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

May 31, 2016

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P.O. Box 1295
Bin 038
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS RE: TECHNICAL SPECIFICATION END STATES (CAC NOS. MF6197 AND MF6198)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 179 to Renewed Facility Operating License No. NPF-68 and Amendment No. 160 to Renewed Facility Operating License No. NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 6, 2015, as supplemented by letters dated October 8, 2015, and May 9, 2016. The changes revise certain TS Required Actions to permit a Required Action end state of MODE 4 instead of MODE 5.

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,

Bob Martin

Bob Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 179 to NPF-68
2. Amendment No. 160 to NPF-81
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 179
Renewed License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility), Renewed Facility Operating License No. NPF-68, filed by Southern Nuclear Operating Company, Inc. (the licensee), dated May 6, 2015, as supplemented by letters dated October 8, 2015, and May 9, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-68, is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protecton Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 179, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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

Attachment:
Changes to the Facility Operating
License and Technical Specifications

Date of Issuance: May 31, 2016

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 160
Renewed License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility), Renewed Facility Operating License No. NPF-81, filed by Southern Nuclear Operating Company, Inc. (the licensee), dated May 6, 2015, as supplemented by letters dated October 8, 2015, and May 9, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

Enclosure 2

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-81 is hereby amended to read as follows:

    (2) Technical Specifications and Environmental Protection Plan

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 160, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: May 31, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 179

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND LICENSE AMENDMENT NO. 160

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

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
NPF-68, page 4	NPF-68, page 4
NPF-81, page 3	NPF-81, page 3
TSs	TSs
3.3.2-1	3.3.2-1
3.3.2-2	3.3.2-2
3.3.2-5	3.3.2-5
3.4.13-1	3.4.13-1
3.4.14-2	3.4.14-2
3.4.15-3	3.4.15-3
3.4.15-4	3.4.15-4
3.4.15-5	3.4.15-5
3.5.4-2	3.5.4-2
3.5.6-1	3.5.6-1
3.6.6-1	3.6.6-1
3.7.7-1	3.7.7-1
3.7.8-1	3.7.8-1
3.7.9-2	3.7.9-2
3.7.10-3	3.7.10-3
3.7.10-4	3.7.10-4
----	3.7.10-5
3.7.11-3	3.7.11-3
---	3.7.11-4
3.7.13-1	3.7.13-1
3.7.13-2	3.7.13-2
3.8.1-6	3.8.1-6
3.8.4-2	3.8.4-2
3.8.4-3	3.8.4-3
3.8.7-1	3.8.7-1
3.8.9-2	3.8.9-2

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 179, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the environmental Protection Plan.

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

(4) Deleted

(5) Deleted

(6) Deleted

(7) Deleted

(8) Deleted

(9) Deleted

(10) Mitigation Strategy License Condition

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

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

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

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

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

(4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

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

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

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter 1 and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 160 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

3.3 INSTRUMENTATION

3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

LCO 3.3.2 The ESFAS instrumentation for each Function in Table 3.3.2-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2 1.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels inoperable.	A.1 Enter the Condition referenced in Table 3.3.2-1 for the channel(s) or train(s).	Immediately
B. One channel inoperable.	B.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.1 Be in MODE 3.	54 hours
	<u>AND</u>	
	B.2.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering MODE 4. ---------- Be in MODE 4.	60 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One train inoperable.	------------------NOTE------------------ One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. --	
	C.1 Restore train to OPERABLE status.	24 hours
	<u>OR</u>	
	C.2.1 Be in MODE 3.	30 hours
	<u>AND</u>	
	C.2.2 -------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 4. --------------------------------- Be in MODE 4.	36 hours
D. One channel inoperable.	------------------NOTE------------------ A channel may be bypassed for up to 12 hours for surveillance testing. --	
	D.1 Place channel in trip.	72 hours
	<u>OR</u>	
	D.2.1 Be in MODE 3.	78 hours
	<u>AND</u>	
	D.2.2 Be in MODE 4.	84 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. One channel inoperable.	------------------NOTE------------------ A channel may be bypassed for up to 12 hours for surveillance testing. --	
	I.1 Place channel in trip.	72 hours
	<u>OR</u>	
	I.2 Be in MODE 3.	78 hours
J. One Main Feedwater Pumps trip channel inoperable.	J.1 Restore channel to OPERABLE status.	48 hours
	<u>OR</u>	
	J.2 Be in MODE 3.	54 hours
K. One RWST Level - Low Low channel inoperable.	------------------NOTE------------------ One additional channel may be bypassed for up to 12 hours for surveillance testing. --	
	K.1 Place channel in bypass.	72 hours
	<u>OR</u>	
	K.2.1 Be in MODE 3.	78 hours
	<u>AND</u>	
	K.2.2 ------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 4. ----------------------------------	
	Be in MODE 4.	84 hours

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1	Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 AND B.2	Be in MODE 3. -------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 4. --------------------------------- Be in MODE 4.	6 hours 12 hours

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	AND		
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	-------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------------------------------	
		Be in MODE 4.	12 hours
C. RHR System suction isolation valve interlock function inoperable.	C.1	Isolate the affected penetration by use of one closed manual or deactivated automatic valve.	4 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Required containment atmosphere radioactivity monitor inoperable. AND Required containment air cooler condensate flow rate monitor inoperable.	E.1	Restore required containment atmosphere radioactivity monitor to OPERABLE status.	30 days
	OR		
	E.2	Restore required containment air cooler condensate flow rate monitor to OPERABLE status.	30 days
---------------NOTE------------- Only applicable when a containment atmosphere gaseous radiation monitor is the only OPERABLE monitor. ------------------------------------- F. Required containment sump monitors inoperable. AND Required containment air cooler condensate flow rate monitor inoperable.	F.1	Analyze grab samples of the containment atmosphere.	Once per 12 hours
	AND		
	F.2.1	Restore required containment sump monitors to OPERABLE status.	7 days
	OR		
	F.2.2	Restore required containment air cooler condensate flow rate monitor to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Required Action and associated Completion Time not met.	G.1	Be in MODE 3.	6 hours
	AND		
	G.2	-------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------------------------------	
		Be in MODE 4.	12 hours
H. All required leakage detection systems inoperable.	H.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of containment normal sumps level and reactor cavity sump level monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.2	Perform CHANNEL CHECK of the required containment atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.3	Perform COT of the required containment atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.15.4	Perform CHANNEL CALIBRATION of the containment sump monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.5	Perform CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor.	In accordance with the Surveillance Frequency Control Program
SR 3.4.15.6	Perform CHANNEL CALIBRATION of the required containment air cooler condensate flow rate monitor.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Required Action and associated Completion Time of Condition A or D not met.	E.1 Be in MODE 3. AND E.2 -------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1	-------------------------NOTE------------------------- Only required to be performed when ambient air temperature is < 40°F. -- Verify RWST borated water temperature is ≥ 44°F and ≤ 116°F.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.2	Verify RWST borated water volume is ≥ 686,000 gallons.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2400 ppm and ≤ 2600 ppm.	In accordance with the Surveillance Frequency Control Program
SR 3.5.4.4	Verify each sludge mixing pump isolation valve automatically closes on an actual or simulated RWST Low-Level signal.	In accordance with the Surveillance Frequency Control Program

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.6 Recirculation Fluid pH Control System

LCO 3.5.6 The Recirculation Fluid pH Control System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Recirculation Fluid pH Control System inoperable.	A.1	Restore system to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------------ Be in MODE 4.	54 hours

3.6 CONTAINMENT SYSTEMS

3.6.6 Containment Spray and Cooling Systems

LCO 3.6.6 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One containment spray train inoperable.	A.1	Restore containment spray train to OPERABLE status.	72 hours
B. One containment cooling train inoperable.	B.1	Restore containment cooling train to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	C.2	-------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 4. --------------------------------- Be in MODE 4.	12 hours

3.7 PLANT SYSTEMS

3.7.7 Component Cooling Water (CCW) System

LCO 3.7.7 Two CCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One CCW train inoperable.	A.1	-------------NOTE------------ Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by CCW. ---------------------------------	
		Restore CCW train to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	-------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 4. ---------------------------------	
		Be in MODE 4.	12 hours

3.7 PLANT SYSTEMS

3.7.8 Nuclear Service Cooling Water (NSCW) System

LCO 3.7.8 Two NSCW trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One NSCW train inoperable.	------------------NOTES------------------ 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for emergency diesel generator made inoperable by NSCW system. 2. Enter applicable Conditions and Required Actions of LCO 3.4.6, "RCS Loops - MODE 4," for residual heat removal loops made inoperable by NSCW system. --- A.1 Restore NSCW system to OPERABLE status.	 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 ------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------- Be in MODE 4.	6 hours 12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One NSCW basin transfer pump inoperable.	D.1	Restore the transfer pump to OPERABLE status.	8 days
	OR		
	D.2.1	Implement an alternate method of basin transfer.	8 days
	AND		
	D.2.2	Restore the transfer pump to OPERABLE status.	31 days
E. Required Action and associated Completion Time not met.	E.1	Be in MODE 3.	6 hours
OR	AND		
UHS inoperable for reasons other than Conditions A, B, C, or D.	E.2	------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------	
		Be in MODE 4.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. Required Action and associated Completion Time of Condition A, B, or D not met.	F.1	-----------NOTE----------- Required Action F.1 is not applicable when entering this Condition from Condition B or D. -------------------------------	
		Lock closed the outside air (OSA) intake dampers of the affected unit and lock open the OSA intake dampers of the unaffected unit.	1 hour
	AND		
	F.2	Place the affected units(s) in MODE 3.	7 hours
	AND		
	F.3	-----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 4. -------------------------------	
		Place the affected unit(s) in MODE 4.	13 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Required Action and associated Completion Time of Condition C or E not met.	G.1 ------------NOTE------------ Required Action G.1 is not applicable when entering this Condition from Condition E. -------------------------------	
	Lock closed the outside air (OSA) intake dampers of the affected unit and lock open the OSA intake dampers of the unaffected unit.	1 hour
	AND	
	G.2 Place the affected units(s) in MODE 3.	7 hours
	AND	
	G.3 Place the affected unit(s) in MODE 5.	37 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.10.1	Verify control room air temperature ≤ 85°F.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.2	Operate each CREFS train for ≥ 10 continuous hours with the heater control circuit energized.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.3	Perform required CREFS filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.10.4	Verify each CREFS train actuates (switches to emergency mode) on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.10.5	Perform required CRE unfiltered air inleakage testing in accordance with the Control Room Envelope Habitability Program.	In accordance with the Control Room Envelope Habitability Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. One or more CREFS trains inoperable due to inoperable CRE boundary.	F.1	Initiate action to implement mitigating actions.	Immediately
	AND		
	F.2	Verify mitigating actions ensure CRE occupant exposures to radiological, chemical, and smoke hazards will not exceed limits.	24 hours
	AND		
	F.3	Restore CRE boundary to OPERABLE status.	90 days
G. Control room air temperature not within limit.	G.1	Restore control room air temperature to within limit.	7 days
H. Required Action and associated Completion Time of Condition A, B, C, or F not met for operating unit.	H.1	Place the unit in MODE 3.	6 hours
	AND		
	H.2	----------NOTE---------- LCO 3.0.4.a is not applicable when entering MODE 4. ------------------------------ Place the unit in MODE 4.	12 hours
I. Required Action and associated Completion Time of Condition D, E, or G not met for operating unit.	I.1	Place the unit in MODE 3.	6 hours
	AND		
	I.2	Place the unit in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	The Surveillance Requirements of Specification 3.7.10 are applicable.	In accordance with applicable SRs.

