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

November 3, 2021

Mr. James Barstow
Vice President, Nuclear Regulatory
  Affairs and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 149 AND 56 REGARDING REVISION TO TECHNICAL SPECIFICATION 5.7.2.19, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM" TO EXTEND CONTAINMENT INTEGRATED AND LOCAL LEAK RATE TEST INTERVALS (EPID L-2020-LLA-0223)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment Nos. 149 and 56 to Facility Operating License Nos. NPF-90 and 96, for Watts Bar Nuclear Power Plant (Watts Bar), Units 1 and 2, respectively. The amendments are in response to your application dated October 2, 2020, as supplemented by letters dated December 15, 2020, and April 29, 2021.

The amendments revise Watts Bar, Units 1 and 2, Technical Specification (TS) 5.7.2.19, "Containment Leakage Rate Testing Program," by replacing the reference to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," with a reference to Nuclear Energy Institute (NEI) 94-01, Revision 3-A, July 2012, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and the conditions and limitations specified in NEI 94-01, Revision 2-A, of the same name, dated October 2008. The amendments extend the Type A primary containment integrated leak rate test interval from 10 to 15 years and extend the Type C local leak rate test interval from 60 to up to 75 months. The amendments also clarify the pressure value for leakage rate testing purposes.

The NRC staff has also reviewed your proposed alternative actions to the required Appendix J, Type A test following a major repair or modification to the containment, and has determined that proposed alternative actions will provide a verification of both the structural integrity as well as the leakage integrity of the restored containment vessel.

The NRC staff finds that the proposed alternative activities provide an acceptable level of quality and safety after major containment repairs and modifications, as required by 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the proposed alternative at Watts Bar, Units 1 and 2, for use during the current 10-year ISI intervals for Watts Bar, Unit 1 that commenced on September 9, 2018, and is scheduled to end on September 8, 2028, and for Watts Bar, Unit 2, that commenced October 19, 2016, and is currently scheduled to end on October 18, 2026.

A copy of our related safety evaluation is also enclosed. Notice of issuance will be included in the Commission's monthly *Federal Register* notice.

Sincerely,

David J. Wrona

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Date: 2021.11.03 11:03:04 -04'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-390 and 50-391

Enclosures:
1. Amendment No. 149 to NPF-90
2. Amendment No. 56 to NPF-96
3. Safety Evaluation

cc: Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-390

WATTS BAR NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 149
License No. NPF-90

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (TVA, the licensee) dated October 2, 2020, as supplemented by letters dated December 15, 2020, and April 29, 2021, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-90 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 149 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

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Date: 2021.11.03 11:00:46 -04'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: November 3, 2021

ATTACHMENT TO AMENDMENT NO. 149

WATTS BAR NUCLEAR PLANT, UNIT 1

FACILITY OPERATING LICENSE NO. NPF-90

DOCKET NO. 50-390

Replace page 3 of Facility Operating License No. NPF-90 with the attached revised page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace the following page of the Appendix A Technical Specifications with the attached page. The revised page is identified by amendment number and contains vertical lines indicating the area of change.

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5.0-24	5.0-24

(4) TVA, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis, instrument calibration, or other activity associated with radioactive apparatus or components; and

(5) TVA, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

TVA is authorized to operate the facility at reactor core power levels not in excess of 3459 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 149 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Safety Parameter Display System (SPDS) (Section 18.2 of SER Supplements 5 and 15)

Prior to startup following the first refueling outage, TVA shall accomplish the necessary activities, provide acceptable responses, and implement all proposed corrective actions related to having the Watts Bar Unit 1 SPDS operational.

(4) Vehicle Bomb Control Program (Section 13.6.9 of SSER 20)

During the period of the exemption granted in paragraph 2.D.(3) of this license, in implementing the power ascension phase of the approved initial test program, TVA shall not exceed 50% power until the requirements of 10 CFR 73.55(c)(7) and (8) are fully implemented. TVA shall submit a letter under oath or affirmation when the requirements of 73.55(c)(7) and (8) have been fully implemented.

5.7 Procedures, Programs, and Manuals

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.7.2.19 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. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008, as modified below:

For containment leakage rate testing purposes, a value of 15.0 psig, which is equivalent to the maximum allowable internal containment pressure, is utilized for P_a to bound a range of peak calculated containment internal pressures from 9.0 to 15.0 psig for the design basis loss of coolant accident.

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

(continued)

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-391

WATTS BAR NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 56
License No. NPF-96

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (TVA, the licensee) dated October 2, 2020, as supplemented by letter dated December 15, 2020, and April 29, 2021, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-96 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 56 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

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Date: 2021.11.03 11:01:19 -04'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: November 3, 2021

ATTACHMENT TO AMENDMENT NO. 56

WATTS BAR NUCLEAR PLANT, UNIT 2

FACILITY OPERATING LICENSE NO. NPF-96

DOCKET NO. 50-391

Replace page 3 of Facility Operating License No. NPF-96 with the attached revised page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace the following page of the Appendix A Technical Specifications with the attached page. The revised page is identified by amendment number and contains vertical lines indicating the area of change.

Remove Pages	Insert Pages
5.0-25	5.0-25

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

TVA is authorized to operate the facility at reactor core power levels not in excess of 3459 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 56 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) TVA shall implement permanent modifications to prevent overtopping of the embankments of the Fort Loudon Dam due to the Probable Maximum Flood by June 30, 2018.

(4) PAD4TCD may be used to establish core operating limits until the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1. FULL SPECTRUM LOCA Methodology shall be implemented when the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

(5) By December 31, 2019, the licensee shall report to the NRC that the actions to resolve the issues identified in Bulletin 2012-01, "Design Vulnerability in Electrical Power System," have been implemented.

(6) The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90 and 50.54(p).

(7) TVA shall fully implement and maintain in effect all provisions of the Commission approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The TVA approved CSP was discussed in NUREG-0847, Supplement 28, as amended by changes approved in License Amendment No. 7.

(8) TVA shall implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Report for the facility, as described in NUREG-0847, Supplement 29, subject to the following provision:

5.7.2.18 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.7.2.19 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. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report in NEI 94-01, Revision 2-A, dated October 2008, as modified below:

For containment leakage rate testing purposes, a value of 15.0 psig, which is equivalent to the maximum allowable internal containment pressure, is utilized for P_a to bound a range of peak calculated containment internal pressures from 9.0 to 15.0 psig for the design basis loss of coolant accident.

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

(continued)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 149 AND 56

TO FACILITY OPERATING LICENSE NOS. NPF-90 AND NPF-96

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-390 AND 50-391

1.0 INTRODUCTION

By application dated October 2, 2020 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML20276A092), as supplemented by letters dated December 15, 2020, and April 29, 2021 (ADAMS Accession Nos. ML20350B799 and ML21119A248, respectively), the Tennessee Valley Authority (TVA or the licensee) requested changes to the technical specifications (TSs) for the Watts Bar Nuclear Plant (Watts Bar), Units 1 and 2. The license amendment request (LAR) proposed changes to TS 5.7.2.19, "Containment Leakage Rate Testing Program," by replacing the TS 5.7.2.19 reference to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program" (ADAMS Accession No. ML003740058), with a reference to Nuclear Energy Institute (NEI) 94-01, Revision 3-A, July 2012, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," (ADAMS Accession No. ML12221A202), and the limitations and conditions specified in NEI 94-01, Revision 2-A, of the same name, dated October 2008 (ADAMS Accession No. ML100620847). More specifically, the proposed amendment would allow extension of the Type A test interval from 10 to 15 years and extension of the Type C test interval from 60 to up to 75 months, based on acceptable performance history as defined in NEI 94-01, Revision 3-A. In addition, a clarification of the value of P_a to be used for containment leakage rate testing purposes is proposed.

The supplement dated December 15, 2020, provided confirmatory test results that were conducted following the licensee's submittal. Additionally, the supplement provided a corrected Enclosure to address a misquote. The supplement dated April 29, 2021, provided additional information in response to a request for additional information. The supplements did not expand the scope of the application as originally noticed and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 2, 2021 (86 FR 7885).

2.0 REGULATORY EVALUATION

2.1 Description of Containment

The licensee describes the Watts Bar containments in the LAR as consisting of a containment vessel and a separate shield building enclosing an annulus. The shield building is a reinforced concrete structure similar in shape to the containment vessel.

The steel containment vessel (SCV) is a low-leakage, freestanding steel structure consisting of a cylindrical wall, a hemispherical dome, and a bottom liner plate encased in concrete. The structure consists of side walls measuring 114 feet 8-5/8 inches in height from the liner on the base to the spring line of the dome and has an inside diameter of 115 feet. The bottom liner plate is 1/4 inch thick, the cylinder varies from 1-3/8 inch thickness at the bottom to 1-1/2 inch thick at the spring line, and the dome varies between 1-3/8 inch thickness and 13/16 inch thickness with 15/16 inch thickness at the apex. The SCV is provided with both circumferential and vertical stiffeners on the exterior of the shell. These stiffeners are required to satisfy design requirements for expansion and contraction, seismic forces, and pressure transient loads. The circumferential stiffeners were installed on approximately 10-foot centers during erection to ensure stability and alignment of the shell. Vertical stiffeners are spaced at 5 degrees between the two lowest circumferential stiffeners. Other locally stiffened areas are provided at the equipment hatch and two personnel locks. During the Watts Bar, Unit 1 steam generator replacement in the Fall of 2006, two construction openings were made in the SCV. These openings were restored by reinstalling the removed steel sections and rewelding them to the remaining structure using full penetration welds. Abandoned-in-place reinforcement and support members to stiffen the SCV during creation and use of the two construction openings were designed to remain attached to the SCV during a seismic event. The integrity of the restored vessel was verified by nondestructive examination (NDE) and leak testing of the welds.

The SCV maximum internal pressure is 15 pounds per square inch gauge (psig) at 250 degrees Fahrenheit (°F), and the design internal pressure is 13.5 psig at 250 °F. The difference between the two is explained in the same section of the LAR by stating: "Paragraph NE-3312(b) of Section III of the [American Society of Mechanical Engineers] ASME Code states that the 'design internal pressure' of the vessel may differ from the 'maximum containment pressure', but in no case shall the design internal pressure be less than 90% of the maximum containment internal pressure."

