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

November 30, 2015

Mr. Joseph W. Shea
Vice President, Nuclear Licensing
Tennessee Valley Authority
1101 Market Street, LP 3R-C
Chattanooga, TN 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATION FOR CONTAINMENT LEAKAGE RATE PROGRAM (CAC NOS. MF5366 AND MF5367)

Dear Mr. Shea:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 335 to Renewed Facility Operating License (RFOL) No. DPR-77, and Amendment No. 328, to RFOL No. DPR-79, for the Sequoyah Nuclear Plant (SQN), Units 1 and 2, respectively. The amendments consist of changes to the RFOLs and Technical Specifications (TSs) in response to your application dated December 2, 2014, as supplemented by letters dated September 11, 2015, and October 23, 2015.

The proposed changes revise TS 6.8.4.h (Improved Standard TS 5.5.14), "Containment Leakage Rate Testing Program," by adopting the approved Nuclear Energy Institute 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," as the implementation document for the performance-based Option B of 10 CFR Part 50, Appendix J. Specifically, the amendment (1) extends the Type A primary containment integrated leak rate test intervals from 10 to 15 years, (2) extends the Type C local leak rate test intervals from 60 to 75 months, and (3) incorporates the regulatory positions stated in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program."

A copy of the staff's related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Sincerely,

Andrew Hon, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 335 to RFOL No. DPR-77
2. Amendment No. 328 to RFOL No. DPR-79
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-327

SEQUOYAH NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 335
Renewed License No. DPR-77

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

   A. The application for amendment by the Tennessee Valley Authority (the licensee), dated December 2, 2014, as supplemented by letters dated September 11, 2015, and October 23, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commissions rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended; 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 Renewed Facility Operating License No. DPR-77 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 335, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and shall be implemented no later than 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley

Benjamin G. Beasley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications
and Renewed Facility Operating License

Date of Issuance: November 30, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 335

RENEWED FACILITY OPERATING LICENSE NO. DPR-77

DOCKET NO. 50-327

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

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

Remove	Insert
5.5-13	5.5-13

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) 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 or instrument calibration or associated with radioactive apparatus or components; and

(5) 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 Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This renewed 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

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 335 are hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods, or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), 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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish 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 NEI 94-01, "Industry Guideline for 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 by the following exceptions:

1. Bypass leakage paths to the auxiliary building leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A, and 49B) for at least 30 days.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-328

SEQUOYAH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 328
Renewed License No. DPR-79

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

   A. The application for amendment by the Tennessee Valley Authority (the licensee), dated December 2, 2014, as supplemented by letters dated September 11, 2015, and October 23, 2015, 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, as amended; 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 Renewed Facility Operating License No. DPR-79 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 328, are hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and shall be implemented no later than 30 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley

Benjamin G. Beasley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications
and Renewed Facility Operating License

Date of Issuance: November 30, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 328

RENEWED FACILITY OPERATING LICENSE NO. DPR-79

DOCKET NO. 50-328

Replace the following page of the Renewed Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
3	3

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

Remove	Insert
5.5-13	5.5-13

(3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) 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 or instrument calibration or associated with radioactive apparatus or components; and

(5) 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 Sequoyah and Watts Bar Unit 1 Nuclear Plants.

C. This renewed 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

The Tennessee Valley Authority is authorized to operate the facility at reactor core power levels not in excess of 3455 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 328 are hereby incorporated into the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Initial Test Program

The Tennessee Valley Authority shall conduct the post-fuel-loading initial test program (set forth in Section 14 of Tennessee Valley Authority's Final Safety Analysis Report, as amended), without making any major modifications of this program unless modifications have been identified and have received prior NRC approval. Major modifications are defined as:

a. Elimination of any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

b. Modification of test objectives, methods or acceptance criteria for any test identified in Section 14 of TVA's Final Safety Analysis Report as amended as being essential;

5.5 Programs and Manuals

5.5.13 Safety Function Determination Program (SFDP) (continued)

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), 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. When a loss of safety function is caused by the inoperability of a single Technical Specification support system, the appropriate Conditions and Required Actions to enter are those of the support system.

5.5.14 Containment Leakage Rate Testing Program

a. A program shall establish 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 NEI 94-01, "Industry Guideline for 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 by the following exceptions:

    1. Bypass leakage paths to the auxiliary building leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (13.2 psig) and the seal system capacity is adequate to maintain system pressure (or fluid head for the containment spray system and RHR spray system valves at penetrations 48A, 48B, 49A, and 49B) for at least 30 days.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 335 AND 328 TO

RENEWED FACILITY OPERATING LICENSE NOS. DPR-77 AND DPR-79

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

1.0 INTRODUCTION

By application dated December 2, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14339A539), as supplemented by letters dated September 11, 2015 (ADAMS Accession No. ML15257A392), and October 23, 2015 (ADAMS Accession No. ML15299A140), Tennessee Valley Authority (TVA, the licensee) submitted a license amendment request (LAR) for the Sequoyah Nuclear Plant (SQN), Units 1 and 2. The LAR would revise the Technical Specification (TS) 6.8.4.h (Improved TS 5.5.14), "Containment Leakage Rate Testing Program," by adopting the approved Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR [Title 10 of the *Code of Federal Regulations*] Part 50, Appendix J," as the implementation document for the performance-based Option B of 10 CFR Part 50, Appendix J, as follows:

- In accordance with the guidance in NEI 94-01, Revision 3-A, and the limitations and conditions for NEI 94-01, Revision 2, the proposed changes would permit the performance-based primary containment integrated leak rate testing (ILRT), also known as a Type A test, interval to be extended from no longer than 10 years to no longer than 15 years, provided acceptable performance history and other requirements stated in this report are maintained.

- In accordance with NEI 94-01, Revision 3-A, the proposed changes would permit the containment isolation valve local leakage rate testing (LLRT), also known as Type C test, intervals to be extended from 60 to 75 months.

The U.S. Nuclear Regulatory Commission (NRC) staff evaluated the LAR against applicable regulatory requirements in the following two aspects:

- Deterministic considerations (i.e., structural and leak-tight integrity of the containment, including the results from the Containment Inservice Inspection (CISI) Program); and

- Probabilistic risk assessment.

This safety evaluation (SE) documents the NRC staff's review, especially how the licensee addressed the limitations and conditions of NEI 94-01, which resulted in the approval of the proposed changes.

2.0 REGULATORY EVALUATION

Section 50.54(o) of 10 CFR requires that the primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Two options are included in 10 CFR Part 50, Appendix J: "Option A – Prescriptive Requirements," and "Option B – Performance-Based Requirements," either of which can be chosen for meeting the requirements of the appendix. 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 TS; and (b) integrity of the containment structure is maintained during the service life of the containment.

In 10 CFR Part 50, Appendix J, Option B, it specifies performance-based requirements and criteria for preoperational and subsequent leakage rate testing. These requirements are met by performing Type A tests to measure the containment system overall integrated leakage rate; Type B pneumatic tests to detect and measure local leakage rates across pressure-retaining leakage-limiting boundaries such as penetrations; and Type C pneumatic tests to measure containment isolation valve leakage rates. After the preoperational tests, these tests are required to be conducted at periodic intervals based on the historical performance of the overall containment system (for Type A tests) and based on the safety significance and historical performance of each boundary and isolation valve (for Type B and Type C tests). These tests ensure integrity of the overall containment system as a barrier to fission product release.

