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

October 29, 2018

Ms. Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P. O. Box 1295, Bin 038
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS TO EXTEND THE CONTAINMENT TYPE A LEAK RATE TEST FREQUENCY TO 15 YEARS AND TYPE C LEAK RATE TEST FREQUENCY TO 75 MONTHS (CAC NOS. MG0240 AND MG0241; EPID L-2017-LLA-0295)

Dear Ms. Gayheart:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 197 to Renewed Facility Operating License NPF-68 and Amendment No. 180 to Renewed Facility Operating License NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2, (Vogtle) respectively. The amendments consist of changes to the License and Technical Specifications (TSs) in response to your application dated September 12, 2017, as supplemented by letter dated April 5, 2018.

The amendments revise TS 5.5.17, "Containment Leakage Rate Testing Program," to allow (1) an increase in the existing Type A Integrated Leakage Rate Testing Program test interval from 10 years to 15 years, in accordance with Nuclear Energy Institute (NEI) Topical Report NEI 94-01, Revision 3-A, "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; (2) adoption of an extension of the containment isolation valve leakage testing (Type C) frequency from 60 months to 75 months for Type C leakage rate testing of selected components, in accordance with NEI 94-01, Revision 3 A; (3) adoption of the use of American National Standards Institute/American Nuclear Society (ANSI/ANS)-56.8-2002, "Containment System Leakage Testing Requirements"; and (4) adoption of a more conservative grace interval of 9 months for Type A, Type B, and Type C leakage tests, in accordance with NEI 94-01, Revision 3-A.

The amendments also make the following administrative change consisting of deleting the information regarding the performance of the next Vogtle Type A test to be performed in March 2002 for Unit 1 and March 1995 for Unit 2, as both Type A tests have already occurred.

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

Sincerely,

Michael Orenak, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 197 to NPF-68
2. Amendment No. 180 to NPF-81
3. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 197
Renewed License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Renewed Facility Operating License No. NPF-68 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated September 12, 2017, as supplemented by letter dated April 5, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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. NPF-68 is hereby amended to read as follows:

Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 197, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-68
and the Technical Specifications

Date of Issuance: October 29, 2018

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 180
Renewed License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Renewed Facility Operating License No. NPF-81 filed by the Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated September 12, 2017, as supplemented by letter dated April 5, 2018, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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. NPF-81 is hereby amended to read as follows:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 180, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-81
and the Technical Specifications

Date of Issuance: October 29, 2018

ATTACHMENT

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

TO LICENSE AMENDMENT NO. 197

RENEWED FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 180

RENEWED FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace the following pages of the Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
License No. NPF-68, page 4	License No. NPF-68, page 4
License No. NPF-81, page 3	License No. NPF-81, page 3
TSs	TSs
5.5-16	5.5-16
5.5-17	5.5-17

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 3625.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 197, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Southern Nuclear Operating Company shall be capable of establishing containment hydrogen monitoring within 90 minutes of initiating safety injection following a loss of coolant accident.

(4) Deleted

(5) Deleted

(6) Deleted

(7) Deleted

(8) Deleted

(9) Deleted

(10) Mitigation Strategy License Condition

The licensee shall develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(a) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training and response personnel

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for Implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy

(2) Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia, pursuant to the Act and 10 CFR Part 50, to possess but not operate the facility at the designated location in Burke County, Georgia, in accordance with the procedures and limitations set forth in this license;

(3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) Southern Nuclear, 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;

(5) Southern Nuclear, 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;

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as my be produced by the operation of the facility authorized herein.

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 1 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

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 3625.6 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 180, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance requirements (SRs) contained in the Appendix A Technical Specifications and listed below are not required to be performed immediately upon implementation of Amendment No. 74. The SRs listed below shall be

5.5 Programs and Manuals (continued)

5.5.16 MS and FW Piping Inspection Program

This program shall provide for the inspection of the four Main Steam and Feedwater lines from the containment penetration flued head outboard welds, up to the first five-way restraint. The extent of the inservice examinations completed during each inspection interval (ASME Code Section XI) shall provide 100% volumetric examination of circumferential and longitudinal welds to the extent practical. This augmented inservice inspection is consistent with the requirements of NRC Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," November 1975 and Section 6.6 of the FSAR.

5.5.17 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 Nuclear Energy Institute (NEI) Topical Report (TR) NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exceptions:

1. Leakage rate testing for containment purge valves with resilient seals is performed once per 18 months in accordance with LCO 3.6.3, SR 3.6.3.6 and SR 3.0.2.

2. Containment personnel air lock door seals will be tested prior to reestablishing containment integrity when the air lock has been used for containment entry. When containment integrity is required and the air lock has been used for containment entry, door seals will be tested at least once per 30 days during the period that containment entry(ies) is (are) being made.

3. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief or alternative has been authorized by the NRC. At the discretion of the licensee, the containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage.

(continued)

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program (continued)

4. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI code, Subsection IWE, except where relief has been authorized by the NRC.

The peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 37 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.2% of primary containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criteria are $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are` $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$,

   2) For each door, the leakage rate is $\leq 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 197 TO RENEWED FACILITY OPERATING LICENSE NPF-68

AND

AMENDMENT NO. 180 TO RENEWED FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By application dated September 12, 2017 (Reference 1), as supplemented by letter dated April 5, 2018 (Reference 2), Southern Nuclear Operating Company, Inc. (SNC, the licensee), requested changes to the technical specifications (TSs) for the Vogtle Electric Generating Plant (Vogtle), Units 1 and 2. The supplement, dated April 5, 2018 provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published the Federal Register on December 5, 2017 (82 FR 57474).

The proposed changes would revise TS 5.5.17, "Containment Leakage Rate Testing Program," to allow (1) an increase in the existing Type A Integrated Leakage Rate Testing (ILRT) Program test interval from 10 years to 15 years, in accordance with Nuclear Energy Institute (NEI) Topical Report (TR) NE1 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (Reference 3), and the conditions and limitations specified in NEI 94-01, Revision 2-A (Reference 4); (2) adoption of an extension of the containment isolation valve leakage testing (Type C) frequency from 60 months to a 75 months for Type C leakage rate testing of selected components, in accordance with NEI 94-01, Revision 3-A; (3) adoption of the use of American National Standards Institute/American Nuclear Society (ANSI/ANS)-56.8-2002, "Containment System Leakage Testing Requirements"; and (4) adoption of a more conservative grace interval of 9 months for Type A, Type B, and Type C leakage tests in accordance with NEI 94-01, Revision 3-A.

The amendments also make the following administrative change consisting of deleting exception number 5 regarding the performance of the next Type A test to be performed in March 2002 for Vogtle, Unit 1, and March 1995 for Vogtle, Unit 2, as both Type A tests have already occurred.

2.0 REGULATORY EVALUATION

2.1 Background

The overall integrity (structural and leak tight integrity) of the primary containment is verified by a Type A ILRT, and the integrity of the penetrations and isolation valves are verified by Type B and Type C local leak rate tests (LLRTs), as required by 10 CFR Part 50, Appendix J. These tests are performed to verify the essential leak tight characteristics of the containment structure at the design basis accident pressure. The Type A test also provides a verification of structural integrity. The leakage rate testing requirements of 10 CFR Part 50 Appendix J, Option B (Type A, Type B and Type C tests) and the containment inservice inspection (CISI) requirements mandated by 10 CFR 50.55a, "Codes and standards," assist in ensuring the continued integrity of the containment during its service life.

Vogtle TS 5.5.17 currently states, in part, that "[t]he peak calculated primary containment internal pressure for the design basis loss of coolant accident, P_a, is 37 psig (pounds per square inch gauge)," and defines "[t]he maximum allowable containment leakage rate, L_a, at P_a, at 0.2% (percent) of primary containment air weight per day." As required by 10 CFR Part 50, Appendix J and TS 5.5.17, the Type A, Type B, and Type C test results must not exceed the L_a with margin. The containment overall leakage rate acceptance criteria is less than or equal to 1.0 L_a. However, during the first unit startup following testing in accordance with 10 CFR Part 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are less than or equal to 0.60 L_a for the combined Type B and Type C tests, and less than or equal to 0.75 L_a for Type A tests.

2.2. Proposed Changes

The requested change would revise TS 5.5.17 to require compliance with NEI 94-01, Revision 3-A, in lieu of Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," September 1995 (Reference 5), including a deletion of exception 5. Additionally, the change would require compliance with the limitations and conditions specified in Section 4.0 of the safety evaluation for NEI 94-01, Revision 2-A.

TS 5.5.17 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. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," dated September 1995, as modified by the following exceptions:

TS 5.5.17 is proposed to state, 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. This program shall be in accordance with the guidelines contained in Nuclear Energy Institute (NEI) Topical Report (TR) 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated

July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008, as modified by the following exceptions:.

Additionally, the deletion of exception number 5, regarding the performance of the next Type A test to be performed in March 2002 for Vogtle, Unit 1, and March 1995 for Vogtle, Unit 2, will occur, as both Type A tests are complete.

The license amendment request (LAR) follows NEI 94-01, Revision 3-A, and the limitations and conditions of Section 4.0 of the NEI 94-01, Revision 2-A, safety evaluation, and Section 4.0 of the NEI 94-01, Revision 3-A, safety evaluation. The licensee proposes an extension of the Type A test interval from 10-year intervals to 15 years from the last Type A test (March 2017 and March 2010 for Vogtle, Units 1 and 2, respectively). The NRC issued Amendment No. 130 for Vogtle, Unit 1, and Amendment No. 108 for Vogtle, Unit 2, on January 12, 2004 (Reference 6), which granted a license amendment for a one-time deferral of the Type A ILRT test interval from 10 to 15 years. The approval of the proposed amendments would allow the next Vogtle, Units 1 and 2, Type A tests to be performed no later than March 2032 and March 2025, instead of no later than March 2027 and March 2020, respectively, based on the current TS requirements. 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 (i.e., performance history) and other requirements stated in Section 9.2.3, "Extended Test Intervals," of NEI 94-01, Revisions 2-A and 3-A. The U.S. Nuclear Regulatory Commission (NRC or the Commission) staff's review of the Vogtle Type A test performance history with respect to meeting the Section 9.2.3 requirements, and the safety evaluation limitations and conditions, is presented in Sections 3.1.1 and 3.2.1, respectively, of this safety evaluation.

