs from the inspection frequency for Alloy 600, and the tube repair processes and products in the existing TSs are not applicable to the RSGs.

The proposed amendment makes revisions to TSs 6.8.4.k, "Steam Generator (SG) Program"; and 6.9.1.16, "Steam Generator (SG) Tube Inspection Report": and is in support of the planned replacement of the SGs at SQN, Unit 2 during the fall 2012 RFO.

The supplement dated October 20, 2011, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's initial proposed no significant hazards consideration determination as published in the *Federal Register* on September 6, 2011 (76 FR 55131).

2.0 REGULATORY EVALUATION

SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and serve to isolate radiological fission products in the primary coolant from the secondary coolant and the environment. For the purposes of this safety evaluation, tube integrity means that the tubes are capable of performing these functions in accordance with the plant design and licensing basis.

The following explains the applicability of general design criteria (GDC) for SQN, Unit 2. The construction permit was issued by the Atomic Energy Commission (AEC) on May 27, 1970, for SQN, Unit 2. The operating license was issued on September 15, 1981. The plant GDC are listed in the Final Safety Analysis Report (FSAR), Section 3.1.2, "Overall Requirements." Section 3.1.2 of the FSAR states, "The Sequoyah Nuclear Plant was designed to meet the intent of the Proposed General Design Criteria for Nuclear Power Plant Construction Permits published in July, 1967. The Sequoyah construction permit was issued in May, 1970. This FSAR, however, addresses the NRC GDC published as Appendix A to 10 CFR 50 in July 1971." In accordance with an NRC staff requirements memorandum from S. J. Chilk to J. M. Taylor, "SECY-92-223 - Resolution of Deviations Identified during the Systematic Evaluation Program," dated September 18, 1992 (ADAMS Accession No. ML003763736), the Commission decided not to apply the Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, GDC to plants with construction permits issued prior to May 21, 1971. Therefore, the GDC that constitute the licensing bases for SQN Unit 2 are those in the FSAR. The GDC applicable to this license amendment, as identified by NRC staff, are consistent with the SQN, Unit 2 FSAR GDC.

The fundamental regulatory requirements with respect to the integrity of the SG tubing are established in 10 CFR. Specifically, Appendix A to 10 CFR, Part 50, "General Design Criteria," states that the reactor coolant pressure boundary (RCPB) shall have "an extremely low probability of abnormal leakage...and gross rupture," (GDC-14); "shall be designed with significant margin" (GDC-15 and -31); shall be of "the highest quality standards practical" (GDC-30); and shall be designed to permit "periodic inspection and testing...to assess...structural and leak tight integrity" (GDC-32). To this end, Section 50.55a of 10 CFR, "Codes and Standards," paragraph (c)(1) specifies that components that are part of the RCPB must meet the requirements of Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Section 50.55a, paragraph (g)(1) further requires that, throughout the service life of a pressurized-water reactor (PWR) facility, ASME Code Class 1 components meet the requirements in Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. Section XI requirements pertaining to inservice inspection of SG tubing are augmented by additional SG tube surveillance requirements in the TSs.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents, such as an SG tube rupture and main steamline break. These analyses consider the primary-to-secondary leakage through the tubing that may occur during these events.

Furthermore, the analyses must show that the offsite radiological consequences do not exceed the applicable limits of the 10 CFR Part 100, "Reactor Site Criteria," guidelines for offsite doses (or 10 CFR 50.67, "Accident Source Term," as appropriate), GDC-19, "Control Room," criteria for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis.

The SQN, Unit 2 TSs are modeled after TS TSTF-449, Revision 4, April 2005. TS 6.8.4.k for SQN, Unit 2 requires that a SG program be established and implemented to ensure that SG tube integrity is maintained. Tube integrity is maintained by meeting specified performance criteria for structural and leakage integrity consistent with the plant design and licensing bases. TS 6.8.4.k requires a condition monitoring assessment be performed during each RFO during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met. TS 6.8.4.k also includes provisions regarding the scope, frequency, and methods of SG tube inspections.

