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

December 19, 2016

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

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2 – ISSUANCE OF AMENDMENTS TO ADOPT TSTF-423, REVISION 1, "TECHNICAL SPECIFICATIONS END STATES, NEDC-32988-A" (CAC NOS. MF7197 AND MF7198)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 281 to Renewed Facility Operating License No. DPR-57 and Amendment No. 225 to Renewed Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant (HNP), Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated December 15, 2015, as supplemented by letter dated April 11, 2016.

The amendments revise TS requirements for end states associated with the implementation of the NRC-approved Technical Specifications Task Force (TSTF) Traveler TSTF-423-A, Revision 1, "Technical Specification End States, NEDC-32988-A," dated December 22, 2009. The TS end states modifications would permit, for some HNP systems, entry into a hot shutdown (Mode 3) end state rather than a cold shutdown (Mode 4) end state, which is the current TS requirement.

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

Sincerely,

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

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 281 to DPR-57
2. Amendment No. 225 to NPF-5
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. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 281
Renewed License No. DPR-57

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

    A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 1 (the facility) Renewed Facility Operating License No. DPR-57 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself; Georgia Power Company; Oglethorpe Power Corporation; Municipal Electric Authority of Georgia; and City of Dalton, Georgia (the owners), dated December 15, 2015, as supplemented by letter dated April 11, 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 as 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 set forth in 10 CFR Chapter I;

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

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

Enclosure 1

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-57 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications (Appendix A) and the Environmental Plan (Appendix B), as revised through Amendment No. 281, are hereby incorporated in the renewed 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 from the date 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 Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: December 19, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 281

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the License and the 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 Pages	Insert Pages
License	License
4	4
TSs	TSs
3.3-69	3.3-69
3.5-1	3.5-1
3.5-2	3.5-2
3.5-9	3.5-9
3.6-17	3.6-17
3.6-18	3.6-18
3.6-19	3.6-19
3.6-25	3.6-25
3.6-27	3.6-27
3.6-34	3.6-34
3.6-41	3.6-41
3.7-1	3.7-1
3.7-2	3.7-2
3.7-3	3.7-3
3.7-4	3.7-4
3.7-9	3.7-9
3.7-13	3.7-13
3.7-16	3.7-16
3.8-6	3.8-6
3.8-27	3.8-27
3.8-37	3.8-37

for sample analysis or instrumentation calibration, or associated with radioactive apparatus or components;

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

(C) This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady sate reactor core power levels not in excess of 2804 megawatts thermal.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Plan (Appendix B), as revised through Amendment No. 281 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance Requirement (SR) contained in the Technical Specifications and listed below, is not required to be performed immediately upon implementation of Amendment No. 195. The SR listed below shall be successfully demonstrated before the time and condition specified:

SR 3.8.1.18 shall be successfully demonstrated at its next regularly scheduled performance.

(3) Fire Protection

Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained in the updated Fire Hazards Analysis and Fire Protection Program for the Edwin I. Hatch Nuclear Plant, Units 1 and 2, which was originally submitted by letter dated July 22, 1986. Southern Nuclear may make changes to the fire protection program without prior Commission approval only if the changes

3.3 INSTRUMENTATION

3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

LCO 3.3.8.2 Two RPS electric power monitoring assemblies shall be OPERABLE for each inservice RPS motor generator set or alternate power supply.

APPLICABILITY: MODES 1, 2, and 3,
MODES 4 and 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies or with both residual heat removal (RHR) shutdown cooling (SDC) isolation valves open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both inservice power supplies with one electric power monitoring assembly inoperable.	A.1 Remove associated inservice power supply(s) from service.	72 hours
B. One or both inservice power supplies with both electric power monitoring assemblies inoperable.	B.1 Remove associated inservice power supply(s) from service.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours

(continued)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six of seven safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. <u>OR</u> One LPCI pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 -----NOTE----- LCO 3.0.4.a is not applicable when entering MODE 3. -------------- Be in MODE 3.	 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE. <u>AND</u> C.2 Restore HPCI System to OPERABLE status.	1 hour 14 days
D. HPCI System inoperable. <u>AND</u> Condition A entered.	D.1 Restore HPCI System to OPERABLE status. <u>OR</u> D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------ Be in Mode 3.	 12 hours
F. Two or more ADS valves inoperable.	F.1 Be in MODE 3. <u>AND</u> F.2 Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
G. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. <u>OR</u> HPCI System and two or more ADS valves inoperable.	G.1 Enter LCO 3.0.3.	Immediately

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

-----NOTE-----
LCO 3.0.4.b is not applicable to RCIC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means high pressure coolant injection (HPCI) System is OPERABLE. <u>AND</u> A.2 Restore RCIC System to OPERABLE status.	1 hour 14 days
B. Required Action and associated Completion Time not met.	B.1 -----NOTE----- LCO 3.0.4.a is not applicable when entering MODE 3. ----- Be in MODE 3.	 12 hours

3.6 CONTAINMENT SYSTEMS

3.6.1.7 Reactor Building-to-Suppression Chamber Vacuum Breakers

LCO 3.6.1.7 Each reactor building-to-suppression chamber vacuum breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more lines with one reactor building-to-suppression chamber vacuum breaker not closed.	A.1 Close the open vacuum breaker.	72 hours
B. One or more lines with two reactor building-to-suppression chamber vacuum breakers not closed.	B.1 Close one open vacuum breaker.	1 hour
C. One line with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	C.1 Restore the vacuum breaker(s) to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------ Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Two lines with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	E.1 Restore all vacuum breakers in one line to OPERABLE status.	1 hour
F. Required Action and Associated Completion Time or Condition A, B, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.7.1	-----NOTES----- 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. ---------- Verify each vacuum breaker is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.7.2	Perform a functional test of each vacuum breaker.	In accordance with the Inservice Testing Program
SR 3.6.1.7.3	Verify the opening setpoint of each vacuum breaker is ≤ 0.5 psid.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.1.8 Suppression Chamber-to-Drywell Vacuum Breakers

LCO 3.6.1.8 Ten suppression chamber-to-drywell vacuum breakers shall be OPERABLE for opening.

AND

Twelve suppression chamber-to-drywell vacuum breakers shall be closed, except when performing their intended function.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1 Restore one vacuum breaker to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
C. One suppression chamber-to-drywell vacuum breaker not closed.	C.1 Close the open vacuum breaker.	2 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
C. Two RHR suppression pool cooling subsystems inoperable.	C.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool spray subsystem inoperable.	A.1 Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days
B. Two RHR suppression pool spray subsystems inoperable.	B.1 Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------ Be in MODE 3.	 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify each RHR suppression pool spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRVs).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
C. Secondary containment inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1 ------------NOTE-------------- LCO 3.0.3 is not applicable. ---------------------------------- Suspend movement of irradiated fuel assemblies in the secondary containment. AND	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----------------NOTE---------------- LCO 3.0.3 is not applicable. --	
	D.1 Place remaining OPERABLE SGT subsystem(s) in operation.	Immediately
	<u>OR</u>	
	D.2.1 Suspend movement of irradiated fuel assemblies in secondary containment.	Immediately
	<u>AND</u>	
	D.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	D.2.3 Initiate action to suspend OPDRVs.	Immediately
E. Two or more required SGT subsystems inoperable in MODE 1, 2, or 3.	E.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours

(continued)

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System

LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHRSW pump inoperable.	A.1 Restore RHRSW pump to OPERABLE status.	30 days
B. One RHRSW pump in each subsystem inoperable.	B.1 Restore one RHRSW pump to OPERABLE status.	7 days
C. One RHRSW subsystem inoperable for reasons other than Condition A.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for RHR shutdown cooling made inoperable by RHRSW System. ---	
	C.1 Restore RHRSW subsystem to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Both RHRSW subsystems inoperable for reasons other than Condition B.	-----------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling made inoperable by RHRSW System. ---	
	E.1 Restore one RHRSW subsystem to OPERABLE status.	8 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	F.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each RHRSW manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.2 Plant Service Water (PSW) System and Ultimate Heat Sink (UHS)

LCO 3.7.2 Two PSW subsystems and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PSW pump inoperable.	A.1 Restore PSW pump to OPERABLE status.	30 days
B. One PSW turbine building isolation valve inoperable.	B.1 Restore PSW turbine building isolation valve to OPERABLE status.	30 days
C. One PSW pump in each subsystem inoperable.	C.1 Restore one PSW pump to OPERABLE status.	7 days
D. One PSW turbine building isolation valve in each subsystem inoperable.	D.1 Restore one PSW turbine building isolation valve to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------ Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One PSW subsystem inoperable for reasons other than Conditions A and B.	---------------NOTES--------------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for diesel generator made inoperable by PSW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for RHR shutdown cooling made inoperable by PSW System. ---	
	F.1 Restore the PSW subsystem to OPERABLE status.	72 hours
G. Required Action and associated Completion Time of Condition F not met. <u>OR</u> Both PSW subsystems inoperable for reasons other than Conditions C and D. <u>OR</u> UHS inoperable.	G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 4.	12 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----------------NOTE----------------- LCO 3.0.3 is not applicable. ---	
	D.1 Place OPERABLE MCREC subsystem in pressurization mode.	Immediately
	OR	
	D.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	D.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	D.2.3 Initiate action to suspend OPDRVs.	Immediately
E. Two MCREC subsystems inoperable in MODE 1, 2, or 3 for reasons other than Condition B.	E.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, or 3.	D.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
E. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----------------NOTE----------------- LCO 3.0.3 is not applicable. --	
	E.1 Place OPERABLE control room AC subsystems in operation.	Immediately
	<u>OR</u>	
	E.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	E.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	E.2.3 Initiate action to suspend OPDRVs.	Immediately

(continued)

3.7 PLANT SYSTEMS

3.7.6 Main Condenser Offgas

LCO 3.7.6 The gross gamma activity rate of the noble gases measured at the main condenser evacuation system pretreatment monitor station shall be ≤ 240 mCi/second.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated and steam jet air ejector (SJAE) in operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gross gamma activity rate of the noble gases not within limit.	A.1 Restore gross gamma activity rate of the noble gases to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	
	B.2 Isolate SJAE.	12 hours
	<u>OR</u>	
	B.3 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------	
	Be in MODE 3.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	OR E.2 Restore required DG to OPERABLE status.	 12 hours
F. Two or more (Unit 1 and swing) DGs inoperable.	F.1 Restore all but one Unit 1 and swing DGs to OPERABLE status	2 hours
G. No DGs capable of supplying power to any Unit 1 LPCI valve load center.	G.1 Restore one DG capable of supplying power to Unit 1 LPCI valve load center to OPERABLE status.	2 hours
H. Required Action and Associated Completion Time of Condition A, B, C, D, E, F, or G not met.	H.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------ Be in MODE 3.	 12 hours
I. One or more required offsite circuits and two or more required DGs inoperable. OR Two or more required offsite circuits and one required DG inoperable.	I.1 Enter LCO 3.0.3.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Unit 1 DG DC electrical power subsystem inoperable. <u>OR</u> Swing DG DC electrical power subsystem inoperable for reasons other than Condition A.	B.1 Restore DG DC electrical power subsystem to OPERABLE status.	12 hours
C. One Unit 1 station service DC electrical power subsystem inoperable.	C.1 Restore station service DC electrical power subsystem to OPERABLE status.	2 hours
D. Required Action and Associated Completion Time of Condition A, B, or C not met.	D.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	 12 hours
E. Two or more DC electrical power subsystems inoperable that result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more (Unit 1 or swing bus) DG DC electrical power distribution subsystems inoperable.	B.1 Restore DG DC electrical power distribution subsystem to OPERABLE status.	12 hours
C. One or more (Unit 1 or swing bus) AC electrical power distribution subsystems inoperable.	C.1 Restore AC electrical power distribution subsystem to OPERABLE status.	8 hours
D. One Unit 1 station service DC electrical power distribution subsystem inoperable.	D.1 Restore Unit 1 station service DC electrical power distribution subsystem to OPERABLE status.	2 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
F. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	F.1 Enter LCO 3.0.3.	Immediately

