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

November 1, 2022

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

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 243 AND 240 TO REVISE TECHNICAL SPECIFICATION 5.5.17, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM," TO INCREASE CALCULATED PEAK CONTAINMENT PRESSURE (EPID L-2021-LLA-229)

Dear Ms. Gayheart:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 243 to Renewed Facility Operating License No. NPF-2 and Amendment No. 240 to Renewed Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively. The amendments are in response to your application dated December 13, 2021, as supplemented by letter dated June 20, 2022.

The amendments revise the Technical Specifications (TS) 5.5.17, "Containment Leakage Rate Testing Program," to update the peak calculated containment internal pressure for the design basis loss-of-coolant accident from 43.8 to 45 pounds per square inch gauge.

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

Sincerely,

/RA

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

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 243 to NPF-2
2. Amendment No. 240 to NPF-8
3. Safety Evaluation

cc: Listserv

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

SOUTHERN NUCLEAR OPERATING COMPANY

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 243
Renewed License No. NPF-2

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

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 1 (the facility), Renewed Facility Operating License No. NPF-2 (the license) filed by Southern Nuclear Operating Company (the licensee), dated December 13, 2021, as supplemented by letter dated June 20, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the Code of Federal Regulations (10 CFR) Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment. Paragraph 2.C.(2) of the license is hereby amended to read as follows:

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 243, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications

3. This amendment is effective as of its date of issuance and shall be implemented within 90 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2022.11.01 11:46:31 -04'00'

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

Attachment:
Changes to Renewed Facility Operating License and Technical Specifications

Date of Issuance: November 1, 2022

SOUTHERN NUCLEAR OPERATING COMPANY

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 240
Renewed License No. NPF-8

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

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 2 (the facility), Renewed Facility Operating License No. NPF-8 (the license) filed by Southern Nuclear Operating Company (the licensee), dated December 13, 2021, as supplemented by letter dated June 20, 2022, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in Title 10 of the Code of Federal Regulations (10 CFR) Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment. Paragraph 2.C.(2) of the license are hereby amended to read as follows:

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 240, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

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Date: 2022.11.01 11:47:13 -04'00'

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

Attachment:
Changes to Renewed Facility Operating License and Technical Specifications

Date of Issuance: November 1, 2022

ATTACHMENT TO JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT NO. 243

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

AND LICENSE AMENDMENT NO. 240

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Renewed Facility Operating Licenses and 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	Insert
License	License
NPF-2, page 4	NPF-2, page 4
NPF-8, page 3	NPF-8, page 3
TSs	TSs
5.5-14	5.5-14

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 243, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

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

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

b. Deleted per Amendment 13

c. Deleted per Amendment 2

d. Deleted per Amendment 2

e. Deleted per Amendment 152

Deleted per Amendment 2

f. Deleted per Amendment 158

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

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

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

3) Identification of process sampling points;

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

5) Procedures defining corrective actions for off control point chemistry conditions; and

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

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

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

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

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

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

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

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2821 megawatts thermal.

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. 240, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Deleted per Amendment 144
(4) Deleted per Amendment 149
(5) Deleted per Amendment 144

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Revision 3-A, dated July 2012, and the conditions and limitations specified in NEI 94-01, Revision 2-A, dated October 2008 as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of frequency specified by the ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B, will be performed in accordance with the requirements of and frequency specified by the ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 45 psig.

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

Leakage rate acceptance criteria are:

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

b. Air lock testing acceptance criteria are:

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

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

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

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 243 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

AND

AMENDMENT NO. 240 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated December 13, 2021, as supplemented by letter dated June 20, 2022 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML21348A733 and ML22171A010, respectively), Southern Nuclear Operating Company (SNC, the licensee) requested changes to the technical specifications (TSs) for the Joseph M. Farley Nuclear Plant, Units 1 and 2 (Farley). The licensee proposed to modify TS 5.5.17, “Containment Leakage Rate Testing Program,” to update the peak calculated containment internal pressure for the design basis loss-of-coolant accident (LOCA) from 43.8 to 45 pounds per square inch gauge (psig).

The supplement dated June 20, 2022, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff’s original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on February 22, 2022, 87 FR 9652.