3.0 TECHNICAL EVALUATION

SQN, Unit 2 has four Westinghouse SGs designated model 51. Each SG contains 3388 mill annealed Alloy 600 tubes. Each tube has a nominal outside diameter of 0.875 inches and a nominal wall thickness of 0.050 inches. The tubes are supported by a number of carbon steel tube support plates and Alloy 600 anti-vibration bars. The tubes were explosively expanded into the tubesheet at both ends for the full length of the tubesheet. The U-bend region of the small radius tubes (i.e., rows 1 and 2) were in situ stress relieved following Cycle 6 (the row 1 tubes were plugged following Cycle 3 and were unplugged, inspected, and stress relieved following Cycle 6).

The NRC has approved a few amendments related to the original SQN SGs, including two alternate repair criteria. The licensee is permitted to repair tubes experiencing predominantly axially oriented outer–diameter stress corrosion cracking confined within the thickness of the tube support plates in accordance with their TSs. In addition, the licensee is permitted to implement the W* Methodology, in that, service induced flaws identified in the W* distance in the tubesheet shall be plugged on detection and those located below the distance may remain in service regardless of size.

The replacement SGs differ from the existing SGs. For example, the tube material is thermally treated Alloy 690 in the replacement SGs versus the mill annealed Alloy 600 in the existing SGs. The replacement SGs are scheduled to be installed during the fall 2012 RFO.

The licensee is proposing to remove the TS requirements associated with alternate tube repair criteria applicable to their original SGs. These requirements are contained in TSs 6.8.4.k.b (performance criteria), 6.8.4.k.c (tube repair criteria), 6.8.4.k.d (tube inspection criteria), and TS 6.9.1.16 (reporting requirements). The alternate tube repair criteria analyses were developed for the licensee's current SGs and not for the replacement SGs. As a result, the analyses used to justify these alternate tube criteria are not applicable to the replacement SGs since the replacement SGs have different design features than the original SGs. Therefore, the NRC staff finds these changes acceptable.

The licensee is also proposing to modify their inspection requirements to adopt those requirements in TSTF-449 applicable to SGs with thermally treated Alloy 690 tubes, which is the material used in their replacement SGs. The modifications include revised inspection intervals. The NRC staff finds the proposed changes to modify the current inspection requirements with those applicable to plants with thermally treated Alloy 690 tubes acceptable since it is consistent with TSTF-449, which the NRC staff has approved (refer to the *Federal Register* on May 6, 2005 (70 FR 24126)).

In summary, the NRC staff finds that the proposed changes to the SG TS requirements are acceptable because the resultant TS are consistent with TSTF-449 and reflect the tube material in the replacement SGs.

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. 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 issued on September 6, 2011 (76 FR 55131). 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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Obodoako

Date: July 10, 2012

July 10, 2012

Mr. Joseph W. Shea
Manager, Corporate Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT REGARDING THE PROPOSED TECHNICAL SPECIFICATION CHANGES FOR REPLACEMENT STEAM GENERATOR TUBE INSPECTION REQUIREMENTS (TAC NO. ME6707) (TS-SQN-2011-01)

Dear Mr. Shea:

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Sincerely,
/RA/
Siva P. Lingam, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-328
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2. Safety Evaluation
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ADAMS Accession No. ML12159A503 * Transmitted by memo dated

OFFICE	LPL2-2/PM	LPL2-2/LA	DE/ESGB/BC*	DSS/STSB/BC	OGC NLO	LPL2-2/BC	LPL2-2/PM
NAME	SLingam	BClayton	GKulesa	RElliott (MHamm for)	AGhosh	DBroaddus (EBrown for)	SLingam
DATE	06/18/12	06/18/12	06/04/12	06/19/12	06/21/12	07/10/12	07/10/12

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