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

August 14, 2026

Ms. Jamie M. Coleman
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway, N 274.EC
Birmingham, AL 35243

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENT NOS. 261 AND 258, TO REVISE TECHNICAL SPECIFICATION 3.3.1 AND 3.3.2 ALLOWABLE VALUES (EPID L-2025-LLA-0145)

Dear Ms. Coleman:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 261 to Renewed Facility Operating License No. NPF-2 and Amendment No. 258 to Renewed Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant (FNP, Farley), Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated September 4, 2025, as supplemented by letter dated June 5, 2026.

The amendments revise selected Allowable Values in TS 3.3.1, “Reactor Trip System (RTS) Instrumentation” and TS 3.3.2, “Engineered Safety Feature Actuation System (ESFAS) Instrumentation.”

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

Sincerely,

/RA/

G. Edward Miller, Senior Project Manager
Operating Reactor Licensing Branch 2
Division of Licensing Projects I
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 261 to NPF-2
2. Amendment No. 258 to NPF-8
3. Safety Evaluation

cc: 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. 261
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 September 4, 2025, as supplemented by letter dated June 5, 2026, 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.C.(2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 261, 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 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

JOHN LAMB Digitally signed by JOHN LAMB
Date: 2026.08.14
15:09:32 -04'00'

Michael Markley, Chief
Operating Reactor Licensing Branch 2
Division of Licensing Projects I
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No, NPF-2
and the Technical Specifications

Date of Issuance: August 14, 2026

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. 258
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 September 4, 2025, as supplemented by letter dated June 5, 2026, 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.C.(2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 258, 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 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

JOHN LAMB Digitally signed by JOHN LAMB
Date: 2026.08.14
15:10:16 -04'00'

Michael Markley, Chief
Operating Reactor Licensing Branch 2
Division of Licensing Projects I
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-8
and the Technical Specifications

Date of Issuance: August 14, 2026

ATTACHMENT TO LICENSE AMENDMENT NO. 261

TO RENEWED FACILITY OPERATING LICENSE NO. NPF2

AND LICENSE AMENDMENT NO. 258

TO RENEWED FACILITY OPERATING LICENSE NO. NPF8

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50364

Replace the following pages of the Renewed Facility Operating License Nos. NPF-2 and NPF-8, 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 Pages	Insert Pages
License	License
NPF-2, Page 4	NPF-2, Page 4
NPF-8, Page 3	NPF-8, Pag 3
TSs	TSs
3.3.1-15	3.3.1-15
3.3.2-9	3.3.2-9
3.3.2-10	3.3.2-10
3.3.2-11	3.3.2-11

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 261, 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 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. 258, 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

Table 3.3.1-1 (page 2 of 8)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
5.	Source Range Neutron Flux	2(d)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.10	≤ 1.2 E5 cps	1.0 E5 cps
		3(a), 4(a), 5(a)	2	J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 1.2 E5 cps	1.0 E5 cps
		3(e), 4(e), 5(e)	1	L	SR 3.3.1.1 SR 3.3.1.10	N/A	N/A
6.	Overtemperature ΔT	1,2	3	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.7 SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.14	Refer to Note 1 (Page 3.3.1-20)	Refer to Note 1 (Page 3.3.1-20)
7.	Overpower ΔT	1,2	3	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.14	Refer to Note 2 (Page 3.3.1-21)	Refer to Note 2 (Page 3.3.1-21)

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication.

Table 3.3.2-1 (page 1 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8	NA	NA
c. Containment Pressure — High 1	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9	≤ 4.3 psig	4.0 psig
d. Pressurizer Pressure — Low	1,2,3[(a)]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9	≥ 1847 psig	1850 psig
e. Steam Line Pressure						
(1) Low	1,2,3[(b)]	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9	≥ 580[(c)] psig	585[(c)] psig
(2) High Differential Pressure Between Steam Lines	1,2,3	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9	≤ 105 psig	100 psig

(a) Above the P-11 (Pressurizer Pressure) interlock.

(b) Above the P-12 (T_{avg} - Low Low) interlock.