The leakage rate test results must not exceed the allowable leakage rate (L_a) with margin, as specified in the TS. Option B also requires that a general visual inspection of the accessible interior and exterior surfaces of the containment system for structural deterioration, which may affect the containment leak-tight integrity, must be conducted prior to each Type A test and at a periodic interval between tests based on the performance of the containment system.

Section V.B.3 of 10 CFR 50, Appendix J, Option B, requires that the regulatory guide or other implementation document used by a licensee to develop a performance-based leakage-testing program must be included, by general reference, in the plant TSs. Furthermore, the submittal for TS revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide.

NEI 94-01, Revisions 2 and 3, have been reviewed by the NRC staff and approved for use. The final SE for Revision 2, issued by letter dated June 25, 2008 (ADAMS Accession No. ML081140105), documents the NRC staff's evaluation and acceptance of Revision 2, subject to six specific limitations and conditions listed in Section 4.1 of the SE for the Type A test. The final SE for Revision 3, issued by letter dated June 8, 2012 (ADAMS Accession No. ML121030286), includes two specific limitations and conditions listed in Section 4.0 of the

SE for the Type C test. This LAR was reviewed with the conditions and limitations presented in the SE for Revisions 2 and 3 of NEI 94-01.

Section 9.2.3.1 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 Electric Power Research Institute (EPRI) Technical Report (TR) 1009325, Revision 2-A, "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 accordance with the guidance in NEI 94-01, Revision 3-A, and the limitations and conditions for NEI 94-01, Revision 2, the licensee proposes to extend the containment Type A test interval from the current approved 10 years to 15 years, based on acceptable performance. This would allow the next Type A test to be performed within 15 years from the last test, instead of the current 10-year interval. In addition, the licensee proposes to extend the containment Type C test interval from the current approved 60 months to 75 months, based on acceptable performance. This would allow the next Type C test to be performed within 75 months from the last test, instead of the current 60 month interval.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Proposed Changes

In its supplement dated October 23, 2015, the licensee superseded the TSs change proposed in its application dated December 2, 2014. The TS was revised to align with the improved TS conversion license amendment[1]. Therefore, the TS 6.8.4.h was renumbered and reformatted to TS 5.5.14.

SQN, Units 1 and 2, TS 5.5.14 currently states, in part:

> 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. Visual examination and testing, including test intervals and extensions, shall be in accordance with Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995 with exceptions provided in the site implementing instructions and the following:

In this request, the licensee proposes the following wording for the first paragraph of TS 5.5.14:

> 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,

[1] By letter dated September 30, 2015 (ADAMS Accession No. ML15238B499), the NRC approved the conversion to Improved TS for SQN, Units 1 and 2.

> Option B, as modified by approved exemptions. Visual examination and testing, including test intervals and extensions, shall be in accordance with NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 2012, and Section 4.1, "Limitations and Conditions for NEI TR 94-01, Revision 2," of the NRC Safety Evaluation Report (SER) in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exceptions:

The licensee proposed to revise the SQN, Units 1 and 2, leakage rate testing program by implementing the guidance in NEI 94-01, Revision 3-A, and the conditions and limitations specified in NEI 94-01, Revision 2-A. The material presented in NEI 94-01, Revision 3-A, as well as the conditions and limitations as noted in the associated SEs for both NEI 94-01, Revisions 2-A and 3-A, were used by the NRC staff to conduct the review for the SQN license amendment application.

The licensee follows NEI 94-01, Revision 3-A, and the limitations and conditions of Section 4.1 of the NEI 94-01, Revision 2-A, SE and Section 4.0 of the NEI 94-01, Revision 3-A, SE. SQN proposes an extension of the Type A test interval, which is currently required by TS to be performed at 10-year intervals, to no longer than 15 years from the last Type A test. To extend the Type A test interval, NEI 94-01, Revision 3-A, provides a guideline that the extension shall be based on two consecutive successful Type A tests (performance history) and other requirements stated in Section 9.2.3 of NEI 94-01, Revisions 2-A and 3-A. The review results of SQN's Type A test performance history, whether meeting the Section 9.2.3 requirements and SE's limitations and conditions, are presented in the following subsections.

The licensee also proposes an extension to the Type C test interval, which is currently required by TS to be performed at 60-month intervals, to no longer than 75-month intervals from the last Type C test. NEI 94-01, Revision 3-A, guidelines explain that extensions of the Type C test intervals are allowed based upon completion of two consecutive periodic as-found tests, where the results of each test are within a licensee's allowable administrative limits and other requirements stated in Section 10.2.3 of NEI 94-01, Revision 3-A. The review of SQN's Type C test performance history, whether meeting the Section 10.2.3 requirements and SE's limitations and conditions, are presented in the following subsections.

3.2 Deterministic Considerations (i.e., structural and leak-tight integrity of the containment)

3.2.1 Historical Type A Test Results Supported the Request

In LAR Section 4.2, the licensee presented the results of the historical Type A tests that are summarized in Tables 1 and 2 below.

The TS acceptance criteria is 0.25 percent of containment air weight at the design basis loss-of-coolant accident pressure (L_a). NEI 94-01, Revisions 2-A and 3-A, state that acceptable performance for a Type A leak rate test is less than 1.0 L_a.

Table 1: SQN, Unit 1, Type A Test Historical Results

Test Date	As-Found Leakage		Acceptance Limit
5/5/90	Mass Point Upper Confidence Limit (UCL) leakage with penalties	0.38 of L_a	1.0 L_a
12/19/93	Mass Point UCL leakage with penalties	0.79 of L_a	1.0 L_a
10/27/07	Mass Point UCL leakage with penalties	0.46 of L_a	1.0 L_a

The preceding Table 1 shows that the Type A tests for SQN, Unit 1, were successful. The licensee states that there are no known modifications for SQN, Unit 1, that will require a Type A test to be performed prior to the Fall 2022 outage, when the next Type A tests will be performed in accordance with the proposed changes. Since the licensee has performed the Type A tests successfully, this supports the licensee's request to extend the Type A testing interval for SQN, Unit 1, in accordance with NEI 94-01, Revision 3-A, and subject to the appropriate limitations and conditions of the NRC SE for NEI 94-01, Revision 2-A.

Table 2: SQN, Unit 2, Type A Test Historical Results

Test Date	As-Found Leakage		Acceptance Limit
3/19/89	Mass Point UCL leakage with penalties	0.32 of L_a	1.0 L_a
4/28/92	Mass Point UCL leakage with penalties	1.31 of L_a	1.0 L_a
12/20/06	Mass Point UCL leakage with penalties	0.36 of L_a	1.0 L_a

The preceding Table 2 shows that the Type A tests for SQN, Unit 2, were less than 1.0 L_a, except in the year 1992. In 1992, the licensee found that a single penetration in the ice condenser glycol system (X-47A) was responsible for the 1.31 of L_a result. The licensee implemented corrective actions to reduce the chance of recurrence.