3.7 PLANT SYSTEMS

3.7.13 Piping Penetration Area Filtration and Exhaust System (PPAFES)

LCO 3.7.13 Two PPAFES trains shall be OPERABLE.

-----NOTE-----
The PPAFES boundary may be opened intermittently under administrative control.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PPAFES train inoperable.	A.1 Restore PPAFES train to OPERABLE status.	7 days
B. Two PPAFES trains inoperable due to inoperable PPAFES boundary.	B.1 Restore PPAFES boundary to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering MODE 4. ----- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.13.1	Operate each PPAFES train for ≥ 15 minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.7.13.2	Perform required PPAFES filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP
SR 3.7.13.3	Verify each PPAFES train actuates on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program
SR 3.7.13.4	Verify one PPAFES train can maintain a negative pressure ≥ 0.250 inches water gauge relative to atmospheric pressure during the post accident mode of operation at a flow rate of 15,500 cfm ± 10%.	In accordance with the Surveillance Frequency Control Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.1	Restore required offsite circuit to OPERABLE status.	12 hours
	<u>OR</u>		
	E.2	Restore DG to OPERABLE status.	12 hours
F. Two DGs inoperable.	F.1	Restore one DG to OPERABLE status.	2 hours
G. One automatic load sequencer inoperable.	G.1	Restore automatic load sequencer to OPERABLE status.	12 hours
H. Required Action and associated Completion Time of Condition A, C, D, E, F, or G not met. OR Required Action B.1, B.3, B.4.1, B.4.2, or B.6 and associated Completion Time not met.	H.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	H.2	-------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------------------------------- Be in MODE 4.	12 hours
I. Three or more required AC sources inoperable.	I.1	Enter LCO 3.0.3.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One DC electrical power source inoperable for reasons other than Condition A or B.	C.1 Restore DC electrical power source to OPERABLE status.	2 hours
D. Required Action and Associated Completion Time not met.	D.1 Be in MODE 3. AND D.2 -------------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 4. --------------------------------- Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is greater than or equal to the minimum established float voltage.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.8.4.2	Verify the battery charger supplies: ≥ 400 amps for System A and B ≥ 300 amps for System C, and ≥ 200 amps for System D at greater than or equal to the minimum established float voltage for ≥ 8 hours for Systems A and B and ≥ 3 hours for Systems C and D. <u>OR</u> Verify each battery charger can recharge the battery to the fully charged state within 12 hours while supplying the largest combined demands of the various continuous steady state loads, after a battery discharge to the bounding design basis event discharge state.	In accordance with the Surveillance Frequency Control Program
SR 3.8.4.3	---------------------------NOTES--------------------------- 1. The modified performance discharge test in SR 3.8.6.6 may be performed in lieu of the service test in SR 3.8.4.3. 2. This Surveillance shall not be performed in MODE 1, 2, 3, or 4. However, credit may be taken for unplanned events that satisfy this SR. --- Verify battery capacity is adequate to supply, and maintain in OPERABLE status, the required emergency loads for the design duty cycle when subjected to a battery service test.	In accordance with the Surveillance Frequency Control Program

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters – Operating

LCO 3.8.7 The required Class 1E 120 V inverters shall be OPERABLE.

-----NOTE-----
Two inverters may be disconnected from their associated DC bus for ≤ 24 hours to perform an equalizing charge on their associated common battery, provided:

a. The associated AC vital bus(es) are energized from their Class 1E regulating transformers; and

b. All other AC vital buses are energized from their associated OPERABLE inverters.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required inverter inoperable.	-----NOTE----- Enter applicable conditions and required actions of LCO 3.8.9 "Distribution Systems - Operating" with any vital bus deenergized. ----------	
	A.1 Restore inverter to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 -----NOTE----- LCO 3.0.4.a is not applicable when entering MODE 4. ----------	
	Be in MODE 4.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more DC electrical power distribution subsystems inoperable.	C.1	Restore DC electrical power distribution subsystems to OPERABLE status.	2 hours
D. Required Action and associated Completion Time not met.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2	-------------NOTE------------- LCO 3.0.4.a is not applicable when entering MODE 4. ---------------------------------	
		Be in MODE 4.	12 hours
E. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.9.1	Verify correct breaker alignments and voltage to required AC, DC, and AC vital bus electrical power distribution subsystems.	In accordance with the Surveillance Frequency Control Program

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 179 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-68

AND

AMENDMENT NO. 160 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By application dated May 6, 2015 (Reference 1), supplemented by letters dated October 8, 2015 (Reference 2), and May 9, 2016 (Reference 3), Southern Nuclear Operating Company, Inc. (SNC, the licensee) proposed changes to the Technical Specifications (TSs) for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2. Specifically, the licensee proposed to adopt U.S. Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications (STS) Change Traveler (TSTF)-432, Revision 1, "Change in Technical Specifications End States (WCAP-16294)," dated November 29, 2010 (Reference 4). By letter of May 9, 2016, SNC withdrew its proposed change to TS 3.7.14, "Engineered Safety Features (ESF) Room Cooler and Safety Related Chiller System." The supplemental letters dated October 8, 2015, and May 9, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 23, 2015 (80 FR 35983).

TSTF-432 incorporates the NRC-approved Nuclear Energy Institute (NEI) Topical Report (TR) WCAP-16294-NP-A, Revision 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS [Nuclear Steam Supply System] PWRs [Pressurized Water Reactors]" (Reference 5), into NUREG-1431, Revision 3, "Standard Technical Specifications Westinghouse Plants" (Reference 6). The licensee stated that its license amendment request (LAR) is consistent with the Notice of Availability of TSTF-432 announced in the *Federal Register* on May 11, 2012 (77 FR 27814).

TSTF-432 is one of the industry initiatives developed under the Risk Management Technical Specifications program. The purpose of risk-informed TS changes is to maintain or improve

safety, while reducing unnecessary burden, and to make TS requirements consistent with the Commission's other risk-informed regulatory requirements.

The Westinghouse STS define the following six operational modes. Of specific relevance to TSTF-432 are Modes 4 and 5:

- Mode 1 - Power operation. The reactor is critical and thermal power is greater than 5 percent of the rated thermal power.
- Mode 2 – Startup. The reactor is critical and thermal power is ≤ 5 percent of the rated thermal power
- Mode 3 - Hot standby. The reactor is subcritical and the average reactor coolant system (RCS) temperature is ≥ 350 degrees Fahrenheit (°F).
- Mode 4 - Hot shutdown. The reactor is subcritical and the average RCS temperature is greater than 200 °F and less than 350 °F. The reactor vessel head closure bolts are fully tensioned.
- Mode 5 - Cold shutdown. The reactor is subcritical and the average RCS temperature is less than or equal to 200°F. The reactor vessel head closure bolts are fully tensioned.
- Mode 6 – Refueling. The reactor in this mode is shut down and one or more reactor vessel head closure bolts are less than fully tensioned.

TR WCAP-16294 identifies and evaluates new TS Required Action end states for a number of TS Limiting Conditions for Operation (LCOs), using a risk-informed approach, consistent with Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 7), and RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (Reference 8). An end state is a condition that the reactor must be placed in if the TS Required Action(s) cannot be met. The end states are currently defined based on placing the unit into a mode or condition in which the TS LCO is not applicable. Mode 5 is the current end state for LCOs that are applicable in Modes 1 through 4. The risk of the transition from Mode 1 to Modes 4 or 5 depends on the availability of alternating current (AC) sources. During the realignment from Mode 4 to Mode 5, there is an increased potential for loss of shutdown cooling and loss of inventory events. Decay heat removal following a loss-of-offsite power event in Mode 5 is dependent on alternating current (AC) power for shutdown cooling, whereas in Mode 4, the potential exists to utilize the turbine driven auxiliary feedwater (AFW) pump for decay heat removal.

Therefore, transitioning to Mode 5 is not always the appropriate end state from a risk perspective. Thus, for specific TS conditions, TR WCAP-16294 justifies Mode 4 as an acceptable alternate end state to Mode 5. The proposed change to the TSs would allow time to perform short-duration repairs, which currently necessitate exiting the original mode of applicability. The Mode 4 TS end state is applied, and risk is assessed and managed in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.65, "Requirements for Monitoring the Effectiveness of

Maintenance at Nuclear Power Plants." Modified end states are limited to conditions where: (1) entry into the shutdown mode is for a short interval, (2) entry is initiated by inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable TS, and (3) the primary purpose is to correct the initiating condition and return to power operation as soon as is practical.

