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

March 9, 2016

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

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT No. 1 - ISSUANCE OF EXIGENT AMENDMENT REGARDING TECHNICAL SPECIFICATION 3.7.1.2 (CAC NO. MF7397)

Dear Mr. Lippard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 203 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1, in response to your application dated March 1, 2016, as supplemented on March 3, 2016.

Specifically, the amendment revises Technical Specification (TS) 3.7.1.2, "Plant Systems – Emergency Feedwater System," action statement b for two emergency feedwater pumps being inoperable by adding a note to the statement "be in at least HOT STANDBY within 6 hours" that extends this time period to 24 hours. The extended action duration is needed to allow the testing of three auxiliary feedwater flow control valves that was missed during the previous refueling outage. This is a one-time change and expires on March 18, 2016.

A copy of the related Safety Evaluation (SE) is also enclosed. The SE provides a no significant hazards consideration determination. The Notice of Issuance, addressing the no significant hazards consideration determination and opportunity for a hearing, will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Shawn Williams

Shawn 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. 203 to Renewed NPF-12
2. Safety Evaluation

cc w/enclosures: Distribution 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. 203
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 March 1, 2016, as supplemented on March 3, 2016, 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.

Enclosure 1

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) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 203, 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 license amendment is effective as of its date of issuance and shall be implemented immediately.

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

Date of Issuance: March 9, 2016

ATTACHMENT TO

LICENSE AMENDMENT NO. 203

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Renewed Facility Operating License No. NPF-12 and Appendix "A" Technical Specifications (TS) 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
NPF-12, Page 3	NPF-12, Page 3
TS	TS
3/4 7-4	3/4 7-4

(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. 203, 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.

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 At least three independent steam generator emergency feedwater pumps and flow paths shall be OPERABLE with:

a. Two motor-driven emergency feedwater pumps, each capable of being powered from separate emergency busses, and

b. One steam turbine driven emergency feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one emergency feedwater pump inoperable, restore the required emergency feedwater pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two emergency feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours.

c. With three emergency feedwater pumps inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops a total head of greater than or equal to 3800 feet at greater than or equal to 90 gpm flow.

2. Verifying that the steam turbine driven pump develops a total head of greater than or equal to 3140 feet at a flow of greater than or equal to 97 gpm when the secondary steam supply pressure is greater than 865 psig. The provisions of Specification 4.0.4 are not applicable.

3. Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

* The ACTION to be in at least HOT STANDBY in 6 hours is extended to 24 hours to test (and perform remedial maintenance on) the motor driven emergency feedwater pump flow control valves per surveillance requirement 4.7.1.2.c.2. This extension expires on March 18, 2016.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 203 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 March 1, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16062A368), as supplemented by letter dated March 3, 2016 (ADAMS Accession No. ML16068A140), South Carolina Electric & Gas Company (SCE&G, the licensee) requested an amendment to Renewed Facility Operating License Number NPF-12, for the Virgil C. Summer Nuclear Station, Unit No. 1 (VCSNS). The amendment would revise Technical Specification (TS) 3.7.1.2, "Plant Systems - Emergency Feedwater System." SCE&G requested the amendment under exigent circumstances as discussed in Section 4.0 of this Safety Evaluation (SE).

The amendment would revise TS 3.7.1.2 action statement b for two emergency feedwater pumps being inoperable by adding a note to the statement "be in at least HOT STANDBY within 6 hours" that extends this time period to 24 hours. The licensee stated that the extended action duration is needed to allow the testing of three auxiliary feedwater flow control valves that were missed during the previous refueling outage. This was requested to be a one-time change and expire on March 18, 2016.

Consistent with the requirements in Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.91(a)(6), exigent circumstances, the U.S. Nuclear Regulatory Commission (NRC, the Commission) posted a Public Notice of the proposed amendment in the local newspaper, "*The State*" located in Columbia, South Carolina, on March 5 and March 6, 2016. The notice provided an opportunity for the public to submit comments on the Commission's proposed no significant hazards consideration (NSHC) determination regarding the proposed amendment.

Enclosure 2

2.0 REGULATORY EVALUATION

2.1 Description of System

2.1.1 Emergency Feedwater System

The VCSNS Emergency Feedwater (EFW) system is described in the VCSNS Updated Final Safety Analysis Report (UFSAR) Section 10.4.9, "Emergency Feedwater System."

