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

September 25, 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 EXIGENT AMENDMENT RE: ONE-TIME EXTENSION TO THE SURVEILLANCE FREQUENCY 4.3.3.6 OF THE CORE EXIT TEMPERATURE INSTRUMENTATION (EPID NO. L-2018-LLA-0226)

Dear Mr. Lippard:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 211 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated August 24, 2018, as supplemented by letters dated August 31, September 11, and September 19, 2018.

The amendment authorizes a one-time extension to Surveillance Requirement 4.3.3.6 of the frequency of the Core Exit Temperature Instrumentation channel calibrations from "every refueling outage" to "every 19 months."

This amendment is being issued under exigent circumstances in accordance with paragraph 50.91(a)(6) of Title 10 of the *Code of Federal Regulations*. The exigent circumstances and final no significant hazards considerations are addressed in Sections 4.0 and 5.0 of the enclosed Safety Evaluation.

The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

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

Docket No. 50-385

Enclosures:
1. Amendment No. 211 to NPF-12
2. Safety Evaluation

cc: via 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. 211
Renewed License No. NPF-12

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

   A. The application for amendment by the South Carolina Electric & Gas Company (the licensee), dated August 24, 2018, as supplemented by letters dated August 31, September 11, and September 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;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-12 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 211, 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. The license amendment is effective as of its date of issuance and shall be implemented upon approval.

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 No. NPF-12
and the Technical Specifications

Date of Issuance: September 25, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 211

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

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

Remove Pages	Insert Pages
License Page 3	License Page 3
TSs 3/4 3-56	TSs 3/4 3-56

(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 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 Part 30, 40 and 70 to receive, possess and use at any time 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 of 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 m[a]y 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) <u>Maximum Power Level</u>

SCE&G is authorized to perathe 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 preoccupation 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 renewed license.

(2) <u>Technical Specifications and Environmental Protection Plant</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 211, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. South Carolina Electric & Gas Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown on Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 30 days or submit a Special Report within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to operable status.

b.1 With the number of OPERABLE Reactor Building radiation monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

i) Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

ii) Submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status.

b.2 Deleted

b.3 With the number of OPERABLE accident monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performing a monthly CHANNEL CHECK and a CHANNEL CALIBRATION every refueling outage. An extension for the CHANNEL CALIBRATION interval for Item 12 in Table 3.3-10 to 19 months is permitted on a one-time basis. This extension expires prior to entering MODE 3, following the RF-24 refueling outage. The Reactor Building Radiation Level Instrumentation CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for the range decades above 10R/hr and a single point calibration of the detector below 10R/hr with an installed or portable gamma source.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 211 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-12

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-395

1.0 INTRODUCTION

By application dated August 24, 2018, as supplemented by letters dated August 31, September 11, and September 19, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML18236A383, ML18243A392, ML18254A406, and ML18262A088, respectively), South Carolina Electric & Gas Company (SCE&G, the licensee) submitted a license amendment request (LAR) that proposes to revise the Virgil C. Summer Nuclear Station (Summer), Unit No. 1, Technical Specifications (TS) for Surveillance Requirement (SR) 4.3.3.6 to allow a one-time extension of the frequency of the Core Exit Temperature Instrumentation channel calibrations from "every refueling outage" to "every 19 months."

The supplemental letters dated August 31, September 11, and September 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) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 10, 2018 (83 FR 45688).

2.0 REGULATORY EVALUATION

2.1 System Description

The licensee provided the following description in Section 2.1 of their August 24, 2018 letter:

> The Incore Temperature Monitoring System is designed to provide rapid monitoring of fuel assembly outlet temperatures and to verify that the core is being adequately cooled (subcooling and natural circulation) during and after an accident.
>
> As a secondary function, the Incore Temperature Monitoring System, is designed to provide an accurate measure of the relative, integrated fuel assembly power

distribution. This function requires the use of the reactor core flux maps from the Incore Neutron Flux Monitoring System to normalize the temperature measurements.

