November 19, 2019

Ms. Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING REVISION TO TECHNICAL SPECIFICATION REQUIREMENTS DURING HANDLING IRRADIATED FUEL AND CORE ALTERATIONS (EPID L-2019-LLA-0091)

Dear Ms. Gayheart:

The Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 301 to Renewed Facility Operating License No. DPR-57 and Amendment No. 246 to Renewed Facility Operating License No. NPF-5 for the Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated April 24, 2019, as supplemented October 17, 2019.

The amendments revise certain TSs to remove the requirements for engineered safety feature systems (e.g., secondary containment, secondary containment valve isolation capability, and standby gas treatment system) to be operable after sufficient radioactive decay of irradiated fuel has occurred following a plant shutdown.

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,

John G. Lamb

John G. Lamb, Senior 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. 301 to DPR-57
2. Amendment No. 246 to NPF-5
3. Safety Evaluation

cc: Listserv

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 301
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 24, 2019, as supplemented October 17, 2019, 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 (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 301, 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, 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 Technical Specifications

Date of Issuance: November 19, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 301

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
3.3-63	3.3-63
3.6-34	3.6-34
3.6-35	3.6-35
3.6-37	3.6-37
3.6-38	3.6-38
3.6-39	3.6-39
3.6-40	3.6-40
3.6-41	3.6-41
3.6-42	3.6-42

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[2] specified or incorporated below:

(1) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B); as revised through Amendment No. 301 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

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level Low - Low, Level 2	1 ,2, 3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ -47 inches
2. Drywell Pressure - High	1, 2, 3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.92 psig
3. Reactor Building Exhaust Radiation - High	1, 2, 3	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 80 mR/hr
4. Refueling Floor Exhaust Radiation - High	1, 2, 3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 80 mR/hr

(a) During movement of recently irradiated fuel assemblies in secondary containment.

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 recently irradiated fuel assemblies in the secondary containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3 due to SR 3.6.4.1.3 not met.	A.1 Verify secondary containment vacuum of ≥ 0.20 inch water gauge can be established in ≤ 10 minutes using one or more OPERABLE standby gas treatment (SGT) subsystem(s).	4 hours
	AND	
	A.2 Restore secondary containment to OPERABLE status.	7 days
B. Secondary containment inoperable in MODE 1, 2, or 3 due to SR 3.6.4.1.4 not met.	B.1 Verify secondary containment vacuum of ≥ 0.20 inch water gauge can be maintained for 1 hour using one or more OPERABLE SGT subsystem(s) at a flow rate ≤ 4000 cfm per subsystem.	8 hours
	AND	
	B.2 Restore secondary containment to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Secondary containment inoperable in MODE 1, 2, or 3 for reasons other than Condition A or B.	C.1	Restore secondary containment to OPERABLE status.	4 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. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment.	E.1	------------NOTE----------- LCO 3.0.3 is not applicable. -------------------------------	
		Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

-----NOTES-----
1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange. <u>AND</u> A.2 -----NOTES----- 1. Isolation devices in high radiation areas may be verified by use of administrative means.	8 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)		2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ---------------------------------	
		Verify the affected penetration flow path is isolated.	Once per 31 days
B. One or more penetration flow paths with two SCIVs inoperable.	B.1	Isolate the affected penetration flow path by use of at least one closed and deactivated automatic valve, closed manual valve, or blind flange.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours
D. Required Action and associated Completion Time of Condition A or B not met during movement of recently irradiated fuel assemblies in the secondary containment.	D.1	-------------NOTE----------- LCO 3.0.3 is not applicable. -------------------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.4.2.1	---------------------------NOTES--------------------------- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for SCIVs that are open under administrative controls. --- Verify each secondary containment isolation manual valve and blind flange that is not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.2.2	Verify the isolation time of each power operated, automatic SCIV is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.2.3	Verify each automatic SCIV actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 The Unit 1 and Unit 2 SGT subsystems required to support LCO 3.6.4.1, "Secondary Containment," shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required Unit 1 SGT subsystem inoperable while: 1. Four SGT subsystems required OPERABLE, and 2. Unit 1 reactor building-to-refueling floor plug not installed.	A.1 Restore required Unit 1 SGT subsystem to OPERABLE status.	30 days from discovery of failure to meet the LCO
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

(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 recently irradiated fuel assemblies in the secondary containment.	-----------------NOTE---------------- LCO 3.0.3 is not applicable. --		
	D.1	Place remaining OPERABLE SGT subsystem(s) in operation.	Immediately
	<u>OR</u>		
	D.2	Suspend movement of recently irradiated fuel assemblies in secondary containment.	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)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two or more required SGT subsystems inoperable during movement of recently irradiated fuel assemblies in the secondary containment.	F.1 -----------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------- Suspend movement of recently irradiated fuel assemblies in secondary containment.	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
SR 3.6.4.3.3	Verify each required SGT subsystem actuates on an actual or simulated initiation signal.	In accordance with the Surveillance Frequency Control Program

UNITED STATES
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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

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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. 246
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 24, 2019, as supplemented October 17, 2019, 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) <u>Technical Specifications</u>

    The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 246 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 Renewed Facility
Operating License No. NPF-5
and Technical Specifications

Date of Issuance: November 19, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 246

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
3.3-63	3.3-63
3.6-33	3.6-33
3.6-34	3.6-34
3.6-36	3.6-36
3.6-37	3.6-37
3.6-38	3.6-38
3.6-39	3.6-39
3.6-40	3.6-40
3.6-41	3.6-41

(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) <u>Maximum Power Level</u>

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) <u>Technical Specifications</u>

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B); as revised through Amendment No. 246 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) <u>Additional Conditions</u>

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) <u>Fire Protection</u>

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

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[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.