The EFW system is required to deliver sufficient feedwater to the steam generators for cooldown upon loss of the normal feedwater supply and during an Anticipated Transient Without Scram (ATWS) event. The EFW system is used to supply feedwater to the steam generators during startup, shutdown, and layup operations. The EFW system operates in conjunction with the turbine bypass system, if available, or the main steam power relief valves and safety valves, to remove thermal energy from the steam generators. The system is designed to automatically deliver feedwater to at least two steam generators. There is sufficient redundancy to establish this flow while sustaining a single active failure in the system in the short-term or a single active or passive failure in the long-term. The EFW system operates until the residual heat removal (RHR) system can be placed in operation.

Sufficient feedwater is available under emergency conditions to bring the plant to a safe shutdown condition. Assuming prior plant operation at engineered safety design rating in the core, the minimum required usable volume for the condensate storage tank is 158,570 gallons based on maintaining the plant at HOT STANDBY conditions for 11 hours. This volume also satisfies the minimum required volume to cooldown the plant to HOT SHUTDOWN conditions assuming the plant is maintained at HOT STANDBY for 2 hours and then cooled down to HOT SHUTDOWN in 4 hours.

The EFW system consists of three pumps, two motor driven and one steam turbine driven. The two motor driven pumps share a common discharge header that splits off into three branches. Each branch has a pneumatic flow control valve, which controls flow to its respective steam generator. The one steam turbine driven pump has a separate header from the motor driven pumps. This header splits off into three branches, which controls flow to its respective steam generator. The three flow control valves for the turbine driven pump have control elements fed from A-train power. The three flow control valves for the motor driven pumps have control elements fed from B-train power. During the performance of testing of the B-train flow control valves, the flow path from both of the motor driven pumps is disrupted. This is due to both pumps sharing a common discharge header prior to branching off to the three flow control valves.

The control systems for the turbine driven and motor driven pump flow control valves are identical. Automatic valve opening signals are generated by the reactor protection and logic system and depend upon the given plant condition that determines whether only the motor driven pump flow control valves open or the turbine driven pump valves opens also. These valves will receive an open signal when their respective pumps receive an auto-start. The exception is the motor driven emergency feedwater pump (MDEFP) flow control valves that do not receive an open signal when all three main feed pumps trip. If the flow control valves are in manual control, the valves fully open in response to an automatic open signal.

The EFW system is designed with three flow control valves at the discharge of the turbine driven emergency feedwater pump (TDEFP) and three flow control valves at the discharges of the MDEFPs. The valves are required to control EFW flow to the steam generators to maintain program level and to produce sufficient main steam to permit main feedwater pump turbine operation and for plant cooldown after main steam is no longer able to drive the main feedwater pump turbines. All six valves are identical, 3-inch Fisher ET, normally open, air-operated, globe valves. The valves are safety-related components that meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, class 2, 1974 edition, Summer, 1975 addenda.

2.1.2 Air Accumulator System

The EFW system flow control valves fail open which is the "safe" position for most accidents. The valves are supported by safety-class air accumulators with sufficient capacity to permit remote valve closure on a high-flow signal and maintain the valve closed for at least three hours during a loss of instrument air system. The air accumulators provide a regulated air supply as needed to close the valves against spring force. The accumulators are supported by a non-safety instrument air system. Each valve has a handwheel to allow manual control. The primary safety function of the air accumulator is to assure a source of safety-related air is available to isolate the flow control valve to a faulted steam generator. The three-hour supply permits automatic or remote manual valve closure following a secondary system break when local valve operation cannot be accomplished due to unsuitable conditions for personnel access.

2.1.2.1 History of TS Surveillance Testing for the EFW Flow Control Valves

Recent performance history was provided by the licensee and is documented in the following task sheets for the three-hour drop test for the motor driven emergency feedwater pump flow control valves. For the last five outages the test has been satisfactory.

STTS 0800070, RF-17 (5/29/08)
STTS 0812592, RF-18 (11/26/09)
STTS 1004151, RF-19 (5/21/11)
STTS 1112452, RF-20 (11/12/12)
STTS 1307841, RF-21 (5/5/14)

2.2 Description of Proposed Change

SCE&G requested an amendment to revise TS 3.7.1.2, "Plant Systems – Emergency Feedwater System."