The licensee also proposes an extension of the Type C test interval. For Vogtle, Units 1 and 2, Type C tests are currently required to be performed at no longer than a 60-month interval. The proposed amendment would extend the Type C test interval to no longer than 75-months from the last Type C test, with a permissible extension period of 9 months (total of 84 months) for non-routine emergent conditions, based on acceptable performance. The NEI 94-01, Revision 3-A, guidelines explain that extensions of 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, "Type C Test Interval," in NEI 94-01, Revision 3-A. The NRC staff's review of the Vogtle Type C test performance history with respect to meeting the Section 10.2.3 requirements, and the safety evaluation limitations and conditions, is presented in Sections 3.1.2 and 3.2.2, respectively, of this safety evaluation.

2.3 Regulations and Guidance

The regulations at Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36(c)(5), "Technical Specifications," require, in part, the inclusion of administrative controls in TSs that are necessary to ensure operation of the facility in a safe manner.

Section 50.55a of 10 CFR, "Codes and standards," contains the CISI program requirements that, in conjunction with the requirements of Appendix J, ensure the continued leaktight and structural integrity of the containment during its service life.

Section 50.65 of 10 CFR, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," 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, as defined in paragraph (b) of this section, 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."

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, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50. Appendix J to 10 CFR Part 50 includes two options: Option A–Prescriptive Requirements, and Option B–Performance-Based Requirements, either of which can be chosen for meeting the requirements of Appendix J.

The testing requirements in 10 CFR Part 50, Appendix J, ensure that leakage through the primary containment and related systems and components penetrating primary containment does not exceed allowable leakage rate value specified in the TSs or associated bases, and that integrity of the containment structure is maintained during its service life.

The licensee has adopted and has been implementing Option B for meeting the requirements of 10 CFR Part 50, Appendix J. Option B specifies the performance-based requirements and criteria for preoperational and subsequent leakage-rate testing. These requirements are met by (1) performance of Type A tests to measure the containment system's 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 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), to ensure the 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 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, 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.

Section V.B.3 of 10 CFR 50, Appendix J, Option B, requires that the RG or other implementation document used by a licensee to develop a performance-based leakage-testing program 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 RG.

The implementation document that is currently referenced in TS 5.5.17 is RG 1.163. RG 1.163 endorsed Nuclear Energy Institute (NEI) TR 94-01, Revision 0 (Reference 7), as a document that provides methods acceptable to the NRC staff for complying with the provisions of Option B, subject to four regulatory positions delineated in Section C of RG 1.163. NEI 94-01, Revision 0, includes provisions that allow the performance-based Type A test interval to be extended for up to 10 years, based upon two consecutive successful tests.

NEI 94-01, Revisions 2 and 3, were reviewed by the NRC and approved for use. The final safety evaluation for Revision 2 (Reference 8), documents the NRC's evaluation and acceptance of Revision 2, subject to six specific limitations and conditions listed in Section 4.1 of the safety evaluation for Type A tests. The safety evaluation also states that the NRC staff agrees with methodology used in ANSI/ANS-56.8-2002 and accepts it as a reference for how licensees should perform the Type A, Type B, and Type C tests. The final safety evaluation for Revision 3 (Reference 9) includes two specific limitations and conditions listed in Section 4.0 of the safety evaluation for Type C tests. The approved versions of NEI 94-01, Revisions 2 and 3, incorporating the NRC staff's safety evaluations, were issued as NEI 94-01, Revision 2-A, and NEI 94-01, Revision 3-A, respectively. Consistent with the requested change, the licensee's submittal was reviewed against the limitations and conditions presented in the safety evaluations included in NEI 94-01, Revisions 2-A and 3-A.

Revision 4.0 of NUREG-1431, "Standard Technical Specifications – Westinghouse Plants: Specifications," contains the improved Standard Technical Specifications (STS) for Westinghouse plants with incorporation of the TS "Containment Leakage Rate Testing Program" (Reference 10), that provided guidance for specific changes to TS for implementation of Option B.

Revision 2 of RG 1.200, "An Approach tor Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (Reference 11), describes an acceptable approach for determining whether the quality of the probabilistic risk assessment (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. Revision 2 of RG 1.200, provides guidance for assessing the technical acceptability of a PRA. Revision 2 of RG 1.200, endorses, with clarifications and qualifications, the use of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard, RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment tor Nuclear Power Plant Applications" (i.e., the ASME/ANS RA-SA 2009 PRA Standard) (Reference 12).

Revision 3 of RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (Reference 13), describes an acceptable risk-informed approach for assessing the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights. This RG also provides risk acceptance guidelines for evaluating the results of such evaluations.

3.0 TECHNICAL EVALUATION

3.1 Deterministic Considerations: Structural and Leak Integrity of the Containment

3.1.1 Historical Type A Test Results

Vogtle, Unit 1

Per TS 5.5.17, the Vogtle, Unit 1, containment was designed for a maximum allowable containment leakage rate (L_a) of 0.2 percent by weight of containment air weight per day at the calculated peak pressure (P_a). TS 5.5.17 indicates that the P_a for the design basis loss of coolant accident (DBLOCA) is 37 psig. Since 1986, a total of five ILRTs have been performed on the Vogtle, Unit 1, containment. All five ILRTs had satisfactory leakage rate results and the most recent two tests were performed at a test pressure within the limitations of American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.8-1994, Section 3.2.11, "Type A Test Pressure," (Reference 14). These five ILRT test results were documented in Enclosure 1, Table 3.2.4-1, "[Vogtle] Unit 1 ILRT Test History" and Table 3.2.4-3, "[Vogtle] ILRT

Test Results – Verification of Current Extended ILRT Interval." The test results are summarized in Table 3.1.1-1 below:

TABLE 3.1.1-1

Vogtle Electric Generating Plant Unit 1 Type A ILRT History

Test Date	Upper 95% Confidence Level (wt.% /day) / (Test Pressure -psig)(1)	Level Corrections (wt.%/day)	As Left Min Pathway Penalty for Isolated Pathways (wt.%/day)	Adjusted As left Leak Rate (wt.%/day)	ILRT Acceptance Criteria(2), L_a (wt.%/day)	Test Method / Data Analysis Techniques
August 1986	0.02489(4)	(3)	(3)	(3)	0.2	(3)
March 1990	0.1048(4)	(3)	(3)	(3)	0.2	(3)
March 1993	0.1344(4)	(3)	(3)	(3)	0.2	(3)
March 2002	0.0332 (5) (37.82 psig)	-0.0063(5)	0.000565(5)	0.0275(5)	0.2	Absolute Method / Mass Point Analysis(6)
March 2017	0.0611 (5) (36.76 psig)	-0.0012(5)	0.0005(5)	0.0604(5)	0.2	Absolute Method / Mass Point Analysis(6)

Table 3.1.1-1 Notes:

(1) P_a is 37 psig. The minimum allowed test pressure is 35.52 psig (i.e. 37 psig x 0.96) per ANSI 56.8-1994, Section 3.2.11 "Type A Test Pressure."
(2) Per TS 5.5.17
(3) Not Available as not provided in the LAR
(4) Data source LAR Table 3.2.4-1
(5) Data source LAR Table 3.2.4-3
(6) Mass Point per ANSI/ANS 56.8-1994

The licensee stated that the data represented in the last two rows of Table 3.1.1-1 of this safety evaluation was gathered consistent with the definition of "Performance Leakage Rate" as defined in Section 5.0 of NEI 94-01, Revisions 2-A and 3-A.

Section 9.1.2 of NEI 94-01, Revision 3-A, states, in part,

> The 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.

Since the elapsed time between the last two Type A tests was 15 years, Section 9.1.2 of NEI 94-01, Revision 3-A, has been satisfied.

TS 5.5.17.a establishes a maximum limit of less than or equal to 0.75 L_a, which equals 0.150 percent of containment air weight per day, for the Vogtle, Unit 1, "As-Left" leakage rate for unit startup following completion of Type A testing. The Vogtle, Unit 1, containment was designed for a leakage rate, L_a, not to exceed 0.2 percent by weight of containment air per day at the calculated peak pressure, P_a. As shown in Table 3.1.1-1 of this safety evaluation, the Vogtle, Unit 1, ILRT results since August 1986 demonstrated ample margin (i.e., greater than 32 percent) between each "As-found" "Upper 95% Confidence Level" value and L_a.

The past five Vogtle, Unit 1, ILRT results have confirmed that the containment leakage rates are acceptable with respect to L_a. The last two Type A tests for Vogtle, Unit 1, had "as found" test results of less than 1.0 L_a at the DBLOCA pressure (P_a) and the guidelines in NEI 94-01, Revisions 2-A and 3-A, regarding acceptable performance history, have been met. Because of these test results, the NRC staff concludes that the results of the Type A ILRTs for Vogtle, Unit 1, provide reasonable assurance that containment overall leakage will be maintained below the design-basis leak rate, consistent with the requirements in TS 5.5.17, and will fulfill the requirements of 10 CFR 50, Appendix J, Option B with having a test frequency of 15 years.

Vogtle, Unit 2

Per TS 5.5.17, the Vogtle, Unit 2, containment was designed for a L_a of 0.2 percent by weight of containment air weight per day at P_a. TS 5.5.17 indicates that the peak calculated peak containment internal pressure for the DBLOCA P_a is 37 psig.