Table 3.3.6.2-1 (page 1 of 1)
Secondary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low Low, Level 2	1, 2, 3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≥ -47 inches
2. Drywell Pressure - High	1, 2, 3	2	SR 3.3.6.2.1 SR 3.3.6.2.2 SR 3.3.6.2.4 SR 3.3.6.2.5	≤ 1.92 psig
3. Reactor Building Exhaust Radiation - High	1, 2, 3	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 80 mR/hr
4. Refueling Floor Exhaust Radiation - High	1, 2, 3, (a)	2	SR 3.3.6.2.1 SR 3.3.6.2.3 SR 3.3.6.2.5	≤ 80 mR/hr

(a) During movement of recently irradiated fuel assemblies in secondary containment.

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 recently irradiated fuel assemblies in the secondary containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Secondary containment inoperable in MODE 1, 2, or 3 due to SR 3.6.4.1.3 not met.	A.1	Verify secondary containment vacuum of ≥ 0.20 inch water gauge can be established in ≤ 10 minutes using one or more OPERABLE standby gas treatment (SGT) subsystem(s).	4 hours
	<u>AND</u>		
	A.2	Restore secondary containment to OPERABLE status.	7 days
B. Secondary containment inoperable in MODE 1, 2, or 3 due to SR 3.6.4.1.4 not met.	B.1	Verify secondary containment vacuum of ≥ 0.20 inch water gauge can be maintained for 1 hour using one or more OPERABLE SGT subsystem(s) at a flow rate ≤ 4000 cfm per subsystem.	8 hours
	<u>AND</u>		
	B.2	Restore secondary containment to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Secondary containment inoperable in MODE 1, 2, or 3 for reasons other than Condition A or B.	C.1	Restore secondary containment to OPERABLE status.	4 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. Secondary containment inoperable during movement of recently irradiated fuel assemblies in the secondary containment.	E.1	-----------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------- Suspend movement of recently irradiated fuel assemblies in the secondary containment.	 Immediately

3.6 CONTAINMENT SYSTEMS

3.6.4.2 Secondary Containment Isolation Valves (SCIVs)

LCO 3.6.4.2 Each SCIV shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

-----NOTES-----

1. Penetration flow paths may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by SCIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more penetration flow paths with one SCIV inoperable.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	8 hours
	<u>AND</u>	
	A.2 -----NOTES----- 1. Isolation devices in high radiation areas may be verified by use of administrative means.	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	2. Isolation devices that are locked, sealed, or otherwise secured may be verified by administrative means. ------------------------------ Verify the affected penetration flow path is isolated.	 Once per 31 days
B. One or more penetration flow paths with two SCIVs inoperable.	B.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met in MODE 1, 2, or 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours
D. Required Action and associated Completion Time of Condition A or B not met during movement of recently irradiated fuel assemblies in the secondary containment.	D.1 -------------NOTE---------- LCO 3.0.3 is not applicable. ------------------------------ Suspend movement of recently irradiated fuel assemblies in the secondary containment.	 Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.4.2.1	-----NOTES----- 1. Valves and blind flanges in high radiation areas may be verified by use of administrative means. 2. Not required to be met for SCIVs that are open under administrative controls. ---------- Verify each secondary containment isolation manual valve and blind flange that is not locked, sealed, or otherwise secured and is required to be closed during accident conditions is closed.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.2.2	Verify the isolation time of each power operated, automatic SCIV is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.6.4.2.3	Verify each automatic SCIV actuates to the isolation position on an actual or simulated actuation signal.	In accordance with the Surveillance Frequency Control Program

3.6 CONTAINMENT SYSTEMS

3.6.4.3 Standby Gas Treatment (SGT) System

LCO 3.6.4.3 The Unit 1 and Unit 2 SGT subsystems required to support LCO 3.6.4.1, "Secondary Containment," shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
During movement of recently irradiated fuel assemblies in the secondary containment.

ACTIONS

---NOTE---
When two Unit 1 SGT subsystems are placed in an inoperable status solely for inspection of the Unit 1 hardened vent rupture disk, entry into associated Conditions and Required Actions may be delayed for up to 24 hours, provided both Unit 2 SGT subsystems are OPERABLE.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required Unit 1 SGT subsystem inoperable while: 1. Four SGT subsystems required OPERABLE, and 2. Unit 1 reactor building-to-refueling floor plug not installed.	A.1 Restore required Unit 1 SGT subsystem to OPERABLE status.	30 days from discovery of failure to meet the LCO

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One required Unit 2 SGT subsystem inoperable. OR 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 recently irradiated fuel assemblies in the secondary containment.	-----------------NOTE----------------- LCO 3.0.3 is not applicable. --		
	D.1	Place remaining OPERABLE SGT subsystem(s) in operation.	Immediately
	OR		
	D.2	Suspend movement of recently irradiated fuel assemblies in secondary containment.	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 recently irradiated fuel assemblies in the secondary containment.	F.1	-----------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------- Suspend movement of recently irradiated fuel assemblies in secondary containment.	 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)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 301 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-57

AND

AMENDMENT NO. 246 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

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

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

By application dated April 24, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19114A456), as supplemented by letter dated October 17, 2019, (ADAMS Accession No. ML19290E958), 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.

