Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

October 4, 2002

TVA-SQN-TS-02-07 10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - UNITS 1 and 2 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 02-07, "ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST (CONTAINMENT INTEGRATED LEAK RATE TEST [CILRT])"

References:

1. TVA letter to NRC dated October 9, 2001, SQN - Unit 2 - Technical Specification (TS) Change No. 01-10, "One-time Frequency Extension for Type A Test (Containment Integrated Leak Rate Test [CILRT])"
2. NRC letter to TVA dated May 7, 2002, "Issuance of Technical Specification Amendment for the Sequoyah Nuclear Plant, Unit 2 - Regarding One-Time Extension of Containment Type A Pressure Test (TAC No. MB3275) (TS 01-10)"
3. TVA letter to NRC dated August 31, 2001, Sequoyah Nuclear Plant (SQN) - Response to Request for Additional Information (RAI) Regarding Risk Informed Inservice Inspection (RI-ISI) Program"

D030

Pursuant to 50.90, TVA hereby requests the following amendment to the SQN licenses DPR-77 and DPR-79 to change the TSs for Units 1 and 2. TVA proposes to revise TS 6.8.4.h, "Containment Leakage Rate Testing Program," to allow a 1-time, 5-year extension to the current 10-year test interval for the performance-based leakage rate test program for 10 CFR 50, Appendix J, Type A tests. TVA's application is a continuation of the information provided previously in References 1 and 3 and as discussed in Reference 2.

The proposed change is 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," and makes use of Revision 2 of the SQN Probabilistic Safety Assessment (PSA). TVA has determined that the resultant increase in Large Early Release Frequency (LERF) for the proposed change is "very small" (i.e., less than 1.0E-07/reactor year [ry]) and satisfies the RG 1.174 criteria.

TVA's American Society of Mechanical Engineers (ASME) IWE program performs containment inspections in order to detect evidence of degradation that may affect either the containment structural integrity or leak tightness. In addition to the IWE examinations, TVA will perform additional non-destructive examinations of the steel containment vessel in the ice condenser region (inaccessible areas) at various elevations. These additional non-destructive examinations will provide added assurance of containment integrity during the 5-year extended interval.

Performance of a Type A test imposes a significant expense to TVA ($225,000 for Unit 1 and $300,000 for Unit 2) while the safety benefit of performing a test within 10 years versus 15 years is minimal.

It should be noted that in spring 2003, SQN will replace steam generators on Unit 1. This project will include cutting the containment structure for removal of the original steam generators. TVA has evaluated requirements associated with post-modification testing (PMT) of the steel containment vessel following steam generator replacement. Based on the evaluation of PMT test requirements, TVA is proposing to perform an ASME code pressure test and local leak rate test of the affected areas in lieu of a full CILRT. This is a technically sound post-modification test that has been performed on similar containment modifications at Turkey Point, Fitzpatrick, Vermont Yankee and St. Lucie Nuclear Stations. The approach complies with 10 CFR 50, Appendix J, NEI 94-01, ANS 56.8, 1994, and ASME Section XI Code, 1992 Edition, 1992 Addenda, Subsection IWE.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that the TS change qualifies for categorical exclusion from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter to the Tennessee State Department of Public Health.

Enclosure 1 to this letter provides description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains marked up copies of the appropriate TS pages from Units 1 and 2 to show the proposed changes. Enclosure 3 contains TVA commitments.

TVA's proposed license amendment is similar to the Duke Energy Corporation submittals for Catawba and McGuire Nuclear Stations as provided by letter dated May 29, 2002.

TVA requests approval of this TS change by February 1, 2003, to allow final planning, scheduling, and preparation for the SQN Unit 1 Cycle 12 refueling outage (scheduled to begin in March 2003). In addition, TVA requests implementation of the revised TS be within 45 days of NRC approval.

This letter is being sent in accordance with NRC RIS 2001-05.

If you have any questions about this change, please telephone me at (423) 843-7170 or J. D. Smith at (423) 843-6672.

