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

August 1, 2023

Mr. R. Keith Brown
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS NOS. 220 AND 203, REGARDING USE OF ACCIDENT TOLERANT FUEL LEAD TEST ASSEMBLIES (EPID L-2022-LLA-0097)

Dear Mr. Brown:

The Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 220 to Renewed Facility Operating License NPF-68 and Amendment No. 203 to Renewed Facility Operating License NPF-81 for the Vogtle Electric Generating Plant (Vogtle), Units 1 and 2, respectively. The amendments consist of changes to the License and technical specifications (TSs) in response to your application dated June 30, 2022, as supplemented by letters dated September 13, 2022, and January 20 and May 5, 2023.

The amendments allow the use of four Accident Tolerant Fuel (ATF) Lead Test Assemblies (LTAs) to be placed in limiting core locations without completion of representative testing for up to two cycles of operation in Vogtle, Unit 2, except that the LTAs may not be placed in core regions that have been shown to be limiting with respect to the control rod ejection analysis. The proposed amendments would revise License Condition 2.D, and the following TSs: (1) TS 3.7.18, " Fuel Assembly Storage in the Fuel Storage Pool," (2) TS 4.2.1, "Fuel Assemblies," and (3) TS 4.3, "Fuel Storage," for Vogtle, Units 1 and 2.

SNC plans to install the four ATF LTAs (7ST1, 7ST2, 7ST3, and 7ST4) in Vogtle, Unit 2, for up to two cycles of operation.

The Vogtle, Units 1 and 2, ATF LTAs amendments and associated exemptions were presented to the Advisory Committee on Reactor Safeguards (ACRS) subcommittee meeting on June 21, 2023, and the ACRS full committee meeting on July 12, 2023. On July 27, 2023 (Agencywide Documents and Access Management System Accession No. ML23200A306), ACRS issued a letter report that stated, in part: "The SE [safety evaluation] report should be issued."

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

Sincerely,

/RA/

John G. Lamb, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-424 and 50-425

Enclosures:
1. Amendment No. 220 to NPF-68
2. Amendment No. 203 to NPF-81
3. Safety Evaluation

cc: Listserv

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-424

VOGTLE ELECTRIC GENERATING PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 220
Renewed License No. NPF-68

1. The Nuclear Regulatory Commission (NRC, 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 June 30, 2022, as supplemented by letters dated September 13, 2022, and January 20 and May 5, 2023, 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:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 220, 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 30 days of issuance. The accident tolerant fuel lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4 may not be installed in the Vogtle, Unit 1, reactor core.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2023.08.01 09:36:28 -04'00'

Michael T. Markley, 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: August 1, 2023

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. 203
Renewed License No. NPF-81

1. The Nuclear Regulatory Commission (NRC, 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 June 30, 2022, as supplemented by letters dated September 13, 2022, and January 20 and May 5, 2023, 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:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 203, 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 30 days of issuance. The accident tolerant fuel lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4 may be installed in the Vogtle, Unit 2, reactor core for up to two cycles of operation.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2023.08.01 09:37:00 -04'00'

Michael T. Markley, 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: August 1, 2023

ATTACHMENT

TO LICENSE AMENDMENT NO. 220

RENEWED FACILITY OPERATING LICENSE NO. NPF-68

DOCKET NO. 50-424

AND

TO LICENSE AMENDMENT NO. 203

RENEWED FACILITY OPERATING LICENSE NO. NPF-81

DOCKET NO. 50-425

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

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-68, page 5	License No. NPF-68, page 5
License No. NPF-81, page 3	License No. NPF-81, page 3
License No. NPF-81, page 4	License No. NPF-81, page 4
License No. NPF-81, page 5	License No. NPF-81, page 5
TSs	TSs
3.7.18-1	3.7.18-1
3.7.18-2	3.7.18-2
4.0-1	4.0-1
4.0-3	4.0-3
4.0-4	4.0-4
4.0-5	4.0-5

(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. 220, 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

7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

(11) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 196, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

D. The facility requires an exemption from the requirements of paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding this exemption are identified in Section 6.2.6 of SSER 5. This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 162, as supplemented by a change approved by License Amendment No. 175.

F. GPC shall comply with the antitrust conditions delineated in Appendix C to this license.

(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 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 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. 203, 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 successfully demonstrated prior to the time and condition specified below for each:

a) DELETED

b) DELETED

c) SR 3.8.1.20 shall be successfully demonstrated at the first regularly scheduled performance after implementation of this license amendment.

(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) <u>Mitigation Strategy License Condition</u>

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
   7. Spent fuel pool mitigation measures

(c) Actions to minimize release to include consideration of:
   1. Water spray scrubbing
   2. Dose to onsite responders

(5) Additional Conditions

The Additional Conditions contained in Appendix D, as revised through Amendment No. 179, are hereby incorporated into this license. Southern Nuclear shall operate the facility in accordance with the Additional Conditions.

D. The facility requires an exemption from the requirements of paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding this exemption are identified in Section 6.2.6 of SSER 8. This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan, which contains Safeguards Information protected under 10 CFR 73.21, is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.

   Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 144, as supplemented by a change approved by License Amendment No. 175.

F. GPC shall comply with the antitrust conditions delineated in Appendix C to this license.

G. Southern Nuclear shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility, as approved in the SER (NUREG-1137) through Supplement 9 subject to the following provision:

   > Southern Nuclear may make changes to the approved fire protection program without prior approval of the Commission, only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

H. Deleted.

I. The Owners shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

3.7 PLANT SYSTEMS

3.7.18 Fuel Assembly Storage in the Fuel Storage Pool

LCO 3.7.18

---NOTE---
Figures 3.7.18-1 and 3.7.18-2 do not apply to lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4.

The combination of initial enrichment burnup and configuration of fuel assemblies stored in the fuel storage pool shall be within the Acceptable Burnup Domain of Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or in accordance with Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

APPLICABILITY: Whenever any fuel assembly is stored in the fuel storage pool.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 ----------NOTE------------ LCO 3.0.3 is not applicable. ------------------------------ Initiate action to move the noncomplying fuel assembly to an acceptable storage location.	 Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.7.18.1	------------------------------NOTE----------------------------- Figures 3.7.18-1 and 3.7.18-2 do not apply to lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4. -- Verify by a combination of visual inspection and administrative means that the initial enrichment, burnup, and storage location of the fuel assembly is in accordance with Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).	Prior to storing the fuel assembly in the fuel storage pool location.

4.0 DESIGN FEATURES

4.1 Site

4.1.1 Site and Exclusion Area Boundaries (EAB)

The VEGP site and EAB consist of approximately 3,169 acres in eastern Georgia on the west side of the Savannah River about 26 miles southeast of Augusta, Georgia, and 15 miles east-northeast of Waynesboro, Georgia, in Burke County, Georgia. The nearest point to the EAB from the VEGP Reactors is the near bank of the Savannah River. Reactor 1 is approximately 3600 feet from the EAB and Reactor 2 is approximately 3900 feet from the EAB.

4.1.2 Low Population Zone (LPZ)

The LPZ is that area falling within a 2-mile radius from the midpoint between the containment buildings.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies (LTAs) that have not completed representative testing may be placed in nonlimiting core regions. In addition, LTAs 7ST1, 7ST2, 7ST3, and 7ST4, which contain fuel rods that include advanced coated cladding features, doped or standard fuel material, and up to four fuel rods with a maximum nominal U-235 enrichment of 6.0 weight percent, are permitted to be placed in limiting core regions for up to two cycles of operation without completion of representative testing. These LTAs cannot be placed in core regions that have been shown to be limiting with respect to the control rod ejection analysis.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver-indium-cadmium, or hafnium metal as approved by the NRC.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

f. A nominal 10.25 inch center to center pitch in the Unit 1 high density fuel storage racks.

g. LTAs 7ST1, 7ST2, 7ST3, and 7ST4 are prohibited from Unit 1 spent fuel pool storage.

(Unit 2) 4.3.1.2

---NOTE--
4.3.1.2a, 4.3.1.2d, and 4.3.1.2e do not apply to LTAs 7ST1, 7ST2, 7ST3, and 7ST4.

The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

b. K_{eff} < 1.0 when fully flooded with unborated water which includes an allowance for uncertainties as described in Section 4.3 of the FSAR.

c. $K_{eff} \leq 0.95$ when fully flooded with water borated to 394 ppm, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

d. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2 may be allowed unrestricted storage in the Unit 2 fuel storage pool.

e. New or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-8 may be stored in the Unit 2 fuel storage pool in a 3-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-1.

New or partially spent fuel assemblies with a maximum initial enrichment of 5.0 weight percent U-235 may be stored in the Unit 2 fuel storage pool in a 2-out-of-4 checkerboard storage configuration as shown in Figure 4.3.1-1.

New or partially spent fuel assemblies with a combination of burnup, decay time, and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-10 may be stored

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

in the Unit 2 fuel storage pool as "low enrichment" fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-2. New or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or which satisfy a minimum IFBA requirement as shown in Figure 4.3.1-9 for higher initial enrichments may be stored in the Unit 2 fuel storage pool as "high enrichment" fuel assemblies in the 3x3 checkerboard storage configuration as shown in Figure 4.3.1-2.

Interfaces between storage configurations in the Unit 2 fuel storage pool shall be in compliance with Figures 4.3.1-3, 4.3.1-4, 4.3.1-5, and 4.3.1-6. "A" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 3.7.18-2. "B" assemblies are new or partially spent fuel assemblies with a combination of burnup and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-8. "C" assemblies are assemblies with initial enrichments up to a maximum of 5.0 weight percent U-235. "L" assemblies are new or partially spent fuel assemblies with a combination of burnup, decay time, and initial nominal enrichment in the "acceptable burnup domain" of Figure 4.3.1-10. "H" assemblies are new or partially spent fuel assemblies with initial nominal enrichments less than or equal to 3.20 weight percent U-235 or which satisfy a minimum IFBA requirement as shown in Figure 4.3.1-9 for higher initial enrichments.

f. A nominal 10.58-inch center to center pitch in the north-south direction and a nominal 10.4-inch center to center pitch in the east-west direction in the Unit 2 high density fuel storage racks.

g. For LTAs 7ST1, 7ST2, 7ST3, and 7ST4, the following requirements apply for storage in the Unit 2 spent fuel storage racks:
   1. Unrestricted storage is allowed in the 2-out-of-4 checkerboard storage configuration as shown in TS Figure 4.3.1-1.
   2. Storage is allowed in the all-cell storage configuration (“A” assemblies as shown on TS Figures 4.3.1-3 and 4.3.1-5) when the LTAs reach 64,000 MWd/MTU of burnup.

