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10 CFR 50.90

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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400
Renewed License No. NPF-63

Subject: License Amendment Request to Revise Technical Specifications Related to Reactor Protection System Instrumentation

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Technical Specifications (TS). 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). The proposed change would decrease the potential of experiencing unnecessary transients on the reactor.

The proposed license amendment has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been concluded that the proposed change does not involve a significant hazards consideration.

Enclosure 1 provides Duke Energy's evaluation of the proposed change. Enclosure 2 provides the existing TS pages marked to show the proposed change. Enclosure 3 provides existing TS Bases pages marked to show the proposed change for information only, as these pages will be implemented in accordance with the TS Bases Control Program upon implementation of the amendment.

Approval of the proposed license amendment is requested within twelve months of acceptance. Duke Energy will implement the amendment prior to startup from the next refueling outage subsequent to the NRC approval date.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated North Carolina State Official.

This letter contains no regulatory commitments.

Please refer any questions regarding this submittal to Art Zaremba, Manager – Nuclear Fleet Licensing, at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 6, 2021.

Sincerely,

Kim E. Maza
Site Vice President
Harris Nuclear Plant

Enclosures:

1. Evaluation of the Proposed Changes
2. Proposed Technical Specification Changes (Mark-up)
3. Technical Specification Bases Changes (Mark-up)

cc: J. Zeiler, NRC Sr. Resident Inspector, HNP
W. L. Cox, III, Section Chief, N.C. DHSR
M. Mahoney, NRC Project Manager, HNP
L. Dudes, NRC Regional Administrator, Region II

ENCLOSURE 1

EVALUATION OF THE PROPOSED CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

44 PAGES PLUS THE COVER

Evaluation of the Proposed Changes
License Amendment Request to Revise Technical Specifications Related to
Reactor Protection System Instrumentation

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Technical Specifications (TS). 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). The proposed change would decrease the potential of experiencing unnecessary transients on the reactor.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

Reactor Trip System

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) by, 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 (or reached). The safe operating region is defined by several considerations such as mechanical/hydraulic limitations on equipment, and heat transfer phenomena. Therefore, the RTS keeps surveillance on process variables which are directly related to equipment mechanical limitations, such as pressure and pressurizer water level (to prevent water discharge through safety valves, and uncovering heaters); and also on variables which directly affect the heat transfer capability of the reactor (e.g., flow and reactor coolant temperatures). Whenever a direct process or calculated variable reaches a setpoint, the reactor will be shut down in order to protect against either gross damage to fuel cladding or loss of system integrity which could lead to release of radioactive fission products into the containment.

One anticipatory trip of the RTS is the reactor trip on turbine trip that 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 reactor trip on turbine trip provides additional protection and conservatism beyond that required for the health and safety of the public. No credit is taken in any of the FSAR Chapter 15 safety analyses for this trip.

Reactor Trip System 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, reactor coolant pump undervoltage, reactor coolant pump underfrequency, pressurizer low pressure, pressurizer high water level or turbine trip signal. 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.

Pressurizer Pressure Control System

The pressurizer pressure control system, as discussed in HNP FSAR Section 7.7.1, maintains or restores the pressurizer pressure 50 psi above to 50 psi below the design pressure following normal operational transients that induce pressure changes by control (manual or automatic) of heaters and spray in the pressurizer. It also provides steam relief by controlling the power operated relief valves (PORVs).

The pressurizer pressure control system maintains a pressure at the set value by four means:

a. Spray
b. PORVs
c. Proportional Heaters
d. Back-up Heaters

Together, the heaters, spray, and PORVs maintain the pressure at the setpoint value and prevent reactor trip as a result of pressure variations caused by operational transients.

Rod Control System

The rod control system, as discussed in HNP FSAR Section 7.7.1, provides for reactor power modulation by manual or automatic control of control rod banks in a preselected sequence as well as the manual operation of individual banks. The automatic rod control system is designed to maintain a programmed average temperature in the reactor coolant by regulating the reactivity within the core. The system is capable of restoring the average temperature to within ± 1.5°F of the programmed temperature following design load changes.

The rod control system is designed to automatically control the reactor in the power range between 15 and 100 percent of rated power for the following design transients:

a. ±10 percent step changes in load,
b. 5 percent/minute ramp loading and unloading.

Steam Dump Control System

As discussed in HNP FSAR Section 7.7.1, HNP is designed to accept a 50 percent load rejection from full power without incurring reactor trip. Steam is dumped to the condenser and/or the atmosphere as necessary to accommodate excess power generation in the reactor during turbine load reduction transients. The steam dump control system 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. Additionally, it maintains the plant at no-load conditions and permits a manually controlled cooldown of the plant.

The steam dump system comprises 14 valves which can bypass steam to the condenser and to the atmosphere. Included in the 14 valves are the eight atmospheric relief valves. Depending on the full power vessel average temperature, the total steam dump capacity is approximately 66 to 86 percent of the rated steam flow.

2.2 Current Technical Specifications Requirements

HNP TS 3.3.1, “Reactor Trip System Instrumentation,” Table 3.3.1-1, “Reactor Trip System Instrumentation,” currently reflects Functional Unit 17 as “Turbine Trip (Above P-7).” The P-7 interlock blocks a reactor trip following a turbine trip when the plant is below approximately 10 percent of full power.

2.3 Reason for the Proposed Change

Changing the interlock for a reactor trip on a turbine trip to a permissive with a higher setpoint decreases the likelihood of experiencing unnecessary transients on the reactor.

2.4 Description of the Proposed Change

Functional Unit 17, “Turbine Trip (Above P-7),” of TS Table 3.3.1-1 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 interlock to the P-8 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.

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.

3.0 TECHNICAL EVALUATION

Duke Energy has performed a best-estimate plant specific analysis for HNP (i.e., with nominal initial conditions and with all control systems functioning per design) to evaluate the RTS modification to change the reactor trip on turbine trip interlock from permissive P-7 to permissive P-8. This analysis addresses the NRC position regarding implementation of any plant features which could increase the probability of a stuck-open pressurizer PORV.

The original design criterion was that turbine load rejections up to the maximum load rejection capability of the plant should not actuate a reactor trip if all control systems function properly as

designed. However, after the Three Mile Island (TMI) incident, the Nuclear Regulatory Commission (NRC) expressed concern regarding implementation of any plant features which could increase the probability of a stuck-open pressurizer PORV. The NRC position is addressed in NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II-K.3.10 (Reference 5).

To address the NRC position in NUREG-0737, Item II-K.3.10, a best-estimate plant specific analysis was performed to demonstrate that overfilling of the pressurizer will not occur. This ensures a Condition II event will not initiate a Condition III event (infrequent events) and ensures the probability of a small break LOCA accident resulting from a stuck-open PORV is substantially unaffected by the modification.

3.1 Analysis Description

An analysis of the turbine trip without reactor trip transient from the P-8 setpoint was performed 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 (Reference 5).

The analysis uses the RETRAN-3D code version (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" (Reference 2), and DPC-NE-3009-PA, "FSAR / UFSAR [Updated FSAR] Chapter 15 Transient Analysis Methodology" (Reference 3), 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 (Reference 4). 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.

3.1.1 Key Assumptions and Inputs

Consistent with the objective of this analysis, the transient conditions simulated were primarily selected to determine the maximum pressurizer pressure following the initiation of the transient. Therefore, the following key assumptions and inputs were utilized to ensure a conservatively high prediction of pressurizer pressure.

The turbine trip without a reactor trip transient is initialized from an indicated initial power level of 49 percent RTP corresponding to the P-8 permissive setpoint. The actual core power level of 51 percent RTP, or 1503.48 MW_{th}, includes a 2 percent power uncertainty.

The turbine trip transient is modeled as a step load decrease from 49 percent to 0 percent load within 0.1 seconds, after a 30 second null transient to establish steady state conditions.

The nominal initial conditions assumed for this analysis are based on a full power loop average temperature (T_{avg}) of 588.8°F, a full power feedwater temperature of 440.0°F, and a nominal steam pressure of 974 psia. A high SG tube plugging of 3 percent is assumed to minimize heat transfer in the SG tube bundle.

Initial primary and secondary side conditions at the P-8 setpoint are consistent with 49 percent RTP; these conditions, such as T_{avg}, pressurizer pressure, pressurizer level, and SG water level, do not account for any uncertainties (i.e., best-estimate 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.

The initial fuel temperature assumed at P-8 is determined using a gap conductance consistent with a nominal 100 percent RTP initial fuel temperature at BOL conditions.

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 HNP RETRAN-3D model includes compensated pressurizer pressure, which is used to control the pressurizer heaters, spray, and a PORV. The other pressurizer PORVs are modeled using un-compensated pressurizer pressure as an input. The initial pressurizer pressure is assumed to be at the reference pressure, or 100 psi below the compensated PORV lift setpoint.

The steam dump control system is comprised of the loss of load controller and the turbine trip controller. The loss of load controller is active while the turbine is operating, and the turbine trip controller is used after turbine trip. The turbine trip controller modulates the condenser dump valves based on the difference between T_{avg} and the no-load temperature (T_{noload}). The current turbine trip controller settings open Bank 1 linearly between 0.0°F and 15.2°F, and Bank 2 linearly between 15.2°F and 30.3°F.

The SG level control system is assumed to be operational and in the automatic mode of control. This system modulates the main feedwater control valves based on the steam/feed flow mismatch and SG level error. The 57 percent narrow range target SG water level is independent of power level.

For the purposes of this analysis, it is assumed that all the pressurizer heaters (backup and proportional), are functioning properly, providing a total capacity of 1401 kW. It should be noted that the peak pressurizer pressure occurs at the initial stages of the transient; the heaters are not actuated during this time and the installed capacity does not have any impact on the peak pressurizer pressure.

Pressurizer spray is controlled using the compensated pressurizer pressure signal. Spray flow is controlled linearly from 0 percent at +25 psi to 100 percent flow at +75 psi. The maximum assumed spray flow is 700 gpm.

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 reactor coolant system (RCS) charging flow and letdown flow remain balanced for the duration of the transient.

3.1.2 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.

3.1.3 Results

A turbine trip without credit for a reactor trip is analyzed 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 pressurizer PORVs are not challenged during a turbine trip without reactor trip transient initiating from the P-8 permissive setpoint.

Sequence of events

- Steady state conditions are established for the initial 30 seconds.
- Turbine trip from an indicated 49 percent RTP at 30 seconds, the turbine stop valves close in 0.1 seconds.
- The initial difference between T_{avg} and T_{noload} is about 15.6°F
- Condenser Dump Banks 1 & 2 open
- Control rods automatically begin to insert
- Peak pressurizer pressure (2301.9 psig) occurs at 42.9 seconds.

The event begins with a turbine trip from the P-8 setpoint power level after a 30 second null transient to establish steady state conditions. Figure 1 shows the initial core power is 51 percent, with indicated core power matching the turbine load. A turbine trip occurs at 30 seconds, removing the available heat sink. Pressurizer pressure (Figure 2) increases immediately following the turbine trip reaching a maximum of 2301.9 psig shortly after the steam dump valves open. The pressure increase is sufficient to require the initiation of pressurizer spray flow, but does not present a challenge to the pressurizer PORVs. Pressurizer level (Figure 3) increases from 42 percent to 48 percent before the RCS begins to cool. The T_{avg} (Figure 4) increases initially until the steam dump controller action, and continued control rod insertion (Figure 8) reduces core power to match the steam dump capacity. Steam pressure (Figure 5) increases due to the turbine trip and remains above the no-load pressure for the duration of the simulation. The steam flow with the steam dump valves open is roughly half of the initial steam flow (Figure 6) resulting in an RCS temperature increase. Feedwater flow (Figure 7) decreases as the main feedwater control valves close. Actual liquid mass in the SG increases as feedwater flow exceeds steam flow until approximately 240 seconds. The longer-term response will restore the initial SG water level.

A turbine trip from the P-8 setpoint power level can occur due to a loss of condenser. For a loss of condenser transient, steam dump control transitions to the turbine trip controller, blocking the atmospheric dump valves from opening. The loss of condenser signal will block the condenser dump valves from opening. For this transient, the heat sink is provided by the steam line PORVs and secondary system safety relief valves. Due to the higher lift setpoints associated with these valves, the RCS temperature increase is greater than with the steam dump valves available. The temperature increase causes the pressurizer PORVs to lift and release steam. The lowest lifting secondary system safety relief valves also lift. Due to the automatic action of the rod control system, the pressurizer does not overfill. Thus, it is concluded this Condition II event will not initiate a Condition III event.