The thermocouple temperature data along with reactor coolant system pressure, loop flows, loop inlet temperatures, average of the loop temperatures, loop differential temperatures, and power level is used to determine:

(1) Saturation margin
(2) Core relative fuel assembly power distribution
(3) Core enthalpy rise nuclear hot-channel factors
(4) Core radial tilting factors

The Incore Temperature Monitoring System provides isolated analog input signals for sixteen (16) thermocouples (two per core quadrant per thermocouple train) to the Core Cooling Monitoring System. The signals from the Train A and Train B isolator cabinets are routed, maintaining electrical separation and utilizing qualified cable and seismically qualified cable trays and conduit, to the respective Core Cooling Monitors "A" and "B".

The Core Cooling Monitoring System is designed to provide information to plant personnel concerning the status of reactor core heat removal capability. This information includes a continuous display of the saturation margin to provide an early warning that core conditions are approaching saturation. In addition, the core outlet temperatures at selected fuel assemblies are displayed to assist in the diagnosis of inadequate core cooling. The Core Cooling Monitoring System was designed in response to the post Three Mile Island requirements documented by the NRC in NUREG-0578, Item 2.1.3b.

The core exit thermocouples measure reactor coolant temperatures which is used to identify inadequate core cooling during normal operation and during post-accident monitoring. Furthermore, the Power Distribution Monitoring System also uses temperature data from the core exit thermocouples. Finally, as part of the plant startup tests, the resistance temperature detector signals are compared with the core exit thermocouple signals.

The core cooling instrumentation includes the Incore Temperature Monitoring System, the core subcooling monitors, and the Reactor Vessel Level Instrumentation System (RVLIS). These systems meet the requirements of NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," item II.F.2 for inadequate core cooling instrumentation (ADAMS Accession No. ML051400209). They are also used to provide Post Accident Monitoring Information in accordance with Regulatory Guide 1.97, Rev. 4, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," (ADAMS Accession No. ML061580448).

Updated Final Safety Analysis Report (FSAR), Section 7.5.5, "INADEQUATE CORE COOLING," states that, "The Incore Temperature Monitoring System consists of 51 thermocouples positioned in the reactor vessel above the core to measure reactor coolant temperature at the fuel assembly outlets." Plant computer system displays and safety parameter display system (SPDS) display, provide indication of core exit thermocouple readings in the Control Room.

The Incore Thermocouples (i.e., Table 3.3-10 Item No. 12) are designed to provide rapid monitoring of fuel assembly outlet temperatures and to verify that the core is being adequately cooled (subcooling and natural circulation) during and after an accident.

The core subcooling monitoring system is designed to provide information to plant personnel concerning the status of reactor core heat removal capability. This information includes a continuous display of the saturation margin to provide an early warning that core conditions are approaching saturation. Two separate core subcooling monitoring system microprocessors calculate the reactor coolant system (RCS) saturation margin based on independent wide range RCS pressure input and RCS temperature inputs and display the results on four main control board analog indicators (two per channel). Temperature inputs are from both hot and cold leg wide range resistance temperature devices (RTDs) and Incore Temperature Monitoring System thermocouples (two per core quadrant). Only the two indicators utilizing incore thermocouple inputs are used for Post-Accident Regulatory Guide 1.97 monitoring functions.

2.2 Licensee's Proposed Changes

The licensee is requesting that a one-time extension, which will expire at the end of Refueling Outage 24, be added for Item 12 of Table 3.3-10 to TS SR 4.3.3.6. This will revise the surveillance requirement of the Core Exit Temperature Instrumentation by extending the CHANNEL CALIBRATION to 19 Months (from the previously interpreted every outage of 18 months) for one surveillance interval. The revised surveillance requirement would state, in part, that:

> Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performing a monthly CHANNEL CHECK and a CHANNEL CALIBRATION every refueling outage. An extension for the CHANNEL CALIBRATION interval for Item 12 in Table 3.3-10 to 19 months is permitted on a one-time basis. This extension expires prior to entering MODE 3, following the RF-24 refueling outage.