(c) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

Table 3.3.2-1 (page 2 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
2. Containment Spray						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8	NA	NA
c. Containment Pressure High - 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9	≤ 27.3 psig	27 psig
3. Containment Isolation						
a. Phase A Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8	NA	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8	NA	NA
(3) Containment Pressure High - 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9	≤ 27.3 psig	27 psig

Table 3.3.2-1 (page 3 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation						
a. Manual Initiation	1,2(d),3(d)	1 per steam line	F	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2(d),3(d)	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.8	NA	NA
c. Containment Pressure - High 2	1,2(d), 3(d)	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9	≤ 16.5 psig	16.2 psig
d. Steam Line Pressure Low	1,2(d),3(b)(d)	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 SR 3.3.2.9	≥ 580(c) psig	585(c) psig
e. High Steam Flow in Two Steam Lines	1,2(d),3(d)	2 per steam line	D, M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(e)	(f)
Coincident with T_{avg} - Low Low	1,2(d),3(d)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 542.6°F	543°F

(b) Above the P-12 (T_{avg} - Low Low) interlock.
(c) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
(d) Except when one MSIV is closed in each steam line.
(e) Less than or equal to a function defined as ΔP corresponding to 40.7% full steam flow below 20% load, ΔP increasing linearly from 40.7% full steam flow at 20% load to 110.3% full steam flow at 100% load.
(f) Less than or equal to a function defined as ΔP corresponding to 40% full steam flow between 0% and 20% load and then a ΔP increasing linearly from 40% steam flow at 20% load to 110% full steam flow at 100% load.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 261 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

AND

AMENDMENT NO. 258 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 September 4, 2025 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML25254A180 (Public, redacted) and ML25247A293 (Not publicly available; proprietary information)), as supplemented by letter dated June 5, 2026 (ML26156A143), Southern Nuclear Operating Company (SNC) (the licensee) submitted a license amendment request (LAR) for Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2.

The proposed amendments would revise selected Allowable Values (AVs) in Technical Specification (TS) 3.3.1, "Reactor Trip System (RTS) Instrumentation" and TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

A notice of consideration of proposed issuance of amendments with proposed no significant hazards consideration determination was published in the *Federal Register* on January 27, 2026 (91 FR 3553) and there has been no public comment on such finding. The supplemental letter dated June 5, 2026, provided additional information that clarified the application, did not expand the scope of the application, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in this notice

1.1 Description of System Design and Operation

The RTS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the fuel design limits and the reactor coolant system (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the engineered safety features (ESF) systems in mitigating accidents.

The protection and monitoring systems are designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RTS, as well as specifying limiting conditions for operation (LCOs) on other reactor system parameters and equipment performance.

The LSSS, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during design-basis accidents (DBAs).

The ESFAS initiates necessary safety systems, based on the values of selected unit parameters, to protect against violating core design limits and the RCS pressure boundary, and to mitigate accidents. This is achieved by specifying LSSS in terms of parameters directly monitored by the ESFAS, as well as specifying LCOs on other reactor system parameters and equipment performance.

1.2 Description of Proposed Change

The licensee has proposed to change the AVs for specified functions in TS tables 3.3.1-1 and 3.3.2-1 as shown in the enclosure to the LAR. Further, the licensee stated that the changes are consistent with the setpoint methodology in WCAP-13751-P, Revision 7, “Westinghouse Setpoint Methodology for Protection Systems Farley Nuclear Plant Units 1 and 2.”

2.0 REGULATORY EVALUATION

2.1 Regulatory Requirements

Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 provides General Design Criteria (GDC) for nuclear power plants. Plant-specific design criteria are described in the plant’s Updated Final Safety Analysis Report (UFSAR). Specifically, section 3.1 of Farley’s UFSAR (ML25311A104) discusses conformance with the following criteria:

Criterion 20, “Protection system functions,” of Appendix A to 10 CFR Part 50 requires in-part, that the protection system be designed to automatically initiate operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences (AOOs) and to sense accident conditions and to initiate the operation of systems and components important to safety. A fully automatic protection system with appropriate redundant channels is provided to cope with events where insufficient time is available for manual action.

Paragraph (c)(1)(ii)(A) of 10 CFR 50.36, “Technical specifications,” requires, in part, that, where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting “be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded. If, during operation, it is determined that the automatic safety system does not function as required, the licensee shall take appropriate action, which may include shutting down the reactor. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence.” The safety analysis establishes an analytical limit in terms of a measured or calculated variable and a specific time after that value is reached to begin protective action. Satisfying these two constraints will ensure that the safety limit in 10 CFR 50.36 will not be exceeded during an AOO and design-basis events. The purpose of an LSSS is to assure that protective action is initiated before the process conditions reach the analytical limit, thereby limiting the consequences of a design-basis event to those predicted by the safety analyses. The LSSS is derived from the analytical limit in a manner determined by the setpoint calculation methodology.

