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

August 9, 2022

Ms. Kim Maza
Site Vice President
Shearon Harris Nuclear Power Plant
5413 Shearon Harris Road
Mail Code NHP0
New Hill, NC 27562-9300

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT NO. 194 TO REVISE TECHNICAL SPECIFICATIONS RELATED TO REACTOR PROTECTION SYSTEM INSTRUMENTATION (EPID L-2021-LLA-0149)

Dear Ms. Maza:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued Amendment No. 194 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). This amendment is in response to your application dated August 6, 2021.

The amendment revises Technical Specification (TS) 3.3.1, "Reactor Trip System Instrumentation," to adjust the reactor trip on turbine trip interlock from P-7 (Low Power Reactor Trips Block) to P-8 (Power Range Neutron Flux).

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

Sincerely,

/RA/

Michael Mahoney, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 194 to NPF-63
2. Safety Evaluation

cc: Listserv

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 194
Renewed License No. NPF-63

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

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated August 6, 2021, 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 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-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 194, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to the startup of the Shearon Harris Nuclear Power Plant, Unit 1, cycle 25.

FOR THE NUCLEAR REGULATORY COMMISSION

Lucas E. Haeg

Digitally signed by Lucas E. Haeg
Date: 2022.08.09 07:21:39 -05'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility License No. NPF-63 and Technical Specifications

Date of Issuance: August 9, 2022

ATTACHMENT TO LICENSE AMENDMENT NO. 194

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following page of the Renewed Facility Operating License with the revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change:

Remove	Insert
Page 4	Page 4

Replace (or remove) the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
3/4 3-4	3/4 3-4

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

(1) Maximum Power Level

Duke Energy Progress, LLC, is authorized to operate the facility at reactor Core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 194, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, LLC. shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company* shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at 15.6.3 Subparts II (1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company* will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

[1]The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.
* On April 29, 2013, the name of “Carolina Power & Light Company” (CP&L) was changed to “Duke Energy Progress, Inc.” On August 1, 2015, the name “Duke Energy Progress, Inc.” was changed to “Duke Energy Progress, LLC.”

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16.	Underfrequency--Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6
17.	Turbine Trip (Above P-8)					
	a. Low Fluid Oil Pressure	3	2	2	1	6
	b. Turbine Throttle Valve Closure	4	4	1	1	10
18.	Safety Injection Input from ESF	2	1	2	1, 2	13
19.	Reactor Trip System Interlocks					
	a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	7
	b. Low Power Reactor Trips Block, P-7					
	1) P-10 Input	4	2	3	1	7
	or					
	2) P-13 Input	2	1	2	1	7
	c. Power Range Neutron Flux, P-8	4	2	3	1	7
	d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
	e. Turbine Inlet Pressure, P-13	2	1	2	1	7

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 194 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DUKE ENERGY PROGRESS, LLC

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC) dated August 6, 2021 (ADAMS Accession No. ML21218A197), Duke Energy Progress, LLC (licensee) proposed changes to the Technical Specifications (TSs) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP). Specifically, the proposed amendment would revise TS 3.3.1, “Reactor Trip System Instrumentation,” to adjust the reactor trip on turbine trip interlock from P-7 (Low Power Reactor Trips Block) to P-8 (Power Range Neutron Flux).

2.0 REGULATORY EVALUATION

2.1 System Design

2.1.1 Reactor Trip System (RTS)

The purpose of the RTS is to limit the consequences of American Nuclear Society (ANS) Condition II events (faults of moderate frequency such as loss of feedwater flow) to, at most, a shutdown of the reactor and turbine. The RTS limits plant operation to ensure that the reactor safety limits are not exceeded during ANS Condition II events and that these events can be accommodated without developing into more severe conditions. As described in Section 7.2.1.1 of the HNP Updated Final Safety Analysis Report (UFSAR), the RTS automatically keeps the reactor operating within a safe region by shutting down the reactor whenever the limits of the region are exceeded. The reactor trip on turbine trip is actuated by two-out-of-three-logic from trip fluid pressure signals or by all-closed signals from the turbine steam stop valves. A turbine trip causes a direct reactor trip above the P-7 interlock. The licensee does not credit this trip in any of the UFSAR, Chapter 15 safety analyses.

2.1.2 RTS Interlocks

Interlock P-7 blocks a reactor trip at low power (below approximately 10 percent of full power) on a low reactor coolant flow in more than one loop. The low power signal is derived from three out of four power range neutron flux signals below the setpoint in coincidence with two out of two turbine first stage pressure signals below the setpoint (low plant load).

