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Serial: RA-17-0022 10 CFR 50.90
October 19, 2017

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

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1
DOCKET NO. 50-400 / RENEWED LICENSE NO. NPF-63

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261 / RENEWED LICENSE NO. DPR-23

SUBJECT: TECHNICAL SPECIFICATION CHANGES TO SUPPORT SELF-PERFORMANCE OF CORE RELOAD DESIGN AND SAFETY ANALYSES

REFERENCES:

1. BAW-10231P-A, *COPERNIC Fuel Rod Design Computer Code*, Revision 1, FRAMATOME ANP, dated January 2004 (ADAMS Accession No. ML042930236)
2. XN-NF-81-58(P)(A), *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Revision 2, and Supplements 1 and 2, dated March 1984
3. NRC Generic Letter 83-11, *License Qualification for Performing Safety Analyses*, Supplement 1
4. NRC letter, *Oconee Nuclear Station, Units 1, 2, and 3 – Issuance of Amendments Allowing the use of the COPERNIC Fuel Performance Code (CAC NOS. MF8158, MF8159, AND MF8160)*, dated May 11, 2017 (ADAMS Accession No. ML17103A509)

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy) is submitting a request to the Nuclear Regulatory Commission (NRC) for an amendment to the Technical Specifications (TS) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP) to support the allowance of Duke Energy to self-perform core reload design and safety analyses. The proposed amendment consists of five changes:

1. Add the NRC-approved COPERNIC topical report (Reference 1) to the list of topical reports in RNP TS 5.6.5.b and HNP TS 6.9.1.6.2.
2. Relocate the following TS parameters to the Core Operating Limits Report (COLR):
   a. (RNP TS 3.5.1) Accumulator boron concentration limits.
   b. (RNP TS 3.5.4) Refueling Water Storage Tank (RWST) boron concentration limits.
   c. (HNP TS 3/4.1.1.1) Shutdown Margin.
   d. (HNP TS 3/4.1.2.5) Boric Acid Tank (BAT) and RWST boron concentration limits.
   e. (HNP TS 3/4.1.2.6) BAT and RWST boron concentration limits.

   f. (HNP TS 3/4.5.1) Accumulator boron concentration limits.
   g. (HNP TS 3/4.5.4) RWST boron concentration limits.
3. Revise the RNP TS 3.1.3 Moderator Temperature Coefficient (MTC) maximum upper limit.
4. Revise HNP TS Chapter 1 definition of Shutdown Margin, consistent with Technical Specification Task Force (TSTF) TSTF-248, Revision 0, "Revise Shutdown Margin Definition for Stuck Rod Exception."
5. Revise the RNP TS 3.2 and HNP TS 3/4.2 Power Distribution Limits LCO Actions and Surveillance Requirements to allow operation of a reactor core designed using the DPC-NE-2011-P methodology.

A meeting between Duke Energy and NRC staff was held on August 3, 2017, in which the above changes were discussed.

The Enclosure provides a description and assessment of the proposed changes. Attachments 1 and 2 provide details regarding proposed change number 5 above. Attachments 3 and 4 provide markup pages of existing TSs, highlighting all of the proposed changes.

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. The bases for these determinations are included in the Enclosure.

Approval of the proposed amendment is requested within one year of the date of this submittal. HNP implementation of this change will be completed prior to startup following the H122 outage, currently scheduled for October 2019. RNP implementation of this change will be completed prior to startup following the R232 outage, currently scheduled for September 2020.

This submittal contains no new regulatory commitments. In accordance with 10 CFR 50.91, Duke Energy is notifying the states of North Carolina and South Carolina of this license amendment request by transmitting a copy of this letter to the designated state officials. Should you have any questions concerning this letter, or require additional information, please contact Art Zaremba, Manager – Nuclear Fleet Licensing, at 980-373-2062.

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

Executed on October 19, 2017.

Sincerely,

Joseph Donahue
Vice President – Nuclear Engineering

JBD

Enclosure: Evaluation of the Proposed Change

Enclosure Attachments:

1. Robinson TS Change Description/Evaluation for DPC-NE-2011-P
2. Harris TS Change Description/Evaluation for DPC-NE-2011-P
3. Robinson Proposed Technical Specification Changes (Markup)
4. Harris Proposed Technical Specification Changes (Markup)

cc: (all with Attachments unless otherwise noted)

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ENCLOSURE

EVALUATION OF THE PROPOSED CHANGE

Subject: TECHNICAL SPECIFICATION CHANGES TO SUPPORT SELF-PERFORMANCE OF CORE RELOAD DESIGN AND SAFETY ANALYSES

Enclosure Attachments:

1. Robinson TS Change Description/Evaluation for DPC-NE-2011-P
2. Harris TS Change Description/Evaluation for DPC-NE-2011-P
3. Robinson Proposed Technical Specification Changes (Markup)
4. Harris Proposed Technical Specification Changes (Markup)

1.0 SUMMARY DESCRIPTION

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy) is submitting a request to the Nuclear Regulatory Commission (NRC) for an amendment to the Technical Specifications (TS) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP) to support the allowance of Duke Energy to self-perform core reload design and safety analyses. The proposed amendment consists of five changes:

1. Add the NRC-approved COPERNIC topical report (Reference 1) to the list of topical reports in RNP TS 5.6.5.b and HNP TS 6.9.1.6.2.
2. Relocate the following TS parameters to the Core Operating Limits Report (COLR):
   a. (RNP TS 3.5.1) Accumulator boron concentration limits.
   b. (RNP TS 3.5.4) Refueling Water Storage Tank (RWST) boron concentration limits.
   c. (HNP TS 3/4.1.1.1) Shutdown Margin.
   d. (HNP TS 3/4.1.2.5) Boric Acid Tank (BAT) and RWST boron concentration limits.
   e. (HNP TS 3/4.1.2.6) BAT and RWST boron concentration limits.
   f. (HNP TS 3/4.5.1) Accumulator boron concentration limits.
   g. (HNP TS 3/4.5.4) RWST boron concentration limits.
3. Revise the RNP TS 3.1.3 Moderator Temperature Coefficient (MTC) maximum upper limit.
4. Revise HNP TS Chapter 1 definition of Shutdown Margin, consistent with Technical Specification Task Force (TSTF) TSTF-248, Revision 0, “Revise Shutdown Margin Definition for Stuck Rod Exception.”
5. Revise the RNP TS 3.2 and HNP TS 3/4.2 Power Distribution Limits LCO Actions and Surveillance Requirements to allow operation of a reactor core designed using the DPC-NE-2011-P methodology.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

Both HNP and RNP are Westinghouse three-loop reactor coolant system (RCS) designs. The reactor core at each site contains 157 fuel assemblies, with each fuel assembly containing a matrix of fuel rods composed of Zircaloy-4 or M5. Uranium dioxide pellets are enclosed in the fuel rods. Typical HNP fuel assemblies consist of 264 fuel rods in a square 17 x 17 array. Typical RNP fuel assemblies consist of 204 fuel rods in a square 15 x 15 array. AREVA currently performs the core design and safety analysis for HNP and RNP.

Relocate TS Parameters to the COLR

HNP and RNP Accumulators:

The Emergency Core Cooling System (ECCS) accumulators are pressure vessels partially filled with borated water and pressurized with nitrogen gas. The operability of each RCS accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are sufficient to ensure the

core remains subcritical during postulated accidents. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break Loss of Coolant Accident (LOCA) is sufficient to keep that portion of the core subcritical.

HNP and RNP Refueling Water Storage Tank:

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of Shutdown Margin (SDM) or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

The operability of the RWST ensures that a sufficient supply of borated water is available for injection into the core by the ECCS. This borated water is used as cooling water for the core in the event of a LOCA and provides sufficient negative reactivity to adequately counteract any positive increase in reactivity caused by RCS cooldown. RCS cooldown can be caused by inadvertent depressurization, a LOCA, or a steam line rupture.

The limits on RWST minimum volume and boron concentration assure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all shutdown and control rods inserted except for the most reactive control assembly. These limits are consistent with the assumption of the LOCA and steam line break analyses and also satisfy boron concentration and volume requirement in non-LOCA events.

HNP Boric Acid Tank:

The BAT at HNP is an additional borated water source, similar to the RWST, that provides negative reactivity for reactor shutdown. The BAT capacity is sized to store sufficient boric acid solution for refueling plus enough for a cold shutdown from full power operation immediately following refueling with the most reactive control rod not inserted.

HNP Shutdown Margin:

A sufficient SDM ensures that: (1) the reactor can be made subcritical from all operating conditions, (2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and (3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

2.2 Current Technical Specifications Requirements

COPERNIC

RNP TS 5.6.5.b and HNP TS 6.9.1.6.2 includes a list of NRC-approved analytical methods used to determine the core operating limits. These lists for HNP and RNP currently do not include the COPERNIC topical report. AREVA currently uses the RODEX2 fuel performance code to perform fuel rod mechanical analyses for HNP and RNP.

Relocate TS Parameters to the COLR

1) RNP TS 3.5.1, "Accumulators," is applicable to Modes 1 and 2, and Mode 3 with pressurizer pressure greater than 1000 psig. It currently includes Surveillance Requirement (SR) 3.5.1.4 to verify boron concentration in each accumulator is between 1950 ppm and 2400 ppm on a frequency of 31 days and once within 6 hours after each solution volume increase of greater than or equal to 70 gallons that is not the result of addition from the RWST.

2) RNP TS 3.5.4, "Refueling Water Storage Tank (RWST)," is applicable to Modes 1 through 4 and currently includes SR 3.5.4.3 to verify RWST boron concentration is between 1950 ppm and 2400 ppm on a frequency of 7 days.

3) HNP TS 3/4.1.1.1, "Boration Control Shutdown Margin – Modes 1 and 2" is applicable to Modes 1 and 2 and currently requires the SDM to be greater than or equal to 1770 pcm for 3-loop operation. If the SDM is less than 1770 pcm, boration must be immediately initiated and continue 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 SDM is restored.

4) HNP TS 3/4.1.2.5, "Borated Water Source – Shutdown," is applicable to Modes 5 and 6 and currently requires a minimum of one borated water source be operable – either the BAT or the RWST. Operability of the BAT is defined, in part, by a boron concentration between 7000 ppm and 7750 ppm. Operability of the RWST is defined, in part, by a boron concentration between 2400 ppm and 2600 ppm. If no borated water source is operable, all operations involving core alterations or positive reactivity changes must be suspended.

5) HNP TS 3/4.1.2.6, "Borated Water Sources – Operating," is applicable to Modes 1 through 4 and currently requires BAT / RWST operability per TS 3.1.2.2, "Flow Paths – Operating." Operability of the BAT and RWST are defined, in part, with the same boron concentration limits as the aforementioned TS 3/4.1.2.5. If the BAT is inoperable and being used as a required borated water source, it must be restored within 72 hours. If the RWST is inoperable, it must be restored within 1 hour.

6) HNP TS 3/4.5.1, "Accumulators," is applicable to Modes 1 through 3 (with RCS pressure above 1000 psig) and currently requires each RCS accumulator to be operable. Operability is defined, in part, by a boron concentration between 2400 ppm and 2600 ppm. With one accumulator inoperable due to boron concentration not within limits, boron concentration must be restored within 72 hours.

7) HNP TS 3/4.5.4, "Refueling Water Storage Tank," requires the RWST to be operable in Modes 1 through 4. Operability of the RWST is defined, in part, by a boron concentration between 2400 ppm and 2600 ppm. If the RWST is inoperable, it must be restored within 1 hour (or within 12 hours if performing surveillance 4.4.6.2.2 for RCS Pressure Isolation Valve leakage testing).

Finally, HNP TS 6.9.1.6.1 and RNP TS 5.6.5.a document the core operating limits that are established prior to each reload.

RNP MTC TS Change

RNP TS 3.1.3 currently requires the MTC to be maintained within the limits specified in the COLR. It also requires the maximum upper MTC limit in the COLR to be less than or equal to +5 pcm/°F at power levels less than 50% rated thermal power (RTP) and less than or equal to 0 pcm/°F at power levels of 50% RTP and above. TS 3.1.3 applies to Modes 1 and 2 with k_{eff} greater than or equal to 1.0 for the upper MTC limit and Modes 1 through 3 for the lower MTC limit. If the MTC exceeds the upper limit, within 24 hours control bank administrative withdrawal limits must be established to maintain the MTC within the limit. If the MTC is below the lower limit, Mode 4 must be entered within 12 hours.

HNP TSTF-248

HNP TS 1.31 currently defines SDM as:

> "SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn."

DPC-NE-2011-P TS Changes

The current RNP TS 3.2 and HNP TS 3/4.2 Power Distribution Limits LCO Actions and SR are based on AREVA power distribution surveillance methodology. In addition, RNP TS 5.6.5.b and HNP TS 6.9.1.6.2 currently include the Duke Energy methodology DPC-NE-2011-P as an NRC-approved analytical method permitted to be used to determine the core operating limits. Details of the existing TS and proposed changes can be found in Attachments 3 and 4.

2.3 Reason for the Proposed Change

The proposed changes will support the allowance of Duke Energy to self-perform core reload design and safety analyses. Some additional reasons are provided below:

COPERNIC

In October 2009, the NRC issued Information Notice (IN) 2009-23 that discusses the impact of irradiation on fuel thermal conductivity. The IN states

> "It is well understood that irradiation damage and the progressive buildup of fission products in fuel pellets result in reduced thermal conductivity of the pellets. However, thermal performance codes approved by NRC before 1999 did not include this reduction in thermal conductivity with increasing irradiation because earlier test data were inconclusive as to the significance of the effect."

RODEX2 was approved prior to 1999 and did not include the fuel thermal conductivity degradation (TCD) modeling. Correction factors have since been developed to compensate for the lack of a TCD model in RODEX2; however, the COPERNIC fuel performance code has the advantage of a TCD model that directly accounts for TCD with burnup.

Relocate TS Parameters to the COLR

A number of TS address limits that generally change with each reload core. License amendments are required to change the values of these cycle-specific limits. The processing of changes to the TS is an unnecessary burden on the licensee and NRC resources. An acceptable alternative of specifying these cycle-specific limits in the COLR results in a resource savings for both the licensees and the NRC by eliminating majority of the license amendment requests on changes in values of cycle-specific parameters, provided that these limits are developed and justified using NRC-approved methodologies. The proposed changes will allow Duke Energy the flexibility of enhancing operating and core design margins without the need for cycle-specific license amendment requests. If any of the limits of safety analyses are not met, prior NRC approval of the change is required.

HNP TSTF-248

In addition, incorporating the new SDM definition into the HNP TS has the potential benefit of decreasing the necessary boration following a reactor trip to maintain SDM, as well as reducing water and acid processing leading up to and following the subsequent startup. Another advantage would be allowing the commencement of cooldown earlier.

RNP MTC TS Change

Furthermore, Duke Energy intends to begin 24-month operating cycles for RNP beginning in Cycle 32. These longer cycles require more energy (in the form of more fresh fuel assemblies at higher enrichment) in the core, which in turn requires more soluble boron in the reactor coolant system to maintain the plant in a just-critical condition. These higher boron concentrations lead to a positive change in MTC values.

Initial 24-month scoping patterns have indicated that positive margin to the current MTC limits still exists. However, the potential for further core design optimization with approved Duke Energy nuclear design methods could change the MTC values in the positive direction, with the potential to exceed the current limit. In particular, the limit of 0.0 pcm/°F at 50% RTP is considered the most at-risk.

2.4 Description of the Proposed Change

COPERNIC

AREVA currently performs fuel rod mechanical analyses for HNP and RNP using the RODEX2 fuel performance code (Reference 2). Per guidelines provided in NRC Generic Letter (GL) 83-11, Supplement 1 (Reference 3), Duke Energy will transition to self-performing fuel rod mechanical analyses using COPERNIC for HNP and RNP. The NRC approved COPERNIC for generic use in licensing applications in the safety evaluation dated June 14, 2002 (Reference 7). Duke Energy has many years of experience using fuel performance codes similar to COPERNIC. Duke Energy has self-performed fuel rod mechanical analyses for Oconee Nuclear Station (ONS) using TACO3 since 1995 and has used the Westinghouse PAD code to perform fuel rod mechanical analyses for McGuire and Catawba Nuclear Stations (MNS and CNS) since 1999. In February 2015, AREVA provided formal training to Duke Energy personnel on the COPERNIC code and methodology. NRC approval to use COPERNIC at ONS was granted on May 11, 2017 (Reference 4). AREVA provides quality assurance and change control for the COPERNIC code. COPERNIC is installed and controlled in accordance with Duke Energy software quality assurance procedures. Duke Energy's software quality

assurance program is in compliance with 10 CFR 50, Appendix B requirements. Documentation consistent with the GL 83-11 guidelines for COPERNIC will be available for NRC audit.

Upon NRC approval, the COPERNIC topical report will be added to RNP TS Section 5.6.5.b and HNP TS Section 6.9.1.6.2, as shown in Attachments 3 and 4. Because the current RNP and HNP TSs are consistent with TSTF-363, "Revise Topical Report References in ITS 5.6.5, COLR [Core Operating Limits Report]" (References 5 and 6), inclusion of revision dates for this topical report in the TS is not required, which is also consistent with Westinghouse Standard Technical Specifications (STS) (NUREG-1431, Reference 21).

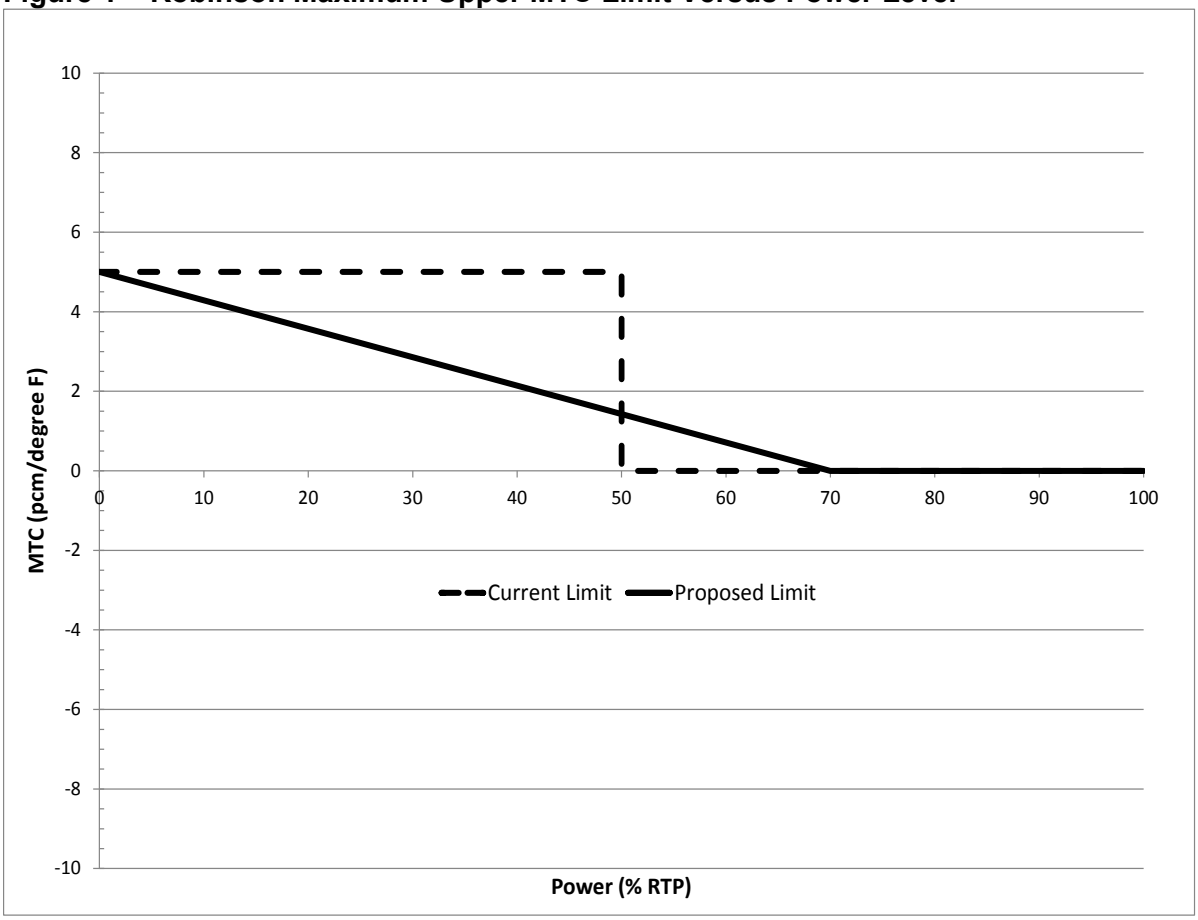
Relocate TS Parameters to the COLR

Each of the boron concentration and SDM limits described in Section 2.2 above are proposed to be replaced with "within the limits specified in the COLR" (or similar), as shown in Attachments 3 and 4. In addition, HNP TS 6.9.1.6.1 and RNP TS 5.6.5.a are modified to include these relocated limits.