The amendment would revise TS 3.7.1.2 action statement b for two emergency feedwater pumps being inoperable by adding a note to the statement "be in at least HOT STANDBY within 6 hours" that extends this time period to 24 hours. The licensee stated that the extended action duration is needed to allow the testing of three auxiliary feedwater flow control valves that were missed during the previous refueling outage. This was requested to be a one-time change and expire on March 18, 2016.

Action b currently states:

> With two emergency feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

TS Surveillance Requirement (SR) 4.7.1.2.c.2 states that, at least once per 18 months during shutdown, the licensee must demonstrate that each emergency feedwater pump is operable by verifying that "[t]he six emergency feedwater control valves can be closed and held closed for three hours with air from the accumulators when the normal instrument air supply is not available."

The licensee proposed TS 3.7.1.2, Action b, to state (with additions underlined):

> With two emergency feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours<u>*</u> and in HOT SHUTDOWN within the following 6 hours.
>
> <u>*The ACTION to be in at least HOT STANDBY in 6 hours is extended to 24 hours to test (and perform remedial maintenance on) the motor driven emergency feedwater pump flow control valves per surveillance requirement 4.7.1.2.c.2. This extension expires on March 18, 2016.</u>

2.3 <u>Description of Regulatory Requirements</u>

VCSNS UFSAR Section 3.1.2, "Conformance with NRC General Design Criteria," defines the principal criteria and safety objectives for the VCSNS design. The following regulation is relevant to the proposed amendment:

10 CFR Part 50 Appendix A, Criterion 34 - Residual Heat Removal

> A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

VCSNS UFSAR Section 10.4.9, "Emergency Feedwater System," explains that the VCSNS EFW system is required to deliver sufficient feedwater to the steam generators for cooldown upon loss of the normal feedwater supply and during an ATWS.

10 CFR 50.62 requires that pressurized water reactors, such as VCSNS, have equipment that is diverse from the reactor protection system to initiate the EFW system under conditions indicative of an ATWS. The EFW system is required to assure adequate removal of heat from the reactor coolant system during an ATWS.

Section 182a of the Atomic Energy Act (the Act) requires applicants for nuclear power plant operating licenses to include TSs as part of the application. The TSs ensure the operational capability of structures, systems and components that are required to protect the health and safety of the public. The regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36. That regulation requires that the TSs include items in the following categories: (1) Safety limits, limiting safety system settings, and limiting control settings (10 CFR 50.36(c)(1)); (2) Limiting conditions for operation (LCOs) (10 CFR 50.36(c)(2)); (3) Surveillance requirements (SRs) (10 CFR 50.36(c)(3)); (4) Design features (10 CFR 50.34(c)(4)); and (5) Administrative controls (10 CFR 50.36(c)(5)).

10 CFR 50.36(c)(2) states, in part, that:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

VCSNS TS 3.7.1.2, Action b, requires the licensee to place the plant in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following six hours when two emergency feedwater pumps are inoperable.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 <u>Background</u>

By letter dated March 1, 2016, as supplemented by letter dated March 3, 2016, SCE&G requested an amendment to Renewed Facility Operating License Number NPF-12 for VCSNS. The amendment would revise TS 3.7.1.2 action statement b for two emergency feedwater pumps being inoperable by adding a note to the statement "be in at least HOT STANDBY within 6 hours" that extends this time period to 24 hours. The licensee stated that the extended action duration is needed to allow the testing of three auxiliary feedwater flow control valves that were missed during the previous refueling outage. This was requested to be a one-time change and expire on March 18, 2016.

The licensee proposed TS 3.7.1.2, Action b, to state (with additions underlined):

> With two emergency feedwater pumps inoperable, be in at least HOT STANDBY within 6 hours<u>*</u> and in HOT SHUTDOWN within the following 6 hours.
>
> <u>*The ACTION to be in at least HOT STANDBY in 6 hours is extended to 24 hours to test (and perform remedial maintenance on) the motor driven emergency feedwater pump flow control valves per surveillance requirement 4.7.1.2.c.2. This extension expires on March 18, 2016.</u>

The licensee stated that it is requesting this license amendment because Surveillance Test Procedure (STP)-120.006, "Emergency Feedwater Valves Backup Air Supply Test," was not performed during the VCSNS fall 2015 outage for the MDEFP flow control valves. The end date for this test is March 17, 2016, based on an 18-month surveillance interval plus 25% per TS 4.0.2.