The P-8 interlock blocks a reactor trip when the plant is below approximately 49 percent of full power, on a low reactor coolant flow in any one loop. The block action occurs when three out of four neutron flux power signals are below the setpoint. Thus, below the P-8 setpoint, the reactor will be allowed to operate with one inactive loop and trip will not occur until two or more loops are indicating low flow.

2.1.3 Pressurizer Pressure Control System

The pressurizer pressure control system, described in HNP UFSAR, Section 7.7.1, maintains, or restores the pressurizer pressure 50 psi above or below the design pressure following normal operational transients that results in pressure changes. The proportional and backup heaters, spray, and pressurizer power operated relief valves (PORVs) maintain the pressure at the setpoint value and prevent reactor trip because of pressure variations caused by operational transients.

2.1.4 Rod Control System

The automatic rod control system, described in HNP UFSAR, Section 7.7.1, is designed to maintain a programmed average temperature in the reactor coolant by regulating the reactivity within the core.

2.1.4 Steam Dump Control System

The steam dump control system, described in HNP UFSAR, Section 7.7.1, is designed to accept a 50 percent load rejection from full power without causing reactor trip. This ensures that stored energy and residual heat are removed following a reactor trip to bring the plant to equilibrium no-load conditions without actuation of the steam generator (SG) safety valves. The steam dump system includes 14 valves which can bypass steam to the condenser and to the atmosphere.

2.2 Proposed TS Change

The proposed change would affect TS 3/4.3.1, “Reactor Trip System Instrumentation.” Specifically, Functional Unit 17, “Turbine Trip (Above P-7),” of TS Table 3.3-1 will be revised to “Turbine Trip (Above P-8).” This proposed change will modify HNP TS to change the reactor trip on turbine trip from the P-7 (Low Power Reactor Trips Block) interlock to the P-8 (Power Range Neutron Flux) interlock. Instead of blocking a reactor trip on a turbine trip signal below 10 percent of full power, the P-8 interlock would block a reactor trip on turbine trip signal below 49 percent of full power.

2.3 Regulatory Requirements

The following Title 10 of *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, “General Design Criterion” (GDC) are applicable to this license amendment request (LAR) evaluation:

2.3.1 GDC Criterion 20, “Protection system functions”

GDC 20 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.

The reactor trip actuated on a turbine trip signal is not assumed in transient and accident analyses since the turbine trip signal originated in the turbine building, which is in a non-seismically qualified area. However, the load rejection event was evaluated to address the Three Mile Island (TMI) Action Item 11.K.3.10 requirements in NUREG-0737, “Clarification of TMI Action Plan Requirements,” dated November 1980 (ML051400209), which states:

> The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss of coolant accident (LOCA) resulting from a stuck-open power operated relief valve (PORV) is substantially unaffected by the modification

The NRC staff's review of the TS changes is to assure the licensee’s compliance with the TMI Action Item II.K.3.10 requirements and compliance with regulations applicable to transient and accident analyses.

2.3.2 GDC Criterion 13, “Instrumentation and control”

GDC 13 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.

2.3.3 10 CFR 50.36, “Technical Specifications”

Paragraph (b) of 10 CFR 50.36 requires in part, each license authorizing operation will include technical specifications. The TSs will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to 10 CFR 50.34.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the following aspects for potential impacts to the current licensing bases:

- Reactor protection system design
- Instrumentation
- Technical Specifications.

3.1 GDC 20 Evaluation

The licensee identified the limiting cases for each event category discussed in the HNP UFSAR Chapter 15 safety analyses and evaluated the effect of the TS change on the LOCA and the transient analysis for each limiting case.

3.1.1 Loss Of Coolant Accidents

Event Definition: A LOCA is defined as a rupture of the reactor coolant pressure boundary in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. Following the break, depressurization of the reactor coolant system (RCS), including the pressurizer occurs. A reactor trip signal occurs when the pressurizer low pressure trip setpoint is reached. Reactor trip and scram are credited for smaller breaks, but conservatively ignored in the large break LOCA analysis.

The following LOCA-related analyses have been evaluated by the licensee for impact due to the proposed change:

- Large and small-break LOCA,
- Reactor vessel and loop LOCA blowdown forces,
- Post-LOCA long-term core cooling subcriticality, and
- Post-LOCA long-term core cooling minimum flow and hot leg switchover to prevent further boron precipitation.

