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

June 28, 2018

Mr. George A. Lippard, III
Vice President, Nuclear Operations
South Carolina Electric & Gas Company
Virgil C. Summer Nuclear Station
P.O. Box 88, Mail Code 800
Jenkinsville, SC 29065

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT RE: INTEGRATED LEAK RATE TEST PEAK CALCULATED CONTAINMENT INTERNAL PRESSURE CHANGE (EPID L-2017-LLA-0348)

Dear Mr. Lippard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 210 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1, in response to your application dated October 6, 2017, as supplemented by letter dated April 19, 2018.

This amendment increases the Integrated Leak Rate Test Peak Calculated Containment Internal Pressure, Pa, listed in Technical Specification 6.8.4.g, "Containment Leakage Rate Testing Program," from 45.1 pounds per square inch gauge (psig) to 46.0 psig. It also removes the references to Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," and American National Standards Institute/American Nuclear Society (ANSI/ANS)-56.8-2002, "Containment System Leakage Testing Requirements," and replaces the reference of Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," July 2012, with NEI 94-01, Revision 2-A, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008.

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

Sincerely,

Shawn Williams

Shawn A. Williams, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-395

Enclosures:
1. Amendment No. 210 to NPF-12
2. Safety Evaluation

cc: Listserv

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

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 210
Renewed License No. NPF-12

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

   A. The application for amendment to the Virgil C. Summer Nuclear Station, Unit No. 1 (the facility), Renewed Facility Operating License No. NPF-12 filed by the South Carolina Electric & Gas Company (the licensee), dated October 6, 2017, as supplemented by letter dated April 19, 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 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 as 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-12 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 210, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This 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 Renewed Facility Operating License and Technical Specifications

Date of Issuance: June 28, 2018

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

ATTACHMENT TO LICENSE AMENDMENT NO. 210

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Renewed Facility Operating License and 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	Insert
License Page 3	License Page 3
TS 6-12b	TS 6-12b

(3) SCE&G, 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 amended through Amendment No. 33;

(4) SCE&G, 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 neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

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

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

(1) Maximum Power Level

SCE&G is authorized to operate the facility at reactor core power levels not in excess of 2900 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this renewed license. The preoperational tests, startup tests and other items identified in Attachment 1 to this renewed license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 210, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measures of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM;

2) A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of the census; and

3) Participation in an Inter-laboratory Comparison Program to ensure that independent checks on the precision and accuracy of measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

g. Containment Leakage Rate Testing Program

A program shall be established to implement leakage rate testing of the containment system as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with NEI 94-01, Revision 2-A, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," October 2008.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 46.0 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.20 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. 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;

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 210 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By letter dated October 6, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17279A715), as supplemented by letter dated April 19, 2018 (ADAMS Accession No. ML18109A317), South Carolina Electric & Gas Company (SCE&G, the licensee) submitted a license amendment request to modify the Virgil C. Summer Nuclear Station (VCSNS), Unit No. 1, Technical Specifications (TSs).

The licensee requested to increase the Integrated Leak Rate Test Peak Calculated Containment Internal Pressure, Pa, listed in Technical Specification (TS) 6.8.4.g, "Containment Leakage Rate Testing Program," from 45.1 pounds per square inch gauge (psig) to 46.0 psig; to remove the references to Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," and American National Standards Institute/American Nuclear Society (ANSI/ANS)-56.8-2002, "Containment System Leakage Testing Requirements"; and to replace the reference of Nuclear Energy Institute (NEI) 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J" (ADAMS Accession No. ML12221A202), with NEI 94-01, Revision 2-A (ADAMS Accession No. ML100620847).

The supplemental letter dated April 19, 2018, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 21, 2017 (82 FR 55409).

2.0 REGULATORY EVALUATION

2.1 Background

The regulation under Section 50.54(o) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that the primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J includes two options, Option A, "Prescriptive Requirements," and Option B, "Performance-Based Requirements." A licensee can choose either option for meeting the requirements of Appendix J.