1.1 System Description

Each Farley unit is a three-loop Westinghouse design pressurized water reactor (PWR) with a large dry containment. The containment is a prestressed, reinforced concrete cylindrical structure with a shallow domed roof and a reinforced concrete foundation slab. The design includes a one-quarter-inch thick (1/4-inch) welded steel liner to the inside face of the concrete with the floor liner installed on top of the foundation and then covered with concrete.

The containment structure is designed to contain radioactive material that may be released from the reactor core following a design basis accident (DBA) and provides shielding from the fission products that may be inside the containment atmosphere following accident conditions. The

containment ensures that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded if a gross failure of the reactor coolant system occurs. The containment encloses the reactor, the reactor coolant systems, the steam generators, and portions of the auxiliary and engineered safeguards systems.

2.0 REGULATORY EVALUATION

The applicable regulatory requirements and guidance is provided in the following subsections.

2.1 Applicable Regulatory Requirements

In 10 CFR 50.36, “Technical specifications,” the NRC establishes its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls. The regulation does not specify the particular requirements to be included in plant’s TSs.

The regulation, 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” requires, in part, that the emergency core cooling system (ECCS) be designed such that an evaluation performed using an acceptable evaluation model demonstrates that acceptance criteria, set forth in 10 CFR 50.46(b), including peak cladding temperature, cladding oxidation, hydrogen generation, maintenance of coolable core geometry, and long-term core cooling are met for a variety of hypothetical LOCAs, including the most severe hypothetical LOCA.

The regulation 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” requires, in part, licensees to establish a program for qualifying the electric equipment important to safety. The electric equipment under the scope of this section includes safety-related equipment, non-safety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified by the safety-related equipment, and certain post-accident monitoring equipment. The regulation in 10 CFR 50.49(e)(1), “Temperature and pressure,” states that “The time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.”

The regulations in 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants" (GDC), establishes the minimum requirements for the principal design criteria for water-cooled nuclear power plants. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) important to safety. The applicable GDCs for this submittal includes:

- Criterion 16 (GDC 16), “Containment design,” requires that the reactor containment and associated systems shall be provided to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.
- Criterion 38 (GDC 38), “Containment heat removal,” requires that a system to remove heat from the reactor containment shall be provided. The system safety function shall be

to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintains them at acceptable low levels.

- Criterion 50 (GDC 50) “Containment design basis,” requires, in part, that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartment can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

The regulations in 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” provide containment leakage test requirements for primary reactor containments.

2.2 Licensing Basis Document

The Farley Updated Final Safety Analysis Report (UFSAR) (ML21313A320), Section 3.11.1, “Equipment Identification and Environmental Conditions,” states:

> Safety-related equipment which is required to function during and subsequent to a DBA, is identified in section 3.2 of the FSAR. Active pumps and valves are discussed in section 3.9 of the FSAR.
>
> The original specifications for safety-related electrical equipment which is subject to a post DBA harsh environment and required to function during and subsequent to a DBA required qualification to IEEE 323-1971. Subsequently, the Farley Nuclear Plant Environmental Qualification (EQ) Program was implemented to comply with the requirements of NRC Inspection and Enforcement Bulletin (IEB) 79-01B, NUREG-0588, Revision 1, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, and 10 CFR 50.49. Based on the dates of the Farley plant operating licenses, Unit 1 was required to comply with the requirements of IEB 79-01B, which provides the NRC Division of Operating Reactors (DOR) Guidelines, and Unit 2 was required to comply with the requirements of NUREG-0588, Category II. The requirements set forth under these programs supplement the requirements of IEEE 323-1971. After implementation of these programs, 10 CFR 50.49 was issued and mandated environmental qualification requirements for safety related electrical equipment. Regulatory Guide 1.89, Revision 1, followed and established IEEE 323-1974 as an acceptable standard to comply with the requirements of 10 CFR 50.49. The provisions of 10 CFR 50.49 waive the need to requalify components previously qualified under the DOR Guidelines or NUREG-0588 unless the components are replaced. The replacement components must comply with the provisions of 10 CFR 50.49 unless there are sound reasons to the contrary. These reasons, when required, will be documented. Accordingly, the EQ program implements the requirements of 10 CFR 50.49 as documented in the EQ master lists and the associated EQ packages. The EQ packages document which version of the IEEE-323 standard was used for the qualification.
>
> Normal operating environmental conditions are defined as conditions existing during routine plant operations. These environmental conditions, as listed in table

3.11-1, represent the normal, maximum, and minimum conditions expected during routine plant operations.

