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

October 4, 2017

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: TSTF-411, REVISION 1, "SURVEILLANCE TEST INTERVAL EXTENSIONS FOR COMPONENTS OF THE REACTOR PROTECTION SYSTEM (WCAP-15376-P-A)." (CAC NO. MF7196)

Dear Mr. Lippard:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 209 to Renewed Facility Operating License (RFOL) No. NPF-12 for the Virgil C. Summer Nuclear Station, Unit No. 1 (VCSNS), in response to your application dated December 16, 2015, as supplemented by letters dated March 7, 2016, February 6, 2017, June 22, 2017, July 6, 2017, and September 27, 2017.

This amendment revises Technical Specification (TS) 3/4.3.1, "Reactor Trip System Instrumentation," and TS 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation," to implement the Allowed Outage Time, Bypass Test Time, and Surveillance Frequency changes approved by the NRC in WCAP-15376-P-A, Rev. 1, "*Risk-Informed Assessment of the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times.*" The proposed changes in this license amendment request are consistent with the NRC approved Technical Specification Task Force (TSTF) Improved Standard Technical Specification Change Traveler TSTF-411, Rev. 1, "*Surveillance Test Interval Extensions for Components of the Reactor Protection System (WCAP-15376-P).*"

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. 209 to 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. 209
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 December 16, 2015, as supplemented by letters dated March 7, 2016, February 6, 2017, June 22, 2017, July 6, 2017, and September 27, 2017, 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.C.(2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 209, 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

James R. Hall for

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: October 4, 2017

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

ATTACHMENT TO LICENSE AMENDMENT NO. 209

RENEWED FACILITY OPERATING LICENSE NO. NPF-12

DOCKET NO. 50-395

Replace the following pages of the Renewed Facility Operating Licenses 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	License
Page 3	Page 3
TS	TS
3/4-3-8	3/4-3-8
3/4-3-11	3/4-3-11
3/4-3-12	3/4-3-12
3/4-3-13	3/4-3-13
3/4-3-14	3/4-3-14
3/4-3-35	3/4-3-35
3/4-3-36	3/4-3-36
3/4-3-37	3/4-3-37
3/4-3-38	3/4-3-38
3/4-3-39	3/4-3-39
3/4-3-40	3/4-3-40

(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. 209, 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.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

ACTION 10 - With the number of OPERABLE Channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

ACTION 12 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A	N.A.	R(11)	N.A.	1, 2, 3* ,4*, 5*
2. Power Range, Neutron Flux						
High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	SA	N.A.	N.A.	1, 2
Low Setpoint	S	R(4)	S/U(18), (16)	N.A.	N.A.	1###, 2
3. Power Range, Neutron Flux High Positive Rate	N.A.	R(4)	SA	N.A.	N.A.	1, 2
4. Deleted						
5. Intermediate Range, Neutron Flux	S	R(4)	S/U(18), (16)	N.A.	N.A.	1###, 2
6. Source Range, Neutron Flux	S	R(4)	S/U(18), (17), (9)	N.A.	N.A.	2##, 3, 4, 5
7. Overtemperature ΔT	S	R	SA	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	SA	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	SA	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	SA	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	SA	N.A.	N.A.	1
12. Loss of Flow	S	R	SA	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Steam Generator Water Level--Low-Low	S	R	SA	N.A.	N.A.	1,2
14. Steam Generator Water Level - Low Coincident with Steam/ Feedwater Flow Mismatch	S	R	SA	N.A.	N.A.	1, 2
15. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	SA	N.A.	1
16. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	SA	N.A.	1
17. Turbine Trip						
A. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
B. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
19. Reactor Trip System Interlocks						
A. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2##
B. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
C. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
D Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1,2
E. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
F. Low Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	(7, 12)	N.A.	1, 2, 3,* 4*, 5*
21. Automatic Trip Logic	N.A.	N.A.	N.A.	N.A.	Q (15)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	(7, 13), R(14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

- Below P-6 (Intermediate Range Neutron Flux Interlock) setpoint.

- Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.

(1) - If not performed in previous 31 days.

(2) - Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) - Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - Detector plateau curves shall be obtained evaluated and compared to manufacturer's data. For the Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(6) - Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) - Each train shall be tested at least every 124 days on a STAGGERED TEST BASIS.

(8) - DELETED

(9) - Surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.

(10) - Setpoint verification is not required.

(11) - The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(12) - The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.

(13) - Local manual shunt trip prior to placing breaker in service.

(14) - Automatic undervoltage trip.

(15) - Each train shall be tested at least every 184 days on a Staggered Test Basis.

(16) 12 hours after reducing power below P-10 and 184 days thereafter.

(17) 4 hours after reducing power below P-6 and 4 hours after entering MODE 3 from MODE 2 and 184 days thereafter.

(18) If not performed in previous 184 days.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL QPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(1)	Q(1)	R(3)	1, 2, 3, 4
c. Reactor Building Pressure-High-1	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure--Low	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Differential Pressure Between Steam Lines--High	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Steam Line Pressure Low	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. REACTOR BUILDING SPRAY								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(1)	Q(1)	R(3)	1, 2, 3, 4
c. Reactor Building Pressure-High-3	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continue)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. CONTAINMENT ISOLATION								
a. Phase "A" Isolation								
1) Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						
3) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(1)	Q(1)	R(3)	1, 2, 3, 4
b. Phase "B" Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A	N.A	N.A.	N.A.	Q(1)	Q(1)	R(3)	1, 2, 3, 4
2) Reactor Building Pressure-High-3	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Purge and Exhaust Isolation								
1) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(1)	Q(1)	R(2,3)	1, 2, 3, 4
2) Containment Radioactivity-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
3) Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. STEAM LINE ISOLATION								
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(1)	Q(1)	R(3)	1, 2, 3
c. Reactor Building Pressure-High-2	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Flow in Two Steam Lines--High Coincident with T_{avg}--Low-Low	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure Low	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. TURBINE TRIP AND FEEDWATER ISOLATION								
a. Steam Generator Water Level--High-High	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2
b. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	Q(1)	Q(1)	R(3)	1, 2
6. EMERGENCY FEEDWATER								
a. Manual	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	Q(1)	Q(1)	R(3)	1, 2, 3
c. Steam Generator Water Level--Low-Low	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
EMERGENCY FEEDWATER (Continued)								
d. Undervoltage - Both ESF Busses	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection		See 1 above for all Safety Injection Surveillance Requirements.						
f. Undervoltage - One ESF Bus	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
h. Suction transfer on low pressure	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. LOSS OF POWER								
a. 7.2 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 7.2 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP								
a. RWST level low-low	S	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	O(1)	O(1)	R(3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERA-TIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS								
a. Pressurizer Pressure, P-11	N.A.	R	SA	N.A.	N.A	N.A.	N.A.	1, 2, 3
b. Low, Low Tavg, P-12	N.A.	R	SA	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Each train shall be tested at least every 184 days on a STAGGERED TEST BASIS.

(2) The 36 inch containment purge supply and exhaust isolation valves are sealed closed during Modes 1 through 4, as required by TS 3.6.1.7. With these valves sealed closed, their ability to open is defeated; therefore, they are excluded from the quarterly slave relay test.

(3) Slave Relay Testing will be conducted every 18 months for Westinghouse type AR relays and preferably during a refueling outage to preclude the risk of actuation. Replacement relays other than Westinghouse type AR or reconciled Cutler-Hammer relays will require further analysis and NRC approval to maintain the established frequency.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 209 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 letter dated December 16, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15356A048), supplemented by letters dated March 7, 2016 (ADAMS Accession No. ML16069A021), February 6, 2017 (ADAMS Accession No. ML17037D369), June 22, 2017 (ADAMS Accession No. ML17174B263), July 6, 2017, (ADAMS Accession No. ML17187A504), and September 27, 2017 (ADAMS Accession No. ML17270A203), South Carolina Electric & Gas Company (SCE&G, the licensee) submitted a License Amendment Request (LAR) to modify the Virgil C. Summer Nuclear Station (VCSNS), Unit No. 1, Technical Specifications (TSs).

This amendment revises TS 3/4.3.1, "Reactor Trip System Instrumentation," and TS 3/4.3.2, "Engineered Safety Feature Actuation System Instrumentation," to implement the allowed surveillance test intervals (STIs), completion time (CT), and bypass test time as approved by the U.S. Nuclear Regulatory Commission (NRC) in WCAP-15376-P-A, Rev. 1, "*Risk-Informed Assessment of the Reactor Trip System (RTS) and Engineered Safety Features Actuation System (ESFAS) Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times*" (ADAMS Package Accession No. ML030870767).

Implementation of the changes in this license amendment request is consistent with the NRC approved Technical Specification Task Force (TSTF) Improved Standard Technical Specification Change Traveler TSTF-411, Rev. 1, "Surveillance Test Interval Extensions for Components of the Reactor Protection System (WCAP-15376-P)" (ADAMS Accession No. ML022470164).

The supplements dated March 7, 2016, February 6, 2017, June 22, 2017, July 6, 2017, and September 27, 2017, 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 April 12, 2016 (81 FR 21601).

