UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 8, 2021

Ms. Kim Maza
Site Vice President
Shearon Harris Nuclear Power Plant
5413 Shearon Harris Road
Mail Code NHP01
New Hill, NC 27562-9300

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT NO. 185 REGARDING REDUCTION OF REACTOR COOLANT SYSTEM MINIMUM FLOW RATE AND UPDATE TO THE CORE OPERATING LIMITS REPORT REFERENCES (EPID L-2020-LLA-0040)

Dear Ms. Maza:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued Amendment No. 185 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). This amendment is in response to your application dated March 6, 2020, as supplemented by letters dated April 23, June 22, and November 24, 2020.

The amendment revises Technical Specification (TS) 3/4.2.5, "DNB [Departure from Nucleate Boiling] Parameters," and TS 6.9.1.6, "Core Operating Limits Report" in support of analysis development for HNP cycle 24 and the introduction of reload batches of Framatome, Inc. (Framatome) GAIA (GAIA) fuel assemblies. TS 3/4.2.5 is revised to reflect a lower minimum Reactor Coolant System (RCS) flow rate, whereas TS 6.9.1.6.2 is revised to reflect the incorporation of the Areva NP, Inc. (AREVA) topical report EMF-2103(P)(A), Revision 3, "Realistic Large Break LOCA [Loss-of-Coolant Accident] Methodology for Pressurized Water Reactors." HNP TS 6.9.1.6.2 is also revised to reflect the removal of analytical methods no longer applicable for the determination of HNP core operating limits.

In addition, as part of the submitted license amendment request, the licensee provided an updated small break LOCA analysis, reflecting the proposed lower minimum RCS flow rate and the use of GAIA fuel assemblies. Therefore, the NRC staff reviewed the submitted material to ensure it properly addressed the effects of the proposed GAIA fuel introduction, including addressing the acceptability for use of GAIA fuel at HNP.

As stated in the NRC staff's related Safety Evaluation, the use of GAIA fuel at HNP is subject to the limitations and conditions of Topical Reports EMF-2103(P)(A), Revision 3, and ANP-10342NP-A, Revision 0, "GAIA Fuel Assembly Mechanical Design." In this regard, it should be noted that Limitation/Condition 3 of ANP-10342NP-A requires that "the final LTA [lead test assembly] program PIE [post-irradiation examination] report shall be submitted to NRC staff prior to any reload batch of GAIA assemblies reaching the third cycle of operation."

A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's regular monthly *Federal Register* notice.

Sincerely,

/RA/

Michael Mahoney, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 185 to NPF-63
2. Safety Evaluation

cc: Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 185
Renewed License No. NPF-63

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated March 6, 2020, as supplemented by letters dated April 23, June 22, and November 24, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Renewed Facility Operating License No. NPF-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 185, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to the startup of the Shearon Harris Nuclear Power Plant, Unit 1, cycle 24 (Spring 2021).

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona

Digitally signed by David J. Wrona
Date: 2021.04.08 13:47:53 -04'00'

David J. Wrona, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility License No. NPF-63
and Technical Specifications

Date of Issuance: April 8, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 185

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following page of the Renewed Facility Operating License with the revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change:

Remove	Insert
Page 4	Page 4

Replace (or remove) the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
iii	iii
2-2	2-2
3/4 2-14	3/4 2-14
6-24	6-24
6-24a	6-24a
6-24b	-
6-24c	-
6-24d	-

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

Duke Energy Progress, LLC, is authorized to operate the facility at reactor Core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 185, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, LLC. shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company* shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at 15.6.3 Subparts II (1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company* will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

[1]The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.
* On April 29, 2013, the name of “Carolina Power & Light Company” (CP&L) was changed to “Duke Energy Progress, Inc.” On August 1, 2015, the name “Duke Energy Progress, Inc.” was changed to “Duke Energy Progress, LLC.”

INDEX

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

FIGURE 2.1-1

REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION

WITH MEASURED RCS FLOW > [290,000 GPM X (1.0 + C_1)]

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

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg} ≤ the limit specified in the COLR, and

b. Pressurizer Pressure ≥ the limit specified in the COLR*, and

c. RCS total flow rate ≥ 290,000 gpm and greater than or equal to the limit specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters shown in Specification 3.2.5 shall be verified to be within its limit at the frequency specified in the Surveillance Frequency Control Program.

4.2.5.2 Verify, by precision heat balance, that RCS total flow rate is within its limit at the frequency specified in the Surveillance Frequency Control Program.**

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

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

6.9.1.6 CORE OPERATING LIMITS REPORT

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

a. SHUTDOWN MARGIN limits for Specification 3/4.1.1.1 and 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 $F_{Q}(X, Y, Z)$ Limits for Specification 3/4.2.2.

g. Enthalpy Rise Hot Channel Factor $F_{\Delta H}(X, Y)$ 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.

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.

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

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

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

d. BAW-10240(P)(A), “Incorporation of M5 Properties in Framatome ANP Approved Methods.”

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

f. Mechanical Design Methodologies

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

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

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

h. 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.

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

j. 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.

k. DPC-NE-3008-P-A, “Thermal-Hydraulic Models for Transient Analysis,” as approved by NRC Safety Evaluation dated April 10, 2018.

l. DPC-NE-3009-P-A, “FSAR / UFSAR Chapter 15 Transient Analysis Methodology,” as approved by NRC Safety Evaluation dated April 10, 2018.

m. ANP-10341P-A, “The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations,” approved version as specified in the COLR.

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.l. The report shall include:

a. The scope of inspections performed on each SG,
b. Degradation mechanisms found,
c. Nondestructive examination techniques utilized for each degradation mechanism,
d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,
e. Number of tubes plugged during the inspection outage for each degradation mechanism,
f. The number and percentage of tubes plugged to date, and the effective plugging percentage in each steam generator, and
g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 DELETED

(PAGES 6-24b THROUGH 6-24d DELETED By Amendment No. 185)

(PAGE 6-25 DELETED By Amendment No. 92)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 185 TO

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DUKE ENERGY PROGRESS, LLC

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By application dated March 6, 2020 (Reference 1), as supplemented by letters dated April 23, June 22 and November 24 (References 2, 3, and 4, respectively), Duke Energy Progress, LLC (the licensee), requested changes to the technical specifications (TSs) for the Shearon Harris Nuclear Power Plant (Harris or HNP), Unit 1.

The amendment revises Technical Specification (TS) 3/4.2.5, "DNB [Departure from Nucleate Boiling] Parameters," and TS 6.9.1.6, "Core Operating Limits Report" in support of analysis development for HNP cycle 24 and the introduction of reload batches of Framatome, Inc. (Framatome) GAIA (GAIA) fuel assemblies. TS 3/4.2.5 is revised to reflect a lower minimum Reactor Coolant System (RCS) flow rate, whereas TS 6.9.1.6.2 is revised to reflect the incorporation of the Areva NP, Inc. (AREVA) topical report EMF-2103(P)(A), Revision 3, "Realistic Large Break LOCA [Loss-of-Coolant Accident] Methodology for Pressurized Water Reactors" (Reference 5). HNP TS 6.9.1.6.2 is also revised to reflect the removal of analytical methods no longer applicable for the determination of HNP core operating limits.