2.2 Licensee's Proposed Changes

The licensee proposed to revise Watts Bar, Units 1 and 2, TS 5.7.2.19 by replacing the reference to RG 1.163 with a reference to NEI 94-01, Revision 3-A, and Section 4.1, "Limitations and Conditions for NEI Topical Report 94-01, Revision 2," of NEI 94-01, Revision 2-A.

The licensee also proposed changes to clarify the value of P_a to be used for containment leakage rate testing purposes. For Watts Bar Unit 1, the licensee proposed to delete an existing statement and add a new paragraph as follows:

> ~~The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 15.0 psig.~~

> For containment leakage rate testing program purposes, a value of 15.0 psig, which is equivalent to the maximum allowable internal containment pressure, is utilized for P_a to bound a range of peak calculated containment internal pressures from 9.0 to 15.0 psig for the design basis loss of coolant accident.

For Watts Bar, Unit 2, the licensee proposed to revise the existing language as follows:

> For containment leakage rate testing purposes, a value of 15.0 psig, which is equivalent to the maximum allowable internal containment pressure, is utilized for P_a to bound ~~the~~ a range of peak calculated containment internal pressures from 9.0 to 15.0 psig for the design basis loss of coolant accident.

2.3 Regulatory Requirements

Section 50.36(c)(5), "Administrative controls," of Title 10 of the *Code of Federal Regulations* (10 CFR) requires, in part, that the TSs include administrative controls necessary to ensure operation of the facility in a safe manner. The LAR requests a change to the "Administrative Controls" section of the Watts Bar TSs.

Section (o) of 10 CFR 50.54, "Conditions of licenses," requires that primary reactor containments for water-cooled power reactors shall be subject to the requirements set forth in 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J contains two options: Option A – Prescriptive Requirements, and Option B – Performance-Based Requirements, either of which can be used to meet Appendix J requirements.

The testing requirements in 10 CFR Part 50, Appendix J ensure that: (a) leakage through containments or systems and components penetrating containments does not exceed allowable leakage rates specified in the TSs, and (b) integrity of the containment structure is maintained during the service life of the containment. Watts Bar, Unit 1 adopted 10 CFR Part 50, Appendix J, Option B for Type A integrated leak rate testing (ILRT), and Type B and Type C local leak rate testing (LLRT) by Amendment No. 5, dated May 27, 1997 (ADAMS Accession No. ML020790130). Watts Bar, Unit 2 was initially licensed to use 10 CFR Part 50, Appendix J, Option B for Types A, B, and C testing on October 22, 2015 (ADAMS Accession No. ML15251A587).

Section V.B.3 of 10 CFR Part 50, Appendix J, Option B, requires the licensee to develop a performance-based leakage-testing program using RG 1.163, or other implementation document, and referencing it in the plant TSs. The submittal for TS revisions must also contain justification, including supporting analyses, if the licensee deviates from methods approved by the NRC and endorsed in RG 1.163, which includes guidance for acceptable leakage rate test methods, procedures, and analyses.

Option B specifies performance-based requirements and criteria for preoperational and subsequent leakage rate testing. These requirements are met by:

1. Type A tests to measure the containment system overall integrated leakage rate,

2. Type B pneumatic tests to detect and measure local leakage rates across pressure retaining leakage-limiting boundaries such as penetrations, and

3. Type C pneumatic tests to measure containment isolation valve leakage rates.

After the containment system has been completed and is ready for operation, Type A tests are conducted at periodic intervals based on the historical performance of the overall containment system to measure the overall integrated leakage rate. The leakage rate test results must not exceed the maximum allowable leakage (L_a) at design basis loss-of-coolant-accident (DBLOCA) pressure (P_a) with margin, as specified in the TSs. Option B also requires that a general visual inspection for structural deterioration of the accessible interior and exterior surfaces of the containment system, which may affect the containment leak-tight integrity, be conducted prior to each Type A test and at a periodic interval between tests based on the performance of the containment system.

Type B and Type C tests are performed based on the safety significance and historical performance of each boundary and isolation valve to ensure integrity of the overall containment system as a barrier to fission product release.

Section 50.55a, "Codes and standards," of 10 CFR contains the containment inservice inspection (ISI) requirements, which, in conjunction with the requirements of 10 CFR Part 50, Appendix J, ensure the continued leak-tight and structural integrity of the containment during its service life.

Section 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," of 10 CFR, paragraph (a)(1), states, in part, that the licensee:

> …shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components … are capable of fulfilling their intended functions. These goals shall be established commensurate with safety and where practical, take into account industry-wide operating experience.

2.4 <u>Regulatory Guidance</u>

The guidance in NEI 94-01, Revision 0 (ADAMS Accession No. ML11327A025), provides methods for complying with the provisions of 10 CFR Part 50, Appendix J, Option B, and includes provisions that address the extension of the performance-based Type A test interval for up to 10 years, based upon two consecutive successful tests.

The NRC staff's final safety evaluation (SE) (ADAMS Accession No. ML081140105), dated June 25, 2008, for NEI 94-01, Revision 2, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (ADAMS Accession No. ML072970206), and Electric Power Research Institute (EPRI) Technical Report (TR)-1009325, Revision 2, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," dated August 2007 (ADAMS Accession No. ML072970208), was incorporated into NEI 94-01, Revision 2-A, and issued on November 19, 2008 (ADAMS Accession No ML100620847). NEI 94-01, Revision 2-A, describes an NRC-approved approach for implementing the optional performance-based requirements of Option B described in 10 CFR Part 50, Appendix J, and incorporates the regulatory positions stated in RG 1.163. NEI 94-01, Revision 2-A, delineates a performance-based approach for determining Type A, Type B, and Type C containment leakage rate testing frequencies, and includes provisions for extending Type A ILRT intervals to up to 15 years. This approach uses industry performance, plant-specific data, and risk insights in determining the appropriate testing frequency, and discusses the performance factors that licensees must consider in determining test intervals.

EPRI TR-1009325, Revision 2[1], provides a generic assessment of the risks associated with a permanent extension of the ILRT surveillance interval to 15 years, and a risk-informed methodology/template to be used to confirm the risk impact of the ILRT extension on a plant-specific basis. Probabilistic risk assessment (PRA) methods are used, in combination with ILRT performance data and other considerations, to justify the extension of the ILRT surveillance interval. This is consistent with guidance provided in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (ADAMS Accession No. ML17317A256), RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," (ADAMS Accession No. ML100910008) to support changes to surveillance test intervals, and RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (ADAMS Accession No. ML090410014).

NEI 94-01, Revision 3-A, provides guidance for extending Type C LLRT intervals beyond 60 months. The NRC published an SE with limitations and conditions for NEI 94-01, Revision 3, by letter dated June 8, 2012 (ADAMS Accession No. ML121030286). In the SE, the NRC concluded that NEI 94-01, Revision 3, describes an acceptable approach for implementing the optional performance-based requirements of Appendix J, and is acceptable for reference by licensees proposing to amend their containment leakage rate testing TSs, subject to two conditions, which are addressed in Section 3.5 of this SE. The SE was incorporated into Revision 3 and subsequently issued as NEI 94-01, Revision 3-A.

RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," provides an acceptable approach for developing risk-informed applications for licensing basis changes that considers engineering issues and applies risk insights. This RG also provides general guidance concerning analysis of the risk associated with proposed changes in plant design and operation.

RG 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," provides guidance for determining the technical adequacy and quality of the PRA, in total or the parts used to support an application, such that the PRA can be used in regulatory decision-making for light-water reactors.

Regulatory Issue Summary (RIS) 2007-06, "Regulatory Guide 1.200 Implementation" (ADAMS Accession No. ML070650428), provides information regarding how the NRC will implement its technical adequacy review of plant specific PRAs in support of RG 1.200.

3.0 TECHNICAL EVALUATION

3.1 ILRT History (Type A Testing)

Per TS 5.7.2.19, Watts Bar specified a maximum allowable containment leakage rate, L_a, of 0.25 percent by weight of the containment air per day at the calculated peak pressure, P_a. Prior to January 22, 2013, L_a was 0.25 percent of containment air weight per day at the calculated

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[1] EPRI TR-1018243 is also identified as EPRI-1009325, Revision 2-A. This report is publicly available and can be found at www.epri.com by typing "1018243" in the search box.

peak pressure. Technical Specification 5.7.2.19 indicates that the peak calculated containment internal pressure for the DBLOCA, P_a, is 15 psig, which bounds a range of peak calculated containment internal pressures from 9.0 to 15.0 psig for the DBLOCA.

The licensee has performed three ILRTs on the Watts Bar, Unit 1, containment since June 1994, and two ILRTs on the Watts Bar, Unit 2, containment since August 2015 (initial Unit 2 startup). The LAR provided the ILRT results showing substantial margin maintained relative to the performance criterion for the most recent Type A tests for both Units, so the extended interval would be permitted by program guidance for Watts Bar. In addition, no adverse trend was apparent that would suggest the performance criterion being exceeded with the requested interval extension to 15 years. The licensee provided the results of these tests in LAR Section 3.3.1, which are summarized in Tables 3.1-1 and 3.1-2 below.

TABLE 3.1-1
Watts Bar, Unit 1 Type A ILRT History

Test Date	**P_a (psig) (1)**	**P_t Test Pressure (psig) (2)**	**P_d Containment Design Pressure (psig)**	**As-Found Leakage (wt%/day)**	**As-Found Acceptance Criteria (wt%/day)**	**As-Left Acceptance Criteria (wt%/day)**
June 1994*	15	15.03	13.5	0.0904135	0.25 (1.0 L_a)	0.1875 (0.75 L_a)
October 1997	15	14.97	13.5	0.10613	0.25 (1.0 L_a)	0.1875 (0.75 L_a)
October 2012	15	14.59	13.5	0.01683	0.25 (1.0 L_a)	0.1875 (0.75 L_a)

Table 3.1.1 Notes:

(1) P_a – As defined in Watts Bar TS 5.7.2.19
(2) P_t – Final test pressure (psig) – minimum allowable P_t is P_a -1 psig = 14 psig
(*) Value from startup is as-left data

TABLE 3.1-2
Watts Bar, Unit 2 Type A ILRT History

Test Date	P_a (psig) (1)	P_t Test Pressure (psig) (2)	P_d Containment Design Pressure (psig)	As-Found Leakage (wt%/day)	As-Found Acceptance Criteria (wt%/day)	As-Left Acceptance Criteria (wt%/day)
August 2015*	15	15.26	13.5	0.0590	0.25 (1.0 L_a)	0.1875 (0.75 L_a)
May 2019	15	15.32	13.5	0.01399	0.25 (1.0 L_a)	0.1875 (0.75 L_a)

Table 3.1.2 Notes:

(1) P_a – As defined in Watts Bar TS 5.7.2.19
(2) P_t – Final test pressure (psig) – minimum allowable P_t is P_a -1 psig = 14 psig
(*) Value from startup is as-left data

The NRC staff notes that Section 9.1.2 of NEI 94-01, Revision 3-A reads, in part, "[t]he elapsed time between the first and the last tests in a series of consecutive passing tests used to determine performance shall be at least 24 months." Per Tables 3.1.1 and 3.1.2 above, the Section 9.1.2 requirement was met.

