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

May 17, 2016

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
P.O. Box 1295
Bin 038
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATION 3.4.14 (CAC NOS. MF6687 AND MF6688)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 201 to Renewed Facility Operating License No. NPF-2 and Amendment No. 197 to Renewed Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated August 31, 2015, as supplemented by letters dated January 28, 2016, and March 11, 2016.

The amendments revise TS 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," to eliminate the requirements for the residual heat removal system suction valve auto closure interlock function.

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

Sincerely,

Shawn Williams

Shawn A. Williams, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 201 to NPF-2
2. Amendment No. 197 to NPF-8
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SOUTHERN NUCLEAR OPERATING COMPANY

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 201
Renewed License No. NPF-2

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

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 1, Renewed Facility Operating License No. NPF-2 filed by Southern Nuclear Operating Company (the licensee), dated August 31, 2015, as supplemented by letters dated January 28, 2016, and March 11, 2016, 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. 201, are hereby incorporated in the renewed facility operating 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 prior to the first entry into Mode 4 following the end-of-cycle refueling outage 27 (scheduled for fall 2016).

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: May 17, 2016

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

SOUTHERN NUCLEAR OPERATING COMPANY

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 197
Renewed License No. NPF-8

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

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 2, Renewed Facility Operating License No. NPF-8 filed by Southern Nuclear Operating Company (the licensee), dated August 31, 2015, as supplemented by letters dated January 28, 2016, and March 11, 2016, 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. 197, are hereby incorporated in the renewed facility operating 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 prior to the first entry into Mode 4 following the end-of-cycle refueling outage 25 (scheduled for fall 2017).

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: May 17, 2016

ATTACHMENT TO

AMENDMENT NO. 201 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

AND AMENDMENT NO. 197 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
NPF-2, page 4	NPF-2, page 4
NPF-8, page 3	NPF-8, page 3
TSs	TSs
3.4.14-2	3.4.14-2
3.4.14-3	3.4.14-3

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 201, 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) Maximum Power Level

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 197, 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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	<u>AND</u>		
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time for Condition A not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours
-------------NOTE--------------- Not applicable to the autoclosure interlock for Unit 1 after restart from 1R27 and for Unit 2 after restart from 2R25. ------------------------------------			
C. RHR System autoclosure or open permissive interlock function inoperable.	C.1	Place the affected valve(s) in the closed position and maintain closed under administrative control.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	-------------------------NOTES------------------------- 1. Not required to be performed in MODES 3 and 4. 2. Not required to be performed on the RCS PIVs located in the RHR flow path when in the shutdown cooling mode of operation. 3. RCS PIVs actuated during the performance of this Surveillance are not required to be tested more than once if a repetitive testing loop cannot be avoided. ---	
	Verify leakage from each RCS PIV is equivalent to $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm at an RCS pressure $\geq$ 2215 psig and $\leq$ 2255 psig.	18 months, prior to entering MODE 2 AND Following valve actuation due to automatic or manual action or flow through the valve (except for RCS PIVs located in the RHR flow path)
SR 3.4.14.2	-------------------------NOTES------------------------- 1. Not required to be met when the RHR System valves are required open in accordance with SR 3.4.12.3. 2. Not applicable to Unit 1 after restart from 1R27 and not applicable to Unit 2 after restart from 2R25. ---	
	Verify RHR System autoclosure interlock causes the valves to close automatically with a simulated or actual RCS pressure signal $\geq$ 700 psig and $\leq$ 750 psig.	In accordance with the Surveillance Frequency Control Program

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 201 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 197 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 application dated August 31, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15261A673), as supplemented by letters dated January 28, 2016 (ADAMS Accession No. ML16028A055), and March 11, 2016 (ADAMS Accession No. ML16071A433), Southern Nuclear Operating Company (SNC, the licensee) submitted a request to change the Joseph M. Farley Nuclear Plant (FNP), Units 1 and 2, Technical Specifications (TSs). The supplemental letters dated January 28, 2016, and March 11, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 27, 2015 (80 FR 65815).

