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

March 23, 2016

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

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING THE RELOCATION OF THE PRESSURE TEMPERATURE LIMIT CURVES FOLLOWING TSTF-419 (CAC NOS. MF6063 AND MF6064)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 277 to Renewed Facility Operating License DPR-57 and Amendment No. 221 to Renewed Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated April 2, 2015, as supplemented by letters dated November 12, 2015, and February 9, 2016.

The amendments revise the HNP, Units 1 and 2, TSs as necessary to relocate the pressure and temperature (P-T or P/T) limit curves and associated references to a Pressure and Temperature Limits Report (PTLR). Specifically, the request modifies Section 1.0, "Definitions," Limiting Conditions for Operation and Surveillance Requirement Applicability Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits," and Section 5.0, "Administrative Controls," of the TS for both HNP units to delete reference to the P-T curves, and to include reference to the unit-specific PTLRs. The request also implements new P-T limits for both HNP units.

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. 277 to DPR-57
2. Amendment No. 221 to NPF-5
3. Safety Evaluation

cc w/encls: Distribution via Listserv

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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 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 277
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 April 2, 2015, as supplemented by letters dated November 12, 2015, and February 9, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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.

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) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 277, are hereby incorporated in the 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

Shawn Williams for

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

Attachment:
Changes to Renewed Facility Operating License No. DPR-57 and the Technical Specifications

Date of Issuance: March 23, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 277

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
1.1-4	1.1-4
3.4-18	3.4-18
3.4-19	3.4-19
3.4-20	3.4-20
3.4-21	3.4-21
3.4-22	3.4-22
3.4-23	5.0-21
3.4-24	5.0-22
5.0-21	5.0-23
5.0-22	

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. 277 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

1.1 Definitions (continued)

MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Section 13.6, Startup and Power Test Program, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2804 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR. The recirculation pump starting temperature requirements shall be maintained within limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. -------------NOTE-------------- Required Action A.2 shall be completed if this Condition is entered. ----------------------------------	A.1 Restore parameter(s) to within limits.	30 minutes
	AND	
Requirements of the LCO not met in MODES 1, 2, and 3.	A.2 Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours
C. -------------NOTE--------------- Required Action C.2 shall be completed if this Condition is entered. -----------------------------------	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.2 Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.9.1	Verify: a. RCS pressure and RCS temperature are within the limits specified in the PTLR during RCS inservice leak and hydrostatic testing, and during RCS non-nuclear heatup and cooldown operations; and b. RCS heatup and cooldown rates are within the limits specified in the PTLR during RCS heatup and cooldown operations, and RCS inservice leak and hydrostatic testing.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	---------------------------NOTE--------------------------- Only required to be met when the reactor is critical and immediately prior to control rod withdrawal for the purpose of achieving criticality. --- Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR.	Once within 15 minutes prior to initial control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	---------------------------NOTE--------------------------- Only required to be met in MODES 1, 2, 3, and 4 during startup of a recirculation pump. --- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to starting an idle recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.9.4	-----------------------------NOTE----------------------------- Only required to be met in MODES 1, 2, 3, and 4 during startup of a recirculation pump. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to starting an idle recirculation pump
SR 3.4.9.5	-----------------------------NOTE----------------------------- Only required to be met when tensioning/ detensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	Once within 30 minutes prior to tensioning/ detensioning the reactor vessel head bolting studs and in accordance with the Surveillance Frequency Control Program

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.6	-----------------------------NOTE---------------------------- Only required to be met when the reactor vessel head is tensioned. --	
	Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	Once within 12 hours after RCS temperature is ≤ 106°F in MODE 4, and in accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 30 minutes after RCS temperature is ≤ 86°F in MODE 4, and in accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1058 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1058 psig.	In accordance with the Surveillance Frequency Control Program

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (applicable amendment specified in the COLR).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates, shall be established and documented in the PTLR for the following:

   i. Limiting Conditions for Operating Section 3.4.9 "RCS Pressure and Temperature (P/T) Limits."

   ii. Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   i. BWROG-TP-11-022-A, Revision 1 (SIR-05-044, Revision 1-A), "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated June 2013.

(continued)

5.6 Reporting Requirements

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

ii. BWROG-TP-11-023-A, Revision 0 (0900876.401, Revision 0-A), "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations," dated May 2013.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601, in lieu of the requirements of 10 CFR 20.1601a, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, measured at 30 cm from the radiation source or from any surface the radiation penetrates, shall be barricaded and conspicuously posted as a high radiation area. Entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates < 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physics supervision in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr, measured at 30 cm from the radiation source or from any surface the radiation penetrates, but less than 500 Rads in 1 hour measured at 1 meter from the radiation source or from any surface that the radiation penetrates, shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervision on duty or Health Physics supervision.

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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. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 221
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 April 2, 2015, as supplemented by letters dated November 12, 2015, and February 9, 2016, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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.

