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

September 26, 2013

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

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, FARLEY NUCLEAR PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING STEAM GENERATOR TUBE INSPECTION PROGRAM (TAC NOS. MF0564, MF0565, MF0562 AND MF0563)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 171 to Renewed Facility Operating License NPF-68 and Amendment No. 153 to Renewed Facility Operating License NPF-81 for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2, and Amendment No. 192 to Renewed Facility Operating License NPF-2 and Amendment No. 188 to Renewed Facility Operating License NPF-8 for the Farley Nuclear Plant (FNP), Units 1 and 2, in response to your letters dated January 23 and July 17, 2013. The amendments revise the following Technical Specifications (TSs) for each facility: 3.4.17, "Steam Generator (SG) Tube Integrity," 5.5.9, "Steam Generator (SG) Program," and 5.6.10, "Steam Generator Tube Inspection Report." These changes summarize and clarify the purpose of the TSs in accordance with TS Task Force Traveler 510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

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,

Robert Martin

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

Docket Nos. 50-424, 50-425, 50-348 and 50-364

Enclosures:
1. Amendment No. 171 to NPF-68
2. Amendment No. 153 to NPF-81
3. Amendment No. 192 to NPF-2
4. Amendment No. 188 to NPF-8
5. Safety Evaluation

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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. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 171
Renewed License No. NPF-68

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 1 (the facility) Renewed Facility Operating License No. NPF-68 filed by the 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 January 23 and July 17, 2013, 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-68 is hereby amended to read as follows:

   C. Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 171, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-68
and the Technical Specifications

Date of Issuance: September 26, 2013

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 153
Renewed License No. NPF-81

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

   A. The application for amendment to the Vogtle Electric Generating Plant, Unit 2 (the facility) Renewed Facility Operating License No. NPF-81 filed by the 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 January 23 and July 17, 2013, 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-81 is hereby amended to read as follows:

C. Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 153, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-81
and the Technical Specifications

Date of Issuance: September 26, 2013

ATTACHMENT

TO LICENSE AMENDMENT NO. 171

RENEWED FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 153

RENEWED FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

Replace the following pages of the Licenses 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
License No. NPF-68, page 4	License No. NPF-68, page 4
License No. NPF-81, page 3	License No. NPF-81, page 3
TSs	TSs
3.4.17-1	3.4.17-1
3.4.17-2	3.4.17-2
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9	5.5-9
5.5-10	5.5-10
5.6-6	5.6-6

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 171 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

(4) Deleted

(5) Deleted

(6) Deleted

(7) Deleted

(8) Deleted

(9) Deleted

(10) Mitigation Strategy License Condition

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

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

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

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

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

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

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

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

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

(1) Maximum Power Level

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

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 153 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

---NOTE---
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1	Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	AND		
	A.2	Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate tube plugging criteria shall be applied as an alternative to the 40% depth based criteria:

Tubes with service-induced flaws located greater than 15.2 inches below the top of the tubesheet do not require plugging. Tubes with service-induced flaws located in the portion of the tube from the top of the tubesheet to 15.2 inches below the top of the tubesheet shall be plugged upon detection.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. Portions of the tube below 15.2 inches below the top of the tubesheet are excluded from this requirement.

The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.

   a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period.

   b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and

   c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

3. If crack indications are found in portions of the SG tube not excluded above, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic nondestructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

(continued)

5.6 Reporting Requirements

5.6.9 Deleted.

5.6.10 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing,

h. The primary to secondary LEAKAGE rate observed in each SG (if it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG) during the cycle preceding the inspection which is the subject of the report; and

i. The calculated accident induced leakage rate from the portion of the tubes below 15.2 inches from the top of the tubesheet for the most limiting accident in the most limiting SG. In addition, if the calculated accident induced leakage rate from the most limiting accident is less than 2.48 times the maximum operational primary to secondary leakage rate, the report should describe how it was determined.

j. The results of monitoring for tube axial displacement (slippage). If slippage is discovered, the implications of the discovery and corrective action shall be provided.