Accident environmental conditions are defined as those deviating significantly from the normal operating environmental conditions as a result of a DBA. These conditions are specified in table 3.11-1 for the postulated accident duration of 30 days. Compatibility of equipment with the specified environmental conditions is provided to fulfill the following design criteria:

A. For normal operation, systems and components required to mitigate the consequences of a DBA or to provide for safe shutdown are designed to remain functional after exposure to the environmental conditions listed in table 3.11-1. Where possible, all safety-related systems and components are designed to withstand the maximum expected 40-year[a] integrated radiation dose at their respective locations within the plant. If it cannot be assured that equipment is designed for the 40-year[a] dose, a replacement maintenance program for that equipment is established. The replacement maintenance program ensures operational integrity of the equipment throughout the life of the plant.

B. In addition to the normal operation environmental requirements given in A. above, systems and components required to mitigate the consequences of a DBA or to provide for safe shutdown of the reactor are designed to remain functional after exposure to the following environmental conditions. Qualification time is based on the operating duration following a DBA.

    1. Such components inside the containment are designed for the temperature, pressure, humidity, and chemical environment inside the containment after a design basis LOCA or main steam line break accident (MSLB).

    2. Such components inside the containment which are required after a LOCA are designed for the post-LOCA radiation dose.

    3. Such components outside the containment which are required to mitigate the consequences of a design basis LOCA are designed for the expected integrated accident radiation dose at the equipment location.

    4. Such components outside the containment are designed for the temperature, pressure, and humidity environmental conditions resulting from a postulated high energy line break (HELB) in areas where such components are located.

(a) The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. The EQ program is credited to continue to manage aging effects associated with EQ equipment for the period of extended operation (see chapter 18, subsections 18.3.1 and 18.4.4). Applicable EQ evaluations based on a 40-year design life were evaluated as time-limited aging analyses (TLAAs) for license renewal and will be revised as necessary to reflect the 60-year plant operating life before the units enter the period of extended operation.

Section 6.2.1.1.1, “Postulated Accident Conditions,” under Section 6.2.1.1, “Design Bases,” under Section 6.2.1, “Containment Functional Design,” of the Farley Updated Final Safety Analysis Report (UFSAR) (ML21313A340) states:

> The containment, in conjunction with engineered safety features, is designed to withstand the internal pressure and coincident temperature resulting from the energy release of the loss-of-coolant accident (LOCA) associated with 2831 MWt [megawatts thermal] and to limit the site boundary radiation dose to within the guidelines set forth in 10 CFR 100. The containment system functional design meets the NRC acceptance criteria contained in General Design Criteria 16 and 50 of 10 CFR Part 50.

From UFSAR Table 6.2-1, “Principal Containment Design Parameters,” the containment design pressure and temperature limits are 54 psig and 280 degrees Fahrenheit (ºF), respectively. SNC is authorized to operate Farley, Units 1 and 2, at reactor core power levels not in excess of 2821 MWt.

The UFSAR Section 6.2.2.1.1, “Containment Spray System,” states:

> The two redundant trains of the containment spray system have been designed to provide sufficient heat removal capacity to prevent exceeding containment design pressure for all piping breaks. Assuming that the water in the RWST [Refueling Water Storage Tank] is at a temperature of 110°F, the containment atmospheric heat removal capability associated with the spray from one containment spray train will initially be 2 x 10^8 Btu/h [British thermal units per hour] at initial spray actuation during the injection phase.

2.3 Applicable Regulatory Guidance

Regulatory Guide 1.89, Revision 1, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants” (ML003740271), describes a method acceptable to the NRC staff for complying with 10 CFR 50.49 with regard to qualification of electric equipment important to safety for service in nuclear power plants to ensure that the equipment can perform its safety function during and after a design-basis accident.

Inspection and Enforcement Bulletin (IEB) 79-01B, “Environmental Qualification [EQ] of Class 1E Equipment” (NRC microfiche Accession No. 7910250528), required the licensee to perform a detailed review of the environmental qualification of Class 1E electrical equipment to ensure that the equipment will function under (i.e., during and following) postulated accident conditions.