(continued)

4.0 DESIGN FEATURES

4.3 Fuel Storage (continued)

4.3.1.3 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent except for LTAs 7ST1, 7ST2, 7ST3, and 7ST4 which may have four rods per assembly enriched up to 6.0 weight percent U-235;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR;

c. $k_{eff} \leq 0.98$ if moderated by aqueous foam, which includes an allowance for uncertainties as described in Section 4.3 of the FSAR; and

d. A nominal 21-inch center to center distance between fuel assemblies placed in the storage racks.

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 194 foot-1 1/2 inch.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1476 fuel assemblies in the Unit 1 storage pool and no more than 2098 fuel assemblies in the Unit 2 storage pool.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 220 TO RENEWED FACILITY OPERATING LICENSE NPF-68

AND

AMENDMENT NO. 203 TO RENEWED FACILITY OPERATING LICENSE NPF-81

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

DOCKET NOS. 50-424 AND 50-425

1.0 INTRODUCTION

By letter dated June 30, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22181B156 (public) and ML22181B155 (proprietary)), as supplemented by letters dated September 13, 2022 (ML22256A198 (public) and ML22256A197 (proprietary)), and January 20 (ML23020A148 (public) and ML23020A147 (proprietary)) and May 5, 2023 (ML23125A269), Southern Nuclear Operating Company, Inc. (SNC, the licensee), requested changes to the technical specifications (TSs) for the Vogtle Electric Generating Plant (Vogtle), Units 1 and 2.

The proposed amendments would allow the use of four Accident Tolerant Fuel (ATF) Lead Test Assemblies (LTAs) to be placed in limiting core locations without completion of representative testing for up to two cycles of operation in Vogtle, Unit 2, except that the LTAs may not be placed in core regions that have been shown to be limiting with respect to the control rod ejection analysis. The proposed amendments would revise License Condition 2.D and the following TSs: (1) TS 3.7.18, " Fuel Assembly Storage in the Fuel Storage Pool," (2) TS 4.2.1, "Fuel Assemblies," and (3) TS 4.3, "Fuel Storage," for Vogtle, Units 1 and 2. SNC plans to install the four ATF LTAs in Vogtle, Unit 2, for up to two cycles of operation.

In addition, SNC is requesting three exemptions. The first and second exemptions are to Title 10 of the *Code of Federal Regulations* (10 CFR), part 50, Section 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," and 10 CFR part 50, Appendix K, "ECCS [emergency core cooling systems] Evaluation Models." The exemptions to 10 CFR 50.46 and 10 CFR part 50, Appendix K would allow the use of coated AXIOM® cladding. The third exemption is from 10 CFR 50.68, "Criticality accident requirements," paragraph (b)(7) to allow greater than 5 weight-percent U-235.

The exemptions to 10 CFR 50.46, 10 CFR part 50, Appendix K, and 10 CFR 50.68 are evaluated in the following ADAMS packages: ML23093A148, ML23096A206, and ML23094A051, respectively.

The U.S. Nuclear Regulatory Commission (NRC) staff identified the need for a regulatory audit to examine the SNC's non-docketed information with the intent to gain understanding, to verify information, or to identify information that will require docketing to support the basis of the licensing or regulatory decision.

By letter dated October 7, 2022 (ML22103A253), NRC issued an audit plan, which provided the list of requested documents and other details pertaining to the audit. By letter dated March 10, 2023, (ML23059A456), the NRC staff issued the Audit Summary.

The supplements dated January 20, and May 5, 2023, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 8, 2022 (87 FR 67508).

2.0 REGULATORY EVALUATION

2.1 Proposed Changes

By letter dated June 30, 2022, as supplemented by letters dated September 13, 2022, January 20, 2023, and May 5, 2023, SNC submitted the proposed changes to RFOL Condition 2.D and TS 3.7.18, "Fuel Assembly Storage in the Fuel Storage Pool," TS 4.2, "Reactor Core," and TS 4.3, "Fuel Storage." The proposed markups are contained in Enclosure 2 of the letters dated June 30, and September 13, 2022. See Sections 3.15 and 3.16 below for the details.

2.2 Regulations and Guidance Considered

The NRC staff considered the following regulatory requirements and guidance in its review of the licensee's application.

The regulation 10 CFR Part 100, "Reactor Site Criteria; Determination of exclusion area: low population zone and population center distance," paragraph (a)(1) of 10 CFR 100.11 states:

> An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem[2] [roentgen equivalent man] or a total radiation dose in excess of 300 rem[2] to the thyroid from iodine exposure.
>
> Footnote 2: The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP [National Council on Radiation Protection and Measurements] recommendations may be disregarded in the determination of their radiation exposure status (see NBS [National Bureau of Standards (now National Institute of Standards and Technology)] Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

Paragraph (a)(2) of 10 CFR 100.11 states:

> A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

The regulation 10 CFR 50.36, "Technical specifications," contain the requirements for the content of TSs. The regulations in 10 CFR 50.36(b) require TSs to be derived from the analyses and evaluations included in the safety analysis report and amendments thereto. In accordance with 10 CFR 50.36(c)(2), "Limiting conditions for operations [LCOs]," the LCOs are the lowest functional capability or performance levels of equipment required for safe operation of the facility. In accordance with 10 CFR 50.36(c)(3), "Surveillance requirements [SRs]," the SRs are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met. The regulation in 10 CFR 50.36(c)(4), "Design features," requires that TSs include design features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (c)(1), (2), and (3) of 10 CFR 50.36.

The regulation 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," section (a)(1)(i) states that:

> Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated. Except as provided in paragraph (a)(1)(ii) of this section, the evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model. This section does not apply to a nuclear power reactor facility for which the certifications required under § 50.82(a)(1) have been submitted.

The regulation 10 CFR 50.46, section (a)(1)(ii) states that:

> Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of appendix K ECCS Evaluation Models.

The specific conditions referenced in paragraph (b)(1) of 10 CFR 50.46 state:

(1) *Peak cladding temperature*. The calculated maximum fuel element cladding temperature shall not exceed 2200° F [Fahrenheit].

(2) *Maximum cladding oxidation*. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

(3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) *Coolable geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) *Long-term cooling*. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The regulation 10 CFR 50.68, “Criticality accident requirements,” states, in part, that:

> (a) Each holder of a construction permit or operating license for a nuclear power reactor issued under this part or a combined license for a nuclear power reactor issued under Part 52 of this chapter, shall comply with either 10 CFR 70.24 of this chapter or the requirements in paragraph (b) of this section.

(b) Each licensee shall comply with the following requirements in lieu of maintaining a monitoring system capable of detecting a criticality as described in 10 CFR 70.24:

(1) Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

(2) The estimated ratio of neutron production to neutron absorption and leakage (k-effective) of the fresh fuel in the fresh fuel storage racks shall be calculated assuming the racks are loaded with fuel of the maximum fuel assembly reactivity and flooded with unborated water and must not exceed 0.95, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such flooding or if fresh fuel storage racks are not used.

(3) If optimum moderation of fresh fuel in the fresh fuel storage racks occurs when the racks are assumed to be loaded with fuel of the maximum fuel assembly reactivity and filled with low-density hydrogenous fluid, the k-effective corresponding to this optimum moderation must not exceed 0.98, at a 95 percent probability, 95 percent confidence level. This evaluation need not be performed if administrative controls and/or design features prevent such moderation or if fresh fuel storage racks are not used.

(4) If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water.

(5) The quantity of SNM [special nuclear material], other than nuclear fuel stored onsite, is less than the quantity necessary for a critical mass.

(6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.

(7) The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight.

(8) The FSAR is amended no later than the next update which § 50.71(e) of this part requires, indicating that the licensee has chosen to comply with § 50.68(b).

The regulation 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants [GDC],” Criterion 10, “Reactor design,” states that:

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC 11, *Criterion 11, - Reactor inherent protection*, states that:

> The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

GDC 12, *Criterion 12* - *Suppression of reactor power oscillations*, states that:

> The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

GDC 19, *Criterion 19 - Control room*, states that:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

GDC 27*, Criterion 27 - Combined reactivity control systems capability*, states that:

> The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

GDC 28, *Criterion 28 -Reactivity limits*, states that:

> The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals

> to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

GDC 35,*Criterion 35- Emergency core cooling,* states that:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

GDC 61, *Criterion 61 - Fuel storage and handling and radioactivity control*, states:

> The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

GDC 62, *Criterion 62 - Prevention of criticality in fuel storage and handling,* states that:

> Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The NRC staff also considered the guidance in NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [light-water reactor] Edition,” Section 4.2, Revision 3, “Fuel System Design” (ML070740002).

Regulatory Guide (RG) 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment [PRA] Results for Risk-Informed Activities,” March 2009 (ML090410014), describes an approach acceptable to the NRC for determining whether the quality of the PRA, in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for light-water reactors. This RG 1.200, Revision 2, provides guidance for assessing the technical adequacy of a PRA.

RG 1.240, Revision 0, “Fresh and Spent Fuel Pool Criticality Analyses” (ML20356A127), describes an approach that the NRC staff considers acceptable to demonstrate that NRC regulatory requirements are met for subcriticality of fuel assemblies stored in fresh fuel vaults and spent fuel pools at light-water reactor (LWR) power plants. It endorses, with clarifications and exceptions, the Nuclear Energy Institute (NEI) guidance document NEI 12-16, “Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants,” Revision 4, (ML19269E069).

RG 1.195, Revision 0, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors” (ML031490640), “Re-Analysis Guidance,” provides guidance to licensees of operating power reactors on acceptable methods and assumptions for performing evaluations of fission product releases and radiological consequences of several postulated light-water reactor design basis accidents. With respect to this LAR, RG 1.195, Section 1.3.2 describes when re-analysis of a design basis radiological analysis is necessary. Specifically, it states: “An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid.”

The analyses and evaluations required to demonstrate compliance with the fundamental acceptance criteria of 10 CFR 50.34 and GDC 19 for an operating license are documented in the facility’s UFSAR. The core radiological inventory and consequence analyses in the Vogtle, Units 1 and 2, current licensing basis (CLB) were last reviewed by the NRC as part of a measurement uncertainty recapture power uprate (MUR-PU) license amendment (ML080350345).

