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

May 18, 2017

Mr. Kelvin Henderson
Senior Vice President
Nuclear Corporate
526 South Church Street, EC-07H
Charlotte, NC 28202

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENTS REVISING TECHNICAL SPECIFICATIONS FOR METHODOLOGY REPORTS DPC-NE-1008-P REVISION 0, "NUCLEAR DESIGN METHODOLOGY USING CASMO-5/SIMULATE-3 FOR WESTINGHOUSE REACTORS," DPC-NF-2010 REVISION 3, "NUCLEAR PHYSICS METHODOLOGY FOR RELOAD DESIGN," AND DPC-NE-2011-P REVISION 2, "NUCLEAR DESIGN METHODOLOGY REPORT FOR CORE OPERATING LIMITS OF WESTINGHOUSE REACTORS" (CAC NOS. MF6648/MF6649 AND MF7693/MF7694)

Dear Mr. Henderson:

By letter dated August 19, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15236A044), as supplemented by letters dated May 4, October 3, and November 17, 2016 (ADAMS Accession Nos. ML16125A420, ML16278A090 and ML16323A102), Duke Energy Progress, LLC (Duke Energy) submitted a request, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.90, for Shearon Harris Nuclear Power Plant, Unit 1 (Harris) and H. B. Robinson Steam Electric Plant, Unit 2 (Robinson) to amend their technical specifications (TSs) to adopt the methodology reports DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," for application specific to Harris and Robinson. In its submittal, Duke Energy requested review and approval of these methodologies for plant-specific use only, allowing these methodologies be used at Harris and Robinson to perform the subject analyses in-house.

The U.S. Nuclear Regulatory Commission (NRC) has issued Amendment No. 157 to Renewed Facility Operating License No. NPF-63 for Harris and Amendment No. 253 to Renewed Facility Operating License No. DPR-23 for Robinson. These amendments change the TSs in response to your submittal dated August 19, 2015, as supplemented by letters dated May 4, October 3, and November 17, 2016.

Enclosures 3 and 5 transmitted herewith contains Sensitive Unclassified Non Safeguard Information. When Separated from Enclosure 3 and 5, this document is decontrolled.

The NRC has completed its review of information provided by the licensee. The NRC staff's safety evaluations (SEs) are enclosed. The NRC staff has determined that the enclosed SEs (Enclosure 3 and Enclosure 5) contain proprietary information pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, the NRC staff has prepared redacted, nonproprietary versions (Enclosure 4 and Enclosure 6). However, the NRC will delay placing the nonproprietary SEs in the public document room for a period of 10-working days from the date of this letter to provide Duke Energy with the opportunity to comment on any proprietary aspects. If you believe that any information in Enclosure 4 and Enclosure 6 is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10-working days, the nonproprietary SEs will be made publicly available.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

If you have any questions concerning this letter, please contact me at 301-415-2760 or by email at Martha.Barillas@nrc.gov.

Sincerely,

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

Docket Nos. 50-261 and 50-400

Enclosures:
1. Amendment No. 157 to NPF-63
2. Amendment No. 253 to DPR-23
3. Safety Evaluation for CASMO-5/SIMULATE3 Methodology PROPRIETARY
4. Safety Evaluation for CASMO-5/SIMULATE3 Methodology NONPROPRIETARY
5. Safety Evaluation for DPC-NF-2010 and DPC-NE-2011-P Methodology PROPRIETARY
6. Safety Evaluation for DPC-NF-2010 and DPC-NE-2011-P Methodology NONPROPRIETARY

cc w/enclosures 1, 2, 4 and 6: Distribution via Listserv **10-working days after issuance**

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. 157
Renewed License No. NPF-63

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

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated August 19, 2015, as supplemented by letters dated May 4, October 3, and November 17, 2016, 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. 157 , 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 within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed License
and the Technical Specifications

Date of Issuance: May 18, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 157

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE

DOCKET NO. 50-400

Replace page 4 of the Renewed Facility Operating License No. NPF-63 with the attached page 4.

Replace 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
6-24c	6-24c
6-24d	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. 157, 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 "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."

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

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

o. Mechanical Design Methodologies

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

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

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

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

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

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

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

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

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

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

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

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

(Methodology for Specification 3.1.1.2 – SHUTDOWN MARGIN – MODES 3, 4, and 5, 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, and 3.9.1 – Boron Concentration).

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

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

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

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

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

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No.253
Renewed License No. DPR-23

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

   A. The application for amendment by Duke Energy Progress, LLC (the licensee), dated August 19, 2015, as supplemented by letters dated May 4, October 3, and November 17, 2016, 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 3.B. of Renewed Facility Operating License No. DPR-23 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 253 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Benjamin G. Beasley, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Operating License No. DPR-23
and the Technical Specifications

Date of Issuance: May 18, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 253

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace page 3 of Renewed Facility Operating License No. DPR-23 with the attached page 3.

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove	Insert
5.0-27	5.0-27

D. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;

E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by operation of the facility.

3. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; 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:

A. Maximum Power Level

The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of 2339 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 253, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(1) For Surveillance Requirements (SRs) that are new in Amendment 176 to Final Operating License DPR-23, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 176. For SRs that existed prior to Amendment 176, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 176.

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

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

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

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

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

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

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

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 157 TO RENEWED FACILITY OPERATING LICENSE

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

AND AMENDMENT NO. 253 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-23

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2, DOCKET NO. 50-261

DUKE ENERGY PROGRESS, LLC

1.0 INTRODUCTION

By application dated August 19, 2015 (Reference 1), as supplemented by letters dated May 4, October 3, and November 17, 2016 (References 2, 3, and 4), Duke Energy Progress, LLC (Duke Energy), the licensee for Shearon Harris Nuclear Power Plant, Unit 1 (Harris), and H. B. Robinson Steam Electric Plant, Unit 2 (Robinson), requested changes to the Harris and Robinson Technical Specifications (TSs). Specifically, Duke Energy requested U.S. Nuclear Regulatory Commission (NRC) site-specific review and approval of DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors (hereafter referred to as DPC-NE-1008-P)" (Reference 5) and adoption of the methodology into Harris TS 6.9.1.6.2 and Robinson TS 5.6.5.b.

The proposed TS revisions that include the DPC-NE-1008-P methodology report would allow Duke Energy to perform reactor physics calculations as part of the core reload design process at Harris and Robinson, and would replace the analysis currently performed by AREVA using CASMO-3/PRISM. Duke Energy considers the methodology described in DPC-NE-1008-P to be an evolution of the CASMO-4/SIMULATE-3 nuclear methodologies used at Catawba, McGuire, and Oconee nuclear stations described in DPC-NE-1005-P-A, Revision 1, "Duke Power Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX [Mixed Oxide]" (Reference 6) and DPC-NE-1006-P-A, "Oconee Nuclear Design Methodology Using CASMO-4/SIMULATE-3" (Reference 7, approved in Reference 8). The DPC-NE-1008-P methodology report submitted by Duke Energy contains benchmarks to McGuire operational data and references to future use at Duke Energy facilities other than Harris and Robinson. However, the NRC staff reviewed the methodology on a plant-specific basis for application to Harris and Robinson only. The NRC staff accepted the license amendment request (LAR) for review by letter dated December 1, 2015 (Reference 9). An audit plan was submitted to the licensee in a letter dated June 16, 2016 (Reference 10), and on July 12 and 13, 2016, the NRC staff audited various materials related to the review of DPC-NE-1008-P. An audit report was sent to the licensee in a letter dated October 17, 2016 (Reference 11), indicating the need to issue requests for additional information (RAIs). These RAIs were issued by the NRC staff by e-mail dated September 7, 2016 (Reference 12). Duke Energy responded to the RAIs in a letter dated October 3, 2016 (Reference 3).

Enclosure 4

Duke Energy also supplemented the LAR with a request for the review of methodology reports DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design" (Reference 13) and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" (Reference 14), and their adoption into the Harris and Robinson TSs. These reports had previously been submitted as an independent licensing action but were withdrawn when it was determined that they were linked to the review of DPC-NE-1008-P. The NRC staff's evaluation of these two methodology reports and their adoption into the Harris and Robinson TSs is presented in a separate safety evaluation (SE) as Enclosure 5 (proprietary SE) and Enclosure 6 (nonproprietary SE) of the transmittal letter to this SE.

On February 2, 2016, the NRC staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (81 FR 5492) for the proposed amendments. Subsequently, by letter dated May 4, 2016, the licensee provided additional information that expanded the scope of the amendment request as originally noticed in the *Federal Register*. Accordingly, the NRC published a second proposed NSHC determination in the *Federal Register* on August 2, 2016 (81 FR 50746), which superseded the original notice in its entirety. The supplemental letters dated October 3, and November 17, 2016, provided additional information that clarified the application, did not expand the scope beyond the second notice, and did not change the staff's proposed no significant hazards consideration determination as published in the *Federal Register*.

2.0 REGULATORY EVALUATION

The Duke Energy report DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," describes a methodology for performing core physics analyses using the CASMO-5 and SIMULATE-3 codes for Harris and Robinson. The CASMO-5 code is a multi-group, two-dimensional (2-D) transport theory code with a microscopic depletion model for burnable absorbers. The CASMO-5 code is used to calculate lattice physics parameters, including cross sections (XS), pin power distributions and other nuclear data, which are utilized by SIMULATE-3 during core design calculations. The SIMULATE-3 code is a two-group, three-dimensional (3-D) coarse-mesh nodal diffusion theory solver, which is capable of combining the nodal solution with the heterogeneous lattice solution from CASMO-5 to calculate pin power distributions.

The use of the DPC-NE-1008-P methodology report for Harris and Robinson reload analyses requires NRC staff review and approval prior to site-specific implementation. In addition to reviewing the methodology report, this SE also evaluates changes to the Harris TS Administrative Controls Section 6.9.1.6 and Robinson TS Administrative Controls Section 5.6.5, which are the respective Core Operating Limits Report (COLR) sections that will add DPC-NE-1008-P to the list of analytical methods used to determine the COLR (hereafter referred to as COLR reference lists).

