UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 22, 2011

Mr. Joseph W. Shea
Corporate Manager, Nuclear Licensing
Tennessee Valley Authority
3R Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNIT 2 – SAFETY EVALUATION OF RELIEF REQUEST NO. 2-APPJ-1, PROPOSED ALTERNATIVE TO THE LEAKAGE TESTING REQUIREMENTS IN PARAGRAPH IWE-5221 OF THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE, SECTION XI, 2001 EDITION WITH 2003 ADDENDA (TAC NO. ME5245)

Dear Mr. Shea:

By letter dated December 22, 2010 (Agencywide Documents Access and Management System Accession No. ML103630044) Tennessee Valley Authority (TVA), the licensee, submitted Relief Request 2-APPJ-1 for the Sequoyah Nuclear Plant (SQN), Unit 2. Pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(i), TVA requested to utilize an alternative to the testing requirements in paragraph IWE-5221, Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 2001 edition with the 2003 addenda. TVA's request for relief proposes alternative test methods (pneumatic leakage test) for containment vessel repair activities following the SQN, Unit 2 Cycle 18 steam generator replacement refueling outage in fall 2012.

The proposed alternative to the requirements of paragraph IWE-5221 will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3(i)), the Nuclear Regulatory Commission authorizes the use of the proposed one-time alternative request during the licensee's steam generator replacement program, which is scheduled during the Cycle 18 refueling outage in fall 2012 for SQN, Unit 2.

All other requirements of the ASME Code, Sections III and XI, for which relief has not been specifically requested and approved remain applicable. If you have any questions, please feel free to contact the SQN Project Manager, Siva P. Lingam, at (301) 415-1564.

Sincerely,

for

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-328

Enclosure: Safety Evaluation

cc w/encl: Distribution via Listserv

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST 2-APPJ-1 ALTERNATIVE TEST METHODS

FOR CONTAINMENT REPAIR ACTIVITIES

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT UNIT 2

DOCKET NO. 50-328

1.0 INTRODUCTION

By the letter dated December 22, 2010 (Agencywide Documents Access and Management System Accession No. ML103630044), Tennessee Valley Authority (TVA, the licensee) proposed an alternative, which would allow the licensee to perform a local leak rate test of the containment in lieu of the "Leakage Test" requirements in Section XI of the American Society of Mechanical Engineers Boiler and Pressure Code (ASME Code). The leakage tests are required as a result of planned repairs to the Sequoyah Nuclear Plant (SQN), Unit 2 steel containment vessel (SCV) in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), 50.55a(a)(3)(i). The licensee will be replacing the SQN, Unit 2 steam generators during the cycle 18 refueling outage, commencing in the fall 2012.

This safety evaluation addresses the ability of the proposed alternative to ensure the continued ability of the SCV to provide an acceptable level of quality and safety after the steam generator replacement activity.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee requested whether they could use an alternative to the requirements in Section XI of the ASME Code for post repair leakage inspection of the SQN, Unit 2 SCV from the scheduled steam generator replacement project scheduled to start in October 2012.

The regulations at 10 CFR 50.55a, "Codes and Standards," incorporate, by reference, the 2001 edition through the 2003 addenda of Section XI of the ASME Code. Paragraph IWE-5221 of Subsection IWE of the ASME Code, Section XI, requires a leakage rate test following any repair and replacement activity. Paragraph IWE-5221 specifies that the leakage rate test be conducted in accordance with the provisions of 10 CFR Part 50, Appendix J, paragraph IV.A, "Containment modification," which states in part, "Any major modification, replacement of a component, which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable for the area affected by the modification."

3.0 TECHNICAL EVALUATION

3.1 Specific Alternative Request

The licensee proposes to perform an "as-left" local leak test on the new pressure boundary welds of the SCV repair welds in lieu of the Type A, Type B, or Type C leakage rate tests as specified in Paragraph IWE-5221 of the ASME Code for this type of repair activity.

3.2 Proposed Alternative Duration

The duration of the proposed alternative is through the completion and approval of all testing associated with the restoration of the containment opening created to support the SQN, Unit 2 steam generator replacement.