The proposed amendments would revise TS 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," to eliminate the requirements for the residual heat removal (RHR) system suction valve auto closure interlock function. The proposed changes would eliminate the current requirement to perform the RHR auto closure interlock Surveillance Requirement (SR) 3.4.14.2 and would revise Action Condition C to support elimination of the RHR auto closure interlock at FNP. The revised Action Condition C would only apply to the open permissive interlock (OPI).

2.0 REGULATORY EVALUATION

2.1 System Description

During normal and emergency conditions, the low pressure RHR system (design pressure of 600 pounds per square inch gauge (psig)) is isolated from the high pressure reactor coolant system (RCS) (normal operating pressure of 2235 psig). Isolation is necessary to (1) avoid RHR system over pressurization, and (2) minimize the potential for loss of integrity of the low pressure system and possible radioactive releases to the environment.

Enclosure 3

Two suction/isolation valves are provided on each inlet line from the RCS to the RHR system inside containment. These motor-operated gate valves are normally closed to keep the low pressure RHR system isolated from the high pressure RCS and are opened only when the RHR system is in operation. The RHR suction isolation valves are interlocked with RCS pressure signals to prevent opening when the RCS pressure is greater than the current OPI setpoint of 402.5 psig and automatically close when the RCS pressure increases above the autoclosure interlock (ACI) setpoint of 700 psig. Thus, the OPI prevents inadvertent opening of the RHR system isolation valves when the RCS pressure is above the valve opening setpoint, and the ACI ensures that the RHR system isolation valves are closed when the RCS is pressurized above the valve closing setpoint. The OPI will not be affected by the proposed removal of the RHR system ACI.

2.2 Description of Proposed Changes

The license amendment request (LAR) proposes to eliminate the TS and SR requirements for the RHR system suction valve ACI function and to add TS notes that state when the current TS requirement is no longer applicable due to the proposed staggered (by unit) implementation.

The proposed change would insert a note in current SR 3.4.14.2. SR 3.4.14.2 requires the verification of RHR system ACI function. The note would state:

> Not applicable to Unit 1 after restart from 1R27 and not applicable to Unit 2 after restart from 2R25.

In addition, the proposed change would insert a note in TS 3.4.14 Action Condition C to eliminate the reference to the RHR system ACI in that condition. The note would state:

> Not applicable to the autoclosure interlock for Unit 1 after restart from 1R27 and for Unit 2 after restart from 2R25.

The applicability of TS 3.4.14 states, "Modes 1, 2, and 3, Mode 4, except valves in the residual heat removal (RHR) flow path when in, or during the transition to or from, the RHR mode of operation." The requirements of TS 3.4.14, including the proposed Action Condition C and SR notes (described above) would not become applicable until the mode of applicability of TS 3.4.14 is entered (i.e., in Mode 4 with the transition from RHR cooling complete). As such, the proposed ACI elimination, alarm installation, and required procedure changes are proposed to be completed prior to entering the applicability of TS 3.4.14, after the applicable refueling outage for each unit.

The licensee stated the RHR system ACI provides an automatic closure for the RHR system suction isolation valves on high RCS pressure; however, rapid overpressure protection of the RHR system is provided by the RHR relief valves (located inside containment) and not by the slow acting suction isolation valves. Ultimately, RHR system overpressure protection is not impacted by the removal of the ACI feature because the functionality of the RHR system ACI feature will be maintained by the adoption of five WCAP-11736 safety evaluation report (SER) recommendations, which are discussed in the following subsection. The removal of the RHR system ACI minimizes the potential for spurious valve closure, which could result in a loss of the

decay heat removal function, RHR system pump damage, and the inability of the RHR system to perform its function of RCS cold over pressurization protection.

