Tanya M. Hamilton
Vice President
Harris Nuclear Plant
5413 Shearon Harris Road
New Hill, NC 27562-9300

10 CFR 50.90

July 30, 2018
Serial: RA-18-0072

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 Modify Reactor Trip System and Engineered Safety Features Actuation System Instrumentation Trip Setpoints

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (Tech Spec) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment modifies Tech Spec Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," and Tech Spec Table 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints," to optimize safety analysis margin in the Final Safety Analysis Report Chapter 15 transient analyses.

The proposed changes have 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 changes involve no significant hazards consideration. Attachment 1 of this license amendment request provides Duke Energy's evaluation of the proposed changes. Attachment 2 provides a copy of the proposed Tech Spec changes. Attachment 3 provides a copy of the Tech Spec Bases markup based on the proposed changes (for information only).

Approval of the proposed license amendment is requested within 12 months of acceptance. The amendment shall be implemented prior to the startup of Cycle 23.

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

This document contains no new Regulatory Commitments.

Please refer any questions regarding this submittal to Jeffrey Robertson, HNP Regulatory Affairs Manager, at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on July 30 , 2018.

Sincerely,

Tanya M. Hamilton

Tanya M. Hamilton

Attachments:

1. Evaluation of the Proposed Change
2. Proposed Technical Specification Changes
3. Proposed Technical Specification Bases Change (For Information Only)

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

I declare under penalty of perjury that the foregoing is true and correct.
Executed on July , 2018.

Sincerely,

Tanya M. Hamilton

Attachments:
1. Evaluation of the Proposed Change
2. Proposed Technical Specification Changes
3. Proposed Technical Specification Bases Change (For Information Only)

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

RA-18-0072

ATTACHMENT 1

EVALUATION OF THE PROPOSED CHANGE

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

18 PAGES PLUS COVER

Evaluation of the Proposed Change

Subject: License Amendment Request to Modify Reactor Trip System and Engineered Safety Features Actuation System Instrumentation Trip Setpoints

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 Technical Specifications (Tech Spec) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment modifies Tech Spec Table 2.2-1, “Reactor Trip System Instrumentation Trip Setpoints,” and Tech Spec Table 3.3-4, “Engineered Safety Features Actuation System Instrumentation Trip Setpoints,” to optimize safety analysis margin in the Final Safety Analysis Report (FSAR) Chapter 15 transient analyses.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

The Reactor Protection System provides an automatic reactor trip function to the reactor trip breakers to protect against unsafe and improper reactor operation during steady state and transient power operation and to provide initiating signals to mitigate the consequences of faulted conditions. The system uses input signals including neutron flux, Reactor Coolant System (RCS) temperature, RCS Flow, pressurizer pressure, pressurizer level, steam generator level, reactor coolant pump under-voltage and under-frequency, turbine trip signals, and safety injection to provide a reactor trip signal.

The reactor trip setpoint limits specified in Tech Spec Table 2.2-1 are the nominal values at which the reactor trips are set for each functional unit. The trip setpoints (TS) have been selected to ensure that the core and RCS are prevented from exceeding their safety limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System (ESFAS) in mitigating the consequences of accidents. The setpoint for a Reactor Trip System (RTS) or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values (AV) for the Reactor TSs have been specified in Tech Spec Table 2.2-1. Operation with setpoints less conservative than the TS but within the AV is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its TS is found to exceed the AV. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Tech Spec Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. As specified in Tech Spec Table 2.2-1, in percent span, Z is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. Total Allowance, TA, is the difference, in

percent span, between the TS and the value used in the safety analyses for reactor trip. Rack Error, R, is the "as measured" deviation, in percent span, for the affected channel from the specified TS. Sensor Error, S, is either the "as measured" deviation of the sensor from its calibration point or the value specified in Tech Spec Table 2.2-1, in percent span, from the analysis assumptions. Use of Tech Spec Equation 2.2-1 for RTS setpoints, and the equivalent Tech Spec Equation 3.3-1 for ESFAS setpoints, allows for a sensor drift factor and an increased rack drift factor, and provides a threshold value for determination of OPERABILITY.

The methodology to derive the TSs is based upon combining all of the uncertainties in the channels. Inherent to the determination of the TSs are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the AV exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The various reactor trip circuits automatically open the reactor trip breakers whenever a condition monitored by the RTS reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a RTS which monitors numerous system variables, therefore providing trip system functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the RTS. The RTS initiates a turbine trip signal whenever reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive RCS cooldown and thus avoids unnecessary actuation of the ESFAS.

2.2 Current Technical Specification Requirements

The HNP Tech Specs are based upon the format and content of the NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," series. As a result, the HNP Tech Spec surveillance numbers and associated Bases numbers differ from those contained in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants." The current values of the Tech Spec Table 2.2-1 and Table 3.3-4 setpoints proposed for change are provided in tabular form in Section 2.4 below in Tables 1 through 7, along with their respective proposed changes.

This amendment will also adjust the content of the following Notes from Tech Spec Table 2.2-1:

NOTE 2: The channel’s maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; 2.0% of ΔT span for T_{avg} input; 0.4% of ΔT span for pressurizer pressure input; and 0.7% of ΔT span for ΔI input.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input and 0.2% of ΔT span for T_{avg} input.

NOTE 5: The sensor error is: 1.3% of ΔT span for $\Delta T/T_{avg}$ temperature measurements; and 1.0% of ΔT span for pressurizer pressure measurements.

2.3 Reason for the Proposed Change

The purpose of this proposed change is to update the safety analysis RTS and ESFAS limits to optimize safety analysis margin in FSAR Chapter 15 transient analyses. Departure from nucleate boiling (DNB) analyses can account for some uncertainties in the DNB limit which precludes the need to account for them deterministically. The method by which uncertainties are included in the DNB limit for HNP is the NRC-approved Statistical Core Design (SCD) methodology described in DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology," as approved by the NRC for use by HNP in Facility Operating License Amendment No. 148 (ADAMS Accession Package No. ML16049A630).

Instrument uncertainties are important in determining the ability of instrument setpoints and designs to be able to achieve their intended functions. In Chapter 15 analyses, the calculated instrument uncertainties are used to establish conservatively bounding initial conditions such as reactor power and temperature, and boundary conditions such as trip setpoints and automatic control system actuation.

The channel statistical allowance (CSA) is calculated for each parameter using a square root sum of the squares of all random uncertainties, plus non-random sources of uncertainty such as seismic or environmental allowances. This CSA is used to determine whether the analytical limits assumed in safety analyses are appropriate. In order for an analytical limit to bound instrument uncertainty, the TA, defined as the difference between the analytical limit and the nominal trip setpoint, must be greater than or equal to the CSA. The difference between the CSA and TA is the calculational margin (CM), and the larger the CM, the more penalizing the analytical limit will be during accident analysis (see Figure 1 at the end of Attachment 1).