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 225
Renewed License No. NPF-5

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

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 2 (the facility) Renewed Facility Operating License No. NPF-5 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself; Georgia Power Company; Oglethorpe Power Corporation; Municipal Electric Authority of Georgia; and City of Dalton, Georgia (the owners), dated December 15, 2015, as supplemented by letter dated April 11, 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 as 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 set forth in 10 CFR Chapter I;

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

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

Enclosure 2

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-5 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B); as revised through Amendment No. 225 are hereby incorporated in the renewed 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 from the date 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 Renewed Facility
Operating License and
Technical Specifications

Date of Issuance: December 19, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 225

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the License and the 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 Pages	Insert Pages
License	License
4	4
TSs	TSs
3.3-69	3.3-69
3.5-1	3.5-1
3.5-2	3.5-2
3.5-10	3.5-10
3.6-17	3.6-17
3.6-18	3.6-18
3.6-19	3.6-19
3.6-25	3.6-25
3.6-27	3.6-27
3.6-33	3.6-33
3.6-40	3.6-40
3.6-41	3.6-41
3.7-1	3.7-1
3.7-2	3.7-2
3.7-3	3.7-3
3.7-4	3.7-4
3.7-9	3.7-9
3.7-13	3.7-13
3.7-16	3.7-16
3.8-6	3.8-6
3.8-27	3.8-27
3.8-38	3.8-38

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

(C) This renewed license shall be deemed to contain, and is subject to, the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions[2] specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady sate reactor core power levels not in excess of 2,804 megawatts thermal, in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B); as revised through Amendment No. 225 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the renewed license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

(a) Fire Protection

Southern Nuclear shall implement and maintain in effect all provisions of the fire protection program, which is referenced in the Updated Final Safety Analysis Report for the facility, as contained

[2] The original licensee authorized to possess, use, and operate the facility with Georgia Power Company (GPC). Consequently, certain historical references to GPC remain in certain license conditions.

3.3 INSTRUMENTATION

3.3.8.2 Reactor Protection System (RPS) Electric Power Monitoring

LCO 3.3.8.2 Two RPS electric power monitoring assemblies shall be OPERABLE for each inservice RPS motor generator set or alternate power supply.

APPLICABILITY: MODES 1, 2, and 3,
MODES 4 and 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies or with both residual heat removal (RHR) shutdown cooling (SDC) isolation valves open.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both inservice power supplies with one electric power monitoring assembly inoperable.	A.1 Remove associated inservice power supply(s) from service.	72 hours
B. One or both inservice power supplies with both electric power monitoring assemblies inoperable.	B.1 Remove associated inservice power supply(s) from service.	1 hour
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours

(continued)

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS - Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six of seven safety/relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

--NOTE--
LCO 3.0.4.b is not applicable to HPCI.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One LPCI pump in both LPCI subsystems inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	 12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. HPCI System inoperable.	C.1 Verify by administrative means RCIC System is OPERABLE.	1 hour
	AND	
	C.2 Restore HPCI System to OPERABLE status.	14 days
D. HPCI System inoperable. AND Condition A entered.	D.1 Restore HPCI System to OPERABLE status.	72 hours
	OR	
	D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
F. Two or more ADS valves inoperable.	F.1 Be in MODE 3.	12 hours
	AND	
	F.2 Reduce reactor steam dome pressure to ≤ 150 psig.	36 hours
G. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition A. OR HPCI System and two or more ADS valves inoperable.	G.1 Enter LCO 3.0.3.	Immediately

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

---NOTE---
LCO 3.0.4.b is not applicable to RCIC.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RCIC System inoperable.	A.1	Verify by administrative means high pressure coolant injection (HPCI) System is OPERABLE.	1 hour
	AND		
	A.2	Restore RCIC System to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	B.1	------------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------	
		Be in MODE 3.	12 hours

3.6 CONTAINMENT SYSTEMS

3.6.1.7 Reactor Building-to-Suppression Chamber Vacuum Breakers

LCO 3.6.1.7 Each reactor building-to-suppression chamber vacuum breaker shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more lines with one reactor building-to-suppression chamber vacuum breaker not closed.	A.1 Close the open vacuum breaker.	72 hours
B. One or more lines with two reactor building-to-suppression chamber vacuum breakers not closed.	B.1 Close one open vacuum breaker.	1 hour
C. One line with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	C.1 Restore the vacuum breaker(s) to OPERABLE status.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
E. Two lines with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening.	E.1 Restore all vacuum breakers in one line to OPERABLE status.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and Associated Completion Time of Condition A, B, or E not met.	F.1 Be in MODE 3. AND F.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.7.1	-----NOTES----- 1. Not required to be met for vacuum breakers that are open during Surveillances. 2. Not required to be met for vacuum breakers open when performing their intended function. --------------- Verify each vacuum breaker is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.1.7.2	Perform a functional test of each vacuum breaker.	In accordance with the Inservice Testing Program
SR 3.6.1.7.3	Verify the opening setpoint of each vacuum breaker is $\leq$ 0.5 psid.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.1.8 Suppression Chamber-to-Drywell Vacuum Breakers

LCO 3.6.1.8 Ten suppression chamber-to-drywell vacuum breakers shall be OPERABLE for opening.

AND

Twelve suppression chamber-to-drywell vacuum breakers shall be closed, except when performing their intended function.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required suppression chamber-to-drywell vacuum breaker inoperable for opening.	A.1 Restore one vacuum breaker to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
C. One suppression chamber-to-drywell vacuum breaker not closed.	C.1 Close the open vacuum breaker.	2 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	12 hours
	AND	
	D.2 Be in MODE 4.	36 hours

3.6 CONTAINMENT SYSTEMS

3.6.2.3 Residual Heat Removal (RHR) Suppression Pool Cooling

LCO 3.6.2.3 Two RHR suppression pool cooling subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHR suppression pool cooling subsystem inoperable.	A.1 Restore RHR suppression pool cooling subsystem to OPERABLE status.	7 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------- Be in MODE 3.	12 hours
C. Two RHR suppression pool cooling subsystems inoperable.	C.1 Restore one RHR suppression pool cooling subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.3.1	Verify each RHR suppression pool cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.2.4 Residual Heat Removal (RHR) Suppression Pool Spray

LCO 3.6.2.4 Two RHR suppression pool spray subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One RHR suppression pool spray subsystem inoperable.	A.1	Restore RHR suppression pool spray subsystem to OPERABLE status.	7 days
B. Two RHR suppression pool spray subsystems inoperable.	B.1	Restore one RHR suppression pool spray subsystem to OPERABLE status.	8 hours
C. Required Action and associated Completion Time not met.	C.1	------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.4.1	Verify each RHR suppression pool spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program

(continued)

3.6 CONTAINMENT SYSTEMS

3.6.4.1 Secondary Containment

LCO 3.6.4.1 The secondary containment shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of irradiated fuel assemblies in the secondary containment,
During CORE ALTERATIONS,
During operations with a potential for draining the reactor vessel (OPDRV).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3.	A.1 Restore secondary containment to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	12 hours
C. Secondary containment inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	C.1 -----------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------- Suspend movement of irradiated fuel assemblies in the secondary containment. <u>AND</u>	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required Unit 2 SGT subsystem inoperable. <u>OR</u> One required Unit 1 SGT subsystem inoperable for reasons other than Condition A.	B.1 Restore required SGT subsystem to OPERABLE status.	7 days
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	 12 hours
D. Required Action and associated Completion Time of Condition A or B not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----------------NOTE---------------- LCO 3.0.3 is not applicable. -- D.1 Place remaining OPERABLE SGT subsystem(s) in operation. <u>OR</u> D.2.1 Suspend movement of irradiated fuel assemblies in secondary containment. <u>AND</u> D.2.2 Suspend CORE ALTERATIONS. <u>AND</u> D.2.3 Initiate action to suspend OPDRVs.	 Immediately Immediately Immediately Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. Two or more required SGT subsystems inoperable in MODE 1, 2, or 3.	E.1	-----------NOTE------------ LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------ Be in MODE 3.	 12 hours
F. Two or more required SGT subsystems inoperable during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	F.1	-----------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------ Suspend movement of irradiated fuel assemblies in secondary containment.	 Immediately
	<u>AND</u>		
	F.2	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	F.3	Initiate action to suspend OPDRVs.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.3.1	Operate each required SGT subsystem for ≥ 15 continuous minutes.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.3.2	Perform required SGT filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

(continued)

3.7 PLANT SYSTEMS

3.7.1 Residual Heat Removal Service Water (RHRSW) System

LCO 3.7.1 Two RHRSW subsystems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RHRSW pump inoperable.	A.1 Restore RHRSW pump to OPERABLE status.	30 days
B. One RHRSW pump in each subsystem inoperable.	B.1 Restore one RHRSW pump to OPERABLE status.	7 days
C. One RHRSW subsystem inoperable for reasons other than Condition A.	------------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for RHR shutdown cooling made inoperable by RHRSW System. ---	
	C.1 Restore RHRSW subsystem to OPERABLE status.	7 days
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 ----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ----------------------------- Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. Both RHRSW subsystems inoperable for reasons other than Condition B.	-----------------NOTE------------------ Enter applicable Conditions and Required Actions of LCO 3.4.7 for RHR shutdown cooling made inoperable by RHRSW System. ---	
	E.1 Restore one RHRSW subsystem to OPERABLE status.	8 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1 Be in MODE 3.	12 hours
	AND	
	F.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	Verify each RHRSW manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	In accordance with the Surveillance Frequency Control Program

3.7 PLANT SYSTEMS

3.7.2 Plant Service Water (PSW) System and Ultimate Heat Sink (UHS)

LCO 3.7.2 Two PSW subsystems and UHS shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One PSW pump inoperable.	A.1 Restore PSW pump to OPERABLE status.	30 days
B. One PSW turbine building isolation valve inoperable.	B.1 Restore PSW turbine building isolation valve to OPERABLE status.	30 days
C. One PSW pump in each subsystem inoperable.	C.1 Restore one PSW pump to OPERABLE status.	7 days
D. One PSW turbine building isolation valve in each subsystem inoperable.	D.1 Restore one PSW turbine building isolation valve to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One PSW subsystem inoperable for reasons other than Conditions A and B.	-----------------NOTES---------------- 1. Enter applicable Conditions and Required Actions of LCO 3.8.1, "AC Sources - Operating," for diesel generator made inoperable by PSW System. 2. Enter applicable Conditions and Required Actions of LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown," for RHR shutdown cooling made inoperable by PSW System. ---	
	F.1 Restore the PSW subsystem to OPERABLE status.	72 hours
G. Required Action and associated Completion Time of Condition F not met. <u>OR</u> Both PSW subsystems inoperable for reasons other than Conditions C and D. <u>OR</u> UHS inoperable.	G.1 Be in MODE 3. <u>AND</u> G.2 Be in MODE 4.	12 hours 36 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------ Be in MODE 3.	12 hours
D. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	-----------------NOTE------------------ LCO 3.0.3 is not applicable. ---	
	D.1 Place OPERABLE MCREC subsystem in pressurization mode.	Immediately
	OR	
	D.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	AND	
	D.2.2 Suspend CORE ALTERATIONS.	Immediately
	AND	
	D.2.3 Initiate action to suspend OPDRVs.	Immediately
E. Two MCREC subsystems inoperable in MODE 1, 2, or 3 for reasons other than Condition B.	E.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------ Be in MODE 3.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, or 3.	D.1 -----------NOTE----------- LCO 3.0.4.a is not applicable when entering MODE 3. ------------------------------ Be in MODE 3.	 12 hours
E. Required Action and associated Completion Time of Condition A not met during movement of irradiated fuel assemblies in the secondary containment, during CORE ALTERATIONS, or during OPDRVs.	------------------NOTE----------------- LCO 3.0.3 is not applicable. --	
	E.1 Place OPERABLE control room AC subsystems in operation.	Immediately
	<u>OR</u>	
	E.2.1 Suspend movement of irradiated fuel assemblies in the secondary containment.	Immediately
	<u>AND</u>	
	E.2.2 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	E.2.3 Initiate action to suspend OPDRVs.	Immediately