RNP MTC TS Change

The maximum upper MTC limit in RNP TS 3.1.3 is proposed to be modified to be less than or equal to +5 pcm/°F at hot zero power with a linear ramp to 0 pcm/°F at 70% RTP, or 0.0 pcm/°F at 70% RTP and above. The current and proposed limits are shown graphically in Figure 1. As can be seen, the new limit represents a reduction in the maximum MTC limit at power levels greater than 0% RTP and less than 50% RTP, and a small increase in the maximum limit at powers greater than or equal to 50% RTP and less than 70% RTP.

Figure 1 – Robinson Maximum Upper MTC Limit Versus Power Level



HNP TSTF-248

The proposed revision to the definition of SDM in HNP TS 1.31 moves the words after "assuming" into new item "a," adds new sentences to item "a," and adds new item "b." This is shown below:

> "SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:
>
> a. All rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn. However, with all rod cluster assemblies verified as fully inserted by two independent means, it is not necessary to account for a stuck rod cluster assembly in the SDM calculation. With any rod cluster assembly not capable of being fully inserted, the reactivity worth of the rod cluster assembly must be accounted for in the determination of SHUTDOWN MARGIN.
> b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level."

The proposed change adopts the definition for shutdown margin included in both TSTF-248 (Reference 10) and the Westinghouse STS (NUREG-1431). Reflected in the second sentence of item "a," TSTF-248 modifies the definition of SDM to include the following new sentence:

"However, with all CONTROL RODS [rod cluster assemblies for HNP] verified as fully inserted by two independent means, it is not necessary to account for a stuck CONTROL ROD [rod cluster assembly for HNP] in the SDM [SHUTDOWN MARGIN for HNP] calculation." The last sentence in item "a" as well as item "b" are added consistent with NUREG-1431.

DPC-NE-2011-P TS Changes

In order to allow operation with the DPC-NE-2011-P methodology (Reference 13), changes are needed to the RNP TS 3.2 and HNP TS 3/4.2 Power Distribution Limits LCO Actions and Surveillance Requirements. For ease of readability and due to the number of changes, the detailed description of the changes is combined with the technical justification and provided in Attachments 1 and 2 for RNP and HNP, respectively. The markup of the TS showing each change is provided in Attachments 3 and 4.

3.0 TECHNICAL EVALUATION

COPERNIC

The NRC staff's June 14, 2002, safety evaluation (Reference 7) approving the COPERNIC code topical report states, in part, that the "COPERNIC computer code is an improved fuel performance code for fuel rod design and analysis of natural, slightly enriched (up to 5 percent) uranium dioxide fuels and urania-gadolinia fuels with the advanced cladding material, M5." The SE goes on to conclude that "the COPERNIC code is acceptable for fuel licensing applications up to a rod average burnup of 62 Gwd/MTU. Licensees that reference this topical report still need to meet 10 CFR 51.52, 'Environmental effects of transportation of fuel and waste' – Table S-4."

Both HNP and RNP are permitted to use fuel with slightly enriched uranium dioxide and M5 cladding. Furthermore, the 62 Gwd/MTU burnup limit will be verified as part of the normal reload design process. Lastly, this proposed change does not seek to modify or authorize transport, nor would it otherwise impact HNP's or RNP's obligations regarding compliance with the requirements of 10 CFR 51.52.

Relocate TS Parameters to the COLR

In Generic Letter (GL) 88-16, the NRC staff concluded that it is essential to safety that the plant is operated within the bounds of cycle specific parameter limits and that a requirement to maintain the plant within the appropriate bounds must be retained in the TS. However, the specific values of these limits may be modified by licensees, without affecting nuclear safety, provided that these changes are determined using an NRC-approved methodology and consistent with all applicable limits of the plant safety analysis that are addressed in the Final Safety Analysis Report (FSAR).

GL 88-16 established an acceptable alternative to specifying the values of cycle-specific parameter limits in the TS. The alternative contained in GL 88-16 controls the values of cycle-specific parameters and assures conformance to 10 CFR 50.36, which calls for specifying the lowest functional performance levels acceptable for continued operation, by specifying the calculation methodology and acceptance criteria. This permits operation at any specific value determined by the licensee, using the specified methodology, to be within the acceptance criteria. The COLR will document the specific values of parameter limits resulting from licensee's calculations including any mid-cycle revisions to such parameter values.

Because plant operation continues to be limited in accordance with the values of the cycle-specific parameter limits that are established using NRC-approved methodologies, this change is considered administrative in nature and there is no effect on plant safety as a consequence of this change.

For each proposed TS relocation described in Sections 2.2 and 2.4 above, DPC-NF-2010 (Reference 12) is the NRC-approved methodology used to calculate the appropriate acceptance criteria to ensure applicable plant safety analysis limits are met.

RNP MTC TS Change

The following Robinson Updated FSAR (UFSAR) non-LOCA Chapter 15 transients are evaluated at Beginning-of-Cycle conditions, with a most-positive MTC:

- UFSAR 15.2.2 Loss of External Electrical Load
- UFSAR 15.2.7 Loss of Normal Feedwater Flow
- UFSAR 15.3.1 Loss of Forced Reactor Coolant Flow
- UFSAR 15.3.2 Reactor Coolant Pump Shaft Seizure (Locked Rotor)
- UFSAR 15.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From Subcritical or Low Power
- UFSAR 15.4.2 Uncontrolled Control Rod Assembly Bank Withdrawal at Power
- UFSAR 15.4.3.1 Withdrawal of a Single Full-Length RCCA
- UFSAR 15.4.8 Spectrum of Rod Cluster Control Assembly (RCCA) Ejection Accidents
- UFSAR 15.6.1 Inadvertent Opening of Pressurizer Safety or Power Operated Relief Valve

Event 15.4.1 is a zero power event, and the analysis of record (AOR) was analyzed with an MTC of +5 pcm/F. In all but two of the remaining AORs, the events were analyzed at hot full power with an MTC of +5 pcm/°F, which bounds all current and proposed MTC limits at full- and part-power conditions. The two exceptions to this are the Loss of External Electrical Load (Peak Secondary Side Pressure case) analysis in UFSAR 15.2.2 and the Locked Rotor analysis in UFSAR 15.3.2, which use a most positive MTC of 0.0 pcm/°F at full power. These events are bounding at hot full power conditions, regardless of MTC. In the case of the former, this is because a complete load rejection from 100% power maximizes the challenge to secondary side pressure. In the latter, minimum departure from nucleate boiling ratio (MDNBR) occurs within seconds of the rotor locking wherein power level does not change significantly from the initial value. Consequently, the analysis performed at full power bounds any analysis performed at part power. Therefore, the proposed TS changes would not invalidate the conclusions of the AOR.

In addition, both Small Break LOCA (SBLOCA) and Realistic Large Break LOCA (RLBLOCA) analyses evaluated moderator reactivity based on an MTC of 0 pcm/°F. Like the previously discussed events, these are limiting at hot full power due to the higher amount of stored energy compared to part-power initial conditions.

The current UFSAR Chapter 15 analyses of record remain bounding with the proposed change to the maximum upper MTC limit. Therefore, the proposed change provides assurance that all of the applicable acceptance criteria continue to be met for each of the analyses with the revised maximum upper MTC limit.

HNP TSTF-248

The new second sentence of item “a” is consistent with NRC approved TSTF-248 (Reference 10) as well as the Westinghouse STS (NUREG-1431). The consideration of a stuck rod is provided only to allow for a single failure of one rod to not insert when a scram is initiated. However, with positive indication that all rods are already fully inserted, such a provision is overly conservative.

The change allows an exception to the highest reactivity worth stuck rod cluster assembly if there are two independent means of confirming that all rod cluster assemblies are fully inserted in the core. Due to the Digital Rod Position Indication (DRPI) system having two redundant trains of indication, if both trains are fully operable on all rod cluster assemblies, and with both trains confirming rods being fully inserted after a reactor trip, then there is adequate verification of the configuration of the rod cluster assemblies such that the assumption of stuck rod cluster assembly is not necessary. The amount of boration required to maintain shutdown margin with and without the stuck rod assumption is controlled with plant procedures.

Finally, item “b” and the last sentence of item “a” are added consistent with NUREG-1431 to ensure that the effects of a stuck rod and the fuel and moderator temperature in Modes 1 and 2 are appropriately accounted for in the determination of SDM.

DPC-NE-2011-P TS Changes

In order to allow operation with the DPC-NE-2011-P methodology (Reference 13), changes are needed to the RNP TS 3.2 and HNP TS 3/4.2 Power Distribution Limits LCO Actions and Surveillance Requirements. For ease of readability and due to the number of changes, the detailed description of the changes is combined with the technical justification and provided in Attachments 1 and 2 for RNP and HNP, respectively. The markup of the TS showing each change is provided in Attachments 3 and 4.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, General Design Criterion (GDC) 10, “Reactor Design,” requires that the reactor core and associated coolant, control, and protection systems 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. HNP is licensed to GDC 10 and this proposed change will not affect the HNP conformance to GDC 10.

RNP was not licensed to the current 10 CFR 50, Appendix A, GDC. Per the RNP UFSAR, it was evaluated against the proposed Appendix A to 10 CFR 50, General Design Criteria for Nuclear Power Plants, published in the Federal Register on July 11, 1967. Criterion 6, “Reactor Core Design,” of the July 11, 1967 proposed Appendix A requires that:

> “The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of

the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power."

This proposed change will not affect the RNP conformance to the July 11, 1967 proposed Appendix A Criterion 6.

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in a licensee's TS. Pursuant to 10 CFR 50.36(c), the TS are required to include the following five categories related to station operation:

(1) Safety limits, limiting safety system settings, and limiting control settings;
(2) Limiting conditions for operation;
(3) Surveillance requirements;
(4) Design features; and
(5) Administrative controls.

Excerpts of 10 CFR 50.36 most applicable to the proposed changes are provided below:

10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met."

10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met."

10 CFR 50.36(c)(5):

"Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4."

The regulatory requirements in 10 CFR 50.36 are not specific regarding the actions to be followed when TS requirements are not met other than a plant shut down. The proposed change revises LCOs, surveillance requirements, and remedial actions in the Technical Specifications to be followed when the LCO is not met based on a newly adopted methodology, DPC-NE-2011-P. In addition, it is required that the analytical methods used to determine the core operating limits be approved and described in the administrative controls section of the TS. The proposed change ensures that these requirements are met and is consistent with the requirements of 10 CFR 50.36.

4.2 Precedent

COPERNIC

The proposed change to adopt the COPERNIC topical report is consistent with the NRC-approved license amendments below:

1. Oconee Nuclear Station, Units 1, 2, and 3 (ONS): NRC safety evaluation dated May 11, 2017 (ADAMS Accession No. ML17103A509) (Reference 4)

   ONS filed an application for a Technical Specification change which, in part, added the COPERNIC topical report to the list of approved topical reports in ONS COLR TS 5.6.5. This approved ONS change is the same as the proposed change in this request.

2. Sequoyah Nuclear Plant, Units 1 and 2 (SQN): NRC safety evaluation dated September 26, 2012 (ADAMS Accession No. ML12249A394) (Reference 8)

   SQN filed an application for a Technical Specification change to allow the use of AREVA's Advanced W17 High Thermal Performance fuel. A part of that application was a change to the Reactor Core Safety Limit associated with the local fuel pin centerline temperature. The change was based on the use of the COPERNIC computer code, which was added to the SQN COLR TS 6.9.1.14 as part of the change. The approved SQN change is similar to the proposed change in this request in that COPERNIC was added to the COLR section of the TS.

3. Arkansas Nuclear One, Unit 1 (ANO): NRC safety evaluation dated July 9, 2014 (ADAMS Accession No. ML14169A475) (Reference 9)

   ANO filed an application for a Technical Specification change to add a provision for the determination of the maximum local fuel pin centerline temperature using the NRC reviewed and approved COPERNIC fuel performance code. The proposed change to ANO TS 2.1.1.1, which was approved by the NRC, is similar to the change proposed in this request in that COPERNIC is being referenced in the TS. However, ANO did not require a change to the COLR section of the TS.

<u>Relocate TS Parameters to the COLR</u>

Each of the proposed TS relocations have already been relocated for the McGuire (MNS) and Catawba (CNS) Nuclear Stations (see the MNS/CNS TS as well as the associated safety evaluation in References 18 through 20). There are no differences between the plant design and licensing basis for MNS/CNS and HNP/RNP that would affect the applicability of these TS relocation changes.

<u>RNP MTC TS Change</u>

Although no specific precedence of licensing actions was identified, there are industry examples of maximum upper MTC TS limits that bound the proposed MTC limit. For example the maximum upper MTC TS limit for MNS and CNS is +7 pcm/°F between hot zero power and 70% RTP, with a linear ramp to 0 pcm/°F between 70% RTP and hot full power. The maximum upper MTC TS limit for Harris Unit 1 is +5 pcm/°F between hot zero power and 70% RTP, with a linear ramp to 0 pcm/°F between 70% RTP and hot full power.

<u>HNP TSTF-248</u>

TSTF-248, Revision 0 was approved by the NRC on October 31, 2000 (Reference 11) and is incorporated into the Westinghouse STS (NUREG-1431).

DPC-NE-2011-P TS Changes

The DPC-NE-2011-P methodology has been in use at MNS and CNS since the 1990's (see the safety evaluations in References 22 and 23). Where appropriate, the proposed TS were created to be consistent with the corresponding MNS and CNS TS. This is discussed in further detail throughout the technical evaluation provided in Attachments 1 and 2.

4.3 No Significant Hazards Consideration Analysis

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy) is submitting a request to the Nuclear Regulatory Commission (NRC) for an amendment to the Technical Specifications (TS) for Shearon Harris Nuclear Power Plant, Unit 1 (HNP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP) to support the allowance of Duke Energy to self-perform core reload design and safety analyses. The proposed amendment consists of five changes:

1. Add the NRC-approved COPERNIC topical report (Reference 1) to the list of topical reports in RNP TS 5.6.5.b and HNP TS 6.9.1.6.2.
2. Relocate the following TS parameters to the Core Operating Limits Report (COLR):
   a. (RNP TS 3.5.1) Accumulator boron concentration limits.
   b. (RNP TS 3.5.4) Refueling Water Storage Tank (RWST) boron concentration limits.
   c. (HNP TS 3/4.1.1.1) Shutdown Margin.
   d. (HNP TS 3/4.1.2.5) Boric Acid Tank (BAT) and RWST boron concentration limits.
   e. (HNP TS 3/4.1.2.6) BAT and RWST boron concentration limits.
   f. (HNP TS 3/4.5.1) Accumulator boron concentration limits.
   g. (HNP TS 3/4.5.4) RWST boron concentration limits.
3. Revise the RNP TS 3.1.3 Moderator Temperature Coefficient (MTC) maximum upper limit.
4. Revise HNP TS Chapter 1 definition of Shutdown Margin, consistent with Technical Specification Task Force (TSTF) TSTF-248, Revision 0, "Revise Shutdown Margin Definition for Stuck Rod Exception."
5. Revise the RNP TS 3.2 and HNP TS 3/4.2 Power Distribution Limits LCO Actions and Surveillance Requirements to allow operation of a reactor core designed using the DPC-NE-2011-P methodology.

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) 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?

   Response: No.

   COPERNIC

   The proposed change adds a topical report for an NRC-reviewed and approved fuel performance code to the list of topical reports in RNP and HNP Technical Specifications (TS), which is administrative in nature and has no impact on a plant configuration or system performance relied upon to mitigate the consequences of an accident. The list of topical reports in the TS used to develop the core operating limits does not impact either the initiation of an accident or the mitigation of its consequences.

Relocate TS Parameters to the COLR

The proposed change relocates certain cycle-specific core operating limits from the RNP and HNP TS to the Core Operating Limits Report (COLR). The cycle-specific values must be calculated using the NRC approved methodologies listed in the COLR section of the TS. Because the parameter limits are determined using the NRC methodologies, they will continue to be within the limit assumed in the accident analysis. As a result, neither the probability nor the consequences of any accident previously evaluated will be affected.

RNP MTC TS Change

The proposed change revises the RNP Technical Specification maximum upper Moderator Temperature Coefficient (MTC) limit. Revision of the MTC limit does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

HNP TSTF-248

The proposed change revises the HNP Technical Specification definition of Shutdown Margin (SDM) consistent with existing NRC-approved definition. The proposed revision to the SDM definition will result in analytical flexibility for determining SDM. Revision of the SDM definition does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

DPC-NE-2011-P TS Changes

The proposed change revises the RNP and HNP TS to allow operation of a reactor core designed using the DPC-NE-2011-P methodology. The DPC-NE-2011-P methodology has already been approved by the NRC for use at RNP and HNP. Revision of the TS to align with the NRC-approved methodology does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident.

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 accident previously evaluated?

   Response: No.

   COPERNIC

   The proposed change adds a topical report for an NRC-reviewed and approved fuel performance code to the list of topical reports in HNP and RNP TS, which is administrative in nature and has no impact on a plant configuration or on system performance. The proposed change updates the list of NRC-approved topical reports used to develop the core

operating limits. There is no change to the parameters within which the plant is normally operated. The possibility of a new or different kind of accident is not created.

Relocate TS Parameters to the COLR

The proposed change relocates certain cycle-specific core operating limits from the RNP and HNP TS to the COLR. No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. In addition, the changes do not impose any new or different requirements or eliminate any existing requirements. The changes do not alter assumptions made in the safety analyses. The proposed changes are consistent with the safety analyses assumptions and current plant operating practice.

RNP MTC TS Change

The proposed change revises the RNP Technical Specification maximum upper MTC limit. The proposed change does not physically alter the plant; that is, no new or different type of equipment will be installed. Therefore the proposed change could also not initiate an equipment malfunction that would result in a new or different type of accident from any previously evaluated. This change does not create new failure modes or mechanisms which are not identifiable during testing, and no new accident precursors are generated.

HNP TSTF-248

Revising the HNP Technical Specification definition of SDM would not require revision to any SDM boron calculations. Rather, it would afford the analytical flexibility for determining SDM for a particular circumstance. The proposed change does not physically alter the plant; that is, no new or different type of equipment will be installed. Therefore the proposed change could also not initiate an equipment malfunction that would result in a new or different type of accident from any previously evaluated. This change does not create new failure modes or mechanisms which are not identifiable during testing, and no new accident precursors are generated.

DPC-NE-2011-P TS Changes

The proposed change revises the RNP and HNP TS to allow operation of a reactor core designed using the DPC-NE-2011-P methodology. The DPC-NE-2011-P methodology has already been approved by the NRC for use at RNP and HNP. The proposed change does not physically alter the plant, that is, no new or different type of equipment will be installed. Therefore the proposed change could also not initiate an equipment malfunction that would result in a new or different type of accident from any previously evaluated. Operating the reactor in accordance with the NRC-approved methodology will ensure that the core will operate within safe limits. This change does not create new failure modes or mechanisms which are not identifiable during testing, and no new accident precursors are generated.

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?

   Response: No.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident. These barriers include the fuel cladding, the reactor coolant system, and the containment system.

COPERNIC

The proposed change adds a topical report for an NRC-reviewed and approved fuel performance code to the list of topical reports in HNP and RNP TS, which is administrative in nature and does not amend the cycle specific parameters presently required by the TS. The individual TS continue to require operation of the plant within the bounds of the limits specified in the COLR. The proposed change to the list of analytical methods referenced in the COLR does not impact the margin of safety.

Relocate TS Parameters to the COLR

The proposed change relocates certain cycle-specific core operating limits from the RNP and HNP TS to the COLR. This change will have no effect on the margin of safety. The relocated cycle-specific parameters will continue to be calculated using NRC-approved methodologies and will provide the same margin of safety as the values currently located in the TS.

RNP MTC TS Change

The proposed change revises the RNP Technical Specification maximum upper MTC limit. The MTC limit change does not impact the reliability of the fission product barriers to function. Radiological dose to plant operators or to the public will not be impacted as a result of the proposed change. The current Updated Final Safety Analysis Report (UFSAR) Chapter 15 analyses of record remain bounding with the proposed change to the maximum upper MTC limit. Therefore, all of the applicable acceptance criteria continue to be met for each of the analyses with the revised maximum upper MTC limit.