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications," states, "each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements section." This regulation requires that the TSs include items in the following specific categories: (1) Safety limits, limiting safety system settings, and limiting control settings;

(2) Limiting conditions for operation; (3) Surveillance requirements; (4) Design features; and (5) Administrative controls.

General Design Criterion (GDC) 10 from 10 CFR Part 50, Appendix A, states that the reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including the effects of anticipated operational occurrences (AOOs).

Duke Energy's application stated that Harris is licensed to the final GDC 10. Duke Energy also stated that Robinson was evaluated against the proposed Appendix A to 10 CFR Part 50 published in the *Federal Register* on July 11, 1967, Criterion 6 of the draft GDC. Criterion 6 of the draft GDC is the equivalent to GDC 10 with respect to the requirements regarding SAFDL protection.

The NRC staff used NUREG-0800, Revision 2, "Standard Review Plan," Section 4.3, "Nuclear Design" in its review of the proposed TS change. Standard Review Plan 4.3 provides criteria for ensuring that the requirements of draft GDC 6 and final GDC 10 are met (Reference 15).

The TS changes requested by Duke Energy modify the references of the COLR in the administrative controls section of the TSs. The concept of the COLR was developed based on the guidance of NRC Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications" (Reference 16), which indicates it is acceptable for licensees to control certain operating parameters by specifying an NRC-approved calculation methodology. These parameter limits may be removed from the TSs and placed in an administratively-controlled document known as the COLR, which is defined in the TSs and required to be submitted to the NRC each operating cycle or when revised. The list of referenced NRC-approved methodologies used to determine these core operating limits is maintained in Harris TS 6.9.1.6.2 and Robinson TS 5.6.5.b. The proposed TS changes would add DPC-NE-1008-P to each of these TSs.

The proposed changes to the plant TSs also include the adoption of the methodology described in DPC-NE-2005-P-A, "Thermal Hydraulic Statistical Core Design Methodology." This licensing action was requested in a separate submittal, which was still under NRC review at the time the LAR requesting approval of DPC-NE-1008-P was submitted. However, it has subsequently been approved by the NRC staff (Reference 17) and added into the Harris and Robinson TSs. Thus, in this SE, the NRC staff is only considering the TS changes needed to implement the DPC-NE-1008-P methodology at Harris and Robinson.

3.0 TECHNICAL EVALUATION

The proposed changes to the Harris and Robinson TSs involve extending the COLR lists to include the DPC-NE-1008-P methodology reviewed as part of this submittal. Therefore, the primary focus of this technical review pertains to the acceptability for the use of the DPC-NE-1008-P methodology in Harris and Robinson reload design analyses.

3.1 Computer Code Models and Methodologies

Within the DPC-NE-1008-P methodology, the codes that the licensee proposed for use in nuclear design calculations are CASMO-5, SIMULATE-3, SIMULATE-3K and CMS-LINK. The

CASMO-5 code is a multi-group, 2-D transport theory model with a microscopic depletion model for burnable absorbers. CASMO-5 is used to calculate lattice physics parameters, including XS, pin power distributions and other nuclear data, which are utilized by SIMULATE-3 during core design calculations. The SIMULATE-3 code is a two-group, 3-D coarse-mesh nodal diffusion theory solver. SIMULATE-3 combines the nodal solution with the heterogeneous lattice solution from CASMO-5 to calculate pin power distributions. SIMULATE-3K is a transient version of SIMULATE-3 that solves the transient version of the diffusion equation including transient phenomena such as delayed neutrons, spontaneous fission, and decay. The other code that is used with this package is CMS-LINK, which processes data from CASMO-5 to produce multi-dimensional tables for the SIMULATE code to reference when performing core design and cycle depletion calculations. Each of the codes is further detailed in the following sections.

3.1.1 CASMO-5

The CASMO-5 code is a multi-group, 2-D transport theory code with a microscopic depletion model for burnable absorbers. CASMO-5 broadly resembles its predecessor code, CASMO-4, which has been previously approved by NRC staff for use in reload design analyses for the McGuire, Catawba, and Oconee nuclear stations (References 6 and 7). The CASMO codes are used to calculate lattice physics parameters, including XS, pin power distributions and other nuclear data, for use as inputs to SIMULATE-3. This is accomplished in both codes by solving the 2-D steady state Boltzmann transport equation for a completely heterogeneous nuclear fuel geometry. Although CASMO-4 and CASMO-5 contain many similarities, a number of differences exist in the details, methods, and capabilities of the two codes.

While Duke Energy provided some information about CASMO-5 in their initial submittal, this information was not sufficient for the NRC staff to fully understand the differences between the previously-approved CASMO-4 and the new CASMO-5. Additional information used by the NRC staff in this review was provided by the code developer for CASMO-5 and SIMULATE-3, Studsvik, and is documented in the Studsvik report SSP-14-P01/12-R, "CASMO5 PWR [Pressurized Water Reactor] Methods and Validation Report," Revision 1 (Reference 18). This report was referenced by Duke Energy in DPC-NE-1008-P for information on the qualification of CASMO-5.

3.1.1.1 Nuclear Data and Micro-group Cross Section Generation

One difference between CASMO-5 and CASMO-4 is the underlying nuclear data used to compute XS. CASMO-5 obtains the majority of its nuclear data from ENDF/B-VII.1 evaluated nuclear data files. Some specific nuclear parameters that are required, but not available in the ENDF files, are taken from TENDL-2012 data. The nuclear data file processing code, NJOY, is used in preprocessing to generate a data library containing 586 neutron groups and 18 gamma groups for use by CASMO-5. Preprocessing state of the art evaluated nuclear data files (like ENDF/B-VII.1) using NJOY is an acceptable industry standard approach for generating a data library to be used for lattice physics calculations, as is supplementing the ENDF data with recent TENDL data when necessary. The NRC staff also reviewed the specific fine group structure and determined that it contains more than sufficient detail and energy resolution for use in CASMO-5 and represents an increase in the fidelity of the underlying nuclear data when compared to the previously approved CASMO-4 methodology.

Input dimensions and material properties are normally provided by the user at cold (room temperature) conditions. At the beginning of its routine, CASMO-5 performs thermal expansion

calculations for the input geometry that adjust the dimensions and densities according to the state conditions of the analysis to be performed. According to the CASMO-5 methods and validation report (Reference 18), this thermal expansion is consistent with CMS-LINK and SIMULATE. Though the XS and pin power data provided to SIMULATE by CASMO are based on hot conditions, the 3-D nodal mesh in SIMULATE uses the cold dimensions (i.e., thermal expansion is not employed to adjust the geometry). This assumes that the effects of thermal expansion are captured in the XS and pin power data, and that the error introduced by using cold dimensions for the 3-D nodal mesh is small. The NRC staff has determined that these assumptions are appropriate and that any error introduced by using a 3-D nodal mesh based on cold dimensions would be negligible.

Once the geometry, isotopic compositions, and state parameters (temperatures, burnups, etc., at which to evaluate the problem) are set, CASMO-5 prepares problem-specific macroscopic XS using the NJOY-created data library's microscopic XS and the specific fuel and material properties (e.g., geometry, number densities). This is performed using first principles definitions for the XS and other details defined by the problem, as is done in many other lattice physics codes, including CASMO-4. Therefore, the NRC staff finds this method for generating problem-specific macroscopic XS in CASMO-5 acceptable.

During the formulation of macroscopic XS, it is necessary to address the resonance energy spectrum. Therefore, effective resonance energy XSs are also calculated during this process. This is performed in CASMO-5 mostly in the same manner as it was done in CASMO-4, though the range of resonance energies covered has been slightly reduced in CASMO-5 due to the increased resolution of the fine group data library used. CASMO-5 now defines the resonance region (where data is explicitly shielded) to be between 10.0 electric volt (eV) and 9118 eV in place of the previous resonance region, 4.0 eV to 9118 eV, defined by CASMO-4. The NRC staff has determined that the increased resolution of the fine group nuclear data appropriately justifies the reduced range over which effective resonance energy XS are calculated. The staff notes that XS resonances located in the removed energy range (4.0 eV to 10.0 eV) are generally broad and sufficiently represented using the high concentration of epithermal groups contained in the 586 group structure of the CASMO-5 nuclear data library. Therefore, the resonances in the removed energy range do not require special treatment in CASMO-5.

In calculating the effective resonance energy XS, CASMO-5 uses an equivalence theorem to relate tabulated resonance integrals to the particular heterogeneous problem, as was done in CASMO-4, albeit over a different energy range. A rational approximation for fuel self-collision probability is used to derive the equivalence expression, and contains the underlying assumption of no resonance scattering. This assumption is generally reasonable since the effective absorption cross section in the resonance energy region is much larger than the scattering cross section.

The final resonance integrals are then used to calculate the effective absorption and fission XS, while accounting for shadowing effects using a Dancoff factor correction. While CASMO-4 and CASMO-5 share the same classical definition for the Dancoff factor, the collision probabilities used to compute the Dancoff factors are obtained differently between the two codes. CASMO-4 uses collision probabilities obtained from a circularized problem (the Wigner-Seitz approximation) with white boundary conditions (BC). Conversely, CASMO-5 obtains these collision probabilities directly from the square pin cell geometry of the problem. In CASMO-5, the total XS is set to a large number in the fuel region (effectively making fuel black to neutrons), a uniform isotropic source is assigned in the fuel regions, and the transport problem for the flux

in the fuel region is solved using the Method of Characteristics (MoC). This solution is repeated twice, once with reflective BC and a second time with vacuum BC to obtain the two collision probabilities needed in the classical definition used to calculate the Dancoff factors. Because the same definition of the Dancoff factor is used as in CASMO-4, because MoC is routinely used to solve transport problems in square geometries (see Section 3.1.1.3 for more details), and because the same correction factor for individual pins is applied as in approved methodologies, the NRC staff determined that the CASMO-5 Dancoff factor calculation is acceptable.