(continued)

3.7 PLANT SYSTEMS

3.7.6 Main Condenser Offgas

LCO 3.7.6 The gross gamma activity rate of the noble gases measured at the main condenser evacuation system pretreatment monitor station shall be ≤ 240 mCi/second.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated and steam jet air ejector (SJAE) in operation.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Gross gamma activity rate of the noble gases not within limit.	A.1 Restore gross gamma activity rate of the noble gases to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	
	B.2 Isolate SJAE.	12 hours
	<u>OR</u>	
	B.3 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	12 hours

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. (continued)	OR E.2 Restore required DG to OPERABLE status.	 12 hours
F. Two or more (Unit 2 and swing) DGs inoperable.	F.1 Restore all but one Unit 2 and swing DGs to OPERABLE status.	2 hours
G. No DGs capable of supplying power to any Unit 2 LPCI valve load center.	G.1 Restore one DG capable of supplying power to Unit 2 LPCI valve load center to OPERABLE status.	2 hours
H. Required Action and Associated Completion Time of Condition A, B, C, D, E, F, or G not met.	H.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	 12 hours
I. One or more required offsite circuits and two or more required DGs inoperable. OR Two or more required offsite circuits and one required DG inoperable.	I.1 Enter LCO 3.0.3.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Unit 2 DG DC electrical power subsystem inoperable. <u>OR</u> Swing DG DC electrical power subsystem inoperable for reasons other than Condition A.	B.1 Restore DG DC electrical power subsystem to OPERABLE status.	12 hours
C. One Unit 2 station service DC electrical power subsystem inoperable.	C.1 Restore station service DC electrical power subsystem to OPERABLE status.	2 hours
D. Required Action and Associated Completion Time of Condition A, B, or C not met.	D.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	 12 hours
E. Two or more DC electrical power subsystems inoperable that result in a loss of function.	E.1 Enter LCO 3.0.3.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more (Unit 2 or swing bus) DG DC electrical power distribution subsystems inoperable.	B.1 Restore DG DC electrical power distribution subsystem to OPERABLE status.	12 hours
C. One or more (Unit 2 or swing bus) AC electrical power distribution subsystems inoperable.	C.1 Restore AC electrical power distribution subsystem to OPERABLE status.	8 hours
D. One Unit 2 station service DC electrical power distribution subsystem inoperable.	D.1 Restore Unit 2 station service DC electrical power distribution subsystem to OPERABLE status.	2 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 ------------NOTE--------- LCO 3.0.4.a is not applicable when entering MODE 3. -------------------------------- Be in MODE 3.	12 hours
F. Two or more electrical power distribution subsystems inoperable that result in a loss of function.	F.1 Enter LCO 3.0.3.	Immediately

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 281 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-57

AND

AMENDMENT NO. 225 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By application dated December 15, 2015 (Reference 1), as supplemented by letter dated April 11, 2016 (Reference 2), Southern Nuclear Operating Company, Inc. (SNC, the licensee), requested changes to the Technical Specifications (TSs) for the Edwin I. Hatch Nuclear Plant (HNP), Unit Nos. 1 and 2 (Unit 1 and Unit 2). The supplement dated April 11, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 16, 2016 (81 FR 7841).

The proposed changes would revise TS requirements for end states associated with the implementation of the NRC-approved Technical Specifications Task Force (TSTF) Traveler TSTF-423-A, Revision 1, "Technical Specification End States, NEDC-32988-A," dated December 22, 2009 (Reference 3). The TS end states modifications would permit, for some HNP systems, entry into a hot shutdown (Mode 3) end state rather than a cold shutdown (Mode 4) end state, which is the current TS requirement.

The following five operational modes are defined in the HNP TSs. Of specific relevance to TSTF-423 are Modes 3 and 4:

Mode 1 - Power Operation: The reactor mode switch is in run position.

Mode 2 - Reactor Startup: The reactor mode switch is in refuel position (with all reactor vessel head closure bolts fully tensioned) or in startup/hot standby position.

Mode 3 – Hot Shutdown: The reactor coolant system (RCS) temperature is above 212 degrees Fahrenheit (°F), and the reactor mode switch is in shutdown position (with all reactor vessel head closure bolts fully tensioned).

Mode 4 - Cold Shutdown: The RCS temperature is equal to or less than 212 °F, and the reactor mode switch is in shutdown position (with all reactor vessel head closure bolts fully tensioned).

Mode 5 - Refueling: The reactor mode switch is in shutdown or refuel position, and one or more reactor vessel head closure bolts are less than fully tensioned.

Most of the current 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 that this perception was incorrect and took corrective actions to improve shutdown operation. At the same time, Standard Technical Specifications (STS) were developed, and many licensees improved their TSs. Since enactment of a shutdown rule was expected, almost all TS changes involving power operation, including a revised end state requirement, were postponed (see, for example, the Final Policy Statement on Technical Specification Improvements (Reference 4)). However, in the mid-1990s, the Commission decided a shutdown rule was not necessary in light of industry improvements.

The STS and most plant TSs provide, as part of the remedial action, a Completion Time (CT) for the plant to either comply with remedial actions or restore compliance with the Limiting Condition for Operation (LCO). If the LCO or the remedial action cannot be met, then the reactor is required to be shut down. When the STS and individual plant TSs were written, the shutdown condition, or "end state," specified was usually cold shutdown.

2.0 REGULATORY EVALUATION

2.1 Proposed Changes

The amendments propose to change certain required end states when the TS actions for remaining in power operation cannot be met within the CTs. Most of the requested TS changes permit an end state of hot shutdown (Mode 3) if risk is assessed and managed, rather than an end state of cold shutdown (Mode 4), contained in the current TSs. The changes were limited to those end states 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 TSs, and (3) the primary purpose is to correct the initiating condition and return to power operation as soon as is practical. Risk insights from both the qualitative and quantitative risk assessments were used in specific TS assessments.

Each proposed TS change is reviewed individually in Section 3.2 of this safety evaluation.

2.2 Regulations and Guidance

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.90 states that whenever a holder of an operating license desires to amend the license (in this case, a TSTF-423 amendment), application for an amendment must be filed with the Commission, fully describing the changes desired, and following as far as applicable, the form prescribed for original applications. As stated in 10 CFR 50.36(a)(1), each applicant for an operating license shall include in its application proposed TSs in accordance with the requirements of 10 CFR 50.36. Further, per 10 CFR 50.36(a)(1), a summary statement of the bases or reasons for such specifications, other than those covering administrative controls shall also be included in the application, but shall not become part of the technical specifications.

In 10 CFR 50.36, "Technical specifications," the Commission established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36(c), TSs, in part, are required to include items in the following specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) surveillance requirements (SRs); (4) design features; and (5) administrative controls. The regulation in 10 CFR 50.36(c)(2)(i) states, in part, that:

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

Licensees control shutdown risk by controlling 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. In practice, the risk during shutdown operations is often addressed by voluntary actions and the application of 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," which is called the Maintenance Rule. The regulation in 10 CFR 50.65(a)(4) states, in part, that:

> Before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), 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.

As described in 10 CFR 50.92(a), in determining whether an amendment to a license will be issued to the applicant, the Commission will be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate. Considerations common to many types of licenses that guide the Commission's determination as to whether a license will be issued are provided in 10 CFR 50.40. The specific findings that the Commission must make to issue an operating license are given in 10 CFR 50.57(a). Therefore, to issue amended TSs containing modified end states, the Commission must find, among other things, that the remedial actions permitted by the TSs (i.e., the modified end states), when considered as part of the overall activities authorized by the license, provide reasonable assurance that the health and safety of the public will not be endangered.

The NRC-approved Boiling Water Reactor (BWR) Owners Group (BWROG) Topical Report (TR) NEDC-32988-A, Revision 2, "Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for Boiling Water Reactor Plants" (hereinafter "NEDC-32988-A") (Reference 5), provides the technical basis to change certain required "end states" when the TS actions for remaining in power operation cannot be met within the CTs. The "end states" are part of the remedial actions described by 10 CFR 50.36(c)(2)(i) in that they are an action other than shutting down the reactor.

Most of the requested TS changes permit an end state of hot shutdown (Mode 3) if risk is assessed and managed rather than an end state of cold shutdown (Mode 4) contained in the current TSs. In describing the basis for changing end states, NEDC-32988-A states, in part, that:

> Cold shutdown is normally required when an inoperable system or train cannot be restored to an operable status within the allowed time. Going to cold shutdown results in the loss of steam-driven systems, challenges the shutdown heat removal systems, and requires restarting the plant. A more preferred operational mode is one that maintains adequate risk levels while repairs are completed without causing unnecessary challenges to plant equipment during shutdown and startup transitions.

The NRC's safety evaluation for TR NEDC-32988, Revision 2, dated September 27, 2002 (Reference 6), states, in part, that:

> In the end state changes considered here, the malfunction of a component or train has generally resulted in a failure to meet a TS and a controlled shutdown has begun because a TS CT has been exceeded.

TSTF-423-A, Revision 1, incorporates the NRC-approved NEDC-32988-A into NUREG-1433, Revision 4, "Standard Technical Specifications – General Electric Plants (BWR/4)" (Reference 7) (and hereby referred to as the STS throughout this safety evaluation), and NUREG-1434, Revision 4, "Standard Technical Specifications – General Electric Plants (BWR/6)" (Reference 8). The conclusions are applicable for all of the BWR products (BWR/2 through BWR/6). The *Federal Register* notice (Reference 9) published on February 18, 2011 (76 FR 9614), announced the availability of this TS improvement as part of the consolidated line item improvement process.

The licensee states that it reviewed BWROG NEDC-32988-A, TSTF-423, Revision 1, and the NRC staff's model safety evaluation (Reference 10), and concluded that the information provided in these three documents is applicable to HNP, Units 1 and 2, and justifies this LAR for incorporation of the changes to the HNP TSs. The TSTF-423 justification references Regulatory Guide (RG) 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants" (Reference 11). On November 27, 2012, the NRC published a *Federal Register* notice stating that RG 1.182 has been withdrawn, and the subject matter has been incorporated into RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

(Reference 12). RG 1.160 endorses NUMARC 93-01, Revision 4A, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Reference 13). SNC assesses the risk of maintenance activities consistent with the guidance in RG 1.160. Therefore, the justification for adoption of TSTF-423 continues to be applicable.

3.0 TECHNICAL EVALUATION

3.1 Risk Assessment

The objective of the BWROG NEDC-32988, Revision 2, risk assessment was to show that any risk increases associated with the proposed changes in TS end states are either negligible or negative (i.e., a net decrease in risk). NEDC-32988 documents a risk-informed analysis of the proposed TS change. Probabilistic risk assessment (PRA) results and insights are used in combination with results of deterministic assessments to identify and propose changes in "end states" for all BWR plants. This is in accordance with guidance provided in RG 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 14), and RG 1.177, "An Approach for Plant Specific Risk-Informed Decisionmaking: Technical Specifications" (Reference 15). The three-tiered approach documented in RG 1.177 was followed. The first tier of the three-tiered approach includes the assessment of the risk impact of the proposed change for comparison to acceptance guidelines consistent with the Commission's Safety Goal Policy Statement, as documented in RG 1.174. The first tier aims at ensuring that there are no unacceptable temporary risk increases as a result of the TS change, such as when equipment is taken out of service. The second tier addresses the need to preclude potentially high-risk configurations that could result if equipment is taken out of service concurrently with the equipment out of service, as allowed by this TS change. The third tier addresses the application of 10 CFR 50.65(a)(4) of the Maintenance Rule for identifying risk-significant configurations resulting from maintenance-related activities and taking appropriate compensatory measures to avoid such configurations.