The Watts Bar TS 5.7.2.19 references RG 1.163. Regulatory Position C of RG 1.163 states in part that NEI 94-01, Revision 0, "provides methods acceptable to the NRC staff for complying with the provisions of Option B in Appendix J to 10 CFR Part 50…" The third paragraph of Section 9.2.3, "Extended Test Intervals" of NEI 94-01, Revision 0 reads, in part:

> In reviewing past performance history, Type A test results may have been calculated and reported using computational techniques other than the Mass Point method from ANSI/ANS-56.8–1994 (e.g., Total Time or Point–to–Point). Reported test results from these previously acceptable Type A tests can be used to establish the performance history. Additionally, a licensee may recalculate past Type A Upper Confidence Limit (UCL) (using the same test intervals as reported) in accordance with ANSI/ANS-56.8–1994 Mass Point methodology and its adjoining Termination criteria in order to determine acceptable performance history …

NEI 94-01, Revision 3-A reads nearly identical except the test standard invoked is ANSI/ANS-56.8–2002.

ILRT History Conclusion

The NRC staff notes that Section 9.2.3 does not state that a licensee must recalculate past Type A test results to demonstrate conformance with the definition of "performance leakage rate" contained in NEI 94-01, Revision 3-A. The staff also notes that the ILRT results since June 1994 (Unit 1 startup) and August 2015 (Unit 2 startup) respectively, demonstrated ample margin (i.e., > 57 percent) between each "as-found" leakage value and La.

The Watts Bar TS 5.7.2.19 establishes the maximum limit for the “as-Left” Leakage Rate for startup following completion of Type A testing at ≤ 0.75 La, or 0.1875 percent of containment air weight per day.

The past ILRT results for both units since startup have confirmed that the primary containment leakage rates are acceptable with respect to the design criterion leakage of containment air weight (La) per day. The last two Type A tests on both Units for Watts Bar had “as found” test results well within the current maximum allowable containment leakage rate specified in TS 5.7.2.19 of 0.25 weight-percent/day (1.0 La).

Based on the Watts Bar ILRT test results, the NRC staff finds that the licensee’s proposed ILRT program is consistent with Section 9.1.2 of NEI 94-01, Revision 3-A.

3.2 LLRT History (Type B and Type C Testing)

The Watts Bar, Units 1 and 2, Appendix J, Type B and Type C LLRT program requires testing of electrical penetrations, airlocks, hatches, bellows, and valves within the scope of the program as required by 10 CFR Part 50, Appendix J, Option B and TS 5.7.2.19. The Type B LLRTs for each unit includes two personnel airlocks, 46 individual bellows, 53 electrical penetrations, 12 resilient seals, and an equipment hatch. The Type C LLRTs for Unit 1 include 81 valve penetrations and the Type C LLRTs for Unit 2 include 73 valve penetrations.

The NRC staff reviewed the LLRT summaries contained in LAR Section 3.3.2, “Type B and Type C Testing.” For Watts Bar, Units 1 and 2, the combined Type B and Type C leakage acceptance criterion is 0.60 La or 147.6 standard cubic feet per hour (scfh). As detailed in NEI 94-01, Revision 3-A, Section 10.2, the combined as-found Type B and C minimum pathway test results and the combined Type B and C as-left maximum pathway test results are evaluated with this acceptance criterion. The NRC staff notes that the Type B and Type C test results showed a large amount of margin between the as-found and as-left outage summations and the respective TS leakage rate acceptance criteria.

With the use of these La values and the data contained in Section 3.3.2 of the LAR, the NRC staff confirmed the accuracy of the “Percentage of 0.6La” values contained in the LAR and determined that:

- The Unit 1 “as-found” minimum pathway leakage rates for the last four refueling outages since 2015 have an average of 4.5 percent of $0.6L_a$ with a high of 6.1 percent of $0.6L_a$.

- The Unit 1 “as-left” maximum pathway leakage rates for the last four refueling outages since 2015 have an average of 12.7 percent of $0.6L_a$ with a high of 15.2 percent of $0.6L_a$.

- The Unit 2 “as-found” minimum pathway leakage rates for the two refueling outages since Unit 2 startup (2015) have an average of 10.9 percent of $0.6L_a$ with a high of 13.4 percent of $0.6L_a$.

- The Unit 2 “as-left” maximum pathway leakage rates for the two refueling outages and the Unit 2 startup (2015) have an average of 17.7 percent of $0.6L_a$ with a high of 21.6 percent of $0.6L_a$.

As conveyed in LAR Section 3.3.2, for Watts Bar, there have been three repeat LLRT failures in Unit 1 and three repeat LLRT failures in Unit 2 over the last two outages. In each case, the failure was a Type C tested valve. These Type C valve test failures during the two most recent refueling outages are described below.

- 1-FCV-90-116 exceeded the administrative limit due to foreign material observed in the seat area. During Unit 1 Refueling Outage 15 (U1R15), the seat area was cleaned, and new diaphragm and packing installed. During U1R16, the valve was cleaned and reworked, and new disc and stem internals were installed. The as-left leakage during U1R15 and U1R16 was 0.034 scfh and 0.0 scfh, respectively.

- 1-FCV-63-23 exceeded the administrative limit. No maintenance was performed based on no active adverse trend present. The as-left leakage during U1R15 and U1R16 was 1.38 scfh and 1.49 scfh, respectively.

- 1-FCV-43-287 exceeded the administrative limit. No maintenance was performed based on post-accident sampling valves maintained in the block position with no possibility for adverse trend possible. The as-left leakage during U1R15 and U1R16 was 5.28 scfh and 3.41 scfh, respectively.

- 2-CKV-31-3392 exceeded the administrative limit due to foreign material observed in the seat area. During Unit 2 Refueling Outage 1 (U2R1), the valve was disassembled, cleaned, and inspected. During U2R2, the valve was cleaned and reworked, and new disc and stem internals were installed. The as-left leakage during U2R1 and U2R2 was 0.005 scfh and 0.39 scfh, respectively.

- 2-FCV-32-111 exceeded the administrative limit. During U2R1, attempts at fixing the valve by machining the plug seat were performed with limited success. During U2R2, no maintenance was performed based on no active adverse trend present from U2R1 as-left test to U2R2 as-found test. The as-left leakage during U2R1 and U2R2 was 4.39 scfh and 4.60 scfh, respectively.

- Electrical penetration X-122E exceeded the administrative limit. During U2R1, no maintenance was performed, but a leakage evaluation was performed to allow leakage to remain as-is. During U2R2, a loose ceramic isolator was discovered, and the isolator was re-torqued to the correct leakage. The as-left leakage during U2R1 and U2R2 was 4.43 scfh and 0.07 scfh, respectively.

In the LAR, the licensee indicated that, following each refueling outage (18-month cycle), LLRT results are reviewed, component performance is evaluated, and LLRT frequencies are reduced accordingly for failure to meet administrative limits and extended, if eligible, following two successful as-found tests. Results of these evaluations and frequency changes are summarized in a post-outage report and this information is used for development of future outage scope of periodic as-found LLRTs. Development of outage scope for conditional as-found and as-left LLRTs is based on planned maintenance and modification activities that have potential to affect containment leakage integrity. The NRC staff notes that this course of action is consistent with the guidance of NEI 94-01, Revision 0, Section 10.2.3, “Type C Test Interval.”

Watts Bar has a total of 114 Type B components on each unit. Of the 114 Type B components, all for Watts Bar, Unit 1, and 113 for Watts Bar, Unit 2, are currently on extended test frequencies. Watts Bar has a total of 189 Type C components on Unit 1, and 184 Type C components on Unit 2. Of the 189 Type C components on Unit 1, 161 components are currently on extended test frequencies. Of the 184 Type C components on Unit 2, 146 components are currently on extended test frequencies. The number of components on extended frequency is adjusted periodically based on valve performance and other plant testing requirements as previously discussed.

LLRT History Conclusion

Due to the aggregate test results at the end of each operating cycle all being well below (i.e., > 95 percent margin) the Type B and Type C test TS leakage rate acceptance criteria of < 0.60 La, the NRC staff determined that the aggregate leakage rate results of the "As-Found Minimum Pathway" for all Watts Bar, Units 1 and 2, Type B and C tests from the last two outages demonstrates a history of adequate maintenance. Further, the NRC staff determined that the licensee provided an adequate explanation regarding the causes of LLRT Type C penetration failures experienced during the most recent refueling outages for both Units.

In summary, the NRC staff concludes that:

- the licensee has been compliant with the guidance of RG 1.163 and NEI 94-01, Revision 0;

- the recent historical combined total Type B and C test results are substantially below the acceptance limit of TS 5.7.2.19; and

- the licensee's corrective action program has been generally effective in addressing poor performing valves and penetrations based on the scope of information provided in the LAR.

Therefore, the NRC staff finds that the licensee is effectively implementing the Watts Bar Type B and Type C LLRT program, as required by Option B of 10 CFR Part 50, Appendix J.

3.3 Containment Inspection Programs

3.3.1 Containment Inservice Inspection (CISI) Program

As described in Section 3.4.1 of the LAR, the Watts Bar CISI Program provides the site specific requirements for implementation of the examinations of the SCV, ASME Code Class MC components, in accordance with ASME Code, Section XI, Subsection IWE, 2013 Edition, as conditioned by 10 CFR 50.55a. The Watts Bar, Unit 1, third interval is effective from September 9, 2018, through September 8, 2028. The Watts Bar, Unit 2, first interval is effective October 19, 2016, through October 18, 2026.

With respect to inaccessible areas, Section 3.4.1 of the LAR indicated that no areas have been identified that would indicate there is an issue that would adversely affect structural integrity or leak tightness of the SCV for inaccessible areas. Further, the inside surface of the two SCV cut outs in the top of the containment dome, which were removed for steam generator replacement, remain uncoated and shall be monitored for wall thickness due to potential corrosion activity. After six cycles of operation, the examination areas have shown no measurable corrosion.