3.1.1.2 The Micro-group Calculation

The final 2-D heterogeneous transport solution is performed by CASMO-5 using a lower number of energy groups (transport group structure), due to the computational requirements of the high fidelity calculation. The accuracy of this transport solution depends heavily on the preservation of the problem specific neutron spectra while collapsing from the Micro-group (586 group) energy structure to the transport group (19 or 35[1] group) energy structure. Therefore, pin cell calculations are performed using the Micro-group structure to provide an approximation for the flux solution. This approximate flux solution is then used to weight the energy condensation, which preserves the spectral fidelity of the micro-group calculations in the transport group structure for the final 2-D transport calculation.

In preparation for the micro-group calculations, the previously-described method for generating problem specific micro-group macroscopic XS, including resonance treatment and Dancoff correction, is repeated by CASMO-5 for each pin type[2] in the problem geometry. A micro-group pin cell calculation is then performed for each pin type.

In the micro-group calculations, each pin type is represented using 3 to 4 regions. Fuel, cladding, coolant, and other surroundings are each modeled using circular concentric regions. For burnable absorber bearing fuel pins, the fuel region is automatically subdivided into multiple annular rings in order to capture the effects of strong variations in the number density of the burnable absorber across the fuel region once the pin begins to deplete. For inert rods (water rods, guide tubes, etc.), a fuel-containing buffer region is placed outside of the coolant region to drive the flux in the cell for the calculation so that the cross section can be determined. Cells next to water gaps or walls (or both) are modelled using an additional region to represent these features. Once modelled, collision probabilities for each pin cell, with flat source in each region, are calculated using the FLURIG-2 method, as in CASMO-4. The resulting pin cell spectra are used to collapse the micro-group XS to the transport group structure.

The micro-group calculation appears to be performed in essentially the same manner as in CASMO-4 and CASMO-5. However, **[[**
]] the micro-group pin-cell calculation and the 2-D transport calculation **[[** **]]** In CASMO-5, the energy structure of the micro-group pin cell calculation (586 energy groups) provides sufficient detail that

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[1] The number of groups is chosen automatically by CASMO based on the problem definition. Nineteen groups are used for uranium dioxide (UO_2) fuel, while 35 groups are used for MOX fuel. Since Duke Energy is not seeking approval of the CASMO-5/SIMULATE-3 methodology for use with MOX fuel, only the 19-group calculation will be performed.

[2] Pins are grouped into types based on geometry and neutronic characteristics (e.g., enrichment, burnable poison loading).

accuracy in the 19-group 2-D transport calculation using the method of characteristics **[[ ]]** The NRC staff therefore determined that the micro-group calculation performed by CASMO-5 is acceptable.

3.1.1.3 The 2-D Transport Calculation

The 2-D heterogeneous transport calculation is performed over the entire lattice geometry. This geometry is spatially discretized into pin cell-sized macro-regions. Each of these macro-regions contains the micro-regions over which the transport calculation is performed. The XS for the micro-regions are condensed to the transport structure using the spectra obtained in the appropriate pin cell micro-group calculation detailed above.

The BC must be specified for each side of the problem (either a single lattice boundary or the edge of the MxN lattice problem being solved). Those available in CASMO-5 are mirror (reflective), periodic, black (vacuum), and rotational. The default is to apply a reflective BC on all four sides of the problem geometry; this is the set of BC most commonly used in conjunction with the SIMULATE codes.

As in CASMO-4, CASMO-5 performs the 2-D heterogeneous transport calculation using the MoC. This commonly used solution technique transforms the Boltzmann transport equation into systems of ordinary differential equations (ODE) that can be solved iteratively. To accomplish this, parallel characteristic lines are overlaid on the problem geometry in discrete angular directions determined by the quadrature set discussed later. The ray tracing routine, also discussed later, then defines the tracks, or segments, laying along the characteristics. In these segments, the flux is assumed constant (or linear) and the material XS is assumed constant. This allows the neutron transport equation to be represented as a 1-D ODE called a "characteristic equation" on each segment. Using an initial/boundary condition at the origin of each characteristic and sweeping along every characteristic in each direction until convergence is achieved yields net neutron currents between tracks, and surface and scalar fluxes for the problem.

There are several differences in the details of the techniques used in CASMO-4 and CASMO-5. One such difference is the source region approximation used. CASMO-4 assumes a flat, constant, source. While this assumption is generally reasonable, problems with large scattering regions such as water reflectors challenge the assumption and necessitate further subdivision of such regions. This in turn reduces computational efficiency. Therefore, CASMO-5 uses a linear source method. This allows for improved accuracy in the solution and addresses the challenge of large scattering regions without further subdividing the problem.

An additional difference is how the effects of anisotropic scattering are treated by and accounted for in the code. CASMO-4 and CASMO-5 both use P_1 approximations for the transport XS used in modelling anisotropic scattering in a problem. CASMO-4 uses the P_1 out-scatter approximation, as do many similar lattice physics codes. This approximation does not require the solution for the P_1 angular flux. CASMO-5, however, uses the P_1 in-scatter approximation to obtain the transport XS, which does require the P_1 angular flux. The treatment of anisotropic scattering is optional (determined by user input), but requires the use of an additional approximation to collapse the transport XS from the library group structure to the 2-D transport group structure. Additionally, the order of P_n equations used can be specified from 0 to 5.

The process for generating quadrature sets also differs slightly between CASMO-4 and CASMO-5. The quadrature set is the set of azimuthal (X-Y radial plane) and polar (Z axial direction) angles over which the parallel characteristics are laid. The spacing between the parallel characteristics is referred to as the ray spacing or track width. Quadrature sets are generated internally by CASMO-5. The polar angles used are generally predetermined by using the Tabuchi-Yamamoto optimal quadrature up to 3 polar angles – this is a departure from CASMO-4, which computes polar angles in such a way as to conserve the mean chord length of an infinite cylinder. If more than 3 polar angles are required, a standard Gauss-Legendre quadrature is used. In determining the optimal quadrature set for the problem, CASMO-5 slightly adjusts the azimuthal angles and ray spacing in order to ensure perfect reflection. Consequently, weights corresponding to each characteristic that are embedded in the systems of ODEs are also adjusted accordingly. Aside from the use of the Tabuchi-Yamamoto optimal quadrature for determination of polar angles, this routine is essentially the same in CASMO-5 as in CASMO-4.

Once the quadrature is determined, the ray tracing routine determines the tracks in each azimuthal direction for each characteristic and performs the sweeping iterations that eventually lead to the problem solution. Initially, the code passes through the problem geometry and assigns cell types according to what lies in and around a particular cell. Each of these cell types then has a specific ray tracing routine that is specialized for that type of cell. Each of the routines sweep along the characteristics, "accumulating" the flux solution, until convergence is achieved. The polar direction is treated by raising the determined azimuthal tracks out of the plane of the problem to the polar angles specified in the quadrature. The downward (negative Z) direction is not simulated, as the problem is assumed symmetric about the problem plane. This overall ray tracing approach is generally referred to as modular ray tracing, and it is understood by the NRC staff that it is applied the same way in both CASMO-4 and CASMO-5.

As discussed previously, when generating data for use in 3-D nodal diffusion codes such as SIMULATE-3, reflective BC are used. However, in reality, assembly surroundings vary and affect the actual flux distribution due to leakage and net currents between different assemblies. Therefore, a fundamental mode adjustment is made. This adjustment essentially modifies the determined flux distribution by adjusting the buckling of the infinite lattice results such that the lattice becomes exactly critical. In effect, this adjustment corrects for the leakage that would occur if the lattice were in a critical reactor rather than the infinite lattice originally simulated. This approach is turned off for calculations involving multiple segments, reflectors, or fuel storage racks, but is automatically activated for calculations designed to provide information to SIMULATE-3.

CASMO-5 uses nonlinear iteration techniques for the 2-D transport calculation, similar to those found in 3-D nodal diffusion codes, to accelerate convergence. This is achieved in CASMO-5 by superimposing a rectangular mesh on the geometry, defined on pin cell boundaries, and tallying averaged scalar fluxes and surface currents during each] sweep. Between sweeps, a set of diffusion-like coarse mesh finite difference (CMFD) equations are constructed on the rectangular mesh. The XS for these mesh cells are obtained using a flux-volume weighting over the cell. Diffusion-like coupling coefficients are constructed to match neutron currents and used to update the initial flux for the next iteration. Additionally, this acceleration technique can be used in multi-assembly calculations. More details on the acceleration technique are provided in Reference 19.

3.1.1.4 Burnup Calculation

CASMO-5 uses a two-step predictor-corrector process to predict burnup in simulated lattices. These two steps are identical in execution and only vary in the input parameters (fluxes, XS, number densities). The predictor step, performed after time step t_{n-1} calculations, uses the flux at time step t_{n-1} to predict number densities and XS for time step t_n. Using these parameters, a new flux spectrum calculation is performed. The corrector step is then performed based on these results before time step t_n. The number densities obtained are averaged with those from the predictor step. These averaged values are taken to be the actual number densities at time step t_n and are used as input for the calculations at time step t_n.

Prior to performing these burn-up calculations, the power normalization factor is determined in order to renormalize the predicted flux distribution to match the user designated power density. This ensures that the lattice is simulated at the correct burn-up rate, given the power that the bundle is expected to produce in its specific application in the core. Additionally, a depletion mesh is specified for use in the calculations. For most fuel pins, depletion is tracked using a single spatial zone per pin. For gadolinia bearing fuels, the fuel region is divided into annular rings in order to simulate layered depletion effects. Control rods and other burnable materials are treated according to user input. Optionally, azimuthal sub-regions (radial quadrants or octants) can also be specified.

Once the flux has been appropriately renormalized for the burnup calculations and the depletion mesh is determined, the two steps described above follow the same approach. Each step solves the Bateman equations, a system of first order ODEs. Each ODE represents the balance (sources and losses) of a nuclide's number density. The net balance then determines the rate of change of the nuclide number density. This system of ODEs is solved using the Chebyshev rational approximation method, a numerical technique that is used to solve the Bateman equations more efficiently than in CASMO-4, which is needed to model the extra fission products and chains that are now modeled in CASMO-5. CASMO-5 also now tracks (n, gamma), (n,2n), (n,3n), and (n,4n) absorption source reactions explicitly in the burnup method, rather than using an approximation like CASMO-4 uses.