For HNP, the safety analyses consider uncertainty in the initial condition and in the trip setpoints. In many instances, the initial condition uncertainty is also included in a corresponding trip setpoint uncertainty. In the case of a DNB SCD analysis, some of those uncertainties (power, pressure, temperature, and flow) are already included in the DNB limit and therefore do not need to be accounted for in either the initial condition or trip setpoint uncertainty. Therefore, for SCD thermal-hydraulic analyses, a separate analytical trip limit is assumed accounting for the fact that part of the CSA is already included in the DNB limit.

2.4 Description of the Proposed Change

The proposed change to HNP Tech Specs requests NRC approval of revised RTS and ESFAS limits that reflect the reduced CM and subsequent reduction in TA. As discussed in Section 2.2 above, the tables below contain both the current and proposed Tech Spec values for RTS and ESFAS Instrumentation. Tables 1 through 6 reflect proposed Tech Spec values for the impacted RTS instrumentation trip setpoints contained within Tech Spec Table 2.2-1, as identified by the Functional Unit number. Only the proposed values in bold font are requested for approval by this license amendment request (LAR). Other values within the table that differ between the current and proposed Tech Spec values that are not bold reflect values that are cycle-specific and will be contained within the Core Operating Limits Report (COLR) for Cycle 23. Table 7 reflects the proposed Tech Spec values for Functional Unit number 1.d, "Safety Injection, Pressurizer Pressure – Low," of Tech Spec Table 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoint."

Duke Energy proposes the following changes (in bold) to the HNP Tech Specs:

Table 1

Functional Unit No. 12 – Reactor Coolant Flow - Low

Tech Spec Term	Current Tech Spec Values	Proposed Tech Spec Values
Total Allowance (TA)	4.58% span	**3.08% span**
Z Term	1.98% span	**1.58% span**
Sensor Error (S)	0.60% span	**0.49% span**
Trip Setpoint (TS)	≥90.5% flow	**≥91.7% flow***
Allowable Value (AV)	≥89.5% flow	**≥90.6% flow**

*Addition of Notes 7 and 8

Table 2

Functional Unit No. 2.a – Power Range, Neutron Flux - High Setpoint

Tech Spec Term	Current Tech Spec Values	Proposed Tech Spec Values
Total Allowance (TA)	5.83% span	**4.58% span**
Z Term	4.56% span	**3.25% span**
Sensor Error (S)	0% span	0% span
Trip Setpoint (TS)	≤108% RTP	≤108% RTP
Allowable Value (AV)	≤109.5% RTP	**≤109.6% RTP**

[RTP = Rated Thermal Power]

Table 3

Functional Unit No. 9 – Pressurizer Pressure - Low

Tech Spec Term	Current Tech Spec Values	Proposed Tech Spec Values
Total Allowance (TA)	5.0% span	**4.625% span**
Z Term	1.52% span	1.52% span
Sensor Error (S)	1.50% span	1.50% span
Trip Setpoint (TS)	≥1960 psig	≥1960 psig
Allowable Value (AV)	≥1948 psig	≥1948 psig

Table 4

Functional Unit No. 10 – Pressurizer Pressure - High

Tech Spec Term	Current Tech Spec Values	Proposed Tech Spec Values
Total Allowance (TA)	7.5% span	**4.625% span**
Z Term	1.52% span	1.52% span
Sensor Error (S)	1.50% span	1.50% span
Trip Setpoint (TS)	≤2385 psig	≤2385 psig
Allowable Value (AV)	≤2397 psig	≤2397 psig

Table 5

Functional Unit No. 7 – Overtemperature ΔT

Tech Spec Term	Current Tech Spec Values	Proposed Tech Spec Values
K_1	1.185	1.185
K_2	0.0224/˚F	0.0224/˚F
K_3	0.0012/psig	0.001/psig
Z Term	7.31 ΔT span	**7.38 ΔT span**

Table 6

Functional Unit No. 8 – Overpower ΔT

Tech Spec Term	Current Tech Spec Values	Proposed Tech Spec Values
K_4	1.12	1.10
K_5 (for increasing T_{avg}) K_5 (for decreasing T_{avg})	0.02/°F 0.00/°F	0.02/°F 0.00/°F
K_6	0.002/˚F	0.002/˚F
Total Allowance (TA)	4.0% ΔT span	**3.33% ΔT span**
Z Term	2.32% ΔT span	**2.43% ΔT span**

Table 7

Functional Unit No. 1.d – Safety Injection, Pressurizer Pressure -- Low

Tech Spec Term	Current Tech Spec Values	Proposed Tech Spec Values
Total Allowance (TA)	18.75% span	**13.5% span**
Z Term	10.47% span	10.47% span
Sensor Error (S)	1.50% span	1.50% span
Trip Setpoint (TS)	≥1850 psig	≥1850 psig
Allowable Value (AV)	≥1838 psig	≥1838 psig

Additionally, the OTΔT and OPΔT allowable values currently listed in Notes 2 and 4 of Tech Spec Table 2.2-1 are expressed as % ΔT span. These allowable values are dependent on cycle specific K_2, K_3, K_6, and fΔI COLR setpoints. Duke Energy proposes expressing these allowable values using their original instrument span as follows:

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; 1.35% of T_{avg} span for T_{avg} input; 0.6% of pressurizer pressure span for pressurizer pressure input; and 0.6% of ΔI span for ΔI input.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; 1.35% of T_{avg} span for T_{avg} input; and 0.6% of ΔI span for ΔI input.

Note 5 of Tech Spec Table 2.2-1 is revised to reflect a change in sensor error for pressurizer pressure measurements from 1.0% to 0.8% of ΔT span.

Duke Energy also proposes the removal of the high power range high negative neutron flux rate trip, Tech Spec Table 2.2-1 Functional Unit 4. This will eliminate a single point vulnerability of a failed rod control fuse which would result in a rod drop and potentially trigger an automatic reactor trip. This trip function will not be credited in any FSAR Chapter 15 accident after HNP Cycle 22, corresponding with the time in which the Chapter 15 analyses will be transitioned from vendor methods to Duke Energy methods.

HNP Tech Spec Bases will be updated to reflect the change in trip setpoint methodology when performing FSAR Chapter 15 safety analyses using the SCD methodology in DPC-NE-2005-P-A and the application of Tech Spec Table 2.2-1 Notes 7 and 8 to the trip setpoint for low reactor coolant flow.