Sincerely,

Pedro Salas
Licensing and Industry Affairs Manager

Subscribed and sworn to before me
on this 4th day of October

Peggy M. Billingsley
Notary Public

My Commission Expires August 12, 2006

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ENCLOSURE 1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT (SQN)
UNITS 1 and 2
DOCKET NOS. 327 and 328

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE NO. 02-07

TVA's EVALUATION

SUBJECT: SEQUOYAH NUCLEAR PLANT (SQN) - DOCKET NOS. 50-327, 50-328 - TECHNICAL SPECIFICATION (TS) CHANGE NO. 02-07, "ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST (CONTAINMENT INTEGRATED LEAK RATE TEST [CILRT])"

1.0 DESCRIPTION

This letter is a request to amend SQN Operating License(s) DPR-77 and -79 to change the TSs for Units 1 and 2.

TVA's proposed change will add a 1-time, 5-year deferral of the Containment Integrated Leak Rate Test (CILRT), also referred to as the 10 CFR 50, Appendix J, Type A test to TS Section 6.8.4.h, "Containment Leakage Rate Testing Program." Section 6.8.4.h contains the general 10 CFR 50, Appendix J test and leakage requirements for the SQN steel containment structure. The Containment Leakage Rate Testing Program refers to requirements contained in 10 CFR 50, Appendix J, Option B and NRC Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995. The RG endorses Nuclear Energy Institute (NEI) 94-01, Revision 0, entitled "Industry Guideline For Implementing Performance Based Option of 10 CFR 50, Appendix J," which requires that Type A tests be performed "at least once per 10-years based on acceptable performance history."

2.0 PROPOSED CHANGE

TVA's proposed change requests, on a one-time basis, an extension to the current 10-year Type A test interval to allow a 15-year test interval. Accordingly, SQN TS Section 6.8.4.h, "Containment Leakage Rate Testing Program," is revised to add the following provision for each unit:

Unit 1

"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."

Unit 2

"Performance of the fall 2003 containment integrated leakage rate (Type A) test may be deferred up to 4 years but no later than spring 2007."

In summary, the proposed change to TS 6.8.4.h will revise the Containment Leakage Rate Test Program requirements to allow a 1-time, 5-year Type A frequency extension.

3.0 BACKGROUND

The SQN primary containment structure for Units 1 and 2 consists of a freestanding steel vessel with an ice condenser and a separate secondary containment that is 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. SQN Final Safety Analysis Report (FSAR) Figure 3.8.2-1 shows the outline and configuration of the steel containment vessel. Section 6.2.1 of the SQN FSAR describes SQN's containment design features.

The SQN TS (Section 6.8.4.h) establishes the requirements for implementing a program to perform containment leakage rate testing in accordance with 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. The types of containment leakage tests include Type A (CILRT), Type B (local leakrate testing for containment penetrations, hatches, personnel air locks, electrical penetrations, etc.) and Type C (local leakrate testing for containment isolation valves). SQN's maximum allowable containment leakage rate is 1.0 L_a which is defined as 0.25 percent of the containment free air volume per day at an accident pressure of 12.0 pounds per square inch.

The Type A test interval for SQN is based on Type A test history and performance and is currently once every 10 years. The test interval for Type A testing is based on NEI 94-01, that states: "Type A testing shall be performed during a period of reactor shutdown at a frequency of at least once per 10 years based on acceptable performance history. Acceptable performance history is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 L_a." Also included with NEI 94-01 is consideration of Plant-Specific Testing Program Factors described in Section 11.3.

The last SQN Unit 1 Type A test was conducted in December 1993 during the Unit 1 Cycle 6 refueling outage. The last SQN Unit 2 Type A test was conducted in April 1992 during the Unit 2 Cycle 5 refueling outage. In accordance with the current SQN TS requirements, Units 1 and 2 are required to perform the next 10-year CILRT during the upcoming Cycle 12 refueling outage (Unit 1 is spring 2003 and Unit 2 is fall 2003). The cost to TVA for performing a CILRT is substantial (estimated cost is $225,000 for Unit 1 and $300,000 for Unit 2). Additional replacement power costs include 20 hours of critical path time for Unit 1 and 36 hours for Unit 2. An estimate of radiological cost to perform a Type A test is 500 millirem of dose for each unit. Accordingly, TVA is proposing a change to TS Section 6.8.4.h to defer the substantial cost of conducting a Type A test and to save critical path time during the upcoming Cycle 12 refueling outages.