1.1 Proposed Actions

As summarized in the table below, the requested TS changes would permit an end state of hot shutdown (Mode 4) rather than an end state of cold shutdown (Mode 5) for the following TS action requirements:

Proposed Changes To End States	
TS & Condition	**Title**
3.3.2-B 3.3.2-C 3.3.2-K	Engineered Safety Feature Actuation System (ESFAS) Instrumentation
3.4.13-B	RCS Operational Leakage
3.4.14-B	RCS PIV [Pressure Isolation Valve] Leakage
3.4.15-G	RCS Leakage Detection Instrumentation
3.5.4-E	Refueling Water Storage Tank (RWST)
3.5.6-B	Recirculation Fluid pH Control System
3.6.6-C	Containment Spray and Cooling Systems
3.7.7-B	Component Cooling Water (CCW) System
3.7.8-B	Nuclear Service Cooling Water (NSCW) System
3.7.9-E	Ultimate Heat Sink (UHS)
3.7.10-F	Control Room Emergency Filtration System (CREFS) – Both Units Operating
3.7.11-H	Control Room Emergency Filtration System (CREFS) – One Unit Operating
3.7.13-C	Piping Penetration Area Filtration and Exhaust System (PPAFES)
3.8.1-H	AC Sources – Operating
3.8.4-D	DC [Direct Current] Sources – Operating
3.8.7-B	Inverters – Operating
3.8.9-D	Distribution Systems – Operating

This LAR is limited to inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable TS, and the primary purpose is to correct the inoperable component(s) and return to power operation as soon as is practical.

2.0 REGULATORY EVALUATION

Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licenses to include TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are contained in 10 CFR 50.36, "Technical specifications." Pursuant to 10 CFR 50.36(c), the TSs are required to include items

in the following specific categories related to plant operation: (1) safety limits, limiting safety systems settings, and limiting control settings; (2) LCOs; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation at 10 CFR 50.36(c)(2) states, in part, "When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

The regulation at 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," requires that the reactor must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents (LOCAs) conforms to the criteria set forth in 10 CFR 50.46(b).

A majority of the TSs and design-basis analyses were developed under the perception that putting a plant in cold shutdown would result in the safest condition, and the design-basis analyses would bound credible shutdown accidents. In the late 1980s and early 1990s, the NRC and licensees recognized the potential significance of events occurring during shutdown conditions, and guidance was issued to improve shutdown operation. Since enactment of a shutdown rule was expected, almost all TS changes involving power operation, including a revised end state requirement, were postponed. However, in the mid-1990s, the Commission decided a shutdown rule was not necessary in light of industry improvements.

Controlling shutdown risk encompasses control of conditions that can cause potential initiating events and responses to those initiating events that do occur. Initiating events are a function of equipment malfunctions and human error. Responses to events are a function of plant sensitivity, ongoing activities, human error, defense-in-depth (DID), and additional equipment malfunctions.

The regulation at 10 CFR 50.65(a)(4) requires that, "Before performing maintenance activities ... the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety." RG 1.182, Revision 0, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants" (Reference 9), which was referenced in the published model safety evaluation (SE) for TSTF-432, provides guidance on implementing the provisions of 10 CFR 50.65(a)(4) by endorsing Section 11 to Nuclear Management and Resources Council (NUMARC) 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated February 22, 2000 (Reference 10).

RG 1.182 was withdrawn subsequent to the issuance of the model SE in November of 2012, and its subject matter has been incorporated into RG 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated May 2012 (Reference 14). RG 1.160 endorses NUMARC 93-01, Revision 4A, dated April 2011 (Reference 15). As stated in RG 1.160, the NRC staff finds that this revision of NUMARC 93-01 provides methods acceptable to the NRC for complying with the provisions of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

The LAR states that the licensee assesses risk of maintenance activities in a manner consistent with RG 1.160, and is, therefore, consistent with the original reference to RG 1.182 in the model SE.

Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," provides, in part, the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

- Criterion 38, "Containment heat removal," requires the provision of a containment heat removal system that will rapidly reduce containment pressure and temperature following any LOCA and maintain them at acceptably low levels. The containment heat removal system supports the containment function by minimizing the duration and intensity of the pressure and temperature increase following a LOCA, thus, lessening the challenge to containment integrity.

- Criterion 41, "Containment atmosphere cleanup," requires that systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

RG 1.174 (Reference 7) describes a risk-informed approach, acceptable to the NRC, for assessing the nature and impact of proposed permanent licensing-basis changes by considering engineering issues and applying risk insights. RG 1.174 also provides risk acceptance guidelines for evaluating the results of such evaluations.

RG 1.177 (Reference 8) describes an acceptable risk-informed approach specifically for assessing proposed permanent allowed outage time and surveillance test interval TS changes, and it also provides risk acceptance guidelines for evaluating the results of such assessments. In addition, RG 1.177 identifies a three-tiered approach for the licensee's evaluation of the risk associated with a proposed Completion Time TS change. Per RG 1.177, the improved STS use the terminology "completion times" and "surveillance frequency" in place of "allowed outage time" and "surveillance test interval."

General guidance for evaluating the technical basis for proposed risk-informed changes is provided in Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," of the NRC Standard Review Plan (SRP), NUREG-0800 (Reference 11). Guidance on evaluating probabilistic risk assessment (PRA) technical adequacy related to risk-informed TS changes is provided in SRP Section 16.1, "Risk-Informed Decisionmaking: Technical Specifications" (Reference 12), which includes Completion Time changes as part of risk-informed decisionmaking.

3.0 TECHNICAL EVALUATION

The changes proposed in TSTF-432 are consistent with the changes proposed and justified in TR WCAP-16294, which was approved by the NRC in an SE on March 29, 2010 (Reference 13). Specifically, end states are prescribed in the TSs when Required Actions are not met. The current TS actions require placing the plant in cold shutdown (Mode 5) based on the expectation that this condition would result in the safest condition, since most design-basis accidents (DBAs) and transients either cannot physically occur during shutdown or would have significantly reduced plant impact and occur much less frequently due to the reduced temperatures and pressures in the plant. Accidents and transients unique to shutdown conditions were anticipated to be of less significance compared to the design-bases events applicable to power operation.

The requested change to the TSs is to allow a Mode 4 end state rather than a Mode 5 end state for selected TS LCO actions. TR WCAP-16294 provides a comparative qualitative assessment of the availability of plant equipment for decay heat removal and accident mitigation in Modes 4 and 5 and considers the likelihood and consequences of initiating events, which may occur in these modes. A quantitative risk assessment of operation in these modes, including the risk associated with the transition from Mode 4 to Mode 5, and then back to Mode 4 to support the return to service, is also provided using a shutdown and transition PRA model developed to support the review of TR WCAP-16294.

TR WCAP-16294 concludes that the availability of steam generator (SG) heat removal capability in Mode 4, and the avoidance of transitioning the plant to and from shutdown cooling, makes Mode 4 the preferred end state over Mode 5 for each of the TS conditions proposed to be changed. This conclusion is further supported by quantitative risk analyses that demonstrate a reduction in plant risk by remaining in Mode 4 compared to the alternative of transitioning to and from Mode 5, in accordance with the existing TS requirements.

Both the qualitative and quantitative analyses of TR WCAP-16294 support a Mode 4 end state. This conclusion is primarily due to the availability of SG cooling in Mode 4 via the turbine driven AFW pump, which is not reliant upon AC power, as compared to the use of shutdown cooling in Mode 5, which requires the availability of AC power. Further, the transition risks associated with establishing shutdown cooling alignments and the resulting potential for loss of inventory or loss of cooling events due to human error during such alignments are avoided by being in Mode 4.

This general assessment is applied as the basis for changing the required end state from Mode 5 to Mode 4 for those TSs which govern plant equipment not included in the PRA models and is supported by qualitative assessments of the plant impact of the unavailability of the TS equipment. For those TSs covering plant equipment included in the PRA models, a quantitative risk assessment is also provided that assesses the comparative risk of completing repairs in Mode 4 or proceeding to Mode 5 for repairs and then returning to Mode 4 for plant startup, while considering the available equipment for accident mitigation.

Changing the required end state to Mode 4 will also result in increased plant availability by decreasing the shutdown time. The additional time required to transition to Mode 5 from Mode 4 when shutting down and also to Mode 4 from Mode 5 when restarting can be eliminated with

the end state change. A typical time for the transition from Mode 4 to Mode 5 during shutdown and from Mode 5 to Mode 4 during startup is 24 hours. Therefore, this change would allow an availability increase of approximately 24 hours.

Changing the end states would allow continued operation with the LCO not met by removing the TS requirement to exit the LCO Applicability. In this case, the requirements of LCO 3.0.4.a would apply unless otherwise stated. LCO 3.0.4.a allows entry into a mode or other specified condition in the Applicability with the LCO not met when the associated Actions to be entered permit continued operation in the mode or other specified condition in the Applicability for an unlimited period of time. Compliance with Required Actions that permit continued operation of the unit for an unlimited period of time in a mode or other specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the unit before or after the mode change. Therefore, in such cases, entry into a mode or other specified condition in the Applicability may be made in accordance with the provisions of the Required Actions.

Thus, implementing modified end states would require adding a Note to the affected Required Actions to prevent using the allowance of LCO 3.0.4.a when entering Mode 4 from Mode 5. This is done to avoid unit operation in a condition that should be prohibited by TS since LCO 3.0.4.a allows entry into a mode or other specified condition in the Applicability when the associated Actions to be entered permit continued operation in the mode or other specified condition in the Applicability for an unlimited period of time. Applying the allowance of LCO 3.0.4.a to modified end states was not analyzed in TR WCAP-16294; therefore, the NRC staff finds that an appropriate limitation is applied by the addition of a Note to the affected TS Required Actions stating that LCO 3.0.4.a is not applicable when entering Mode 4 from Mode 5.

3.1 Technical Analysis

This section provides the NRC staff evaluation of the impact of each proposed end state change on DID and safety margins as applied to the corresponding safety systems. The NRC staff's evaluation applies only to the proposed changes to the TSs as described below. Consistent with its March 29, 2010, SE, the NRC staff finds that the TR WCAP-16294 used realistic assumptions regarding the plant conditions and the availability of various mitigating systems in analyzing the risks and considering the DID and safety margins. Thus, the NRC staff concludes that TR WCAP-16294 uses realistic assumptions to justify the proposed changes in the end state. However, during the proposed Mode 4 end state, due to the safety injection (SI) signal blockage and non-availability of accumulators, operator actions will be required to mitigate potential events.

During the proposed Mode 4 end state, risk is assessed and managed consistent with 10 CFR 50.65. The NRC staff's review is based on the knowledge of lower RCS pressure in Mode 4, which reduces the severity of a LOCA, and limits any coolant inventory loss in the event of a LOCA.

3.1.1 Proposed Required Actions

The proposed changes would add a Note stating that, "LCO 3.0.4.a is not applicable when entering Mode 4," to each Required Action listed in the table below. In general, the end state for

each Required Action shown in the table below would be is revised to be in Mode 4 instead of in Mode 5. The following table provides: (1) the TS number and title, (2) the Required Action to be revised, (3) the current end state and Completion Time, and (4) the proposed end state and Completion Time.