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SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 192
Renewed License No. NPF-2

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

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated January 23, 2013, as supplemented by a letter dated July 17, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 192, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 26, 2013

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 188
Renewed License No. NPF-8

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

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated January 23, 2013, as supplemented by a letter dated July 17, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-8 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 188, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 26, 2013

ATTACHMENT

TO LICENSE AMENDMENT NO. 192

RENEWED FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

AND

TO LICENSE AMENDMENT NO. 188

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the License and Appendix 'A Technical Specifications (TSs) with the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
NPF-2, page 4	NPF-2, page 4
NPF-8, page 3	NPF-8, page 3
TS Page	TS Page
3.4.17-1	3.4.17-1
3.4.17-2	3.4.17-2
5.5-5	5.5-5
5.5-6	5.5-6
5.5-7	5.5-7
5.5-8	5.5-8
5.5-9 through 5.5-16	5.5-9 through 5.5-16
5.6-6	5.6-6

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 192, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(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 renewed license supported by a favorable evaluation by the Commission.

a. Southern Nuclear shall not operate the reactor in Operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

b. Deleted per Amendment 13

c. Deleted per Amendment 2

d. Deleted per Amendment 2

e. Deleted per Amendment 152

Deleted per Amendment 2

f. Deleted per Amendment 158

g. Southern Nuclear shall maintain a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical parameters and control points for these parameters;

2) Identification of the procedures used to quantify parameters that are critical to control points;

3) Identification of process sampling points;

4) A procedure for the recording and management of data;

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license.

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

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

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

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

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

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2775 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 188 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube plugging criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next inspection. AND A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program (continued)

b. A surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program.

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following:

a. Provisions for condition monitoring assessments. Condition monitoring

(continued)

5.5.9 Steam Generator (SG) Program (continued)

assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby and cooldown), all anticipated transients included in the design specification, and design basis accidents. This includes retaining a safety factor of 3.0 ($3\Delta P$) against burst under normal steady state full power operation primary to secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Accident induced leakage is not to exceed 1 gpm total for all three SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

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5.5.9 Steam Generator SG Program (continued)

c. Provisions for SG tube plugging criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube plugging criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG installation.

2. After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the

(continued)

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5.5.9 Steam Generator SG Program (continued)

conclusion of the included SG inspection outage.

a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;

b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;

c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and

d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3. If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the condenser hotwells for evidence of condenser in leakage;

d. Procedures for the recording and management of data;

(continued)

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5.5.10 Secondary Water Chemistry Program (continued)

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 3, and in accordance with ASME N510-1989. The FNP Final Safety Analysis Report identifies the relevant surveillance testing requirements.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.5% when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Flowrate (CFM)
CREFS Recirculation	2,000 ± 10%
CREFS Filtration	1,000 ± 10%
CREFS Pressurization	300 + 25% to - 10%
PRF Post LOCA Mode	5,000 ± 10%

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.5% when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Flowrate (CFM)
CREFS Recirculation	2,000 ± 10%
CREFS Filtration	1,000 ± 10%
CREFS Pressurization	300 + 25% to - 10%
PRF Post LOCA Mode	5,000 ± 10%

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in ASME N510-1989, shows the methyl iodide penetration less than the value

(continued)

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5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
CREFS Recirculation	2.5%	70%
CREFS Filtration	2.5%	70%
CREFS Pressurization	0.5%	70%
PRF Post LOCA Mode	5%	95%

NOTE: CREFS Pressurization methyl iodide penetration limit is based on a 6-inch bed depth.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Delta P (in. water gauge)	Flowrate (CFM)
CREFS Recirculation	2.3	2,000 ± 10%
CREFS Filtration	2.9	1,000 ± 10%
CREFS Pressurization	2.2	300 + 25% to - 10%
PRF Post LOCA Mode	2.6	5,000 ± 10%

e. Demonstrate that the heaters for the CREFS Pressurization System dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF Ventilation System	Wattage (kW)
CREFS Pressurization	2.5 ± 0.5

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

(continued)

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5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design;

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of $\geq$ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than 10 curies.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to the emergency diesel generator storage tanks by deterrnining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a clear and bright appearance with proper color; or a water and sediment content within limits.

b. Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

(continued)

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5.5.13 Diesel Fuel Oil Testing Program (continued)

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance test frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

1. a change in the TS incorporated in the license; or

2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

(continued)

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5.5.15 Safety Function Determination Program (SFDP) (continued)

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Main Steamline Inspection Program

The three main steamlines from the rigid anchor points of the containment penetrations downstream to and including the main steam header shall be inspected. The extent of the inservice examinations completed during each inspection interval (IWA 2400, ASME Code, 1974 Edition, Section XI) shall provide 100 percent volumetric examination of circumferential and longitudinal pipe welds to the extent practical. The areas subject to examination are those defined in accordance with examination category C-G for Class 2 piping welds in Table IWC-2520.