Paragraph (c)(3) of 10 CFR 50.36, "Surveillance requirements," states that surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

Technical Specification Task Force (TSTF) Traveler TSTF-355, Revision 1, "Make changes to RTS and ESF tables," describes modifications to RTS TS table 3.3.1-1 and ESFAS TS table 3.3.2-1 to replace the requirement for a trip setpoint with a requirement for a nominal trip setpoint (NTSP in the LAR and NTS in the calculations). FNP Units 1 and 2 incorporated TSTF-355 through amendment Nos. 203 and 199, respectively, on August 3, 2016 (ML15233A448). The result of implementing TSTF-355 at FNP is that each process parameter has two associated TS values: (1) AV, and (2) NTS. The AVs specified in the TS tables serve as the 10 CFR 50.36 LSSS such that a channel is OPERABLE if the automatic actuation occurs before exceeding the AV during the Channel Operational Test (COT). As such, the AV differs from the NTS by an amount primarily equal to the expected instrument loop uncertainties, such as drift during the surveillance interval (The setpoint methodology specifies how to calculate the AV from the NTS, as described below). In this manner, the actual setting of the device will still meet the LSSS definition and ensure that a safety limit is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval (defined as the AV in the TS). If the actual setting of the device is found to have exceeded the AV, the device would be considered inoperable from a TS perspective. This requires corrective action including those actions required by 10 CFR 50.36 when automatic protective devices do not function as required. If the actual setting is found to exceed the NTS but not the AV, then the instrument is considered OPERABLE and is adjusted so the setting is more conservative than the NTS.

2.2 Regulatory Guidance

Regulatory Guide (RG) 1.105, Revision 4, "Setpoints for Safety-Related Instrumentation," February 2021 (ML20330A329) describes the method acceptable to the NRC staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the technical specification limits. This regulatory guide endorses American National Standards Institute (ANSI)/International Society of Automation (ISA) Standard 67.04.01-2018, "Setpoints for Nuclear Safety-Related Instrumentation," which provides a basis for establishing setpoints for nuclear instrumentation for safety systems.

The NRC staff notes that the Revision of record for Farley is RG 1.105, Revision 3 (ML993560062). The evaluation in this SE considers aspects of Revision 4 for the purpose of determining acceptability of the proposed changes but does not imply any change in the version of the RG to which the licensee's methodology is committed.

3.0 TECHNICAL EVALUATION

3.1 Evaluation of TS Allowable Value Changes

The NRC staff reviewed the licensee's analyses for the proposed changes to the AVs. The licensee provided these analyses in its LAR, as supported by attachments 5 (WCAP-13751-NP (non-proprietary)) and 6 (WCAP-13751-P (proprietary)) of the LAR. The staff examined the licensee's methodology per WCAP-13751-P, Revision 7, as provided in its LAR. The observed calculations for each of the NTSs associated with each proposed changed AV(s) for RTS TS table 3.3.1-1 (Function 5), and ESFAS TS table 3.3.2-1 (Functions 1.c, 1.e.1, 1.e.2, 2.c, 3.b.3, 4.c, 4.d, and 4.e). The NRC staff evaluated the licensee's proposed AV changes against the guidance in RG 1.105, Revision 3.

The NRC staff determined that the new AVs are within the proposed TS limits established in RG 1.105, as discussed in the following section.

3.1.1 Licensee's Methodology for Calculating Allowable Values

As described in section 2.4, "Process Allowances," of its methodology (WCAP-13751-P), the licensee's determination of its AVs for the RTS and ESFAS functions are listed below:

For <u>decreasing</u> parameter functions: AV = NTS + (2*RCA) – RD
For <u>increasing</u> parameter functions: AV = NTS – (2*RCA) + RD

where, as defined from WCAP-13751-P:

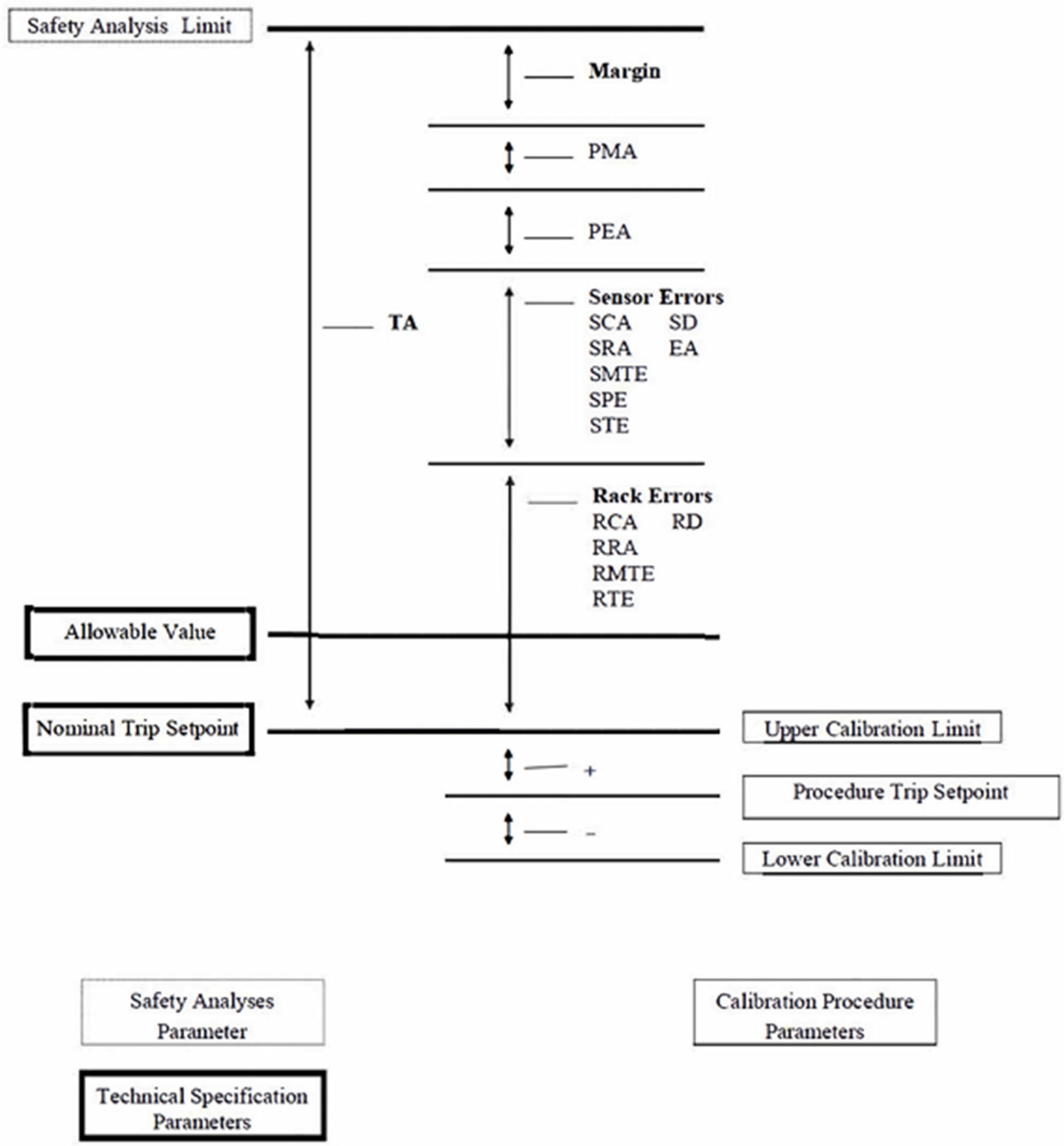
- Rack Calibration Accuracy (RCA; in engineering units) – The two-sided calibration tolerance of the process racks as reflected in the plant calibration procedures.

- Rack Drift (RD; in engineering units) – An allowance to account for the change in input-output relationship over a period of time at reference conditions.

In Revision 7 of WCAP-13751-P, the licensee notes that Revision 6 of WCAP-13751-P, complies with TSTF-355, which sets the actual trip setpoint to be more conservative than the defined NTS in the TS. Visual representations of the conservative trip setpoint (Labeled as "Procedure Trip Setpoint" [PTS]) are shown in figures 4-1 and 4-2 of WCAP-13751-P.

Figure 4-1

Farley Nuclear Plant Setpoint Error and Margin Diagram (for increasing parameter function)



The licensee's methodology follows an arithmetic summation method, which considers the uncertainty allowance of the RCA and RD in reference to the actual trip setpoint Upper Calibration Limit (UCL). The licensee states its calibration tolerance has been reduced to a value that bounds its expected RD for an operable instrument channel. This results in an expectation where the AV is limited to the RD calculated from the Lower Calibration Limit (LCL) of the PTS.

