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919.362.2502

10 CFR 50.90

December 2, 2016
Serial: HNP-16-066

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 Relocate Technical Specification Cycle-Specific Parameters to the Core Operating Limits Report

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment would relocate selected figures and values from the TS to the Core Operating Limits Report (COLR), including TS Figure 2.1-1 cited in TS 2.1.1, "Safety Limits – Reactor Core," selected portions of Table 2.2-1 Note 1 on Overtemperature ΔT (OTΔT) and Note 3 on Overpower ΔT (OPΔT) cited in TS 2.2.1, "Limiting Safety System Settings – Reactor Trip System Instrumentation Setpoints," and Departure from Nucleate Boiling (DNB) values cited in TS 3.2.5, "DNB Parameters." These changes are consistent with the intent of Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-339, "Relocate TS Parameters to COLR." As a result of the above changes, TS 6.9.1.6, "Core Operating Limits Report," and associated TS Bases will be revised to reflect the above proposed changes.

Furthermore, editorial changes are proposed throughout the HNP TS to remove all reference to plant procedure PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report," as it pertains to the COLR. Going forward, the COLR will no longer be contained in PLP-106, aligning with the Duke Energy fleet procedure governing core design and design deliverable documents.

This amendment request also proposes to delete requirements from HNP Administrative Control TS 6.7, Safety Limit Violation, that duplicate requirements found in regulation 10 CFR 50.36. The proposed changes are consistent with the intent of NRC-approved TSTF-5, "Delete Safety Limit Violation Notification Requirements," Revision 1.

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 determined 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 TS changes. Attachment 3 provides a copy of the TS Bases markup based on the proposed changes.

Approval of the proposed license amendment is requested by December 1, 2017, to support the planned refueling outage scheduled for April 2018. The amendment shall be implemented within 90 days following approval.

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 Jeff Robertson, HNP Regulatory Affairs Manager, at (919) 362-3137.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on December 2, 2016.

Sincerely,

Tanya M. Hamilton

Tanya M. Hamilton

Attachments:
1. Evaluation of the Proposed Change
2. Proposed Technical Specification Changes
3. Proposed Technical Specification Bases Changes

cc: Mr. C. Jones, NRC Sr. Resident Inspector, HNP
Mr. W. L. Cox, III, Section Chief N.C. DHSR
Ms. M. Barillas, NRC Project Manager, HNP
Ms. C. Haney, NRC Regional Administrator, Region II

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HNP-16-066

ATTACHMENT 1

EVALUATION OF THE PROPOSED CHANGE

SHEARON HARRIS NUCLEAR POWER PLANT / UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

Evaluation of the Proposed Change

Subject: License Amendment Request to Relocate Technical Specification Cycle-Specific Parameters to the Core Operating Limits Report

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 (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The proposed license amendment would relocate selected figures and values from the TS to the Core Operating Limits Report (COLR), including TS Figure 2.1-1 cited in TS 2.1.1, "Safety Limits – Reactor Core," selected portions of Table 2.2-1 Note 1 on Overtemperature ΔT (OTΔT) and Note 3 on Overpower ΔT (OPΔT) cited in TS 2.2.1, "Limiting Safety System Settings – Reactor Trip System Instrumentation Setpoints," and Departure from Nucleate Boiling (DNB) values cited in TS 3.2.5, "DNB Parameters." These changes are consistent with the intent of Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-339, "Relocate TS Parameters to COLR." As a result of the above changes, TS 6.9.1.6, "Core Operating Limits Report," and associated TS Bases will be revised to reflect the above proposed changes.

Furthermore, editorial changes are proposed throughout the HNP TS to remove all reference to plant procedure PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report," as it pertains to the COLR. Going forward, the COLR will no longer be contained in PLP-106, aligning with the Duke Energy fleet procedure governing core design and design deliverable documents.

This amendment request also proposes to delete requirements from HNP Administrative Control TS 6.7, Safety Limit Violation, that duplicate requirements found in regulation 10 CFR 50.36. The proposed changes are consistent with the intent of NRC-approved TSTF-5, "Delete Safety Limit Violation Notification Requirements," Revision 1.

2.0 DETAILED DESCRIPTION

Duke Energy proposes changes to the HNP TS as follows:

- TS Figure 2.1-1 is deleted and relocated to the COLR.

