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Serial: RA-18-0016 10 CFR 50.90
June 5, 2018

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: RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING TECHNICAL SPECIFICATION CHANGES TO SUPPORT SELF-PERFORMANCE OF CORE RELOAD DESIGN AND SAFETY ANALYSES

REFERENCES:

1. Duke Energy letter, *Technical Specification Changes to Support Self-Performance of Core Reload Design and Safety Analyses,* dated October 19, 2017 (ADAMS Accession No. ML17292A040)
2. NRC email, *Harris/Robinson RAIs – Change Technical Specifications to Support Performance of Core Reload Design and Safety Analyses (L-2017-LLA-0356)*, dated April 25, 2018 (ADAMS Accession No. ML18116A105)

Ladies and Gentlemen:

In Reference 1, Duke Energy Progress, LLC (formerly referred to as Duke Energy Progress, Inc.), referred to henceforth as “Duke Energy,” submitted a request for an amendment to the Technical Specifications (TS) to support self-performance of core reload design and safety analyses at the Shearon Harris Nuclear Power Plant, Unit 1 (HNP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (RNP). In Reference 2, the NRC requested additional information regarding Reference 1.

Attachment 3 provides Duke Energy’s response to the Reference 2 RAIs. Attachment 3 contains information that is proprietary to Duke Energy. In accordance with 10 CFR 2.390, Duke Energy requests that Attachment 3 be withheld from public disclosure. An affidavit is included (Attachment 1) attesting to the proprietary nature of Attachment 3. A non-proprietary version of Attachment 3 is included in Attachment 2.

Attachments 4 and 5 provide updated markups of the RNP and HNP TS pages, respectively, that are affected by the RAI responses in this letter. All TS markup pages and associated inserts in Attachments 4 and 5 supersede the respective TS markup pages and inserts from Reference 1, except for RNP TS Pages 3.3-18 and 3.3-19, which are new markup pages that were not included in Reference 1.

This submittal contains no new regulatory commitments. Duke Energy is notifying the states of North Carolina and South Carolina 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 June 5, 2018.

Sincerely,

Joseph Donahue
Vice President – Nuclear Engineering

JBD

Attachments:
1. Affidavit of Joseph Donahue
2. Responses to the NRC Request for Additional Information (Redacted)
3. Responses to the NRC Request for Additional Information (Proprietary)
4. Robinson Proposed Technical Specification Changes (Markup)
5. Harris Proposed Technical Specification Changes (Markup)

cc: (all with Attachments unless otherwise noted)

C. Haney, Regional Administrator USNRC Region II
J. Zeiler, USNRC Senior Resident Inspector – HNP
J. Rotton, USNRC Senior Resident Inspector – RNP
M. C. Barillas, NRR Project Manager – HNP
D. J. Galvin, NRR Project Manager – RNP
W. L. Cox, III, Section Chief, NC DHSR (Without Attachment 3)
S. E. Jenkins, Manager, Radioactive and Infectious Waste Management Section (SC) (Without Attachment 3)
A. Wilson, Attorney General (SC) (Without Attachment 3)
A. Gantt, Chief, Bureau of Radiological Health (SC) (Without Attachment 3)

Attachment 1

Affidavit of Joseph Donahue

AFFIDAVIT of Joseph Donahue

1. I am Vice President of Nuclear Engineering, Duke Energy Corporation, and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing and am authorized to apply for its withholding on behalf of Duke Energy.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.390 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke Energy's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke Energy in designating information as proprietary or confidential. I am familiar with the Duke Energy information contained in Attachment 3 to Duke Energy RAI response letter RA-18-0016 regarding application to revise technical specifications to support self-performance of core reload design and safety analyses.

4. Pursuant to the provisions of paragraph (b) (4) of 10 CFR 2.390, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke Energy and has been held in confidence by Duke Energy and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke Energy. Information is held in confidence if it falls in one or more of the following categories.

      (a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by a vendor or consultant, without a license from Duke Energy, would constitute a competitive economic advantage to that vendor or consultant.

      (b) The information requested to be withheld consist of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage for example by requiring the vendor or consultant to perform test measurements, and process and analyze the measured test data.

      (c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation assurance of quality or licensing of a similar product.

      (d) The information requested to be withheld reveals cost or price information, production capacities, budget levels or commercial strategies of Duke Energy or its customers or suppliers.

(e) The information requested to be withheld reveals aspects of the Duke Energy funded (either wholly or as part of a consortium) development plans or programs of commercial value to Duke Energy.

(f) The information requested to be withheld consists of patentable ideas.