In 1992, the licensee was using NRC Information Notice 85-71, "Containment Integrated Leak Rate Tests," dated August 22, 1985, to determine its containment integrated leak rate test (CIRLT) test results. The practice in this document was to add all of the minimum path leakage savings for Type B and Type C tests to the CILRT results. Minimum path leakage savings are the difference between the as-found and the as-left leakage rates. For X-47A, the as-found minimum path leak rate was 211.1943 standard cubic feet per hour (SCFH), and the as-left minimum path leak rate was 0.00 SCFH after the licensee took corrective actions. The licensee's corrective actions adequately addressed this leakage path.

NEI 94-01, Revision 3-A, Section 5.0, "Definitions," states:

> The performance leakage rate is calculated as the sum of the Type A UCL and as-left minimum pathway leakage rate (MNPLR) leakage rate for all Type B and Type C pathways that were in service, isolated, or not lined up in their test position (i.e., drained and vented to containment atmosphere) prior to performing the Type A test. In addition, leakage pathways that were isolated during performance of the test because of excessive leakage must be factored into the performance determination. The performance criterion for Type A tests is a performance leak rate of less than $1.0L_a$.

By using the above definition, the 1992 test would not be a failure, as the Type A test would be less than 1.0 L_a. The NRC staff finds that this is an acceptable approach to meet the two consecutive periodic Type A tests less than 1.0 L_a criterion in order to grant a frequency extension to use the guidance in NEI 94-01, Revision 3-A, and subject to the appropriate limitations and conditions of the NRC SE for NEI 94-01, Revision 2-A.

The licensee submitted an LAR on October 4, 2002, where the April 1992 CILRT as-found mass leak rate (MLR) result was reported as 0.05854 percent/day (0.2342 L_a). This leakage rate is the sum of the calculated 95 percent upper confidence limit (UCL) of 0.05773 percent/day and the leakage due to sump level increase of 0.00081 percent/day. In a September 11, 2015, request for additional information (RAI) response, the April 1992 results only included the 95 percent UCL result. The more appropriate result to report would be the total as-found MLR of 0.05854 percent/day when using the above definition from NEI 94-01, Revision 3-A, Section 5.0.

The licensee stated that there are no known modifications for SQN, Unit 2, that will require a Type A test to be performed prior to the Fall 2022 outage, when the next Type A tests will be performed in accordance with the proposed changes. Since the licensee has performed the Type A tests successfully, this supports the licensee's request to extend the Type A testing interval for SQN, Unit 2, in accordance with NEI 94-01, Revision 3-A, and subject to the appropriate limitations and conditions of the NRC SE for NEI 94-01, Revision 2-A.

3.2.2 <u>Historical Type B and Type C Leak Rate Results Supported the Request</u>

In LAR Section 4.3, the licensee presented the results of its Type B and Type C testing. The results are summarized in the following Tables 3 and 4.

Table 3: SQN, Unit 1, Historical Type B and Type C Leak Rate Results

Refueling Outage	As-Found Minimum Pathway Leakage	As-Left Maximum Pathway Leakage
U1R17 Fall 2010	13.5 SCFH	21.6 SCFH
U1R18 Spring 2012	16.9 SCFH	28.5 SCFH
U1R19 Fall 2013	25.1 SCFH	78.73 SCFH

(SCFH = standard cubic feet per hour)

Table 4: SQN, Unit 2, Historical Type B and Type C Leak Rate Results

Refueling Outage	As-Found Minimum Pathway Leakage	As-Left Maximum Pathway Leakage
U2R17 Spring 2011	15.4 SCFH	31.8 SCFH
U2R18 Fall 2012	18.4 SCFH	45.8 SCFH
U2R19 Spring 2014	44.4 SCFH	43.6 SCFH

(SCFH = standard cubic feet per hour)

The leak rate limit allowed for containment integrity is 0.6 L_a for both the as-found minimum pathway leak rate and the as-left maximum pathway leak rate. L_a is equivalent to approximately 225 SCFH, making 0.6 L_a approximately 135 SCFH. At the time of the submittal, the current total penetration leakage on a minimum path basis was approximately 31 percent of 0.6 L_a for SQN, Unit 1, and 16 percent of 0.6 L_a for SQN, Unit 2. Since the licensee has performed the Type B and Type C tests successfully, the NRC staff found it supports the licensee's request to extend the Type B and Type C testing intervals for SQN, Units 1 and 2, in accordance with NEI 94-01, Revision 3-A, and subject to the appropriate limitations and conditions.

3.2.3 NRC Conditions in NEI 94-01, Revision 2-A, are Satisfied

In the NRC SE dated June 25, 2008, the NRC staff concluded that the guidance in NEI 94-01, Revision 2, is acceptable for reference by the licensee's proposal to amend its TS in regard to containment leakage rate testing, subject to the six limitations and conditions noted in Section 4.1 of the NRC SE for NEI 94-01, Revision 2-A (ADAMS Accession No. ML100620847). The NRC staff evaluated and found that the licensee adequately addressed and satisfied these conditions in the LAR as discussed below.

3.2.3.1 NRC Condition 1

> For calculating the Type A leakage rate, the licensee should use the definition in the NEI TR 94-01, Revision 2, in lieu of that in ML13329A881 [American National Standards Institute]/ANS [American Nuclear Society]-56.8-2002. (Refer to SE Section 3.1.1.1).

The licensee stated in Section 4.0 of the LAR (Reference 5.1) that following the NRC staff's approval of this LAR, SQN will use the definition in Section 5.0 of NEI 94-01, Revision 3-A, for calculating the Type A leakage rate when future SQN, Units 1 and 2, Type A tests are performed. NEI 94-01, Revision 3-A, contains the same definition as in Revision 2-A for calculating the Type A test leakage rate. Therefore, the NRC staff finds that the licensee has adequately addressed this condition.

3.2.3.2 NRC Condition 2

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

Section 9.2.3.2 of NEI 94-01, Revisions 2-A and 3-A, "Supplemental Inspection Requirements," states that in order 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 of the Type A test is extended to 15 years.

In Section 4.4 of the LAR, the licensee stated that the examinations performed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Subsection IWE/IWL program satisfy the general visual examination requirements specified in 10 CFR 50, Appendix J, Option B. The licensee stated that the frequency of examinations pursuant to Subsection IWE (three examinations over a 10-year interval) assures that at least three general visual examinations of metallic components will be conducted between the Type A tests, if the Type A test interval is extended to 15 years.

Furthermore, the licensee stated that (1) TVA performs a visual inspection of the accessible interior and exterior of the SQN, Units 1 and 2, containment vessel prior to each Type A test. This examination is performed in sufficient detail to identify any evidence of deterioration that may affect the containment vessel structural integrity or leak tightness. The examination is conducted in accordance with approved plant procedures to satisfy the requirements of 10 CFR Part 50, Appendix J.

Additionally, the licensee provided the dates of completed and scheduled ILRT and completed general visual examinations of containment surfaces and an approximate schedule for future general visual examinations of containment interior and exterior surfaces, representative of a typical 15-year period between Type A tests for both SQN, Units 1 and 2.

The licensee concluded that together, these examinations assure that at least three general visual examinations of the accessible exterior and interior containment vessel surfaces and one visual examination immediately prior to the next Type A test will be conducted, if the Type A test interval is extended to 15 years, thereby meeting the requirements of Section 9.2.3.2 of NEI 94-01, Revision 3-A, as well as Condition 2 in Section 4.1 of the NRC SE for NEI 94-01, Revision 2.