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) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 221 are hereby incorporated in the 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

Shawn Williams for

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

Attachment:
Changes to Renewed Facility
Operating License No. NPF-5
and the Technical Specifications

Date of Issuance: March 23, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 221

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
1.1-5	1.1-5
3.4-18	3.4-18
3.4-19	3.4-19
3.4-20	3.4-20
3.4-21	3.4-21
3.4-22	3.4-22
3.4-23	5.0-21
3.4-24	5.0-22
5.0-21	5.0-23
5.0-22	

(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. 221 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.

1.1 Definitions (continued)

PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Tests and Operation, of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.7.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2804 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical throughout the operating cycle assuming that: a. The reactor is xenon free; b. The moderator temperature is ≥ 68°F, corresponding to the most reactive state; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR. The recirculation pump starting temperature requirements shall be maintained within limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------------NOTE-------------- Required Action A.2 shall be completed if this Condition is entered. ------------------------------------ Requirements of the LCO not met in MODES 1, 2, and 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours
C. -------------NOTE-------------- Required Action C.2 shall be completed if this Condition is entered. ------------------------------------ Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.9.1	Verify: a. RCS pressure and RCS temperature are within the limits specified in the PTLR during RCS inservice leak and hydrostatic testing, and during RCS non-nuclear heatup and cooldown operations; and b. RCS heatup and cooldown rates are within the limits specified in the PTLR during RCS heatup and cooldown operations, and RCS inservice leak and hydrostatic testing.	In accordance with the Surveillance Frequency Control Program
SR 3.4.9.2	----------------------------NOTE---------------------------- Only required to be met when the reactor is critical and immediately prior to control rod withdrawal for the purpose of achieving criticality. --- Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR.	Once within 15 minutes prior to initial control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	----------------------------NOTE---------------------------- Only required to be met in MODES 1, 2, 3, and 4 during startup of a recirculation pump. --- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to starting an idle recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.4	----------NOTE---------- Only required to be met in MODES 1, 2, 3, and 4 during startup of a recirculation pump. ------------------------------ Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to starting an idle recirculation pump
SR 3.4.9.5	----------NOTE---------- Only required to be met when tensioning/ detensioning the reactor vessel head bolting studs. ------------------------------ Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	Once within 30 minutes prior to tensioning/ detensioning the reactor vessel head bolting studs and in accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.6	-----------------------------NOTE----------------------------- Only required to be met when the reactor vessel head is tensioned. --	
	Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	Once within 12 hours after RCS temperature is ≤ 120°F in MODE 4, and in accordance with the Surveillance Frequency Control Program <u>AND</u> Once within 30 minutes after RCS temperature is ≤ 100°F in MODE 4, and in accordance with the Surveillance Frequency Control Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1058 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify reactor steam dome pressure is ≤ 1058 psig.	In accordance with the Surveillance Frequency Control Program

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (applicable amendment specified in the COLR).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates, shall be established and documented in the PTLR for the following:

   i. Limiting Conditions for Operating Section 3.4.9 "RCS Pressure and Temperature (PT) Limits"

   ii. Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   i. BWROG-TP-11-022-A, Revision 1 (SIR-05-044, Revision 1-A), "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," dated June 2013.

(continued)

5.6 Reporting Requirements

5.6.7 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

ii. BWROG-TP-11-023-A, Revision 0 (0900876.401, Revision 0-A), "Linear Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations," dated May 2013.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601, in lieu of the requirements of 10 CFR 20.1601a, each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, measured at 30 cm from the radiation source or from any surface the radiation penetrates, shall be barricaded and conspicuously posted as a high radiation area. Entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates < 1000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physics supervision in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas with radiation levels ≥ 1000 mrem/hr, measured at 30 cm from the radiation source or from any surface the radiation penetrates, but less than 500 Rads in 1 hour measured at 1 meter from the radiation source or from any surface that the radiation penetrates, shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the Shift Supervision on duty or Health Physics supervision.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 277 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-57

AND

AMENDMENT NO. 221 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By letter dated April 2, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15092A856), as supplemented by letters dated November 12, 2015 (ADAMS Accession No. ML15322A089), and February 9, 2016, (ADAMS Accession No. ML16041A244), Southern Nuclear Operating Company, Inc. (SNC, the licensee), requested to amend their license to make changes to the Technical Specifications (TS) for the Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2. The supplements dated November 12, 2015, and February 9, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 7, 2015 (80 FR 38760).

The proposed changes would revise the HNP, Units 1 and 2, TS as necessary to relocate the pressure and temperature (P-T or P/T) limit curves and associated references to a Pressure and Temperature Limits Report (PTLR). Specifically, the request would modify Section 1.0, "Definitions," Limiting Conditions for Operation and Surveillance Requirement Applicability Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits," and Section 5.0, "Administrative Controls," of the TS for both HNP units to delete reference to the P-T curves, and to include reference to the unit-specific PTLRs.