The information in this submittal is held in confidence for the reasons set forth in paragraphs 4(ii)(a) and 4(ii)(c) above. Rationale for this declaration is the use of this information by Duke Energy provides a competitive advantage to Duke Energy over vendors and consultants, its public disclosure would diminish the information's marketability, and its use by a vendor or consultant would reduce their expenses to duplicate similar information. The information consists of analysis methodology details, analysis results, and supporting data, relative to a method of analysis that provides a competitive advantage to Duke Energy.

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.390, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld is that which is marked in Attachment 3 to Duke Energy RAI response letter RA-18-0016 regarding application to revise technical specifications to support self-performance of core reload design and safety analyses. This information enables Duke Energy to:

(a) Support license amendment requests for its Harris and Robinson reactors.

(b) Support reload design calculations for Harris and Robinson reactor cores.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke Energy.

(a) Duke Energy uses this information to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke Energy can sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke Energy.

5. Public disclosure of this information is likely to cause harm to Duke Energy because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke Energy to recoup a portion of its expenditures or benefit from the sale of the information.

Joseph Donahue affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

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

Executed on June 5, 2018.

Joseph W Donahue

Joseph Donahue

Attachment 2

Responses to the NRC Request for Additional Information (Redacted)

Note: Text that is within brackets with an "a,c" superscript is proprietary to Duke Energy and has been removed.

SRXB-RAI-1:

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.

Robinson was not licensed to the current 10 CFR 50, Appendix A, GDC. Per the Robinson 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."

LAR Enclosure Attachments 1 and 2 describe the changes to the Robinson and Harris TSs, respectively, for the heat flux hot channel factor F_Q. The proposed TSs will use the designation $F_Q^M(X, Y, Z)$ for the measured F_Q and the designation $F_Q^L(X, Y, Z)$ for the limit F_Q. LAR Enclosure Attachments 1 and 2 also introduce the use of a variable "KSLOPE" to ensure positive margin exists to the centerline fuel melt (CFM) limit during transient conditions when core peaking may be greater than the design value. The variable KSLOPE is not currently defined or used in DPC-NE-2011, the Harris or Robinson TS, or the Harris or Robinson core operating limits report (COLR). DPC-NE-2011, Section 6.4.3, describes an adjustment for the CFM limits in one sentence but does not designate this adjustment using the variable KSLOPE:

> If $F_Q^M(X, Y, Z)$ exceeds $F_Q^L(X, Y, Z)^{RPS}$ (CFM limits), then a reduction is made to the OT [over temperature] ΔT trip setpoints, or the $f_1(\Delta I)$ or $f_2(\Delta I)$ breakpoints are adjusted.

LAR Enclosure Attachment 1 (Page 2 of 14) states that:

> "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."

A similar statement is also in LAR Enclosure Attachment 2.

Also, DPC-NE-2011, Section 6.1, defines the factor K(Z) as the "normalized F_Q as a function of core height" and the factor K(BU) as representing "the normalized burnup dependency" of the F_Q limit at rated thermal power. However, DPC-NE-2011 does not specify how these factors are determined or how whether they are sensitive to fuel types applicable to Harris and Robinson.

The NRC staff has identified the following questions to ensure the appropriate considerations are made in the determination of KSLOPE and that the CFM and loss of coolant accident limits are protected.

a. Identify the steps taken and the parameters considered in determining KSLOPE.

b. Explain how the normalized $F_Q(X, Y, Z)$ as a function of core height (K(Z)) is calculated. Provide reason(s) for any assumptions in calculating this factor.

c. Explain how the normalized $F_Q(X, Y, Z)$ as a function of burnup (K(BU)) is calculated. Provide reason (s) for any assumptions in calculating this factor.

d. Address fuel type dependencies as applicable to Harris and Robinson in your response to the above questions.

SRXB-RAI-1 Response:

Part (a):

If the measured F_Q exceeds the centerline fuel melt limit, $F_Q^L(X, Y, Z)^{RPS}$, the overpower ΔT $f_2(\Delta I)$ trip reset function breakpoints from the Core Operating Limits Report (COLR) are reduced by KSLOPE for each 1% that the measured F_Q exceeds its limit. The variable KSLOPE is the derivative that relates the change in centerline fuel melt peaking margin to a change in ΔI. This derivative has the units of $\%F_Q/\%\Delta I$. The process used to calculate this variable is described below.