HNP TSTF-248

The proposed revision to the HNP Technical Specification definition of SDM does not impact the reliability of the fission product barriers to function. Radiological dose to plant operators or to the public will not be impacted as a result of the proposed change. Adequate SDM will continue to be ensured for all operational conditions.

DPC-NE-2011-P TS Changes

The proposed change revises the RNP and HNP TS to allow operation of a reactor core designed using the DPC-NE-2011-P methodology. As a portion of the overall Duke Energy methodology for cycle reload safety analyses, DPC-NE-2011-P has already been approved by the NRC for use at RNP and HNP. The proposed change will continue to ensure that applicable design and safety limits are satisfied such that the fission product barriers will continue to perform their design functions. Operation of the reactor in accordance with the DPC-NE-2011-P methodology will ensure the margin of safety is not reduced.

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 change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

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

6.0 REFERENCES

1. BAW-10231P-A, *COPERNIC Fuel Rod Design Computer Code*, Revision 1, FRAMATOME ANP, dated January 2004 (ADAMS Accession No. ML042930236)
2. XN-NF-81-58(P)(A), *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Revision 2, and Supplements 1 and 2, dated March 1984
3. NRC Generic Letter 83-11, *Licensee Qualification for Performing Safety Analyses*, Supplement 1
4. NRC letter, *Oconee Nuclear Station, Units 1, 2, and 3 – Issuance of Amendments Allowing the use of the COPERNIC Fuel Performance Code (CAC NOS. MF8158, MF8159, AND MF8160)*, dated May 11, 2017 (ADAMS Accession No. ML17103A509)
5. Carolina Power & Light Company letter, *Request for Technical Specification Change Revision to Core Operating Limits Report (COLR) References*, dated June 14, 2000 (ADAMS Accession No. ML003725331)
6. Carolina Power & Light Company letter, *Revised Technical Specification Pages for License Amendment Request – Addition of Methodology References to Core Operating Limits Report*, dated January 11, 2000 (ADAMS Accession No. ML003676878)
7. NRC letter, *FRAMATOME ANP Topical Report BAW-10231, "COPERNIC Fuel Rod Design Computer Code" - Correction of Error in Safety Evaluation (TAC No. MA6792)*, dated June 14, 2002 (ADAMS Accession No. ML021360461)
8. NRC letter, *Sequoyah Nuclear Plant, Units 1 and 2 - Issuance of Amendments to Revise the Technical Specification to Allow Use of AREVA Advanced W17 High Thermal Performance Fuel (TS-SQN-2011-07) (TAC NOS. ME6538 and ME6539)*, dated September 26, 2012 (ADAMS Accession No. ML12249A394)
9. NRC letter, *Arkansas Nuclear One, Unit 1 - Issuance of Amendment RE: Revision to Technical Specification 2.1.1.1, Reactor Core Safety Limits (TAC No. MF2277)*, dated July 9, 2014 (ADAMS Accession No. ML14169A475)
10. TSTF-248-A, *Revise Shutdown Margin definition for stuck rod exception*, Revision 0

11. NRC letter from W. D. Beckner to A. R. Pietrangelo, Informing on the disposition of sixteen travelers, dated October 31, 2000 (ADAMS Accession No. ML003775261)
12. DPC-NF-2010-A, *Nuclear Physics Methodology for Reload Design*, Revision 3, as approved by NRC Safety Evaluation dated May 18, 2017 (ADAMS Accession No. ML17102A923)
13. DPC-NE-2011-P-A, *Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*, Revision 2, as approved by NRC Safety Evaluation dated May 18, 2017 (ADAMS Accession No. ML17102A923)
14. NRC letter, *Shearon Harris Nuclear Power Plant, Unit 1 and H.B. Robinson Steam Electric Plant, Unit No. 2 - Issuance of Amendments Revising Technical Specifications for Methodology Reports DPC-NE-1008-P Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," DPC-NF-2010 Revision 3, "Nuclear Physics Methodology for Reload Design," And DPC-NE-2011-P Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" (CAC Nos. MF6648/MF6649 and MF7693/MF7694)*, dated May 18, 2017 (ADAMS Accession No. ML17102A923)
15. DPC-NE-2005-P-A, *Thermal-Hydraulic Statistical Core Design Methodology*, Revision 5, as approved by NRC Safety Evaluation dated March 8, 2016 (ADAMS Accession No. ML16049A630)
16. ANF-88-054(P), *PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2*
17. Westinghouse Nuclear Advisory Letter, NSAL-15-1, *Heat Flux Hot Channel Factor Technical Specification Surveillance*, dated February 3, 2015
18. Catawba Nuclear Station Technical Specifications (ADAMS Accession No. ML052990150)
19. McGuire Nuclear Station Technical Specifications (ADAMS Accession No. ML052870418)
20. NRC letter, *Issuance of Amendments – Catawba Nuclear Station, Units 1 And 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473)*, dated March 25, 1994
21. NUREG-1431, *Standard Technical Specifications Westinghouse Plants*, Revision 4, Published April 2012 (ADAMS Accession No. ML12100A222)
22. NRC letter, *Issuance of Amendments – Catawba Nuclear Station, Units 1 and 2 (TACs M83173 and M83174)*, dated September 14, 1992
23. NRC letter, *Issuance of Amendment No. 128 to Facility Operating License NPF-9 and Amendment No. 110 to Facililty Operating License NPF-17 – McGuire Nuclear Station, Units 1 and 2 (TACs M80694)*, dated November 27, 1991

ENCLOSURE

Attachment 1

Robinson TS Change Description/Evaluation for DPC-NE-2011-P

Proposed Revision to Table of Contents

The following editorial revisions to the Table of Contents were made to be consistent with the Core Operating Limits Methodology described in DPC-NE-2011-P, *Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors* (Enclosure Reference 13).

a. Revised the section 3.2.1 title to show the heat flux hot channel factor, F_Q, is a three-dimensional quantity.

b. Revised the nuclear enthalpy rise hot channel factor nomenclature to show, $F_{\Delta H}$, is a two-dimensional quantity.

c. Removed the reference to AREVA's Axial Flux Difference (AFD) PDC-3 Axial Offset Control Methodology.

Technical Justification for Revision to Table of Contents

The AREVA Core Operating Limits Methodology is replaced by the Duke Energy methodology described in DPC-NE-2011-P. The Duke Energy method is based on three-dimensional power distribution analyses where the heat flux hot channel factor (F_Q) is measured and analyzed in three dimensions (X,Y,Z), and the nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) is measured and analyzed in two dimensions (X,Y). Margin to limit calculations are performed in three dimensions for F_Q and in two dimensions for $F_{\Delta H}$ consistent with measurement of each of these parameters. The specific reference to AREVA's PDC-3 Axial Offset Control Methodology is not applicable to Duke Energy's Core Operating Limits Methodology, and is therefore removed.

Proposed Revision to Technical Specification (TS) 3.2.1, Heat Flux Hot Channel Factor

TS 3.2.1 is revised to reflect the power peaking surveillance method described in the Duke Energy Core Operating Limits Methodology Report, DPC-NE-2011-P. Revisions to TS 3.2.1 include:

1) Revised all instances of $F_Q(Z)$ to reflect new nomenclature for the heat flux hot channel factor, $F_Q(X,Y,Z)$, used in the DPC-NE-2011-P methodology.

2) Revised all instances of $F_Q^V(Z)$, which is associated with the PDC-3 Axial Offset Control Methodology, to reflect the new measured peaking factor, $F_Q^M(X,Y,Z)$, required by the DPC-NE-2011-P methodology.

3) Action A in the current specification is replaced with new Actions A, B, and C. The current Action B is renumbered to become Action D, but is otherwise unchanged. New Action A requires the measured $F_Q(X,Y,Z)$ (designated as $F_Q^M(X,Y,Z)$) to be within its

"steady state" limit. The margin calculated represents the current operating margin of the reactor core. This comparison was not performed in the current specification. If the measured F_Q(X,Y,Z) exceeds its limit, both a thermal power reduction and a reduction in the Power Range Neutron Flux-High and OPΔT trip setpoints is required to maintain protection against consequences of postulated transients. A flux map is required to verify the acceptability of the measured F_Q prior to increasing thermal power above the reduced power level imposed by new Action A.1. The action times to perform power reductions and modify trip limits are based on operating experience and the low probability of an event occurring in the time allotted to complete each action. The allotted completion times are consistent with NUREG-1431 (Enclosure Reference 21) and with those previously approved for McGuire and Catawba Nuclear Stations.

Action B.1 is essentially equivalent to the old Action A.1. This requirement is used to verify that F_Q^M(X,Y,Z) is within the transient power distributions encountered during normal operation. This comparison is the same as in the current specification in that it limits the allowable AFD if F_Q^M(X,Y,Z) exceeds its transient limit, $F_Q^L(X,Y,Z)^{OP}$. If the operational margin calculation indicates negative margin, then a reduction in the AFD limits and/or the THERMAL POWER level is performed as specified in the Core Operating Limits Report (COLR). The THERMAL POWER reduction is controlled by Action B.2. Actions B.3 and B.4 require a reduction in the Power Range Neutron Flux – High trip and Overpower ΔT trip setpoints by greater than or equal to the magnitude of power reduction specified in Action B.2. The action times to perform power reductions and to modify the trip limits are the same as those proposed for Action A and are based on operating experience and the low probability of an event occurring in the time allotted to complete each action. They are also consistent with NUREG-1431 and with those previously approved for McGuire and Catawba Nuclear Stations. Action B.5 specifies that a flux map is required to verify the acceptability of the measured F_Q prior to increasing thermal power above the reduced power level imposed by Action B.2.

Action C is a new comparison. It compares the measured F_Q against the centerline fuel melt limit ($F_Q^L(X,Y,Z)^{RPS}$) to confirm that positive margin exists to the fuel melt limit during transient conditions. If the RPS margin calculation indicates negative margin, then the overpower ΔT $f_2(\Delta I)$ breakpoints from the COLR are reduced by KLSOPE for each 1% that the measured F_Q exceeds its limit. The variable KSLOPE is determined in the maneuvering analysis and is specified in the COLR. This requirement ensures the centerline fuel melt criterion is satisfied when core peaking may be greater than the design value.

Measured F_Q comparisons against these limits are described in Section 6 of DPC-NE-2011-P.

4) Surveillance Requirement (SR) 3.2.1.1 in the current specification is replaced with SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.1.3 requiring the measured F_Q(X,Y,Z) to satisfy three

separate limits. SR 3.2.1.1 compares the measured $F_Q(X,Y,Z)$ against the steady state limit where the measured peak is increased by measurement uncertainty and manufacturing tolerance (i.e. engineering hot channel factor). This check confirms the acceptability of the current condition of the reactor core. The measured $F_Q(X,Y,Z)$ comparisons in SR 3.2.1.2 and 3.2.1.3 refer back to new LCO's B and C where the measured $F_Q(X,Y,Z)$ must satisfy the transient Loss of Coolant Accident (LOCA) and centerline fuel melt limits. The limits to which measurements are compared correspond to the design peak at steady state conditions increased by a factor that represents the maximum amount the power at a given core location (both axially and radially) can increase above the design value before the measured value may become limiting during power maneuvers or transients. Margins to both the LOCA peaking limit (operational margin) and the centerline fuel melt (RPS margin) are calculated. The operational margin forms the basis for restricting AFD limits and/or THERMAL POWER level, while the RPS margin forms the basis for reducing the OPΔT trip $f_2(\Delta I)$ breakpoints. Allowances for measurement uncertainty and the engineering hot channel factor are included in the limits.

5) SR 3.2.1.2 and SR 3.2.1.3 require extrapolation of the measured $F_Q(X,Y,Z)$ to 31 Effective Full Power Days (EFPD) beyond the most recent measurement. The intent of this requirement is to make projections of the measurement to determine at what point the measured $F_Q(X,Y,Z)$ would exceed allowable limits if the current trend continues. In the new surveillance, an incore flux map is obtained and power distribution information from the current and previous measurements are extrapolated to project the power distribution 31 EFPD in the future. Operational and RPS $F_Q(X,Y,Z)$ margins are calculated based on the projected power distribution to determine the point in time where the measured $F_Q(X,Y,Z)$ would exceed allowable limits. If the extrapolation indicates the measured $F_Q(X,Y,Z)$ would exceed allowable limits prior to the next scheduled surveillance (31 EFPD beyond the most recent measurement), then either a flux map is performed prior to the projected point in time where the surveillance limits are projected to be exceeded or the measured $F_Q(X,Y,Z)$ is increased by an appropriate factor specified in the COLR. This requirement ensures the core is monitored at a frequency that considers the conditions where measured peaks are under-predicted or trending in an unexpected manner. The trending of measured peaking factors and margins provides the necessary information so appropriate actions can be taken prior to allowable limits being exceeded before the next 31 EFPD measurement interval.

The revised surveillance requirements address concerns identified in the Westinghouse Nuclear Advisory Letter, NSAL-15-1 (Enclosure Reference 17). The technical concern identified in this NSAL pertains to the condition where the trend in the measured F_Q is decreasing and margin to the allowable limit is also decreasing. In the Westinghouse method, if the measured F_Q decreases from the previous flux map, the measured F_Q is not penalized to ensure that F_Q is within allowable limit prior to the next measurement. In this scenario, the Technical Specification surveillance may not be sufficient to ensure

that F_Q remains within its applicable limits prior to the next surveillance interval (31 EFPD).

In the revised surveillance, the measured three-dimensional power distribution is trended and extrapolated to 31 EFPD into the future. Operational and RPS $F_Q(X,Y,Z)$ margin calculations are repeated at this future state to confirm $F_Q(X,Y,Z)$ does not exceed allowable limits. The margin calculation takes into account how transient margin changes with burnup through the use of burnup dependent operational and RPS $F_Q(X,Y,Z)$ allowable limits. This method accounts for the actual behavior the measured power distribution and the predicted behavior in $F_Q(X,Y,Z)$ margin to ensure $F_Q(X,Y,Z)$ remains within applicable limits.

Technical Justification for Revision to TS 3.2.1

TS 3.2.1 was revised to provide required actions and surveillance requirements consistent with Duke Energy's methodology for core power distribution control and surveillance of the heat flux hot channel factor. This surveillance methodology is described in DPC-NE-2011-P and is currently implemented at McGuire and Catawba Nuclear Stations.

The heat flux hot channel factor, $F_Q(X,Y,Z)$ is a specified acceptable fuel design limit that preserves the initial conditions for the Emergency Core Cooling System (ECCS) analysis. $F_Q(X,Y,Z)$ is defined as the maximum local heat flux on the surface of a fuel rod at a given core elevation (Z) in an assembly located at radial location (X,Y), divided by the average fuel rod heat flux, allowing for manufacturing tolerances on the fuel pellets and fuel rods. Since $F_Q(X,Y,Z)$ is the ratio of local surface heat fluxes, it is related to the total local power density in a fuel rod. Operation within the $F_Q(X,Y,Z)$ limits specified in the COLR prevents power peaking that would exceed the LOCA peaking limits derived in the ECCS analysis. The limit varies inversely with power for power levels above 50% rated thermal power.

The $F_Q(X,Y,Z)$ limit is the product of the peaking limit at rated thermal power (F_Q^{RTP}) and the normalized peaking limit as a function of core elevation, K(Z), and burnup, K(BU). The F_Q^{RTP}, K(Z) and K(BU) limits are provided by the fuel vendor. Analysis values are specified in the COLR.

The reload maneuvering analysis defines the AFD power level space, the rod insertion limits and the f(ΔI) penalty function(s) employed in the OPΔT and/or the OTΔT trip functions. Limits on the above parameters provide assurance that core peaking limits are bounded and thermal limits are not challenged. Appropriate uncertainties (described in DPC-NE-2011-P) are applied to calculated peaks used to establish the AFD power level space and the f(ΔI) penalty functions. Measurement of the core power distribution at steady-state conditions is performed using the incore detectors to obtain a three-dimensional flux map. The flux map is used to confirm the measured heat flux hot channel factor, $F_Q^M(X,Y,Z)$, is within the values of the designed core

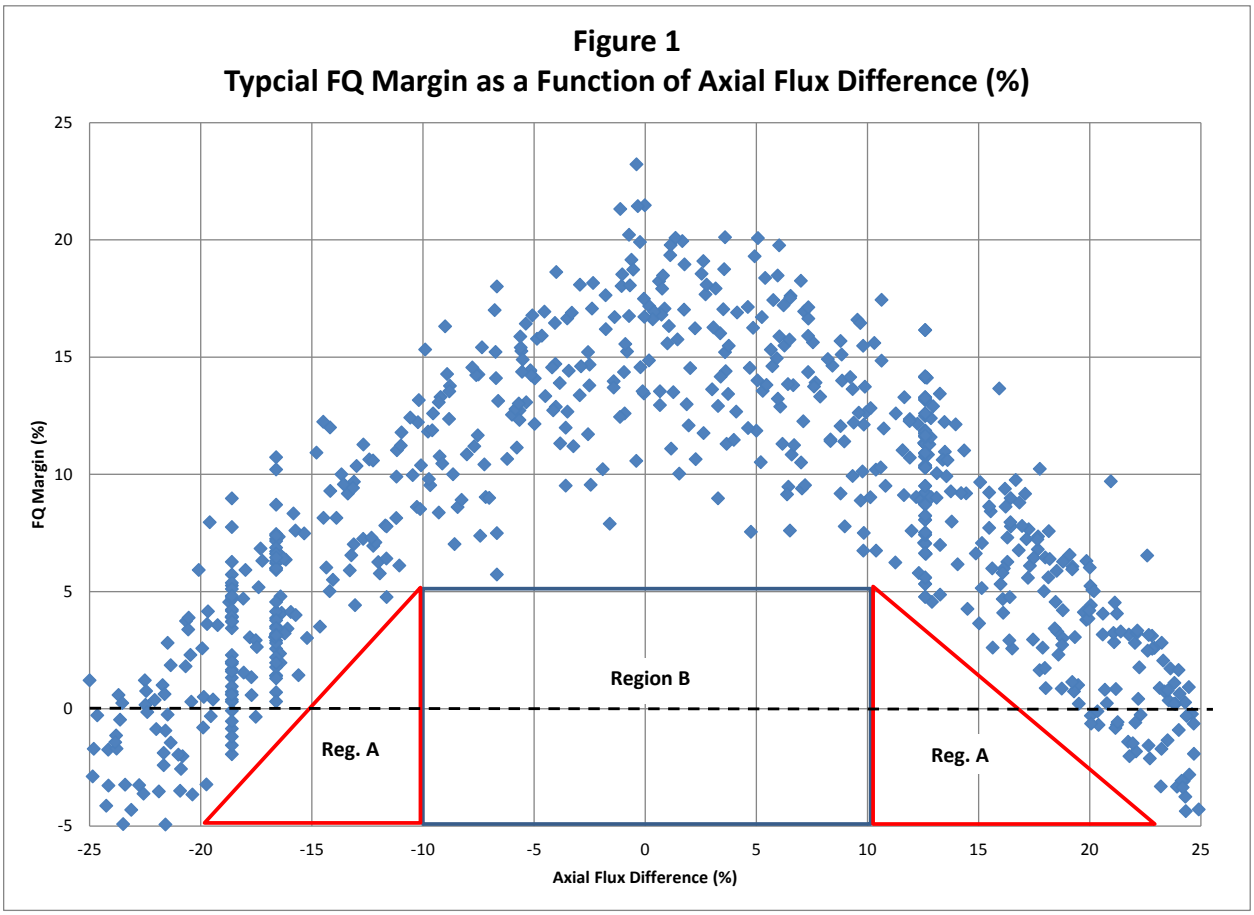
power distribution. Three comparisons are performed. The first verifies $F_Q^M(X,Y,Z)$ is within its limit for the current state of the core. The second verifies that $F_Q^M(X,Y,Z)$ is within the transient power distributions encountered during normal operation. In this comparison, the measured $F_Q^M(X,Y,Z)$ is compared against the transient limit, $F_Q^L(X,Y,Z)^{OP}$, which is the maximum peaking factor increase above steady state that preserves the LOCA limit in the transient condition. The transient limit $F_Q^L(X,Y,Z)^{OP}$ is determined in the maneuvering analysis. The third comparison verifies $F_Q^M(X,Y,Z)$ is within the transient centerline fuel melt limit, $F_Q^L(X,Y,Z)^{RPS}$. This limit is also determined in the maneuvering analysis. Together, these comparisons verify the acceptability of LOCA peaking and centerline fuel melt limits for the measured conditions as constrained by design power level, control bank insertion, AFD, and excore quadrant power tilt ratio limits.