3.0 TECHNICAL EVALUATION

3.1 Background

The reactors for Vogtle, Units 1 and 2, each contain 193 fuel assemblies. Each assembly consists of a matrix of 264 Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material, not to exceed 5 weight-percent enrichment. The proposed change is to load four LTAs with advanced ATF features, including Advanced Doped Pellet Technology (ADOPT) fuel (ML20132A015), AXIOM cladding (ML21090A110), chromium coating, and four rods per LTA with up to 6 weight-percent enrichment U-235, in limiting core locations for up to two cycles of operation, except that the LTAs may not be placed in core regions that have been shown to be limiting with respect to the control rod ejection analysis.

3.2 Current Licensing Basis

Vogtle, Units 1 and 2, TS 3.7.18, “Fuel Assembly Storage in the Fuel Storage Pool,” has a limiting condition for operation (LCO) that states:

> The combination of initial enrichment burnup and configuration of fuel assemblies stored in the fuel storage pool shall be within the Acceptable Burnup Domain of Figures 3.7.18-1 (Unit 1), 3.7.18-2 (Unit 2), or in accordance with Specification 4.3.1.1 (Unit 1) or 4.3.1.2 (Unit 2).

Vogtle, Units 1 and 2, TS have an existing provision in TS 4.2.1 describing the fuel assemblies that may be loaded into the reactor core, including LTAs. Vogtle, Units 1 and 2, TS 4.2.1, “Fuel

Assemblies," states that:

> The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide ($U0_2$) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

TS 4.3 "Fuel Storage," provides requirements for new and spent fuel storage.

3.3 Description of Lead Test Assemblies (LTAs)

The LTAs are Westinghouse 17 X 17 PRIME™ Optimized Fuel Assembly designs (ML22059B071) and each contains:

- Up to 132 rods with Westinghouse ADOPT uranium dioxide pellets having a maximum of 5 weight-percent U-235 [Uranium 235] enrichment and coated AXIOM cladding.

- Three rods with Westinghouse ADOPT uranium dioxide pellets having a maximum of 6 weight-percent U-235 enrichment and coated AXIOM cladding.

- One rod with Westinghouse ADOPT uranium dioxide pellets having a maximum of 6 weight-percent U-235 enrichment and uncoated AXIOM cladding.

- All other rods will have Westinghouse uranium dioxide pellets having a maximum of 5 weight-percent U-235 enrichment, Zirconium Diboride (ZrB2) Integral Fuel Burnable Absorber (IFBA) coated pellets and coated AXIOM cladding.

The cladding coating will consist of chromium (Cr) applied to the outer surface of the AXIOM cladding. There are no other changes to the existing fuel assembly design.

The four LTAs will include 16 rods (four rods per LTA) with initial enrichment of up to 6 weight-percent.

3.4 Reactor Coolant System (RCS) Chemistry

Increases in Reactor Coolant System (RCS) activity, caused by fuel oxidation from RCS water getting into the fuel rod during normal operation, is detected and monitored by existing plant equipment in accordance with approved procedures. The chromium (Cr) coating on the LTA rods requires consideration of increased Cr-51 isotope generation. The formation and possible release of Cr-51 is an issue that is already monitored through chemistry surveillance procedures at the plant. The impact of fast neutron irradiation on Cr mechanical properties from previous research detailed in ATF-ISG-2020-01, "Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept – Interim Staff Guidance," (ML19343A121) was considered and is not expected to challenge plant chemistry systems during the LTA utilization, at which time more data on Cr-51 generation from the LTAs themselves can be considered. Because of the limited number of LTAs, even if increased Cr-51 is observed during the LTA implementation, existing plant systems would be expected to reduce this radioisotope from effluents and radiation protection programs would address any new occupational exposure issues.

NRC staff concluded that the existing plant radiochemistry systems and processes and radiation protection programs are adequate to deal with anticipated RCS chemistry effects of LTA implementation. Systems and procedures are in place to monitor and reduce system Cr-51 from the plant, as necessary.

3.5 Core Physics

The licensee, SNC, uses WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON" (ML030760104), for the lattice physics calculations and modeling. PARAGON does not have generic approval for modeling fuel with enrichments greater than 5 wt% U-235. However, PARAGON, in part, was used to validate WCAP-18443, "Qualification of the Two-Dimensional Transport Code PARAGON2" (ML19308C031), which does have generic approval for modeling fuel with enrichments greater than 5 wt% U-235. The code-to-code comparisons between PARAGON and PARAGON2 showed sufficient agreement in the shared range of applicability. Differences in the neutronic characteristics of the LTAs are expected to be limited to the vicinity of the LTAs such that PARAGON performance will be unaffected when modeling co-resident fuel. Additionally, the LTA average enrichment is still below 5 wt% U-235. Therefore, the NRC staff expects PARAGON to adequately model the LTAs which contain only four higher enriched fuel rods per assembly. Any differences between the co-resident fuel and the LTAs are modeled explicitly, namely ADOPT pellets, chromium coating, and increased enrichment. The licensee determined that the neutronically significant features of ADOPT fuel pellets and chromium coating are explicitly modeled but the additives have a negligible neutronic impact. The licensee determined that all parameters associated with the fuel pellets and rods are modeled conservatively and that no core physics impacts are anticipated. SNC has also determined that there is no change to the standard overall nuclear design process in terms of incore fuel management, safety analyses, or evaluation of operational data.

Increased enrichment is known to have an effect on the neutron flux spectrum. An increase in U-235 content will cause spectral hardening, reducing the proportion of thermal neutrons to non-thermal neutrons. Only 16 out of 50,952 rods will have a fuel enrichment greater than 5 wt%, thus the average fuel enrichment is essentially unchanged. The core-wide neutron flux spectrum will continue to be dominated by the co-resident fuel. There may be some local impact to the flux spectrum, but those impacts will be confined to the LTA.

Differences in the flux spectrum can affect detector functionality. However, as described above, the neutronic impact of increased enrichment is confined to the LTA; therefore, there is no expected decrease in detector functionality caused by the LTAs. The licensee stated that "The small number of LTA fuel rods with enrichment above 5 wt% are placed such as to have a negligible effect on the incore flux detectors. Technical Specification Surveillance Requirements are not impacted, and design basis peaking factor limits will be met at all times." Therefore, the detectors will continue to adequately monitor the core and provide reliable neutronic information despite the presence on the LTAs.

The LTAs will experience higher peaking factors compared to co-resident fuel because the LTAs will be located in limiting locations, except for core regions that have been shown to be limiting with respect to the control rod ejection analysis. The peaking factors in the analysis of record, as described in the updated final safety analysis report (UFSAR), will continue to bound the peaking factors for the LTAs.

The NRC staff has determined that the licensee's evaluation of the core physics and nuclear design is acceptable because any effects caused by the LTA are confined to the LTA; therefore, the core performance will be dominated by the co-resident fuel. Additionally, any local effects within the LTA are modeled conservatively.

3.6 Loss-of-Coolant Accidents (LOCAs) and Steam Line Break (SLB) Mass and Energy Release

The licensee, SNC, evaluated the effects of the LTAs on the short- and long-term loss of coolant accidents (LOCA) and steam line break (SLB) mass and energy (M&E) release analyses of record (AOR). In its letter dated June 30, 2022, the licensee stated that M&E releases can be affected by the following parameters:

- Fuel dimensions (rod outer diameter)
- Pressure drops through the core
- Core Stored Energy
- Changes in Decay Heat
- Initial RCS Temperature and Pressure Conditions
- Initial Steam Generator Temperature and Pressure Conditions
- Break Location and Break Area

In its letter dated June 30, 2022, the licensee stated that:

> The short-term LOCA M&E releases are most impacted by changes in the initial RCS pressure and temperature conditions, the break location and break area. None of these parameters are changing for the LTA program, so there is no impact on short-term LOCA M&E releases for the addition of four LTAs to the core.

The long-term LOCA M&E AOR was reperformed for the updated LTA conditions and compared with the existing AOR results. The licensee also stated that:

> None of the fuel dimensions were impacted by the LTAs and the overall core pressure drop change due to four LTAs was determined to be negligible. The core stored energy in the analysis of record was determined to be bounding for the core with the four LTAs. Finally, the decay heat curve used in the analysis of record was determined to be bounding for the four higher enriched rods present in the LTAs.

The SLB M&E AOR was reperformed for the updated LTA conditions and compared with the existing AOR results. The licensee stated that:

> The SLB M&E release analyses model core-average parameters such as fuel heat transfer characteristics (UAs), decay heat, initial stored energy, and reactivity feedback. However, based on the total number of the fuel rods to be

> inserted into the core (four LTAs of 193 total fuel assemblies and up to 16 rods enriched to 6 wt% out of 50,952 total fuel rods), the impact on core average effects such as fuel UAs, decay heat, initial core stored energy, and reactivity feedback are judged to be negligible. Therefore, the analysis of record for the SLB M&E releases inside and outside containment remain bounding and applicable with the addition of the four LTAs.

The licensee also stated that:

> Because the LOCA and SLB M&E releases are not impacted by the LTA program, the downstream containment and compartment response analyses are also not impacted.”

The NRC staff determined that the results of SNC’s supporting analyses for the implementation of the Vogtle LTA program on the LOCA and SLB M&E release are sufficient. The NRC agrees that the impact of 16 slightly enriched rods on a core of 50,952 active fuel rods will not have a significant impact on the overall core fuel dimensions, pressure drops, stored energy, decay heat, initial operating temperature and pressure conditions, or break sizes and locations that could impact the LOCA and SLB AORs. The quantitative analysis of the impact of the small number of rods remains bounding within the existing AORs and maintains conservative assumptions in the analyses. Based on the above, the existing LOCA and SLB AORs can be reasonably maintained with the addition of the LTAs.

3.6.1 Small and Large Break Loss-of-Coolant Accident (LOCAs)

Vogtle uses two separate vendor (Westinghouse) evaluation model (EM) computer codes to evaluate large and small break LOCAs respectively. These models meet the requirements of Appendix K to 10 CFR Part 50 and have been reviewed and approved previously for use at Vogtle by the NRC staff.

In its letter dated June 30, 2022, the licensee stated:

> To support insertion of the four LTAs, Large Break LOCA and Small Break LOCA evaluations were performed to justify safe operation of the LTAs and estimate the impact of the LTAs on co-resident fuel .
>
> [The licensee’s vendor]…. reviewed the ….EM to assess the impact of the LTA features and determined that the approved codes and methods are adequate to evaluate the LTAs without any modification. The Large Break and Small Break LOCA evaluations demonstrate that the acceptance criteria for LOCAs, given in 10 CFR 50.46, continue to be met.