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J,

(continued)

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program (continued)

Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

Section 9.2.3: The next Type A test, after the March 1994 test for Unit 1 and the March 1995 test for Unit 2, shall be performed during refueling outage R22 (Spring 2009) for Unit 1 and during refueling outage R20 (Spring 2010) for Unit 2. This is a one time exception.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

c. During plant startup following testing in accordance with this program, the leakage rate acceptance criterion for each containment purge penetration flowpath is $\leq 0.05\ L_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

5.5 Programs and Manuals

5.5.18 Control Room Integrity Program (CRIP)

A Control Room Integrity Program (CRIP) shall be established and implemented to ensure that the control room integrity is maintained such that a radiological event, hazardous chemicals, or a fire challenge (e.g., fire byproducts, halon, etc.) will not prevent the control room operators from controlling the reactor during normal or accident conditions. The program shall require testing as outlined below. Testing should be performed when changes are made to structures, systems and components which could impact Control Room Impact (CRE) integrity. These structures, systems and components may be internal or external to the CRE. Testing should also be conducted following a modification or a repair that could affect CRE inleakage. Testing should also be performed if the conditions associated with a particular challenge result in a change in operating mode, system alignment or system response that could result in a new limiting condition. Testing should be commensurate with the type and degree of modification or repair. Testing should be conducted in the alignment that results in the greatest consequence to the operators.

A CRIP shall be established to implement the following:

a. Demonstrate, using Regulatory Guide (RG) 1.197 and ASTM E741, that CRE inleakage is less than the below values. The values listed below do not include 10 cfm assumed in accident analysis for ingress / egress.

   i) 43 cfm when the control room ventilation systems are aligned in the emergency recirculation mode of operation,

   ii) 600 cfm when the control room ventilation systems are aligned in the isolation mode of operation, and

   iii) 2,340 cfm when the control room ventilation systems are aligned in the normal mode of operation;

b. Demonstrate that the leakage characteristics of the CRE will not result in simultaneous loss of reactor control capability from the control room and the hot shutdown panels;

c. Maintain a CRE configuration control and a design and licensing bases control program and a preventative maintenance program. As a minimum, the CRE configuration control program will determine whether the i) CRE differential pressure relative to adjacent areas and ii) the control room ventilation system flow rates, as determined in accordance with ASME N510-1989 or ASTM E2029-99, are consistent with the values measured at the time the ASTM E741 test was performed. If item i or ii has changed, determine how this change has affected the inleakage characteristics of the CRE. If there has been degradation in the inleakage characteristics of the

(continued)

5.5 Programs and Manuals

5.5.18 Control Room Integrity Program (CRIP) (continued)

CRE since the E741 test, then a determination should be made whether the licensing basis analyses remain valid. If the licensing basis analyses remain valid, the CRE remains OPERABLE.

d. Test the CRE in accordance with the testing methods and at the frequencies specified in RG 1.197, Revision 0, May 2003.

The provisions of SR 3.0.2 are applicable to the control room inleakage testing frequencies.

5.5.19 Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.

b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.

c. The provisions of Surveillance Requirements 3.0.2 and 3.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

5.6 Reporting Requirements

5.6.8 PAM Report

When a report is required by Condition B or F of LCO 3.3.3, "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.

5.6.9 Deleted

5.6.10 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each degradation mechanism,

f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator,

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NO. 171 TO RENEWED FACILITY OPERATING LICENSE NPF-68

AMENDMENT NO. 153 TO RENEWED FACILITY OPERATING LICENSE NPF-81

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

AND

AMENDMENT NO. 192 TO RENEWED FACILITY OPERATING LICENSE NPF-2

AMENDMENT NO. 188 TO RENEWED FACILITY OPERATING LICENSE NPF-8

FARLEY NUCLEAR PLANT, UNITS 1 AND 2

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

1.0 INTRODUCTION

By application dated January 23 and July 17, 2013 (Agencywide Documents and Management System (ADAMS) Accession No. ML13025A163 and ML13199A181), Southern Nuclear Operating Company (SNC, the licensee) proposed changes to the Technical Specifications (TSs) for Joseph M. Farley Nuclear Plant, Units 1 and 2 (FNP) and Vogtle Electric Generating Plant, Units 1 and 2 (VEGP), to adopt U.S. Nuclear Regulatory Commission (NRC)-approved Revision 2 to Technical Specifications Task Force (TSTF) Standard Technical Specifications (STS) change traveler TSTF-510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

The proposed changes, which apply to the same section numbers for each facility's technical specifications, revise TS Limiting Condition for Operation (LCO) 3.4.17, "Steam Generator (SG) Tube Integrity," Specification 5.5.9, "Steam Generator (SG) Program," and Specification 5.6.10, "Steam Generator Tube Inspection Report." The specific changes concern SG inspection periods, and address applicable changes and clarifications. The licensee stated that its license amendment request (LAR) is consistent with the Notice of Availability of TSTF-510, Revision 2, announced in the *Federal Register* on October 27, 2011 (76 FR 66763) as part of the consolidated line item improvement process (CLIIP).