Since November 1988, a total of four ILRTs have been performed on the Vogtle, Unit 2, containment and they all had satisfactory leakage rate results. The most recent two tests were performed at a test pressure within the limitations of ANSI 56.8-1994, Section 3.2.11. The test results are documented in Enclosure 1, LAR Table 3.2.4-2, "[Vogtle] Unit 2 ILRT Test History" and LAR Table 3.2.4-3, "[Vogtle] ILRT Test Results – Verification of Current Extended ILRT Interval." The test results are summarized in Table 3.1.1-2 below:

TABLE 3.1.1-2
Vogtle Electric Generating Plant Unit 2 Type A ILRT History

Test Date	Upper 95% Confidence Level (wt.% /day) / (Test Pressure -psig)[1]	Level Corrections (wt.%/day)	As Left Min Pathway Penalty for Isolated Pathways (wt.%/day)	Adjusted As left Leak Rate (wt.%/day)	ILRT Acceptance Criteria[2], L_a (wt.%/day)	Test Method / Data Analysis Techniques
November 1988	0.03154[4]	[3]	[3]	[3]	0.2	[3]
April 1992	0.1373[4]	[3]	[3]	[3]	0.2	[3]
March 1995	0.0938 [5] (37.3048 psig)	[7]	0.0001[5]	0.0939[5]	0.2	Absolute Method / BN-TOP-1, Rev. 1[6]
March 2010	0.1116 [5] (37.05 psig)	-0.0032[5]	0.0009[5]	0.1093[5]	0.2	Absolute Method / BN-TOP-1, Rev. 1[6]

Table 3.1.1-2 Notes:

[1] P_a is 37 psig. The minimum allowed test pressure is 35.52 psig (i.e. 37 psig x 0.96) per ANSI 56.8-1994, Section 3.2.11 "Type A Test Pressure."
[2] Per TS 5.5.17
[3] Not Available as not provided in the LAR
[4] Data source LAR Table 3.2.4-1
[5] Data source LAR Table 3.2.4-3
[6] Total Time (per BN-TOP-1, 1972) [Reference 5.4]
[7] No correction was included, yielded a slight reduction in the leakage rate

The licensee indicated that the data represented in the last two rows of Table 3.1.1-2 of this safety evaluation, was gathered consistent with the Definition of "Performance leakage rate" as defined in Section 5.0 of NEI 94-01 Revisions 2A and 3A.

Since the elapsed time between the last two Type A tests was 15 years, Section 9.1.2 of NEI 94-01, Revision 3-A, has been satisfied.

TS 5.5.17 references RG 1.163. Regulatory Position C of RG 1.1.63 in turn states that NEI 94-01, Revision 0 "[p]rovides methods acceptable to the NRC staff for complying with the provisions of Option B in Appendix J to 10 CFR Part 50." Section 9.2.3 "Extended Test Intervals" of NEI 94-01, Revision 0, states, 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, is identical, except that ANSI 56.8-1994 is replaced by ANSI 56.8-2002 (Reference 14). Section 9.2.3 of NEI 94-01, Revision 0, does not mandate that a licensee recalculate past Type A test results to demonstrate conformance with the definition of "performance leakage rate" contained in NEI 94-01, Revision 3-A. Because the results since November 1988 demonstrated ample margin (i.e., less than 31 percent) between each "As-found" "Upper 95% Confidence Level" value and L_a, the NRC staff finds it acceptable that the licensee did not recalculate the Vogtle, Unit 2, Type A test results from earlier than March 1995.

TS 5.5.17.a establishes a maximum limit of less than or equal to 0.75 L_a, which equals 0.150 percent of containment air weight per day, for the Vogtle, Unit 2, "As-Left" leakage rate for unit startup following completion of Type A testing. The Vogtle, Unit 2, containment was designed for L_a not to exceed 0.2 percent by weight of containment air per day at P_a. As shown in Table 3.1.1-2 of this safety evaluation, there has been adequate margin to L_a for all historical ILRTs spanning a period of time greater than twenty years.

The past four Vogtle, Unit 2, ILRT results dating back to 1988 have confirmed that the containment leakage rates are acceptable with respect to the design L_a. The last two Type A tests for Vogtle, Unit 2, had "as found" test results of less than 1.0 L_a and the guidelines in NEI 94-01, Revisions 2-A and 3-A, regarding acceptable performance history, have been met. The NRC staff concludes that the results of the Type A ILRTs for Vogtle, Unit 2, provide reasonable assurance that containment overall leakage will be maintained below the design-basis leak rate, consistent with the requirements in TS 5.5.17, and will fulfill the requirements of 10 CFR 50, Appendix J, Option B with a test frequency of 15 years.

3.1.2 <u>Types B and C Leak Rate Test History</u>

Type B testing ensures that the leakage rate of individual containment penetration components is acceptable. Type C testing ensures that individual containment isolation valves (CIVs) are essentially leak tight. In addition, aggregate Type B and Type C leakage rates support the leakage tightness of both containments by minimizing potential leakage paths.

Vogtle, Unit 1

The leakage rate acceptance criteria stated in TS 5.5.17 is less than or equal to 0.60 L_a (where 0.60 L_a = 0.6 x 380,445 standard cubic centimeters per minute (sccm) = 228,273 sccm) for the combined Type B and Type C tests during the first unit startup following testing.

The NRC staff reviewed the local leak rate summaries contained in LAR Table 3.4.5-1, "[Vogtle] Unit 1 Types B and C LLRT Combined As-Found/As-Left Trend Summary," of the LAR. Using the provided L_a values and the data contained in LAR Table 3.4.5-1, the NRC staff confirmed the accuracy of the "Fraction of 0.6 L_a" values contained in Table 3.4.5-1 and finds that:

- The Vogtle, Unit 1, "As-Found" minimum pathway leakage rates for the last seven refueling outages since 2008 have an average of 3.61 percent of 0.6 L_a with a high of 8.64 percent of 0.6 L_a.

- The Vogtle, Unit 1, "As-Left" maximum pathway leakage rates for the last seven refueling outages since 2008 have an average of 6.19 percent of 0.6 L_a with a high of 8.24 percent of 0.6 L_a.

As stated in Section 3.4.6, "Type B and Type C Local Leak Rate Testing Program Implementation Review," of the LAR, all Vogtle, Unit 1, penetrations eligible for extended intervals are on extended intervals except Type C Penetration 40, which failed its "As-Found" LLRT during refueling outage 1R19. This "As-Found" LLRT penetration failure was attributed to test error as subsequent disassembly of the CIVs found nothing wrong with the valves. The testing error consisted of the failure to verify that the valves associated with the Type C test boundary were not leaking before establishing a leakage rate for the CIVs. The licensee stated that the testing frequency for both valves was reset to 18 months. After two consecutive, successive tests, the licensee extended it to 60 months. As reflected in LAR Table 3.4.6-1, "[Vogtle] Unit 1 Types B and C LLRT Program Implementation Review," other than the Penetration 40 Type C LLRT failure, there were no other failures of Type B and Type C penetrations eligible for extended performance intervals during refueling outage 1R19 in 2015 and refueling outage 1R20 in 2017.

The licensee provided a sufficient explanation about the cause of failure of the sole LLRT Type B and Type C penetration failure experienced during the last two Vogtle, Unit 1, refueling outages. In addition, consistent with 10 CFR Part 50, Appendix J, Option B, requirements, the licensee reestablished a maximum test frequency of 18 months for the CIVs associated with Penetration 40. Furthermore, based on the review of LAR Table 3.4.5-1 of the LAR, the aggregate results of the "As-Found Minimum Pathway" for all Vogtle, Unit 1, Type B and C tests from 2008 through 2017 demonstrate a history of adequate maintenance since the aggregate test results at the end of each operating cycle were all well below (i.e., greater than 91 percent margin) the Type B and Type C test leakage rate acceptance criteria of less than or equal to 0.60 L_a. Based on the review of the historical information, the NRC staff finds that the licensee is adequately implementing the requirements of its Appendix J, Option B, performance-based testing program.

The NRC staff finds that the Type B and Type C tests for Vogtle, Unit 1, were less than the design-basis leak rate and the guidelines in NEI 94-01, Revisions 2-A and 3-A, regarding acceptable performance history, have been met. Therefore, the NRC staff concludes that the results of the Type B and Type C tests provide reasonable assurance that as-found minimum pathway totals will be maintained below the design-basis leak rate at Vogtle, Unit 1, consistent with the requirements in TS 5.5.17 and fulfill the requirements of 10 CFR 50, Appendix J, Option B, when the Type C test frequency is 75 months.

Vogtle, Unit 2

The leakage rate acceptance criteria stated in TS 5.5.17 is less than or equal to 0.60 L_a (where 0.60 L_a = 228,273 sccm) for the combined Type B and Type C tests during the first unit startup following testing.

The NRC staff reviewed the local leak rate summaries contained in LAR Table 3.4.5-2, "[Vogtle] Unit 2 Types B and C LLRT Combined As-Found/As-Left Trend Summary," of the LAR. Using the provided L_a values and the data contained in LAR Table 3.4.5-2, the NRC staff confirmed the accuracy of the "Fraction of 0.6 L_a" values contained in LAR Table 3.4.5-2 and finds that:

- The Vogtle, Unit 2, "As-Found" minimum pathway leakage rates for the last seven refueling outages since 2007 have an average of 1.59 percent of 0.6 L_a with a high of 2.33 percent of 0.6 L_a.

- The Vogtle, Unit 2, "As-Left" maximum pathway leakage rates for the last seven refueling outages since 2007 have an average of 4.44 percent of 0.6 L_a with a high of 6.21 percent of 0.6 L_a.

As stated in Section 3.4.6 of the LAR, all but one Vogtle, Unit 2, Type B and Type C penetrations eligible for extended test intervals are on extended intervals. The sole exception was Type C Penetration 78, which failed its "As-Found" LLRT during three consecutive refueling outages 2R16, 2R17, and 2R18. The failure during 2R18 was attributed to trash being stuck inside valve 2-HV-0781. Due to similar Penetration 78 failures experience during 2R16 and 2R17, the licensee stated that the air operated valve (AOV) program engineer activated the program logistics to rebuild the valve actuator and to determine the cause of failure during either refueling outages 2R19 or 2R20. Consistent with Appendix J, Option B, requirements, the current test interval for Penetration 78 currently remains at 18 months. Prior to returning Penetration 78 Type C testing to an extended test interval, the licensee stated that:

- A determination and the resolution of the cause of the failure must take place; and
- The resolution will be followed by the performance of two successful LLRTs during consecutive refueling outages.