A limitation and condition in the NRC approval of the Large Break LOCA EM indicates a use constraint for changes that would be expected to significantly exacerbate downcomer boiling. The licensee states that:

> Downcomer boiling is not exacerbated since the core-wide thermal-hydraulic response is negligibly impacted by the four LTAs.

The NRC staff determined that the results of SNC’s supporting analyses for the implementation of the Vogtle LTA program on Small Break and Large Break LOCA EMs are sufficient. The

Large Break and Small Break LOCA evaluations have considered the impact of installing four LTAs at Vogtle and operating the LTAs up to the current licensed burnup limit. NRC staff concludes that the existing LOCA EM AOR for Vogtle is bounding for the LTAs, and the presence of the LTAs will have a negligible impact on the co-resident fuel, and 10 CFR 50.46 acceptance criteria would continue to be met.

3.7 Non-LOCA Transients

While evaluating non-LOCA transients for the addition of LTAs, SNC considered two categories: events which are dependent on core-average effects and those that are impacted by local effects on the fuel.

In its letter dated June 30, 2022, the licensee stated:

> For the first category, events are analyzed to address gross plant criteria, such as loss of shutdown margin, margin to hot leg saturation, overpressurization of the reactor coolant system, overpressurization of the secondary system, or overfilling of the pressurizer. Based on the total number of the fuel rods to be inserted into the core (four LTAs of 193 total fuel assemblies and up to 16 higher enrichment rods out of 50,952 total fuel rods), the impact on core-average effects such as core-average fuel heat transfer characteristics, decay heat, and initial core stored energy were evaluated and determined to be negligible. Any small effects caused by the LTAs would be more than offset by existing margins in the safety analyses. As such, the LTAs do not impact the core-average events.
>
> Events in the second category are potentially impacted by local effects in the fuel rods and could be affected more significantly by the LTAs. These events include:
>
> - Zero and full power steamline breaks – core response cases
> - Locked rotor
> - Loss of reactor coolant flow (complete and partial)
> - RCCA withdrawal from subcritical
> - Rod ejection

The NRC notes that this removed consideration of loss of shutdown margin to hot leg saturation, over-pressurization of the RCS, over-pressurization of the secondary system, and overfilling of the pressurizer.

In its letter dated June 30, 2022, the licensee stated:

> Westinghouse completed an evaluation to address the potential effects of the LTAs and concluded the following:
>
> - The LTAs have no impact on the current, approved non-LOCA computer codes, methodology, or relevant acceptance criteria for each event.
> - LTA geometry, material properties, and reactivity feedback characteristics were confirmed to have no impact on the non-LOCA safety analyses. Any small effects caused by differences in the geometry, material properties, and/or reactivity feedback characteristics of the LTAs are more than offset by existing margins in the safety analyses.

- While the LTAs may lead the core, they will be placed in core locations that have been shown to be non-limiting with respect to the rod ejection analysis.
- Event-specific statepoints used as input to departure from nucleate boiling ratio (DNBR) calculations are not impacted by the LTAs.
- The relevant fuel-specific acceptance criteria continue to be met with consideration for the LTAs for events concerned with local effects: minimum DNBR (loss of flow events and RCCA withdrawal from subcritical), percent of rods in DNB and peak clad temperature (locked rotor), and peak fuel enthalpy (rod ejection).

> In summary, the LTAs have been evaluated against the non-LOCA safety analyses and were determined to be acceptable. All acceptance criteria are met and the conclusions documented in the applicable UFSAR sections remain valid.

WCAP-18482-P-A, "Westinghouse Advanced Doped Pellet Technology (ADOPT) Fuel," contains the following condition:

> Licensees must demonstrate that CRE [control rod ejection] models, methods, and acceptance criteria are applicable to fuel designs containing ADOPT fuel pellets and capture all relevant fuel burnup and cladding corrosion related phenomena.

The LAR does not address this condition as the LAR was submitted concurrent with the review of WCAP-18482-P. SNC asserted, and the NRC staff has determined, that the above condition does not need to be addressed for the following reasons. The licensee is not adopting WCAP-18482-P-A in its entirety. Only the four LTAs will contain ADOPT fuel pellets. The LTAs are placed in positions non-limiting with respect to CRE and there is significant margin in rod worth for the LTAs. The LTAs do not have a significant effect on the local conditions of the limiting assemblies for CRE. Effects on the global core parameters, which the LTAs would be included in, caused by CRE behave similarly as category one events, i.e., events that are dependent on core-average effects, as described above. The NRC staff has determined that this assessment appropriately addresses the WCAP-18482-P-A condition described above by justifying that it is not necessary to demonstrate applicability of the methodology due to the limited impact of the LTAs on CRE.

The NRC staff determined that the results of SNC's supporting analyses for the implementation of the Vogtle LTA program on non-LOCA transient design basis accidents are adequate. The impact of 4 rods enriched to a maximum of 6 %wt U-235 on a fuel assembly of 264 totals rods as well as the impact of 16 enriched rods on a core of 50,952 active fuel rods will not have a significant impact on the overall assembly and core fuel parameters and when averaged out are less than the bounding parameter values in the respective AORs. For non-LOCA DBA events which can be subject to more localized fuel effects, the applicable NRC-approved computer codes are not affected; LTA geometry, material properties, and reactivity feedback characteristics are bounded by the existing AOR margins; and relevant fuel-specific acceptance criteria continue to be met so long as LTAs are placed in the analyzed non-limiting core locations. In all cases bounding assumptions of the existing AORs ensure continued conservatism. Based on the above, the existing non-LOCA transient analyses would be reasonably maintained.

3.8 Thermal-Hydraulic

The codes and methods used by SNC for thermal-hydraulic analyses are largely unaffected by the technologies used in the LTAs. Increased enrichment and ADOPT fuel pellets do not alter any inputs into these methodologies with the exception of increased peaking factors. The peaking factors assumed by the UFSAR continue to bound the higher peaking factors in the LTAs. There are no geometric parameters or characteristics that differ with the inclusion of AXIOM cladding. The chromium coating will only slightly increase the outer diameter of the fuel rod; therefore, there is no significant reduction in flow area. Coated and uncoated cladding have been shown to perform similarly with respect departure from nucleate boiling (DNB) performance.

In its letter dated June 30, 2022, the licensee stated:

> As the LTAs are designed to have a lower power peaking factor than that used in the T/H analysis of record for the UFSAR, sufficient DNBR margin is available to offset the potential mixed core penalty on the LTAs.

Because the LTAs do not present any adverse effects in DNB performance, the licensee is not implementing any changes to codes and methods used in thermal-hydraulic analyses.

The NRC staff has determined that SNC's thermal-hydraulic analysis is acceptable because the codes and methods remain applicable, as well as their acceptance criteria, and there is no significant loss of margin associated with the LTAs. The peaking factors assumed by the UFSAR remain conservative.

3.9 Fuel Rod Design

As stated in Section 3.2, "Description of LTAs," of this safety evaluation (SE), the LTAs will contain four significant design changes relative to the neighboring fuel assemblies. The design changes and their effects on fuel rod design are described below. The assembly design used for the LTAs is consistent with the 17x17 PRIME Optimized Fuel Assembly design.

3.9.1 AXIOM Cladding

AXIOM cladding is a zirconium alloy similar to the ZIRLO® and Optimized ZIRLO™ alloy but with additional alloying elements to improve specific properties. AXIOM cladding has improved corrosion resistance, lower hydrogen pickup, and lower creep growth compared to current Westinghouse cladding designs. WCAP-18546-P/NP, "Westinghouse AXIOM cladding for Use in Pressurized Water Reactor Fuel" (ML21090A110) was reviewed and approved by the NRC concurrent with this LAR.

There are no limitations and conditions within WCAP-18456-P/NP-A (ML23089A065 and ML23089A066) that conflict with the requests made in this LAR. The NRC staff has determined that the use of AXIOM cladding in this application is acceptable because there is no expected loss of safety margin associated with the use of AXIOM cladding in pertinent limiting core locations.[1]

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[1] As stated in the May 5, 2023, supplement, the LTAs will not be placed in core regions that have been shown to be limiting with respect to the control rod ejection analysis.

3.9.2 Chromium Coating

The chromium coating is a thin layer of chromium that will be applied over the AXIOM cladding substrate.

In its letter dated June 30, 2022, the licensee states that:

> Fuel rod Cr-coating also provides improved corrosion resistance to the cladding; however, no corrosion benefits are taken for the fuel performance evaluations of the Vogtle LTA program. For the LTA program, the fuel rod Cr-coating is assumed to have the same material properties and behaviors as the substrate material (AXIOM cladding), with no credit taken for additional corrosion benefits.

The licensee's coating is modeled as a small increase in outer diameter of the AXIOM cladding substrate. This approach to modeling is acceptable because, the licensee's analysis indicates the chromium coating is expected to improve overall cladding performance. Therefore, there is no loss of safety margin expected by modeling the coating as the substrate cladding with the exception of cladding emissivity, as described below.

The chromium coating surface remains shinier than typical cladding designs due to the chromium's higher corrosion resistance.

In its letter dated September 13, 2022, the licensee stated that:

> In general, shinier surfaces have lower emissivity and therefore lower radiative heat transfer. As chromium coatings resist oxidation and retain their surface appearance, it is likely that the coating will negatively impact cladding temperature for transients where radiation to steam is the dominant mode of heat transfer.

When radiative heat transfer is the dominant mode of heat transfer, such as radiation to steam, there is potential for an increase in PCT when compared to uncoated cladding.

SNC's analysis states that in conditions, such as LOCA, where radiative heat transfer is the dominant mode of heat transfer, the outer surface of the coated cladding will still oxidize, resulting in an increase in cladding emissivity. The licensee concludes that there is no need to explicitly model the coated cladding beyond an increase in the outer diameter of the substrate.

Coated cladding oxidation is expected under LOCA conditions, but this does not preclude an increase in PCT compared to uncoated rods. The time required during a LOCA for sufficient coated cladding oxidation, such that there is no increase in the resulting PCT is not apparent. If there is sufficient oxidation before the time of PCT, then there may be a substantial increase in PCT.

SNC provided data demonstrating that the chromium coating will oxidize during LOCA conditions such that the emissivity will increase to a point where there will be no significant increase in PCT. The data provided supports the licensee's original conclusions related to cladding emissivity during a LOCA.

The NRC staff has determined that SNC's evaluation of coated cladding emissivity and its effect on LOCA PCT is sufficient, because data was provided demonstrating no significant difference in PCT.