The current STS requirements in the above specifications were established in May 2005 with the NRC staff's approval of TSTF-449, Revision 4, "Steam Generator Tube Integrity" (NRC *Federal Register* Notice of Availability (70 FR 24126)). The TSTF-449 changes to the STS incorporated a new, largely performance-based approach for ensuring the integrity of the SG tubes is maintained. The performance-based requirements were supplemented by

prescriptive requirements relating to tube inspections and tube repair limits to ensure that conditions adverse to quality are detected and corrected on a timely basis. As of September 2007, the TSTF-449, Revision 4, changes were adopted in the plant TS for all PWRs.

The proposed changes in TSTF-510, Revision 2, reflect licensees' early implementation experience with respect to the TSTF-449, Revision 4. TSTF-510 characterizes the changes as editorial corrections, changes, and clarifications intended to improve internal consistency, consistency with implementing industry documents, and usability without changing the intent of the requirements. The proposed changes are an improvement to the existing SG inspection requirements and continue to provide assurance that the plant licensing basis will be maintained between SG inspections.

2.0 REGULATORY EVALUATION

The SG tubes in pressurized water reactors (PWRs) have a number of important safety functions. These tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system and are relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damage during severe accidents.

10 CFR 50.36, "Technical Specifications," establishes the requirements related to the content of the TS. Pursuant to 10 CFR 50.36, TSs are required to include items in the following categories related to this issue: LCOs, SRs, and administrative controls. LCOs, SRs and administrative controls in the FNP and VEGP TS relevant to SG tube integrity are in 3.4.17 SG Tube Integrity, 5.5.9 Steam Generator Program, and 5.6.10 Steam Generator Tube Inspection Report .

10 CFR 50.36(c)(5) defines administrative controls as "the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure the operation of the facility in a safe manner." Programs established by the licensee to operate the facility in a safe manner, including the SG Program, are listed in the administrative controls section of the TS. The SG Program is defined in Specification 5.5.9, while the reporting requirements relating to implementation of the SG Program are in Specification 5.6.10.

VEGP and FNP Specification 5.5.9 requires that an SG Program be established and implemented to ensure that SG tube integrity is maintained. Specification 5.5.9.a requires that a condition monitoring assessment be performed during each outage in which the SG tubes are inspected, to confirm that the performance criteria are being met. SG tube integrity is maintained by meeting the performance criteria specified in TS 5.5.9.b for structural and leakage integrity, consistent with the plant design and licensing basis. The applicable tube repair criteria, specified in TS 5.5.9.c, are that tubes found during ISI to contain flaws with a depth equal to or exceeding 40 percent of the nominal wall thickness shall be plugged, unless the tubes are permitted to remain in service through application of the alternate repair criteria. Specification 5.5.9.d includes provisions regarding the scope, frequency, and methods of SG tube inspections. These provisions require

that the inspections be performed with the objective of detecting flaws of any type as described in the TS.

3.0 TECHNICAL EVALUATION

Each proposed change to the TS is described individually below, followed by the NRC staff's assessment of the change. This evaluation applies to both FNP and VEGP, except where noted.

3.1 Specification 5.5.9, "Steam Generator (SG) Program"

Delete the last two words in the last sentence in the introductory paragraph as follows: "In addition, the Steam Generator Program shall include: ~~the following provisions~~."

The subsequent paragraph starts with "Provisions for..." and stating "provisions" in the introductory paragraph is duplicative. The NRC staff has reviewed the licensee's proposed change to Specification 5.5.9 and agrees that this editorial change to eliminate duplicative text is acceptable.

3.2 Specification 5.5.9, Paragraph 5.5.9 b.1, "Structural integrity performance criterion"

The first sentence currently states:

> All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down, and all anticipated transients included in the design specification) and design basis accidents.

Proposed Change: Revise the sentence to relocate the end colon to after the word "down "as follows:

> All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down), all anticipated transients included in the design specification, and design basis accidents.

The basis for the change is that the sentence inappropriately includes anticipated transients in the description of normal operating conditions. The NRC staff finds this editorial change to differentiate anticipated transients from normal operating conditions to be acceptable.

3.3. Paragraph 5.5.9.c, "Provisions for SG tube repair criteria," Paragraph 5.5.9.d, "Provisions for SG tube inspections," LCO 3.4.17, "Steam Generator (SG) Tube Integrity"

Proposed Change: Change all references to "tube repair criteria" to "tube plugging criteria." This change is intended to be consistent with the treatment of SG tube repair throughout Specification 5.5.9.

Assessment:

The following paragraph is from TSTF-510, Revision 2 (ML110610350), regarding clarification of "tube repair" versus "tube plugging" in the updating of the STSs.

> Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. Steam generator tubes have experienced tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. When the degradation of the tube wall reaches a prescribed criterion for action, the tube is considered defective and corrective action [tube plugging or tube repair] is taken. ...

Neither FNP nor VEGP have repair techniques included in their TS. As a result, they can only plug defective tubes. Accordingly, the licensee has elected to adopt the change to "tube plugging" in accordance with TSTF-510, Revision 2, in all applicable places in the FNP and VEGP TS. The proposed change provides a more accurate description of the response to tube degradation, adding clarity to the specification. Therefore, the NRC staff finds the change acceptable.

3.4 Paragraph 5.5.9.d, "Provisions for SG tube inspections"

Proposed Change: Change the term "an assessment of degradation" to "a degradation assessment" to be consistent with the terminology used by the industry.

Assessment: The NRC staff agrees that the terminology should be consistent and finds the change acceptable.

3.5 Paragraph 5.5.9.d.1

The paragraph currently states: "Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement."

Proposed change: The change would replace "SG replacement" with "SG installation." The basis for the change is that "SG installation" encompasses "SG replacement."

Assessment: The NRC staff finds that the change involves no change in the inspection requirement, will provide more generic terminology from plant to plant and is acceptable.

3.6 Paragraph 5.5.9.d.2 – FNP and VEGP

3.6.1 Paragraph 5.5.9.d.2 for FNP – SGs with alloy 690 thermally-treated (TT) tubes

> NOTE: As described in FNP's Updated Final Safety Analysis Report (UFSAR) Revision 23, Table 5.2-22, "Reactor Coolant Pressure Boundary Materials," both Unit 1 and Unit 2 use steam generator tubes made of ASME SB-163, Alloy 690 TT material.

The paragraph currently states:

> Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

Proposed Change: Revise paragraph 5.5.9.d.2 as follows:

> After the first refueling outage following SG installation, inspect each SG at least every 72 effective full power months or at least every third refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, c, and d below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.
>
> a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 144 effective full power months. This constitutes the first inspection period;
>
> b) During the next 120 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period;
>
> c) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the third inspection period; and
>
> d) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the fourth and subsequent inspection periods.

3.6.2 Paragraph 5.5.9.d.2 for VEGP – SGs with alloy 600 TT tubes

NOTE: VEGP, Units 1 and 2, both utilize 600TT SG tubes as described in VEGP UFSAR, Revision 17, Section 5.4.2.4.1, "Selection and Fabrication of Materials."

The paragraph currently states:

> Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

Proposed Change: Revise paragraph 5.5.9.d.2 as follows:

> After the first refueling outage following SG installation, inspect each SG at least every 48 effective full power months or at least every other refueling outage (whichever results in more frequent inspections). In addition, the minimum number of tubes inspected at each scheduled inspection shall be the number of tubes in all SGs divided by the number of SG inspection outages scheduled in each inspection period as defined in a, b, and c below. If a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube repair criteria, the minimum number of locations inspected with such a capable inspection technique during the remainder of the inspection period may be prorated. The fraction of locations to be inspected for this potential type of degradation at this location at the end of the inspection period shall be no less than the ratio of the number of times the SG is scheduled to be inspected in the inspection period after the determination that a new form of degradation could potentially be occurring at this location divided by the total number of times the SG is scheduled to be inspected in the inspection period. Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage.
>
> a) After the first refueling outage following SG installation, inspect 100% of the tubes during the next 120 effective full power months. This constitutes the first inspection period;
>
> b) During the next 96 effective full power months, inspect 100% of the tubes. This constitutes the second inspection period; and
>
> c) During the remaining life of the SGs, inspect 100% of the tubes every 72 effective full power months. This constitutes the third and subsequent inspection periods.

3.6.3 Assessment for Sections 3.6.1 and 3.6.2 above:

The proposed changes to TS 5.5.9.d.2 are similar for each of the tube alloy types used in domestic steam generators, but with differences that reflect the improved resistance of alloy 690 TT to stress corrosion cracking relative to both alloy 600 mill annealed (MA) and alloy 600 TT. These differences include progressively larger maximum inspection interval requirements and sequential inspection periods (during which 100% of the tubes must be inspected) for alloy 600 MA, 600 TT, and alloy 690 TT tubes, respectively. In addition, because of the longer maximum inspection intervals allowed for alloy 600 TT and 690 TT tubes, paragraph 5.5.9.d.2 includes a restriction on the distribution of sampling over each sequential inspection period for alloy 600 TT and 690 TT tubes that is not included for alloy 600 MA tubes.