2.0 REGULATORY EVALUATION

2.1 Background

The Pressurized-Water Reactor Owners Group (PWROG), formerly the Westinghouse Owners Group (WOG), Technical Specifications Optimization Program (TOP) evaluated changes to STIs and CTs (allowed outage time) for the analog channels, logic cabinets, master and slave relays, and reactor trip breakers (RTBs). The methodology evaluated increases in surveillance intervals, test and maintenance out-of-service times, and the bypassing of portions of the reactor protection system (RPS) during test and maintenance. As stated in NRC staff's safety evaluation dated December 20, 2002, for the approval of WCAP-15376-P, Revision 0, (ADAMS Accession No. ML023540534), "In 1983, WOG submitted WCAP-10271-P, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," which provided a methodology to be used to justify revisions to a plant's [RPS] TS. ... The WOG stated in WCAP-10271-P that plant staff devoted significant time and effort to perform, review, document, and track surveillance activities that, in many instances, may not be required on the basis of the high reliability of the equipment. The justification for the changes was the small impact that the changes would have on plant risk."

By letter dated February 21, 1985, the NRC accepted WCAP-10271, including Supplement 1, with conditions. In 1989, the NRC staff issued a safety evaluation report (SER) for WCAP-10271, Supplement 2 that approved similar relaxations for the ESFAS. An additional supplemental SER issued in 1990 provided consistency between the RTS and ESFAS STIs and CTs. The NRC subsequently adopted the TS changes proposed by WCAP-10271 into NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 0, issued September 1992. After the approval of WCAP-10271 and its supplements, the PWROG submitted WCAP-14333- P, "Probabilistic Risk Assessment of the RPS and ESFAS Test Times," in May 1995. The purpose of this topical report was to provide justification for the following TS relaxations beyond those approved in WCAP-10271:

- Increase the bypass test times and CTs for both the solid-state and relay protection system RTS and ESFAS designs for the analog channels, increase the CT from 6 hours to 72 hours and the bypass test time from 4 hours to 12 hours for the logic cabinets, master relays, and slave relays.

- For cases in which the logic cabinet and RTB both cause their train to be inoperable when in test or maintenance, allow bypassing of the RTB for the period of time equivalent to the bypass test time for the logic cabinets, provided that both are tested at the same time and the plant design is such that both the RTB and the logic cabinet cause their associated electrical trains to be inoperable during test or maintenance.

The NRC staff accepted WCAP-14333 by letter dated July 15, 1998. Following the approval of WCAP-14333, PWROG submitted WCAP-15376-P, Revision 0, to the NRC staff on November 8, 2000, which the staff subsequently approved by letter dated December 20, 2002 (ADAMS Accession No. ML023540534). By letter dated March 19, 2003, the Westinghouse Owners Group submitted WCAP-15376-P-A, Revision 1, which incorporated the NRC staff's December 20, 2002, safety evaluation and all Requests for Additional Information and responses thereto. The WCAP-15376-P-A, Revision 1, is referenced as WCAP-15376 in this safety evaluation.

WCAP-15376 provides justification for changes in TSTF-411. Functions which were included in WCAP-15376 are considered as "generically approved," while those not included in WCAP-15376 are considered as "plant-specific." WCAP-15376 specifically evaluated the analog channels, logic cabinets, master relays, and RTBs, and evaluated both the solid-state protection system (SSPS) and the relay protection system. WCAP-15376 also included justification for the following TS relaxations:

- Additional extension of the STIs for components of the RPS and ESFAS to those previously approved in WCAP-10271
- Extension of the STI, CT, and bypass test times for the RTBs

2.2 <u>Description of Changes</u>

This license amendment request changes to TSs 3/4.3.1 and 3/4.3.2 are based on the changes approved in WCAP-15376. In general, the RTB bypass test time is relaxed from 2 hours to 4 hours, the Allowed Outage Time (AOT) is relaxed from 1 hour to 24 hours, and the Surveillance Frequency is relaxed from 2 months to 4 months in TSs 3/4.3.1. The Surveillance Frequencies for the Logic Cabinet are relaxed from 2 months to 6 months; the Master Relays are relaxed from 2 months to 6 months (for plants with solid state protection); and the Analog Channels are relaxed from 3 months to 6 months in both TSs 3/4.3.1 and 3/4.3.2.

The proposed TS changes include some functions in the scope of TSTF-411 and some changes that are not within the scope of TSTF-411. TSTF-411 states that the STI changes will reduce the required testing on the RPS components without significantly impacting their reliability, and reduce the potential for reactor trips and actuation of engineered safety features (ESFs) associated with the testing of these components. The CT extensions for the RTBs will provide the licensee additional time to complete test and maintenance activities while at power, potentially reducing the number of forced outages related to compliance with RTB CTs, and provide consistency with the CTs for the logic cabinets.

A summary of the changes in WCAP-15376 is contained in the two tables below.

Table 1
Summary of WCAP-15376 RTS and ESFAS
STI and CT Changes for the Solid State Protection System

Component	Surveillance Test Intervals	Completion Times and Bypass Times
Logic Train	2 months to 6 months	No changes
Master Relays	2 months to 6 months	No changes
Analog Channels	3 months to 6 months	No changes
Reactor Trip Breakers	2 months to 4 months	Allowed Outage Times: 1 hour to 24 hours; Bypass Time: 2 hours to 4 hours

Table 2
Summary of WCAP-15376 RTS and ESFAS
STI and CT Changes for the Relay Protection System

Component	Surveillance Test Intervals	Completion Times and Bypass Times
Logic Train	2 months to 6 months	No changes
Master Relays	No Changes	No changes
Analog Channels	3 months to 6 months	No changes
Reactor Trip Breakers	2 months to 4 months	Allowed Outage Times: 1 hour to 24 hours; Bypass Time: 2 hours to 4 hours

The NRC staff compared the licensee's proposed changes to WCAP-15376. Any variations from WCAP-15376 or requested changes that were not approved in WCAP-15376 are explained in the technical evaluation.

The VCSNS TS use the term "Allowed Outage Time (AOT)." The corresponding term used in NUREG-1431 is "Completion Time (CT)." The CT is the specified time by which Required Actions must be completed for a designated Condition. For the purpose of this safety evaluation, the terms AOT and CT may be used interchangeably.

2.3 Applicable Regulatory Requirements and Guidance

The NRC staff evaluation is based on the following regulatory requirements and guidance documents.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50.36, "Technical specifications," states, "Each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements of this section." Specifically, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether a limiting condition for operation is required to be included in the TS.

10 CFR Part 50, Appendix A, General Design Criteria (GDC) 13, "*Instrumentation and control*" requires 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 as appropriate to assure adequate safety, including those variables and systems, affecting the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

10 CFR Part 50, Appendix A, GDC 21, "*Protection system reliability and testability*," requires that the protection system be designed for high functional reliability and in service testability, with redundancy and independence sufficient to preclude loss of the protection function from a single failure and preservation of minimum redundancy despite removal from service of any component or channel.

10 CFR Part 50, Appendix A, GDC 22, "*Protection system independence*," requires that the protection system be designed so that natural phenomena, normal operating, maintenance, testing, and postulated accident conditions do not result in loss of the protection function.

10 CFR 50.65, "*Requirements for monitoring the effectiveness of maintenance at nuclear power plants*" requires monitoring the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that these structures, systems, and components are capable of fulfilling their intended functions.

Changes to implement the bypass test time, Completion Time, and Surveillance Frequency changes that were approved by the NRC in WCAP-15376-P-A, Revision 1, "*Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,*" are incorporated in TSTF-411, Revision 1, "*Surveillance Test Interval Extensions for Components of the Reactor Protection System (WCAP- 15376).*"

NUREG-1431, Revision 4, "*Standard Technical Specifications: Westinghouse Plants – Specifications,*" (ADAMS Accession No. ML12100A222) contains the improved standard technical specifications (STSs) for Westinghouse plants.

Regulatory Guide (RG) 1.174, Revision 2, "*An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,*" May 2011 (ADAMS Accession No. ML100910006) describes a risk-informed approach with associated acceptance guidelines for licensees to assess the nature and impact of proposed permanent licensing basis changes by considering engineering issues and applying risk insights.

Regulatory Guide 1.177, "*An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications,*" August 1998 (ADAMS Accession No. ML003740176) describes an acceptable risk-informed approach and additional acceptance guidance geared toward the assessment of proposed permanent TS CT changes.

A three-tiered approach for the licensee's evaluation of the risk associated with a proposed CT TS change are identified in RG 1.177, as discussed below:

- Tier 1 assesses the risk impact of the proposed change in accordance with acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RGs 1.174 and 1.177. The first tier assesses the impact on operational plant risk based on the change in core damage frequency (ΔCDF) and change in large early release frequency (ΔLERF). It also evaluates plant risk while equipment covered by the proposed CT is out of service, as represented by incremental conditional core damage probability (ICCDP) and incremental conditional large early release probability (ICLERP). Tier 1 also addresses probabilistic risk assessment (PRA) quality, including the technical adequacy of the licensee's plant-specific PRA for the subject application. Tier 1 also considers the cumulative risk of the present TS change in light of past (related) applications or additional applications under review along with uncertainty/sensitivity analysis with respect to the assumptions related to the proposed TS change.

- Tier 2 identifies and evaluates any potential risk-significant plant equipment outage configurations that could result if equipment, in addition to that associated with the proposed license amendment, is taken out of service simultaneously, or if other risk-significant operational factors, such as concurrent system or equipment testing, are also involved.