3.2 NRC Staff Evaluation

The proposed TS change would allow the testing of the three EFW flow control valves downstream of the two motor driven EFW pumps during operational MODE 1, 2, and 3. There are three flow control valves, one each suppling the A steam generator, B steam generator, and C steam generator. Specifically, flow control valve 3531 feeds the A steam generator, flow control valve 3541 feeds the B steam generator, and flow control valve 3551 feeds the C steam generator.

The EFW flow control valves fail open, which is the "safe" position for most accidents but they are also required to be closed due to a faulted steam generator. The EFW flow control valves are held closed for three hours by their associated air accumulators following an EFW high-flow signal to a faulted steam generator. The three-hour time period permits automatic valve closure following a secondary system break when local valve operation cannot be accomplished because local conditions are unsuitable for personnel access. The EFW flow control valves are supported by safety class air accumulators with sufficient capacity to permit remote valve closure for at least three hours during a loss of the instrument air system. The air accumulators provide a regulated air supply as needed to close the EFW flow control valves against spring force. The accumulators are supported by a non-safety instrument air system.

The focus of the NRC staff review of the proposed TS change is related to the following three key areas:

- Duration of the test, repairs, and possible retest
- Proposed test methodology during MODE 1, 2, and 3
- Compensatory measures and defense-in-depth

3.2.1 Duration of the test, repairs, and possible retest

The proposed TS change from the 6-hour action statement to HOT STANDBY to a 24-hour action statement to HOT STANDBY is based on accommodating the potential testing duration for three EFW flow control valves. The extended time period may be necessary because, if unacceptable leakage is found during the testing of the three EFW flow control valves and the surveillance fails, the SCE&G staff will have to determine the failed component(s) and make the necessary repairs and retests in order for the TS surveillance to pass.

The NRC staff responded to the licensee's March 1, 2016, application with a request for additional information (RAI) in an email dated March 3, 2016 (ADAMS Accession No. ML16064A033). The RAI was with respect to Section 2.1, "Possible Repairs Timeline," of the application that describes component failures that could be encountered following the surveillance testing. The estimated repair times for these failures were based on repairing each item identified below and included tagging out the appropriate isolation devices. The time also accounted for retesting of the EFW flow control valves to ensure the capability to hold the valve closed for three hours as required by TS 4.7.1.2.c.2.

Air accumulator check valve replacement – 18 hours.
Air pressure regulator rebuild and calibration – 10 hours.
Air actuator diaphragm casing bolts torque adjustment – 8 hours.
Air actuator diaphragm replacement – 12 hours.
Air solenoid valve replacement – 14 hours.
Air relief valve replacement and setup – 8 hours.

The NRC staff RAI stated that the application did not describe if the 24-hour time allowance to placing the plant into hot standby has adequate margin to allow a controlled reactor shutdown without initiating a reactor trip. Also, the amount of time to perform a controlled shutdown should be clarified. In addition, the application did not describe the timeline that supports the 24-hour time allowance, given the worst-case component failure noted above (the 18 hours to replace the accumulator check valve).

SCE&G responded to the NRC staff's RAI in a letter dated March 3, 2016. The licensee stated that conservatism is built into the timeline provided in the application and that there is adequate margin to conduct a controlled reactor shutdown without initiating a reactor trip. The worst-case timeline results from failing the initial test, replacing an air accumulator check valve, subsequently failing the post-repair test, and then initiating shutdown. The ACTION statement will be entered when the test is initiated, with the pressure drop test interval for the flow control valve commencing within 15 minutes. VCSNS engineering has provided guidance and psi/min pressure drop criteria for test personnel to determine if the leakrate will exceed acceptance criteria after the first 15 minutes. The test setup opens the air vent valve just upstream of the check valve; therefore, excessive check valve leakage will be detectable at that vent. Based on these times, it is reasonable to assume that a test failure and check valve leakage would be detected within 1 hour of entering the ACTION statement. The worst-case repair time, based on check valve replacement, is 18 hours. This time, however, assumes that the post-maintenance check valve leakrate is acceptable and the re-test takes 4 hours. This scenario postulates unacceptable post maintenance check valve leakage, which would be detected within the first hour of the re-test. Therefore, this time is conservative by 3 hours in addition to the 7-hours of conservatism inherent in the repair time. Also, the amount of time typically allotted at the station for a controlled shutdown to Hot Standby is 3 hours.