In addition, as part of the submitted license amendment request, the licensee provided an updated small break LOCA analysis, reflecting the proposed lower minimum RCS flow rate and the use of GAIA fuel assemblies. The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the submitted material to ensure it properly addressed the effects of the proposed GAIA fuel introduction, including addressing the acceptability for use of GAIA fuel at HNP.

The supplement dated November 24, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's initial proposed no significant hazards consideration determination as published in the *Federal Register* on November 3, 2020 (85 FR 69660).

2.0 REGULATORY EVALUATION

2.1 System Description

From the HNP Updated Final Safety Analysis Report (UFSAR), Chapter 5 (Reference 6), a description of the RCS is provided, as follows:

> The Reactor Coolant System (RCS) … consists of three similar heat transfer loops connected in parallel to the reactor pressure vessel. Each loop contains a reactor coolant pump, steam generator and associated piping and valves. In addition, the system includes a pressurizer, pressurizer relief and safety valves, interconnecting piping and instrumentation necessary for operational control. All the above components are located in the Containment Building.
>
> During operation, the RCS transfers the heat generated in the core to the steam generators where steam is produced to drive the turbine generator. Borated demineralized water is circulated in the RCS at a flow rate and temperature consistent with achieving the reactor core thermal-hydraulic performance. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber (boron) used in chemical shim control.
>
> The RCS pressure boundary provides a barrier against the release of radioactivity generated within the reactor, and is designed to ensure a high degree of integrity throughout the life of the plant.

2.2 Description of Changes

The licensee for HNP proposed changes to Technical Specification (TS) 3/4.2.5, "DNB Parameters," and TS 6.9.1.6, "Core Operating Limits Report." The change to TS 3/4.2.5 would be revised to reflect a lower minimum Reactor Coolant System (RCS) flow rate and TS 6.9.1.6 would be revised to incorporate of EMF-2103(P)(A), Revision 3, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors."

2.2.1 Reactor Coolant System Flow Rate Reduction

The current TS limiting condition for operation (LCO) 3.2.5 states, as follows:

> 3.2.5 The following DNB-related parameters shall be maintained within the following limits:
>
> a. Reactor Coolant System T_{avg} ≤ the limit specified in the COLR, and
>
> b. Pressurizer Pressure ≥ the limit specified in the COLR*, and
>
> c. RCS total flow rate ≥ 293,540 gpm and greater than or equal to the limit specified in the COLR.

The requested change is to revise TS LCO 3.2.5 to state, as follows:

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

a. Reactor Coolant System T_{avg} ≤ the limit specified in the COLR, and

b. Pressurizer Pressure ≥ the limit specified in the COLR*, and

c. RCS total flow rate ≥ 290,000 gpm and greater than or equal to the limit specified in the COLR.

The licensee proposed a change to the TS Index as follows:

Current TS Index for Section 2.1 states, as follows:

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

Revised TS Index for Section 2.1 will state, as follows:

FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS – THREE LOOPS IN OPERATION WITH MEASURED RCS FLOW > [290,000 GPM X (1.0 + C_1)]

Additionally, TS page 2-2 is also changed to reflect the above change of the title of FIGURE 2.1-1.

2.2.2 Core Operating Limits Report References Revisions

The licensee proposed to revise and consolidate the COLR reference list in TS 6.9.1.6.2. The changes would consolidate and renumber the list of references.

The current list in TS 6.9.1.6.2 states, as follows:

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).

d. XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," 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, 3.2.2 – Heat Flux Hot Channel Factor, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

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, 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

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).

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 2.1.1 – Reactor Core Safety Limits, 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).

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 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).

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

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 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, 3.2.5 – DNB Parameters, 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.

(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.

BAW-10231P-A, “COPERNIC Fuel Rod Design Computer Code,” 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 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).

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).

t. DPC-NE-3008-P-A, "Thermal-Hydraulic Models for Transient Analysis," as approved by NRC Safety Evaluation dated April 10, 2018.

(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).

u. DPC-NE-3009-P-A, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology," as approved by NRC Safety Evaluation dated April 10, 2018.

(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).

[v. ANP-10341P-A, "The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations," 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).

Note, item v (above) was added by license amendment 180 (ADAMS Accession No. ML20281A279), and was not part of the TS when this license amendment request was submitted by the licensee and therefore not shown in the TS markups submitted by the licensee]

The revised list in TS 6.9.1.6.2 will state, as follows:

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

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

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

d. BAW-10240(P)(A), “Incorporation of M5 Properties in Framatome ANP Approved Methods.”

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

f. Mechanical Design Methodologies

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

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

h. 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.

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

j. 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.

k. DPC-NE-3008-P-A, “Thermal-Hydraulic Models for Transient Analysis,” as approved by NRC Safety Evaluation dated April 10, 2018.

l. DPC-NE-3009-P-A, “FSAR / UFSAR Chapter 15 Transient Analysis Methodology,” as approved by NRC Safety Evaluation dated April 10, 2018.

m. ANP-10341P-A, “The ORFEO-GAIA and ORFEO-NMGRID Critical Heat Flux Correlations,” approved version as specified in the COLR.

[note that item b, above, is proposed to be added to TS 6.9.1.6.2 with this license amendment, all other remaining items are only renamed (i.e., currently exist in TS 6.9.1.6.2). Additionally, the information previously found below each item, the referenced TS sections, is also proposed to be deleted by this license amendment]

2.3 Applicable Regulatory Requirements and Guidance

Regulations

The regulation at Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36, “Technical specifications,” establishes the regulatory requirements related to the content of TSs. Section 50.36(a)(1) requires an application for an operating license to include proposed TSs. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, must also be included in the application, but shall not become part of the TSs.

The regulation at 10 CFR 50.36(b) requires that:

> Each license authorizing operation of a …utilization facility …will include technical specifications. The technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to [10 CFR] 50.34 [“Contents of applications; technical information”]. The Commission may include such additional technical specifications as the Commission finds appropriate.

The regulation at 10 CFR 50.36(c)(2), “Limiting conditions for operation,” states:

> (i) 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.

The regulation at 10 CFR 50.36(c)(5) requires that:

> 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 [10 CFR] 50.4.

The requirements contained in 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” state, in part, that the Emergency Core Cooling System (ECCS) shall be designed such that an evaluation performed using an acceptable evaluation model demonstrates that acceptance criteria, set forth in 10 CFR 50.46(b), including peak cladding temperature, cladding oxidation, hydrogen generation, maintenance of coolable core geometry, and long-term core cooling are met for a variety of hypothetical loss-of-coolant accidents (LOCAs), including the most severe hypothetical LOCA.