- TS Safety Limit 2.1.1 is revised to read:

    2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits specified in the COLR; and the following Safety Limits shall not be exceeded:

    a. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.141 for the HTP DNB correlation.

b. The peak centerline temperature shall be maintained < [(2790 – 17.9 x P – 3.2 x B) x 1.8 + 32] ˚F where P is the maximum weight percent of Gadolinia (%) and B is the maximum pin burnup (GWD/MTU).

APPLICABILITY: MODES 1 and 2.

ACTION:

If Safety Limit 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

- The overpower and overtemperature ΔT trip constants and time constants in Table 2.1-1 of TS 2.2.1 are deleted and relocated to the COLR.
- TS 3.2.5 DNB-related parameters are deleted and relocated to the COLR.
- Administrative Control TS 6.9.1.6 is revised to reflect the above changes.
- Editorial changes throughout TS to remove reference to HNP plant procedure PLP-106 as it pertains to the COLR.
- Administrative Control TS 6.7.1 is deleted to remove duplicative reporting and restart requirements to those already contained in the regulations.

Relocation of cycle-specific parameters from the TS to the COLR, a licensee-controlled document subject to the requirements of TS 6.9.1.6 and the provisions of 10 CFR 50.59, would afford Duke Energy the flexibility to revise cycle-specific parameters that are in accordance with NRC-approved methodologies without the need for license amendments. The COLR is required to be submitted to the NRC for each reload cycle per TS 6.9.1.6, including any mid-cycle revisions or supplements to the NRC, unless otherwise approved by the Commission.

3.0 TECHNICAL EVALUATION

Basis for Proposed Change

NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits From Technical Specifications," dated October 4, 1988, provides guidance to licensees for the removal of cycle-dependent variables from the TS provided that these values are included in a COLR and are determined with NRC-approved methodologies referenced in the TS. Westinghouse Electric Company (Westinghouse) subsequently developed WCAP-14483, "Generic Methodology for Expanding Core Operating Limits Report," describing how cycle-specific parameters may be relocated to the COLR. WCAP-14483 was accepted for referencing by the NRC on January 19, 1999. The Safety Evaluation Report, contained in the January 19, 1999, NRC letter approving WCAP-14483-A, concluded that additional information contained in the TS may be relocated to the COLR.

The limits on the parameters which are removed from the TS and added to the COLR must be developed and justified using NRC-approved methodologies. All accident analyses, performed

in accordance with these methodologies, must meet the applicable NRC-approved limits of the safety analysis. The removal of parameter limits from the TS and their addition to the COLR does not obviate the requirement to operate within these limits. Furthermore, any changes to those limits must be performed in accordance with TS 6.9.1.6.3. If any of the applicable limits of the safety analysis are not met, prior NRC approval of the change is required, as is the case for a license amendment request. For more routine modifications, where NRC-approved methodologies and limits of the safety analysis remain applicable, the potentially burdensome and lengthy process of amending the TS may be avoided. The requested changes are essentially administrative in nature; therefore, the required level of safety will be maintained.

Applicability of TSTF-339

The following requested changes are based upon the NRC-approved Westinghouse Owner's Group (WOG) Technical Specifications Task Force (TSTF)-339, "Relocated TS Parameters to the COLR Consistent with WCAP-14483," Revision 2, and Westinghouse WCAP-14483-A:

1. Revise TS 3.2.5 to relocate the pressurizer pressure, RCS average temperature (T_{avg}), and RCS total flow rate values to the COLR. The minimum limit for total flow based on that used in the reference safety analysis will be retained in the TS.
2. Revise TS Table 2.2-1, Notes 1 and 3, to relocate the overtemperature ΔT and overpower ΔT (K) constant values and dynamic compensation (τ) values, and the breakpoint and slope values for the f(ΔI) penalty function(s) to the COLR.
3. Revise TS 2.1 Safety Limits, and the associated bases, to relocate Figure 2.1-1 to the COLR and replace it with more specific requirements regarding the safety limits (i.e., the fuel DNB design basis and the fuel centerline melt design basis). The NRC-approved methodologies used to derive the parameters in the figure will be referenced in the Reporting Requirements section of the TS.