Specific worst-case testing, repair, and retesting timelines are as follows:

Time to perform the initial, unsatisfactory, air drop test	1 hour
Time to complete worst-case repair (includes 4 hours re-test)	18 hours
Time to shutdown to Mode 3, assuming repair is unsuccessful	3 hours
Total time for ACTION compliance (1 hour + 18 hours + 3 hours)	22 hours (2-hour margin for the 24-hour LCO)

A breakdown on the 18 hours (noted above) for the worst-case repair is further provided.

System tag out	<1 hour per tag-out (rounded to 3 hours) – 2 hours of margin
Check valve replacement	5 hours (rounded to 8 hours) – 3 hours of margin
Remove tag out	<1 hour per tag-out (rounded to 3 hours) – 2 hours of margin
Post maintenance testing	4 hours
Total	11 hours (rounded to 18 hours) – 7 hours of margin

The NRC staff reviewed the RAI response and finds the response acceptable. The licensee has provided a realistic timeline of the testing, possible worst-case repair plan, retesting, and tag-out durations and has provided a reasonable margin that allows, in the event of unsuccessful repairs, a means to shut down the unit without needing to trip the reactor. Also, the worst-case repair plan has a built-in 7-hour margin. Therefore, the NRC staff finds the proposed duration to be acceptable.

3.2.2 Proposed test methodology during Modes 1, 2, and 3

The proposed change from the 6-hour action statement to HOT STANDBY to a 24-hour action statement to HOT STANDBY is based on testing methodology for the three EFW flow control valves. The NRC staff determined that the closure of all three of the EFW flow control valves at the same time may place the plant at a higher risk than the testing the EFW flow control valves one at a time and declaring each flow path functional once that valve passes the surveillance. Performing the test one valve at a time would also provide defense-in-depth open flow paths to the other two steam generators.

The NRC staff responded to the licensee's March 1, 2016, application with an RAI in an email dated March 3, 2016. The RAI stated that the license amendment request did not have details on how the test is to be conducted, that is, with all three EFW flow control valves closed at the same time for the three-hour hold time, or with each of the three valves closed one at a time for the three-hour hold time. The NRC staff requested, with respect to the test method, that the licensee:

a. Provide specific details of how the flow control valves are tested.
b. Describe if each of the three flow controls valves share any air accumulators.
c. Describe the risk and or benefit for testing all three flow control valves (3 hours hold time) at power versus testing each of the three flow control valves individually.

SCE&G responded to the NRC staff's RAI in a letter dated March 3, 2016. The licensee responded to part 'a' stating that STP-120.006 Revision 7F currently allows testing the valves simultaneously or one at a time (Precaution 2.2). Motor driven flow control valve closure in Modes 1-3 requires entry into limiting condition for operation (LCO) Action Statement 3.7.1.2.b (Precaution 2.1). Performance in Modes 1, 2, and 3 also requires both diesel generators to be operable with no maintenance or testing in progress on either diesel generator. A Pre-Job brief is conducted to stress Human Performance Tool use. Following the brief, an AS FOUND equipment lineup is performed and test instrumentation is installed. If a flow control valve is closed (normally open at power, refer to TS SR 4.7.1.2.a.4), then it is opened. With the flow control valve open, the non-safety-related air supply to the accumulator is isolated. Air is slowly bled-off upstream of the safety-related check valve constituting the boundary between the safety-related (accumulator) and non-safety air systems. The flow control valve is then closed using air from the safety-related air accumulator, and once closed, the drop test is conducted for a period of three hours. Test acceptance is based on meeting established pressure drop (PSI/HR) criterion and the valve being held in the closed position for the three-hour test duration. Subsequent to test completion, manipulated valves are restored to their pre-test positions, the flow control valve is re-opened, and field standards are removed.