Proposed Changes to End States			
TS Number and Title	**TS Required Action**	**Current End State and Completion Time**	**Proposed End State and Completion Time**
3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions: 1.a, 2.a, and 3.a	B.2.2	Mode 5 in 84 hours	Mode 4 in 60 hours
3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation Functions: 1.b, 2.b, 3.b, and 7.a	C.2.2	Mode 5 in 60 hours	Mode 4 in 36 hours
3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation Function: 7.b	K.2.2	Mode 5 in 108 hours	Mode 4 in 84 hours
3.4.13 RCS Operational Leakage	B.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.4.14 RCS Pressure Isolation Valve (PIV) Leakage	B.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.4.15 RCS Leakage Detection Instrumentation	G.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.5.4 Refueling Water Storage Tank (RWST)	E.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.5.6 Recirculation Fluid pH Control System	B.2	Mode 5 in 84 hours	Mode 4 in 54 hours
3.6.6 Containment Spray and Cooling Systems	C.2	Mode 5 in 84 hours	Mode 4 in 12 hours
3.7.7 Component Cooling Water (CCW) System	B.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.7.8 Nuclear Service Cooling Water (NSCW) System	B.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.7.9 Ultimate Heat Sink (UHS)	E.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.7.10 Control Room Emergency Filtration System (CREFS) – Both Units Operating	F.3	Mode 5 in 37 hours	Mode 4 in 13 hours
TS Number and Title	**TS Required Action**	**Current End State and Completion Time**	**Proposed End State and Completion Time**
3.7.11 Control Room	H.2	Mode 5 in 36 hours	Mode 4 in 12 hours

Emergency Filtration System (CREFS) – One Unit Operating			
3.7.13 Piping Penetration Area Filtration and Exhaust System (PPAFES)	C.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.8.1 AC Sources – Operating	H.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.8.4 DC Sources – Operating	D.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.8.7 Inverters – Operating	B.2	Mode 5 in 36 hours	Mode 4 in 12 hours
3.8.9 Distribution Systems – Operating	D.2	Mode 5 in 36 hours	Mode 4 in 12 hours

3.1.2 Evaluation of Proposed Changes

TR WCAP-16294 does not address entry into Mode 4 from Mode 5 when the Required Actions are in effect. Such a mode change would be permissible since the proposed revised actions permit continued operation in Mode 4 for an unlimited period of time and, therefore, transitioning from Mode 5 to Mode 4 would be permissible using LCO 3.0.4.a. Since applying LCO 3.0.4.a to modified end states was not analyzed in TR WCAP-16294, an appropriate note is proposed to be added to each affected Required Action that denotes the provisions of LCO 3.0.4.a are not applicable to Mode 4 entry from Mode 5.

In some instances, the TSs proposed to be revised by the LAR had different TS numbers, titles, language, or content as compared to TSTF-432. If the NRC staff determined that these differences had an impact on the technical content of the amendment, an evaluation documenting the acceptability of the differences is included in this SE. The remaining differences that were deemed administrative in nature by the staff are considered acceptable and consistent with the intent of TSTF-432.

There are TS and LCO requirements that are revised by TSTF-432 but do not exist in the VEGP TSs. TSTF-432 modifies the end states for existing TS Required Actions; but it is not intended to add new TSs or LCO requirements that do not exist in a licensee's TSs. Based on this, the subject LAR, which represents a plant-specific adoption of TSTF-432, only proposes changing end states for existing VEGP Required Actions. These differences between the LAR and TSTF-432 are deemed administrative in nature by the staff and are, therefore, acceptable and consistent with the intent of TSTF-432.

In select cases, the licensee chose not to adopt the TSTF-432 changes. Since the changes specified by TSTF-432 relax existing requirements in the TSs, not adopting the changes results in the TSs retaining the same level of conservatism. Also, each change specified by TSTF-432 is independent of the other changes; thus, by choosing not to adopt a change, the other changes will not be affected. The staff considers these differences to be administrative in nature and are, therefore, acceptable and consistent with the intent of TSTF-432.

A brief description of the systems and components covered by this LAR, along with the NRC staff's evaluation of the proposed changes to the VEGP TSs, are provided below. The plant-specific system descriptions are based on the VEGP TS Bases, Revision 33, which were submitted to the NRC under oath and affirmation on April 10, 2015 (Reference 18).

3.1.2.1 TS 3.3.2 Engineered Safety Feature Actuation System (ESFAS) Instrumentation

ESFAS initiates necessary safety systems based on the values of selected unit parameters (1) to protect against violating core design limits and the RCS pressure boundary and (2) to mitigate accidents. The ESFAS instrumentation functions are listed in TS Table 3.3.2-1, and those relevant to the LAR are described below:

Function 1.a Safety Injection - Manual Initiation
Function 1.b Safety Injection - Automatic Actuation Logic and Actuation Relays

SI system: The SI system provides two primary functions: (1) primary side water addition to ensure maintenance or recovery of reactor vessel water level (covering the active fuel for heat removal, clad integrity, and for limiting the peak clad temperature to < 2200 °F) and (2) boration to ensure recovery and maintenance of shutdown margin (keff < 1.0). These functions mitigate the effects of high energy line breaks both inside and outside of containment.

Manual initiation causes actuation of all components in both trains in the same manner as any of the automatic actuation signals. The automatic actuation logic and actuation relays must be operable in Mode 4 to support system level manual initiation.

These functions in VEGP TS 3.3.2 require two channels for manual initiation and two trains for automatic actuation logic, and actuation relays shall be operable in Modes 1, 2, 3, and 4.

Function 2.a Containment Spray - Manual Initiation
Function 2.b Containment Spray - Automatic Actuation Logic and Actuation Relays

Containment spray system: The containment spray system provides two primary functions: (1) lowers containment pressure and temperature after a high energy line break in containment and (2) reduces the amount of radioactive iodine in the containment atmosphere. These functions are necessary to ensure the containment structure pressure boundary integrity and limit the radioactive iodine release to the environment in the event of failure of the containment structure.

The operator can initiate containment spray in the control room by simultaneously turning the two containment spray actuation hand-switches from the same channel resulting in both trains of containment spray starting. Two switches are used to prevent inadvertent actuation of containment spray.

In Mode 4, adequate time is available to manually actuate required components in the event of a DBA. However, because of the large number of components actuated, actuation is simplified by the use of the manual actuation hand-switches. Automatic actuation logic, and actuation relays must be operable in Mode 4 to support system level manual initiation.

These functions in VEGP TS 3.3.2 require two channels for manual initiation and two trains for automatic actuation logic and actuation relays shall be operable in Modes 1, 2, 3, and 4.

Function 3.a Phase A Containment Isolation - Manual Initiation
Function 3.b Phase A Containment Isolation - Automatic Actuation Logic and Actuation Relays

Phase A containment isolation (CI) system: The Phase A CI signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity from containment in the event of a DBA. Phase A CI is actuated automatically by SI or manually via the automatic actuation logic and actuation relays.

These functions in VEGP TS 3.3.2 require two channels for manual initiation and two trains for automatic actuation logic, and actuation relays shall be operable in Modes 1, 2, 3, and 4.

Function 7.a Semi-Automatic Switchover to Containment Sump - Automatic Actuation Logic and Actuation Relays

Semi-automatic switchover to containment sump: At the end of the injection phase of an LOCA when the RWST is nearly empty and continued cooling must be provided by the ECCS to remove decay heat, the source of water for the ECCS pumps is switched to the containment recirculation sump by a combination of automatic and manual actions. This switchover must occur before the RWST empties to prevent damage to the residual heat removal (RHR) pumps and a loss of cooling capability. Switchover must not occur before there is sufficient water in the containment sump to support engineered safety feature (ESF) pump suction. Also, early switchover must not occur to ensure that sufficient borated water is injected from the RWST.

There are two trains for the automatic actuation logic and actuation relays. This function consists of the same features and operates in the same manner as described in Function 1.b. The LCO for this function requires that two trains shall be operable in Modes 1, 2, and 3, and in Mode 4, only one train is required to be operable to support semi-automatic switchover for the single RHR pump that is required by TS 3.5.3, "ECCS - Shutdown."

Evaluation of SI, Containment Spray, Phase A CI, and Semi-Automatic Switchover to Containment Sump

Functions 1.a, 2.a, and 3.a require two channels to be operable, and Functions 1.b, 2.b, and 3.b require two trains to be operable. For Function 1.a, if one channel is inoperable, the other channel is available to initiate SI. For Function 1.b, if one train is inoperable, the other train is available to initiate SI if the associated ECCS train is operable, in accordance with TS 3.5.3, "ECCS – Shutdown." For Functions 2.a and 2.b, if one channel or train is inoperable, the other channel or train is available to initiate containment spray. In addition, the containment, CI valves, containment spray system, and the containment cooling system are available and required to be operable in Mode 4. For Functions 3.a and 3.b, if one channel or train is inoperable, the other channel or train is available to initiate Phase A CI. In addition, the CI valves, containment spray system, and the containment cooling system are available in Mode 4. For Function 7.a, if one train is inoperable in Modes 1 through 3, the other train is available to initiate switchover to the containment sump. In addition, the operator can perform the switchover manually in Modes 1 through 4.

A cooldown to Mode 4 leaves the unit in a state in which transients progress slower than at power, backup core cooling is available via RHR, there is increased time for operator actions and mitigation strategies, and there is a lower overall risk than proceeding to Mode 5. Placing the unit in Mode 5 does not increase the instrumentation available for event mitigations, and, therefore, there is no benefit with regard to monitoring plant status by proceeding to Mode 5. Sufficient DID is maintained when the end state is changed from Mode 5 to Mode 4. In addition, the NRC staff anticipates that equipment repairs requiring plant shutdown and entry into Mode 4 would be infrequent events of short duration. Therefore, the NRC staff concludes that the proposed change is acceptable.

<u>Function 7.b Semi-Automatic Switchover to Containment Sump - Refueling Water Storage Tank (RWST) Level - Low Low, Coincident with Safety Injection</u>

Semi-automatic switchover to containment sump: During the injection phase of an LOCA, semi-automatic switchover from the RWST to the containment sump occurs only if the RWST low low level signal is coincident with an SI signal. This prevents accidental switchover during normal operation. The semi-automatic switchover results in the containment sump to RHR pump suction valves opening automatically, which is then followed by operator manual actions to complete the switchover.