The NRC staff determined that if the licensee maintains its proposed settings at the PTS, the values proposed for the AVs would be more conservative than those described as the nonconservative direction of the “as-found tolerance” depicted in figure 1 of ANSI/ISA 67.04.012018.

The NRC staff notes that, in reviewing certain TS-controlled calculated instrument channel uncertainties in WCAP-13751-P, the methodology for determining AVs is as defined in the AV formula above. Through the use of this methodology, it is mathematically possible for an instrument channel’s “as found” measured value to be within the calculated drift allowance yet beyond the TS controlled instrument channel’s AV.

However, based on the NRC staff review of the TS-controlled instrument channel AVs in WCAP-13751-P, it determined that for those cases where the magnitude of departure beyond an instrument channel’s AV, the departure is unlikely and minimal. Thus, while the possibility exists for certain TS-controlled instrument channels to behave normally and slightly exceed the AV, the NRC staff notes that in the past 20 years there have been no occurrences of a TS-controlled instrument channel exceeding its AV at Farley as a result of the WCAP-13751-P calculation method. This fact is coupled with the requirement that should any TS-controlled instrument channel exceed its AV, the required corrective actions would extend beyond mere instrument channel recalibration. It would also require a licensee event report (LER) and an evaluation to “preclude recurrence” of exceeding the instrument channel’s AV in accordance with 10 CFR 50.36(c)(1)(ii)(A). In the event the “as-found” measured value of a TS-controlled instrument channel exceeds its AV, the licensee should review the guidance in RIS 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36, ‘Technical Specifications,’ Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels,” dated August 24, 2006 (ML0518100770), and potentially adapt its setpoint methodology accordingly.

Thus, when taken in totality, when using the licensee’s setpoint methodology, while the possibility exists that in some cases the TS-controlled instrument channel may exceed its AV slightly due to the combination of the full tolerance of the RCA and RD values, the NRC staff finds this approach appropriate in this application, with the licensee’s understanding that it assumes the potential for the as-found measured value to be above the AV resulting in additional corrective action steps associated with the use of this methodology.

3.1.2 Protective Function Capability

As part of its evaluation, the NRC staff performed an independent confirmatory review to determine whether the licensee’s AV calculation values are adequate to provide reasonable assurance that required protective actions initiate before the associated plant process parameters exceed their analytical limits based upon the margins established between AV and NTS. Table 1 below lists the current and revised AVs.

Table 1: *Licensee AV Modification*

RTS Function (TS Table 3.3.1-1)	Current AV	Revised AV
5	≤ 1.3 E5 cps	≤ 1.2 E5 cps
ESFAS Function (TS Table 3.3.2-1)	**Current AV**	**Revised AV**
1.c	≤ 4.5 psig	≤ 4.3 psig
1.e.1	≥ 575 psig	≥ 580 psig
1.e.2	≤ 112 psig	≤ 105 psig
2.c	≤ 28.3 psig	≤ 27.3 psig
3.b.3	≤ 28.3 psig	≤ 27.3 psig
4.c	≤ 17.5 psig	≤ 16.5 psig
4.d	≥ 575 psig	≥ 580 psig
4.e	40.3%	40.7%

*These values represent the percent change respective to their 'decreasing' and 'increasing' parameter functions.

Based on review of the licensee's calculations and NRC staff's independent confirmatory results, the NRC staff determined the following with respect to the proposed AVs:

- If the licensee maintains its proposed settings at the PTS, the values proposed for the AVs would be more conservative than those described as the non-conservative direction of the "as-found tolerance" depicted in figure 1 of ANSI/ISA 67.04.01-2018.

- The licensee incorporated TSTF-355, which takes a conservative approach by establishing a "Procedure Trip Setpoint" in relation to the NTS. The AV equation uses uncertainty terms RCA and RD, which are the same terms used to calculate the channel statistical allowance that helps support the calculation of the LSSS (i.e., AV) as required per 10 CFR 50.36(c)(1)(ii)(A). This approach maintains conformity across the variables used to evaluate trip setpoints and conservative margins for the associated protective functions.

- Per table 1 above, separate from ESFAS function 4.e, each RTS and ESFAS function AVs were adjusted closer to their associated NTS, which slightly increases a stricter channel calibration check for operability but also increases the confidence that the protection channel safety analysis limits will not be violated and supports defense-in-depth designs for protection channels with unspecified safety analysis limits (TS table 3.3.1-1, Function 5; and TS table 3.3.2-1, Functions 1.e.2, and 4.e). The change for ESFAS Function 4.e is minor and does not delineate from accuracy improvements set by older WCAP13751P revisions.