On the basis that the licensee's schedule of general visual examinations described in the LAR results in at least three examinations between Type A tests and one examination immediately prior to the Type A test for containment vessel inner and outer surfaces, the NRC staff finds that the licensee's inspection schedule plan meets this condition.

3.2.3.3 NRC Condition 3

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

In Section 4.0 of the LAR, the licensee stated that (1) general visual examination of accessible interior and exterior surfaces of the containment is conducted in accordance with the SQN, Units 1 and 2, CISI Program, which implements the requirements of the ASME Code, Section XI, Subsections IWE and IWL, as required by 10 CFR 50.55a(g); and (2) the SQN, Units 1 and 2, CISI Program contains requirements to evaluate the acceptability of the inaccessible areas if such conditions were identified, in accordance with 10 CFR 50.55a(b)(2)(ix)(A).

In Section 4.4.1 of the LAR, the licensee provided the following information relative to the moisture barrier at the junction of containment vessel wall and the concrete floor:

(a) The SQN, Units 1 and 2, steel containment vessels (SCVs) have a moisture barrier that is examined in accordance with IWE-2500-1, Category E-A, Item E1.30. Examinations were conducted in 2000, 2003, 2006, 2011, and 2014 for SQN, Unit 2, and 2000, 2004, 2006, and 2012 for SQN, Unit 1. In each of the examinations, areas were noted where the moisture barrier was not completely adhered to the concrete interface. The moisture barrier was removed and reapplied in the subject areas.

(b) In 2011, four locations were identified in the SQN, Unit 2, SCV where the moisture barrier was not completely adhered to the concrete interface. Supplemental ultrasonic testing (UT) thickness measurements were taken to validate the remaining wall thickness in the excavated area. The thickness was above the nominal wall thickness, and no further actions were required.

By letter dated September 11, 2015, in response to Mechanical and Civil Engineer Branch (EMCB) RAI-3, the licensee stated that the examinations of the moisture barrier have identified areas with lack of bonding of the sealant. When these areas are identified, the sealant is excavated to allow inspection of the SCV under the moisture barrier and the condition is entered in the corrective action program (CAP) for evaluation. Examinations have identified no instances of suspect areas or detrimental flaws that would affect the structural integrity of leak tightness of the SCV. After examination of the excavated areas, the sealant is reapplied and reexamined prior to the end of the outage.

By letter dated September 11, 2015, in response to EMCB RAI-1, the licensee stated that the general visual examination of the moisture barrier performed for SQN, Unit 1, in April and May of 2015 during refueling outage U1R20, identified five areas with indications of loss of adhesion. The areas were excavated after the initial examination. Excavated areas were examined and found to be acceptable. Following excavation and examination, the sealant was reapplied and the areas were reexamined. The resealed areas were found to be acceptable.

By letter dated October 23, 2015, the licensee supplemented the response to EMCB RAI-3 and stated that in January 2000, TVA processed a design change to provide for alternative methods of sealing the seal joint between the concrete and SCV. The alternative method allows the removal of the existing fiberglass filler and filling the crevice completely with a polyurethane elastomeric sealant when repairs to the moisture barrier seal are required during regular maintenance or during IWE inspections. TVA also stated that due to possible repeat of lack of adhesion from repairs made during SQN, Unit 2, Cycle 17 examination, Condition Report 1094801 has been initiated in the TVA CAP to determine the appropriate corrective measures for lack of moisture barrier adhesion.

By letter dated September 11, 2015, the licensee stated that the SCV to concrete interface is visible in the annulus (the area between the SCV and the concrete shield building). No moisture barrier covers the SCV to concrete interface in the annulus. The metal to concrete interface is examined in the annulus each inspection period. By letter dated October 23, 2015, TVA stated that the review of the past two examination reports for both SQN, Units 1 and 2, determined no "notices-of-indications" have been issued that list any findings with respect to the SCV to concrete interface of the annulus exposed accessible surface.

By letter dated September 11, 2015, in response to EMCB RAI-4, the licensee stated that the areas around inaccessible areas, including those inaccessible due to the ice condenser configuration, are examined and, to date, no areas have been identified that indicate there is any issue adversely affecting structural integrity or leak tightness of the SCV in inaccessible areas. Remote visual techniques are used, if possible, to eliminate inaccessibility. A section of the stainless steel sheet metal thermal barrier is removed during moisture barrier examinations to allow access, and the accessible areas beneath the thermal barrier are examined. No problems have been identified during those examinations.

NRC Information Notice 2004-09, "Corrosion of Steel Containment and Containment Liner," describes deterioration of the coating and rusting of the SQN SCV, due to the localized water ponding at the clogged drain in the annulus area behind the emergency gas treatment system (EGTS) duct work. By letter dated September 11, 2015, in response to the RAI, the licensee stated that the examinations behind the EGTS duct work are performed every inspection period during the inspection of the outside surface of the SCV in the annulus. No new indications have been identified during these examinations since NRC Information Notice 2004-09 was issued.

In Section 4.5 of the LAR, the licensee stated that consistent with the guidance provided in NEI 94-01, Revision 3-A, Section 9.2.3.3, abnormal degradation of the primary containment structure identified during the conduct of IWE program examinations, or at any other times, would be entered into the CAP for evaluation to determine the cause of the degradation and to initiate appropriate corrective actions. In addition, the NRC staff's review of licensee's containment inspection program to support the inspection results will be discussed in Section 3.2.5.

Based on the above information regarding the IWE examinations of the SQN, Units 1 and 2, containment structures and the SQN operating experience, to date, no conditions that would indicate the presence of any significant degraded condition in the accessible or inaccessible areas of the containment vessel affecting the leak tightness or structural integrity of the SQN, Units 1 and 2, containment structures have been identified. In addition, the SQN, Units 1 and 2, CISI Program contains requirements to evaluate the acceptability of the inaccessible areas if such conditions were identified, in accordance with 10 CFR 50.55a(b)(2)(ix)(A). As such, the NRC staff concludes that the licensee has adequately addressed this condition.

3.2.3.4 NRC Condition 4

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

The licensee stated in Section 4.0 of the LAR that SQN, Units 1 and 2, have already replaced the steam generators. By letter dated September 11, 2015, in response to RAI-6, the licensee stated that two SCV cutouts were removed and replaced in both SQN, Units 1 and 2, SCV to allow access to the steam generators during the replacement outages. Grid locations for UT thickness examinations have been selected. Each grid is 12 inches wide and 24 inches long and spaced equally along the cut lines. Areas with less than the required minimum thickness will be evaluated. These examinations are performed every other outage. The last performance of this examination on SQN, Unit 1, was in 2013 during refueling outage U1R19. No thickness readings were less than the required minimum thickness. The last performance of this examination for Unit 2, was in 2014, during refueling outage U2R19. One grid was found to be less than the required minimum thickness, and that grid was evaluated and found to be acceptable. By letter dated October 23, 2015, the licensee supplemented the response to EMCB RAI-6 and stated that the 2014 Unit 2 UT thickness examination was the first UT examination conducted on the SCV cutouts for steam generator replacement; therefore, no baseline information exists. However, the licensee stated that no degradation was noted in the examination report.