Duke Energy has reviewed TSTF-339, Revision 2, and concluded that the TS changes as outlined in WCAP-14483-A are applicable to HNP. The above proposed changes are also consistent with Revision 4 of NUREG-1431, "Standard Technical Specifications – Westinghouse Plants" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12100A222), with minor format and structure differences due to HNP having not converted to Improved Technical Specifications (ITS), including the TS 2.1.1 numbering scheme (alphabetical versus numerical) and the use of TS 6.7.1 to identify Safety Limit Violation actions.

Editorial Changes related to PLP-106

The HNP COLR is currently contained within plant procedure PLP-106 as an attachment. To maintain consistency with the Duke Energy fleet procedure governing core design deliverable documents, the COLR will no longer be contained within PLP-106, but will rather be its own entity. As such, all reference to PLP-106 in HNP TS pertaining to the COLR will be deleted. Reference to PLP-106 in TS as it pertains to the Technical Specification Equipment List Program is not affected by this license amendment request, aside from those instances where the current procedure title is given.

The COLR is a licensee-controlled document, subject to the requirements of TS 6.9.1.6 and the provisions of 10 CFR 50.59. It will continue to be submitted to the NRC for each reload cycle, including any mid-cycle revisions or supplements to the NRC, unless otherwise approved by the Commission. Therefore, the deletion from TS of reference PLP-106 as it pertains to the COLR is

an administrative change that aligns with the corresponding wording of NUREG-1431, Volume 1, Revision 4.

Applicability of TSTF-5

The requested deletion of requirements from Administrative Control TS 6.7, Safety Limit Violation, is based upon the NRC-approved TSTF-5, “Delete Safety Limit Violation Notification Requirements,” Revision 1.

Duke Energy has reviewed TSTF-5, Revision 1, and concluded that the intent of the TS changes to be applicable to HNP. The proposed changes delete administrative actions from the TS that duplicate the requirements to report safety limit violations and requirements to preclude restart after a safety limit violation without NRC approval. These requirements are also more restrictive than those already contained in the regulations in that they require the report to be submitted to the NRC within 14 days of the violation. The 10 CFR 50.36 reporting requirements require the licensee to notify the NRC as required by 10 CFR 50.72 and submit a Licensee Event Report to the NRC as required by 10 CFR 50.73. Therefore, appropriate reporting would be made to the NRC in accordance with the regulations in the event a TS safety limit was violated. In addition, 10 CFR 50.36 states that operations must not be resumed until authorized by the Commission. The removal of the duplicate reporting and restart requirements from TS is a simplification of the TS and a reduction in administrative burden to track duplicated requirements. It also aligns with the requirements as presented in NUREG-1431, Volume 1, Revision 4.

4.0 REGULATORY ANALYSIS

4.1 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 (TS). The proposed LAR relocates selected figures and values from the TS to the Core Operating Limits Report (COLR), including TS Figure 2.1-1 cited in TS 2.1.1, “Safety Limits – Reactor Core,” selected portions of Table 2.2-1 Note 1 on Overtemperature ΔT (OTΔT) and Note 3 on Overpower ΔT (OPΔT) cited in TS 2.2.1, “Limiting Safety System Settings – Reactor Trip System Instrumentation Setpoints,” and Departure from Nucleate Boiling (DNB) values cited in TS 3.2.5, “DNB Parameters.” These changes are consistent with the intent of Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-339, “Relocate TS Parameters to COLR.” This LAR also proposes the deletion of duplicative notification, reporting, and restart requirements from the Administrative Controls section of TS, consistent with the intent of NRC-approved TSTF-5, “Delete Safety Limit Violation Notification Requirements.” Furthermore, editorial changes are proposed throughout the HNP TS to remove all reference to plant procedure PLP-106, “Technical Specification Equipment List Program and Core Operating Limits Report,” as it pertains to the COLR. As a result of the above changes, TS 6.9.1.6, “Core Operating Limits Report,” and associated TS Bases will be revised to reflect the above proposed changes.

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 proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes are administrative in nature, facilitate improved content and presentation of Administrative controls, and alter only the format and location of cycle-specific parameter figures and limits from the TS to the COLR. This relocation does not result in the alteration of the design, material, or construction standards that were applicable prior to the change. The proposed changes will not result in modification of any system interface that would increase the likelihood of an accident since these events are independent of the proposed change. The proposed amendment will not change, degrade, or prevent actions, or alter any assumptions previously made in evaluating the radiological consequences of an accident described in the Final Safety Analysis Report (FSAR).