2.3 Applicable Regulations and Guidance

Title 10 of the *Code of Federal Regulations* (CFR), Section 50.36(c)(3) states, "Surveillance Requirements," of 10 CFR Part 50 requires: "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

Chapter 3 of the updated FSAR describes the extent to which structures, systems, and components important to safety meet Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR, Part 50, "Domestic Licensing of Production and Utilization Facilities."

- Criterion 13, "Instrumentation and control," requires, in part, that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions.

Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

- Criterion 19, "Control room," requires, in part, that a Control Room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents.

Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident", provides guidance for selection of readouts to monitor plant variables and systems during and following a design basis event.

3.0 TECHNICAL EVALUATION

Information from the core exit thermocouples are used: (1) to inform the operators of core exit temperatures, and (2) as a surrogate for core power which is used by the core modeling software. There is a different form of calibration associated with each of these uses. This LAR is requesting an extension to the surveillance interval for calibration of the core exit thermocouples, but not for the power monitoring application (because this operation will remain within the current calibration interval for this application).

Current TS Table 3.3-10 states that the CHANNEL CHECK AND CHANNEL CALIBRATION should be performed every 18 months. To justify the one-time extension to 19 months, the licensee provided: (1) a description of the ways that information from the core exit thermocouples is used, (2) a description of how instrument uncertainties are calculated, (3) the calculated total amount of drift expected at the end of the extended calibration interval, and (4) an explanation of why the additional drift has an undiscernible safety impact. By email on September 5, 2018 (ADAMS Accession No. ML18249A193), the NRC staff sent requests for additional information (RAI), to which the licensee responded by letter dated September 11, 2018 (ADAMS Accession No. ML18254A406).

In the response to RAI no. 1 a, the licensee described its contractor's analyses using the Square Root of the Sum of the Square Roots (SRSS) uncertainty calculation methodology for the core exit thermocouples. This is an acceptable methodology employed for the calculation of setpoints that actuate automatic protective action of the reactor trip system and the engineered safety features actuation system. Although there are no automatic protective actions associated with the Core Exit Thermocouples, the NRC staff considers the calculation methodology described to be adequate for the core exit thermocouple instrument channels.

The magnitude of instrument drift is one of the uncertainties, when combined with all other instrument channel uncertainties, used to determine the required allowance for overall instrument loop accuracy to support the channel's designated design functions (i.e., the instrument channel must be sufficiently accurate to meet the requirements of each of the functions and applications the channel serves). During calibration activities, the performance of the instrument channel is measured under calibration conditions to determine if the measured output of the channel has deviated beyond its expected performance. This information is used to determine whether it is sufficiently accurate to support performance requirements.

In response to RAI no. 1 c, the licensee described how the possible additional instrument channel drift due to the extended calibration interval will not have a significant impact on any of the design functions associated with the core exit thermocouples. The additional drift is more than one order of magnitude smaller than half of the minor division on the displays the operator will use, and over two orders of magnitude below the normal operating margins.

Additionally, in response to RAI no. 4, the licensee stated that there is no impact on core monitoring parameters such as core relative fuel assembly power distribution, core enthalpy rise nuclear hot-channel factors, or core radial tilting factors. These parameters rely on the Power Distribution Monitoring System (PDMS) calculation. The methodology for the PDMS requires consistent temperature indication, however, not absolute temperature accuracy to correlate the indicated temperatures from the core exit thermocouples with the measured power in those core locations from an incore flux map. The uncertainty of core peaking factors calculated from the PDMS is based on the time since flux map calibration and uncertainty increases with time since calibration. However, the PDMS has a required 180 day calibration frequency per TS 4.3.3.11.2 and remains within the calibration interval until the refueling outage begins in October 2018.

The NRC staff reviewed the as-found and as-left surveillance data from the last two surveillances performed by the licensee. This data indicates that during the last calibrations, the majority of the as-found values of the temperature monitoring channels were within the procedural as-found allowances, and the as-left values following completion of the calibration did not encroach on the allowances of the calibration procedures. Therefore, the NRC staff has concluded that there is reasonable assurance that public health and safety will be maintained during the one-time surveillance extension.