The initial step requires generation of three-dimensional core power distributions. These power distributions are generated with SIMULATE-3 **[**

]^{a,c}

An example CFM margin plot is shown in Figure RAI-1-1 at 117.5% full power (FP). Using this information, conservative **[**

]^{a,c}

The variable KSLOPE is specified in the COLR. An illustration of its use follows. If the measured F_Q exceeds the centerline fuel melt limit by 2%, then the positive and negative $f_2(\Delta I)$ trip reset function breakpoints would have to be reduced by greater than or equal to [

][a,c]

Figure RAI-1-1
Example 117.5% FP CFM Margins

a, c

Part (b) and (c):

The generation of the loss of coolant accident (LOCA) analysis limits for the fuel designs in operation in Duke Energy's reactors is the responsibility of the fuel vendor performing the LOCA analysis. These limits are determined using a NRC-approved methodology, and include a reference F_Q value which may be modified by axial and burnup dependent functions. The reference F_Q, including any axial and burnup dependencies, limits the amount of stored energy assumed at the initiation of a LOCA and the peak location heat generation rate to acceptable levels that prevent peak clad temperatures from exceeding design limits. The LOCA F_Q limits specified by the fuel vendor are verified analytically using the methodology defined in DPC-NE-2011, and by periodic power distribution measurements by satisfying the following relationship:

$$F_Q^M(X,Y,Z) \leq (F_Q^{RTP}/P) \times K(Z) \times K(BU) \qquad (P > 0.5)$$

where, $F_Q^M(X,Y,Z)$ is the measured F_Q
F_Q^{RTP} is the F_Q limit at rated thermal power (RTP) conditions provided in the COLR
P is the fraction of rated thermal power (Thermal Power/RTP)
K(Z) is the normalized F_Q^{RTP} as a function of core height provided in the COLR
K(BU) is the normalized F_Q^{RTP} as a function of burnup provided in the COLR

In general, the K(Z) function accounts for the lower region of the reactor core being less limiting than the top because it quenches faster. This allows for a higher F_Q to be tolerated in this region of the core without impacting peak clad temperature margin. A burnup dependent function may be required to limit the initial amount of stored energy in the fuel to account for phenomena such as fuel thermal conductivity degradation (TCD). The need for either a K(Z) or K(BU) function is dependent upon the fuel design, the nuclear steam supply system (NSSS) design, the thermal output of the reactor core, and the modeling assumptions specified in the NRC-licensed LOCA methodology. Any assumptions made in the development of the K(Z) and K(BU) functions are based on the licensed method under the control of the fuel vendor performing the LOCA analysis. Configuration control is maintained by the transmittal of F_Q^{RTP}, and K(Z) and K(BU), if applicable, from the fuel vendor to Duke Energy for each reload core. Example K(Z) and K(BU) functions are presented Tables RAI-1-1 and RAI-1-2 to illustrate the calculation of each function.

Table RAI-1-1
Example K(Z) Function

Axial Elevation (ft.)	F_Q Limit	K(Z) Note 1
0	2.60	1.0
4	2.60	1.0
> 4	2.50	0.9615
12	2.50	0.9615

Note 1: Assumes $F_Q^{RTP} = 2.60$

Table RAI-1-2
Example K(BU) Function

Burnup (MWD/MTU)	F_Q Limit	K(BU) Note 2
0	2.60	1.0
35,000	2.60	1.0
55,000	2.34	0.9
62,000	2.08	0.8

Note 2: Assumes $F_Q^{RTP} = 2.60$

Part (d):

Fuel type dependencies applicable to the calculation of KSLOPE, and K(Z) and K(BU) are addressed separately below.

KSLOPE:

Cycle-specific CFM margin calculations are performed to confirm the acceptability of the centerline fuel melt design criteria using the methodology described in Reference RAI-1-1. These same calculations are used to confirm the acceptability of KSLOPE. CFM linear heat generation limits are generated for each unique fuel design present in the reactor core using a NRC-approved fuel rod mechanical analysis methodology and fuel performance code. The core model used to generate power distributions for CFM margin comparisons also explicitly models the characteristics of each unique fuel design in the reactor core. Because both the predictions and limits account for the characteristics of the fuel in operation, the results from the CFM margin calculations implicitly include any fuel type dependencies, thus preserving the fidelity of the analysis used to confirm the acceptability of KSLOPE, or used as the basis to calculate a new value.

K(Z) and K(BU):

LOCA limits are typically developed for each unique fuel design present in the reactor. They consist of a reference value, F_Q^{RTP}, which may be modified by axial and burnup dependent functions to limit the initial amount of stored energy to preclude peak clad temperature limits from being exceeded. The need for axial and burnup dependent functions, designated as K(Z) and K(BU), is dependent upon the characteristics of fuel design, the thermal output of the reactor, and the response of the NSSS during a LOCA. For the remainder of this discussion, the term LOCA limits refers to the aggregate of the reference LOCA value and any K(Z) or K(BU) functions. Because LOCA limits are dependent on the NSSS design, the fuel design and the thermal output of the reactor core, there are separate LOCA analysis limits applicable to Harris and Robinson. If a new fuel design is introduced to one of these reactors, the acceptability of the existing LOCA limits would be determined, and if needed, new LOCA limits defined for the new fuel design. The cycle-specific confirmation of these limits would either use bounding limits that envelope all fuel designs, or limits which are mapped to the appropriate core locations of each unique fuel design in the reactor core. The core model used in reload design calculations also models each unique fuel design present in the reactor core. As a result, the power distributions used to confirm the acceptability of the LOCA limits, and the LOCA limits themselves, include the impact of the fuel designs in operation. For these reasons, the cycle-specific reload analyses performed to establish Technical Specification limits (i.e. AFD and rod insertion limits) that preserve the LOCA analysis F_Q assumptions appropriately account for the impact of any fuel design present in the reactor core.