The amendments revise certain TSs to remove the requirements for engineered safety feature (ESF) systems (e.g., secondary containment, secondary containment valve isolation capability, and standby gas treatment system) to be operable after sufficient radioactive decay of irradiated fuel has occurred following a plant shutdown. Following sufficient radioactive decay, these systems are no longer required during a design-basis accident (DBA) fuel handling accident (FHA) to ensure that the evaluated radiological consequences at the exclusion area boundary (EAB), low population zone (LPZ), and the main control room (MCR) remains below the Title 10 of the *Code of Federal Regulation* (10 CFR), Section 50.67(b)(2) dose limits and the accident dose criteria specified in the U.S. Nuclear Regulatory Commission (NRC) standard review plan, which represents a small fraction of 10 CFR 50.67 dose limits.

The affected TS Limiting Conditions for Operations (LCOs) are as follows:

- 3.3.6.2, Secondary Containment Isolation Instrumentation
- 3.6.4.1, Secondary Containment
- 3.6.4.2, Secondary Containment Isolation Valves (SCIVs)
- 3.6.4.3, Standby Gas Treatment (SGT) System

The corresponding sections of the TS Bases are also affected. SNC used NRC-approved Technical Specification Task Force (TSTF)-51, Revision 2, as the model for its requested changes.

The purpose of this license amendment request (LAR) is to improve the performance of activities during refueling outages. Keeping containment penetrations open for equipment and personnel access during outages contributes greatly to improving performance. With the current TS requirements, SNC has to close the equipment hatch to complete core alterations and fuel handling activities. Doing this interrupts several outage tasks.

The application dated April 24, 2019, provided a markup of the affected TS and basis documents that are current at the time of the application. However, the NRC staff noted that one of the affected TSs, TS 3.6.4.1 "Secondary Containment," was revised by letter dated September 4, 2019 (ADAMS Accession No. ML19198A104), containing the NRC issued Amendment Nos. 298 and 243 for HNP, Units 1 and 2, respectively. Amendment Nos. 298 and 243 changed TS 3.6.4.1. Amendment Nos. 298 and 243 revised the existing Condition A to Condition C, revised the existing Condition B to Condition D, revised existing Condition C to Condition E, and added Condition A and Condition B to TS 3.6.4.1. By supplement dated October 17, 2019, the licensee resubmitted a markup of the amended TS 3.6.4.1 and corresponding basis documents to the NRC.

The supplement dated October 17, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC (or the Commission) staff's original proposed no significant hazards consideration determination as published the *Federal Register* on July 16, 2019 (84 FR 33987).

2.0 REGULATORY EVALUATION

The regulation in 10 CFR 50.36, "Technical specifications," section (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. When a limiting condition for operation of any process step in the system of a fuel reprocessing plant is not met, the licensee shall shut down that part of the operation or follow any remedial action permitted by the technical specifications until the condition can be met.

The regulation 10 CFR 50.67, "Accident source term," subsection (b)(2) states:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) [Roentgen equivalent man (rem)] total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

The HNP, Unit 1, secondary containment and SGT systems were designed to the following applicable Atomic Energy Commission preliminary general design criteria (GDC) identified in *Federal Register* at 32 FR 10213, published on July 11, 1967 (ADAMS Accession No. ML043310029):

> 1967 GDCs: GDC 10, Containment (Category A); GDC 17, Monitoring Radioactive Releases (Category B); GDC 19, Protection Systems Reliability (Category B); GDC 20, Protection Systems Redundancy and Independence (Category B); GDC 21, Single Failure Definition (Category B); GDC 22, Separation of Protection and Control Instrumentation Systems (Category B); GDC 62, Inspection of Air Cleanup Systems (Category A); GDC 63, Testing of Air Cleanup Systems Components (Category A); GDC 64, Testing of Air Cleanup Systems (Category A); GDC 65, Testing of Operational Sequence of Air Cleanup Systems (Category A); and GDC 70, Control of Releases of Radioactivity to the Environment (Category B).

The HNP, Unit 2, secondary containment and SGT systems were designed to the following 10 CFR Part 50, Appendix A GDCs for Nuclear Power Plants:

GDC 13, Instrumentation and control; GDC 16, Containment design; GDC 19, Control room; GDC 20, Protection system functions; GDC 21, Protection system reliability and testability; GDC 22, Protection system independence; GDC 23, Protection system failure modes; GDC 24, Separation of protection and control systems; GDCs 41, 42, and 43, Containment atmosphere cleanup, inspection, and testing; and GDC 64, Monitoring radioactivity releases.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Rev. 0, published July 2000 (ADAMS Accession No. ML003716792), provides the methodology for analyzing the radiological consequences of several design basis accidents (DBAs) to show compliance with 10 CFR 50.67. RG 1.183 provides guidance to licensees on acceptable application of alternate source term (AST)

submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition, (SRP) Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Rev. 0, July 2000 (ADAMS Accession No. ML003734190), provides guidance to the NRC staff for the review of alternative source term amendment requests. Section 15.0.1 states that the NRC reviewer should evaluate the proposed change against the guidance in RG 1.183. The dose acceptance criteria for the fuel handling accident are a TEDE of 6.3 rem at the exclusion area boundary for the worst 2 hours, 6.3 rem at the exclusion area boundary low population zone, and 5 rem in the control room for the duration of the accident.

Guidance for the NRC staff's review of Technical Specifications is provided in Section 16, Technical Specifications, of NUREG-0800, Revision 3, dated March 2010 (ADAMS Accession No. ML100351425).