The following 10 CFR Part 50, Appendix A, General Design Criteria (GDC) are applicable:

> GDC 10 - Reactor design. The reactor core and associated coolant, control, and protection systems shall 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.
>
> GDC 15 - Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.
>
> GDC 35 - Emergency core cooling. A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Regulatory Guidance

The NRC staff considered guidance contained in NRC Generic Letter (GL) 88-16, “Removal of Cycle-Specific Parameter Limits from Technical Specifications” (Reference 7), insofar as it specifies acceptable ways to reference, in the TS, methodologies used to determine cycle-specific parameter limits.

3.0 TECHNICAL EVALUATION

3.1 REACTOR COOLANT SYSTEM FLOW RATE REDUCTION

The licensee requested to change the required minimum RCS flow rate from 293,540 gallons per minute (gpm) to 290,000 gpm in order to provide additional operating margin for the minimum RCS flow rate.

The minimum RCS flow rate is specified in TS limiting condition for operation (LCO) 3.2.5.c. The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed in the departure from nucleate boiling (DNB) analyses. A lower RCS flow will cause the core to approach DNB limits. For anticipated operational occurrences (AOOs) in which the minimum RCS flow is a conservative assumption, the safety analyses show that the AOO will result in meeting the DNB criterion, when initiated at the minimum flow rate. This demonstration, in turn, assures compliance with GDC 10[1].

The licensee evaluated the Final Safety Analysis Report (FSAR) transient analyses to determine if they were affected by a change in minimum RCS flow. The transient analyses were grouped into three categories - Category 1: Transient Not Applicable, Bounded, or Insensitive to RCS Flow; Category 2: Transients Bounded by Current RCS Flow Assumptions; and Category 3: Transient-Specific Evaluations.

A full list of Category 1 transients is in Table 1 of Enclosure 1 to the licensee's March 6, 2020 letter. Transients in this category include such analyses as: Protection Against Dynamic Effects Associated with Postulated Rupture of Piping (LOCA only), Inadvertent Closure of Main Steam Isolation Valves, Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip, and Partial Loss of Forced Reactor Coolant Flow. The NRC staff reviewed the Category 1 transients and determined the licensee's evaluation that the change in RCS minimum flow does not impact the transient analyses listed in Table 1, is appropriate. Since the analyses for Category 1 transients are not affected by the RCS flow rate, the NRC staff determined that the requirements of GDC 10 and 15 remain satisfied for events in this category, relative to the proposed RCS flow rate reduction.

The Category 2 transients are bounded by the current analysis of record RCS flow assumptions. Events in which the analysis of record assumes an RCS flow rate of less than 290,000 gpm are not affected by the requested RCS flow change and therefore, do not need to be re-analyzed. Events in which a maximum RCS flow rate is conservative also do not need to be re-analyzed at the requested lower RCS flow rate.

The licensee uses the Duke Energy Statistical Core Design (SCD) methodology for many FSAR Chapter 15 DNB analyses. The SCD methodology is used to assess compliance with the short-term core cooling acceptance criteria. According to Section 15.0.3.1 of the UFSAR (Reference 11), the SCD methodology reassigns the instrument uncertainty on RCS flow, among other things, from the thermal-hydraulic analysis and incorporates the RCS flow uncertainty into the statistical design limit (SDL) on DNB. The licensee states in its March 6, 2020 letter, "The proposed reduction in minimum RCS flow rate from 293,540 gpm to 290,000 gpm will allow the surveillance limit to be reduced to 1.022 × 290,000 gpm = 296,380 gpm. FSAR DNB analyses of record performed with the SCD methodology must therefore consider a total RCS flow rate of 296,380 gpm."

A measurement uncertainty of ± 2.2 percent (%) flow is associated with the precision heat balance. This uncertainty corresponds to a value of ± 6520 gpm, suggesting that the SCD analyses support operating statepoints that range from 289,860 gpm to 302,900 gpm. The NRC staff observed that the proposed safety limit of ≥ 290,000 gpm is at the lower end, but is addressed, within this analyzed range.

The limit provided in a recent core operating limits report (COLR) is listed as ≥ 293,840 gpm, after subtraction for measurement uncertainty. The licensee stated, on Page 4 of Enclosure 1 of its March 6, 2020, letter, that "The surveillance limit… is set to the TS minimum RCS flow rate plus an allowance for flow measurement uncertainty. The surveillance limit is currently

[1] While GDC 10 directly applies to DNB analyses to which this minimum flow rate is directly applicable, the requirements of GDC 15 relate to the effects of system pressurization transients, which are also included in the licensee's disposition of UFSAR Chapter 15 events.

calculated as 1.022 x 293,540 gpm = 299,998 gpm." The licensee also stated, on Page 8 of Enclosure 1 of its March 6, 2020 letter, that the "reactor core safety limits in the HNP Cycle 23 COLR… have been generated assuming a measured RCS flow rate of 290,000 gpm x (1.0 + 0.022) = 296,380 gpm… The Cycle 24 COLR will assume the same."

The NRC staff inferred that the discussion on Page 8 on Enclosure 1 of the licensee's March 6, 2020, letter, confirms the information that the NRC staff also reviewed within the HNP UFSAR, indicating that the nominal RCS flow rate, 296,380 gpm, supports operating statepoints slightly lower than 290,000 gpm. This information is an indication that the current surveillance limit contained in the cycle 23 COLR is conservative because it requires a higher measured RCS flow rate than supported by the safety analyses. Therefore, the existing SCD analyses include margin to allow for the proposed RCS flow rate safety limit reduction, and the proposed TS change would allow the licensee to establish a cycle-specific surveillance limit that more closely aligns with the analyzed, nominal RCS flow rate. Based on this evaluation, the NRC staff determined that the licensee's conclusion that the SCD analyses need not be revised to support the lower safety limit is acceptable.

A complete list of Category 2 transients and the analyzed flow rate can be found in Table 2 of Enclosure 1 of the licensee's March 6, 2020, letter.

Since the licensee adequately demonstrated that the effects of the RCS flow rate reduction are bounded by the current analysis of record for Category 2 transients, the NRC staff determined that the requirements of GDC 10 and 15 remain satisfied for events in this category, relative to the proposed RCS flow rate reduction.

Category 3 transients consist of the FSAR transients which need to be re-analyzed with the decreased TS minimum RCS flow rate. With one exception discussed in Section 3.1.4 below, the licensee will reanalyze each Category 3 transient using NRC-approved safety analysis methods that are applicable to HNP, as the licensee states in its letter dated March 6, 2020 "[r]eanalysis with Duke Energy methods is in progress and will account for the proposed reduction in the TS minimum RCS flow rate." For the reasons discussed in the following sections of this Safety Evaluation, the NRC staff finds this to be acceptable.