3.1.4 AMSAC Considerations

The following Anticipated Transient without Reactor Scram (ATWS) Mitigation System Actuation Circuitry (AMSAC) considerations are included as additional information, but since ATWS is not a Condition II event it is not considered relative to the acceptance criteria for this evaluation. AMSAC provides an independent and diverse backup to the RTS. AMSAC is armed when both Turbine First Stage pressure channel inputs indicate a reactor power above 35 percent. When the level in 2/3 SGs decreases below 20 percent for 25 seconds, the circuitry is activated (if armed). AMSAC remains armed for 360 seconds after turbine load drops below 35 percent.

In the event of an ATWS, AMSAC performs two functions: (1) trips the turbine, and (2) initiates auxiliary feedwater (AFW) flow. AMSAC is normally armed for actuation at 35 percent turbine power as sensed by turbine impulse pressure. However, changing the reactor trip on turbine trip to the P-8 setpoint of 49 percent reactor power will allow the turbine to be shut down at power levels less than 49 percent without resulting in a reactor trip signal. With the turbine off-line, turbine impulse pressure approaches condenser pressure disarming AMSAC after 6 minutes, and as a result AMSAC may not be armed for automatic actuation when SG level decreases to the setpoint. The proposed revision to the P-8 setpoint has the potential to create a condition in which AMSAC will not automatically initiate AFW flow in the event of an ATWS occurring below 49 percent reactor power with the turbine off-line.

Two scenarios are evaluated to examine the transient response and address the operation of AMSAC without RTS actuation from the P-8 setpoint. Best-estimate initial conditions corresponding to the P-8 power level are assumed. The normal plant control systems including rod control and steam dump are assumed to be operational. The first scenario is the traditional loss of main feedwater ATWS. The second scenario postulates a coincident turbine trip with the loss of main feedwater. Each are described further below.

The loss of main feedwater ATWS scenario from the P-8 setpoint (49 percent reactor power) begins with the trip of the operating main feedwater pump. The turbine is assumed to remain operating at the initial power level during the initial phase of the event. The initial RCS response to the loss of feedwater flow is an increase in temperature. Steam dump valves begin to open as the RCS T_{avg} increases. Rod control similarly responds to the T_{avg} increase by inserting control rods. The reactor trip on the SSPS low-low SG level is not credited. When SG level reaches the AMSAC low-low SG level setpoint, AMSAC is credited to trip the turbine and initiate AFW flow. All three AFW pumps are assumed to start and provide flow to the SGs. Following turbine trip and AFW actuation, steam line pressure decreases to the point where a main steam isolation signal occurs. The low steam line pressure signal is credited to close the main steam

isolation valves (MSIV) but is not credited to initiate either a safety injection signal or a reactor trip. The steam dumps cease to be effective following MSIV closure. The longer-term transient response is characterized by AFW refilling the SGs, and steam line PORVs controlling SG pressure. The pressurizer PORVs do lift during this scenario. If Emergency Core Cooling System (ECCS) initiates following the low steam line pressure safety injection signal, the pressurizer would be expected to fill and liquid relief result. This result is consistent with the limiting ATWS event. The limiting ATWS event occurs at full power conditions and is mitigated by AMSAC.

The second loss of main feedwater ATWS scenario from the P-8 setpoint (49 percent reactor power) considered assumes the turbine trips coincident with main feedwater pump trip. The initial response is similar to the first scenario and includes an RCS T_{avg} increase with steam dump valve and rod control responding to this temperature increase. Due to the turbine already being tripped, the difference between T_{avg} and the turbine reference temperature (T_{ref}) is greater, resulting in a reduced RCS T_{avg} relative to the first scenario. Rod control also uses the lower T_{ref} contributing to a lower core power level. When SG level reaches the low-low SG level setpoint, AMSAC is credited to initiate AFW flow since the elapsed time after turbine trip is less than the 6-minute reset time. Steam pressure does not decrease below initial, the SGs do not dry out, and the minimum liquid mass in each SG is greater than the first scenario evaluated. The longer-term transient response is characterized by AFW refilling the SGs, and condenser steam dump valves controlling RCS temperature and SG pressure. The pressurizer PORVs do not lift during this scenario.

In both scenarios, AMSAC performs its intended functions of providing a turbine trip signal and initiating AFW flow. It is conservative to assume rod control is in automatic as this extends the time to reaching SG low-low level. The low-low level setpoints assumed in the evaluation are adjusted to ensure conservative results. The AFW is initiated by AMSAC prior to the AMSAC reset time.

3.1.5 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 demonstrate the proposed change does not significantly affect the defense in depth protection afforded 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.

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 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 in Reference 4. It should be noted the full power FSAR Section 15.2.3 analyses do not overfill the pressurizer. Thus, it is concluded this Condition II event will not initiate a Condition III event.

3.2 Methods and Modeling Changes

3.2.1 Method Changes

The Duke Energy response to DPC-NE-3009-PA Request for Additional Information (RAI) 1b indicates that Duke Energy will utilize the single node SG model in the Chapter 15 Loss of Normal Feedwater transient analysis for the post-trip phase of the event if significant tube bundle uncovery is predicted to occur. Use of the single node SG model is intended to ensure a conservative prediction of primary to secondary heat transfer in the transient analysis. The RAI 1b response addresses an issue identified for the Catawba Nuclear Station (Catawba) Unit 2 model in DPC-NE-3002-A (Reference 6). The identified concern was an over-prediction of primary-to-secondary heat transfer when the minimum post-trip SG water inventory decreased below 10 percent of the full-power inventory in the Loss of Normal Feedwater Flow event.

There are significant nodalization differences between the Catawba Unit 2 SG model developed in DPC-NE-3000-PA (Reference 7) and the model used in the DPC-NE-3002-A methodology. The Catawba Unit 2 SG model uses 3 nodes to model the pre-heat tube bundle and the Harris RETRAN model uses 12 nodes to model the feedring tube bundle. The DPC-NE-3002 method uses RETRAN-02 with enthalpy transport specified in the tube bundle, whereas the DPC-NE-3009 method uses RETRAN-3D with enthalpy transport disabled in the tube bundle. As noted in DPC-NE-3009 SE section 3.2.1.1.3, the improved accuracy of the tube bundle nodalization is needed to justify deactivation of the enthalpy transport model on the SG primary and secondary sides. The DPC-NE-3009 method uses the RETRAN-3D algebraic slip model with the Chexal-Lellouche drift flux correlation specified in the tube bundle. The strengths of the algebraic slip model are a result of close attention to the physical characteristics that must exist over the complete range of the simulation and in the vast amount of data that has been used in its validation. As noted in DPC-NE-3009-PA SE section 3.2.1.2.4, the algebraic slip model provides a good prediction of the void fraction for predictions of experimental test data. The algebraic slip model improves the prediction of the void distribution in the tube bundle.

The proposed usage of the simplified model for the loss of normal feedwater flow event described in DPC-NE-3009-PA RAI 1b preceded completion of the HNP and H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP) loss of normal feedwater analyses. The transient response with the single and multiple node SG models is compared for the loss of normal feedwater analysis. The comparison demonstrates that the multiple node and single node models do not produce significantly different results and that the overall transient results line up very well.

The original concern identified with the DPC-NE-3002-A Catawba Unit 2 SG model related to the over-prediction of heat transfer in the post-trip transient response. To illustrate the impact on the transient response using the RETRAN-3D multiple node and single node models, figures providing a comparison of RCS T_{avg} and SG liquid inventory are included. Note that the analysis with the single node model uses reactor power versus time from the multiple node analysis for the pre-trip response as well as the time of reactor trip and AFW actuation due to low SG level. Figure 9 provides the RCS T_{avg} comparison. This figure illustrates the thermal balance in the RCS between core power and SG heat transfer. Figure 10 provides the SG liquid inventory comparison. These figures demonstrate good agreement between the models and indicate the original concern with heat transfer is not evident. Figures 11 and 12 provide the same

comparison for the RNP analysis and further support the conclusion that primary-to-secondary heat transfer is not overpredicted even with significant tube bundle uncovery.

The analyses performed to support changing the reactor trip on turbine trip interlock from permissive P-7 to permissive P-8 include best-estimate loss of main feedwater ATWS scenarios. These scenarios may include a significant decrease in secondary inventory. For these scenarios the prediction of SG liquid level is important to the sequence of events, which requires use of the multiple node SG model. The single node SG model does not contain a sufficient level of detail to provide an accurate SG water level indication to facilitate an RTS, engineered safety feature (ESF), or AMSAC actuation. Therefore, the multiple node model is better suited for simulating the loss of main feedwater ATWS scenarios.

Conclusion

In DPC-NE-3009-PA RAI 1b, Duke Energy proposed an update to the Loss of Normal Feedwater Flow methodology described in Section 5.2.4 of DPC-NE-3009-PA. This update is discussed in SE Section 3.2.2.2. Specifically, if the minimum post-trip SG water inventory indicates significant tube bundle uncovery, then the post-trip phase of the event will be analyzed using the simplified SG model.

It is proposed that based on the results of the comparison of the single and multiple node SG models for the HNP and RNP loss of normal feedwater flow analyses, the change proposed in RAI 1b be modified to allow the use of either the multiple node or single node SG model for the DPC-NE-3009-PA Section 5.2.4 methods for HNP.

3.2.2 Modeling Changes

The HNP RETRAN-3D model as described in DPC-NE-3008-PA (Reference 2) 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 (IRHTC) 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. Each of these additions is described below.

The feedwater model described in Reference 2 is expanded to add a volume to connect the three feedwater lines illustrated in Figure 4.1-2 of Reference 2 to represent the physical arrangement of the feedwater system. This enables feedwater modeling to be consistent with normal plant operations at the P-8 setpoint.

The pressurizer IRHTC is modeled consistent with the description provided in RAI 2 of Reference 3 for best-estimate modeling of pressurizer spray. A minimum value is assumed when pressurizer spray is off. A nominal value is assumed with full pressurizer spray flow. The value is linearly interpolated between these values to minimize the IRHTC.

Steam dump control is comprised of two controllers; the loss of load controller that is active when the turbine is operating and the turbine trip controller following turbine trip. The loss of load controller modulates the condenser dump and atmospheric dump banks based on the difference between T_{avg} and T_{ref}. The turbine trip controller modulates the condenser dump

banks based on the difference between T_{avg} and the T_{noload} of 557°F. Steam dump control is credited to provide a heat sink and mitigate the event.

The steam dump valves are depicted in Figure 4.1-2 of Reference 2. Section 3.2.1.1.7 of the Reference 4 discusses the main steam line and the steam dump valves. The valve flow characteristics for the condenser steam dump valves and atmospheric dump valves are defined consistent with the RAI 16 response for Reference 2. The condenser steam dump valves are used by the turbine trip controller to mitigate the turbine trip event. Section 5.2.2 of Reference 3 discusses how the steam dumps are treated for the turbine trip analyses; evaluated for core cooling and disabled for primary and secondary peak pressure.

3.3 Review of FSAR Transients

A review of FSAR Chapter 15 safety analyses has been performed in order to confirm that the safety analyses results are not adversely affected by this proposed change of moving the reactor trip on turbine trip function from P-7 (approximately 10 percent RTP) to P-8 (approximately 49 percent RTP). A description is provided for each event, with the reactor trip functions identified in FSAR Table 15.0.8-1 for that event. The following provides an assessment of the proposed change with respect to the safety analyses.

The HNP FSAR Chapter 15 safety analyses are performed using the approved methodologies presented in DPC-NE-3008-PA (Reference 2) and DPC-NE-3009-PA (Reference 3), as approved for use by HNP in Reference 4. Guidance for the NRC staff's review of models and methodologies and the associated design criteria are provided in NUREG-0800, the Standard Review Plan (Reference 8). The HNP FSAR Chapter 15 events are analyzed at the limiting power levels to satisfy the NUREG-0800 criteria. The proposed change does not affect a reactor trip signal credited in the FSAR Chapter 15 safety analyses. The proposed change affects a defense-in-depth anticipatory trip. Thus, the HNP FSAR Chapter 15 safety analyses remain bounding and conservative.

15.1.1 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN A DECREASE IN FEEDWATER TEMPERATURE

Event Definition: Reductions in feedwater temperature will cause an increase in core power by decreasing reactor coolant temperature. Such transients are attenuated by the thermal capacity of the secondary plant and of the RCS. The overpower/overtemperature protection (overtemperature and overpower ΔT trips) prevents any power increase which could lead to a departure from nucleate boiling ratio (DNBR) less than the safety analysis limit.