- Tier 3 addresses the licensee's overall configuration risk management program (CRMP) to ensure that adequate programs and procedures are in place for identifying risk- significant plant configurations resulting from maintenance or other operational activities and that the licensee takes appropriate compensatory measures to avoid risk-significant configurations that may not have been considered during the Tier 2 evaluation. Compared with Tier 2, Tier 3 provides additional coverage to ensure that the licensee identifies risk-significant plant equipment outage configurations in a timely manner and appropriately evaluates the risk impact of out-of-service equipment before performing any maintenance activity over extended periods of plant operation. Tier 3 guidance can be satisfied by the Maintenance Rule (Section (a)(4)), which requires a licensee to assess and manage the increase in risk that may result from activities such as surveillance testing and corrective and preventive maintenance, subject to the guidance provided in RG 1.177, Section 2.3.7.1, and the adequacy of the licensee's program and PRA model for this application. The purpose of the CRMP is to ensure that the licensee will appropriately assess from a risk perspective equipment removed from service before or during the proposed extended CT.

RGs 1.174 and 1.177 also describe acceptable implementation strategies and performance monitoring plans to help ensure that the assumptions and analyses used to support the proposed TS changes will remain valid. The monitoring program should include means to adequately track the performance of equipment that, when degraded, can affect the conclusions of the licensee's evaluation for the proposed licensing basis change. RG 1.174 states that monitoring performed in accordance with the Maintenance Rule can be used when such monitoring is sufficient for the structures, systems, and components (SSCs) affected by the risk-informed application.

RG 1.174 includes 5 Key Principles that a risk-informed application should be evaluated to:

1) The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.

2) The proposed change is consistent with the defense-in-depth philosophy.

3) The proposed change maintains sufficient safety margins.

4) When proposed changes increase CDF or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5) The licensee should monitor the impact of the proposed change using performance measurement strategies.

NUREG-0800, "*Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants*", SRP Section 19.1, "*Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities*," (ADAMS Accession No. ML071700657) addresses the technical adequacy of a baseline PRA used by a licensee to support license amendments for an operating reactor.

SRP Section 19.2, "*Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance*," (ADAMS Accession No. ML071700658) provides general guidance for evaluating the technical basis for proposed risk-informed changes.

SRP Section 16.1, "*Risk-Informed Decision Making: Technical Specifications,*" (ADAMS Accession No. ML070380228) provides more specific guidance related to risk-informed TS changes, including CT changes as part of risk-informed decision-making.

3.0 TECHNICAL EVALUATION

3.1 Proposed Changes

The licensee requests changes consistent with TSTF-411, "*Surveillance Test Interval Extensions for Components of the Reactor Protection System* (WCAP-15376-P)." WCAP 15376-P-A provides the technical basis for functions in the scope of TSTF-411 unless specified, in which case a plant-specific analysis is to be performed for those plant-specific functions. The LAR provides the revised marked-up TS changes and the related functional units subject to the proposed changes. The licensee provided the below description of the proposed changes:

Proposed Change 1:

> TS 3/4.3.1, Table 3.3-1 - Action 8: The proposed change for the AOT is from 1 hour to 24 hours. In addition, the Reactor Trip Breaker bypass test time is relaxed from 2 hours to 4 hours. Also, due to the extension of the 1 hour AOT to 24 hours and the time allowed to bypass one channel is being extended from 2 hours to 4 hours, the last provisions of Action 8 allowing additional time for maintenance and an extended bypass time have been deleted.

Proposed Change 2:

TS 3/4.3.1, Table 4.3.1 - Analog Channel Operational Test (ACOT): The proposed change to the ACOT is from Quarterly (92 days) to Semi-Annually (184 days) for the following functions in Table 4.3-1 of the VCSNS TSs: 2 (High Setpoint only), 3, 7, 8, 9, 10, 11, 12, 13, and 14. The proposed change to the ACOT frequency is changed to 184 days for the following functions in Table 4.3-1 of the VCSNS TSs: 2 (low Setpoint only), 5, and 6 by the addition of Notes 16, 17, and 18.

Trip Actuating Device Operational Test (TADOT): The proposed change to the TADOT is from Quarterly to Semi-Annually for the following functions in Table 4.3-1 of the VCSNS TSs: 15 and 16. The description for each of the above functions is in Table 4.3-1 of the current technical specifications.

Proposed Change 3:

TS 3/4.3.1, Table 4.3-1 –TADOT: The proposed change to the TADOT for the Reactor Trip Breaker (Function 20) is from Monthly (62 days on a Staggered Test Basis) to every 2 months (124 days on a Staggered Test Basis) as shown in revised Note 7. Note because of the difference in the Staggered Test Basis definition (discussed above) the number of days used for the VCSNS surveillance frequency is larger than the number used in TSTF-411 surveillance frequency.

Proposed Change 4:

TS 3/4.3.1, Table 4.3-1 – TADOT: The proposed change to the TADOT for the Reactor Trip Bypass Breaker (Function 22) is from Monthly (62 days on a Staggered Test Basis) to every 2 months (124 days on a Staggered Test Basis) as shown in revised Note 7 (62 days to 124 days), which is being inserted into the TADOT column. Note that this change is different than what is shown in the markups for TSTF-411. This change is necessary because TSTF-411 does not provide a change to the Reactor Trip Bypass Breaker as the Bypass Breaker is treated as a part of the Reactor Trip Breaker function in the standard TS in NUREG-1431 and has the same surveillance frequency assigned. In the VCSNS TS, the Reactor Trip Breakers and the Reactor Trip Bypass Breakers are separate Functions consistent with the standard TS in NUREG-0452. In the VCSNS TS, the two separate Functions are assigned the same frequency specified in Note 7 to be consistent with the change for the corresponding NUREG-1431 Reactor Trip Breaker Function in TSTF-411. Thus, both the VCSNS Reactor Trip Breakers and Reactor Trip Bypass Breakers will be tested at the same surveillance frequency consistent with NUREG-1431 and TSTF-411.

Proposed Change 5:

TS 3/4 3.1, Table 4.3-1 - Actuation Logic Test (ALT): The proposed change to the ALT for the Automatic Trip Logic (Function 21) is from Monthly to Quarterly on a Staggered Test Basis as specified in new Note 15 (184 days).

Proposed Change 6:

TS 3/4.3.2, Table 4.3-2 – ACOT: The proposed change to the ACOT is from Quarterly to Semi-Annually for the following functions in Table 4.3-2 of the VCSNS TSs: 1.c, 1.d, 1.e, 1.f, 2.c, 3.b.2, 4.c, 4.d, 4.e, 5.a, 6.c, 6.h, 8.a, 9.a, and 9.b (see Attachment 1 TS mark-up for the title of each Function).

The NRC staff notes that Function 6.h, "Emergency Feedwater, suction transfer on low pressure, Analog Channel Operational Test," and Function 8.a, "Automatic Switchover to Containment Sump, refueling water storage tank (RWST) level lo-lo, Analog Channel Operational Test," provide swap-over functions to an alternate source from the condensate storage tank and RWST, respectively, and are independent of the RTS. The NRC staff's SER for WCAP-10271-P-A functions did not include approval for these two functions. However, in Amendment No.101 for VCSNS's implementation of WCAP-10271-P-A, the NRC staff approved an increase in the VCSNS STIs from monthly to quarterly based on a plant-specific analysis. The licensee's LAR proposes to extend these two STIs from quarterly to semi-annually. The NRC staff considers these two functions to be plant-specific and require a plant-specific analysis as they were not included in the WCAP-15376-P-A analysis.

Proposed Change 7:

TS 3/4.3.2, Table 4.3-2 – ALT: The proposed change to the ALT is from Monthly to Quarterly on a Staggered Test Basis as shown in the revised Note 1 (62 days to 184 days) for the following functions in Table 4.3-2 of the VCSNS TSs: 1.b, 2.b, 3.a.3, 3.b.1, 3.c.1, 4.b, 5.b, 6.b, and 8.b.

Proposed Change 8:

TS 3/4.3.2, Table 4.3-2 - Master Relay Test: The proposed change to the Master Relay Test is from Monthly to Quarterly on a Staggered Test Basis as shown in the revised Note 1 (62 days to 184 days) for the following functions in Table 4.3-2 of the VCSNS TSs: 1.b, 2.b, 3.a.3, 3.b.1, 3.c.1, 4.b, 5.b, 6.b, and 8.b.

<u>The following Proposed Changes were not included in the scope of TSTF-411</u>

Proposed Change 9:

TS 3/4.3.1, Table 4.3-1, RTS Function 15: Reactor Trip on Reactor Coolant Pump Undervoltage from Quarterly to Semi-Annually.

Proposed Change 10:

TS 3/4.3.1, Table 4.3-1, RTS Function 16: Reactor Trip on Reactor Coolant Pump Underfrequency from Quarterly to Semi-Annually.

The NRC staff notes that in the NRC's letter to the Westinghouse Owners Group, "Acceptance for Referencing of Licensing Topical Report WCAP-10271, Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection System Instrumentation Systems," dated February 21, 1985, the NRC staff stated that reactor coolant pump undervoltage and underfrequency units were included in the unavailability models, and the NRC staff approved their use for analog channels. In VCSNS Amendment No.101 (ADAMS Accession No. ML012250025) regarding VCSNS's implementation of WCAP-10271-P-A, the NRC approved the trip actuating device check from monthly to quarterly. The licensee's LAR proposes to extend these STIs from quarterly to semi-annually. The NRC staff considers the

WCAP-15376-P-A analysis to be applicable to these two functions since they had been included in the unavailability models for WCAP-10271-P-A.

The following table summarizes the components and the proposed VCSNS TS changes.