3.0 TECHNICAL EVALUATION

FSAR Chapter 15 analyses must account for the effect of instrument uncertainty on any given transient. Duke Energy has two methods of accounting for uncertainty in the safety analyses: deterministic, referred to as “non-SCD” method, and statistical, known as the SCD method. The non-SCD method accounts for uncertainty in key parameters by biasing the initial conditions up to the maximum calculated uncertainty in the conservative direction for a given transient. The NRC-approved SCD method, described in DPC-NE-2005-P-A, statistically combines the effects of uncertainty on key parameters which are important in the calculation of statistical DNB limit, referred to as the SDL. The SCD methodology thus utilizes nominal initial conditions for key DNB parameters in safety analysis. The SDL is used in the accident analyses to evaluate whether DNB has occurred. The parameter uncertainties considered in the determination of the SDL are:

1. Reactor Power
2. Core Flow and Core Bypass Flow
3. Core Exit Pressure
4. Core Inlet Temperature
5. Radial Power Distribution (Measurement and Manufacturing / Hot Channel Factors)
6. Axial Peak Magnitude and Location
7. DNB Correlation and Code/Model

The radial power distribution, axial peak magnitude and location, and DNB correlation and code/model uncertainties are not considered in this analysis as they are not typically input to system thermal-hydraulic analyses. In addition, core flow rate, core exit pressure, and core inlet temperature are not directly measured or controlled in the plant. As such, uncertainty in the following four parameters is evaluated instead:

1. Reactor Power
2. RCS Flow Rate
3. Pressurizer Pressure
4. Reactor Average Temperature

The SCD method fully accounts for the initial condition uncertainties in the SDL. Incorporating initial condition uncertainty into any facet of SCD analyses beyond that already included in the SDL would therefore double count uncertainty and produce an overly conservative result. As the initial condition uncertainty may be a component of boundary conditions such as trip setpoints or control systems, these must be evaluated to ensure the analysis is not overly penalizing.

HNP utilizes Westinghouse Letter Report FCQL-355 as the original engineering methodology and operability determination bases for values defined in Tech Spec RTS/ESFAS Trip setpoint tables. There are five listed terms related to uncertainty for each of the trip setpoints in the HNP Tech Specs. These are TA, Z, S, TS, and AV, and are defined as follows:

- TS: Considered a nominal Reactor Trip value setting.
- AV: Accommodates instrument drift assumed between operational tests and the accuracy to which TS can be measured and calibrated.
- TA: Difference (in percent of span) between TS and Safety Analysis Limit [SAL] assumed for Reactor Trip function; e.g., TA = |TS - SAL|. Defined within Tech Spec Equation 2.2-1 [Z + R + S < TA]; where 'R' includes Rack Drift and Calibration Uncertainties.
- 'Z' Term: Statistical summation of analysis errors excluding Sensor and Rack Drift and Calibration Uncertainties.
- 'S' Term: Sensor Drift and Calibration Uncertainties.

The CSA is the total uncertainty in a given instrument channel, also known as the total loop uncertainty. The CSA is calculated using a square root sum of the squares of all random sources of uncertainty, plus non-random sources of uncertainty, such as seismic or environmental allowances.

The SAL (analytical limit) is defined as the uncertainty adjusted reactor trip setpoint assumed by safety analysis when performing FSAR Chapter 15 transient analyses. In order for a given analytical limit to fully account for sources of uncertainty, the TA must be greater than or equal to the calculated CSA for that trip.

The difference between the TA and the CSA is the CM. A large CM will protect analysis assumptions in the event that the CSA increases, but will result in a more penalizing accident analysis result by delaying reactor trip. Any change to the TA or CSA resulting in negative margin would invalidate the safety analysis limit assumed in the safety analyses. For this request, the total allowance was set such that a CM of approximately +1% instrument span is maintained. This is judged adequate to protect the safety analyses against small changes in the CSA while not producing overly restrictive safety analysis limits. Figure 1 provides a visualization of the terms previously discussed in this section.

The method of calculating the parameter uncertainties and the setpoints remains unchanged. The methodologies for calculating the as-found tolerances and as-left tolerances about the TS or more conservative actual field setpoint are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints," which is incorporated by reference into the FSAR. The actual field setpoint and the associated as-found and as-left tolerances are specified in PLP-106, "Technical Specification Equipment List Program," the applicable section of which is incorporated by reference into the FSAR. Changes to the setpoints are primarily due to updated component uncertainty values and harvesting excess calculational margin in the setpoint total allowance.

3.1 <u>RTS Low Reactor Coolant System Flow Trip</u>

The Reactor Coolant Low Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

The current Tech Spec minimum RCS flow is 293,540 gpm (Tech Spec 3.2.5.c). The associated RTS low RCS flow trip setpoint is 90.5% flow of the RCS loop flows with a TA of 4.58% span. The TA includes excess CM of 2.5% span to the SAL.

An increase in the low RCS flow trip setpoint from 90.5% flow to 91.7% is proposed. This increase in the trip setpoint is intended to offset some of the impact of a future decrease in HNP minimum Tech Spec RCS flow by producing an earlier reactor trip upon a decrease in reactor coolant system flow rate. The safety analyses will bound operation at the current minimum Tech Spec RCS flow of 293,540 gpm with the revised trip setpoints planned for Cycle 23 included in this LAR.

The total allowance of the proposed low RCS flow trip setpoint is reduced to 3.08% span to capture the excess CM resulting in an updated CM of 1.02% span. A CM of approximately 1% is adequate to protect against unexpected small changes to the CSA without being overly penalizing to the safety analyses. In calculating the S and Z terms, their values changed with the removal of excess margin and the AV term changed with the low RCS flow trip setpoint change.

The setpoint changes are summarized in Table 1 in Section 2.4 and the markup of the affected Tech Spec Table 2.2-1 changes is in Attachment 2.

As a change in the nominal setpoint is proposed, the Tech Spec markups for the Low RCS flow TS refer to Tech Spec Table 2.2-1 Notes 7 and 8 for implementation of Technical Specification Task Force Traveler (TSTF-)493, “Clarify Application of Setpoint Methodology for LSSS Functions” (ADAMS Accession No. ML100060064). The methodologies for calculating the as-found tolerances and as-left tolerances about the TS or more conservative actual field setpoint are specified in EGR-NGGC-0153. The actual field setpoint and the associated as-found and as-left tolerances will be specified in PLP-106.

When surveillance test results exceed these tolerances, specific additional review actions are required on the part of the technicians, operations staff and engineering prior to and following returning the affected channels to service. The intent is to ensure that during testing the instruments and loop perform in accordance with "expected" capability rather than within allowable values, which can include additional margin. These actions are described in EGR-NGGC-0153 and are invoked by existing Tech Spec Table 2.2-1, Notes 7 and 8.