License Amendment Nos. 256 and 200, dated August 28, 2008 (ADAMS Accession No. ML081770075), for HNP, Units 1 and 2, respectively, approved implementation of AST radiological methodology. Full AST implementation replaced the previous AST used in HNP design basis radiological analyses and incorporated the TEDE dose criteria.

3.0 TECHNICAL EVALUATION

3.1 System Design and Operation

The HNP Units 1 and 2 secondary containments are comprised of structures that completely enclose the respective primary containments and those components that may contain primary system fluid. The secondary containment includes the reactor building, standby gas treatment (SGT) system, reactor building isolation control system, and main stack. The function of the secondary containment is to contain, dilute, and hold up fission products that may leak from primary containment following a DBA.

During refueling operations, the secondary containment structure forms a control volume that serves to hold up fission product radioactivity that may be released during a fuel handling accident (FHA). In conjunction with operation of the SGT System and closure of certain secondary containment isolation valves (SCIVs) in lines that 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.

Four exhaust radiation detectors from the refueling floor are located near the respective ventilation exhaust ductwork. Upon detection of high radiation, secondary containment isolation and actuation of the SGT system are initiated automatically.

For further details of the system design and operation, see pages E-1 and E-2 of the letter dated April 24, 2019.

3.2 Current TS Requirements

The following HNP, Units 1 and 2, TSs currently include the Applicability, in part, "During movement of irradiated fuel assemblies..." and "During CORE ALTERATIONS:"

• TS 3.3.6.2, Secondary Containment Isolation Instrumentation
• TS 3.6.4.1, Secondary Containment
• TS 3.6.4.2, Secondary Containment Isolation Valves (SCIVs)
• TS 3.6.4.3, Standby Gas Treatment (SGT) System

These TSs also include applicable actions requiring immediate suspension of movement of irradiated fuel assemblies in the secondary containment and core alterations.

3.3 Reason for the Proposed Change

The licensee stated the reason for the proposed change:

> After sufficient radioactive decay of irradiated fuel following a reactor shutdown, the proposed amendment will allow HNP Units 1 and 2 the flexibility to move personnel and equipment and perform work which would affect secondary containment and SGT system operability during the handling of irradiated fuel. The proposed amendment would also align the HNP TSs more closely, as technically practicable, with the STS [Standard Technical Specifications] described in NUREG-1433, Revision 4.0 (Ref. 3) [sic] [(Ref. 2) ADAMS Accession No. ML12104A192].

3.4 Description of the Proposed Change

The licensee stated the described the proposed change:

> The proposed change revises the applicability requirements of several TSs to require these specifications, "During movement of recently irradiated fuel assemblies," and eliminate the applicability requirement, "During CORE ALTERATIONS." The proposed term "recently," as it relates to irradiated fuel, is described in the associated TS Bases as fuel that has occupied part of a critical reactor core within the previous 24 hours. The TS actions are revised to reflect the change to the TS applicability requirements.

3.4.1 Revision to TS 3.3.6.2

The applicable modes or other specified conditions in Table 3.3.6.2-1 in TS 3.3.6.2, "Secondary Containment Isolation Instrumentation," for Function 4, "Refueling Floor Exhaust Radiation High," are Modes 1, 2, 3, 5, and (a). Note (a) states:

> (a) During CORE ALTERATIONS and during movement of irradiated fuel assemblies in secondary containment.

The applicable modes or other specified conditions are modified to be modes 1, 2, 3, and (a). The specified condition during CORE ALTERATIONS is deleted and the word *recently* is added.

Revised note (a) would state:

(a) During movement of recently irradiated fuel assemblies in secondary containment.

3.4.2 Revision to TS 3.6.4.1

The applicable modes in TS 3.6.4.1, "Secondary Containment," are Modes 1, 2, 3, during movement of irradiated fuel assemblies in the secondary containment, and during CORE ALTERATIONS. The following revisions are proposed:

- Applicability is revised to state, in part, "During movement of *recently* irradiated fuel assemblies in the secondary containment," and *During CORE ALTERATIONS* is deleted.

- Condition E is revised to state, "Secondary containment inoperable during movement of *recently* irradiated fuel assemblies in the secondary containment" *or during CORE ALTERATIONS* is deleted.

- Required Action E.1 is revised to state, "Suspend movement of *recently* irradiated fuel assemblies in the secondary containment," and Required Action E.2, "Suspend CORE ALTERATIONS," and its associated logical connector and Completion Time immediately are deleted.

3.4.3 Revision to TS 3.6.4.2

The applicable Modes in TS 3.6.4.2, "Secondary Containment Isolation Valve (SCIVs)," are Modes 1, 2, 3, during movement of irradiated fuel assemblies in the secondary containment, and during CORE ALTERATIONS. The following revisions are proposed:

- Applicability is revised to state, in part, "During movement of *recently* irradiated fuel assemblies in the secondary containment," and *During CORE ALTERATIONS* is deleted.

- Condition D is revised to state, "Required Action and associated Completion Time of Condition A or B not met during movement of *recently* irradiated fuel assemblies in the secondary containment" *or during CORE ALTERATIONS* is deleted.

- Required Action D.1 is revised to state, "Suspend movement of *recently* irradiated fuel assemblies in the secondary containment," and Required Action D.2, "Suspend CORE ALTERATIONS," and its associated logical connector and Completion Time immediately are deleted.

3.4.4 Revision to TS 3.6.4.3

The applicable modes in Technical Specification 3.6.4.3, "Standby Gas Treatment (SGT) System," are modes 1, 2, 3, during movement of irradiated fuel assemblies in the secondary containment, and during core alterations. The following revisions are proposed:

- Applicability is revised to state, in part, "During movement of recently irradiated fuel assemblies in the secondary containment," and *During CORE ALTERATIONS* is deleted.