SRXB-RAI-1 References

RAI-1-1. DPC-NE-2011-P-A, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", Revision 2

SRXB-RAI-2

LAR Enclosure Attachments 1 and 3 describe the proposed revision to Robinson TS 3.2.1, Heat Flux Hot Channel Factor (F_Q). The LAR proposes a new LCO Action statement, LCO 3.2.1 Action C, that compares the measured $F_Q^M(X, Y, Z)$ against the centerline fuel melt limit ($F_Q^L(X, Y, Z)^{RPS}$). If $F_Q^M(X, Y, Z)$ exceeds $F_Q^L(X, Y, Z)^{RPS}$, the LAR proposes the following.

> C.1 Reduce the OPΔT $f_2(\Delta I)$ breakpoints from the COLR limit by KSLOPE for each 1% $F_Q^M(X, Y, Z)$ exceeds limit.

The proposed LCO 3.2.1 Action C has a proposed completion time of 72 hours.

However, the technical justification in LAR Enclosure Attachment 1 for LCO 3.2.1 Action C is insufficient. Please provide a technical justification for LCO 3.2.1 Action C, including addressing the following.

a. The LAR does not address the completion time for LCO 3.2.1 Action C. Please provide a basis for the LCO 3.2.1 Action C completion time.

b. Robinson TS Table 3.3.1-1, "Reactor Protection System Instrumentation," Note 2, "Overpower ΔT," includes the Overpower ΔT (OPΔT) function. The Robinson OPΔT function includes an axial imbalance function that is designated as f(ΔI). However, in the proposed Robinson LCO 3.2.1 Action C.1, the axial imbalance function is designated as $f_2(\Delta I)$, not f(ΔI). Please revise the proposed Robinson LCO 3.2.1 Action C.1 to be consistent with TS Table 3.3.1-1, Note 2, including providing the appropriate TS page markup, or justify the discrepancy.

SRXB-RAI-2 Response:

Part (a):

If the measured F_Q, $F_Q^M(X, Y, Z)$, exceeds the centerline fuel melt limit, $F_Q^L(X, Y, Z)^{RPS}$, which preserves the centerline fuel melt limit in the transient condition, LCO 3.2.1 Action C.1 requires a reduction in the OPΔT $f_2(\Delta I)$ breakpoints by KSLOPE for each 1% $F_Q^M(X, Y, Z)$ exceeds the centerline fuel melt limit. A 72 hour completion time is specified for this task. This completion time allows for an evaluation of the flux map measurement to confirm the validity of the out-of-limit condition, and if necessary, performance of confirmatory flux map prior to the implementation of the RPS setpoint change. If the out-of-limit condition is confirmed, the 72 hour completion time supports the orderly performance of the RPS setpoint modification, including the generation of a work order, work plan, job brief, rescheduling of any conflicting work, and the actual $f_2(\Delta I)$ breakpoint modification for each instrument channel. The 72 hour completion time is also consistent with completions times required to modify other RPS trip functions such as those required by TS 3.2.1 Actions B.3 and B.4. For these reasons, and because the probability of a limiting transient that would challenge the centerline fuel melt limit in this time period is highly unlikely, the 72 hour completion time is considered acceptable.

Part (b):

The Table 3.3.1-1 f(ΔI) trip reset function nomenclature specified in Note 2 is revised by adding a "2" subscript to be consistent with the nomenclature used in Action C.1. Additionally, the OTΔT f(ΔI) trip reset function nomenclature specified in Note 1 is also revised by adding a "1" subscript. These revisions make the f(ΔI) trip reset function nomenclature contained in these Notes consistent with that used at Harris, McGuire Nuclear Station and Catawba Nuclear Station. Attachment 4 contains the Table 3.3.1-1 markups identifying the proposed changes.

SRXB-RAI-3:

The LAR proposes to replace the Robinson SR 3.2.1.1, and add two new SRs, SR 3.2.1.2, and SR 3.2.1.3. The LAR does not technically justify the surveillance frequencies for the replaced SR 3.2.1.1 or for the new SRs, SR 3.2.1.2, and SR 3.2.1.3. Please discuss and provide a technical justification for the surveillance frequencies of the revised and new SRs.