Notes 7 and 8 were added to Tech Spec Table 2.2-1 under License Amendment 139 approved on May 30, 2012 (ADAMS Accession No. ML11356A096) that require verifying both TS setting as-found and as-left values during surveillance testing. In accordance with 10 CFR 50.36, these functions are Limiting Safety System Settings. This LAR proposes to add Notes 7 and 8 to the functional unit Reactor Coolant Flow – Low. The existing Notes 7 and 8 correspond to TSTF-493 option A, Notes 1 and 2, respectively. HNP Tech Spec Bases for Tech Spec Table 2.2-1 states, “adding test requirements ensures that instruments will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis. These notes address NRC staff concerns with Tech Spec Allowable Values. Specifically, calculated Allowable Values may be non-conservative depending upon the evaluation of instrument performance history, and the as-left requirements of the calibration procedures could have an adverse effect on equipment operability. In addition, using Allowable Values as the limiting setting for assessing instrument channel operability may not be fully in compliance with the intent of 10 CFR 50.36, and the existing surveillance requirements would

not provide adequate assurance that instruments will always actuate safety functions at the point assumed in the applicable safety analysis. In the HNP Technical Specifications, the term Trip Setpoint is analogous to Nominal Trip Setpoint (NTSP) in TSTF-493."

Note 7 requires a channel performance evaluation when the as-found setting is outside its as-found tolerance. The performance evaluation verifies that the channel will continue to behave in accordance with safety analysis and instrument performance assumptions in the setpoint methodology. The purpose of this evaluation is to provide confidence in the performance prior to returning the channel to service. If the as-found setting is non-conservative with respect to the AV, the channel is INOPERABLE. If the as-found setting is conservative with respect to the AV but is outside the as-found tolerance band, the channel is OPERABLE but degraded. The degraded channel condition will be further evaluated during performance of the surveillance. This evaluation will consist of resetting the channel setpoint to within the as-left tolerances applicable to the actual setpoint implemented in the surveillance procedures (field setting), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition is entered into the corrective action program for further analysis and trending.

Note 8 requires that the as-left channel setting be reset to a value that is within the as-left tolerances about the TS in Tech Spec Table 2.2-1 or within as-left tolerances about a more conservative actual (field) setpoint. As-left channel settings outside the as-left tolerances of PLP-106 and the surveillance procedures cause the channel to be INOPERABLE.

A tolerance is necessary because no device perfectly measures the process. Additionally, it is not possible to read and adjust a setting to an absolute value due to the readability and/or accuracy of the test instruments or the ability to adjust potentiometers. The as-left tolerance is considered in the setpoint calculation. Failure to set the actual plant trip setpoint to within the as-left tolerances of the NTSP or within as-left tolerances of a more conservative actual field setpoint would invalidate the assumptions in the setpoint calculation, because any subsequent instrument drift would not start from the expected as-left setpoint. The determination will consider whether the instrument is degraded or is capable of being reset and performing its specified safety function. If the channel is determined to be functioning as required (i.e., the channel can be adjusted to within the as-left tolerance and is determined to be functioning normally based on the determination performed prior to returning the channel to service), then the channel is OPERABLE and can be restored to service. If the as-left instrument setting cannot be returned to a setting within the prescribed as-left tolerance band, the instrument would be declared INOPERABLE.

Markups indicating the addition of the Low RCS flow trip to the Tech Spec Bases section description of TSTF-493 are provided in Attachment 3.

3.2 RTS High Power Range Neutron Flux Trip

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a high and low range trip setting. The low setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the high setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The current high neutron flux trip setpoint of 108% RTP, with a TA of 5.83% span, is based on a 5% NI system component uncertainty. According to vendor estimates and experience, this uncertainty encompasses the reactor vessel downcomer water density and radial power redistribution effects. The HNP safety analysis methodology in DPC-NE-3009-P-A, "FSAR/UFSAR Chapter 15 Transient Analysis Methodology," however, as approved by the NRC for use by HNP in Facility Operating License Amendment No. 164 (ADAMS Accession Package No. ML18060A404), stipulates that safety analysis explicitly model effects such as downcomer attenuation, rod shadow, and power tilt. Thus, the 5% NI system component uncertainty is overly conservative. Based on an evaluation, a NI system component uncertainty of 3.2% is determined to be adequate. Setpoint analysis yielded a TA of 4.58% span with the CM maintained at 1.12% span. A CM of approximately 1% is adequate to protect against unexpected small changes to the CSA without being overly penalizing to the safety analyses. Changes to the Z and AV terms are due to the change in the NI system component uncertainty.

The setpoint changes are summarized in Table 2 as found in Section 2.4 and the markup of the affected Tech Spec Table 2.2-1 changes is in Attachment 2.

3.3 Pressurizer Pressure Trips

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a high and low pressure trip, thus limiting the pressure range in which reactor operation is permitted. The low setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

The high setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the RCS against system overpressure.

(a) RTS Low and High Pressurizer Pressure Trip

The current pressurizer low pressure TS of 1960 psig with a TA of 5.0% span includes excess CM of 1.84% span. Likewise, the current pressurizer high pressure trip setpoint of 2385 psig with a TA of 7.5% span includes excess CM of 4.34% span.

The TA for both the high and low pressurizer pressure trip setpoints are reduced to 4.625% span to capture excess margin resulting in an updated CM of 1.46% span. A CM of approximately 1% is adequate to protect against unexpected small changes to the CSA without being overly penalizing to the safety analyses.

The setpoint changes are summarized in Tables 3 and 4 in Section 2.4 and the markup of the affected Tech Spec Table 2.2-1 changes is in Attachment 2.

(b) ESFAS Low Pressurizer Pressure Safety Injection Trip

The current safety injection trip setpoint of 1850 psig with a TA of 18.75% span includes excess CM of 6.64% span.

The TA for the safety injection trip setpoint is reduced from 18.75 to 13.5% span to capture excess margin, resulting in an updated CM of 1.39% span. A CM of approximately 1% is adequate to protect against unexpected small changes to the CSA without being overly penalizing to the safety analyses.

The setpoint changes are summarized in Table 7 of Section 2.4 and the markup of Tech Spec Table 3.3-4 changes is in Attachment 2.

3.4 Reactor Average Temperature

(a) RTS Overtemperature ΔT Trip

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transport to and response time of the temperature detectors (about 4 seconds), and pressure is within the range between the pressurizer high and low pressure trips. The setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for transport to and response time of the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in the COLR. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Tech Spec Table 2.2-1.

The proposed reduction of setpoint K_3 from 0.0012/psig to 0.001/psig necessitates a change in both the Tech Spec ΔT span for pressurizer pressure input from 0.4% to 0.3% ΔT span (equal to 0.06% of pressurizer pressure span) reflected in the proposed change to Tech Spec Table 2.2-1 Note 2 and the ΔT span for pressurizer pressure measurements from 1.0% to 0.8% ΔT span, the latter of which is reflected in the proposed change to Tech Spec Table 2.2-1, Note 5.