3.9.3 ADOPT Fuel Pellets

In its letter dated June 30, 2022, the licensee stated "ADOPT fuel is a modified uranium dioxide UO_2 fuel pellet doped with small amounts of chromia (Cr_2O_3) [chromium oxide] and alumina (Al_2O_3) [aluminum dioxide]. The additives facilitate greater densification and diffusion during sintering, resulting in a higher density and an enlarged grain size as compared to undoped UO_2." ADOPT pellets have a higher density, thus more U-235, than non-ADOPT fuel pellets. This could result in the fuel rod being more sensitive to reactivity-initiated accidents, such as a rod ejection accident (REA). SNC has stated that the LTAs will not be placed in positions limiting with respect to a REA. Otherwise, ADOPT fuel pellets are expected to improve fuel performance with no significant reduction in safety margin.

The NRC staff has determined that SNC's evaluation of ADOPT fuel pellets in the LTAs is acceptable, because there will be no apparent reduction in fuel performance and the applicable limitations and conditions have been addressed as described in Section 3.7, "Non-LOCA Events," of this SE.

3.9.4 Increased Enrichment

Each LTA will contain four fuel rods enriched up to a maximum of 6 wt% U-235, for a total 16 fuel rods with increased enrichment. Many of the models and codes used to analyze the LTAs have been either approved or contain data and models for enrichments greater than 5 wt% U-235. Increased enrichment is known to have a significant effect on the neutron flux, namely a reduction in thermal flux due to spectral hardening and a lower overall flux. However, no significant changes in the neutron flux spectrum throughout the core are expected due to the limited number of fuel rods with enrichments exceeding 5 wt% U-235. Any changes in the flux spectrum would be local to the LTAs and any changes are expected to be minimal. Therefore, the NRC staff has determined that the use of four fuel rods with enrichments up to 6 wt% U-235 per LTA is acceptable.

3.10 Fuel Handling and Storage

There is no comprehensive, NRC-approved generic methodology for performing nuclear criticality safety (NCS) analyses for fuel storage and handling. SNC did not perform a typical standalone criticality safety analysis (CSA) for the LTAs. Instead, SNC relied upon a combination of analysis and engineering judgement to evaluate the reactivity impact of the LTA and the reactivity margin present in the prescribed storage limitations on the LTAs, while utilizing the new fuel storage racks (NFSR) AOR (ML20244D565) and the Vogtle Spent Fuel Pool (SFP) AOR (ML042320397 for Vogtle, Unit 1), and (ML042320413 for Vogtle, Unit 2) as a baseline.

3.10.1 Lead Test Assembly Description

The LTAs are described in the licensee's letter dated June 30, 2022:

- 260 ~4.95 wt% U-235 enriched fuel rods (uncertainty to 5 wt%)
- Four ~5.95 wt% U-235 enriched fuel rods (assumed at 6 wt% nominal)

- 128 Integral Fuel Burnable Absorbers (IFBA) rods with 1.5X standard loading (^{10}B reduced 5% were applicable in SFP models only)[2] and 8" cutback region (IFBA modeled as full length in depletion analysis)
- IFBA rods contain annular blankets but are modeled as solid rods.
- Non-IFBA rods contain ADOPT doped pellets: ADOPT doped pellets can be bounded by a maximum fuel percent of Theoretical Density (TD) of UO2 of 98.3% and are modeled as such without dopants.
- All rods except one higher enriched rod are Chromium coated AXIOM cladding, while the additional rod is uncoated AXIOM cladding. SFP and NFSR models contain uncoated zircaloy-4 (neutronically equivalent to AXIOM cladding) or AXIOM cladding. Depletion analysis input applied a 10 μm chromium coating.
- 16 Wet Annular Burnable Absorber (WABA) rodlets modeled conservatively as full length in the depletion analysis to 24 GWd/MTU.
- Non-reactive assembly structures like mixing and spacer grids, sleeves, and top/bottom nozzles are not modeled.

SNC needs an exemption to Paragraph 50.68(b)(7) for the fuel rods enriched above 5.0 wt% U-235. Each LTA is limited to four rods at 6.0 wt% U-235. Those four rods add about 0.015% more U-235 to the LTAs relative to the Vogtle NFSR AOR and the Vogtle SFP AOR maximum enrichment of 5.0 wt% U-235.

The increased TD in the ADOPT pellets adds much more U^{235} to the fuel assemblies. Vogtle NFSR AOR used 96% TD. The Vogtle SFP AOR used 97.5% TD. The ADOPT pellets are limited to 136 rods per LTA. The increased TD adds about 1.185% more U-235 to the LTAs relative to Vogtle's NFSR AOR and about 0.412% more U-235 to the LTAs relative to Vogtle's Vogtle SFP AOR.

The total increase in U-235 for the Vogtle NFSR and SFP relative to the AORs is 1.2% and 0.427%, respectively. The relative increase in U-235 content is determined to aid in evaluating the potential impact on reactivity in the NFSR and SFP.

3.10.2 Code Versions and Applications

The analysis methodology employs the following computer codes and cross-section libraries: the two-dimensional (2-D) transport lattice code PARAGON for the in-reactor depletion calculations and SCALE Version 6.2.3 for the NFSR and SFP k_{eff} calculations. PARAGON two-dimensional (2-D) transport lattice code has been approved by the NRC Final SE for *Westinghouse Topical Report WCAP-16045-P, Revision 0 "Qualification of the Two-Dimensional Transport Code Paragon,"* (ML040780402). SCALE is a comprehensive modeling and simulation suite for nuclear safety analysis and design developed and maintained by Oak Ridge National Laboratory under contract with the NRC, U.S. Department of Energy, and the National Nuclear Security Administration to perform reactor physics, criticality safety, radiation shielding, and spent fuel characterization for nuclear facilities and transportation/storage package designs and has been used extensively for NFSR and SFP criticality analysis.

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[2] This parenthetical indicates that the B^{10} in the IFBA was reduced 5% for conservatism when modeling the fuel assemblies in the SFP.

3.10.3 New Fuel Vault

The regulation 10 CFR 50.68(b) has two paragraphs that directly address storage of fuel in fresh fuel storage racks. Both are accident focused as the fresh fuel storage racks are dry/unmoderated. Paragraph 50.68(b)(2) addresses the scenario where the fresh fuel storage racks become fully flooded. Paragraph 50.68(b)(3) addresses the scenario where the fresh fuel storage racks are the subject of an optimum moderation condition. Typically, the source of the optimum moderation is thought to be firefighting water or aqueous foam from firefighting efforts in the building holding the fresh fuel storage racks. If the building housing the fresh fuel storage racks is susceptible to environmental damage, that could be another source of moderating medium. The license's term for its fresh fuel storage racks is new fuel storage racks (NFSR).

The Vogtle NFSR AOR (ML20244D565) calculated a k_{eff} of 0.9364 and 0.9434 for the fully flooded and optimum moderation cases respectively. The Vogtle NFSR AOR did not credit any burnable poison in the fuel rods. The requested exemption to 10 CFR 50.68 credits 128 IFBA rods with 1.5X standard loading (^{10}B reduced 5% were applicable in SFP models only) and 8" cutback region (IFBA modeled as full length in depletion analysis). SNC's letter dated June 30, 2022, indicates the LTAs (with the IFBA) are approximately 0.12 Δkeff less reactive than its current fresh fuel assemblies (w/o any IFBA) in a fully moderated scenario and approximately 0.10 Δkeff less reactive in an optimum moderated scenario. These Δkeff values represent significant margins. SNC's letter dated September 13, 2022, provided additional details on how these estimates were derived. The individual calculations performed by the licensee to estimate the reactivity margin provided by crediting 128 IFBA rods with 1.5X standard loading were not performed to a 95 percent probability, 95 percent confidence level. The large margin did, however, provide reasonable assurance that the k_{eff} of Vogtle NFSR will not exceed 0.95, at a 95 percent probability, 95 percent confidence level if fully flooded or 0.98, at a 95 percent probability, 95 percent confidence level if optimally moderated.

3.10.4 Spent Fuel Pool

The regulation 10 CFR 50.68(b) has one paragraph that directly addresses storage of fuel in the SFP. Paragraph 50.68(b)(4) of 10 CFR requires, *"If no credit for soluble boron is taken, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water*." The Vogtle SFP NCS analysis does contain soluble boron, so the 10 CFR 50.68(b)(4) requirements regarding soluble boron do apply. There is no optimum moderation paragraph for SFPs since TS 4.3.2, "Drainage," ensures a minimum water level in the SFP providing reasonable assurance an optimum moderation will not occur.

With respect to the SFP, the proposed LAR would prohibit the LTAs from being stored in the Vogtle, Unit 1, SFP. The LTAs would be limited to being stored in only two of the Vogtle, Unit 2, SFP storage configurations; the two-out-of-four (2oo4) configuration, which is a repeating 2x2 array of alternating fresh unburned and unpoisoned fuel assemblies with empty storage cells, and the all-cell (4oo4) configuration, which is a repeating 2x2 array with each storage cell filled with a fuel assembly meeting the specified burnup/enrichment requirements.

As with the NFSR the justification for allowing storage of the LTAs in the 2oo4 configuration relies primarily on crediting the 128 IFBA rods with 1.5X standard loading. SNC's letter dated June 30, 2022, states, "The 2oo4 storage configuration was modeled with and without 128 IFBA rods and does not specifically credit any installed WABA. The 2oo4 model with reference 5.0 wt% fuel and no IFBA produced a calculated k_{eff} of 0.9455. The same 2oo4 configuration model with the LTA assemblies with 128 IFBA rods yielded a k_{eff} of 0.7970. Thus, the LTAs have a reactivity margin of about 15% at fresh conditions. Peak reactivity will rise early in the assembly life as IFBA burns out but will not challenge the 15% margin. As a result, sufficient reactivity hold-down is present to conclude it is safe to load the LTAs in the 2oo4 SFP storage configuration." The k_{eff} of 0.9455 is comparable to the zero boron k_{eff} 0.94574 calculated in the licensee's analysis of record (ML042320413). NUREG/CR-6760, *Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit"* (ML020770436), supports the licensee's assertion that crediting the IFBAs for fresh fuel does provide substantial margin. However, NUREG/CR-6760 does not support the licensee's assertion that margin calculated for fresh fuel will not be challenged as the fuel assembly is used in the reactor. NUREG/CR-6760 does support that a 128 IFBA rod fuel assembly will never be as reactive as an equivalent fresh fuel assembly without the 128 IFBA rods. While the licensee's LTAs may not always have the 15% Δk margin, the margin will always be substantial. That substantial large margin provides reasonable assurance that the keff of the Vogtle, Unit 2, SFP 2oo4 storage configuration will not meet or exceed a k_{eff} of 1.0 at a 95 percent probability, 95 percent confidence level if fully flooded with unborated water or exceed a k_{eff} of 0.95, at a 95 percent probability, 95 percent confidence level if fully flooded with borated water.