NUREG-0588, Revision 1, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment” (ML031480402), provides interim staff positions on selected areas of environmental qualification of electrical equipment including guidance on (1) how to establish environmental service conditions, (2) how to select methods which are considered appropriate for qualifying the equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation.

2.4 Proposed TS Change

The changes to the TS are indicated in **BOLD** font below.

The current TS 5.5.17 contains the sentence:

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

The revised TS 5.5.17 sentence would state:

> The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is **45** psig.

3.0 TECHNICAL EVALUATION

3.1 LOCA Mass and Energy Release Analysis

The current Farley LOCA containment mass and energy (M&E) release analysis uses the NRC-approved methodology documented in Westinghouse topical report WCAP-10325-P-A, "Westinghouse LOCA Mass and Energy [M&E] Release Model for Containment Design March 1979 Version," May 1983 (ML080640615 – proprietary/non-public). The properties of the reactor coolant system (RCS) material used in WCAP-10325-P-A were found to have non-conservatively lower values compared to the current American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The justification for the continued use of the previously approved LOCA M&E analysis methodology contained in WCAP-10325-P-A is provided in PWROG-17034-P-A, Rev 0 (ML20083K090).

From 2006 until 2014, Westinghouse issued Nuclear Safety Advisory Letters (NSALs) 06-6 (ML22195A159), NSAL 11-5 (ML13239A479), and NSAL 14-2 (ML22195A177) reporting errors that impact the M&E release and consequently the LOCA containment pressure, vapor temperature, and sump temperature response analyses. The extent to which each NSAL affects Farley is described in Section 3.2 of the enclosure to the letter dated December 13, 2021. The licensee proposed updated LOCA M&E release analysis and the associated containment response analysis. The licensee proposed a revised value of containment internal pressure or the integrated leak rate test (ILRT) pressure, P_a, as result of the updated analysis.

SNC used the current methodology based on the NRC-approved WCAP-10325-P-A for the proposed LOCA M&E release analysis. The M&E data was generated using inputs that addressed the errors reported in NSALs 06-6, 11-5, and 14-2.

Consistent with the analysis of record (AOR) documented in Section 6.2, "Containment Systems," of the Farley UFSAR, the licensee reanalyzed the following three double-ended guillotine breaks in the RCS that are limiting because of the large mass flowrates from these breaks during the LOCA blowdown phase:

- hot leg break between reactor vessel and steam generator (SG), also called double-ended hot leg (DEHL) break,

- cold leg discharge break between reactor coolant pump (RCP) and the reactor vessel, and

- pump suction break between the SG and the RCP, also called double-ended pump suction (DEPS) break.

Among these break locations, the licensee determined that the DEHL and the DEPS break generated the most limiting short- and long-term peak containment conditions, respectively. The licensee calculated the M&E releases assuming minimum safeguards (safety injection (SI)) flow based on the postulated single failure of an emergency diesel generator for the DEPS case as it results in the loss of one train of safeguards equipment. The licensee did not reanalyze the cold leg discharge break because it was not limiting.

SNC used American Nuclear Society (ANS) Standard 5.1, "American National Standard for Decay Heat Power in Light Water Reactors," August 1979 and applied two-sigma uncertainty (two time the standard deviation) to conservatively calculate the decay heat energy in the LOCA M&E release model. The licensee listed key assumptions and inputs in its letter dated December 13, 2021, to ensure conservative calculation of M&E release and stated that these assumptions remain unchanged from the current AOR.

SNC listed the inputs that were updated in the computer codes used for the LOCA M&E release calculation to correct the errors discussed in the NSALs. The changes include treatment of inactive metal mass, turning on the drift flux model and break flow with inertia model in SATAN-VI code which is a code in WCAP-10325-P-A methodology for the calculation of M&E release during the LOCA blowdown phase. The licensee stated that the drift flux model and the break flow with inertia model have been previously reviewed and approved for use within the conditions and limitations specified by the NRC. The licensee used the RWST maximum temperature of 110°F, which is consistent with UFSAR Section 6.2.2.1.1.