EPT-156, "Reactor Coolant System (RCS) ΔT Scaling At 100% Reactor Power," is performed quarterly to protect the ΔT trip scaling following normalization. The tolerance for ΔT indication is -1.0 to 3.0 % ΔT error where % ΔT error is calculated as follows:

$$\Delta\mathrm{T}_{error} = \frac{\Delta\mathrm{T}_{indicated} - \Delta\mathrm{T}_0}{\Delta\mathrm{T}_0} * 100\%$$

Based on recent performances of EPT-156, typical indicated 100% power ΔT values range from 63.0 to 65.0 ˚F, meeting the acceptance criteria. Assuming a high ΔT_0 of 70 ˚F to protect against future changes, the negative tolerance can be converted from % ΔT error to ˚F error as -1 % ΔT error = 70 * 0.01 = -0.7 ˚F. The positive tolerance is not considered in safety analysis as a high indicated ΔT results in an earlier reactor trip. The adjustment in ΔT tolerance results in a need to increase the corresponding bias term in the OTΔT and OPΔT trip uncertainty calculations from 0.6 to 0.7˚F. This adjustment results in an increase in the Z term from 7.31% ΔT span to 7.38% ΔT span.

The setpoint changes are summarized in Table 5 in Section 2.4 and the markup of the affected Tech Spec Table 2.2-1 changes is in Attachment 2.

(b) RTS Overpower ΔT Trip

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the high neutron flux trip. The setpoint is

automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for transport to and response time of the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded, and (3) axial power distribution.

The proposed change of setpoint K_4 in the COLR from 1.12 to 1.10 with an analytical limit of 1.15 results in a TA of 3.33% ΔT span. As discussed above, the adjustment in ΔT tolerance results in a need to increase the corresponding bias term in the OTΔT and OPΔT trip uncertainty calculations from 0.6 to 0.7˚F. This in turn results in an increase in the Z term from 2.32% ΔT span to 2.43% ΔT span.

The setpoint changes are summarized in Table 6 in Section 2.4 and the markup of the affected Tech Spec Table 2.2-1 changes is in Attachment 2.

(c) RTS ΔT Allowable Values

The OTΔT and OPΔT allowable values are currently listed in Notes 2 and 4 of Tech Spec Table 2.2-1. The allowable values are the rack uncertainty terms for the inputs to the trips and are listed as a percentage of ΔT span. The conversion factors from pressurizer pressure span, T_{avg} span, or ΔI span (the inputs to OTΔT and OPΔT) to ΔT span include the K_2, K_3, K_6, and fΔI setpoints. As a result, a change in any of these setpoints in the COLR also changes the Tech Spec allowable value despite the rack uncertainty terms remaining the same. As this poses a significant barrier to changing these COLR setpoints, Notes 2 and 4 should be specified in original instrument span instead of ΔT span. As such, the allowable values are as follows, neglecting the conversion to ΔT span:

- T_{avg} input changes from 2% ΔT span to 1.35% T_{avg} span
- Pressurizer pressure span input changes from 0.4% ΔT span to 0.6% Pressurizer pressure span
- ΔI span input changes from 0.7% ΔT span to 0.6% ΔI span

As each ΔT trip uses the same instrument channels, the rack uncertainty and therefore allowable values are equal, neglecting conversion to ΔT span. The allowable value for ΔI is added to Tech Spec Table 2.2-1, Note 4, to support implementation of an f_2ΔI function. As the allowable value is in terms of ΔI span, it is independent of the selected slope. Tech Spec markups have been added to Attachment 2 to change the units for the OTΔT and OPΔT allowable values.

3.5 RTS High Power Range Negative Neutron Flux Rate Trip

The proposed removal of the high power range negative neutron flux rate trip will eliminate a single point vulnerability of a failed rod control fuse which would result in a rod drop and potentially trigger an automatic reactor trip. While the negative flux rate trip is currently credited in the vendor-provided dropped rod analysis of record (HNP FSAR Section 15.4.3.1), this trip function will no longer be credited in any FSAR Chapter 15 accident after HNP Cycle 22, following the replacement of the current FSAR Chapter 15 dropped rod analysis with that performed using the Duke Energy methodologies documented in DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis," and DPC-NE-3009-P-A, both of which were approved for HNP use by the NRC in Amendment No. 164.

The markups of Tech Spec Table 2.2-1 changes are in Attachment 2.

3.6 Update Technical Specification Bases Section 2.2.1

The Tech Spec Bases are being updated to reflect the TS methodology when performing FSAR Chapter 15 safety analyses using the SCD methodology and the application of Tech Spec Table 2.2-1 Notes 7 and 8 to the trip setpoint for low reactor coolant flow.

The SCD methodology contained within DPC-NE-2005-P-A statistically combines the effects of initial condition uncertainty on DNB to determine the DNB ratio SDL. The SDL is set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNB ratio is at the SDL, accounting for initial condition uncertainty. Thus, the setpoint methodology separates the initial condition uncertainty from the setpoint drift and trip module uncertainty.

The markup to the Tech Spec Bases is provided for information purposes in Attachment 3.

3.7 Conclusion

It is requested that the TA for several reactor trips be reduced to take advantage of excess margin in those allowances. The proposed changes to the TAs, and any associated changes to the Z term and AV have been provided in Attachment 2. It was also determined that for reactor power, pressurizer pressure, reactor average temperature, and RCS flow, a significant portion of the uncertainty incorporated into the analysis limit is accounted for in the calculation of the SDL. As such, SCD analyses may assume a TS less conservative than the analytical limit assumed for non-SCD analyses.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.34(b) requires a FSAR to be submitted as part of an application for an operating license. In particular, 10 CFR 50.34(b)(2) requires the FSAR to contain “evaluations required to show that safety functions will be accomplished” for plant structures, systems, and components. The models and methods as approved by the NRC in DPC-NE-2005-P-A, DPC-NE-3008-P-A, and DPC-NE-3009-P-A will be used to perform safety analyses in Chapter 15 of the HNP FSAR.

According to 10 CFR 50.36, licensees are required to include Tech Specs “derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34.” The analyses were performed using models and methods approved by the NRC, and may therefore be used to set Tech Spec limits or verify that Tech Spec limits are met for a given cycle.

HNP was licensed to the standard of the general design criteria (GDC) included in Appendix A to 10 CFR Part 50. The analyses were performed using models and methods previously approved by the NRC (DPC-NE-2005-P-A, DPC-NE-3008-P-A, and DPC-NE-3009-P-A), and therefore allowed for use in evaluating compliance with several GDC, including:

- Criterion 10, which requires the reactor core and associated coolant, control and protection systems to be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.
- Criterion 15, which requires the reactor coolant system and associated systems to be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.
- Criterion 20, which requires protection systems to 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.
- Criterion 28, which requires reactivity control systems to be designed with appropriate limits on the potential amount and rate of reactivity releases to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

The proposed changes will not affect HNP's conformance to the GDC listed above.