4.0. TECHNICAL ANALYSIS

The testing requirements of 10 CFR 50, Appendix J, provide assurance that leakage through the containment, including systems and components that penetrate the containment, does not exceed the allowable leakage value specified in the SQN TSs (L_a). The limitation of containment leakage provides assurance that the containment would perform its design function following a design basis accident.

The 10 CFR 50, Appendix J rule was revised (effective October 26, 1995) to allow licensees to choose containment leakage testing under Option A, "Prescriptive Requirements" or Option B, "Performance-Based Requirements." TVA requested a license amendment for SQN to allow implementation of Option B and was granted approval by NRC letter dated February 5, 1996. The SQN TS was subsequently revised to include Option B. The SQN TS revision included a reference to NRC RG 1.163 for performing Type A, B, and C testing. RG 1.163 specifies a method acceptable to NRC for complying with Option B by endorsing the use of NEI 94-01 and ANSI/ANS 56.8-1994, subject to specific regulatory positions in the RG.

Exceptions to the requirements of RG 1.163 are allowed by 10 CFR 50, Appendix J, Option B, Section V.B, "Implementation," which states:

> *The Regulatory Guide or other implementing document used by a licensee, or applicant for an operation license, to develop a performance based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.*

The adoption of the Option B performance-based containment leakage rate testing program did not alter the basic method by which Appendix J leakage rate testing is performed, but it did alter the frequency of measuring primary containment leakage in Type A, B and C tests. Frequency is based upon an evaluation which looks at the "as found" leakage history to determine the frequency for leakage testing that provides assurance that leakage limits will be maintained. The changes to the Type A test frequency did not directly result in an increase in containment leakage. Similarly, the proposed change to the Type A test frequency will not directly result in an increase in containment leakage.

The allowed frequency for testing was based upon a generic evaluation documented in NUREG-1493, "Performance-Based Containment Leakage-Test Program." Section 10.1.2 of this NUREG provided the following observations with regard to the Type A test frequency:

> *Reducing the frequency of Type A tests (ILRTs) from the current three per 10 years to one per 20 years was found to lead to an imperceptible increase in risk. The estimated increase in risk is very small because ILRTs identify only a few potential leakage paths that cannot be identified by Type B and C testing and the leaks that have been found by Type A tests have been only marginally above the existing requirements.*
>
> *Given the insensitivity of risk to containment leakage rate (Chapter 5) and the small fraction of leakage paths detected solely by Type A testing, increasing the interval between integrated leakage-rate tests is possible with minimal impact on public risk.*
>
> *The findings to date strongly support earlier indications that Type B and C testing can detect a very large fraction of containment leaks. The fraction of leaks that can be detected only by integrated containment leakage test is small, on the order of a few percent.*

TVA Risk Assessment

TVA, by Enclosure 4 of Reference 2, provided a risk assessment for a one-time frequency extension on SQN Unit 2. The Reference 2 risk assessment is also applicable to SQN, Unit 1.

The Reference 2 risk assessment showed the increase in the Large Early Release Frequency (LERF) to be greater than 1.0E-07/reactor year (ry) when the frequency of an Type A test was decreased from 3/10 year to 1/15 year. This increase did not meet the guidelines of RG 1.174 that defines very small changes as increases in Core Damage Frequencies (CDF) less than 10E-6/ry and increases in LERF less than 10E-7/ry. By Reference 3, NRC noted that changes that result in small increases in LERF are generally judged to be acceptable if the plant baseline total LERF is less than 10E-5/ry. Because TVA's risk assessment did not explicitly quantify the LERF for external events, the staff could not conclude that the total LERF would be less than 10E-5/ry. Based on the increase in LERF of greater than 1.0E-07/ry and without a quantification of LERF from external events, NRC granted approval for a single cycle deferment of the Type A test for SQN, Unit 2. Section 3.2 of Reference 3 indicated that all other requirements of RG 1.174 had been met for a one-time Type A test frequency of 1/15 year.