The decrease in feedwater temperature transient is less severe than the increase in secondary steam flow event (Section 15.1.3). Consequently, no analysis is presented in the FSAR.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.1.2 FEEDWATER SYSTEM MALFUNCTIONS THAT RESULT IN AN INCREASE IN FEEDWATER FLOW

Event Definition: Addition of excessive feedwater will cause an increase in core power by decreasing reactor coolant temperature. An example of excessive feedwater flow would be a full opening of a feedwater control valve due to a feedwater control system malfunction or an operator error. At power, this excess flow causes a greater load demand on the RCS due to increased subcooling in the SG. With the plant at no-load conditions, the addition of cold feedwater may cause a decrease in RCS temperature and thus an effective reactivity insertion due to the effects of the negative moderator coefficient of reactivity.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Power range neutron flux (high setting),
2. Manual trip.

The ESF actuation functions identified in FSAR Table 15.0.8-1 are:

1. High-high steam generator level-produced feedwater isolation and turbine trip

This event is analyzed at both hot full power (HFP) conditions and hot zero power (HZP) conditions.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.1.3 EXCESSIVE INCREASE IN SECONDARY STEAM FLOW

Event Definition: An excessive increase in secondary system steam flow (excessive load increase incident) is defined as a rapid increase in steam flow that causes a power mismatch between the reactor core power and the SG load demand. The Reactor Control System is designed to accommodate a 10 percent step load increase or a 5 percent per minute ramp load increase in the range of 15 to 100 percent of full power. Any loading rate in excess of these values may cause a reactor trip actuated by the RTS. Steam flow increases greater than 10 percent are analyzed in Sections 15.1.4 and 15.1.5.

This event is predominantly a cooldown event and is evaluated at full power conditions. At full power, the margin to limits is the smallest and, therefore, bounds operation at lower power levels.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Power range neutron flux (high setting),
2. Overtemperature ΔT,
3. Overpower ΔT, or
4. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

Event Definition: The most severe core conditions resulting from an accidental depressurization of the Main Steam System are associated with an inadvertent opening of a single steam dump, power operated relief or safety valve. The analyses performed, assuming a rupture of a main steam line, are given in Section 15.1.5.

For this event initiating from Mode 1, the maximum steam flow through a single steam dump, power operated relief, or safety valve drives a thermal load increase less than that considered in Section 15.1.3. Ultimate reactor power level and the potential challenge to the departure from nucleate boiling (DNB) specified acceptable fuel design limit is greater for Section 15.1.3. The event is therefore bounded by Section 15.1.3 before trip and by Section 15.1.5 after trip since the Condition II criteria are met by the more challenging Condition IV event.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Low pressurizer pressure,
2. Power range neutron flux (high setting),
3. Overpower ΔT,
4. Safety Injection System (SIS), or
5. Manual.

The ESF actuation functions identified in FSAR Table 15.0.8-1 are:

1. Low pressurizer pressure,
2. Low compensated steam line pressure,
3. Hi-1 containment pressure, and
4. Manual.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.1.5 STEAM SYSTEM PIPING FAILURE

Event Definition: The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of reactor coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power.

Four base scenarios are considered. The reactor is assumed to be initially operating at either HFP or HZP conditions. From both of these initial conditions, the transient is assumed to occur either with or without offsite power.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. SIS,
2. Low pressurizer pressure,
3. Overpower ΔT,
4. Power range neutron flux (high setting), or
5. Manual trip.

The ESF actuation functions identified in FSAR Table 15.0.8-1 are:

1. Low pressurizer pressure,
2. low compensated steam line pressure,
3. Hi-1 and Hi-3 containment pressure, and
4. Manual.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on these accident scenarios and the conclusions of the FSAR remain valid.

15.2.1 STEAM PRESSURE REGULATOR MALFUNCTION OR FAILURE THAT RESULTS IN DECREASING STEAM FLOW

There are no steam pressure regulators in HNP whose failure or malfunction could cause a steam flow transient.

15.2.2 LOSS OF EXTERNAL ELECTRICAL LOAD

Event Definition: A major load loss on the plant can result from loss of external electrical load due to some electrical system disturbance. Offsite alternating current (AC) power remains available to operate plant components such as the reactor coolant pumps; as a result, the onsite emergency diesel generators are not required to function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. This will cause a sudden reduction in steam flow, resulting in an increase in pressure and temperature in the SG shell. As a result, the heat transfer rate in the SG is reduced, causing the reactor coolant temperature to rise, which in turn causes reactor coolant expansion, pressurizer insurge, and RCS pressure rise.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. High pressurizer pressure,
2. Overtemperature ΔT,
3. Steam generator low-low level,
4. High pressurizer water level, or

5. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. This scenario is bounded by the results of the analysis for Section 15.2.3, which was further evaluated as discussed below for an event initiating from the P-8 permissive and does not result in a reactor trip. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.2.3 TURBINE TRIP

Event Definition: For a turbine trip event, the reactor would be tripped directly (unless below an interlock setpoint) from a signal derived from the turbine stop emergency trip fluid pressure and turbine stop valves. This interlock is currently set at the P-7 setpoint (approximately 10 percent power) and is the subject of this request to increase it to the P-8 setpoint (approximately 49 percent power).

This event is analyzed with the limiting assumption of no credit for a direct reactor trip on turbine trip. Three cases are analyzed for this event: one challenging the primary overpressurization criterion, one challenging the secondary overpressurization criterion, and one challenging the fuel design limits. In all cases, the input parameters are biased (BOL kinetics) to maximize the increase in reactor power during the transient.

Pressurizer PORVs are not credited in the primary overpressurization case but are assumed to be available in the secondary overpressurization and fuel design limits cases. Pressurizer level remains on-scale for each of the cases analyzed.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. High pressurizer pressure
2. Overtemperature ΔT,
3. Steam generator low-low level,
4. High pressurizer water level, or
5. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. This scenario is evaluated for an event initiating from the P-8 permissive and does not result in a reactor trip. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.2.4 INADVERTENT CLOSURE OF MAIN STEAM ISOLATION VALVES

Event Definition: Inadvertent closure of the MSIVs would result in a complete loss of steam flow similar to but less severe than the turbine trip event analyzed in Section 15.2.3. The main steam line isolation valves close more slowly than the turbine stop valves resulting in a less severe transient. Therefore, this event is bounded by the results of the analysis for Section 15.2.3.

15.2.5 LOSS OF CONDENSER VACUUM AND OTHER EVENTS RESULTING IN TURBINE TRIP

Event Definition: Loss of condenser vacuum is one of the events that can cause a turbine trip. Turbine trip initiating events are described in Section 15.2.3. A loss of condenser vacuum would preclude the use of steam dump to the condenser and would block operation of the atmospheric steam dump valves. However, since steam dump is assumed not to be available in the turbine trip analysis, no additional adverse effects would result if the turbine trip were caused by loss of condenser vacuum. Therefore, the analysis results and conclusions contained in Section 15.2.3 apply to loss of condenser vacuum.

An evaluation of a loss of condenser event from the P-8 setpoint is discussed in Section 3.1, above. This event is mitigated using the steam line PORVs and secondary system safety relief valves as the heat sink. The evaluation concludes that due to the margin between the initial pressurizer pressure and the lift setpoints for the pressurizer PORVs, and the higher temperature of the available heat sink, the pressurizer PORVs are expected to lift. Due to the automatic action of the rod control system, the pressurizer will not overfill. The PORV lift and overfill conclusions are consistent with the full power Section 15.2.3 turbine trip analysis.

15.2.6 LOSS OF NONEMERGENCY AC POWER TO THE STATION AUXILIARIES

Event Definition: A complete loss of non-emergency power (i.e., offsite power) may result in the loss of all power to the plant auxiliaries: i.e., the reactor coolant pumps, condensate pumps, etc. The loss of power may be caused by a complete loss of the offsite grid accompanied by a turbine generator trip at the station, or by a loss of onsite non-emergency AC distribution system.

The reactor will trip: 1) due to turbine trip; 2) upon reaching one of the trip setpoints in the primary and secondary systems, as a result of the flow coastdown and decrease in secondary heat removal; or 3) due to loss of power to the control rod drive mechanisms, as a result of the loss of power to the plant. The analysis presented in Section 15.2.6.2 credits a reactor trip on low-low SG level and does not credit either a turbine trip or an immediate release of the control rod drive mechanisms caused by the loss of power.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Steam generator low-low level,
2. Overtemperature ΔT, or
3. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

Event Definition: A Loss of Normal Feedwater Flow transient is initiated by either a main feedwater pump failure or a malfunction in the feedwater control valves. The loss of main

feedwater flow decreases the amount of subcooling in the secondary-side downcomer which diminishes the primary-to-secondary system heat transfer and leads to an increase in the primary system coolant temperature. The increase in primary coolant temperature results in overpressurization of the RCS.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Steam generator low-low level,
2. Overtemperature ΔT, or
3. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.2.8 FEEDWATER SYSTEM PIPE BREAK

Event Definition: A major feedwater line rupture is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the SGs to maintain shell side fluid inventory in the SGs. Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either an RCS cooldown (by excessive energy discharge through the break) or an RCS heatup. Potential RCS cooldown resulting from a secondary pipe rupture is evaluated in Section 15.1.5. Therefore, only the RCS heatup effects are evaluated for a feedwater line rupture.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Steam generator low-low level,
2. High pressurizer pressure,
3. Overtemperature ΔT,
4. SIS, or
5. Manual trip.

The ESF actuation functions identified in Table 15.0.8-1 are:

1. High Containment pressure,
2. Steam generator low-low water level,
3. Low compensated steam line pressure, and
4. High steam line differential pressure

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.3.1 PARTIAL LOSS OF FORCED REACTOR COOLANT FLOW

Event Definition: A partial loss of forced reactor coolant flow accident can result from a mechanical or electrical failure in a reactor coolant pump, or from a fault in a reactor coolant pump bus. If the reactor is at power at the time of the accident, the immediate effect of loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Low RCS flow,
2. Undervoltage,
3. Underfrequency, or
4. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.3.2 COMPLETE LOSS OF FORCED REACTOR COOLANT FLOW

Event Definition: A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of forced reactor coolant flow is a rapid increase in the reactor coolant temperature.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Low RCS flow,
2. Undervoltage,
3. Underfrequency, or
4. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.3.3 REACTOR COOLANT PUMP SHAFT SEIZURE (LOCKED ROTOR)

Event Definition: An instantaneous seizure of a reactor coolant pump rotor is assumed to occur which rapidly reduces flow through the affected reactor coolant loop. A reactor trip on a low flow signal is credited.

The transient may be terminated by the following reactor trip functions:

1. Low RCS flow, or
2. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.3.4 REACTOR COOLANT PUMP SHAFT BREAK

Event Definition: The accident is postulated as an instantaneous failure of a reactor coolant pump shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip is initiated on a low flow signal in the affected reactor coolant loop.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Low RCS flow, or
2. Manual trip

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.4.1 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

Event Definition: A RCCA withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCA's resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control or rod control systems. This could occur with the reactor subcritical or during startup.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Power range neutron flux (low or high setting),
2. High positive flux rate,
3. Overtemperature ΔT,
4. Overpower ΔT, or
5. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.4.2 UNCONTROLLED ROD CLUSTER CONTROL ASSEMBLY BANK WITHDRAWAL AT POWER

Event Definition: This event is defined as the inadvertent addition of positive reactivity to the core caused by the uncontrolled withdrawal of an RCCA bank(s) while at power. Unless

terminated by manual or automatic action, the power mismatch and resultant reactor coolant temperature rise could eventually result in DNB.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Power range neutron flux (high setting),
2. Overtemperature ΔT,
3. Overpower ΔT,
4. High pressurizer pressure,
5. High pressurizer level, or
6. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.4.3 ROD CLUSTER CONTROL ASSEMBLY MISOPERATION (SYSTEM MALFUNCTION OR OPERATOR ERROR)

Event Definition: Rod cluster control assembly misoperation accidents include:

1) One or more dropped assemblies within the same group.
2) A dropped full length assembly bank.
3) Statically misaligned full length assembly.
4) Withdrawal of a single full length assembly.

The dropped RCCA and dropped RCCA bank events are initiated by a de-energized control rod drive mechanism or by a malfunction associated with a RCCA bank during power operation. The result is that a single RCCA or RCCA bank falls into the core. The dropped RCCA promptly inserts negative reactivity which reduces reactor power and disturbs the power distribution, resulting in an increase (augmentation) of local power peaking.

The static misalignment events occur when a malfunction of the control rod drive mechanism causes a control rod to be out of alignment with its bank. Misalignment occurs when the rod is either higher or lower than any of the other control rods in the same bank. During this event, the reactor is at steady-state rated full power conditions, and no excursion of core temperature, pressure, flow, or power occurs.