The proposed license amendment request (LAR) would also implement new P-T limits for HNP, Unit 1, that are valid up to peak internal diameter (ID) fluence values of 2.43×10^{18} neutrons per centimeter-squared (n/cm^2) and 3.08×10^{18} n/cm^2 with energy greater than one million electron volts (E > 1 MeV), which correspond to 38 and 49.3 Effective Full Power Years (EFPY) of core operation, respectively. Similarly, the proposed LAR would implement new P-T limits for HNP, Unit 2, that are valid up to peak ID fluence values of 1.95×10^{18} n/cm^2 and 3.28×10^{18} n/cm^2, which correspond to 37 and 50.1 EFPY of core operation, respectively. The revised P-T limits adopt the U.S. Nuclear Regulatory Commission (NRC) approved methodology addressed in the

PTLR and would replace the current P-T limits for both HNP units that were valid for up to 54 EFPY.

Throughout this safety evaluation (SE), no specific reference is made to the proprietary content of the attachments to the licensee's April 2, 2015, November 12, 2015, and February 9, 2016, submittals to avoid the need for proprietary markings in this SE.

2.0 REGULATORY EVALUATION

2.1 System Description

All components of the reactor coolant system (RCS) are designed to withstand the effects of cyclic loads resulting from system pressure and temperature changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. In accordance with Appendix G to Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR), the TS limit the P-T changes during RCS heatup and cooldown to within the design assumptions and the stress limits for cyclic operation. These limits are defined by P-T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing. Each curve defines an acceptable region for normal operation. The curves are used for operational guidance during heatup and cooldown maneuvering, when P-T indications are monitored and compared to the applicable curve, to determine that operation is within the allowable region.

2.2 Regulations and Guidance

In 10 CFR 50, Section 50.36, "Technical Specifications," the U.S. Nuclear Regulatory Commission (NRC) established its regulatory requirements related to the content of TS. Pursuant to 10 CFR 50.36(c), TS are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TS. For this review, the NRC staff interprets the requirements in 10 CFR 50.36 using the accumulation of generically approved guidance that is provided in the improved standard technical specifications (ISTS). For this review, the NRC staff used NUREG-1433, Volume 1, Revision 4.0, "Standard Technical Specifications, General Electric BWR/4 Plants, Revision 4.0, Volume 1, Specifications," April 2012 (ADAMS Accession No. ML12104A192).

The NRC evaluates the acceptability of a facility's proposed PTLR based on the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (ADAMS Accession No. ML031110004), as supplemented by Industry/TS Task Force (TSTF) Standard TS Change Traveler TSTF-419, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR" (ADAMS Accession No. ML012690234). GL 96-03 delineates the requirements for both the methodology and the PTLR including, but not limited to, the requirements of Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements." According to GL 96-03, licensees must perform three separate licensee actions to relocate P-T curves and setpoints to a licensee-controlled document. The licensee must (1) have a methodology approved by the NRC to reference in its TS; (2) develop a report such as a PTLR or a similar document to contain the figures, values, parameters, and any explanation necessary; and (3) modify the applicable sections of the TS accordingly. Additionally, Attachment 1 of GL 96-03 requires that the licensee evaluate seven criteria to demonstrate the acceptability of its PTLRs, as follows:

(1) The PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluence values.
(2) The PTLR methodology describes the surveillance program.
(3) The PTLR methodology describes how the low temperature overpressure protection (LTOP) system limits are calculated applying system/thermal hydraulics and fracture mechanics.
(4) The PTLR methodology describes the method for calculating the adjusted reference temperature (ART) values using Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," (ADAMS Accession No. ML003740284).
(5) The PTLR methodology describes the application of fracture mechanics in the construction of P-T limits based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Appendix G, and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock," (ADAMS Accession No. ML070380185).
(6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T limits for boltup temperature and hydrotest temperature.
(7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation.

The NRC has established fracture prevention criteria requirements in 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation." These requirements mandate that all light-water nuclear power reactors, except those with 10 CFR 50.82(a)(1) certifications, must meet the fracture toughness requirements for the reactor coolant pressure boundary (RCPB) set forth in 10 CFR Part 50, Appendix G. The provisions of 10 CFR Part 50, Appendix G, require that the P-T limits for an operating light-water nuclear power reactor be at least as conservative as those that would be generated if the methods of Appendix G, "Fracture Toughness Criteria for Protection Against Failure," to Section XI of the ASME B&PV Code were used to generate the P-T limits. The provisions of 10 CFR Part 50, Appendix G, also require that applicable surveillance data from reactor pressure vessel (RPV) material surveillance programs developed in accordance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," be incorporated into the calculations of plant-specific P-T limits. Appendix G to 10 CFR 50 also requires that the P-T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials. Finally, Table 1 of 10 CFR Part 50, Appendix G, provides the NRC staff's criteria for meeting the P-T limit requirements of ASME B&PV Code, Section XI, Appendix G, as well as the minimum temperature requirements for the RPV during normal heatup, cooldown, and pressure test operations.