The justification for allowing storage of the LTAs in the 4oo4 configuration relies on setting a large burnup limit for storage in the burnup credit configuration. SNC's letter dated June 30, 2022, states, "For storage of the LTAs in the all-cell configuration, a burnup limit of 64 GWd/MTU was selected. The AOR burnup requirement is about 40 GWd/MTU and a 64 GWd/MTU LTA burnup limit provides a 24 GWd/MTU or greater than 8% in k_{eff} margin for the LTAs. At 64 GWd/MTU, no additional analysis is needed to allow safe storage of the LTAs in the all cell [aka 4oo4] storage configuration." SNC's letter dated September 13, 2022, provided additional details on how these estimates were derived. The NRC staff consulted the licensee's analysis of record (ML042320413). The 40 GWd/MTU burnup requirement the licensee cites is just a little more than their current TS limit for a fuel assembly with 5 wt/% U235. While it is reasonable to believe the LTAs with a small amount of additional fissile material would require a small amount of additional burnup, it is unclear that 40 GWd/MTU burnup would be sufficient to meet the regulatory requirements. To overcome that and preclude the need for a more detailed analysis, SNC proposed setting the burnup limit for the LTAs at 64 GWd/MTU. The licensee's analysis of record has a third order polynomial relating required burnup to initial fuel assembly enrichment for the 4oo4 (aka All-Cell) configuration. While there are certainly differences between the fuel assemblies modeled in the licensee's AOR and the proposed LTAs, the NRC staff concludes the aforementioned polynomial can provide some insight. The NRC staff used the polynomial to estimate how much burnup a fuel assembly with all rods enriched to 6 wt/% U-235 would need for storage in the Vogtle, Unit 2, 4oo4 storage configuration. That estimate is approximately 53.7 GWd/MTU. This estimated requirement is for a fuel assembly with 20 percent more fissile material than SNC's current fuel whereas the LTAs will only have about 0.427 percent more fissile material. These estimates indicate the licensee's use of a 64 GWd/MTU burnup limit for storage in the Vogtle, Unit 2, 4oo4 storage configuration provides substantial margin. That substantial large margin provides reasonable assurance that the k_{eff} of the Vogtle, Unit 2, SFP 4oo4 storage configuration will not meet or exceed a k_{eff} of 1.0 at a 95 percent probability, 95 percent confidence level if fully flooded with unborated water or exceed a

k_{eff} of 0.95, at a 95 percent probability, 95 percent confidence level if fully flooded with borated water.

SNC evaluated the multiple misloading accident where all four fresh LTAs are collocated in a 2x2 array. SNC's letter dated June 30, 2022, states, "The limiting accident in the SFP is a multiple misload [14]. An infinitely modeled multiple misload (4 ATF LTA assemblies in a 2x2 reflected storage array) with TS-required 2000 ppm of soluble boron results in total reactivity of 0.9469, including a total bias and uncertainty of 0.045, bounding all analysis of record bias and uncertainty totals. This reactivity is without any IFBA (which was shown to provide significant hold-down) or WABA within the model. Additionally, the analysis considered an infinite misload of LTAs when only four will be operated. As a result, the LTAs do not create an accident condition concern." The estimated k_{eff} of 0.9469 is close to the regulatory limit and in of itself does not represent substantial margin. Additionally, there is insufficient information for the NRC staff to determine whether that number alone would meet the regulatory limit of k_{eff} less than or equal to 0.95, at a 95 percent probability, 95 percent confidence level if fully flooded with borated water. However, as the licensee stated, and as previously shown on the 2oo4 configuration analysis the 128 IFBA rods will provide substantial reactivity margin. When the 128 IFBA rods are considered, the NRC concludes there is reasonable assurance the regulatory requirement for k_{eff} to be less than or equal to 0.95, at a 95 percent probability, 95 percent confidence level if fully flooded with borated water.

3.10.5 Fuel Handling and Storage Conclusion

The NRC staff has evaluated SNC's request for four LTAs having a maximum of 6 wt% U-235. The NRC staff has found that there is reasonable assurance the four LTAs as described in SNC's letter dated June 30, 2022, will meet 10 CFR 50.68(b)(2), 10 CFR 50.68(b)(3), and 10 CFR 50.68(b)(4). SNC will update its plant procedure(s), if necessary, to meet 10 CFR 50.68(b)(1) that states:

> Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water.

3.11 <u>Design Transients</u>

In its letter dated June 30, 2022, the licensee states that "The core reactivity parameters were reviewed, and it was determined that any differences caused by the inclusion of the LTAs have a negligible impact on the margin to trip and control systems operability analyses. The results and conclusions of the analysis of record remain valid for the LTA program."

NRC staff review concluded that transient AOR would not be expected to be impacted by the reactivity changes when the 16 slightly enriched LTA rods are averaged out over the 50,952 active fuel rods in the core; the overall change in reactivity across the core is negligible and the existing AOR remains bounding. Management of local fuel effects are sufficient, as described in Section 3.7 of this SE, so long as LTAs are placed in the analyzed non-limiting core locations for rod ejection.

3.12 Best Estimate Analyzer for Core Operations Nuclear (BEACON™) Core Monitoring System

In its letter dated June 30, 2022, the licensee states that the BEACON™ core monitoring system will be unaffected by the LTAs. The LTAs will be placed to have a negligible effect on the measurements of the power distribution monitoring system. The NRC staff finds this acceptable, based on the limited number of LTAs in the core, coupled with the explicit modeling of new materials, which will ensure that the power distribution in the core can be monitored.

3.13 Radiological Review

3.13.1 Core Source Term

In its letter dated June 30, 2022, the licensee stated that Supporting analyses for the implementation of the Vogtle lead test assembly (LTA) program include the analysis of variations in the isotopic inventory of the core. An evaluation was performed to determine the impact of 16 higher enrichment lead test rods (four LTAs, each with four higher enrichment fuel rods) on the core radionuclide inventory used for radiological/dose consequences.

The evaluation was performed using an established computer code sequence used to simulate nuclear fuel cycles and compositions, taking into account transmutation of isotopes over core life of commercial light water applications. The evaluation considered bounding ranges of enrichment, burnup, and rod power and determined that the impact of the 16 higher enriched lead test rods on the overall core radionuclide inventory was inconsequential. The licensee concluded that "For significant isotopes which contribute to dose, the core inventory for the for the core design implementing the LTAs was determined to be bounded by the existing core inventory used for radiological/dose consequences."

The NRC staff review determined that the results of SNC's supporting analyses for the implementation of the Vogtle LTA program were sufficient. The impact of 16 slightly enriched rods on a core of 50,952 active fuel rods will not have a significant impact on the overall core enrichment level, power, and burnup factors which impact radiological dose consequences. The quantitative analysis of the impact of the small number of rods remains within the existing analysis of record and maintains the conservative bounding assumptions of the analysis. Based on the above, the existing source term analysis can be reasonably maintained with LTA addition.

3.13.2 Radiological and Dose Consequences

In its letter dated June 30, 2022, the licensee stated that:

> It has been determined that the LTAs do not impact the radiological consequences analyses for the following design basis accidents:
>
> - Loss of Coolant Accident (LOCA)
> - Steam Generator Tube Rupture (SGTR)
> - Main Steam Line Break (MSLB)
> - Loss of Offsite Power (LOOP)
> - Locked Rotor (LR)
> - Control Rod Ejection (CRE)

- Small Line Break Outside Containment (SLBOC)
- Waste Gas Decay Tank Rupture (WGDTR)
- Liquid Waste Tank Failure (LWTF)
- Fuel Handling Accident (FHA)

This determination is based on the following confirmations:

- The LTAs do not impact the reactor coolant system (RCS) and gas and liquid waste tank nuclide activities (SGTR, MSLB, LOOP, SLBOC, WGDTR, LWTF).
- The RCS mass released during the assumed small line break outside containment is calculated based on the assumed flow rate and is not impacted by changes in the fuel (SLBOC).
- The calculations of the steam releases from the steam generators to the environment used in the radiological consequences analyses model the core-wide fuel average temperature, the total mass in the core, and the core decay heat. None of these are impacted by the inclusion of the four LTAs (SGTR, MSLB, LOOP, LR, CRE).
- It is assumed that the LTAs lead the core and therefore could be postulated to fail following a locked rotor or rod ejection accident. It has been confirmed that inclusion of the LTAs does not increase the amount of fuel damage considered in the radiological consequences analyses of the locked rotor or rod ejection accident in the analyses of record, i.e., 5% for locked rotor with no fuel melting and 10% for rod ejection with melting limited to less than the innermost 10% of the fuel pellet at the hot spot (LR, CRE).
- It has been confirmed that the LTAs do not impact the core average nuclide activities used to determine the activity released from fuel assumed to fail following a locked rotor, rod ejection accident, or LOCA (LR, CRE, LOCA).
- It has been confirmed that the gap fractions used to define the activity released from fuel assumed to fail following a locked rotor, rod ejection, or fuel handling accident are not impacted by the differences in the LTAs from current fuel (LR, CRE, FHA). The cladding material and fuel enrichment do not impact the mechanisms of fission gas release. ADOPT fuel changes the fuel microstructure by increasing the grain size. Increased fuel grain size increases the diffusion distance of gases, resulting in lower transient fission gas release. Steady-state fission gas release is approximately the same as standard UO2 fuel. This is consistent with the evaluation of gap release fractions in Section 6.1.1 of WCAP-18482 [2].
- The activities of dose significant radionuclides postulated for release in a fuel handling accident (FHA) involving the LTAs (e.g., Xe-133, Xe-135, I-131, I-132, I-133) are bounded by the activities of the same radionuclides in the existing FHA analyses. Therefore, the dose potential of an FHA involving the LTAs is bounded by the existing fuel handling accident radiological consequence analyses when evaluated at the same decay time (FHA).

The NRC staff determined that the results of SNC's supporting analyses for the implementation of the Vogtle LTA program on radiological and dose consequences of design basis accidents are sufficient. The impact of 4 slightly enriched rods on a fuel assembly of 264 totals rods as well as the impact of 16 slightly enriched rods on a core of 50,952 fuel rods will not have a significant impact on the overall assembly and core fuel parameters and when averaged out are less than the bounding parameter values in the respective AORs. Since the quantitative analysis impact of the small number of rods remains within the existing AORs, and maintains the conservative bounding assumptions of the analysis. Based on the above, the existing DBA radiological dose consequence analyses can be reasonably maintained with LTA addition.