   Therefore, the proposed changes do 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?*

   The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes do not involve any change to the configuration or method of operation of any plant equipment. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes. Also, there will be no change in types or increase in the amounts of any effluents released offsite.

   Therefore, the proposed changes do 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 involve a significant reduction in a margin of safety. Previously-approved methodologies will continue to be used in determination of cycle-specific core operating limits that are present in the COLR. The proposed changes are administrative in nature and will not affect the plant design or system operating parameters. As such, there is no detrimental impact on any equipment design parameter and the plant will continue to be operated within prescribed limits.

   Therefore, the proposed changes do 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.

4.2 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 CONSIDERATION

Duke Energy has concluded that the proposed amendment meets the criteria provided by 10 CFR 51.22(c)(9) for categorical exclusion from the requirement for an Environmental Impact Statement. The proposed amendment does 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.

HNP-16-066

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT / UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits ~~shown in Figure 2.1-1 for 3-loop operation~~.

specified in the COLR; and the following Safety Limits shall not be exceeded:

ADD: INSERT 1

APPLICABILITY: MODES 1 and 2.

ACTION:

ADD: INSERT 2

~~a.~~ ~~Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1.~~

~~b.~~ ~~Operation with less than 3 loops is governed by Specification 3.4.1.1.~~

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig except during hydrostatic testing.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of Specification 6.7.1.

MODES 3, 4, and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The Reactor Trip System Instrumentation and Interlock Setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

INSERT 1

a. The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ 1.141 for the HTP DNB correlation.

b. The peak centerline temperature shall be maintained < [(2790 - 17.9 x P - 3.2 x B) x 1.8 + 32] °F where P is the maximum weight percent of Gadolinia (%) and B is the maximum pin burnup (GWD/MTU).

INSERT 2

If Safety Limit 2.1.1 is violated, restore compliance and be in HOT STANDBY within 1 hour.