If $F_Q^M(X,Y,Z)$ is not within its limit for the current state of the core, thermal power is reduced at least 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 15 minutes. Similarly, the Power Range Neutron Flux-High trip setpoints and OPΔT trip setpoints are reduced at least 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit. The current specification also reduces the OTΔT trip setpoints. This reduction is not necessary because the OPΔT trip function provides the primary overpower protection along with the Power Range Neutron Flux-High trip. Reducing only the OPΔT trip setpoints aligns with the current McGuire and Catawba specification, NUREG-1431, and the current and proposed Harris specification.

The current specification requires the AFD target band (i.e. AFD limits) to be reduced to a more restrictive band if the measured $F_Q(X,Y,Z)$ exceeds its transient LOCA limits. The new specification requires AFD and/or thermal power limits to be reduced depending upon the magnitude $F_Q^M(X,Y,Z)$ exceeds its transient limit. SR 3.2.1.2 is meant to provide protection against $F_Q\ (X,Y,Z)$ exceeding its limit during normal operational transient conditions and is similar to the $F_Q^V(Z)$ verification performed in the current specification. The proposed change the actions to restore $F_Q\ (X,Y,Z)$ to within its transient operational limit, $F_Q^L(X,Y,Z)^{OP}$, were made to account for the potential condition where a reduction in AFD limits alone may not be sufficient to restore $F_Q^M(X,Y,Z)$ to within its transient limit. In some instances, both a reduction in AFD and the THERMAL POWER level is required. This condition typically occurs when transient F_Q is not located in the top or bottom regions of the core where an AFD reduction can affect the magnitude of the peak.

Figure 1 illustrates F_Q margins as a function of AFD and represents typical margins for a 3-loop core. The plot was generated using the core operating limits analysis methodology described in DPC-NE-2011-P. Each point on this plot represents the minimum margin for a particular combination of xenon, control rod positon and burnup. The required action to restore the transient F_Q to within its limits are dependent on the magnitude of the negative margin, and corresponding margin improvement required. Region A illustrates a region where a conservative reduction in AFD for a given F_Q margin violation can be determined to restore F_Q

within its limits. However, if the margin improvement required moves into the AFD space defined by Region B, both an AFD and THERMAL POWER reduction would be required to restore F_Q to within its transient limits.



The AFD and THERMAL POWER reductions required to restore the transient F_Q to within its limits are moved from Technical Specifications to a new COLR Table. Relocating these values to the COLR is proposed because the AFD and THERMAL POWER reductions required to compensate for a given negative margin may be influenced by the axial variation in the F_Q limit (i.e. K(Z) penalty), fuel management, and fuel assembly and burnable absorber deigns. Consequently, compensatory actions defined in this table will be determined (or validated) on cycle-specific basis using the approved methodology defined in DPC-NE-2011-P listed in TS 5.6.5.b. An example table (Table 1) is shown below. The values in this table are representative reload values.

Table 1
(Typical)
Thermal Power and AFD Limit Reductions Required When $F_Q^L(X,Y,Z)^{OP}$ is Exceeded

Negative FQ-Op Margin (%)	Required THERMAL POWER Limit (%RTP)	Negative Limit AFD Reduction (%)	Positive Limit AFD Reduction (%)
< 2.0 % Note 1	≤ 100%	≥ 2.0%	≥ 4.0%
≥ 2.0% and < 4.0 %	≤ 97%	≥ 4.0%	≥ 6.0%
≥ 4.0% and < 6.0 %	≤ 94%	≥ 6.0%	≥ 8.0%
≥ 6.0%	≤ 50%	N/A	N/A

Note 1: Confirm positive margin exists at the reduced AFD limits by re-calculating margin using updated Monitoring Factors. If the out-of-limit condition is not resolved, reduce THERMAL POWER by greater than 3% for each 1% negative margin.

The completion time to reduce the AFD limits is changed from 15 minutes to 4 hours to be consistent with the McGuire and Catawba specifications and NUREG-1431 (Standard Technical Specifications). This change is appropriate because the AFD reduction is in response to the condition where the steady state limit is satisfied, but an operational transient would have to occur for the transient limit to be exceeded. Because normal operational transients that produce transient F_Q's are constrained by plant maneuvering limits and fission product time constants associated with the transient production and decay of iodine and xenon, a 4 hour time limit is sufficient to reduce the AFD limits to ensure transient F_Q limits are not exceeded. The 4 hour time limit for the reduction in THERMAL POWER is made for the same reasons, and to allow for an orderly decrease in the THERMAL POWER level by reactor operators.
The time allotted for the reductions in the Power Range Neutron Flux – High and Overpower ΔT trip setpoints is consistent with the current specification.

The proposed requirement to demonstrate through incore flux mapping that $F_Q^M(X,Y,Z)$ is within its transient operational limit is intended to ensure the out-of-limit condition has been resolved prior to increasing thermal power. Last, if the required actions and associated completion times cannot be met, the reactor must be brought to MODE 2 within 6 hours and defined by Action D.1 which is the same requirement as action B.1 in the current specification. This action is consistent with the McGuire and Catawba specifications, NUREG-1431, and current action B.1.

The new requirement in SR 3.2.1.2 and SR 3.2.1.3 to extrapolate the measured $F_Q(X,Y,Z)$ to 31 Effective Full Power Days (EFPD) beyond the most recent measurement is based on the methodology described in DPC-NE-2011-P. Power distribution information from the current and previous measurements are extrapolated to project the power distribution 31 EFPD in the future. The determination of both operational and RPS $F_Q(X,Y,Z)$ margins at this future state ensures appropriate actions can be taken prior to allowable limits being exceeded before the next 31 EFPD measurement interval if the current trend continues. The revised surveillance requirements addresses the concerns identified in the Westinghouse Nuclear Advisory Letter, NSAL-15-1 because the method accounts for the actual behavior of the measured power

distribution and the predicted behavior in $F_Q(X,Y,Z)$ margin to ensure $F_Q(X,Y,Z)$ remains within applicable limits.

Proposed Revision to TS 3.2.2, Nuclear Enthalpy Rise Hot Channel Factor

TS 3.2.2 was revised to reflect the power peaking surveillance method described in the Core Operating Limits Methodology Report, DPC-NE-2011-P. These revisions are summarized as follows:

1) All instances of the measured $F^N_{\Delta H}$ are revised to reflect the new nomenclature for the nuclear enthalpy rise hot channel factor ($F^M_{\Delta H}(X,Y)$) required by the DPC-NE-2011-P methodology. One exception is the TS 3.2.2 title and header, where $F^N_{\Delta H}$ was replaced with the general nomenclature $F_{\Delta H}(X,Y)$, as it is more appropriate in this context.

2) Required Actions for Condition A have been revised to reflect the methodology in DPC-NE-2011-P. The reduction in thermal power for the condition where $F^M_{\Delta H}(X,Y)$ exceeds its limit is specified by the factor RRH. RRH is the factor by which the power level is decreased for each percent $F_{\Delta H}(X,Y)$ is above its limit. RRH is defined in the COLR. The inverse of this factor is the fraction increase in the maximum allowable radial peaking (MARP) limits allowed when thermal power is decreased by 1% RTP. The RRH factor also determines the amount of reduction in the Power Range Neutron Flux-High trip setpoint. This maintains core protection and an operability margin at the reduced power level commensurate to that at rated thermal power conditions. A new action item is added to reduce the OTΔT trip setpoint if $F^M_{\Delta H}(X,Y)$ exceeds its limit. The trip setpoint reduction is governed by the factor TRH, which is defined in the COLR. This requirement ensures that protection margin is maintained at the reduced power level for departure from nucleate boiling (DNB) related transients not covered by the reduction in the Power Range Neutron Flux-High trip setpoint.

   The required completion time to reduce thermal power and the Power Range Neutron Flux-High trip setpoint if $F^M_{\Delta H}(X,Y)$ exceeds its limit were both decreased. The time required to decrease thermal power was reduced from 4 to 2 hours, and the time required to change the Power Range Neutron Flux-High trip setpoints was reduced from 72 to 8 hours. The reductions were made to maintain consistency with the Harris proposed specification and the currently approved McGuire and Catawba specifications. A 72 hour completion time is set for reducing the OTΔT trip setpoints. The time difference between changing the Power Range Neutron Flux-High trip setpoints and the OTΔT trip setpoints is due to the relative complexity of modifying each setpoint.

3) Surveillance Requirement 3.2.2.1 is replaced with two new surveillance requirements: SR 3.2.2.1 and SR 3.2.2.2. This modification was made to reflect that the

measured $F^{M}_{\Delta H}(X,Y)$ in the DPC-NE-2011-P methodology must be evaluated against both a steady state limit and a transient limit. SR 3.2.2.1 requires that $F^{M}_{\Delta H}(X,Y)$ satisfy the steady-state limit, $F^{L}_{\Delta H}(X,Y)^{LCO}$, while SR 3.2.2.2 requires that $F^{M}_{\Delta H}(X,Y)$ satisfy the transient surveillance limit, $F^{L}_{\Delta H}(X,Y)^{Surv}$. The steady-state limits bound steady-state operation, and the surveillance limits ensure the measured $F_{\Delta H}$ is within the transient power distributions encountered during normal operation. SR 3.2.2.2 also has the added requirement to trend (extrapolate) the measured nuclear enthalpy rise hot channel factor to determine at what point $F_{\Delta H}(X,Y)$ will exceed its allowable limit.

The required completion time to reduce thermal power if $F^{M}_{\Delta H}(X,Y)$ exceeds its limit was reduced from 4 hours to 2 hours. The completion time to reduce the Power Range Neutron Flux-High trip setpoint was maintained at 8 hours. A 72 hour completion time is established for reducing the OTΔT trip setpoints and is consistent with the current specification. The difference in the completion times for reducing the Power Range Neutron Flux-High trip setpoints and the OTΔT trip setpoints is because of the complexity of changing the OTΔT trip setpoints relative to the Power Range Neutron Flux-High trip setpoints. The completion times proposed are also consistent with the McGuire and Catawba specifications.

SR 3.2.2.1 requires the measured nuclear enthalpy rise hot channel factor, $F^{M}_{\Delta H}(X,Y)$, be within its steady state limit. Steady state nuclear enthalpy rise hot channel factor limits are based on the limiting Condition II transient. These limits, $F^{L}_{\Delta H}(X,Y)^{LCO}$, are MARP limits developed with the methodology described in DPC-NE-2005-P, *Thermal-Hydraulic Statistical Core Design Methodology* (Enclosure Reference 15). MARP limits are constant departure from nucleate boiling ratio (DNBR) limits which are a function of both the magnitude and location of the axial peak, F(Z). For each radial core location, the peak F(Z) and its axial location are determined from the measured 3-D power distribution and used to determine the steady state LCO limit, $F^{L}_{\Delta H}(X,Y)^{LCO}$. The measured nuclear enthalpy rise hot channel factor, $F^{M}_{\Delta H}(X,Y)$, at each radial core location is compared against the steady state LCO limit ($F^{L}_{\Delta H}(X,Y)^{LCO}$) to ensure positive DNB margin.

SR 3.2.2.2 is a new check that accounts for the variation in power distribution produced from normal operational maneuvers that are not present in the steady state measured power distribution. This check is similar to the $F^{L}_{Q}(X,Y,Z)^{OP}$ check. The transient nuclear enthalpy rise surveillance limit is designated by the parameter, $F^{L}_{\Delta H}(X,Y)^{Surv}$. This limit is based on the allowable MARP limits, and represents the design peak radial power at steady state conditions increased by a factor that represents the maximum amount that $F_{\Delta H}$ at a given core (assembly) location can increase above the design value before the measured value may become limiting.

SR 3.2.2.2 also contains a requirement to trend (extrapolate) the measured nuclear enthalpy rise hot channel factor, $F^{M}_{\Delta H}(X,Y)$, and its limits. The intent of this requirement is to determine at what point peaking would exceed allowable limits if the current trend continues. In the new surveillance, an incore flux map is obtained and a determination is made as to whether the measured $F^{M}_{\Delta H}(X,Y)$ will exceed allowable peaking at 31 EFPD beyond the most recent measurement. If the extrapolated $F^{M}_{\Delta H}(X,Y)$ measurement exceeds the allowable $F_{\Delta H}(X,Y)$ limit, then either the surveillance interval to the next power distribution map is decreased based on the available margin, or the $F^{M}_{\Delta H}(X,Y)$ measurement is increased by a penalty specified in the COLR and the peaking margin calculation is repeated. This process ensures the core is monitored at an appropriate frequency that considers the conditions where measured peaks are under-predicted or trending in an unexpected manner. The trending of measured peaking factors and margins provides the necessary information so appropriate actions can be taken prior to allowable limits being exceeded before the next 31 EFPD measurement interval.

Technical Justification for Revision to TS 3.2.2

TS 3.2.2 was revised to provide the required actions and surveillance requirements consistent with the DPC-NE-2011-P methodology for core power distribution control and surveillance of the nuclear enthalpy rise hot channel factor.

The nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, is a specified acceptable fuel design limit that preserves the initial conditions for the most limiting non-OTΔT DNB transient (i.e. primary DNB protection that is not provided by the OTΔT trip function). $F_{\Delta H}(X,Y)$ is defined as the ratio of the integral of linear power, along the rod with the highest integrated power, to the value of this integral along the average rod. Since $F_{\Delta H}(X,Y)$ integrates the power along the length of the rod, it is related to the linear heat generation rate of the fuel rod, averaged over the length of the rod. $F_{\Delta H}(X,Y)$ limits are preserved by the design analysis performed in accordance with DPC-NE-2011-P. When power distribution measurements from the incore detectors are obtained, the measured value of the assembly radial peaking factor is designated as $F^{M}_{\Delta H}(X,Y)$. Operation within MARP and $F^{L}_{\Delta H}(X,Y)^{Surv}$ limits specified in the COLR ensures the measured peaking factors are within the limits assumed in the design calculations. The $F^{L}_{\Delta H}(X,Y)^{Surv}$ limit is based on the allowable MARP limit and represents the design radial power at steady state conditions increased by a factor that represents the maximum amount that the power at a given assembly location can increase above the design value before the measured value may become limiting. Comparison of the measurement against this limit ensures the design DNB limit is preserved in the transient condition.

MARP limits are a family of maximum allowable radial peaking curves, typically plotted as maximum allowable radial peak versus axial location of peak, parameterized by the axial peaking factor, F(Z). The family of curves is the locus of points for which the minimum DNBR is equivalent to that calculated for the most limiting non-OTΔT transient (based on the reference

design peaking). The MARPs in the COLR are based on the state point that represents the point of minimum DNBR during this transient. These limits ensure the DNB design basis is satisfied for normal operation, operational transients and transients of moderate frequency.

The reload maneuvering analysis determines limits on global core parameters that can be measured directly. The primary parameters used to monitor and control the core power distribution are control bank insertion, AFD, and quadrant power tilt ratio. Limits are placed on these parameters to ensure the core power peaking factors remain bounded during power operation. Uncertainties for the nuclear design model, $F_{\Delta H}$ measurement, engineering hot channel factor, rod spacing, axial peaking factor, and other uncertainties in the CHF correlation were statistically combined to produce an overall DNBR uncertainty. This overall uncertainty was used to establish the statistical DNBR limit (SDL), as described in DPC-NE-2005-P. Since the MARP limits link the peaking limits to the DNBR limit, these uncertainties are accounted for in the MARP limits, and are therefore not applied to the predicted peaking factors in the maneuvering analysis.

Comparison of the measured core power distribution at steady-state conditions to the design power distribution provide confirmation that the measured $F_{\Delta H}(X,Y)$ is within the values of the design core power distribution. This comparison verifies the applicability of the measured core condition to the designed condition so operation within control bank insertion, AFD, and excore quadrant power tilt ratio limits preserves the initial condition DNB peaking criteria. Since the measurement uncertainty and engineering hot channel factor are included in the MARP Limits, there is no need to increase the measured core power distribution by these factors prior to comparison to limits. If $F_{\Delta H}(X,Y)$ is not within its limit, the core power level is reduced by RRH for each percent the measured $F_{\Delta H}(X,Y)$ exceeds its limit. The value of RRH is specified in the COLR and is determined based on how the $F_{\Delta H}(X,Y)$ limit changes with power. Following the power reduction, an additional 6 hours is allowed to restore $F_{\Delta H}(X,Y)$ within its limit. If this cannot be accomplished, the Power Range Neutron Flux-High trip setpoint is reduced by greater than or equal to RRH for each percent $F_{\Delta H}(X,Y)$ exceeds its limit. In addition, within 72 hours, $F_{\Delta H}(X,Y)$ is either restored within its limit or a reduction in the OTΔT setpoint is performed. Both the Power Range Neutron Flux-High trip and OTΔT trip setpoints are reduced because both these reactor trips provide DNB protection. These reductions maintain margin to core protection limits at the reduced power level commensurate to that at rated thermal power. This action limits the consequences of a transient by limiting the transient power level achievable during a postulated event. Action times for the thermal power adjustment and trip setpoint changes are based on operating experience. Setpoint adjustments associated with the power range channels are less complex than corresponding adjustments for the OTΔT trip function. The acceptability of the proposed completion times are also based on the increased DNB margin from the reduced power condition combined with the low probability of a DNB event occurring in the allotted completion time frames. The action times are consistent with the completion times proposed for Harris, and with the currently approved McGuire and Catawba completion times.

When the measurement is obtained, values of measured $F_{\Delta H}(X,Y)$ are compared directly to the MARP limits for the LCO and compared to the $F^{L}_{\Delta H}(X,Y)^{SURV}$ limit in the surveillance portion of the specification. $F^{L}_{\Delta H}(X,Y)^{SURV}$ is obtained by adjusting the design radial peaking factor, $F^{D}_{\Delta H}(X,Y)$, by a factor representative of the minimum margin at each core location (determined in the maneuvering analysis) corresponding to the transient condition. The resulting limit is the maximum peaking factor increase above steady state that preserves the DNB limit in the transient condition. The surveillance methodology used to determine $F^{L}_{\Delta H}(X,Y)^{SURV}$ is described in DPC-NE-2011-P. Both the $F^{L}_{\Delta H}(X,Y)^{SURV}$ and the MARP limits are provided in the COLR. If $F^{M}_{\Delta H}(X,Y)$ is less than $F^{L}_{\Delta H}(X,Y)^{SURV}$, then positive margin exists and $F_{\Delta H}(X,Y)$ is within its limit.

The implementation of SR 3.2.2.2 is used to determine at what point the measured $F^{M}_{\Delta H}(X,Y)$ will exceed allowable limits if the current trend continues. Power distribution information from the current and previous measurements are extrapolated to project the power distribution 31 EFPD in the future. The determination of $F^{M}_{\Delta H}(X,Y)$ margin at a future state ensures appropriate actions can be taken prior to allowable limits being exceeded before the next 31 EFPD measurement interval. The revised surveillance requirement addresses the concerns identified in the Westinghouse Nuclear Advisory Letter 15-1 except as they relate to $F_{\Delta H}(X,Y)$ versus $F_{Q}(X,Y,Z)$.

Proposed Revision to TS 3.2.3, Axial Flux Difference

The proposed change replaces AREVA's PDC-3 Axial Offset Control Methodology, *PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H.B. Robinson Unit 2* (Enclosure Reference 16), with Duke Energy's Core Operating Limits methodology described in DPC-NE-2011-P. The Duke Energy methodology produces an envelope of allowable AFD values at various power levels. Cycle-specific AFD limits are provided in the COLR, and replace the operating space referred to in the current Specification. Since AFD limits for the new method are not dependent on a target AFD, the requirement to maintain the AFD within a band about the target AFD has been removed along with the requirement to determine and update the target AFD. The AFD envelope is applicable between 50% and 100% rated thermal power. The allowable operating AFD space is anywhere within the AFD envelope.

Technical Justification for Revision to TS 3.2.3

The Core Operating Limits Methodology described in DPC-NE-2011-P is based on the performance of a three-dimensional analysis (maneuvering analysis) to determine AFD limits. The AFD limits prevent the core power distribution from exceeding allowable values determined by LOCA peaking limits, and initial condition DNB MARP limits during power operation. The AFD limits are defined by a three-dimensional core maneuvering analysis that determines core peaking dependence on core loading, fuel depletion, thermal-hydraulic state point, control rod

position, and xenon distribution. Correlations between peaking margin and axial power offset are developed that allow determination of negative and positive offset limits at selected power levels. The resulting offset limits preclude operation with negative margin (either LOCA or DNB), and are translated from offset to corresponding AFD limits. Peaking margins are calculated by augmenting nodal peaks with uncertainties and allowances as described in DPC-NE-2011-P. The margin database comprises calculations from the entire range of power distributions generated in the maneuvering analysis, including control bank insertion to the insertion limit and transient xenon conditions.

