TVA

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December 22, 2010

10 CFR 50.55a

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

Sequoyah Nuclear Plant, Unit 2
Facility Operating License No. DPR-79
NRC Docket No. 50-328

Subject: **Request to Employ Alternative Testing to IWE-5221 Requirements in American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI (2001 Edition with 2003 Addenda) - Request Number 2-APPJ-1**

Pursuant to 10 CFR 50.55a(a)(3)(i), the Tennessee Valley Authority (TVA) requests NRC approval of the following request to utilize alternative testing instead of that mandated by the IWE-5221 requirements in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. TVA's IWE Program is based on the 2001 Edition of the ASME Code with the 2003 Addenda. TVA's enclosed request for relief proposes alternative test methods (pneumatic leakage test) for repair activities associated with the Sequoyah Nuclear Plant (SQN), Unit 2, steel containment vessel following the SQN, Unit 2 Cycle 18 steam generator replacement outage in fall 2012. The details of the 10 CFR 50.55a request are presented in the Enclosure.

To support the SQN, Unit 2, steam generator replacement project scheduled to start in October 2012, TVA requests approval of the proposed alternative by March 2012.

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There are no commitments contained in this letter. Please direct any questions concerning this matter to Rod Cook at (423) 51-2834.

Respectfully,

R. M. Krich

Enclosure: 10 CFR 50.55a Request Number 2-APPJ-1

cc (Enclosure):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

ENCLOSURE

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMERICAN SOCIETY OF MECHANICAL ENGINEERS CODE RELIEF REQUEST

10 CFR 50.55a REQUEST NUMBER 2-APPJ-1

10 CFR 50.55a Request Number 2-APPJ-1

Request for Proposed Alternative
in Accordance with 10 CFR 50.55a(a)(3)(i)

1. American Society of Mechanical Engineers (ASME) Code Component Affected

Sequoyah Nuclear Plant (SQN), Unit 2, Steel Containment Vessel (SCV).

2. Applicable Code Edition and Addenda

ASME Section XI 2001 Edition with the 2003 Addenda are applicable to the Inservice Inspection (ISI) program interval for this request.

3. Applicable Code Requirement

The applicable ASME Code requirement is Section XI, Paragraph IWE-5221, and requires that that an appropriate 10 CFR 50, Appendix J test be performed following a repair or modification of the pressure retaining boundary. Specifically, the ASME Code requires that repair/replacement activities performed on the pressure retaining boundary of Class MC or Class CC components shall be subjected to a pneumatic leakage test in accordance with the provisions of Title 10, Part 50 of the Code of Federal Regulations, Appendix J, Paragraph IV.A, which states that a Type A, Type B, or Type C test be performed, as appropriate, for the repaired or modified pressure boundary component.

4. Reason for Request

For the repairs that will be made to the SCV following the replacement of the SQN, Unit 2, steam generators (SGs), performance of an integrated leak rate test provides no additional assurance of containment integrity as compared to local leak rate testing of the restoration welds with containment pressurized to P_a (12.0 pounds per square inch gauge [psig]). However, performing an integrated leak rate test (Type A test) requires additional schedule time, manpower, dose, and test instrumentation to be installed throughout containment. The integrated leak rate test takes longer to perform and requires stopping most work activities inside of containment for several days. The integrated leak rate test does not provide any additional assurance of the quality of the repair welds of the SCV.

5. Proposed Alternative and Basis for Use

To facilitate the SQN, Unit 2, SG replacement, access openings will be made in the dome of the SQN, Unit 2, SCV to allow rigging of the SGs out of and into containment. Following welding that restores these SCV openings, it is proposed that instead of performing an integrated leak rate test (Type A test) as required by ASME Section XI, Paragraph

IWE-5221, a local leak rate test be performed on the new pressure boundary welds of the SCV.

The sections of the SCV that are removed will be rewelded in place by qualified personnel in accordance with the SQN code of record requirements. The code of record for the SCV is ASME Section III, 1968 Edition through the Winter 1968 Addenda. Consistent with the code of record requirements, examinations will be performed on the steel vessel repair welds. As a minimum, a magnetic particle test of the back gouge of the root pass will be performed, and 100 percent radiography will be performed on the pressure boundary containment SCV final repair welds. In addition, ASME Section XI, as modified by 10 CFR 50.55a(b)(2)(ix)(G), requires both a General Visual VT-3 and a Detailed Visual VT-1 examination of the SCV pressure boundary welds. These are preservice examinations. The SCV repair welds will be tested by a local leakage/pressure test by pressurizing the containment vessel to the required test pressure of at least P_a (12.0 pounds per square inch gauge [psig]) and performing a bubble test of the repair welds after a hold time of at least 10 minutes. The test pressure will be held between 12.2 psig and 12.5 psig. Pressurizing containment to P_a will structurally test the SCV repair weld. Zero detectable leakage is the acceptance criterion, and is determined by the absence of bubble formation. Any leakage identified will be corrected, and the test will be reperformed. The SCV will be pressurized through an existing penetration using an external air compressor. It takes approximately 4 hours to pressurize the SCV to the test pressure. Once attaining test pressure, the pressure will be held for 10 minutes before testing and during the bubble testing and visual examination. It will require approximately 1-2 hours to perform the bubble test and visual examination. After the bubble test is completed, the SCV will be depressurized in a controlled manner which takes approximately 2-4 hours. Qualified personnel will conduct all examinations. The combination of the 100 percent radiography, demonstrating that the repair welds meet the construction code radiography acceptance criteria, and the local leak rate test of the repair welds by performing the bubble test while the SCV and repair welds are at accident pressure are more than sufficient to prove the integrity of the steel containment vessel.