SRXB-RAI-3 Response:

The proposed justifications for the replaced SR 3.2.1.1 and for the new SR 3.2.1.2 and SR 3.2.1.3 are provided below. SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.1.3 are modified by the following 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.

This note applies during the first power ascension after a refueling and allows for an increase in thermal power until an equilibrium power level has been achieved at which time a power distribution map can be obtained. This allowance, however, is modified by one of the three frequency conditions shown below.

- 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 $F_Q^M(X,Y,Z)$ was last verified, and
- 31 EFPD [effective full power days] thereafter

All three frequency conditions are applicable to SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.1.3. The first condition requires a F_Q measurement following a refueling outage prior to thermal power exceeding 75% RTP. The second condition requires verification that the measured F_Q, $F_Q^M(X,Y,Z)$, is within specified limits after a power rise of ≥ 10% RTP over the thermal power at which it was last verified to be within specified limits. Because $F_Q^M(X,Y,Z)$ could not have previously been measured in this reload core, power may be increased to RTP prior to an equilibrium verification of $F_Q^M(X,Y,Z)$ provided a measurement of $F_Q^M(X,Y,Z)$ is performed at one or more power levels during startup physics testing. As required by the first condition, these measurements are performed prior to reaching 75% RTP. This ensures that a determination of $F_Q^M(X,Y,Z)$ is made at a lower power level at which adequate margin is available before power is increased to 100% RTP. Without the first and second frequency conditions, reactor power could be increased to the RTP condition and operated for 31 EFPD prior to verification that $F_Q^M(X,Y,Z)$ is within limits. The second frequency condition is not intended to require verification of the surveillance parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is at least 10% higher than that power at which F_Q was last measured.

The 12 hour completion time after achieving equilibrium conditions allows for the orderly setup and performance of a flux map and the associated processing and review of the results used to verify each limit.

The 31 EFPD frequency is acceptable for verifying that $F_Q^M(X, Y, Z)$ is within limits because of the slow and predictable behavior of the core power distribution with increasing burnup and burnable absorber depletion. Provisions of SR 3.2.1.2 and SR 3.2.1.3 also extrapolate the measured power distribution for 31 EFPD beyond the most recent measurement and repeats the margin calculations at this future state to ensure $F_Q(X, Y, Z)$ remains within applicable limits.

SRXB-RAI-4:

For Robinson TS 3.2.3, “Axial Flux Difference (AFD),” SR 3.2.3.1, SR 3.2.3.2, and SR 3.2.3.3 are proposed to be replaced with SR 3.2.3.1.

a. While the surveillance portion of SR 3.2.3.1 was retained, the frequency portion of SR 3.2.3.1 was revised. However, the LAR does not address this change.

b. The existing SR 3.2.3.2 includes the following note in the frequency portion of the SR

   Only required to be performed if AFD monitor is inoperable

   The existing SR 3.2.3.2 was partially included in the frequency column of SR 3.2.3.1, however, the following was not included: (1) the need to log the AFD for each operable channel, (2) the note in the surveillance portion of the SR, and (3) the condition to perform the SR on 15 minutes intervals. The LAR does not address this change. The staff notes that the LAR proposed to retain the logging provision and the equivalent of the note in the surveillance portion of the Harris AFD TS but the LAR does not describe the reason the provisions are deleted for Robinson but retained for Harris.

   Please provide a discussion of and a technical justification for the proposed revision to the Robinson SR 3.2.3.1 and the elimination of SR 3.2.3.2

SRXB-RAI-4 Response:

A single response is provided to answer parts (a) and (b).

Current SR 3.2.3.1 and the SR 3.2.3.2 specify the requirement to monitor AFD for the condition where each excore channel is operable (SR 3.2.3.1) and for the condition where the AFD monitor alarm is inoperable (SR 3.2.3.2). To accommodate the proposed deletion of SR 3.2.3.2, the frequency portion of SR 3.2.3.1 is being revised to add the requirement to monitor AFD more frequently when the AFD monitor alarm is inoperable. The AFD logging requirements and variable monitoring frequency requirements that depend upon power level for SR 3.2.3.2 were removed. Justifications for each of these changes are presented below.

Current SR 3.2.3.2 requires an AFD log to be established to monitor the accumulation of penalty minutes for the time AFD was outside of the target band. This requirement is a carryover from the AREVA PDC-3 Axial Offset Control Methodology and is not required in the Duke Energy methodology. Therefore, the note that states, “Assume logged values of AFD exist during the previous time interval” is no longer required. For the same reason, the requirement to log AFD for each operable excore channel if the AFD monitor alarm monitor is inoperable is also removed.