LOCA margin is calculated using fuel vendor supplied F_Q LOCA limits which include any axial or burnup dependency as defined by normalized F_Q K(Z) and K(BU) functions. The AFD limits determined are cycle-specific and are adjusted for measurement uncertainty and also include the peaking increase corresponding to a quadrant power tilt ratio of 1.02. These adjustments are applied at each power level between 50% and 100% of rated thermal power. Adjusted cycle-specific AFD limits are specified in the COLR.

Initial condition DNB peaking margins are computed from the augmented peaks and the MARP limits based on Statistical Core Design (SCD) methodology described in DPC-NE-2005-P, *Thermal-Hydraulic Statistical Core Design Methodology*. The MARP limits are a family of peaking limits for which either the minimum DNBR is equal to the deign DNBR limit, or the coolant quality at the minimum DNBR location is equal to the critical heat flux correlation quality limit. The MARP limits provide linkage between the reference DNBR analyses, with their design peaking distributions, and the core operating limits. The initial condition MARP limits are based on the state point that represents the point of minimum DNBR during the most limiting non-OTΔT DNB transient.

The maneuvering analysis methodology does not require the establishment of a target AFD and therefore reference to this band was removed. However, a target operating band is typically used to control axial power distribution in day-to-day operation. Control within this target band (operating space) constrains the variation of axial xenon and power distributions during normal operation and unit power maneuvers reducing the potential for abnormal power distributions. The action time to restore AFD to within its limits is consistent with the McGuire and Catawba specifications and NUREG-1431.

<u>Proposed Revision to TS 3.2.4, Quadrant Power Tilt Ratio</u>

TS 3.2.4 is being revised to reflect the power peaking surveillance method described in the Core Operating Limits Methodology Report, DPC-NE-2011-P. The following changes are proposed.

1) The quadrant power tilt ratio (QPTR) at which a thermal power reduction is calculated is changed from 1.0 to 1.02. This change is permissible because the power distribution analysis includes a peaking allowance for QPTRs up to 1.02. For the condition where the QPTR increases above 1.02, a reduction in thermal power is required to limit the

maximum local linear heat rate. The actions required to reduce thermal power are provided in the current specification. This change reflects a QPTR of 1.02 as the "reference" value, above which a thermal power reduction is required.

2) The phrase “Determine QPTR” in Action A.2 is clarified by replacing it with “Perform SR 3.2.4.1.” The new phrase provides more specificity on performing the intended action. Thus the intent of the action is not changed.

3) A note is added to the Applicability statement to indicate that the specification is not applicable until completion of excore detector calibration subsequent to refueling.

4) A clarification of the Completion Time notes for Actions A.5 and A.6 is made to clarify that the “more restrictive” limit of Required Action A.1 or A.2 should be applied. This clarification does not change the intent of the original note.

Technical Justification for Revision to TS 3.2.4

TS 3.2.4 is being revised to reflect required actions consistent with the Core Operating Limits Methodology described in DPC-NE-2011-P.

The three-dimensional maneuvering analysis is used to confirm the acceptability of LOCA, DNB and centerline fuel melt limits. This analysis includes an allowance for the quadrant power tilt in the core. Calculated power distributions are increased by an amount corresponding to a 2% quadrant power tilt, equivalent to a QPTR of 1.02. Therefore, the resulting rod insertion limits, AFD limits, and f(ΔI) safety limit implicitly include an allowance for excore quadrant power tilt ratios up to the TS value of 1.02. Consequently, actions are only required if the QPTR exceeds 1.02.

A clarification is also made to Action A.2 to specify performance of SR 3.2.4.1. This SR requires determination of the QPTR from either the nuclear instrumentation or by using incore movable detectors.

Finally, the note added to the Applicability statement accounts for the fact that QPTR’s following a refueling outage are unreliable until the excore nuclear instrumentation is calibrated.

ENCLOSURE

Attachment 2

Harris TS Change Description/Evaluation for DPC-NE-2011-P

Proposed Revision to Table of Contents

The section title for 3/4.2.2 is revised to show the heat flux hot channel factor, F_Q, is a three-dimensional quantity. The acronym for the nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, is added.

Technical Justification for Revision to Table of Contents

The AREVA Core Operating Limits Methodology is replaced by the Duke Energy methodology described in DPC-NE-2011-P, *Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors* (Enclosure Reference 13). This calculation method is based on three-dimensional analyses where the heat flux hot channel factor is measured and analyzed in three dimensions, (X,Y,Z), and the nuclear enthalpy rise hot channel factor is measured and analyzed in two dimensions.

Proposed Revision to Technical Specification (TS) 3/4.2.1, Axial Flux Difference

The proposed change replaces AREVA's Axial Offset Control Methodology, *PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H.B. Robinson Unit 2* (Enclosure Reference 16), with Duke Energy's Core Operating Limits Methodology described in DPC-NE-2011-P. The Duke Energy methodology produces an envelope of allowable Axial Flux Difference (AFD) values that do not change with changes in the target AFD. The AFD limits are applicable between 50% and 100% rated thermal power. Since AFD limits for the new method are not dependent on a target AFD, the requirement to maintain the AFD within a band about the target AFD for LCO 3.2.1 has been removed along with the Surveillance Requirement 4.2.1.3 to determine and update the target AFD. The allowable operating AFD space is anywhere within the AFD envelope.

Technical Justification for Revision to TS 3/4.2.1

The Core Operating Limits Methodology described in DPC-NE-2011-P is based on the performance of a three-dimensional analysis (maneuvering analysis) to determine AFD limits. The AFD limits prevent the core power distribution from exceeding allowable values determined by the loss of coolant accident (LOCA) peaking limits, and initial condition departure from nucleate boiling (DNB) maximum allowable radial peaking (MARP) limits during power operation. The AFD limits are defined by a three-dimensional core maneuvering analysis that determines the core peaking dependence on core loading, fuel depletion, thermal-hydraulic state point, control rod position, and xenon distribution. Correlations between peaking margin and axial power offset are developed that allow determination of negative and positive offset limits at selected power levels. The resulting offset limits preclude operation with negative margin (either LOCA or DNB). These limits are translated from offset to corresponding AFD

limits. Peaking margins are calculated by augmenting nodal peaks with uncertainties and allowances as described in DPC-NE-2011-P. The margin database comprises calculations from the entire range of power distributions generated in the maneuvering analysis, including control bank insertion to the insertion limit and transient xenon conditions.

LOCA margin is calculated using fuel vendor supplied F_Q LOCA limits which include any axial or burnup dependency as defined by normalized F_Q K(Z) and K(BU) functions. The AFD limits determined are cycle-specific and are adjusted for measurement uncertainty and also include the peaking increase corresponding to a quadrant power tilt ratio of 1.02. These adjustments are applied at each power level between 50% and 100% rated thermal power. Cycle-specific AFD limits are specified in the Core Operating Limits Report (COLR).

Initial condition DNB peaking margins are computed from the augmented peaks and MARP limits based on Statistical Core Design (SCD) methodology described in DPC-NE-2005-P, *Thermal-Hydraulic Statistical Core Design Methodology* (Enclosure Reference 15). The MARP limits are a family of peaking limits for which either the minimum departure from nucleate boiling ratio (DNBR) is equal to the design DNBR limit, or the coolant quality at the minimum DNBR location is equal to the critical heat flux correlation quality limit. The MARP limits provide linkage between the reference DNBR analyses, with their design peaking distributions, and the core operating limits. The initial condition MARP limits are based on the state point that represents the point of minimum DNBR during the most limiting non-OTΔT DNB transient.

The maneuvering analysis methodology does not require the establishment of a target AFD and therefore reference to this band was removed. However, a target operating band is typically used to control the axial power distribution in day-to-day operation. Control within this target band (operating space) constrains the variation of axial xenon and power distributions during normal operation and unit maneuvers reducing the potential for abnormal power distributions.

<u>Proposed Revision to TS 3/4.2.2, Heat Flux Hot Channel Factor</u>

TS 3/4.2.2 is revised to reflect the power peaking surveillance method described in the Core Operating Limits Methodology Report, DPC-NE-2011-P. Revisions to TS 3/4.2.2 include:

1) Revised the section title of 3/4.2.2, Surveillance Requirement (SR) 4.2.2.3, and the title of Figure 3.2-2 so that the heat flux hot channel factor, F_Q, is shown to vary radially (X,Y) and axially (Z) in the core.

2) All instances of the measured $F_Q(Z)$ are revised to reflect the new nomenclature in DPC-NE-2011-P for measured heat flux hot channel factor, $F_Q^M(X,Y,Z)$.

3) The action statement is revised to specify the "steady-state" limit and two additional "transient" limits. The first comparison requires the measured F_Q (designated as $F_Q^M(X,Y,Z)$), for the current state of the reactor core to be within its steady state limit. This comparison is not performed in the current specification. The second comparison is used to verify that $F_Q^M(X,Y,Z)$ is within the transient power distributions encountered during normal operation. This comparison is the same as in the current specifications in that it limits the allowable AFD if the $F_Q^M(X,Y,Z)$ exceeds its transient limit, $F_Q^L(X,Y,Z)^{OP}$. If the operational margin calculation indicates negative margin, then a reduction in the AFD limits and/or the THERMAL POWER level is performed as specified in the COLR. The third comparison is new (relative to the current AREVA method). It compares $F_Q^M(X,Y,Z)$ against the transient centerline fuel melt limit, $F_Q^L(X,Y,Z)^{RPS}$ to confirm positive margin is maintained to the centerline fuel melt limit during transient conditions. If the RPS margin calculation indicates negative margin, then the overpower ΔT $f_2(\Delta I)$ breakpoints are reduced by KLSOPE. The variable KSLOPE is also determined in the maneuvering analysis and is specified in the COLR. This requirement ensures the centerline fuel melt criterion is satisfied when core peaking may be greater than the design value. Measured $F_Q(X,Y,Z)$ comparisons against these limits are described in section 6 of DPC-NE-2011-P.

   The time to reduce the Power Range Neutron Flux-High Trip Setpoints was increased from 8 to 72 hours.

4) SR 4.2.2.2 is revised to specify that the measured $F_Q(X,Y,Z)$ is evaluated against multiple limits.

5) SR 4.2.2.2.c is revised to specify that the measured $F_Q(X,Y,Z)$ must satisfy three separate limits. The first comparison is against the steady state limit where the measured peak is increased by measurement uncertainty and the manufacturing tolerance (i.e. engineering hot channel factor). This comparison verifies the current operating condition of the reactor core. The second and third comparisons are to confirm the measured $F_Q(X,Y,Z)$ satisfies transient LOCA and centerline fuel melt limits. With the new methodology, the limits to which measurements are compared correspond to the design peak at steady state conditions increased by a factor that represents the maximum amount the power at a given core location (both axially and radially) can increase above the design value before the measured value may become limiting during power maneuvers or transients. Margins to both the LOCA peaking limit (operational margin) and the centerline fuel melt (RPS margin) are calculated. The operational margin forms the basis for restricting AFD limits and/or the thermal power level, while the RPS margin forms the basis for reducing the OPΔT trip $f_2(\Delta I)$ breakpoints. Allowances for measurement uncertainty and the engineering hot channel factor are included in the limits.

6) SR 4.2.2.2.e and SR 4.2.2.2.f are revised to implement the power peaking extrapolation method described in DPC-NE-2011-P. The intent of this surveillance requirement is to make projections of the measurement to determine at what point the measured $F_Q(X,Y,Z)$ would exceed allowable limits if the current trend continues. Power distribution measurements are performed every 31 effective full power days (EFPD) through performance of incore flux maps. Power distribution information from the current and previous measurements are extrapolated to project the power distribution 31 EFPD in the future. Operational and RPS $F_Q(X,Y,Z)$) margins are calculated based on the projected power distribution to determine the point in time where the measured $F_Q(X,Y,Z)$ would exceed allowable limits. If the extrapolation indicates the measured $F_Q(X,Y,Z)$ would exceed allowable limits prior to the next scheduled surveillance (31 EFPD beyond the most recent measurement), then either a flux map is performed prior to the point in time where the surveillance limits are projected to be exceeded or the measured $F_Q(X,Y,Z)$ is increased by an appropriate factor specified in the COLR. These requirements ensure the core is monitored at a frequency that considers the conditions where measured peaks are under-predicted or trending in an unexpected manner. The trending of the measured peaking factors and margins provides the information necessary so appropriate actions can be taken prior to allowable limits being exceeded before the next 31 EFPD measurement interval.

   The revised surveillance requirements address concerns identified in the Westinghouse Nuclear Advisory Letter, NSAL-15-1 (Enclosure Reference 17). The technical concern identified in this NSAL pertains to the condition where the trend in the measured F_Q is decreasing and margin to the allowable limit is also decreasing. In the Westinghouse method, if the measured F_Q decreases from the previous flux map, the measured F_Q is not penalized to ensure that F_Q is within allowable limit prior to the next measurement. In this scenario, the Technical Specification surveillance may not be sufficient to ensure that F_Q remains within its applicable limits prior to the next surveillance interval (31 EFPD).

   In the revised surveillance, the measured three-dimensional power distribution is trended and extrapolated to 31 EFPD into the future. Operational and RPS $F_Q(X,Y,Z)$ margin calculations are repeated at this future state to confirm $F_Q(X,Y,Z)$ does not exceed allowable limits. The margin calculation takes into account how transient margin changes with burnup through the use of burnup dependent operational and RPS $F_Q(X,Y,Z)$ allowable limits. This method accounts for the actual behavior the measured power distribution and the predicted behavior in $F_Q(X,Y,Z)$ margin to ensure $F_Q(X,Y,Z)$ remains within applicable limits.

7) The proposed change specifies that the core plane regions for which the limits specified in Specifications 4.2.2.2c, 4.2.2.2e and 4.2.2.2f are not applicable will be defined in the BASES.

Technical Justification for Revision to TS 3/4.2.2

TS 3/4.2.2 was revised to provide required actions and surveillance requirements consistent with Duke Energy's methodology for core power distribution control and surveillance of the heat flux hot channel factor. This surveillance methodology is described in DPC-NE-2011-P and is currently implemented at McGuire and Catawba Nuclear Stations.

The heat flux hot channel factor, $F_Q(X,Y,Z)$, is a specified acceptable fuel design limit that preserves the initial conditions for the Emergency Core Cooling System (ECCS) analysis. $F_Q(X,Y,Z)$ is defined as the maximum local heat flux on the surface of a fuel rod at a given core elevation (Z) in an assembly located at radial location (X,Y), divided by the average fuel rod heat flux, allowing for manufacturing tolerances on the fuel pellets and fuel rods. Since $F_Q(X,Y,Z)$ is the ratio of local surface heat fluxes, it is related to the total local power density in a fuel rod. Operation within the $F_Q(X,Y,Z)$ limits specified in the COLR prevents power peaking that would exceed the LOCA peaking limits derived in the ECCS analysis. The limit varies inversely with power for power levels above 50% rated thermal power.

The $F_Q(X,Y,Z)$ limit is the product of the peaking limit at rated thermal power (F_Q^{RTP}) and the normalized peaking limit as a function of core elevation, K(Z), and burnup, K(BU). F_Q^{RTP}, K(Z) and K(BU) limits are provided by the fuel vendor. Analysis values are specified in the COLR.

The reload maneuvering analysis defines the AFD power level space, the rod insertion limits and the f(ΔI) penalty function(s) employed in the OPΔT and/or the OTΔT trip functions. Limits on the above parameters provide assurance that core peaking limits are bounded and thermal limits are not challenged. Appropriate uncertainties described in DPC-NE-2011-P are applied to calculated peaks used to establish the AFD power level space and the f(ΔI) penalty functions.

Measurement of the core power distribution at steady-state conditions is performed using the incore detectors to obtain a three-dimensional flux map. The flux map is used to confirm the measured heat flux hot channel factor, $F_Q^M(X,Y,Z)$, is within the values of the designed core power distribution. Three comparisons are performed. The first verifies $F_Q^M(X,Y,Z)$ is within its limit for the current state of the core. The second verifies that $F_Q^M(X,Y,Z)$ is within the transient power distributions encountered during normal operation. In this comparison, the measured $F_Q^M(X,Y,Z)$ is compared against the transient limit, $F_Q^L(X,Y,Z)^{OP}$, which is the maximum peaking factor increase above steady state that preserves the LOCA limit in the transient condition. The transient limit, $F_Q^L(X,Y,Z)^{OP}$, is determined in the maneuvering analysis. The third comparison verifies $F_Q^M(X,Y,Z)$ is within the transient centerline fuel melt limit, $F_Q^L(X,Y,Z)^{RPS}$. This limit is also

determined in the maneuvering analysis. Together, these comparisons verify the acceptability of LOCA peaking and centerline fuel melt limits for the measured conditions as constrained by design power level, control bank insertion, AFD, and excore quadrant power tilt ratio limits.

If $F_Q^M(X,Y,Z)$ is not within its limit for the current state of the core, thermal power is reduced at least 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 15 minutes. Similarly, the Power Range Neutron Flux-High Trip Setpoints are reduced at least 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit. The time frame to reduce the Power Range Neutron Flux-High Trip Setpoints was increased from 8 to 72 hours. This change was made to align this requirement with the current McGuire and Catawba specifications, NUREG-1431 and the current and proposed Robinson specification. The 72 hour completion time is based on the increased LOCA margin from the power reduction combined with the low probability of a LOCA occurring in the 72 hour allotted time frame. The time requirement (72 hours) to reduce the OPΔT trip setpoint is not changed.

The current specification requires AFD limits to be reduced 1% for each 1% the measured $F_Q(X,Y,Z)$ exceeds its transient LOCA limits. The new specification requires AFD and/or thermal power limits to be reduced depending upon the magnitude $F_Q^M(X,Y,Z)$ exceeds its transient limit. SR 4.2.2.2.c is meant to provide protection against $F_Q(X,Y,Z)$ exceeding its limit during normal operational transient conditions. The proposed change to the SR 4.2.2.2.c actions to restore $F_Q(X,Y,Z)$ to within its transient operational limit, $F_Q^L(X,Y,Z)^{OP}$, were made to account for the potential condition where a reduction in AFD limits alone may not be sufficient to restore $F_Q^M(X,Y,Z)$ to within its transient limit. In some instances, both a reduction in AFD and the THERMAL POWER level is required. This condition typically occurs when transient F_Q is not located in the top or bottom regions of the core where an AFD reduction can affect the magnitude of the peak.

Figure 1 illustrates F_Q margins as a function of AFD. This plot was generated using the core operating limits analysis methodology described in DPC-NE-2011-P. Each point on this plot represents the minimum margin for a particular combination of xenon, control rod positon and burnup. The required action to restore the transient F_Q to within its limits are dependent on the magnitude of the negative margin, and corresponding margin improvement required. Region A illustrates a region where a conservative reduction in AFD for a given F_Q margin violation can be determined to restore F_Q within its limits. However, if the margin improvement required moves into the AFD space defined by Region B, both an AFD and thermal power reduction would be required to restore F_Q to within its transient limits.

The AFD and THERMAL POWER reductions required to restore the transient F_Q to within its limits are moved from Technical Specifications to a new COLR Table. Relocating these values to the COLR is proposed because the AFD and THERMAL POWER reductions required to compensate for a given negative margin may be influenced by the axial variation in the F_Q limit

(i.e. K(Z) penalty), fuel management, and fuel assembly and burnable absorber deigns. Consequently, compensatory actions defined in this table will be determined (or validated) on cycle-specific basis using the approved methodology defined in DPC-NE-2011-P listed in TS 6.9.1.6.2. An example table (Table 1) is shown below. The values in this table are representative reload values.

