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

August 29, 2024

Jamie M. Coleman
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. 250 AND 247, REGARDING LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION 3.6.5, “CONTAINMENT AIR TEMPERATURE,” ACTIONS (EPID L-2024-LLA-0098)

Dear Jamie Coleman:

The Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 250 to Renewed Facility Operating License No. NPF-2 and Amendment No. 247 to Renewed Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant (Farley), Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated July 18, 2024, as supplemented by your letters dated August 16 and August 27, 2024.

The amendments revise TS 3.6.5, “Containment Air Temperature,” Actions upon exceeding the TS 3.6.5 Limiting Condition for Operation (LCO) limit of containment average air temperature ≤120 degrees Fahrenheit (°F), and removes an expired LCO Note. Specifically, the proposed amendment would relocate existing TS 3.6.5 Required Action A.1 as proposed Required Action A.3 and revise the associated Completion Time and add a clarifying Completion Time Note, add proposed Required Actions and Completion Times for A.1 and A.2, for Condition A when containment average air temperature is not within the LCO limit.

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

Enclosure 3 to this letter contains proprietary information. When separated from Enclosure 3, this document is DECONTROLLED.

If you have questions, you can contact me at 301-415-2258 or at Zachary.Turner@nrc.gov.

Sincerely,

/RA/

Zachary M. Turner, 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. 250 to NPF-2
2. Amendment No. 247 to NPF-8
3. Proprietary Safety Evaluation
4. Nonproprietary Safety Evaluation

cc: without Enclosure 3 Listserv

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 250
Renewed License No. NPF-2

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

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated July 18, 2024, as supplemented by letters dated August 16, 2024 and August 27, 2024, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

    (2) Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 250, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to the Renewed Facility
Operating License and Technical
Specifications

Date of Issuance: August 29, 2024

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 247
Renewed License No. NPF-8

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

   A. The application for amendment by Southern Nuclear Operating Company, Inc. (Southern Nuclear), dated July 18, 2024, as supplemented by letters dated August 16, 2024 and August 27, 2024, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 247, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to the Renewed Facility
Operating License and Technical
Specifications

Date of Issuance: August 29, 2024

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

LICENSE AMENDMENT NO. 250

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348,

AND LICENSE AMENDMENT NO. 247

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
3.6.5-1	3.6.5-1
---------	3.6.5-2

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 250, 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 byproducts, 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 incorporate 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. 247, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Delete per Amendment 144

(4) Delete Per Amendment 149

(5) Delete per Amend 144

3.6 CONTAINMENT SYSTEMS

3.6.5 Containment Air Temperature

LCO 3.6.5 Containment average air temperature shall be ≤ 120°F.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment average air temperature not within limit.	A.1 Verify containment average air temperature ≤ 122°F.	8 hours AND Once per 8 hours thereafter
	AND	
	A.2 Verify refueling water storage tank temperature ≤ 100°F.	8 hours AND Once per 8 hours thereafter
	AND	
	A.3 Restore containment average air temperature to within limit.	----------NOTE---------- Not to exceed 7 days cumulative in calendar year ---------------------------- 7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.5.1	Verify containment average air temperature is within limit.	In accordance with the Surveillance Frequency Control Program

ENCLOSURE 4

(NONPROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 250 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

AND

AMENDMENT NO. 247 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

This document contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390.

Redacted information is identified by blank space enclosed within double brackets. [[ ]].

1.0 INTRODUCTION

By application dated July 18, 2024, (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML24201A107, non-publicly available; ML24201A108, publicly available), as supplemented by letters dated August 16 (ML24229A244, non-publicly available; ML24229A245, publicly available) and August 27, 2024 (ML24240A081), Southern Nuclear Operating Company, Inc. (SNC, the licensee) requested changes to the Technical Specifications (TSs) for Renewed Facility Operating License Nos. NPF-2 and NPF-8 for Joseph M. Farley Nuclear Plant (Farley), Units 1 and 2, respectively.

The proposed amendments would revise the Farley renewed facility operating licenses and, Appendix A, TS 3.6.5, "Containment Air Temperature," Actions upon exceeding the TS 3.6.5 Limiting Condition for Operation (LCO) limit that containment average air temperature shall be ≤120 degrees °F (less than or equal to 120 degrees Fahrenheit) and remove an expired LCO Note. Specifically, the proposed amendment would relocate existing TS 3.6.5 Required Action A.1 as proposed Required Action A.3 and revise the associated Completion Time and add a clarifying Completion Time Note, add proposed Required Actions and Completion Times for A.1 and A.2, for Condition A when containment average air temperature is not within the LCO limit.

The U.S. Nuclear Regulatory Commission (NRC, the Commission) staff participated in a regulatory audit from July 30 through August 16, 2024 to ascertain the information needed to support its review and determine request for additional information, as needed. By letter dated August 16, 2024, the licensee supplemented its request and responded to the audit providing additional information associated with the NRC questions discussed in the audit. The supplemental letter dated August 16, 2024, provided additional information that clarified the application. The supplemental letter dated August 27, 2024, revised the proposed TS 3.6.5 Completion Time A.3 format to add a Note with clarifying information. These supplemental letters did not expand the scope of the application as originally noticed and did not change NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 29, 2024 (89 FR 60930). By letter dated August 22, 2024, the NRC staff issued its audit summary (ML24226B211).

1.1 System Description

In Section 2.1 of the Enclosure to its submittal dated July 18, 2024, the licensee stated:

> The containment is a prestressed, reinforced concrete cylindrical structure with a shallow domed roof and a reinforced concrete foundation slab. A 1/4-in.-thick welded steel liner is attached to the inside face of the concrete. The floor liner is installed on top of the foundation slab and is then covered with concrete. The containment completely encloses the reactor, the reactor coolant systems, the steam generators, and portions of the auxiliary and engineered safeguards systems. It ensures that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded even if gross failure of the reactor coolant system occurs. The structure is designed to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). Additionally, this structure provides shielding from the fission products that may be present in the containment atmosphere following accident conditions.

As described in Final Safety Analysis Report (FSAR) subsection 6.2.2, three systems are provided to reduce containment atmosphere temperature and pressure and/or to remove heat from the containment under post-accident conditions. These are the low-head safety injection/residual heat removal system, the containment spray system, and the containment cooling system. 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. The containment cooling system has been designed to remove heat which will be released to the containment atmosphere during any Main Steam Line Break (MSLB) or Loss of Coolant Accident (LOCA) up to and including the double-ended rupture of the largest system pipe. This is accomplished by one of four containment air coolers.

As described in FSAR subsection 6.2.1.3.3, Containment Pressure Transient Analysis, and shown in Table 6.2-3, Initial Conditions for Pressure Analysis, the analyses for containment pressure assumed an initial containment temperature of 127°F while the accumulator is assumed at 120°F and the refueling water storage tank (RWST) is assumed at 110°F.

1.2 Proposed Change

Current TS 3.6.5 LCO requires that "[c]ontainment average air temperature shall be ≤ 120°F." Additionally, there is an expired TS LCO Note that reads, "[c]ontainment average air temperature shall be ≤ 122°F until 0600 hours on September 9, 2023." If LCO 3.6.5 is not met, under Condition A, the plant has 8 hours to restore the temperature to within limits. If Condition A, Required Action A.1 and Completion Time is not met, then Condition B, Required Action and Completion Time B.1 requires the plant to be in Mode 3 within 6 hours, and Required Action and Completion Time B.2 requires the plant to be in Mode 5 within 36 hours.