TVA's calculation of LERF as provided by Reference 2 was based on Revision 1 of the SQN Probabalistic Safety Assessment (PSA). Revision 2 of the SQN PSA has recently been completed and is described in Reference 4 (see TVA response to NRC Question 14, item 2). When Revision 2 of the SQN PSA is used with the same risk assessment methodology described in Enclosure 4 of Reference 2, the following results are obtained:

1. The estimate of population dose in Enclosure 4 of Reference 2 remains bounding,

2. the estimate of conditional containment failure probability (CCFP) in Enclosure 4 of reference 2 remains unchanged, and

3. the increase in LERF when the frequency of a Type A test is decreased from 3/10 year to 1/15 year is 6.5E-08/ry.

The increase in LERF in item 3 above is calculated simply as the product of the increase in LERF from reference 2 (2.05E-07/ry) and the ratio of CDFs from Revision 1 and Revision 2 of the SQN PSA (1.27E-5/4.02E-5). This increase in LERF is less than 1.0E-07/ry and is considered a "very small" increase in accordance with RG 1.174. Accordingly, the above described results from Revision 2 of the SQN PSA, along with the information in references 2 and 3 fully support a one-time Type A test frequency of 1/15 year for Sequoyah Units 1 and 2.

TVA Deterministic Evaluation

In addition to TVA's risk assessment, TVA's proposed TS change is based on performance history from previous Type A tests, SQN's American Society of Mechanical Engineers (ASME) Section XI, Subsection IWE examination and inspection program. TVA has also evaluated performance of additional non-destructive examinations of the steel containment vessel in the ice condenser region. A description of Type A test history, inspection results, and future examinations are provided as follows:

Test History Information

Previous Unit 1 Type A test results have shown leakage to be below the 1.0 L_a leakage limit. The performance leak rate of the last two consecutive Unit 1 tests were May 1990 (0.066%/ day = 0.2640 L_a) and December 1993 (0.1306%/day = 0.5224 La). Margins to date from previous tests indicate at least 10 percent margin (worst case).

Previous Unit 2 Type A test results have shown leakage to be below the 1.0 L_a leakage limit. The last two consecutive Unit 2 test results were March 1989 (0.20191%/day = 0.8076 L_a) and April 1992 (0.05854%/ day = 0.2342 L_a). Margins to date from previous tests indicate at least 10 percent margin (worst case).

The risk is further minimized by continued 10 CFR 50, Appendix J Type B and Type C testing. SQN's inservice inspection (ISI) program provides additional confidence in containment structural integrity and leak tightness. Accordingly, the proposed extension of the Type A test for Units 1 and 2 represent minimal risk for increased leakage.

Containment Penetrations with Mechanical Bellows

The SQN containment penetration mechanical bellows are within the scope of containment inspection and Appendix J Type A, B, or C leak testing and are two-ply laminated testable bellows. Each bellow is local leak rate tested (Type B) by pressurizing between the two plies. These bellows incorporate a screen mesh between the inner and outer plies to ensure separation is maintained. This design prevents a "pinch" from occurring at the folds and ensures that the entire space between the plies is pressurized and leak tested during Type B testing.

Following the issuance of NRC Information Notice 92-20, a representative sample of bellows was tested at all three TVA plant sites to confirm adequate separation and communication exists across the entire testable volume. This test verified flow through the annulus between the plies of the bellows.

Option B of 10 CFR 50, Appendix J would allow extended test intervals up to 120 months for Type B components, based on acceptable performance. Due to industry concerns, SQN has limited extended test intervals for bellows to 60 months. Additionally, penetrations with bellows are tested on a staggered basis such that a portion are tested each refueling outage.