FNP and VEGP propose to move the first two sentences of paragraph 5.5.9.d.2 to the end of the paragraph and make editorial changes to improve clarity. The NRC staff finds these changes to be of a clarifying nature, not changing the current intent of these two sentences. However, the LAR also includes changes to when inspections are performed for each SG tube type as follows:

FNP (690 TT tubes)

- The second inspection period would be revised from 108 to 120 EFPM.
- The third inspection period would be revised from 72 to 96 EFPM.
- The fourth and subsequent inspection periods would be revised from 60 to 72 EFPM.

VEGP (600 TT tubes)

- The second inspection period would be revised from 90 to 96 EFPM.
- The third and subsequent inspection periods would be revised from 60 to 72 EFPM.

The licensee characterizes these changes as marginal increases for consistency with typical fuel cycle lengths that better accommodate the scheduling of inspections. The NRC staff observes that this is clearly the case for plants operating with 18- or 36-month inspection intervals (one or two fuel cycles, respectively). With these intervals, the last scheduled inspection during the first inspection period would coincide with the end of the first, third, and subsequent inspection periods. The NRC staff finds that for plants operating with 54-month inspection intervals (three fuel cycles); the end of each inspection period will not generally coincide with a scheduled inspection outage.

The proposed changes would generally increase the number of inspections for FNP in each of the third and subsequent inspection periods by up to one additional inspection, for VEGP in each of the second and subsequent inspection periods by up to one additional inspection. This could reduce the required average minimum sample size during these periods. However, inspection sample sizes will continue to be subject to paragraph 5.5.9.d which states that in addition to meeting the requirements of paragraph 5.5.9.d.1, d.2, and d.3, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure SG tube integrity is maintained until the next scheduled inspection. Therefore, the NRC staff concludes that with the proposed changes to the length of the second and subsequent inspection periods, compliance with the SG program requirements in Specification 5.5.9.d will continue to ensure both adequate inspection scopes and tube integrity.

For each inspection period, paragraph 5.5.9.d.2 currently requires that at least 50 percent of the tubes be inspected by the refueling outage nearest to the mid-point of the inspection period and the remaining 50 percent by the refueling outage nearest the end of the inspection period. The NRC staff notes that if there are not an equal number of inspections in the first half and second half of the inspection period, the average minimum sampling requirement may be markedly different for inspections in the first half of the inspection period compared to those in the second half, even when there are uniform intervals between each inspection.

For example, a plant in the 120 EFPM inspection period with a scheduled 36-month interval (two fuel cycles) between each inspection would currently be required to inspect 50 percent of the tubes by the refueling outage nearest the midpoint of the inspection which would be the third refueling outage in the period, six months before the mid-point (assuming an inspection was performed at the very end of the 144 EFPM inspection period). However, since no inspection is scheduled for that outage, then the full 50 percent sample must be performed during the inspection scheduled for the second refueling outage in the period. Two inspections would be scheduled to occur in the second half of the inspection period, at 72 and 108 months into the inspection period. Thus, the current sampling requirement could be satisfied by performing a 25 percent sample during each of these inspections or other combinations of sampling (e.g., 10 percent during one and 40 percent in the other) totaling 50 percent.

Based on operational experience, the NRC staff finds there is no longer a basis to require the minimum initial sample size to vary so much from inspection to inspection. The licensee proposes to revise this requirement such that the minimum sample size for a given inspection in a given inspection period is 100 percent divided by the number of scheduled inspections during that inspection period. For the above example, the proposed change would result in a uniform initial minimum sample size of 33.3 percent for each of the three scheduled inspections during the inspection period. The NRC staff concludes this proposed revision to be an improvement to the existing requirement since it provides a more consistent minimum initial sampling requirement.

The proposed changes to paragraph 5.5.9.d.2 include two new sentences addressing the prorating of required tube sample sizes if a degradation assessment indicates the potential for a type of degradation to occur at a location not previously inspected with a technique capable of detecting this type of degradation at this location and that may satisfy the applicable tube plugging criteria. For example, new information from another similar plant becomes available indicating the potential for circumferential cracking at a specific location on the tube. Previous degradation assessments had not identified the potential for this type of degradation at this location. Thus, previous inspections of this location had not been performed with a technique capable of detecting circumferential cracks. However, now that the potential for circumferential cracking has been identified at this location, paragraph 5.5.9.d requires a method of inspection to be performed with the objective of detecting circumferential cracks which may be present at this location and that may satisfy the applicable tube plugging criteria.