2.3 Previous NRC Approval of RHR System ACI Removal and Applicability to FNP

In WCAP-11736, "Residual Heat Removal System Autoclosure Interlock Removal Report for the Westinghouse Owner's Group [(WOG)]," issued on August 8, 1989 (ADAMS Legacy Library Accession No. 8911150023), Westinghouse evaluated the removal of the ACI from RHR system suction/isolation valves at four reference plants: the Salem Nuclear Generating Station, Unit No. 1; Callaway Plant, Unit No. 1; North Anna Power Station, Unit No. 1; and Shearon Harris Nuclear Power Plant, Unit 1 (Shearon Harris). The WOG plants participating in the program were categorized into one of four groups based on RHR system configuration and design characteristics that are similar to one of the four reference plants. The choice of the four particular reference plants was intended to provide the maximum number of the other WOG members with the best possible fit, should they choose to delete their ACI in the future and reference WCAP-11736. FNP was categorized as being part of Group 4 corresponding to Shearon Harris. The NRC staff's SER of WCAP-11736 concluded that a net safety benefit would result from removal of the ACI, provided that five recommendations delineated in the SER are implemented. The referenced recommendations are listed below.

1. An alarm will be added to each RHR suction valve, which will actuate if the valve is open and the pressure is greater than the open permissive setpoint and less than the RHR system design pressure minus the RHR pump head pressure.

2. Valve position indication to the alarm must be provided from the stem-mounted limit switches, and power to the limit switches must not be affected by power lockout of the valve.

3. The procedural improvements described in WCAP-11736 should be implemented. Procedures themselves are plant-specific.

4. Where feasible, power should be removed from the RHR suction valves prior to their being leak-checked.

5. The RHR suction valve operators should be sized so that the valves cannot be opened against full system pressure.

In addition, the SER concluded that the information contained in WCAP-11736 may be referenced to supplement licensee plant-specific submittals requesting removal of the ACI. However, such reference would only be used for compliance with those items that are generic to the WOG plants.

2.4 Regulatory Requirements

Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licenses to include TSs as part of the license. In Section 50.36, "Technical specifications," of Title 10 of the *Code of Federal Regulations* (10 CFR), the NRC established its regulatory requirements related to the content of TSs. Pursuant to

10 CFR 50.36, TSs are required to include items in the following five specific categories related to plant operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation (LCOs); (3) SRs; (4) design features; and (5) administrative controls.

With regard to LCOs, the regulation in 10 CFR 50.36(c)(2)(i) states, in part:

> Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

With regard to SRs, the regulation in 10 CFR 50.36(c)(3) states:

> 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.

Part 50 to 10 CFR establishes the fundamental regulatory requirements with respect to the domestic licensing of nuclear production and utilization facilities. The "Introduction" section to Appendix A to 10 CFR Part 50 states, in part:

> Under the provisions of [10 CFR] 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. ... The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.
>
> These General Design Criteria [(GDC)] establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission.

Those GDC applicable to the submitted LAR include:

> GDC 13, "Instrumentation and control," which states:
>
>> Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

GDC 14, "Reactor coolant pressure boundary," which states:

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 19, "Control room," which states, in part:

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

.... [H]olders of operating licenses using an alternative source term under [10 CFR] 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv [sievert] (5 rem) total effective dose equivalent (TEDE) as defined in [10 CFR] 50.2 for the duration of the accident.

Furthermore, with regard to the applicability of GDC 19 to the submitted LAR, NUREG-0800, Standard Review Plan (SRP) Section 5.4.7, "Residual Heat Removal (RHR) System," numbered item 4 under the "Technical Rationale" subsection states:

GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit during both normal operating and accident conditions, including the [loss-of-coolant accident]. [Branch Technical Position (BTP)] 5-4 provides guidance for compliance with GDC 19 with regard to achieving cold shutdown from the control room using only safety-grade equipment. The RHR system is required for safe shutdown and cooldown of the reactor during normal and accident conditions. Compliance with GDC 19 enhances plant safety by ensuring the availability of adequate instrumentation and controls in the control room to perform the required safety functions of the RHR system under all anticipated conditions.