The proposed LAR invokes the methodology documented in NRC-approved Boiling Water Reactor Owners' Group (BWROG) Licensing Topical Report (TR) BWROG-TP-11-022-A, Revision 1, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," (ADAMS Accession No. ML13277A557), and TR BWROG-TP-11-023-A, Revision 0, "Linear

Elastic Fracture Mechanics Evaluation of General Electric Boiling Water Reactor Water Level Instrument Nozzles for Pressure-Temperature Curve Evaluations," (ADAMS Accession No. ML13183A017). The NRC staff approved the 2004 Edition of the ASME B&PV Code with no Addenda for PTLRs developed in accordance with the methodology in BWROG-TP-11-022-A .

The NRC staff evaluates the acceptability of a facility's proposed PTLR based on the following NRC regulations and guidance:

- GL 96-03 (ADAMS Accession No. ML031110004).
- 10 CFR 50 Appendix G & H;
- RG 1.99, Revision 2
- GL 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)" (ADAMS Accession No. ML031200626);
- GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," (ADAMS Accession No. ML031070449);
- RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (ADAMS Accession No. ML010890301);
- Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components," (ADAMS Accession No. ML14149A165);
- NUREG-0800, Section 5.3.2, Revision 2

RG 1.99, Revision 2, contains methodologies for calculating the adjusted RT_{NDT} (ART) due to neutron irradiation. The ART is dependent upon a chemistry factor (CF), which in turn is dependent upon the amount of copper and nickel in the material of the RPV. The CF, copper, and nickel contents that are inputs into the ART calculation may be revised based on the test results from 10 CFR 50, Appendix H, surveillance programs.

GL 92-01, Revision 1, requested that licensees submit their plant-specific RPV data to the NRC staff for review, and GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

Fluence calculations for use in ART and P-T limit curve analyses are acceptable if they are performed with approved methodologies or with methods that are shown to conform to the guidance in RG 1.190.

RIS 2014-11 clarified that the beltline definition in 10 CFR Part 50, Appendix G, is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1×10^{17} n/cm^2 with E > 1 MeV, and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period.

NUREG-800, Section 5.3.2, provides an acceptable method for determining the P-T limits for ferritic materials in the beltline of the RPV based on the ASME B&PV Code, Section XI, Appendix G, methodology.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The licensee proposed the following changes in their LAR:

(1) Modify the definitions section of the TS to include a definition of a PTLR. The figures, values, and parameters for P-T limits will be relocated to the PTLR on a unit-specific basis in accordance with the methodology approved by the NRC in BWROG-TP-11-022-A that maintains the acceptance limits and the limits of the safety analysis. As noted in the definition, plant operation within these limits is addressed by individual specifications. The PTLR provides the explanations, figures, values, and parameters of the P-T limits for the applicable effective operational period.

(2) Revise Limiting Conditions for Operation and Surveillance Requirements Section 3.4.9 to replace the P-T system limits with a reference to the PTLR.

(3) Add Specification 5.6.7, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," to the reporting requirements of TS Section 5.0.

The licensee stated that relocation of the P-T limit curves to the PTLRs adopts the methodology provided in BWROG-TP-11-022-A and BWROG-TP-11-023-A. The licensee also stated that the proposed TS changes are consistent with the guidance provided in GL 96-03, as supplemented by TSTF-419, which allows the licensee to relocate their P-T curves and associated numerical limits (such as heatup and cooldown rates) from the plant TS to PTLRs, which are licensee-controlled documents. In order for the licensee to implement the PTLRs, the analytical methods used to develop the P-T limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the Administrative Controls section of the plant TSs.

The proposed PTLR for HNP, Unit 1, contains new P-T limit curves that are valid for peak ID fluence values of 2.43×10^{18} n/cm^2 and 3.08×10^{18} n/cm^2, which correspond to 38 and 49.3 EFPY of core operation, respectively. Similarly, the proposed PTLR for HNP, Unit 2, contains new P-T limits that are valid up to peak ID fluence values of 1.95×10^{18} n/cm^2 and 3.28×10^{18} n/cm^2, which correspond to 37 and 50.1 EFPY of core operation, respectively.

The purpose of the BWROG methodology is to provide boiling water reactors (BWRs) with an NRC-approved report that can be referenced in plant TS to establish BWR fracture mechanics methods for generating P-T curves/limits, and other associated numerical limits, thereby allowing BWR plants to adopt the PTLR option. The licensee stated that the P-T curves were developed in accordance with the methodology and template in the BWROG methodology, as documented in the PTLRs provided in Enclosures 5 and 6 for HNP Units 1 and 2, respectively, of the licensee's April 2, 2015, submittal. The BWROG methodology does not include development or licensing of vessel fluence methods, but the PTLRs state that the methodology used to calculate the RPV neutron fluence values utilized in the development of the HNP, Units 1 and 2, P-T limit curves is in accordance with RG 1.190 methods.