Trending of the examination results showed that corrosion of the liner equal to 10 percent wall loss would not occur for approximately 125 years in one grid location and more than 300 years in all other locations. Based on these trending results, the licensee extended the examination frequency to once every 10 years.

3.3.2 CISI Program History

The licensee provided the CISI Program scope within LAR Tables 3.4.1 1 and 3.4.1 2, "Components Subject to Examination," for Units 1 and 2, respectively. Additionally, a general visual and moisture barrier examination summary was provided.

The staff reviewed the history and results of the CISI program for both units, and found the following were important to the maintenance and safety of the SCV:

- The licensee stated that the most recent visual examinations indicated that there were no relevant indications that required evaluation.

- The two SCV cut outs in the top of the Unit 1 containment dome have shown no measurable corrosion. Trending of the examination results showed that corrosion of the liner equal to 10% wall loss would not occur for approximately 125 years in one grid location and more than 300 years in all other locations.

- The areas around inaccessible areas were examined and no areas have been identified that would indicate there is an issue that would adversely affect structural integrity or leak tightness of the SCV in inaccessible areas.

- No breach or puncture in the metal flashing was identified. The moisture seal barrier was observed not to be separated or cracked, and no indication of moisture intrusion.

- Although moisture barrier material had failed in some of the leak chase channel (LCC) boxes, water chemistry analysis was conducted and found that the water was not ground water, the visual examination concluded that there was no accelerated corrosion on the bottom liner plate, and the moisture barrier materials were replaced and all LCC boxes were re sealed.

The NRC staff observed that there was no indication of the licensee's failure to adequately implement the requirements of its Appendix J, Option B performance based testing program. When the moisture barrier material was found to have failed, the licensee took actions to perform water chemistry analysis, visually examine the liner plate, replace the existing barrier materials, and reseal the LCC boxes.

3.3.3 Containment Protective Coatings Program

As stated in Section 3.4.2 of the LAR, the Watts Bar Protective Coatings Program is designed to install and maintain all Coating Service Level I, II, III, and corrosive environment protective coatings at the quality required to perform their intended function. The licensee stated that these inspections assure conformance to the Watts Bar response to NRC Generic Letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized Water Reactors" (ADAMS Accession No. ML042360586). Recent coatings inspection results were provided in Section 3.4.2 of the LAR.

Based on the licensee's recent coating inspection results and the conclusion that "the total quantity of unqualified coatings remains within the bounds required by design basis, with considerable margin remaining," the staff finds the licensee's response to GL 2004-02 acceptable.

3.3.4 Operating Experience Review

Section 3.5 of the LAR described several NRC Information Notices (INs) and a Regulatory Issue Summary (RIS) concerning containment corrosion and the licensee's review to determine the impact on the Watts Bar, Unit 1 and Unit 2, containments.

The NRC staff reviewed the licensee's evaluations of the INs and RIS concerning containment corrosion. The licensee stated that one IN did not apply to the Watts Bar containments, two INs had already been covered by the current inspection procedure, and revisions were made to the inspection procedure for the other two INs and the RIS. Based on its review of the licensee's evaluation, the NRC staff notes that the licensee had appropriately considered the relevant operating experience for containment corrosion.

3.3.5 Containment Inspection Conclusion

Based on its review of the licensee's containment inspection programs, the NRC staff finds that the licensee has been compliant with the requirements for Watts Bar, Units 1 and 2.

3.4 NEI 94-01, Revision 2-A, Limitations and Conditions

In the NRC SE dated June 25, 2008, the staff concluded that the guidance in NEI 94-01, Revision 2, was acceptable for reference by licensees proposing to amend their TSs to permanently extend the Type A surveillance test interval to 15 years, subject to the limitations and conditions noted within the SE (i.e., Section 4.1). Table 3.6.1-1 of the LAR provides a response to each of these limitations and conditions which are summarized below with NRC staff assessments of each response.

Limitation and Condition 1

> For calculating the Type A leakage rate, the licensee should use the definition in NEI 94-01, Revision 2, in lieu of that in ANSI/ANS [American National Standards Institute/American Nuclear Society]-56.8-2002. (Refer to SE Section 3.1.1.1).

Licensee's Response to Limitation and Condition 1

> TVA will utilize the definition in NEI 94-01, Revision 3-A, Section 5. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.

Staff Assessment of Licensee's Response to Limitation and Condition 1

Section 3.2.9, "Type A test performance criterion," of ANSI/ANS-56.8-2002 defines the "performance leakage rate" and reads in part:

> The performance criterion for a Type A test is met if the performance leakage rate is less than La. The performance leakage rate is equal to the sum of the

> measured Type A test UCL and the total as-left [minimum pathway leakage rate] of all Type B or Type C pathways isolated during performance of the Type A test.

The NRC staff reviewed the definitions of "performance leakage rate" contained in NEI 94-01, Revision 2 and Revision 3-A, and determined that the definitions contained in both documents are identical.

Therefore, the NRC staff finds that because Watts Bar will use the definition found in Section 5.0 of NEI 94-01, Revision 3-A for calculating the Type A leakage rate in the Watts Bar Containment Leakage Rate Testing Program, the licensee has satisfied Limitation and Condition 1.

<u>Limitation and Condition 2</u>

> The licensee submits a schedule of containment inspections to be performed prior to and between Type A tests. (Refer to SE Section 3.1.1.3).

<u>Licensee's Response to Limitation and Condition 2</u>

> TVA will perform a general visual inspection of the accessible interior and exterior surfaces of the primary containment and components prior to the Type A test. Inspections performed between Type A tests are described further in Section 3.4 of the LAR.

<u>Staff Assessment of Licensee's Response to Limitation and Condition 2</u>

The NRC staff's SE Section 3.1.1.3, Enclosure Page 7, for NEI 94-01 Revision 2, reads, in part:

> NEI TR 94-01, Revision 2, Section 9.2.3.2, states that: "To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years." NEI TR 94-01, Revision 2, recommends that these inspections be performed in conjunction or coordinated with the examinations required by ASME Code, Section XI, Subsections IWE and IWL. The NRC staff finds that these visual examination provisions, which are consistent with the provisions of regulatory position C.3 of RG 1.163, are acceptable considering the longer 15-year interval. Regulatory Position C.3 of RG 1.163 recommends that such examination be performed at least two more times in the period of 10 years. The NRC staff agrees that as the Type A test interval is changed to 15 years, the schedule of visual inspections should also be revised. Section 9.2.3.2 in NEI TR 94-01, Revision 2, addresses the supplemental inspection requirements that are acceptable to the NRC staff.

Page 10 of NEI 94-01, Revision 3-A Section 9.2.1 "Pretest Inspection and Test Methodology" reads, in part:

> Prior to initiating a Type-A test, a visual examination shall be conducted of accessible interior and exterior surfaces of the containment system for structural

> problems that may affect either the containment structure leakage integrity or the performance of the Type A test. This inspection should be a general visual inspection of accessible interior and exterior surfaces of the primary containment and components. It is recommended that these inspections be performed in conjunction or coordinated with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE/IWL required examinations.

Page 12 of NEI 94-01, Revision 3-A Section 9.2.3.2 “Supplemental Inspection Requirements” reads:

> To provide continuing supplemental means of identifying potential containment degradation, a general visual examination of accessible interior and exterior surfaces of the containment for structural deterioration that may affect the containment leak-tight integrity must be conducted prior to each Type A test and during at least three other outages before the next Type A test if the interval for the Type A test has been extended to 15 years. It is recommended that these inspections be performed in conjunction or coordinated with the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE/IWL required examinations.

The NRC staff reviewed Section 3.4, “Containment Inspections,” and the Tables contained in Section 3.4.1, “Containment Inservice Inspections,” of the LAR. Tables 3.4.1-1 and 3.4.1-2 of Section 3.4.1 provide a descriptive table of all components which require Subsection IWE and pre-ILRT inspections for both Watts Bar units. Based on this review, the NRC staff has confirmed that the guidance pertaining to IWE inspections and the pre-ILRT primary containment inspections described in the NRC staff SE, Section 3.1.1.3, for NEI 94-01, Revision 3-A, can be satisfied for Watts Bar.

Based on the foregoing discussion, the NRC staff finds that Watts Bar intends to comply with the guidance contained in NEI 94-01, Revision 3-A, Sections 9.2.1 and 9.2.3.2 and to satisfy the provisions contained in NRC staff SE, Section 3.1.1.3.

Accordingly, the NRC staff finds that the licensee has satisfied Limitation and Condition 2.

<u>Limitation and Condition 3</u>

> The licensee addresses the areas of the containment structure potentially subjected to degradation. (Refer to SE Section 3.1.3).

<u>Licensee’s Response to Limitation and Condition 3</u>

> TVA will perform general visual observations of the accessible interior and external surfaces of the containment structure in accordance with containment inspection procedures. Any evidence of structural deterioration is recorded and evaluated or repaired as required. These inspections are described further in Section 3.4 of this enclosure [to the LAR].

<u>Staff Assessment of Licensee’s Response to Limitation and Condition 3</u>

The licensee has conducted containment inservice inspection in accordance with requirements for implementation of the examinations of the SCV, ASME Code Class MC components, in accordance with ASME Code, Section XI, Subsection IWE, 2013 Edition, as conditioned by

10 CFR 50.55a. The most recent examinations performed indicated that there were no relevant indications that required evaluation. The areas around inaccessible areas were examined and, to date, no areas have been identified that would indicate there is an issue that would adversely affect structural integrity or leak tightness of the SCV in inaccessible areas. The licensee recorded no measurable corrosion problem with the steel containments.

Based on its review of the licensee's inspection results, the staff finds that the licensee has satisfied Limitation and Condition 3.

<u>Limitation and Condition 4</u>

> The licensee addresses any tests and inspections performed following major modifications to the containment structure, as applicable. (Refer to SE Section 3.1.4).