3.1.1.5 Output and Edits

Following the 2-D transport solution and burnup calculations, CASMO-5 prepares the output. Output parameters include:

- Lattice and region averaged constants; XS and diffusion coefficients
- Neutron balances
- Reaction rates for individual nuclides in the fuel
- Power distribution
- Detector reaction rates
- Kinetics data
- Neutron poison data
- Discontinuity factors (DFs)

In preparing lattice and region averaged XS and diffusion coefficients, CASMO-5 first expands the flux obtained from the 2-D transport calculation to the micro-group (586 group) structure using the spectra from the micro-group pin cell calculations. This expanded flux is also used to

calculate reaction rates and the neutron balance. Lattice and region averaged XS may be printed in any available group structure, and are condensed and homogenized using a standard volume and flux weighting. Diffusion coefficients are obtained using the classical inverse of three times the transport XS relationship, and are likewise condensed and homogenized. Aside from the group structure, the same general approach is used in CASMO-4 and CASMO-5.

The neutron balance, normalized such that the lattice average Nu-fission rate is 1, may be printed for any part of the lattice and includes:

- Region-averaged fluxes
- Absorption rates
- Fission rates
- Nu-fission rates
- Leakages
- Out-scatter
- In-scatter
- Sources
- Currents

Fuel rod reaction rate edits are also available, along with fission rate distribution, heat surface flux, maximal linear loading, and maximal heat surface flux.

Assuming the power distribution from gamma irradiation is flat over the lattice, the gamma smeared power distribution can be obtained from a relationship relying on the fraction of fission energy released as gamma irradiation that escapes the fuel rod when fission occurs. This fraction is determined by comparison of power distributions from energy deposition calculations, and because slightly different relationships were used in CASMO-5 and CASMO-4, the value of the constant employed in the two codes is slightly different. Explicit gamma transport calculations are compared to those obtained from the previously mentioned relationship.

Detector reaction rates are also determined from the detector cell flux and detector material microscopic XS. This is the same as in CASMO-4. CASMO-5 also has the ability to model rhodium detectors, though little detail was provided in the materials from Studsvik. Because of this, and because a review of the licensing basis documentation for Harris and Robinson showed that rhodium detectors are not in use at the two sites, the NRC staff did not review the rhodium detector Y factor models in CASMO-5.

The basic delayed neutron data for the relevant nuclides is based on the commonly-used Tuttle data set. The effective delayed neutron yield, β_{eff}, for the standard 6 delayed-group precursors is computed in CASMO-5 using a spatially averaged and adjoint spectrum weighted formulation. Inverse neutron velocities may be calculated using adjoint spectrum weighting, flux spectrum weighting, or a combination of both. The adjoint spectrum is also used to compute the mean neutron generation time, which in turn yields the prompt neutron lifetime. As in CASMO-4, the adjoint flux can be calculated from the adjoint fundamental mode equation. However, a new option is available in CASMO-5 where the transport-equivalent CMFD data can be used to solve the 2-D adjoint diffusion equation – this option is recommended for specific circumstances involving reflectors.

Identical to CASMO-4, CASMO-5 contains an option to edit output for xenon and samarium data, if desired. The equilibrium xenon input option can be used to replace the poison's number densities that were calculated from the burnup calculations with equilibrium values based on the problem solution.

Finally, DFs are calculated such that the net currents at lattice interfaces are preserved in SIMULATE calculations. For single assembly calculations, net currents on the boundaries are 0, so the DFs are determined by the ratio between the surface and average fluxes in order to provide this information for use in SIMULATE. As discussed previously, the fundamental mode adjustment is applied for CASMO-5 calculations used to provide data to SIMULATE-3.

3.1.1.6 Additional Capabilities

CASMO-5 contains a number of additional capabilities that are outside the scope of this review, including, sub-mesh data generation, gamma transport calculations, energy deposition calculations, and fuel storage rack capabilities. During the NRC staff's audit during this review, Duke indicated that some of these capabilities may be used in future licensing applications. In general, the NRC staff will review these capabilities on a case-by-case basis as they are used to support future Duke Energy licensing submittals.

Traditionally, 3-D nodal diffusion codes like SIMULATE ignored treatment of the baffle and reflector and relied on user adjusted albedos to treat neutron reflection in these regions. However, modern nodal codes have reflector and baffle modeling capabilities and rely on data generated from lattice codes like CASMO-5. For the Studsvik CMS, CASMO-5 treats the baffle and reflector regions using a multi-assembly transport calculation. In these calculations, the baffle and reflector regions are treated as an assembly composed of slabs located adjacent to a fuel lattice. Black BC are used on the reflector edge opposite the fuel lattice to provide accurate flux shapes, reaction rates, and leakage effects. XS, currents, surface and average fluxes, and DFs are calculated for reflector and baffle regions as they are for any other region for use in SIMULATE. SIMULATE then models a homogenized baffle/reflector with the XS and DFs provided by CASMO.

Following the 2-D neutron transport calculation, gamma sources from neutron capture, fission and inelastic scattering can be calculated for each region. A characteristics-based gamma transport calculation largely similar to the MoC calculation used to solve the neutron transport problem can be performed. The gamma library for CASMO-5 is tabulated in 18 groups and is based on 70 group ENDF/B.VII.0 data, as this data remained unchanged in ENDF/B.VII.1. The calculation is performed directly in the 18 group energy structure, treating prompt and delayed gammas together or separately, as desired. The gamma XS are temperature independent, and therefore the data is evaluated at room temperature.

Although gamma fluxes are calculated for all parts of the lattice, only detector fluxes and response are normally included in output. As in CASMO-4, CASMO-5 has the ability to calculate the energy deposited in any micro-region from the slowing down and absorption of neutrons and from gamma absorption.

Similar to CASMO-4, CASMO-5 has the ability to perform calculations over more than one assembly, though implementation in CASMO-5 appears to be more flexible, with the capability to extend beyond 2x2 bundle matrices. In the multi-assembly calculation, each assembly may have its own design, rotation, control rod status, and physical and thermal hydraulic conditions.

Additionally, empty assembly locations (all moderator) can be modelled. The code maintains its full capability for these calculations, and generally follows the same calculational methodology. The fundamental mode calculation is turned off by default for multi-assembly calculations.

Though CASMO-5 has capabilities for modelling fuel storage racks, the NRC staff did not review the models and methods associated with fuel storage rack calculations as part of the present effort.

3.1.1.7 Conclusion On CASMO-5 Models And Methods

As discussed in the preceding sections, the NRC staff reviewed the models and methods employed in CASMO-5 and found them to be acceptable for the uses planned by Duke Energy as described in the DPC-NE-1008-P methodology report. Where CASMO-5 differs from its predecessor, CASMO-4 (which as discussed previously has been approved in plant-specific applications), the departures are understood by the NRC staff to be enhancements to either the accuracy of the solution (particularly with regard to the use of the updated ENDF/B-VII.1 data library and the finer energy group structure) or computational efficiency. The use of these models hinges on appropriate validation of CASMO-5, both as a standalone code and as part of a code system with SIMULATE-3.

3.1.2 CMS-LINK

CMS-LINK is a data processing tool used by the Studsvik CMS package of codes. It serves as a link between the major codes CASMO-5 and SIMULATE-3. Specifically, it takes the CASMO-5 output and generates a nuclear data library that SIMULATE-3 uses. The information collected from CASMO-5 output includes macroscopic two-group XS, two-group DFs at assembly boundaries, yields and microscopic XS for important fission products, detector constants, kinetics data, and pin-by-pin power distributions.

From this information, CMS-LINK arranges the data into multi-dimensional tables that characterize the effects of instantaneous and integrated perturbations to local core conditions. Specifically, this means it creates and tabulates partial differentials based on the effects of individual parameter perturbations calculated using branching and histories that are performed in CASMO-5 as supplemental (perturbed and repeated) calculations. The SIMULATE codes are designed to use this information to accurately determine the appropriate inputs to use to simulate a specific core condition. In this way, SIMULATE can accurately adjust and cover a wide range of core calculations without requiring new 2-D lattice calculations every time the state of the system changes. It is the NRC staff's understanding that the CMS-LINK code is a post-processing code that allows information to be transmitted between CASMO and SIMULATE. Traditionally, such codes do not receive detailed NRC staff review, because they do not perform detailed technical calculations. Additionally, the NRC staff understands from Duke Energy's description of the CMS-LINK code in DPC-NE-1008-P, that its use in the new methodology is unchanged from previously-approved Duke Energy neutronics calculations methodologies. Thus, the NRC staff did not re-review the CMS-LINK code as part of the review of the DPC-NE-1008-P methodology and determined that its use with CASMO-5 and SIMULATE-3 is acceptable.

3.1.3 SIMULATE-3

SIMULATE-3 is a two-group, 3-D nodal solver based on the quartic polynomial analytic nonlinear diffusion accelerated neutronics model that employs fourth-order polynomial representations of the intra-nodal flux distributions in both the fast and thermal neutron groups. SIMULATE-3 has been approved by the NRC in plant-specific applications for other Duke Energy sites in References 6 and 7. The code is based on modified coarse mesh diffusion theory calculational technique, with coupled thermal-hydraulic and Doppler feedback. The program explicitly models the baffle and reflector regions, eliminating the need to normalize to higher-order fine mesh calculations. SIMULATE-3 is capable of pin-power reconstruction calculations, which were used for validation calculations that are discussed later in this SE.

In order to ensure flux continuity at nodal interfaces and perform an accurate determination of pin-wise power distribution, SIMULATE-3 uses assembly DFs that are pre-calculated by CASMO-5. These factors are related to the ratio of the nodal surface flux in the actual heterogeneous geometry to the cell averaged flux in an equivalent homogeneous model and are determined for each energy group as a function of exposure, moderator temperature, fuel temperature, boron concentration, and control rod state.