GDC 20, "Protective system functions," which states:

> The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC 30, "Quality of reactor coolant pressure boundary," which states:

> Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

GDC 34, "Residual heat removal," which states:

> A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

3.0 TECHNICAL EVALUATION

3.1 Background

The licensee prepared the LAR in accordance with the NRC-approved WCAP-11736. Enclosed with the submittal was, "Residual Heat Removal System Autoclosure Interlock Removal Report for the Joseph M. Farley Nuclear Plant Units 1 and 2," which addresses the implementation of WCAP-11736 at FNP and the differences between FNP and its reference plant.

FNP is one of the Group 4 plants referenced by WCAP-11736 as being a participant in the program for removal of ACI functions. The reference plant for Group 4 is Shearon Harris and the ACI function at Shearon Harris was removed under License Amendment No. 24 (ADAMS Accession No. ML020580201). Therefore, this NRC-approved report was considered as part of the basis for justifying the licensee's proposed action.

Section 2.6, "Probabilistic Risk Assessment of the Event V Sequence and Safety Analyses of Transients," of the NRC staff's WCAP-11736 SER states:

> The staff has no requirements based on the absolute values in the PRA [probabilistic risk assessment] analyses and will not require a plant-specific PRA from each licensee proposing to remove the ACI. However, the licensee should do sufficient PRA and safety analyses to ensure that its plant will not show results that will invalidate the conclusions of WCAP-11736.

The licensee further states that, "Removal of the RHR ACI is not a risk-informed application based on the requirements provided in RG [Regulatory Guide] 1.174 but a deterministic assessment with probabilistic insights."

The licensee provided a PRA and a safety analysis in its August 31, 2015, application, consistent with Section 2.6 of WCAP-11736. However, because the PRA was not submitted under Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," the NRC staff did not use the risk information as a basis for a staff decision. Instead, the NRC staff review was based on a deterministic evaluation of the licensee's application of the approved methodology in WCAP-11736 to FNP. Specifically, the staff reviewed the licensee's application to ensure that the five recommendations delineated in the WCAP-11736 SER will be implemented with the removal of the ACI at FNP. Where operating experience indicated that a deviation from one of the five recommendations would be beneficial, the licensee provided adequate justification for this deviation, which is discussed further below.

Operating experience has shown that ACI circuitry associated with RHR system suction (or isolation) valves has the potential to fail, causing inadvertent isolation of the RHR system and subsequent loss of decay heat removal during shutdown and refueling operations. This loss of cooling capability could result in a loss of low temperature overpressure protection (LTOP), potentially leading to failure of the reactor coolant pressure boundary (RCPB). An NRC staff-approved generic methodology for ACI removal documented in WCAP-11736 concludes that RHR system availability is increased, and plant risk is reduced, by removing the ACI. The NRC staff findings in the WCAP-11736 SER, which FNP has considered and plans to implement as discussed in its LAR, support removal of the ACI and are directly applicable to FNP.

The intent of the ACI at FNP is to provide automatic closure of the RHR system suction valves when pressure exceeds a setpoint of 700 psig so that over-pressurization of the RHR system does not occur. Action Condition C of FNP TS LCO 3.4.14 currently requires action to be taken if the ACI is inoperable. Additionally, SR 3.4.14.2 currently requires that verification be performed to test the functionality of the ACI. The proposed revisions would remove all references to the ACI, based on removal of the ACI feature from the FNP design. The NRC staff notes that NUREG-0800, BTP 5-4, "Design Requirements of the Residual Heat Removal System," contains an RHR system isolation requirement that states:

> The valves should have independent diverse interlocks to protect against one or both valves being open during an RCS increase above the design pressure of

the RHR system, to the extent that such interlocks will not degrade high system reliability during shutdown operations (see Generic Letter 88-17).