The licensee proposes to relocate existing TS 3.6.5 Required Action A.1 as Required Action A.3 and revise the associated Completion Time and add a clarifying Completion Time Note, add Required Actions and Completion Times A.1 and A.2, as well as remove the expired TS 3.6.5 LCO Note described above. Specifically, the proposed change would modify the TS 3.6.5 Required Actions if containment average air temperature exceeds the LCO of 120°F to allow continued operation for up to seven days (not to exceed seven days cumulative in calendar year), provided that the containment average air temperature does not exceed 122°F (verified within eight hours of containment average air temperature exceeding 120°F, and once per eight hours thereafter), and that RWST temperature remains less than or equal to 100°F (verified within eight hours of containment average air temperature exceeding 120°F, and once per eight hours thereafter).

2.0 REGULATORY EVALUATION

The applicable regulatory requirements and guidance are provided in the following subsections:

2.1 Applicable Regulatory Requirements

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include TSs as part of the application. The NRC's regulatory requirements related to the content of the TSs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." Pursuant to 10 CFR 50.36, each operating license issued by the Commission includes TSs and includes items in the following categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) LCOs, (3) SRs, (4) design features, (5) administrative controls, (6) decommissioning, (7) initial notification, and (8) written reports.

10 CFR Part 50 provides the general provisions for "Domestic Licensing of Production and Utilization Facilities." Under 10 CFR 50.90, whenever a holder of a license wishes to amend the license, including technical specifications in the license, an application for amendment must be filed, fully describing the changes desired. Under 10 CFR 50.92(a), determinations on whether to grant an applied-for license amendment are to be guided by the considerations that govern the issuance of initial licenses or construction permits to the extent applicable and appropriate. Both the common standards in 10 CFR 50.40(a), and those specifically for issuance of operating licenses in 10 CFR 50.57(a)(3), provide that there must be reasonable assurance that the activities at issue will not endanger the health and safety of the public.

The NRC staff has identified the following applicable sections within 10 CFR Part 50 for the staff's review of the licensee's application to revise TS 3.6.5, "Containment Air Temperature," Required Actions:

- 10 CFR 50.36, *Technical specifications,*" paragraph (c)(2), "Limiting conditions for operation"

- 10 CFR 50.46, "*Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors*"

- 10 CFR 50.49, "*Environmental qualification of electric equipment important to safety for nuclear power plants*"

The regulations in 10 CFR 50, Appendix A, "*General Design Criteria [GDC] for Nuclear Power Plants*", 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 4* [*GDC* 4], "Environmental and dynamic effects design bases"
- *Criterion 16* [GDC 16], "Containment design"
- *Criterion 38* [GDC 38], "Containment heat removal"
- *Criterion 50* [GDC 50] "Containment design basis"

2.2 Licensing Basis Document

The Farley FSAR Section 3.11, "Environmental Design of Mechanical and Electrical equipment," states:

> 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 [Institute of Electrical and Electronics Engineers] IEEE 323-1971, [IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear

Power Generating Stations.] 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 [Environmental Qualification of Class IE Equipment,] NUREG-0588, Revision 1, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, [ML031480402] 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 [IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations.] 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, [Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants, ML003140271] followed and established IEEE 323-1974, [IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations] 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.

2.3 Applicable Regulatory Guidance

NRC regulatory guides (RGs) provide a method acceptable to the NRC staff for complying with the regulations. RGs often describe industrial standards that are acceptable to the NRC in satisfying applicable regulations. The NRC also issues guidance in certain technical areas and review guidance for the NRC staff's use in evaluating licensee submittals. The NRC staff considered the following guidance during its review of the proposed changes:

- RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Revision 1, June 1984 (ML003740271)
- RG 1.155, "Station Blackout," Revision 0, August 1988 (ML003740034)
- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3, January 2018 (ML17317A256)
- RG 1.177, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Revision 2, January 2021 (ML20164A034)
- NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1, July 1981 (ML031480402)

- NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition” (SRP), Section 3.11, “Environmental Qualification of Mechanical and Electrical Equipment, Revision 3, March 2007 (ML070720037)
- Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” September 1996 (ML031110021)
- Inspection and Enforcement Bulletin 79-01B, “Environmental Qualification [EQ] of Class 1E Equipment”, January 1980 (NRC Microfiche 7910250528)

3.0 TECHNICAL EVALUATION

The NRC staff evaluated the amendment request to determine whether the proposed changes are consistent with the regulatory requirements and the plant-specific design and licensing basis. The NRC staff evaluations are detailed in the following subsections:

3.1 Containment Design Basis Analysis (DBA)

The containment average air temperature limit is an initial condition for the DBA analysis to ensure that the containment spray (CS) and containment cooling systems are capable of maintaining the total amount of energy in the containment within the specified limits and are able to remove heat from the containment during post-accident conditions. It is an important consideration in establishing the containment environmental qualification (EQ) operating envelope for both pressure and temperature.

The LCO 3.6.5 for containment average air temperature limit ensures that the initial conditions assumed for the containment response for a main steam line break (MSLB) or loss-of-coolant accident (LOCA) event are not violated during plant operation. During post-accident conditions, the total amount of energy removed from the containment by the CS and the containment cooling system is dependent upon the energy released into the containment during the postulated event. Exceeding the containment design limits for pressure and temperature may result in leakage rates greater than that assumed for the accident analysis. The licensee is proposing to change the TS Actions to allow continued operation up to seven days (not to exceed seven days cumulative in calendar year) above the LCO 3.6.5 limit of ≤120°F, as long as the containment temperature remains ≤122°F and the refueling water storage tank (RWST) temperature is ≤100°F. The containment DBA evaluations performed by the licensee assessed the impact of these changes.

3.1.1 LOCA Mass and Energy Release

For the LOCA Mass and Energy (M&E) release inside the containment, the licensee used NRC-approved analytical methods in WCAP-10325-P-A, “Westinghouse LOCA Mass and Energy Release Model for Containment Design March 1979 Version,” May 1983 (ML080640615 non-publicly available), corrected based on the Westinghouse Nuclear Safety Advisory Letters (NSALs) 06-6, 11-5, and 14-2 (ML22195A159, ML13239A479, and ML22195A177, respectively). The licensee considered the following limiting postulated accidents:

- Double-ended pump suction guillotine (DEPSG), minimum engineered safety feature (ESF).

- Double-ended hot leg guillotine (DEHLG), blowdown phase only.

- Spectrum of MSLBs.

The revised M&E release analysis is based on an accumulator temperature increased conservatively from an input value of 120°F in the current analysis to 124°F. In addition, the LOCA M&E release is based on a change in the RWST temperature from its current value of 110°F to a realistic value of 100°F, based on operational data provided by the licensee indicating that the RWST temperature is typically less than 95°F, to provide additional margin for the M&E release. The proposed change requires the licensee to verify the RWST temperature ≤100°F when TS 3.6.5 Condition A is entered. Therefore, the NRC staff finds the licensee's proposed change of the analysis input for RWST temperature to 100°F acceptable because it is the revised TS limit for the relevant analysis when TS 3.6.5 Condition A is entered, and is conservative based on the RWST operational data for temperature.

