May 29, 2003

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

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS REGARDING RISK INFORMED INTEGRATED LEAK RATE TESTING EXTENSION (TAC NOS. MB6987 AND MB6988)

Dear Mr. Scalice:

The Commission has issued the enclosed Amendment No. 287 to Facility Operating License No. DPR-77 and Amendment No. 276 to Facility Operating License No. DPR-79 for the Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively. These amendments are in response to your application dated October 4, 2002, as supplemented February 19, 2003, and May 19, 2003.

The Amendments revise Technical Specification 6.8.4.h, Containment Leakage Rate Testing Program, to allow the licensee to postpone its Appendix J, Type A, Containment Integrated Leak Rate Test (ILRT) up to 5 years for SQN, Units 1 and 2. Specifically, for Unit 1 the performance of the spring 2003 ILRT may be deferred up to 5 years but no later than spring 2008, and for Unit 2 performance of the fall 2003 ILRT may be deferred up to an additional 3.5 years but no later than spring 2007. In Amendment No. 265 to the Facility Operating License No. DPR-79 for SQN, Unit 2, Technical Specification 6.8.4.h was revised to allow the licensee to postpone the ILRT one cycle (i.e., 1.5 years) from spring 2002. Therefore, the total deferral for SQN, Unit 2 from the original requirement to perform an ILRT in spring 2002 will be up to 5 years.

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/

Michael L. Marshall, Jr., Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 287 to License No. DPR-77
2. Amendment No. 276 to License No. DPR-79
3. Safety Evaluation

cc w/enclosures: See next page

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ADAMS ACCESSION: ML031500140

cc w/enclosures: See next page

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TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 287
License No. DPR-77

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 4, 2002, as supplemented February 19, 2003, 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. 287, 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 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Allen G. Howe, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 29, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 287

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
6-10a	6-10a

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 276
License No. DPR-79

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

   A. The application for amendment by Tennessee Valley Authority (the licensee) dated October 4, 2002, as supplemented February 19, 2003, and May 19, 2003, 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. 276, 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 45 days after issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/***RA***/

Allen G. Howe, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 29, 2003

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 276</u>

<u>FACILITY OPERATING LICENSE NO. DPR-79</u>

<u>DOCKET NO. 50-328</u>

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.

<u>REMOVE</u>	<u>INSERT</u>
6-9	6-9

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 287 TO FACILITY OPERATING LICENSE NO. DPR-77

AND AMENDMENT NO. 276 TO FACILITY OPERATING LICENSE NO. DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By application dated October 4, 2002, as supplemented February 19, 2003, and May 19, 2003, Tennessee Valley Authority (the licensee) proposed an amendment to the Technical Specifications (TSs) for Sequoyah Nuclear Plant (SQN) Units 1 and 2. The requested changes would revise TS 6.8.4.h, Containment Leakage Rate Testing Program, to allow the licensee to postpone its Appendix J, Type A, Containment Integrated Leak Rate Test (ILRT). Specifically, the proposed TSs state that for Unit 1 the performance of the spring 2003 ILRT may be deferred up to 5 years but no later than spring 2008, and for Unit 2 performance of the fall 2003 ILRT may be deferred up to an additional 3.5 years but no later than spring 2007. In Amendment No. 265 to the Facility Operating License No. DPR-79 for SQN, Unit 2, Technical Specification 6.8.4.h was revised to allow the licensee to postpone the ILRT one cycle (i.e., 1.5 years) from spring 2002. Therefore, the total deferral for SQN, Unit 2 from the original requirement to perform a ILRT in spring 2002 will be up to 5 years.

The proposed changes are submitted on a risk-informed basis as described in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The February 19, 2003, and May 19, 2003, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination or expand the scope of the application.

2.0 REGULATORY EVALUATION

RG 1.174 provides guidance on the use of Probabilistic Risk Assessment findings and risk insights in support of licensee requests for changes to a plant's licensing basis, as in requests for license amendments and technical specification changes.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix J, Option B, requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. SQN Units 1 and 2, TS 6.8.4.h requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, with exceptions provided in the site implementing instructions. RG 1.163 endorses, with certain exceptions, Nuclear Energy Institute (NEI) report 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at each SQN unit have been successful, so the current interval requirement is 10 years.