A review of TVA records since 1979 has revealed no failures of these bellow tests for either SQN Unit 1 or Unit 2.

Plant Operational Performance

During power operation, instrument air from air-operated valves is vented inside containment and provides pressurization of the containment structure. Instrumentation monitors containment pressure and annunciation is provided for conditions approaching the limits allowed by the TSs. This cycling of the containment pressure during operation amounts to periodic integrated pressure testing of the containment structure at low differential pressures. Although pressurization is not as significant as would be created during a design basis accident, pressurization of containment does provide assurance that the containment structure is leak tight. The periodic cycling of containment pressure also complements the visual inspection of interior and exterior boundaries in the containment structure that may be inaccessible for visual examination.

ASME Code Examination and Inspection (Subsection IWE)

TVA engineers and inspectors perform inspection activities on the containment structure to support performance of the required Type A test. SQN also performs containment inspections in accordance with the ASME Section XI Subsection IWE ISI program. The IWE program will continue to perform inspection activities on SQN Units 1 and 2 containment through the proposed Appendix J test extension interval.

TVA's IWE program is performed in accordance with 10CFR 50.55a and ASME Section XI. Additional general visual examinations of containment are performed in accordance with Appendix J.

TVA's IWE program is based on the applicable portions of Subsections IWA and IWE of the 1992 Edition, 1992 Addenda, of ASME Section XI. The first inspection interval for the containment ISI program began September 9, 1996. The first inspection period ended September 8, 2001, in accordance with 10 CFR 50.55a(g)(6)(ii)(B)(1). The second and third inspection periods will end September 8, 2005 and September 8, 2008, respectively, in accordance with ASME Section XI. The second inspection interval for containment will begin September 9, 2008.

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 assurance that degradation of the containment structure will be detected and corrected before it can affect the structural integrity or leak tightness.

A general visual examination was performed on the Unit 1 SCV during the Cycle 10 refueling outage and Unit 2 during the Cycle 9 refueling outage. This examination was performed to meet the ASME Section XI, Subsection IWE, Table IWE-2500-1, Examination Category E-A, Item Number E1.11 requirements and the SQN TS 4.6.1.6 requirements. This general visual examination is required to be performed once per inspection period on the accessible exterior surface areas of the SCV per 10CFR50.55a(b)(2)(x)(E). The TS general visual examination is performed prior to any 10 CFR 50, Appendix J, Type A, integrated leak rate test and during two other refueling outages before the next Type A test, if the Type A test has been extended to

10 years, on the accessible interior and exterior surface areas of the SCV. There were no conditions identified during these general visual examinations that affected the leak tightness or structural adequacy of the SCV.

There is an ongoing effort to repair coatings (general rust and discoloration) on the SCV exterior side for both units. Areas identified have been visually inspected and evaluated after surface preparation. These areas were not considered suspect and did not impact the structural integrity or leak tightness of the SCV.

The 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 Section XI, Subsection IWE, 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 and thickness data showed that the remaining thickness was acceptable. Accordingly, the areas identified to date for augmented examination have not impacted the structural integrity or leak tightness of the steel containment vessel.

A VT-3 visual examination 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 Section XI, Subsection IWE, 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 adhered to the concrete and 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 in accordance with the requirements of IWE-2500(b). The examination identified conditions consisting of mild uniform corrosion, discoloration and minor pitting below the floor surface 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 original nominal wall plate thickness in this location. On Unit 2 the area between azimuth 170 degrees to 177 degrees that was examined during Cycle 9 refueling outage identified 11 areas of pitting and during 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 Engineering 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.

TVA feels the actions described above will arrest any SCV degradation and will preserve containment integrity beyond the 5-year extension interval.

Future Code Inspections

A VT-3 examination to meet the ASME Section XI (i.e., Subsection IWE, Table IWE-2500-1, Examination Category E-A, item number E1.12 requirement to examine the accessible surface areas at the end of the interval from one side of the SCV) is scheduled during the Units 1 and 2 Cycle 15 refueling outages (third period of the second ISI interval).