The RWST has four level transmitters. The logic requires two out of four channels to initiate the semi-automatic switchover from the RWST to the containment sump. The LCO for this function requires that four channels shall be operable in Modes 1, 2, 3, and 4.

The licensee identified a deviation from TSTF-432 concerning Function 7.b. Specifically, Condition K for VEGP TS 3.3.2, the condition for an inoperable RWST Level – Low Low channel, has the same Required Actions but different Completion Times as compared to TSTF-432, which is based on Revision 3 of NUREG-1431. TSTF-432 specifies placing the affected channel in bypass within 6 hours or transitioning to Mode 3 within 12 hours, followed by Mode 5 in 42 hours, as compared to the VEGP Completion Times of 72 hours, 78 hours, and 108 hours, respectively. The current VEGP Completion Times, which are consistent with Revision 4 to NUREG-1431 (Reference 16), were revised in a previous license amendment (Reference 17) that was based on NRC-approved TR WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS [Reactor Protection System] and ESFAS Test Times and Completion Times."

The NRC staff concludes that the deviation is acceptable since (1) current VEGP Completion Times for Condition K represent previously approved changes that reflect the current revision of the STS, (2) the proposed change is consistent with TSTF-432 by providing an additional 6 hours to be in Mode 4 after the time requirement to be in Mode 3, and (3) the justification provided by the TSTF-432 model SE for changing this end state is still applicable and encompasses the deviation.

Evaluation of the Semi-Automatic Switchover to Containment Sump – Refueling Water Storage Tank (RWST) Level - Low Low, Coincident with Safety Injection

If one channel is inoperable, the other three channels are available to initiate the semi-automatic switchover to the containment sump. Placement of the unit in Mode 5 does not increase the instrumentation available for event mitigation; therefore, there is no benefit with regard to monitoring plant status by proceeding to Mode 5.

The redundancy is such that a single channel failure and one channel being inoperable will not defeat the initiation of the semi-automatic switchover from the RWST to the containment sump. As a result, sufficient DID is maintained when the end state is changed from Mode 5 to Mode 4. In addition, the NRC staff anticipates that equipment repairs requiring plant shutdown and entry into Mode 4 would be infrequent events of short duration. Therefore, the NRC staff concludes the proposed change is acceptable.

3.1.2.2 TS 3.4.13 RCS Operational Leakage

The safety significance of RCS operational leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring RCS leakage into the containment area is necessary. A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100 percent leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, so as not to interfere with RCS leakage detection.

TS LCO 3.4.13 pertains to the protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling and, additionally, to preventing the accident analyses radiation release assumptions from being exceeded.

In Modes 1, 2, 3, and 4, RCS operational leakage shall be limited to:

a. No pressure boundary leakage;

b. 1 gallon per minute (gpm) unidentified leakage;

c. 10 gpm identified leakage; and

d. 150 gallons per day primary to secondary leakage through any one steam generator (SG).

NRC Staff Evaluation:

RCS leakage that is not large enough to be a small-break LOCA should be treated as an event leading to a controlled shutdown that is not modeled in the quantitative risk analysis.

In Mode 4, the RCS pressure is significantly reduced, which results in reduced leakage. All LOCA mitigating systems, with the exception of the accumulators, are available, and RHR serves as the backup to AFW for decay heat removal. If RCS operational leakage is not within limits for reasons other than pressure boundary leakage or primary-to-secondary leakage, then

the leakage must be reduced to within limits in 4 hours, consistent with Required Action A.1. If operational leakage is not restored to within limits in 4 hours, pressure boundary leakage exists, or primary-to-secondary leakage is not within the limit, then Required Actions B.1 and B.2 become applicable. The proposed Required Actions B.1 and B.2 require that the unit be placed in Mode 3 within 6 hours and Mode 4 within 12 hours. Thus, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequence. In addition, the NRC staff anticipates that equipment repairs requiring plant shutdown and entry into Mode 4 would be infrequent events of short duration. Therefore, the NRC staff concludes the proposed change to revise Required Action B.2, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.3 <u>TS 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage</u>

The regulations at 10 CFR 50.2, "Definitions," and 10 CFR 50.55a(c), "Codes and standards," define RCS pressure isolation valves (PIVs) as any two normally closed valves in series within the RCPB, which separate the high pressure RCS from an attached low pressure system. The RCS PIV leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance. A known component of the identified leakage before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing. Leakage measured through one PIV in a line is not RCS operational leakage if the other is leaktight.

The main purpose of this specification is to prevent overpressure failure of the low pressure portions of the connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. The failure consequences could be an LOCA outside of containment, an unanalyzed accident that could degrade the ability for low pressure injection.

TS LCO 3.4.14 requires RCS PIV leakage to be within limits in Modes 1, 2, and 3, and within limits in Mode 4, with the exception of valves in the RHR flow path, when in or during the transition to or from the RHR mode of operation.

In Modes 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for an LOCA outside containment.

<u>NRC Staff Evaluation</u>:

TS 3.4.14 limits RCS leakage because of the concern of over-pressurization of a lower pressure system that can lead to an interfacing system LOCA. In Mode 4, the RCS pressure is significantly reduced, which reduces the PIV leakage. All LOCA mitigating systems, with the exception of the accumulators, are available, and RHR serves as the backup to AFW for decay heat removal. Therefore, sufficient DID is maintained when the end state is changed from Mode 5 to Mode 4. In addition, the NRC staff anticipates that equipment repairs requiring plant shutdown and entry into Mode 4 would be infrequent events of short duration. Therefore, the

NRC staff concludes the proposed change to revise Required Action B.2, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.4 TS 3.4.15 RCS Leakage Detection Instrumentation

General Design Criterion 30 of Appendix A to 10 CFR Part 50, "Quality of reactor coolant pressure boundary," requires means for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. Leakage detection systems must have the capability to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified RCS leakage.

This LCO requires that the following RCS leakage detection instrumentation shall be operable in Modes 1, 2, 3 and 4:

a. The containment normal sumps level and reactor cavity sump monitors,

b. One containment atmosphere radioactive monitor (gaseous or particulate), and

c. Either the containment air cooler condensate flow rate or a containment atmosphere gaseous or particulate radioactivity monitoring system not taken credit for in item b.

The licensee identified a deviation from TSTF-432 concerning VEGP TS 3.4.15. Specifically, VEGP TS 3.4.15 includes conditions that reflect combinations of inoperable instrumentation not included in the STS. Even with the differences, the existing and proposed end state Completion Times for this TS are in agreement with TSTF-432, and the justification provided by the TSTF-432 model SE for changing this end state bounds the deviation. Therefore, the NRC staff concludes the deviation is acceptable.

NRC Staff Evaluation:

If one function is inoperable, the other functions are available to provide an indication of RCS leakage. In the unlikely event that Condition H (all required monitors inoperable) occurs, the likelihood of an initiating event in Mode 4 is not higher than in Mode 5. Placing the unit in Mode 5 does not increase the instrumentation available for detecting RCS leakage; therefore there is no benefit with regard to monitoring plant status by proceeding to Mode 5. Sufficient DID will be maintained when the end state is changed from Mode 5 to Mode 4. In addition, the NRC staff anticipates that equipment repairs requiring plant shutdown and entry into Mode 4 would be infrequent events of short duration. Therefore, the NRC staff concludes the proposed change to revise Required Action G.2, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.5 TS 3.5.4 Refueling Water Storage Tank (RWST)

The RWST supplies borated water to the chemical and volume control system during abnormal operating conditions to the refueling pool during refueling and to the ECCS and the containment spray system during accident conditions. The RWST supplies both trains of the ECCS and the

containment spray system through separate, redundant supply headers during the injection phase of an LOCA recovery.

During normal operation in Modes 1, 2, and 3, the SI and RHR pumps are aligned to take suction from the RWST. LCO 3.5.4 requires that the RWST shall be operable in Modes 1, 2, 3, and 4.

The licensee identified a deviation from TSTF-432 concerning VEGP TS 3.5.4. Specifically, VEGP TS 3.5.4 includes conditions that reflect combinations of inoperable isolation valves for the sludge mixing equipment that are not included in the STS. Even with the differences, the existing and proposed end state Completion Times for this TS are in agreement with TSTF-432, and the justification provided by the TSTF-432 model SE for changing this end state bounds the deviation. Therefore, the NRC staff concludes the deviation is acceptable.

NRC Staff Evaluation:

Since SI and recirculation may not be available due to an inoperable RWST, any loss of inventory events that cannot be isolated can lead to core damage. From Table 3.2.1, in the final SE of WCAP-16294, remaining in Mode 4 instead of cooling down to Mode 5 reduces the Core Damage Probability (CDP) by more than a factor of three. The primary accidents such as LOCAs and steam line breaks (SLBs) are less likely to occur in Mode 4. Since control rods are inserted in Mode 4, the SLB analysis assumption of the highest worth rod stuck is an unlikely scenario. In the lower part of Mode 4 transients progress slower than at power, backup cooling is available via RHR and there is increased time for operator action and mitigation strategies. Proceeding to Mode 5 may add additional risk by switching from AFW cooling to RHR cooling. Based on Table 3.2.1 in the final SE of WCAP-16294 a shutdown to Mode 4 is appropriate if the RWST is inoperable.

In Mode 4, the transient conditions are less severe than at power so that variations in the RWST parameters or other reasons of inoperability are less significant. In addition, if the boron concentration is low, the emergency boration equipment is likely to be available to increase the RCS boron concentration. By changing the end state for Required Action E.2 to Mode 4, the possibility of a loss of inventory event due to switching to RHR cooling is eliminated, reducing the possibility that the RWST inventory would be required. Therefore, sufficient DID is maintained when the unit remains in Mode 4 rather than transitioning to Mode 5. In addition, the NRC staff anticipates that equipment repairs requiring plant shutdown and entry into Mode 4 would be infrequent events of short duration. Therefore, the NRC staff concludes the proposed change to revise TS 3.5.4, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.6 TS 3.5.6.ECCS Recirculation Fluid pH Control System

The VEGP recirculation fluid pH control system is a passive system designed to raise the long-term pH of the solution in the containment sump following a DBA. The system consists of three storage baskets located in containment that contain crystalline trisodium phosphate (TSP) in the dodecahydrate form. In the event of an LOCA, the TSP contained in the storage baskets will be dissolved in the RCS and RWST inventories lost through the pipe break. The resulting increase in the recirculation solution pH into the range of 7.5 to 10.5 assures that iodine is retained in solution and that chloride induced stress corrosion is minimized.