3.1.2 <u>LOCA Containment Pressure and Temperature Response Analysis</u>

For the containment response analysis, the licensee used the Generation of Thermal-Hydraulic Information for Containment (GOTHIC), version 8.1 code. NRC staff has previously approved Farley, Units 1 and 2, to use the GOTHIC version 8.1 code via the issuance of Amendment Nos. 243 and 240 (ML22263A225), respectively, in response to SNC's license amendment request (LAR) dated December 13, 2021 (ML21348A733), as supplemented June 20, 2022 (ML22171A010). The licensee stated in the LAR to revise TS 3.6.5 that the containment response for the DEPSG with minimum ESF case bounds the response for the DEHLG case. As stated in the FSAR Table 6.2-3, the current MSLB containment response is based on initial containment air temperature of 127°F, therefore the MSLB accident does not require evaluation because its current analysis results are already bounded for a containment average air temperature of 124°F, as confirmed by reviewing information pertaining to the GOTHIC code during the audit.

The licensee's containment response analysis for the DEPSG case used a conservative initial containment air temperature of 127°F, and the M&E release is based on an accumulator temperature of 124°F, as stated in the licensee's submittal and confirmed by reviewing information pertaining to M&E release calculations during the audit. In the licensee's supplement dated August 16, 2024, the licensee provided the following revised analysis results and the analysis of record (AOR) values:

- Maximum pressure = 45.02 psig, AOR value =44.83 psig

- Maximum vapor temperature = 264.64°F, AOR value = 264.36°F

- Maximum sump temperature = 259°F, AOR value = 258.66°F

- Maximum sump temperature during LOCA recirculation phase = 251°F

The NRC staff finds the containment pressure and temperature response analysis and results acceptable because the licensee used the NRC-accepted GOTHIC methodology using a conservative M&E release for a limiting break and conservative inputs for the initial accumulator water temperature (124°F) and containment air temperature (127°F), which bound the proposed TS value 122°F. The licensee stated that the current TS 5.5.17 integrated leak rate test (ILRT) pressure 'Pa' value 45 psig is not affected because the rounding of maximum containment pressure 45.02 psig to 45 psig is appropriate. The NRC staff finds the proposed rounding acceptable because it is insignificant compared to measurement uncertainties.

3.1.3 Net Positive Suction Head (NPSH) Evaluation

The NPSH analysis consists of an evaluation of the NPSH margin for the residual heat removal (RHR) system and CS system pumps that draw water from the containment sump during the LOCA recirculation phase. The equation for NPSH Available (NPSHA) is as follows:

$$NPSHA = \frac{144}{\rho}\left(P_{ctm} - P_{vapor}\right) + h_{sump} - h_{pump} - h_{loss}$$

where,

P_{ctm} = Containment pressure (psia)

P_{vapor}= Vapor pressure at the sump temperature (psia)

ρ = Density of the sump fluid (lbm/ft^3)

h_{sump} = Minimum sump fluid elevation (ft.)

h_{pump} = Elevation of pump suction (ft.)

h_{loss} = Suction line losses, including strainer loss (ft.)

The value of P_{vapor} in the above equation is affected because it depends on the sump water temperature response, while the remaining parameters do not change from their AOR values. The GOTHIC analysis results for the limiting DEPSG break case shows the maximum sump temperature 259°F, which is a small increase of 0.34°F from its AOR value of 258.66°F. The licensee stated that the strainer head losses are shown to decrease as the sump temperature increases above 140°F. The licensee demonstrated this in its supplement dated August 16, 2024, in which the licensee summarized the total strainer head loss for temperatures at 120°F, 140°F, 160°F, 180°F, and 212°F, and conservatively assumed the head loss to be the same at higher temperatures as that corresponding to 212°F. In the NPSHA calculation, consistent with the guidance in RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-Of-Coolant-Accident" (ML22152A114), the licensee did not credit containment accident pressure (CAP) above the vapor pressure at the sump water temperature.

The NPSH margin defined as:

$$NPSH_{margin} = NPSHA - NPSHR$$

Table 1, below, shows the values of NPSHA, NPSH Remaining (NPSHR), and the NPSH margin for the RHR and CS pumps at Farley, Units 1 and 2. This information is taken from Table 3.3-1 of the licensee’s submittal dated July 18, 2024.

Table 1: NPSHA, NPSHR, and NPSH Margin Results for RHR and CS Pumps

Pump	NPSHA (ft.)	NPSHR (ft.)	Minimum NPSH Margin (ft.)
Unit 1, A Train, RHR	20.7	18	2.7
Unit 1, B Train, RHR	19.6	18	1.6
Unit 1, A Train, CS	20.8	18	2.8
Unit 1, B Train, CS	21.2	18	3.2

Pump	NPSHA (ft.)	NPSHR (ft.)	Minimum NPSH Margin (ft.)
Unit 2, A Train, RHR	19.2	18	1.2
Unit 2, B Train, RHR	19.3	18	1.3
Unit 2, A Train, CS	20.7	18	2.7
Unit 2, B Train, CS	19.1	18	1.1

The NRC finds the licensee NPSH analysis and results showing NPSH margin for the RHR and CS pump, as indicated above in Table 1, acceptable because they are based on a conservative LOCA M&E release for the limiting break and conservative sump temperature response and the results demonstrate that there is adequate NPSH margin available.

3.1.4 Containment DBA Conclusions

The NRC staff independently evaluated the licensee's LOCA containment response analysis and notes that:

- The M&E release analysis was performed using conservative input for the accumulator temperature and is based on NRC-approved WCAP-10325-P-A methodology after correction of errors documented in Westinghouse NSALs.

- The containment pressure and temperature response was performed using conservative inputs for the initial accumulator temperature and initial containment air temperature and is based on NRC-accepted GOTHIC methodology. The current TS 5.5.17 ILRT pressure of 45 psig is not affected.

- The NPSH analysis for the RHR and CS pumps was performed with a sump temperature response that uses conservative inputs for the initial accumulator temperature and initial containment air temperature.

Based on the evaluation performed, the NRC staff finds that the 10 CFR 50, Appendix A, GDCs 16 and 50 continue to be satisfied because the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. The staff also finds that the GDC 38 continues to be satisfied because the containment heat removal systems (RHR and CS) pumps have adequate NPSH margin so that these systems will remove heat from the reactor containment and rapidly reduces, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintains them at acceptable low levels.

3.2 LOCA Evaluation

3.2.1 Large Break LOCA (LBLOCA)

The maximum accumulator temperature is an input parameter to the LBLOCA analysis. If the accumulator temperature is higher during a LBLOCA, fuel temperatures may tend to increase for a longer period, ultimately causing the peak cladding temperature (PCT) to be higher. An increase in containment air temperature will increase the accumulator temperature because the accumulators are located inside the containment. The licensee performed a study of accumulator temperature sensitivities for similar pressurized water reactor (PWR) plant designs with similar fuel assembly designs and power levels. These studies were used to predict the cladding temperature response and determine an estimated PCT effect for Farley. The licensee stated that the accumulator temperature sensitivities used were executed prior to the incorporation of modeling of fuel thermal conductivity degradation (TCD) in best-estimate LOCA analyses and fuel performance codes. In Attachment 6 of the licensee's submittal dated July 18, 2024, the licensee stated that the studies remain valid for the purpose of estimating the effect of the increase accumulator temperature range and that appropriate conservatism is applied to the estimate of effect to account for the use of pre-TCD transient results from representative plants for the Farley evaluation.

The NRC staff independently evaluated the licensee's analysis to assess the impact of a 2°F change in the accumulator temperature on LBLOCA PCT. The NRC staff evaluation focused on the change in the PCT rack-up due to the proposed action and ensuring the proposed change maintains adequate margin to the 10 CFR 50.46 PCT acceptance criterion of 2200°F.