The licensee in its analysis does not credit the interlock in the small and large break analysis; therefore, the proposed change does not impact HNP UFSAR LOCA analysis. The long-term core cooling and post-LOCA subcriticality analyses are independent of the RTS actuation signals. Therefore, the proposed change does not impact HNP UFSAR long-term and post-LOCA analyses.

Therefore, the the NRC staff finds proposed change has no effect on the above LOCA-related accident scenarios and the conclusions in the HNP UFSAR remain valid.

3.1.2 Non-LOCA Accidents

For the Non-LOCA Transient Analyses, the licensee considered the following HNP UFSAR transients:

1. Feedwater system malfunctions that result in a decrease in feedwater temperature (15.1.1)

2. Feedwater system malfunctions that result in an increase in feedwater flow (15.1.2)
3. Excessive increase in secondary steam flow (15.1.3)
4. Inadvertent opening of a SG relief or safety valve (15.1.4)
5. Steam system piping failure (15.1.5)
6. Steam pressure regulator malfunction or failure that results in decreasing steam flow (15.2.1)
7. Loss of external electrical load (15.2.2)
8. Turbine trip (15.2.3)
9. Inadvertent closure of main steam isolation valves (15.2.4)
10. Loss of condenser vacuum and other events resulting in turbine trip (15.2.5)
11. Loss of nonemergency alternate current power to the station auxiliaries (15.2.6)
12. Loss of normal feedwater (15.2.7)
13. Feedwater system pipe break (15.2.8)
14. Partial loss of forced reactor coolant flow (15.3.1)
15. Complete loss of forced reactor coolant flow (15.3.2)
16. Reactor loss of forced pump shaft seizure (Locked Rotor) (15.3.3)
17. Reactor coolant pump shaft break (15.3.4)
18. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition (15.4.1)
19. Uncontrolled rod cluster control assembly bank withdrawal at power (15.4.2)
20. Rod cluster control assembly mis-operation (15.4.3)
21. Startup of an inactive reactor coolant pump at an incorrect temperature (15.4.4)
22. Chemical and volume control system malfunction that results in a decrease in boron concentration in the reactor coolant (15.4.6)
23. Inadvertent loading and operation of a fuel assembly in an improper position (15.4.7)
24. Spectrum of rod cluster control assembly ejection accident (15.4.8)
25. Inadvertent operation of the emergency core cooling system during power operation (15.5.1)
26. Chemical and volume control system malfunction that increases reactor coolant inventory (15.5.2)
27. Inadvertent opening of a pressurizer safety or PORV (15.6.1)
28. Break in instrument line or other line from reactor coolant pressure boundary that penetrate containment (15.6.2)
29. SG tube rupture (15.6.3)
30. Radioactive waste gas system leak or failure (15.7.1)
31. Liquid waste system leak or failure (15.7.2)
32. Postulated radioactive releases due to liquid tank failure (15.7.3)
33. Design basis fuel handling accidents (15.7.4)
34. Spent fuel cask drop accidents (15.7.5)
35. Anticipated transients without scram (15.8)

Based upon the NRC staff's review of the evaluation performed by the licensee in its LAR and the staff's previous evaluations of other nuclear power plants such as Indian Point Nuclear Generating Unit No. 3 (ML003780834), and the North Anna Power Station, Unit Nos. 1 and 2 (ML013460457), which are similar in design to the HNP Unit 1, the NRC staff concludes that the above referenced transients are not affected by the proposed change.

3.1.3 TMI Action Item II.K.3.10 Analysis

The NRC staff position for TMI Action Item II.K.3.10 in NUREG-0737 states that:

> The anticipatory trip modification proposed by some licensees to confine the range of use to high-power levels should not be made until it has been shown on a plant-by-plant basis that the probability of a small-break loss-of-coolant accident (LOCA) resulting from a stuck-open power operated relief valve (PORV) is substantially unaffected by the modification.

The licensee performed an analysis of the turbine trip without reactor trip transient from the P-8 setpoint to determine if the pressurizer PORVs are challenged. The turbine trip without a reactor trip transient was initialized from an indicated initial power level of 49 percent rated thermal power (RTP) corresponding to the P-8 permissive setpoint. The actual core power level of 51 percent RTP includes a 2 percent power uncertainty to allow for the Leading Edge Flow Meters to be out of service. The normal plant control systems are assumed to be operational. This best-estimate analysis addresses the NRC position in NUREG-0737, Item II.K.3.10.

3.1.4 Anticipated Transient Without Scram (ATWS) Mitigating System Actuation Circuitry (AMSAC) Consideration

In Section 3.1.4 of the LAR, the licensee included additional information related to ATWS Mitigating System Actuation Circuitry or AMSAC. AMSAC is not part of acceptance criteria for the licensee's analysis results and evaluation. The NRC staff did not review the information provided by the licensee related to HNP's AMSAC system, since the licensee in its application did not request approval of the HNP's AMSAC system.