SNC used the NRC-approved methodology in WCAP-10325-P-A for the LOCA M&E release after correcting errors reported in NSALs 06-6, 11-5, and 14-2, and the NRC-accepted Generation of Thermal Hydraulic Information for Containments (GOTHIC) Code for the containment response analyses for determining the revised containment pressure and temperature response, and the peak containment internal pressure, P_a.

The NRC staff reviewed the licensee's proposed LOCA M&E release analysis and found that the assumptions that the licensee applied to the new M&E analysis in the letter dated December 13, 2021, are consistent with the values provided in Section 6.2.1 of the Farley UFSAR, including the analyzed core power of 2830.5 MWt from UFSAR Table 6.2-7. The NRC staff finds the use of the RWST maximum temperature of 110°F RWST temperature is consistent with the containment response analysis presented in Farley UFSAR Section 6.2.2.1.1. SNC is authorized to operate Farley, Units 1 and 2, at steady state reactor core power levels not in excess of 2821 MWt.

SNC calculated the M&E releases for the blowdown, reflood, and post-reflood phases of the LOCA for the DEPS cases, considering the minimum SI. The DEHL case calculated the M&E releases only for the blowdown phase. The licensee used the M&E release data as an input to the containment response analysis to determine the peak containment pressure and vapor temperature.

The NRC staff reviewed the DEHL break case and notes that the licensee calculated M&E releases for the blowdown phase only. The generic sensitivity study documented in WCAP-10325-P-A, Section 3.3, provides sensitivity of M&E release to the break location. The study notes that the DEHL break during the blowdown phase released the highest M&E

compared to the reflood, and post-reflood phases and, therefore, generated the highest peak containment pressure. The NRC staff also notes that for the post-blowdown phase, although the emergency core cooling system (ECCS) flooding rate is significant, the M&E released from the break is minimal because the large flow resistance of the affected SG loop's RCP plus the SG tube flow resistance limiting the flow through the SG side of the break. Through its review, the NRC finds that analyzing the M&E releases and the containment response for only the DEHL blowdown phase is acceptable based on the generic evaluation provided in the NRC-approved methodology WCAP-10325-P-A.

The resulting maximum peak pressure of 44.8 psig occurs for the DEPS case which is greater than the current TS 5.5.17 value of 43.8 psig for P_a but remains below the design limit of 54 psig. Similarly, the calculated peak containment vapor temperature 264°F is bounded by the design limit of 280°F.

The NRC staff reviewed the licensee’s assumption of minimum SI flow based on single failure of an emergency diesel generator for the DEPS case and found it acceptable, because it is the limiting analysis case.

The NRC staff finds the LOCA M&E release analysis acceptable because the licensee applied the currently used NRC-approved WCAP-10325-P-A methodology while addressing the errors specified in the NSALs listed above.

3.2 LOCA Containment Response Analysis

SNC stated that the proposed containment pressure and temperature response analyses to determine the limiting LOCA were performed using version 8.1 of the computer code GOTHIC. As stated in UFSAR Section 6.2.1, the licensee used NRC-accepted GOTHIC Version 6.0 for the AOR containment response analysis. The licensee stated that the change in code version does not result in a numerically significant departure from the previous analysis and does not yield a benefit to the containment peak pressure and temperature results.

Since GOTHIC Version 6.0 is the NRC-accepted code for containment response analysis for Farley, the NRC staff requested the licensee confirm that using the GOTHIC Version 8.1 does not result in a numerically significant departure from the AOR results.

In the supplement dated June 20, 2022, SNC stated that it has performed a thorough benchmark between version 6.0 and version 8.1 of the GOTHIC Code utilizing the most limiting case. The benchmark for LOCA was performed ahead of the NSAL re-analysis using the AOR DEPS leg (minimum SI) case. The licensee presented graphs for the benchmarks performed for the containment pressure, containment temperature and the sump temperature for this case.

SNC stated that none of the changes which impacted the containment model yielded a numerically significant difference in the reported results. The licensee noted that version 8.1 of the GOTHIC code calculated unreasonably high sump water temperatures in the very beginning of the transient because the AOR did not model any initial liquid volume in the sump. The licensee stated that these high sump water temperatures at the beginning of the transient can be alleviated by realistically initializing the model with a small amount of water in the sump. The NRC staff finds this acceptable because the resulting sump water temperature will be less with an initial water volume in the sump as compared to an unrealistic case with no initial water volume in the sump.