The rod withdrawal event is initiated by an electrical or mechanical failure in the Rod Control System that causes the inadvertent withdrawal of a single RCCA. A rod is withdrawn from the reactor core causing an insertion of positive reactivity which results in a power excursion transient, increasing the core heat flux and creating a challenge to DNB margin.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Overtemperature ΔT,
2. High pressurizer level, or

3. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on these accident scenarios and the conclusions of the FSAR remain valid.

15.4.4 STARTUP OF AN INACTIVE REACTOR COOLANT PUMP AT AN INCORRECT TEMPERATURE

Event Definition: The inadvertent startup of an idle loop while operating would result in the sudden introduction of colder water into the core from the idle loop which could cause an unplanned reactivity insertion and power increase. Starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature would result in the injection of cold water into the core, which would cause a reactivity insertion and subsequent power increase.

The event is not credible in Modes 1 and 2 because the HNP TS require that three reactor coolant pumps operate in Modes 1 and 2. In Mode 3 and below, the reactor is subcritical and there is no significant load on the plant. The potential for a significant reactivity excursion due to this event is negligible.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Low flow interlocked with P-8, or
2. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.4.5 A MALFUNCTION OR FAILURE OF THE FLOW CONTROLLER IN A BWR LOOP THAT RESULTS IN AN INCREASED REACTOR COOLANT FLOW RATE

This section is not applicable to HNP.

15.4.6 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT

Event Definition: Reactivity can be added to the core by feeding primary grade water into the RCS via the reactor makeup portion of the Chemical and Volume Control System, resulting in decreasing boron concentration in the RCS. The dilution of primary system boron adds positive reactivity to the core. This event can lead to an erosion of shutdown margin for subcritical initial conditions, or a slow power excursion for at-power conditions. A boron dilution for at-power conditions behaves in a similar manner to a slow Uncontrolled RCCA Bank Withdrawal event (Section 15.4.2).

Boron dilution is a manual operation under strict administrative controls with procedures calling for a limit on the rate and duration of dilution. The rate of addition of unborated makeup water to the RCS is limited by operator response to alarm setpoints for reactor makeup water flow, charging flow, and letdown flow.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Power range high flux,
2. Overtemperature ΔT, or
3. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

Event Definition: Fuel and core loading errors can arise from the inadvertent loading of one or more fuel assemblies into improper positions, or the improper addition or removal of discrete burnable absorber rod assemblies, when applicable. These loading errors can result in severe changes in the core power distribution which may be undetectable by the incore instrumentation.

Steady state power distributions are calculated in three dimensions for several fuel misload cases. For each case analyzed, the assembly powers in instrumented core locations are compared to a normally loaded core to determine if the case would be detected at the time of the initial low-power flux map used in verifying that the core is properly loaded. Full power operation with the most severe peaking at any core location resulting from undetected misloadings are analyzed.

Misloadings which exceed the criteria from the low-power flux map are detectable with the incore instrumentation system. Misloadings which are undetectable at the time of the low-power flux map are analyzed to ensure fuel failures in excess of allowed limits will not occur as a result of this event.

Effect of Proposed Change: This event is not mitigated by a reactor trip function. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.4.8 SPECTRUM OF ROD CLUSTER CONTROL ASSEMBLY EJECTION ACCIDENTS

Event Definition: This event is defined as the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The transient is terminated by the Doppler reactivity effects of increased fuel temperature, and by an automatic reactor trip. This

event challenges deposited enthalpy, radiological consequences, and pressurization acceptance criteria.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Power range high flux,
2. High positive flux rate,
3. Overtemperature ΔT,
4. Overpower ΔT,
5. Low pressurizer pressure, or
6. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.4.9 SPECTRUM OF ROD DROP ACCIDENTS IN A BWR

This section is not applicable to HNP.

15.5.1 INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM DURING POWER OPERATION

Event Definition: Spurious ECCS operation at power could be caused by operator error or a false electrical actuation signal. A spurious signal may originate from any of the SIS actuation channels. If a reactor trip does not occur coincident with safety injection actuation, the turbine throttle valves will open to offset the addition of negative reactivity from the SIS. The transient is eventually terminated by the RTS due to low pressurizer pressure or manual trip.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Low pressurizer pressure,
2. Manual trip, or
3. Safety injection trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.5.2 CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

Event Definition: An increase in reactor coolant inventory which results from the addition of cold, unborated water to the RCS is analyzed in Section 15.4.6. An increase in reactor coolant inventory which results from the injection of highly borated water into the RCS is analyzed in Section 15.5.1.

15.5.3 A NUMBER OF BWR TRANSIENTS

This section is not applicable to HNP.

15.6.1 INADVERTENT OPENING OF A PRESSURIZER SAFETY OR POWER OPERATED RELIEF VALVE

Event Definition: An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer power operated relief or safety valve. Initially the event results in a rapidly decreasing RCS pressure until this pressure reaches a value corresponding to the hot leg saturation pressure. At this time, the pressure decrease is slowed considerably. The pressure continues to decrease throughout the transient.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Pressurizer low pressure,
2. Overtemperature ΔT, or
3. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.6.2 BREAK IN INSTRUMENT LINE OR OTHER LINE FROM REACTOR COOLANT PRESSURE BOUNDARY THAT PENETRATE CONTAINMENT

Event Definition: A primary sample or instrument line break provides a release path for reactor coolant outside Containment. The line break selected for analysis is the letdown line which penetrates the Containment. This is the largest penetration whose failure could result in an event in this category. This failure would result in larger releases than would be the case for the smaller instrument and sample lines.

Following such a break, the RCS pressure decreases due to the loss of reactor coolant. When the pressurizer pressure has reached the low pressure setpoint, a reactor trip is initiated. A turbine trip follows a reactor trip and results in an increase in secondary side pressure to the SG safety valve set pressure. The safety injection signal on low pressurizer pressure terminates the break flow by isolating the letdown line inside Containment.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Pressurizer low pressure, or
2. Manual trip.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.6.3 STEAM GENERATOR TUBE RUPTURE

Event Definition: A SG tube rupture results in the leakage of contaminated reactor coolant into the secondary system and the subsequent release of a portion of that activity to the atmosphere. The accident examined is the complete severance of a single SG tube. The accident is assumed to take place at power with the reactor coolant contaminated with fission products corresponding to continuous operation with a limited number of defective fuel rods.

The transient may be terminated by the following reactor trip functions identified in FSAR Table 15.0.8-1:

1. Low pressurizer pressure, or
2. Overtemperature ΔT.

Effect of Proposed Change: The reactor trip functions identified for this event are not affected by the proposed change. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.6.4 SPECTRUM OF BWR STEAM SYSTEM PIPING FAILURES OUTSIDE CONTAINMENT

This section is not applicable to HNP.

15.6.5 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 reviewed for impact by 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 analyses to determine acceptable core cooling. Thus, the peak cladding temperature and clad oxidation analyses are not affected by the proposed change.

The reactor vessel and loop LOCA blowdown force calculations are independent of the RTS actuations due to the relatively short time frames involved. The interlock being modified is not credited in any of these analyses, and thus they are not affected.

The long term post-LOCA subcriticality analysis evaluates the containment sump average boron concentration relative to that required to ensure the core remains subcritical. This analysis is independent of the RTS actuation signals. Thus, the subcriticality analysis is not affected by the proposed change.

The long-term core cooling minimum flow and hot leg switchover analyses are dependent on the core decay heat as a function of time. The flow rates determined ensure adequate core cooling. The timing for hot leg switchover ensures the boron concentration within the core remains acceptable post-LOCA. These analyses are independent of the RTS actuation signals. Thus, these analyses are not affected by the proposed change.

Effect of Proposed Change: The proposed change has no effect on these accident scenarios and the conclusions of the FSAR remain valid.

15.7.1 RADIOACTIVE WASTE GAS SYSTEM LEAK OR FAILURE

Event Definition: The most limiting waste gas accident is defined as an unexpected and uncontrolled release to the atmosphere of the radioactive xenon and krypton fission gases that are stored in one operating waste gas decay tank. The analysis assumes that the plant has been operating at the power level of 2958 MWt (rated power of 2948 MWt with 0.34% uncertainty) with one percent failed fuel for an extended period sufficient to achieve equilibrium radioactive concentrations in the RCS. As soon as possible after shutdown, all noble gases have been removed from the RCS and transferred to the gas decay tank which is assumed to release its contents in an uncontrolled manner.

Effect of Proposed Change: This event assumes the reactor is shutdown, and the turbine generator is not on-line. No reactor trip functions are credited. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.7.2 LIQUID WASTE SYSTEM LEAK OR FAILURE

HNP Liquid Waste System leaks or failures are bounded by the Analyzed Liquid Tank Failure in Section 15.7.3.

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID TANK FAILURE

Event Definition: The consequences of the postulated failure of a tank containing potentially contaminated liquid on the nearest potable water supply and the nearest surface water in an unrestricted area are discussed in FSAR Sections 2.4.12 and 2.4.13.

Effect of Proposed Change: This event is independent of reactor operation. No reactor trip functions are credited. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.7.4 DESIGN BASIS FUEL HANDLING ACCIDENTS

Event Definition: The analyzed Fuel Handling Accident (FHA) inside containment involves dropping a spent fuel assembly resulting in the rupture of the cladding of all the fuel rods (264) in the assembly. The projected worst-case FHA in the Fuel Handling Building involves dropping

a recently discharged pressurized water reactor (PWR) assembly (including the handling tool) on top of another recently discharged PWR assembly in a fuel storage rack.

Effect of Proposed Change: This event is independent of reactor operation. No reactor trip functions are credited. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

Event Definition: As discussed in FSAR Section 9.1, the cask handling crane is prohibited from traveling over the new and spent fuel pools or any unprotected safety related equipment. Thus, an accident resulting from dropping a cask or other major load into the new or spent fuel pools is not credible.

Effect of Proposed Change: This event is independent of reactor operation. No reactor trip functions are credited. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

Event Definition: An ATWS is an anticipated operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power) that is accompanied by a failure of the RTS to shut down the reactor. The HNP plant-specific analyses performed for the Measurement Uncertainty Recapture (MUR) Power Uprate demonstrated that the effects of the MUR would not result in unacceptable consequences (License Amendment No. 139 and Safety Evaluation provided Reference 9).

Effect of Proposed Change: This event does not credit reactor trip functions to mitigate the event. Therefore, the proposed change has no effect on this accident scenario and the conclusions of the FSAR remain valid.

3.4 Summary

The HNP FSAR analyses of record do not credit the anticipatory reactor trip on turbine trip in the analyses that show the accident analysis acceptance criteria are met. The conclusions of the FSAR remain valid following the proposed change to the RTS where the turbine trip interlock is changed from P-7 to P-8. The P-7 interlock receives input from the power range neutron flux instrumentation and the turbine first stage pressure. The P-8 interlock only receives input from the power range neutron flux instrumentation. This change is acceptable because the P-8 interlock will continue to receive reliable input from the power range neutron flux instrumentation.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements and Guidance

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.

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. The regulation does not specify the particular requirements to be included in a plant's TS.

Specifically, 10 CFR 50.36(c)(2)(ii) requires that a TS LCO be established for each item meeting one or more of the following criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The change to the enabling interlock for a reactor trip on turbine trip continues to meet this regulation.

4.2 Precedents

The NRC has previously approved similar submittals for plants changing the interlock at which the reactor trip from turbine trip is enabled. Specific precedents include the following:

- Seabrook Station, Unit No. 1 (ADAMS Accession No. ML080460674)
- H.B. Robinson Steam Electric Plant, Unit No. 2 (ADAMS Accession No. ML100120068)
- Indian Point Nuclear Generating Unit No. 3 (ADAMS Accession No. ML003780834)
- North Anna Power Station, Unit No. 1 and No. 2 (ADAMS Accession No. ML013460457)
- Salem Nuclear Generating Station, Unit Nos. 1 and 2 (ADAMS Accession No. ML011690022)
- Braidwood Station, Unit 1 and Byron Station, Units 1 and 2 (ADAMS Accession No. ML020850675)
- Donald C. Cook Nuclear Plant, Units 1 and 2 (ADAMS Accession No. ML062840162)

4.3 Significant Hazards Consideration

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), proposes a revision to Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Technical Specifications (TS). 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).

Duke Energy has evaluated whether or not a significant hazards consideration is warranted with the proposed amendment by addressing the three criterion set forth in 10 CFR 50.92(c) as discussed below:

(1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed amendment revises requirements for the reactor protection system interlock associated with the turbine trip protection function. Specifically, the proposed change allows the interlock for turbine trip function to be raised from the current interlock setting of nominally 10 percent reactor power (P-7) to nominally 49 percent reactor power (P-8).