As reflected in LAR Table 3.4.6-2, "[Vogtle] Unit 2 Types B and C LLRT Program Implementation Review," there were no other failures of eligible Type B and Type C penetrations during refueling outage 2R17 in 2014 and refueling outage 2R18 in 2016.

Based on the NRC staff's review of the historical information provided in Sections 3.4.5 "Containment Leakage Rate Testing Program - Type B and Type C Testing Program" and Section 3.4.6 of the LAR, there was no indication of the licensee's failure to adequately implement the requirements of its Appendix J Option B performance-based testing program.

The licensee provided an adequate preliminary explanation about the cause of the only LLRT Type B and Type C penetration failure experienced during the last two Vogtle, Unit 2, refueling outages. Furthermore, a preventative maintenance program ticket was initiated by the AOV program engineer to rebuild the valve actuator and establish a definitive cause of failure. In addition, consistent with Appendix J, Option B, requirements, the licensee is maintaining a maximum test frequency of 18 months for the CIVs associated with Penetration 78. Furthermore, based on the review of Table 3.4.5-2 of the LAR, the NRC staff finds that the aggregate results of the "As-Found Minimum Pathway" for all Vogtle, Unit 2, Type B and C tests from 2007 through 2016 demonstrates a history of adequate maintenance since the aggregate test results at the end of each operating cycle were all well below (i.e. greater than 97 percent margin) the Type B and Type C test leakage rate acceptance criteria of less than or equal to 0.60 L_a. Based on the review of the above, the NRC staff finds that the licensee is adequately implementing the requirements of its Appendix J, Option B, performance-based testing program for Type B and Type C tests.

The NRC staff finds that the Type B and Type C tests for Vogtle, Unit 2, were less than the design-basis leak rate, and the guidelines in NEI 94-01, Revisions 2-A and 3-A, regarding acceptable performance history, have been met. Therefore, the NRC staff concludes that the results of the Type B and Type C tests provide reasonable assurance that as-found minimum pathway totals will be maintained below the design-basis leak rate at Vogtle, Unit 2, consistent with the requirements in TS 5.5.17 and fulfill the requirements of 10 CFR 50, Appendix J, Option B, when the Type C test frequency is 75 months.

3.1.3 Deletion of Exception 5 to TS 5.5.17

The licensee proposed to delete the exception 5 to TS 5.5.17, which states:

> The next Type A test, after the March 2002 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years.

As indicated in Table 3.2.4-3 of the LAR, the first Vogtle, Unit 1, Type A test after the March 2002 ILRT was completed during the Vogtle, Unit 1, refueling outage 1R20 in March 2017. Similarly as indicated by the Table 3.2.4-3 of the LAR, the first Vogtle, Unit 2, Type A test after the March 1995 ILRT was completed during Vogtle, Unit 2, refueling outage 2R14 in March 2010. Since the ILRTs associated with exception 5 were performed, NRC staff concludes that the deletion of TS 5.5.17 exception 5 is acceptable.

3.1.4 Containment Inservice Inspection (CISI) Program

The containment for Vogtle, Units 1 and 2, is a steel-lined, post-tensioned reinforced concrete cylinder with a hemispherical dome supported on a flat, conventionally reinforced concrete basemat with a central cavity and instrumentation tunnel to the reactor vessel. The inside face of the containment is lined with one quarter inch thick steel plates welded together to form a leaktight barrier. The tendon gallery is attached to the underside of the basemat which provides access to the vertical tendons for installation, tensioning, and inservice inspection.

The licensee stated that it is implementing its CISI Program in accordance with the applicable edition/addenda of Subsections IWE/IWL of Section XI, Division 1, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, subject to the applicable regulatory conditions as required by 10 CFR 50.55a(g)(4)(iv), "Applicable ISI Code: Use of subsequent editions and addenda." The CISI Program for the third 10-year ISI interval was developed utilizing the 2007 Edition through the 2008 Addenda of the ASME BPV Code, Subsections IWE and IWL (Reference 16) and 10 CFR 50.55a. Subsection IWL provides the rules and requirements for inservice inspection of Class CC (concrete containment) components. Subsection IWE provides the rules and requirements for inservice inspection of Class MC (metal containment) pressure-retaining components and requires general visual examination of 100 percent of accessible metallic surfaces of the containment pressure boundary three times over a 10-year inspection interval, pursuant to 10 CFR 50.55a(b)(2)(ix)(E).

Pursuant to IWL-2410(a) and (c), general visual examinations of accessible surfaces of containment concrete and post-tensioning system components of the containment are conducted every 5 years, which would be two examinations over a 10-year interval. The licensee is required to perform general visual examinations for Subsection IWE and general and detailed examinations for Subsection IWL In the LAR, the licensee provided a summary of recent results of IWE and IWL inspections and containment coating assessments.

In Section 3.4.4 of the LAR, the licensee provided a discussion of the results of recent containment examinations performed during refueling outages 1R20 and 2R18 (May 31, 2014 to May 30, 2017) for containment coatings and IWE steel liner examinations; and IWL examinations for the containment structure post-tensioning system for Vogtle, Unit 1 (30th Year), and Vogtle, Unit 2 (25th Year), performed during the period of August 1, 2014, to August 1, 2016. During the containment coating assessment for 1R20, the licensee identified rust on structural steel bolting and some cracking paint. Several condition reports were generated related to spots on the liner plate with exposed bare metal, and flaking and missing paint. During 2R18, the coating assessment was performed in conjunction with the examination of the containment liner plate and concrete floor integrity. Condition reports were generated to create work orders for missing paint and damaged and flaking coatings. The areas noted during 1R20 and 2R18 were identified at elevation 171' for both inside and outside of the concrete bioshield.

The results of IWE examinations performed for certain portions of the auxiliary and containment buildings during 1R20 and 2R18 identified minor surface rust on liner plate welds, areas of exposed metal, and cracked coatings. The licensee stated that the IWE examinations performed were completed with satisfactory results, which demonstrated that the containment pressure boundary is capable of performing its intended function as a leak tight barrier.

The results of the IWL examinations performed on the containment structure post-tensioning system pursuant to IWL-3221 for Vogtle, Unit 1 (30th Year), and Vogtle, Unit 2 (25th Year), led the licensee to conclude that the containment structures have experienced no abnormal degradation of their post-tensioning systems, and all non-conformance items have been identified, documented, and reported as required.

In Section 3.6 of the LAR, the licensee also identified several license renewal aging management programs for the primary containment, which are identified in Chapter 19 of the Vogtle, Units 1 and 2, final safety analysis report (FSAR) (Reference 17). As part of the license renewal effort, SNC had to demonstrate that commitments related to the aging effects applicable for the systems, structures, and components within the scope of the license renewal would be managed adequately during the period of extended operation. The renewed operating licenses for Vogtle, Units 1 and 2, were issued on June 3, 2009, extending the original licensed operating term by 20 years (Reference 18). The following programs and activities are credited with the aging management of the primary containment: FSAR Section 19.2.29, "10 CFR 50 Appendix J Program," which monitors leakage rates through the containment pressure boundary including penetrations and access openings; Section 19.2.30, "ISI-IWE Program," which manages aging effects for the containment liners and their integral attachments, including connecting penetrations and parts forming the leaktight boundary; and Section 19.2.31, "ISI-IWL Program," which manages the reinforced concrete and unbonded post-tensioning systems of the containment structures.

Based on the above, the NRC staff finds that the licensee has an adequate CISI program in place to periodically examine, monitor and manage structural deterioration and aging degradation of the pressure-boundary components of the Vogtle containment. The NRC staff finds that the licensee is satisfactorily monitoring and managing the Vogtle, Units 1 and 2, containments and performing supplemental inspections to periodically examine and monitor aging degradation, thereby providing reasonable assurance that the containment structural and leak-tight integrity will continue to be maintained. Therefore, the NRC staff concludes that the licensee continues to meet the requirements of 10 CFR 50.55a(a)(1)(ii), which endorses ASME BPV Code, Section XI, Division 1, Subsections IWE and IWL.

3.2 NRC Staff Evaluation of the Conditions and Limitations

As discussed in Section 2.0 of this safety evaluation, and in accordance with the guidance in NEI 94-01, Revision 2-A, the licensee proposes to extend the containment Type A test interval from the current approved 10 years to 15 years, based on acceptable performance. The NRC staff's evaluation of the proposed LAR against the limitations and conditions in NEI 94-01, Revision 2-A, is discussed in Section 3.2.1 of this safety evaluation.

As discussed in Section 2.0 of this safety evaluation, and in accordance with the guidance in NEI 94-01, Revision 3-A, the licensee proposes to extend the containment Type C test interval from the current approved 60 months to 75 months, with a permissible extension period of 9 months (total of 84 months) for non-routine emergent conditions, based on acceptable performance. The NRC staff's evaluation of the LAR against the limitations and conditions in NEI 94-01, Revision 3-A, is discussed below in Section 3.2.2 of this safety evaluation.

3.2.1 NEI 94-01, Revision 2, Conditions

Currently, the Vogtle, Units 1 and 2, Appendix J containment leakage rate testing program invokes RG 1.163 as the plan implementation document. The licensee proposes to revise the Appendix J containment leakage rate testing program by replacing RG 1.163 with the guidance contained in NEI 94-01, Revision 3-A, and the conditions and limitations of NEI 94-01, Revision 2-A. The NRC staff found in its June 25, 2008, safety evaluation (Reference 8) that the use of NEI 94-01, Revision 2-A, is acceptable for referencing by licensees proposing to amend their TS to permanently extend the ILRT surveillance interval to 15 years, provided the following applicable six conditions are satisfied:

NRC Condition 1

NRC Condition 1 states:

> 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 ANSI/ANS-56.8-2002. (Refer to SE [safety evaluation] Section 3.1.1.1).

In Table 3.7.1-1, "NEI 94-01 Revision 2-A, Limitations and Conditions," of the September 12, 2017, LAR submittal, the licensee states:

> VEGP [Vogtle] will utilize the definition in NEI 94-01 Revision 3-A, Section 5.0. This definition has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.

The NRC staff reviewed the definitions of "performance leakage rate" contained in NEI 94-01, Revision 2 and Revision 3-A and finds that the definitions contained in both documents are identical. Therefore, the NRC staff concludes that the licensee has adequately addressed and satisfied "Condition 1".