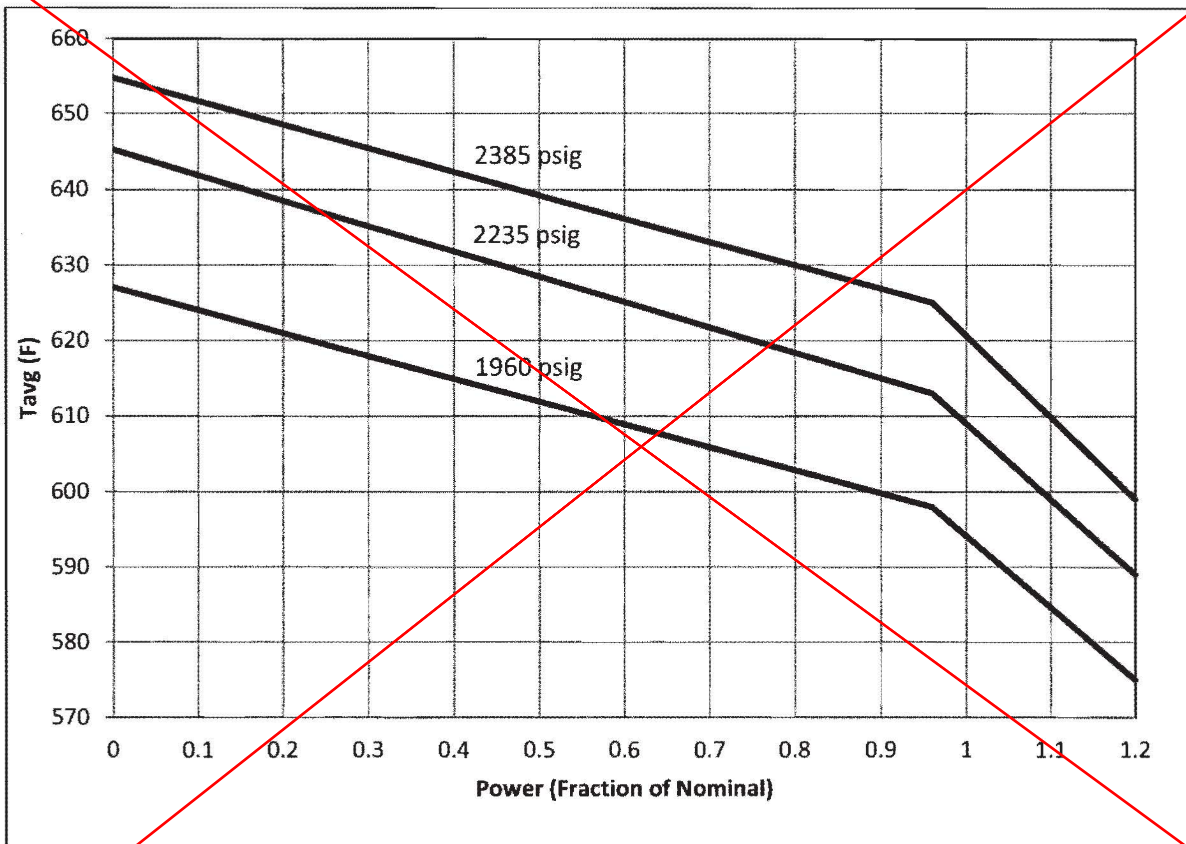

FIGURE 2.1-1
REACTOR CORE SAFETY LIMITS – THREE LOOPS IN OPERATION
WITH MEASURED RCS FLOW > [293,540 GPM X (1.0 + C_1)]

ADD:
"This figure is deleted from Technical Specifications and relocated to the COLR."

No Changes made to this page

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	4.56	0	≤ 108% of RTP** See NOTES 7, 8	≤ 109.5% 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.	Power Range, Neutron Flux, High Negative 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
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	Note 5	See Note 1	See Note 2
8.	Overpower ΔT	4.0	2.32	1.3	See Note 3	See Note 4
9.	Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10.	Pressurizer Pressure-High	7.5	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)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT ADD: "The values denoted with [*] are specified in the COLR."

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)}\left[\frac{1}{1+\tau_3 S}\right] \le \Delta T_o \left\{K_1 - K_2 \frac{(1+\tau_4 S)}{(1+\tau_5 S)}\left[T\left[\frac{1}{1+\tau_6 S}\right] - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Instrumentation;

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = Lead-lag compensator on measured ΔT;

[*] [*]

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = ~~0~~ s, τ_2 = ~~0~~ s;

$\frac{1}{1+\tau_3 S}$ = Lag compensator on measured ΔT;

[*]

τ_3 = Time constants utilized in the lag compensator for ΔT, τ_3 = ~~4~~ s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

[*]

K_1 = ~~1.185~~;

[*]

K_2 = ~~0.0224~~/°F;

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

[*] [*]

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = ~~22~~ s, τ_5 = ~~4~~ s;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued) ADD: "The values denoted with [*] are specified in the COLR."

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 = ~~0~~ [*] s;

T' = Reference T_{avg} at RATED THERMAL POWER (≤~~588.8~~ [*]°F);

K_3 = [*] ~~0.0012~~/psig;

P = Pressurizer pressure, psig;

P' = [*] ~~2235~~ 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 [*] ~~-21.6%~~ and [*] ~~+12.0~~%, 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 [*] ~~-21.6~~%, the ΔT Trip Setpoint shall be automatically reduced by [*] ~~1.75~~% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds [*] ~~+ 12.0~~%, the ΔT Trip Setpoint shall be automatically reduced by [*] ~~1.50~~% 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.

TABLE NOTATIONS

NOTE 3: OVERPOWER ΔT ADD: "The values denoted with [*] are specified in the COLR."

$$\Delta T \frac{(1+\tau_1 S)}{(1+\tau_2 S)} \frac{(1)}{(1+\tau_3 S)} \le \Delta T_o \left\{ K_4 - K_5 \frac{(\tau_7 S)}{(1+\tau_7 S)} \frac{(1)}{(1+\tau_6 S)} T - K_6 \left[T \frac{(1)}{(1+\tau_6 S)} - T'' \right] - f_2(\Delta I) \right\}$$

Where:	ΔT	=	As defined in Note 1.
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	As defined in Note 1.
	τ_1, τ_2	=	As defined in Note 1.
	$\frac{1}{1+\tau_3 S}$	=	As defined in Note 1.
	τ_3	=	As defined in Note 1.
	ΔT_o	=	As defined in Note 1.
	K_4	[*] =	~~1.12~~.
	K_5	[*] =	~~0.02~~/°F for increasing average temperature and [*] ~~0~~ for decreasing average temperature.
	$\frac{\tau_7 S}{1+\tau_7 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation.
	τ_7	=	Time constants utilized in the rate-lag compensator for T_{avg}, τ_7 = ~~13~~ [*] s.
	$\frac{1}{1+\tau_6 S}$	=	As defined in Note 1.
	τ_6	=	As defined in Note 1.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued) ADD: "The values denoted with [*] are specified in the COLR."