3.1.1 <u>FSAR Section 15.1.2 – Feedwater System Malfunctions that Result in an Increase in Feedwater Flow</u>

The Feedwater System Malfunctions that Result in an Increase in Feedwater Flow transient is non-limiting for DNB and centerline fuel melt (CFM). The licensee states that, "the proposed reduction in the TS minimum RCS flow rate from 293,540 gpm to 290,000 gpm will result in no or negligible effect on the DNB and CFM results." This transient is not analyzed on a cycle-specific basis, but the licensee stated that this event will be analyzed with the new minimum RCS flow rate using the NRC-approved Duke Energy methods. Since the proposed flow rate reduction would have no or negligible effect on the DNB and CFM results, the NRC staff determined that the licensee's approach of addressing the effects of this event with analyses using the applicable, NRC-approved methods, assuming the reduced RCS flow rate, was acceptable.

3.1.2 <u>FSAR Section 15.1.5 – Steam System Piping Failure</u>

The steam system piping failure event is analyzed at hot full power (HFP) and hot zero power (HZP) conditions. The analysis of record (AOR) for the HZP case was performed with an RCS flow rate of 290,000 gpm and therefore, does not need to be re-analyzed. In the AOR, the HZP case is limiting; however, the HFP AOR case was not analyzed at the lower RCS flow rate. The HFP condition re-analysis will be performed to ensure the HZP condition is limiting.

There are short- and long-term analysis conditions for the HFP cases. For the short-term, HFP analysis, the licensee indicated that the present cycle analysis indicated greater than 20-percent margin to CFM, and that the minimum DNB ration remains well above the limit. Thus, the licensee concluded that the short-term analysis is non-limiting and is unlikely to become limiting based on the reduction in RCS flow rate. The licensee stated that the long-term, post-trip return to power is bounded by the existing HZP analysis of record, which is also unlikely to change with the RCS flow rate reduction. However, the licensee stated that both the short-term and long-term analyses will be performed using the applicable, NRC-approved methods and assuming the RCS flow rate reduction. Since the results of both HFP case analyses are unlikely to change significantly when the analyses are performed using the reduced RCS flow rate, the NRC staff has determined that the licensee's approach is acceptable.

3.1.3 FSAR Section 15.2.7 – Loss of Normal Feedwater Flow

The AOR for the Loss of Normal Feedwater Flow transient demonstrates that fluid mass in each steam generator exceeds 10,000 pounds-mass (lbm) for the duration of the event and subcooling margin exceeds 40 degree Fahrenheit (°F) after reactor trip. The long-term core cooling capability would be negligibly impacted by a small reduction in TS minimum RCS flow rate. The licensee stated that this event will be analyzed with the new minimum RCS flow rate using the NRC-approved Duke Energy methods. Since the event pertains more directly to the main steam system, with an allowance for primary system subcooling margin, the NRC staff agreed that this event is unlikely to be significantly affected by the proposed RCS flow rate reduction. Therefore, the NRC staff determined that the licensee's approach of addressing the effects of this event with analyses using the applicable, NRC-approved methods, assuming the reduced RCS flow rate, was acceptable.

3.1.4 FSAR Section 15.4.2 – Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

For the Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal at Power transient, two analyses of record exist – short-term core cooling and peak primary pressure. The short-term core cooling AOR assumes a total RCS flow rate of 296,380 gpm and uses the SCD methodology, as evaluated in Section 3.1 of this SE.

The peak primary pressure AOR is bounded by the turbine trip event and has a greater than 50 pounds per square inch (psi) margin to the acceptance criteria. The licensee stated that a reduction in RCS flow rate would have a negligible impact on the pressurization transient, and the licensee did not indicate that the event would be reanalyzed. The NRC accepted this disposition because the pressurization transient is analyzed in a low-power condition and the event is terminated by the high pressurizer pressure reactor trip. Key considerations governing this event are the rate of reactivity insertion and the total amount of reactivity inserted before the pressurizer pressure trip setpoint is reached, and as such, the NRC staff agrees that this event would be reasonably insensitive to the assumed RCS flow rate. Therefore, because the licensee's evaluation indicated that the uncontrolled RCCA bank withdrawal would be reasonably insensitive to the assumed RCS flow rate, the NRC staff determined that the pressurization aspects of this transient do not need to be reanalyzed at a lower RCS flow rate.

3.1.5 Summary – Category 3 Non-LOCA Events

The NRC staff determined that the licensee adequately addressed the effects of the RCS flow rate reduction on the Category 3 non-LOCA events. The licensee provided an evaluation for each that indicated that the proposed reduction was unlikely to affect the results significantly. In addition, for all but the uncontrolled RCCA withdrawal pressurization event, the licensee stated that these events will be analyzed using the NRC-approved, Duke Energy analysis methods (Reference 13). For the uncontrolled RCCA withdrawal, the NRC staff determined that the event was reasonably insensitive to RCS flow, consistent with the licensee's evaluation provided in Reference 1. Based on these considerations, the NRC staff determined that the proposed reduction in RCS flow rate was acceptable with respect to the Category 3, non-LOCA events. Relative to these events, the NRC staff determined the licensee's evaluations, as described in its letter dated March 6, 2020, demonstrated that the proposed, RCS flow rate reduction would remain consistent with the requirements of GDCs 10 and 15.

The Category 3 LOCA events are described in the following subsections.

3.1.6 FSAR Section 15.6.5.3 – Small Break LOCA Transient

The Small Break LOCA (SBLOCA) was re-analyzed for two reasons: to support the reduced TS minimum RCS flow rate and to support operation with GAIA fuel. Information regarding the transition to GAIA fuel is provided in Section 3.3 of this SE. The HNP, Unit 1, SBLOCA analysis with GAIA fuel can be found in Framatome licensing reports ANP-3766P and ANP-3766NP, proprietary and non-proprietary, respectively. These reports were included as Enclosures 6 and 7 of the licensee's March 6, 2020 letter (Reference 1). The licensee completed its SBLOCA analysis in accordance with EMF-2328(P)(A), Revision 0 and EMF-2328(P)(A), Revision 0, Supplement 1(P)(A), Revision 0 (References 8 and 9, respectively)[2].

The acceptance criteria for a SBLOCA analysis are provided in 10 CFR 50.46(b)(1-4). In order to determine the limiting analysis, the licensee completed a spectrum of cold-leg breaks with equivalent diameters ranging from 1.0 in to 8.7 inches. The NRC staff reviewed the results for the limiting break, provided in a table on Page 6 of Enclosure 1 of the licensee's March 6, 2020 letter, which indicated that the analytic results satisfied the acceptance criteria contained in 10 CFR 50.46(b)(1-4).