The NRC staff review was focused on ensuring that the similarly designed plants used in the sensitivity studies (referred to as plants A and B) realistically model the Farley reactor coolant system (RCS) and that an acceptable evaluation model was used to capture the effect of the change, consistent with the requirements of 10 CFR 50.46.

In response to the NRC staff audit questions, the licensee provided additional information regarding the accumulator temperature sensitivities performed in its supplement dated August 16, 2024. In the supplement, the licensee stated that the ASTRUM evaluation model, used in the existing LBLOCA AOR for Farley, relies on a statistical sampling technique to demonstrate compliance to the 10 CFR 50.46 acceptance criteria under postulated LBLOCA conditions. The licensee further stated that the statistical sampling of the uncertainty contributors occurs simultaneously in the uncertainty analysis, leading to scatter in the analysis results when plotted as a function of a single uncertainty contributor. Thus, the licensee did not consider execution of the ASTRUM evaluation model framework to be the best option to estimate the effect of the maximum accumulator temperature increase from 120°F to 122°F. The NRC staff finds use of similar PWR plant designs to estimate the effect to be acceptable as long as the plant designs used realistically model the Farley RCS system. In Table 1-1 of the supplement dated August 16, 2024, the licensee provided comparison of input parameters pertinent to LBLOCA for reference Plants A and B to the Farley design. The parameters compared included information on physical plant parameters, plant operating conditions, fuel assembly designs, core power and peaking factors, fluid conditions, accident boundary conditions and the accumulator design information. The NRC staff notes that although Table 1-1 of the supplement to the LAR shows Farley's rated thermal power as 2,775 megawatts thermal (MW_t), the plant is actually licensed to 2821 MW_t. However, the comparisons made with Plant A and B have a power of 2,900 MW_t, which bound the Farley licensed power. The NRC staff reviewed the information and notes that while the plants used in the accumulator sensitivity study were comparable to the Farley design, there were several design variations that could lead to changes in the overall PCT.

The licensee provided details of the errors/changes to the WCOBRA/TRAC code version which was used in the ASTRUM evaluation model for the LBLOCA AOR for Farley and as well as the sensitivity studies. The changes were reported previously to the NRC as part of 10 CFR 50.46 yearly reporting. The NRC staff reviewed the changes for their potential effect on the estimate of the PCT and found the evaluation model used to be acceptable. The NRC staff notes that the LBLOCA AOR for the Farley plant as well as the sensitivity studies performed using the WCOBRA/TRAC code for similar plants did not include an estimate of the effect of explicitly accounting for TCD when determining the PCT impact associated with the accumulator temperature change. There is currently a total of 200°F worth of penalties, including a 150°F penalty for TCD, on the LBLOCA PCT rack-up listed in Table 15.4-3 of the Farley FSAR (ML23318A074). These penalties must be accounted for since they could potentially impact the initial stored energy during the transient and could lead to variations in the PCT estimates associated with the proposed increase in accumulator temperature. In its supplement dated August 16, 2024, following discussion on this topic during a regulatory audit, the licensee increased its estimate of the impact of the accumulator temperature change from 2°F to 14°F, in part to address staff questions and concerns in this analysis.

The licensee discussed 18 accumulator temperature sensitivity studies performed for the Westinghouse-fueled nuclear plants using the WCOBRA/TRAC code to further confirm the magnitude of the impact of the accumulator temperature change on the PCT. These studies included variations in the number of RCS loops, fuel assembly design, power level, and peaking factors as well as accumulator temperature ranges from approximately 40°F to 130°F. The studies showed that average impact on PCT for each 1°F change in the accumulator temperature varied from **[[**

]] The licensee estimated that the 2°F increase in the maximum accumulator temperature will have a 14°F effect on the Farley analysis PCT for LBLOCA.

The NRC staff finds the 14°F effect on the Farley analysis PCT for the 2°F increase in the maximum accumulator temperature to be acceptable because the licensee; (1) used an acceptable evaluation model, (2) used plants that realistically model the Farley RCS, and (3) applied appropriate conservatisms to the PCT impact to account for variabilities due to application of similar plant models and not accounting for the penalties on the UFSAR PCT rack-up, such as fuel TCD. The NRC staff notes that with the additional 14°F impact, the PCT rack-up is estimated to be 2048°F, which maintains sufficient margin to the 10 CFR 50.46 acceptance criterion of 2200°F.

3.2.2 Small Break LOCA

A small break LOCA (SBLOCA) for Farley is defined in Section 15.3.1.1, “Identification of Causes and Accident Description,” of the Farley FSAR as a breach in the reactor coolant pressure boundary with a total cross-sectional area less than 1.0 ft^2. An increase in the containment air temperature leads to an increase in the accumulator temperature. The licensee calculated that a 2°F increase in accumulator temperature leads to an enthalpy increase of ~2 Btu/lbm [~2.23%]), which corresponds to a small reduction in total energy removal capability of the accumulator fluid. The licensee concluded that the proposed temperature increase will have a negligible impact on the SBLOCA AOR due to the accumulator injection characteristics remaining unaffected as well as the small impact of stored energy on the cladding heatup transient for a small break.

In its supplement dated August 16, 2024, the licensee provided additional details to support this conclusion regarding SBLOCA. In the supplement, the license provided a summary of SBLOCA results for various break sizes. The summary included comparison of accumulator injection times with the PCT times.

The NRC staff evaluated the impact of the change in the containment air temperature on the accumulator actuation timing to ensure a different break size does not become limiting. Based on the summary data, the accumulator does not actuate for a 2-inch break, and hence this break is not impacted by a containment temperature increase. For a break size of 2.25-inch, accumulator injection does not occur until 346.7 seconds after the PCT occurs; therefore, the 2°F change in accumulator water temperature will have negligible impact on the PCT. The limiting break size occurs at 2.75-inch, where the PCT occurs over 250 seconds after accumulator injection. For break sizes 2.75-inch and above, the accumulators actuate well prior to the PCT time, and thus any change in accumulator injection timing due to the containment temperature change would have minimal effect on the PCT due to a small reduction in total energy removal capability of the accumulator fluid. The data showed that the smallest difference between accumulator injection time and PCT time occurs for the 2.5-inch break (26.3 seconds). In this case, the PCT is over 300°F lower than the limiting PCT, and thus any impact of the 2°F containment temperature increase on the accumulator timing would not cause this break size to become limiting.

In its supplement dated August 16, 2024, the licensee also stated that an increase in containment temperature from 120°F to a conservative accumulator temperature input of 124°F, corresponds to an enthalpy increase of ~4 Btu/lbm. The licensee indicated that the change in heat removal capacity associated with a 4°F increase in containment air temperature would remain small relative to the total heat removal capacity of the accumulator fluid.

The NRC staff confirmed independently that there will be negligible impact on the SBLOCA AOR based on the analysis provided by the licensee for different break sizes and the relatively small reduction in total energy removal capability of the accumulator fluid associated with the change in the containment air temperature.

3.3 Post LOCA Evaluation

3.3.1 Subcriticality Assessment

The post LOCA subcriticality analyses minimize liquid mass inventories for boron injection sources such as the accumulators. The potential increase in maximum accumulator temperature will change the accumulator mass due to the density decrease. The licensee stated that the change in density due to the increase in temperature from 120°F to 122°F is small enough to not have any measurable impact on the accumulator mass used in the subcriticality calculations to the level of precision used for the density in those calculations compared to the magnitude of the change.