- The values described in columns 16 and 17 of table 3-32A, per WCAP-13751-P, are in engineering units (i.e., reactor thermal power (RTP), pounds per square inch gauge (psig), Flow, Fahrenheit, etc.). Regarding ESFAS table 3.3.2-1, Function 4.e (Specifically linked to Protection Channel row 26 in table 3-32A), the 40.7 percent AV is in "%

Nominal Flow." To verify the new proposed AV value for Function 4.e, a simple span conversion is not used for accuracy reasons. The licensee provides a detailed conversion method outlined in table 3-35 of its LAR, which provides the conversion product Eq. 3-35.7. The specific focus is the product $(\frac{Fmax}{Fn})$^2 in Eq. 3-35.7, which is used to convert "% span" value uncertainty terms, such as RCA and RD, to the appropriate engineering unit "% Nominal Flow" used for Function 4.e. The final conversion layout to obtain AV becomes: AV = NTS – [(2*RCA) + RD] * $(\frac{Fmax}{Fn})$^2.-

The NRC staff notes per section 4.3, "Application to the Technical Specifications," of WCAP-13751-P, the licensee follows three criteria when evaluating a channel calibration for the associated plant rack and field device. Once the as-found measured value is recorded, these acceptance criteria determine if the channel is operable, operable with need to adjust, or inoperable with a need for further investigation/evaluation, based on as-found trip setpoints in reference to their calibration tolerance and the AV.

3.2 Technical Conclusion

The NRC staff determined that if the licensee maintains its proposed settings at the PTS, the values proposed for the AVs would be more conservative than those described as the nonconservative direction of the "as-found tolerance" depicted in figure 1 of ANSI/ISA 67.04.01-2018. The AVs established and surveillance testing will continue to be performed to ensure continued TS controlled instrument channel operability.

The LAR requested approval of the AVs and not the underlying methodology used to calculate the values. Therefore, the NRC staff approval of the proposed AVs does not generically endorse the WCAP setpoint methodology but, for the reasons outlined in section 3.1.1 of this SE, the proposed changes maintain sufficient margin for the protection system to automatically actuate safety-related systems and components. Therefore, the changes continue to meet GDC 20 by ensuring automatic protective actions are initiated before safety limits are reached. The NRC staff also concludes that the requirements of 10 CFR 50.36(c)(1)(ii)(A) and 10 CFR 50.36(c)(3), "Surveillance requirements," remain satisfied because adequate margin for automatic protective action will continue to exist prior to exceeding established safety limits.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC notified the State of Alabama official of the proposed issuance of the amendments on April 7, 2026. On April 8, 2026, the State official confirmed that the State of Alabama had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

This action relates to authorizations under 10 CFR Part 50 with respect to installation or use of a facility component and changes to surveillance requirements. The NRC staff has determined that any ground disturbance is limited to previously disturbed areas. Additionally, the NRC staff has determined that the action involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, no significant increase in individual or cumulative public or occupational radiation exposure, and no significant increase in the potential for or consequences from radiological accidents. Finally, the NRC staff has determined that a categorical exclusion applies and that special circumstances under 10 CFR 51.22, "Categorical exclusions," are not present that would preclude reliance on the categorical exclusion. Accordingly, this action meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(d)(1) and (d)(8). Pursuant to 10 CFR 51.22, no environmental impact statement or environmental assessment need be prepared in connection with the action.

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.

Date: August 14, 2026

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2, ISSUANCE OF AMENDMENT NOS. 261 AND 258, TO REVISE TECHNICAL SPECIFICATION 3.3.1 AND 3.3.2 ALLOWABLE VALUES (EPID L-2025-LLA-0145) DATED AUGUST 14, 2026

DISTRIBUTION:
PUBLIC
RidsNrrPMFarley Resource
RidsNrrLAPBlechman Resource
RidsACRS_MailCTR Resource
RidsRgn2MailCenter Resource
RidsNrrDeIceb Resource
RidsDrssNebb Resource
RidsDrssTspsb Resource
RidsNrrDlpOrlb2 Resource

ADAMS Accession Nos.:
Package: ML26210A316
Amendment: ML26210A317
eConcurrence Case: 20260729-01598