Furthermore, in Section 4.2 of the LAR, the licensee stated, in part:

> (a) TVA plans to modify the SQN, Units 1 and 2, containment liners to attach a track near the equipment hatch on the refuel floor during U1R20 (Spring 2015) and U2R20 (Fall 2015), respectively. This modification is minor, and the Type A post-maintenance test will be deferred until the next regularly scheduled Type A test as allowed by NEI 94-01, Revision 0, Section 9.2.4. The associated post-maintenance tests are tracked in the SQN CAP and in the surveillance program for completion.

The NRC staff notes that the acceptability of attaching a track to the containment vessel wall near the equipment hatch of SQN, Units 1 and 2, and deferral of the required post-modification testing, is outside the scope of this SE. The NRC staff also notes that as stated in the current SQN TS, the containment leakage rate testing program is established in accordance with the guidelines contained in RG 1.163, which endorses NEI 94-01, Revision 0; and that it is appropriate for the licensee to use Revision 0 until this license amendment is implemented.

TVA is also considering modifications to replace several electrical penetrations on SQN, Units 1 and 2. If replaced, the associated welds on the containment liner will be locally tested and not require a Type A test.

The NRC staff notes that the acceptability of replacing several electrical penetrations on SQN, Units 1 and 2, and performance of the required post-modification testing, is outside the scope of this SE.

(a) There are no known modifications that will require a Type A test to be performed prior to U1R25 (Fall 2022) and U2R24 (Fall 2021), when the next Type A tests will be performed in accordance with this proposed change.

(b) Any unplanned modifications to the containment prior to the next scheduled Type A test would be subject to the testing requirements of NEI 94-01, Revision 0, Section 9.2.4, or NEI 94-01, Revision 3-A, Section 9.2.4, as applicable.

(c) There have been no pressure or temperature excursions in either SQN, Units 1 or 2, containments which could have adversely affected containment integrity since the performance of the last Type A tests. There is no other anticipated addition or removal of plant hardware within either Units 1 or 2 containments that could affect leak tightness.

By letter dated September 11, 2015, in response to RAI-6, the licensee stated that TVA will implement the NRC staff position with regard to any future post-repair pressure testing following major and minor SQN, Units 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.

Based on the above and the fact that the proposed revision to TS incorporates the conditions and limitations specified in NEI 94-01, Revision 2-A, the NRC staff concludes that the licensee's containment leak rate program will implement the staff position with regard to post-repair pressure testing following major and minor containment repairs and modifications, as explained in Section 3.1.4 of the NRC staff SE for NEI 94-01, Revision 2. Therefore, the staff concludes that the licensee has adequately addressed this condition.

3.2.3.5 NRC 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 TR 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 ADAMS Accession No. ML081140105, Section 3.1.1.2 of the SE approving this TR.)

The licensee stated in Section 4 of the LAR that TVA acknowledges and accepts the NRC staff's position in Condition 5, as communicated to the nuclear industry in NRC Regulatory Issue Summary (RIS) 2008-27, "Staff Position on Extension of the Containment Type A Test Interval Beyond 15 Years Under Option B of Appendix J to 10 CFR Part 50," dated December 8, 2008 (ADAMS Accession No. ML080020394).

Accordingly, the NRC staff finds that the licensee has confirmed its understanding that any extension of the Type A test interval beyond the performance-based limit of 15 years should be infrequent and should be requested only for compelling reasons, and that the NRC staff will implement the position in RIS 2008-27 in reviewing such LARs. Therefore, the licensee has adequately addressed this condition.

3.2.3.6 NRC 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 TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, including the use of past ILRT data.

This condition is not applicable to SQN, Units 1 and 2. The licensee was not licensed under 10 CFR Part 52.

3.2.4 <u>NRC Conditions in NEI 94-01, Revision 3-A, are Satisfied</u>

In the NRC SE dated June 8, 2012, the staff concluded that the guidance in NEI 94-01, Revision 3, is acceptable for reference by licensees proposing to amend their TSs in regard to containment leakage rate testing, subject to the two limitations and conditions noted in Section 4.0 of the NRC SE for NEI 94-01, Revision 3-A (ADAMS Accession No. ML12221A202). The NRC staff evaluated and found the licensee adequately addressed and satisfied these conditions in the LAR as discussed below.

3.2.4.1 NRC Condition 1

> The staff is allowing the extended interval for Type C LLRT 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 CAP 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 [boiling-water reactor] MSIVs [main steam isolation valves]), 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.

The licensee states in this LAR that it will follow the guidance of NEI 94-01, Revision 3-A, to assess and monitor margin between the Type B and Type C leakage rate summation and the regulatory limit. The licensee also confirms it will take corrective actions to restore margin to an acceptable level, as needed. Therefore, the licensee has adequately addressed this condition.

3.2.4.2 NRC Condition 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.

The licensee will estimate the amount of understatement for both SQN, Units 1 and 2, in the Type B and Type C total and include this determination of the acceptability in a post-outage report. This is consistent with Section 11.3.2 of NEI 94-01, Revision 3-A.

Based on the review of the licensee's submittal, the NRC staff concludes that the primary containment leakage rate testing program contains provisions for trending and monitoring that conservatively applies a leakage understatement factor to account for the extended interval. In addition, the post-outage report will contain the necessary information. Therefore, the licensee addressed and satisfied NRC Condition 2.

3.2.5 Containment Inservice Inspection Program Supported Containment Inspection Results

In Section 4.0 of the LAR, the licensee stated that general visual examinations of accessible interior and exterior surfaces of the containment vessel for structural problems are conducted in accordance with the SQN, Units 1 and 2, CISI Program and schedule, which implement the requirements of the ASME Code, Section XI, Subsections IWE as required by 10 CFR 50.55a(g). The IWE/IWL inspections and supplemental inspections, in accordance with other approved plant procedures, are used to satisfy the general visual examination requirements of Appendix J, Option B, and to monitor and manage the age-related degradations of the primary containment to ensure that containment structural and leak-tight integrity is maintained through its service life.

The operating experience relative to the moisture barrier at the interface of the concrete base floor and the containment vessel wall has been discussed in Section 3.2.3.3 of this SE.

In Section 4.4.1 of the LAR, the licensee stated the following regarding IWE examinations:

(a) The visual examinations of the SQN, Units 1 and 2, SCV interior and exterior surfaces performed during the first 10-year interval revealed minor rusting, discoloration, and flaking, which did not affect the structural integrity or leak tightness of the SCV. The areas were cleaned and recoated in accordance with site procedures.

(b) SQN, Units 1 and 2, have completed requirements of the first period, second 10-year interval IWE examination. The second 10-year interval IWE examination requirements use the 2001 Edition, 2003 Addenda, of ASME Section XI, as modified by the 10 CFR 50.55a(b) limitations for both units.

(c) The conditions noted during the second interval IWE examinations consist of some flaking and discoloration of coatings along with rust, unpainted areas, arc strikes, scratches, small gouges, surface corrosion, scrapes, and minor pitting, which did not affect the leak tightness or structural integrity of the containment boundary. No detrimental flaws have been observed to date.

(d) Currently, SQN, Units 1 and 2, have no areas requiring augmented examination per the requirements of Category E-C.

By letter dated September 11, 2015, in response to EMCB RAI-1, the licensee stated that since the LAR was submitted on December 2, 2014, a general visual examination of accessible interior and exterior surfaces of the SQN, Unit 1, containment vessel was completed in April and May of 2015, during refueling outage U1R20. The licensee stated that there was no indication that an adverse condition exists.