The applicable accident analyses, as described in the HNP Final Safety Analysis Report (FSAR), have been reviewed and the turbine trip input to reactor trip has been verified as not used in the accident analyses. Therefore, it is concluded that the consequences as described in the FSAR accident analyses are unaffected by the proposed change.

A best-estimate analysis of plant response to a turbine trip at nominally 49 percent power provided with the amendment request shows that the applicable acceptance criteria continue to be met. Specifically, it is shown that a turbine trip without a reactor trip below 49 percent power does not challenge the pressurizer power operated relief valves (PORVs) or the steam generator safety valves. Consequently, the proposed change does not adversely affect the probability of a small break loss of coolant accident due to a stuck-open PORV. The evaluation of a turbine trip due to a loss of condenser from the current P-8 setpoint of 49 percent reactor thermal power concluded that the PORVs and the secondary system safety relief valves are expected to be challenged. However, overfilling of the pressurizer will not occur and this Condition II event will not initiate a Condition III event. The challenge to the PORVs with a loss of condenser does not violate design or licensing criteria.

Therefore, operation of the facility in accordance with the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

No new accident initiators or precursors are introduced by the proposed amendment. Changing the interlock for the reactor trip on turbine trip from P-7 to P-8 changes the power level associated with enabling and disabling the reactor trip on turbine trip function. While the reactor trip on turbine trip provides additional protection and conservatism beyond that required for the health and safety of the public, it is an anticipatory trip and no credit is taken in any of the FSAR safety analyses. The change does not affect how the associated trip functional units operate or function. The changes do not create the possibility of a new or different kind of accident from any previously evaluated since these interlock changes do not affect the way that the associated trip functional units operate or function.

Therefore, operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any previously evaluated.

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

Response: No.

No safety analyses were changed or modified as a result of the proposed change in reactor trip on turbine trip interlock from P-7 to P-8. The applicable FSAR accident analyses were reviewed and it was concluded that the accident analyses are unaffected by the proposed change. An analysis of plant response to a turbine trip at nominally 49 percent power shows that the applicable acceptance criteria are met. Furthermore, the reactor trip on turbine trip provides additional protection and conservatism beyond that required for the health and safety of the public and the safety analyses in Chapter 15 of the FSAR do not take credit for this reactor trip.

Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant reduction in the margin of safety.

Based upon the above evaluation, Duke Energy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATIONS

Duke Energy has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined by 10 CFR 20, or it would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs be prepared in connection with the proposed amendment.

6.0 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,” September 2014.

2. DPC-NE-3008-PA, “Thermal-Hydraulic Models for Transient Analysis,” Revision 0, as approved by NRC Safety Evaluation dated April 10, 2018 (see ADAMS Accession No. ML16278A080 (public), ML16278A082 (non-public)).

3. DPC-NE-3009-PA, “FSAR / UFSAR Chapter 15 Transient Analysis Methodology,” Revision 0, as approved by NRC Safety Evaluation dated April 10, 2018 (see ADAMS Accession No. ML16278A080 (public), ML16278A082 (non-public)).

4. NRC Letter to Steven Capps (Duke Energy), “Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications for Methodology Reports DPC-NE-3008-P, Revision 0, “Thermal-Hydraulic Models for Transient Analysis,” and DPC-NE-3009-P, Revision 0, “FSAR / UFSAR Chapter 15 Transient Analysis Methodology” (CAC NOS. MF8439 and MF8440; EPID L-2016-LLA-0012), April 10, 2018 (ADAMS Accession No. ML18060A401 (public), ML18060A318 (non-public)).

5. NUREG-0737, “Clarification of TMI Action Plan Requirements,” Item II.K.3.10, Proposed Anticipatory Trip Modification, November 1980 (ADAMS Accession No. ML051400209).

6. DPC-NE-3002-A, Revision 4b, “UFSAR Chapter 15 System Transient Analysis Methodology,” September 2010 (ADAMS Accession No. ML16102A159 (public), ML16102A169 (non-public)).

7. DPC-NE-3000-A, Revision 5a, “Thermal-Hydraulic Transient Analysis Methodology,” October 2012 (ADAMS Accession No. ML16032A004 (public), ML16032A005 (non-public)).

8. U.S. Nuclear Regulatory Commission, NUREG-0800, “Standard Review Plan,” Section 15.0, “Introduction - Transient and Accident Analyses,” Revision 3, March 2007 (ADAMS Accession No. ML070710376)

9. NRC Letter to Chris Burton (Progress Energy Carolinas, Inc.), “Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate (TAC No. ME6169)”, dated May 30, 2012 (ADAMS Accession No. ML11356A096).

7.0 FIGURES

1. Nuclear Power and Turbine Load vs Time
2. Pressurizer Pressure vs Time
3. Pressurizer Level vs Time
4. Loop T_{avg} vs Time
5. Steam Generator Pressure vs Time
6. Steam Mass Flow vs Time
7. Feedwater Mass Flow vs Time
8. Control Rod Position vs Time
9. HNP Loss of Normal Feedwater – RCS Loop Average Temperature Comparison
10. HNP Loss of Normal Feedwater – SG Liquid Inventory Comparison
11. RNP Loss of Normal Feedwater – RCS Loop Average Temperature Comparison
12. RNP Loss of Normal Feedwater – SG Liquid Inventory Comparison