Removal of the ACI at FNP, and adoption of the five WCAP-11736 SER recommendations, is consistent with this BTP 5-4 requirement, as the net change is expected to increase system reliability during shutdown and refueling operations, while maintaining the same functionality as found in WCAP-11736.

Operation in Modes 1, 2, 3, and 4 are not affected by removal of the ACI, as RHR system isolation is maintained by the RHR suction valve OPI that protects the integrity of the RHR system at high pressure. In its LAR, the licensee notes that the OPI, which prevents the opening of RHR suction valves when RCS pressure signals indicate a pressure greater than 402.5 psig, is not affected by the proposed TS changes.

To replace the ACI function of RHR system isolation upon pressurizing the RCS when transitioning out of shutdown operations during plant startup, a control room alarm will be added, alerting operators if any RHR system suction valve is not fully closed. This alarm will ensure that all RHR system suction valves will be closed before proceeding with plant startup. Therefore, FNP will rely on operator action and enhanced operator training instead of the ACI. The licensee stated that it will implement the following to address the five WCAP-11736 recommendations:

1. An alarm will be added to each RHR suction isolation valve, which will actuate if the valve is open and the RCS pressure is greater than the open permissive setpoint and less than the RHR system design pressure minus the RHR pump head pressure at minimum flow.

2. Valve position indication to the alarm will be provided from the stem-mounted limit switches and power to the stem mounted limit switches will not be affected by power lockout of the valve.

3. Alarm response procedures will be implemented to support the addition of the alarm for the RHR suction isolation valves, and other procedures will be revised as necessary to address the deletion of the ACI.

4. Procedures will be revised to eliminate the current requirement to lockout power to the open RHR suction isolation valves below 180°F.

5. Procedures will be implemented to require that power to all four closed RHR suction isolation valves be locked out in Modes 1, 2, and 3.

3.2 <u>NRC Staff Evaluation</u>

3.2.1 WCAP-11736 Recommendations

The NRC staff's SER of WCAP-11736 concluded that a net safety benefit would result from removal of the ACI, provided that five recommendations delineated in the WCAP-11736 SER are implemented. The licensee addressed the five WCAP-1136 SER recommendations. The

NRC staff's review of this LAR is based on confirming that the licensee has adequately addressed the five recommendations.

WCAP-11736 Recommendation No: 1: An alarm will be added to each RHR suction valve, which will actuate if the valve is open and the pressure is greater than the open permissive setpoint and less than the RHR system design pressure minus the RHR pump head pressure.

FNP proposal to address Recommendation No. 1:

A control room alarm will be added which will alert operators if an RHR System suction isolation valve is open and the RCS pressure exceeds the alarm setpoint. This setpoint will be greater than the open permissive setpoint and less than the RHR System design pressure minus the RHR System pump head pressure at minimum flow.

NRC staff evaluation of Recommendation No. 1: The staff reviewed the FNP proposal to address Recommendation No. 1 and determined that it is acceptable because it meets the recommendation to add an alarm to each RHR suction valve with a setpoint greater than the open permissive setpoint and less than the RHR system design pressure minus the RHR system pump head pressure at minimum flow.

WCAP-11736 Recommendation No. 2: Valve position indication to the alarm must be provided from the stem-mounted limit switches, and power to the stem-mounted limit switches must not be affected by power lockout of the valve.

FNP proposal to address Recommendation No. 2:

The four RHR System suction isolation valves for each unit will utilize the existing limit switches located in the valve operator for valve position indication to the new alarm. These limit switches are actuated by a gear arrangement off the motor actuator rotor shaft. The new contacts on the existing limit switches utilized for position indication to the new alarms are different from the limit switch contacts which presently provide valve position to the main control board. As a result, diversity in valve position indication is achieved. In addition, the alarm circuit is powered by a supply which is separate from the supply that powers the valve control and position indication circuits. Thus, the alarm will remain functional during a power lockout of the valve.