Furthermore, suppose this inspection is performed for the first time during the third of four SG inspections scheduled for one of the inspection periods. Paragraph 5.5.9.d.2 currently does not specifically specify whether this location needs to be 100 percent inspected by the end of the inspection period, or whether a prorated approach may be taken. The NRC staff addressed this question in Issue 1 of NRC Regulatory Information Summary (RIS) 2009-04, "Steam Generator

Tube Inspection Requirements," dated April 3, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML083470557), as follows:

> Issue 1: A licensee may identify a new potential degradation mechanism after the first inspection in a sequential period. If this occurs, what are the expectations concerning the scope of examinations for this new potential degradation mechanism for the remainder of the period (e.g., do 100 percent of the tubes have to be inspected by the end of the period or can the sample be prorated for the remaining part of the period)?
>
> [NRC Staff Position:] The TS contain requirements that are a mixture of prescriptive and performance-based elements. Paragraph "d" of these requirements indicates that the inspection scope, inspection methods, and inspection intervals shall be sufficient to ensure that SG tube integrity is maintained until the next SG inspection. Paragraph "d" is a performance-based element because it describes the goal of the inspections but does not specify how to achieve the goal. However, paragraph "d.2" is a prescriptive element because it specifies that the licensee must inspect 100 percent of the tubes at specified periods.
>
> If an assessment of degradation performed after the first inspection in a sequential period results in a licensee concluding that a new degradation mechanism (not anticipated during the prior inspections in that period) may potentially occur, the scope of inspections in the remaining portion of the period should be sufficient to ensure SG tube integrity for the period between inspections.
>
> In addition, to satisfy the prescriptive requirements of paragraph "d.2" that the licensee must inspect 100 percent of the tubes within a specified period, a prorated sample for the remaining portion of the period is appropriate for this potentially new degradation mechanism. This prorated sample should be such that if the licensee had implemented it at the beginning of the period, the TS requirement for the 100 percent inspection in the entire period (for this degradation mechanism) would have been met. A prorated sample is appropriate because (1) the licensee would have performed the prior inspections in this sequential period consistently with the requirements, and (2) the scope of inspections must be sufficient to ensure that the licensee maintains SG tube integrity for the period between inspections.

The NRC staff finds that relocation of information in sentences 3 and 4 as described above clarifies the existing requirement consistently with the NRC staff's position from RIS 2009-04 quoted above and is, therefore, acceptable.

The proposed fifth sentence in paragraph 5.5.9.d.2 states, "Each inspection period defined below may be extended up to 3 effective full power months to include a SG inspection outage in an inspection period and the subsequent inspection period begins at the conclusion of the included SG inspection outage." Allowing extension of the inspection periods by up to an additional 3 EFPMs potentially impacts the average tube inspection sample size to be implemented during a given inspection in that period. For example, if four SG inspections are scheduled to occur within the nominal 120-EFPM period, then the minimum sample size for each of the four inspections

could average as little as 25 percent of the tube population. If a fifth inspection can be included within the period by extending the period by 3 EFPM, then the minimum sample size for each of the five inspections could average as little as 20 percent of the tube population. Since the subsequent period begins at the end of the included SG inspection outage, the proposed change does not impact the required frequency of SG inspection.

Required tube inspection sample sizes are also subject to the performance-based requirement in paragraph 5.5.9.d, which states, in part, that in addition to meeting the requirements of paragraph 5.5.9.d.1, d.2, and d.3, "the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next scheduled SG inspection." This requirement remains unchanged under the proposal. The NRC staff concludes the proposed fifth sentence involves only a relatively minor relaxation to the existing sampling requirements in paragraph 5.5.9.d.2. However, the performance-based requirements in 5.5.9.d ensure that adequate inspection sampling will be performed to ensure tube integrity is maintained. The NRC staff concludes that the proposed change is acceptable.

Finally, the first sentence of the proposed revision to paragraph 5.5.9.d.2 replaces the last sentence of the current paragraph 5.5.9.d.2. This sentence establishes the minimum allowable SG inspection frequency. This minimum inspection frequency is unchanged from the current requirement in FNP's and VEGP's technical specifications. The NRC staff finds that the wording changes in the sentence are of an editorial and clarifying nature and are not material, such that the current intent of the requirement is unchanged. Thus, the NRC staff concludes the first sentence of proposed paragraph 5.5.9.d.2 is acceptable.

3.7 Paragraph 5.5.9.d.3

The first sentence of FNP paragraph 5.5.9.d.3 currently states:

> If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less).

Proposed Change: Revise this sentence as follows:

> If crack indications are found in any SG tube, then the next inspection for each affected and potentially affected SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever results in more frequent inspections).

This change is the same for VEGP except the sentence begins, "If crack indications are found in portions of the SG tube not excluded above..." with reference to the alternate plugging criteria in VEGP TS 5.5.9.c.