Table 2: TSTF-411 Proposed Changes

RPS/ESFAS Components	Surveillance Frequency			
	Current (Mo.)		Proposed (Mo.)	
Analog Channel	3		6	
Logic Cabinet	2		6	
Master Relay	2		6	
Reactor Trip Breaker	2		4	
	Completion Time (Hr.)		Bypass Test Time (Hr.)	
	Current	Proposed	Current	Proposed
Reactor Trip Breaker	1	24	2	4

3.2 <u>NRC Staff Evaluation</u>:

In accordance with SRP Sections 16.1, 19.1, and 19.2, the NRC staff reviewed the VCSNS incorporation of WCAP-15376 using the Key Principles of the risk-informed decision-making presented in RGs 1.174 and 1.177, the RG 1.177 three-tiered approach, and the WCAP-15376 SER conditions and limitations.

3.2.1 Key Principles of the risk-informed decision-making presented in RGs 1.174 and 1.177

Key Principle 1: The proposed change meets the current regulations, unless it explicitly relates to a requested exemption or rule change.
Key Principle 2: The proposed change is consistent with the defense-in-depth philosophy.
Key Principle 3: The proposed change maintains sufficient safety margins.

The NRC staff evaluated these Key Principles in the WCAP-15376-P, Rev 0, "*Risk-Informed Assessment of RTS & ESFAS Surveillance Test Intervals & Reactor Trip Breaker Test & Completion Times*," *NRC Safety Evaluation* (ADAMS Accession No. ML030870542). The NRC staff's evaluation found that WCAP-15376 was consistent with the accepted guidelines of RG 1.174 and RG 1.177, and NRC staff guidance as outlined in NUREG-0800, "*Standard Review Plan.*" From traditional engineering insights, the NRC staff found that the proposed changes in WCAP-15376 continue to meet the regulations, have no impact on the defense-in-depth philosophy, and would not involve a significant reduction in the margin of safety.

Key Principle 4: When proposed changes increase CDF or risk, the increase(s) should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

The NRC staff evaluation found that the changes proposed by the licensee employ a risk-informed approach to justify changes to STIs, CT, and bypass test time. The risk metrics, ΔCDF, ΔLERF, ICCDP, and ICLERP that were developed in the topical report and used to evaluate the impact of the proposed changes are consistent with those presented in RGs 1.174 and 1.177.

Key Principle 5: The licensee should monitor the impact of the proposed change using performance measurement strategies.

RGs 1.174 and 1.177 also establish the need for an implementation and monitoring program to ensure that extensions to TS CT, bypass test times, or reduction in TS safety features do not degrade operational safety over time and that no adverse effects occur from unanticipated degradation or common-cause mechanisms. The purpose of an implementation and monitoring program is to ensure that the impact of the proposed TS change continues to reflect the reliability and availability of SSCs impacted by the change. In addition, the application of the three-tiered approach in evaluating the proposed CT, bypass test times, and STI extensions provides additional assurance that the changes will not significantly impact the key principle of defense in depth.

The licensee monitors the reliability and availability of the RTS and ESFAS instrumentation under the Maintenance Rule (Section (a)(1)), which requires a licensee to monitor the performance or condition of SSCs against licensee-established goals. In response to RAI 11, the licensee confirmed that VCSNS follows the current Maintenance Rule guidance in NUMARC 93-01, Revision 4A, "*Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,*" April 2011 (ADAMS Accession No. ML11116A198) and RG 1.160, Revision 3, "*Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,*" May 2012 (ADAMS Accession No. ML113610098). Based on the above, the NRC staff finds that VCSNS satisfies the RG 1.174 and RG 1.177 guidelines for an implementation and monitoring program for the proposed change.

3.2.2 Staff Evaluation of the Three-Tiered Risk Assessment

WCAP-15376 is consistent with the NRC approach for using probabilistic risk assessment in risk-informed decisions on plant-specific changes to the current licensing basis as presented in Regulatory Guides 1.174, "*An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis,*" and 1.177, "*An Approach for Plant Specific, Risk-Informed Decision-making: Technical Specifications.*" The WCAP-15376 approach addresses the impact on defense-in-depth, the impact on safety margins, and the impact on risk. The risk evaluation considered the three-tiered approach as presented by the NRC in Regulatory Guide 1.177 for the extension to the RTB Completion Time. Tier 1, PRA Capability and Insights, assesses the impact of the proposed Completion Time (AOT) change on core damage frequency (CDF), incremental conditional core damage probability (ICCDP), large early release frequency (LERF), and incremental conditional large early release probability (ICLERP). Tier 2, Avoidance of Risk Significant Plant Configurations, considers potential risk-significant plant operating configurations. Tier 3, Risk-Informed Plant Configuration Control and Management, is addressed when the Technical Specification Completion Time change is implemented.

Tier 1: Probabilistic Risk Assessment Capability and Insights

The first tier evaluates the impact of the proposed changes on plant operational risk based on the VCSNS implementation of WCAP-15376. In Tier 1, the NRC staff review involves (1) evaluation of the validity of the PRA and its application to the proposed changes, and (2) evaluation of the PRA results and insights based on the licensee's proposed application.

PRA Technical Adequacy

The objective of the PRA technical adequacy review is to determine whether the PRA model in WCAP-15376, which is used in evaluating the proposed RPS and ESFAS STI extensions, is of sufficient scope and detail for this application. The WCAP-15376 provided a generic PRA model for the evaluation of the STI extensions, and RTB CT and bypass test time. The NRC staff found this generic model and the WCAP-15376 evaluations to be acceptable on a generic basis in its SER dated December 20, 2002. Although the SER accepted the use of a representative model as generally reasonable, the application of the representative model and the associated results to a specific plant introduces a degree of uncertainty because of modeling, design, and operational differences. Therefore, each licensee adopting WCAP-15376 would need to confirm that the topical report analyses and results are applicable to its plant.

In the SER for WCAP-15376, the NRC staff found that the applicability of the generic PRA analysis for the proposed TS extensions to other Westinghouse plants may not be representative based on design variations in actuated systems and the contribution to plant risk from accident classes impacted by the proposed change. The NRC staff therefore concluded that each licensee would need to address any differences between its plant and the representative plant that could increase the STI, CT, or bypass test time.

To determine that WCAP-15736 is applicable to VCSNS, the licensee addressed the implementation guidance developed by the PWROG in the LAR, Attachment 5 by comparing plant-specific data to the generic analysis assumptions. The evaluation compared the general baseline assumptions including surveillance, maintenance, procedures, operator actions, transient and ATWS frequencies, actuation signals, safety functions, and certain component failure probabilities to confirm that the generic evaluation assumptions used in the topical reports are also applicable to VCSNS.

The WCAP-15376 was based on a large dry containment and assumed that the only contributions to LERF would come from containment bypass events and core damage events with the containment not isolated. The contributions from containment failure events are not considered in WCAP-15376. The licensee concluded in LAR Attachment 5, Section 3.1.3, that the WCAP analysis and determination of LERF is based on a large dry containment; therefore, the results are applicable.

In response to RAI 4 to address the plant-specific containment assessment of Condition 1, the licensee provided the LERF profile for containment failure contributors which included containment failure events other than bypass or failure of containment isolation. The licensee found that the containment isolation and bypass events were dominant contributors to the LERF and concluded that the plant-specific containment design has no major differences. According to the response to RAI 3 regarding functions 15 and 16, the licensee's assessment for RAI 4 is also applicable to these two signals. In response to RAI 7.k.i, the licensee concluded that functions 6.h and 8.a do not have a unique impact on containment failure events. Based on the applicability of the topical report and the risk insights from the LERF profile, the NRC staff finds the licensee's containment assessment conclusions to be reasonable.

In response to RAI 5, the licensee provided justification for the smaller plant-specific transient frequency compared to that used in the topical report. The licensee provided a profile of transient initiating events based on plant-specific frequency calculations, and stated that the frequencies of turbine trip and loss of main feedwater have been significantly reduced. Based on the plant-specific analysis, the NRC staff finds that the transient frequency was justified and the applicability of the topical report is not changed.

In response to RAI 3, the licensee explained that WCAP-15376-P-A is applicable for functions 15 and 16 because their TADOT surveillance frequencies were justified in WCAP-10271-P-A and its supplements. NRC staff notes that the NRC Safety Evaluation dated February 21, 1985 "*Acceptance for Referencing of Licensing Topical Report WCAP-10271, Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection System Instrumentation Systems*," approved reactor coolant pump undervoltage and underfrequency functions for WCAP-10271-P-A. WCAP-15376-P-A, Section 11, states that its recommendations are applicable to all signals evaluated in WCAP-10271-P-A. Further, TSTF-411 did not identify functions 15 and 16 as requiring a plant-specific analysis. Therefore, the NRC staff agrees that WCAP-15376-P-A is applicable to functions 15 and 16.

WCAP-15376-P-A did not evaluate ESFAS functions 6.h and 8.a; therefore, the licensee performed a plant-specific analysis for its proposed STI extensions. In response to RAI 7.b, 7.c.ii and RAI 9.ii, the licensee described the PRA modeling for function 6.h and function 8.a. The NRC staff finds that the PRA model level for functions 6.h and 8.a is adequate for the application because it is generally consistent with the PRA model level evaluated for functions in WCAP-15376-P-A, and includes the instrumentation and control components associated with the proposed STI extensions.