The testing requirements in Appendix J ensure that leakage through the primary reactor containment and related systems and components penetrating primary containment do not exceed allowable leakage rate values specified in the TSs or associated bases; and periodic surveillance of reactor containment penetrations and isolations valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. The licensee has adopted and implemented Option B for meeting the requirements of Appendix J. Option B identifies the performance-based requirements and criteria for preoperational and subsequent periodic leakage-rate testing. These requirements are met by performance of Type A, Type B, and Type C tests.

Type A tests (also referred to as the integrated leak rate test) are tests intended to measure the primary reactor containment overall integrated leakage rate after the containment has been completed and is ready for operations and at periodic intervals thereafter.

2.2 Description of Changes

The licensee proposed to revise TS 6.8.4.g, "Containment Leakage Rate Testing Program," as follows:

- Remove the reference to RG 1.163, "Performance- Based Containment Leak-Test Program," September 1995. The licensee stated that this change is administrative in nature because the regulatory positions stated in RG 1.163 are incorporated into NEI 94-01, Revision 2-A.

- Remove the reference to ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements." The licensee stated that this change is administrative in nature because NEI 94-01, Revision 2-A, incorporates ANSI/ANS-56.8-2002 by reference for testing methodology guidance.

- Replace the reference to NEI 94-01, Revision 3-A, with NEI 94-01, Revision 2-A. The licensee stated this change is administrative in nature because the conditions and limitations required for NEI 94-01, Revision 2-A, were previously submitted and approved by the NRC under Amendment No. 194 issued February 5, 2014 (ADAMS Accession No. ML13326A204).

- Revise the current Pa value of 45.1 psig with 46.0 psig (60.7 pounds per square inch absolute). The licensee stated that this proposed change reflects updated calculations to the loss-of-coolant accident (LOCA) mass and energy (M&E) releases that addressed modeling and material property errors that were raised in various Westinghouse Nuclear Safety Advisory Letters (NSALs).

2.3 Applicable Regulatory Requirements and Guidance

The NRC staff based its review of the proposed changes on the following requirements in 10 CFR Part 50:

- Appendix A, General Design Criterion (GDC) 16, Containment design, states that "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

- Appendix A, GDC 38, Containment heat removal, states that, "A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels."

- Appendix A, GDC 50, Containment design basis, states, in part, that, "The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident.

- Appendix J, Option B, Performance-Based Requirements, Section V.B.3, states, in part, that "The regulatory guide or other implementation document used by a licensee ... to develop a performance-based leakage-testing program must be included, by general reference, in the plant technical specifications. The submittal for technical specification revisions must contain justification, including supporting analyses, if the licensee chooses to deviate from methods approved by the Commission and endorsed in a regulatory guide."

3.0 TECHNICAL EVALUATION

The VCSNS Unit No. 1 is a Westinghouse three-loop pressurized-water reactor. Containment is provided by the reactor building (RB). The RB is a steel-lined, post tensioned, reinforced concrete structure that provides a barrier against the escape of fission products should a LOCA occur. Consistent with the Updated Final Safety Evaluation Report (UFSAR) (ADAMS Package Accession No. ML17215A008) and the license amendment request, hereinafter in this safety evaluation, the abbreviation RB is alternatively used for the term "containment".

The VCSNS LOCA M&E release analysis uses an NRC-approved methodology in Westinghouse WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version," May 1983 (proprietary). In 2006, 2011, and 2014, Westinghouse issued NSALs 06-6, 11-5, and 14-2[1] reporting errors that affects the M&E release and consequently the LOCA containment pressure, vapor temperature, and sump temperature response analyses. The licensee proposed to correct the LOCA M&E release analysis from the errors reported in the above NSALs and the associated containment response analyses. Since the calculated peak RB internal pressure related to the design basis LOCA TS value of Pa is affected, the licensee proposed a new value of Pa.

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[1] Westinghouse Electric Company, LLC, Nuclear Safety Advisory Letter (NSAL) 06-6, "LOCA Mass and Energy Release Analysis," dated June 6, 2006; NSAL 11-5, "LOCA Mass and Energy Release Analysis," dated July 25, 2011; and NSAL 14-2, "Westinghouse Loss-of-Coolant Accident Mass and Energy Release Calculation Issue for Steam Generator Tube Material Properties," dated March 31, 2014.