The methodologies for calculating the as-found tolerances and as-left tolerances about the TS or more conservative actual field setpoint are specified in EGR-NGGC-0153. This procedure, in its entirety, implements, in part, the Harris commitment to Regulatory Guide 1.105, "Instrument Setpoints", Revision 1 (ADAMS Accession No. ML13064A112), as described in the Harris FSAR, Section 1.8. This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to ensuring that the instrument setpoints in systems important to safety initially are within and remain within the specified limits.

4.2 Precedent

HNP has previously implemented setpoint-related Tech Spec changes subject to TSTF-493 applicability. License Amendment No. 139 was approved on May 30, 2012, for changes to Table 2.2-1 functions 2, 3 and 4 (power range neutron flux) values in support of the Leading Edge Flow Meter Measurement Uncertainty Recapture Power Uprate (ADAMS Accession No. ML11356A096). This amendment added the Tech Spec Table 2.2-1 footnotes 7 and 8, to which extended applicability is now proposed within this LAR. The associated Tech Spec Bases, Section 2.2.1, was also updated to describe the plant methodology for compliance with TSTF-493 requirements for applicable functions. Additionally, License Amendment No. 146 was approved on June 30, 2015, that modified Tech Spec Table 3.3-4, revising the Functional Unit 9.a, "Loss-of-Offsite Power 6.9 Kilovolt Emergency Bus Undervoltage – Primary," instrumentation trip setpoint and associated allowable value, and adding two notes regarding channel setpoint surveillance (ADAMS Accession No. ML15163A056).

4.3 No Significant Hazards Consideration Determination

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), proposes a license amendment request (LAR) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP) Technical Specifications (Tech Spec). The proposed license amendment modifies Tech Spec Table 2.2-1, “Reactor Trip System Instrumentation Trip Setpoints,” and Tech Spec Table 3.3-4, “Engineered Safety Features Actuation System Instrumentation Trip Setpoints,” to optimize safety analysis margin in the Final Safety Analysis Report (FSAR) Chapter 15 transient analyses.

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below.

1. *Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?*

The Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) provide plant protection and are part of the accident mitigation response. The RTS and ESFAS functions do not themselves act as a precursor or an initiator for any transient or design basis accident. Therefore, the proposed change does not significantly increase the probability of any accident previously evaluated.

The proposed change does not alter the design assumptions, conditions, or configuration of the facility. The structural and functional integrity of the RTS and ESFAS, or any other plant system, is unaffected. The proposed change does not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems, and components from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of any accident previously evaluated. The proposed changes are consistent with safety analysis assumptions and resultant consequences. The methods used to calculate the parameter uncertainties and the setpoints remain unchanged. Changes to the setpoints are primarily due to updated component uncertainty values and harvesting excess calculational margin (CM) in the setpoint total allowance (TA).

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?*

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. The proposed changes will not affect the normal method of plant operation. No performance requirements will be affected or eliminated. The proposed changes will not result in physical alteration to any plant system nor will there be any change in the method by which any safety-related plant system performs its safety function. The proposed changes do not alter assumptions made in the safety analysis but ensures that the instruments behave as assumed in the accident analysis. The proposed change is consistent with the safety analysis assumptions. The methods used to calculate the parameter uncertainties and the setpoints remain unchanged. Changes to the setpoints are primarily due to updated component uncertainty values and harvesting excess CM in the setpoint TA.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *Does the proposed change involve a significant reduction in a margin of safety?*

The proposed changes do not alter the manner in which safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not negatively impacted by these changes. Redundant RTS and ESFAS trains are maintained, and diversity with regard to the signals that provide reactor trip and engineered safety features actuation is also maintained. All signals credited as primary or secondary, and all operator actions credited in the accident analyses will remain the same. The proposed changes will not result in plant operation in a configuration outside the design basis. The methods used to calculate the parameter uncertainties and the setpoints remain unchanged. Changes to the setpoints are primarily due to updated component uncertainty values and harvesting excess CM in the setpoint TA.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92, and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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.