In response to RAI 7.c.ii and 7.c.iii, the licensee explained the PRA modeling of applicable initiating events and plant responses. The licensee stated that if the initiating event requires the functions 6.h and 8.a in response to the initiating event, then the functions are available for that initiating event. Furthermore, the licensee stated that there is no model incompleteness with respect to initiating events or plant responses, and confirmed that the PRA model for functions 6.h and 8.a reflect the as-built, as-operated plant. Based on these responses, the NRC staff finds that the PRA model sufficiently includes associated initiating events and plant responses for functions 6.h and 8.a.

In response to RAI 7.e, 7.g.iii and follow up RAI 7.d, 7.f, 7.g, 7.i, 7.j, and 12, the licensee provided the sources of the PRA data and explained data uncertainties used for functions 6.h and 8.a. The licensee stated that the source of PRA data for the components modeled for functions 6.h and 8.a come from the following references: (1) WCAP-15376-P-A, (2) NUREG/CR-5500, "*Reliability Study: Combustion Engineering Reactor Protection System, 1984 – 1998*" (ADAMS Accession No. ML003773958), and (3) NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants." Regarding the data uncertainties, the licensee stated that some analog channel basic event probabilities were conservatively doubled from the three month probabilities and the logic cabinet and master relay components were increased by a factor of three, in accordance with their proposed STI extensions. This method assumes a linear relation between the failure probability and the STI, similar to the topical report. The licensee stated that the other components' basic event probabilities in the signal path have uncertainties based on the sources from which they were derived. The NRC staff finds the data and data uncertainty sources used for functions 6.h and 8.a are appropriate for the application because they are consistent with those used in the topical report or use updated data.

Peer Review

The Westinghouse Owners Group performed a peer review of the VCSNS internal events PRA model in August 2002, conducted in accordance with NEI 00-02, "Probabilistic Risk Assessment (PRA) Peer Review Guidance." In 2005, the licensee performed a gap assessment of the internal events PRA model against RG 1.200, Revision 0, *"An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,"* February 2004. Following work to address the facts and observations (F&Os) from the gap assessment, a focused scope peer review against RG 1.200, Revision 1, was performed by the licensee in 2007. In 2011, the licensee completed a gap assessment against American Society for Mechanical Engineers / American Nuclear Society (ASME/ANS) PRA Standard ASME/ANS RA-Sa-2009, Addenda to ASME RA-S-2008, "*Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications*" and RG 1.200, Revision 2, "*An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,"* March 2009 (ADAMS Accession Number ML090410014). In response to RAI 1, the licensee provided F&Os from the 2002, 2007, and 2011 reviews which were (1) open, or (2) closed by self-assessment for supporting requirements that the review or gap assessment found was not met or met at capability category I, and their resolutions. The NRC staff reviewed these F&Os and resolutions and found either no impact or adequate disposition. The evaluation of open F&Os or issues is discussed below.

The following observations were identified to have the resolution not completed from the 2011 gap assessment.

> F&O 4_6 stated that there were no repeated failures noted within a short time interval, but there is no guidance to ensure such failures are counted as single failures. The licensee determined that this was a documentation/ guidance issue because there were no examples identified and there is no impact on the LAR. Based on this resolution, the NRC staff agrees that this F&O does not impact the application.
>
> F&O 3_3 provided a documentation suggestion for a human reliability analysis to provide additional guidance on pre-initiators. The licensee determined this was a documentation issue and does not impact the LAR. The NRC staff agrees this documentation issue does not impact the application.
>
> F&O 1_43 provided a suggestion for monitoring sources for technology changes. The licensee determined that this suggestion does not impact the LAR. The NRC staff agrees this suggestion does not impact the application.
>
> F&O 6_9 stated that "several component and component failure modes were identified as being screened from the model without meeting the justification specified in SY-A15, SY-A11, SY-B13, SY-A13, and SY-A14. Flow diversion pathways were screened based on relative cross sectional areas rather than pressures/flows and some components were screened based on assumed low failure probabilities rather than quantifications. For all systems, provide quantitative justification for screening components, failure modes, and flow diversion paths from the model."

The licensee stated in the F&O 6_9 resolution that re-screening of systems was not completed. In response to follow up RAI 1, the licensee addressed this F&O. The licensee reviewed all modeled systems that initially screened out, evaluated components and failure modes using new screening criteria, and performed sensitivity analyses which showed insignificant impact on CDF and LERF. The licensee concluded that this screening issue does not impact the actions in the WCAP-15376 human reliability analysis, does not increase the ATWS contribution to CDF, and does not impact the total transient frequency. The NRC staff finds that this F&O does not have an impact on the application.

In response to internal flooding F&Os, licensee resolutions stated that due to the numerous internal flooding PRA observations from the 2011 self-assessment, VCSNS performed a complete update. In response to RAI 2.ii, the licensee clarified that the internal events PRA model is maintained to the requirements of ASME/ANS-RA-Sa-2009, and that the internal flood PRA model has been updated to the requirements of the ASME/ANS RA-Sb-2013 standard. VCSNS considers the internal flooding model requirements in the ASME/ANS RA-Sb-2013 standard to better represent the components needed to identify risk vulnerabilities due to flooding. The NRC staff has not endorsed the ASME/ANS RA-Sb-2013 standard; however, the NRC staff finds that using the ASME/ANS RA-Sb-2013 standard (instead of ASME/ANS-RA-Sa-2009) results in a small risk increase for this application due to the available margin to the acceptance guidelines. According to the response to RAI 1 in the supplement dated February 6, 2017, in June 2016 the Westinghouse Owners Group performed a peer review of the internal events PRA, including internal flooding. In response to follow up RAI 1 in the supplement dated July 6, 2017, regarding the impact of the June 2016 peer review, the licensee stated that it had no impact on the PRA model of record used to support the LAR, the LAR supplement response, or the LAR RAI responses.

Based on the NRC staff's review of the F&Os provided in response to RAI 1 and the licensee's determination that the 2016 peer review does not impact the LAR, the NRC staff finds the internal events PRA, excluding the internal flooding PRA model, to be of sufficient technical adequacy for the application. While the internal flooding PRA model was not reviewed against an NRC-endorsed PRA standard for technical adequacy, the NRC staff does not expect the risk impact contribution from the internal flooding to change the conclusions of "small" risk increase due to the available margin to the acceptance guidelines in RG 1.174.

PRA Update/Procedures

VCSNS has a PRA procedure for maintaining the model, configuration control, and a process for tracking issues potentially affecting the PRA model. PRA updates include a review of plant changes, plant procedures, plant operating and equipment history data, and plant-specific frequencies, failure rates, and other data-driven parameters. PRA maintenance serves to keep the PRA current between model updates. Both maintenance and updates are documented and verified in accordance with procedures. In addition, plant procedures address computer control, software testing, and deficiency tracking.

PRA Results and Insights

Cumulative Risk

In response to follow up RAI 7.d, 7.f, 7.g, 7.i, 7.j, and 12 in its supplement dated June 22, 2017, the licensee provided delta and cumulative delta for both CDF and LERF. The topical reports evaluated functions with either a 2/4 or a 2/3 logic; however, the licensee used the results for 2/3 logic because this is the predominant logic at VCSNS. The plant-specific evaluation results for functions 6.h and 8.a are added to the 2/3 logic results. The plant-specific results included the STI extensions associated with the analog channels, the logic cabinets, and the master relays; however, no credit was taken for decreases in component unavailability due to maintenance or test. The licensee reported the cumulative risk from pre-TOP (i.e., before WCAP-10271) to the WCAP-15376-P-A values to be 2.0E-6/yr (ΔCDF) and 1.1E-7/yr (ΔLERF), including the plant-specific function evaluations. The licensee stated that changes from pre-TOP to WCAP-10271-P-A were not risk-informed and gave no contribution to the cumulative risk results. The reported change in CDF and LERF results are shown in the table below, and include only the internal events risk contribution.

In response to the NRC staff's request for supplemental information, the change in CDF for function 6.h was reported as an increase and the change in LERF was a decrease. In response to RAI 7.g.ii, the licensee explained that these results were due to post-processing calculations for very low frequency cut sets. Since the PRA calculation does not provide meaningful results under those conditions, the NRC staff evaluates the change in LERF for function 6.h using a bounding evaluation. If the reported change in CDF for function 6.h of 8.7E-9/yr was to also be assumed to be the change in LERF, the impact on the LERF results of the application would be negligible.

External events considered in LAR Attachment 5 were seismic, fire, high winds, external flood, and transportation and nearby facility accidents. The licensee presented arguments that low seismic frequencies, coupled with small or decreased signal unavailabilities associated with implementing WCAP-15376-P-A, operator actions, system mitigation, and the reactor trip function from ATWS mitigation actuation circuitry (AMSAC), are expected to result in very small risk increase, or even a risk benefit for some seismic events. In addition, implementation of WCAP-15376-P-A was also reported to result in a small reduction in fire CDF (i.e., a risk benefit). In response to RAI 13 in its supplement dated February 6, 2017, the licensee explained that this CDF reduction is due to the ESFAS signal unavailability reduction for the single train emergency feed water pump start signal in fire areas where the fire could impact one train. The licensee's analysis found that this decrease in fire risk was greater in these areas than in areas where two trains could be affected. With regard to the other external hazards, the licensee reviewed the VCSNS Individual Plant Examination for External Events studies and qualitatively concluded that the proposed TS changes have no impact or no significant impact on plant risk. Based on the information provided in LAR Attachment 5 as supplemented by the response to RAI 13, the NRC staff expects external events to have an overall negligible change-in-risk contribution compared to that from internal events.