The licensee made a few exceptions to the approved methodology, as described on pages 3-5 and 3-6 of ANP-3766P/NP. The first was made to correct for an error that had been identified after the model was approved. This error correction has been acceptably implemented for other, 3-loop Pressurized Water Reactors (PWRs) that Framatome supports and is acceptable because it provides a more accurate representation of thermal hydraulic phenomena during the transient. The second reflects an update to an empirically-based model used in the evaluation, which was made to reflect an update to that model's existing data set. This is also acceptable because it provides a more accurate representation of the associated phenomena during the transient.

[2] Note that the licensee has already received NRC approval to implement methods described in EMF-2328(P)(A), Revision 0 (Harris License Amendment No. 114 (ADAMS Accession No. ML030870796), so this portion of the NRC staff's review was limited to assuring that (a) the effects of the RCS flow rate reduction were accounted for in the analysis, and (b) issues specific to the implementation of the recently approved Supplement 1(P)(A) to EMF-2328(P)(A) were appropriately addressed (i.e., RCP trip timing).

A number of additional studies were completed in the SBLOCA analysis including delayed reactor coolant pump (RCP) trip, attached piping breaks, and ECCS temperature sensitivity. In Reference 4, the licensee provided supplemental analyses that evaluated the sensitivity of the PCT and oxidation to RCP trip time delay to ensure that a limiting delay time had been identified. The licensee determined, based on these sensitivity studies, that the limiting PCT from the break spectrum analysis documented in its March 6, 2020 letter, remained bounding. In the attached piping break study, found in Section 4.4 of Enclosures 6 and 7 (proprietary and non-proprietary versions, respectively), to its March 6, 2020 letter, the licensee analyzed the effects of an accumulator line and safety injection (SI) line breaks. The results of these piping breaks were determined to be less limiting than those of the break spectrum analysis. In the ECCS temperature sensitivity study discussed in Reference 1, the temperatures of various sources of emergency coolant were varied to demonstrate that a bounding temperature had been used.

Based on the sensitivity studies described in References 1 and 4, the NRC staff determined that the licensee had analyzed a number of break sizes, locations, and other properties sufficient to provide assurance that the acceptance criteria contained in paragraph (b) to 10 CFR 50.46 were met, as required by 10 CFR 50.46(a)(1)(i).

Therefore, the NRC staff determined that the limiting results provided by the licensee were acceptable. Because the applicable 10 CFR 50.46(b) acceptance criteria were met, the NRC staff also determined that the licensee addressed the impact of the proposed RCS flow rate reduction in a manner that was consistent with GDC 35 requirements, with regard to SBLOCA.

3.1.7 Realistic Large Break LOCA Analysis with GAIA Fuel

The licensee submitted Framatome licensing report ANP-3767P, Revision 0, “Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis with Framatome GAIA Fuel Design Licensing Report,” to support operation with GAIA fuel. This report presents the Harris-specific implementation of the NRC-approved methodology, EMF-2103(P)(A) “Realistic Large Break LOCA Methodology for Pressurized Water Reactors,” Revision 3.

HNP received NRC approval to implement an ECCS performance evaluation that was based on Revision 0 of EMF-2103(P)(A), (by adding a reference to the plant-specific methodology in ANP-3011(P), “Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis,” Revision 1, to TS 6.9.1.6.2), in a license amendment issued in May 2012 (Reference 10). That analysis reflected certain plant-specific modeling approaches that were unique to HNP at the time. In the current LAR, the licensee proposes to replace its reference to the plant-specific analysis with a reference to the generic methods described in EMF-2103(P)(A), Revision 3. This is acceptable, because the NRC staff has reviewed and approved Revision 3 to EMF-2103(P)(A) (Reference 15), which includes generically-approved modeling features that account for the plant-specific adaptations of EMF-2103(P)(A), Revision 0, that HNP had previously used.

Since HNP already received approval to implement EMF-2103(P)(A)-based methods, this portion of the NRC staff review was limited to ensuring that the licensee addressed the conditions and limitations associated with EMF-2103(P)(A), Revision 3, and that the analysis results satisfied the applicable acceptance criteria, i.e., the criteria contained in paragraphs (b)(1-3) of 10 CFR 50.46. The NRC staff reviewed Table 3-4 of Enclosure 4 to the licensee’s March 6, 2020 letter, which describes how each of the conditions and limitations associated with EMF-2103(P)(A), Revision 3, is addressed. The NRC staff determined that each limitation and condition was satisfied, as described in that table. The NRC staff also reviewed Table 3-5 of

Enclosure 4 to the March 6, 2020 letter, which indicated that the analytic results satisfied the applicable acceptance criteria, i.e., the criteria contained in 10 CFR 50.46(b)(1-3). Because the applicable 10 CFR 50.46(b) acceptance criteria were met, the NRC staff also determined that the licensee addressed the impact of the proposed RCS flow rate reduction in a manner that was consistent with GDC 35 requirements, with regard to Large Break LOCA (LBLOCA).

Based on these considerations, the NRC staff determined that the licensee had acceptably implemented the generically-approved version of EMF-2103(P)(A), Revision 3, and on this basis, the NRC staff has determined that the proposed revision to the HNP TS, replacing the HNP-specific LBLOCA analysis with a reference to EMF-2103(P)(A), is acceptable.

3.1.8 Conclusion – RCS Flow Rate Reduction

The RCS flow rate is an initial condition in the Harris UFSAR, Chapter 15, safety analyses. A reduction to its minimum value can have a non-conservative effect within the safety analyses, especially with regard to DNB safety analysis. The licensee reviewed the effect that the RCS flow rate would have on each licensing basis event and described those effects. Based on the review described above, the NRC staff determined that the proposed effects were acceptably addressed. Because the licensee addressed the effects of the proposed flow reduction for the licensing basis safety analyses, the NRC concludes that the proposed RCS flow rate reduction would remain consistent with the requirements of GDCs 10, 15, and 35, and with 10 CFR 50.46 requirements.

3.2 CORE OPERATING LIMITS REPORT REFERENCE REVISIONS

The licensee proposed to remove analytical methods that will no longer be used at HNP, following a recent transition to NRC-approved, Duke Energy analytic methods. The licensee also proposed to update the formatting and structure of the TS COLR references to eliminate TS cross-referencing between the list of methods in use, contained in TS 6.9.1.6.2.

NRC guidance provides the framework for licensees to relocate cycle-specific parameter limits to licensee administrative control in GL 88-16. The guidance indicates that, in order to control these cycle-specific parameter limits, a new reporting requirement should be added to existing reporting requirements. In GL 88-16, the sample specification provides necessary information for inclusion in the administrative reporting requirements. The exact text is provided below, with a key phrase emphasized in bold text:

> [6.9.X] [Core] operating limits shall be established and documented in the [CORE] OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle. **(If desired, the individual specification that address [core] operating limits may be referenced.)** The analytical methods used to determine the [core] operating limits shall be those previously reviewed and approved by the NRC in [identify the Topical Report(s) by number, title, and date, or identify the staff's safety evaluation report for a plant-specific methodology by NRC letter and date]. The [core] operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal0hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The [CORE] OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to

> the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

The cross-referencing between methods in use and specific core operating limits in the present Harris TS is consistent with the emphasized parenthetical phrase, which indicates that cross-referencing is optional. Therefore, the NRC determined that, because the cross-referencing the licensee proposes to eliminate is optional, its proposed deletion is acceptable.