The TSs invoke a risk assessment because 10 CFR 50.65(a)(4) is applicable to maintenance-related activities and does not cover other operational activities beyond the effect they may have on existing maintenance-related risk.

The BWROG risk assessment approach was found to be acceptable in the safety evaluation for NEDC-32988, Revision 2. In addition, the analyses show that the three-tiered approach criteria for allowing TS changes are met as follows:

- Risk Impact of the Proposed Change (Tier 1):

  The risk changes associated with the TS changes in TSTF-423 in terms of mean yearly increases in core damage frequency (CDF) and large early release frequency (LERF) are risk neutral or risk beneficial. In addition, there are no significant temporary risk increases as defined by RG 1.177 criteria associated with the implementation of the TS end state changes.

- Avoidance of Risk-Significant Configurations (Tier 2):

  The performed risk analyses, which are based on single LCOs, indicate that there are no high-risk configurations associated with the TS end state changes. The reliability of redundant trains is normally covered by a single LCO. When multiple LCOs occur, which affect trains in several systems, the plant's risk-informed configuration risk management program, or the risk assessment and management program implemented in response to the Maintenance Rule (10 CFR 50.65 (a)(4)), shall ensure that high-risk configurations are avoided. As part of the implementation of TSTF-423, the licensee has committed to follow Section 11 of NUMARC 93-01, Revision 3 (Reference 16), and include guidance in appropriate plant procedures and/or administrative controls to preclude high-risk plant configurations when the plant is at the proposed end state. This commitment shall be incorporated into the licensee's Final Safety Analysis Report (FSAR), as discussed in Section 3.3 of this safety evaluation. The NRC staff finds that such guidance is adequate for preventing risk-significant plant configurations.

- Configuration Risk Management (Tier 3):

  The licensee has a program in place to ensure compliance with 10 CFR 50.65(a)(4) to assess and manage the risk from maintenance activities. This program can support the licensee's decision in selecting the appropriate actions to control risk for most cases in which a risk-informed TS is entered.

  The generic risk impact of the end state mode change was evaluated, subject to the following assumptions and TSTF-IG-05-02, "Implementation Guidance for TSTF-423, Revision 0" (Reference 17).

  1. The entry into the end state is initiated by the inoperability of a single train of equipment or a restriction on a plant operational parameter, unless otherwise stated in the applicable TS.
  2. The primary purpose of entering the end state is to correct the initiating condition and return to power as soon as is practical.
  3. When Mode 3 is entered as the repair end state, the time the reactor coolant pressure is above 500 pounds per square inch gauge (psig) will be minimized. If reactor coolant pressure is above 500 psig for more than 12 hours, the associated plant risk will be assessed and managed.

These assumptions are consistent with typical entries into Mode 3 for short duration repairs, which is the intended use of the TS end state changes. The NRC staff concludes that going to Mode 3 (hot shutdown) instead of going to Mode 4 (cold shutdown) to carry out equipment repairs, which are of short duration, does not have any adverse effect on plant risk.

3.2 Assessment of TS Changes

Addition of Note Regarding LCO 3.0.4.a:

The existing TSs for HNP, Units 1 and 2, include the following requirement in LCO 3.0.4:

> When an LCO is not met, entry into a MODE or other specified Condition in the Applicability shall only be made:
>
> a. 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,
>
> b. After performance of a risk assessment addressing inoperable systems and components, consideration of the results, determination of the acceptability of entering the MODE or other specified condition in the Applicability, and establishment of risk management actions, if appropriate; exceptions to this specification are stated in the individual Specifications, or
>
> c. When an allowance is stated in the individual value, parameter, or other Specification.
>
> This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the unit.

Adoption of TSTF-423 requires the following Note be added to each Required Action where the end state is changed to Mode 3:

> LCO 3.0.4.a is not applicable when entering MODE 3.

The Note prohibits entry into Mode 3 within the applicability using the provision of LCO 3.0.4.a. The purpose of this Note is to provide assurance that entry into Mode 3 is not made without the appropriate risk assessment described in LCO 3.0.4.b.

The addition of this Note is acceptable because it prevents an inappropriate use of the LCO 3.0.4.a allowance to go into Mode 3 with the specified system being inoperable.

Since the basis for the Note is the same for all affected HNP LCOs, the NRC staff's discussion on the basis for acceptance is not repeated in each assessment below. In most cases, Units 1 and 2 are identical. Where differences exist, they will be noted below.

3.2.1 TS 3.3.8.2, "Reactor Protection System (RPS) Electric Power Monitoring"

The Reactor Protection System (RPS) Electric Power Monitoring System is provided to isolate the RPS bus from the normal uninterruptible power supply or an alternate power supply in the event of over voltage, under voltage, or under frequency. This system protects the load connected to the RPS bus against unacceptable voltage and frequency conditions and forms an important part of the primary success path of the essential safety circuits. Some of the essential equipment powered from the RPS buses includes the RPS logic, scram solenoids, and various valve isolation logic.

Proposed Modifications for End State Required Actions and Completion Times:

Current TS 3.3.8.2 Required Actions C.1 and C.2 state:

C. 1 Be in MODE 3

AND

C.2 Be in MODE 4

Revised TS 3.3.8.2 Required Action C.1 would state:

---NOTE-------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

C.1 Be in MODE 3

Additionally, TS 3.3.8.2 Required Action C.2 and CT C.2 would be deleted.

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.3.8.2, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

To reach Mode 3, per the TSs, there must be a functioning power supply with protective circuitry in operation. However, the overvoltage, undervoltage, or underfrequency condition must exist for an extended time period to cause damage. There is a low probability of this occurring in the short period of time that the plant would remain in Mode 3 without this protection.

The specific failure condition of interest is not risk significant in BWR PRAs. If the required restoration actions cannot be completed within the specified time, going into Mode 4 at HNP would cause loss of the high pressure reactor core isolation cooling (RCIC) system and loss of the power conversion system (condenser/feedwater) and would require activating the RHR system. In addition, Emergency Operating Procedures (EOPs) direct the operator to take control of the depressurization function if low pressure injection/spray systems are needed for reactor pressure vessel (RPV) water makeup and cooling.

Based on the low probability of loss of the RPS power monitoring system during the infrequent and limited time in Mode 3, and the number of systems available in Mode 3, the NRC staff has determined that the risks of staying in Mode 3 are approximately the same as, and in some cases lower than, the risks of going to Mode 4 end state; therefore, the NRC staff concludes that the change is acceptable.

3.2.2 <u>TS 3.5.1, "Emergency Core Cooling System (ECCS) – Operating"</u>

The emergency core cooling system (ECCS) is designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a loss-of-coolant accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS network consists of the high pressure coolant injection (HPCI) system, the core spray (CS) system, the low pressure core injection (LPCI) mode of the RHR system, and the automatic depressurization system (ADS). The suppression pool provides the required source of water for the ECCS. Although no credit is taken in the safety analyses for the condensate storage tank, it is capable of providing a source of water for the HPCI and CS systems.

<u>Proposed Modifications for End State Required Actions and Completion Times</u>:

Current TS 3.5.1 Required Actions B.1 and B.2 state:

B.1 Be in MODE 3

<u>AND</u>

B.2 Be in MODE 4

Revised TS 3.5.1 Required Action B.1 would state:

--NOTE---
LCO 3.0.4.a is not applicable when entering MODE 3.

B.1 Be in MODE 3

Additionally, TS 3.5.1 Required Action B.2 and CT B.2 would be deleted.

New Condition E is proposed as follows (existing Condition E is renumbered as Condition F):

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

Current TS 3.5.1 Condition E (renumbered as Condition F) states:

Two or more ADS valves inoperable.

<u>OR</u>

Required Action and associated Completion Time of Condition C or D not met.

Revised TS 3.5.1 Condition E (renumbered as Condition F) deletes the OR statement and is modified to state:

Two or more ADS values inoperable.

Current TS 3.5.1 Required Action E.1 and E.2 are renumbered as F.1 and F.2.

Current TS 3.5.1 Condition F is renumbered to be new Condition G with no change in Required Actions, but TS 3.5.1 Required Action F.1 is renumbered to G.1.

Deviations from TSTF-423-A, Revision 1, or the STS:

HNP TS 3.5.1 does not include conditions that are equivalent to STS Conditions E and F. The changes identified in TSTF-423-A, Revision 1, for STS 3.5.1, new Condition G are, therefore, incorporated into HNP, Units 1 and 2, TS 3.5.1 as new Condition E, and references to Conditions E and F are omitted. Similarly, changes identified in TSTF-423-A for STS 3.5.1, new Condition H, are incorporated into HNP, Units 1 and 2, TS 3.5.1, new Condition F. The end state CT for modified Conditions E and F in HNP, Units 1 and 2, TS 3.5.1, are consistent with those evaluated in TR NEDC-32988 and provided in TSTF-423-A, Revision 1.

NRC Staff Assessment:

The BWROG performed a comparative PRA evaluation in NEDC-32988-A of the core damage risks of operation in the current Mode 4 end state and the proposed Mode 3 end state. The NRC staff's conclusion described in the safety evaluation (Reference 18) for NEDC-32988, Revision 2, on the BWROG PRA evaluation indicates that the core damage risks are lower in Mode 3 than in Mode 4. For HNP, going to Mode 4 for one ECCS subsystem would cause loss of the high pressure core cooling HPCI/RCIC systems and loss of the power conversion system (condenser/feedwater) and would require activating the RHR system. In addition, plant EOPs direct the operator to take control of the depressurization function if low pressure injection/spray systems are needed for RPV water makeup and cooling.

Based on the low probability of loss of the reactor coolant inventory and the number of systems available in Mode 3, the NRC staff concludes that the risks of staying in Mode 3 are approximately the same as, and in some cases lower than, the risks of going to Mode 4 end state; therefore, the change is acceptable.

Additionally, the NRC staff reviewed the differences between the HNP TSs and the TSs in the TSTF-423 safety evaluation regarding the ECCS and determined that the differentiation does not invalidate the applicability of the TSTF-423 changes; therefore, the NRC staff's assessment for the proposed change remains unaffected.

3.2.3 TS 3.5.3, "RCIC System"

The RCIC system is not part of the ECCS; however, the RCIC system is included with the ECCS section because of its similar functions.

The RCIC system is designed to operate either automatically or manually following RPV isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of the RPV water level. Under these conditions, the HPCI and RCIC systems perform similar functions.

Proposed Modifications for End State Required Actions and Completion Times:

Current TS 3.5.3 Required Action B.1 and B.2 state:

B.1 Be in Mode 3

AND

B.2 Reduce reactor steam dome pressure to $\leq$ 150 psig

Revised TS 3.5.3 Required Action B.1 would state:

B.1 ------------------------------------NOTE--
LCO 3.0.4.a is not applicable when entering MODE 3.

Be in Mode 3

Additionally, TS 3.5.1 Required Action B.2 and CT B.2 would be deleted.

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.5.3, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

This change would allow the inoperable RCIC system to be repaired in a plant operating mode with lower risk and without challenging the normal shutdown systems. NEDC-32988-A did a comparative PRA evaluation of the core damage risks of operation in the current end state and in the Mode 3 end state. The evaluation indicates that the core damage risks are lower in Mode 3 than in Mode 4. Going to Mode 3 with reactor steam dome pressure less than or equal to 150 psig for inoperability of RCIC would also cause loss of the high pressure steam-driven injection system (RCIC/HPCI) and loss of the power conversion system (condenser/feedwater) and would require activating the RHR system. In addition, EOPs direct the operator to take control of the depressurization function if low pressure injection/spray systems are needed for RPV water makeup and cooling. Based on the low probability of loss of the necessary overpressure protection function and the number of systems available in Mode 3, the NRC staff concludes that the risks of staying in Mode 3 are approximately the same as, and in some cases lower than, the risks of going to Mode 4 end state; therefore, the change is acceptable.