The proposed change is replacing the words "for each SG" with words "for each affected and potentially affected SG." The licensee states that the existing wording can be misinterpreted. The licensee further states that the intent is that those SGs that are affected and those SGs that are potentially affected must be inspected for the degradation mechanism that caused the crack indication. However, some licensees have questioned whether the current reference to "each SG"

requires only the SGs that are affected to be inspected for the degradation mechanism. The proposed revision is intended to clarify the intent of the requirement.

Assessment: Proposed changes in paragraph 5.5.9.d.2 (Section 3.6 above) permits SG inspection intervals to extend over multiple fuel cycles for SGs with alloy 600 TT and alloy 690 TT tubing, assuming that such intervals can be implemented while ensuring tube integrity is maintained in accordance with paragraph 5.5.9.d. However, stress corrosion cracks may not become detectable by inspection until the crack depth approaches the tube plugging limit. In addition, stress corrosion cracks may exhibit high growth rates. For these reasons, once cracks have been found in any SG tube, paragraph 5.5.9.d.3 restricts the allowable interval to the next scheduled inspection to 24 EFPMs or one refueling outage (whichever is less). The intent of this requirement is that it applies to the affected SG and to any other SG which may be potentially affected by the degradation mechanism that caused the known crack(s).

For example, a root cause analysis in response to the initial finding of one or more cracks might reveal that the crack(s) are associated with a manufacturing anomaly which causes locally high residual stress which in turn caused the early initiation of cracks at the affected locations. If it can be established that the extent of condition of the manufacturing anomaly applies only to one SG and not the others, then the NRC staff agrees that only the affected SG needs to be inspected within 24 EFPMs or one refueling cycle in accordance with paragraph 5.5.9.d.3. The next scheduled inspections of the other SGs will continue to be subject to all other provisions of paragraph 5.5.9.d. The NRC staff finds the proposed change to paragraph 5.5.9.d.3 acceptable, because it clarifies the intent the paragraph.

3.8 Specification 5.6.10, "Steam Generator Tube Inspection Report"

The specification lists items to be included in a report which shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, "Steam Generator (SG) Program."

Proposed Changes: Item b., which currently reads: "Active degradation mechanisms found..." would be revised to read: "Degradation mechanisms found..."

Item e., which currently reads: "Number of tubes plugged during the inspection outage for each active degradation mechanism..." would be revised to read: "Number of tubes plugged during the inspection outage for each degradation mechanism..."

Item f., which currently reads, "Total number and percentage of tubes plugged to date..." would be revised to read: "The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator..."

Assessment: This proposal would delete the word "active" in items b. and e. above. Thus, all degradation mechanisms found, whether deemed to be active or not, would now be reportable.

TSTF-510 describes the proposed change to item f. as the combination of Westinghouse STS specification 5.6.7 items f. and h. Item h. requires, "The effective plugging percentage for all plugging [and tube repairs] in each SG..." Neither FNP nor VEGP TS 5.6.10 currently implements

item h. The proposed change to item f. would be added to the SG reporting requirements to align TS requirements with TSTF-510.

The proposed change is more conservative than the current requirements due to the insertion of additional requirements. The NRC staff finds the change acceptable.

4.0 STATE CONSULTATION

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

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendment for the FNP. 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 for the VEGP (78 FR 28254, May 14, 2013).

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 for the FNP (78 FR 28253, May 14, 2013).

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 Contributor: K. Hemphill
R. Grover

Date of issuance: September 26, 2013

September 26, 2013

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

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, FARLEY NUCLEAR PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING STEAM GENERATOR TUBE INSPECTION PROGRAM (TAC NOS. MF0564, MF0565, MF0562 AND MF0563)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 171 to Renewed Facility Operating License NPF-68 and Amendment No. 153 to Renewed Facility Operating License NPF-81 for the Vogtle Electric Generating Plant (VEGP), Units 1 and 2, and Amendment No. 192 to Renewed Facility Operating License NPF-2 and Amendment No. 188 to Renewed Facility Operating License NPF-8 for the Farley Nuclear Plant (FNP), Units 1 and 2, in response to your letters dated January 23 and July 17, 2013. The amendments revise the following Technical Specifications (TSs) for each facility: 3.4.17, "Steam Generator (SG) Tube Integrity," 5.5.9, "Steam Generator (SG) Program," and 5.6.10, "Steam Generator Tube Inspection Report." These changes summarize and clarify the purpose of the TSs in accordance with TS Task Force Traveler 510, "Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."

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/

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

Docket Nos. 50-424, 50-425, 50-348 and 50-364

Enclosures:
1. Amendment No. 171 to NPF-68
2. Amendment No. 153 to NPF-81
3. Amendment No. 192 to NPF- 2
4. Amendment No. 188 to NPF- 8
5. Safety Evaluation

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