The licensee responded to part 'b' stating that the flow control valves do not share an accumulator.

The licensee responded to part 'c' stating that the licensee will be performing the test one flow control valve at a time; thus, entering and exiting the ACTION statement for each valve test. The core damage frequency (CDF) was compared between performing the test one flow control valve at a time versus performing the test when all three flow control valves are closed together. Closing the EFW flow control valves one at a time is more favorable to plant risk.

The NRC staff reviewed the RAI responses and finds them acceptable for each of the three parts noted above. The licensee has elected to perform testing one flow control valve at a time and the procedure, STP-120.006, allows testing to be performed one flow control valve at a time. Since the EFW valves do not share accumulators, the staff finds that there is no reason that the three EFW flow valves cannot be tested individually. The NRC staff finds that, by testing the EFW flow control valves individually, two open flow paths will retain functionality from the motor driven EFW pumps to provide the core cooling function via the other two steam generators, as necessary for any emergency conditions. Therefore, the NRC staff concludes the proposed test methodology is acceptable.

3.2.3 Compensatory measures and defense-in-depth

The proposed change from the 6-hour action statement to HOT STANDBY to a 24-hour action statement to HOT STANDBY is based on the testing duration for each of the three EFW flow control valves.

The staff considered the compensatory actions and defense-in-depth strategies in place to support extended TS limited conditions for operation action times. Compensatory actions such as the station's review for adverse weather conditions before the test, operator actions, and protecting essential systems, structures, and components are key elements for granting time extensions.

The NRC staff responded to the licensee's March 1, 2016, application with an RAI in an email dated March 3, 2016. The RAI asked the licensee to describe the defense-in-depth strategies for core cooling and protection for other safety trains and SSCs appropriate to maintain the safety function of the EFW system during testing. The licensee was requested to provide the following items:

a. Describe if during the performance of the three flow control valve test (all at the same time) if one dedicated operator is required or three dedicated operators are required.
b. Describe the actions for 'a' above and if these actions will be added to the surveillance test.
c. Describe if any electrical busses will be operable (versus available) and protected in support of this test including the turbine driven EFW pump such as direct current busses for valves and governor controls.
d. Describe all other defense-in-depth, special operator training, and protected equipment strategies that are necessary for the performance of Surveillance Test Procedure (STP)-120.006, such as feed and bleed SSCs and just-in time operator training.
e. Management oversight such as management controls related to Infrequently Performed Test or Evolution (IPTE)

SCE&G responded to the NRC staff's RAI in a letter dated March 3, 2016. The licensee responded to part 'a' stating that only one valve will be tested at a time; therefore, one dedicated operator stationed at the flow control valves will be sufficient to perform local manual valve operation, if required.

The licensee responded to part 'b' stating that the actions required to restore functionality to a flow control valve during this test are to close the two air vent valves previously opened, and open the air isolation valve previously closed. These actions are described in STP-120.006, Enclosure A, "Tech Spec/EOOS/Functionality Review." There will be a dedicated local operator stationed at the flow control valves. The dedicated operator is required to be briefed on these actions prior to the test.

The licensee responded to part 'c' stating that the station will protect the following equipment per OAP-114.1: the turbine driven emergency feedwater pump and both emergency diesel generators. Switchyard work that places the plant in a grid risk condition will be restricted. The turbine driven emergency feedwater pump room is normally locked closed. All engineered safety feature busses will be operable during the performance, up to and including instrument busses for valve control.

The licensee responded to part 'd' stating that operators are trained on local manual operation of these valves. Operators are trained on initiation of bleed and feed during a loss of heat sink, within the requalification training matrix every two years. Required operator actions are within the scope of continuous operator training, at regular intervals; therefore, no specialized training is required.

The licensee responded to part 'e' stating that this evolution will be designated as an Infrequently Performed Test or Evolution (IPTE). An IPTE requires a member of Station Management to conduct a pre-job briefing emphasizing key aspects of the evolution. For this evolution, the additional requirements for Critical Infrequently Performed Tests or Evolutions (CIPTE) will be implemented. The Management Duty Supervisor (MDS) will be designated to provide additional management oversight assistance during the evolution.