NRC staff evaluation of Recommendation No: 2: The staff reviewed the FNP proposal to address Recommendation No. 2 and determined that it is acceptable because it meets the recommendation to use stem-mounted limit switches to provide valve position indication to the new alarm, and power to the limit switches will not be affected by power lockout of the valve.

WCAP-11736 Recommendation No. 3: The procedural improvements described in WCAP-11736 should be implemented. Procedures themselves are plant-specific.

FNP proposal to address Recommendation No. 3:

> Plant procedures will be reviewed and revised as appropriate to reflect the deletion of the RHR System ACI. Procedures will also be revised to address appropriate operator response to the control room alarm which is being added as part of this modification.

NRC staff evaluation of Recommendation No. 3: The staff reviewed the FNP proposal to address Recommendation No. 3 with respect to the WCAP-11736 generic procedural improvements and issued requests for additional information (RAIs) on December 17, 2015 (ADAMS Accession No. ML15337A484), and February 16, 2016 (ADAMS Accession No. ML16041A360). The licensee responded with supplemental letters dated January 28, 2016, and March 11, 2016.

Section 2.3, "Procedural Changes," of the WCAP-11736 SER, states that, "WCAP-11736 proposes generic procedural requirements." However, with respect to Recommendation No. 3, the NRC staff noted that the WCAP-11736 SER also states that, "The staff agrees with this generic guidance assuming a surveillance procedure for the [RHR System] suction valve alarms is added to ensure these alarms remain operable." Therefore, the NRC staff issued an RAI to the licensee to provide a description of the mentioned surveillance procedure for the new RHR system suction valve alarm at FNP. In its January 28, 2016, supplemental letter, the licensee focused on the new annunciator response procedure rather than on the surveillance procedure. Therefore, the NRC staff issued a followup RAI requesting that the licensee describe the relevant surveillance procedure (e.g., calibration procedure) that will apply to the new alarm to ensure that it will remain functional. In its March 11, 2016, supplemental letter, the licensee discussed the RCS pressure transmitters that currently provide the pressure signals for the RHR system ACI and will subsequently provide the pressure signals for the new alarms.

The licensee specifically mentioned the calibration procedures for each of the pressure transmitters for each unit that will be revised and the frequency at which they are currently performed. The licensee further explained that there is currently a section in each of these procedures that verifies that the expected response occurs when RCS pressure approaches the current ACI setpoint, and that these sections will be updated to ensure that, instead of valve autoclosure at the ACI setpoint, the new alarm will actuate at the desired setpoint. The licensee identified the following plant-specific procedures that will be revised to include calibration and functional testing of the new alarm: FNP-1-STP-201.16, FNP-1-STP-201.17, FNP2-STP-201.16, and FNP-2-STP-201.17.

The NRC staff finds that the licensee has adequately described the relevant surveillance procedures associated with the new alarm, including the specific procedures that will be updated, and the process by which they will be updated at FNP. Therefore, the NRC staff finds the surveillance procedure associated with the new alarm, consistent with the WCAP-11736 SER stipulation discussed above, to be acceptable.

The NRC staff also reviewed the FNP proposal to address Recommendation No. 3 with respect to the plant-specific procedural improvements and determined that it is acceptable because it meets the recommendation to review and revise plant-specific procedures as appropriate, including operator response procedures related to the new alarm.

WCAP-11736 Recommendation No. 4: Where feasible, power should be removed from the RHR suction valves prior to their being leak-checked.

FNP proposal to address Recommendation No. 4:

> Technical Specification 3.4.14, "RCS Pressure Isolation Valve Leakage," contains the requirements for leakage testing for the RHR System suction Isolation valves. SR 3.4.14.1 specifies the leak testing requirements for the pressure isolation valves. FNP will continue to verify the RHR suction isolation valve leakage is within the required limits in accordance with SR 3.4.14.1. Ensuring proper valve position will continue to be accomplished by use of valve position indication and administrative controls.

