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

November 1, 2005

10 CFR 50.55a

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of )
Tennessee Valley Authority) Docket No. 50-327

**SEQUOYAH NUCLEAR PLANT (SQN) UNIT 1, RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION RELATED TO ONE-CYCLE REQUEST TO EXTEND THE SECOND 10-YEAR INSERVICE INSPECTION INTERVAL FOR REACTOR VESSEL WELD EXAMINATION - REQUEST NO. 1-ISI-27**

Reference: 1) NRC letter to TVA dated October 24, 2005, Sequoyah Nuclear Plant, Unit 1 - Request for Additional Information to Support Authorization to Extend the Second 10-Year Inservice Inspection Interval for Reactor Vessel Weld Examination (TAC NO. MC7561)

2) TVA letter to NRC dated July 8, 2005, "Sequoyah Nuclear Plant (SQN) Unit 1 - Subject: American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection (ISI) Program Relief Request to Extend the Second 10-Year ISI Interval For Unit 1 Reactor Vessel (RV) Weld Examination - Request No. 1-ISI-27"

By the Reference 2 letter, TVA requested NRC approval, pursuant to 10 CFR.50.55a(a)(3)(i), for the use of an alternative to the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Paragraph IWB-2412, Inspection Program B.

NRC staff requested additional information in Reference 1 to complete review of TVA's submittal. Enclosed is the additional information as requested.

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If you have any questions concerning this matter, please contact Jim Smith at (423) 843-6672.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 1st day of November, 2005.

Sincerely,

for

P. L. Pace
Manager, Site Licensing and
Industry Affairs

Enclosure
cc (Enclosure):
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ENCLOSURE

TENNESSEE VALLEY AUTHORITY (TVA)
SEQUOYAH NUCLEAR PLANT (SQN), UNIT 1

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

***NRC Request***

*In your July 8, 2005, request for authorization to extend the second 10-year ISI interval for reactor pressure vessel retaining weld examinations, you stated that the technical justification for your request is consistent with the guidance provided in a January 27, 2005, letter from the U.S. Nuclear Regulatory Commission to Westinghouse Electric Company (Summary of Teleconference with the Westinghouse Owners Group Regarding Potential One Cycle Relief of Reactor Pressure Vessel Shell Weld Inspections at Pressurized Water Reactors Related to WCAP-16168-NP, "Risk Informed Extension of Reactor Vessel In-Service Inspection Intervals"). Item number six of this guidance is repeated below.*

> *The licensee could then provide a discussion of how, based on its plant operational experience, fleet-wide operational experience and plant characteristics, the likelihood of an event (in particular, a significant pressurized thermal shock event) over the next operating cycle which could challenge the integrity of the reactor vessel pressure vessel (RPV), if a flaw was present, is very low.*

*Section 5.5 of your submittal includes general statements indicating that the likelihood of pressurized thermal shock (PTS) events is small. The submittal briefly describes the strategy of the Sequoyah Nuclear Plant, Unit 1's emergency operating procedures intended to allow the operators to identify the onset of PTS conditions and provide the steps required to mitigate any cold pressurization challenges to the reactor vessel integrity.*

*The staff is reevaluating the risk from PTS events in a study done to develop a technical basis for revising Title 10 of the Code of Federal Regulations, Part 50, Section 61.*

*Although the staff has not yet completed its evaluation, the current results indicate that the following three types of accident sequences cause the more severe PTS events, and thereby dominate the risk. Please describe the characteristics of your plant (design and operating procedures) that provide assurance that the likelihood of a severe PTS event over the next operating cycle which could challenge the integrity of the RPV, if a flaw was present, is very low.*

*Sequence 1*

> *Any transient with reactor trip followed by one stuck-open pressurizer safety relief valve that re-closes after about one hour. Severe PTS events also require the failure to properly control high head injection.*

*Sequence 2*

> *Large loss of secondary steam from steam line break or stuck-open atmospheric dump valves. Severe PTS events also require the failure to properly control auxiliary feedwater flow rate and destination (e.g., away from affected steam generators) and failure to properly control high pressure injection.*

*Sequence 3*

> *Four to nine-inch loss-of-coolant accidents. Severity of PTS event depends on break location (worst location appears to be in the pressurizer line) and primary injection systems flowrate and water temperature.*

**TVA Response**

Design Features

Pressurized thermal shock (PTS) occurs when a severe primary system overcooling event is followed by repressurization of the reactor vessel. The propagation of crack-like defects in the reactor vessel wall during these events is related to a reduction in the fracture toughness of the vessel material due to neutron flux exposure during plant operation. As long as the fracture resistance of the reactor vessel material remains relatively high, system

pressurization following an overcooling event will not cause a failure of vessel integrity.

To challenge reactor vessel integrity and core cooling during a PTS event, a number of contributing factors must be present. These factors include 1) a reactor vessel flaw of correct size to propagate, 2) material with a high copper content, 3) a relatively high level of irradiation, 4) a severe overcooling transient with repressurization, and 5) a resulting crack of such size and location that the ability of the reactor vessel to maintain core cooling is affected.

To address these factors, the final PTS rule issued in 10 CFR 50.61 dated July 23, 1985, established a PTS screening criteria for all domestic operation PWR plants based on a PTS reference temperature ($RT_{PTS}$). The rule required a $RT_{PTS}$ to be calculated for all vessel beltline materials such that it could be compared to a conservative screening criterion. As defined by the rule, the $RT_{PTS}$ is sensitive to vessel material properties (i.e., the initial material nil-ductility transition reference temperature [$RT_{NDT}$]as well as copper and nickel content) and the neutron irradiation exposure level. The $RT_{PTS}$ values were established using beltline material properties and material irradiation for the inner surfaces of the beltline materials projected to the end of plant life.