3.2 NRC Staff Evaluation

Per GL 96-03, relocating P-T curves and setpoints to a licensee-controlled document requires three separate licensee actions: (1) having an NRC approved methodology to reference in its TS; (2) develop a report such as a PTLR or a similar document to contain the figures, values, parameters, and any explanation necessary; and (3) modify the applicable sections of the TS accordingly.. The NRC staff review of the licensee's application against these three actions, surveillance program data, calculated ART values, and calculated neutron fluence values, is provided below in Sections 3.2.1 through 3.2.6 of this SE.

Relocation of the P-T curves does not eliminate the requirement to operate in accordance with the limits specified in Appendix G to 10 CFR Part 50. The requirement to operate within the limits in the PTLR is specified in and controlled by the TS. Only the figures, values, and parameters associated with the P-T limits are to be relocated to the PTLR. In order for the curves to be relocated to a PTLR, a methodology for their development must be approved in advance by the NRC and use the guidance of GL 96-03. The PTLR review process requires that changes to the methodology be approved by the NRC. Furthermore, when changes are made to the figures, values, and parameters contained in the PTLR, the PTLR is to be updated and submitted to the NRC upon issuance.

3.2.1 PTLR Acceptability

The NRC staff examined the proposed PTLRs and determined that they were developed from the template PTLR found in Appendix B of BWROG-TP-11-022-A. Furthermore, the NRC staff determined that the seven criteria specified in Attachment 1 of GL 96-03 were satisfied, as documented in Section 1.3 and Table 1-1 of BWROG-TP-11-022-A and summarized below:

(1) The PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluence values.

    The NRC staff's evaluation of the methodology of calculating the neutron fluence is contained in Section 3.2.5 of this SE. As detailed in Section 3.2.5 of this SE, the NRC staff found the computer codes and methods used to calculate the neutron fluence acceptable. Therefore, this criterion is satisfied.

(2) The PTLR methodology describes the surveillance program.

    Appendix A to both of the HNP, Units 1 and 2, PTLRs indicates that HNP, Units 1 and 2, participate in the approved Boiling Water Reactor Vessel and Internals Program (BWRVIP) Integrated Surveillance Program (ISP), which meets the requirements of 10 CFR Part 50, Appendix H. Hence, this criterion is satisfied.

(3) The PTLR methodology describes how the LTOP system limits are calculated applying system/thermal hydraulics and fracture mechanics.

    The LTOP criterion is not applicable to HNP, Units 1 and 2, because they are BWRs.

(4) The PTLR methodology describes the method for calculating the ART values using RG 1.99, Revision 2.

The HNP, Units 1 and 2, PTLRs indicate that RG 1.99, Revision 2, provides the methods for determining the ARTs for the beltline materials, with their chemistry factors (CFs) determined by surveillance data information from the BWRVIP ISP or RG 1.99, Revision 2, as appropriate. Hence, this criterion is satisfied.

(5) The PTLR methodology describes the application of fracture mechanics in the construction of P-T limits based on the ASME B&PV Code, Section XI, Appendix G, and NUREG-0800, Section 5.3.2.

The PTLR states that the P-T limits were calculated in accordance with BWROG-TP-11-022-A and BWROG-TP-11-023-A. This description is sufficient as the NRC staff's SE of BWROG-TP-11-022-A documents that the BWROG methodology meets the fifth criterion. Also, as documented in Section 3.2.4 of this SE, the staff found the HNP, Units 1 and 2, PTLRs appropriately implement the BWROG-TP-11-022-A and BWROG-TP-11-023-A methodologies. Hence, this criterion is satisfied.

(6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P-T limits for boltup temperature and hydrotest temperature.

The PTLR states that the P-T limits were calculated in accordance with the BWROG-TP-11-022-A and BWROG-TP-11-023-A. This description is sufficient as BWROG-TP-11-022-A contains detailed information regarding the minimum temperature requirements for boltup temperature and hydrotest temperature, and the NRC staff's SE of BWROG-TP-11-022-A documents that the BWROG methodology meets the sixth criterion. Also, as documented in Section 3.2.4 of this SE, the NRC staff found that the HNP, Units 1 and 2, P-T limits meet the minimum temperature requirements of 10 CFR 50, Appendix G. Hence, this criterion is satisfied.

(7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation.

The NRC staff reviewed and approved the use of the ISP described in Appendix A of BWROG-TP-11-022-A as documented in its SE of the BWROG methodology. In addition, Appendix A of both PTLRs describes the specifics of the ISP as it relates to HNP, Units 1 and 2. Tables 7 and 8 of both PTLRs show how the ART values were calculated considering CFs derived from ISP data, if applicable, for the materials represented in the ISP. Also, the licensee's November 12, 2015, supplement describes specifics of the ISP representative materials for HNP, Units 1 and 2, and how these materials were used in the ART calculations for HNP, Units 1 and 2. Hence, this criterion is satisfied.