The two-group model solves the neutron diffusion equation in 3-D, and the assembly homogenization employs the flux discontinuity factors from CASMO-5 to combine the global (nodal) flux shape and the assembly heterogeneous flux distribution. The flux discontinuity concept is also applied to the baffle/reflector region in both radial and axial directions to eliminate the need for normalization, or other adjustments at the core/reflector interface.

The SIMULATE-3 fuel depletion model uses tabular and functionalized macroscopic or microscopic, or both XS to account for fuel exposure without tracking the individual nuclide concentrations. Depletion history effects are calculated by CASMO-5 and then processed by the CMS-LINK code for generation of the XS library used by SIMULATE-3.

SIMULATE-3 is used to calculate the 3-D pin-by-pin power distribution in a manner that accounts for individual pin burnup and spectral effects. SIMULATE-3 also calculates control rod worth and moderator, Doppler, and xenon feedback effects.

As previously discussed, SIMULATE-3 has been approved by the NRC for use in plant specific applications (References 6 and 7). Because of this, the NRC staff did not re-review the models and methods associated with the SIMULATE-3 code in this safety evaluation. However, the NRC staff did implicitly review the use of SIMULATE-3 in the validation of the overall CASMO-5/SIMULATE-3 methodology and the development of power distribution uncertainty factors. Thus, while SIMULATE-3 has previously been found acceptable in plant-specific applications, its use with CASMO-5 in the DPC-NE-1008-P methodology is contingent on acceptable results as will be evaluated in the following sections of this SE.

3.1.4 SIMULATE-3K

SIMULATE-3K is an extension of SIMULATE-3, which is used for analysis of core transients. The spatial neutronics models in SIMULATE-3K are identical to those in SIMULATE-3. SIMULATE-3K solves the transient neutron diffusion equation incorporating effects of delayed neutrons, spontaneous fission in fuel, alpha-neutron interactions from actinide decay, and gamma-neutron interactions from long-term fission product decay. For the applications

reviewed in DPC-NE-1008-P, SIMULATE-3K is used only as part of the dynamic rod worth measurement (DRWM) methodology.

The SIMULATE-3K code was only used in DPC-NE-1008-P as part of the DRWM technique for providing control rod worth for McGuire, as will be discussed in more detail in Section 3.2.2.3 of this SE. Control bank worth predictions performed for benchmarking purposes were calculated only using CASMO-5, CMS-LINK, and SIMULATE-3 and did not involve the use of SIMULATE-3K. Just as the NRC staff does not consider approval of DPC-NE-1008-P to extend approval of the DRWM technique to Harris and Robinson, the use of benchmarking data from McGuire that was reduced from nuclear instrumentation signals using SIMULATE-3K does not imply approval of the use of SIMULATE-3K for Harris and Robinson. The NRC staff therefore did not review SIMULATE-3K for this application, and any future use of SIMULATE-3K at Harris and Robinson in licensing methodologies would require NRC review and approval.

3.2 Computer Code Validation

In order to demonstrate that the neutronics codes can accurately predict key parameters such as core reactivity, burnup and fuel isotopic composition, core physics parameters, and power distributions, code vendors and licensees must validate them against measurements. As DPC-NE-1008-P includes the first submittal of CASMO-5 for the NRC's review and approval, this section will discuss validation of CASMO-5 on its own. Though this validation only considers CASMO-5, it is restricted to demonstrating the code's effectiveness for calculating parameters relevant to the fuel system design in use at Harris and Robinson and does not constitute generic validation of the code. Since SIMULATE-3 has been previously approved for use by the NRC and is widely used in the industry, this SE will not consider validation of SIMULATE-3 alone. However, validation of the combined CASMO-5/SIMULATE-3 methodology, performed in DPC-NE-1008-P, will be discussed.

3.2.1 CASMO-5 Validation

Studsvik, the code vendor for CASMO-5, performed benchmarks against critical experiments, isotopic measurements from burned fuel pins, and higher order codes. Studsvik considers CASMO-5 to be validated for a wide variety of pressurized water reactor (PWR) lattice geometries fueled with low enriched uranium (LEU), gadolinia and integral fuel burnable absorber (IFBA), and various discrete absorbers and control rod types.

3.2.1.1 Critical Experiment Comparisons

Studsvik provided comparisons to the Babcock & Wilcox (B&W) 1810, B&W 1484, and KRITZ-3 critical experiments, which demonstrate performance in reactor lattice and pseudo-core configurations. The lattices modeled in these experiments were similar to those at the Harris and Robinson plants and included various combinations of UO_2 and MOX fuel rods, silver-indium-cadmium and boron carbide control rods, and gadolinia burnable absorbers. Eigenvalue (K_{eff}) predictions from these experiments were generally good, with a low and generally conservative bias and small standard deviation. B&W 1484 tests with boral sheets were predicted with a wider uncertainty band than others in the series, and tended to be **[[ ]]**. A substantial portion of the uncertainty is due to uncertainty in the boron content in the sheets, **[[ ]]** Because Duke Energy did not request approval to use CASMO-5 for spent fuel storage rack calculations, any **[[ ]]** for this core configuration does not

represent a concern to the NRC staff. In the analysis of the KRITZ-3 test, there appeared to be a slight increase in K_{eff} as a function of moderator temperature. The NRC staff observed that there are several possible explanations for this trend, including that the code may tend to predict a **[[** **]]** than the testing (which is consistent with other observations in the validation). Since the trend is very small and would result in a slight **[[** **]]** of the reactivity at reactor operating conditions, the NRC staff determined that it was acceptable. Fission rate predictions compared well to the KRITZ experiments and showed no significant bias. Calculated **[[** **]]** but since the CASMO-5/SIMULATE-3 methodology developed by Duke is not for use with MOX fuel at Harris and Robinson, this does not represent a substantial issue.

Studsvik also provided comparisons to the AEA Winfrith DIMPLE and TCA reflector critical experiments, which were intended to demonstrate CASMO-5's capability in modeling baffle and reflector regions. The DIMPLE experiments simulated the rectangular corner configuration of a PWR with and without a baffle, and the TCA experiments simulated a single fuel assembly with a steel/water reflector region. In modeling the DIMPLE experiment, the computed values of K_{eff} were close to unity for the cores with and without the baffle. The fission rate errors were slightly lower in the core with the baffle than the core without, though the mean and standard deviation of the error was very small in both cases. When compared to the TCA experiment, the reactivity effect of varying thickness steel baffles were very well predicted, and values of the delayed neutron fraction and ratio of effective delayed neutron yield to neutron generation time (β_{eff}/l^*) were very close to those determined in the experiment.

3.2.1.2 Isotopic Measurement Comparisons

In order to evaluate CASMO-5's ability to correctly deplete isotopes, Studsvik provided comparisons to the isotopic concentrations of destructively analyzed fuel pins. These pins were operated at Yankee Rowe, a PWR in the United States, and several PWRs and boiling water reactors in Japan.

The Yankee Rowe fuel rods were from cycles that used control rods for reactivity control rather than boron. However, in order to generate the isotopic compositions to compare to the measured data, Studsvik modeled single assemblies using boron (rather than control rods) for reactivity control. The CASMO-5 model also used the core average power history, since local power history was not available. **[[**

]], the NRC staff found the comparison to the measurements to be acceptable.

The staff found CASMO-5 also provided reasonably good predictions of the isotopic compositions of the fuel rods analyzed by the Japanese Atomic Energy Research Institute (JAERI). The Uranium (U)-235 and U-238 concentrations were predicted with a high degree of accuracy, often within 1%. Other actinides and most fission products, however, vary significantly in how well they are predicted, and showed as much as a **[[** **]]** difference from the burned rod concentrations. Often these larger errors were from isotopes that had very small

concentrations in the sample. Overall, CASMO-5 predicted those isotopes that are important to determining the core power distribution with an acceptable level of accuracy.

3.2.1.3 Higher-Order Computational Benchmarks

CASMO-5 was also benchmarked against higher-order codes in order to confirm the accuracy of the neutron transport calculation implementation and determine the reactivity effect of operating conditions in a plant, where high-fidelity test data does not exist, to augment the measurement data discussed in Section 3.2.1.1. The neutron transport benchmark is the Organization for Economic Cooperation and Development C5G7 MOX lattice benchmark, which uses MCNP4.1, while the benchmarks at operating conditions were calculated in MCNP6 by Studsvik.

The CG57 MOX lattice benchmark models a 16-assembly geometry containing 17x17 fuel lattices. Cross-sections are supplied for the benchmark (in a 7 energy group structure) to ensure that the benchmark results are only reflective of the accuracy of the CASMO-5 neutron transport solution implementation. The K_{eff} calculated by CASMO-5 and MCNP4.1 differs only by 5 percent mille, with a maximum power distribution error of 0.5%. The small error demonstrates that the transport solution has been properly implemented in CASMO-5.

The other benchmarks performed by Studsvik were comparisons to MCNP6 computations of a variety of lattice designs with varying boron concentrations, fuel enrichments, and burnable absorber loading (including both gadolinia and IFBA). The two codes both used the ENDF-B-VII nuclear data library for the calculations. Based on these comparisons, CASMO-5 tends to underpredict the boron worth and control rod worth. **[[**

]] Burnable absorber reactivity at beginning of life compares well to MCNP calculations with a high degree of accuracy for both IFBA and gadolinia. The NRC staff observed that fission rates also compare well, with very low average errors for standard rods and a slightly negative bias for IFBA rods. Generally, normalized fission rate predictions are within 1% of measurements. Overall, the comparison to MCNP6 shows that CASMO-5 provides acceptable calculation of a variety of integral parameters, including boron and control rod worths, and moderator temperature and fuel temperature defects.