Figure 1
Nuclear Power and Turbine Load vs Time

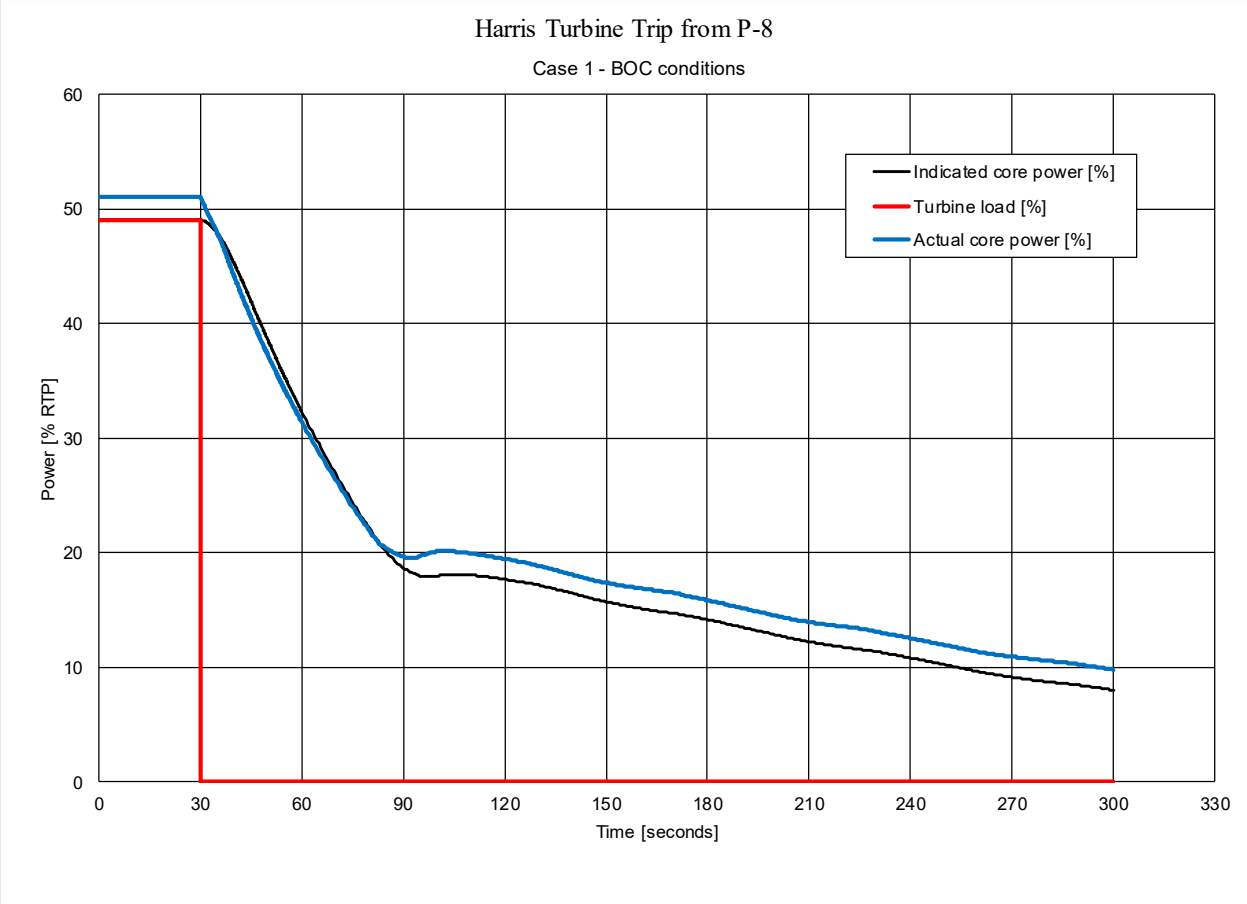


Figure 2
Pressurizer Pressure vs Time

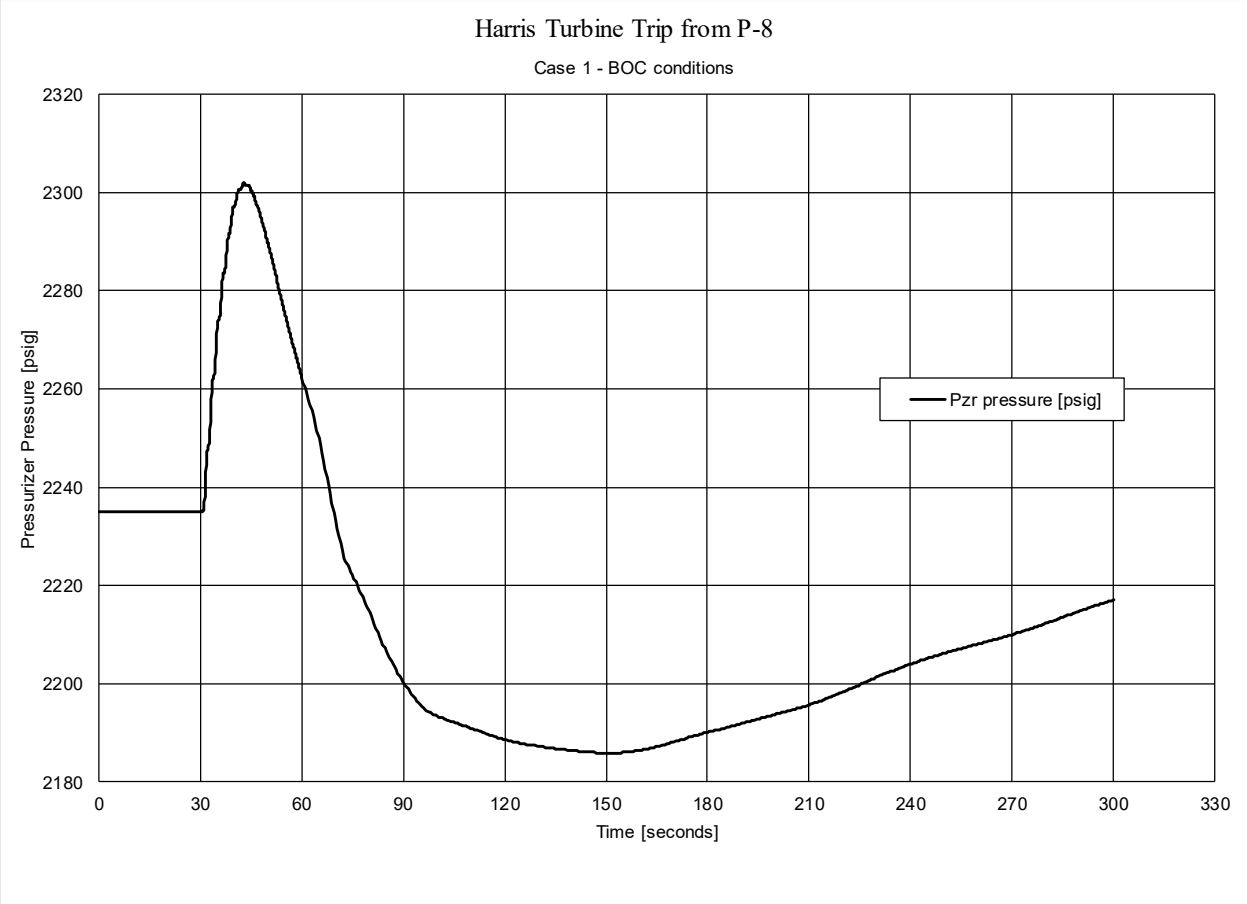


Figure 3
Pressurizer Level vs Time

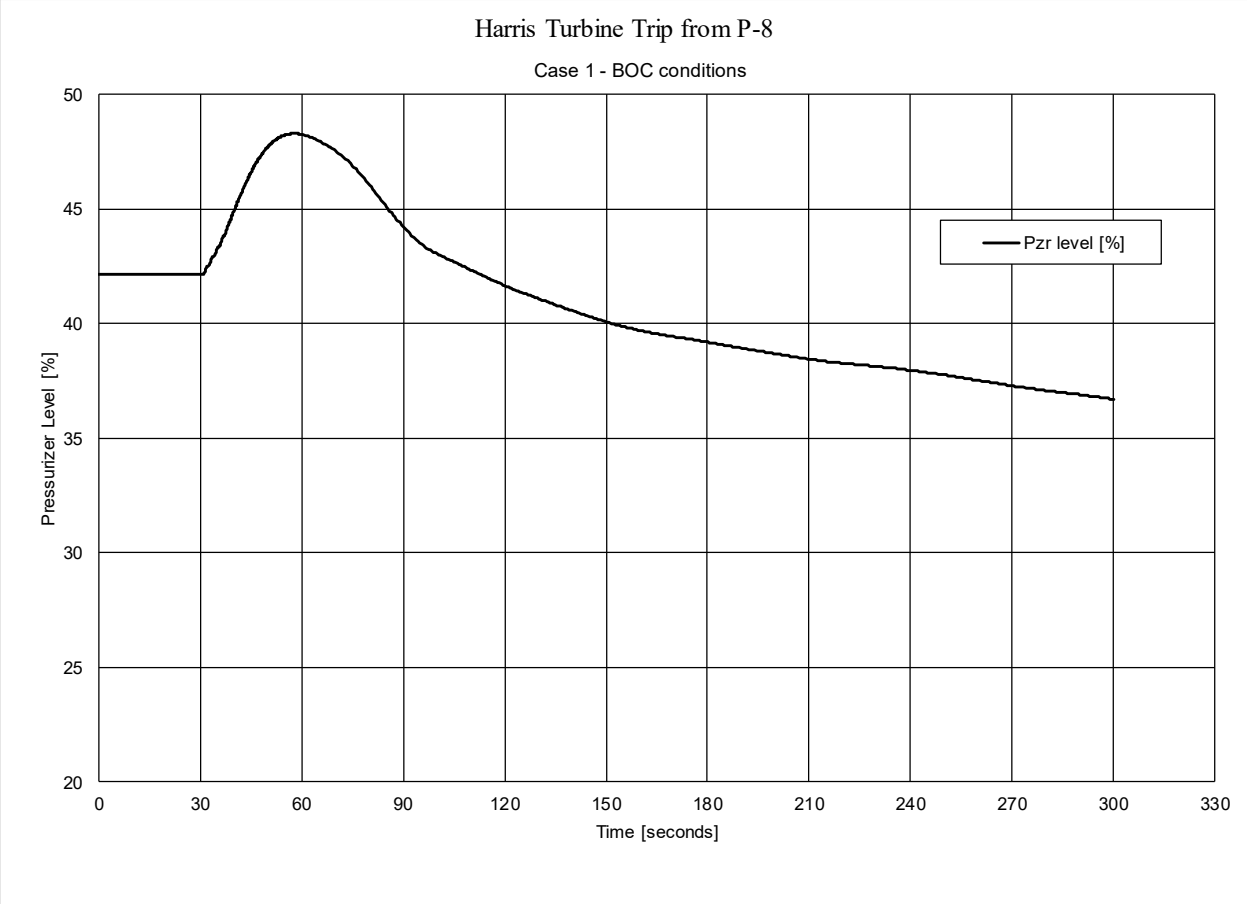


Figure 4
Loop T_{avg} vs Time

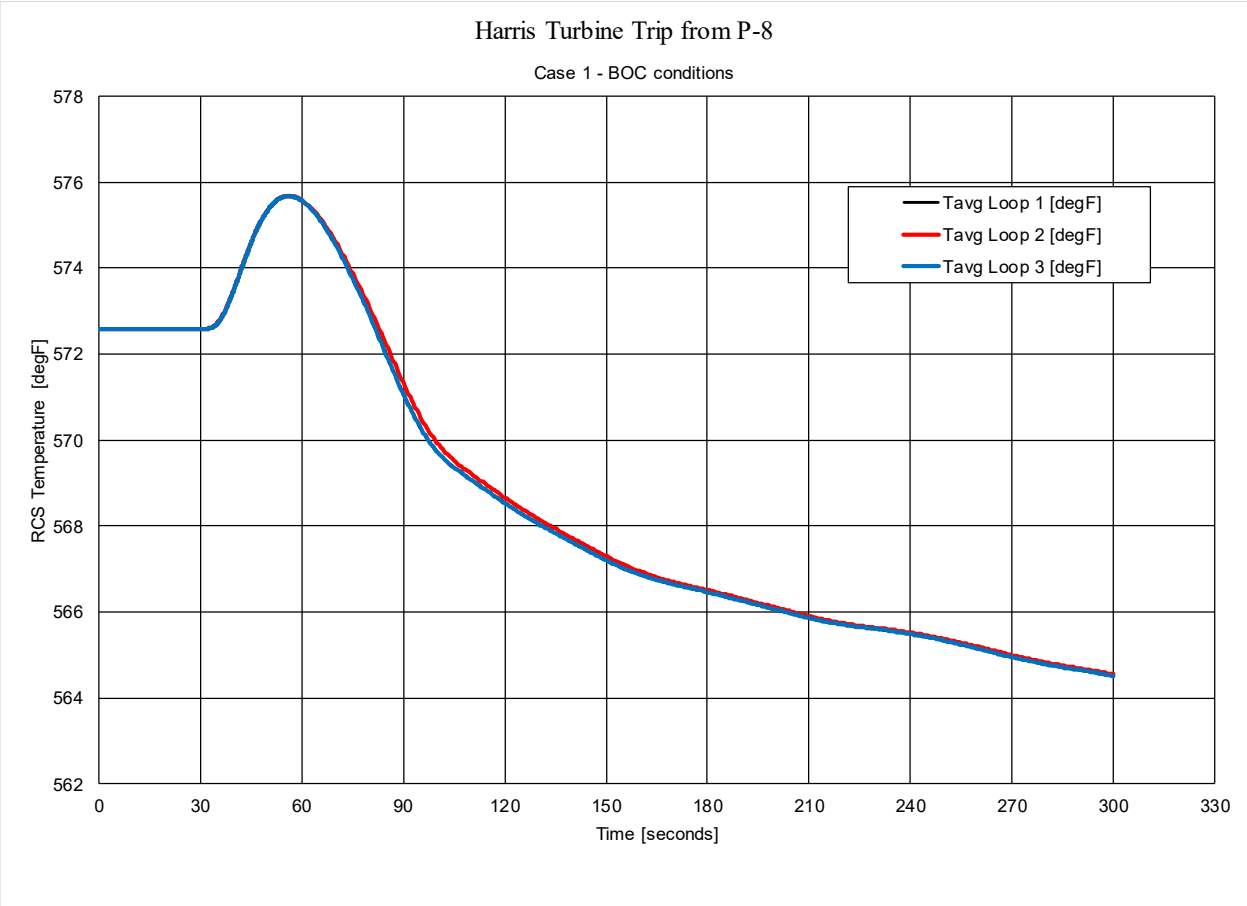


Figure 5
Steam Generator Pressure vs Time

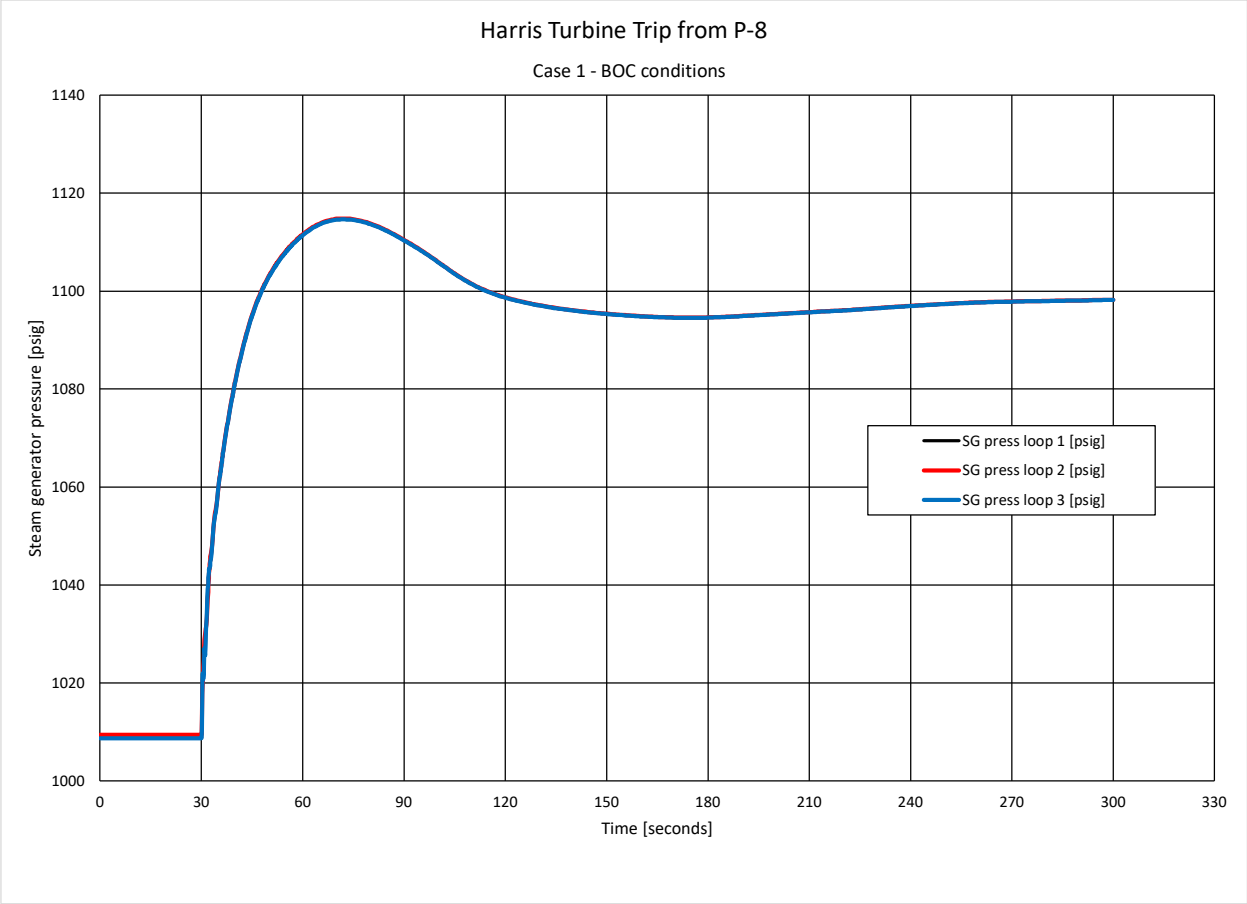


Figure 6
Steam Mass Flow vs Time

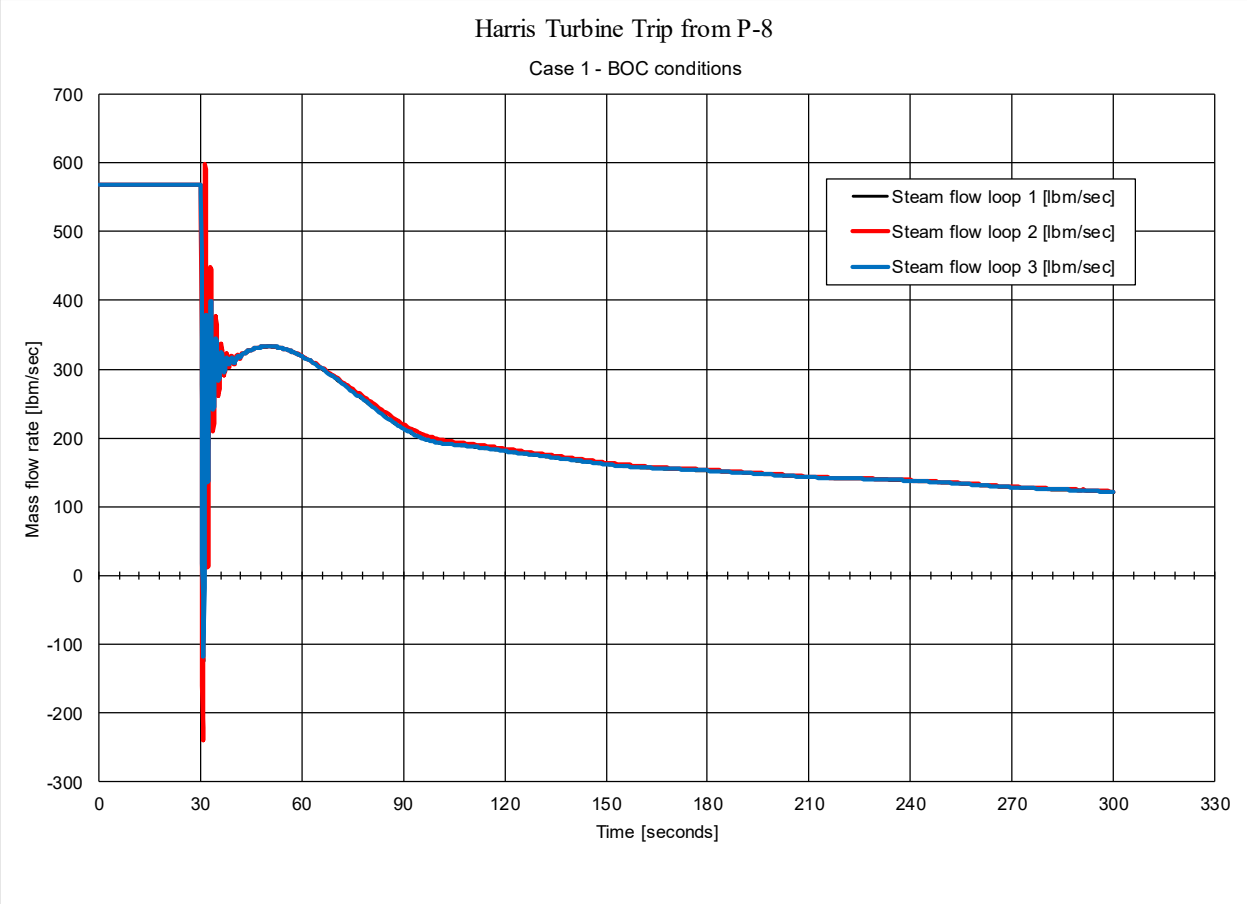


Figure 7
Feedwater Mass Flow vs Time

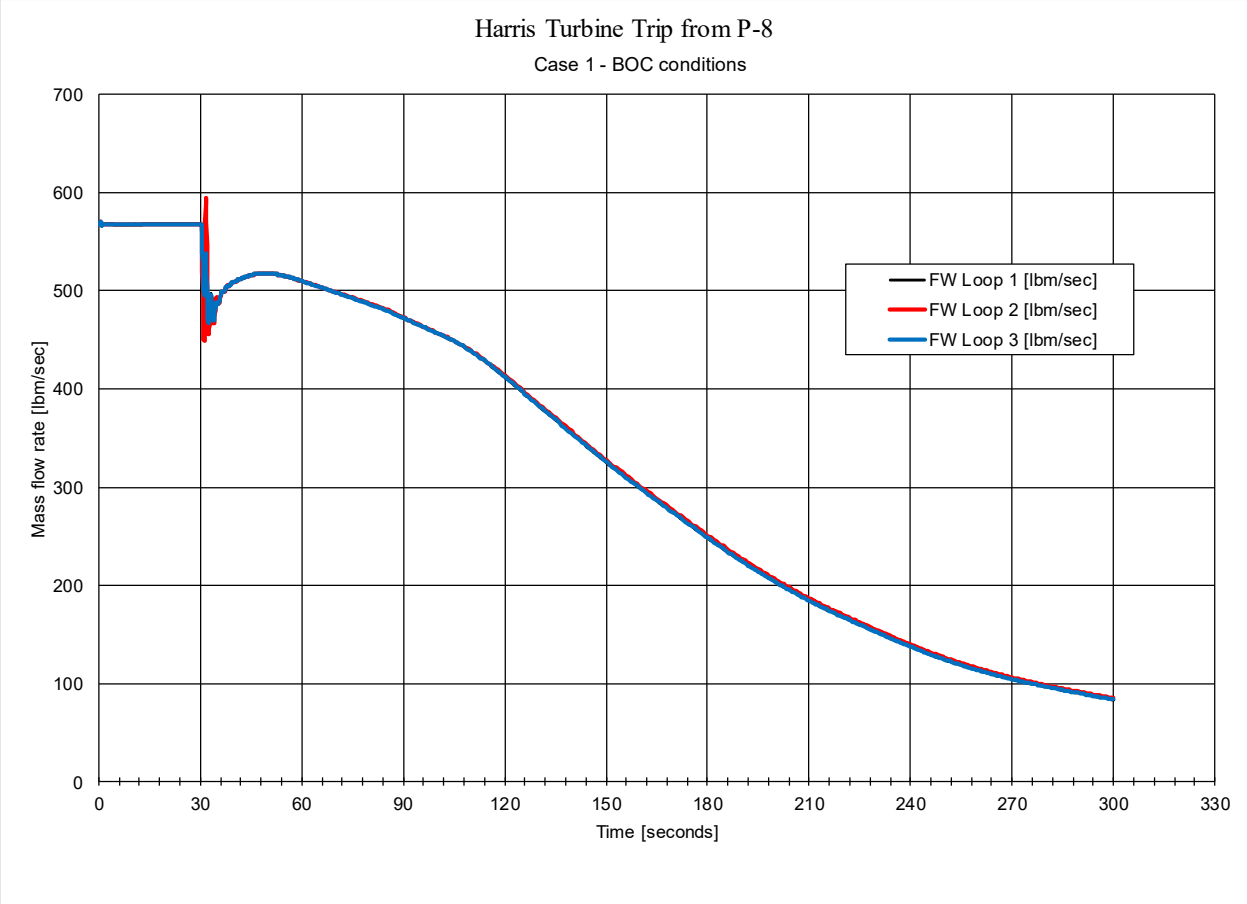


Figure 8
Control Rod Position vs Time

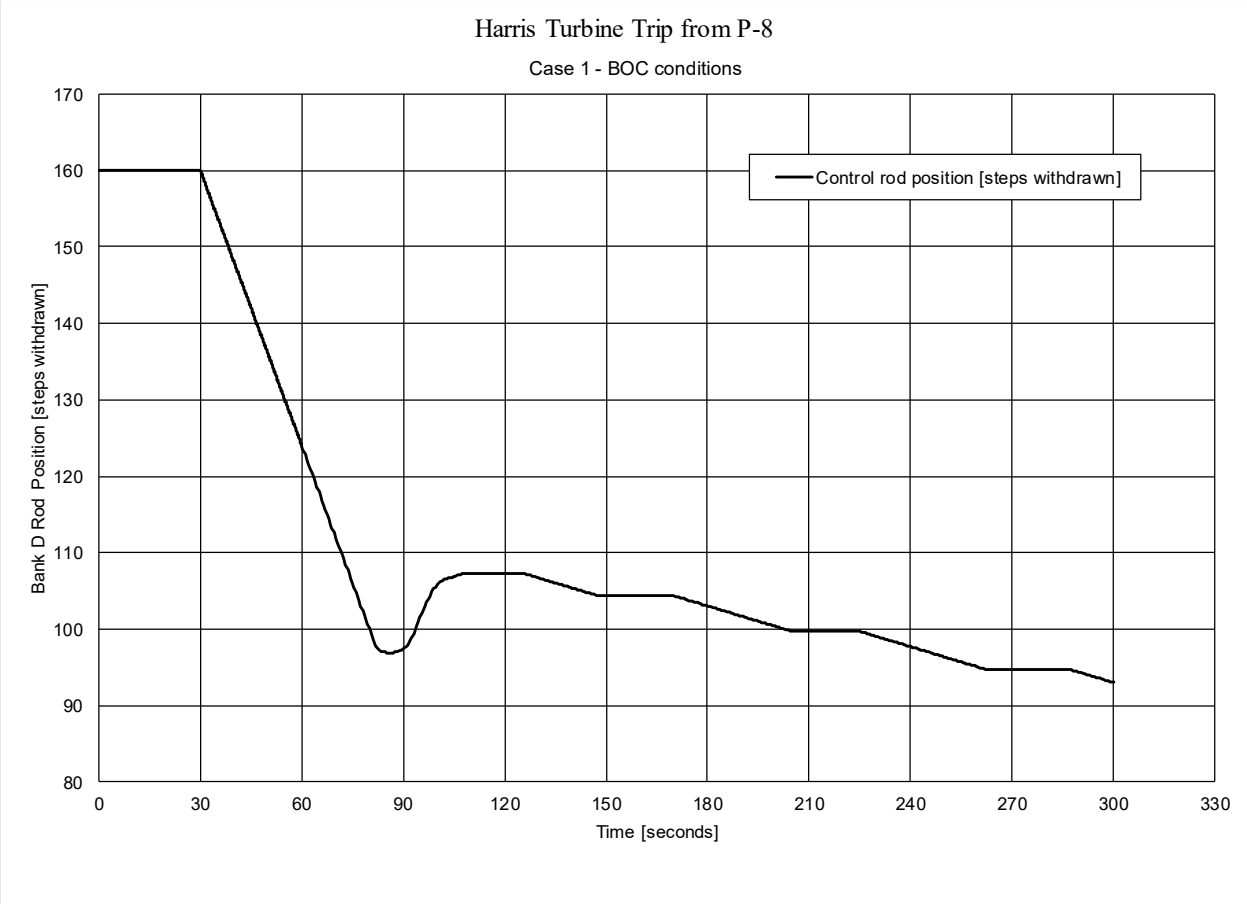


Figure 9
HNP Loss of Normal Feedwater
RCS Average Temperature Comparison

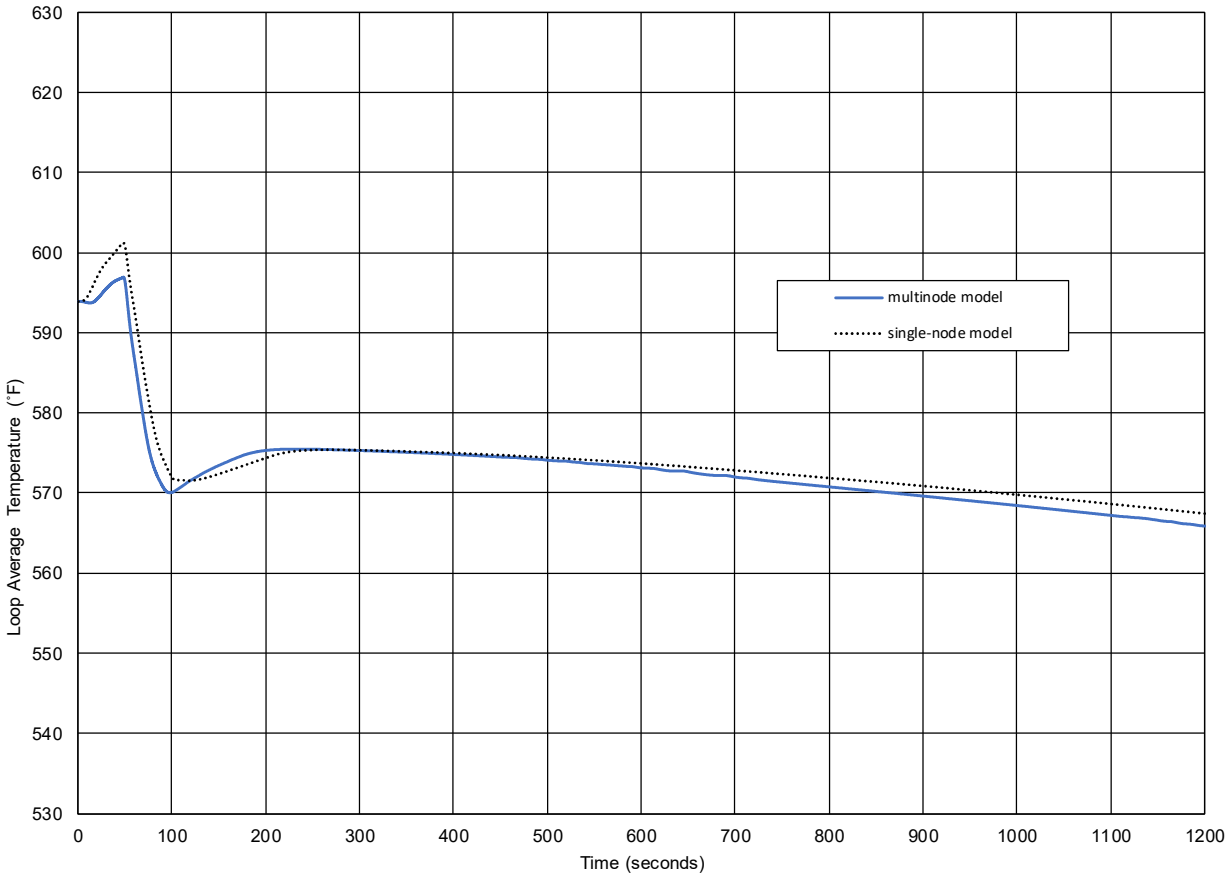


Figure 10
HNP Loss of Normal Feedwater
SG Liquid Inventory Comparison

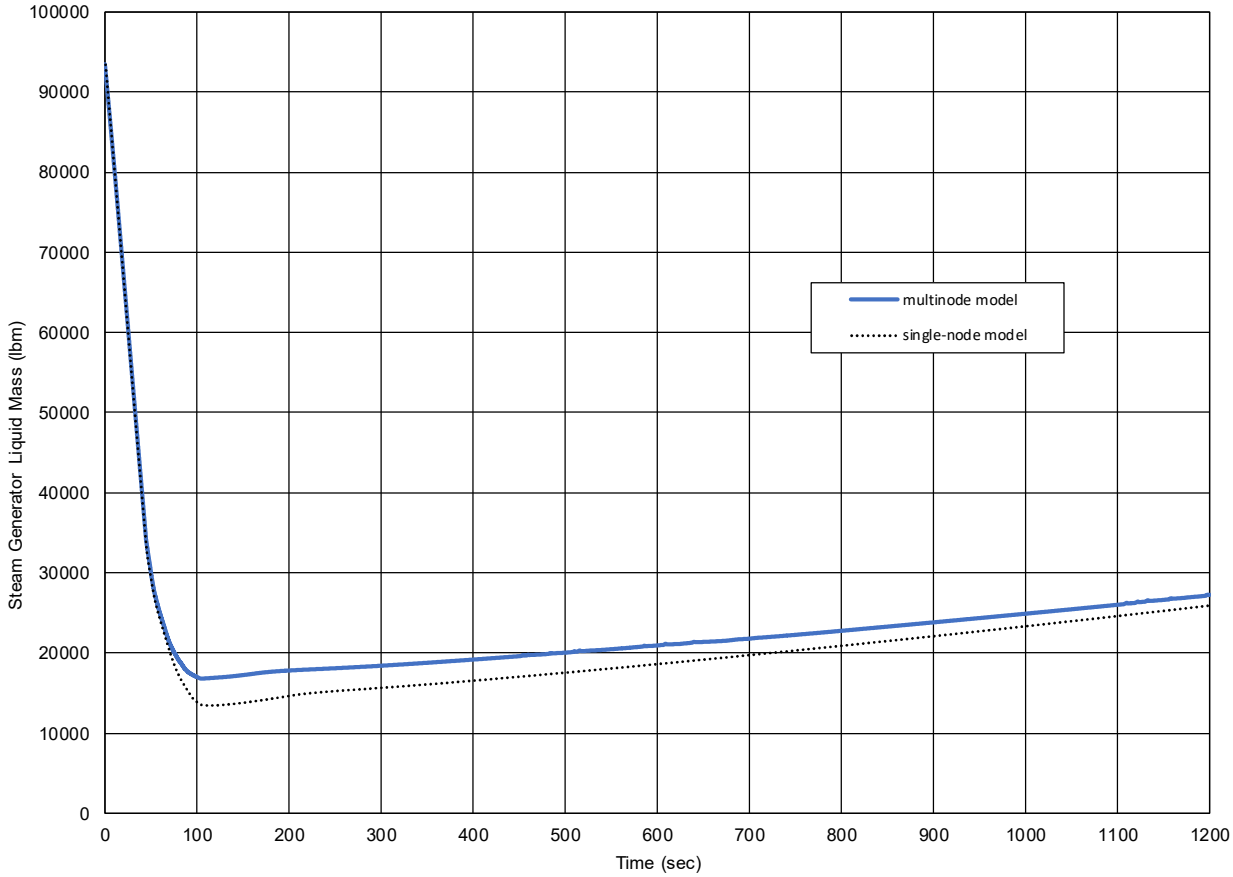


Figure 11
RNP Loss of Normal Feedwater
RCS Average Temperature Comparison