3.1 Revise LOCA Containment Peak Pressure 'Pa' in TS Section 6.8.4.g

The proposed M&E release analysis followed by the revised RB response analyses resulted in a Pa change in TS Section 6.8.4.g Pa from 45.1 psig to 46.0 psig. The NRC staff's evaluation of these analyses is provided below.

3.1.1 LOCA M&E Release Analysis

The licensee used the current methodology based on the NRC approved WCAP-10325-P-A for LOCA M&E release analysis. The M&E data was generated using inputs that addressed the NSALs 06-6, 11-5, and 14-2.

Consistent with the analysis of record (AOR) reported in the UFSAR, the licensee re-analyzed the following three double-ended guillotine breaks in the reactor coolant system (RCS) that are limiting because of the large mass flowrates from these breaks during the LOCA blowdown phase:

(a) hot leg between reactor vessel and steam generator (SG), also called double-ended hot leg (DEHL) break;

(b) cold-leg discharge between reactor coolant pump (RCP) and reactor vessel; and

(c) pump suction between the SG and the RCP, also called double-ended pump suction (DEPS) break.

Among these break locations, the licensee determined that the DEHL break, (a), and the DEPS break, (c), generated the most limiting short-term and long-term pressure and temperature transients respectively. The licensee calculated the M&E releases for the blowdown, reflood, and the post-reflood phases of the DEPS break for the maximum and minimum safety injection sensitivity cases.

For the DEHL break case, the licensee calculated the M&E releases for the blowdown phase only. In a request for additional information (RAI) dated March 8, 2018 (ADAMS Accession No. ML18066A000), the NRC staff requested that the licensee provide justification for not calculation the M&E releases and the containment response for the remaining three phases of the DEHL break loss-of coolant accident, and , and explain quantitatively why the DEPS cases bound the plant analysis or record for the long term containment and sump temperature responses. By letter dated April 19, 2018, the licensee stated that the generic sensitivity study documented in WCAP-10325-P-A, Section 3.3, provides sensitivity of M&E release to the break location. The licensee stated that among the three break locations identified above, the DEHL break blowdown phase released the highest M&E and, therefore, generated the highest peak containment pressure. However, for this break when the post-blowdown emergency core cooling system (ECCS) flow is established, initially two-phase break flow exits from the vessel side and becomes a saturated liquid flow in the long term. Although the ECCS flooding rate is significant, the M&E released from the break is minimal because the large flow resistance of the affected SG loop's RCP plus the SG tube flow resistance limits the flow through the SG side of the break and therefore significantly limits the post-blowdown M&E releases. Therefore, the licensee concluded that consistent with the generic evaluation in WCAP-10325-P-A, Section 3.3, the VCSNS plant-specific analysis is not necessary for the DEHL break post-blowdown phase because of limited M&E contribution from the SG side of the break of the affected SG.

For the DEPS break, due to the location of the break, the flow would have lesser enthalpy relative to the DEHL in the blowdown phase and therefore its peak RB pressure would be bounded by same from the DEHL break. However, after ECCS initiation during the reflood phase, the DEPS break generates additional RB pressure peaks because flow continues from the vessel side as well as the SG side releasing decay heat and SG metal heat from the break. Therefore, the licensee concluded, that in the VCSNS plant-specific analysis, the subsequent DEPS subsequent peaks are bounded by the DEPS blowdown peak.

NRC Staff Evaluation

The NRC staff finds the licensee's justification for not analyzing the M&E releases and the RB response for the DEHL post-blowdown phase acceptable because the licensee's rationale is consistent with the generic evaluation provided in the NRC-approved WCAP-10325-P-A. The NRC staff also agrees that the RB peak pressure generated during the DEHL blowdown phase bounds the peak pressures for the DEPS break because the DEHL blowdown M&E release bounds the DEPS M&E releases during the blowdown and post-blowdown phases.

In an RAI dated March 8, 2018, the NRC staff referred to Section 3.3 of the October 6, 2017, enclosure, which states, in part, "... modeling corrections were applied that slightly altered certain initial conditions in FSAR Table 6.2-2..." The NRC staff requested the licensee to describe the corrections in modeling the M&E release analysis that resulted in a change in the initial conditions.

