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

March 13, 2002

10 CFR 50.90

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

Gentlemen:

In the Matter of) Docket No. 50-328
Tennessee Valley Authority)

SEQUOYAH NUCLEAR PLANT (SQN) - RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING TECHNICAL SPECIFICATION (TS) CHANGE 01-10, "ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST (CONTAINMENT INTEGRATED LEAK RATE TEST [CILRT])"

Reference:

1. Letter from TVA to NRC dated October 9, 2001, "Sequoyah Nuclear Plant (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. Letter from NRC to TVA dated February 14, 2002, "Request for Additional Information on Technical Specification Change No. 01 10, 'One-Time Frequency Extension for Type A Test (Containment Integrated Leak Rate Test [CILRT])'(TAC No. MB3275)"

3. Letter from TVA to the 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

This letter provides the additional information requested by the reference 2 letter to support NRC review of SQN TS Change 01-10. The enclosure provides TVA responses to the NRC staff questions.

Based on discussion with NRC staff concerning TVA's Probabalistic Safety Assessment (PSA) calculations, any reduction in the calculation of delta Large Early Release Frequency (LERF) from "small" to "very small" is preferred for Type A test interval extensions. TVA has evaluated SQN's PSA calculations that were submitted in Enclosure 4 to TVA's reference 1 letter. It may be noted that the calculations are based on Revision 1 of the SQN PSA. Revision 2 of the SQN PSA has recently been completed. Revision 2 of the SQN PSA is described in TVA's response to Question 14 (Item 2) contained in Reference 3. The mean core damage frequency (CDF) of the Revision 2 SQN PSA is 1.25E-5/year. Using Revision 2 of the SQN PSA and the same methodology described in Enclosure 4 of TVA's reference 1 letter, the increase in LERF when the frequency of an CILRT is decreased from 3/10 year to 1/15 year is less than 1.0E-7/year. Accordingly, this increase in LERF is a "very small" increase in LERF per Regulatory Guide 1.174.

This letter is being sent in accordance with NRC RIS 2001-05. Please direct questions concerning this issue to 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 13th day of March 2002

Notary Public

My Commission Expires May 9, 2005

Enclosures
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ENCLOSURE

**RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
SEQUOYAH (SQN) TECHNICAL SPECIFICATION CHANGE NO. 01-10,
UNIT 2 ONE-TIME FREQUENCY EXTENSION FOR TYPE A TEST
CONTAINMENT INTEGRATED LEAK RATE TEST**

NRC Question 1:

On page 8 of Enclosure 4 to your October 9, 2001, letter, you make reference to "containment liner" in several places. However, in the background description in your Enclosure 1, you describe the containment as a "freestanding steel vessel." Please clarify this discrepancy and discuss whether this difference in containment configuration would make any difference in the calculations related to a preexisting leak.

TVA Response:

The discussion on Page 8 of Enclosure 4 develops the equations used to determine the probability of a preexisting containment leak. The preexisting leak probability consists of essentially two terms:

- the probability of a preexisting leak through a containment penetration, and

- the probability of a preexisting leak in the containment vessel (free standing steel shell).

It is the later that is referred to as the containment liner on page 8 of Enclosure 4. The reference to the containment liner is simply to distinguish the two preexisting leakage paths.

The description of the containment vessel as a free standing steel structure is provided in Section 3.8.2.1 of the SQN Final Safety Analysis Report (FSAR) and is shown on Figure 3.8.2-1 of the FSAR. The containment was evaluated as a free standing steel structure in Enclosure 4. Therefore, there is no difference in the containment configuration between that evaluated in the risk analysis contained in Enclosure 4 and the SQN Unit 2 containment configuration.

NRC Question 2:

In describing test history information in your Enclosure 1, you state, "[P]revious Unit 2 Type A test results have shown leakage to be below the 1.0 L_a leakage limit." Per the guidelines in Nuclear Energy Institute (NEI) Report NEI 94-01, an acceptable performance history (for extending Type A test interval) is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage rate is less than 1.0 L_a. The method of determining the performance leakage rate is provided in Section 9.2.3 of the NEI report. Please provide the values of performance leakage rates for the last two consecutive Type A CILRT tests performed at SQN Unit 2.