3.2.1.4 CASMO-5 Validation Conclusion

In reviewing the CASMO-5 validation provided by Studsvik, the NRC staff found that the code compared well to the B&W and KRITZ critical experiments, demonstrating general capability to predict reactivity for a variety of lattice configurations. As demonstrated by comparison to the AEA Winfrith DIMPLE and TCA reflector critical experiments, CASMO-5 also provides good predictions for baffle and reflector regions. The ability of the code to perform burnup calculations was demonstrated by comparison to the Yankee Rowe and JAERI fuel rod isotopic measurements, which showed that the key isotopes for predicting power distributions were well predicted. Comparisons to MCNP6 showed that other integral parameters, such as boron worth, control rod worth, moderator and Doppler defects, and lattice fission rates are predicted well by the code.

The NRC staff therefore determined that CASMO-5 will provide acceptable lattice physics parameters to SIMULATE-3 for the Westinghouse 15x15 and 17x17 lattices in use at Harris and Robinson fueled with LEU (less than or equal to 5 weight %) and using IFBA, gadolinia, and discrete burnable absorbers. These parameters will be used as part of the broader CASMO-5/SIMULATE-3 methodology discussed in DPC-NE-1008-P, the benchmarking and evaluation of which will be discussed in the subsequent sections of this SE.

3.2.2 CASMO-5/SIMULATE-3 Benchmarks

To evaluate the performance of the CASMO-5/SIMULATE-3 code system, Duke performed benchmarks to five recent cycles at Harris Unit 1, five recent cycles at Robinson Unit 2, and four recent cycles at McGuire Unit 1. The units and cycles were chosen to encompass a wide range of reactor designs (3- and 4-loop Westinghouse plants), fuel designs, lattice sizes (15x15 and 17x17), burnable absorber types (gadolinia and IFBA), and fuel management strategies. Details about the plant and fuel designs are provided in Section 3.1 and Table 3-1 of DPC-NE-1008-P.

CASMO-5/SIMULATE-3 results were compared to measured hot zero power (HZP) and hot full power (HFP) critical boron concentration (CBC), HZP control bank worth, and HZP isothermal temperature coefficients (ITCs). Comparisons made to power distribution data were used to develop peaking factor uncertainties and will be discussed in Section 3.2.2.4.

3.2.2.1 Critical Boron Concentration Comparisons

The HZP CBC measurements are made during initial cycle startup physics testing at a condition with zero power, peak samarium, and no xenon. The HZP measurements are corrected to an all-rods-out (ARO) condition. At-power measurements are taken at or near HFP. Both HZP and HFP measurements are corrected for boron-10 depletion to ensure a common basis for comparing the computed results to the measurements. As discussed in the response to RAI 1 (Reference 3), predicted values of HFP CBC are calculated assuming that the soluble boron present in the reactor coolant system (RCS) contains a boron-10 isotopic concentration of 19.76 atom percent. Each measurement is corrected by multiplying it by the ratio of the best-estimate boron-10 concentration to the reference boron-10 concentration of 19.76 atom percent. This best-estimate boron-10 concentration is determined according to equation 1 in the RAI response, **[[**

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The means of the measured minus predicted (M-P) deviation (i.e., the bias) for both the HZP and HFP CBC are negative. The mean and standard deviation of HZP CBC M-P values presented in Section 3.2.1 of DPC-NE-1008-P are representative of the data presented in Table 3-2. This data set includes predictions from Harris, Robinson, and McGuire. The Harris and Robinson M-P data is consistently negative, while the McGuire M-P data is consistently positive. For the HFP CBC comparison, the mean and standard deviation of the M-P values cover the data provided in Table 3-3 except early in the cycle. The Harris and Robinson M-Ps also show clear trends over the course of the cycle, increasing from a negative number, rising to near or above zero around the middle of the cycle, and slowly decreasing towards the end of the cycle.

Soluble boric acid in the RCS is used to compensate for reactivity changes caused by fuel depletion as well as depletion of burnable absorbers. Differences in the fuel system design and

different burnable absorber use can thus account for substantial differences both in the actual CBC over the course of the cycle as well as the code's ability to predict the CBC. The differences seen in the predictive capability of the CASMO-5/SIMULATE-3 code system between the three plants are therefore expected, since McGuire uses IFBA and Harris and Robinson both use gadolinia burnable absorbers. Additionally, other core characteristics differ between the sites, and may contribute to differences in the ability to predict CBC. McGuire is a four-loop Westinghouse plant with 193 assemblies, while Harris and Robinson are three-loop Westinghouse plants with 157 assemblies. McGuire and Harris have fuel with a 17x17 lattice while Robinson has fuel with a 15x15 lattice. There are also differences in the number and specific design of the control rods at each site.

The NRC staff was initially concerned that the differences in M-P values among the plants would make them unsuitable for statistical pooling and thus invalidate conclusions regarding uncertainty in the CBCs. However, during the audit (Reference 11), the NRC staff learned that CBC calculational uncertainties are addressed on a plant-specific basis. As discussed in the response to RAI 2 (Reference 3), a measured-to-predicted CBC bias is developed for each plant that uses the CASMO-5/SIMULATE-3 methodology. The bias is developed based on operating history, and may vary as a function of burnup. The acceptability of the bias is confirmed as part of the core reload design process, and if necessary, it is adjusted to reflect changes in model performance.

Overall, though not without bias or trends, the predictions of CBC provided by the CASMO-5/SIMULATE-3 code system yield results within an acceptable range of the measured value. Typical startup test criteria from American Nuclear Society (ANS)-19.6.1 require the HZP CBC to be within plus or minus 50 parts per million of the predicted value; all the predicted HZP points and nearly all of the HFP points provided were within plus or minus **[[ ]]** parts per million of the measurement. Additional confidence in the predictive capability of the CASMO-5/SIMULATE-3 code system is introduced by the development of plant-specific biases, as discussed above, which further reduce these uncertainties.

3.2.2.2 Isothermal Temperature Coefficient Comparisons

ITCs are measured during startup physics testing at the beginning of the cycle, at HZP near-ARO conditions. To measure ITC, moderator temperature is changed and the corresponding reactivity change is measured by the reactivity computer. M-P deviations provided by Duke were consistently negative, which is conservative for HZP ITC. On average, McGuire HZP ITC M-P values were more negative than those for Harris and Robinson. The NRC staff found the predictions to be accurate, with an average bias and standard deviation both smaller than previously approved Duke Energy methodologies (References 6 and 7) for calculating core physics parameters.

3.2.2.3 Control Bank Worth Comparisons

Control bank worth is another parameter measured during startup physics testing. Each plant compared to in DPC-NE-1008-P uses a different system of control rod worth measurement – Robinson uses the boron dilution technique, Harris uses the rod swap technique, and McGuire uses the DRWM technique.[3] Harris and McGuire measure both shutdown and control rod bank

[3] NRC approval of the DPC-NE-1008-P methodology does not constitute approval of the dynamic rod worth measurement technique for Harris and Robinson.

worth, while Robinson only measures the worth of the regulating control banks (the ones used during power operations).[4] All the measurement techniques employed at the Duke Power sites have been previously found to be acceptable by the NRC staff and are in conformance with the ANS-19.6.1 startup physics testing standard.

In the dilution measurement technique, the RCS boron concentration is decreased continuously. This is compensated for by incremental insertion of the control rods in sequence. The reactivity computer is then used to calculate the reactivity added by each rod insertion. In the rod swap measurement technique, the differential rod worth of the predicted highest worth bank is determined using the boron dilution technique. Other banks are compared to the reference bank by inserting the test bank and withdrawing the reference bank to offset the negative reactivity addition. In the DRWM, described in DPC-NE-2012A (Reference 20), the banks are inserted and withdrawn at maximum speed and the excore detector signals are recorded. These signals are then post-processed with the reactivity computer and correction factors derived from SIMULATE-3 and SIMULATE-3K to solve the inverse point kinetics equation and determine the bank worth.

From the data presented by Duke, the relative error decreases as the predicted control bank worth increases. However, when looking at the absolute value of the control bank worth error, the spread is essentially constant as worth increases. On average, bank worth is slightly over predicted at McGuire and under predicted at Harris and Robinson. Because there is more data from Harris and Robinson than McGuire, the overall bias is negative. Except for one outlier point provided from McGuire, predicted bank worths are within plus or minus 100 percent mille of measured bank worths, and even the outlier point is within 15% of the measurement as required by test criteria. Overall, the predictive capabilities of the CASMO-5/SIMULATE-3 code system for control bank worth are similar to those of the previous CASMO-4/SIMULATE-3 system.

As with the CBC measurements and predictions discussed above in Section 3.2.2.1 above, there are sufficient differences between the plants, so the NRC staff did not consider the data sets to suitable for statistical pooling. This is expected due to the differences in measurement techniques between the sites as well as differences in fuel and core design. However, as discussed in Duke Energy's response to RAI 3 (Reference 3), the ability to pool the control rod worth data among the sites is not necessary. The data presented is used as a benchmark to demonstrate the overall performance of the CASMO-5/SIMULATE-3 methodology, but the uncertainty shown is not the uncertainty assumed in the safety analysis calculations.

The uncertainty assumed in the safety analysis is 10% of the total available control rod worth, which corresponds to the acceptance criterion from the ANS-19.6.1 standard for startup physics testing. In shutdown margin and trip reactivity calculations, Duke Energy thus only credits 90% of the total control rod worth. The total rod worth uncertainty presented for CASMO-5/SIMULATE-3 is lower than 10%, even when differences between plants are accounted for. As discussed in the response to RAI 3, the 10% uncertainty value is confirmed each cycle through startup physics testing. The NRC staff determined the CASMO-5/SIMULATE-3 code system uncertainty for control rod worth prediction is acceptable and ensures that the uncertainty assumed in the safety analysis will be met.

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[4] Testing of shutdown bank worth, while providing useful data for benchmarking nuclear physics methodologies, is not required by the ANS-19.6.1 standard used by plants for reload startup physics testing.

3.2.2.4 Power Distribution Measurement Comparisons

Power distribution measurements are made throughout the cycle by inserting fission chambers in the center instrument tube in assemblies around the core. The raw data from the fission chambers is then normalized and post-processed using signal-to-power conversion factors determined from the CASMO-5/SIMULATE-3 core models. This results in a 3-D relative power distribution for the entire core.