NRC staff evaluation of Recommendation No. 4: The staff reviewed the FNP proposal to address Recommendation No. 4. The licensee has deviated from the recommendation to remove power from the RHR suction valves prior to their being leak-checked. However, the licensee's response is acceptable because, as documented in a February 17, 1988, SER for the Diablo Canyon Nuclear Power Plant ACI removal (ADAMS Legacy Accession No. 8802240328), "Both the staff and the licensee agreed that [removal of power to isolation valves during shutdown] would be a bad practice since the valves would not be available to perform their isolation function should the need arise during shutdown."

Furthermore, the RHR system provides an LTOP safety function during shutdown and refueling operations by providing pressure relief to the pressure relief tank inside containment. Previously, power lockout of the RHR system suction valves was implemented at FNP to reduce the frequency of inadvertent valve closure due to the presence of the ACI. With the removal of the ACI circuitry, FNP is also removing the requirement to lockout power to the RHR system suction valves. LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," requires that two RHR suction relief valves with setpoints less than or equal to 450 psig be available, or alternatively, that the RCS be depressurized with an RCS vent of greater than or equal to 2.85 square inches in Modes 4, 5, and 6. The presence of LCO 3.4.12 does not inherently prevent the closure of the two RHR suction relief valves during Modes 4, 5, and 6; appropriate plant procedures are also required to ensure that the LCO will be met. With power no longer being removed and manual valve closure possible, the NRC staff requested that the licensee describe plant procedures associated with the control of the RHR suction relief valves to provide assurance that isolation will not occur during Modes 4, 5, and 6. In its January 28, 2016, supplemental letter, the licensee provided assurance that appropriate operator training has been adopted and that appropriate procedures will be implemented to ensure that RHR system relief valves remain operable in Modes 4, 5, and 6. The licensee also notes, in its LAR, that failure of a pressure transmitter or loss of power to the solid state protection system would result in the RHR System suction valves remaining open allowing the RHR System to maintain its LTOP safety function.

Based on the discussion above, the NRC staff finds that the licensee has adequately addressed Recommendation No. 4.

<u>WCAP-11736 Recommendation No. 5</u>: The RHR suction valve operators should be sized so that the valves cannot be opened against full system pressure.

> FNP Proposal to Address Recommendation No. 5:
>
> The motors for the RHR suction valve operators are sized to open against a differential pressure of 700 psid. However, the ability of the MOV to open is dependent on parameters such as supplied voltage and friction coefficients. Based on the fact that the valves have a small motor sized for less than 1/3 full RCS system pressure, even with full voltage and a conservative stem coefficient of friction, there is reasonable assurance that the MOV will not open at full RCS system pressure. No credit was taken for the capability to open the valve against full system pressure in either the generic analysis of WCAP-11736 or the FNP specific evaluations in Enclosure 2 of this LAR. Furthermore, power will be removed from all four of these valves In Modes 1, 2, and 3, and the OPI will continue to function to prevent opening of these valves when RCS pressure is greater than 402.5 psig.

NRC staff evaluation of Recommendation No. 5: The staff reviewed the FNP proposal to address Recommendation No. 5 and determined that it is acceptable because the valves in question are designed to open at less than one-third of RCS system pressure, meaning that they are not sized to be opened against full RCS system pressure. Therefore, the FNP RHR suction valve operator sizing meets the recommendation that the RHR suction valve operators be sized so that the valves cannot be opened against full RCS system pressure.

Furthermore, procedures will be implemented to require that power to all four closed RHR suction isolation valves be locked out in Modes 1, 2, and 3, and the OPI will continue to function to prevent the opening of these valves when RCS pressure is greater than 402.5 psig, providing additional defense-in-depth to Recommendation No. 5.