<u>Licensee's Response to Limitation and Condition 4</u>

> TVA will implement the staff position with regard to any future post-repair pressure testing following major WBN [Watts Bar], Unit 1 and 2, containment repairs and modifications, as explained in Section 3.1.4 of the NRC staff SE for NEI 94-01, Revision 2. Specifically, with regards to major repairs and modifications, TVA recognizes that, since the issuance of the SE for NEI 94-01, Revision 2, the requirement to perform a Type A test has been removed from ASME Section XI, Subsection IWE-5000, and is now NRC Condition 10CFR50.55a(b)(2)(ix)(J). Following a major modification, as an alternative to performing a Type A test, TVA will perform the following activities:
>
> a) Perform all NDE required by the construction code.
>
> b) Examine the locally welded areas for essentially zero leakage using a bubble test or equivalent.
>
> c) Subject the entire containment to P_a pressure specified in the Technical Specifications for a minimum of 10 minutes.
>
> d) Perform a general visual examination of the accessible portions of the interior and exterior surfaces of containment in accordance with ASME B&PV [Boiler and Pressure Vessel] Code, Subsection IWE.

<u>Staff Assessment of Licensee's Response to Limitation and Condition 4</u>

As indicated in TVA's response to Limitation and Condition 4 above, the licensee plans to implement an alternative to the requirement in the regulatory condition on IWE-5000 in 10 CFR 50.55a(b)(2)(ix)(J) of performing an Appendix J, Type A, test following a major containment repair/replacement activity. The regulation at 10 CFR 50.55a(z) permits alternatives to the requirements of paragraphs (b) through (h) of 10 CFR 50.55a or portions thereof when authorized by the NRC.

In order to adopt NEI 94-01, Revisions 2 and 3, a licensee has to address the limitations and conditions of both SEs. Because Limitation and Condition 4 in the SE for Revision 2 refers to the staff position in SE Section 3.1.4, which focuses major and minor containment repairs and modifications and observes that a short duration test may be an acceptable alternative to a Type A test, the NRC staff has evaluated TVA's proposed alternative. The staff's evaluation is provided below.

Staff Evaluation of Alternative to Type A Test

Section 50.55a(g)(4) to 10 CFR requires that the inservice inspection (ISI) of the pressure retaining components of steel (Class MC) and concrete (Class CC) containments meet the requirements set forth in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PV) Code and Addenda that are incorporated by reference in 10 CFR 50.55a(b), subject to the condition listed in paragraph (b)(2)(vi), and the conditions listed in paragraphs (b)(2)(viii) and (b)(2)(ix).

As stated in 10 CFR 50.55a(b)(2)(ix)(J), a repair/replacement activity such as cutting a large construction opening in the containment pressure boundary to replace steam generators (SGs), reactor vessel heads, pressurizers, or other major equipment is considered a major containment modification. The requirement goes on to say that "when applying IWE–5000 to Class MC pressure-retaining components, any major containment modification or repair/replacement must be followed by a Type A test to provide assurance of both containment structural integrity and leak-tight integrity prior to returning to service." This regulatory condition is intended to ensure that the post-repair pressure test provides a verification of both the structural and leak-tight integrity of the restored containment.

The Watts Bar, Unit 1 third 10-year ISI began on September 9, 2018, and is scheduled to end on September 8, 2028. The Watts Bar, Unit 2, initial 10-year ISI began on October 19, 2016, and is scheduled to end on October 18, 2026. The applicable ASME B&PV Code, Section XI, edition and addenda for both units, for Article IWE-5000, is the 2013 Edition (ADAMS Accession No. ML18227A599), and is in effect until the end of the current containment ISI intervals.

As stated in the LAR, TVA had planned to replace the Watts Bar, Unit 2, SGs during the Unit 2 Cycle 5 refueling outage. The replacement is now scheduled to take place during the Cycle 4 refueling outage in spring 2022. The replacement activities will require the creation and restoration of construction openings in the shield building and the pressure boundary of the free-standing steel containment vessel to provide access for the removal of the original SGs as well as the installation of the replacement SGs.

As stated in the response to Limitation and Condition 4, the post-maintenance testing following a major repair will include all non-destructive examinations required by the construction code (Item a of the proposed alternative). The NRC staff finds that the performance of these examinations will provide verification of the quality of the repair welds prior to being subject to the leakage test and structural integrity test.

The licensee has proposed to perform a local leakage test of the welds affected by the repair/replacement activities using a bubble test with acceptance criterion of essentially zero detectable leakage (Item b of the proposed alternative). The NRC staff finds that a bubble leakage test will provide a post-repair verification of the leakage integrity of the welds affected by the repair/replacement and, therefore, the leakage integrity of the restored containment vessel.

In addition, the licensee has proposed to subject the entire containment to the P_a pressure specified in its technical specifications for a minimum of 10 minutes (Item c of the proposed alternative). The NRC staff finds that an uneventful structural response, in general, and specifically in the repair area, during and after pressurizing the entire containment to the test pressure and holding for at least 10 minutes will provide verification of the structural integrity of the repaired area as well as the restored containment vessel.

Lastly, the licensee has proposed to perform a general visual examination of the accessible portions of the interior and exterior surfaces of containment in accordance with ASME B&PV Code, Subsection IWE (Item d of the proposed alternative). The NRC staff finds that visual examination of accessible internal and external surfaces of the steel containment following local leak testing and pressurization of the containment will permit the identification of any flaws or other degraded or abnormal conditions (e.g., deformation, cracks, etc.) that may have occurred during these tests.

Based the staff's evaluation above, and the fact that Watts Bar, Units 1 and 2, have successfully completed two consecutive Appendix J, Type A, tests (see LAR Tables 3.3.1-1 and 3.3.1-2), the NRC staff finds that the licensee's proposed alternative to the Appendix J, Type A, test will provide a verification of both the structural integrity as well as the leakage integrity of the restored containment vessel.

The NRC staff finds that the proposed alternative activities provide an acceptable level of quality and safety after major containment repairs and modifications, as required by 10 CFR 50.55a(z)(1). Therefore, the NRC staff authorizes the proposed alternative at Watts Bar, Units 1 and 2, for use during the current 10-year ISI intervals for Watts Bar, Unit 1 that commenced on September 9, 2018, and is scheduled to end on September 8, 2028, and for Watts Bar, Unit 2, that commenced October 19, 2016, and is currently scheduled to end on October 18, 2026.

With the authorization of the alternative activities described above, the NRC staff finds that the licensee has addressed the tests and inspections for both containment structural integrity and leak-tight integrity prior to returning to service after a major modification and, therefore, has satisfied Limitation and Condition 4.

Limitation and Condition 5

> The normal Type A test interval should be less than 15 years. If a licensee has to utilize the provision of Section 9.1 of NEI 94-01, Revision 2, related to extending the ILRT interval beyond 15 years, the licensee must demonstrate to the NRC staff that it is an unforeseen emergent condition. (Refer to SE Section 3.1.1.2).

Licensee's Response to Limitation and Condition 5

> TVA will comply with this condition. Extensions of the Type A test frequency beyond 15 years are used only to accommodate unforeseen emergent conditions.

Staff Assessment of Licensee's Response to Limitation and Condition 5

The staff concludes that the licensee's statement that it will comply with this condition is acceptable and the licensee has satisfied Limitation and Condition 5.

Limitation and Condition 6

> For plants licensed under 10 CFR Part 52, applications requesting a permanent extension of the ILRT surveillance interval to 15 years should be deferred until after the construction and testing of containments for that design have been completed and applicants have confirmed the applicability of NEI 94-01, Revision 2, and EPRI Report No. TR-1009325, Revision 2, including the use of past containment ILRT data.

Licensee's Response to Limitation and Condition 6

> Not applicable. WBN [Watts Bar], Units 1 and 2, were not licensed pursuant to 10 CFR Part 52.

NRC Staff Assessment of Licensee's Response to Limitation and Condition 6

Limitation and Condition 6 is only applicable to plants licensed under 10 CFR Part 52. The Watts Bar, Units 1 and 2, licenses were issued under 10 CFR Part 50; therefore, this item is not applicable.

Conclusion Related to the Six Limitations and Conditions Listed in NEI 94-01, Revision 2-A, Section 4.1, of the NRC SE

The NRC staff evaluated each of the six limitations and conditions listed above and determined that the licensee adequately satisfied all of the limitations and conditions identified in NEI 94-01, Revision 2-A, Section 4.1, of the NRC SE. Therefore, the NRC staff finds it acceptable for Watts Bar to adopt the "conditions and limitations" of NEI 94-01, Revision 2-A, SE Section 4.1, as part of the implementation documents listed in TS 5.7.2.19.

3.5 NEI 94-01, Revision 3-A, "Conditions"

The NRC published an SE with limitations and conditions for NEI 94-01, Revision 3, by letter dated June 8, 2012. In the SE, the NRC concluded that NEI 94-01, Revision 3, describes an acceptable approach for implementing the optional performance-based requirements of Appendix J, and is acceptable for reference by licensees proposing to amend their containment leakage rate testing TSs, subject to two conditions discussed below. The SE was incorporated into Revision 3 and subsequently issued as NEI 94-01, Revision 3-A, on July 31, 2012.

The LAR proposed to use NEI 94-01, Revision 3-A, as the implementation document for the leak rate testing program. Accordingly, Watts Bar, Units 1 and 2, will be adopting, in part, the testing criteria of ANSI/ANS 56.8–2002 as part of its licensing basis. As stated in NEI 94-01, Revision 3-A, Section 2.0, "Purpose and Scope," where technical guidance overlaps between NEI 94-01, Revision 3-A, and ANSI/ANS 56.8–2002, the guidance in NEI 94-01, Revision 3-A, takes precedence.

Topical Report Condition 1

The June 8, 2012, NEI 94-01, Revision 3, SE, Section 4.0, Condition 1, stipulates that:

> NEI 94-01, Revision 3, is requesting that the allowable extended interval for Type C LLRTs be increased to 75 months, with a permissible extension (for

> non-routine emergent conditions) of nine months (84 months total). The staff is allowing the extended interval for Type C LLRTs be increased to 75 months with the requirement that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit. In addition, a corrective action plan shall be developed to restore the margin to an acceptable level. The staff is also allowing the non-routine emergent extension out to 84-months as applied to Type C valves at a site, with some exceptions that must be detailed in NEI 94-01, Revision 3. At no time shall an extension be allowed for Type C valves that are restricted categorically (e.g. BWR MSIVs), and those valves with a history of leakage, or any valves held to either a less than maximum interval or to the base refueling cycle interval. Only non-routine emergent conditions allow an extension to 84 months.