- Condition D is revised to state, "Required Action and associated Completion Time of Condition A or B not met during movement of *recently* irradiated fuel assemblies in the secondary containment" *or during CORE ALTERATIONS* is deleted.

- Required Action D.2.1 is re-numbered as D.2 and revised to state, "Suspend movement of *recently* irradiated fuel assemblies in the secondary containment," and Required Action D.2.2, Suspend CORE ALTERATIONS," and its associated logical connector and Completion Time are deleted.

3.5 Current Licensing Basis and Accident Analysis

The fuel handling accident (FHA) involves the drop of a spent fuel assembly during refueling operations. The analysis assumes 172 fuel rods will be damaged as a result of the postulated FHA, and thus instantaneously release of their available gap activity to the environment, taking no credit for reactor building closure or isolation. The depth of water over the damaged fuel is not less than 21 feet and is controlled by TS 3.7.8, "Spent Fuel Storage Pool Water Level," and TS 3.9.6, "Reactor Pressure Vessel (RPV) Water Level."

For further details of the current licensing basis and accident analysis are on pages E-4 through E-7 of the letter dated April 24, 2019.

3.6 NRC Staff Evaluation

The NRC staff evaluated the licensee's application to determine if the proposed changes are consistent with the guidance, regulations, and licensing information discussed in Section 2.0 of this safety evaluation and the approved Traveler TSTF-51, Revision 2.

As described in the Reviewer's Note incorporated by TSTF-51 into the Standard Technical Specifications, information is to be provided by the licensee describing the evaluation of recently irradiated fuel demonstrating that after sufficient radioactive decay has occurred (from the time of shutdown) the radiological doses resulting from a fuel handling accident (FHA) remain below the regulatory limits specified in GDC 19, and well within the offsite radiation dose values specified in 10 CFR 50.67, without crediting the systems not required to be operable.

3.6.1 Evaluation of Recently Irradiated Fuel

Following reactor shutdown, decay of short-lived fission products greatly reduces the fission product inventory present in the irradiated fuel. Radiological dose analyses take credit for the normal decay of irradiated fuel fission products.

The licensing basis evaluation implementing the AST is consistent with RG 1.183, Appendix B, "Assumptions for Evaluating the Radiological Consequences of a Fuel Handling Accident." The licensee's evaluation is based on a core source term inventory using GE14 (10x10) fuel that was run for a 24-month fuel cycle at the power level of 2,818 MWt (100.5 percent of the power level of 2,804 MWt) and includes an additional 10 percent margin and a cycle average maximum bundle radial peaking factor of 1.5.

The FHA analysis evaluated radiological dose with a fission product decay period of 24 hours after shutdown. Two FHA cases analyzed were the:

- An FHA assuming secondary containment isolates automatically, and the SGT system draws a vacuum and filters the radioactivity prior to release to the environment. The radioactive release is filtered and elevated.

- An FHA with no credit taken for secondary containment isolation or operation of the SGT system. The unfiltered radioactive release is assumed to be at ground level for the duration of the accident.

The postulated FHA involves a drop of a fuel assembly on top of other fuel assemblies in the reactor core during refueling operations. The licensee determined that the drop distance associated with this location bounds the maximum height that is allowed by the HNP refueling equipment configuration and this is the limiting case because it results in the maximum release of fission products to the secondary containment. Also, SNC has determined that damage due to a fuel assembly drop over the core in the reactor vessel bounds a drop onto fuel assemblies in the spent fuel pool.

The licensee did not propose any changes to the transport or the control room habitability inputs and assumptions for the FHA, which are documented in the alternative source term license amendment request and its supplements.

3.6.1.1 Source Term

The fission product inventory that constitutes the source term for this event is the gap activity in the fuel rods assumed to be damaged as a result of the postulated design basis FHA. In the licensee's original alternate source term (AST) LAR, which incorporated an additional 10-percent margin as set forth in license amendment Nos. 256 and 200, the NRC staff found this uncertainty allowance and the licensee's implementation of the approved isotope generation and depletion computer code to be acceptable for establishing the AST core inventory for accident analyses. Also, a cycle average maximum bundle radial peaking factor of 1.5 was assumed in the analysis and approved for the damaged bundles. In this regard, volatile constituents of the core fission product inventory migrate from the fuel pellets to the gap between the pellets and the fuel rod cladding during normal power operations. The fission

product inventory in the fuel rod gap of the damaged fuel rods is assumed to be released instantaneously to the surrounding water because of the accident.

The licensee's current LAR uses the source term from the FHA design basis accident analysis. All inputs and assumptions are consistent with the FHA analysis documented in the alternative source term license amendment request and its supplements. In addition, the licensee used inputs and assumptions that are consistent with RG 1.183. Therefore, the NRC staff finds the licensee's FHA source term to be acceptable.

The licensee describes a total of four updates to the HNP design basis FHA radiological consequences analysis as other fuel designs were introduced after the original implementation of the AST. Re-evaluations performed by the licensee demonstrated that due to the 10-percent margin, the analysis of record continued to be bounding. These updates are as follows:

- Version 2 dated February 23, 2015:

    - Corrected the number of full-length rods in a GE14 fuel bundle where the value of equivalent full-length fuel rods was replaced with the correct value. The corrected value increased the fraction of the core with cladding failure (FC) resulting in an increase in the activity released from the fuel by 2-percent.