The NRC staff's independent evaluation finds that a potential small increase in the accumulator temperature will not have a measurable impact in the accumulator mass because the change in density due to the increase in temperature from 120°F to 122°F does not have any measurable impact on the accumulator mass used in the subcriticality calculations to the level of precision used for the density. Further, the NRC staff notes that an increase in accumulator temperature will lead to an increase in the solubility of the boric acid. Hence, the NRC staff finds that the proposed small increase in accumulator temperature has a negligible effect on the subcriticality assessment for this LAR.

3.3.2 Sump Dilution and Hot Leg Switchover Assessment

In its submittal dated July 18, 2024, the licensee states that the accumulator mass calculation is based on a higher density value and is not dependent on the maximum accumulator temperature. Hence the licensee states that there is no impact on the post LOCA sump dilution calculation due to the potential small increase in accumulator temperature from 120°F to 122°F.

Further, as stated in the previous section, the decrease in density due to an increase in temperature from 120°F to 122°F is small enough to not have any measurable impact on the accumulator mass. The licensee also states that the hot leg switchover analysis uses the same accumulator mass value used in the sump dilution calculation and thus is not impacted by the small increase in accumulator temperature. Based on the discussion provided by the licensee, the NRC staff finds the sump dilution and hot leg switchover assessment is not impacted by the increase in accumulator temperature and is therefore acceptable.

3.3.3 Decay Heat Removal Assessment

In its letter dated July 18, 2024, the licensee stated that any minor changes in core voiding and core boil-off rates resulting from the 2°F accumulator temperature increase are relatively short-term effects that do not persist into the long-term cooling phase of the emergency core cooling system (ECCS) performance evaluations.

Based on the above, the NRC staff concludes that any changes in the core voiding or boil-off rates due to increase in accumulator temperature from 120°F to 122°F are short-term effects and thus do not impact the long-term cooling phase of the ECCS and is, therefore, acceptable.

3.4 Containment Cooling Systems Evaluation

Containment temperature is maintained by using four safety-related forced-air heat exchangers. The containment cooling system consists of four containment air coolers, each with a one-third cooling capacity during normal operation, with up to four units operating. Each air cooler consists of a fan and finned tube coil supplied by water from the service water system.

The heat sink for the containment air cooling units is the service water system. In the licensee's submittal dated July 18, 2024, the licensee stated that:

> The ultimate heat sink (UHS) analysis for the service water pond considers various combinations of units in shutdown and accident conditions. The service water discharge from both units is aligned back to the pond, resulting in the pond absorbing decay heat from both units for the potential 7-day period. The small increase in containment temperature at the start of the event represents an insignificant effect compared to the magnitude of heat transferred to the service water from both units.

The licensee further states that, while the UHS might be expected to rise in conjunction with the increased ambient temperatures, SNC does not expect the UHS to exceed its TS limit of 95°F as verified by Surveillance Requirement 3.7.9.2.

FSAR Section 6.2.2.3.2, “Containment Cooling System,” indicates that the containment cooling systems components are designed to operate and withstand the post-accident environment. The criteria for the selection of the design service lifetime for the air coolers in the accident environment is based primarily on the service lifetime for the fan motor. FSAR Section 6.2.2.4, “Testing and Inspection,” states that “[c]ontainment air cooling unit fan motors are certified by the manufacturer to operate during and/or subsequent to a LOCA.”

In Section 3.3 “Safety Margins,” of its submittal dated July 18, 2024, the licensee stated that the SNC response to Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," continues to be based on the AOR, and remains unaffected by the proposed change to TS 3.6.5. The licensee stated that its analysis showed that no water hammer will occur within this piping. Following a LOCA coincident with loss-of-offsite power, the containment fans and service water pumps are deenergized. The licensee stated, in part, that:

> In the time interval of interest (25 seconds or less) following initiation of this event the service water downstream of the containment coolers has been calculated to reach a maximum of 119°F. Even assuming that the service water downstream of the containment coolers picks up an additional 2°F based on the increased initial containment average air temperature, the maximum temperature will still be less than the 164°F temperature required to form a vapor cavity.
>
> Service water temperature, along with its associated vapor pressure, will rise rapidly following containment cooler fan restart. However, the increase in the service water's vapor pressure, caused by the service water's temperature rise, will not reach the increased pressure in the containment cooler discharge piping. Therefore, two phase flow conditions still will not occur.

Based on the above, a short-term (i.e., up to 7 days) containment air temperature excursion of 2°F on a yearly basis will not impact the capability of each of the nuclear unit’s four safety-related forced-air heat exchangers to satisfy its safety function.

As described in FSAR Section 6.2.3.2.3, “Containment Ventilation Systems,” a portion of the containment air cooler discharge is ducted directly to the reactor cavity to provide cooling to the cavity, instrumentation, and equipment within.

With respect to the impact on operability of instrumentation and equipment within the reactor cavity of a short-term (i.e., up to 7 days) containment average air temperature excursion of up to 2°F on a yearly basis, the licensee concluded in its July 18, 2024, submittal that:

> SNC has reviewed the specifications of the non-EQ electric equipment (i.e., electric equipment not subject to the requirements in 10 CFR 50.49) within containment that is expected to perform a design function under normal operation (e.g., electrical equipment that is either relied upon by the plant operators to inform operational decisions or provides a signal input to other plant systems or processes) whose failure could mislead a plant operator or cause a

plant transient and determined that this electrical equipment will not be adversely impacted by the proposed temporary increase in plant temperature.

In its letter dated July 18, 2024, the licensee states in part, that:

The containment cooling and ventilating functions are augmented by the four containment recirculation fans, which take suction from the containment dome and discharge downward to help provide mixing of the containment atmosphere during normal operation to augment heat removal and maintain uniform temperature distributions throughout the containment volume.

The control rod drive mechanism (CRDM) cooling system consists of fans and ducting to draw air through the CRDM shroud and eject it to the main containment atmosphere. One hundred-percent redundancy is provided by a standby fan.

The reactor vessel support cooling system, consisting of two 100% capacity dans and ducting, is arranged to cool the reactor vessel supports by drawing air through the supports. One hundred percent redundancy of the active components is provided.

FSAR 6.2.3.2.4 states that:

In the event of a failure of the CRDM cooling system, the airflow to the CRDMs may be lost. Loss of airflow to the CRDM would increase the operating coil temperatures. In the event of high CRDM temperatures, the control rods fall to the safe position, i.e., inserted into the core.

FSAR 6.2.3, "Containment Air Purification and Cleanup Systems," states, in part:

The containment minipurge system provides continuous ventilation and filtration of the containment atmosphere so as to allow periodic occupation of the containment during normal power operation.

FSAR 6.2.3.3.4, "Containment Ventilation Systems," states, in part, that:

The normal operation of the containment air coolers and the operation of the remaining containment ventilation systems are not required to reduce accident doses or maintain offsite effluent concentrations during normal operation within established guidelines.

In Section 3.5, "Implementation and Monitoring Plan," of its submittal dated July 18, 2024, the licensee proposes that the following compensatory measures be initiated prior to exceeding 120°F as defense-in-depth efforts in order to prevent containment average air temperature exceeding 122ºF

- Operate available containment coolers on high speed with service water aligned to the service water wet pit, (i.e., to the emergency mode);
- Operate available containment mini-purge continuously;
- Operate available containment recirculation fans in high speed;
- Put in place work controls to prevent removal of containment cooling system components and supporting systems from service; and
- Put in place work controls to protect the containment cooling systems.