K_6 = [*] ~~0.002~~/°F for T > T" and K_6 = [*] ~~0~~ for T≤ T",

T = As defined in Note 1,

T" = Reference T_{avg} at RATED THERMAL POWER (≤ [*] ~~588.8~~°F)

S = As defined in Note 1, and

$f_2(\Delta I)$ = [*] ~~0 for all ΔI~~.

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.

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~~, "Technical Specification Equipment List Program and Core Operating Limits Report."~~

SHUTDOWN MARGIN - MODES 3, 4, AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limit specified in the CORE OPERATING LIMITS REPORT (COLR)~~, plant procedure PLP-106~~.

APPLICABILITY: MODES 3, 4, AND 5.

ACTION:

With the SHUTDOWN MARGIN less than the required value immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the required value:

a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and

b. At least once per 24 hours by consideration of the following factors:

   1) Reactor Coolant System boron concentration,
   2) Control rod position,
   3) Reactor Coolant System average temperature,
   4) Fuel burnup based on gross thermal energy generation,
   5) Xenon concentration, and
   6) Samarium concentration.

FIGURE 3.1-1

SHUTDOWN MARGIN VERSUS RCS BORON CONCENTRATION
MODES 3, 4, AND 5/DRAINED

This figure is deleted from Technical Specifications and is controlled by the CORE OPERATING LIMITS REPORT~~, plant procedure PLP-106~~.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR)~~, plant procedure PLP-106~~. The maximum positive limit shall be less than or equal to +5 pcm/°F for power levels up to 70% RATED THERMAL POWER and a linear ramp from that point to 0 pcm/°F at 100% RATED THERMAL POWER.

APPLICABILITY: Positive MTC Limit - MODES 1 and 2* only**.
Negative MTC Limit - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the Positive MTC Limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within the Positive MTC Limit specified in the COLR within 24 hours, or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

b. With the MTC more negative than the Negative MTC Limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

*With k_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the Positive MTC Limit specified in the COLR~~, plant procedure PLP-106~~, prior to initial operation above 5% of RATED THERMAL POWER after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the Negative MTC Limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tank via a boric acid transfer pump and a charging/safety injection pump to the Reactor Coolant System (RCS), and

b. Two flow paths from the refueling water storage tank via charging/safety injection pumps to the RCS.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one of the above required boron injection flow paths to the RCS OPERABLE, restore at least two boron injection flow paths to the RCS to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR)~~, plant procedure PLP-106~~ at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the flow path between the boric acid tank and the charging/safety injection pump suction header tank is greater than or equal to 65°F when a flow path from the boric acid tank is used;

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position;

c. At least once per 18 months by verifying that each automatic valve in the flow path actuates to its correct position on a safety injection test signal; and

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. delivers at least 30 gpm to the RCS.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging/safety injection pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With only one charging/safety injection pump OPERABLE, restore at least two charging/safety injection pumps to OPERABLE status within 72 hours* or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN as specified in the CORE OPERATING LIMITS REPORT (COLR)~~, plant procedure PLP-106~~ at 200°F within the next 6 hours; restore at least two charging/safety injection pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

-- NOTE --

*The 'A' Train charging/safety pump is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the 'A' Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the 'A' Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the 'A' Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in the HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging/safety injection pumps shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the Reactor Coolant System and reactor coolant pump seals, that a differential pressure across each pump of greater than or equal to 2446 psid is developed when tested pursuant to the Inservice Testing Program.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 As a minimum, the following borated water source(s) shall be OPERABLE as required by Specification 3.1.2.2:

a. The boric acid tank with:

1. A minimum contained borated water volume of 24,150 gallons, which is ensured by maintaining indicated level of greater than or equal to 74%,

2. A boron concentration of between 7000 and 7750 ppm, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank (RWST) with:

1. A minimum contained borated water volume of 436,000 gallons, which is equivalent to 92% indicated level.

2. A boron concentration of between 2400 and 2600 ppm,

3. A minimum solution temperature of 40°F, and

4. A maximum solution temperature of 125°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With the boric acid tank inoperable and being used as one of the above required borated water sources, restore the boric acid tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN specified in the CORE OPERATING LIMITS REPORT (COLR)~~, plant procedure PLP-106~~ at 200°F; restore the boric acid tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn as specified in the CORE OPERATING LIMITS REPORT (COLR)~~, plant procedure PLP-106~~.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With a maximum of one shutdown rod not fully withdrawn as specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Fully withdraw the rod, or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown rod shall be determined to be fully withdrawn as specified in the COLR:

a. Within 15 minutes prior to withdrawal of any rods in Control Bank A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.
**With K_{eff} greater than or equal to 1.