    - Analyzed the effect of using Global Nuclear Fuel-Americas (GNF-A), GNF2 fuel where two effects were evaluated: (1) the differing number of equivalent full-length rods in a GNF2 bundle; and, (2) the core source term. The reduction in the number of equivalent full-length rods increases the FC value. Some of the GNF2 core source terms are greater than and some are less than those used in Version 1 of the FHA analysis of record. The combined effects were determined to increase onsite and offsite doses 2-percent compared to the base case.

  The calculated onsite and offsite doses increased by 2-percent compared to the base case, and did not alter the radiological consequence results of the FHA analysis of record. This increase was absorbed by the 10-percent margin applied in Version 1 of the FHA analysis of record. Thus, the radiological consequence results of the FHA analysis of record remained bounding.

- Version 3 dated January 6, 2016:

    - For FHA case crediting the SGT system, the reactor building drawdown time was increased from 2 minutes to 10 minutes. The FHA case crediting the SGT system is not the limiting case. Therefore, this change had no impact on the worst-case calculated dose results of the FHA analysis of record.

    - Updated the dose calculation program LOCADOSE cases containing zero in-leakage for the control room and the technical support center with in-leakage values from the loss of coolant accident (LOCA) analysis. This change had no apparent impact on the worst-case calculated dose results because the cases that were updated with LOCA in-leakage values were not the limiting cases for the control room and the technical support center. The limiting cases assumed a conservative in-leakage of 10,000 cfm, which has not been changed.

  - Increased technical support center filter efficiency from 90-percent to 95-percent. This change did not noticeably change the calculated technical support center radiological consequence results of the FHA analysis of record.

  In summary, these changes did not adversely impact the limiting radiological dose consequences of the HNP FHA analysis of record.

- Version 4 dated September 14, 2017:

  - Analyzed the effect of using Contaminated Enriched Uranium Product (CEUP) fuel. Replacing GNF2 fuel with CEUP fuel results in a change in isotopic concentrations. The only change of significance is the increase in Pu-238 concentration. As Pu-238 is not released as part of the FHA, the evaluation conducted for GNF2 fuel is still applicable. Thus, the radiological dose consequences of the HNP FHA analysis of record are not impacted.

- Version 5 dated November 19, 2018:

  - Analyzed the effect of using GNF3 fuel, involving the combined effects of the GNF3 fuel equivalent number of full-length rods, the number of damaged fuel rods, and the core source term, which would be expected to decrease onsite and offsite doses 1-percent compared to the base case. This increases the margin applied in the HNP FHA analysis of record to account for future fuel changes or power uprates. The radiological dose consequences of the HNP FHA analysis of record remain bounding.

  - Updated the effect of using GE14 and GNF2 fuel - the calculated number of damaged equivalent GE14 full length rods decreased and the number of equivalent GNF2 full length rods decreased. As a result, the cladding failure in both cases decreased, thereby reducing the expected onsite and offsite doses compared to the base case and improving the margin applied in the HNP FHA analysis of record. The radiological dose consequences of the HNP FHA analysis of record remain bounding.

  - Updated the effect of using CEUP fuel - replacing GNF3 fuel with CEUP fuel results in a change in isotopic concentrations. The change of significance is the increase in some Group 12 Cerium isotope concentrations. However, the Group 12 Cerium isotopes are not released as part of the FHA and the evaluation for GNF3 fuel is still applicable. Thus, the radiological dose consequences of the HNP FHA analysis of record are not impacted.

These updates to the HNP FHA analysis of record did not result in a change to the methodology or an adverse change in the radiological dose consequences to offsite or control room personnel documented previously in the NRC safety evaluation approving full implementation of AST radiological methodology at HNP by letter dated August 28, 2008 (ADAMS Accession No. ML081770075). The worst-case FHA dose to individuals at the exclusion area boundary and low population zone continues to be 1.2 rem TEDE and remains below the 10 CFR

50.67(b)(2)(i) and (ii) radiation dose criteria of 25 rem TEDE, and below the radiation dose criterion of 6.3 rem TEDE specified in Table 1, "Accident Dose Criteria," of NRC NUREG-0800, Section 15.0.1 and Table 6, "Accident Dose Criteria," of NRC RG 1.183. The worst-case FHA dose to personnel in the control room continues to be 3.5 rem TEDE. This FHA dose assumes isolation, pressurization, and filtration of the control room. The worst-case FHA dose to the control room continues to remain below the 10 CFR 50.67(b)(2)(iii) radiation dose criterion of 5 rem TEDE.

3.6.1.2 Recently Irradiated FHA Conclusion

The licensee evaluated the radiological consequences resulting from a FHA with recently irradiated fuel and concluded that the radiation doses at the exclusion area boundary, low population zone, and control room are well within the radiation dose guidelines provided in 10 CFR 50.67 and accident-specific dose criteria specified in SRP Section 15.0.1. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified in Section 2.3 of this safety evaluation. The NRC staff performed independent confirmatory dose evaluations, as necessary, to ensure a thorough understanding of the licensee's methods. The NRC staff finds that the exclusion area boundary, low population zone, and control room doses estimated by the licensee for the FHA with recently irradiated fuel meets the applicable accident dose criteria and are, therefore, acceptable.

3.6.2 Heavy Load Drop Analysis

The NRC staff evaluated the impact of the proposed changes on the heavy load drop analysis. The control of movement of loads heavier than a fuel assembly over irradiated fuel is described in the licensee's responses to Generic Letter 81-07, "Control of Heavy Loads," dated February 3, 1981, which references NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," (ADAMS Accession Number ML070250180). Subsection 10.20, "Overhead Handling Systems," of the HNP, Unit 1, final safety analysis report (FSAR) describes the licensing basis regarding control of heavy loads for both units, and HNP technical requirements manual (TRM) Section T 3.9.4, "Crane Travel," and associated bases provide administrative controls governing movement of heavy loads over fuel assemblies in the spent fuel storage pool racks.