The NRC staff did not perform independent confirmatory dose evaluations because of the limited impact that LTAs would have on the Vogtle, Units 1 and 2, CLB core inventory. As described in RG 1.195, "An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid." Therefore, the NRC staff reviewed SNC's accident AOR, including descriptions in Chapter 15 of the UFSAR and related guidance and references. The purpose of this review was to determine if the installation of LTAs would impact the assumptions and results of those analyses and if the analyses would continue to be bounding following approval of the proposed LAR.

The fission product inventory available for release, if fuel becomes damaged during an accident, is described in Table 15A-3 of the licensee's UFSAR and is called the "current licensing basis (CLB) core inventory" for the remainder of this evaluation. The NRC guidance provides instruction on what source term is to be assumed to have been released when conducting design basis radiological analyses. In some analyses, a fraction of the CLB core inventory is assumed to be released. In other analyses, no fuel damage is expected to occur, so the released source term is assumed to be from radioactivity that is present in the reactor coolant system (RCS) during normal plant operations.

In response to questions from the NRC staff during an audit, the licensee supplemented its letter dated June 30, 2022, with a letter dated January 20, 2023. In the letter dated January 20, 2023, SNC compared the CLB core inventory to the expected inventory that would result if LTAs were installed in the core. The licensee calculated the LTA inventory using methods acceptable to the NRC (i.e., ORIGEN ARP computer code). Table 2 of the letter dated January 20, 2023, shows that the LTA inventory is bounded by the CLB inventory for the radionuclides of concern that contribute to dose.

Section 3.3 of SNC's letter dated June 30, 2022, provides the licensee's considerations in determining the impacts of LTAs on the design basis radiological analyses. SNC evaluated ten design basis accidents for radiological consequences. None of the accident analyses are impacted by LTAs, such that re-analysis would be required; therefore, the analyses reflected in the CLB continue to apply when determining compliance with acceptance criteria. Additionally, the NRC staff notes that RCS chemistry may be impacted by the use of Cr coated fuel as described in Interim Staff Guidance, ATF-ISG-2020-1, "Supplemental Guidance Regarding the Chromium-Coated Zirconium Alloy Fuel Cladding Accident Tolerant Fuel Concept," dated July 2020 (ML19343A121); however, TS 3.4.16 "RCS Specific Activity," ensures that RCS specific activity will remain within acceptable bounds as assumed in applicable accident analyses.

The NRC staff finds that the proposed use of LTAs does not impact the assumptions or inputs used in the CLB accident analyses, or conclusions drawn on those results; therefore, re-analysis is not required as described in section 1.2.3 of RG 1.195.

Furthermore, the NRC staff finds that SNC has demonstrated that, as it pertains to radiological consequences of DBAs, adequate protection will be maintained during the use of LTAs, as proposed.

Based on the above, the NRC staff finds that the CLB design basis radiological analysis continues to satisfy applicable acceptance criteria that were included to satisfy 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC 19.

3.14 PRA Insights

Probabilistic risk assessments (PRAs) estimate risk and identify what could go wrong, how likely it is, and what the consequences could be. PRA results give insight into the strengths and weaknesses of the design and operation of a nuclear power plant. PRAs are used in a wide range of activities, including the current risk informed programs such as risk informed technical specification initiatives 4b (risk-informed completion times), and 5b (surveillance frequency control program) (ML18183A493 and ML090850642, respectively). During the LAR process, the NRC staff reviews PRA-related information in the licensing application. A key tenet of the NRC's risk-informed decision-making is that these models reflect the as-built, as-operated plant.

For this reason, the PRA models should be updated to reflect significant plant modifications. The introduction of different fuel into the reactor core may affect these models, particularly if the reactor core composition strongly influences the plant's response to a postulated accident (e.g., impact to the success criteria, required time associated with operator actions, mission time, or accident sequences). The success criteria establish the minimum number or combinations of structures, systems, or components (SSCs) required to operate to ensure that the safety functions are satisfied.[3] The required time is the time needed by operators to successfully perform and complete an action.[4] Mission time is the time that an SSC is required to operate in order to successfully perform its function.[5] An accident sequence is a representation in terms of an initiating event followed by a sequence of failures or successes of events (such as SSC operation, or operator action) that can lead to undesired consequences.[6]

In section 3.14.1 to 3.14.6 below, the NRC staff evaluates the impact of implementing ATF LTAs on the technical adequacy of the licensee's PRA models with regard to the NRC-approved risk informed programs described in license amendment numbers 158 and 188 to RFOL NPF-68 (i.e., risk-informed completion time program) and amendment numbers 140 and 171 for RFOL NPF-81 (i.e., surveillance frequency control program) for Vogtle Electric Generating Plant, Units 1 and 2 (ML102520083 and ML15127A669, respectively).

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[3] This term is defined, in part, in the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications."

[4] This term is defined, in part, in ASME/ANS RA-Sa-2009.

[5] This term is defined, in part, in ASME/ANS RA-Sa-2009.

[6] This term is defined, in part, in ASME/ANS RA-Sa-2009.

3.14.1 PRA Insights

In the letter dated June 30, 2022, the licensee evaluated the following four parameters that could impact the PRA models due to placement of LTAs in the reactor core:

- Decay heat level at the time of reactor trip due to initiating events
- The hottest core node temperature
- Core exit thermocouple temperature
- Unfavorable exposure time in anticipated transient without scram (ATWS)

3.14.2 Decay Heat level

The licensee estimated the change in core averaged decay heat generation level due to the new LTAs. Their analyses result in an increase of less than 0.01-percent to the core averaged decay heat generation level.

The NRC staff agrees that a less than 0.01-percent increase in the core averaged decay heat generation level is a negligible change that does not impact the success criteria, required time associated with operator actions, mission time associated with SSCs, or the accident sequences in the licensee's PRA model.

3.14.3 Hottest Core Node Temperature

SNC estimated that, following a reactor trip, the LTA would be less than 0.6% hotter than the current fuel assembly at that location, and it has a negligible to minor impact on the core peak temperature response.

The NRC staff agrees that a less than 0.6-percent increase in LTA temperature as compared to the current fuel assembly and resultant increase in core peak temperature heat generation level of 0.6-percent is a negligible change that does not impact the success criteria, required time associated with operator actions, mission time associated with SSCs, or the accident sequences in the licensee's PRA model.

3.14.4 Core Exit Thermocouple Temperature

There are many core exit thermocouples which measure coolant outlet temperatures at preselected positions and provide indication of inadequate core cooling and core subcooling margin monitoring. They are divided into 2 trains and are shown on a plasma display by the plant safety monitoring system. The plasma display shows minimum, average, and maximum core exit thermocouple temperatures for each reactor quadrant. The plasma display receives input from all the core exit thermocouples throughout all four reactor quadrants.

SNC uses the core exit thermocouple temperature to enter functional recovery emergency operating procedures and severe accident management guidelines. The licensee's assessment of LTA bundle power shows that the LTA power increase is in the order of 1.006 and corresponds to a local assembly temperature increase of approximately 5-degrees Fahrenheit. The 5-degree Fahrenheit increase is within the core exit thermocouple measurement uncertainty and therefore, does not affect the ability to display the core exit thermocouple temperature on the plasma display in the main control room.

Because the 5-degree increase in core exit thermocouple temperature can be displayed on the plasma display in the main control room for operator monitoring, the NRC staff finds that the increase in core exit thermocouple measurement has a negligible impact on the success criteria, mission time associated with SSCs, the accident sequences, or required time associated with operator actions related to entering emergency operating procedures or severe accident management guidelines and taking recovery actions in the licensee's PRA model.

3.14.5 Unfavorable exposure time in ATWS

An unfavorable exposure time is the duration of time in a cycle during which pressure relief by pressurizer power operated relief valves and safety valves is not sufficient to prevent reactor coolant system pressure from exceeding ASME service level C limit during the initial pressure transient after ATWS. The unfavorable exposure time durations are one of the most important variables in determining core damage in the ATWS PRA model.

SNC considered two categories of non-LOCA events for the LTAs: those that are dependent on core-average effects and those that are impacted by local effects in the fuel rods. ATWS events are included in the first category. For the first category, the licensee performed an analysis to address gross plant criteria, such as loss of shutdown margin, margin to hot leg saturation, over pressurization of the reactor coolant system, over pressurization of the secondary system, or overfilling of the pressurizer. Based on the total number of the fuel rods inserted into the core (four LTAs of 193 total fuel assemblies and up to 16 higher enrichment rods out of 50,952 total fuel rods), the licensee evaluated the impact on core-average effects such as core average fuel heat transfer characteristics, decay heat, and initial core stored energy and determined that they were negligible and that any small effects caused by the LTAs is less than the existing margins in the safety analyses. Specifically, the licensee amendment request states that the LTA geometry, material properties, and reactivity feedback characteristics have no impact on the safety analyses for non-LOCA events and any small effects caused by differences in the geometry, material properties, and/or reactivity feedback characteristics of the LTAs are also less than the existing margins in the safety analyses. As such, the licensee determined that the LTAs do not impact the core-average effects for non-LOCA events (includes ATWS).

In Enclosure 2 of the LAR dated June 30, 2022, SNC concluded that the LTAs do not impact the safety analyses for non-LOCA events such that all their acceptance criteria are met, and the conclusions documented in the applicable final safety analysis report, as updated (UFSAR) remains valid. The LTAs do not impact the moderator temperature coefficient limit over 95 percent of the operating cycle. Additionally, as discussed in the LAR Enclosure 2, Section 3.9 of the letter dated June 30, 2022, the LTAs are designed to have a lower power peaking factor than that used in the thermal-hydraulic analysis of record for the UFSAR, and there is no change in UFSAR Section 4.4, “Thermal and Hydraulic Design,” along with no change to the thermal-hydraulic input to the plant TSs.

Because the LTAs do not change the (1) thermal-hydraulic analyses of record, (2) the safety analyses for the non-LOCA events (includes ATWS), or (3) the operation of any plant SSCs, and considering that the licensee performs evaluations of the LTAs as part of the cycle specific reload safety analysis to confirm that the acceptance criteria of the existing safety analyses continues to be met, the NRC finds that there is a negligible impact on unfavorable exposure time in the ATWS PRA model and there is a negligible impact on the success criteria, required time associated with operator actions, mission time associated with SSCs, or the accident sequences in the licensee's PRA model.