Section 4.4.2 of the LAR states that the Class CC equivalent components at SQN (i.e., the structural base slab and metal liner) are exempt from examination based on the exemptions of IWL-1220(b) and IWE-1220(b). The structural base slab and metal liner are covered with concrete, which forms the reactor building floor and results in these components being inaccessible for examination. By letter dated September 11, 2015, in response to the EMCB RAI-2, the licensee stated that as part of SQN structures monitoring program, inspection is performed on a 5-year frequency that includes all structures in the scope of the SQN 10 CFR 50.65 Maintenance Rule Program. During this inspection, a walkdown of the containment is performed looking for obvious degradation or the presence of aging mechanisms on visible/accessible portions of the shield building, containment interior concrete, and the containment vessel.

The licensee stated that (1) inspections of the shield building interior and exterior surfaces are also performed under a preventative maintenance surveillance for shield building integrity verification. These inspections are performed on a 10-year frequency and also serve to identify concrete aging mechanisms or degradation; (2) there has been no abnormal degradation of the SQN, Units 1 and 2, shield buildings or containment interior concrete structures that would indicate the potential for significant concrete degradation in inaccessible areas; and (3) the most recent inspections of the SQN, Units 1 and 2, shield buildings were performed under the maintenance rule inspection program during U1R18 and U2R18, for SQN, Units 1 and 2, respectively.

Based on the above evaluation, the NRC staff finds that there is reasonable assurance that the licensee is adequately implementing its CISI Program to monitor and manage age-related degradation of the SQN containment structures. Thus, the results of the recent IWE inspections and concrete inspections discussed above supported that there has not been evidence to date of significant degradation of the SQN, Units 1 and 2, containment structures. In addition, the degradations noted have been entered into the SQN CAP and appropriately managed.

3.2.6 Summary of Deterministic Considerations of Structural and Leak-Tight Integrity

Based on the regulatory and technical evaluations above, the NRC staff finds that the licensee has adequately implemented its Reactor Building Leakage Rate Testing Program consisting of ILRT and LLRT. The results of the recent ILRT and LLRT demonstrate acceptable performance of the SQN, Units 1 and 2, containment and demonstrate that the structural and leak-tight integrity of the containment structure is adequately managed and will continue to be periodically monitored and managed by the ILRT and LLRT. The NRC staff finds that the licensee has addressed all of the NRC conditions to demonstrate acceptability of adopting NEI 94-01, Revision 3-A, and all of the limitations and conditions identified in the NRC staff SE incorporated in NEI 94-01, Revision 2-A.

Based on the review of the licensee's submittal of December 2, 2014 (Reference 5.1); supplemental information dated September 11, 2015 (Reference 5.2), and October 23, 2015 (Reference 5.5); and the regulatory and technical evaluation above, the NRC staff concludes that the licensee has effectively implemented adequate containment leakage rate testing (ILRT and LLRT) and CISI Program to periodically examine, monitor, and manage age-related degradation of the SQN, Units 1 and 2, SCVs. The results of the past ILRT, LLRT, and the CISI Program, demonstrate that the structural and leak-tight integrity of the SQN, Units 1 and 2, containments is adequately managed. The structural and leak-tight integrity of the SQN, Units 1 and 2, containments will continue to be periodically monitored and managed by the LLRT and CISI Program if the current ILRT interval is extended from 10 years to 15 years. Therefore, the NRC staff concludes that there is reasonable assurance that the SQN, Units 1 and 2, containment structural and leak-tight integrity will continue to be maintained, without undue risk to public health and safety, if the current ILRT interval is extended to 15 years. The next Type A test interval at SQN, Unit 1, may be conducted no later than October 26, 2022, instead of the current due date of October 26, 2017.

3.3 Probabilistic Risk Assessment

3.3.1 Background

Section 9.2.3.1 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. NEI 94-01, Section 9.2.3.4, "Plant-Specific Confirmatory Analyses," 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 SER, 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. These conditions, set forth in Section 4.2 of the SE for EPRI TR-1009325, Revision 2, stipulate that:

1. The licensee submit documentation indicating that the technical adequacy of its probabilistic risk assessment is consistent with the requirements of RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," relevant to the ILRT extension application.

2. The licensee submit 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 the SER for EPRI TR-1009325, Revision 2.

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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.

[3] The SER 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 actually provided in Section 3.2.4.6.

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

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

3.3.2 Plant-Specific Risk Analysis

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 SQN, Units 1 and 2, were provided in Enclosures 2 and 3 of the LAR dated December 2, 2014. In Section 4.6.1 of Enclosure 1 to the LAR, the licensee stated that the plant-specific risk assessment follows the guidance in:

- NEI 94-01, Revision 3-A;

- The methodology described in EPRI TR-1018243, Revision 2-A, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals, Revision 2-A of 1009325," dated October 2008;

- The NRC regulatory guidance on the use of risk insights in support of a request for a plant's licensing basis as outlined in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"; and

- The methodology used for Calvert Cliffs Nuclear Plant to estimate the likelihood and risk implications of corrosion-induced leakage of steel liners going undetected during the extended test interval.

The licensee stated in the LAR that SQN, Units 1 and 2, have a Level 2 PRA model that provides representative results for internal events, including internal floods.

The licensee addressed each of the four conditions for the use of EPRI TR-1009325, Revision 2, which are listed in Section 4.2 of the NRC SER. A summary of how each condition has been met is provided in the sections below.

3.3.3 Technical Adequacy of the Probabilistic Risk Assessment is Sufficient

The first condition stipulates that the licensee submit documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200 relevant to the ILRT extension application.

In Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," the NRC staff clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200 (ADAMS Accession No. ML070240001) to assess the technical adequacy of the PRA used to support risk-informed applications. Revision 2 of RG 1.200 (ADAMS Accession No. ML090410014) will be used for all risk-informed applications received after March 2010. In Section 3.2.4.1 of the SER to EPRI TR-1009325, Revision 2, the NRC staff stated, in part, that:

> [l]icensee requests for a permanent extension of the ILRT surveillance interval to 15 years pursuant to NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, will be treated by NRC staff as risk-informed LARs. Consistent with information provided to industry in Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation," the NRC staff will expect the licensee's supporting Level 1/LERF [large early release frequency] PRA to address the technical adequacy requirements of RG 1.200, Revision 1... Any identified deficiencies in addressing this standard shall be assessed further in order to determine any impacts on any proposed decreases to surveillance frequencies. If further revisions to RG 1.200 are issued which endorse additional standards, the NRC staff will evaluate any application referencing NEI TR 94-01, Revision 2, and EPRI Report No. 1009325, Revision 2, to examine if it meets the PRA quality guidance per the RG 1.200 implementation schedule identified by the NRC staff.

In the same section of the SER, the NRC staff states that Capability Category (CC) I of ASME PRA standards shall be applied as the standard for assessing PRA quality for ILRT extension applications, as approximate values of core damage frequency (CDF) and LERF, and their distribution among release categories is sufficient to support the evaluation of changes to ILRT frequencies.

Per Section 4.6.2 "PRA Technical Adequacy," of the submittal, the plant-specific confirmatory analysis for both Units 1 and 2 uses accident sequence results from the Level 2 analysis. The licensee's PRA model includes internal events and calculates the CDF contribution of external events (seismic-induced CDF and fire-induced CDF).