As summarized in HNP, Unit 1, FSAR sub-subsection 10.20.2, "Power Generation Design Bases," the Unit 1 reactor building crane provides service to both units. The HNP, Unit 1, reactor building crane is designed to withstand a single failure and maintain the capability to safely retain its load. The HNP, Unit 2, reactor building crane is not a single-failure-proof crane and therefore is used under strict administrative controls over the refueling floor. A load drop analysis has been performed to determine maximum lifting heights above the floor and load paths to be followed whenever the HNP, Unit 2, reactor building crane is used over the refueling floor. The TRM Section T 3.9.4 and associated bases require heavy load lifts over the fuel assemblies in the spent fuel storage pool racks to use the HNP, Unit 1, single-failure-proof crane in conjunction with the specified lifting devices in compliance with ANSI B30.9-1971, "Slings," and ANSI N14.6-1978, "Standard For Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4,500 kg) or More for Nuclear Materials."

The load drop analysis for handling heavy loads over the refueling floor with the HNP, Unit 2, reactor building crane demonstrates that the moving of heavy loads within established safe load heights and paths is acceptable. Use of any hoist other than the HNP, Unit 1, reactor building crane over any equipment required to reach and maintain cold shutdown of either unit is prohibited. Therefore, core cooling capability and the integrity of the fuel cladding will be maintained. An inadvertent drop of a heavy load using the HNP, Unit 1, reactor building crane is precluded due to the single failure design of the crane, and a load drop analysis demonstrates that an inadvertent drop of a heavy load using the HNP, Unit 1, reactor building crane would have no impact on the health or safety of the public.

The licensing basis for heavy load drop analysis demonstrates that moving a heavy load within the established safe load paths is acceptable and that there will be no consequential damage to the integrity of the fuel cladding or core cooling capability. The NRC staff finds there is no impact on the heavy load drop analysis and that the proposed changes are acceptable.

3.6.3 Removal of the Core Alterations TSs Applicability

As documented in a letter from the NRC staff to Technical Specifications Task Force (TSTF) (ADAMS Accession No. ML17346A587), the NRC staff concluded after considerable review and analysis, that for certain facilities, adopting TSTF-51 could result in exceeding the bounding licensing basis FHA analysis of record dose for the control room. Following approval of the licensee's current LAR, the secondary containment, SCIVs, secondary containment instrumentation, and SGT systems are no longer required to be operable during core alterations or as a primary success path for mitigating the FHA to ensure off-site and control room doses remain below the 10 CFR 50.67 dose limits. The isolation, pressurization, and filtration of the control room continues to be assumed in the FHA, and, therefore, these requirements are not modified by the proposed amendment request.

The NRC staff identified that dropping a source, fuel assembly, or component during core alterations could damage a recently irradiated fuel assembly creating a radioactive source term that may result in exceeding the resultant radiological doses calculated by the licensing basis FHA analysis of record. Therefore, the NRC staff recommended that when adopting TSTF-51, that licensees provide one of the following discussions in their LAR for TSs revised to remove the defined term "core alterations" from their applicability:

- Confirm that the length of time defined as "recently" is less than the time required to remove the reactor vessel head and internals and expose the irradiated fuel after a shutdown;

- Provide an analysis that demonstrates that the dropping of any unirradiated fuel assembly, sources, reactivity control component, or other component affecting reactivity within the reactor vessel onto irradiated fuel assemblies prior to the period of time defined as "recently" will not result in a radioactive release from the irradiated fuel;

- Describe the limitations or controls that would prevent movement of any unirradiated fuel assembly, source, reactivity control component, or other component affecting reactivity within the reactor vessel capable of damaging a fuel assembly prior to the time period defined as "recently"; or

- Provide an analysis that demonstrates that the dose consequences of a failure of a single irradiated fuel assembly with no technical specification-required mitigation systems available remain below the regulatory limits and the regulatory guidance limits for a fuel handling accident.

In the licensee's current LAR, the licensee confirmed that the length of time defined as "recently" is less than the time required to remove the reactor vessel head and internals and expose the irradiated fuel after a shutdown. The licensee states that, "based on the design of HNP, Units 1 and 2, the time required to disassemble the reactor (e.g., remove the reactor vessel head, steam dryer, steam separators, core shroud head, and internals) and expose the irradiated fuel after a reactor shutdown is greater than 24 hours." The NRC staff has determined that the radiological consequences of dropping a source, fuel assembly, or component during core alterations will not exceed the radiological consequences resulting from damage to a recently irradiated fuel assembly. Therefore, the radiological dose consequences will remain bounded by the analysis of an FHA with recently irradiated fuel as described in the HNP, Unit 2, updated (FSAR) subsection 15.3.5 and, therefore, the proposed changes are acceptable.

3.7 Variations from TSTF-51

The proposed amendment is based on the STS changes described in TSTF–51, Revision 2, except insofar as SNC proposes variations from the NUREG-1433 markups in TSTF-51, as identified below. The LAR includes differing TS numbers and TS titles, where applicable, as discussed below.