Based on the benchmark between of the two GOTHIC Code versions performed by licensee and the results from the benchmark presented in the supplement dated June 20, 2022, the NRC staff finds the licensee's assertion that the change from Version 6.0 to Version 8.1 of the GOTHIC Code does not result in a numerically significant departure from the previous analysis to be acceptable.

SNC presented results for the DEHL break and minimum SI DEPS break scenarios in the letter dated December 13, 2021. The initial conditions used in the analysis are also presented in the letter. The licensee stated that the initial conditions are consistent with the AOR. The licensee stated that the input for post-LOCA service water pond temperature calculation is revised and a bounding value of 107°F is used for the entire LOCA transient as input into the residual heat removal/component cooling water model in the containment evaluation model. The licensee also updated the containment fan cooler (CFC) performance curves to accommodate the change in the service water temperature. The licensee performed a new analysis of the containment fan cooler to regenerate the CFC performance curve at a service water temperature of 97.3°F to use in the first hour of accident and created a CFC performance curve at a service water temperature of 107°F to use for the remaining portion of the accident. The licensee conservatively increased the CFC delay time from 92 to 115 seconds for the analysis.

Farley UFSAR Table 9.2-8, "Component Cooling Water/Service Water Temperatures (°F)," lists service water inlet temperatures as 97.3°F for the LOCA injection and LOCA recirculation phases. The peak containment pressure and temperature occur within the first hour as presented in the results in the letter dated December 13, 2021, for the proposed containment analysis. The NRC staff finds use of the AOR value of 97.3°F for the first hour for CFC performance curve and updating to a conservative value of 107°F for the remaining portion of the transient acceptable as the value used for the first hour is based on the 97.3°F from the existing AOR and a conservative temperature of 107°F is used after the first hour.

The licensee stated the changes to service water pond temperature and CFC performance curve inputs are conservative and that the changes were evaluated under SNC's 10 CFR 50.59 process and found not to need prior NRC approval. The NRC staff agrees that the proposed input changes by the licensee are conservative and are therefore acceptable.

Table 1 below shows the results of containment pressure and temperature provided by the licensee in the letter dated December 13, 2021. The table below shows that the peak containment pressures and the peak temperatures for the proposed AOR are below their respective design limits. Based on the results of the updated AOR, the NRC staff finds the proposed change to the peak calculated containment internal pressure for the design basis LOCA to be acceptable.

Table 1 – Containment Peak Pressure and Temperature Results

Case	Proposed AOR	AOR	Design Limit
DEPS Peak Pressure (psig)	44.86	43.8	54
DEPS Peak Vapor Temperature (°F)	264.39	263	280
DEHL Peak Pressure (psig)	43.42	43.6	54
DEHL Peak Vapor Temperature (°F)	263.57	264	280

The NRC staff finds the proposed TS 5.5.17 change acceptable because the containment design conditions important to safety are not exceeded during a postulated LOCA. Based on

this, the NRC staff concludes that the proposed change meets the requirements of 10 CFR Part 50, Appendix A, GDC 16.

3.3 LOCA Sump Temperature Response and Net Positive Suction Head (NPSH) Analysis

Farley UFSAR Section 6.2.1.3.6, "Long-Term Containment Performance," states that during the LOCA sump water recirculation mode, a third pressure peak occurs due to steam evolution from the reactor core because of the boiloff of the hotter core injection water. The UFSAR lists the maximum sump water temperature of 260°F, which occurs at 1,252 seconds from LOCA initiation. In the supplement dated June 20, 2022, SNC provided the following information on the sump water temperature response and net positive suction head (NPSH) of the pumps that draw water from the sump during LOCA recirculation mode:

- Sump water temperature profile for the minimum SI DEPS break case showing a peak temperature of 259°F occurring at 1,370 seconds from LOCA initiation, switchover to sump recirculation mode at 2,139 seconds, and a peak sump water temperature of 253°F during the sump recirculation mode.