TVA Response:

The performance leak rate of the last two consecutive Unit 2 tests were:

March, 1989 CILRT - 0.20191 %/day = 0.8076 L_a
April, 1992 CILRT - 0.05854 %/day = 0.2342 L_a

NRC Question 3:

Please provide a description of the ISI that provides assurance that in the absence of a CILRT for 15 years, the containment structural and leak-tight integrity will be maintained, beyond the partial description provided on page E1-6 of your letter. Please identify the Edition and Addenda of the American Society of Mechanical Engineers (ASME) Code, Section XI, used for containment ISI together with the start dates of the first and second SQN Unit 2 containment ISI intervals, and the future inspection periods.

TVA Response:

Containment Inservice Inspection (ISI) is performed in accordance with 10 CFR 50.55a and ASME Section XI. Additional general visual examinations of containment are performed in accordance with Appendix J. No additional ISI of containment is performed.

The containment ISI 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.

NRC Question 4:

IWE-1240 of Subsection IWE of Section XI of the ASME Boiler and Pressure Vessel Code requires you to identify the surface areas requiring augmented examinations. Please provide the locations of the containment surfaces which you have identified as requiring augmented examination and a summary of findings of the augmented examinations you performed for SQN Unit 2 in these areas.

TVA Response:

The SQN Unit 2 augmented examination areas identified are at chilled water system penetrations X-64, X-65, X-66, X-67 on the outboard side. The nozzle reinforcement on the outboard side of the penetrations had severe corrosion due to moisture absorbed and held against the nozzle reinforcement by black foam insulation. These areas were ultrasonically examined and thickness data showed that the remaining thickness was acceptable and did not exceed the design basis requirement. Accordingly, the areas identified to date for augmented examination have not impacted the structural integrity or leak tightness of the steel containment vessel.

It may be noted that these areas were recoated and the ultrasonic thickness examinations repeated to establish baseline data for successive examinations to be conducted in accordance with IWE-1240.

NRC Question 5:

For the examination of seals and gaskets, and testing of bolts associated with the primary containment pressure boundary (Examination Categories E-D and E-G), you had requested relief from the requirements of the Code. As an alternative, you plan to examine them during the leak rate testing of the primary containment. With the flexibility provided in Option B of Appendix J for Type B and Type C testing (as per NEI 94-01 and RG 1.163), and the extension requested in this amendment for Type A testing, please provide the examination schedule for examining and testing seals, gaskets, and bolts related to the integrity of the containment pressure boundary.

TVA Response:

The Request for Relief CISI-01 was submitted to and approved by NRC for Examination Category E-D, seals and gaskets. The alternate examination for the request for relief was that the leak tightness of seals and gaskets are tested in accordance with 10 CFR Part 50, Appendix J Type B testing. Type B tests are performed at least once each ISI interval as required by 10CFR Part 50, Appendix J, in addition to the Type B tests performed prior to disassembly and after reassembly. As identified in the request for relief, there are no examinations of seals and gaskets which will be performed in accordance Subsection IWE. No additional alternatives were included with the request for relief.

The Request for Relief CISI-04 was submitted and approved by NRC for Examination Category E-G bolting. CISI-04 pertained to bolt torque and tension tests (Item No. E8.20). The VT-1 visual examinations required by Item No. E8.10 of Examination Category E-G will continue to be performed. Examination deferrals were not utilized during the first inspection period.

NRC Question 6:

Inspections of some reinforced and steel containments have indicated degradation from the uninspectable (embedded) side of the steel shell and liner of primary containments. In case of SQN-2, the major uninspectable areas would be those behind the ice baskets and part of the shell embedded in the basemat. Please discuss how potential leakages due to age-related degradation mechanisms described above are factored into the risk-informed assessment related to the CILRT interval extension. Please note that, as discussed in NUREG-1493, it takes only 3.5 sq. in. of leak area for the entire containment (100%) air to leak in 24 hrs.

TVA Response:

As discussed in the response to Question 1 of this RAI, the equations for the probability of a preexisting containment leak as a function of LLRT and ILRT intervals, are developed on page 8 of Enclosure 4 of the T/S Submittal. Referring to page 8 of Enclosure 4, it can be seen that the probability of a preexisting containment leak consists of factors that account for a preexisting leak in:

1. a containment penetration (i.e., through the isolation valve/device), and,

2. the free standing steel shell (i.e., through the welds connecting the containment shell steel plates, see Figure 3.8.2-9 of the SQN FSAR).