NRC Condition 2

NRC Condition 2 states:

> 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).

In Table 3.7.1-1 of the LAR, the licensee states:

> Reference Section 3.4.2 (Tables 3.4.2-5, 3.4.2-6, 3.4.2-10 and 3.4.2-11) of this LAR submittal.

NEI 94-01, Section 9.2.3.2, "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.

The licensee states in Section 3.4.2, "Containment Inservice Inspection (CISI) Program" of the LAR that the components utilized in the design of the containment that are scoped into the IWL program under item L1.11 are the accessible surface areas of the concrete, and a general visual examination is performed every five years per IWL-2410.

Additionally, Section 3.4.2 of the LAR states, in part:

> The VEGP [Vogtle] Units meet the criteria of IWL-2421(a) allowing for examinations per IWL-2421(b). Consequently, the following Class CC nonexempt items will be subject to examination at the following frequencies:
>
> - **Frequency of 10 years ± 1 year**
>   Tendons (Item L2.10)
>   Tendon wire or strands (Item L2.20)
>
> - **Frequency of 5 years ± 1 year**
>   Tendon anchorages hardware and surrounding concrete (Item L2.30)
>   Tendon corrosion protection medium (Item L2.40)
>   Free Water (Item L2.50)

Vogtle, Unit 1

Table 3.4.2-5, "[Vogtle] Unit 1, IWE Examination Schedule," of the LAR indicates that that the third period of the second CISI interval closed on May 30, 2017. The most recent ILRT occurred in March 2017 during 1R20 and was completed shortly before the close of the second 10-year CISI interval. With approval of the proposed LAR, the next Vogtle, Unit 1, ILRT would need to be completed before or during March 2032. Therefore, projecting 15 years onto the IWE schedule of Table 3.4.2-5 of the LAR, the next ILRT would need to be completed during second period of the fourth CISI interval.

Consistent with the ASME BPV Code, Section XI, Tables 3.4.2-2, "IWE Examination Requirements for Class MC Components - Examination Category E-A, Containment Surfaces," and 3.4.2-3, "IWE Examination Requirements for Class MC Components - Examination Category EC Containment Surfaces Requiring Augmented Examination," of the LAR indicate 100 percent visual inspections of the relevant "Items" for each period of each interval. Table 3.4.2-5 of the LAR indicates that there will be four entire CISI program periods without a completed Type A test between March 2017 and the next required ILRT of March 2032.

Tables 3.4.2-8, "IWL Examination Requirements for Class CC Components Examination Category L-A, Concrete," and 3.4.2-9, "IWL Examination Requirements for Class CC Components Examination Category L-B, Unbonded Post-Tensioning System," provide the details of the ASME BPV Code, Section XI, IWL examination requirements for the concrete surface and the tendons, respectively, for the Vogtle, Unit 1, containment.

Table 3.4.2-10, "[Vogtle] Unit 1, IWL Examination Schedule," of the LAR currently shows no suspected areas (Item No. L1.12) of concrete degradation and all accessible concrete surface area (Item No L1.11) are being inspected on a five-year frequency consistent with the IWL-2421(a) and (b).

Table 3.4.2-10 of the LAR states that the tendons (Item No. L2.10) and wire or strand are being inspected on a ten-year frequency consistent with IWL-2421(a) and (b). Table 3.4.2-10 also states that the following actions take place or are scheduled on a five-year frequency consistent with IWL-2421(a) and (b):

- an inspection of each tendon's "Anchorage, hardware and surrounding concrete" (Item No. L2.30);
- an evaluation of each tendon's "Corrosion protection medium" (Item No. L2.40); and
- a determination of the "Free Water" (Item No. L2.50) associated with each tendon;

Based on the above, the NRC staff finds that the licensee has submitted a schedule of the containment inspections to be performed prior to and between Type A tests. Therefore, the NRC staff concludes that the licensee has addressed and satisfied NRC Condition 2 for Vogtle, Unit 1.

Vogtle, Unit 2

Table 3.4.2-6, "[Vogtle] Unit 2, IWE Examination Schedule," of the LAR indicates that the first period of the second CISI interval closed on May 30, 2010. The most recent ILRT occurred in March 2010 during 2R14 and was completed shortly before the close of the first period of the second 10-year CISI interval. With approval of the proposed LAR, the next Vogtle, Unit 2, ILRT would need to be completed before or during March 2025. Therefore, projecting 15 years onto the IWE schedule of Table 3.4.2-6 of the LAR, the next ILRT would need to be completed during the third period of the third CISI interval.

Consistent with ASME BPV Code, Section XI, Tables 3.4.2-2 and 3.4.2-3 of the LAR indicate 100 percent visual inspections of the relevant "Items" for each period of each interval. Table 3.4.2-6 of the LAR indicates that there will be four entire CISI program periods without a completed Type A test between March 2010 and the next required ILRT of March 2025.

Tables 3.4.2-8 and 3.4.2-9 of the LAR provide the details of the ASME BPV Code, Section XI, IWL examination requirements for the concrete surface and the tendons, respectively, for the Vogtle, Unit 2, containment.

Table 3.4.2-11, "[Vogtle] Unit 2, IWL Examination Schedule," of the LAR shows no suspected areas (Item No. L1.12) of concrete degradation and all accessible concrete surface areas (Item No L1.11) are being inspected on a five-year frequency consistent with IWL-2421(a) and (b).

Table 3.4.2-11 of the LAR states that the tendons (Item No. L2.10) and wire or strand (Item No. L2.20) are being inspected on a ten-year frequency consistent with IWL-2421(a) and (b). The table also states that the following actions take place or are scheduled on a five-year frequency consistent with IWL-2421(a) and (b):

- an inspection of each tendon's "Anchorage, hardware and surrounding concrete" (Item No. L2.30);
- an evaluation of each tendon's "Corrosion protection medium" (Item No. L2.40); and
- a determination of the "Free Water" (Item No. L2.50) associated with each tendon;

Based on the above, the NRC staff finds that the licensee has submitted a schedule of the containment inspections to be performed prior to and between Type A tests. Therefore, the NRC staff concludes that the licensee has addressed and satisfied NRC Condition 2 for Vogtle, Unit 2.

NRC Condition 3

NRC Condition 3 states:

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

In Table 3.7.1-1 of the LAR, the licensee states:

> Reference Section 3.4.2 (Tables 3.4.2-2, 3.4.2-3, 3.4.2-4, 3.4.2-8, and 3.4.2-9) of this LAR submittal.

Inaccessible Areas/ Augmented Examinations

The licensee stated they will use ASME BPV Code, Section XI, paragraph IWE-1232(c) to define containment surface area accessibility.

Additionally, as a result of examinations and analysis, the licensee identified one area during 2R16 as requiring to be placed under augmented examination. This area is located on level 1 of the Vogtle, Unit 2, containment and is located near the personnel hatch. An augmented examination for the liner plate bulge was completed in 2R18 with flaws/areas of degradation remaining essentially unchanged. This area no longer requires augmented examination per IWE-2420(c). No additional areas have been identified requiring augmented examination.

Bellows

Section 3.1, "Description of Primary Containment System," of the LAR states that the bellows associated with the sleeve assemblies of the fuel transfer tube are not subject to ASME BPV Code, Section III, Code Class MC requirements.

Electrical Penetrations

The Vogtle Types B and C testing program requires testing of Type B electrical penetrations in accordance with 10 CFR 50, Appendix J, Option B and RG 1.163. The results of the test program are used to demonstrate that proper maintenance and repairs are made on these components throughout their service life.

Section 3.4.6 of the LAR states that all Vogtle, Units 1 and 2, Type B containment electrical penetrations eligible for extended test intervals are on extended intervals as demonstrated by satisfactory performance within the constraints of 10 CFR 50, Appendix J, Option B and RG 1.163.

Bolting

Table 3.4.2-4, "IWE Examination Requirements for Class MC Components Examination Category E-G, Pressure Retaining Bolting," of the LAR states that the completion of 100 percent VT-1 visual examination will be performed for pressure retaining bolted connections before the end of each CISI Interval. LAR Table 3.4.2-4 lists IWE-3530 as the acceptance standard.

Moisture Barriers

Item No. E1.30, "Moisture Barriers," listed in Table 3.4.2-2 of the LAR requires the completion of 100 percent visual examination during each period of each CISI interval. Table 3.4.2-2 of the LAR lists IWE-3510 as the "Acceptance Standard."

Containment Liner Backed By Concrete

Item No. L1.11 "All accessible surface areas," as listed in Table 3.4.2-8 of the LAR, requires a general visual examination of accessible containment concrete surface areas every five years, per IWL-2410. Item No. L1.12, "Suspect Areas," of Table 3.4.2-8 requires, per the discretion of the responsible engineer for IWL related activities, a detailed visual examination of suspect areas that were identified during the general visual examination.

Summary

The NRC staff finds that the licensee has addressed the containment structure areas that are potentially subject to degradation. Therefore, the NRC staff concludes that the licensee has satisfied NRC Condition 3.

NRC Condition 4

NRC Condition 4 states:

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

In Table 3.7.1-1 of the LAR, the licensee states:

> There have been no major or minor containment repairs or modifications performed nor are any repairs or modifications planned for the containment structure.

The licensee stated that no major repairs or modifications have been performed, nor are repairs or modifications planned for the Vogtle, Units 1 and 2, containment structures. Therefore, the NRC staff concludes that the licensee has adequately addressed and satisfied NRC Condition 4.

NRC Condition 5

NRC Condition 5 states:

> 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 SE [safety evaluation] Section 3.1.1.2).

In Table 3.7.1-1 of the LAR, the licensee states:

> VEGP [Vogtle] will follow the requirements of NEI 94-01 Revision 3-A, Section 9.1. This requirement has remained unchanged from Revision 2-A to Revision 3-A of NEI 94-01.
>
> In accordance with the requirements of NEI 94-01, Revision 2-A, SER [safety evaluation report] Section 3.1.1.2, [Vogtle] will also demonstrate to the NRC staff that an unforeseen emergent condition exists in the event an extension beyond the 15-year interval is required.