Condition 1 identifies three issues that are required to be addressed:

(1) The allowance of an extended interval for Type C LLRTs of 75 months requires that a licensee's post-outage report include the margin between the Type B and Type C leakage rate summation and its regulatory limit;

(2) A corrective action plan is to be developed to restore the margin to an acceptable level; and

(3) Use of the allowed 9-month extension for eligible Type C valves is only allowed for non-routine emergent conditions, but not for valves categorically restricted and other excepted valves.

<u>Licensee's Response to Condition 1, Issue 1</u>

> The post-outage report will include the margin between the Type B and Type C Minimum Pathway Leak Rate (MNPLR) summation value, as adjusted to include the estimate of applicable Type C leakage understatement, and its regulatory limit of 0.60 La. TVA will establish an administrative limit to provide margin to the regulatory limit of 0.60 La.

<u>Licensee's Response to Condition 1, Issue 2</u>

> When the potential leakage understatement adjusted Type B and Type C MNPLR total is greater than the WBN [Watts Bar] administrative leakage summation limit, but less than the TS limit of 0.60 La, then an analysis and determination of a corrective action plan will be prepared to restore the leakage summation margin to less than the WBN [Watts Bar] leakage limit. This plan will focus on those components which have contributed the most to the increase in the leakage summation value and the manner of timely corrective action that best focuses on the prevention of future component leakage performance issues so as to maintain an acceptable level of margin.

<u>Licensee's Response to Condition 1, Issue 3</u>

> TVA will apply the 9-month extension period only to eligible Type C components for non-routine emergent conditions. Such occurrences will be documented in the record of the tests. This non-routine extension is not allowed for valves

> specifically restricted to a maximum 30 month interval or any valve held to less than a maximum interval or the base interval (30 months).

NRC Staff Assessment of Licensee's Response to Condition 1

The NRC staff has reviewed the requirements of NEI 94-01, Revision 3 against the TVA response for Issues (1), (2), and (3) of Topical Report Condition 1. The Licensee response indicates that following approval of the subject amendment, TVA's actions will be consistent with the guidance of NEI 94-01, Revision 3-A. The NRC staff notes that revised guidance contained in Revision 3-A, Section 10.1, "Introduction"; Section 10.2.3, "Corrective Actions"; Section 11.3.2, "Programmatic Controls"; and Section 12.1, "Report Requirements," reflects the NRC staff's SE input pertaining to Issues (1), (2), and (3). The NRC staff finds that TVA acknowledges the requirements of Condition 1 and that the licensee intends for Watts Bar to comply with these requirements.

Topical Report Condition 2

The NRC SE dated June 8, 2012, Section 4.0, Condition 2, stipulates that:

> The basis for acceptability of extending the ILRT interval out to once per 15 years was the enhanced and robust primary containment inspection program and the local leakage rate testing of penetrations. Most of the primary containment leakage experienced has been attributed to penetration leakage and penetrations are thought to be the most likely location of most containment leakage at any time. The containment leakage condition monitoring regime involves a portion of the penetrations being tested each refueling outage, nearly all LLRT's being performed during plant outages. For the purposes of assessing and monitoring or trending overall containment leakage potential, the as-found minimum pathway leakage rates for the just tested penetrations are summed with the as-left minimum pathway leakage rates for penetrations tested during the previous 1 or 2 or even 3 refueling outages. Type C tests involve valves which, in the aggregate, will show increasing leakage potential due to normal wear and tear, some predictable and some not so predictable. Routine and appropriate maintenance may extend this increasing leakage potential. Allowing for longer intervals between LLRTs means that more leakage rate test results from farther back in time are summed with fewer just tested penetrations and that total used to assess the current containment leakage potential. This leads to the possibility that the LLRT totals calculated understate the actual leakage potential of the penetrations. Given the required margin included with the performance criterion and the considerable extra margin most plants consistently show with their testing, any understatement of the LLRT total using a 5-year test frequency is thought to be conservatively accounted for. Extending the LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI 94-01, Revision 3, Section 12.1.
>
> When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the primary containment leakage rate testing program trending or monitoring must include an estimate of the amount of understatement in the Type B & C total, and must be included in a licensee's post-outage report. The

report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Condition 2 identifies two issues that are required to be addressed:

(1) Extending the Type C LLRT intervals beyond 5 years to a 75-month interval should be similarly conservative, provided an estimate is made of the potential understatement and its acceptability determined as part of the trending specified in NEI 94-01, Revision 3, Section 12.1; and

(2) When routinely scheduling any LLRT valve interval beyond 60-months and up to 75-months, the Primary Containment Leakage Rate Testing Program trending or monitoring must include an estimate of the amount of understatement in the Type B and Type C total, and must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations.

Additionally, NEI 94-01, Revision 3-A has a margin related requirement contained in Section 12.1, "Report Requirements":

> A post–outage report shall be prepared presenting results of the previous cycle's Type B and Type C tests, and Type A, Type B, and Type C tests, if performed during that outage. The technical contents of the report are generally described in ANSI/ANS-56.8–2002, and shall be available on–site for NRC review. The report shall show that the applicable performance criteria are met, and serve as a record that continuing performance is acceptable. The report shall also include the combined Type B and Type C leakage summation, and the margin between the Type B and Type C leakage rate summation and its regulatory limit. Adverse trends in the Type B and Type C leakage rate summation shall be identified in the report and a corrective action plan developed to restore the margin to an acceptable level.

Licensee's Response to Condition 2, Issue 1

TVA will apply a potential leakage understatement adjustment factor to the actual As-Left leak rate. This will result in a combined conservative Type C total for all 75-month LLRTs being "carried forward" and will be included when the total summation is required to be updated (either while online or following an outage).

Licensee's Response to Condition 2, Issue 2

A post-outage report is prepared with results of the tests performed during that outage. The report will show that the applicable performance criteria are met and serve as a record that continuing performance is acceptable. If an adverse trend in the potential leakage understatement is identified, then a corrective action plan is prepared, focused on those components which have contributed the most to the adverse trend in the leakage summation value.

Licensee's Response to Margin-Related Requirement in NEI 94-01, Revision 3-A, Section 12.1

> At WBN [Watts Bar], in the event an adverse trend in the aforementioned potential leakage understatement adjusted Type B and C summation is identified, then an analysis and determination of a corrective action plan will be prepared to restore the trend and associated margin to an acceptable level. The corrective action plan will focus on those components that have contributed the most to the adverse trend in the leakage summation value and the manner of timely corrective action, as deemed appropriate that best focuses on the prevention of future component leakage performance issues.

NRC Staff Assessment of Licensee's Response to Condition 2

The NRC staff has reviewed the requirements of NEI 94-01, Revision 3 against the TVA response to Issues (1) and (2) of Condition 2, as well as to the margin related requirement in Section 12.1 of Revision 3-A. The licensee's response indicates that following approval of the proposed amendment, TVA's actions will be consistent with the guidance of NEI 94-01, Revision 3-A. The NRC staff notes that revised guidance contained in Revision 3-A, Section 11.3.2 "Programmatic Controls," and Section 12.1, "Report Requirements," reflects the NRC staff's SE input pertaining to both Issues (1) and (2). The NRC staff finds that TVA acknowledges the requirements of Condition 2 and that the licensee intends for Watts Bar to comply with Condition 2.

3.6 NEI 94-01, Revision 3-A, SE Section 4.0, "Conclusion"

Based on the above evaluation of each condition, the NRC staff determined that the licensee has adequately addressed both conditions in Section 4.0 of the NRC SE for NEI 94-01, Revision 3. Therefore, the NRC staff finds it acceptable for TVA to adopt NEI 94-01, Revision 3-A, as the implementation document in TS 5.7.2.19 for Watts Bar.

3.7 Probabilistic Risk Assessment of the Proposed Extension of the ILRT Test Intervals

3.7.1 Background

Section 9.2.3.1, "General Requirements for ILRT Interval Extensions beyond Ten Years," of NEI 94-01, Revision 2-A, states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," of NEI 94-01 states that the assessment should be performed using the approach and methodology described in EPRI TR-1009325, Revision 2-A[2], "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals." The analysis is to be performed by the licensee and retained in the plant documentation and records as part of the basis for extending the ILRT interval.

In the SE, dated June 25, 2008, the NRC staff found the methodology in EPRI TR-1009325, Revision 2, acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied.

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[2] It should be noted that EPRI TR-1009325, Revision 2-A, is also identified as EPRI TR-1018243. This report is publicly available and can be found at www.epri.com by typing "1018243" in the search field box.

These conditions, set forth in Section 4.2 of the SE for EPRI TR-1009325, Revision 2, stipulate that:

1. The licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

2. The licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT surveillance interval to 15 years is small and consistent with the clarification provided in Section 3.2.4.6[3] of this SE.

3. The methodology in EPRI TR-1009325, Revision 2, is acceptable provided the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by licensees is assigned a value of 100 times the maximum allowable leakage rate (L_a) instead of 35 L_a.

4. A LAR is required in instances where containment over-pressure is relied upon for emergency core cooling system (ECCS) performance.

3.7.2 Plant-Specific Risk Evaluation

The licensee performed a risk impact assessment for extending the Type A containment ILRT interval from 10 years to 15 years. The risk analyses for Watts Bar was provided in Enclosure 2 of the LAR.

In Section 3.7.1 of the LAR, the licensee stated that the plant-specific risk assessment for Watts Bar follows the guidance in:

- Appendix H of Electric Power Research Institute, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals: Revision 2-A of 1009325," EPRI TR-1018243, October 2008

- Electric Power Research Institute, "Risk Impact Assessment of Revised Containment Leak Rate Testing Intervals," EPRI TR-104285, August 1994

- Nuclear Regulatory Commission, "Performance-Based Containment Leak-Test Program," NUREG-1493, September 1995 (ADAMS Accession No. ML20098D498)

- Calvert Cliffs liner corrosion analysis described in a letter to the NRC dated March 27, 2002 (ADAMS Accession No. ML020920100)

The LAR stated that the plant-specific risk assessment follows the guidance in NEI 94-01, Revision 3-A, the methodology described in EPRI TR-1018243 (also identified as EPRI TR-1009325, Revision 2-A), and the NRC regulatory guidance outlined in RG 1.174.

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[3] The SE for EPRI TR-1009325, Revision 2, indicates that the clarification regarding small increases in risk is provided in Section 3.2.4.5; however, the clarification is provided in Section 3.2.4.6.