Table 1
(Typical)
Thermal Power and AFD Limit Reductions Required When $F_Q^L(X,Y,Z)^{OP}$ is Exceeded

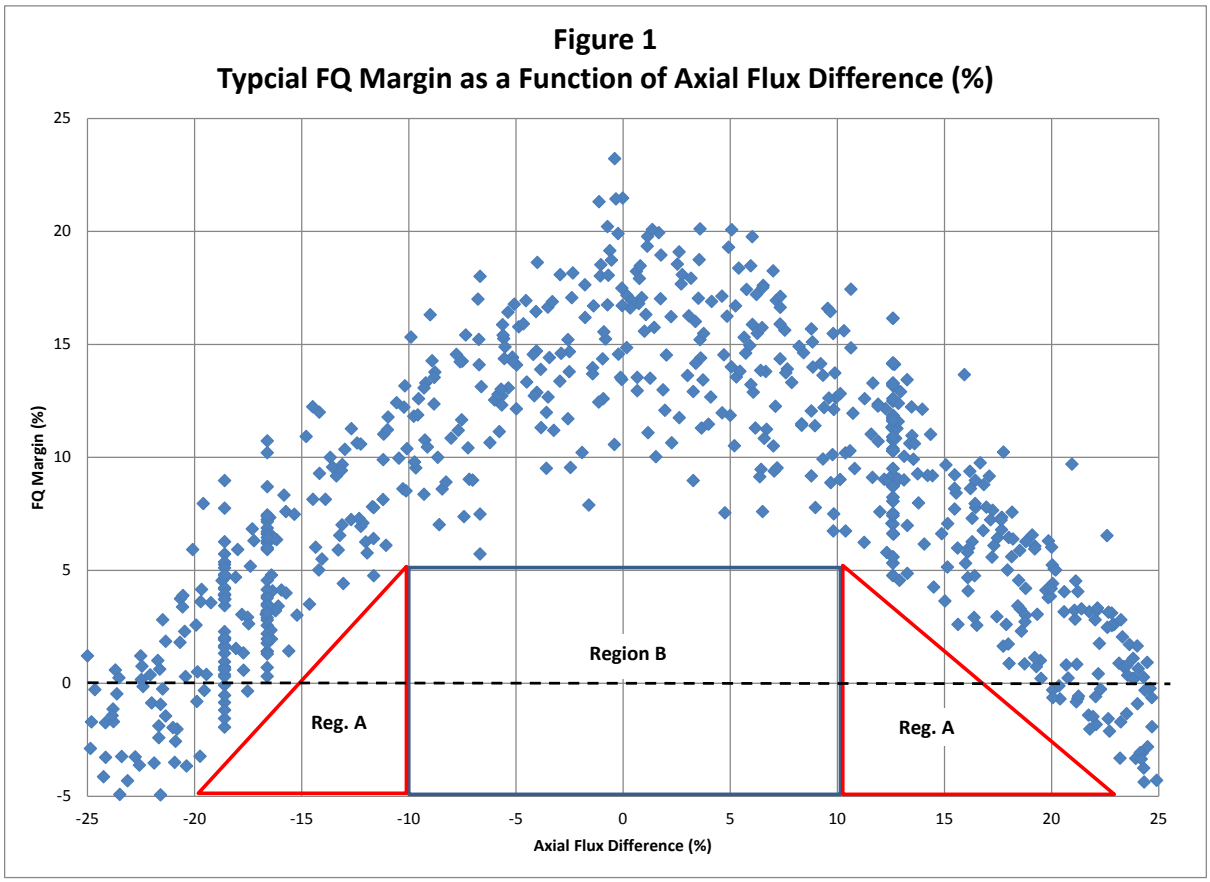
Negative FQ-Op Margin (%)	Required THERMAL POWER Limit (%RTP)	Negative Limit AFD Reduction (%)	Positive Limit AFD Reduction (%)
< 2.0 % Note 1	≤ 100%	≥ 2.0%	≥ 4.0%
≥ 2.0% and < 4.0 %	≤ 97%	≥ 4.0%	≥ 6.0%
≥ 4.0% and < 6.0 %	≤ 94%	≥ 6.0%	≥ 8.0%
≥ 6.0%	≤ 50%	N/A	N/A

Note 1: Confirm positive margin exists at the reduced AFD limits by re-calculating margin using updated Monitoring Factors. If the out-of-limit condition is not resolved, reduce THERMAL POWER by greater than 3% for each 1% negative margin.

The completion time to reduce the AFD limits is changed from 15 minutes to 4 hours to be consistent with the McGuire and Catawba specifications and NUREG-1431 (standard Technical Specifications). This change is appropriate because the AFD reduction is in response to the condition where the steady state limit is satisfied, but an operational transient limit would have to occur for the transient limit to be exceeded. Because normal operational transients that produce transient F_Q's are constrained by plant maneuvering limits and fission product time constants associated with the transient production and decay of iodine and xenon, a 4 hour time limit is sufficient to reduce the AFD limits to ensure transient F_Q limits are not exceeded. The 4 hour time limit for the reduction in THERMAL POWER is made for the same reasons, and to allow for an orderly decrease in the THERMAL POWER level by reactor operators.

The time allotted for the reductions in the Power Range Neutron Flux – High and Overpower ΔT trip setpoints is consistent with the current specification. The requirement to reset the AFD alarm setpoints to the modified AFD limits within 8 hours is removed since this is an administrative change and not a protection setpoint.

The proposed requirement to demonstrate through incore flux mapping that $F_Q^M(X,Y,Z)$ is within its transient operational limit is intended to ensure the out-of-limit condition has been resolved prior to increasing thermal power. Last, if the required actions and associated completion times cannot be met, the reactor must be brought to MODE 2 within 6 hours. This action is consistent with the McGuire and Catawba specifications, NUREG-1431 and current actions associated with TS 3.2.2.



The replacement of SR 4.2.2.2.e and SR 4.2.2.2.f with the power peaking extrapolation method described in DPC-NE-2011-P is used to determine at what point the measured $F_Q(X,Y,Z)$ would exceed allowable limits if the current trend continues. Power distribution information from the current and previous measurements are extrapolated to project the power distribution 31 EFPD in the future. The determination of both operational and RPS $F_Q(X,Y,Z)$ margins at this future state ensures appropriate actions can be taken prior to allowable limits being exceeded before the next 31 EFPD measurement interval. The revised surveillance requirements addresses the concerns identified in the Westinghouse Nuclear Advisory Letter, NSAL-15-1 because the method accounts for the actual behavior of the measured power distribution and the predicted behavior in $F_Q(X,Y,Z)$ margin to ensure $F_Q(X,Y,Z)$ remains within applicable limits.

SR 4.2.2.2g identifies the regions of the core in which Specifications 4.2.2.2c, 4.2.2.2e and 4.2.2.2f do not apply. The change relocates the regions of the core in which these Specifications do not apply to the Technical Specification Bases. This change is consistent with NUREG-1431. Cycle-specific analyses performed in the maneuvering analyses verify the acceptability of peaking limits over the entire core height.

Proposed Revision to TS 3/4.2.3, Nuclear Enthalpy Rise Hot Channel Factor

TS 3/4.2.3 was revised to reflect the power peaking surveillance method described in the Core Operating Limits Methodology Report, DPC-NE-2011-P. These revisions are summarized as follows:

1) All instances of the measured $F_{\Delta H}$ are revised to reflect the new nomenclature for the nuclear enthalpy rise hot channel factor, $F^{M}_{\Delta H}(X,Y)$, used by the DPC-NE-2011-P methodology. In addition, the general nomenclature of $F_{\Delta H}(X,Y)$ is added to the TS 3/4.2.3 title.

2) "Action a" is revised to reduce THERMAL POWER by greater than RRH from rated thermal power for each 1% $F^{M}_{\Delta H}(X,Y)$ exceeds its limit, and within the next 6 hours either restore $F^{M}_{\Delta H}(X,Y)$ to within its limit for RTP or reduce the Power Range Neutron Flux-High trip setpoints greater than RRH for each 1% $F^{M}_{\Delta H}(X,Y)$ exceeds its limit. RRH is the factor by which the thermal power level and Power Range Neutron Flux-High trip setpoints are decreased for each percent $F_{\Delta H}(X,Y)$ is above its limit. RRH is defined in the COLR. The inverse of this factor is the fraction increase in the MARP limits allowed when thermal power is decreased by 1% RTP. Reducing the Power Range Neutron Flux-High trip setpoint maintains core protection and an operability margin at the reduced power level commensurate to that at rated thermal power conditions. A new action item is added to either restore $F^{M}_{\Delta H}(X,Y)$ to within its limit for RTP or reduce the OTΔT trip setpoint by greater than TRH for each 1% $F^{M}_{\Delta H}(X,Y)$ exceeds its limit. The factor TRH is specified in the COLR. This requirement ensures that protection margin is maintained at the reduced power level for DNB related transients not covered by the reduction in the Power Range Neutron Flux-High trip setpoint.

   The required completion time to reduce thermal power if $F^{M}_{\Delta H}(X,Y)$ exceeds its limit was reduced from 4 hours to 2 hours. The completion time to reduce the Power Range Neutron Flux-High trip setpoint was maintained at 8 hours. A 72 hour completion time is established for reducing the OTΔT trip setpoints because of the complexity of changing the OTΔT trip setpoints relative to the Power Range Neutron Flux-High trip setpoints. The completion times proposed are also consistent with the McGuire and Catawba specifications.

3) SR 4.2.3.2 is modified to specify that $F_{\Delta H}(X,Y)$ in the DPC-NE-2011-P methodology is evaluated against multiple limits. Sub-steps a and b have been added to specify that $F_{\Delta H}(X,Y)$ must satisfy a steady-state limit ($F^{L}_{\Delta H}(X,Y)^{LCO}$), and a surveillance limit ($F^{L}_{\Delta H}(X,Y)^{Surv}$). The steady-state limits bound steady-state operation, and the surveillance limits ensure the measured $F_{\Delta H}(X,Y)$ is within the transient power

distributions encountered during normal operations. Sub-step c has been added to describe the new requirement to trend (extrapolate) the measured nuclear enthalpy rise hot channel factor to determine at what point $F_{\Delta H}(X,Y)$ will exceed its allowable limit.

The proposed SR 4.2.3.2.a requires the measured nuclear enthalpy rise hot channel factor, $F^{M}_{\Delta H}(X,Y)$, be within the steady state limit. Steady state nuclear enthalpy rise hot channel factor limits are based on the limiting Condition II transient. These limits, $F^{L}_{\Delta H}(X,Y)^{LCO}$, are MARP limits developed with the methodology described in DPC-NE-2005-P, *Thermal-Hydraulic Statistical Core Design Methodology.* MARP limits are constant DNBR limits which are a function of both the magnitude and location of the axial peak, F(Z). For each radial core location the peak F(Z) and its axial location are determined from the measured 3-D power distribution and used to determine the steady state LCO limit, $F^{L}_{\Delta H}(X,Y)^{LCO}$. The measured nuclear enthalpy rise hot channel factor, $F^{M}_{\Delta H}(X,Y)$, at each radial core location is compared against the steady state LCO limit ($F^{L}_{\Delta H}(X,Y)^{LCO}$) to ensure positive DNB margin.

The proposed SR 4.2.3.2.b is a new check that accounts for the variation in power distribution produced from normal operational maneuvers that are not present in the steady state measured power distribution. This check is similar to the $F^{L}_{Q}(X,Y,Z)^{OP}$ check. The transient nuclear enthalpy rise surveillance limit is designated by the parameter, $F^{L}_{\Delta H}(X,Y)^{Surv}$. This limit is based on the allowable MARP limit and represents the design radial power at steady state conditions increased by a factor that represents the maximum amount that $F_{\Delta H}$ at a given core (assembly) location can increase above the design value before the measured value may become limiting.

SR 4.2.3.2.c is added to trend the measured nuclear enthalpy rise hot channel factor, $F^{M}_{\Delta H}(X,Y)$, and its limits. The intent of this requirement is to determine at what point peaking would exceed allowable limits if the current trend continues. In the new surveillance, an incore flux map is obtained and a determination is made as to whether the measured $F^{M}_{\Delta H}(X,Y)$ will exceed allowable peaking at 31 Effective Full Power Days (EFPD) beyond the most recent measurement. If the extrapolated $F^{M}_{\Delta H}(X,Y)$ measurement exceeds the allowable $F_{\Delta H}(X,Y)$ limit, then either the surveillance interval to the next power distribution map is decreased based on the available margin, or the $F^{M}_{\Delta H}(X,Y)$ measurement is increased by a penalty specified in the COLR and the peaking margin calculation is repeated. This process ensures the core is monitored at an appropriate frequency that considers the conditions where measured peaks are under-predicted or trending in an unexpected manner. The trending of measured peaking factors and margins provides the necessary information so appropriate actions can be taken prior to allowable limits being exceeded before the next 31 EFPD measurement interval.

4) SR 4.2.3.2.d is added to appropriately format the pre-existing surveillance schedule.

Technical Justification for Revision to TS 3/4.2.3

TS 3/4.2.3 is revised to provide the required actions and surveillance requirements consistent with the DPC-NE-2011-P methodology for core power distribution control and surveillance of the nuclear enthalpy rise hot channel factor.

The nuclear enthalpy rise hot channel factor, $F_{\Delta H}(X,Y)$, is a specified acceptable fuel design limit that preserves the initial conditions for the most limiting non-OTΔT DNB transient (i.e. primary DNB protection that is not provided by the OTΔT trip function). $F_{\Delta H}(X,Y)$ is defined as the ratio of the integral of linear power, along the rod with the highest integrated power, to the value of this integral along the average rod. Since $F_{\Delta H}(X,Y)$ integrates the power along the length of the rod, it is related to the linear heat generation rate of the fuel rod, averaged over the length of the rod. $F_{\Delta H}(X,Y)$ limits are preserved by the design analysis performed in accordance with DPC-NE-2011-P. When power distribution measurements from the incore detectors are obtained, the measured value of the assembly radial peaking factor is designated as $F^{M}_{\Delta H}(X,Y)$. Operation within the MARP and $F^{L}_{\Delta H}(X,Y)$ limits specified in the COLR ensures the measured peaking factors are within the limits assumed in the design calculations. The $F^{L}_{\Delta H}(X,Y)^{Surv}$ limit is based on the allowable MARP limit and represents the design radial power at steady state conditions increased by a factor that represents the maximum amount that the power at a given assembly location can increase above the design value before the measured value may become limiting. Comparison of the measurement against this limit ensures the design DNB limit is preserved in the transient condition.

MARP limits are a family of maximum allowable radial peaking curves, typically plotted as maximum allowable radial peak versus axial location of peak, parameterized by the axial peaking factor, F(Z). The family of curves is the locus of points for which the minimum DNBR is equivalent to that calculated for the most limiting non-OTΔT transient (based on the reference design peaking). The MARPs in the COLR are based on the state point that represents the point of minimum DNBR during this transient. These limits ensure the DNB design basis is satisfied for normal operation, operational transients and transients of moderate frequency.

The reload maneuvering analysis determines limits on global core parameters that can be measured directly. The primary parameters used to monitor and control the core power distribution are control bank insertion, AFD, and quadrant power tilt ratio. Limits are placed on these parameters to ensure the core power peaking factors remain bounded during power operation. Uncertainties for the nuclear design model, $F_{\Delta H}$ measurement, engineering hot channel factor, rod spacing, axial peaking factor, and other uncertainties in the CHF correlation were statistically combined to produce an overall DNBR uncertainty. This overall uncertainty was used to establish the statistical DNBR limit (SDL), as described in DPC-NE-2005-P. Since

the MARP limits link the peaking limits to the DNBR limit, these uncertainties are accounted for in the MARP limits, and are therefore not applied to the predicted peaking factors in the maneuvering analysis.

Comparison of the measured core power distribution at steady-state conditions to the design power distribution provide confirmation that the measured $F_{\Delta H}(X,Y)$ is within the values of the design core power distribution. This comparison verifies the applicability of the measured core condition to the designed condition, so that if the control bank insertion and AFD are at their most limiting values, then the initial condition DNB peaking criteria are preserved. Since the measurement uncertainty and engineering hot channel factor are included in the MARP Limits, the measured core power distribution does not have to be increased by these factors prior to comparison to limits. If $F_{\Delta H}(X,Y)$ is not within its limit, the core power level is reduced by RRH for each percent the measured $F_{\Delta H}(X,Y)$ exceeds its limit. The value of RRH is specified in the COLR and is determined based on how the $F_{\Delta H}(X,Y)$ limit changes with power. Following the power reduction, an additional 6 hours is allowed to restore $F_{\Delta H}(X,Y)$ within its limit. If this cannot be accomplished, the Power Range Neutron Flux-High trip setpoint is reduced by greater than or equal to RRH for each percent $F_{\Delta H}(X,Y)$ exceeds its limit. In addition, within 72 hours, $F_{\Delta H}(X,Y)$ is either restored within its limit or a reduction in the OTΔT setpoint is performed. Both the Power Range Neutron Flux-High trip and OTΔT trip setpoints are reduced because both of these reactor trips provide DNB protection. The trip setpoint reductions also maintain margin to core protection limits at the reduced power level commensurate to that at rated thermal power, and limits consequences of a transient by limiting the transient power level achievable during a postulated event. Actions times are based on operating experience, and the increased DNB margin from the reduced power condition combined with the low probability of a DNB event occurring in the allotted completion time. They are also consistent with the completion times currently approved for McGuire and Catawba Nuclear Stations.

When the measurement is obtained, values of measured $F_{\Delta H}(X,Y)$ are compared directly to the MARP limits for the LCO and compared to the $F^{L}_{\Delta H}(X,Y)^{SURV}$ limit in the surveillance portion (SR 4.2.3.2) of the specification. $F^{L}_{\Delta H}(X,Y)^{SURV}$ is obtained by adjusting the design $F^{D}_{\Delta H}(X,Y)$ for the least amount of margin obtained from all the power distributions analyzed in the maneuvering analysis. The resulting limit is the maximum peaking factor increase above steady state that preserves the DNB limit in the transient condition. The surveillance methodology used in obtaining $F^{L}_{\Delta H}(X,Y)^{SURV}$ is described in DPC-NE-2011-P. Both the resulting $F^{L}_{\Delta H}(X,Y)^{SURV}$ and the MARP limits are provided in the COLR. If $F^{M}_{\Delta H}(X,Y)$ is less than $F^{L}_{\Delta H}(X,Y)^{SURV}$, then positive margin exists, and $F_{\Delta H}(X,Y)$ is within its limit.

The implementation of SR 4.2.3.2.c is used to determine at what point the measured $F^{M}_{\Delta H}(X,Y)$ will exceed allowable limits if the current trend continues. Power distribution information from the current and previous measurements are extrapolated to project the power distribution 31 EFPD in the future. The determination of $F^{M}_{\Delta H}(X,Y)$ margin at a future state ensures appropriate

actions can be taken prior to allowable limits being exceeded before the next 31 EFPD measurement interval. The revised surveillance requirement addresses the concerns identified in the Westinghouse Nuclear Advisory Letter 15-1 except as they relate to $F_{\Delta H}(X,Y)$ versus $F_Q(X,Y,Z)$.

Proposed Revision to TS 3/4.2.4, Quadrant Power Tilt Ratio

TS 3/4.2.4 is revised to reflect the power peaking surveillance method described in the Core Operating Limits Methodology Report, DPC-NE-2011-P. The following change is proposed:

1) The quadrant power tilt ratio (QPTR) at which a thermal power reduction is calculated is changed from 1.0 to 1.02. This change is permissible because the power distribution analysis includes a peaking allowance for quadrant power tilt ratios up to 1.02. For the condition where the quadrant power tilt ratio increases above 1.02, a reduction in thermal power is required to limit the maximum local linear heat rate. The actions required to reduce thermal power are provided in the current specification. This change reflects a quadrant power tilt ratio of 1.02 as the "reference" value, above which a thermal power reduction is required. Action statements a.2.b) and b.2 are affected by this change.

Technical Justification for Revision to TS 3/4.2.4

TS 3/4.2.4 is revised to reflect required actions consistent with the Core Operating Limits Methodology described in DPC-NE-2011-P.

The three-dimensional maneuvering analysis is used to confirm the acceptability of LOCA, DNB and centerline fuel melt limits. This analysis includes an allowance for the quadrant power tilt in the core. Calculated power distributions are increased by an amount corresponding to a 2% quadrant power tilt, equivalent to a QPTR of 1.02. Therefore, the resulting rod insertion limits, AFD limits, and f(ΔI) safety limits implicitly include an allowance for excore quadrant power tilt ratios up to the TS value of 1.02. Consequently, actions are only required if the QPTR exceeds 1.02.

Proposed Revision to TS 6.9.1.6, Core Operating Limits Report

TS 6.9.1.6.1.f and 6.9.1.6.1.g are revised by removing the specific list of parameters in items f and g that are currently stated as being included in the COLR. The specific items removed are F_Q^{RTP}, K(Z), and V(Z) for TS 3/4.2.2 and $F_{\Delta H}^{RTP}(X,Y)$ and Power Factor Multiplier, $PF_{\Delta H}$ for TS 3/4.2.3.

Technical Justification for Revision to TS 6.9.1.6

The list of parameters is removed because they are already identified in TS 3/4.2.2 and 3/4.2.3 as being located in the COLR and the text that will remain in TS 6.9.1.6.1.f and 6.9.1.6.1.g continues to identify the appropriate TS that have parameters located in the COLR. This change is made to be consistent with McGuire, Catawba, Robinson, and NUREG-1431.