The total estimated area of the SCV from the base concrete floor slab to the top of the SCV on the exterior side is approximately 61,000 square feet. The inaccessible surface area is estimated to be approximately 6800 square feet (1800 square feet of this area due to insulation attached to the SCV, and 5000 square feet due to ventilation duct work and electrical cable trays). It is estimated that 89 percent of the SCV exterior side is examined each period for IWE and will be VT-3 examined during the Cycle 15 refueling outage.

The area below the floor is not included in the area for examination because the embedded metal liner and concrete base slab are exempt from examination in accordance with IWE-1220(b) and IWL-1220(b) of Subsections IWE and IWL of ASME Section XI.

Additional Inspections

During the Unit 1 Cycle 10 and Unit 2 Cycle 9 refueling outages ultrasonic thickness measurements were taken at three locations (2-foot x 3-foot 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. These inaccessible areas are not specifically susceptible to degradation, however, TVA plans to perform additional inspections in these areas 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. Degraded areas will be evaluated by Engineering for inclusion under the augmented program per IWE-1240 of Subsection IWE of Section XI of ASME. 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 are 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 - elevation- 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 outages 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 Section XI code.

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 signs of active corrosion. There were no detrimental flaws or significant degradation noted during the examination. The SCV at these locations was recoated.

Related Relief Requests

TVA Request for Relief CISI-01 was submitted and approved by NRC for Examination Category E-D, seals and gaskets. TVA's CISI-01 included alternative requirements for ensuring leak tightness of seals and gaskets. Alternative leak testing is performed in accordance with 10 CFR Part 50, Appendix J (Type B testing). A Type B test is performed at least once each ISI interval as required by 10 CFR Part 50, Appendix J, and during each disassembly and re-assembly sequence. As identified in TVA's request for relief, there are no examinations of seals and gaskets which will be performed in accordance with Subsection IWE.

TVA Request for Relief CISI-04 was submitted and approved by NRC for Examination Category E-G, bolting. TVA's CISI-04 pertained to bolt torque and tension tests (Item No. E8.20). The CISI-04 was approved to waive performance of bolt torque and tension tests for bolted connections that have not been disassembled and reassembled during the inspection interval. The VT-1 visual examinations required by Item No. E8.10 of Examination Category E-G, will continue to be performed. Examinations required by Item E8.10 were not deferred during the first period.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

TVA's proposed revision to the technical specifications (TSs) for Sequoyah Nuclear Plant (SQN) Units 1 and 2, adds notation to Section 6.8.4.h, "Containment Leakage Rate Testing Program," to allow a 1-time, 5-year extension to the current 10-year interval for 10 CFR 50, Appendix J, Type A testing.

TVA has evaluated the proposed change and concluded that it does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation of the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," and as discussed below:

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No. The proposed change for extending Type A test frequency does not significantly increase the probability of an accident previously evaluated since the change is not a modification to plant systems, nor a change to plant operation that could initiate an accident.

TVA performed an evaluation of the risk significance for the proposed increase to the SQN Units 1 and 2 Type A test frequency. The results of the TVA risk evaluation indicates that the increase in Large Early Release Frequency (LERF) remains below the level of risk significance defined in NRC

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." TVA's evaluation indicates that the increase in frequency for all releases (small, large, early and late) and the increase in radiation dose to the population is also non-risk significant.

The proposed test interval extension does not involve a significant increase in the consequences of an accident. Research documented in NUREG-1493 determined that generically, very few potential containment leakage paths fail to be identified by Type A tests. An analysis of 144 Type A test results, including 23 failures, found that no failures were due to containment liner breach. The NUREG concluded that reducing the Type A test frequency to once per 20 years would lead to an imperceptible increase in risk. Furthermore, the NUREG concluded that Type B and C testing provides assurance that containment leakage from penetration leak paths (i.e., valves, flanges, containment air-locks) identify any leakage that would otherwise be detected by the Type A tests.