The PRA technical adequacy for SQN is discussed in Section 4.6.2 of the LAR. The licensee stated that a full-scope internal events, including internal flooding, PRA peer review of the SQN PRA model, was conducted in 2011. In Section 4.6.2 of the LAR, the licensee provided a brief summary of the SQN final resolutions to the Facts and Observations (F&Os) from the 2011 SQN PRA full scope peer review of internal events, including internal flooding. The licensee identified F&Os 1-15, 4-1, and 4-13 as F&Os that were not incorporated as recommended by the peer review team. F&Os 4-1 and 4-13 are peer review suggestions as identified by the peer review team. F&O 4-1, which was also judged by the peer review team to meet the optional CC II, was a documentation enhancement suggestion. Likewise, F&O 4-13 was also judged to meet CCI by the peer review team. The NRC staff finds F&Os 4-1 and 4-13 acceptable for the ILRT extension application because F&Os 4-1 and 4-13 meet the ASME/ANS standard to CCI, as required for the application.

F&O 1-15 was considered not met by the peer team for SR AS-A10 because the general transients event tree that was employed by the licensee did not provide sufficient detail to capture important system requirements and required operator interactions for all initiating events. The peer review team offered three recommendations as to what the licensee could do in response to the F&O. The possible resolutions were:

(1) Subdivide the General Transients event tree to better represent the unique challenges presented by specific initiating events (e.g., Transient with Loss of [Power Conversion System] PCS, Transient with PCS Available, [Loss of Station Power] LOSP) or document how those challenges are addressed in the top logic model.

(2) Modify the existing event sequence and/or linked fault tree to ensure that the challenge to the Pressurizer Safeties is captured for initiating events that would prevent the [Power Operated Relief Valves] PORVs from opening.

(3) Explicitly model the [Station Blackout] SBO sequences to ensure that the necessary mitigating systems are addressed following power recovery.

To provide a disposition for this F&O, the licensee stated in Enclosure 3, "Resolutions to Probabilistic Risk Assessment Peer Review Team Facts and Observations," of the LAR that "[t]he tree was updated to explicitly ask demand for PORVs and Safeties." The licensee did not explain the rationale for choosing this resolution to this F&O except to provide some background on why the inclusion of the failure rates of mitigating systems following power recovery due to SBO has a negligible effect on the CDF. However, the licensee did reference the explanation for SBO system failure after power recovery in another LAR RAI responses KNH-008 and KNH-009 (ADAMS Accession No. ML15176A678). That LAR (ADAMS Accession No. ML13329A881) was approved to convert the SQN TS to Improved Standard TS. The NRC staff evaluated the licensee's responses to these RAIs and found that the licensee used recommended resolution #2 by the peer review team and provided extended discussion about why they would not use recommended resolutions #1 and #3. Thus, the NRC staff finds the licensee's disposition adequately resolves F&O 1-15 for this ILRT extension application because the licensee followed the peer-review recommendation, and the associated supporting requirement meets CCI, which is acceptable for this application.

The full list of the remaining F&Os and their resolutions are provided in Enclosure 3. The licensee further stated that the peer review F&Os have been resolved by incorporating the peer review team recommendations, and those F&Os have no effect on the proposed application. The NRC staff reviewed the resolutions and dispositions to the F&Os and confirmed that the supporting requirement associated with those F&Os meet CCI of the ASME/ANS standard.

In Section 3.2.4.2 of the SER for NEI 94-01, Revision 2, and EPRI TR-1009325, Revision 2, the NRC staff states:

> Although the emphasis of the quantitative evaluation is on the risk impact from internal events, the guidance in EPRI Report No. 1009325, Revision 2, Section 4.2.7, "External Events," states that: "Where possible, the analysis should include a quantitative assessment of the contribution of external events (e.g., fire and seismic) in the risk impact assessment for extended ILRT intervals." This section also states that: "If the external event analysis is not of sufficient quality or detail to directly apply the methodology provided in this

document [(i.e., EPRI Report No. 1009325, Revision 2)], the quality or detail will be increased or a suitable estimate of the risk impact from the external events should be performed. This assessment can be taken from existing, previously submitted and approved analyses or other alternate method of assessing an order of magnitude estimate for contribution of the external event to the impact of the changed interval.

In Section 9.0 the submittal, the licensee stated that the SQN seismic and internal fire PRA model lacks detailed Level 2 PRA modeling and "their potential contribution is limited to a conservative estimate of the change in LERF associated with the ILRT interval extension." The seismic and fire risk contribution is calculated by summing the seismic-induced CDF with the fire-induced CDF. The licensee stated that the consequences associated with the other external events (e.g., high winds) "are relatively insignificant to the consequences of seismic and internal fire events." The licensee also stated that it reviewed any plant changes against "the screening described in Supplement 4 to Generic Letter 88-20 and NUREG-1407 to address other external hazards." The Seismic CDF was obtained using the Generic Issue-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," "simple average" column from the seismic core damage frequency in the 2008 United States Geological Survey seismic hazard curves. The fire CDF was obtained from an analysis of the fire areas in the final phase of the FIVE screening approach. The NRC staff concludes that the licensee has adequately considered external events because a conservative and quantitative assessment of external events contribution to the risk impact of extending ILRT intervals has been provided, as described in Section 3.2.2, "Estimated Risk Increase," of this SE.

The NRC staff reviewed Section 4.6.2 and Enclosure 3 of the LAR to determine the technical adequacy of SQN's internal events model for this application. Given that the peer review of the licensee's PRA model against Revision 2 of RG 1.200 and the ASME PRA standard was completed, and the licensee dispositioned the peer review findings, the NRC staff concludes that the PRA model used by the licensee is of sufficient technical adequacy to support the evaluation of changes to ILRT frequencies. Accordingly, the first condition is met.

3.3.4 Estimated Risk Increase is Small

The second condition stipulates that the licensee submit documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small and is consistent with the guidance in RG 1.174 and the clarification provided in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2-A. Specifically, a small increase in population dose should be defined as an increase in population dose of less than or equal to either 1.0 person-rem per year or 1 percent of the total population dose, whichever is less restrictive. In addition, a small increase in conditional containment failure probability (CCFP) should be defined as a value marginally greater than that accepted in previous one-time 15-year ILRT extension requests. This would require that the increase in CCFP be less than or equal to 1.5 percentage points. Thus, the associated risk metrics include LERF, population dose, and CCFP.

The licensee reported the results of the plant-specific risk assessment in Section 4.6.1 of the LAR. Details of the risk assessment for SQN, Units 1 and 2, are provided in Section 4.6.2 and

Enclosures 2 and 3 of the LAR. The reported risk impacts are risk impact from baseline, which estimates the impact of a change in test frequency from three tests in 10 years (the test frequency under 10 CFR 50 Appendix J, Option A) to one test in 15 years, and risk impact from current, which estimates the impact of a change from one test in 10 years to one test in 15 years. The following conclusions can be drawn based on the licensee's analysis associated with extending the Type A ILRT frequency:

1. The reported increase in LERF for a change in test frequency from three tests in 10 years to one test in 15 years is 5.81×10^{-7} per year for SQN, Unit 1, and 5.90×10^{-7} per year for SQN, Unit 2 (from Table 36 of Enclosure 2 of the LAR). These numbers include both internal and external events (internal fires and seismic events) and the impacts from corrosion and loss of containment overpressure due to a large pre-existing containment liner leak. These changes in internal and external events risk are considered to be "small" (i.e., between 1×10^{-6} per year and 1×10^{-7} per year) per acceptance guidelines in RG 1.174. According to RG 1.174, an assessment of baseline LERF is required to show that the total LERF is less than 1×10^{-5} per reactor year. In Enclosure 3 of the LAR, the licensee estimated the total base LERF to be 6.68×10^{-6} per year for SQN, Unit 1, and 6.83×10^{-6} per year for SQN, Unit 2. Thus, the new total LERFs, given the increase in ILRT interval, would be approximately 7.26×10^{-6} per year for SQN, Unit 1, and 7.43×10^{-6} per year for SQN, Unit 2, which are below the total LERF value of 1.0×10^{-5} per reactor year in RG 1.174.