The licensee is requesting additions to TS 6.8.4.h, "Containment Leakage Rate Testing Program," which would add another exception from the guidelines of RG 1.163, regarding the Type A test interval. Specifically, the proposed technical specifications state that the performance of the Unit 1 Spring 2003 Type A test may be deferred up to 5 years, but no later than spring 2008, and the performance of the fall 2003 containment integrated leakage rate (Type A) test may be deferred to no later than spring 2007. The licensee's proposed TS change is an extension of the currently specified 10-year interval for an ILRT to a 15-year interval on a one-time basis. The local leakage rate tests (Type B and C tests), including their schedule, are not affected by this request.

3.0 TECHNICAL EVALUATION

3.1 TS Administrative Controls Section 6.8.4.h, Containment Leakage Testing Program

The licensee has proposed to revise TS Administrative Control Section 6.8.4.h to add another exception from the guidelines of RG 1.163, regarding the Type A test interval. The revised TS Administrative Control Section 6.8.4.h is as follows (proposed changes are underlined):

Sequoyah Unit 1:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B as modified by approved exemptions. Visual examination and testing, including test intervals and extensions, shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions. <u>Performance of the spring 2003 containment integrated leakage rate (Type A) test may be deferred up to 5 years but no later than spring 2008.</u>

Sequoyah Unit 2:

> A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. Visual examination and testing, including test intervals and extensions, shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with

exceptions provided in the site implementing instructions. Performance of the fall 2003 containment integrated leakage rate (Type A) test may be deferred to no later than spring 2007.

The leak rate testing requirements of Option B of Appendix J, and the containment inservice inspection (ISI) requirements mandated by the 10 CFR 50.55a complement each other in ensuring the leak-tightness and structure integrity of the containment. Therefore, a detailed evaluation related to the ISI of the containment and potential areas of weaknesses in the containment is performed in the following section.

3.2 Inservice Inspection for Primary Containment Integrity

The SQN, Units 1 and 2, are Westinghouse pressurized-water reactors (PWRs) with a primary containment structure consisting of a freestanding steel vessel with an ice condenser, and a separate secondary containment—a reinforced-concrete, shield building. The primary containment vessel consists of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete.

The SQN TS (Section 6.8.4.h) establishes the requirements for implementing a program to perform containment leak rate testing in accordance with 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions.

The integrity of the penetrations and isolation valves are verified through Type B and Type C local leak rate tests as required by 10 CFR Part 50, Appendix J, and the overall leak-tight integrity of the primary containment is verified through an ILRT. These tests are performed to verify the essentially leak-tight characteristics of the containment at the design-basis accident pressure. The last ILRT for SQN Unit 1 was performed in December 1993. The next ILRT is scheduled to begin in spring 2003. With the extension of the ILRT time interval, the next overall verification will be performed no later than spring 2008. The last ILRT for SQN Unit 2 was performed in April 1992. The next ILRT is scheduled to begin in fall 2003. With the extension of the ILRT time interval, the next overall verification will be performed no later than spring 2007. The licensee provided information related to the ISI of the containment and discussed potential areas of weaknesses in the containment that may not be apparent in the risk assessment. In addition, in its February 19, 2003, letter the licensee provided responses to the staff's request for additional information to explicitly address five issues related to ISI of the containment. The U. S. Nuclear Regulatory Commission (NRC) staff's evaluation of the licensee's responses to these issues is discussed in the following paragraphs.

ISI Program at SQN Units 1 and 2: The licensee stated that the ISI program was established in 1995, in accordance with Subsection IWE of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (ASME Code), Section XI, 1992 Edition with the 1992 Addenda, to assure detection of degradation affecting containment integrity. The first 10-year containment inspection (IWE) interval is divided into three periods as follows:

First Period:	September 9, 1996 - September 8, 2001
Second Period	September 9, 2001 - September 8, 2005
Third Period	September 9, 2005 - September 8, 2008

The licensee stated that first credited examination for the first period IWE program was performed for SQN Unit 1 on January 19, 2000, during the Cycle 10 refueling outage and for Unit 2 on March 24, 1999, during the Cycle 9 refueling outage. The visual examinations of the Units 1 and 2 steel containment vessel (SCV) have been performed in accordance with the IWE program. To date, no major indications of containment degradation have been found. These periodic IWE examinations provide reasonable assurance that degradation of the containment structure will be detected and corrected before it can affect the structural integrity or leak tightness of the containment.