ENCLOSURE

Attachment 3

Robinson Proposed Technical Specification Changes (Markup)

(continued)

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be maintained within the limits specified in the COLR. The maximum upper limit shall be ≤ +5.0 pcm/°F at ~~less than 50% RTP~~ or 0.0 pcm/°F at ~~50~~% RTP and above.

Replace with "70"

Replace with "hot zero power with a linear ramp to 0 pcm/°F at 70% RTP,"

APPLICABILITY: MODE 1 and MODE 2 with k_{eff} ≥ 1.0 for the upper MTC limit, MODES 1, 2, and 3 for the lower MTC limit.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. MTC not within upper limit.	A.1	Establish administrative withdrawal limits for control banks to maintain MTC within limit.	24 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 2 with k_{eff} < 1.0.	6 hours
C. MTC not within lower limit.	C.1	Be in MODE 4.	12 hours

Replace with "X,Y,Z"

$F_Q(\cancel{Z})$
3.2.1

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Heat Flux Hot Channel Factor ($F_Q(\cancel{Z})$)

LCO 3.2.1 $F_Q(Z)$, as approximated by $F_Q^V(Z)$, shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F_Q^V(Z)$ not within limit.	A.1 Reduce AFD target band limits to restore $F_Q^V(Z)$ to within limit.	15 minutes
	OR	
	A.2.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F_Q^V(Z)$ exceeds limit.	30 minutes
	AND	
	A.2.2 Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q^V(Z)$ exceeds limit.	72 hours
	AND	
	A.2.3 Reduce Overpower and Overtemperature ΔT trip setpoints ≥ 1% for each 1% $F_Q^V(Z)$ exceeds limit.	72 hours
	AND	

(continued)

Replace with Inserts #1a through #1f

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.4 Perform SR 3.2.1.1.	Prior to increasing THERMAL POWER above the limit of Required Action A.2.1
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

Delete

SURVEILLANCE REQUIREMENTS

----------NOTE----------
During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify $F_Q^V(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^V(Z)$ was last verified AND 31 EFPD thereafter

Delete

LCO 3.2.1 $F^M_Q(X,Y,Z)$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. $F^M_Q(X,Y,Z)$ not within steady state limit.	A.1 Reduce THERMAL POWER ≥ 1% RTP for each 1% $F^M_Q(X,Y,Z)$ exceeds limit.	15 minutes
	AND	
	A.2 Reduce Power Range Neutron Flux — High trip setpoints ≥ 1% for each 1% $F^M_Q(X,Y,Z)$ exceeds limit.	72 hours
	AND	
	A.3 Reduce Overpower ΔT trip setpoints ≥ 1% for each 1% $F^M_Q(X,Y,Z)$ exceeds limit.	72 hours
	AND	
	A.4 Perform SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.1.3.	Prior to increasing THERMAL POWER above the limit of Required Action A.1

(continued)

Insert #1a

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. $F^M_Q(X,Y,Z) > F^L_Q(X,Y,Z)^{OP}$	B.1 Reduce the Negative and Positive AFD limits as specified in the COLR to restore $F^M_Q(X,Y,Z)$ to within limit.	4 hours
	AND	
	B.2 Reduce THERMAL POWER as specified in the COLR to restore $F^M_Q(X,Y,Z)$ to within limit.	4 hours
	AND	
	B.3 Reduce Power Range Neutron Flux - High trip setpoints by ≥ 1% for each 1% THERMAL POWER level reduced in Required Action B.2.	72 hours
	AND	
	B.4 Reduce Overpower ΔT trip setpoints by ≥ 1% for each 1% THERMAL POWER level reduced in Required Action B.2.	72 hours
	AND	
	B.5 Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the limit of Required Action B.2
C. $F^M_Q(X,Y,Z) > F^L_Q(X,Y,Z)^{RPS}$	C.1 Reduce the OPΔT $f_2(\Delta I)$ breakpoints from the COLR limit by KSLOPE for each 1% $F^M_Q(X,Y,Z)$ exceeds limit.	72 hours

Insert #1b

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 2.	6 hours

Insert #1c

SURVEILLANCE REQUIREMENTS

----------NOTE----------

During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify $F^M_Q(X,Y,Z)$ is within steady state limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified <u>AND</u> 31 EFPD thereafter

(continued)

Insert #1d

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	----------NOTE---------- 1. Extrapolate $F^M_Q(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_Q(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_Q(X,Y,Z)_{EXTRAPOLATED} \geq F^L_Q(X,Y,Z)^{OP}{}_{EXTRAPOLATED}$, and $\frac{F^M_Q(X,Y,Z)_{EXTRAPOLATED}}{F^L_Q(X,Y,Z)^{OP}{}_{EXTRAPOLATED}} > \frac{F^M_Q(X,Y,Z)}{F^L_Q(X,Y,Z)^{OP}}$ then: a. Increase $F^M_Q(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$; or b. Repeat SR 3.2.1.2 prior to the time at which $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$ is extrapolated to not be met. 2. Extrapolation of $F^M_Q(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions. ---------- Verify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified AND 31 EFPD thereafter

(continued)

Insert #1e

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.3	----------NOTES---------- 1. Extrapolate $F^M_Q(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_Q(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_Q(X,Y,Z)_{EXTRAPOLATED} \geq F^L_Q(X,Y,Z)^{RPS}{}_{EXTRAPOLATED}$, and $\frac{F^M_Q(X,Y,Z)_{EXTRAPOLATED}}{F^L_Q(X,Y,Z)^{RPS}{}_{EXTRAPOLATED}} > \frac{F^M_Q(X,Y,Z)}{F^L_Q(X,Y,Z)^{RPS}}$ then: a. Increase $F^M_Q(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$; or b. Repeat SR 3.2.1.3 prior to the time at which $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$ is extrapolated to not be met. 2. Extrapolation of $F^M_Q(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions. ---------- Verify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified AND 31 EFPD thereafter

Insert #1f

Replace with: $F_{\Delta H}(X,Y)$

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($\cancel{F^N_{\Delta H}}$ $F^N_{\Delta H}$)

LCO 3.2.2 $F^N_{\Delta H}$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Required Actions A.2 and A.3 must be completed whenever Condition A is entered. ---------- $F^N_{\Delta H}$ not within limit.	A.1.1 Restore $F^N_{\Delta H}$ to within limit.	4 hours
	OR	
	A.1.2.1 Reduce THERMAL POWER to < 50% RTP.	4 hours
	AND	
	A.1.2.2 Reduce Power Range Neutron Flux-High trip setpoints to ≤ 55% RTP.	72 hours
	AND	
	A.2 Perform SR 3.2.2.1.	24 hours
	AND	
		(continued)

Replace with Inserts #3a through #3e

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 --------NOTE--------- THERMAL POWER does not have to be reduced to comply with this Required Action. ---------- Perform SR 3.2.2.1.	Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

Delete

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	----------NOTE---------- If $F^N_{\Delta H}$ is within limits and measurements indicate that $F^N_{\Delta H}$ is increasing with exposure then: a. Increase $F^V_Q(Z)$ by a factor of 1.02 and reverify $F^V_Q(Z)$ is within limits; or b. Perform SR 3.2.1.1 and SR 3.2.3.3 once per 7 EFPD until two successive measurements indicate $F^N_{\Delta H}$ is not increasing. ---------- Verify $F^N_{\Delta H}$ is within limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> 31 EFPD thereafter

Delete

LCO 3.2.2 $F^M_{\Delta H}(X,Y)$ shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Required Actions A.3.2.2 and A.4 must be completed whenever Condition A is entered. ---------- $F^M_{\Delta H}(X,Y)$ not within limit.	A.1 Reduce THERMAL POWER ≥ RRH% from RTP for each 1% $F^M_{\Delta H}(X,Y)$ exceeds limit.	2 hours
	AND	
	A.2.1 Restore $F^M_{\Delta H}(X,Y)$ to within limit for RTP.	8 hours
	OR	
	A.2.2 Reduce Power Range Neutron Flux — High trip setpoints ≥ RRH% for each 1% $F^M_{\Delta H}(X,Y)$ exceeds limit.	8 hours
	AND	
	A.3.1 Restore $F^M_{\Delta H}(X,Y)$ to within limit for RTP.	72 hours
	OR	
	A.3.2.1 Reduce OTΔT Trip Setpoint by ≥ TRH for each 1% $F^M_{\Delta H}(X,Y)$ exceeds limit.	72 hours
	AND	(continued)

Insert #3a

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3.2.2 Perform SR 3.2.2.1. AND A.4 ----------NOTE---------- THERMAL POWER does not have to be reduced to comply with this Required Action. ---------- Perform SR 3.2.2.1.	24 hours Prior to THERMAL POWER exceeding 50% RTP AND Prior to THERMAL POWER exceeding 75% RTP AND 24 hours after THERMAL POWER reaching ≥ 95% RTP
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

Insert #3b

SURVEILLANCE REQUIREMENTS

----------NOTE----------

During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify $F^{M}_{\Delta H}(X,Y)$ is within steady state limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP <u>AND</u> Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F^{M}_{\Delta H}(X,Y)$ was last verified <u>AND</u> 31 EFPD thereafter

(continued)

Insert #3c

SURVEILLANCE	FREQUENCY
SR 3.2.2.2 ----------NOTES---------- 1. Extrapolate $F^M_{\Delta H}(X,Y)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_{\Delta H}(X,Y)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_{\Delta H}(X,Y)_{EXTRAPOLATED} \geq F^L_{\Delta H}(X,Y)^{SURV}_{EXTRAPOLATED}$ and $\frac{F^M_{\Delta H}(X,Y)_{EXTRAPOLATED}}{F^L_{\Delta H}(X,Y)^{SURV}_{EXTRAPOLATED}} > \frac{F^M_{\Delta H}(X,Y)}{F^L_{\Delta H}(X,Y)^{SURV}}$ then: a. Increase $F^M_{\Delta H}(X,Y)$ by the appropriate factor specified in the COLR and reverify $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$; or b. Repeat SR 3.2.2.2 prior to the time at which $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$ is extrapolated to not be met. 2. Extrapolation of $F^M_{\Delta H}(X,Y)$ is not required for the initial flux map taken after reaching equilibrium conditions. ----------	
Verify $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND (continued)

Insert #3d

SR 3.2.2.2 (continued)	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F^{M}_{\Delta H}(X,Y)$ was last verified AND 31 EFPD thereafter

Insert #3e

3.2 POWER DISTRIBUTION LIMITS

Delete

3.2.3 AXIAL FLUX DIFFERENCE (AFD) ~~(PDC 3 Axial Offset Control Methodology)~~

LCO 3.2.3 The AFD:

a. Shall be maintained within the target band about the target flux difference. The allowable values of the target band are specified in the COLR.

----------NOTE----------
The AFD shall be considered outside the target band when two or more OPERABLE excore channels indicate AFD to be outside the target band.
---

b. May deviate outside the target band with THERMAL POWER < 90% RTP or 0.9 APL, whichever is less, but ≥ 50% RTP, provided AFD is within the acceptable operation limits and cumulative penalty deviation time is ≤ 1 hour during the previous 24 hours. The acceptable operation limits are specified in the COLR.

----------NOTES----------
1. Penalty deviation time shall be accumulated on the basis of a 1 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

2. The Allowable Power Level (APL) is the limitation placed on THERMAL POWER for the purposes of applying the AFD target flux and operational limit curves. The APL is as follows:

APL = minimum over Z of (100%)($F_Q^{RTP}(Z)$)(K(Z))/$F_Q^V(Z)$
---

c. May deviate outside the target band with THERMAL POWER < 50% RTP.

----------NOTE----------
Penalty deviation time shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside the target band.

Replace with Insert #4

APPLICABILITY: MODE 1 with THERMAL POWER > 15% RTP.

----------NOTE----------
A total of 16 hours of operation may be accumulated with AFD outside the target band without penalty deviation time during surveillance of power range channels in accordance with SR 3.3.1.6, provided AFD is maintained within acceptable operation limits.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less. AND AFD not within the target band.	A.1 Restore AFD to within target band.	15 minutes
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to < 90% RTP or 0.9 APL, whichever is less.	15 minutes

(continued)

Delete

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- Required Action C.1 and C.2 must be completed whenever Condition C is entered. ---------- THERMAL POWER < 90% RTP or 0.9 APL, whichever is less, and ≥ 50% RTP with cumulative penalty deviation time > 1 hour during the previous 24 hours. <u>OR</u> THERMAL POWER < 90% RTP or 0.9 APL, whichever is less, and ≥ 50% RTP with AFD not within the acceptable operation limits.	C.1 Reduce THERMAL POWER to < 50% RTP. <u>AND</u> C.2 Restore cumulative penalty deviation time to less than 1 hour.	30 minutes Prior to increasing THERMAL POWER to ≥ 50% RTP
D. ----------NOTE---------- Required Action D.1 must be completed whenever Condition D is entered. ---------- Required Action and associated Completion Time for Condition C not met.	D.1 Reduce THERMAL POWER to < 15% RTP.	9 hours

Delete

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD is within limits for each OPERABLE excore channel.	7 days
SR 3.2.3.2	----------NOTE---------- Assume logged values of AFD exist during the preceding time interval. ---------- Verify AFD is within limits and log AFD for each OPERABLE excore channel.	-----NOTE------ Only required to be performed if AFD monitor alarm is inoperable ---------- Once within 15 minutes and every 15 minutes thereafter when THERMAL POWER ≥ 90% RTP or 0.9 APL, whichever is less <u>AND</u> Once within 1 hour and every 1 hour thereafter when THERMAL POWER < 90% RTP or 0.9 APL, whichever is less

(continued)

Delete

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.3.3	----------NOTE---------- 1. The initial target flux difference after each refueling may be determined from design predictions. 2. The target flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ in accordance with SR 3.2.1.1. ---------- Determine, by measurement, the target flux difference of each OPERABLE excore channel.	Once within 31 EFPD after each refueling AND 31 EFPD thereafter

Delete

LCO 3.2.3 The AFD in % flux difference units shall be maintained within the limits specified in the COLR.

----------NOTE----------
The AFD shall be considered outside limits when two or more OPERABLE excore channels indicate AFD to be outside limits.

APPLICABILITY: MODE 1 with THERMAL POWER ≥ 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AFD not within limits.	A.1 Reduce THERMAL POWER to < 50% RTP.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AFD within limits for each OPERABLE excore channel.	7 days AND Once within 1 hour and every 1 hour thereafter with the AFD monitor alarm inoperable

Insert #4

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT RATIO (QPTR)

LCO 3.2.4 The QPTR shall be ≤ 1.02.

Add Insert #5

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. QPTR not within limit.	A.1	Reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > ~~1.00~~.	2 hours
	AND		
	A.2	~~Determine QPTR~~ and reduce THERMAL POWER ≥ 3% from RTP for each 1% of QPTR > ~~1.00~~.	Once per 12 hours
	AND		
	A.3	Perform SR 3.2.1.1 and SR 3.2.2.1.	24 hours AND Once per 7 days thereafter
	AND		
	A.4	Reevaluate safety analyses and confirm results remain valid for duration of operation under this condition.	Prior to increasing THERMAL POWER above the limit of Required Action A.1
	AND		
			(continued)

Replace with "1.02"

Replace with "Perform SR 3.2.4.1"

Replace with "1.02"

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.5	----------NOTE---------- Perform Required Action A.5 only after Required Action A.4 is completed. ---------- Normalize excore detectors to show zero QPTR.	Insert "more restrictive" Prior to increasing THERMAL POWER above the limit of Required Action A.1 or A.2
	AND		
	A.6	----------NOTE---------- Perform Required Action A.6 only after Required Action A.5 is completed. ---------- Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after reaching RTP OR Within 48 hours after increasing THERMAL POWER above the limit of Required Action A.1 or A.2
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 50% RTP.	4 hours Insert "more restrictive"

----------NOTE----------

Not applicable until calibration of the excore detectors is completed subsequent to refueling.

Insert #5

SURVEILLANCE REQUIREMENRTS (continued)

SURVELLANCE		FREQUENCY
SR 3.5.1.4	Verify boron concentration in each accumulator is ~~≥ 1950 ppm and ≤ 2400 ppm~~. Replace with "within the limits specified in the COLR"	31 days <u>AND</u> -----NOTE------ Only required to be performed for affected accumulators ---------- Once within 6 hours after each solution volume increase of ≥ 70 gallons that is not the result of addition from the refueling water storage tank
SR 3.5.1.5	Verify control power is removed from each accumulator isolation valve operator.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	----------NOTE---------- Only required to be performed when ambient air temperature is < 45ºF or > 100ºF. ---------- Verify RWST borated water temperature is ≥ 45ºF and ≤ 100ºF.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 300,000 gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ~~≥ 1950 ppm and ≤ 2400 ppm~~.	7 days

Replace with "within the limits specified in the COLR"

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 DELETED

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. Shutdown Margin (SDM) for Specification 3.1.1;

2. Moderator Temperature Coefficient limits for Specification 3.1.3;

3. Shutdown Bank Insertion Limits for Specification 3.1.5;

4. Control Bank Insertion Limits for Specification 3.1.6;

5. Heat Flux Hot Channel Factor ($F_Q(Z)$) limit for Specification 3.2.1;

6. Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) limit for Specification 3.2.2;

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. Axial Flux Difference (AFD) limits for Specification 3.2.3;

8. Boron Concentration limit for Specification 3.9.1;

9. Reactor Core Safety Limits Figure for Specification 2.1.1;

10. Overtemperature ΔT and Overpower ΔT setpoint parameter values for Specification 3.3.1; and

11. Reactor Coolant System pressure, temperature and flow Departure from Nucleate Boiling (DNB) limits for Specification 3.4.1.

Add Insert #6

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The approved version shall be identified in the COLR. These methods are those specifically described in the following documents:

1. Deleted

2. XN-NF-84-73(P), "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," approved version as specified in the COLR.

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

4. Deleted

5. XN-75-32(A), "Computational Procedure for Evaluating Rod Bow," approved version as specified in the COLR.

6. Deleted

7. Deleted

8. XN-NF-78-44(A), "Generic Control Rod Ejection Analysis," approved version as specified in the COLR.

9. XN-NF-621(A), "XNB Critical Heat Flux Correlation," approved version as specified in the COLR.

10. Deleted

(continued)

12. ECCS Accumulators boron concentration limits for Specification 3.5.1.

13. ECCS Refueling Water Storage Tank boron concentration limits for Specification 3.5.4.

Insert #6

No changes to this page. Included for information only.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements (continued)

11. XN-NF-82-06(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

12. Deleted

13. Deleted

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

14. Deleted

15. Deleted

16. ANF-88-054(P), “PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2,” approved version as specified in the COLR.

17. ANF-88-133 (P)(A), “Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU,” approved version as specified in the COLR.

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

19. EMF-92-081(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.

20. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

21. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results,” approved version as specified in the COLR.

22. EMF-96-029(P)(A), "Reactor Analysis System for PWRs," approved version as specified in the COLR.

23. EMF-92-116, “Generic Mechanical Design Criteria for PWR Fuel Designs,” approved version as specified in the COLR.

24. EMF-2103(P)(A), “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” approved version as specified in the COLR.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

25. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," approved version as specified in the COLR.

26. BAW-10240(P)(A), "Incorporation of M5 Properties in Framatome ANP Approved Methods," approved version as specified in the COLR.

27. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," approved version as specified in the COLR.

28. DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology," approved version as specified in the COLR.

29. DPC-NE-1008-P-A, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," as approved by NRC Safety Evaluation dated May 18, 2017.

30. DPC-NF-2010-A, "Nuclear Physics Methodology for Reload Design," as approved by NRC Safety Evaluation dated May 18, 2017.

31. DPC-NE-2011-P-A, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" as approved by NRC Safety Evaluation dated May 18, 2017.

Add Insert #7

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status,

(continued)

32. BAW-10231P-A, “COPERNIC Fuel Rod Design Computer Code,” approved version as specified in the COLR.

Insert #7

ENCLOSURE

Attachment 4

Harris Proposed Technical Specification Changes (Markup)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2948 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.29 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.31 ~~SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.~~

SITE BOUNDARY

1.32 For these Specifications, the SITE BOUNDARY shall be identical to the EXCLUSION AREA BOUNDARY defined above.

Replace with Insert #1

SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. All rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn. However, with all rod cluster assemblies verified as fully inserted by two independent means, it is not necessary to account for a stuck rod cluster assembly in the SDM calculation. With any rod cluster assembly not capable of being fully inserted, the reactivity worth of the rod cluster assembly must be accounted for in the determination of SHUTDOWN MARGIN, and

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.