The ECCS recirculation fluid pH control system is required to be operable in Modes 1, 2, 3, and 4 when a DBA could cause the release of radioactive material in containment requiring operation of the system. The system assists in reducing the amount of radioactive material available for release to the outside atmosphere after a DBA.

VEGP TS 3.5.6 currently requires the unit to be in Mode 3 in 6 hours and Mode 5 in 84 hours if the system is inoperable with the Required Action and associated Completion Time for restoring system operability not being met.

The licensee identified a deviation from TSTF-432 concerning VEGP TS 3.5.6. Specifically, this VEGP TS is not included in the STS. Although not included in the STS, both TR WCAP-16294, Revision 1, which is the basis for TSTF-432, and the TSTF-432 model SE address this system. TR WCAP-16294 states, "The justification for changing the end state to Mode 4 for TS 3.6.7, 'Spray Additive System,' is also applicable to the recirculation fluid pH control system, since they perform the same function." Therefore, VEGP TS 3.5.6 has been evaluated consistent with its inclusion in TR WCAP-16294 and the model SE, and the NRC staff concludes this deviation is acceptable.

<u>NRC Staff Evaluation</u>:

TR WCAP-16294 provides the technical basis for the proposed change by indicating that, "Events, such as an LOCA or a secondary side break, are less likely in Mode 4 due to the limited time in the mode and less severe thermal-hydraulic conditions. ... Therefore, sufficient defense-in-depth is maintained when the end state is changed from Mode 5 to Mode 4." TR WCAP-16294 further states that, "proceeding to Mode 5 does not increase the protection available," and it is highly unlikely that all of the baskets would be empty; therefore, an inoperable recirculation fluid pH control system would still provide some pH control.

In Mode 4, the RCS pressures and temperatures are lower, ECCS operation is maintained so the criteria of 10 CFR 50.46 are met, and the containment spray systems and containment cooling systems are available to depressurize and reduce the airborne radioiodine in containment. Therefore, the NRC staff concludes the proposed change to revise TS 3.5.6, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.7 <u>TS 3.6.6 Containment Spray and Cooling Systems</u>

The containment spray and cooling systems provide containment atmosphere cooling to limit post-accident pressure and temperature in containment to less than the design values. The containment spray system consists of two separate trains of equal capacity, each capable of meeting the design bases. Each train includes a containment spray pump, spray headers, nozzles, valves, and piping. Each train is powered from a separate emergency bus. The RWST supplies borated water to the containment spray system during the injection phase of operation. In the recirculation mode of operation, containment spray pump suction is transferred from the RWST to the containment sump.

Two trains of containment cooling, each of sufficient capacity to supply 100 percent of the design cooling requirement, are provided. Each train of four fan units is supplied with cooling

water from a separate train of NSCW. Air is drawn into the coolers through the fan and discharged to the SG compartments, pressurizer compartment, instrument tunnel, and outside the secondary shield in the lower areas of containment.

One train of containment spray and one containment cooling train are required to maintain the containment peak pressure and temperature below design limits in the event of a DBA. The LCO requires that two containment spray trains and two containment cooling trains shall be operable in Modes 1, 2, 3, and 4.

The licensee identified a deviation from TSTF-432 concerning VEGP TS 3.6.6. Specifically, the STS includes conditions that reflect combinations of containment spray and cooling inoperability that are not included in the VEGP TSs. Since the VEGP TSs do not include these conditions representing the different combinations of inoperable equipment, entrance into LCO 3.0.3 is required upon their occurrence. Even with the differences, the proposed end state Completion Time for this TS is in agreement with TSTF-432, and the justification provided in the TSTF-432 model SE for changing this end state bounds the deviation. Therefore, the NRC staff concludes the deviation is acceptable.

NRC Staff Evaluation:

The containment spray and containment cooling systems are designed for accident conditions initiated at full power. One train of each system satisfies the assumptions in the safety analyses. If one train of either containment spray or containment cooling is inoperable, the other train is available to mitigate the accident along with both trains of the other system. If two containment spray trains or two containment cooling units are inoperable, the plant must immediately enter LCO 3.0.3. The requirements of GDC 38 from 10 CFR Part 50, Appendix A, will still be met. Therefore, the NRC staff concludes the proposed change to revise TS 3.6.6, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.8 TS 3.7.7 Component Cooling Water (CCW) System
TS 3.7.8 Nuclear Service Cooling Water (NSCW) System

CCW system: The CCW system provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation, the CCW system also provides this function for the spent fuel storage pool. The CCW system serves as a barrier to the release of radioactive byproducts between potentially radioactive systems and the NSCW and, thus, to the environment.

The CCW system is arranged as two independent, full capacity cooling loops. Each safety-related train includes three 50 percent capacity pumps, a surge tank, a heat exchanger, piping, valves, and instrumentation. Each train is powered from a separate bus. An open surge tank in the system provides pump trip protective functions to ensure that sufficient net positive suction head is available. The pumps in each train are automatically started on receipt of an SI signal. Only two out of the available pumps are required to be operable. The third pump serves as a standby to allow maintenance.

The principal safety-related function of the CCW system is the removal of decay heat from the reactor via the RHR system. This may be during a normal or post-accident cooldown and

shutdown. The safety-related function is covered by TS LCO 3.7.7, which requires that two CCW trains shall be operable in Modes 1, 2, 3, and 4.

NSCW: The NSCW system consists of two separate, 100 percent capacity, safety-related, cooling water trains. Each train consists of three 50 percent capacity pumps, various safety and non-safety-related component heat exchangers, piping, valving, and instrumentation. The pumps and valves are remote and manually aligned, except in the unlikely event of an LOCA. The pumps are automatically started upon receipt of an SI signal, and all essential valves are aligned to their post-accident positions.

The NSCW system provides a heat sink for the removal of process and operating heat from safety-related components during a DBA or transient. During normal operation and a normal shutdown, the NSCW system also provides this function for various safety-related and non-safety-related components. The principal safety-related function of the NSCW system is the removal of decay heat from the reactor via the CCW system. The safety-related function is covered by TS LCO 3.7.8, which requires that two NSCW trains shall be operable in Modes 1, 2, 3, and 4.

NRC Staff Evaluation of the CCW and NSCW Systems

The CDP values listed in Table 3.2.1 of the final SE of TR WCAP-16294, from the evaluation for the scenarios, show that there is slightly less risk associated with Mode 4 than there is with a cooldown to Mode 5 when a train of CCW or NSCW is inoperable. One CCW or NSCW train will be operating when the unit enters Mode 4. Each train is designed to handle 100 percent of the heat loads during power operation and accident conditions. The heat loads will be significantly less in the shutdown modes, and some accidents are less likely to occur. As a result, sufficient DID is maintained when the end state is changed from Mode 5 to Mode 4. Therefore, the NRC staff concludes the proposed change to revise TS 3.7.7 Required Action B.2 and TS 3.7.8 Required Action B.2, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.9 TS 3.7.9 Ultimate Heat Sink (UHS)

The UHS provides a heat sink for processing and operating heat from safety-related components during a transient or accident, as well as during normal operation. This is done by utilizing the NSCW and the CCW systems.

The UHS consists of the NSCW system mechanical draft towers. Two 100 percent capacity redundant NSCW towers are provided for each unit. One tower is associated with each train of the NSCW system. Each NSCW tower consists of a basin that contains the UHS water supply and an upper structure that contains four individual fan spray cells where the heat loads are transferred to the atmosphere. The two principal functions of the UHS are the dissipation of residual heat after reactor shutdown and after an accident. TS LCO 3.7.9 requires that the UHS shall be operable in Modes 1, 2, 3, and 4.

The licensee identified a deviation from TSTF-432 concerning VEGP TS 3.7.9. Specifically, the VEGP TSs include conditions (e.g., low level, fan/spray cell inoperability, basin transfer pump inoperability) that are not included in the STS. Also, the STS include a condition for the UHS

temperature that is not reflected in the VEGP TSs. The end state Completion Times for the additional VEGP conditions reflect plant-specific attributes of the VEGP UHS design and are consistent with those evaluated in WCAP-16294 and provided in TSTF-432. The omitted Condition for the UHS temperature reflects plant-specific attributes of the VEGP design and licensing basis. Even with the differences, the existing and proposed end state Completion Times for this TS are in agreement with TSTF-432, and the justification provided in the TSTF-432 model SE for changing this end state bounds the deviation. Therefore, the NRC staff concludes the deviation is acceptable.

NRC Staff Evaluation:

TS 3.7.9 addresses degradations to the cooling capability of the UHS. The most likely scenario for entering Condition E is that the cooling capability of the UHS is only partially degraded. A cooldown to Mode 4 places the unit in a state where heat loads are significantly less than at full power.

The UHS is designed to remove 100 percent of the heat loads generated during power operation and accident conditions. The heat load will be significantly less in the shutdown modes. Some accidents are less likely to occur during shutdown modes. As a result, sufficient DID is maintained when the end state is changed from Mode 5 to Mode 4. Therefore, the NRC staff concludes the proposed change to revise Required Action E.2, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.10 TS 3.7.10 Control Room Emergency Filtration System (CREFS) – Both Units Operating

TS 3.7.11 Control Room Emergency Filtration System (CREFS) – One Unit Operating

CREFS provides a protected environment from which operators can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke. CREFS consists of four independent, redundant full capacity air filtration trains for the common Unit 1 and Unit 2 control room. Each train consists of carbon filter moisture eliminators, high efficiency particulate air (HEPA) filters, electric heaters, a cooling coil, and supply/return fans. The filter trains for Unit 1 and Unit 2 are powered from their respective 'A' and 'B' train safety feature buses.

Upon receipt of (1) a manual initiation signal or (2) a control room isolation signal due to an SI signal or a high radiation signal in the outside air intake, the normal air supply to the control room is isolated, and the stream of ventilation air is recirculated through the system filter trains. Also, outside air required for pressurization is mixed with the return air before it enters the filtration unit. Both TS LCO 3.7.10 and 3.7.11 require that four CREFS trains shall be operable in Modes 1, 2, 3, and 4.