Regarding risk relating to Type C testing, the NRC SE in NEI 94-01, Revision 3-A stated:

> EPRI TR-104285 provided a risk impact assessment of alternative testing intervals for both ILRT and LLRT. Risk involved in conducting Type C LLRTs on extended intervals (using population dose as the metric) was determined using valve leakage performance data obtained from industry by the NEI. This pre-1995 data was very conservatively applied in the risk impact assessment by assuming that the leakage magnitude for a penetration would be that associated with the valve in the penetration that exceeded its administrative limit. As stated above, the recent (post-1995) failure rate data indicates that the failure rate of Type C valves tested on extended intervals was significantly less than the failure rate for the general population of Type C valves tested pre-1995. NEI states that this 1994 risk impact assessment remains conservative and valid based on the application of this recent data to assess the risk involved with testing valves that qualify for testing on extended intervals in accordance with NEI 94-01. The NRC staff agrees.

The licensee addressed each of the four conditions for the use of EPRI TR-1009325, Revision 2-A, which are listed in Section 4.2 of the NRC SE. The four conditions that are being evaluated are (1) PRA Quality (referred to as Technical Adequacy of PRA), (2) Estimated Risk Increases, (3) Leak Rate for the Large Pre-Existing Containment Leak Rate Case, and (4) Determining if Containment Overpressure is Relied Upon for ECCS Performance. A summary of how each condition has been met is provided in the sections below.

3.7.2.1 Technical Adequacy of PRA – Condition 1

Condition 1 in Section 4.2 of the SE for EPRI TR-1009325, Revision 2, stipulates that the licensee submits documentation indicating that the technical adequacy of their PRA is consistent with the guidance of RG 1.200 relevant to the ILRT extension application. This regulatory guide describes one acceptable approach for determining whether the technical adequacy of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for light-water reactors.

Also in Section 3.2.4.1 of the SE to EPRI TR-1009325, the NRC staff stated that Capability Category (CC) I of ASME PRA standard shall be applied as the standard for assessing PRA quality for ILRT extension applications, as approximate values of core damage frequency (CDF) and large early release frequency (LERF) and their distribution among release categories are sufficient to support the evaluation of changes to ILRT frequencies.

The Watts Bar, Units 1 and 2, PRA technical adequacy is addressed in Section 4.0 of Enclosure 2, “PRA Evaluation,” of the LAR. The LAR stated that the PRA model includes an Internal Events (IE) PRA model, which includes Internal Flooding (IF), and a Seismic PRA (SPRA) model for the Watts Bar power plant. The IE model is stated to have been developed in accordance with the requirements in RG 1.200, Revision 2, and has been subjected to peer reviews and the Facts and Observations (F&O) closure process. The SPRA model is stated to be peer reviewed to the ASME/PRA Standard RA-Sb-2013, “Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications.”

The Watts Bar IE and IF PRA were subjected to a full scope peer review in November of 2009, and the peer review was conducted using NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using ASME/ANS PRA Standard." The conclusions of the IE peer review are stated in the LAR, Enclosure 2, Section 4.4. The peer review was performed against the RA-Sa-2009 standard, which is endorsed in RG 1.200, Revision 2. The peer review resulted in 50 F&Os. The 50 F&Os were subjected to a Peer Review Closure process in June 2017 in accordance with RG 1.200. Of the 50 F&Os, the closure peer review team concluded that 43 F&Os met the requirements for closure leaving 7 open F&Os, these 7 F&Os are found in Table 6 of Enclosure 2. The licensee stated that these seven open F&Os have no or negligible impact on the LERF, delta LERF, dose, or CCDP results. The NRC staff reviewed the seven open F&Os and determined that there is negligible or no impact on the Watts Bar, Units 1 and 2, risk results, and that these open F&Os are not risk significant for this review.

The seismic PRA is developed using and subjected to a peer review following the guidance contained in NEI 12-13, "External Hazards PRA Review Process Guidelines" (ADAMS Accession No. ML12240A027) and the guidance contained in the ASME/PRA Standard RA-Sb-2013. In 2016, the Watts Bar seismic PRA model was subjected to a peer review. In this peer review 74 "finding" level F&Os were identified. In April of 2017, the licensee's closure review team determined that all but one of the findings could be closed. As for the remaining finding, the licensee determined the seismic finding only pertains to documentation enhancements and causes no change to the calculated seismic PRA CDF or LERF. With no changes to these values, there would be no changes to the PRA of the containment ILRT extension. The NRC staff reviewed the open F&O and determined that there is negligible or no impact on the Watts Bar, Units 1 and 2 risk results, and that the one open F&O is not risk significant for this review.

In Section 4.6 and 4.7 of Enclosure 2 of the LAR, the licensee stated that the IF, high winds, external flooding, and other hazards that were not modelled were evaluated and the results of those evaluations are summarized in Table 9 of the LAR. The NRC staff reviewed the results in Table 9 and confirmed there are no unique vulnerabilities identified for these additional hazards relevant to this LAR.

Based on the review of the above information, the NRC staff finds that the licensee has addressed the relevant findings and gaps from the peer reviews and that they have no impact on the results of this application. Therefore, the NRC staff concludes that the PRA models used by the licensee are of sufficient quality to support the evaluation of changes to ILRT frequencies. Accordingly, Condition 1 is met.

3.7.2.2 Estimated Risk Increase – Condition 2

Condition 2 stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is "small," consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2-A. Additionally, for plants that rely on containment over-pressure for net positive suction head for ECCS injection, both CDF and LERF will be considered in the ILRT evaluation and compared with the risk acceptance guidelines in RG 1.174. This last point is not applicable given that Watts Bar, Units 1 and 2, does not rely on containment over-pressure for ECCS performance and the effect of extending the containment ILRT interval has no effect on core damage frequency.

Regulatory Guide 1.174 defines “very small” changes in risk as resulting in increases of LERF of less than 1.0E-07/reactor year. The RG also states that when the calculated increase in LERF is in the “small” range of 1.0E-07 per reactor year to 1.0E-06 per reactor year, applications will be considered only if it can be reasonably shown that the total LERF is less than 1.0E-05 per reactor year. Given the above information, the associated risk metrics include LERF, population dose, and conditional containment failure probability (CCFP).

The LAR provided the results of the plant-specific risk assessment for a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years in Section 3.7 of Enclosure 1 and provided the details of the calculations in Enclosure 2, Section 5.4. The LAR stated that the increase in LERF resulting from a change in the Type A ILRT test interval from 3 in 10 years to 1 in 15 years is estimated to be 1.90E-7/reactor year for both Watts Bar units using the EPRI guidance. In addition, the LAR provided the calculated total LERF is 1.61E-6/reactor year and 1.60E-6/reactor year, for Units 1 and 2, respectively. The NRC finds that these values are within the ranges specified in RG 1.174, as such, the estimated change in LERF is determined to be “small” using the acceptance guidelines of RG 1.174.

The LAR provided the resulting population doses for the Type A test frequency change to 1 per 15 years to be 5.76E-2 person-rem/year for Watts Bar, Unit 1, and 5.73E-2 person-rem/year for Watts Bar, Unit 2. These population doses correspond to roughly 0.4 percent increases for both units. The LAR provided the details of this assessment in Section 7.3 of Enclosure 2. The guidance contained within NEI 94-01 states that a “small” total population dose is defined as an increase of ≤ 1.0 person-rem/year, or ≤ 1 percent of the total population dose, whichever is less restrictive for the risk impact assessment of the extended ILRT intervals. The reported increase in total population dose is below the acceptance criteria provided in EPRI TR-1009325, Revision 2-A, and defined in Section 3.2.4.6 of the NRC SE for NEI 94-01, Revision 2. Given the analysis showing an increase of less than 1 person-rem/year for Watts Bar, Units 1 and 2, the increase to population dose is “small.”

The CCFP is discussed in Section 7.4 of Enclosure 2 of the LAR. The increase in the CCFP due to the change in test frequency from 3 in 10 years to 1 in 15 years is 0.908 percent for Watts Bar, Units 1 and 2. NEI 94-01 states that increases in CCFP of ≤ 1.5 percent is considered “small.” The values provided in the LAR are below the acceptance guidelines in Section 3.2.4.6 of the NRC SE for NEI 94-01 and, therefore, can be considered “small.”

Based on the risk assessment results, the NRC staff finds that the increase in LERF is “small” and consistent with the acceptance guidelines of RG 1.174, and the increase in the total population dose and the magnitude of the change in the CCFP for the proposed change are also “small.” The defense-in-depth philosophy is maintained as the independence of barriers will not be degraded because of the requested change, and the use of the quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, Condition 2 is met.

3.7.2.3 Leak Rate for the Large Pre-Existing Containment Leak Rate Case – Condition 3

Condition 3 stipulates that in order to meet the guidance in EPRI TR-1009325, Revision 2, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensee shall be 100 L_a instead of 35 L_a. As noted in Section 3.7.1 of Enclosure 1 of the LAR, the methodology in EPRI TR-1009325, Revision 2-A, incorporated the use of 100 L_a as the average leak rate for the pre-existing containment large leakage rate accident case (i.e., accident case 3b). This value has been used in the Watts Bar risk assessments. Accordingly, the NRC staff finds that Condition 3 is met.

3.7.2.4 Applicability if Containment Over-Pressure is Credited for ECCS Performance – Condition 4

Condition 4 stipulates that in instances where containment over-pressure is relied upon for ECCS performance, a LAR is required to be submitted. In Section 3.7.3 of Enclosure 1 of the LAR, the licensee stated that Watts Bar does not rely upon containment over-pressure for ECCS performance. Because the licensee does not rely upon containment over-pressure for ECCS performance, Condition 4 does not apply.

3.8 Staff Evaluation of Proposed TS Changes

3.8.1 Proposed Changes to TS 5.7.2.19

As noted above, the LAR proposed to revise Watts Bar, Units 1 and 2, TS 5.7.2.19, by adopting NEI 94-01, Revision 3-A, as the implementation document for the performance-based Option B of 10 CFR Part 50, Appendix J. The proposed LAR would extend the Type A CILRT interval from 10 years to 15 years, and the Type C LLRT intervals from 60 months to 75 months.

For both Watts Bar, Units 1 and 2, the LAR proposed to modify TS 5.7.2.19 to replace the reference to RG 1.163 to NEI 94-01, Revision 3-A, and Section 4.1, “Limitations and Conditions for NEI 94-01, Revision 2,” of the NRC SE in NEI 94-01, Revision 2-A, dated October 2008.