3.3 GAIA FUEL ASSEMBLY INTRODUCTION

In concert with this license amendment request, the licensee is planning to transition from the current High Thermal Performance (HTP) fuel assembly design to the Framatome GAIA fuel assembly design, which is described in NRC-approved licensing topical report ANP-10342NP-A, Revision 0, "GAIA Fuel Assembly Mechanical Design" (Reference 12). The licensee's April 23, 2020, supplemental letter (Reference 2), provided information demonstrating the acceptability of the GAIA fuel assembly for use at HNP.

Relative to the current HTP fuel, the GAIA fuel assembly has the following differences in design characteristics: spacer grids, bottom nozzle, and guide tube material. Of these, the design feature with the primary potential to affect the accident and transient analyses is the spacer grid. The new grid design has lower incident pressure drop than the HTP fuel assembly does. Thus, the GAIA fuel assembly will have improved DNB performance, and transition cores will have flow redistribution between HTP and GAIA fuel because of the pressure drop mismatch and resulting crossflow.

3.3.1 Effect of GAIA Fuel on Safety Analysis

In its April 23, 2020, supplement, the licensee provided information addressing the effect of the GAIA fuel assembly introduction on the UFSAR, Chapter 15, safety analyses. This information was provided in a format largely similar to that provided justifying the RCS flow rate reduction. The licensee separated the UFSAR Chapter 15 design basis events into two categories: those that do not require any type of reanalysis, and those that require a re-analysis either of the subchannel thermal hydraulics, or of the system transient.

In summary, the licensee stated:

> The differences between the HTP and GAIA fuel assembly designs are expected to have no significant impact or provide a beneficial impact on most aspects of the UFSAR Chapter 15 non-LOCA analyses. The mixed core is expected to cause a relatively minor penalty in DNB results for HTP fuel assemblies.[3] Confirmation of the expected effect and demonstration of positive analytical margin will be documented as part of the reload design process.

The table of events (Table 1.1 of the licensee's April 23, 2020 letter) not requiring re-analysis contained 27 licensing basis events, and these events fell into six categories based on the licensee's disposition. These were:

[3] In the NRC staff's assessment, an improvement in DNB performance resulting from the modified GAIA spacer grids will cause crossflow to divert from the HTP assemblies, as the GAIA spacer grids have lower flow resistance. However, the NRC staff observed that the VIPRE-01 subchannel analysis code is capable of simulating this mixed core effect accurately.

A. Bounded by another event
B. Non-limiting VIPRE-01 results that do not require cycle-specific monitoring
C. Does not require VIPRE-01 calculations.
D. Does not apply to HNP.
E. Does not involve a nuclear steam supply system transient.
F. Is not credible during power operation; no analysis necessary at zero-power.

For the events in category A, the NRC staff reviewed the HNP UFSAR and verified that the current licensing basis indicates that the event is bounded by another event. In the case of the Loss of Non-Emergency AC Power to the Station Auxiliaries, an American Nuclear Society (ANS) Condition II[4] event, is described in Section 15.2.6 of the UFSAR, which contains a description of a legacy analysis performed using the ANF-RELAP computer code. However, Table 1 of Enclosure 1 to the licensee's March 6, 2020 letter explains that various aspects of the event in Section 15.2.6 of the UFSAR are bounded by specific, other licensing basis events. In some cases, these events are in more severe and less frequent categories (e.g., the complete loss of forced reactor coolant flow is an ANS Category III Infrequent Fault), however, the licensee noted that the bounding events are analyzed using the more restrictive criteria applied to the ANS Category II events, which are anticipated operational occurrences. Since the bounding analyses apply similar acceptance criteria as would be applied to the Section 15.2.6 UFSAR transient, the NRC staff determined this disposition was acceptable. Based on its confirmatory review of the HNP UFSAR and consideration of the additional information in the original LAR, the NRC staff determined that the licensee's disposition for Category A events, relative to the GAIA fuel assembly design, was acceptable.

Three licensing basis events fell into Category B. These were Excessive Increase in Secondary Steam Flow, Turbine Trip, and Feedwater System Pipe Break. Each of these events is a transient initiated in the secondary coolant system and belongs to a class of transients (i.e., increases or decreases in secondary cooling) where numerous initiating events must be considered, but only one or several in the anticipated operational occurrence or accident category are bounding. Thus, the NRC staff accepted the disposition for the events in Category B because the licensee will analyze those events that are bounding, when necessary.

For the remaining, unanalyzed events listed in Categories C through F of the licensee's disposition table, the NRC staff reviewed the events and determined that the licensee's disposition was appropriate, as described in three examples below. First, the licensee determined the inadvertent startup of the ECCS during power operation did not require VIPRE-01 calculations, i.e., Category C. This is an appropriate disposition, because the inadvertent ECCS startup transient is a mass addition transient, the DNB performance for which is bounded by another event, and the primary and secondary pressurization effects would not require a VIPRE-01 analysis. Second, the licensee used Category D to disposition events that are reserved for boiling water reactors, and hence do not apply to HNP. Third, Category E was used to disposition radiological consequences of plant system or equipment failures, which do not involve transients of the primary or secondary coolant systems. Similar considerations apply to the remaining events and categories in Table 1 of the licensee's April 23, 2020 letter. Based on these considerations, the NRC staff determined that the licensee's disposition for not explicitly analyzing the events in Categories C through F of Table 1 of the April 23, 2020 letter was acceptable, relative to batch implementation of the GAIA fuel assembly design.

[4] ANS Condition II events are analogous to anticipated operational occurrences. ANS Condition III and IV events, by comparison, are less frequent and are hence permitted to have more severe consequences.

The licensee's disposition of Category II events requires reanalysis or cycle-specific evaluation in accordance with the NRC-approved reload safety evaluation process. As these events will be specifically considered using NRC-approved reload safety evaluation criteria and, if necessary, will be reanalyzed using NRC-approved methods, the NRC staff finds the licensee's approach for disposition of these events acceptable.

3.3.2 Limitations and Conditions in ANP-10342NP-A

To evaluate the acceptability of the GAIA fuel assembly at HNP, the NRC staff assessed whether the licensee had acceptably satisfied the applicable limitations and conditions in Framatome's ANP-10342NP-A, Revision 0, "GAIA Fuel Assembly Mechanical Design." In Section 4.0 of the NRC's SE approving ANP-10342NP-A, Revision 0 for use (Reference 14), the NRC staff provided four limitations and conditions. Each is repeated below, followed by a summary of information provided by the licensee addressing the limitation/condition and a brief staff evaluation.