3.1 Technical Evaluation Summary

The licensee provided adequate supporting information for the NRC staff to conclude that the extension of a month to the surveillance interval will result in an insignificant increase in drift of the instrument channels with no impact on their design functions. The NRC staff has also concluded that there is reasonable assurance that the core exit thermal instrumentation will remain operable during the extended surveillance interval period.

Based on the above and past performance, the affected Incore Temperature monitors affected by this one-time surveillance interval extension continue to provide reasonable assurance that the core exit thermocouples are sufficiently accurate to perform their required design functions. The NRC staff finds that this change meets the regulatory requirements described in 10 CFR 50.36(c)(3) and GDCs 13 and 19. The NRC staff, therefore, concludes that the proposed change TS surveillance 4.3.3.6 related to the calibration frequency of the core exit thermocouples is acceptable.

4.0 EXIGENT CIRCUMSTANCES

The NRC's regulations contain provisions for issuance of amendments when the usual 30-day public comment period cannot be met. These provisions are applicable under exigent circumstances. Consistent with the requirements in 10 CFR 50.91(a)(6), exigent circumstances exist when: (1) a licensee and the NRC must act quickly; (2) time does not permit the NRC to publish a *Federal Register* notice allowing 30 days for prior public comment; and (3) the NRC determines that the amendment involves no significant hazards considerations. As discussed in the licensee's application dated August 31, 2018, the licensee requested that the proposed amendment be processed by the NRC on an exigent basis.

The licensee stated that the need for the amendment request to be processed on an exigent basis is because the licensee was unable to complete SR 4.3.3.6 within the required 18 month (every refueling outage) schedule. SR 4.3.3.6 was started on June 19, 2018, but was halted due to inadequate fire watch resources. However, before the test could be rescheduled, unidentified leakage (in the movable incore detection system) was noted to be elevated on June 21, 2018. Reactor building entries were made on June 21, 2018, and determined that movable incore thimble B-7 was leaking around the 'A' 10-path, and the leak was isolated. During system functional testing, it was determined that water in some of the incore movable detector thimble tubes prevented the moveable incore detection system from being restored to an operable status.

During the performance of TS SR 4.3.3.6, when a core exit thermocouple string is found to be out of tolerance the string is adjusted to bring it back to within the specified calibration tolerance. The adjustment of the thermocouple string then makes it unavailable for use in Power Distribution Monitoring System (PDMS) due to the fact that PDMS was calibrated to the previous thermocouple reading. In order to use the thermocouple string in PDMS again, it must be calibrated to neutron power using moveable incore detection system (which is inoperable).

The licensee stated that if TS SR 4.3.3.6 is performed, and any thermocouple strings were found out of tolerance and needed adjustment, they would be required to be removed from PDMS. If more than TS allowable thermocouples are removed, the requirements of TS 3.3.3.11.b.2 would no longer be satisfied and PDMS would become inoperable. With an inoperable PDMS, plant maneuvers during the performance of planned pre-outage surveillances and testing would be performed without power distribution monitoring capability and increase the probability of a TS required plant shutdown.

The licensee submitted this amendment request on August 24, 2018, however, due to the impracticality of restoring the surveillance capability described above, the licensee revised its request to preclude a plant shutdown and submitted a supplement on the basis of exigent circumstances. By letter dated August 31, 2018, the licensee requested approval by September 24, 2018.

The licensee identified that unidentified leakage was elevated on June 21, 2018 and made multiple attempts to restore the system to complete the SR. The last reactor building entry was made on August 2, 2018. The licensee made a decision on August 7, 2018 to not perform dewatering of the thimbles online due to the radiological and industrial risk associated with the task. The NRC staff finds that the licensee made a timely application for the proposed amendment following its identification and attempted repairs of the issue.

The NRC staff finds that the licensee made reasonable efforts to attempt to restore the incore detector system to operable status by making ten reactor building entries from when the issue was first identified until August 2, 2018. The NRC staffs finds that the licensee could not have avoided the need for an exigent amendment.