As required by 10 CFR 50.36(c)(5), “Administrative controls,” the proposed change ensures that the TSs direct operation of the facility in a safe manner as described in this safety evaluation. Changes to other parts of the TS are not proposed. Due to the above assessment, the NRC staff finds the proposed changes to the TSs acceptable.

3.8.2 Use of a Bounding Value for P_a

As part of the LAR, the licensee proposed a clarification of the value of P_a to be used for containment leakage testing purposes. Appendix J to 10 CFR Part 50 defines P_a as “the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specification or associated bases.” Watts Bar TS 5.7.2.19 currently defines P_a as 15.0 psig. To support the LAR, TVA conducted an evaluation for a range of P_a pressures, and proposes to allow the use of a lower pressure in future LLRT testing. TVA is requesting a bounding value for P_a to minimize the impact on related documents when the calculated P_a is changed. As an example, TVA state that if the calculated P_a changes, then revisions of TS 5.2.7.19 and approximately 43 containment leak rate test procedures, among others, would be required. Use of a bounding value for P_a eliminates that burden and provides a similar level of quality and safety.

With respect to the LLRT test pressure, TVA contracted Kalsi Engineering to perform a detailed evaluation to determine whether an increase in seat leakage is expected when the differential pressure (LLRT test pressure) is reduced from 16.5 psig to 9.0 psig. TVA indicated that the maximum LLRT test pressure of 16.5 psig equates to 1.1 times the historical P_a value of 15.0 psig, which is the current maximum LLRT test pressure allowed by the Watts Bar specific LLRT procedures. Watts Bar TS 5.7.2.19 currently defines P_a as the peak calculated containment internal pressure for the DBLOCA of 15.0 psig. TVA selected a lower limit of

9.0 psig for the evaluation to provide margin below the current calculated P_a of 9.36 psig in case of future changes to the calculations.

TVA provided proprietary and non-proprietary versions of Kalsi Engineering Report No. 3960C (Revision 0), “Evaluation of Higher Test Pressure on Leakage for Watts Bar,” for NRC staff review. The non-proprietary version of the Kalsi Engineering report (Enclosure 4 of the LAR) described the evaluation of potential seat leakage based on seat contact force for a wide range of valves. From the use of analysis, industry data, and supporting test data (as available), the Kalsi Engineering report demonstrated that an increase in seat leakage is not expected when the LLRT test pressure is reduced from 16.5 psig to a lower limit of 9.0 psig. However, the Kalsi Engineering report indicated that hard-seated check valves in three design groups had low seating stress and a significant decrease in seat load from the 16.5 psig to 9.0 psig test pressure and recommended that one check valve from each group be tested to support the calculation-based conclusions.

In its letter dated December 15, 2020, TVA submitted a LAR supplement to provide the results of the confirmatory tests recommended in the Kalsi Engineering report. In the LAR supplement, TVA described its actions to modify the LLRT procedures to measure leakage as test pressure was increased from zero to the range of 9.36 to 9.5 psig, and then to measure leakage again as the test pressure was raised to the normal LLRT pressure of 15.5 to 16.0 psig. Based on the measured leakage rates, the confirmatory tests demonstrated that the valve leakage rates increase as pressure increases. Therefore, the confirmatory tests support the assumption that testing of the hard-seated check valves at a higher pressure of 15.5 to 16.0 psig versus 9.36 psig will not decrease the component leakage rate, and will not adversely affect the total containment leakage rate in comparison to the TS leakage rate acceptance criteria.

As part of the evaluation of potential seat leakage for specific valves, the Kalsi Engineering report included tables indicating that the LLRT leakage history for some valves in various groups at Watts Bar had “unfavorable” leakage history. However, the Kalsi Engineering report did not describe an evaluation or resolution of this leakage history. In response to a request for additional information from the NRC staff, the licensee submitted a letter dated April 29, 2021 (ADAMS Accession No. ML21119A248), which described its activities to address the unfavorable LLRT leakage history for the specific valves indicated in the Kalsi Engineering report. For example, the licensee stated that for each LLRT performance where the measured leakage rate exceeded the specified administrative limit, the issue was entered into the TVA corrective action program. In such a situation, the licensee indicated that the LLRT personnel at Watts Bar would perform troubleshooting activities to determine the source of the leakage. If an individual component LLRT exceeded the specified administrative limit during two consecutive periodic performances, the licensee stated that those components were entered into the 10 CFR 50.65 Maintenance Rule Program for cause determination and evaluation at Watts Bar, except in one described circumstance without an impact on this evaluation. In its submittal, the licensee provided a brief description of the activities performed or planned for each valve identified in the Kalsi Engineering report with an unfavorable leakage history. For example, the range of licensee activities to address unfavorable leakage for the specific valves included: (1) valve disassembly, cleaning, and inspection; (2) corrective maintenance, such as valve disc modifications, internal surface refurbishing, and parts replacement; and (3) evaluations of the acceptability of specific valve leakage based on overall containment performance.

Based on its review, the NRC staff finds that the licensee has provided adequate support for the proposed changes to the range of LLRT test pressures included in the LAR. This support is based on the determination that an increase in seat leakage will not occur for the subject valves when the LLRT test pressure is reduced from 16.5 psig to 9.0 psig. Further, the NRC staff finds that the licensee has demonstrated that the LLRT leakage history for specific valves indicated in the Kalsi Engineering report was properly addressed at Watts Bar without impacting the proposed plant-specific changes in the LAR.

3.8.2.1 Deviation No. 1 from ANSI/ANS 56.8-2002

In Section 3.8.3 of the LAR, the licensee described Deviation No. 1 from ANSI/ANS 56.8-2002 as related to the allowable leakage rate (La). ANSI/ANS 56.8-2002 defines La (wt%/24 h) as the "maximum allowable Type A test leakage rate at P_a." Further, ANSI/ANS 56.8-2002 defines P_a (psig) as the "calculated peak containment internal pressure related to the design-basis loss-of-coolant accident (LOCA)." As illustrated in the LAR, L_a is a mass leakage rate with a direct correlation to P_a pressure when converted to a volumetric leak rate. As a result, a bounding P_a of 15.0 psig will result in a higher allowable leakage rate, L_a, than what would be permitted if using the calculated P_a value of 9.36 psig.

The licensee stated that the value for La at Watts Bar has been based on a P_a value of 15.0 psig since initial startup and commercial operation of the nuclear power plant. As described in TS Bases B3.6.1, "Applicable Safety Analysis," the associated dose analysis has used an L_a based-on P_a of 15.0 psig. Based on this historical use of P_a of 15.0 psig in the dose analysis, the NRC staff finds Deviation No. 1 from ANSI/ANS 56.8-2002 to be acceptable as described in the LAR where the bounding P_a of 15.0 psig would result in a higher allowable leakage rate, L_a, than what would be permitted if using the calculated P_a value of 9.36 psig.

3.8.2.2 Deviation No. 2 from ANSI/ANS 56.8-2002

In Section 3.8.4 of the LAR, the licensee described Deviation No. 2 from ANSI/ANS 56.8-2002 as related to the maximum test pressure. ANSI/ANS 56.8-2002 limits the maximum Type B and Type C test pressure to 1.1 times P_a for those components where a higher differential pressure results in increased sealing. The licensee considered this restriction to be generically worded to apply to a broad range of component designs and is intended to prevent the use of a higher test pressure as a means to reduce the component leakage rate.

The licensee reported that it reviewed all Watts Bar components that are Type B and C tested to identify the scope of components where a higher differential pressure might increase leakage sealing. As discussed above, the licensee contracted Kalsi Engineering to perform a detailed evaluation to determine whether an increase in seat leakage is expected when the differential pressure (LLRT test pressure) is reduced from 16.5 psig to 9.0 psig. The licensee stated that the Kalsi Engineering evaluation determined that an increase in seat leakage is not expected when the LLRT test pressure is reduced from 16.5 psig to a bounding lower limit of 9.0 psig. The NRC staff discusses its review of the Kalsi Engineering report in Section 3.8.2 above. Based on its review, the NRC staff finds Deviation No. 2 from ANSI/ANS 56.8-2002 as described in the LAR to be acceptable.

3.9 Technical Evaluation Conclusion

Based on the preceding regulatory and technical evaluations, the NRC staff finds that the licensee has adequately implemented its existing primary containment leakage rate testing program consisting of ILRT and LLRT. The results of the recent ILRTs and of the LLRT combined totals demonstrate acceptable performance and support a conclusion that the structural and leak-tight integrity of the primary containment is adequately managed and will continue to be periodically monitored and managed effectively with the proposed changes. The NRC staff finds that the licensee has addressed the NRC conditions to demonstrate the acceptability of adopting NEI 94-01, Revision 3-A, and the limitations and conditions identified in the staff SE incorporated in NEI 94-01, Revision 2-A. Therefore, the NRC staff finds that the proposed changes to the Watts Bar, Units 1 and 2, TS 5.7.2.19 regarding the containment leakage rate testing program are acceptable and meet the requirements of 10 CFR 50.36(c)(5). Additionally, the NRC finds that the licensee has adequately addressed the four conditions for the use of EPRI TR-1009325, Revision 2, by addressing PRA quality, demonstrated through calculating the “small” increases in LERF, used the 100 L_a instead of 35 L_a for average leak rates, and addressed containment overpressure by not relying on overpressure for ECCS performance. Accordingly, the NRC staff finds the PRA assessment for the extension of the ILRT intervals acceptable. Lastly, the NRC staff finds that the proposed change to use a bounding value of 15.0 psig for P_a (instead of the calculated P_a) to bound the range of peak calculated containment internal pressures from 9.0 to 15.0 psig for the DBLOCA is acceptable because it will not adversely affect the total containment leakage rate in comparison to the TS leakage rate acceptance criteria.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment on June 3, 2021. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to installation or use of a facility’s components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding published in the *Federal Register* on February 2, 2021 (86 FR 7885). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: B. Lee, NRR
J. Ma, NRR
Z. Gran, NRR
T. Scarbrough, NRR

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SUBJECT: WATTS BAR NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT NOS. 149 AND 56 REGARDING REVISION TO TECHNICAL SPECIFICATION 5.7.2.19, “CONTAINMENT LEAKAGE RATE TESTING PROGRAM” TO EXTEND CONTAINMENT INTEGRATED AND LOCAL LEAK RATE TEST INTERVALS (EPID L-2020-LLA-0223) DATED NOVEMBER 3, 2021

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