The rate of occurrence of a preexisting leak in a containment penetration $-\lambda_p$, is based on the information in NUREG-1493.

The rate of occurrence of a preexisting leak in the containment liner $-\lambda_l$, is estimated in the Enclosure 4 evaluation as equivalent to the mean failure rate for a storage tank rupture (see page 8 of Enclosure 4). Therefore, the Enclosure 4 analysis does implicitly account for potential leakage due to age-related degradation mechanisms as explained below:

Factors to account for age-related failures (e.g., corrosion, bearing wear, spring fatigue, etc.) are not specifically included in risk assessments because it is assumed that structures, system and components are designed and maintained such that environmental and wear induced failures do not occur. However, when age-related failures do occur, they are included in the basis for the probabilities used in the risk assessment.

Based on this, the degradation in the containment steel shell of some plants noted in this question should not be included in determining the probability of a preexisting leak because for these degradations, the containment pressure boundary was intact. The only preexisting leaks found to date by ILRTs are those due to a failure-to-seal of an isolation device. There have been no occurrences of a containment shell leak detected by ILRTs. This is why the NUREG-1493 analysis and the risk assessment methodology used by the NRC Staff to date, considers the probability of a preexisting leak in the containment steel shell to be insignificant compared to the probability of a failure-to-seal of an isolation device.

In summary, the potential leakage due to age-related degradation mechanisms [i.e., containment shell corrosion] are factored into the risk-informed assessment [Enclosure 4] that supports the SQN Unit 2 containment ILRT interval extension. This is accounted for by assigning a failure probability to the containment shell. The assurance that any age-related degradation will not result in a preexisting leak in the SQN-2 containment steel shell is based on the following:

1. Recent inspections of the corrosion susceptible areas of the steel shell found no significant degradation or wear as described below:

   A VT-3 visual examination was performed during the last refueling outage on Unit 2 (U2C10). The inspection looked at the steel containment vessel (SCV) interior surface in the vicinity of the moisture barrier at the interface of the SCV and raceway floor (see Figure 6.2.1-63 of the FSAR). This inspection was a result of a VT-3 visual examination of the moisture barrier integrity. The moisture barrier and fiberglass filler in the crevice was removed and the SCV surface was examined in accordance with the requirements of Table IWE-2500-1 Examination Category E-A and (IWE-2500(b)). The examination was performed from 12 inches above the floor to 6 inches below the floor interface. The examination identified no detrimental flaws or significant degradation of the SCV. 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. This polyurethane elastometric material will serve to fill the crevice area, act as the protective coating for the SCV, and provide a leak tight barrier.

In addition, the SCV located in seal area of the ice condenser (again, see Figure 6.2.1-3 of the SQN FSAR) was ultrasonically examined during U2C9 refueling outage at three locations (2' x 3' grids). All SCV locations examined were at nominal wall thickness with no evidence of degradation.

2. The containment is continuously being pressurized by instrument air during power operation and must be routinely vented to maintain containment pressure within T/S. As discussed on page E1-6, pressurization is not as significant as would be created during a design basis accident, however this pressurization of containment does provide assurance that the containment structure is leak tight.

   The median failure pressure for the SQN-2 containment is estimated to be 70 psig (see Figure 4.4-1 of the SQN IPE). This compares to a maximum pressure during a DBA of 12 psig. The large difference in the maximum containment pressure produced during a DBA and the median containment failure pressure implies that only large flaws in the steel shell will result in a through-wall crack (leak) in containment during a DBA. A corrosion induced flaw could not develop in a sufficiently uniform manner such that some portion of the flaw would not be through-wall and detectable by this routine pressurization.

The overall conclusion of NUREG-1493, NRC Staff analyses, and the analysis in Enclosure 4 of TVA's TS change 01-10 submittal are consistent and are summarized below:

1. Type B & C tests detect essentially all preexisting containment leaks.
2. Type A tests (ILRTs) have not detected leakage through a containment shell.
3. The probability of a preexisting leak in the containment steel shell is negligible compared to a preexisting leak in a containment isolation device.
4. The probability that age-related degradation which could lead to a preexisting leak in the containment steel shell is negligible compared to a preexisting leak in a containment isolation device.
5. Preexisting containment leaks are not risk-significant. Large early releases are dominated by severe accident phenomena which result in containment failure and releases which bypass containment.