By letter dated April 19, 2018, the licensee's response provided (1) a list of the specific NSALs 06-6, 11-5, and 14-2 items applicable to VCSNS M&E analysis that affected the initial conditions for M&E analysis, and (2) the M&E modeling corrections that overlapped the NSAL-related changes but did not prompt a change to TS 6.8.4.g. The licensee stated that the M&E analysis modelling corrections are based on the following:

(a) an increase in uncertainty in the current initial RCS average temperature 587.4 degrees Fahrenheit (°F), conservatively increased from 5.3 °F to 6.3 °F, resulting in a higher initial RCS average temperature of 593.7 °F, which conservatively increases the initial RCS energy;

(b) a conservative increase in the piping volume by 44 square foot average for each accumulator from the tank to the cold-leg injection point in the RCS model. The licensee stated that this also conservatively increases the RCS mass by 280 pounds, or by 0.066 percent; and

(c) the modification of reactor vessel internals performed for minimizing or eliminating the baffle flow jetting and its potential impact on fuel integrity. The modification reduces the pressure differentials across the baffle joints by altering the reactor vessel lower internals such that the coolant's downflow path in the baffle-barrel region is converted into an upflow path.

The NRC staff finds the M&E release analysis acceptable because the licensee applied the currently used NRC-approved WCAP-10325-P-A methodology while addressing VCSNS-related specific errors reported in the NSALs and the M&E updated model overlapping corrections.

3.1.2 LOCA RB Response Analyses

For the LOCA RB response analyses, the licensee used the currently used CONTEMPT-LT/28 methodology stated in UFSAR Section 6.2.1.3.3.1. The analyses is performed for the DEHL break LOCA case, and DEPS break LOCA cases with maximum and minimum safety injection. The DEHL break results in the higher peak pressure that occurs in the short term, while the DEPS minimum safety injection case results in the limiting long-term conditions. The resulting maximum peak pressure of 45.5 psig occurs for the DEHL case during the blowdown phase, which is greater than the TS 6.8.4.g value of 45.1 psig for Pa.

In an RAI dated March 8, 2018, the licensee was requested to confirm that the inputs and assumptions for the RB pressure, vapor temperature, and sump temperature response analyses are the same as in the AOR and to justify any differences that reduced conservatism. By letter dated April 19, 2018, the licensee stated that besides the changes in the M&E analysis described above, the remaining key inputs and assumptions are the same as in the AOR. These key items are: (a) containment free volume and heat sink parameters, (b) residual heat removal heat exchanger parameters, (c) reactor building fan cooler start time, (d) reactor building spray start time, and (e) reactor building spray design flow.

NRC Staff Evaluation

The NRC staff finds the licensee's response acceptable because the same inputs and assumptions for the key items as in the AOR are used in the revised RB response analyses without reducing conservatism.

In the October 6, 2017, application, for comparing the revised analysis with the AOR results, the licensee developed composite graphs to create maximum envelopes for RB pressure (Figure 1), RB vapor temperature (Figure 2), and RB sump temperature (Figure 3) for the DEHL break, DEPS break cases, and showed the AOR graphs [Figures 6.2-1 (DEPS, Minimum safety injection), 6.2-2 (DEPS Maximum safety injection), and 6.2-3 (DEHL)] of the UFSAR superposed on the composite graphs.

A review of Figure 1 shows the maximum peak pressure of 45.5 psig for the short-term DEHL case which is greater than AOR DEHL peak pressure of 45.1 psig same as the current TS 6.8.4.g value for Pa. The licensee proposed a bounding Pa of 46.0 psig in the revised TS.

Regarding the RB vapor temperature and sump temperature, a review of Figures 2 and 3 shows the revised graphs follow closely with AOR graphs with insignificant differences that would not affect the equipment environmental qualification profile, and the net positive suction head analysis for the pumps that draw water from the RB sump during the LOCA recirculation phase.