3.2.2 Summary of Regulatory Compliance

Based on the review of the RHR ACI removal LAR, the NRC staff concludes that:

- The instrumentation and control requirements of GDC 13 have been satisfied by including an alarm on each RHR suction isolation valve, which will actuate if the valve is open and the RCS pressure is greater than the open permissive setpoint and less than the RHR system design pressure minus the RHR pump head pressure at minimum flow.

- The RCPB requirements of GDC 14 have been satisfied because the leakage from the RHR system suction isolation valves will continue to be tested and verified in the same manner as before the proposed change.

- The control room requirements of GDC 19 have been satisfied based on the BTP 5-4 guidance for compliance with GDC 19 with regard to achieving cold shutdown from the control room using only safety-grade equipment, as the removal of this interlock is expected to increase system reliability during shutdown and refueling operations as was found in WCAP-11736.

- The protective system function requirements of GDC 20 have been satisfied because the removal of the RHR system ACI will not adversely affect the ability of the plant instrumentation and systems to assure that the specified acceptable fuel design limits are not exceeded and to respond to accident conditions and initiate the operation of systems and components important to safety.

- The quality of RCPB requirements of GDC 30 have been satisfied because the leakage from the RHR system suction isolation valves will continue to be tested and verified in the same manner as before the proposed change.

- The RHR requirements of GDC 34 have been satisfied because the removal of the RHR system ACI does not adversely affect the capability of the RHR system to perform its intended safety function. The removal of the RHR system ACI minimizes the possibility of spurious valve closure. Rapid overpressure protection of the RHR system is provided by the RHR relief valves and not by the slow acting suction isolation valves. The RHR system overpressure protection is not affected by the removal of the ACI feature. Therefore, the NRC staff finds that the RHR ACI removal satisfies the requirements of GDC 34.

- The TS requirements of 10 CFR 50.36 have been satisfied. The proposed change eliminates the requirement to perform the SR for the RHR ACI since the ACI is being removed. Rapid overpressure protection of the RHR system is provided by the RHR relief valves and not by the slow acting suction isolation valves. The RHR system overpressure protection is not affected by the removal of the ACI feature. As such, the proposed change does not adversely affect the RHR system's capability to maintain facility operation within the required safety limits.

3.3 NRC Staff Conclusion

The NRC staff finds that the proposed changes to TS 3.4.14 and SR 3.4.14.2, which remove reference to and reliance on the RHR system ACI function, including the addition of notes indicating the applicability to FNP, Units 1 and 2, are acceptable because the licensee has adequately addressed the five WCAP-11736 SER recommendations described in the preceding section as part of the NRC staff's generic approval of WCAP-11736. The staff concludes that the proposal satisfies the requirements of 10 CFR 50.36, and that the systems will continue to meet the requirements of GDCs 13, 14, 19, 20, 33, and 34, and that there is reasonable assurance that the equipment will continue to perform its required safety functions.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of Alabama official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change

surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on October 27, 2015 (80 FR 65815). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributors: Amrit Patel
Daniel Warner

Date: May 17, 2016

May 17, 2016

Mr. C. R. Pierce
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
P.O. Box 1295
Bin 038
Birmingham, AL 35201-1295

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS TO REVISE TECHNICAL SPECIFICATION 3.4.14 (CAC NOS. MF6687 AND MF6688)

Dear Mr. Pierce:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 201 to Renewed Facility Operating License No. NPF-2 and Amendment No. 197 to Renewed Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated August 31, 2015, as supplemented by letters dated January 28, 2016, and March 11, 2016.

The amendments revise TS 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," to eliminate the requirements for the residual heat removal system suction valve auto closure interlock function.

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

Sincerely,
/RA/
Shawn A. Williams, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 201 to NPF-2
2. Amendment No. 197 to NPF-8
3. Safety Evaluation

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