The PTS screening criteria established by 10 CFR 50.61 is as follows:

$RT_{PTS}$ = 270°F for plates, forgings, axial welds
$RT_{PTS}$ = 300°F for circumferential weld materials

For those materials with $RT_{PTS}$ values less than the screening criteria, no additional evaluation was required.

The beltline materials in the SQN Unit 1 reactor vessel consist of an intermediate forging, lower forging and an intermediate-to-lower forging circumferential weld. The $RT_{PTS}$ values were recently evaluated for reactor vessel irradiation through the end of the current operating license (32 Effective Full Power Years [EFPYs]) and a projected 20 year plant license extension [48 EFPY]). This evaluation is summarized in Appendix B of Reference 1 (which was submitted to NRC by Reference 2). The

evaluation established the following $RT_{PTS}$ values for the SQN Unit 1 reactor vessel beltline materials.

| | $RT_{PTS}$ at 32 EFPY | $RT_{PTS}$ at 48 EFPY |
|---|---|---|
| Intermediate Shell Forging | 209°F | 221°F |
| Lower Shell Forging | 231°F | 241°F |
| Circumferential Weld Metal | 204°F | 220°F |

The actual reactor vessel irradiation projected for the end of the requested inspection period extension is 18.1 EFPY

Since these results indicate that the SQN Unit 1 reactor vessel beltline materials meet the current PTS screening criteria for reactor vessel irradiation projected through the end of the current operating license (and beyond) and the expected reactor vessel irradiation at the end of the requested extension period will be significantly less than that evaluated, the potential for a PTS event to challenge the integrity of the reactor vessel during the extension period is very low.

Additionally, SQN Unit 1 was one of the plants compared to the plants analyzed in the PTS generalization study discussed in the Request for Additional Information (RAI) (see Reference 3). Additional information regarding SQN Unit 1 design features as they relate to PTS transients is documented in Appendix A of Reference 3.

Operating Procedures

As discussed in Section 5.5 of TVA's July 8, 2005, submittal, SQN Unit 1 has implemented Emergency Operating Procedures (EOPs) and operator training to mitigate a PTS event. The EOPs are based on the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs) for Westinghouse-designed nuclear steam supply systems. Each of the three accident sequences discussed in the RAI involve a reactor trip. Upon receipt of a reactor trip, operators implement Emergency Procedure E-0, "Reactor Trip or Safety Injection." This procedure requires immediate verification of a reactor trip and emergency core cooling system actuation. Following performance of the immediate actions, additional steps are taken to determine the availability of a secondary heat sink, check main steam line isolation, check reactor coolant pump trip criteria, and monitor primary system temperature.

For the Sequence 1 event described in the RAI (open pressurizer safety valve), Step 10 of E-0 requires monitoring of the Critical Safety Function Status Trees and the transition to Emergency Procedure E-1, "Loss of Reactor or Secondary Coolant." Similarly, Step 11 of the procedure requires monitoring of the status trees and transition to Emergency Procedure E-2, "Faulted Steam Generator Isolation" for the Sequence 2 event (secondary pressure boundary not intact). The Sequence 3 event (primary system small break loss of coolant accident) is addressed in Step 13 of the procedure which also requires monitoring of the status trees and the transition to Emergency Procedure E-1.

The Critical Safety Function Status Trees (Function Restoration Procedure FR-0) provide for the monitoring of critical safety functions to ensure the integrity of the plant fission product barriers. Status trees have been developed to address subcriticality (F-0.1), core cooling (F-0.2), heat sink (F-0.3), containment integrity (F-0.5), primary system inventory (F-0.6), as well as pressurized thermal shock (F-0.4). The status trees for PTS (F-0.4) are used to identify the approach to a PTS condition based on monitored primary system temperatures and pressures. The status trees establish red, orange, yellow or green operator action paths based on plant conditions. If conditions exist which establish a red or orange action path, the status tree will be monitored continuously. If no red or orange path conditions are present, then status tree monitoring is reduced to once every 10 to 20 minutes unless a significant change in plant status occurs. Status tree monitoring is only terminated when the plant reaches cold shutdown (Mode 5) conditions or emergency procedures are exited and automatic safety injection is armed.

The PTS Safety Function Status Tree directs operator action to Function Restoration Procedure FR-P.1, "Pressurized Thermal Shock" for all red and orange path conditions. This procedure contains the detailed instructions for the PTS mitigation strategies of 1) terminating the primary system cooldown, 2) terminating emergency core cooling system flow (if proper criteria are met), 3) depressurization of the primary system, 4) establishment of stable primary system conditions in the normal operating range, and 5) implementation of a thermal "soaking" period prior to any cooldown outside of the normal operating region.

In summary, for the three types of accident sequences which result in the more severe PTS events, current TVA Emergency Operating Procedures promptly direct operators to monitor primary conditions for PTS. Based on actual plant conditions, the procedures provide for the proper level of monitoring and operator action required to mitigate conditions which could potentially result in reactor vessel PTS. Operator training ensures that operators are sufficiently knowledgeable and capable of implementing these procedures. Based on operator actions performed in accordance with these procedures, the likelihood of a severe PTS event over the next operating cycle is very low.

References


1. Topical Report No. WCAP-15293, Revision 02, "Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," July 2003.

2. TVA Letter to NRC, "Sequoyah Nuclear Plant Units 1 and 2 - Updated Pressure-Temperature Limit Reports (PTLRs) and Topical Reports for SQN Technical Specification Change No. 00-14" dated September 3, 2003 (S64 030903 801).

3. Sandia National Laboratories Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants" dated December 14, 2004.