3.14.6 PRA Insights Conclusion

The NRC staff finds that the four parameters discussed above have a negligible impact on the success criteria, required time, mission time, and the accident sequences in the licensee's PRA models. Additionally, the NRC staff finds that because there is a negligible impact on the SNC's PRA models, there is a negligible impact on the licensee's surveillance frequency control program and the risk-informed completion time program. Therefore, the NRC staff concludes that there is reasonable assurance that SNC's PRA models continue to support the risk-informed completion time and surveillance frequency control programs and therefore, the placement of LTAs in the Vogtle Electric Generating Plant, Unit 2, reactor core is acceptable.

3.15 RFOL Condition 2.D Changes

3.15.1 Current RFOL Condition 2.D

The Vogtle, Unit 1, RFOL NPF-68, License Condition 2.D states:

> The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption b are identified in Section 6.2.6 of SSER 5.
>
> An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1967, issued August 21, 1986, and relieved GPC from the requirement of having a criticality alarm system. GPC and Southern Nuclear are hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.
>
> These exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. The exemptions in items b and c above are granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

The Vogtle, Unit 2, RFOL NPF-81, License Condition 2.D states:

> The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. These include (a) an exemption from the requirements of 10 CFR 70.24 for two criticality monitors around the fuel storage area, and (b) an exemption from the requirements of Paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding exemption b are identified in Section 6.2.6 of SSER 8.
>
> An exemption was previously granted pursuant to 10 CFR 70.24. The exemption was granted with NRC materials license No. SNM-1981, issued July 13, 1988,

> and relieved GPC from the requirement of having a criticality alarm system. GPC and Southern Nuclear are hereby exempted from the criticality alarm system provision of 10 CFR 70.24 so far as this section applies to the storage of fuel assemblies held under this license.
>
> These exemptions are authorized by law, will not present an undue risk to the public health and safety and are consistent with the common defense and security. The exemption in item b above is granted pursuant to 10 CFR 50.12. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

3.15.2 Proposed RFOL Condition 2.D

SNC proposes to delete the current Condition 2.D, for Vogtle, Units 1 and 2, and replace it with the following:[7]

> The facility requires an exemption from the requirements of paragraph III.D.2(b)(ii) of Appendix J of 10 CFR 50, the testing of containment air locks at times when containment integrity is not required. The special circumstances regarding this exemption are identified in Section 6.2.6 of SSER 5. This exemption is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. This exemption is granted pursuant to 10 CFR 50.12. With this exemption, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

3.15.3 Evaluation of RFOL 2.D

By letters dated June 30, and September 13, 2022, SNC is requesting to voluntarily change its licensing basis for Vogtle, Units 1 and 2, from 10 CFR 70.24, “Criticality accident requirements,” to 10 CFR 50.68, “Criticality accident requirements.”

The regulation 10 CFR 50.68(a) states:

> Each holder of a construction permit or operating license for a nuclear power reactor issued under this part or a combined license for a nuclear power reactor issued under Part 52 of this chapter, shall comply with either 10 CFR 70.24 of this chapter or the requirements in paragraph (b) of this section.

Since SNC is the holder of the operating licenses for Vogtle, Units 1 and 2, SNC is voluntarily changing its licensing basis from 10 CFR 70.24 to 10 CFR 50.68. The regulation 10 CFR 50.68(a) allows this change; therefore, the NRC staff finds it acceptable for SNC to voluntarily adopt 10 CFR 50.68 for Vogtle, Units 1 and 2.

SNC is proposing to delete License Condition 2.D and replace it with the language in Section 3.14.2 for RFOL NPF-68 and RFOL NPF-81, since it is adopting 10 CFR 50.68. The NRC finds it acceptable to revise License Condition 2.D for RFOL NPF-68 and RFOL NPF-81,

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[7] The Unit 1 version of this license condition follows. The Unit 2 version is the same, except that “SSER 5” is “SSER 8.”

because the licensee is adopting 10 CFR 50.68 as its licensing basis in lieu of 10 CFR 70.24, which is allowed by 10 CFR 50.68(a).

The exemption from 10 CFR 50.68(b)(7) to allow four LTAs greater than 5 wt/% U235 is addressed under ADAMS Accession No. ML23094A051.

3.16 Technical Specifications Additions and/or Changes

3.16.1 TS 3.7.18, " Fuel Assembly Storage in the Fuel Storage Pool"

The current LCO 3.7.18 does not have a NOTE.

The revised LCO 3.7.18 would add the following NOTE:

> ---NOTE---
> Figures 3.7.18-1 and 3.7.18-2 do not apply to lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4.
> --

The current Surveillance Requirement (SR) 3.7.18.1 does not have a NOTE.

The revised SR 3.7.18.1 would add the following NOTE:

> ---NOTE---
> Figures 3.7.18-1 and 3.7.18-2 do not apply to lead test assemblies 7ST1, 7ST2, 7ST3, and 7ST4.
> --

3.16.2 TS 4.2, "Reactor Core"

The current TS 4.2.1, "Fuel Assemblies," states:

> The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

The revised TS 4.2.1 would state:

> The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy, ZIRLO®, or Optimized ZIRLO™ clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have

been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies (LTAs) that have not completed representative testing may be placed in nonlimiting core regions. In addition, LTAs 7ST1, 7ST2, 7ST3, and 7ST4, which contain fuel rods that include advanced coated cladding features, doped or standard fuel material, and up to four fuel rods with a maximum nominal U-235 enrichment of 6.0 weight percent, are permitted to be placed in limiting core regions for up to two cycles of operation without completion of representative testing. These LTAs cannot be placed in core regions that have been shown to be limiting with respect to the control rod ejection analysis.

3.16.3 TS 4.3, "Fuel Storage"

The current TS 4.3.1.1 does not have an item g.

The revised TS 4.3.1.1 would add an item g that states:

g. LTAs 7ST1, 7ST2, 7ST3, and 7ST4 are prohibited from Unit 1 spent fuel pool storage.

The current TS 4.3.1.2 does not have a NOTE.

The revised TS 4.3.1.2 would add a NOTE that states:

-------------------------------------NOTE---
4.3.1.2a, 4.3.1.2d, and 4.3.1.2e do not apply to LTAs 7ST1, 7ST2, 7ST3, and 7ST4.

The current TS 4.3.1.2 does not have an item g.

The revised TS 4.3.1.2 would add an item g that states:

g. For LTAs 7ST1, 7ST2, 7ST3, and 7ST4, the following requirements apply for storage in the Unit 2 spent fuel storage racks:
   1. Unrestricted storage is allowed in the 2-out-of-4 checkerboard storage configuration as shown in TS Figure 4.3.1-1.
   2. Storage is allowed in the all-cell storage configuration ("A" assemblies as shown on TS Figures 4.3.1-3 and 4.3.1-5) when the LTAs reach 64,000 MWd/MTU of burnup.

The current TS 4.3.1.3 a states:

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent;

The revised TS 4.3.1.3 a would state:

a. Fuel assemblies having a maximum U-235 enrichment of 5.05 weight percent except for LTAs 7ST1, 7ST2, 7ST3, and 7ST4 which may have four rods per assembly enriched up to 6.0 weight percent U-235;

3.16.4 NRC Staff Review of Proposed TS Additions and Changes

Based on the review results detailed in technical evaluations sections above, the NRC staff has determined that the proposed TSs would continue to be derived from the analyses and evaluations included in the UFSAR and amendments thereto in accordance with 10 CFR 50.36(b).

The proposed amendments also comply with 10 CFR 50.36(c)(2) and (3). The TS LCO 3.7.18 and SR 3.7.18.1 still apply to the LTAs; only the figures referenced in the TS notes (Figures 3.7.18-1 and 3.7.18-2) do not apply. As referenced in TS LCO 3.7.18 and SR 3.7.18.1, the TS 4.3.1.1 (Unit 1) and 4.3.1.2 (Unit 2) would still apply to LCO 3.7.18, and SR 3.7.18.1. The proposed changes to TS 4.3.1.1 and 4.3.1.2 add storage requirements related to the spent LTA assemblies. Therefore, LCO 3.7.18, as amended, would continue to require the lowest functional capability or performance levels of equipment required for safe operation of the facility, in accordance with 10 CFR 50.36(c)(2). Therefore, this TS change is acceptable. Similarly, the proposed SR revision to SR 3.7.18.1 will continue to provide an adequate way to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met, in accordance with 10 CFR 50.36(c)(3). Therefore, this TS change is acceptable.

As required by 10 CFR 50.36(c)(4), the proposed design features revisions in TS 4.2.1, 4.3.1.1, 4.3.1.2 and 4.3.1.3 adequately describe the features of the ATF LTAs, such as materials of construction and geometric arrangements, which, if altered or modified, could have a significant effect on safety. In addition, the proposed changes describe information that is not covered by the LCOs, safety limits or surveillance requirements. The proposed revisions provide an equivalent level of detail for the description of the LTAs and their new and spent fuel storage requirements as is provided for the existing fuel assemblies. TS 4.2.1 does not specify that the LTAs are limited to Vogtle, Unit 2, only; however, Vogtle, Unit 1, is subject to 10 CFR 50.46 and 10 CFR Part 50, Appendix K requirements. Vogtle, Unit 1, is not receiving an exemption from the specified 10 CFR 50.46 and 10 CFR Part 50, Appendix K requirements to allow these particular LTAs to be placed in the core. These regulations apply to prevent the LTAs 7ST1, 7ST2, 7ST3, and 7ST4 from being placed in the Vogtle, Unit 1, reactor core.

Therefore, the NRC staff determined that these changes are acceptable because the proposed TSs will continue to include required design features of the facility, such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety, in accordance with 10 CFR 50.36(c)(4). Therefore, the NRC staff concludes that the proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Georgia State official was notified of the proposed issuance of the amendments on April 12, 2023. On May 16, 2023, the Georgia State official confirmed did not have any comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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 published in the *Federal Register* on November 8, 2022 (87 FR 67508), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

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Date: August 1, 2023

SUBJECT: VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENTS NOS. 220 AND 203, REGARDING USE OF ACCIDENT TOLERANT FUEL LEAD TEST ASSEMBLIES (EPID L-2022-LLA-0097) DATED AUGUST 1, 2023

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OFFICE	NRR/DSS/SNSB/BC	NRR/DSS/STSB/BC	NRR/DSS/SCPB/BC	NRR/DRA/ARCB/BC
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