In the November 12, 2015, supplement, the licensee provided additional information regarding missing information in the PTLRs. The licensee stated that the initial RT_{NDT} values were taken from surveillance materials test reports, and that those values were derived using the

General Electric (GE) RT_{NDT} Estimation Method submitted by the BWROG and assessed in a December 16, 1994, NRC SE (ADAMS Legacy Library Accession No. 9501170165). The licensee also stated that Procedure 1 from Appendix A of BWROG-TP-11-022-A was used to evaluate surveillance data for HNP, Unit 1, for the surveillance plate material, and Procedure 2 was used for the weld material. For HNP, Unit 2, Procedure 2 was used to evaluate surveillance data for both the weld and plate materials since credible surveillance data was not available. In the February 9, 2016, supplement, the licensee provided revised PTLRs containing the above information. The NRC staff found this supplemental information to be acceptable.

Based on the above supplemental information and the fact that the seven GL 96-03 criteria are satisfied, the NRC staff finds that the HNP, Units 1 and 2, PTLRs are acceptable to be referenced in TS 3.4.9 and fulfill licensee action number 2 of GL 96-03 (i.e., developing a report such as a PTLR or a similar document to contain the figures, values, parameters, and any explanation necessary).

3.2.2 Consideration of Surveillance Program Data

Appendix H to 10 CFR 50 requires that plants have surveillance programs and use the test results from those programs in the development of 10 CFR 50, Appendix G P-T limits. Furthermore, BWROG-TP-11-022-A requires the use of data from the BWRVIP ISP, as documented in the NRC's SE of that report.

HNP, Units 1 and 2, are both part of the BWRVIP ISP that combines all the surveillance programs for U.S. BWRs into a single integrated program. Under the ISP, the limiting plate and weld (the target materials) in the participating BWR plants are represented by similar materials under irradiation in other BWR plants.

HNP, Unit 1, is a host plant under the ISP, so the representative plate and weld materials are contained in the HNP, Unit 1, surveillance capsules. Section 5.0 of the HNP, Unit 1, PTLR states that the limiting plate heat is based on credible surveillance data that bounds the RG 1.99, Revision 2, CF. Therefore, the fitted CF is used for calculations related to the limiting beltline plate. In addition, an archival plate heat (Heat No. C3985-2) from the HNP, Unit 1, vessel was included in the BWR Supplemental Surveillance Program (SSP) and those data were also determined to be credible, and, consequently, a reduced margin term was used for this material. The representative weld material (20291) was contained in the Cooper and SSP Capsule C capsules. However, since the material heats for the limiting weld material and representative surveillance capsule weld material do not match, the licensee calculated the CF using the tables in RG 1.99, Revision 2. Finally, per Appendix A of the HNP, Unit 1, PTLR, two more HNP, Unit 1, the licensee is scheduled to remove and test the capsules under the ISP in approximately 2016 and 2029. The NRC staff reviewed the CF information included in the licensee's November 12, 2015, letter and enclosures and found the revised CFs and margin terms to be acceptable for the HNP, Unit 1, limiting materials.

HNP, Unit 2, is a host plant under the ISP, so the representative plate and weld materials are contained in the HNP, Unit 2, surveillance capsules. Section 5.0 of the HNP, Unit 2, PTLR states that the representative plate heat does not match the target plate heat; however, it does match the heat for plate material used in other beltline plates. Since the licensee has tested only one surveillance capsule containing this plate heat, no fitted CF is available; therefore, the licensee calculated the CF using the tables in RG 1.99, Revision 2. The representative weld material heat does not match any weld material heats used in the HNP, Unit 2, beltline; therefore, the CF was also calculated using the tables in RG 1.99, Revision 2. Finally, per Appendix A of the HNP, Unit 2, PTLR, the licensee is schedule to remove and test two more HNP, Unit 2, capsules under the ISP in approximately 2017 and 2027. The NRC staff independently reproduced the CF values for HNP, Unit 2, using the tables in RG 1.99, Revision 2. The NRC staff found that their calculated CF values and the licensee's CF values were consistent; therefore, the NRC staff finds them to be acceptable for the HNP, Unit 2, limiting materials.

3.2.3 Beltline Materials Adjusted Reference Temperature (ART)

Limiting material ART values are used in the development of P-T limits. These values are to be calculated using RG 1.99, Revision 2, methods.

In Section 5.0 of the HNP, Unit 1, PTLR, the licensee stated that the lower intermediate shell plate heat No. C4114-2 is the limiting material for the beltline region, and detailed ART calculations for all beltline and extended beltline materials (i.e., fluence at one-quarter RPV thickness is greater than 1×10^{17} n/cm^2, E > 1 MeV) were provided in Tables 7 and 8 of the PTLR for 38 and 49.3 EFPY, respectively.