CASMO-5/SIMULATE-3 was used to model the core power distributions measured during recent cycles at Robinson, Harris, and McGuire, and the relative error between the predictions and the cycle measurements was calculated. A sample of the relative error between measured and predicted assembly average relative powers was provided in Figures 3-12 through 3-14 of DPC-NE-1008-P, which show the errors for a quarter core over the course of a cycle at each of the three plants. The NRC staff analyzed this data and found that the mean and standard deviation were both small and consistent between the three plants. The distribution of relative errors was centered very close to zero and was spread with a near-normal distribution almost entirely within ± 2%. There were no noticeable trends in assembly average relative power error with respect to cycle exposure and minimal trends with respect to core location. Only the assemblies at the core periphery showed a noticeable increase in the spread of the data, but the predictions were still accurate. The NRC staff thus determined that the CASMO-5/SIMULATE-3 methodology's predictions of three-dimensional core power distributions are acceptable.

3.3 Assembly Uncertainty Factors

Duke developed assembly uncertainty factors (or Observed Nuclear Reliability Factors (ORNFs)) for the power peaking factors $F_{\Delta H}$, F_q, and F_z, based on the dataset of measured and predicted relative powers discussed in Section 3.2.2.4. In DPC-NE-1008-P Figures 3-12 through 3-14, Duke Energy provided representative results of comparisons of measured and predicted assembly powers for recent operating cycles. However, in developing assembly ONRFs, Duke Energy used the full dataset, which included a wider range of operating history. Core locations where both the predicted and measured normalized assembly average power were less than one were excluded from the analysis. The NRC staff determined this to be reasonable because the assembly power uncertainties are more important for higher-powered assemblies.

The full dataset used to generate ONRFs was not provided to the NRC staff in the licensee's initial submittal. However, during the audit (Reference 11), the staff had the opportunity to review the calculation notebooks supporting the power distribution uncertainty factor calculations, which contained all of the relevant data. The amount of data supporting the ONRFs developed by Duke Energy was found to be so voluminous that it would be impractical to evaluate it other than on a statistical basis. Thus, though the licensee's original submittal only summarized the dataset through statistical analysis, the NRC staff determined that this level of information provided in the LAR was acceptable.

Duke Energy calculates the assembly uncertainty factors such that 95% of predictions augmented by the factor will be greater than the measurement (at a 95% confidence level, thus constructing a 95/95 one-sided tolerance limit). The statistical method used requires the data to pass a normality test at a 1% significance level or a non-parametric method is used to

determine the uncertainty factor. Regardless of whether or not non-parametric methods are used, the technique used for calculating the ONRFs for $F_{\Delta H}$, F_q, and F_z, based on the statistics determined from the data is consistent with previously approved Duke Energy methodologies (e.g., Reference 6). The formula for determining the ONRF is:

$$\text{ONRF} = \begin{cases} 1 - \text{Bias} + K_a\sigma_a, & \text{normal} \\ 1 - E_{mth}, & \text{non-parametric} \end{cases}$$

For the methodology based on normal statistics, K_a is the 95/95 one-sided upper tolerance factor based on the sample size, σ_a is the standard deviation of the relative error, and $K_a\sigma_a$ is presented as the "statistical deviation." For the non-parametric methodology E_{mth} is defined as the mth smallest relative error for a sample size *n* such that there is a 95% confidence level that 95% of the population has a standard error greater than this value. This is consistent with other NRC-approved methodologies for calculating non-parametric tolerance intervals. The statistical deviation for the non-parametric case is determined by subtracting E_{mth} from the bias.

Information regarding the datasets, including normality test results, evaluations of the biases and statistical deviations, and resulting ONRFs are provided in Table 3-6 of DPC-NE-1008-P. A summary of this data, and comparison to the results from the previously-approved DPC-NE-1005-PA methodology, is provided in Table 1 below. The data for all three parameters ($F_{\Delta H}$, F_q, and F_z) were found to not be normally distributed, so non-parametric methods were used to generate the ONRFs. Biases for the parameters were found to be of similar magnitude or smaller than in previous Duke methodologies and were less than or equal to zero. Statistical deviations are generally substantially smaller than previous methodologies, resulting in smaller ONRFs. This is as expected based on the NRC staff's review of the CASMO-5 code discussed above. Because the methodology used to generate the assembly uncertainty factors was unchanged compared to previously-approved Duke Energy methodologies, and because the reduction in the ONRFs is as expected based on the changes in the underlying nuclear physics calculation methodologies, and because the ONRF values were determined to lie within an acceptable range, the NRC staff determined both the ONRFs and the techniques used to generate them are acceptable.

Table 1. Comparison of Observed Nuclear Reliability Factors between CASMO-5/SIMULATE-3 and the previously-approved CASMO-4/SIMULATE-3 methodology.

Parameter	ONRF	
	CASMO-5/SIMULATE-3[5]	CASMO-4/SIMULATE-3[6]
$F_{\Delta H}$	[[ ]]	[[ ]]
F_q	[[ ]]	[[ ]]
F_z	[[ ]]	[[ ]]

3.4 Pin Power Uncertainty Factors

Duke also developed pin power uncertainty factors for LEU and gadolinia fuel pins using CASMO-5/SIMULATE-3. The LEU pin power uncertainty factor was developed based solely on comparisons between CASMO-5/SIMULATE-3 calculations and the B&W Urania Gadolinia

[5] Reference 5, Table 3-6
[6] Reference 6, Table B3-10

(B&W 1810) critical experiments discussed in Section 3.2.1.1 of this SE. These critical experiments were conducted with fuel rods at near-beginning of cycle (BOC) conditions, where gadolinia rods are not limiting due to substantially lower power density. Therefore, the gadolinia pin power uncertainty factor was developed based on uncertainties from CASMO-5 calculations of the B&W critical experiments combined with a SIMULATE-3 to CASMO-5 pin power reconstruction uncertainty, which was derived from 2x2 color-set calculations. The staff determined both the LEU and gadolinia pin power uncertainty factor methodologies to be consistent with the previously approved Duke methodologies.

3.4.1 B&W 1810 Critical Experiment Benchmarks

Duke modeled six of the critical configurations from the B&W 1810 critical experiments using both CASMO-5 alone and CASMO-5/SIMULATE-3. These six experiments were chosen because power distribution measurements were taken during the experiment at the mid-plane of the central assembly. Four of the experiments represent the 15x15 B&W fuel assembly lattice, which is very similar to the Westinghouse fuel lattice, and two represent the 16x16 Combustion Engineering fuel assembly lattice, which has larger gaps for control rods. Three of the cores have only LEU fuel and three of the cores contain LEU and gadolinia fuel. Fuel enrichments ranged from 2.46 to 4.02 weight % U-235.

When modeling the B&W critical experiments using SIMULATE-3, peripheral fuel pins were relocated to better model the partial fuel assemblies at the exterior of the experiment. As discussed in Section 4.4 of DPC-NE-1008-P, these changes were necessary because SIMULATE-3 was not designed to model the partial fuel assemblies that were present in the experiment. Duke stated that these changes were confined to the core periphery, to minimize impacts to predicted powers in the central region where measurements were taken. During the audit (Reference 11), NRC staff verified the changes made to model the experiment in SIMULATE-3 by reviewing the calculation notebook and confirmed that they were minor and acceptable. Additionally, the low uncertainty of the SIMULATE-3 pin-power distributions relative to the measurements supports Duke's conclusion that the changes have a minimal impact on the power prediction.

Predicted power distributions were normalized to an assembly average of 1.0 to compare to the measured power distribution. Relative errors were then calculated according to equation 4-1 in DPC-NE-1008-P. This is almost identical to the calculation performed in DPC-NE-1005-PA, Revision 1 (Reference 6), though DPC-NE-1008-P includes more cores from the critical experiment. Mean and standard deviations of the relative error were calculated for each core, for both LEU and gadolinia fuel rods, and with both CASMO-5 and SIMULATE-3. The mean and standard deviations of the error were comparable in magnitude to those presented in DPC-NE-1005-PA, Revision 1, though both the bias and the standard deviation were slightly reduced, demonstrating an increase in predictive capability. The comparison to the B&W critical experiments thus demonstrates the acceptability of the CASMO-5/SIMULATE-3 methodology's capability in predicting pin powers relative to experiments.

3.4.2 SIMULATE-3 LEU Pin Power Uncertainty

The method used to develop the SIMULATE-3 LEU pin power uncertainty is essentially the same as previously approved Duke neutronics methods and as the method discussed earlier for determining the ONRF. The uncertainty developed is a 95/95 one-sided tolerance limit on the error of the SIMULATE-3 pin power prediction relative to the B&W 1810 critical experiment data.

As before, the technique used to evaluate the uncertainty varies depending on whether or not the error is normally distributed. In this case, for the LEU pin power uncertainty evaluation, the error was normally distributed and only one technique for calculating the pin power uncertainty was discussed in the topical report. The LEU pin power uncertainty was defined as

$$\text{LEU Pin Power Uncertainty} = -\text{Bias} + K_p\sigma_p$$

where K_p is the 95/95 one-sided tolerance factor for a sample of pin power errors of a given size, σ_p is the standard deviation of the predicted-to-measured pin power error, and $K_p\sigma_p$ is once again defined as the statistical deviation.

LEU pin power uncertainties for CASMO-5 and SIMULATE-3 were developed using the measured and predicted relative powers from all of the LEU rods in the B&W 1810 critical experiments. The SIMULATE-3 LEU pin power bias and standard deviation were very close to the CASMO-5 values, which demonstrates the accuracy of the SIMULATE-3 pin power reconstruction methodology. Both SIMULATE-3 and CASMO-5 LEU pin power uncertainties were acceptably small.