Table 3: Cumulative ΔCDF and ΔLERF for Proposed STI Extensions

	WCAP-10271-P-A to WCAP-14333-P-A	WCAP-14333-P-A to WCAP-15376-P-A	Pre-TOP to WCAP-15376-P-A (note)
	ΔCDF (/yr)	ΔCDF (/yr)	ΔCDF (/yr)
2/3 logic	6.1E-7	8.5E-7	
Plant-specific Function Evaluation			
EFW Suction ESFAS 6.h	0	0	
RWST level ESFAS 8.a	3.8E-8	5.0E-7	
Total ΔCDF (/yr)	6.5E-7	1.3E-6	2.0E-6
	ΔLERF (/yr)	ΔLERF (/yr)	ΔLERF (/yr)
2/3 logic	2.2E-8	5.7E-8	
Plant-specific Function Evaluation			
EFW Suction ESFAS 6.h	0	0	
RWST level ESFAS 8.a	4E-10	3.1E-8	
Total ΔLERF (/yr)	2.2E-8	8.8E-8	1.1E-7

Note: 2/4 logic results from the WCAP studies are not used for the total ΔCDF and ΔLERF because 2/3 logic is the predominant logic for VCSNS.

The estimated total baseline CDF was reported to be 6.87E-5/yr and total LERF to be 3.56E-6/yr. These frequencies include internal events/internal flooding, seismic, and fire events. The NRC staff finds that given these total baseline frequencies, the ΔCDF and the ΔLERF satisfy the RG 1.174 acceptance guidelines for "small" increases. As shown in RG 1.174 Figures 4 and 5, the increase in risk is considered small if the ΔCDF is in the range of 10E-6/yr to 10E-5/yr and the ΔLERF is in the range of 10E-7/yr to 10E-6/yr.

WCAP-15376-P-A also evaluated the reactor trip breaker proposed extensions from WCAP-14333-P-A as a base case. The RTB results were reported in the LAR, Table 1, for predominantly 2/3 logic plants which bounds the 2/4 logic results, and were less than the RG 1.177 acceptance guidelines of 5E-7 for ICCDP and 5E-8 for ICLERP.

Tier 2: Avoidance of Risk-Significant Plant Configurations

A licensee should provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when specific plant equipment is taken out of service in accordance with the proposed TS change.

In LAR Section 4.2, the licensee provided Tier 2 restrictions. The NRC staff reviewed these restrictions and found that they are consistent with the Tier 2 restrictions identified in the NRC staff's SER, Section 3.3, on WCAP-15376-P. In response to RAI 3 the licensee evaluated the RTS functions 15 and 16, and identified no changes to the Tier 2 restrictions provided in the LAR. The licensee also evaluated Tier 2 for ESFAS functional units 6.h and 8.a in response to RAI 7.k.ii and follow up RAIs 7.d, 7.f, 7.g, 7.j, and 12, and identified no additional Tier 2 restrictions. The licensee stated that Tier 2 restrictions identified when a RTB is unavailable will be flagged in the VCSNS Equipment Out of Service (EOOS) tool.

The licensee evaluated concurrent component outage configurations and confirmed the applicability of Tier 2 restrictions for VCSNS. Based on the above, the NRC staff finds the licensee's Tier 2 analysis supports the implementation of WCAP-15376 at VCSNS and satisfies the condition of the staff SERs for WCAP-15376 regarding Tier 2.

Tier 3: Risk-Informed Configuration Risk Management Program (CRMP)

The CRMP provides a proceduralized risk-informed assessment to manage the risk associated with equipment inoperability. The CRMP assessment tool utilizes at least a Level 1 at-power internal events PRA model. The CRMP assessment may use any combination of quantitative and qualitative input. VCSNS has the capability to perform a configuration dependent assessment of the overall impact on risk of proposed plant configurations prior to, and during, the performance of maintenance activities that remove equipment from service. VCSNS re-assesses risk if an equipment failure or malfunction or emergent condition produces a plant configuration that has not been previously assessed.

In response to RAI 8, RAI 9, and RAI 10, the licensee explained how the CRMP meets RG 1.177 guidance. VCSNS uses the EOOS risk monitor for the CRMP which calculates CDF and LERF based on the internal events PRA model and is modified to support maintenance risk evaluations. VCSNS monitors and assesses plant modifications and procedure changes to determine if the PRA model, and, therefore, EOOS, should be revised. The procedures for the PRA model are applicable to the CRMP model, the Maintenance Rule (10 CFR 50.65), and implementation procedures, and are controlled in accordance with 10 CFR Part 50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The CRMP guidance in RG 1.177 allows external hazards to be treated qualitatively and/or quantitatively. In response to RAI 9.i, the licensee explained that EOOS does not quantify fire-related risk; however, it notifies the user to take actions when equipment needed to safely shut down the plant during a fire is taken out of service. These steps are taken as part of the VCSNS Fire Emergency Procedure Risk program, which considers equipment removed from service for planned maintenance, emergent work, and testing. For external events not modeled, administrative procedures are relied on to limit risk, and EOOS may reflect the external event condition using a surrogate such as increased loss of offsite power frequency.

In response to RAI 9.ii and RAI 9.iii, the licensee explained the CRMP model for evaluating the signals and functions in the LAR. The licensee stated that for reactor trip signals modeled in the PRA, the function is modeled from the sensors/detectors through the reactor trip and bypass breakers. For ESFAS signals modeled, components are analyzed from the sensors/detectors through the slave relays or the output steps of the engineered safeguard features loading sequencer. VCSNS will assign conservative surrogate events in EOOS for non-modeled signals as described in the response to RAI 9.iii; or, alternatively, VCSNS will explicitly model the inputs in lieu of using surrogates.

According to the response to RAI 9.ii, for analog channels placed in trip, the licensee uses a multiple of initiating event frequencies. The increase in initiator frequency during these tests is performed for all tests that trip bistables which input to a safety injection signal or a reactor trip signal, regardless of whether the signal being tested is explicitly modeled in the CRMP. The CRMP models SSPS actuation logic and master relay testing by removing the affected train's

SSPS cabinet from service which prevents taking credit for that train's RPS/ESFAS mitigation functions whether or not the input signals are modeled. For master relays, this approach is also used and results in removing credit for additional functions other than those impacted by the relay, which is a conservative approach.

The NRC staff finds that the licensee's program to control risk is capable of adequately assessing the activities being performed to ensure that high-risk plant configurations do not occur and/or compensatory actions are implemented if a high-risk plant configuration or condition should occur. As such, the licensee's program provides for the assessment and management of increased risk during maintenance activities as required by the Maintenance Rule (Section (a)(4)) and satisfies the RG 1.177 guidelines for a CRMP for the proposed change.

3.2.3 Limitations and Conditions

The licensee's evaluation of the five NRC staff SER conditions and limitations are discussed below.

Condition No. 1: A licensee is expected to confirm the applicability of the topical report to the plant and perform a plant-specific assessment of containment failures and address any design or performance differences that may affect the proposed changes.

To address the applicability of WCAP-15376, the licensee provided three tables in Attachment 5 of the LAR. Table 1 compares the general plant parameters with the parameters in the WCAP. The results show that all the parameters meet the guidance of WCAP. Table 2 compares the applicability of reactor trip actuation signals, and concludes that all the reactor trip signals agree with the approved WCAP. Table 3 compares the applicability of ESFAS signals and concludes that all the ESFAS signals agree with the approved WCAP. Based on the above, the licensee concluded that impact of implementation of the WCAP is acceptable per the guidance of Regulatory Guide 1.174.

Applicability of the master relay and safeguards driver card failure probabilities

The component failure probabilities developed as part of WCAP-15376 are applicable to VCSNS. For Solid State Protection System plants, this includes the master relay and safeguards driver card failure probabilities. The failure probabilities for these components are based on data collected from a number of Westinghouse nuclear steam supply system plants. The failure probabilities are:

Master Relays: 1.1E-05
Safeguards Driver Cards: 5.9E-04

A summary of the experience for these components at VCSNS from 2009 to 2013 is provided in the Table below.

Table 4: Summary of Actuation and Failure Experience on the Safeguards Driver Cards and Master Relays		
Parameter	Safeguards Driver	Master Relays
Actuations	254	1572
Failures	0	0

An analysis based on the binomial distribution was used to determine the number of expected failures for the given failure probabilities and actuations. For both components, either 0 or 1 failures would be expected. Based on the data provided in Table 4, it is concluded that the failure probabilities for these components used in the WCAP analysis are applicable to VCSNS.

Applicability of the containment failure assessment

The WCAP-15376 analysis and determination of LERF is based on a large dry containment. The containment building at VCSNS is a large dry containment; therefore, the NRC staff finds the WCAP-15376 containment failure assessment is applicable to VCSNS.

Based on the above and the NRC staff evaluation of Tier 1 presented in Section 3.2.2 of this SE, the staff finds Condition No. 1 to be met.

Condition No. 2: Address the Tier 2 and Tier 3 analyses including risk significant configuration insights and confirm that these insights are incorporated into the plant-specific configuration risk management program.

Based on the evaluation presented in Section 3.2.2 of this SE, the NRC staff finds Condition No. 2 to be met.

Condition No. 3: The risk impact of concurrent testing of one logic train and associated reactor trip breaker needs to be evaluated on a plant-specific basis to ensure conformance with the WCAP-15376 evaluation, and Regulatory Guides 1.174 and 1.177.