In Section 5.0 of the HNP, Unit 2, PTLR, the licensee stated that lower shell plate heat No. C8553-1 is the limiting material for the beltline region. The licensee provided detailed ART calculations for all beltline and extended beltline materials (i.e., fluence greater than 1×10^{17} n/cm^2, E > 1 MeV) in Tables 7 and 8 of the PTLR for 37 and 50.1 EFPY, respectively.

The NRC staff performed independent calculations for the HNP, Unit 1, limiting beltline material ART values at 38 and 49.3 EFPY, and for the HNP, Unit 2, limiting beltline material ART values at 37 and 50.1 EFPY using the methods of RG 1.99, Revision 2. The NRC staff reviewed the copper and nickel contents, the CFs, and initial RT_{NDT} values for consistency with the license renewal application (LRA) for HNP, Units 1 and 2, dated March 1, 2000, which is the most recent material information approved by the NRC staff. In some cases, the licensee revised the copper content, the CF, or the margin terms from the LRA because of recent test results obtained from the BWRVIP ISP. In their November 12, 2015, supplement, the licensee provided the NRC staff with the data from the BWRVIP necessary for the NRC staff to complete their independent ART assessment.

The NRC staff independently calculated the ART values using the methods of RG 1.99, Revision 2, which resulted in a finding of one-quarter RPV thickness ART values. The resulting one-quarter RPV thickness (1/4t) ART values agreed with the licensee's calculated values.

3.2.4 P-T Limit Confirmatory Calculations

The HNP PTLRs contain new P-T curves for Hydrostatic Pressure and Leak Test (Curve A), Normal Operation Core Not Critical (Curve B), and Normal Operation - Core Critical (Curve C) for Unit 1 at 38 EFPY and 49.3 EFPY, and for Unit 2 at 37 EFPY and 50.1 EFPY. The NRC staff noted that the new P-T curves are different than the proposed to be removed P-T curves in the existing TS and requested the licensee to explain the factors that lead to the differences between the PTLR P-T curves and the TS P-T curves. In their November 12, 2015, supplement, the licensee indicated that the previous P-T curves were developed using GE methods, as opposed to the BWROG method that was used to develop the curves in the PTLRs, which includes some notable differences from the prior methodology:

- Water level instrumentation (WLI) nozzle and feedwater nozzle curves were developed from plant-specific nozzle finite element analysis and boundary integral equation/influence function Linear Elastic Fracture Mechanics nozzle solutions.
- The bottom head was treated by applying a conservative stress concentration factor (SCF) value of 3.0.

The licensee also noted differences in the resulting P-T limit curves are caused by the following additional factors:

- The fluence used in the PTLR analysis is higher than the previous fluence and is based on a more recent evaluation.
- The ART calculation in the current analysis uses the latest information from the BWRVIP ISP, as well as the Combustion Engineering Owner's Group best estimate chemistry results.

The BWROG methodology states that separate P-T curves be generated for the upper vessel region (including the feedwater nozzle), the beltline region, and the bottom head. Using the information provided in the November 12, 2015, and February 9, 2016, supplements, the NRC staff performed confirmatory P-T curves for HNP, Units 1 and 2, using the methodology of the 2004 Edition of ASME B&PV Code, Section XI, Appendix G; BWROG-TP-11-022-A; BWROG-TP-11-023-A; and the ART values reported in the PTLRs for the bottom head, feedwater nozzle, WLI nozzle, and beltline shell regions. The NRC staff performed detailed calculations for Curves A, B, and C for HNP, Unit 1, for 38 EFPY, and for HNP, Unit 2, for 50.1 EFPY, and independently reproduced the licensee's P-T curves for both HNP units. The NRC staff also verified that the proposed P-T limits meet the minimum temperature requirements of 10 CFR 50, Appendix G, which contains additional requirements determined from the most limiting material (highest RT_{NDT}) in closure head flange and vessel flange regions.

Based on its confirmatory calculations for 38 and 50.1 EFPY, the NRC staff determined that the HNP, Units 1 and 2, proposed P-T limits for Curves A, B, and C meet the minimum temperature requirements of 10 CFR 50, Appendix G; are as conservative or more conservative than P-T limits generated using the methods of the ASME B&PV Code, Section XI, Appendix G; and implement the methodologies of NRC-approved topical reports BWROG-TP-11-022-A and BWROG-TP-11-023-A. Therefore, the NRC staff finds that the licensee's revised P-T limits are acceptable for HNP, Unit 1, at 38 EFPY and 49.3 EFPY and for Unit 2 at 37 EFPY and 50.1 EFPY and fulfill licensee action number 1 of GL 96-03.