The licensee stated that it will follow the requirements of NEI 94-01, Revision 3-A, Section 9.1. NEI 94-01, Revision 3-A, Section 9.1, "Introduction," contains the relevant passage from the NRC staff safety evaluation for NEI 94-01, Revision 2, and states in part:

> ...required surveillance intervals for recommended Type A testing given in this section may be extended by up to 9 months to accommodate unforeseen emergent conditions, but should not be used for routine scheduling and planning purposes.

Therefore, the licensee has demonstrated its understanding that any extension of the Type A test interval beyond the upper-bound performance-based limit of 15 years should be infrequent and that any requested permission (i.e. for such an extension) will demonstrate to the NRC staff that an unforeseen emergent condition exists. Therefore, the NRC staff concludes that the licensee has addressed and satisfied NRC Condition 5.

NRC Condition 6

NRC Condition 6 states:

> 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 [Electric Power Research Institute] Report No. 1009325, Revision 2, including the use of past containment ILRT data.

This condition is not applicable to Vogtle since it was not licensed under 10 CFR Part 52.

Summary

Based on the above, the NRC staff finds that the licensee adequately addresses the six conditions identified in Section 4.1 of the NRC safety evaluation for NEI 94-01, Revision 2-A. Therefore, the NRC staff finds it acceptable for SNC to adopt the "conditions and limitations" of NEI 94-01, Revision 2-A, as part of the implementation documents in TS 5.5.17 for Vogtle, Units 1 and 2.

3.2.2 NEI 94-01, Revision 3, Conditions

The licensee proposes to use NEI 94-01, Revision 3-A, as the implementation document for TS 5.5.17 to govern its Type B and Type C LLRT programs. The licensee stated that they will meet the limitations and conditions in Section 4.0 of the NRC safety evaluation for NEI 94-01, Revision 3-A. Accordingly, Vogtle, Units 1 and 2, will be adopting, in part, the testing criteria ANSI/ANS 56.8-2002 in its licensing basis. The NRC staff has evaluated whether the licensee addressed and satisfied these conditions for Vogtle, Units 1 and 2, as applicable, as discussed below.

NRC Condition 1

NRC Condition 1 states, in part, that:

> NEI TR 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 [NRC] 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 [NRC] 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 [boiling-water reactor 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 stated that their post-outage report shall 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 L_a. When the potential leakage understatement adjusted Types B and C MNPLR total is greater than the Vogtle leakage summation limit of 0.5 L_a, but less than the regulatory limit of 0.6 L_a, the licensee stated it shall perform an analysis and a corrective action plan shall be prepared to restore the leakage summation margin to less than the Vogtle, Units 1 and 2, leakage limit. Additionally, the licensee stated that they will apply the 9-month allowable interval extension period only to eligible Type C components and only for non-routine emergent conditions.

The NRC staff compared the requirements of NEI 94-01, Revision 3, against the statements in the LAR and finds that SNC will meet the requirements. Therefore, the NRC staff concludes that the licensee addressed and satisfied NRC Condition 1.

<u>NRC Condition 2</u>

NRC Condition 2 states, in part, that:

> ISSUE 1 - 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 TR 94-01, Revision 3, Section 12.1.
>
> ISSUE 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 Types B and C total leakage, 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 stated that;

> The change in going from a 60-month extended test interval for Type C tested components to a 75-month interval, as authorized under NEI 94-01, Revision 3-A, represents an increase of 25% [percent] in the LLRT periodicity. As such, VEGP [Vogtle], Units 1 and 2 will conservatively apply a potential leakage understatement adjustment factor of 1.25 to the actual As-Left leak rate, which will increase the As-Left leakage total for each Type C component currently on greater than a 60-month test interval up to the 75-month extended test interval.

Should the Type B and Type C combined totals exceed an administrative limit of 0.5 L_a, but be less than the TS acceptance value (performance criterion) of 0.6 L_a, an analysis will be performed and a corrective action plan prepared to restore and maintain the leakage summation margin to less than the administrative limit. The corrective action plan should focus on those components which have contributed the most to the increase in the leakage summation value and what manner of timely corrective action, as deemed appropriate, best focuses on the prevention of future component leakage performance issues.

NEI 94-01, Revision 3-A, also has a margin-related requirement as contained in Section 12.1, Report Requirements.

The licensee stated that:

> 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 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.

The NRC staff reviewed and compared the requirements of NEI 94-01, Revision 3, against the statements in the LAR and finds that SNC will meet the requirements. Therefore, the NRC staff concludes that the licensee addressed and satisfied NRC Condition 2.

<u>Summary</u>

Based on the above, the NRC staff finds that the licensee adequately addresses both conditions in Section 4.0 of the NRC safety evaluation for NEI 94-01, Revision 3-A. Therefore, the NRC staff finds it acceptable for Vogtle to adopt NEI 94-01, Revision 3-A, as the implementation document in TS 5.5.17 for Vogtle, Units 1 and 2.

3.3 <u>Risk-informed Considerations</u>

An acceptable approach for making risk-informed decisions about proposed TS changes, including both permanent and temporary changes, is to show that the proposed licensing basis (LB) changes meet the five Key Principles stated in RG 1.174, Revision 3, Section C. Key "Principle 4" states:

> When the proposed LB change results in an increase in risk, the increases should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants.

Key "Principle 4" is centered on risk considerations. For proposed LB changes resulting in an increase in risk, the increase should be small and consistent with the intent of the Commission's policy statement on safety goals for the operations of nuclear power plants. The licensee stated that the Vogtle model of record (MOR), Version 5, was utilized to perform the ILRT interval extension analysis. The Key "Principle 4" was evaluated using the risk-informed decision making framework for technical specifications described in the Standard Review Plan (SRP) Chapter 16.1 (Reference 19), RG 1.200, and RG 1.174, Revision 3.

Section 9.2.3.1 of NEI 94-01, Revision 3-A, states that plant-specific confirmatory analyses are required when extending the Type A ILRT interval beyond 10 years. Section 9.2.3.4 of NEI 94-01, Revision 3-A, 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, "Risk Impact Assessment of Extended Integrated Leak Rate Testing Intervals," August 2007 (Reference 20). 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 final safety evaluation for Revision 2 of NEI 94-01, the NRC staff found the methodologies in Revision 2-A of NEI 94-01, and EPRI Final TR-1009325, Revision 2, are acceptable for referencing by licensees proposing to amend their TSs to permanently extend the ILRT interval to 15 years, provided certain conditions are satisfied. The NRC staff set forth the following conditions related to referencing the methodology in EPRI Final TR-1009325, Revision 2:

1. The licensee submits documentation indicating that the technical adequacy of its probabilistic risk assessment (PRA) is consistent with the requirement 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 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 of this SE [the final safety evaluation for Revision 2 of NEI 94-01.]

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

3.3.1 Plant-Specific Risk Evaluation

For its review, the NRC staff used the licensee's risk evaluation contained in Attachment 1 of the LAR (Reference 1) and the supplemental letter dated April 5, 2018 (Reference 2). Additionally, the NRC staff reviewed the safety evaluation in support of amendments 188 and 171 for Vogtle, Units 1 and 2, respectively, dated August 8, 2017 (Reference 21), regarding the implementation of risk-managed TS.

In Section 2, "Methodology," and Section A.2.3, "Risk Assessment Methodology Summary," of Attachment 1 of the LAR, the licensee stated that the plant-specific risk assessment followed the guidance of NEI 94-01, Revision 3-A; the methodology described in EPRI Report 1018243, Revision 2-A; and the NRC regulatory guidance outlined in RG 1.174, Revision 3, on the use of PRA and risk insights in support of a LAR for changes to a plant's licensing basis. In addition, the methodology approved by the NRC for Calvert Cliffs Nuclear Power Plant in the safety evaluation dated May 1, 2002, (Reference 22) to estimate the likelihood and risk implication of undetected corrosion-induced leakage of steel containment liners was used. This methodology considered the additional window of vulnerability from extending the ILRT interval to estimate the conditional containment failure probability and its effect on the Large, Early Release Frequency (LERF) and the estimated population dose.

The licensee's responses to the four limitations and conditions for EPRI Report No. 1009325, Revision 2, contained within Section 4.2 of the safety evaluation for NEI 94-01, Revision 2, are reviewed below.

3.3.2 Evaluation of PRA Acceptability

The first condition stipulates that the licensee will submit documentation indicating that the technical adequacy of its PRA is consistent with the requirements of RG 1.200, Revision 2, relevant to the ILRT extension application.

RG 1.200, Revision 2, describes one acceptable approach for determining whether the technical acceptability of a PRA is sufficient for use in regulatory decision making for light-water reactors. The purpose of RG 1.200, Revision 2, is: (a) to provide guidance to licensees for use in determining the technical acceptability of the base PRA used in a risk-informed regulatory activity, and (b) to endorse industry standards and peer-review guidance.

In Regulatory Issue Summary 2007-06, "Regulatory Guide 1.200 Implementation" (Reference 23), the NRC clarified that for all risk-informed applications received after December 2007, the NRC staff will use Revision 1 of RG 1.200 (Reference 24) to assess technical adequacy of the PRA used to support risk-informed applications. Revision 2 of RG 1.200 (Reference 11) will be used for all risk-informed applications received after March 2010. In Section 3.2.4.1 of the safety evaluation for NEI 94-01, Revision 2 (Reference 8), and EPRI TR-1009325, Revision 2, the NRC staff states, in part, that:

> Licensee 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 license amendment requests. 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 PRA to address the technical adequacy requirements of RG 1.200, Revision 1. Capability category I of ASME RA-Sa-2003 shall be applied as the standard, since approximate values of [core damage frequencies] CDF and LERF and their distribution among release categories are sufficient for use in the EPRI methodology. 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.

3.3.2.1 Scope of PRA

The scope of the licensee's evaluation should include an assessment of the change in risk for internal events (IE), internal flood (IF), internal fires, seismic events, and other external hazards. The Vogtle PRA model is composed of an internal events PRA (IEPRA) (including IF), FPRA and a seismic PRA (SPRA). The Vogtle IEPRA model is an at-power, Level I and LERF PRA that includes internal events and internal floods.