1. This GAIA fuel assembly design is approved for use with low enrichment uranium (LEU) fuel which has been enriched to less than or equal to 5 percent.

The licensee stated that the HNP TS limit the maximum enrichment of fuel loaded into the HNP reactor core to 5.0 weight percent Uranium-235 (U-235) or less.

The NRC staff reviewed the limitation and verified that the specified limit is contained in HNP TS 5.3.1. Since the TS limit is consistent with limitation/condition 1 of the SE approving ANP-10342NP-A for use, the NRC staff determined that limitation/condition 1 has been acceptably addressed.

2. The GAIA fuel assembly design is licensed for a maximum fuel rod burnup of 62,000 Megawatt-days/metric ton of Uranium [MWD/MTU].

The licensee stated that the fuel rod design code in use at HNP, COPERNIC, is similarly limited to 62,000 MWD/MTU [equivalent to 62 gigawatt-days per metric ton of uranium (GWD/MTU) as stated by the licensee]. The licensee also stated, in concert with the license amendment approving of the licensee's use of the fuel rod design code at HNP, that adherence to this limit is verified as part of the normal reload design process.

Based on the consistency between the GAIA fuel assembly burnup limitation and that of the fuel rod design code in use, and on the fact that the licensee will verify that this limit is met as a part of the normal reload design process, the NRC staff determined that the licensee has acceptably addressed limitation/condition 2 on the use of the GAIA fuel assembly design at HNP.

3. The final LTA [lead test assembly] program PIE [post-irradiation examination] report shall be submitted to NRC staff prior to any reload batch of GAIA assemblies reaching the third cycle of operation.

The licensee clarified that this limitation/condition requires the fuel vendor to submit a final LTA program report to the NRC staff prior to any reload batch of GAIA fuel assemblies reaching a third cycle of operation. The licensee also stated that a third batch reload of GAIA fuel assemblies would not be introduced until 2024 at HNP.

This limitation applies to and was established to assure that LTAs had been irradiated to sufficient exposure to provide meaningful data to the fuel vendor to qualify the assemblies for a third batch exposure. The NRC staff agrees that this limitation need not be satisfied until sometime in the future, following operation in cycles 1 and 2. Additionally, limitation/condition 3 was intended to provide information to validate analyses indicating that the instrument tubes and guide tubes would not, in high exposure conditions, experience unanticipated fuel assembly axial growth. Such a condition is expected to be unlikely to occur based on the analyses and evaluations described in ANP-10342P-A, and in Section 3.3.1.7 of the NRC staff's SE for that report. Based on this consideration, the NRC staff determined that it is acceptable for the licensee to implement use of GAIA fuel for cycles 1 and 2 pending submission of the information required to fulfill limitation/condition 3.

In the absence of the fuel vendor's report, the licensee would need to perform a safety analysis under 10 CFR 50.59, "Changes, Tests, and Experiments," and/or seek a license amendment, in order to implement a cycle 3 core reload using GAIA fuel.

Limitation/condition number 4 was removed from ANP-10324NP-A, therefore limitation/condition 5 listed in Section 4.0 of the SE approving ANP-10342NP-A for use, is the fourth limitation/condition applicable to HNP.

5. As part of the plant-specific LAR [license amendment request] implementing GAIA, the licensee must demonstrate acceptable performance of GAIA under RIA [reactivity initiated accident (e.g., rod cluster control assembly ejection accident)] conditions, including fuel damage, coolable geometry, and radiological consequences, using approved methods. Current guidance and analytical methods are found in SRP [Standard Review Plan] 4.2 Appendix B. Newer guidance is expected soon (e.g., DG [draft regulatory guide]-1327). The licensee should consider the most up-to-date guidance and analytical limits at the time of submittal. Alternative means to demonstrate compliance will be considered on a case-by-case basis.

In its supplement dated April 23, 2020, the licensee stated that the methods and acceptance criteria in use to perform the rod cluster control assembly ejection accident are specified in an request for additional information (RAI) response associated with the NRC staff review of DPC-NE-3009, "FSAR / UFSAR Chapter 15 Transient Analysis Methodology." This RAI response was submitted by letter dated October 30, 2017 and was found acceptable as documented in Section 3.5.21.1 of the NRC SE approving DPC-NE-3009 (Reference 13).

The NRC staff reviewed the supplemental information provided and referenced by the licensee. The NRC staff confirmed that the methods and acceptance criteria are consistent with those specified in limitation/condition 5; however, the limitation requires a demonstration that the acceptance criteria are satisfied. The licensee's response to the referenced RAI provided this demonstration for then-current fuel designs, while the proposed GAIA fuel assembly design may differ from those designs. In its supplemental letter dated November 24, 2020, the licensee provided an engineering evaluation describing the design similarities and differences between GAIA and HTP fuel, as well as the applicability of the analyses referenced above to the GAIA fuel assembly design. Among the key considerations, the licensee stated that the transition from the HTP to the GAIA fuel design has negligible impact on the neutronic characteristics of the fuel and, therefore, negligible impact on the transient power response of the event and power distribution.

The licensee's evaluation also provided information summarizing the effect GAIA fuel introduction would have on the rod ejection analysis, which comprises six independent analyses using various computer codes and evaluating the accident with regard to different fuel damage mechanisms. According to the licensee's evaluation, the thermal-mechanical differences between HTP and GAIA fuel assemblies offer DNB benefits to offset potential compromises in DNB performance that could come about because of differences in the fuel rod mechanical design. Outside of DNB-related fuel cladding integrity, the licensee concluded that the GAIA fuel would perform either comparably or better relative to HTP fuel, meaning the analyses that assume use of HTP fuel are conservative. The licensee concluded that a minor change in the uranium loading of GAIA fuel would cause a change to the source term predicted for the reactivity insertion accident (RIA), but that cycle-specific calculations are performed to demonstrate that applicable dose limits will remain satisfied for each reload core.

Based on its review of the licensee's evaluation, the NRC staff determined that the licensee acceptably addressed the effects that the GAIA fuel assembly would have on the RIA analysis, and that GAIA fuel could be expected to perform essentially the same as, or marginally better than, HTP fuel such that the existing RIA analysis provides a reasonable confirmation that the acceptance criteria used within Duke's analytic methods would remain satisfied. Therefore, the NRC staff determined that the licensee satisfied limitation/condition 5.

3.4 TECHNICAL EVALUATION SUMMARY

The NRC staff has determined the following, based on its review:

- Concerning the proposed RCS flow rate reduction, the licensee acceptably addressed the effects of the reduction on the safety analyses and demonstrated that the reduction can be accomplished while meeting applicable design criteria.

- Concerning the implementation of updated SB and LB LOCA analyses, the licensee has used acceptable methods and demonstrated conformance to the applicable acceptance criteria contained in paragraph (b) of 10 CFR 50.46.