<u>Implementing IWE-1240 at SQN</u>: The licensee states that the SQN Units 1 and 2 augmented examination areas identified are at chilled water system penetrations X-64, X-65, X-66, and X-67 on the exterior side of the SCV. These areas are examined once per period in accordance with ASME Code, Section XI, Table IWE-2500-1, Examination Category E-C, Item Number E4.12. The nozzle reinforcement on the exterior side of the penetrations had corrosion due to moisture absorbed and held against the nozzle reinforcement by foam insulation. These areas were ultrasonically examined for minimum wall thickness. The initial examinations on the penetrations were performed on March 7, 2000, for Unit 1 (Cycle 10 refueling outage), and May 6, 1999, for Unit 2 (Cycle 9 refueling outage). The licensee states that the thickness data showed that the remaining thickness was acceptable. The next scheduled examination of these areas is during the next period (the next period contains both Cycles 12 and 13 refueling outages). Accordingly, the areas identified to date for augmented examination have not impacted the structural integrity or leak tightness of the steel containment vessel.

<u>IWE Table-2500-1, Examination Categories E-D and E-G for seals and gaskets, and examination and testing of bolts</u>: The licensee states that Relief Requests CISI-01 and CISI-04 for examination Categories E-D and E-G were previously authorized by NRC staff. In Relief Requests CISI-01 and CISI-04, an alternative test requirement (i.e., 10 CFR Part 50, Appendix J, Option B) eliminates the need to perform visual examination of seals and gaskets for containment penetrations. This test provides reasonable assurance of the leak-tightness of ASME Class MC (metal containment) pressure-retaining, bolted connections. The initial test frequency for performing a leak test on seal, gaskets and bolts, which are Type B components, is at least once every 30 months. If two consecutive as-found Type B tests are within their administrative limit, the test interval may be extended to 60 months. If three consecutive as-found Type B tests are within their administrative limit, the test interval may be extended to 120 months. If a test result is greater than the administrative limit for the components, the components are restored to a leak rate below the administrative limit, and the test interval is re-established at 30 months. The licensee states that at SQN, seals and gaskets are tested in accordance with 10 CFR Part 50, Appendix J, Option B, and are Type-B-tested during a 60-month period for the full population. They are tested on a staggered basis such that a portion is tested each refueling outage. Since Option B was first implemented at SQN (spring 1997 for Unit 1 and fall 1996 for Unit 2), seals and gaskets on both units have been tested at least once and are undergoing their second round of testing on a staggered basis. In addition to the 60-month tests, testing is performed prior to and following disassembly of the containment penetration. Testing of seals and gaskets will also occur as part of the ILRT (Type A test) at the end of the 15-year extended interval because the Type A test will challenge all Test B test barriers. All the exposed surfaces of bolted connections are visually examined (VT-1) once each inspection interval in accordance with the requirements of the ASME Code, Section XI, Table IWE-2500-1.

Integrity of stainless steel bellows: In the past, the staff has found that two-ply stainless steel bellows are susceptible to trans-granular stress corrosion cracking, and the leakage through them is not detectable by Type B testing (see NRC Information Notice 92-20, Inadequate Local Leak Rate Testing). The licensee states that the bellows are tested under the 10 CFR Part 50, Appendix J program (Option B) which would allow extended test intervals up to 120 months for Type B components based on acceptable performance. Due to industry concerns, SQN has limited the extended test intervals of its bellows to 60 months. Additionally, penetrations with bellows are tested on a staggered basis such that a portion is tested each refueling outage. If the bellows test fails the Appendix J Option B test, the bellows' sheet metal cover is removed, the bellows are pressurized to test pressure, and visually inspected for leakage using a bubble solution (snoop), lights, mirror, etc. The bellows are repaired or replaced as necessary if the bellows are found to be leaking. The extension of ILRT frequency from 10 to 15 years has no effect on this testing since frequency of inspection and testing of these bellows is limited to 60 months.

Inspection of embedded side of the containment steel shell: The staff is concerned that inspections of some reinforced concrete and steel containment structures at other nuclear power plants have identified degradation (e.g., corrosion) on the uninspectable (embedded) side of the containment steel shell of the primary containment. The major uninspectable areas of the ice condenser containment include those behind the ice baskets and part of the shell embedded in the basemat. A summary of work performed in the inaccessible region of the containment follows:

A visual examination (VT-3) was performed on the SCV interior surface in the vicinity of the moisture barrier at the interface of the SCV and raceway floor for Unit 1 during the Cycle 10 refueling outage and Unit 2 during Cycles 9 and 10 refueling outages. This examination was a result of the periodic VT-3 visual examination of the moisture barrier to meet the ASME Code, Section XI, Table IWE-2500-1, Examination Category E-D, Item Number E5.30. The examination results identified degradation of the moisture barrier at various locations, where the seal was not adhering to the concrete and at the SCV interface on both units. A VT-3 examination of the SCV was performed from 12 inches above the floor to 6 inches below the floor during the Unit 1 Cycle 10 refueling outage and Unit 2 Cycles 9 and 10 refueling outages, over the full length of the moisture barrier. The VT-3 examination was performed in accordance with the requirements of the ASME Code, Section XI, paragraph IWE-2500(b). The examination identified conditions consisting of mild uniform corrosion, discoloration and minor pitting below the floor surfaces on both units. One area on Unit 1 was identified at 30 degrees azimuth where the SCV wall thickness was slightly reduced due to corrosion mechanisms. However, ultrasonic thickness measurements verified that there was no wall loss below the original nominal wall plate thickness in this location. On Unit 2, the area between 170 and 177 degrees azimuth that was examined during the Cycle 9 refueling outage identified 11 areas of pitting, and during the Cycle 10 refueling outage one area at 273.5 degrees azimuth where the SCV wall thickness was slightly reduced due to corrosion mechanisms. However, ultrasonic thickness measurements verified that there was no significant wall loss at these locations and each area was within the design minimum wall thickness. All areas were evaluated by the licensee's engineering staff and no detrimental flaws or significant degradation of the SCV liner were noted during the evaluation. All of the existing moisture barrier, along with the fiberglass filler in the crevice (6 inches below the surface), was removed and replaced with a polyurethane elastomeric material during the Units 1 and 2 Cycle 10 refueling outages. This polyurethane elastomeric material will serve to fill the crevice area, act as the protective coating for the SCV, and provide a leak-tight barrier.

The licensee states that the action described above will arrest any SCV degradation and will preserve containment integrity beyond the 5-year extension interval.

During the Unit 1 Cycle 10 and Unit 2 Cycle 9 refueling outages, ultrasonic thickness measurements were taken at three locations (2-feet x 3-feet grids) on the exterior side of the SCV at the seal area between the ice condenser and the SCV. These ultrasonic thickness measurements revealed no areas below the original nominal wall-plate thickness. There was no material degradation noted in these examination areas.

The SQN steel containment vessel contains areas that are inaccessible inside containment due to the ice condenser system design configuration. Although these areas are not susceptible to any particular degradation mechanism, TVA plans to perform additional inspections to validate integrity of the steel containment vessel. Additional ultrasonic thickness measurements on the SCV inaccessible areas will be performed during the Units 1 and 2 Cycle 12 refueling outages to assess potential degradation. The ultrasonic thickness measurements will be taken at the 4-inch spacing line intersections in each 12-inch x 12-inch grid. Any degraded areas will be evaluated by the licensee's engineering staff for inclusion in the augmented program per ASME Code, Section XI, paragraph IWE-1240. These grids are randomly selected at the following areas:

Two inaccessible areas are behind the ice condenser wall panels and behind the insulation on the exterior of the SCV outside the incore instrument room. A sampling of 24 grids is planned for these areas.

- 796 elevation - SCV area at the interface to the top deck panel (6 grids)
- 778-788 elevations - SCV area behind the ice condenser where sweating on the exterior side of the SCV has been observed (6 grids)
- 721 elevation - SCV area at the vapor barrier for the ice condenser floor (6 grids)
- 691-721 elevations - SCV area behind the insulation on the exterior side (6 grids)

The inaccessible SCV exterior area behind the emergency gas treatment system (EGTS) duct work at the floor-to-SCV interface will be VT-3 examined when the duct work is removed to allow access during the Cycle 12 refueling outage on each unit. Following examination, this area will be examined when the general visual examination for the SCV is scheduled in accordance with the ASME Code, Subsection IWE.

During the Unit 2 Cycle 11 refueling outage, 12 feet of the EGTS duct work was removed and the SCV examined. Minor corrosion and pitting were identified with no visible sign of active corrosion. There were no detrimental flaws or significant degradation noted during the examination. The SCV at these locations was re-coated.

The licensee states that the potential leakage due to age-related degradation in uninspectable areas is factored into TVA's risk assessment which supports the requested ILRT interval extension from 10 to 15 years.