- Minimum NPSH available (NPSHA) for the pumps that draw water from the sump during the recirculation mode with a minimum NPSH margin (NPSHA minus NPSH required (NPSHR)) of 1.1 feet occurring at the sump water temperature range of 212-291°F.

- Maximum NPSHR pumps that draw water from the sump during the LOCA recirculation phase is 18 feet at flow rates at 4,500 gallons per minute (gpm) for the residual heat removal pump and 3,400 gpm for the containment spray pump.

- Containment accident pressure (CAP) is not used in the calculation of minimum NPSHA for the pumps that draw water from the sump.

- NPSH analysis used the pressure loss in the current sump strainer design installed to address Generic Safety Issue (GSI)-191. This NPSH analysis does not impact Surveillance Requirement (SR) 3.6.10.1.

- The maximum sump water temperature is below the current value in the UFSAR and there is positive NPSH margin for the pumps that draw water from the sump during the LOCA sump recirculation mode. The NPSH analysis is performed using the pressure loss in the current sump strainer design installed to address GSI-191. This NPSH analysis does not impact Surveillance Requirement (SR) 3.6.10.1.

Section 6.2.1.3.3, "Containment Pressure Transient Analysis," of the Farley UFSAR states that maximum containment wall liner plate is approximately 250°F, which is below the peak vapor temperature value in the AOR. Since the proposed containment peak vapor temperature is below the design limit, the NRC staff concludes that the containment wall temperatures will remain below its design limit of 280°F.

Since the peak containment pressure and temperature for the DBA are bounded by their design limits and there is positive NPSH margin without considering the CAP for pumps that draw water from the sump during LOCA sump recirculation mode, the NRC staff finds the results for proposed analysis acceptable.

Based on its review of the information provided in the letter dated December 13, 2021, as supplemented by letter dated June 20, 2022, the NRC staff finds the LOCA sump temperature response and NPSH analysis acceptable, because the licensee's sump temperature response and NPSH analysis remain consistent with the requirements of GDC 38 in that the containment pressure and temperature following any design basis LOCA will be maintained within the acceptable limits.

The NRC staff finds the proposed TS 5.5.17 change acceptable, because the containment heat removal system is designed such that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from DBA, in compliance with the intent of 10 CFR Part 50, Appendix A, GDC 50.

3.4 Plant Impacts of Change in Containment Pressure

SNC proposed to revise the peak calculated containment internal pressure for the design basis LOCA, described in TS 5.5.17, from 43.8 psig to 45 psig. The containment design limit is 54 psig.

3.4.1 10 CFR 50, Appendix J Program Review

In its letter dated December 13, 2021, SNC stated:

> Upon NRC approval, this increase in P_a would be reflected in the 10 CFR Part 50, Appendix J containment leak rate testing procedures.

Because the licensee must meet the regulation 10 CFR 50, Appendix J for Type A, Type B, and Type C tests, and these tests will remain consistent with and within the constraints of ANS/ANSI 56.8-2002, the NRC staff finds the proposed change to TS 5.5.17 "Containment Leakage Rate Testing Program," acceptable.

3.4.2 Dose Consequence

In the letter dated December 13, 2021, the licensee stated that the change in P_a does not affect the offsite radiological consequences of a LOCA as previously analyzed in the Farley UFSAR Section 15.4.1.7, "Environmental Consequences of Postulated Loss-of-Coolant Accident."

The NRC staff reviewed the letter dated December 13, 2021, and the Farley UFSAR Section 15.4.1.7, and determined that the maximum allowable containment leakage analysis for LOCA is not changed due to the licensee's proposed change in P_a. Therefore, the NRC staff concludes that the increase in the P_a does not impact the offsite radiological consequences of the LOCA accident analysis.

In the letter dated December 13, 2021, the licensee stated that the calculated control room operator dose during a LOCA is dependent on the maximum allowable containment atmosphere leakage rate and is unaffected by the change in P_a. The NRC staff reviewed the letter dated December 13, 2021, and determined that since the maximum allowable containment leakage rate is unaffected by the change in P_a, dose to the control room operators is not affected by the licensee's proposed change.

The licensee stated that its proposed change in P_a does impact the post-LOCA sump temperature profile due to the service water temperature input change. The licensee stated that the impact of the change in the sump water temperature profile was evaluated in a revision to the LOCA dose consequence analysis and that the results negligibly increased due to higher sump water temperatures later in the accident.

