<u>SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATIONAMENDMENT TO</u>

<u>RENEWED FACILITY OPERATING LICENSE NO. NPF-63</u>

<u>Shearon Harris Nuclear Power Plant, Unit 1</u>

DOCKET NOS. 50-400

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC) dated August March 6, 2021 (ADAMS Accession No. ML21218A197), Duke Energy (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). In addition, changes to TS Bases 3.3.1 are required to reflect enabling the reactor trip on turbine trip at the P-8 interlock versus the P-7 interlock. These changes will be made in accordance with the HNP Technical Specifications Bases Control Program.

2.0 REGULATORY EVALUATION

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, 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 U.S. Nuclear Regulatory Commission (NRC) staff's review of the TS changes is to assure the licensee's compliance with the TMI Action Item II.K.3.1 0 requirements and compliance with regulations applicable to transient and accident analyses.

10 CFR 50.36, “Technical specifications”:

The NRC's regulatory requirements related to the content of the TS are set forth in Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TS include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) limiting conditions for operation (LCOs), (3) surveillance requirements (SRs), (4) design features, and (5) administrative controls.

10 CFR 50 Appendix A, General Design Criteria 13 and 20

10 CFR Part 50 Appendix A, General Design Criterion (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."

10 CFR Part 50 Appendix A, 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.

These criteria will continue to be met with the proposed changes. The ability of the instrumentation to perform their required functions will not be impacted.

3.0 SYSTEM DESIGN

<u>Reactor Trip System</u>

The purpose of the Reactor Trip System (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, such that the plant is capable of returning to operation after corrective action is taken. 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 identified in Section 7.2.1.1 of the HNP Final Safety Analysis Report (FSAR), 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. No credit is taken in any of the FSAR Chapter 15 safety analyses for this trip.

<u>Reactor Trip System Interlocks</u>

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 loops are indicating low flow.

<u>Pressurizer Pressure Control System</u>

The pressurizer pressure control system, as discussed in HNP FSAR 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 operating valves (PORVs) maintain the pressure at the setpoint value and prevent reactor trip because of pressure variations caused by operational transients.

<u>Rod Control System</u>

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

Steam Dump Control System

The steam dump control system as discussed in HNP FSAR 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.

4.0 PROPOSED TS CHANGES

The proposed changes would affect TS 3.3.1, "Reactor Trip Instrumentation." Specifically, the proposed TS Table 3.3.1-1 would change the turbine trip closure of the turbine valve signal to be Applicable when power levels are above the P-8 (Power Range Neutron Flux) setting rather than the current P-7 (Low Power Reactor Trips Block) setting. The licensee proposed TS also would modify the HNP TSs Bases, "Reactor Trip System Instrumentation Setpoints" to address the P7 to P8 change setting.

5.0 TECHNICAL EVALUATION

The licensee identified the limiting cases for each event category discussed in the safety analysis sections of the HNP FSAR and evaluated the effect of TS changes on the loss of coolant accident (LOCA) and the transient analysis for each limiting case.

5.1 LOSS OF COOLANT ACCIDENTS

Event Definition: A loss of coolant accident (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 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 neglected in the large break LOCA analysis.

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

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

The interlock being modified is not credited in the small and large break analysis; therefore, the proposed change does not impact HNP FSAR 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 FSAR long-term and post LOCA analyses.

Therefore, the proposed change has no effect on the above accident scenarios and the conclusions of the HNP FSAR remain valid.

5.2 NON -LOCA ACCIDENTS

For the Non-LOCA Transient Analyses, the licensee considered the following HNP FSAR events:

1. Feedwater system malfunction that result in a decrease in feedwater temperature (15.1.1)
2. Feedwater system malfunction 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 steam generator 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 AC 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 misoperation (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 power operated relief valve (15.6.1)
28. Break in instrument line or other line from reactor coolant pressure boundary that penetrate containment (15.6.2)
29. Steam generator 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 (ADAMS Accession No. ML21218A197) and previous evaluations of the plants such as Indian Point Nuclear Generating Unit No. 3 (ADAMS Accession No. ML003780834) , and the North Anna Power Station, Unit No. 1 and No. 2 (ADAMS Accession No. ML013460457) which are similar in design to the HNP Unit 1, the NRC staff concluded that the above referenced transients are not affected by the proposed increase in power level required for a reactor trip following a turbine signal for HNP Unit 1.

5.3 TMI ACTION ITEM II.K.3.10 ANALYSIS

The 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."

An analysis of the turbine trip without reactor trip transient from the P-8 setpoint was performed by the licensee 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 (LEFM) 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 (ADAMS Accession No. ML051400209).

The analysis uses the RETRAN-3D code (Reference 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" (ADAMS Accession No. ML16278A082), and DPC-NE-3009-PA, FSAR / UFSAR [Updated FSAR] Chapter 15, "Transient Analysis Methodology" (ADAMS Accession No. ML16278A082), as approved for use by HNP per license amendment number 164 and documented in the Safety Evaluation Report per NRC letter dated April 10, 2018 (ADAMS Accession No. ML18060A318). The HNP RETRAN-3D base model simulates the overall thermal-hydraulic and nuclear response of the Nuclear Steam Supply System (NSSS) 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 steam generator (SG) level control system is also modeled to represent the runback in feedwater flow during the transient.

The following assumption are conservatively used in the analysis:

- Least negative Beginning-Of-Life (BOL) reactivity parameters are used. Beginning-Of-Life 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 FSAR

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 Tavg, respectively. The initial pressurizer level corresponding to the P-8 setpoint is determined using this function. It is assumed that the reactor coolant system (RCS) charging flow and letdown flow remain balanced for the duration of the transient

Methods and Modeling Changes

The HNP RETRAN-3D model as described in DPC-NE-3008-PA (ADAMS Accession No, ML16278A082) 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 is previously approved by the NRC (ADAMS Accession No, ML16278A082) 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 (ADAMS Accession No, ML16278A082), and the Catawba Unit 2 (ADAMS Accession No ML16102A169.)

Acceptance Criteria

The 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

Analysis Results

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 FSAR Section 15.2.3 analysis results consistent with the Safety Evaluation (SE) Section 3.5.8 (ADAMS Accession No. ML18060A318.) It should be noted the full power FSAR Section 15.2.3 analyses do not overfill the pressurizer. Thus, this Condition II event will not initiate a Condition III event. The NRC staff finds the proposed analysis modelling changes as acceptable since it is previously approved by the NRC.

6.0 CONCLUSIONS

The proposed change affects a defense-in-depth anticipatory reactor trip signal not credited in the HNP FSAR Chapter 15 safety analyses. The results of the evaluation performed by the licensee demonstrate the proposed change does not significantly affect the defense in depth protection resulted by this anticipatory reactor trip. 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. Therefore, the proposed TSs Changes as summarized in Section 4.0 of this safety evaluation and as described in detail in the license amendment request (ADAMS Accession No. ML21218A197) is acceptable.

7.0 REFERENCES

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,”