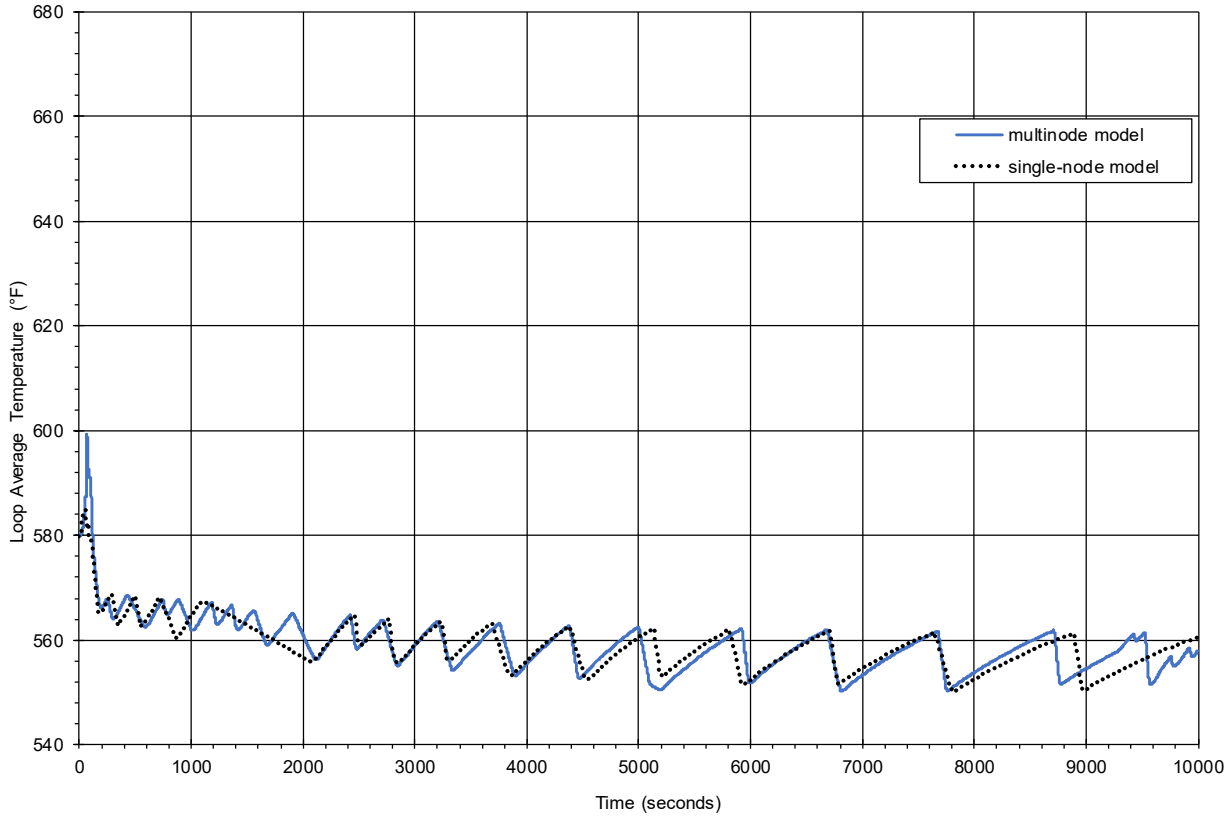
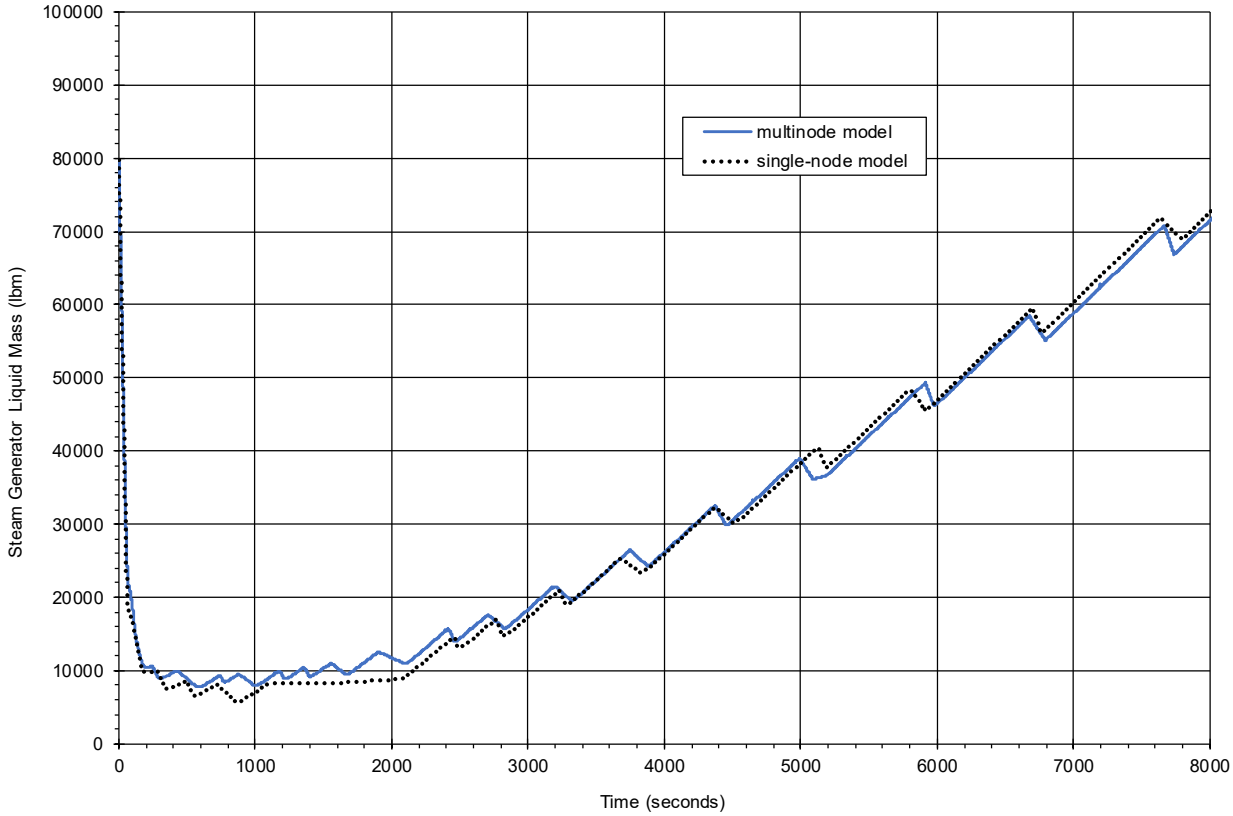


Figure 12
RNP Loss of Normal Feedwater
SG Liquid Inventory Comparison



ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

1 PAGE PLUS THE COVER

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-7~~) 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

ENCLOSURE 3

TECHNICAL SPECIFICATION BASES CHANGES (MARK-UP)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

1 PAGE PLUS THE COVER

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power the Reactor trip from the Turbine trip is automatically blocked by the loss of ~~P-7~~ P-8 (a power level of approximately ~~10%~~ 49% of RATED THERMAL POWER ~~or a turbine inlet pressure at approximately 10% of full power equivalent~~); and on increasing power, reinstated automatically by ~~P-7~~ P-8.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip), and deenergizes the high voltage to the detectors. On decreasing power, Source Range Level trips are automatically reactivated and high voltage restored.

P-7 On increasing power P-7 automatically enables Reactor trips on low flow in more than one reactor coolant loop, reactor coolant pump motor undervoltage and underfrequency, ~~turbine trip,~~ pressurizer low pressure and pressurizer high level. On decreasing power, the above listed trips are automatically blocked.

P-8 On increasing power, P-8 automatically enables Reactor trips on low flow in one or more reactor coolant loops and turbine trip. On decreasing power, the P-8 automatically blocks the above listed trips.

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. Provides input to P-7.

P-13 Provides input to P-7.

References

1. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."
2. BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code.”
3. ANP-10341P-A, Revision 0, “The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations,” September 2018.