The NRC staff reviewed the licensee's letter dated December 13, 2021, and the NRC staff concludes that the LOCA dose consequence analysis is not affected by the proposed change of P_a and that the increase in sump temperature as a result of the service water input change does not impact the offsite consequences of the LOCA analysis.

3.4.3 EQ of Electrical Equipment Important to Safety

The NRC staff reviewed the letter dated December 13, 2021, to determine the impact of the proposed change in P_a on the EQ of electrical equipment. In the letter dated December 13, 2021, SNC stated that it performed a review of the effects of the proposed change in mass and energy release on the EQ of equipment. The licensee provided figures (Figure 5, "Environmental Qualification and LOCA Containment Pressure Profiles," and Figure 6, "Environmental Qualification and LOCA Containment Temperature Profiles") that showed the revised pressure and temperature profiles inside containment during a design-basis LOCA and the representative EQ test profiles.

SNC also stated that the temperature and pressure margins in the Farley EQ program are applied in accordance with 10 CFR 50.49 and Regulatory Guide 1.89. These temperature and pressure margins are in addition to the EQ bounding temperature and pressure profiles for equipment qualified in accordance with 10 CFR 50.49. The licensee stated that EQ program documents have been revised based on these new calculated pressure and temperature profiles with no negative impacts.

Based on its review of the letter dated December 13, 2021, the NRC staff finds that the licensee has shown that the revised containment pressure and temperature values remain bounded by the profiles used to qualify equipment inside containment. Furthermore, SNC stated that the proposed change in calculated peak containment internal pressure for the design-basis LOCA does not result in any impact on any area of the plant outside containment. Given no changes to environmental conditions outside containment, the NRC staff finds that equipment located outside containment should remain qualified under the revised design conditions.

Based on its review of the information in the letter dated December 13, 2021, the NRC staff finds that the environmental parameters (i.e., temperature and pressure) remain bounded by the existing EQ of electrical equipment due to the proposed changes and that the proposed changes will have no adverse impact on the Farley EQ program or its ability to continue to meet the requirements of 10 CFR 50.49, and Section 3.11.1 of the Farley UFSAR.

3.5 Evaluation of TS Change

Based on the evaluations presented in Sections 3.1 through 3.4 above, the NRC staff finds the proposed TS 5.5.17 change acceptable for the following reasons:

- The containment design conditions important to safety are not exceeded during a postulated LOCA. Based on this, the NRC staff concludes that the proposed change meets the requirements of 10 CFR Part 50, Appendix A, GDC 16.

- The containment heat removal system would reduce the containment pressure and temperature with the other associated systems, following DBA and would maintain them at acceptable levels, thus meeting the intent of 10 CFR Part 50 Appendix A, GDC 38.

- The containment heat removal system is designed such that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from DBA, in compliance with the intent of 10 CFR Part 50, Appendix A, GDC 50.

- The increase in containment pressure from 43.8 psig to 45 psig remains in TS 5.5.17 and continues to meet 10 CFR 50.36.

- The increase in containment pressure from 43.8 psig to 45 psig continues to meet 10 CFR 50.46.

- Since SNC is required to meet 10 CFR 50, Appendix J for Type A, Type B, and Type C testing consistent within the constraints of ANS/ANSI 56.8-2002, the NRC staff finds the proposed change acceptable.

- Since the environmental parameters (i.e., temperature and pressure) remain bounded by the existing EQ of electrical equipment due to the proposed changes, the NRC staff concludes that the proposed change will have no adverse impact on the Farley EQ program or its ability to continue to meet the requirements of 10 CFR 50.49, and its licensing basis in UFSAR Section 3.11.1.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendments on September 20, 2022. On September 21, 2022, the State official confirmed that the State of Alabama had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes 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 (87 FR 9652, February 22, 2022), 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 of Issuance: November 1, 2022

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT NOS. 243 AND 240 TO REVISE TECHNICAL SPECIFICATION 5.5.17, "CONTAINMENT LEAKAGE RATE TESTING PROGRAM," TO INCREASE CALCULATED PEAK CONTAINMENT PRESSURE (EPID L-2021-LLA-229) DATED NOVEMBER 1, 2022

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