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR)~~, plant procedure PLP-106~~.

APPLICABILITY: MODES 1* and 2* **.

ACTION:

With the control banks inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2:

a. Restore the control banks to within the insertion limit specified in the COLR within 2 hours, or

b. Reduce THERMAL POWER within 2 hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limit specified in the COLR at least once per 12 hours except during time intervals when the rod insertion limit monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

**With K_{eff} greater than or equal to 1.

FIGURE 3.1-2

ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER, THREE LOOP OPERATION

This figure is deleted from Technical Specifications, and is controlled by the CORE OPERATING LIMITS REPORT~~, plant procedure PLP-106~~.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a band about the target AFD as specified in the CORE OPERATING LIMITS REPORT (COLR)~~, plant procedure PLP-106~~.

APPLICABILITY: MODE 1 above 50% of RATED THERMAL POWER*.

ACTION:

a. With the indicated AFD outside of the limits specified in the COLR, either:

 1. Restore the indicated AFD to within the limits specified in the COLR within 15 minutes, or

 2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the limits specified in the COLR.

---

* See Special Test Exception 3.10.2

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

This figure is deleted from Technical Specifications and is controlled by the CORE OPERATING LIMITS REPORT~~, plant procedure PLP-106~~.

FIGURE 3.2-2

K(Z) - THE NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

This figure is deleted from Technical Specifications and is controlled by the CORE OPERATING LIMITS REPORT~~, plant procedure PLP-106~~.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg} ≤ ~~594.8°F after addition for instrument uncertainty, and~~

b. Pressurizer Pressure ≥ ~~2185 psig* after subtraction for instrument uncertainty, and~~

c. RCS total flow rate ≥ 293,540 gpm ~~after subtraction for instrument uncertainty.~~

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters not within its specified limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown in Specification 3.2.5 shall be verified to be within its limit at least once per 12 hours.

4.2.5.2 Verify, by precision heat balance, that RCS total flow rate is within its limit at least once per 18 months.**

the limit specified in the COLR, and

the limit specified in the COLR*, and

and greater than or equal to the limit specified in the COLR.

* This limit is not applicable during either a THERMAL POWER Ramp in excess of ±5% RATED THERMAL POWER per minute or a THERMAL POWER step change in excess of ±10% RATED THERMAL POWER.

** Required to be performed within 24 hours after ≥ 95% RATED THERMAL POWER.

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

* Time constants utilized in the lead-lag controller for Steam Line Pressure--Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

** The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate--High is ≥ 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

The indicated values are the effective, cumulative, rate-compensated pressure drops as seen by the comparator.

NOTE 1: 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 2: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 3.3-4 (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~~, "Technical Specification Equipment List Program and Core Operating Limits Report~~.~~"~~

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PNSC, and the results of this review shall be submitted to the Manager - Nuclear Assessment Section and the Vice President - Harris Nuclear Plant.

6.7 SAFETY LIMIT VIOLATION

ADD: "Deleted."

6.7.1 ~~The following actions shall be taken in the event a Safety Limit is violated:~~

~~a.~~ ~~The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President - Harris Nuclear Plant shall be notified within 24 hours;~~

~~b.~~ ~~A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PNSC. This report shall describe: (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems, or structures, and (3) corrective action taken to prevent recurrence;~~

~~c.~~ ~~The Safety Limit Violation Report shall be submitted, within 14 days of the violation, to the Commission, the Manager - Nuclear Assessment Section, and the Vice President - Harris Nuclear Plant; and~~

~~d.~~ ~~Operation of the unit shall not be resumed until authorized by the Commission.~~

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR)~~, plant procedure PLP-106,~~ prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. SHUTDOWN MARGIN limits for Specification 3/4.1.1.2.

b. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.

c. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5.

d. Control Bank Insertion Limits for Specification 3/4.1.3.6.

e. Axial Flux Difference Limits for Specification 3/4.2.1.

f. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(Z), and V(Z) for Specification 3/4.2.2.