3.1.5 RETRAN Computer Code Consideration

The licensee's analysis uses the RETRAN-3D code[1] to simulate the system thermal-hydraulic response for a turbine trip at reduced power without an immediate reactor trip. The analysis uses the HNP RETRAN-3D methodology presented in DPC-NE-3008-PA, "Thermal- Hydraulic Models for Transient Analysis" (ML16278A080), and DPC-NE-3009-PA, FSAR / UFSAR Chapter 15, "Transient Analysis Methodology," as approved for use by HNP in License Amendment number 164 and documented in the NRC's safety evaluation dated April 10, 2018 ((ML18060A401) (public), (ML18060A318) (nonpublic)). The HNP RETRAN-3D base model simulates the overall thermal-hydraulic and nuclear response of the Nuclear Steam Supply System as well as the various control and protection systems. The control systems that act to mitigate this transient are the pressurizer pressure control system, rod control system, and steam dump control system. The steam dump control is comprised of the loss of load controller which is active while the turbine is operating, and the turbine trip controller when the turbine is tripped. The turbine trip controller is used in this analysis, inherently crediting the condenser as being available. The SG level control system is also modeled to represent the runback in feedwater flow during the transient.

The following assumption are conservatively used by the licensee in the analysis:

---

[1] Electric Power Research Institute (EPRI) Topical Report NP-7450(A), "RETRAN-3D – A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems"

- Least negative Beginning-Of-Life (BOL) reactivity parameters are used. BOL reactivity parameters have lower differential rod worth and the least negative moderator temperature coefficient. Using BOL parameters in the analysis yields more conservative results that bound the full cycle of operation. Selection of BOL is consistent with the UFSAR, Section 15.2.3 analysis.

- A high SG tube plugging of 3 percent is assumed to minimize heat transfer in the SG tube bundle.

- Rod control is assumed to be operational and in the automatic mode of control for the duration of the transient. Since the turbine trip transient is a load decrease, the rods are automatically inserted to mitigate the transient.

- The pressurizer pressure control system is assumed to be operational and in the automatic mode of control.

- The pressurizer level program low and high setpoints are a function of no-load and full load T_{avg}, respectively. The initial pressurizer level corresponding to the P-8 setpoint is determined using this function. It is assumed that the RCS charging flow and letdown flow remain balanced for the duration of the transient.

<u>Methods and Modeling Changes</u>

The HNP RETRAN-3D model as described in DPC-NE-3008-PA (ML16278A080) is modified for the best-estimate turbine trip from P-8 setpoint analysis. Two models are added to provide additional detail required for this analysis; additional detail is added to the feedwater model, and a variable best-estimate inter-region heat transfer coefficient model is used in the pressurizer. Additional detail is provided for the steam dump control system and steam dump valves used to mitigate this transient. The modelling change was previously approved by the NRC in License Amendment 164 for HNP ((ML18060A401) (public), (ML18060A318) (nonpublic)) and is used and is approved by the NRC by plants with similar design, such as H.B. Robinson Steam Electric Plant, Unit No. 2 (ML18060A401).

<u>Analysis Acceptance Criteria</u>

The licensee’s acceptance criterion for the best-estimate transient initiating from the P-8 setpoint is that overfilling of the pressurizer will not occur. This ensures a Condition II event will not initiate a Condition III event and ensures the probability of a small-break LOCA resulting from a stuck-open PORV is substantially unaffected by the modification.

<u>Analysis Results</u>

A turbine trip without credit for a reactor trip is analyzed by the licensee from the current P-8 setpoint of 49 percent power with nominal initial conditions and with all control systems functioning per design (i.e., best-estimate conditions). This case models the current turbine trip controller settings and credits the condenser as being available. The analysis performed by the licensee shows that pressurizer PORVs are not challenged during a turbine trip without reactor trip transient initiating from the P-8 permissive setpoint.

The evaluation of a turbine trip due to a loss of condenser from the current P-8 setpoint of 49 percent RTP concluded that the pressurizer PORVs, steam line PORVs, and secondary system safety relief valves are not expected to be challenged. The consequences of this transient would be bounded by the full power UFSAR, Section 15.2.3 analysis results. UFSAR, Section 15.2.3 analyses noted that turbine trip from the full power does not result in overfilling the pressurizer. Thus, this Condition II event will not initiate a Condition III event. The NRC staff finds the analysis modeling result to support the requested TS changes acceptable since the model is previously approved by the NRC.