- Concerning the TS COLR Reference updates, the licensee indicated that references proposed for removal are no longer used and demonstrated that the updated formatting remains consistent with the guidance contained in GL 88-16.

- Since the updated analyses reflect the batch introduction of the Framatome GAIA fuel assembly design, the NRC staff also confirmed that the GAIA fuel assembly design introduction will be accomplished acceptably, and that the conditions and limitations for the applicable topical report have been addressed.

- The licensee appropriately addressed the effects of the proposed reduced minimum RCS flowrate for the licensing basis safety analyses, and the revision of TS LCO 3.2.5(c) continues to meet 10 CFR 50.36(c)(2), to ensure when a LCO 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.

Based on the considerations discussed above, the NRC staff concludes that the proposed license amendment revising the minimum RCS flow rate and updating the COLR references list to eliminate outdated references, revise formatting, and update the TS with new analytic methods, is therefore, acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the State of North Carolina official was notified of the proposed issuance of the amendment on February 2, 2020 (Agencywide Document Access and Management System (ADAMS) Accession No. ML21033A845). The State of North Carolina official responded on February 2, 2020, with no comments (ADAMS Accession No. ML21033B106).

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes the requirements with respect to installation or use of a facility's components located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration (85 FR 69660, dated November 24, 2020), and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need to be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Duke Energy, "License Amendment Request to Reduce the Minimum Required Reactor Coolant System Flow Rate and Update the List of Analytical Methods Used in the Determination of Core Operating Limits," Docket No. 50-400, March 6, 2020, (ADAMS Accession No. ML20066L112, part of ADAMS Package No. ML20066L108).

2. Duke Energy, "Supplemental Information for License Amendment Request to Reduce the Minimum Required Reactor Coolant System Flow Rate and Update the List of Analytical Methods Used in the Determination of Core Operating Limits," Docket No. 50-400, April 23, 2020 (ADAMS Accession No. ML20114E131).

3. Duke Energy, "Supplemental Information for License Amendment Request to Reduce the Minimum Required Reactor Coolant System Flow Rate and Update the List of Analytical Methods Used in the Determination of Core Operating Limits," Docket No. 50-400, June 22, 2020 (ADAMS Accession No. ML20174A640).

4. Duke Energy, “Response to Request for Additional Information Regarding License Amendment Request to Reduce the Minimum Required Reactor Coolant System Flow Rate and Update the List of Analytical Methods Used in the Determination of Core Operating Limits,” Docket No. 50-400, November 24, 2020 (ADAMS Accession No. ML20329A387, part of ADAMS Package No. ML20329A386).

5. Areva NP, Inc. topical report EMF-2103(P)(A), Revision 3, “Realistic Large Break LOCA [Loss-of-Coolant Accident] Methodology for Pressurized Water Reactors” (ADAMS Package Accession No. ML16286A579).

6. Duke Energy, Shearon Harris Nuclear Power Plant, Unit 1 – “Shearon Harris Nuclear Plant, Unit 1, Amendment 63 to Final Safety Analysis Report, Chapter 5, Reactor Coolant System and Connected Systems,” (ADAMS Accession No. ML20147A022), Part of “Submittal of Updated Final Safety Analysis Report (Amendment 63), Technical Specification Bases Revision, Report of Changes Pursuant to 10 CFR 50.59 and Summary of Commitment Changes,” Dated May 15, 2020 (ADAMS Package No. ML20147A016).

7. NRC Generic Letter (GL) 88-16, “Removal of Cycle-Specific Parameter Limits from Technical Specifications,” dated October 4, 1988 (ADAMS Accession No. ML031130447).

8. Framatome Advanced Nuclear Power Richland, “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” Transmittal Letter NRC:01:019, Proprietary Report EMF-2328(P)(A), Revision 0, and Nonproprietary Report EMF-2328(NP)(A), Revision 0, Project Docket No. 728, May 9, 2001 (ADAMS Package No. ML011410426).

9. AREVA, Inc. “PWR Small Break LOCA Evaluation Model, S-RELAP5 Based,” Transmittal Letter NRC:17:013, Proprietary Report EMF-2328PA, Revision 0, Supplement 1PA, Revision 0, and Nonproprietary Report EMF-2328NPA, Revision 0, Supplement 1NPA, Revision 0, Project Docket No. 728, March 2017 (ADAMS Package No. ML17082A170).

10. NRC, “Shearon Harris Nuclear Plant, Unit 1 – Issuance of Amendment Re: The Revision to Technical Specification Core Operating Limits Report References for Realistic Large Break Loss-of-Coolant Accident Analysis (TAC No. ME6999),” Docket No. 50-400, May 30, 2012 (ADAMS Accession No. ML12076A103).

11. Duke Energy, Shearon Harris Nuclear Power Plant, Unit 1 – “Shearon Harris Nuclear Plant, Unit 1, Amendment 63 to Final Safety Analysis Report, Chapter 15, Accident Analysis,” (ADAMS Accession No. ML20147A032), Part of “Submittal of Updated Final Safety Analysis Report (Amendment 63), Technical Specification Bases Revision, Report of Changes Pursuant to 10 CFR 50.59 and Summary of Commitment Changes,” Dated May 15, 2020 (ADAMS Package No. ML20147A016).

12. Framatome, “GAIA Fuel Assembly Mechanical Design,” Transmittal Letter NRC:19:028, Proprietary Report ANP-10342P-A, Revision 0, and Nonproprietary Report ANP-10342NP-A, Docket No. 99902041, November 2019 (ADAMS Package No. ML19309D913).

13. NRC, “Shearon Harris Nuclear Power Plant, Unit 1 and H. B. Robinson Steam Electric Plant, Unit No. 2 – Issuance of Amendments Revising Technical Specifications for Methodology Reports DPC-NE-3008-P, Revision 0, ‘Thermal-Hydraulic Models for Transient Analysis,’ and DPC-NE-3009-P, Revision 0, ‘FSAR / UFSAR Chapter 15 Transient Analysis Methodology,” Docket No. 50-400, April 10, 2018 (ADAMS Accession No. ML18060A401).

14. NRC, “Final Safety Evaluation for Framatome, Inc. Topical Report ANP-10342P, Revision 0, ‘GAIA Fuel Assembly Mechanical Design’ (CAC No. MF9078/EPID: L-2016-TOP-0016),” September 24, 2019 (ADAMS Package Accession No. ML19204A048).

15. NRC, “Final Safety Evaluation for Areva NP Inc. Topical Report EMF-2103(P), Revision 3, ‘Realistic LOCA Methodology For Pressurized Water Reactors’ (TAC No. MF2904),” June 29, 2016 (ADAMS Package Accession No. ML16172A329).

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SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT NO. 185 REGARDING REDUCTION OF REACTOR COOLANT SYSTEM MINIMUM FLOW RATE AND UPDATE TO THE CORE OPERATING LIMITS REPORT REFERENCES (EPID L-2020-LLA-0040) DATED APRIL 8, 2021

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