ADD: INSERT 3

g. Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$ for Specification 3/4.2.3.

h. Boron Concentration for Specification 3/4.9.1.

6.9.1.6.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC at the time the reload analyses are performed, and the approved revision number shall be identified in the COLR.

a. XN-75-27(P)(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

b. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

c. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.

(Methodology for Specification 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

INSERT 3

i. Reactor Core Safety Limits Figure for Specification 2.1.1.

j. Overtemperature ΔT and Overpower ΔT setpoint parameters and time constant values for Specification 2.2.1.

k. Reactor Coolant System pressure, temperature, and flow Departure from Nucleate Boiling (DNB) limits for Specification 3/4.2.5.

HNP-16-066

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT / UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-063

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux is indicative of the margin to DNB~~.~~

~~The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (in this application, the HTP correlation for Siemens Fuel. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.~~

The DNBR safety limit for high thermal performance fuel is 1.141 for the Siemens HTP correlation (Reference 1).

~~The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature below which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.~~

Insert new paragraph 1:

The restrictions of this safety limit also prevent fuel centerline melting. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The fuel centerline temperature limit is a function of weight percent of Gadolinia and pin burnup as presented in Reference 2 and approved for use at HNP per Reference 3.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE (Continued)

~~These curves are based on an enthalpy hot channel factor, $F_{\Delta H}$, specified in the CORE OPERATING LIMITS REPORT (COLR) and a limiting axial power shape. An allowance is included for an increase in calculated $F_{\Delta H}$ at reduced power based on the expression:~~

$$\cancel{F_{\Delta H} = F_{\Delta H}^{RTP} [1 + PF_{\Delta H} (1-P)]}$$

~~Where P is the fraction of RATED THERMAL POWER,~~

~~$F_{\Delta H}^{RTP}$ = $F_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR, and~~

~~$PF_{\Delta H}$ = Power Factor Multiplier for $F_{\Delta H}$ specified in the COLR.~~

~~These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the $f_1(\Delta I)$ function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.~~

Insert new paragraph 2:

The safety limit figure provided in the COLR shows the loci of points of Fraction of Rated Thermal power, RCS Pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, and that the exit quality is within the limits defined by the DNBR correlation. The reactor core safety limits are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (the 95 / 95 DNB criteria) that the hot fuel rod in the core does not experience DNB; and

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor core safety limits are used to define the various RPS functions such that the above criteria are satisfied during steady state operation and Condition I and II events. To ensure that the RPS precludes the violation of the above criteria, additional criteria are applied to the Over Temperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core safety limits will be satisfied during steady state operation and Condition I and II events.

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, ~~"Technical Specification Equipment List Program and Core Operating Limits Report,"~~ 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

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

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 ~~Figure 2.1-1.~~ 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 Table 2.2-1.

Undervoltage and Underfrequency - Reactor Coolant Pump Buses (Continued)

at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

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 (a power level of approximately 10% 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.

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. 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.

Insert new Section:

References

1. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel."
2. XN-NF-79-56(P)(A), Revision 1, "Gadolinia Fuel Properties for LWR Safety Evaluation."
3. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results."

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER
TIL
map
sym
loc
mus
acc

adjusted for measurement uncertainty then compared to the analytical limits specified in the COLR.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR that is equal to or greater than the design DNBR value throughout each analyzed transient. The indicated T_{avg} value and the indicated pressurizer pressure value are ~~compared to analytical limits of 594.8°F and 2185 psig, respectively, after an allowance for measurement uncertainty is included~~.

When RCS flow rate is measured, an additional allowance is necessary prior to comparison with the limit of Specification 3.2.5.c. Specifically for the precision calorimetric heat balance, a normal RCS flow rate error of 2.1% will be included.

Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% for undetected fouling of the feedwater venturi, raises the nominal flow measurement allowance to 2.2% for no venturi fouling. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant parameters.

Note 2 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 3.3-4 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, ~~“Technical Specification Equipment List Program and Core Operating Limits Report,”~~ 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 NTSP in place of the generic term LTSP. The NTSP is the LTSP with 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 2. 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