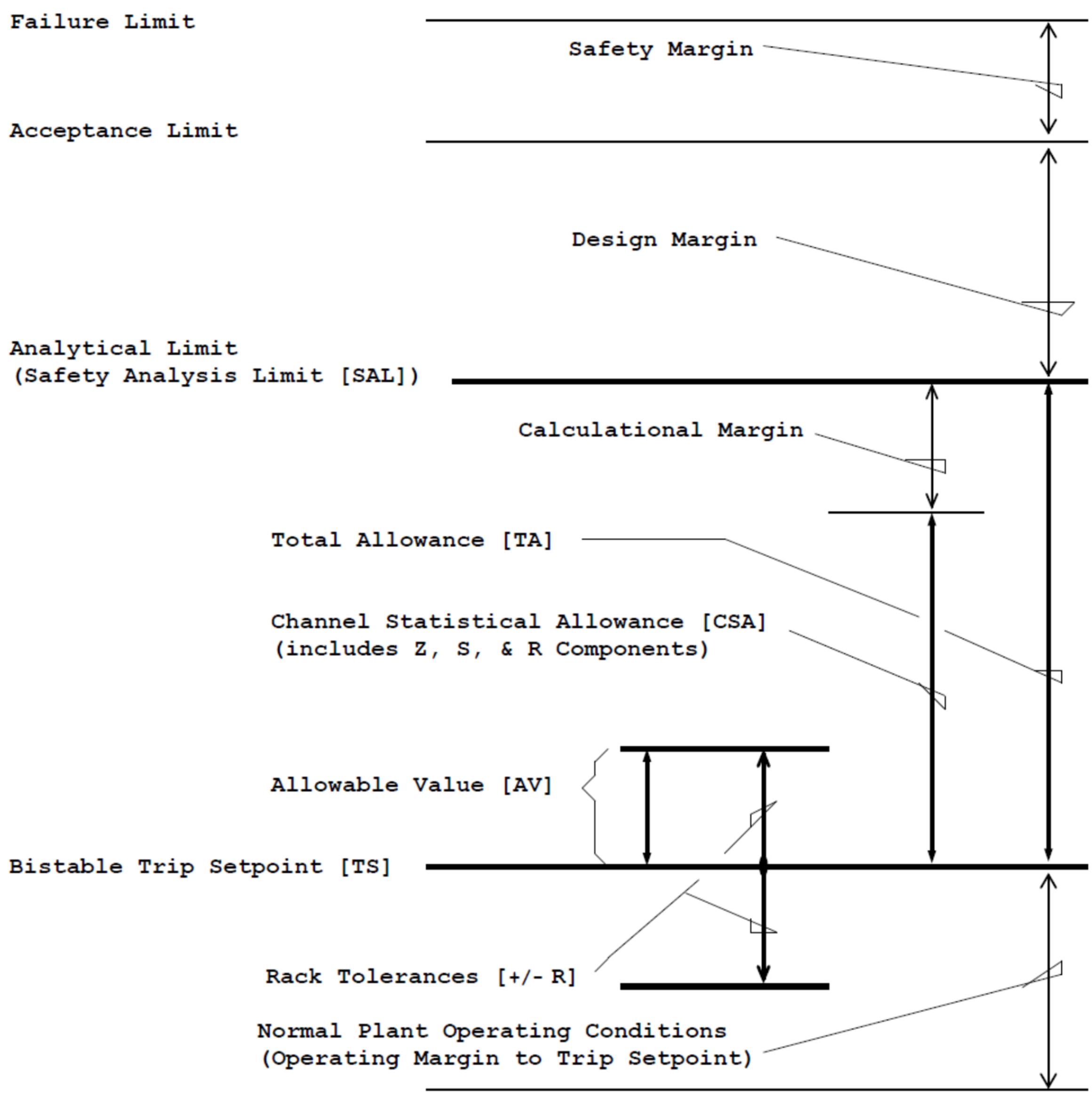
5.0 ENVIRONMENTAL CONSIDERATION

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 in 10 CFR Part 20, and 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 effluents 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 need be prepared in connection with the proposed amendment.

Figure 1: Operating Conditions, Uncertainties, and Margins Relative to Safety Analysis Limit, Allowable Value, and Trip Setpoint



RA-18-0072

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

5 PAGES PLUS COVER

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1.	Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2.	Power Range, Neutron Flux					
	a. High Setpoint	~~5.83~~ 4.58	~~4.56~~ 3.25	0	≤ 108% of RTP** See NOTES 7, 8	~~≤ 109.5% of RTP**~~ ≤ 109.6% of RTP**
	b. Low Setpoint	7.83	4.56	0	≤ 25% of RTP** See NOTES 7, 8	≤ 26.8% of RTP**
3.	Power Range, Neutron Flux, High Positive Rate	2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds See NOTES 7, 8	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4.	Not Used ~~Power Range, Neutron Flux, High Negative Rate~~	~~2.33~~ N/A	~~0.83~~ N/A	~~0~~ N/A	N/A ~~≤ 5% of RTP** with a time constant ≥ 2 seconds~~ ~~See NOTES 7, 8~~	N/A ~~≤ 6.3% of RTP** with a time constant ≥ 2 seconds~~
5.	Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6.	Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4 x 10^5 cps
7.	Overtemperature ΔT	9.0	~~7.31~~ 7.38	Note 5	See Note 1	See Note 2
8.	Overpower ΔT	~~4.0~~ 3.33	~~2.32~~ 2.43	1.3	See Note 3	See Note 4
9.	Pressurizer Pressure-Low	~~5.0~~ 4.625	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10.	Pressurizer Pressure-High	~~7.5~~ 4.625	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11.	Pressurizer Water Level-High	8.0	3.42	1.75	≤ 87% of instrument span See NOTES 7, 8	≤ 88.5% of instrument span

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
12. Reactor Coolant Flow-Low	~~4.58~~ 3.08	~~1.98~~ 1.58	~~0.6~~ 0.49	~~≥ 90.5% of loop full indicated flow~~ ≥ 91.7% of loop full indicated flow See NOTES 7, 8	~~≥ 89.5% of loop full indicated flow~~ ≥ 90.6% of loop full indicated flow
13. Steam Generator Water Level Low-Low	25.0	17.45	2.0	≥ 25.0% of narrow range instrument span	≥ 23.5% of narrow range instrument span
14. Steam Generator Water Level - Low	8.9	5.95	2.0	≥ 25.0% of narrow range instrument span	≥ 24.05% of narrow range instrument span
Coincident With Steam/Feedwater Flow Mismatch	20.0	3.01	Note 6	≤ 40% of full steam flow at RTP**	≤ 43.1% of full steam flow at RTP**
15. Undervoltage - Reactor Coolant Pumps	14.0	1.3	0.0	≥ 5148 volts	≥ 4920 volts
16. Underfrequency - Reactor Coolant Pumps	5.0	3.0	0.0	≥ 57.5 Hz	≥ 57.3 Hz
17. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	≥ 1000 psig	≥ 950 psig
b. Turbine Throttle Valve Closure	N.A.	N.A.	N.A.	≥ 1% open	≥ 1% open
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

The values denoted with [*] are specified in the COLR.

NOTE 1: (Continued)

T	=	Average temperature, °F;
$\frac{1}{1+\tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = [*] s;
T'	=	Reference T_{avg} at RATED THERMAL POWER (≤ [*]°F);
K_3	=	[*]/psig;
P	=	Pressurizer pressure, psig;
P'	=	[*] psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between [*]% and [*]%, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds [*]%, the ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds [*]%, the ΔT Trip Setpoint shall be automatically reduced by [*]% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; ~~2.0% of ΔT span for T_{avg} input; 0.4% of ΔT span for pressurizer pressure input; and 0.7% of ΔT span for ΔI input.~~

1.35% of T_{avg} span for T_{avg} input; 0.6% of pressurizer pressure span for pressurizer pressure input; and 0.6% of ΔI span for ΔI input.

The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input; 1.35% of T_{avg} span for T_{avg} input; and 0.6% of ΔI span for ΔI input.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

The values denoted with [*] are specified in the COLR.

NOTE 3: (Continued)

K_6	=	[*]/°F for T > T" and K_6 = [*] for T ≤ T",
T	=	As defined in Note 1,
T"	=	Reference T_{avg} at RATED THERMAL POWER (≤ [*]°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	[*].

NOTE 4: ~~The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input and 0.2% of ΔT span for T_{avg} input.~~

NOTE 5: The sensor error is: 1.3% of ΔT span for ΔT/T_{avg} temperature measurements; and ~~1.0%~~ 0.8% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

NOTE 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 2.2-1 (Nominal Trip Setpoint (NTSP)) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints." The as-found and as-left tolerances are specified in PLP-106.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-1	3.64	0.71	1.5	≤ 3.0 psig	≤ 3.6 psig
d. Pressurizer Pressure--Low	~~18.75~~ 13.5	10.47	1.5	≥ 1850 psig	≥ 1838 psig
e. Steam Line Pressure--Low	4.52	0.71	2.0	≥ 601 psig	≥ 581.5 psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure--High-3	3.64	0.71	1.5	≤ 10.0 psig	≤ 11.0 psig

RA-18-0072

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES
(FOR INFORMATION ONLY)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

5 PAGES PLUS COVER

No changes to this page.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel, pressurizer, and the RCS piping, pumps, valves and fittings are designed to Section III, Division I of the ASME Code for Nuclear Power Plants, which permits a maximum transient pressure of 110% to 125% of design pressure (2485 psig) depending on component. The Safety Limit of 2735 psig (110% of design pressure) is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as measured" Setpoint is within the band allowed for calibration accuracy. For example, if a bistable has a trip setpoint of 100%, a span of 125%, and a calibration accuracy of 0.5% of span, then the bistable is considered to be adjusted to the trip setpoint as long as the "as measured" value for the bistable is ≤ 100.62%.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor and an increased rack drift factor, and provides a threshold value for determination of OPERABILITY.