3.2 GDC 13 Evaluation

The NRC staff reviewed the LAR against GDC 13. The instrumentation utilized to monitor and initiate reactor trip from turbine trip are unchanged and currently operate at power levels greater than the P-7 interlock (approximately 10 percent of RTP). The proposed change will enable reactor trip from turbine trip at power levels greater than the P-8 interlock (approximately 49 percent RTP).

Based on NRC staff's review, the proposed changes do not impact the ability of the instrumentation to monitor and control 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. The proposed changes do not impact the ability to provide appropriate controls to maintain these variables and systems within prescribed operating ranges. Additionally, since the proposed operating range is a subset of the current operating range, the instrument uncertainty is unchanged and still bounded by the original analysis. The NRC staff also reviewed the USFAR and determined the proposed changes are consistent with the current design basis. Therefore, the staff concludes the proposed changes meet the requirements of 10 CFR Part 50, Appendix A, GDC 13.

3.3 Technical Specification Change Evaluation

Under 10 CFR 50.36(c), TSs must include items in the following five categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls. The proposed change would affect TS 3/4.3.1, "Reactor Trip System Instrumentation." Specifically, Functional Unit 17, "Turbine Trip (Above P-7)," of TS Table 3.3-1 will be revised to "Turbine Trip (Above P-8)." This proposed change will modify HNP TS to change the reactor trip on turbine trip from the P-7 (Low Power Reactor Trips Block) interlock to the P-8 (Power Range Neutron Flux) interlock. Instead of blocking a reactor trip on a turbine trip signal below 10 percent of full power, the P-8 interlock would block a reactor trip on turbine trip signal below 49 percent of full power. The NRC staff determined that the proposed TS change was acceptable because it was consistent with the technical analysis and evaluation discussed in sections 3.1 through 3.2 of this safety evaluation. The NRC staff finds the analysis modeling result to support the requested TS changes acceptable because the model is consistent with those previously reviewed and approved by the NRC. Thus, 10 CFR 50.36 will continue to be met.

Technical Evaluation Summary

The proposed change affects a defense-in-depth anticipatory reactor trip signal not credited in the HNP UFSAR, Chapter 15 safety analyses. The turbine trip without reactor trip transient analysis from the current P-8 setpoint of 49 percent RTP concluded that the pressurizer PORVs will not be challenged during a best-estimate simulation (i.e., all control systems performing as designed) with the current configuration of the steam dump control system, crediting the condenser. The NRC staff finds the analysis input assumptions, modeling and its result support the requested TS changes and is therefore acceptable.

Based on the above, the NRC staff finds that the proposed changes continue to meet the requirements of GDC 13 and GDC 20, as well as 10 CFR 50.36. The NRC staff concludes that the proposed TS change as summarized in Section 2.2 of this safety evaluation and as described in detail in the LAR is, therefore, acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, published in the *Federal Register* on November 30, 2021, (86 FR 67984), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the North Carolina State official on May 26, 2022, of the proposed issuance of the amendments. The State official had no comments.

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: F. Forsaty
C. Cheung
C. Ashley

Date: August 9, 2022

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT NO. 194 TO REVISE TECHNICAL SPECIFICATIONS RELATED TO REACTOR PROTECTION SYSTEM INSTRUMENTATION (EPID L-2021-LLA-0149) DATED AUGUST 9, 2022

DISTRIBUTION:
PUBLIC
PM File Copy
RidsNrrDorlLpl2-2
RidsNrrLARButler
RidsACRS_MailCTR
RidsNrrPMShearonHarris
RidsRgn2MailCenter
RidsNrrDssStsb
RidsNrrDssSnsb
RidsNrrDexEicb
FForsaty, NRR
CCheung, NRR
CAshley, NRR

ADAMS Accession No.: ML22126A008

OFFICE	DORL/LPL2-2/PM	DORL/LPL2-2/LA	DSS/SNSB/BC	DEX/EICB/BC
NAME	MMahoney	RButler	SKrepel	MWaters
DATE	05/27/2022	5/12/2022	03/23/2022	03/31/2022
OFFICE	DSS/STSB/BC	OCG – NLO	DORL/LPL2-2/BC	DORL/LPL2-2/PM
NAME	VCusumano	JEzell	DWrona (LHaeg for)	MMahoney
DATE	04/25/2022	07/28/2022	08/09/2022	08/09/2022

OFFICIAL RECORD COPY