3.2.4 TS 3.6.1.7, "Reactor Building-to-Suppression Chamber Vacuum Breakers"

The function of the reactor building-to-suppression chamber vacuum breakers is to relieve vacuum when primary containment depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is mitigated by flow through the reactor building-to-suppression chamber vacuum breakers and through the suppression-chamber-to-drywell vacuum breakers. The design of the external (reactor building-to-suppression chamber) vacuum relief provisions consists of two vacuum breakers (a mechanical vacuum breaker and an air-operated butterfly valve) located in series in each of two lines from the reactor building to the suppression chamber airspace.

Proposed Modifications for End State Required Actions and Completion Times:

New TS 3.6.1.7 Condition D is added as follows:

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

Current Condition D is renumbered to be new Condition E with no change to the end state. Required Action D.1 is renumbered to be E.1.

Current Condition E (renumbered as Condition F) states:

> Required Action and Associated Completion Time not met.

Revised Condition E (renumbered as Condition F) would state:

> Required Action and Associated Completion Time of Conditions A, B, or E not met.

Current Required Action E.1 and E.2 are renumbered to be F.1 and F.2.

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.6.1.7, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

NEDC-32988-A has determined that the specific failure condition of interest is not risk significant in BWR PRAs. The reduced end state would only be applicable to the situation where the vacuum breaker(s) in one line with one or more reactor building-to-suppression chamber vacuum breakers inoperable for opening with the remaining operable vacuum breakers capable of providing the necessary vacuum relief function. The existing end state remains unchanged, as established by new Condition D, for conditions involving one line with one or more vacuum breakers inoperable for opening, since they are needed in Modes 1, 2, and 3. In Mode 3, for other accident considerations, HPCI, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and decay heat removal. Additionally, the EOPs direct the operators to take control of the depressurization function if low pressure injection/spray systems are needed for reactor coolant makeup and cooling. Because

DID is maintained with respect to water makeup and decay heat removal by remaining in Mode 3, the NRC staff concludes that the change is acceptable.

3.2.5 TS 3.6.1.8, "Suppression Chamber-to-Drywell Vacuum Breakers"

The function of the suppression chamber-to-drywell vacuum breakers is to relieve vacuum in the drywell. There are 12 internal vacuum breakers located on the vent header of the vent system between the drywell and the suppression chamber, which allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Therefore, suppression chamber-to-drywell vacuum breakers prevent an excessive negative differential pressure across the wetwell drywell boundary. Each vacuum breaker is a self-actuating valve, similar to a check valve, which can be remotely operated for testing purposes.

Proposed Modifications for End State Required Actions and Completion Times:

New TS 3.6.1.8 Condition B is added, as follows:

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

Current Condition B is renumbered to be new Condition C with no change to the end state. Required Action B.1 is renumbered as C.1

Current TS 3.6.1.8 Condition C (renumbered as Condition D) states:

Required Action and associated Completion Time not met.

Revised TS 3.6.1.8 Condition C (renumbered as Condition D) would state:

Required Action and associated Completion Time of Condition C not met.

Current Required Actions C.1 and C.2 are renumbered as D.1 and D.2.

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.6.1.8, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

NEDC-32988-A has determined that the specific failure of interest is not risk significant in BWR PRAs. The reduced end state would only be applicable to the situation where one required suppression chamber-to-drywell vacuum breaker is inoperable for opening, with the remaining operable vacuum breakers capable of providing the necessary vacuum relief function, since they are required in Modes 1, 2, and 3. By remaining in Mode 3, HPCI, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and decay heat removal. Additionally, the EOPs direct the operators to take control of the depressurization function if low pressure injection/spray systems are needed for RCS makeup and cooling. Therefore, DID is maintained with respect to water makeup and decay heat removal by remaining in Mode 3. The existing end state remains unchanged for conditions involving any suppression chamber-to-drywell vacuum breakers that are stuck open, as established by new Conditions C and D; therefore, the NRC staff concludes that the change is acceptable.

3.2.6 TS 3.6.2.3, "Residual Heat Removal Suppression (RHR) Pool Cooling"

Following a design-basis accident (DBA), the RHR suppression pool cooling system removes heat from the suppression pool. The suppression pool is designed to absorb the sudden input of heat from the primary system. In the long term, the pool continues to absorb residual heat generated by fuel in the reactor core. Some means must be provided to remove heat from the suppression pool so that the temperature inside the primary containment remains within design limits. This function is provided by two redundant RHR suppression pool cooling subsystems. The purpose of this LCO is to ensure that both subsystems are operable in applicable modes.

Proposed Modifications for End State Required Actions and Completion Times:

New TS 3.6.2.3 Condition B is added that states:

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

Condition B is renumbered to be new Condition C with no change to the end state. Required Action B.1 is renumbered as C.1.

Current TS 3.6.2.3 Condition C (renumbered as Condition D) states:

Required Action and associated Completion Time not met.

Revised TS 3.6.2.3 Condition C (renumbered as Condition D) would state:

Required Action and associated Completion Time of Condition C not met.

Current Required Actions C.1 and C.2 are renumbered as D.1 and D.2.

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.6.2.3, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

BWROG completed a comparative PRA evaluation of the core damage risks of operation in the current end state versus operation in the Mode 3 end state. The results described in NEDC-32988-A, and as evaluated by the NRC staff in the associated safety evaluation, indicated that the core damage risks while operating in Mode 3 (assuming the individual failure conditions) are lower or comparable to the current end state. One loop of the RHR suppression pool cooling system is sufficient to accomplish the required safety function. By remaining in Mode 3, HPCI, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and RHR. Additionally, the plant EOPs direct the operators to take control of the depressurization function if low pressure injection/spray are needed for RCS makeup and cooling. Since DID is improved with respect to water makeup and RHR by remaining in Mode 3, the NRC staff concludes that the change is acceptable.

3.2.7 TS 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray"

Following a DBA, the RHR suppression pool spray system removes heat from the suppression chamber airspace. The suppression pool is designed to absorb the sudden input of heat from the primary system from a DBA, or a rapid depressurization of the RPV, through safety/relief valves. The heat addition to the suppression pool results in increased steam in the suppression chamber, which increases primary containment pressure. Steam blowdown from a DBA can also bypass the suppression pool and end up in the suppression chamber airspace. Some means must be provided to remove heat from the suppression chamber so that the pressure and temperature inside primary containment remain within analyzed design limits. This function is provided by two redundant RHR suppression pool spray subsystems.

Proposed Modifications for End State Required Actions and Completion Times:

Current TS 3.6.2.4 Required Action C.1 and C.2 states:

C. 1 Be in MODE 3.

AND

C.2 Be in MODE 4.

Revised TS 3.6.2.4 Required Action C.1 would state:

--NOTE--------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

C.1 Be in MODE 3.

Additionally, TS 3.6.2.4 Required Action C.2 and CT C.2 would be deleted.

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.6.2.4, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

BWROG completed a comparative PRA evaluation of the core damage risks of operation in the current end state versus operation in the Mode 3 end state. The results described in NEDC-32988-A, and as evaluated by the NRC staff in the associated safety evaluation, indicated that the core damage risks while operating in Mode 3 (assuming the individual failure conditions) are lower or comparable to the current end state. The main function of the RHR suppression spray system is to remove heat from the suppression chamber so that the pressure and temperature inside primary containment remain within analyzed design limits. The RHR suppression spray system was designed to mitigate potential effects of a postulated DBA, that is, a large break LOCA, which is assumed to occur concurrently with the most limiting single failure and conservative inputs, such as for initial suppression pool water volume and temperature. Under the conditions assumed in the DBA, steam blown down from the break could bypass the suppression pool and end up in the suppression chamber air space, and the RHR suppression spray system could be needed to condense such steam so that the pressure and temperature inside primary containment remain within analyzed design-basis limits. However, the frequency of a DBA is very small, and the containment has considerable margin to fail above the design limits. For these reasons, the unavailability of one or both RHR suppression spray subsystems has no significant impact on CDF or LERF, even for accidents initiated during operation at power. Therefore, it is very unlikely that the RHR suppression spray system will be challenged to mitigate an accident occurring during power operation. This probability becomes extremely unlikely for accidents that would occur during a small fraction of the year (less than 3 days) during which the plant would be in Mode 3 (associated with lower initial energy level and reduced decay heat load as compared to power operation) to repair the failed RHR suppression spray system. The argument for staying in Mode 3 instead of going to Mode 4 to repair the RHR suppression pool spray system (one or both trains) is also supported by DID considerations; therefore, the NRC staff concludes that the change is acceptable.

3.2.8 TS 3.6.4.1, "Secondary Containment"

The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a DBA. In conjunction with operation of the SGT system and closure of certain valves whose lines penetrate the secondary containment, the secondary containment is designed to reduce the activity level of the fission products prior to release to the environment and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside primary containment.

Proposed Modifications for End State Required Actions and Completion Times:

Current TS 3.6.4.1 Required Action B.1 and B.2 states:

B.1 Be in MODE 3.

AND

B.2 Be in MODE 4.

Revised TS 3.6.4.1 Required Action B.1 would state:

---NOTE--------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

B.1 Be in MODE 3.

Additionally, TS 3.6.4.1 Required Action B.2 and CT B.2 would be deleted.

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.6.4.1, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

This LCO entry condition does not include gross leakage through an un-isolable release path. BWROG concluded in NEDC-32988-A that previous generic PRA work related to Appendix J to 10 CFR 50 requirements has shown that containment leakage is not risk significant. The primary containment and all other primary and secondary containment-related functions would still be operable, including the SGT system, thereby minimizing the likelihood of an unacceptable release. By remaining in Mode 3, HPCI, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and RHR. Additionally, the plant EOPs direct the operators to take control of the depressurization function if low pressure injection/spray is needed for RCS makeup and cooling. Therefore, the NRC staff concludes that the change is acceptable because DID is improved with respect to water makeup and RHR by remaining in Mode 3.

The NRC staff notes that as stated in the safety evaluation for NEDC-32988-A, the NRC staff's approval relies upon the primary containment and all other primary and secondary containment-related functions to still be operable, including the SGT system, for maintaining DID while in Mode 3.

3.2.9 TS 3.6.4.3, "Standby Gas Treatment (SGT) System"

The function of the SGT system is to ensure that radioactive materials that leak from the primary containment into the secondary containment following a DBA are filtered and adsorbed prior to exhausting to the environment. The Unit 1 and Unit 2 SGT systems each consist of two fully redundant subsystems, each with its own set of dampers, charcoal filter train, and controls.

As stated in the application, STS 3.6.4.3, Condition A, provides requirements when one standby gas treatment (SGT) subsystem is inoperable. For HNP, the Unit 1 and Unit 2 SGT systems each consist of two fully redundant subsystems, each with its own set of dampers, charcoal filter train, and controls. The Unit 1 and Unit 2 SGT systems are shared between Units 1 and 2, and the SGT system on both units automatically starts and operates in response to actuation signals indicative of conditions or an accident that could require operation of the system.

Proposed Modifications for End State Required Actions and Completion Times:

Current TS 3.6.4.3 Required Action C.1 and C.2 states:

C. 1 Be in MODE 3.

AND

C.2 Be in MODE 4.

Revised TS 3.6.4.3 Required Action C.1 would state:

---NOTE-------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

C.1 Be in MODE 3.

Additionally, TS 3.6.4.3 Required Action C.2 and CT C.2 would be deleted.

Current TS 3.6.4.3 Required Action E.1 states:

E.1 Enter LCO 3.0.3

Revised TS 3.6.4.3 Required Action E.1 would state:

---NOTE-------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

E.1 Be in MODE 3.