> These actions may be initiated prior to exceeding 120°F consistent with the best efforts to avoid exceeding the TS limiting containment average air temperature.
>
> Operating the containment coolers, the mini-purge, and recirculation fans in this manner maximizes containment cooling. Aligning the service water system for the emergency mode to the containment coolers provide higher flow rates and maximizes containment cooling. The work controls maintain the systems in an operating condition until the high temperature situation is resolved.
>
> Should an emergency event occur, these systems will revert to their emergency alignment and function as assumed in the safety analysis.

As the post-accident containment atmosphere, which consists of a steam-air mixture, is circulated through the bank of cooling coils, it is cooled, and a portion of the steam is condensed. FSAR section 6.2.2.2, "Containment Cooling System," states, in part, that "[t]he capacity of one cooler in conjunction with one containment spray train is adequate to maintain pressure and temperature below peak calculated LOCA conditions." FSAR Table 6.2-19, "Containment Results For The Design Basis LOCA," indicates that containment "Steam and air" temperature "Prior to LOCA" can be ≤ 127°F. Therefore, the current licensing bases for both Farley Unit 1 and Unit 2 bounds the proposed changes of the LAR.

Section 3.4.4, "Risk Results," of its submittal dated July 18, 2024, the licensee states that:

> Containment Spray would likely occur earlier in a LOCA scenario, but RWST cues are still available to inform operators of low level. Since there is minimal impact to peak containment pressures and containment sump water temperatures, no changes to RHR success criteria are noted.

In Section 3.2, "Defense-in-Depth," of its submittal dated July 18, 2024, the licensee states, in part, that:

> The containment heat removal systems are designed such that the failure of any single active component, assuming the availability of either onsite or offsite power exclusively, does not prevent the systems from accomplishing their design safety functions.
>
> There are no changes to the design of any plant system. There are no changes to the redundancy inherent in the containment heat removal design.
>
> Therefore, both plant designs of Farley Unit 1 and Unit 2 provide reasonable assurance of the continued availability of the containment heat removal systems to perform their intended function after an anticipated operational occurrence or a postulated design basis accident.

Based on the above, the NRC staff concludes allowing containment average air temperature increase above 120°F to 122°F for up to 7 days cumulative per calendar year does not impact the layers of defense-in-depth and would continue to meet GDC 38, "Containment heat removal," and is, therefore, acceptable.

3.5 Equipment Qualification of Electrical Equipment Important to Safety

The NRC staff reviewed the licensee's LAR, as supplemented, to determine the impact of the proposed change on the electrical equipment within containment. In its submittal, as supplemented, the licensee noted that the scope of the Farley EQ Program includes electrical equipment that is important to safety. In the licensee's submittal dated July 18, 2024, the licensee stated that, "[e]quipment that is important to safety involves safety-related and non-safety-related electrical equipment whose failure can prevent satisfactory accomplishment of safety functions as described in 10 CFR 50.49 (b)(1) and (b)(2) and certain post-accident monitoring equipment as described in 10 CFR 50.49(b)(3)."

Additionally, the licensee stated that the proposed "2°F increase in the containment average air temperature is bounded by the existing [EQ] analyses, due to conservatisms and margins in the existing test programs and calculations," and goes on to conclude that "an increase from 120°F to 122°F in the containment average [air] temperature for up to [seven days (not to exceed seven days cumulative in calendar year)] will have no impact on the qualification status or qualified lives of existing equipment located in containment in the EQ Program scope."

Based on the review of the submittal, as supplemented, and information related to the licensee's EQ Program reviewed during the audit, the NRC staff finds that the temperature excursions resulting from the licensee's proposed TS 3.6.5 change in containment average air temperature would be negligible in comparison to the time and temperatures considered for qualifying electric equipment. Therefore, the EQ of electric equipment at Farley is unlikely to be adversely impacted by the proposed increase in containment average air temperature.

In the July 18, 2024 submittal, the licensee stated that they had also "reviewed the specifications of the non-EQ electric equipment (i.e., electric equipment not subject to the requirements in 10 CFR 50.49) within containment that is expected to perform a design function under normal operation (e.g., electrical equipment that is either relied upon by the plant operators to inform operational decisions or provides a signal input to other plant systems or processes) whose failure could mislead a plant operator or cause a plant transient and determined that this electrical equipment will not be adversely impacted by the proposed temporary increase in [containment average air temperature]." The NRC staff reviewed the licensee's evaluation of this equipment during the audit to gain a better understanding of the scope of equipment and the methodology used to assess equipment potentially impacted by the proposed change.

Based on the above, the NRC staff finds that non-EQ electric equipment is unlikely to be adversely impacted by the proposed short-term temperature excursions based on the sufficient design qualifications and specifications of those equipment.

The NRC staff also finds that the licensee's proposed changes are unlikely to have an adverse impact on the Farley, Units 1 and 2, EQ Program or its ability to continue to meet the requirements of 10 CFR 50.49, Category II of NUREG-0588, Revision 1, IEB 79-01B, and GDC 4. The NRC staff concludes that the proposed changes would have no adverse impact on non-EQ electric equipment at Farley, Units 1 and 2, or their ability to meet the requirements of GDC 4.

Based on the above, the NRC staff's independent evaluation concludes that there is reasonable assurance that the proposed changes are bounded by the plant AOR for electrical equipment qualification and is, therefore, acceptable.

3.6 Other Considerations (Defense-in-Depth)

The licensee presented several risk-insights related to the proposed change including the defense-in-depth design features consistent with the criteria identified in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis". The licensee presented compensatory measures as well as features that provide layers of defense applicable to containment heat removal. The NRC staff reviewed the seven items presented in the LAR and finds them consistent with Section 2.1.1.2 of RG 1.174. Based on the licensee's provided discussion, the NRC staff concludes that the increase in containment average air temperature from 120°F to 122°F for up to seven days (not to exceed seven days cumulative in calendar year) does not impact the layers of defense-in-depth of the containment heat removal systems at Farley. Thus, the NRC staff finds the requested change maintains adequate defense-in-depth capability.

3.7 Evaluation of Technical Specification Change

The licensee proposed the following changes to LCO 3.6.5:

- The existing Condition A, Required Action A.1, Restore containment temperature to within limit, is renumbered as proposed Required Action A.3, and a new Completion Time of 7 days is proposed.
- A new Required Action A.1 is proposed. The proposed Required Action A.1, Verify containment average air temperature ≤ 122°F. The proposed Completion Time is 8 hours AND once per 8 hours thereafter.
- A new Required Action A.2 is proposed. The proposed Required Action A.2, Verify refueling water tank temperature ≤ 100°F. The proposed Completion Time is 8 hours AND once per 8 hours thereafter.
- Proposed Required Action A.3 (renumbering current Required Action A.1), Restore containment average air temperature to within limit. The Completion Time is proposed to be changed from 8 hours to 7 days. A note is added specifying that the Completion Time is "[n]ot to exceed 7 days cumulative in calendar year."
- Delete the expired Note associated with LCO 3.6.5. The expired Note reads, "Containment average air temperature shall be ≤122°F until 0600 hours on September 9, 2023."

The temperature limits contained in the proposed Required Actions A.1 and A.2 were evaluated by the NRC, as described in Sections 3.1 through 3.5 of this Safety Evaluation (SE), and were found to meet the applicable regulatory requirements as listed in Section 3.9 of this SE. Accordingly, the staff concludes that the proposed TS changes provide reasonable assurance that the plant will operate safely while the limits are adhered to. The proposed Completion Times for Required Actions A.1 and A.2 provide assurance that the temperature limits contained in the actions will not be exceeded because the temperature verifications are initially prompt, and thereafter frequent enough, to ensure that the temperatures are verified to be within the prescribed limits. Therefore, the proposed Required Actions A.1 and A.2 are acceptable.