The NRC staff reviewed the RAI response and finds the response acceptable for each of the five parts noted above. Since the EFW flow control valves will be tested individually, one dedicated operator will be in the field near the EFW control valves to perform any needed actions for quick restoration of the EFW flow path to the affected steam generator. The operator actions are described in the procedure. Related to defense-in-depth actions, during EFW flow control valves testing in Modes 1, 2, and 3, the turbine driven emergency feedwater pump (normally in a locked room) and both emergency diesel generators will be protected. In addition, switchyard work that places the plant in a grid risk condition will be restricted. Related to operator training, the licensee stated that training is current for the operating staff in the event that there is a complete loss of EFW flow to the steam generators. In addition, the local operators are trained on manual operations of the EFW flow control valves.

Also, the licensee stated that management oversight will be involved with this testing evolution in the way of an IPTE and CIPTE. This includes an emphasis in maintaining the highest margins of safety, the need for exercising caution and conservatism during the test or evolution, particularly when uncertainties are encountered, assigned responsibilities for the activity, and the need for open communications. Therefore, the NRC staff finds the proposed compensatory measures and defense-in-depth to be acceptable.

3.3 Technical Conclusion

In conclusion, the NRC staff has determined that the requested exigent license amendment is acceptable. Specifically, the testing of one EFW flow control valve in Modes 1, 2, and 3 provides two other open functional flow paths from the motor driven pumps to the steam generator in the event of an emergency. The licensee provided time studies demonstrating that, under a worst-case equipment failure and repair plan for one flow control valve at a time, the plant can be safely shut down without tripping the unit within the allotted proposed 24-hour LCO. The licensee also described strategies in place related to protected equipment (EFW turbine driven pump and both emergency diesels), along with management oversight during the test, a review of adverse weather conditions before the test such as severe weather considerations, and positioning of dedicated operators for quick system restoration for the steam generator flow path, if needed for steam generator cooling. Based on the above, the NRC concludes that the requirements in 10 CFR 50.36 are met and the license amendment request is, therefore, acceptable.

4.0 EXIGENT CIRCUMSTANCES

The NRC's regulations contain provisions for the issuance of amendments when the usual 30-day prior 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 March 1, 2016, the licensee requested that the proposed amendment be processed by the NRC on an exigent basis.

Under the provisions in 10 CFR 50.91(a)(6), the NRC notifies the public in one of two ways when exigent circumstances exist:

(1) by issuing a *Federal Register* notice providing notice of an opportunity for hearing and allowing at least 2 weeks from the date of the notice for prior public comments; or (2) by using local media to provide reasonable notice to the public in the area surrounding the licensee's facility. In this case, the NRC notified the public using local media. A Public Notice was published in "*The State*" located in Columbia, South Carolina, on March 5 and March 6, 2016. 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 March 8, 2016. No comments were received.

The licensee stated that the need for the amendment request to be processed on an exigent basis is because, due to an oversight, the licensee had failed to conduct the EFW flow control valve surveillance during the Fall of 2015 startup from refueling outage 22. The licensee did not detect this error until the plant was in HOT SHUTDOWN at approximately 345 degrees Fahrenheit (F) and, therefore, rescheduled the performance of the test until after the plant had entered Mode 1. Subsequently, on January 30, 2016, during a normal process schedule review, the licensee considered how the six-hour action to HOT STANDBY required by TS 3.7.1.2 might impact its completion of this test in Mode 1. On February 23, 2016, the licensee completed a contingency matrix for the test and determined that a reasonable repair and retest, if required, could not be completed during the six-hour window provided by TS 3.7.1.2. Therefore, the licensee prepared a license amendment request to extend this time to 24-hours and submitted it to the NRC on March 1, 2016.

Based on the above, the NRC staff finds that the licensee made a timely application for the proposed amendment following its identification of the issue. Based on these findings, and the determination that the amendment involves no significant hazards considerations as discussed below, the NRC staff concludes that the exigent provisions of 10 CFR 50.91(a)(6) apply to this license amendment request.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The NRC's regulations in 10 CFR 50.92(c) state that the NRC may make a determination that a license amendment involves no significant hazards consideration if operation of the facility, in accordance with the proposed 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, which is presented below:

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

   No. A onetime change to the action statement of TS 3.7.1.2, ACTION b, does not increase the probability or consequences of any analyzed accident addressed within

FSAR [Final Safety Analysis Report] Chapter 15. The EF [Emergency Feedwater] system is not an initiator of any Chapter 15 accidents, and the one-time change does not make it an initiator. Therefore, there cannot be an increase in the probability of an accident previously evaluated.