Current TS 3.6.4.3 CT E.1 states:

Immediately

Revised TS 3.6.4.3 CT E.1 would state:

12 hours

Deviations from TSTF-423-A, Revision 1, or the STS:

The licensee states that as a result of the ability to share the four HNP, Units 1 and 2, SGT subsystems between units, additional combinations of SGT equipment inoperability are possible that are not reflected in STS 3.6.4.3, and the requirements in STS 3.6.4.3, Condition A, are represented in HNP, Units 1 and 2, TS 3.6.4.3, as Conditions A and B. This difference results in renumbering of the Conditions in HNP, Units 1 and 2, TS 3.6.4.3, relative to those in TSTF-423-A. Changes identified in TSTF-423-A, Revision 1, for STS 3.6.4.3, Condition B, are incorporated into the HNP, Units 1 and 2, TS 3.6.4.3, Condition C. The changes identified in TSTF-423-A for STS 3.6.4.3, Condition D, are incorporated into HNP TS 3.6.4.3, Condition E.

The end state Completion Time for modified Conditions C and E in HNP, Units 1 and 2, TS 3.6.4.3, are consistent with those evaluated in NEDC-32988-A, and provided in TSTF-423-A, Revision 1.

NRC Staff Assessment:

The unavailability of one or both SGT subsystems has no impact on CDF or LERF, independent of the mode of operation at the time of the accident. Furthermore, the challenge frequency of the SGT system (i.e., the frequency with which the system is expected to be challenged to mitigate offsite radiation releases resulting from materials that leak from the primary to the secondary containment above TS limits) is less than 1.0E-6/year (yr). Consequently, the conditional probability that this system will be challenged during the repair time interval, while the plant is at either the current or the proposed end state (i.e., Mode 4 or Mode 3, respectively) is less than 1.0E-8/yr. This probability is considerably smaller than the probabilities considered negligible in RG 1.177 for much higher consequence risks such as large early release.

The results described in NEDC-32988-A, and as evaluated by the NRC staff in the associated safety evaluation, summarize the NRC staff's risk argument for approval of TS LCO 3.6.4.3, "Standby Gas Treatment (SGT) System." The argument for staying in Mode 3 instead of going to Mode 4 to repair the SGT system (one or both trains) is also supported by DID considerations. The NRC staff's evaluation makes a comparison between the current (Mode 4) and proposed (Mode 3) end state with respect to the means available to perform critical functions (i.e., functions contributing to the DID philosophy) whose success is needed to prevent core damage and containment failure and mitigate radiation releases. The risk and DID arguments used according to the integrated decisionmaking process of RG 1.174 and RG 1.177 support the conclusion that Mode 3 is as safe as Mode 4 for repairing an inoperable SGT system; therefore, the NRC staff concludes that the change is acceptable.

Additionally, the NRC staff has reviewed the design differences between the HNP SGT system and the SGT systems described in the TSTF-423 safety evaluation, and determined that the differentiation does not invalidate the applicability of the TSTF-423 changes; therefore, the NRC staff's assessment for the proposed change remains unaffected.

3.2.10 TS 3.7.1, "Residual Heat Removal Service Water (RHRSW) System"

The residual heat removal service water (RHRSW) system is designed to provide cooling water for the RHR system heat exchangers required for a safe reactor shutdown following a DBA or transient. The RHRSW system is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR system.

Proposed Modifications for End State Required Actions and Completion Times:

New TS 3.7.1 Condition D is added that states:

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

Current Condition D and Required Action D.1 is renumbered to be new Condition E and Required Action E.1 with no change to the end state.

Current TS 3.7.1 Condition E (renumbered as Condition F) states:

Required Action and associated Completion Time not met.

Revised TS 3.7.1 Condition E (renumbered as Condition F) would state:

Required Action and associated Completion Time of Condition E not met.

Current Condition E and Required Action E.1 and E.2 are renumbered to be new Condition F and Required Actions F.1 and F.2 with no change to the end state.

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.7.1, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

BWROG performed a comparative PRA evaluation of the core damage risks when operating in the current end state versus the proposed Mode 3 end state. The results indicated that the core damage risks while operating in Mode 3 (assuming the individual failure conditions) are lower or comparable to the current end state. By remaining in Mode 3, HPCI, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and decay heat removal. Additionally, the EOPs direct the operators to take control of the depressurization function if low pressure injection/spray are needed for RCS makeup and cooling. Therefore, DID is improved with respect to water makeup and decay heat removal by remaining in Mode 3, and the required safety function can still be performed with the RHRSW subsystem components that are still operable; therefore, the NRC staff concludes that the change is acceptable.

3.2.11 TS 3.7.2, "Plant Service Water (PSW) System and Ultimate Heat Sink (UHS)"

The PSW system is designed to provide cooling water for the removal of heat from equipment, such as the diesel generators (DGs), RHR pump coolers, and room coolers for ECCS equipment required for a safe reactor shutdown following a DBA or transient. The PSW system also provides cooling to unit components, as required, during normal operation. Upon receipt of a loss of offsite power or LOCA signal, nonessential loads are automatically isolated, the essential loads are automatically divided between PSW Divisions 1 and 2, and one PSW pump is automatically started in each division.

Proposed Modifications for End State Required Actions and Completion Times:

New TS 3.7.2 Condition E is added that states:

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

Current Condition E and Required Action E.1 is renumbered to be new Condition F and Required Action F.1 with no change to the end state.

Current TS 3.7.2, Condition F, states:

> Required Action and associated Completion Time of Condition A, B, C, D, or E not met.
>
> OR
>
> Both PSW subsystems inoperable for reasons other than Conditions C and D.
>
> OR
>
> UHS inoperable.

Revised TS 3.7.2, Condition F (renumbered as Condition G), would state:

> Required Action and associated Completion Time of Condition F not met.
>
> OR
>
> Both PSW subsystems inoperable for reasons other than Conditions C or D.
>
> OR
>
> UHS inoperable.

Current Required Actions F.1 and F.2 are renumbered to Required Action G.1 and G.2 with no change to the end state.

Deviations from TSTF-423-A, Revision 1, or the STS:

HNP TS 3.7.2, Conditions B and D, provide requirements when one PSW turbine building isolation valve is inoperable in one or each PSW subsystem. These conditions reflect plant-specific features of the HNP, Units 1 and 2, design, and STS 3.7.2 does not include similar conditions. Additionally, STS TS 3.7.2 includes conditions for inoperability of cooling tower fans (Condition C) and ultimate heat sink water temperature (Condition D) that are not included or necessary in the HNP, Units 1 and 2, TSs due to plant-specific differences in the PSW system design.

These differences result in renumbering of the conditions in HNP TS 3.7.2 relative to those in TSTF-423-A. Changes identified in TSTF-423-A, Revision 1, for STS 3.7.2, Condition C, are incorporated into HNP, Units 1 and 2, TS 3.7.5, Condition E, and plant-specific Conditions B and D are adopted within this condition. The referenced conditions in Condition G are also renumbered to reflect these differences. The end state Completion Time for modified Conditions E and G in HNP, Units 1 and 2, TS 3.7.2 are consistent with those evaluated in TR NEDC-32988 and provided in TSTF-423-A, Revision 1.

NRC Staff Assessment:

BWROG performed a comparative PRA evaluation of the core damage risks associated with operating in the current end state versus the proposed Mode 3 end state. The results indicated that the core damage risks while operating in Mode 3 (assuming the individual failure conditions) are lower or comparable to the current end state. With one PSW pump inoperable in one or more subsystems, the remaining pumps are adequate to perform the PSW heat removal function. By remaining in Mode 3, HPCI, RCIC, and the power conversion system (condensate/feedwater) remain available for water makeup and decay heat removal. Additionally, the EOPs direct the operators to take control of the depressurization function if low pressure injection/spray are needed for RCS makeup and cooling. Therefore, DID is improved with respect to water makeup and decay heat removal by remaining in Mode 3; therefore, the NRC staff concludes that the change is acceptable.

Additionally, the NRC staff has reviewed the design differences between HNP and the systems described in the TSTF-423-A safety evaluation regarding the PSW system and UHS. The NRC staff determined that the differentiation does not invalidate the applicability of the TSTF-423 changes; therefore, the staff's assessment for the proposed change remains unaffected.

3.2.12 TS 3.7.4, "Main Control Room Environmental Control (MCREC) System"

The main control room environmental control (MCREC) system provides a radiologically controlled environment from which the unit can be safely operated following a DBA.

The safety-related function of MCREC system includes two independent and redundant high efficiency air filtration subsystems for emergency treatment of recirculated air and outside supply air. Each subsystem consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section, a second HEPA filter, a booster fan, and the associated ductwork and dampers.

Proposed Modifications for End State Required Actions and Completion Times:

Current TS 3.7.4 Required Action C.1 and C.2 states:

C.1 Be in MODE 3

AND

C.2 Be in MODE 4.

Revised TS 3.7.4 Required Action C.1 would state:

---NOTE-------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

C.1 Be in MODE 3.

Additionally, TS 3.7.4 Required Action C.2 and CT C.2 would be deleted.

Current TS 3.7.4 Required Action E.1 states:

E.1 Enter LCO 3.0.3.

Current TS 3.7.4 CT E.1 states:

Immediately

Revised TS 3.7.4 Required Action E.1 would state:

---NOTE-------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

E.1 Be in MODE 3.

Revised TS 3.7.4 CT E.1 would state:

12 hours

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.7.4, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

The unavailability of one or both MCREC subsystems has no significant impact on CDF or LERF, irrespective of the mode of operation at the time of the accident. Furthermore, the challenge frequency of the MCREC system (i.e., the frequency with which the system is expected to be challenged to maintain a dose of less than 5 rem in the main control room following a DBA with radiation leaking from the containment) is less than 1.0E-6/yr. The challenge frequency will ultimately adjust the plant risk to a higher value for a specified period of time during the repair time interval. The change in plant risk can be quantified for this specific 7-day interval and produce a conditional event probability. The conditional event probability that the MCREC system will be challenged during the repair time interval while the plant is in Mode 4, or in the proposed Mode 3, is less than 1.0E-8. This probability is considerably smaller than probabilities considered "negligible" in RG 1.177 for much higher consequence risks, such as large early release.

The NRC safety evaluation for TR NEDC-32988, Revision 2, summarizes the NRC staff's risk argument for approval of TS 4.5.1.16, and LCO 3.7.4, "Main Control Room Environmental Control (MCREC) System (BWR-4 only)." The argument for staying in Mode 3 instead of going to Mode 4 to repair the MCREC system (one or both trains) is also supported by DID considerations. Section 6.2 of Reference 17 makes a comparison between the Mode 3 and Mode 4 end state with respect to the means available to perform critical functions (i.e., functions contributing to the DID philosophy) whose success is needed to prevent core damage and containment failure and mitigate radiation releases. The risk and DID arguments used according to the integrated decisionmaking process of RG 1.174 and RG 1.177 support the conclusion that Mode 3 is as safe as Mode 4 for repairing an inoperable MCREC system. Based on the above, the NRC staff concludes that the change is acceptable.

3.2.13 <u>TS 3.7.5, "Control Room Air Conditioning (AC) System"</u>

The control room air conditioning (AC) system portion of the MCREC (hereafter referred to as the control room AC system) provides temperature control for the control room following isolation of the control room. The control room AC system consists of three 50 percent capacity subsystems that provide cooling and heating of control room supply air. Each subsystem consists of an air handling unit (i.e., cooling coils and fan), water cooled condensing units, refrigerant compressors, ductwork, dampers, and instrumentation and controls to provide for control room temperature control.

<u>Proposed Modifications for End State Required Actions and Completion Times</u>:

Current TS 3.7.5 Required Action D.1 and D.2 states:

D.1 Be in MODE 3.

<u>AND</u>

D.2 Be in MODE 4.

Revised TS 3.7.5 Required Action D.1 would state:

---NOTE--------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

D.1 Be in MODE 3.

Additionally, TS 3.7.5 Required Action D.2 and CT D.2 would be deleted.