The NRC staff finds the revised RB pressure, vapor temperature, and sump temperature response analyses acceptable because the licensee based it on the current CONTEMPT-LT/28 methodology using conservative inputs and assumptions. The change of Pa from 45.1 psig to 46.0 psig in TS Section 6.8.4.g is, therefore, acceptable. The Pa value modestly above the actual peak calculated maximum accident pressure is acceptable as it is intended to represent the minimum test pressure for Appendix J testing and specifying a greater pressure value is expected to yield conservative test results for determining containment leakage potential.

3.2 Remove Reference to RG 1.163 and ANSI/ANS-56.8-2002 from TS 6.8.4.g

The licensee proposed to remove reference to RG 1.163, "Performance- Based Containment Leak Test Program", dated September 1995 and ANSI/ANS-56.8-2002, "Containment System Leakage Testing Requirements" from the TS Section 6.8.4.g.

The NRC staff finds removing the reference to RG 1.163 from TS 6.8.4.g acceptable because the regulatory positions given in this regulatory guide have been incorporated into the NRC-approved topical reports NEI 94-01, Revisions 2-A and 3-A. These topical reports describe an acceptable approach for implementing the performance based requirements of Option B to 10 CFR Part 50, Appendix J. They include provisions for extending Type A integrated leak rate test intervals to up to 15 years.

The NRC staff finds removing the reference to ANSI/ANS-56.8-2002 from TS 6.8.4.g acceptable because this standard is already referred in NEI 94-01, Revision 2-A and 3-A for Type A testing.

3.3 Replace Reference of NEI 94-01, Revision 3-A, with NEI 94-01, Revision 2-A, in TS 6.8.4.g

The licensee proposes to replace reference of NEI 94-01, Revision 3-A, with Revision 2-A, in TS 6.8.4.g.

The NRC staff finds it acceptable and accurate to change the reference to NEI 94-01, Revision 2-A, because the conditions and limitations required for NEI 94-01, Revision 2-A, were submitted previously and approved by the NRC under Amendment No. 194.

Regarding the 10 CFR Part 50 Appendix J Local Leakage Rate Tests, also termed as either Type B or Type C tests, the licensee will continue to follow the guidance in NEI 94-01, Revision 2-A. The difference between NEI 94-01, Revision 2-A and Revision 3-A regarding the Type C test intervals is that NEI 94-01, Revision 2-A, limits the Type C test to 60 months rather than the 75 months allowed by NEI 94-01, Revision 3-A. The licensee has not requested to increase the Type C test to 75 months.

3.4 NRC Staff Conclusion

The licensee used the NRC-approved methodology for LOCA M&E release after correction of errors reported in NSALs 06-6, 11-5, and 14-2 and the RB response analyses for determining the revised RB pressure and temperature response and 'Pa'.

The NRC staff concludes that the proposed change meets the requirements of 10 CFR Part 50: (1) Appendix A, GDC 16, because the licensee showed that the containment design conditions important to safety are not exceeded during a postulated LOCA; (2) Appendix A, GDC 38, because the licensee showed that the containment heat removal system would reduce the containment pressure and temperature rapidly with the other associated systems, following design-basis accident and would maintain them at acceptable levels; (3) Appendix A, GDC 50, because the licensee showed that the containment heat removal system is designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from design-basis accident; and (4) Appendix J, Option B, because the licensee referenced the NRC-approved NEI guidance documents with limitations and conditions for Type A and C tests. Therefore, the NRC staff finds the proposed TS changes acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of South Carolina official was notified of the proposed issuance of the amendment on June 1, 2018. On June 1, 2018, the State official confirmed that the State of South Carolina had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, published in the *Federal Register* on November 21, 2017 (82 FR 55409), and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: Ahsan Sallman, NRR
Jerome Bettle, NRR

Date: June 28, 2018

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 – ISSUANCE OF AMENDMENT RE: INTEGRATED LEAK RATE TEST PEAK CALCULATED CONTAINMENT INTERNAL PRESSURE CHANGE (EPID L-2017-LLA-0348) DATED JUNE 28, 2018

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OFFICE	OGC/NLO	DORL/LPL2-1/BC	DORL/LPL2-1/PM	
NAME	AGosh	MMarkley	SWilliams	
DATE	6/8/18	6/28/18	6/28/18	

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