The proposed Required Action A.3 is unchanged from existing Required Action A.1. Therefore, the proposed Required Action A.3 is acceptable because it requires that the containment temperature be returned to within the LCO limit. However, the Completion Time is proposed to be significantly longer than the existing time. In addition, an added note ensures that the cumulative time in Condition A in a calendar year is limited to the Completion Time.[1] The Completion Time is seven days, which can be expended in a single time period or accumulated over several entries into Condition A. Once the seven days is accumulated within a single calendar year, the Completion Time is expired for the remainder of that year and immediate entry into Action B is required if the LCO is entered. The NRC staff found that the limits contained in Required Actions A.1 and A.2 ensure that the plant is operated safely and continues to meet regulatory requirements while temperatures are maintained within the action limits. Even though the limits in Required Actions A.1 and A.2 were found to be protective of plant safety, the proposed Action A.3 would still require the licensee to return the containment temperature to within the LCO limit. The proposed Completion Time for Action A.3 ensures that the containment temperature is restored within a reasonable time considering the NRC deterministic conclusions discussed in Sections 3.2, 3.3, and 3.4 of this SE regarding safe plant operation while in Required Actions A.1 and A.2. Therefore, the proposed Completion Time of 7 days (not to exceed 7 days cumulative in calendar year) is acceptable.

The licensee proposed to remove an expired LCO Note that is no longer applicable to the TS LCO and was previously approved for a one time use under emergency circumstances with the issuance of Amendment Nos. 247 and 244 for Farley, Units 1 and 2, respectively (ML23235A296). Removal of the Note is an administrative change and is, therefore, acceptable.

Regulations in 10 CFR 50.36(c)(2) require that the TS contain LCOs that represent the lowest functional capability of the equipment required for safe operation, and further allows the licensee to follow remedial actions allowed by the TS until the LCO condition can be met. Since the LCO for TS 3.6.5 is not changed, the LCO remains acceptable. The NRC staff determined that the Required Actions are remedial actions that will provide reasonable assurance of adequate protection of public health and safety when the LCO is not met. Therefore, the Required Actions are acceptable remedial actions. Based on the above, the NRC staff concludes that the proposed changes to TS 3.6.5 LCO would continue to meet 10 CFR 50.36 and are, therefore, acceptable.

The NRC staff determined, based on discussions outlined in sections 3.1 through 3.3 of this SE, that the maximum PCT does not exceed 2200°F, that the ECCS cooling performance evaluation model is acceptable, and that the analytical techniques described by the licensee realistically describes the behavior of Farley's reactor system during a LOCA. Based on the above, the NRC staff concludes that the proposed changes to TS 3.6.5 LCO would continue to meet 10 CFR 50.46 and are, therefore, acceptable.

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1. As described in the associated change to the TS bases in this LAR, "[t]he cumulative time is tracked as actual time operating in Condition A and initially begins from the initial Condition A entry in the calendar year. Each entry and exit time for Condition A are tracked and added to the prior cumulative time(s)."

3.8 Evaluation of Risk-Insights

In Section 3.4 of its submittal dated July 18, 2024, the licensee used the Farley Probabilistic Risk Assessment (PRA) model to evaluate the change in risk associated with the proposed change to TS LCO 3.6.5 action statements to allow operation with containment average air temperature >120°F and ≤122°F for a period of up to 7 days cumulative in a calendar year.

Because this LAR provided risk insights and is not a risk-informed application, the NRC staff did not review the licensee's submittal against the principles in RGs 1.174 and 1.177. The NRC staff evaluated the use of risk insights to support its deterministic review of the licensee's plant-specific circumstances to address elevated site ambient temperatures for the proposed changes to TS 3.6.5. As such, the NRC evaluation of risk does not alter the NRC's position on defense-in-depth for protecting the containment as the last barrier to the release of radioactivity or constitute an expansion of the scope of approval or authority provided in NRC approval of amendments for Initiative 4b (TSTF-505) and 10 CFR 50.69.

The licensee used the Modular Accident Analysis Program (MAAP) to identify aspects of the PRA that are potentially impacted by the proposed TS change, including accident sequences, success criteria, human reliability analysis, and characteristics of release to the environment following an accident. The licensee used the maximum acceptable containment air temperature (122°F) as an initial condition for the MAAP analysis to develop input to key elements of the PRA. The credited cases were reviewed by comparing the base cases with the updated containment temperature case. For cases that may impact the PRA, analyses were completed to determine the values to be used in the updated PRA model.

The change in core damage frequency (CDF) and large early release frequency (LERF) were calculated to evaluate the impact of the change to TS LCO 3.6.5 on plant risk.

3.8.1 PRA Model Acceptability

NRC staff evaluated the acceptability of the Farley PRA model as part of its approval of the amendments to adopt NEI 06-09, "Risk-Informed Technical Specifications Initiative 4b, 'Risk-Managed Technical Specification (RMTS) Guidelines'" (ML19175A243), that preceded Technical Specification Task Force (TSTF) 505, Revision 2, and 10 CFR 50.69 (ML21137A247) for risk-informed categorization and treatment of structures, systems, and components (SSCs). The NRC staff found that an independent assessment for the closure of Findings and Observations (F&Os) was performed consistent with Appendix X to NEI 05-04/07-12/12-13, "Close Out of Facts and Observations (F&Os)," and no additional peer reviews were required (ML17086A451).

The licensee stated in Section 3.4.2, "PRA Model Acceptability" of the licensee's submittal dated July 18, 2024, that, following the approval of 10 CFR 50.69, it performed a Focused Scope Peer Review in January 2023 to review an upgrade to the Farley PRA. Items implemented in the Farley PRA since the approval of 10 CFR 50.69 include updated fire PRA methods and updated diverse and flexible coping strategies (FLEX) modeling consistent with the 2022 NRC staff memorandum on crediting FLEX strategies in PRA (ML22014A084).

Based on a review of Section 3.4.2 of the licensee's submittal dated July 18, 2024, the staff confirmed that the licensee addressed and closed all open F&Os resulting from the Focused Scope Peer Review.

This review, in addition to the previous review of the licensee's PRA model used in the Initiative 4b and 10 CFR 50.69 amendments, the NRC staff finds that the licensee's PRA model is acceptable, with respect to scope, level of detail, technical elements, and plant representation, to support the TS change evaluation, consistent with the guidelines in RG 1.177, Revision 2 "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications."

3.8.2 PRA Modeling of the Proposed Change

The scope of the Farley PRA model for the assessment of risk impact due to the increase in containment air temperature is limited to the applicable hazards that would be impacted by the proposed TS change including internal events, internal flooding, internal fire, and seismic. The licensee identified candidate human failure events (HFEs) and success criteria that may be affected by an increase in initial containment temperature based on the review of its current human reliability analysis (HRA) calculator and the PRA success criteria records. For example, the licensee postulated that an increase in containment temperature would affect the timing of CS actuation which could affect the timing of ECCS recirculation mode related HFEs.