The licensee identified deviations from TSTF-432 concerning VEGP TS 3.7.10 and 3.7.11. Specifically, the VEGP CREFS system is configured differently as compared to the STS system. The VEGP system has four trains (two trains per unit) protecting the common unit control room, which is governed by two separate TSs (TS 3.7.10, two units in the mode of applicability, versus TS 3.7.11, one unit in the mode of applicability). In contrast, the STS has one TS LCO (i.e.,

TS 3.7.10) representing the CREFS system, which only includes two trains. The result of the different equipment configuration is that the VEGP TSs 3.7.10 and 3.7.11 include additional Condition statements to reflect the increased redundancy as compared to the STS. The additional VEGP TS Condition statements affected by the proposed new end state and not associated with an inoperable control room envelope (CRE) boundary are bounded by TS LCO 3.7.10, Condition A (i.e., one CREFS train inoperable) from the STS. These conditions from VEGP TSs 3.7.10 and 3.7.11 are bounded because their Required Actions and Completion Times ensure that the CRE is protected by at least two full capacity CREFS trains in a single failure proof configuration within a Completion Time that is equivalent to the time required to restore an inoperable train in the STS. Both the VEGP TS and the STS are also equivalent from the standpoint that if the Required Action is not met within the Completion Time for the aforementioned conditions, then the operating unit(s) will be required to shut down. Since the TSTF-432 model SE and STS conditions bound the previously described deviations, the NRC staff concludes the deviations are acceptable.

Another deviation from TSTF-432 was identified by the licensee for the Required Actions and Completion Times associated with VEGP Condition D from TS LCO 3.7.10 and Condition F from TS LCO 3.7.11. Specifically, these conditions represent an inoperable CRE boundary. The differences are based on a previous license amendment that allowed the licensee to establish mitigating actions within 24 hours so that (1) CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences and (2) CRE occupant exposures to chemical and smoke hazards will not exceed limits. After mitigating actions are established within 24 hours, 90 days is allowed to restore CRE boundary operability before requiring a plant shutdown. In contrast, the STS allow 24 hours to restore operability before requiring a plant shutdown. Since this deviation is based on a previously approved amendment that is in agreement with Revision 4 to NUREG-1431 (Reference 16) and is bounded by the TSTF-432 model SE, the NRC staff concludes the deviation is acceptable.

The final deviation identified by the licensee pertains to Required Action F.2 from VEGP TS 3.7.10, which currently stipulates that the unit be placed in Mode 3 within 7 hours and which the LAR proposed to stipulate that the unit be placed in Mode 4 in 13 hours versus the STS requirement of 6 hours and 12 hours, respectively. Since (1) the 7-hour limit to transition to Mode 3 is the current approved time for VEGP and (2) the licensee added an additional 6 hours to be in Mode 4 as part of its LAR, which is consistent with the TSTF-432 time differentials of 6 hours between Modes 3 and 4, the NRC staff concludes this deviation is acceptable.

End state requirements for conditions from VEGP TSs 3.7.10 and 3.7.11 that are not bounded by TSTF-432 (i.e., control room temperature not within limit, two CREFS trains inoperable in one unit) were outside the scope of the LAR and, as a result, a new Condition statement was created to ensure that their current end state requirements remain the same, which the NRC staff concludes is acceptable.

NRC Staff Evaluation:

If one CREFS train on one unit is inoperable, the other train remains available to provide control room filtration. If all CREFS trains are inoperable, an independent initiating event and radioactive release must occur for filtration to be required in Modes 4 and 5. As a result, sufficient DID is maintained when the end state is changed from Mode 5 to Mode 4. Therefore,

the NRC staff concludes the proposed change to revise Required Action F.3 from TS LCO 3.7.10 and Required Action H.2 from TS LCO 3.7.11, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.11 TS 3.7.13 Piping Penetration Area Filtration and Exhaust System (PPAFES)

PPAFES maintains a negative pressure in the piping penetration area and ESF pump rooms and filters the exhaust from the negative pressure boundary. PPAFES minimizes the release of airborne radioactivity to the outside atmosphere resulting from recirculation line and component leakage into the piping penetration area ECCS and ESF pump rooms during an accident condition.

PPAFES consists of two independent and redundant trains. Each train consists of a heater, a prefilter or demister, a HEPA filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, and instrumentation, as well as demisters, functioning to reduce the relative humidity of the air stream, also form part of the system. A second bank of HEPA filters, which follows the adsorber section, collects carbon fines and provides backup in case of failure of the main HEPA filter bank. The downstream HEPA filter is not credited in the accident analysis. The system initiates filtered ventilation following receipt of a containment ventilation isolation signal.

PPAFES is a standby system, parts of which may also operate during normal unit operations. During emergency operations, the PPAFES dampers are realigned and fans are started to initiate filtration. Upon receipt of the actuating signal(s), the penetration room is isolated, and the stream of ventilation air discharges through the system filter trains. The prefilters remove any large particles in the air, as well as any entrained water droplets, to prevent excessive loading of the HEPA filters and charcoal adsorbers. TS LCO 3.7.13 requires that two PPAFES trains shall be operable in Modes 1, 2, 3, and 4.

The licensee identified a deviation from TSTF-432 concerning VEGP TS 3.7.13. Specifically, the TSTF-432 changes that encompass this TS are associated with TS LCOs 3.7.12, "Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)," and 3.7.14, "Penetration Room Exhaust Air Cleanup System (PREACS)," from the STS. Since (1) VEGP TS 3.7.13 is bounded by TSs 3.7.12 and 3.7.14 from the STS and (2) the existing and proposed end state Completion Times are the same, the NRC staff concludes the deviation is acceptable.

NRC Staff Evaluation:

If one PPAFES train is inoperable, the other train remains available to provide penetration and pump room air filtration. If two trains are inoperable due to an inoperable penetration or ECCS pump room boundary, an LOCA must also occur to require the operation of PPAFES. The severity of the postulated accidents during shutdown modes is limited due to the slower pace of the accident event progression and increased time for operator actions and mitigation strategies.

A LOCA is less likely to occur during shutdown modes because the following are significantly reduced or eliminated: energy contained within the reactor pressure boundary, reactor coolant

temperature and pressure, and the corresponding stresses. As a result, sufficient DID is maintained when the end state is changed from Mode 5 to Mode 4. Therefore, the NRC staff concludes the proposed change to revise TS 3.7.13 Required Action C.2, which allows the plant to remain in Mode 4, is acceptable.

3.1.2.12 TS 3.8.1 AC Sources – Operating
TS 3.8.4 DC Sources – Operating
TS 3.8.7 Inverters – Operating
TS 3.8.9 Distribution Systems – Operating

AC sources: The Class 1E AC electrical power distribution system, AC sources, consists of the offsite power sources (preferred power sources, normal and alternates), and the onsite standby power sources (Train 'A' and Train 'B' diesel generators (DGs)). The AC electrical power system provides independent and redundant sources of power to the ESF system.

VEGP also includes a standby auxiliary transformer (SAT), which is in addition to the two sources from the offsite transmission network, that is a common offsite power source capable of connecting to any safety bus on either unit. It can be energized by combustion turbines that have an enhanced black start capability.[1]

The onsite Class 1E AC distribution system is divided into redundant load groups (trains) so that the loss of any one group does not prevent the minimum safety functions from being performed. Each train has connections to two offsite power sources and a single DG. An offsite circuit consists of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network or the 13.8 kilovolt (kV) SAT to the onsite Class 1E ESF buses. Certain required unit loads are returned to service in a predetermined sequence in order to prevent overloading the transformer supplying offsite power to the onsite Class 1E distribution system.

After the DG has started, it will automatically tie to its respective bus after offsite power is tripped as a consequence of ESF bus undervoltage or degraded voltage, independent of or coincident with an SI signal. The DGs will also start and operate in the standby mode without tying to the ESF bus on an SI signal alone. Following the trip of offsite power, a sequencer signal strips nonpermanent loads from the ESF bus. When the DG is tied to the ESF bus, loads are then sequentially connected to its respective ESF bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading the DG by automatic load application.

In Modes 1, 2, 3, and 4, TS LCO 3.8.1 requires to be operable: (1) two qualified circuits

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[1] This evaluation is based on the availability of this alternate power source as currently included in VEGP TS 3.8.1. However, in the licensee's letter dated April 18, 2016, (ADAMS Accession No. ML16109A338), related to the review of a separate license amendment other than the one being reviewed in this safety evaluation, the licensee proposes to remove the TSs for the enhanced black start combustion turbine generators and the black start diesel generator. Any concerns regarding the acceptability of the TSTF-432 changes in end states to the changes proposed in the licensee's April 18, 2016, letter will be addressed as a part of that forthcoming review.

between the offsite transmission network and the onsite Class 1E AC electrical power distribution system, (2) two DGs capable of supplying the onsite Class 1E power distribution subsystem(s), and (3) automatic load sequencers for Train 'A' and Train 'B' ESF buses.

The licensee identified a deviation from TSTF-432 concerning VEGP TS 3.8.1. Specifically, the VEGP configuration for offsite power sources includes an additional qualified source as compared to the STS (i.e., the 13.8 kV SAT). This additional offsite source allows the licensee more time to restore an inoperable DG before being required to shut down the unit and results in an additional TS Condition statement. Even with the differences, the existing and proposed end state Completion Times for this TS are in agreement with TSTF-432, and the justification provided in the TSTF-432 model SE for changing this end state bounds the deviation. Therefore, the NRC staff concludes the deviation is acceptable.

Direct current (DC) sources: The station DC electrical power system provides the AC emergency power system with control power. It also provides both motive and control power to selected safety-related equipment and preferred AC vital bus power (via inverters). There are four safety features 125 volts direct current (VDC) systems (identified as A, B, C, and D) per unit. The 125 VDC systems A and C form the train 'A' safety features DC system, and their associated battery chargers receive power from two Class 1E train 'A' motor control centers. The 125 VDC systems B and D form the train 'B' safety features DC system, and their battery chargers receive power from two Class 1E train B motor control centers. As required by GDC 17 from 10 CFR Part 50, Appendix A, the DC electrical power system is designed so that no single failure in any 125 VDC system will result in conditions that will prevent the safe shutdown of the reactor plant.