In addition to the NUREG conclusions, TVA's American Society of Mechanical Engineers (ASME) IWE program performs containment inspections in order to detect evidence of degradation that may affect either the containment structural integrity or leak tightness. In addition to the IWE examinations, TVA will perform additional non-destructive examinations of the steel containment vessel in the ice condenser region (inaccessible areas) at various elevations. These additional non-destructive examinations will provide added assurance of containment integrity during the 5-year extended interval.

Accordingly, TVA's proposed extension of the Type A test interval does not increase the probability or consequences of an accident previously evaluated.

2. **Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

Response: No. The proposed change to extend the Type A test interval does not create the possibility of a new or different type of accident because there are no physical changes made to the plant or plant equipment governing normal plant operation. There are no changes to the operation of the plant that would introduce a new failure mode creating the possibility of a new or different kind of accident.

TVA will perform additional non-destructive examinations of the steel containment vessel in the ice condenser region (inaccessible areas)at various elevations. These additional non-destructive examinations will provide added assurance of containment integrity during the 5 year extended interval.

3. **<u>Does the proposed change does not involve a significant reduction in a margin of safety.</u>**

Response: No. The proposed change to extend the Type A test interval will not significantly reduce the margin of safety. A generic study documented in NUREG-1493 indicates that extending the Type A leak test interval to 20 years would result in an imperceptible increase in risk to the public. The NUREG also found that, generically, the containment leakage rate contributes a very small amount to the individual risk and that the decrease in the Type A test frequency would have a minimal affect on risk because most potential leakage paths are detected by Type C testing.

Previous Type A leakage tests conducted on SQN Units 1 and 2 indicate that leakage from containment have been less than the 10 CFR 50, Appendix J leakage limit of 1.0 L_a. A review of the previous Type A test results indicate a stable trend with a 10 percent margin below the 1.0 L_a leakage limit.

Accordingly, these test results, in conjunction with the research findings from NUREG-1493, provide assurance that the proposed extension to the Type A test interval would not significantly reduce the margin of safety.

Based on the above, TVA concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 <u>Applicable Regulatory Requirements/Criteria</u>

Title 10 of the *Code of Federal Regulations*, Part 50 (10 CFR 50), Section 50.54(o) and 10 CFR 50, Appendix J, contain primary reactor containment leakage test requirements for water-cooled power reactors. The 10 CFR 50, Appendix J requirements are divided into Option A (prescriptive requirements) and Option B (performance-based requirements). The Option B rulemaking in 1995 provided licensees with an alternative approach to determine test intervals for containment leakage rate testing. The Option B approach was based on system and component performance in lieu of compliance with prescriptive requirements. NRC RG 1.163, September 1995, "Performance-Based Containment Leak Test Program," was developed as a method acceptable to the NRC staff for implementing Option B. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) document NEI 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) leak 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. TVA, by letter dated December 8, 1995, submitted a license amendment (TS Change 95-24) to request use of Option B at SQN that was subsequently approved by NRC letter dated February 5, 1996.

The SQN TSs (Section 6.8.4.h) currently contain program requirements for implementing leak rate testing of containment as required by 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions. Section 6.8.4.h further states that this program shall be in accordance with the guidelines contained in RG 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by exceptions set forth in the site implementing instructions. Accordingly, the SQN Type A test interval is currently prescribed in TSs as once per 10 years.

The proposed change is submitted on a risk informed basis as described in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" and makes use of Revision 2 of the SQN Probabilistic Safety Assessment (PSA). TVA has determined that the resultant increase in LERF for the proposed change is "very small" [i.e., less than 1.0E-07/reactor year (y)] and satisfies the RG 1.174 criteria.

TVA's American Society of Mechanical Engineers (ASME) IWE program performs containment inspections in order to detect evidence of degradation that may affect either the containment structural integrity or leak tightness. In addition to the IWE examinations, TVA will perform additional non-destructive examinations of the steel containment vessel in the ice condenser region (inaccessible areas) at various elevations. These additional non-destructive examinations will provide added assurance of containment integrity during the 5-year extended interval.