The proposed revision to SR.3.2.3.1 adds the requirement to monitor AFD more frequently if the AFD monitor alarm is inoperable. The monitoring frequency is 7 days when the alarm monitor is operable, and is reduced to 1 hour if it is not. The variable frequency requirement when the AFD alarm monitor is inoperable in the current SR 3.2.3.2 is removed because there is no basis for the increased monitoring frequency depending upon the operating power level (15 minutes with thermal power $\geq$ 90% RTP and 1 hour when thermal power is < 90% FTP). This multi-tiered requirement adds administrative burden to the operations staff and is replaced in the proposed SR 3.2.3.1 with the requirement to monitor AFD within 1 hour of the AFD monitor alarm being inoperable, and every 1 hour thereafter. This singular surveillance frequency is acceptable during normal operation because AFD changes slowly over time. During a transient condition where AFD may change at a faster rate, the indicated AFD on the control board would be monitored by the operations staff as part of normal practice. A 1 hour surveillance frequency is also consistent with the McGuire and Catawba Technical Specifications.

To maintain fleet consistency, a change to Harris SR 4.2.1.1 is proposed. The proposed change encompasses removing the logging provision when the AFD Monitor Alarm is inoperable. This is justified because there are no requirements in the Duke Energy methodology requiring the logging of AFD.

The proposed revision to Harris SR 4.2.1.1 is shown below. Attachment 5 contains the updated Technical Specification markup that includes this proposed revision.

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 **within 1 hour and every 1 hour** ~~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~~.

SRXB-RAI-5

Robinson TS 5.6.5.a lists TSs with parameters to be documented in the core operating limits report (COLR). The LAR proposes to revise the names of TS of 3.2.1 and 3.2.2 but does not propose corresponding changes in TS 5.6.5.a.5 and TS 5.6.5.a.6. The effect is that the COLR will have parameters not applicable to the TS. Please clarify TS 5.6.5.a to be consistent with the proposed changes to TS of 3.2.1 and 3.2.2 and provide corresponding TS markups as applicable.

SRXB-RAI-5 Response:

Robinson Technical Specification 5.6.5.a is revised to maintain consistency in the naming convention used in the revised TS 3.2.1 and TS 3.2.2 and TS 5.6.5.a items 5 and 6. The revised TS 5.6.5.a is shown below. The updated Technical Specification markup that includes this proposed revision is presented in Attachment 4.

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~~ $\mathbf{F_Q(X, Y, Z)}$ **Limits** for Specification 3.2.1;

6. Nuclear Enthalpy Rise Hot Channel Factor ~~($F^N_{\Delta H}$) limit~~ $\mathbf{F_{\Delta H}(X, Y)}$ **Limits** for Specification 3.2.2;

Harris TS 6.9.1.6.1 is also revised to maintain consistency between TS 3/4.2.2 and TS 3/4.2.3, and TS 6.9.1.6.1 items “f” and “g”. This proposed revision to TS 6.9.1.6.1 is shown on the next page. The updated Technical Specification markup that includes this proposed revision is presented in Attachment 5.

<u>6.9.1.6</u> <u>CORE OPERATING LIMITS REPORT</u>

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) 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)~~ $\mathbf{F_Q(X, Y, Z)}$ **Limits** for Specification 3/4.2.2.

g. Enthalpy Rise Hot Channel Factor~~, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, PFΔH~~ $\mathbf{F_{\Delta H}(X, Y)}$ **Limits** for Specification 3/4.2.3.

h. Boron Concentration for Specification 3/4.9.1.

i. Reactor Core Safety Limits Figure for Specification 2.1.1.

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

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

SRXB-RAI-6:

In LAR Enclosure Attachments 2 and 4, the LAR describes the proposed revision to Harris TS 3/4.2.2, "Heat Flux Hot Channel Factor - $F_Q(Z)$." The LAR proposes a new TS 3/4.2.2 LCO Action a.1 (from Insert #2b) that states:

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

However, Harris TS 3/4.2.2 LCO Action a.1 (from insert 2b) is not fully discussed in the corresponding technical justification section. Please provide a technical justification for Harris TS 3/4.2.2 LCO Action a.1, including addressing the following.

a. The LAR proposes 72 hours to complete the action for LCO 3.2.2 Action a.1 (from insert 2b). The LAR does not discuss the time period. Please provide a justification for the time period.

b. DPC-NE-2011, in the section designated as "Technical Justification of Changes for Revision 2 (Redacted)," for Change 6-10, "Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$ (Section 6.4.3, third paragraph, page 6-10)," states

   The option to adjust the breakpoints of the OTΔT $f_1(\Delta I)$ trip reset penalty function is added because the OPΔT $f_2(\Delta I)$ trip reset penalty function at Harris is not active. As a result, the OTΔT $f_1(\Delta I)$ trip reset function breakpoints must be adjusted to compensate for the condition where $F_Q^M(X,Y,Z)$ exceeds $F_Q^L(X,Y,Z)^{RPS}$ limit.