(1) The main control room environmental control (MCREC) system instrumentation and the MCREC system continue to be assumed to provide isolation, pressurization, and filtration of the MCR in the event of an FHA. Since this system and associated isolation instrumentation are mitigation systems necessary to maintain dose to personnel in the MCR below the regulatory and regulatory guidance limits for an FHA, the following TSs and associated Bases are not modified:

    - TS 3.3.7.1, "Main Control Room Environmental Control (MCREC) System Instrumentation,"
    - TS 3.7.4, "Main Control Room Environmental Control (MCREC) System,"
    - TS 3.7.5, "Control Room Air Conditioning (AC) System,"
    - TS 3.8.2, "AC Sources – Shutdown,"
    - TS 3.8.5, "DC Sources – Shutdown," and
    - TS 3.8.8, "Distribution Systems – Shutdown."

This is a plant-specific variation from TSTF-51, in which the HNP TS numbers and titles differ from TSTF-51.

(2) Mode 5 is deleted from the Applicable Modes or Other Specified Conditions column for Function 4, "Refueling Floor Exhaust Radiation – High," in Table 3.3.6.2-1 as described in Section 3.3 herein. Mode 5 is not specified in NUREG-1433 TS Table 3.3.6.2-1 for the equivalent function. Therefore, this is a plant-specific variation from TSTF-51. The NRC staff reviewed this proposed change to determine the impact on the proposed radiological

consequences for the design basis accidents that occur in Mode 5. The secondary containment, its associated instrumentation, secondary containment isolation valves, and the standby gas treatment systems are no longer required to be operable during core alterations or as a primary success path for mitigating the FHA to ensure off-site and MCR doses remain below the 10 CFR 50.67 dose limits. In addition, the FHA is the only design basis accident applicable in Mode 5. The NRC staff finds that the proposed change has no impact on the inputs, assumptions, or methodologies used to determine the radiological consequences for the FHA with recently irradiated fuel, therefore the proposed change is acceptable from a radiological dose perspective.

(3) Function 5, "Manual Initiation," of NUREG-1433, TS Table 3.3.6.2-1 is not applicable to HNP, Units 1, and 2, and, therefore, marked up pages of the associated TS Bases are not included. This is an administrative variation from TSTF-51.

(4) Condition G of NUREG-1433, TS 3.6.1.3, "Primary Containment Isolation Valves (PCIVs)," and NUREG-1433, TS 3.8.8, "Inverters – Shutdown," are not applicable to HNP, Units 1 and 2, and, therefore, marked up pages of the associated TSs and Bases are not included. This is an administrative variation from TSTF-51.

(5) TSTF-542, "Reactor Pressure Vessel Water Inventory Control," was approved for HNP, Units 1 and 2, in License Amendments 290 and 235, respectively (ADAMS Accession No. ML18123A368) eliminating requirements related to operations with a potential for draining the reactor vessel. Therefore, variations related to the NUREG-1433 marked up TS and Bases pages consider these changes. This is an administrative variation from TSTF-51.

(6) The TS Bases are revised, where applicable, consistent with TSTF-51. Plant specific changes are made (additions, deletions, and/or changes) to reflect the plant specific nomenclature, number, reference, system description, analysis, or licensing basis description, including the addition of the radioactive decay period of 24 hours assumed in the FHA radiological dose analysis. The proposed changes are considered administrative variations from TSTF-51.

The differences from TSTF-51 listed herein are either: (1) necessary variations to maintain the requirements for required safety systems assumed in the HNP FHA analysis; or (2) minor variations or deviations that are administrative in nature. Therefore, the NRC staff finds these differences from TSTF-51 acceptable.

3.8 NRC Staff Technical Evaluation Conclusion

The licensee evaluated the radiological consequences resulting from a FHA and concluded that the radiological consequences at the exclusion area boundary, low population zone, and control room are within the radiological dose guidelines provided in 10 CFR 50.67, 10 CFR 50 Appendix A, Criterion 19, and accident-specific dose criteria specified in SRP Section 15.0.1 and RG 1.183. The NRC staff's review has found that the licensee used analysis assumptions and inputs consistent with applicable regulatory guidance identified above in Section 2.0. To verify the licensee's analyses, the NRC staff performed confirmatory radiological consequence dose calculations and compared the results to those calculated by the licensee. The radiological consequences calculated by both the licensee and the NRC staff provided results

that are well within the radiation dose criteria set forth in 10 CFR 50.67. The NRC staff finds that the exclusion area boundary, low population zone, and control room doses estimated by the licensee for the FHA with recently irradiated fuel meets the applicable accident-specific dose criteria, there is no impact to the heavy load drop analysis of record, and a new source term will not be created during core alterations. Therefore, the licensee's proposed changes are acceptable.

The NRC staff reviewed the technical basis provided by the licensee to assess the radiological impacts of the changes to the FHA analysis in the licensee's technical specifications. The NRC staff finds that the licensee's proposed TS changes in Section 3.4 of this safety evaluation are consistent with the regulatory requirements and guidance identified in Section 2.0 of this safety evaluation. The NRC staff finds, with reasonable assurance that the licensee's changes to the TSs will continue to comply with these criteria. Therefore, the proposed TSs changes in Section 3.4 of this safety evaluation are acceptable with regard to the radiological consequences of the postulated DBAs. Based on the above, the NRC staff concludes that the requirements of 10 CFR 50.36 will 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 27, 2019. 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 (84 FR 33987; July 16, 2019). 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.

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Date: November 19, 2019

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING REVISION TO TECHNICAL SPECIFICATION REQUIREMENTS DURING HANDLING IRRADIATED FUEL AND CORE ALTERATIONS (EPID L-2019-LLA-0091 DATED NOVEMBER 19, 2019

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