The risk impact of concurrent testing of one logic train and the associated reactor trip breaker (RTB) is addressed by demonstrating that the WCAP-15376 analysis is applicable to VCSNS. The WCAP analysis assumes that if a RTB is out of service, its associated logic train is also out of service. Limitations for various configurations when the RTB is removed from service are included in Tier 2 and Tier 3 assessments. Tier 2 requirements provide reasonable assurance that risk-significant plant equipment outage configurations will not occur when equipment is out of service. These requirements place limitations on additional equipment that can be removed from service during one of the risk-informed extended CTs. Tier 3 ensures that risk significant out-of-service equipment is evaluated prior to performing any maintenance activities. Tier 3 evaluations are addressed by the plant's Configuration Risk Management Program used to comply with 10 CFR 50.65(a)(4). Therefore, concurrent testing is addressed in the licensee's analysis. VCSNS testing and analysis is consistent with the WCAP approach.

WCAP-15376 did not specifically evaluate or preclude concurrent testing of one logic cabinet and associated RTB. The NRC staff questioned the applicability of the topical report to this particular maintenance configuration. In response to NRC staff requests for additional information (RAIs) on WCAP-15376, the PWROG provided risk estimates for this more limiting configuration. The ICCDP and ICLERP estimates were within the guidelines of RG 1.177. Since these incremental risk metrics are met for a 30-hour maintenance time, they are also met for a 4-hour bypass time. In the LAR Section 4.4, the licensee stated that concurrent testing is addressed in the WCAP analysis and that the VCSNS testing is consistent with this approach. The licensee showed that the generic analysis presented in WCAP-15376 is applicable to VCSNS.

In response to RAI 7.k.iii for plant-specific functions 6.h and 8.a, the licensee stated that there is no relation between RTB unavailability and the ACOT. The RTBs remain available throughout these operational tests. However, the VCSNS SSPS actuation logic and master relay testing verifies the operability of the RTBs, RTS, and ESFAS in a single test. The surveillance tests all of the ESFAS and RTS signals, not only the functions 6.h and 8.a. The reactor trip bypass breaker is racked in so that the function of the reactor trip breaker is unavailable during the test. This test is included in the licensee's evaluation of the cumulative ΔCDF and ΔLERF, which were found to meet RG 1.174 guidance.

Based on the applicability of WCAP-15376 to VCSNS, and the fact that plant-specific signal assessments are expected to be within the acceptance guidelines of RG 1.174 and RG 1.177, the NRC staff finds Condition No. 3 to be met.

Condition No. 4: To ensure consistency with the reference plant, the model assumptions for human reliability in WCAP-15376-P, Rev. 0 should be confirmed to be applicable to the plant- specific configuration.

In LAR Attachment 5, the licensee identified that the backup operator action to trip the reactor by interrupting power to the motor generator sets may not always be effective due to the short time available for all plant conditions. The licensee determined that not crediting this operator action was offset by the increased reliability estimate for RTBs from an updated study, NUREG/CR-6928, "*Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants*" (ADAMS Accession No. ML070650650). Due to the updated RTB data, the anticipated transient without scram (ATWS) frequency with no credit for this operator action was smaller than the topical report ATWS frequency with credit for the operator action. Therefore, the NRC staff finds that the plant-specific ATWS CDF remains less than that in the topical report (TR) so that there is no impact on its applicability to the TR.

The licensee concluded that the human reliability associated with the relevant operator actions in the TR are applicable to VCSNS based on a plant-specific assessment, and confirm that plant procedures are in place for the operator actions credited in the WCAP analysis. In response to follow up RAI 7.k.ii (supplement dated June 22, 2017), the licensee described backup operator actions and procedures for the automatic actions associated with plant-specific functions 6.h and 8.a. The licensee stated that the PRA model does not credit backup operator actions for functions 6.h and 8.a.

The NRC staff finds that Condition 4 is met because the licensee has confirmed the backup operator actions are a success path or provided acceptable justification, procedures are in place for the action, and backup operator actions for the plant-specific signals are not credited in the PRA model.

Condition No. 5: For future digital upgrades with increased scope, integration, and architectural differences beyond those of Eagle 21, the staff finds that the generic applicability of WCAP-15376-P, Rev. 0, to future digital systems is not clear and should be considered on a plant-specific basis.

The licensee determined that this condition is not applicable because there are presently no plans to implement digital upgrades to the Reactor Protection or ESF Systems at VCSNS at this time. The NRC staff finds Condition No. 5 is not applicable.

WCAP-15376 Response to a Question Regarding Setpoints

An additional commitment from the response to NRC RAI requires each plant to review their setpoint calculation methodology to determine the impact of extending the [Analog] Channel Operational Test (ACOT) Surveillance Frequency from 92 days to 184 days.

The licensee addressed this by stating the following in the December 16, 2015, LAR:

> The response to this RAI in Reference 5 noted that plant-specific RTS and ESFAS setpoint uncertainty calculations and assumptions, including instrument drift, will be reviewed to determine the impact of extending the Surveillance Frequency of the Channel Operational Test (COT) from 92 days to 184 days.
>
> The VCSNS personnel reviewed "as found" and "as left" data for the RTS and ESFAS setpoints for a 24-month period and concluded that sufficient margin is present to offset the change in drift anticipated as a result of increasing the operational test surveillance frequencies to 184 days (semi-annual). Based on the licensee's review of this data, the allowable margin present in the setpoints is more than adequate to offset the predicted increase in uncertainty/drift resulting from the increased interval between operational tests.
>
> While SCE&G does not anticipate any impact in going from 92 days to 184 days, VCSNS will trend the "as found" and "as left" data for the three representative trip functions analyzed in WCAP-15376-P-A (Over temperature Delta-T, Steam Generator Level, and Pressurizer Pressure) for two years (four operational tests) after implementation of the amendment granting the semi-annual operational tests.

Based on the above statement, the licensee will trend the data to identify any unanticipated impacts from the extended internal; therefore, the NRC staff finds that the licensee's response is acceptable.

3.3 Summary

3.3.1 Proposed Changes

Proposed Change 1:

> TS 3/4.3.1, Table 3.3-1 - Action 8: The proposed change for the AOT is from 1 hour to 24 hours. In addition, the Reactor Trip Breaker bypass test time is relaxed from 2 hours to 4 hours. Also, due to the extension of the 1 hour AOT to 24 hours and the time allowed to bypass one channel is being extended from 2 hours to 4 hours, the

last provisions of Action 8 allowing additional time for maintenance and an extended bypass time have been deleted.

The changes noted above are consistent with the changes approved in WCAP-15376, Revision 1. Based on the NRC staff's previous approval of WCAP-15376, the NRC staff finds that the proposed changes are acceptable.

Proposed Changes 2, 9, and 10:

> TS 3/4.3.1, Table 4.3.1 - Analog Channel Operational Test (ACOT): The proposed change to the ACOT is from Quarterly (92 days) to Semi-Annually (184 days) for the following functions in Table 4.3-1 of the VCSNS TSs: 2 (High Setpoint only), 3, 7, 8, 9, 10, 11, 12, 13, and 14. The proposed change to the ACOT frequency is changed to 184 days for the following functions in Table 4.3-1 of the VCSNS TSs: 2 (low Setpoint only), 5, and 6 by the addition of Notes 16, 17, and 18.
>
> Trip Actuating Device Operational Test (TADOT): The proposed change to the TADOT is from Quarterly to Semi-Annually for the following functions in Table 4.3-1 of the VCSNS TSs: 15 and 16. The description for each of the above functions is in Table 4.3-1 of the current technical specifications.

All of the above changes except for the changes to functions 15 and 16 pertaining to TADOT are directly based on approved WCAP-15376 for the analog trip channels and are therefore acceptable. The changes proposed to TADOT functions 15 and 16 are related to reactor coolant pumps undervoltage and underfrequency, which are not analog trip channels.

The subject of increasing the TADOT time was reviewed and accepted in WCAP-10271, Supplement 1-P-A dated July 1983. This report states the following on page 12:

> Reactor Coolant Pump undervoltage and underfrequency
>
> The reactor coolant pump undervoltage and underfrequency functional units are not analog channels. These functions are relays which change state ("drop out" or open circuit) when the voltage or frequency are not appropriate to hold the relay in-state. However, the output of these units and the output of the analog channels are fed into the RTS [Reactor Trip System] actuation logic in the same manner. Also, these units were included in the unavailability calculation models of the WCAP and of the staff along with the analog channels. Therefore, the approvals made by the staff in this SER [safety evaluation report] for analog channels also apply to these functional units.

The NRC staff reviewed and accepted the analog channel surveillance test interval from one month to three months in WCAP-10271-P-A. The NRC staff's safety evaluation included the change from one month to 3 months for undervoltage and underfrequency surveillance requirements.

Because the applicable TADOT frequencies were justified to be extended from 1 month to 3 months in WCAP-10271-P-A and its supplements, and the changes that were justified in WCAP- 14333-P-A, Revision 1, and WCAP-15376, Revision 1, are applicable to all of the signals included in WCAP-10271-P-A and its supplements, the extension of the above listed TADOT Frequencies from 92 days to 184 days was also justified by WCAP-15376. This is stated in Section 11 of WCAP-15376, Revision 1, as "These recommendations are applicable to all the signals evaluated in WOG TOP [Technical Specifications Optimization Program] for both solid state and relay protection systems ..." (i.e., all signals evaluated in WCAP-10271-P-A and its supplements). Therefore, the extension of the TADOT Frequencies from 92 days to 184 days justified in WCAP-14333 and WCAP-15376 are applicable to the RTS Functions 15 and 16 listed above. The NRC staff reviewed the explanation provided by the licensee and finds it is acceptable based on the intent of WCAP-14333 and WCAP-15376.