Insert #1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN – MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~1770 pcm~~ for 3-loop operation.

APPLICABILITY: MODES 1 and 2*.

ACTION:

Replace with "the limit specified in the COLR"

With the SHUTDOWN MARGIN less than ~~1770 pcm~~, 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

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~1770 pcm~~:

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 above required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s);

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1 at the frequency specified in the Surveillance Frequency Control Program by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6;

c. Within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6; and

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors below, with the control banks at the maximum insertion limit of Specification 3.1.3.6:

*See Special Test Exceptions Specification 3.10.1.

LIMITING CONDITION FOR OPERATION

3.1.2.5 As a minimum, one of the following borated water sources shall be OPERABLE:

a. The boric acid tank with:

1. A minimum contained borated water volume of 7150 gallons which is ensured by maintaining indicated level of greater than or equal to 23%,
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 106,000 gallons, which is equivalent to 12% indicated level,
2. A boron concentration ~~of between 2400 and 2600 ppm~~, and
3. A minimum solution temperature of 40°F.

Replace with "within the limits specified in the COLR"

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required borated water source shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying the boron concentration of the water,
2. Verifying the contained borated water volume, and
3. Verifying the boric acid tank solution temperature when it is the source of borated water.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature when it is the source of borated water and the outside air temperature is less than 40°F.

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.

Replace with "within the limits specified in the COLR"

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.

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.

Replace with "the limits"

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

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at the frequency specified in the Surveillance Frequency Control Program when the AFD Monitor Alarm is OPERABLE, and

b. Monitoring and logging the indicated AFD for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AFD Monitor Alarm is inoperable. The logged values of the indicated AFD shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

~~4.2.1.3 The target AFD of each OPERABLE excore channel shall be determined by excore measurement at the frequency specified in the Surveillance Frequency Control Program in conjunction with the requirements of Specification 4.2.2.2. The target AFD may be updated between measurements by adding the most recently measured value and the change in the predicted value since the measurement. The provisions of Specification 4.0.4 are not applicable.~~

Delete

Replace "Z" with "X,Y,Z"

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - F_Q~~(Z)~~

LIMITING CONDITION FOR OPERATION

3.2.2 ~~$F_Q(Z)$~~ shall be within the limits specified in the COLR.

Replace with: $F_Q^M(X,Y,Z)$

APPLICABILITY: MODE 1.

ACTION:

~~With $F_Q(Z)$ exceeding its limit:~~

~~a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 8 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit. Otherwise, be in at least MODE 2 within 6 hours.~~

~~b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.~~

Replace with Inserts #2a and 2b

a. With specification 4.2.2.2.c.1 not being satisfied ($F_Q^M(X,Y,Z)$ exceeding its steady-state limit):

1. Reduce THERMAL POWER ≥ 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 15 minutes.

2. Reduce the Power Range Neutron Flux-High Trip setpoints by ≥ 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 72 hours.

3. Reduce the Overpower ΔT trip setpoints by ≥ 1% for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 72 hours.

4. Prior to increasing THERMAL POWER above the maximum allowable power level from action 3.2.2.a.1, demonstrate through incore flux mapping that $F_Q\ (X,Y,Z)$ is within its steady-state limit.

5. If the required Actions and associated completion times are not met, be in MODE 2 within 6 hours.

b. With specification 4.2.2.2.c.2 not being satisfied ($F_Q^M(X,Y,Z)$ exceeding its transient Operational limit, $F_Q^L(X,Y,Z)^{OP}$):

1. Reduce AFD limits by the amount specified in the COLR to restore $F_Q\ (X,Y,Z)$ to within its limits within 4 hours.

2. Reduce THERMAL POWER by the amount specified in the COLR to restore $F_Q\ (X,Y,Z)$ to within its limits within 4 hours.

3. Reduce Power Range Neutron Flux – High trip setpoints ≥ 1% for each 1% that the THERMAL POWER level is reduced within 72 hours.

4. Reduce the Overpower ΔT trip setpoints by ≥ 1% for each 1% that the THERMAL POWER level is reduced within 72 hours.

5. Prior to increasing THERMAL POWER above the maximum allowable power level from action 3.2.2.b.2, demonstrate through incore flux mapping that $F_Q\ (X,Y,Z)$ is within its transient operational limit, $F_Q^L(X,Y,Z)^{OP}$.

6. If the required Actions and associated completion times are not met, be in MODE 2 within 6 hours.

Insert #2a

a. With specification 4.2.2.2.c.3 not being satisfied ($F_Q^M(X,Y,Z)$ exceeding its transient Reactor Protection System limit, $F_Q^L(X,Y,Z)^{RPS}$):

1. Reduce Overpower ΔT $f_2(\Delta I)$ breakpoints by KSLOPE for each 1% $F_Q^M(X,Y,Z)$ exceeds the limit within 72 hours.

2. If the required Actions and associated completion times are not met, be in MODE 2 within 6 hours.

Insert #2b

** Replace with: $F_Q^M(X,Y,Z)$

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

replace with "limits"

4.2.2.2 ~~$F_Q(Z)$~~ shall be evaluated to determine if it is within its ~~limit~~ by:

**

a. Using the movable incore detectors to obtain a power distribution map at any THERMAL POWER greater than 5% of RATED THERMAL POWER.

**

b. Increasing the ~~measured $F_Q(Z)$ component~~ of the power distribution map by 3% to account for manufacturing tolerances and further increasing the value by 5% to account for measurement uncertainties. Verify the requirements of Specification 3.2.2 are satisfied.

Replace with "relationships"

c. Satisfying the following ~~relationship~~:

Replace with Insert #3

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{P \times V(Z)} \quad \text{for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{F_Q^{RTP} \times K(Z)}{V(Z) \times 0.5} \quad \text{for } P \leq 0.5$$

where $F_Q^M(Z)$ is the measured $F_Q(Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty, F_Q^{RTP} is the F_Q limit, K(Z) is the normalized $F_Q(Z)$ as a function of core height, P is the fraction of RATED THERMAL POWER, and V(Z) is the function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(Z), and V(Z) are specified in the COLR.

d. Measuring ~~$F_Q^M(Z)$~~ according to the following schedule:

**

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which ~~$F_Q(Z)$~~ was last determined,* or

**

2. At the frequency specified in the Surveillance Frequency Control Program, whichever occurs first.

* During power escalation at the beginning of each cycle, power level may be increased until a power level for extended operation has been achieved and a power distribution map obtained.

1. <u>Steady-state Limit:</u>

$$F_Q^M(X,Y,Z) \leq \frac{F_Q^{RTP}}{P} K(Z) * K(BU) \qquad \text{for P > 0.5}$$

$$F_Q^M(X,Y,Z) \leq \frac{F_Q^{RTP}}{0.5} K(Z) * K(BU) \qquad \text{for P ≤ 0.5}$$

where $F_Q^M(X,Y,Z)$ is the measured $F_Q(X,Y,Z)$ increased by the allowances for manufacturing tolerances and measurement uncertainty. F_Q^{RTP} is the $F_Q(X,Y,Z)$ limit at RATED THERMAL POWER provided in the COLR. K(Z) is the normalized $F_Q(X,Y,Z)$ as a function of core height and P is the fraction of RATED THERMAL POWER. K(BU) accounts for degradation of thermal conductivity. F_Q^{RTP}, K(Z) and K(BU) are specified in the COLR.

2. <u>Transient Operational Limit:</u>

$$F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{OP}$$

$$F_Q^L(X,Y,Z)^{OP} = F_Q^D(X,Y,Z) * M_Q(X,Y,Z)$$

where $F_Q^L(X,Y,Z)^{OP}$ is the cycle dependent maximum allowable design peaking factor which ensures that the $F_Q(X,Y,Z)$ limit will be preserved for operation within the LCO limits. $F_Q^L(X,Y,Z)^{OP}$ includes allowances for calculational and measurement uncertainties.

$F_Q^D(X,Y,Z)$ is the design power distribution for $F_Q(X,Y,Z)$ provided in the COLR.

$M_Q(X,Y,Z)$ is the margin remaining in core location X,Y,Z to the LOCA limit in the transient power distribution and is provided in the COLR for normal operating conditions and power escalation testing during startup operations.

3. <u>Transient Reactor Protection System Limit:</u>

$$F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{RPS}$$

$$F_Q^L(X,Y,Z)^{RPS} = F_Q^D(X,Y,Z) * M_C(X,Y,Z)$$

where $F_Q^L(X,Y,Z)^{RPS}$ is the cycle dependent maximum allowable design peaking factor which ensures that the centerline fuel melt limit will be preserved for operation within the LCO limits. $F_Q^L(X,Y,Z)^{RPS}$ includes allowances for calculational and measurement uncertainties.

$M_C(X,Y,Z)$ is the margin remaining to the centerline fuel melt limit in core location X,Y,Z from the transient power distribution and is provided in the COLR for normal operating conditions and power escalation testing during startup operations.

Insert #3

SURVEILLANCE REQUIREMENTS (Continued)

e. With measurements indicating $F_{\Delta H}^{N}$ has increased since the previous determination of $F_{Q}^{M}(Z)$ either of the following actions shall be taken:

1) $F_{Q}^{M}(Z)$ shall be increased by 2% over that specified in Specification 4.2.2.2c, or

2) $F_{Q}^{M}(Z)$ shall be measured and a target AFD reestablished at least once per 7 Effective Full Power Days until two successive maps indicate that $F_{\Delta H}^{N}$ is not increasing.

f. With the relationships specified in Specification 4.2.2.2c above not being satisfied:

1) Calculate the percent $F_{Q}(Z)$ exceeds its limit by the following expression:

$$\left\{ \left[\text{maximum} \left[\frac{F_{Q}^{M}(Z) \times V(Z)}{\frac{F_{Q}^{RTP}}{P} \times K(Z)} \right] \right] -1 \right\} \times 100 \text{ for } P \geq 0.5$$

$$\left\{ \left[\text{maximum} \left[\frac{F_{Q}^{M}(Z) \times V(Z)}{\frac{F_{Q}^{RTP}}{0.5} \times K(Z)} \right] \right] -1 \right\} \times 100 \text{ for } P < 0.5$$

2) One of the following actions shall be taken:

a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the AFD limits specified in the COLR by 1% AFD for each percent $F_{Q}(Z)$ exceeds its limits as determined in Specification 4.2.2.2f.1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

b) Comply with the requirements of Specification 3.2.2 for $F_{Q}(Z)$ exceeding its limit by the percent calculated above.

Replace with Insert #4

e. Extrapolating $F_Q^M(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement.* If $F_Q^M(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates:

$$F_Q^M(X,Y,Z)_{EXTRAPOLATED} \geq F_Q^L(X,Y,Z)_{EXTRAPOLATED}^{OP},$$

and

$$\frac{F_Q^M(X,Y,Z)_{EXTRAPOLATED}}{F_Q^L(X,Y,Z)_{EXTRAPOLATED}^{OP}} > \frac{F_Q^M(X,Y,Z)}{F_Q^L(X,Y,Z)^{OP}}$$

then:

1. Increase $F_Q^M(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{OP}$; or
2. Repeat Surveillance Requirement 4.2.2.2.c.2 prior to the time at which $F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{OP}$ is extrapolated to not be met.

f. Extrapolating $F_Q^M(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement.* If $F_Q^M(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates:

$$F_Q^M(X,Y,Z)_{EXTRAPOLATED} \geq F_Q^L(X,Y,Z)_{EXTRAPOLATED}^{RPS},$$

and

$$\frac{F_Q^M(X,Y,Z)_{EXTRAPOLATED}}{F_Q^L(X,Y,Z)_{EXTRAPOLATED}^{RPS}} > \frac{F_Q^M(X,Y,Z)}{F_Q^L(X,Y,Z)^{RPS}}$$

then:

1. Increase $F_Q^M(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{RPS}$; or
2. Repeat Surveillance Requirement 4.2.2.2.c.3 prior to the time at which $F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{RPS}$ is extrapolated to not be met.

---

* Extrapolation of $F_Q^M(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions.

Insert #4

g. The limits specified in Specifications 4.2.2.2c, 4.2.2.2e, and 4.2.2.2f above are not applicable in the ~~following core plane regions:~~

~~1. Lower core region from 0 to 15%, inclusive.~~
~~2. Upper core region from 85 to 100%, inclusive.~~

Replace with "core plane regions specified in the BASES"

Replace "Z" with "X,Y,Z"

4.2.2.3 When $F_Q(\cancel{Z})$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(\cancel{Z})$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

(Pages 3/4 2-7c and 3/4 2-7d have been deleted)

FIGURE 3.2-2

K(Z) - THE NORMALIZED $F_Q(\cancel{Z})$ AS A FUNCTION OF CORE HEIGHT

Replace "Z" with "X,Y,Z"

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

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

Insert: - $F_{\Delta H}(X,Y)$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}$ shall be within the limits specified in the COLR.

Replace with: $F^{M}_{\Delta H}(X,Y)$

APPLICABILITY: MODE 1.

ACTION:

a. With $F_{\Delta H}$ outside the limits given in 3.2.3:

1. Within 4 hours either:

   a) Restore $F_{\Delta H}$ to within the limits given in 3.2.3, or

   b) Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and, reduce Power Range Neutron Flux Trip setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

Replace with Insert #5

1. Within 2 hours reduce THERMAL POWER ≥ RRH%* from RATED THERMAL POWER for each 1% $F_{\Delta H}^{M}(X,Y)$ exceeds limit.

2. Within 8 hours either:
   a. Restore $F_{\Delta H}^{M}(X,Y)$ to within the limit for RATED THERMAL POWER, or
   b. Reduce Power Range Neutron Flux - High trip setpoints ≥ RRH%* for each 1% $F_{\Delta H}^{M}(X,Y)$ exceeds limit.

3. Within 72 hours either:
   a. Restore $F_{\Delta H}^{M}(X,Y)$ to within limit for RATED THERMAL POWER, or
   b. Reduce Overtemperature ΔT Trip Setpoints by ≥ TRH* for each 1% $F_{\Delta H}^{M}(X,Y)$ exceeds limit.

---

* RRH% and TRH are specified in the COLR.

Insert #5

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

Replace with: $F^{M}_{\Delta H}(X,Y)$

Replace with "4"

~~2~~. Within 24 hours of $\cancel{F_{\Delta H}}$ initially being outside the limits of 3.2.3, verify through incore flux mapping that $\cancel{F_{\Delta H}}$ is within the limits given in 3.2.3.

Replace with "5"

~~3~~. Subsequent POWER OPERATION may proceed provided that $\cancel{F_{\Delta H}}$ is demonstrated through incore flux mapping to be within acceptable limits prior to exceeding the following THERMAL POWER levels*:

a) 50% RATED THERMAL POWER
b) 75% RATED THERMAL POWER
c) Within 24 hours of attaining greater than or equal to 95% RATED THERMAL POWER

b. With the requirements of ACTION 3.2.3.a not met, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 6 hours.

* THERMAL POWER does not have to be reduced to comply with this ACTION.

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

Replace with: $F^{M}_{\Delta H}(X,Y)$

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 ~~$F_{\Delta H}$~~ shall be ~~determined to be within acceptable limits~~:

Add Insert #6

~~a.~~ Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

~~b.~~ At the frequency specified in the Surveillance Frequency Control Program thereafter.

Replace with "evaluated to determine if it is within its limits by:"

Increase indent and replace with "1"

Increase indent and replace with "2"

a. Verifying $F_{\Delta H}^{M}(X,Y)$ is within the steady state limit.

b. Verifying $F_{\Delta H}^{M}(X,Y)$ is within the transient Surveillance limit, $F_{\Delta H}^{L}(X,Y)^{SURV}$

c. Extrapolating $F_{\Delta H}^{M}(X,Y)$ using at least two measurements to 31 EFPD beyond the most recent measurement.* If $F_{\Delta H}^{M}(X,Y)$ is within limits and the 31 EFPD extrapolation indicates:

$$F_{\Delta H}^{M}(X,Y)_{EXTRAPOLATED} \geq F_{\Delta H}^{L}(X,Y)_{EXTRAPOLATED}^{SURV},$$

and

$$\frac{F_{\Delta H}^{M}(X,Y)_{EXTRAPOLATED}}{F_{\Delta H}^{L}(X,Y)_{EXTRAPOLATED}^{SURV}} > \frac{F_{\Delta H}^{M}(X,Y)}{F_{\Delta H}^{L}(X,Y)^{SURV}}$$

then:

1. Increase $F_{\Delta H}^{M}(X,Y)$ by the appropriate factor specified in the COLR and reverify $F_{\Delta H}^{M}(X,Y) \leq F_{\Delta H}^{L}(X,Y)^{SURV}$; or
2. Repeat Surveillance Requirement 4.2.3.2.b prior to the time at which $F_{\Delta H}^{M}(X,Y) \leq F_{\Delta H}^{L}(X,Y)^{SURV}$ is extrapolated to not be met.*

---

* Extrapolation of $F_{\Delta H}^{M}(X,Y)$ is not required for the initial flux map taken after reaching equilibrium conditions.

d. Measuring $F_{\Delta H}^{M}(X,Y)$ according to the following schedule:

Insert #6

3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but less than or equal to 1.09:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Within 2 hours either:

   a) Reduce the QUADRANT POWER TILT RATIO to within its limit, or

   b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of ~~1~~ and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

Replace with "1.02"

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

Insert Amendment No.

*See Special Test Exceptions Specification 3.10.2.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of ~~1.00~~, within 30 minutes;

Replace with "1.02"

3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours 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; and

4. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Calculate the QUADRANT POWER TILT RATIO at least once per hour until either:

   a) The QUADRANT POWER TILT RATIO is reduced to within its limit, or

   b) THERMAL POWER is reduced to less than 50% of RATED THERMAL POWER.

Insert Amendment No.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open with power supply circuit breaker open,

b. A contained borated water volume of between 66 and 96% indicated level,

c. A boron concentration ~~of between 2400 and 2600 ppm~~, and

d. A nitrogen cover-pressure of between 585 and 665 psig.

Replace with "within the limits specified in the COLR"

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve or boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With one accumulator inoperable due to boron concentration not within limits, restore the boron concentration within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2. Verifying that each accumulator isolation valve is open.

*RCS pressure above 1000 psig.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

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

b. A boron concentration ~~of between 2400 and 2600 ppm of boron~~,

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

d. A maximum solution temperature of 125°F.

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

Replace with "within the limits specified in the COLR"

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 125°F.

* Except that while performing surveillance 4.4.6.2.2, the tank must be returned to OPERABLE status within 12 hours.

6.9.1.6 CORE OPERATING LIMITS REPORT

Insert "3/4.1.1.1 and"

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.

Delete

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

Add Insert #7

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).

i. Shutdown and Operating Boric Acid Tank and Refueling Water Storage Tank boron concentration limits for Specification 3/4.1.2.5 and 3/4.1.2.6.

j. ECCS Accumulators and Refueling Water Storage Tank boron concentration limits for Specification 3/4.5.1 and 3/4.5.4.

Insert #7

No changes to this page. Included for information only.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," approved version as specified in the COLR.

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

e. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," 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, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

f. ANP-3011(P), "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," Revision 1, as approved by NRC Safety Evaluation dated May 30, 2012.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

g. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," approved version as specified in the COLR.

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

No changes to this page. Included for information only.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

h. ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).

i. EMF-92-081(P)(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," 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).

j. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

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

k. BAW-10240(P)(A), "Incorporation of M5 Properties in Framatome ANP Approved Methods."

(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).

l. EMF-96-029(P)(A), "Reactor Analysis Systems for PWRs," 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).

m. EMF-2328(P)(A) PWR Small Break LOCA Evaluation Model, S-RELAP5 Based, approved version as specified in the COLR.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

n. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors", approved version as specified in the COLR.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

(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).

o. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

Add Insert #8

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

p. DPC-NE-2005-P-A, “Thermal-Hydraulic Statistical Core Design Methodology,” approved version as specified in the COLR.

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

q. DPC-NE-1008-P-A, “Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors,” as approved by NRC Safety Evaluation dated May 18, 2017.

(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).

r. DPC-NF-2010-A, “Nuclear Physics Methodology for Reload Design,” as approved by NRC Safety Evaluation dated May 18, 2017.

(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, and 3.9.1 – Boron Concentration).

s. DPC-NE-2011-P-A, “Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors” as approved by NRC Safety Evaluation dated May 18, 2017.

(Methodology for Specification 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).

BAW-10231P-A, “COPERNIC Fuel Rod Design Computer Code,” approved version as specified in the COLR.

Insert #8