ASME Section XI, Paragraph IWE-5221 requires that an appropriate 10 CFR 50, Appendix J test be performed following a repair or modification of the pressure retaining boundary. Specifically, the ASME Code requires a Type A, Type B, or Type C test, as appropriate, for the repaired or modified pressure boundary component.

10 CFR 50, Appendix J, Option B provides guidelines for meeting the safety objectives of the Appendix J requirements. Section 9.2.4 of Nuclear Energy Institute (NEI) 94-01 (Reference c) states that "repairs and modifications that affect the containment leakage rate require leak rate testing (Type A testing or local leak rate testing) prior to returning the containment to operation."

A local leak rate test provides the most accurate and direct method of assuring the leak tight integrity of the repair welds. The local leak rate test is considered a superior test for determining leakage at the repaired area as compared to a Type A test. The local leak rate test will pressurize the entire containment to greater than P_a and directly quantify the leakage at the repair area, while a Type A test measures total containment leakage. This test is being performed to reestablish the leak-tight integrity of the SCV following application of the repair welds. Also, the acceptance criterion for leakage of the repair welds will be zero leakage. This acceptance criterion is a more stringent criterion than that of a Type A test. Therefore, if there is any leakage of the SCV at the repair welds, it would be identified by the local leak rate test and would be corrected.

Additionally, the containment pressure test, performed at P_a, will reestablish the structural integrity of the SCV. Therefore, the required pressure test at P_a and the local leak rate test of the SCV repair welds satisfy or exceed the intent of a Type A test to establish containment integrity after a repair activity.

TVA has determined that a local leak rate test is the most appropriate test to perform on the SCV to meet the testing requirements of the ASME Code. A Type A test is a less sensitive test than a local leak rate test. TVA considers that the local leak rate test, in conjunction with the planned containment pressure test, will continue to provide for an acceptable level of quality and safety.

6. Duration of Proposed Alternative

This 10 CFR 50.55a request proposes one-time alternate testing in place of performing an integrated leak rate test (Type A test) as is required by ASME Code Section XI, paragraph IWE-5221, specifically to test the welds that will restore the access openings that will be made in the SQN, Unit 2, SCV dome to facilitate the replacement of the SQN, Unit 2, steam generators. Use of this proposed alternative does not alter the ISI or containment leak rate testing requirements that apply to the regularly scheduled performance of integrated leak rate testing of the SQN, Unit 2, SCV in accordance with ASME Code and 10 CFR 50, Appendix J. The duration of the proposed alternative local leak rate testing at P_a is limited, therefore, to this specific application during the SQN, Unit 2, steam generator replacement outage.

7. Precedents

NRC approval has been previously granted for the same type of alternative leak testing following repair of the respective plant's SCV:

a) NRC Letter to Vice President, Operations, Entergy Operations, Inc., Waterford Steam Electric Station, Unit 3 – Request for Relief from Requirements of ASME Code Section XI, IWE-5221 Re: Post-Repair Leakage Inspection of Steel Containment Vessel (TAC No. ME3345), dated May 18, 2010 – ADAMS Accession No. ML100850089

b) NRC Letter to Mr. Karl W. Singer (TVA CNO and Executive VP), Watts Bar Nuclear Plant, Unit 1 – One Time Request for Relief from ASME Section XI Code Requirements – Tests Following Repair, Modification, or Replacement (IWE-5221)(TAC No. MC8920), dated August 30, 2006 – ADAMS Accession No. ML061590111

c) NRC Letter to Mr. J. A. Scalice (TVA CNO and Executive VP), Sequoyah Nuclear Plant, Unit 1 – Request for Relief from American Society of Mechanical Engineers, Section XI Code Requirements for Tests following Repair, Modification, or Replacement (TAC No. MB8431), dated May 9, 2003 – ADAMS Accession No. ML031320320

8. References

a) ASME Section III, 1968 Edition through the Winter 1968 Addenda.

b) 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," 2010.

c) NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 0, July 26, 1995.