Based on the explanation described above, the NRC staff agrees that the TADOT tests for reactor trip on reactor coolant pump undervoltage and underfrequency can be conducted every 6 months. Therefore, the NRC staff finds the proposed changes are acceptable.

Proposed Change 3:

> TS 3/4.3.1, Table 4.3-1 –TADOT: The proposed change to the TADOT for the Reactor Trip Breaker (Function 20) is from Monthly (62 days on a Staggered Test Basis) to every 2 months (124 days on a Staggered Test Basis) as shown in revised Note 7. Note because of the difference in the Staggered Test Basis definition (discussed above) the number of days used for the VCSNS surveillance frequency is larger than the number used in TSTF-411 surveillance frequency.

This change is consistent with the approved WCAP-15376 which requires a TADOT every 62 days on a staggered test basis in SR 3.3.1.4. Testing of each channel every 62 days on a staggered test basis and testing of two channels (in two equal intervals) of 62 days is the same. Therefore, the NRC staff finds the proposed changes are acceptable.

Proposed Change 4:

> TS 3/4.3.1, Table 4.3-1 – TADOT: The proposed change to the TADOT for the Reactor Trip Bypass Breaker (Function 22) is from Monthly (62 days on a Staggered Test Basis) to every 2 months (124 days on a Staggered Test Basis) as shown in revised Note 7 (62 days to 124 days), which is being inserted into the TADOT column. Note that this change is different than what is shown in the markups for TSTF-411. This change is necessary because TSTF-411 does not provide a change to the Reactor Trip Bypass Breaker as the Bypass Breaker is treated as a part of the Reactor Trip Breaker function in the standard TS in NUREG-1431 and has the same surveillance frequency assigned. In the VCSNS TS, the Reactor Trip Breakers and the Reactor Trip Bypass Breakers are separate. Functions consistent with the standard TS in NUREG-0452. In the VCSNS TS, the two separate Functions are assigned the same frequency specified in Note 7 to be consistent with the change for the corresponding NUREG-1431 Reactor Trip Breaker Function in TSTF-411. Thus, both the VCSNS Reactor Trip Breakers and Reactor Trip Bypass Breakers will be tested at the same surveillance frequency consistent with NUREG-1431 and TSTF-411.

TSTF-411 does not provide specific guidance for the Reactor Trip Bypass Breakers because the Bypass Breakers are treated as a part of the RTBs. The NRC staff finds it acceptable to use the same surveillance requirements for the Reactor Trip Bypass Breakers as the RTBs. Because the same surveillance frequencies are used for both the RTBs and the Reactor Trip Bypass Breakers, the NRC staff finds the proposed changes are acceptable.

Proposed Change 5:

> TS 3/4 3.1, Table 4.3-1 - Actuation Logic Test (ALT): The proposed change to the ALT for the Automatic Trip Logic (Function 21) is from Monthly to Quarterly on a Staggered Test Basis as specified in new Note 15 (184 days).

The frequency of the proposed test is in compliance with the approved WCAP-15376 and TSTF-411. Therefore, the NRC staff finds the proposed changes are acceptable.

Proposed Change 6:

> TS 3/4.3.2, Table 4.3-2 – ACOT: The proposed change to the ACOT is from Quarterly to Semi-Annually for the following functions in Table 4.3-2 of the VCSNS TSs: 1.c, 1.d, 1.e, 1f, 2.c, 3.b.2, 4.c, 4.d, 4.e, 5.a, 6.c, 6.h, 8.a, 9.a, and 9.b (see Attachment 1 TS mark-up for the title of each Function).

The analog channel surveillance test interval has been changed from 3 months to 6 months, which is consistent with TSTF-411 and approved in WCAP-15376. Therefore, NRC staff finds the proposed changes are acceptable.

Proposed Change 7:

> TS 3/4.3.2, Table 4.3-2 – ALT: The proposed change to the ALT is from Monthly to Quarterly on a Staggered Test Basis as shown in the revised Note 1 (62 days to 184 days) for the following functions in Table 4.3-2 of the VCSNS TSs: 1.b, 2.b, 3.a.3, 3.b.1, 3.c.1, 4.b, 5.b, 6.b, and 8.b (see Attachment 1 TS mark-up for the title of each function).

The frequency of the proposed test is consistent with TSTF-411 and approved in WCAP-15376. Therefore, NRC staff finds the proposed changes are acceptable.

Proposed Change 8:

> TS 3/4.3.2, Table 4.3-2 - Master Relay Test: The proposed change to the Master Relay Test is from Monthly to Quarterly on a Staggered Test Basis as shown in the revised Note 1 (62 days to 184 days) for the following functions in Table 4.3-2 of the VCSNS TSs: 1.b, 2.b, 3.a.3, 3.b.1, 3.c.1, 4.b, 5.b, 6.b, and 8.b (see Attachment 1 TS mark-up for the title of each function).

For Solid State Protection Systems (SSPS), the Master Trip Relay surveillance test interval has been changed from 3 months to 6 months consistent with TSTF-411 and approved in WCAP-15376. As clarified in the licensee letter dated March 7, 2016 (ML16069A021), VCSNS uses SSPS for RTS as well as for ESFAS. VCSNS employs SSPS for both RPS and ESFAS. Therefore, NRC staff finds the proposed changes are acceptable.

3.3.2 TSTF-411 Changes Not Incorporated Into VCSNS TS

The licensee provided the statement below for the TSTF-411 changes that were not incorporated into VCSNS TS:

> <u>TS 3/4 3.1, Table 4.3-1 - TADOT for the following RTS Function:</u>
> Manual Reactor Trip (Function 1). The current VCSNS TSs require this Function to have a TADOT performed once per Refueling. This TADOT is also required to independently verify the undervoltage and shunt trip circuits' Operability. The current licensing basis frequency has proven adequate to ensure this Function performs as designed. Therefore, the TSTF-411 change to a TADOT once every 62 days on a Staggered Test Basis (ITS) is not being incorporated into the VCSNS TSs at this time; and
>
> <u>TS 3/4 3.1, Table 4.3-1 - ALT for the following RTS Function:</u>
> Reactor Trip System Interlocks - P-7 (Function 19.B). The current VCSNS TSs require this Function to have a Channel Calibration and ACOT performed once per Refueling. The current licensing basis frequency has proven adequate to ensure this Function performs as designed. Therefore, the TSTF-411 change to an ALT once every 92 days on a Staggered Test Basis (ITS) is not being incorporated into the VCSNS TSs at this time.

The licensee stated that it has not changed the existing TS format to ITS format and the surveillance frequencies have not been changed. The NRC staff finds that there is no requirement to implement the TSTF-411 in its entirety because the changes can be evaluated independently. The NRC staff finds the licensee's position acceptable.

TSTF-411 surveillance frequency changes that are not proposed for VCSNS are:

- ALT and Master Relay Test for the Control Room Emergency Filtration System - ESFAS
- Channel Operation Test for the Boron Dilution Protection System - ESFAS
- Channel Operation Test for the Steam Line Isolation on Steam Line Pressure Negative Rate - High ESFAS function
- Channel Operation Test for the Automatic Switchover to Containment Sump on RWST Level - Low - Low Coincident with Safety Injection and Containment Sump Level - High ESFAS function

The NRC staff finds the licensee's position is acceptable because the above surveillance frequencies are not applicable to VCSNS TSs.

3.4 NRC Staff Conclusion

The NRC staff concludes that the application meets GDC 13, 21, and 22 and the requirements in 10 CFR 50.36 and 10 CFR 50.65. The NRC staff concludes the licensee has demonstrated the applicability of WCAP-15376 to VCSNS. The NRC staff found the risk impacts for ΔCDF, ΔLERF, ICCDP, and ICLERP as estimated by WCAP-15376 to be applicable to VCSNS and the plant-specific function contribution to be within the acceptance guidelines for RG 1.174 and RG 1.177. The licensee showed the applicability of the specified functional units to the topical report evaluations and results, and performed plant-specific function analyses. The licensee's Tier 2 analysis evaluated concurrent outage configurations and confirmed the applicability of the risk-significant configurations identified by the staff SER limitations and conditions and topical report analysis to ensure control of these configurations. The NRC staff found the licensee's Tier 3 CRMP to be consistent with the RG 1.177 CRMP guidelines and the Maintenance Rule for the implementation of WCAP-15376. The licensee monitors the reliability and availability of the RTS and ESFAS instrumentation under the Maintenance Rule (Section (a)(1)). Therefore, the NRC staff concludes the TS revisions proposed by the licensee are consistent with the STI, CT, and bypass extensions approved for WCAP-15376 and meet the staff's SER conditions and limitations as outlined in WCAP-15376.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendment on August 23, 2017. The State official 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, and changes surveillance requirements. 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 April 12, 2016 (81 FR 21601), 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: October 4, 2017

SUBJECT: VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: TSTF-411, REVISION 1, "SURVEILLANCE TEST INTERVAL EXTENSIONS FOR COMPONENTS OF THE REACTOR PROTECTION SYSTEM (WCAP-15376-P-A)." (CAC NO. MF7196) DATED OCTOBER 4, 2017

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