3.3.2.2 Internal Events and Internal Flooding

The licensee stated that evaluation of the technical adequacy of its IEPRA model consisted of a peer review (May 2009), and an internal flooding review (2011). In May 2009, a full-scope peer review of the Vogtle IEPRA (including IF) was performed against the requirements of ASME PRA Standard RA-Sb-2005 (Reference 25) and RG 1.200, Revision 1. The results identified three "not met" facts and observations (F&Os). A gap assessment was performed between ASME RA-Sb-2005, as qualified by RG 1.200, Revision 1, and the ASME/ANS RA-Sa-2009 PRA Standard, as qualified by RG 1.200, Revision 2.

In Table A-1, "Resolution of the Vogtle Internal Events PRA Peer Review F&Os," in Appendix A of the LAR, the licensee reconciled each F&O for the IEPRA by either providing a description of how the F&O was resolved or providing an assessment of the impact of resolution of the F&O on the results for the ILRT extension. The NRC staff evaluated each F&O and the licensee's disposition in Table A-1 and found that none of the F&Os had any significant impact for the application.

A summary of issues identified during the NRC staff's review is provided below along with the associated resolution.

In the April 5, 2018, supplement, SNC stated that the reactor coolant pump (RCP) shutdown seal (SDS) was modeled in the Vogtle PRA models (both IEPRA and SPRA) by adding events and operator actions with corresponding human error probabilities consistent with the Topical Report PWROG-14001-P, Revision 1, and the associated NRC safety evaluation including limitations and conditions in Section 5 (Reference 26). The licensee stated that Limitations and Conditions Nos. 2, 4, and 5 were addressed with explicit modeling in a sensitivity study model or in a revised MOR, as explained below.

- For Limitation and Condition No. 2, the licensee stated that for the Vogtle IEPRA model, a sensitivity study was performed to determine the CDF and LERF impacts of RCP seal loss of coolant accidents (LOCAs) if the rated temperature of the RCP SDS is exceeded. The results of the sensitivity study demonstrated an increase of 13.6 percent in CDF and an increase of 6.2 percent in LERF in comparison to the IEPRA CDF and LERF values submitted in the LAR.

The licensee provided clarification that for the initial analysis performed by Westinghouse for asymmetric cooling, the information used to model the impact of asymmetric cooling in the Vogtle IEPRA sensitivity analysis was not based on a modular accident analysis program (MAAP). For the Vogtle IEPRA model, additional time was assumed beyond the 45 minutes, to account for (1) time for the water in the idle cold leg to heat up after the steam generator dried out and (2) time for the polymer ring in the RCP SDS to heat up and lose its material properties. The licensee stated that given the additional time assumed for the sensitivity analysis, the model was more realistic in the event an operator failed to initiate cooldown within one hour and the RCP SDS would fail. The NRC staff finds that the sensitivity analysis performed is reasonable because the licensee considered additional time assumed beyond the 45 minutes for Limitation and Condition No. 2 for modeling the RCP SDS in the IEPRA to support the ILRT extension.

- To address Limitation and Condition No. 4, the licensee stated the limitation and condition is not applicable because the limitation is specifically for RCP model 93A. The licensee confirmed the Vogtle has RCP model 93A-1. The licensee stated that Model 93A-1 seals directly onto the shaft with no shaft sleeve O-ring to consider.

The NRC staff finds that Limitation and Condition No. 4 does not apply to Model 93A-1.

- For Limitation and Condition No. 5, the licensee stated that the limitation and condition had been addressed by modeling plant-specific operator actions with corresponding human error probabilities in the IEPRA, as described in PWROG-14001-P, Revision 1.

Additionally, the licensee stated in the April 5, 2018, supplement that,

> Incorporation of the RCP shutdown seals into the VEGP [Vogtle] PRA models is PRA maintenance as defined in the ASME/ANS RA-Sa-2009 PRA Standard and qualified by RG 1.200, Revision 2.
>
> The peer-reviewed Vogtle IEIF PRA [IEPRA] and SPRA did not include the Westinghouse Generation III low-leakage (shutdown) seals. However, the peer-reviewed PRA model did include an RCP seal leakage model (WOG 2000 model) to assess the plant response to events that result from a total loss of cooling to the RCP seals. Implementation of the new low-leakage RCP seal model into the IEIF PRA [IEPRA] was performed consistent with the PRA method, modeling, and framework that had already been peer-reviewed.

The ASME/ANS RA-Sa-2009 PRA Standard defines a PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. PRA maintenance is described as the update of the PRA models to reflect plant changes such as modifications, procedure changes, or plant performance (data). The licensee stated that the change in the seal leakage model is not a new methodology, the new seal leakage model is an expansion of the current peer-reviewed model with different failure probabilities and associated human actions. The licensee stated that the equipment, dependencies, and types of accident sequences remain the same, therefore, there is no change in the model scope. In addition, the licensee stated that the peer-reviewed PRA model can still evaluate the risk associated with station blackout and total loss of cooling events related to RCP seal failures, resulting in no change in PRA modeling capabilities. The NRC staff finds the incorporation of the RCP seal leakage model into the IEPRA model has been performed in accordance with PWROG-14001-P, Revision 1, with the NRC limitations and conditions and RG 1.200, Revision 2.

Table 6-2, "Vogtle Units 1 and 2 Internal and External Events Summary," of Attachment 1 of the LAR provides LERF values for Vogtle, Units 1 and 2. The LERF values for the Vogtle, Units 1 and 2, IEPRA were both 6.45E-09 per year. For fire events, the LERF values for Vogtle, Units 1 and 2, were 1.39E-06 per year and 1.56E-06 per year, respectively. The LERF values across the IEPRA and FPRA hazards are approximately two orders of magnitude (i.e., 10^{-2}) in difference, while that for CDF is one order of magnitude. In the April 5, 2018, supplement, the licensee provided clarifying information to address the major contributions unique to the FPRA that are not applicable to the IEPRA and, therefore, result in a higher FPRA LERF value. Some major contributions the licensee provided include: (1) IEPRA LERF does not have contributions from the main control room abandonment scenarios (MCRAB), and (2) containment isolation is not credited in the MCRAB scenarios because there are no directions for containment isolation in the remote shutdown panel (RSP) operation procedures. The licensee confirmed that the MCRAB model only credits systems/components available from the RSP and LERF scenarios related to fire-induced multiple spurious operation scenarios or common cause scenarios that are major contributors for FPRA LERF. The NRC staff finds that such unique contributors for FPRA modeling would contribute to higher LERF values; therefore, the variable differences in LERF values across the IEPRA and FPRA were reasonable due to the unique contributions considered in the FPRA model.

In the Section 3.3.2 of the LAR, the licensee stated that in 2013 a significant upgrade in MAAP capabilities was initiated for the IEPRA. In the April 5, 2018, supplement, the licensee clarified, in part, that:

> The use of the term "upgrade" was improperly used to designate updates/changes to the MAAP version 4.0.5 parameter file. MAAP4.0.8 included many enhancements to the reactor core, reactor coolant system, containment models, engineered safeguards, and other miscellaneous models in MAAP4.

The licensee further confirmed that the enhancements included new parameters and stated the update from MAAP 4.0.5 to MAAP 4.0.8 has a small, but insignificant, impact on the results, such that the success criteria used in the PRA models was not revised due to the MAAP update. The NRC staff finds that the MAAP update to version 4.0.8 from 4.0.5 is consistent with the ASME/ANS RA-Sa-2009 PRA standard definition for PRA maintenance.

In Attachment 1, Table 6-2, of the LAR, a CDF value of 2.52E-06/year and a LERF value of 6.45E-09/year for Vogtle, Units 1 and 2 are provided for the internal events hazard. These CDF and LERF values are identical for each unit. Typically, differences in CDF and LERF results exist for multiple-unit plants, even if the differences are not significant. SNC stated the Vogtle PRA is a single logic model that represents both units. The licensee also stated that the Vogtle units are constructed almost identically and that the PRA models for both units were found to be identical during the Initial Plant Examination (IPE). Plant changes are reviewed during model updates to identify unit differences that may require creating separate unit PRAs. Some of the unit differences identified include spent fuel pool size, room numbers for internal flooding, and room cooling. However, these differing items are not explicitly modeled in the PRA. In addition, the licensee demonstrated that the shared systems for the two-unit site would not have the capability of causing a dual-unit trip. The NRC staff finds that the licensee appropriately considered the design differences between Vogtle, Units 1 and 2, and that plant changes are reviewed to assess model updates to determine if separate unit representation in a PRA model is necessary. Therefore, the NRC staff finds that a single logic model is acceptable for the ILRT risk evaluation.

Based on the review of the LAR and supplement, the NRC staff concludes that the licensee demonstrated that the IEPRA meets the guidance in RG 1.200, Revision 2 and that it is reviewed against the applicable supporting requirements (SRs) in ASME/ANS RA-Sa-2009 PRA Standard. Accordingly, the Vogtle IEPRA is acceptable for use in performing the risk impact assessment for extending the Type A containment ILRT interval.

3.3.2.3 Internal Fire Hazards

The Vogtle FPRA model was peer reviewed in 2012 by the Pressurized Water Reactors Owners Group against ASME/ANS RA-Sa-2009 PRA Standard, RG 1.200, Revision 2, and NEI 07-12, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," November 2008

(Reference 27). Following the Pressurized Water Reactors Owners Group peer review, a focused-scope peer review also was conducted for the qualitative and quantitative screening elements that were dispositioned as not applicable. The focused-scope peer review did not identify any additional F&Os.

Based on the above, the NRC staff concludes that the FPRA has been adequately peer reviewed against the current version of the PRA standard and RG 1.200, and that the licensee has adequately dispositioned the F&Os to support the technical adequacy of the FPRA for the requested ILRT extension.

3.3.2.4 Seismic and Other External Events

External hazards were evaluated against Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities – 10 CFR 50.54(f)" in response to the NRC IPEEE program (Reference 28). The assessment included the risk contribution from internal fire and seismic events. The IPEEE evaluation screened out other external hazards, including high winds, floods, transportation accidents, nearby facility accidents, and other activities associated with hazardous materials.