During normal operation, the 125 VDC load is typically powered from two battery chargers, using load sharing circuitry, with the batteries floating on the system. Only one charger is required to be operable to support the associated DC power system. In the case of loss of normal power to the battery charger, the DC load is automatically powered from the station batteries.

In Modes 1, 2, 3, and 4, TS LCO 3.8.4 requires that four class 1E 125 VDC electrical power sources shall be operable.

Inverters: The function of the inverter is to convert DC power to AC power. Through use of an inverter, the station batteries can provide AC electrical power to the vital buses. The inverters can be powered from the battery chargers or from the station batteries. The inverters provide the preferred source for the reactor protection system (RPS), ESFAS, the nuclear steam supply system control and instrumentation, the post-accident monitoring system, and the safety-related radiation monitoring system. The station batteries ensure that an uninterruptible power source is available for these end devices.

Maintaining the required inverters operable ensures that the redundancy incorporated into the design of the RPS and ESFAS instrumentation and controls is maintained so that the fuel, RCS, and containment design limits are not exceeded. TS LCO 3.8.7 specifies that the required Class 1E 120 V inverters shall be operable in Modes 1, 2, 3, and 4.

Distribution systems: The onsite Class 1E AC electrical power distribution systems are divided by train into two redundant and independent AC electrical power distribution subsystems. The DC and AC vital buses are divided into four channels of distribution, two channels of which are associated with each train.

The AC electrical power subsystem for each train consists of a primary ESF 4.16 kV bus and secondary 480 and 120 V buses, distribution panels, motor control centers, and load centers.

Each 4.16 kV ESF bus has at least one separate and independent offsite source of power, as well as a dedicated onsite DG source. Each 4.16 kV ESF bus is normally connected to a preferred offsite source. A transfer to the alternate offsite source can be made manually. If all offsite sources are unavailable, the onsite emergency DG supplies power to the 4.16 kV ESF bus. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries.

The secondary AC electrical power distribution subsystem for each train includes the safety-related load centers, motor control centers, and distribution panels shown in Table B 3.8.9-1 of the VEGP TS bases. The 120 VAC vital buses are arranged in two load groups per train and are normally powered from the inverters. The alternate power supply for the vital buses is Class 1E regulating transformers powered from the same train as the associated inverter. There are four independent 125 VDC electrical power distribution subsystems (two for each train).

TS LCO 3.8.9 specifies that the required AC, DC, and AC vital bus electrical power distribution subsystems shall be operable in Modes 1, 2, 3, and 4.

NRC Staff Evaluation of AC Sources, DC Sources, Inverters, and Distribution Systems

The final SE of TR WCAP-16294, Table 3.2.1, shows that the CDP decreases slightly when the unit is cooled down to Mode 4 instead of Mode 5 for each condition in TS 3.8.1, TS 3.8.4, TS 3.8.7, and TS 3.8.9.

For TS 3.8.1, two trains of DGs are available if two offsite power circuits are inoperable and two offsite power circuits are available if two DGs are inoperable. If an offsite power circuit and/or a DG are inoperable, at least one of each remains available. For TS 3.8.4, there are two redundant trains of DC power; if one is inoperable, the other is available to provide the necessary DC power. For TS 3.8.7, there are two redundant trains of inverters; if one is inoperable, the other train is available to provide the necessary AC power.

The slower nature of event progression during shutdown modes provides increased time for operator actions and mitigation strategies if an event were to occur. In addition, some events are less likely to occur during shutdown modes. As a result, sufficient DID is maintained when the end state is changed from Mode 5 to Mode 4. Therefore, the NRC staff concludes the proposed change to revise TS 3.8.1 Required Action H.2, TS 3.8.4 Required Action D.2, TS 3.8.7 Required Action B.2, and TS 3.8.9 Required Action D.2, which allows the plant to remain in Mode 4, is acceptable.

3.2 Risk Evaluation

The licensee stated in its LAR that the information in the Westinghouse TR WCAP-16294 and TSTF-432 is applicable to VEGP. As stated in the final SE for TR WCAP-16294 (Reference 13), the NRC staff reviewed TR WCAP-16294 using SRP Chapters 19.2 and 16.1, and the five key principles of risk-informed decisionmaking presented in RG 1.174 and RG 1.177. The final SE also stated that design and operational differences among Westinghouse plants were identified, and appropriate sensitivity studies were performed, which show that the conclusions of the quantitative risk assessment apply to all Westinghouse plants. The NRC staff concludes that the risk evaluation as discussed in the NRC staff's final SE for TR WCAP-16294 is applicable to VEGP; therefore, the risk evaluation related to the LAR is acceptable.

3.3 TS Bases Changes

TSTF-432 included, and the licensee submitted, the following TS Bases changes:

- A reference to the NRC-approved TR WCAP-16294 has been added to the reference section of the TS Bases for each TS affected in TSTF-432.

- The following statement was added to each TS Bases Action section affected:

> Remaining within the Applicability of the LCO is acceptable to accomplish short duration repairs to restore inoperable equipment because the plant risk in MODE 4 is similar to or lower than MODE 5 (Ref. [ ]). In MODE 4 the steam generators and residual heat removal system are available to remove decay heat, which provides diversity and defense in depth. As stated in Reference [ ], the steam turbine driven auxiliary feedwater pump must be available to remain in MODE 4. Should steam generator cooling be lost while relying on this Required Action, there are preplanned actions to ensure long-term decay heat removal. Voluntary entry into MODE 5 may be made as it is also acceptable from a risk perspective.

The NRC staff generally does not approve TS bases changes; however, the staff does review the changes for consistency with the proposed TS changes. The NRC staff determined that the TS Bases changes are consistent with the proposed TS changes and the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993 (58 FR 39132).

3.4 Summary

The NRC staff has reviewed the SNC proposed adoption of TSTF-432 to modify certain TS requirements to permit an end state of hot shutdown mode (i.e., Mode 4) with the implementation of TR WCAP-16294 and found the proposed changes to be consistent with the

NRC-approved TR WCAP-16294. The NRC staff also determined that the requirements of 10 CFR 50.36 are met, and the proposed changes to the TSs are, therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Georgia official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on June 23, 2015 (80 FR 35983). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Southern Nuclear Operating Company application dated May 6, 2015, regarding adoption of TSTF-432, Revision 1, "Change in Technical Specifications End States (WCAP-16294)" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15128A239).

2. Southern Nuclear Operating Company supplemental letter dated October 8, 2015 (regarding the licensee's response to the NRC staff letter dated September 8, 2015 (ADAMS Accession No. ML15240A232), for Request for Additional Information) (ADAMS Accession No. ML15281A179).

3. Southern Nuclear Operating Company supplemental letter dated May 9, 2016 (regarding the licensee's response to the NRC staff letter dated April 7, 2016 (ADAMS Accession No. ML16077A177) for Request for Additional Information) (ADAMS Accession No. ML16130A690).

4. TSTF-432, Revision 1, "Change in Technical Specifications End States (WCAP-16294)," dated November 29, 2010 (ADAMS Accession No. ML103360003).

5. WCAP-16294-NP-A, Revision 1, "Risk-Informed Evaluation of Changes to Technical Specification Required Action Endstates for Westinghouse NSSS PWRs," dated June 2010 (ADAMS Accession No. ML103430249).

6. NUREG-1431, Volume 1, Revision 3, "Standard Technical Specifications Westinghouse Plants, Specifications," dated June 2004 (ADAMS Accession No. ML041830612).

7. Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," dated November 2002 (ADAMS Accession No. ML023240437)

8. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," August 1998 (ADAMS Accession No. ML003740176).

9. Regulatory Guide 1.182, "Assessing and Managing Risk before Maintenance Activities at Nuclear Power Plants, "dated May 2000 (ADAMS Accession No. ML003699426).

10. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," dated February 22, 2000 (ADAMS Accession No. ML003704489).

11. NUREG-0800, Standard Review Plan, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June 2007 (ADAMS Accession No. ML071700658)

12. NUREG-0800, Standard Review Plan, Section 16.1, Revision 1, "Risk-Informed Decisionmaking: Technical Specifications," dated March 2007 (ADAMS Accession No. ML070380228).

13. Final Safety Evaluation of NEI Topical Report WCAP-16294-NP, Revision 0, "Risk-Informed Evaluation of Changes to Technical Specification Required Endstates for Westinghouse NSSS PWRs," dated March 29, 2010 (ADAMS Package Accession No. ML100820533).

14. Regulatory Guide 1.160, Revision 3, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated May 2012 (ADAMS Accession No. ML113610098).

15. NUMARC 93-01, Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated April 2011 (ADAMS Accession No. ML11116A198).

16. NUREG-1431, Volume 1, Revision 4, "Standard Technical Specifications, Westinghouse Plants," dated April 2012 (ADAMS Accession No. ML12100A222).

17. License Amendment, "Vogtle Electric Generating Plant, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MA6911 and MA6912)," dated December 22, 2000 (ADAMS Accession Package No. ML010120468).

18. Vogtle Units 1 and 2 Improved Technical Specification Bases, Revision 33, dated April 10, 2015 (ADAMS Accession Package No. ML15114A101).

Principal Contributor: William Satterfield

Date: May 31, 2016

May 31, 2016

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Company, Inc.
P.O. Box 1295
Bin 038
Birmingham, AL 35201-1295

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TECHNICAL SPECIFICATION END STATES (CAC NOS. MF6197 AND MF6198)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 179 to Renewed Facility Operating License No. NPF-68 and Amendment No. 160 to Renewed Facility Operating License No. NPF-81 for the Vogtle Electric Generating Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 6, 2015, as supplemented by letters dated October 8, 2015, and May 9, 2016. The changes revise certain TS Required Actions to permit a Required Action end state of MODE 4 instead of MODE 5.

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,
/RA/

Bob Martin, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 179 to NPF-68
2. Amendment No. 160 to NPF-81
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

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WSatterfield, NRR
RecordsAmend
RidsNrrDssStsb Resource

ADAMS Accession No.: ML16130A577 *by memo dated

OFFICE	DORL/LPL2-1/PM	DORL/LPL2-1/LA	DSS/STSB/BC*
NAME	BMartin	LRonewicz	AKlein
DATE	5/20/2016	5/27/2016	5/11/2016
OFFICE	OGC – NLO subject to e-mail comments	DORL/LPL2-1/BC	DORL/LPL2-1/PM
NAME	JWachutka	MMarkley	BMartin
DATE	5/18/2016	5/27/2016	5/31/2016

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