Deviations from TSTF-423-A, Revision 1, or the STS:

STS 3.7.5 assumes that the control room AC system design consists of two 100 percent control room AC subsystems. For HNP, Units 1 and 2, the control room AC safety function is provided by three 50 percent capacity subsystems. With one control room AC subsystem inoperable, the two remaining control room AC subsystems provide 100 percent of the required capacity. With two subsystems inoperable, control room AC capability is diminished but remains capable of providing 50 percent of the required capacity. This plant-specific design difference, and the historical licensing basis for HNP, Units 1 and 2, are reflected in HNP, Units 1 and 2, TS 3.7.5.

The changes identified in TSTF-423-A, Revision 1, for STS 3.7.5, Condition B, are incorporated into HNP, Units 1 and 2, TS 3.7.5, as Condition D. The end state CT for modified Conditions D in HNP, Units 1 and 2, TS 3.7.5 are consistent with those evaluated in TR NEDC-32988 and provided in TSTF-423-A, Revision 1.

STS 3.7.5 Condition D provides actions when all control room AC subsystems are inoperable. The analogous Condition was removed from HNP, Units 1 and 2, TS 3.7.5, in License Amendment Nos. 270 and 2142 (HNP, Units 1 and 2, respectively (Reference 18), with the adoption of changes described in TSTF-477. The changes identified in TSTF-423-A, Revision 1, for STS 3.7.5, Condition D, are, therefore, not incorporated into HNP, Units 1 and 2, TS 3.7.5.

NRC Staff Assessment:

The unavailability of one or more AC subsystems has no significant impact on CDF or LERF, irrespective of the mode of operation at the time of the accident. Additionally, the challenge frequency of the AC system (i.e., the frequency with which the system is expected to be challenged to provide temperature control for the control room following control room isolation after a DBA that leads to core damage) is less than 1.0E-6/yr. The challenge frequency will ultimately adjust the plant risk to a higher value for a specified period of time during the repair time interval. The change in plant risk can be quantified for this specific 7-day interval and produce a conditional event probability. The conditional event probability that the AC subsystem will be challenged during the repair time interval while the plant is in Mode 4, or in the proposed Mode 3, is less than 1.0E-8. This probability is considerably smaller than probabilities considered "negligible" in RG 1.177 for much higher consequence risks, such as large early release.

The NRC staff's safety evaluation of TR NEDC-32988 summarizes its risk basis for approval of LCO 3.7.4, "Control Room Air Conditioning (CRAC) System." The NRC staff determined that the CRAC system is similar to the HNP CRAC system. The basis for staying in Mode 3 instead of going to Mode 4 to repair the CRAC system (one or both trains) is supported by DID considerations. Section 6.2 of the NRC staff's safety evaluation for NEDC-32988-A, makes a comparison between the Mode 3 and Mode 4 end states with respect to the means available to perform critical functions (i.e., functions contributing to the DID philosophy) whose success is needed to prevent core damage and containment failure and to mitigate radiation releases. The risk and DID arguments used according to the integrated decisionmaking process of RG 1.174 and RG 1.177 support the conclusion that Mode 3 is as safe as Mode 4 for repairing an inoperable control room AC system. Therefore, the NRC staff concludes that the change is acceptable.

Additionally, the NRC staff's review of the design and functions of the HNP AC system versus that assessed in the NRC staff's safety evaluation for NEDC-32988-A for AC determined that slight differences have no impact on the NRC staff's assessment since the functions of both systems are similar.

3.2.14 TS 3.7.6, "Main Condenser Offgas"

During operation, steam from the low pressure turbine is exhausted directly into the condenser. Air and noncondensable gases are collected in the condenser and then exhausted through the steam jet air ejectors (SJAEs) to the main condenser offgas system. The offgas from the main condenser normally includes radioactive gases. The main condenser offgas system has been incorporated into the unit design to reduce the gaseous radwaste emission. This system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. The gaseous mixture is cooled by the offgas condenser; the water and condensables are stripped out by the offgas condenser and moisture separator. The radioactivity of the remaining gaseous mixture (i.e., the offgas recombiner effluent) is monitored downstream of the moisture separator prior to entering the holdup line.

Proposed Modifications for End State Required Actions and Completion Times:

Current TS 3.7.6 Required Action B.3.1 and B.3.2 states:

B.3.1 Be in MODE 3.

AND

B.3.2 Be in MODE 4.

Revised TS 3.7.6 Required Action B.3.1 will be renumbered to Required Action B.3 and would state:

---NOTE--------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

3.1 Be in MODE 3.

Additionally, TS 3.7.6 Required Action B.3.2 and CT B.3.2 would be deleted.

Deviations from TSTF-423-A, Revision 1, or the STS:

For TS 3.7.6, the application does not specify any deviation from TSTF-423, Revision 1, or the STS.

NRC Staff Assessment:

The failure to maintain the gross gamma activity rate of the noble gases in the main condenser offgas system within limits has no significant impact on CDF or LERF, irrespective of the mode of operation at the time of the accident. Additionally, the challenge frequency of the main condenser offgas system (i.e., the frequency with which the system is expected to be challenged to mitigate offsite radiation releases following a DBA) is less than 1.0E-6/yr. The challenge frequency will ultimately adjust the plant risk to a higher value for a specified period of time during the repair time interval. The change in plant risk can be quantified for this specific 7-day interval and produce a conditional event probability. The conditional event probability that the offgas system will be challenged during the repair time interval while the plant is in Mode 4, or in the proposed Mode 3, is less than 1.0E-8. This probability is considerably smaller than probabilities considered "negligible" in RG 1.177 for much higher consequence risks, such as large early release.

The NRC staff's safety evaluation of NEDC-32988-A summarizes the NRC staff's risk argument for approval of TS 4.5.1.18 and LCO 3.7.6, "Main Condenser Offgas." The argument for staying in Mode 3 instead of going to Mode 4 to repair the main condenser offgas system (one or both trains) is also supported by DID considerations. Section 6.2 of Reference 17 makes a comparison between the Mode 3 and Mode 4 end state with respect to the means available to perform critical functions (i.e., functions contributing to the DID philosophy) whose success is needed to prevent core damage and containment failure and mitigate radiation releases. The risk and DID arguments used according to the integrated decisionmaking process of RG 1.174 and RG 1.177 support the conclusion that Mode 3 is as safe as Mode 4 for repairing an inoperable main condenser offgas system. Therefore, the NRC staff concludes that the change is acceptable.

3.2.15 TS 3.8.1, "AC Sources - Operating"

The Unit 1 Class 1E AC electrical power distribution system AC sources consist of the offsite power sources (preferred power sources, normal and alternate) and the onsite standby power sources (DGs 1A, 1B, and 1C for HNP, Unit 1, and DGs 2A, 2C, and 1B for HNP, Unit 2). As required by the 1967 proposed 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits" General Design Criterion (GDC) 39, "Emergency Power for Engineered Safety Features (Category A)" (Reference 19) for HNP, Unit 1, and 10 CFR 50, Appendix A, GDC 17, "Electric power systems," for HNP, Unit 2, the design of the AC electrical power system provides independence and redundancy to ensure an available source of power to the engineered safety feature (ESF) systems.

The Class 1E AC distribution system is divided into redundant load groups, so loss of any one group does not prevent the minimum safety functions from being performed. Each load group has connections to two preferred offsite power supplies and a single DG.

Offsite power is supplied to the 230 kV and 500 kV switchyards from the transmission network by eight transmission lines. From the 230 kV switchyards, two electrically and physically separated circuits provide AC power through startup auxiliary transformers 1C and 1D, to 4.16 kV ESF buses 1E, 1F, and 1G for HNP, Unit 1, and buses 2E, 2F, and 2G for HNP, Unit 2.

Proposed Modifications for End State Required Actions and Completion Times:

Current TS 3.8.1 Required Action H.1 and H.2 states:

H.1 Be in MODE 3.

AND

H.2 Be in MODE 4.

Revised TS 3.8.1 Required Action H.1 would state:

---NOTE---------------------------------------
LCO 3.0.4.a is not applicable when entering MODE 3.

H.1 Be in MODE 3.

Additionally, TS 3.8.1 Required Action H.2 and CT H.2 would be deleted.

Deviations from TSTF-423-A, Revision 1, or the STS:

The format and content of Conditions for HNP TS 3.8.1 differs from that provided in TSTF-423-A, Revision 1, for STS 3.8.1. HNP, Units 1 and 2, TS 3.8.1, provides an additional condition (Condition C) for inoperability of one required DG on the opposite unit that does not appear in STS 3.8.1. HNP, Units 1 and 2, TS 3.8.1, also provides an additional condition (Condition G) for the situation where there are no DGs capable of providing power to any LPCI valve load centers that do not appear in STS 3.8.1. Additionally, STS 3.8.1 includes a condition (Condition F) for automatic load center inoperability that does not appear in HNP, Units 1 and 2, TS 3.8.1. These conditions reflect plant-specific attributes of the HNP design and/or the historical HNP licensing basis.

These differences result in renumbering of the conditions in HNP TS 3.8.1 relative to those in TSTF-423-A. Changes identified in TSTF-423-A, Revision 1, for STS 3.8.1, Condition G, are incorporated into HNP, Units 1 and 2, TS 3.8.1, Condition H, and plant-specific Conditions C and G are incorporated within this Condition. The end state Completion Time for modified Condition H in HNP, Units 1 and 2, TS 3.8.1, is consistent with those evaluated in TR NEDC-32988 and provided in TSTF-423-A, Revision 1.

NRC Staff Assessment:

Entry into any of the Conditions for the AC power sources implies that the AC power sources have been degraded, and the single failure protection for the safe shutdown equipment may be ineffective. Consequently, as specified in TS 3.8.1 at present, the plant operators must bring the plant to Mode 4 when the Required Action is not completed by the specified time for the associated action.

NEDC-32988-A provides a comparative PRA evaluation of the core damage risks of operation in the current end state and in the Mode 3 end state. Events initiated by the loss of offsite power are dominant contributors to CDF in most BWR PRAs, and the steam-driven core cooling systems (RCIC and HPCI) play a major role in mitigating these events. The evaluation indicates that the core damage risks are lower in Mode 3 than in Mode 4 for one inoperable AC power source. Going to Mode 4 for one inoperable AC power source would cause loss of the high pressure steam-driven injection system (RCIC/HPCI) and loss of the power conversion system (condenser/feedwater) and require activating the RHR system. In addition, EOPs direct the operator to take control of the depressurization function if low pressure injection/spray systems are needed for RPV water makeup and cooling.

Based on the low probability of loss of the AC power and the number of steam-driven systems available in Mode 3, the NRC staff determined that the risks of staying in Mode 3 are lower than going to Mode 4 end state. Therefore, the NRC staff concludes that the change is acceptable.

Additionally, the NRC staff's review of the design and functions of the HNP AC sources system versus that assessed in the staff's safety evaluation for NEDC-32988 for the same system determined that slight differences have no impact on the staff's assessment since functions of the safety systems specified in the assessment are identical.

3.2.16 <u>TS 3.8.4, "DC Sources - Operating"</u>

The direct current (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. As required by the 1967 proposed GDC 39 (Reference 19) for HNP, Unit 1, and, GDC 17, for HNP, Unit 2, the DC electrical power system is designed to have sufficient independence, redundancy, and testability to perform its safety functions, assuming a single failure.

The station service DC power sources provide both motive and control power to selected safety-related and nonsafety-related equipment. Each DC subsystem is energized by one 125/250 volt (V) station service battery and three 125 V battery chargers (two normally inservice chargers and one standby charger). Each battery is exclusively associated with a single 125/250 volt direct current (VDC) bus. Each set of battery chargers associated exclusively with a 125/250 VDC subsystem cannot be interconnected with any other 125/250 VDC subsystem. The normal and backup chargers are supplied from the same alternating current (AC) load groups for which the associated DC subsystem supplies the control power. The loads between the redundant 125/250 VDC subsystem are not transferable except for the ADS, the logic circuits and valves of which are normally fed from the Division 1 DC system.