In Section 3.2.4.2 of the final safety evaluation for NEI 94-01, Revision 2, the NRC staff states that:

> 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.

The NRC staff finds the licensee appropriately considered other external hazards consistent with NEI 94-01, Revision 2.

The NRC staff's review of the risk contribution from seismic events for this application is framed by the context that an order of magnitude estimate is sufficient. As stated in Section 3.3.2 of Enclosure 1 of the LAR, the licensee used a SPRA model to assess the contribution from the seismic risk for this application. The NRC staff's review of the licensee's SPRA used for this application was supported by the review performed for amendment number 196 for Vogtle, Unit 1, and amendment number 179 for Vogtle, Unit 2, regarding the addition of the SPRA to the 10 CFR 50.69 categorization process (Reference 29). The NRC staff reviewed the applicable docketed information from these issued amendments because the same SPRA model and corresponding peer-review were used to support both applications.

Section 3.3.2 of Enclosure 1 of the LAR states that the SPRA has undergone a peer review in 2014 against ASME/ANS RA-Sb-2013 PRA Standard (Addendum B) (Reference 30). Regulatory Guide 1.200, Revision 2, endorses the ASME/ANS RA-Sa-2009 PRA Standard (Addendum A). Tables A-4.1 through A-4.3 of Attachment 1 to Enclosure 1 of the LAR provide a comparison and "basis of assessment" of the differences between each SR of Part 5 of Addendum B and those in Addendum A in the context of the licensee's SPRA. Section 3.2 of the request to incorporate SPRA into the 10 CFR 50.69 categorization process, dated June 22, 2017 (Reference 31) addressed the NRC staff's comments on ASME/ANS RA-Sb-2013 PRA Standard Addendum B in the NRC letter to ASME (Reference 32) in the context of establishing the technical capability of the Vogtle SPRA. In the "basis for assessment" for the difference between Addenda A and B, the licensee stated that its SPRA conformed to accepted current practices for SR SFR-C6. The licensee stated in a letter dated February 21, 2018

(Reference 33), that the soil-structure interaction (SSI) input response spectra were from the site Probabilistic Safety Hazard Analysis (PSHA). The licensee further stated that site-specific dynamic soil profile properties that were developed were strain-compatible with the SSI input response spectra. The licensee also provided additional details about the approach used for the SSI analysis, including the guidance followed and the justification for deviation from the guidance. The licensee stated that confirmatory analyses were performed to validate the accuracy of the SSI analysis of embedded structures.

Based on the review of (1) the licensee's discussion in Section 3.2 of the June 22, 2017, LAR where the licensee addressed the NRC staff's comments on Addendum B in the context of the SPRA; (2) the licensee's comparison of the supporting requirements of Part 5 of Addendum B of the ASME/ANS 2009 PRA Standard to those in Addendum A, and (3) the details of the licensee's approach for performing the SSI analysis for use in its SPRA, the NRC staff finds that the licensee's use of ASME/ANS RA-Sb-2013 PRA Standard Addendum B adequately addresses the technical elements for the development of a SPRA. Therefore, the NRC staff concludes that the licensee's approach is an acceptable alternative to the NRC-endorsed approach for the licensee's SPRA used to support this application.

Table A-3 of Attachment 1 to Enclosure 1 of the LAR provides a list of finding-level F&Os from the peer-review of the SPRA and the corresponding dispositions by the licensee. The NRC staff reviewed the licensee's resolution of all finding-level F&Os provided in Table A-3 and considered the potential impact of the findings on this application and the acceptability of the reported resolution for this application. Based on the review of the licensee's resolutions of the finding-level F&Os from the peer-review of the SPRA in the context of this application, the NRC staff finds that the licensee has adequately resolved the finding-level F&Os for this application.

3.3.2.5 Summary of PRA Acceptability

The licensee evaluated its IEPRA and FPRA against the ASME/ANS RA-Sa-2009 PRA Standard and RG 1.200, Revision 2; addressed or evaluated the impact of the findings developed during the peer reviews for the PRAs for applicability to the ILRT interval extension; and included a quantitative assessment of the contribution for external events. The NRC staff finds the technical acceptability of the SPRA used by the licensee to be sufficient to provide an order of magnitude estimate of the seismic risk contribution for this application. Based on the above, the NRC staff concludes that the PRA (i.e., IEPRA, FPRA, and SPRA) used by the licensee is acceptable to support the evaluation of changes for the requested ILRT extension. Accordingly, the first condition, as discussed in Section 3.3 of this safety evaluation, is met.

3.3.2.6 Estimated Risk Increase

The second condition, as discussed in Section 3.3 of this safety evaluation, stipulates that the licensee submits documentation indicating that the estimated risk increase associated with permanently extending the ILRT interval to 15 years is small and consistent with the guidance in RG 1.174, Revision 3 (Reference 13), and the clarification provided in Section 3.2.4.6 of the final safety evaluation for Revision 2 of NEI 94-01 (Reference 8). 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-roentgen equivalent man (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. Additionally, for plants that rely on containment overpressure 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, Revision 3. The associated risk metrics include LERF, population dose, and CCFP, but not CDF, because SNC does not rely on containment overpressure for ECCS performance, as discussed in Section 3.1.2 of the LAR.

The licensee provided a plant-specific risk impact report in Attachment 1 to the LAR. Additionally, in the April 5, 2018, supplement, the licensee provided clarifying information that confirmed the values reported in the LAR are correct.

In Section 4.2.4 of Attachment 1 of the LAR, for the containment allowable leakage factor, the licensee states that the reference plant has an allowable leakage rate of 0.1 percent/day and Vogtle, Units 1 and 2, has a 0.20 percent/day allowable leakage rate. Therefore, consistent with the EPRI Report 1009325, Vogtle has a factor of 2.0 greater than the reference plant.

The reported risk impacts are based on 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. The following conclusions can be drawn from the licensee's analysis associated with extending the Type A ILRT frequency:

1. The reported increase in LERF values for internal events is 2.32E-08/year for Vogtle, Units 1 and 2. These values, as a result of including the risk impact of corrosion-induced leakage of the steel liners occurring and going undetected during the extended test interval, are 2.10E-12/year for Vogtle, Units 1 and 2. The total LERF for combined internal and external events is 1.73E-06/year and 1.90E-06/year for Vogtle, Units 1 and 2, respectively. The risk contribution from external events includes the effects of internal fires and seismic events, as discussed in Sections 3.3.2.3 and 3.3.2.4.1 of this safety evaluation. These changes in risk are considered to be "very small" per the acceptance guidelines in RG 1.174, Revision 3.

2. The reported increase in the total population dose rate (PDR) is 1.79E-03 person-rem/year for Vogtle, Units 1 and 2, which included the increases in Classes 3a and 3b. The reported increase in PDR is below the acceptance guidelines in Section 3.2.4.6 of the final safety evaluation for Revision 2 of NEI 94-01. 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 Type A ILRT frequency from three in 10 years to once in 15 years is approximately 0.92 percent for Vogtle, Units 1 and 2. These values are below the acceptance guidelines in Section 3.2.4.6 of the final safety evaluation for Revision 2 of NEI 94-01.

Based on the above, the NRC staff concludes that the estimated risk increase associated with permanently extending the Type A ILRT interval to once in 15 years is small, and is consistent with the guidance in RG 1.174, Revision 3, and the clarification provided in Section 3.2.4.6 of the final safety evaluation for Revision 2 of NEI 94-01. The defense-in-depth philosophy is maintained, as the independence of barriers will not be degraded as a result of the requested change, and the use of the three 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, as discussed in Section 3.3 of this safety evaluation, and Key "Principle 4" in RG 1.174, Revision 3, are met.

3.4 Leak Rate for the Large Preexisting Containment Leak Rate Case

The third condition, as discussed in Section 3.3 of this safety evaluation, stipulates that in order to make the methodology in EPRI TR-1009325, Revision 2-A, acceptable, the average leak rate for the preexisting 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 in Table 3.3.1-1 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 preexisting containment large leakage rate accident case, and this value has been used in the Vogtle, Units 1 and 2, specific risk assessment. Accordingly, the third condition is met.

3.5 Applicability if Containment Overpressure is Credited for ECCS Performance

The fourth condition, as discussed in Section 3.3 of this safety evaluation, stipulates that in instances where containment overpressure is relied upon for ECCS performance, a LAR shall be submitted. The licensee stated that containment overpressure is not relied upon for ECCS performance for Vogtle; therefore, the fourth condition is met.

3.6 Technical Conclusion

Based on the above, the NRC staff concludes that the licensee has adequately implemented its primary containment leakage rate testing program consisting of the ILRT and LLRT. The results of the recent ILRTs and the LLRT (Type B and Type C tests) combined totals demonstrate acceptable performance and support a conclusion that the structural and leak-tight integrity of the primary containment vessel is adequately managed and will continue to be periodically monitored and managed effectively. The NRC staff concludes that the licensee has addressed the NRC conditions to demonstrate acceptability of adopting TR NEI 94-01, Revision 3-A, and the limitations and conditions identified in the NRC staff's safety evaluation that was incorporated into NEI 94-01, Revision 2-A. The NRC staff concludes that the risk impact for extending the integrated leak rate testing intervals is consistent with the acceptance guidelines of RG 1.174, Revision 3, and that the proposed changes continue to meet the requirements in 10 CFR 50.36(c)(5). Therefore, the NRC staff concludes that the proposed changes to TS 5.5.17 regarding the primary containment leakage rate testing program are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments on September 19, 2018. The NRC staff verified that the State official had no comments on September 20, 2018.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility 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 has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on that finding, which was published in the *Federal Register* on December 5, 2017 (82 FR 57474). 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.

7.0 REFERENCES

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Date: October 29, 2018

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS TO EXTEND THE CONTAINMENT TYPE A LEAK RATE TEST FREQUENCY TO 15 YEARS AND TYPE C LEAK RATE TEST FREQUENCY TO 75 MONTHS (CAC NOS. MG0240 AND MG0241; EPID L-2017-LLA-0295) DATED OCTOBER 29, 2018

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