2. Given a change in Type A ILRT frequency from three tests in 10 years to one test in 15 years, the reported increase in the total population dose for SQN, Unit 1, is 6.43×10^{-2} person-rem per year, and for SQN, Unit 2, is 6.74×10^{-2} person-rem per year, or a maximum of 0.57 percent of the total population dose (Table 36 of Enclosure 2). These values are based on the population dose data from NUREG/CR-4551, Revision 1, Volume 5, "Evaluation of Severe Accident Risks: Sequoyah, Unit 1" (ADAMS Accession No. ML070540074). SQN uses the population dose methodology for the Surry collapsed accident progression bins to determine the applicable population dose. The licensee states that the site specific dose estimates are based on analyses performed in support of license renewal assessment of severe accident management alternatives, adjusted to reflect the current SQN population estimates from year 2010. The reported increase in total population dose is below the values provided in EPRI TR-1009325, Revision 2 A, and defined in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2. These numbers only include internal events. However, the NRC staff considered the external events' contribution to LERF and concluded that the change in the total population dose would not impact the NRC staff conclusions in this SER. Thus, this increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

3. The increase in CCFP due to change in test frequency from three tests in 10 years to one test in 15 years is 0.841 percent for SQN, Unit 1, and 0.842 percent for SQN, Unit 2. These values are small and are below the

acceptance guidelines in Section 3.2.4.6 of the NRC SER for NEI 94-01, Revision 2.

Based on the review of the SQN risk assessment results, the NRC staff finds that for SQN, Units 1 and 2, the increase in LERF is small and consistent with the risk acceptance guidelines of RG 1.174. In addition, the increase in the total population dose and the CCFP for the requested change are small to support the LAR. The defense-in-depth philosophy is maintained because the independence of barriers will not be degraded as a result of the requested change, and the use of quantitative risk metrics collectively ensures that the balance between prevention of core damage, prevention of containment failure, and consequence mitigation is preserved. Accordingly, the second condition is met.

3.3.5 Leak Rate for the Large Pre-Existing Containment Leak Rate Case is Properly Used

The third condition stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2, acceptable, the average leak rate for the pre-existing containment large leak rate accident case (i.e., accident case 3b) used by the licensees shall be 100 L_a instead of 35 L_a.

As noted by the licensee in Section 4.6.1 of the LAR, the methodology in EPRI TR-1009325, Revision 2-A, incorporates the use of 100 L_a as the average leak rate for the pre-existing containment large leak rate accident case, and this value has been used in the SQN, Units 1 and 2, plant-specific risk assessments. Accordingly, the third condition is met.

3.3.6 Containment Over-Pressure is Credited for ECCS Performance is Not Applicable

The fourth condition stipulates that in instances where containment over-pressure is relied upon for ECCS performance, an LAR is required to be submitted. In Section 5.5.1 of the LAR, the licensee stated that the plant does not rely on containment overpressure aid in net positive suction head for ECCS injection. Accordingly, the fourth condition does not apply.

3.3.7 Probabilistic Risk Assessment Summary

Based on the above, the NRC staff finds that the LAR for a permanent extension of the Type A containment ILRT frequency from once in 10 years to once in 15 years for SQN, Units 1 and 2, satisfies the applicable PRA requirements, and is, thus, acceptable.

4.0 SUMMARY

Based on the regulatory and technical evaluations above, the NRC staff finds that the licensee has adequately implemented its Reactor Building Leakage Rate Testing Program consisting of ILRT and LLRT. The results of the recent ILRT and LLRT demonstrate acceptable performance of the SQN, Units 1 and 2, containment and demonstrate that the structural and leak-tight integrity of the containment structure is adequately managed and will continue to be periodically monitored and managed by the ILRT and LLRT. The NRC staff finds that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting NEI 94-01, Revision 3-A, and the limitations and conditions identified in the NRC staff SE incorporated in NEI 94-01, Revision 2-A. In addition, the NRC staff finds that the license has adequately performed the required PRA in accordance with approved EPRI TR-1009325, Revision 2-A. The increased risk from the proposed change in test interval in LERF is small and consistent with the risk acceptance guidelines of RG 1.174.

Therefore, the NRC staff concludes that it is acceptable to approve the proposed license amendment in References 1, 2 and 5 for SQN, Units 1 and 2, as follows:

1. Revise TS 5.5.14, "Containment Leakage Rate Testing Program," to adopt NEI 94-01, Revision 3-A, and the limitations and conditions identified in the NRC staff SE incorporated into NEI 94-01, Revision 2-A, as the implementation document.

2. Extend the current performance-based Type A test interval to 15 years and Type C test interval to 75 months.

5.0 REFERENCES

1. Letter dated December 2, 2014, from J.W. Shea, Tennessee Valley Authority, to USNRC regarding License Amendment Request to Revise Sequoyah Nuclear Plant Units 1 and 2 Technical Specification Section 6.8.4.h, "Containment Leakage Rate Testing Program (SQN-TS-14-03)" (ADAMS Accession No. ML14339A539).

2. Letter dated September 11, 2015, from J.W. Shea, Tennessee Valley Authority, to USNRC regarding Application to Revise Technical Specification 6.8.4.h, "Containment Leakage Rate Testing Program (SQN-TS-14-03), Response to Request for Additional Information" (ADAMS Accession No. ML15257A392).

3. Nuclear Energy Institute 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," July 2012 (ADAMS Accession No. ML12221A202).

4. Nuclear Energy Institute 94-01, Revision 2-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," October 2008 (ADAMS Accession No. ML100620847).

5. Letter dated October 23, 2015, from J.W. Shea, Tennessee Valley Authority, to USNRC regarding Application to Revise Technical Specification 6.8.4.h, "Containment Leakage Rate Testing Program (SQN-TS-14-03), Supplement to Response to Request for Additional Information" (ADAMS Accession No. ML15299A140).

6.0 STATE CONSULTATION

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

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in amounts, and no significant change in the types of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (80 FR 13914). Accordingly, the amendment meets 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 amendment.

8.0 CONCLUSION

The NRC staff 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.

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Date: November 30, 2015

A copy of the staff's related Safety Evaluation is enclosed. Notice of Issuance will be included in the Commission's Biweekly *Federal Register* Notice.

Sincerely,

/RA/

Andrew Hon, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures:
1. Amendment No. 335 to RFOL No. DPR-77
2. Amendment No. 328 to RFOL No. DPR-79
3. Safety Evaluation

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