<u>Proposed Modifications for End State Required Actions and Completion Times</u>:

Unit 1 is described herein; Unit 2 is similar.

Current TS 3.8.4 Required Action D.1 and D.2 states:

D.1 Be in MODE 3.

<u>AND</u>

D.2 Be in MODE 4.

Revised TS 3.8.4 Required Action D.1 would state:

---NOTE---
LCO 3.0.4.a is not applicable when entering MODE 3.

D.1 Be in MODE 3.

Additionally, TS 3.8.4 Required Action D.2 and CT D.2 would be deleted.

Deviations from TSTF-423-A, Revision 1, or the STS:

The structure and content of conditions in HNP TS 3.8.4 differ from that provided in TSTF-423-A, Revision 1, for STS 3.8.4. STS 3.8.4, Conditions A, B, and C, provide requirements when conditions exist that render one or more DG or station service DC electrical subsystems inoperable, including battery or charger inoperability. End state requirements are provided in STS 3.8.4, Condition D, when the requirements of Condition A, B, or C are not met for a station service DC electrical subsystem, and in ISTS 3.8.4, Condition E, when the requirements of Conditions A, B, or C are not met for a DG DC electrical subsystem.

Similarly, HNP TS 3.8.4, Conditions A, B, and C, provide requirements when conditions exist that render a DG or station service DC electrical subsystem inoperable. End state requirements are provided in STS 3.8.4, Condition D, when the requirements of Conditions A, B, or C are not met for a station service DC electrical subsystem, and STS 3.8.4, Condition E, provides actions when the requirements of Conditions A, B, or C are not met for a DG DC electrical subsystem.

The changes identified in TSTF-423-A, Revision 1, for STS 3.8.1, Condition D, are incorporated into HNP, Units 1 and 2, TS 3.8.1, Condition D. The end state Completion Time for modified Condition D in HNP, Units 1 and 2, TS 3.8.4, is consistent with those evaluated in TR NEDC-32988 and provided in TSTF-423-A, Revision 1.

NRC Staff Assessment:

If one of the DC electrical power subsystems is inoperable, the remaining DC electrical power subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. BWROG did a comparative PRA evaluation in NEDC-32988 of the core damage risks of operation in the current end state and in the proposed Mode 3 end state. Events initiated by the loss of offsite power are dominant contributors to CDF in most BWR PRAs, and the steam driven core cooling systems IC, RCIC, and HPCI play a major role in mitigating these events. The evaluation indicates that the core damage risks are lower in Mode 3 than in Mode 4. Going to Mode 4 for one inoperable DC power source would cause loss of the high pressure steam-driven injection system (RCIC and HPCI) and loss of the power conversion system condenser/feedwater) and require activating the RHR system. In addition, the EOPs direct the operator to take control of the depressurization function if low pressure injection/spray systems are needed for RPV water makeup and cooling. Based on the low probability of loss of the DC power and the number of systems available in Mode 3, the NRC staff determined that the risks of staying in Mode 3 are approximately the same as, and in some cases lower than, the risks of going to Mode 4 end state. Therefore, the NRC staff concludes that the change is acceptable.

Additionally, the NRC staff's review of the design and functions of the HNP AC sources system versus that assessed in the NRC staff's safety evaluation for the same system in NEDC-32988, Revision 2, determined that slight differences have no impact on the NRC staff's assessment since functions of the safety systems specified in the assessment are identical.

3.2.17 TS 3.8.7, "Distribution Systems - Operating"

The onsite Class 1E AC and DC electrical power distribution system is divided into redundant and independent AC and DC electrical power distribution subsystems.

The primary AC distribution system consists of three 4.16 kV ESF buses, each having an offsite source of power and a dedicated onsite DG source. Each 4.16 kV ESF bus is normally connected to a normal source startup auxiliary transformer (SAT) (1D). During a loss of the normal offsite power source to the 4.16 kV ESF buses, the alternate supply breaker from SAT 1C attempts to close. If all offsite sources are unavailable, the onsite emergency DGs supply power to the 4.16 kV ESF buses. The secondary plant distribution system includes 600 VAC emergency buses 1C and 1D and associated load centers and transformers.

Proposed Modifications for End State Required Actions and Completion Times:

Unit 1 is described herein; Unit 2 is similar.

Current TS 3.8.7 Required Action E.1 and E.2 states:

E.1 Be in MODE 3.

AND

E.2 Be in Mode 4.

Revised TS 3.8.7 Required Action E.1 would state:

-----NOTE-----
LCO 3.0.4.a is not applicable when entering MODE 3.

E.1 Be in MODE 3.

Additionally, TS 3.8.7 Required Action E.2 and CT E.2 would be deleted.

Deviations from TSTF-423-A, Revision 1, or the STS:

The structure and content of conditions in HNP TS 3.8.7 differ from that provided in TSTF-423-A, Revision 1, for STS 3.8.9. STS 3.8.9, Conditions A, B, and C, provide requirements when one or more AC or DC electrical power distribution subsystems are inoperable or one or more AC vital buses are inoperable. End state requirements are provided in STS 3.8.9, Condition D, when the requirements of Conditions A, B, or C are not met.

Similarly, HNP TS 3.8.7, Conditions A, B, C, and D, provide requirements when conditions exist that render a Unit 1, Unit 2, or swing bus AC or DC electrical power distribution subsystem inoperable. End state requirements are provided in HNP TS 3.8.7, Condition E, when the requirements of Conditions A, B, C, or D are not met.

The changes identified in TSTF-423-A, Revision 1, for STS 3.8.9, Condition D, are incorporated into HNP, Units 1 and 2, TS 3.8.7, Condition E. The end state Completion Time for modified Condition E in HNP, Unit 1 and 2, TS 3.8.7, is consistent with those evaluated in TR NEDC-32988 and provided in TSTF-423-A, Revision 1.

NRC Staff Assessment:

If one of the AC/DC/AC vital subsystems is inoperable, the remaining AC/DC/AC vital subsystems have the capacity to support a safe shutdown and to mitigate an accident condition. NEDC-32988, Revision 2, did a comparative PRA evaluation of the core damage risks of operation in the current end state and in the Mode 3 end state with one of the AC/DC/AC vital subsystems inoperable. Events initiated by the loss of offsite power are dominant contributors to CDF in most BWR PRAs, and the steam-driven core cooling systems (RCIC and HPCI) play a major role in mitigating these events. The evaluation indicates that the core damage risks are lower in Mode 3 than in Mode 4. Going to Mode 4 for one inoperable AC/DC/AC vital subsystem would cause loss of the high pressure steam-driven injection system (RCIC/HPCI) and loss of the power conversion system (condenser/feedwater) and require activating the RHR system. In addition, EOPs direct the operator to take control of the depressurization function if low pressure injection/spray systems are needed for RPV water makeup and cooling.

Based on the low probability of loss of the AC/DC/AC vital electrical subsystems during the infrequent and limited time in Mode 3 and the number of systems available in Mode 3, the NRC staff determined that the risks of staying in Mode 3 are approximately the same as, and in some cases lower than, the risks of going to Mode 4 end state. Therefore, the NRC staff concludes that the change is acceptable.

Additionally, the NRC staff's review of the differences in the design of HNP distribution systems versus that assessed in the NRC staff's safety evaluation for ITS distribution systems in NEDC-32988, Revision 2, has no impact on the staff's assessment since the functions of the mitigating systems specified in the assessment are similar and, therefore, the change is acceptable.

3.3 Regulatory Commitments

In its application, the licensee made the following regulatory commitments:

REGULATORY COMMITMENTS	DUE DATE/EVENT
SNC will follow the guidance established in Section 11 of NUMARC 93-01, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Nuclear Management and Resource Council, Revision 4A, April 2011.	To be implemented with amendment
SNC will follow the guidance established in TSTF-IG-05-02, Revision 2, "Implementation Guidance for TSTF-423, Revision 1, 'Technical Specifications End States, NEDC-32988-A'," with the exception the SNC will follow Regulatory Guide (RG) 1.160 in lieu of RG 1.182, and SNC will follow Revision 4A of NUMARC 93-01 in lieu of Revision 3 of NUMARC 93-01.	To be implemented with amendment

The NRC staff concludes that reasonable controls for the implementation and subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program.

3.4 Summary

Because the time spent in Mode 3 to perform the repair on any of the systems described above would be infrequent and limited, and in light of the DID considerations (discussed above and in NEDC-32988-A, and as evaluated in the NRC staff's safety evaluation for NEDC-32988), the NRC staff concludes that the proposed changes to the HNP TSs are acceptable and the requirements of the 10 CFR 50.36 continue to be met.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments on August 24, 2016. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components 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 (81 FR 7841). 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, letter to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant License Amendment Request for Adoption of TSTF-423-A, Revision 1, Technical Specification End States, NEDC-32988-A," dated December 15, 2015 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML15351A023).

2. Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Power Plant Unit 1 and 2 - License Amendment Request for Adoption of TSTF-423-A, Revision 1, Supplemental Information," dated April 11, 2016 (ADAMS Accession No. ML16102A453).

3. Technical Specifications Task Force, letter to U.S. Nuclear Regulatory Commission, "Transmittal of Revised Risk-Informed End State Travelers," dated December 22, 2009 (ADAMS Accession No. ML093570241); includes TSTF-423, Revision 1, "Technical Specifications End States, NEDC-32988-A."

4. *Federal Register*, Vol. 58, No. 139, p. 39136, "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Plants," dated July 22, 1993.

5. BWR Owners Group, NEDC-32988-A, Revision 2, "Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants," December 2002 (ADAMS Accession No. ML030170084).

6. U.S. Nuclear Regulatory Commission, Safety Evaluation of Topical Report NEDC-32988, Revision 2, "Technical Justification to Support Risk-Informed Modification to Selected Required Action End States for BWR Plants," dated September 27, 2002 (ADAMS Accession No. ML022700603).

7. U.S. Nuclear Regulatory Commission, NUREG-1433, Revision 4.0, "Standard Technical Specifications – General Electric BWR/4 Plants," April 2012 (ADAMS Accession No. ML12104A192).

8. U.S. Nuclear Regulatory Commission, NUREG-1434, Revision 4.0, "Standard Technical Specifications - General Electric BWR/6 Plants," April 2012 (ADAMS Accession No. ML12104A195)

9. *Federal Register*, Vol. 76, No. 34, p. 9614, "Notice of Availability of the Proposed Models for Plant-Specific Adoption of Technical Specifications Task Force (TSTF) Traveler TSTF-423, Revision 1, "Technical Specification End States, NEDC-32988-A,'" for Boiling Water Reactor Plants Using the Consolidated Line Item Improvement Process," dated February 18, 2011.

10. U.S. Nuclear Regulatory Commission, "Model Application and Model Safety Evaluation for Technical Specification End States, NEDC-32988-A," dated February 2, 2011 (ADAMS Accession No. ML102730688).

11. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," May 2000 (ADAMS Accession No. ML003699426).

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

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

14. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998 (ADAMS Accession No. ML003740133).

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

16. Nuclear Management and Resource Council, NUMARC 93-01, Revision 3, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," July 2000 (ADAMS Accession No. ML031500684).

17. BWR Owners Group, "TSTF-IG-05-02, Implementation Guidance for TSTF-423, Revision 0, 'Technical Specifications End States, NEDC-32988-A'," September 2005 (ADAMS Accession No. ML052700156).

18. U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, Issuance of Amendments Regarding the Control Room Air Conditioning System," dated December 10, 2014 (ADAMS Accession No. ML14279A261).

19. Proposed "General Design Criteria for Nuclear Power Plant Construction Permits," for 10 CFR Part 50, "Licensing of Production and Utilization Facilities," published July 11, 1967 (32 FR 10213) (ADAMS Accession No. ML043310029).

Principal Contributors: L. Wheeler
R. Grover

Date: December 19, 2016

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

Sincerely,

/RA/

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

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 281 to DPR-57
2. Amendment No. 225 to NPF-5
3. Safety Evaluation

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