3.2.5 Fluence Methodology

The licensee stated that fluence calculations supporting the proposed P-T limits were performed in accordance with RG 1.190 using the Radiation Analysis Modeling Application (RAMA) fluence methodology discussed in BWRVIP-114-A, "BWR Vessel and Internals Project: RAMA Fluence Methodology Theory Manual" (not publicly available; proprietary information). The RAMA methodology has been generically approved by the NRC for use at BWRs, subject to the condition/limitation that plant geometry-specific validation must be performed for plants that do not have geometries similar to either Hope Creek Generating Station, Unit 1, or Susquehanna Steam Electric Station, Units 1 and 2 (i.e., BWR/4s).

The licensee provided proprietary reports on the fluence calculations for HNP, Units 1 and 2, in the November 12, 2015, supplement. These reports state that all reactor design and operating data inputs used to develop the model were plant-specific to HNP, Units 1 and 2. The licensee validated the fluence calculation methods by a comparison with the measurements from HNP dosimetry, as well as with measurements from similar BWRs. For HNP, Unit 1, the licensee calculated the fluence values at end of cycle 25 at 28.4 EFPY and projected end of reactor design life at 49.3 EFPY. For HNP, Unit 2, the licensee calculated the fluence values at beginning of cycle 22 at 26.6 EFPY and projected end of the reactor's extended design life at 50.1 EFPY.

The NRC staff reviewed the fluence calculations supporting the proposed P-T limits. The licensee calculated reactor vessel neutron fluence using a methodology that has been generically approved by the NRC and adhered to the recommendations set forth in RG 1.190. Based on the fact that the licensee used NRC-approved methods and guidance, the NRC staff finds that their fluence calculations are acceptable for use in the development of the proposed P-T limits.

3.2.6 Compliance with TSTF-419

Industry/TSTF Standard Technical Specification Change Traveler TSTF-419, as modified by the changes proposed in an NRC letter dated March 7, 2011 (ADAMS Accession No. ML110660285), and approved by the staff in a March 21, 2002, letter (ADAMS Accession No. ML020800488), requires that references to NRC-approved topical reports for the PTLR methodology must include the revision number and date of the topical report. These changes were incorporated into Section 5.6.4 of NUREG-1433. The proposed markup of TS 5.6.7 in the licensee's April 2, 2015, LAR references Revision 1 of TR BWROG-TP-11-022-A dated June 2013 and Revision 0 of TR BWROG-TP-11-023-A dated May 2013. Therefore, the requirements of TSTF-419, as modified by the March 7, 2011 and March 21, 2002 NRC letters, are satisfied.

If the licensee wishes to revise the PTLRs to use a different revision of the TRs, it must submit a new LAR.

In addition to the revised TS 5.6.7, the NRC staff reviewed the licensee's revised pages for the applicable sections of the TS, TS 1.1 and 3.4.9, and finds that they are appropriately modified to account for the implementation of the PTLR, thereby fulfilling licensee action number 3 of GL 96-03 (i.e., licensee must modify the applicable sections of the TS accordingly).

3.2.7 Summary

The NRC staff verified that the licensee's application satisfies the three required actions and the seven criteria of GL 96-03. Additionally, the NRC staff verified that the licensee properly incorporated surveillance program data, and properly calculated the ART and neutron fluence values. The NRC staff concludes that the licensee provided an acceptable means of establishing and maintaining the detailed values of the P-T limit curves for both HNP units in PTLRs. Furthermore, because plant operation continues to be limited in accordance with the requirements of Appendix G to 10 CFR Part 50, and the P-T limits in the TS will be established using a methodology approved by the NRC, these changes will not adversely impact plant safety.

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 March 1, 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 (80 FR 38760). 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.

Principal Contributors: G. Stevens
R. Beaton

Date: March 23, 2016

March 23, 2016

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

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING THE RELOCATION OF THE PRESSURE TEMPERATURE LIMIT CURVES FOLLOWING TSTF-419 (CAC NOS. MF6063 AND MF6064)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 277 to Renewed Facility Operating License DPR-57 and Amendment No. 221 to Renewed Facility Operating License NPF-5 for the Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated April 2, 2015, as supplemented by letters dated November 12, 2015, and February 9, 2016.

The amendments revise the HNP, Units 1 and 2, TSs as necessary to relocate the pressure and temperature (P-T or P/T) limit curves and associated references to a Pressure and Temperature Limits Report (PTLR). Specifically, the request modifies Section 1.0, "Definitions," Limiting Conditions for Operation and Surveillance Requirement Applicability Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits," and Section 5.0, "Administrative Controls," of the TS for both HNP units to delete reference to the P-T curves, and to include reference to the unit-specific PTLRs. The request also implements new P-T limits for both HNP units.

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. 277 to DPR-57
2. Amendment No. 221 to NPF-5
3. Safety Evaluation

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Amendment No. ML16062A099

*via memo

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/DE/EVIB/BC*	NRR/DSS/STSB/BC	OGC
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OFFICE	NRR/LPL2-1/BC	NRR/LPL2-1/PM			
NAME	MMarkley	MOrenak			
DATE	3/21/16	3/23/16			

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