The licensee has made preparations to restore the incore detector system to operable status during the upcoming refueling outage which is scheduled to start on October 6, 2018. Considering the multiple attempts to restore the incore detector system to operable status, and after the licensee decided that further attempts would create a radiological risk, the licensee made a timely decision to submit a license amendment request. The NRC staff finds the licensee has not abused the exigent amendment process.

Pursuant to 10 CFR 50.91 (a)(6)(1)(A), the NRC notified the public using a *Federal Register* notice, which was published on September 10, 2018 (83 FR 45688). The notice provided an opportunity for the public to submit comments on the Commission's proposed no significant hazards consideration (NSHC) determination. All comments were requested to be provided to the NRC by September 24, 2018.

One comment was received, and is discussed in section 7.0 of this safety evaluation (SE).

Based on the above and the determination that the amendment involves no significant hazards considerations as discussed Section 5.0 of this SE, the NRC staff concludes that the exigent provisions of 10 CFR 50.91(a)(6) apply to this license amendment request.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Under the Commission's regulations in 10 CFR 50.92, the NRC may make a final determination that a license amendment involves NSHC if operations of the facility in accordance with the amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration in its letter of August 24, 2018. The licensee also provided an analysis in their August 31, 2018, letter, and it was unchanged from the analysis provided in the August 24, 2018, letter. The licensee's analysis is as follows:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change is a short duration, one-time extension to the surveillance frequency requirement of channel calibrations of the Core Exit Temperature Instrumentation. The performance of the surveillance, or the failure to perform the surveillance, is not a precursor to an accident. An extension in performing the surveillance does not result in the system being unable to perform its function. The systems required to mitigate accidents will remain capable of performing their required functions. No new failure modes have been introduced because of this action and the consequences remain consistent with previously evaluated accidents.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

The proposed change only affects the surveillance frequency requirement for the channel calibrations of the Core Exit Temperature Instrumentation. This proposed change does not involve a change to any physical features of the plant, or the manner in which these functions are utilized. No few failure mechanisms will be introduced by the one-time surveillance extension being requested.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed change does not alter any plant setpoints or functions that are assumed to actuate in the event of postulated accidents. The proposed change does not alter any plant feature and only alters the frequency which the surveillance tests must be performed.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff reviewed the licensee's no significant hazards consideration analysis. Based on this review and on the NRC staff's safety evaluation of the underlying license amendment request, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff makes a determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

6.0 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 September 17, 2018. The State of South Carolina official had no comments.

7.0 PUBLIC COMMENTS

On September 10, 2018, the NRC staff published an "Individual Notice of Consideration of Issuance of Amendment to Renewed Facility Operating License NPF-12, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing," in the *Federal Register* associated with the proposed amendment request (83 FR 45688). In accordance with the requirements in 10 CFR 50.91, the notice provided a 14-day period for public comment on the proposed no significant hazards consideration determination. One comment from a member of the public was received, however it was not related to the proposed no significant hazards consideration determination or to the proposed amendment request. The comment can be found at www.regulations.gov, reference NRC-2018-0198.

8.0 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 made a determination that no significant hazards consideration is involved for the proposed amendment as discussed above in SE Section 5.0. 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.

9.0 CONCLUSION

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

Principal Contributors: Nobert Carte, NRR
Diana Woodyatt, NRR
Michael Mahoney, NRR

Date: September 25, 2018

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 - ISSUANCE OF EXIGENT AMENDMENT RE: ONE-TIME EXTENSION TO THE SURVEILLANCE FREQUENCY 4.3.3.6 OF THE CORE EXIT TEMPERATURE INSTRUMENTATION (EPID NO. L-2018-LLA-0226) DATED SEPTEMBER 25, 2018

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ADAMS Accession No.: ML18260A027 ***By Memo Dated**

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OFFICE	NRR/DSS/STSB/BC	OGC-NLO	NRR/DORL/LPL2-1/BC	NRR/DORL/LPL2-1PM
NAME	VCusamano	BHarris	MMarkley	MMahoney
DATE	09/19/18	09/21/18	09/24/18	09/25/18

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