The relevant consequences stem from the ability to maintain core cooling. The change is not detrimental to the ability to remove core heat, because while the maintenance is being performed affects the two MDEFPs [motor driven emergency feedwater pumps], the TDEFP [turbine driven emergency feedwater pump] remains available for Steam Generator cooling. A review of the Chapter 15 analyses shows that for single failure considerations, only one safety train is credited for accident mitigation. Events crediting EF flow assume one EF pump is able to deliver flow to the Steam Generators. This is preserved by maintaining the availability and operability of the TDEFP. The only specific circumstance in which TDEFP operation could be potentially affected is the occurrence of a break of the Main Steam 4" branch line that supplies steam to the TDEFP. Since the activity does not involve a change to the main steam system, or otherwise affects the ability of the main steam system to supply the TDEFWP, there cannot be an increase in the probability of such a break. Nonetheless, in the unlikely event that the MDEFP flowpaths cannot be restored quickly because of that break, and with the area potentially inaccessible, core cooling can still be assured by initiating safety injection to establish feed and bleed cooling, as directed by the Emergency Operating Procedures (EOPs).

Failure to automatically isolate EF to the affected Steam Generator is an important consideration within two secondary pipe break analyses. For secondary side pipe breaks inside containment (FSAR Section 6.2), operator action at 30 minutes is credited to isolate EF to the affected Steam Generator. Local or remote operator action within 30 minutes is required to prevent overpressurizing the containment. Secondly, for secondary side pipe breaks outside containment (FSAR Section 3.11.2.2.2.2 and 10.4.9.3), credit is taken for operator action at 10 minutes to isolate EF to the affected Steam Generator. Since the harsh environment will limit local access and manual actions, operator action from the control room is required for secondary pipe breaks outside containment to preserve environment conditions for equipment qualification. Extending the action statement from six hours to 24 hours does not increase 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?

   No. Extension of the action statement does not create the possibility of a new or different kind of accident from any accident previously evaluated. In the case of secondary breaks outside the reactor building, which would make the flow control valves inaccessible for local operation, procedural guidance outside of the EOPs directs the operators to take alternative action (secure the MDEFPs) if the flow control valve associated with the faulted Steam Generator cannot be closed from the control board. Increasing the duration of the allowed action from six hours to 24 hours does not result in a new or different kind of accident.

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

   No. The relevant margin of safety stems from the ability to maintain core cooling using the Steam Generators. As described previously, the continued operability of the TDEFP preserves the core cooling function in the event of an emergency. The postulation of a single failure is not required while in the LCO Action Time. Nonetheless, because of the availability of safety injection and the ability to perform feed and bleed cooling, core cooling will be assured. Therefore, there will not be a significant reduction in the 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 South Carolina State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to 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.

8.0 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:
Larry Wheeler, NRR/DSS/SBPB

Date: March 9, 2016

March 9, 2016

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

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 - ISSUANCE OF EXIGENT AMENDMENT REGARDING TECHNICAL SPECIFICATION 3.7.1.2 (CAC NO. MF7397)

Dear Mr. Lippard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 203 to Renewed Facility Operating License No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1, in response to your application dated March 1, 2016, as supplemented on March 3, 2016.

Specifically, the amendment revises Technical Specification (TS) 3.7.1.2, "Plant Systems – Emergency Feedwater System," action statement b for two emergency feedwater pumps being inoperable by adding a note to the statement "be in at least HOT STANDBY within 6 hours" that extends this time period to 24 hours. The extended action duration is needed to allow the testing of three auxiliary feedwater flow control valves that was missed during the previous refueling outage. This is a one-time change and expires on March 18, 2016.

A copy of the related Safety Evaluation (SE) is also enclosed. The SE provides a no significant hazards consideration determination. The Notice of Issuance, addressing the no significant hazards consideration determination and opportunity for a hearing, will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Shawn 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. 203 to Renewed NPF-12
2. Safety Evaluation

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