The licensee eliminated HFEs that are not considered impacted by the increased containment temperature using these screening criteria: (1) HFEs calculated with the screening method, (2) HFEs without a supporting MAAP analysis used for timing due to containment average air temperature only being an input to MAAP cases, (3) HFEs related to steam generator overfill due to the possibility of a steam line break being eliminated as the steam generator overfill would cause water to replace steam preventing a steam line break from occurring, and (4) HFEs with long time windows and recovery timings (>1 hr) due to those HFEs not being time sensitive enough to impact the success rate of completing required actions.

Using the above screening criteria, the licensee identified the remaining HFEs with MAAP parameters of interest and subsequently evaluated them by re-running the associated MAAP case. The three MAAP cases used by the licensee to support the evaluations affected by the change in containment air temperature included: (1) MSLBX: main steam line break inside containment with one of the three steam generators feeding the break, (2) LLOCA: double-ended cold leg break with two trains of ECCS and three accumulators available, and (3) MLOCA: six-inch diameter cold leg break with two trains of ECCS available.

The NRC staff reviewed the licensee's screening criteria and analysis results and finds them acceptable because they are consistent with thermal hydraulic logic and PRA practices relative to HRA.

3.8.3 Risk Results and Insights

In Section 3.4.4 of its submittal dated July 18, 2024, the licensee provided the results of the plant risk evaluation associated with the proposed TS change and is summarized as follows: (1) negligible impact on MAAP analysis of timing and success criteria used in the Farley PRA model; (2) negligible impact on human error probabilities (HEPs) for post-initiator operator actions credited in the PRA; (3) no changes to RHR success criteria needed due to minimal impact to peak containment pressures and containment sump water temperatures identified from MAAP analysis; (4) sensitivity analysis with more restrictive success criteria for containment fan coolers from 2/4 to 3/4 showing negligible impact on plant risk; and (5) quantification results from Farley PRA, accounting for internal events, internal flooding, internal fire, and seismic, showing negligible impact on plant risk.

The licensee concluded that the cumulative risk impact from a 2°F increase in containment average air temperature results in: (1) less than $1x10^{-7}$ incremental conditional core damage probability (ICCDP) and less than $1x10^{-8}$ incremental conditional large early release probability (ICLERP), (2) less than a $1x10^{-7}$/yr change in core damage frequency (DCDF) and less than a $1x10^{-8}$/yr change in large early release frequency (DLERF), and (3) new baseline CDF less than $1x10^{-4}$/yr and baseline LERF less than $1x10^{-5}$/yr, when the risk impact of the proposed TS change is incorporated.

The NRC staff finds that the licensee's risk results are within the applicable NRC guidelines, the ICCDP and ICLERP meet the criteria in RG 1.177, and the CDF/DCDF and LERF/DLERF meet the criteria in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

Based on the above, the staff finds that the licensee's PRA results support the proposed TS LCO 3.6.5 action statements to allow operation with containment average air temperature >120°F and ≤122°F for up to seven days (not to exceed seven days cumulative in calendar year) and is, therefore, acceptable.

3.9 Technical Evaluation Conclusion

The NRC staff's review covered the pressure and temperature conditions in the containment due to a spectrum of postulated LOCAs and secondary line breaks. The NRC staff evaluated the technical justifications and the analyses presented by the licensee. Based on the review of the LAR and the supplemental information provided, the NRC staff concludes that:

- 10 CFR, Appendix A, GDC 4 continues to be met because the small increase in containment average air temperatures for a short period of time has negligible effect on the capability of Farley, Units 1 and 2, electrical equipment to perform their design function,

- 10 CFR 50, Appendix A, GDCs 16 and 50 continue to be met because the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require,

- 10 CFR 50, Appendix A, GDC 38 continues to be met because the containment heat removal system (RHR and CS) pumps have adequate NPSH margin so that these systems will remove heat from the reactor containment and rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptable low levels,

- The proposed changes will continue to meet the 10 CFR 50.36 regulatory requirements as the proposed LAR does not change LCO 3.6.5 and the Required Actions and Completion Times provide reasonable assurance of safety, and

- The licensee will continue to meet the 10 CFR 50.46 regulatory requirements as the licensee is using an acceptable evaluation model that realistically describes the Farley RCS with an explicit accounting for uncertainties to account for any variabilities.

The proposed LAR provides sufficient technical justifications to demonstrate compliance with applicable regulations and an acceptable evaluation of risk insights to support managed risk during any period when the plant exceeds the LCO.

Based on the above, the NRC staff finds the request to relocate existing TS 3.6.5 Required Action A.1 as proposed Required Action A.3 and revise the associated Completion Time, add proposed Required Actions and Completions Times for A.1 and A.2, as well as remove an expired Note in TS 3.6.5 LCO acceptable. The NRC staff's independent evaluation concluded that the proposed changes are bounded by the plant AOR and is, therefore, acceptable.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The NRC's regulation in 10 CFR 50.92(c) states that the NRC may make a final determination, under the procedures in 10 CFR 50.91, that a license amendment involves no significant hazards consideration (NSHC) if the operation of the facility, in accordance with the amendment, would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

An evaluation of the licensee's proposed no significant hazards consideration, as shown below, was completed by the NRC staff. The Commission finds that the licensee's analyses, consistent with 10 CFR 50.91, "Notice for public comment; State consultation," is sufficient to support the proposed determinations that the amendment requests involve no significant hazards consideration is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed changes do not adversely affect the operation of any structures, systems, or components (SSCs) associated with an accident initiator or initiating sequence of events. The proposed changes do not affect the design of the containment heat removal systems.

   The proposed amendment does not affect accident initiators or precursors nor adversely alter the design assumptions, conditions, and configuration of the facility. The proposed amendment does not alter any plant equipment or operating practices with respect to such initiators or precursors in a manner that the probability of an accident is increased. The proposed amendment to allow exceeding the containment average air temperature above the current limit of 120°F up to a maximum of 122°F for up to 7 cumulative days in the calendar year and remove an expired Note does not adversely affect the operation of the assumed mitigation systems or the containment fission product barrier assumptions. As demonstrated in the SNC request, the potential increase in containment average air temperature coupled with limiting the refueling water storage tank (RWST) temperature to ≤100 °F is more than offset by existing margins in the safety analyses. As such, the proposed change will not alter assumptions relative to the mitigation of an accident or transient event. The proposed amendment does not increase the likelihood of the malfunction of an SSC or adversely impact analyzed accidents.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed amendment does not introduce any new or unanalyzed modes of operation. The proposed changes do not involve a physical alteration to the plant (i.e., no new or different type of equipment will be installed) or a change to the methods governing normal plant operation. The changes do not alter the limiting assumptions made in the safety analysis.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The margin of safety is related to the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment. The fission product barriers continue to be able to perform their required functions; based on the pre-existing margins and conservatisms currently assumed in the safety analyses. Therefore, the margins to the onsite and offsite radiological dose limits are not significantly reduced.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the NRC's review of the above NSHC evaluation proposed by the licensee, the NRC staff concludes that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff has made a final determination that no significant hazards consideration is involved for the proposed amendment and that the amendment should be issued as allowed by the criteria contained in 10 CFR 50.91.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified via email of the proposed issuance of the amendments on August 19, 2024, and the State official confirmed that the State of Alabama had no comments (ML24240A137).

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on July 29, 2024 (89 FR 60930). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Additionally, the removal of the expired TS 3.6.5 LCO Note is editorial and meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10)(v). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 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 amendment 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 29, 2024

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENT NOS. 250 AND 247, REGARDING LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATION 3.6.5, “CONTAINMENT AIR TEMPERATURE,” ACTIONS (EPID L-2024-LLA-0098) DATED AUGUST 29, 2024

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