   The OPΔT $f_2(\Delta I)$ trip reset penalty function at Harris being not active means $f_2(\Delta I)$ is defined as 0.0 for all ΔI and thus $f_2(\Delta I)$ has no breakpoints or slopes defined. Prior to $f_2(\Delta I)$ being moved to the COLR by Harris Amendment 161 (ADAMS Accession No. ML17250A202) $f_2(\Delta I)$ was defined as 0.0 for all ΔI for Harris. In the Harris Cycle 21 Core Operating Limits Report, Revision 1, submitted February 14, 2018 (ADAMS Accession No. ML18045A648), $f_2(\Delta I)$ was also defined as 0.0 for all ΔI.

   Please describe how the proposed action statement addresses the condition of the measured $F_Q^M(X,Y,Z)$ exceeds $F_Q^L(X,Y,Z)^{RPS}$ if the OPΔT $f_2(\Delta I)$ trip reset penalty function at Harris is not active. In particular, please describe what it means to adjust the breakpoints for OPΔT $f_2(\Delta I)$ when no breakpoints and slopes are defined.

c. Two actions a. are being proposed for TS 3/4.2.2 LCO Action a., one each in inserts 2a and 2b. Please clarify the designation of the actions for Harris TS 3/4.2.2 LCO Actions.

SRXB-RAI-6 Response:

Part (a):

Refer to the response for SRXB-RAI-2 part a.

Part (b):

At the time Revision 2 to the DPC-NE-2011 methodology report was written, the Harris OPΔT trip function lacked the hardware for the $f_2(\Delta I)$ trip reset function. Accordingly, a method change was made to allow for the adjustment of the OTΔT $f_1(\Delta I)$ trip reset function to regain CFM margin for the condition where $F_Q^M(X,Y,Z)$ exceeds the centerline fuel melt limit, $F_Q^L(X,Y,Z)^{RPS}$. An option to install the OPΔT $f_2(\Delta I)$ trip reset function was being pursued at the time Revision 2 to DPC-NE-2011 was submitted on May 4, 2016 (ADAMS Accession No. ML16125A420), however, the plant modification to install this trip reset penalty function had not been approved. Since the DPC-NE-2011 submittal, the plant modification to install the $f_2(\Delta I)$ functionality to the OPΔT trip function has been completed (performed during the spring 2018 refueling outage). Because the full functionality of the OPΔT trip function, including the $f_2(\Delta I)$ trip reset function, will be available prior the implementation of Duke Energy reload methods, the action associated with specification 4.2.2.2.c.3 to reduce the overpower ΔT $f_2(\Delta I)$ breakpoints if the measured F_Q is not within limits is acceptable. Breakpoints and slopes assumed in reload analyses for this function will be specified in the Core Operating Limits Report.

Part (c):

Insert 2b should specify action “c”, not action “a”. The corrected insert is shown below. Attachment 5 contains the updated Technical Specification markup for TS 3/4.2.2 including the corrected insert.

c. 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.

SRXB-RAI-7:

Harris TS 6.9.1.6.1 lists the core operating limits that shall be established and documented in the COLR. Harris TS 6.9.1.6.2 lists the analytical methods to be used to determine the core operating limits and the TS each analytical method is being used to determine. The LAR proposes to add to Harris TS 6.9.1.6.1 core operating limits in TSs 3/4.1.1.1, 3/4.1.2.5, 3/4.1.2.6, 3/4.5.1, and 3/4.5.4; however, no corresponding changes were proposed to TS 6.9.1.6.2 to identify the analytical methods applicable to the proposed added TS. LAR Enclosure Section 3.0 states that or each proposed TS relocation described in LAR Enclosure Sections 2.2 and 2.4, DPC-NF-2010 is the NRC-approved methodology used to calculate the appropriate acceptance criteria to ensure applicable plant safety analysis limits are met. Please provide a markup of TS 6.9.1.6.2 to reflect the additional TS DPC-NF-2010 is being used to determine.

SRXB-RAI-7 Response:

DPC-NF-2010-A was added to the list of analytical methods used to determine core operating limits as part of the NRC approval of the license amendment request for this report (ADAMS Accession No. ML17102A923). The citation for DPC-NF-2010-A in Technical Specification 6.9.1.6.2 (item r) is modified to add the core operating limits associated with Technical Specifications 3/4.1.1.1, 3/4.1.2.5, 3/4.1.2.6, 3/4.5.1 and 3/4.5.4. The modified citation is shown below. Attachment 5 contains the updated Technical Specification markup that includes this proposed revision to TS 6.9.1.6.2.