In conclusion, based on the considerations discussed above, (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 ENVIRONMENTAL CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. NRC letter to TVA dated February 5, 1996, "Issuance of Technical Specification Amendments for the Sequoyah Nuclear Plant, Units 1 and 2, (TAC Nos. M94239 and M94240)(TS 95-24)"

2. TVA letter to NRC dated October 9, 2001, SQN - Unit 2 - Technical Specification (TS) Change No. 01-10, "One-time Frequency Extension for Type A Test (Containment Integrated Leak Rate Test [CILRT])"

3. NRC letter to TVA dated May 7, 2002, "Issuance of Technical Specification Amendment for the Sequoyah Nuclear Plant, Unit 2 - Regarding One-Time Extension of Containment Type A Pressure Test (TAC No. MB3275) (TS 01-10)"

4. TVA letter to NRC dated August 31, 2001, Sequoyah Nuclear Plant (SQN) - Response to Request for Additional Information (RAI) Regarding Risk Informed Inservice Inspection (RI-ISI) Program"

5. TVA letter to NRC dated April 11, 2002, Sequoyah Nuclear Plant (SQN) - Additional Information for Technical Specification (TS) Change 01-10, "One Time Frequencey Extension for Type A test (Containment Integrated Leak Rate Test [CILRT])"

ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 and 2

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE
MARKED PAGES

I. AFFECTED PAGE LIST

Unit 1

6-10a

Unit 2

6-9

II. MARKED PAGES

See attached.

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

h. Containment Leakage Rate Testing Program

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

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

Leakage rate acceptance criteria are:

a Containment overall leakage rate acceptance criteria is $\leq 1.0\ L_a$ During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0\ 60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are.

1) Overall air lock leakage rate is $\leq 0\ 05\ L_a$ when tested at $\geq P_a$.

2) For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 6 psig for at least two minutes.

The provisions of SR 4.0 2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

i Configuration Risk Management Program

The Configuration Risk Management Program (CRMP) provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The program applies to Technical Specification structures, systems, or components for which a risk-informed allowed outage time has been granted. The program shall include the following elements:

a Provisions for the control and implementation of a Level 1 at-power internal events PRA-informed methodology. The assessment shall be capable of evaluating the applicable plant configuration.
b. Provisions for performing an assessment prior to entering the Limiting Condition for Operation (LCO) Action for preplanned activities.
c. Provisions for performing an assessment after entering the LCO Action for unplanned entry into the LCO Action.
d Provisions for assessing the need for additional actions after the discovery of additional equipment out of service conditions while in the LCO Action.
e. Provisions for considering other applicable risk significant contributors such as Level 2 issues and external events, qualitatively or quantitatively.

6 8.4 f Radioactive Effluent Controls Program (Cont)

of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the SITE BOUNDARY SHALL BE LIMITED to the following:

1. For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin, and
2. For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/year to any organ.

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radio-nuclides in particulate form with half-lives greater than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50, and

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

g Radiological Environmental Monitoring Program (DELETED)

h Containment Leakage Rate Testing Program

fall 2003

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 ~~spring 2002~~ containment integrated leakage rate (Type A) test may be deferred up to ~~one cycle~~ but no later than ~~fall 2003~~

4 years

spring 2007

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 12.0 psig

The maximum allowable containment leakage rate, L_a, at P_a, is 0.25% of the primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria is $\leq 1.0\ L_a$ During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
SEQUOYAH PLANT (SQN)
UNITS 1 AND 2

TVA COMMITMENTS

Unit 1

1. Additional ultrasonic thickness measurements on the Steel Containment Vessel (SCV) inaccessible areas will be performed during the Unit 1 Cycle 12 refueling outage, to assess potential degradation.

2. 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.

Unit 2

1. Additional ultrasonic thickness measurements on the SCV inaccessible areas will be performed during the Unit 2 Cycle 12 refueling outage, to assess potential degradation.

2. The inaccessible SCV exterior area behind the 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 outages.