3.3 Basis for Relief

The licensee committed to perform the activities described below as a part of the SCV restoration effort. The sections of the SCV that were removed will be rewelded in place in accordance with the requirements of Section III, of the ASME Code, 1968 Edition through the winter 1968 addenda. Before performing the repair weld, the surfaces to be welded will be cleaned. 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 of SCV final repair welds. In addition, ASME Code, Section XI, as modified by 10 CFR 50.55a(b)(2)(ix)(G), requires both a general visual, VT-3 examination and a detailed visual, VT-1 examination of the SCV pressure boundary welds. To perform a weld leak test, the containment will be pressurized to a test pressure P_a of at least 12.0 pounds per square inch gauge (psig) for a minimum of 10 minutes. The test pressure will be held between 12.2 psig and 12.5 psig. A zero leakage criterion will be used for weld acceptance, which is determined by the absence of any bubbles formation. Any leakage identified will be corrected, and the test will be repeated.

Based on the above procedure, the licensee stated that any local leakage will be directly identified by the bubble test and it will confirm the findings of the prior nondestructive examination tests. 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, will provide a more effective examination than the Type A test, as required by the ASME Code, paragraph IWE-5221. Therefore, an alternative to the requirement of paragraph IWE-5221 is requested pursuant to 10 CFR 50.55a(a)(3)(i), in that the proposed alternative provides an acceptable level of quality and safety.

3.4 Nuclear Regulatory Commission (NRC) Staff Evaluation

To facilitate the SQN, Unit 2 steam generator replacement, the SCV will be breached. An opening will be cut in the SCV in order to remove and replace the steam generators. After the steam generator replacement, the SCV sections removed will be reattached through welding. Paragraph IWE-5221 of the ASME Code requires that leakage rate testing be conducted to ensure the integrity of the repairs before returning the SCV to operable status. In lieu of the Type A, Type B, or Type C leakage rate testing, the licensee proposed to perform a series of examinations and a leak test subjecting the SCV to accident pressure, to verify the leak tightness and integrity of the liner welds and the SCV.

The licensee's proposed relief request includes the detailed examination and test sequence that is summarized herein. The licensee proposed to perform the activities described below as a part of the SCV restoration effort. The sections of the SCV that were removed will be rewelded in place in accordance with the requirements of the ASME Code, Section III, 1968 edition through the winter 1968 addenda (TVA's code of record requirements). Before performing the repair weld, the surfaces to be welded will be cleaned and examined by magnetic particle testing of the weld preparation area, and 100 percent radiography of the final repair weld will be performed. In addition, both a general visual, VT-3 examination and a detailed visual, VT-1 examination of the SCV pressure boundary welds will be conducted. To perform a weld leak test, the containment will be pressurized to a test pressure P_a of at least 12 psig for a minimum of 10 minutes. The test pressure will be held between 12.2 psig and 12.5 psig. A zero leakage criterion will be used for weld acceptance, which is determined by the absence of any bubbles formation. Any leakage identified will be corrected, and the test will be repeated

The magnetic particle or liquid penetrate testing, the 100 percent radiography of the repair weld, and the subsequent bubble test will provide adequate assurance that the repair welds do not leak or have any structural defects. The zero leakage acceptance criterion for the bubble test will ensure that the SCV leakage rate is not altered by the steam generator replacement activity, and the pressurization of the SCV to the accident pressure will confirm the integrity of the SCV after the repair. Therefore, the NRC staff concludes that the proposed alternative will provide adequate assurance of structural integrity.

4.0 CONCLUSION

On the basis of the information provided in the relief request, and an independent NRC staff evaluation, the NRC staff concludes that the proposed alternative of a local leak rate test in lieu of the requirements in paragraph IWE-5221 of the ASME Code will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3(i)), the NRC authorizes the use of the proposed one-time alternative request during the licensee's steam generator replacement program, which is scheduled during the Cycle 18 refueling outage in fall 2012 for SQN, Unit 2.

All other requirements of the ASME Code, Section III and Section XI, for which relief has not been specifically requested and approved, remain applicable.

Principal Contributor: D. Hoang

Date: December 22, 2011

All other requirements of the ASME Code, Sections III and XI, for which relief has not been specifically requested and approved remain applicable. If you have any questions, please feel free to contact the SQN Project Manager, Siva P. Lingam, at (301) 415-1564.

Sincerely,

/RA by TOrf for/

Douglas A. Broaddus, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

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