Based on its review of the information provided by the licensee, the staff finds that (1) the structural integrity of the containment vessel is adequately verified through periodic ISIs as required by Subsection IWE of the ASME Code, Section XI; (2) the integrity of the penetrations, containment isolation valves and bellows is periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J and the TS, and (3) the potential leakages from uninspectable areas are factored into and addressed in its risk assessment. The staff finds that implementation of the licensee's containment ISI program, including the areas covered by augmented inspections, will provide adequate assurance that the containment structural integrity will be maintained during the extended ILRT period.

3.3 Risk Impact of Extending Type A Test Interval

The licensee performed a risk impact assessment of extending the Type A test interval to 15 years. An updated risk assessment based on Revision 2 of the SQN Probabilistic Safety Analysis (PSA) was provided in the licensee's application. The updated assessment provides a revised estimate of the impact of the extended test interval on large early release frequency (LERF). The estimated impact of the extended test interval on population dose and on conditional containment failure probability were not revised and remains bounding. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) Project Report TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174.

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, industry undertook a similar study. The results of that study are documented in EPRI TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for SQN early in the plant's life, required a Type A test frequency of three tests in 10 years. The EPRI study estimated that relaxing the test frequency from three tests in 10 years to one test in 10 years would increase the average time that a leak that was detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of the leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10-percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the PWR and boiling-water reactor representative plants in the EPRI study confirmed a NUREG-1493 conclusion that a reduction in the frequency of Type A tests from three tests in 10 years to one test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of one person-rem per year in increased public dose.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem per year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking was completed in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per year and increases in LERF less than 10^{-7} per year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original three tests in 10 years frequency. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability for the proposed change to demonstrate that the defense-in-depth philosophy is met.

The licensee provided analyses, as discussed below. The following comparisons of risk from a change in test frequency from three tests in 10 years to one test in 15 years are considered to be bounding for the SQN comparative frequencies of one test in 10 years to one test in 15 years. The following conclusions can be drawn from the analysis associated with extending the Type A test interval:

1. Given the change from a three in 10-year test frequency to a one in 15-year test frequency, the increase in the total integrated plant risk, in person-rem per year, is estimated to be about 0.37 percent based on information provided in the licensee's original risk assessment (October 9, 2001, letter). This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from three in 10 years to one in 20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original three in 10 years to one in 15 years is estimated to be 6.5×10^{-8} per year based on Revision 2 of the SQN PSA. However, there is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE/IWL). The most recent visual examination of the SQN containment was performed in 2000 and 1999 for Units 1 and 2, respectively. The next scheduled IWE/IWL containment inspection is in 2003 for both Units 1 and 2. Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events

is estimated to be less than 1 x 10^{-7} per year. The staff concludes that increasing the Type A test interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. Based on information provided in the licensee's original risk assessment (October 9, 2001, letter), the staff estimates that the change in test frequency from three in 10 years to one in 15 years would result in a 0.3 percent increase in the conditional containment failure probability. The staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174.

3.4 Summary

On the basis of the considerations discussed above, the NRC staff concludes that the licensee has adequate procedures to examine and monitor potential age-related and environmental degradations of the pressure retaining components of the SQN, Units 1 and 2 primary containment. Also, the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174. Thus, granting a one-time extension of performing the ILRT as proposed by the licensee in Section 6.8.4.h of its proposed TS change request for SQN Units 1 and 2 is acceptable.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change an inspection or a surveillance requirement. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (68 FR 5681). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 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.

6.0 REFERENCES

1. Title 10 of the *Code of Federal Regulations* (CFR), Part 50, Appendix J

2. ASME *Boiler and Pressure Vessel Code*, Section XI, 1992 Edition including 1992 Addenda.

3. NRC Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," September 1995.

4. Nuclear Energy Institute Document, NEI 94-01, Revision0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

5. NRC Information Notice 92-20, "Inadequate Local Leak Rate Testing."

6. Letter from Pedro Salas (Tennessee Valley Authority) to NRC, "Sequoyah Nuclear Plant - Units 1 and 2 - Technical Specification Change No. 02-07, One-time Frequency Extension for Type A Test (Containment Integrated Leak Rate Test [CILRT])," October 4, 2002.

7. Letter from Pedro Salas (Tennessee Valley Authority) to NRC, "Sequoyah Nuclear Plant - Units 1 and 2 - Response to Request for Additional Information (RAI) Regarding Technical Specification Change No. 02-07, One-time Frequency Extension for Type A Test (Containment Integrated Leak Rate Test [CILRT])," February 19, 2003.

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