The methodology to derive the Trip Setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

ADD: <INSERT>

Reactor Trip System Instrumentation Setpoints and TSTF-493

This section applies only to the Functional Units to which Notes 7 and 8 in the Trip Setpoint Column are applicable. Those Functional Units have revisions in accordance with Technical Specification Task Force Traveler 493 (TSTF-493). "Clarify Application of Setpoint Methodology for LSSS Functions." Those Functional Units are limited to

- Power Range, Neutron Flux High Setpoint
- Power Range, Neutron Flux Low Setpoint
- Power Range, Neutron Flux High Positive Rate
- ~~Power Range, Neutron Flux High Negative Rate~~
- Pressurizer Water Level – High Setpoint

ADD:
• Reactor Coolant Flow - Low Setpoint

Notes 7 and 8 have been added to Table 2.2-1 that require verifying both trip setpoint setting as-found and as-left values during surveillance testing. In accordance with 10 CFR 50.36, these functions are Limiting Safety System Settings. Adding test requirements ensures that instruments will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis. These notes address NRC staff concerns with Technical Specification Allowable Values. Specifically, calculated Allowable Values may be non-conservative depending upon the evaluation of instrument performance history, and the as-left requirements of the calibration procedures could have an adverse effect on equipment

<INSERT>

The statistical core design (SCD) methodology presented in DPC-NE-2005 statistically combines the effects of initial condition uncertainty and other uncertainties on DNB to determine a DNBR statistical design limit (SDL). The SDL is set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the calculated minimum DNBR is at the DNBR limit, accounting for uncertainty. The initial condition uncertainty contained in the SDL comprises some or all of the channel uncertainty for some reactor trip functions. The Total Allowances given in Table 2.2-1 satisfy Equation 2.2-1, accounting for all channel uncertainty. FSAR Chapter 15 analyses performed using the SCD methodology account for some or all of the channel uncertainty in the SDL. As such, the RPS trip setpoints used in SCD analyses may assume a smaller Total Allowance which satisfies Equation 2.2-1 after removal of uncertainty terms accounted for in the SDL.

No changes to this page.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

operability. In addition, using Allowable Values as the limiting setting for assessing instrument channel operability may not be fully in compliance with the intent of 10 CFR 50.36, and the existing surveillance requirements would not provide adequate assurance that instruments will always actuate safety functions at the point assumed in the applicable safety analysis. In the Harris Technical Specifications, the term Trip Setpoint is analogous to Nominal Trip Setpoint (NTSP) in TSTF-493.

Note 7 requires a channel performance evaluation when the as-found setting is outside its as-found tolerance. The performance evaluation verifies that the channel will continue to behave in accordance with safety analysis and instrument performance assumptions in the setpoint methodology. The purpose of this evaluation is to provide confidence in the performance prior to returning the channel to service. If the as-found setting is non-conservative with respect to the Allowable Value, the channel is INOPERABLE. If the as-found setting is conservative with respect to the Allowable Value but is outside the as-found tolerance band, the channel is OPERABLE but degraded. The degraded channel condition will be further evaluated during performance of the surveillance. This evaluation will consist of resetting the channel setpoint to within the as-left tolerances applicable to the actual setpoint implemented in the surveillance procedures (field setting), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition is entered into the corrective action program for further analysis and trending.

Note 8 requires that the as-left channel setting be reset to a value that is within the as-left tolerances about the Trip Setpoint in Table 2.2-1 or within as-left tolerances about a more conservative actual (field) setpoint. As-left channel settings outside the as-left tolerances of PLP-106 and the surveillance procedures cause the channel to be INOPERABLE.

A tolerance is necessary because no device perfectly measures the process. Additionally, it is not possible to read and adjust a setting to an absolute value due to the readability and/or accuracy of the test instruments or the ability to adjust potentiometers. The as-left tolerance is considered in the setpoint calculation. Failure to set the actual plant trip setpoint to within as-left the tolerances of the NTSP or within as-left tolerances of a more conservative actual field setpoint would invalidate the assumptions in the setpoint calculation, because any subsequent instrument drift would not start from the expected as-left setpoint. The determination will consider whether the instrument is degraded or is capable of being reset and performing its specified safety function. If the channel is determined to be functioning as required (i.e., the channel can be adjusted to within the as-left tolerance and is determined to be functioning normally based on the determination performed prior to returning the channel to service), then the channel is OPERABLE and can be restored to service. If the as-left instrument setting cannot be returned to a setting within the prescribed as-left tolerance band, the instrument would be declared INOPERABLE.

The methodologies for calculating the as-found tolerances and as-left tolerances about the Trip Setpoint or more conservative actual field setpoint are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints," which is incorporated by reference into the FSAR. The actual field setpoint and the associated as-found and as-left tolerances are specified in PLP-106, the applicable section of which is incorporated by reference into the FSAR.

Limiting Trip Setpoint (LTSP) is generic terminology for the setpoint value calculated by means of the setpoint methodology documented in EGR-NGGC-0153. HNP uses the plant-specific term Nominal Trip Setpoint (NTSP) in place of the generic term LTSP. The NTSP is the LTSP with

No changes to this page.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

margin added, and is always equal to or more conservative than the LTSP. The NTSP may use a setting value that is more conservative than the LTSP, but for Technical Specification compliance with 10 CFR 50.36, the plant-specific setpoint term NTSP is cited in Note 8.

The NTSP meets the definition of a Limiting Safety System Setting per 10 CFR 50.36 and is a predetermined setting for a protective channel chosen to ensure that automatic protective actions will prevent exceeding Safety Limits during normal operation and design basis anticipated operational occurrences, and assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Allowable Value is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is within the as-found tolerance and is conservative with respect to the Allowable Value during a CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual NTSP setting ensures that a Safety Limit is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance band, in accordance with uncertainty assumptions stated in the setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

Field setting is the term used for the actual setpoint implemented in the plant surveillance procedures, where margin has been added to the calculated field setting. The as-found and as-left tolerances apply to the field settings implemented in the surveillance procedures to confirm channel performance. A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the instrument operability must be verified based on the field setting and not the NTSP.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.