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

   (Methodology for Specification**s 3.1.1.1 – SHUTDOWN MARGIN – MODES 1 and 2,** 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4, and 5, 3.1.1.3 – Moderator Temperature Coefficient, **3.1.2.5 – Borated Water Source – Shutdown, 3.1.2.6 – Borated Water Sources – Operating,** 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, **3.5.1 – ECCS Accumulators – Cold Leg Injection, 3.5.4 – ECCS Refueling Water Storage Tank,** and 3.9.1 – Boron Concentration).

Attachment 3

Responses to the NRC Request for Additional Information (Proprietary)

Note: Text that is within brackets with an “a,c” superscript is proprietary to Duke Energy.

Attachment 4

Robinson Proposed Technical Specification Changes (Markup)

Table 3.3.1-1 (page 6 of 7)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Nominal Trip Setpoint by more than 2.96% of ΔT span.

$$\Delta T_{setpoint} \le \Delta T_0 \{ K_1 - K_2 [(1 + \tau_1 S) / (1 + \tau_2 S)] (T - T') + K_3(P - P') - \cancel{f(\Delta I)} \}$$

$f_1(\Delta I)$

Where: ΔT_0 is the indicated ΔT at RTP, ºF.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, ºF.
T' is the reference T_{avg} at RTP, ≤ [*]ºF.

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, ≥ [*] psig

K_1 ≤ [*] K_2 = [*]/ºF K_3 = [*]/psig
τ_1 ≥ [*] sec τ_2 ≤ [*] sec

$\cancel{f(\Delta I)}$ =	[*] {(q_b - q_t) - [*]}	when q_t - q_b < - [*] RTP
$f_1(\Delta I)$	0% of RTP	when - [*] RTP ≤ q_t - q_b ≤ [*] RTP
	[*] {(q_t - q_b) - [*]}	when q_t - q_b > [*] RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and q_t + q_b is the total THERMAL POWER in percent RTP.

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

Table 3.3.1-1 (page 7 of 7)
Reactor Protection System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Nominal Trip Setpoint by more than 3.17% of ΔT span.

$$\Delta T_{setpoint} \leq \Delta T_0 \{ K_4 - K_5 [\tau_3 S / (1 + \tau_3 S)] T - K_6(T - T') - \cancel{f(\Delta I)} \}$$

$\boxed{f_2(\Delta I)}$

Where: ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the reference T_{avg} at RTP, ≤ [*] °F.

$K_4 \leq$ [*]	$K_5 \geq$ [*]/°F for increasing T_{avg}	$K_6 \geq$ [*]/°F when T > T'
	[*]/°F for decreasing T_{avg}	[*]/°F when T ≤ T'
		$t_3 \geq$ [*] sec

$\cancel{f(\Delta I)}$ = $\boxed{f_2(\Delta I)}$	[*] {(q_b - q_t) - [*]}	when q_t - q_b < - [*] RTP
	0% of RTP	when - [*] RTP ≤ q_t - q_b ≤ [*] RTP
	[*] {(q_t - q_b) - [*]}	when q_t - q_b > [*] RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and q_t + q_b is the total THERMAL POWER in percent RTP.

The values denoted with [*] are 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;

$F_Q(X,Y,Z)$ Limits

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;

$F_{\Delta H}(X,Y)$ Limits

(continued)

Attachment 5

Harris Proposed Technical Specification Changes (Markup)

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

within 1 hour and every 1 hour

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

c. 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

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) 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.

$F_Q(X,Y,Z)$ Limits

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

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

$F_{\Delta H}(X,Y)$ Limits

h. Boron Concentration for Specification 3/4.9.1.

i. Reactor Core Safety Limits Figure for Specification 2.1.1.

Add Insert #7

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

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

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 2.2.1 – Reactor Trip System Instrumentation Setpoints, 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 3.2.5 – DNB Parameters).

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 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

l. 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.

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

Insert #7

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 2.2.1 – Reactor Trip System Instrumentation Setpoints, 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 3.2.5 – DNB Parameters).

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.

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.

Add Insert #8

(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).

Replace with Insert #9

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

(Methodology for Specifications 3.1.1.1 – SHUTDOWN MARGIN – MODES 1 and 2, 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4, and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.2.5 – Borated Water Source – Shutdown, 3.1.2.6 – Borated Water Sources – Operating, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.5.1 – ECCS Accumulators – Cold Leg Injection, 3.5.4 – ECCS Refueling Water Storage Tank, and 3.9.1 – Boron Concentration).

Insert #9