3.4.3 Gadolinia Pin Power Uncertainty

The relative pin power errors for the gadolinia rods from the B&W critical experiments are substantially higher, for both CASMO-5 and SIMULATE-3, than the error for the LEU rods. Duke stated in the methodology report that this is because the average relative power for the gadolinia-loaded fuel pins from the experiment was 0.158, and the absolute value of the uncertainty is therefore much larger compared to the average power. This is an artifact of the near-BOC conditions at which the B&W critical experiments are performed. Later in the cycle, when the gadolinia has burned out of the fuel rods, their power would approach the LEU rod power and the gadolinia-loaded fuel rods could potentially become limiting.

To estimate the CASMO-5/SIMULATE-3 uncertainty at the time in core life when the gadolinia rods would potentially become limiting, Duke used a technique that is different from the one used to evaluate the LEU uncertainty but consistent with previously approved Duke Energy methodologies (References 6 and 7). Rather than using SIMULATE-3 for direct computation of the pin power uncertainty, this technique computed the gadolinia pin power uncertainty from CASMO-5 and combined it with an estimate of the uncertainty added by SIMULATE-3 through modeling and pin power reconstruction. Because the overall approach is consistent with precedent Duke Energy methodologies, the NRC staff finds it to be acceptable. Acceptability of the gadolinia pin power uncertainty developed using the approach will be discussed in the following sections.

3.4.3.1 CASMO-5 Gadolinia Pin Power Uncertainty from Critical Experiments

The method used to evaluate the CASMO-5 pin power uncertainty for gadolinia rods in the B&W 1810 critical experiments is similar to that described for LEU rods in Section 3.4.2 of this SE. However, there is one main difference – the bias and standard deviation are constructed from error (predicted minus measured) rather than relative error (predicted minus measured divided by measured). The pin uncertainty for CASMO-5 is then defined as:

$$\text{Gadolinia Critical Experiment Pin Uncertainty} = -\frac{\text{Bias}}{\bar{M}} + \frac{K\sigma}{\bar{M}}$$

Here, $\bar{M}$ is the average of the measured powers. This allows Duke to specify an average power that is greater than that actually experienced by the gadolinia rods in the critical experiment, in order to calculate the uncertainty in terms of the relative power expected when the gadolinia rods may become limiting. Because gadolinia fuel has a decreased thermal conductivity relative to LEU fuel, gadolinia fuel has a higher initial fuel temperature during transients. This in turn causes the centerline fuel melt limits to be challenged at lower power densities than LEU fuel, resulting in a decreased linear heat rate limit for gadolinia fuel. Duke Energy's proposed value for $\bar{M}$ reflects this decrease in the linear heat rate limit, and is consistent with previous Duke Energy methodologies (References 6 and 7). The NRC staff therefore finds the value of $\bar{M}$ to be acceptable for evaluating the gadolinia critical experiment pin uncertainty using the equation provided above.

The resulting CASMO-5 gadolinia pin power uncertainty from the B&W critical experiments is presented in Table 4-6 of DPC-NE-1008-P. The bias is much larger than for the LEU fuel, though the standard deviation is smaller. Combining these with the 95/95 one-sided tolerance limit factor, which for gadolinia is much larger than LEU due to the reduced number of data points, results in a gadolinia uncertainty that is slightly higher than the CASMO-5 LEU pin power uncertainty. This is expected and consistent with past approved licensing applications using this approach, and is therefore acceptable. The CASMO-5 gadolinia pin power uncertainty is also slightly smaller than the CASMO-4 gadolinia pin power uncertainty presented in DPC-NE-1005-P-A, Rev 1 (Reference 6). This is expected because of the overall improvement in predictive capability provided by moving from CASMO-4 to CASMO-5, and is acceptable. Therefore, the NRC staff determined the CASMO-5 gadolinia pin power uncertainty is acceptable.

3.4.3.2 SIMULATE-3 Pin Power Reconstruction Uncertainty

To estimate the uncertainty associated with the SIMULATE-3 pin power reconstruction, Duke analyzed eleven infinite lattice 2x2 colorset cases with both CASMO-5 and SIMULATE-3. These simulations encompassed a variety of combinations of fresh and burned fuel and varied the assembly burnup, fuel enrichment, number and enrichment of gadolinia rods, and the fuel lattice design (15x15 vs 17x17) and associated fuel rod diameter. Gadolinia concentrations of 2 to 8 weight % were considered, with between zero and 24 gadolinia rods per assembly. Initial uranium enrichment varied from 4.25% to 4.95%. The maximum assembly-averaged burnup reached in the calculation was approximately 40 gigawatt days per metric ton of uranium.

The data used to evaluate the CASMO-5 to SIMULATE-3 pin power reconstruction uncertainty is summarized in Table 4-9 of DPC-NE-1008-P. Biases for gadolinia pins were slightly higher than for LEU pins, but standard deviations were generally similar or slightly lower. Because the dataset was seen to be not conform to a normal distribution, the pin power reconstruction uncertainty was calculated with a non-parametric method like the one discussed in Section 3.3 of this SE. Both bias and standard deviation for the gadolinia fuel pins were lower than those of the previously approved DPC-NE-1005-P-A, Revision 1, methodology (Reference 6), and were acceptably low. A comparison of LEU pin uncertainties between the previously-approved methodology and the DPC-NE-1008-P methodology is provided in Table 2, below.

3.4.3.3 Combined SIMULATE-3 Gadolinia Pin Power Uncertainty

The CASMO-5 to SIMULATE-3 pin power reconstruction uncertainty described in Section 3.4.3.2 was combined with the uncertainty developed from the CASMO-5 analysis of the gadolinia critical experiments described in Section 3.4.3.1. The statistical combination of the uncertainties is defined as:

$$\text{SIMULATE-3 Gadolinia Pin Power Uncertainty} = -(\text{Bias}_c + \text{Bias}_{pr}) + \sqrt{(K_c\sigma_c)^2 + (K_{pr}\sigma_{pr})^2}$$

where the subscript *c* denotes statistics from the critical experiments and the subscript *pr* denotes statistics from the color-set analysis, and other terms have been previously defined. The resulting SIMULATE-3 pin power uncertainty for gadolinia-fueled rods is similar to, but slightly less than, that of the previously-approved DPC-NE-1005-PA, Revision 1 (Reference 6), and slightly greater than the LEU pin power uncertainty (as anticipated). A comparison of gadolinia pin uncertainties between DPC-NE-1008-P and DPC-NE-1005-PA, Revision 1, is provided in Table 2, below. Because Duke Energy used the same methodology in determining the gadolinia pin power uncertainty as in the previously-approved methodologies, and because the new fuel pin uncertainties are within an acceptable range of values, the NRC staff determined that the pin power uncertainties and the techniques used to generate them are acceptable.

Table 2. Comparison of fuel pin uncertainties between CASMO-5/SIMULATE-3 and the previously-approved CASMO-4/SIMULATE-3 methodology.

Fuel Type	Fuel Pin Uncertainty	
	CASMO-5/SIMULATE-3[7]	CASMO-4/SIMULATE-3[8]
LEU	[[ ]]	[[ ]]
Gadolinia	[[ ]]	[[ ]]

3.5 Statistically Combined Power Distribution Uncertainty Factors

Because pin power cannot be directly measured during power operation, Duke combined the assembly and pin uncertainties into a statistically combined power distribution uncertainty factor (SCUF) in order to provide an estimate of the complete uncertainty associated with the core model's ability to predict pin powers. This factor is to be applied both during the design of reload cores and during surveillance of an operating fuel cycle. Regardless of whether it is for use with LEU fuel or gadolinia, the SCUF is defined as:

$$\text{SCUF} = 1 - \sum_{i=1}^{n} \text{Bias}_i + \sqrt{\sum_{i=1}^{n} (K_i\sigma_i)^2}$$

[7] Reference 5, Section 4.8.
[8] Reference 6, Section 4.2 and 4.3.

where the biases and the statistical uncertainties ($K_i\sigma_i$) were derived for the assembly and pin in previous sections. This is the same methodology for computing the SCUF as was approved by the NRC in DPC-NE-1005-PA, Revision 1 (Reference 6).

The resulting LEU and gadolinia SCUFs for $F_{\Delta H}$, F_q, and F_z were presented in Table 5-1 of DPC-NE-1008-P. The assembly biases and uncertainties for each parameter were discussed in Section 3.3 of this SE, while the pin biases and uncertainties (which are not unique to the peaking factor under consideration) were discussed in Sections 3.4.2 and 3.4.3 for LEU and gadolinia pins, respectively. The resulting SCUFs were substantially smaller than the SCUFs presented in previously-approved Duke methodologies, due to the smaller biases and uncertainties at both the pin and assembly level. A comparison of the SCUFs between DPC-NE-1008-P and DPC-NE-1005-PA, Revision 1, is provided in Table 3, below. Because the methodology used to determine the SCUFs was the same as in the previously-approved methodology, and because the new SCUFs were comparable to the SCUFs generated using the previous methodology and were within an acceptable range of values, the NRC staff determined that both the values of the SCUFs and the method used to generate them were acceptable.

Table 3. Comparison of SCUFs between CASMO-5/SIMULATE-3 and the previously-approved CASMO-4/SIMULATE-3 methodology.

Fuel Type	LEU Fuel SCUF		Gadolinia Fuel SCUF	
	CASMO-5/SIMULATE-3[9]	CASMO-4/SIMULATE-3[10]	CASMO-5/SIMULATE-3	CASMO-4/SIMULATE-3
$F_{\Delta H}$	[[ ]]	[[ ]]	[[ ]]	[[ ]]
F_q	[[ ]]	[[ ]]	[[ ]]	[[ ]]
F_z	[[ ]]	[[ ]]	[[ ]]	[[ ]]

3.6 Conclusion Regarding DPC-NE-1008-P

The NRC staff reviewed the CASMO-5 models and methodologies, as well as its validation to experimental and higher-order codes, and found it to be an appropriate lattice physics code for the purposes discussed in DPC-NE-1008-P. It is therefore acceptable to provide information to SIMULATE-3 for the Westinghouse 15x15 and 17x17 lattices in use at Harris and Robinson fueled with LEU (≤ 5 weight %) and using IFBA, gadolinia, and discrete burnable absorbers.

The NRC staff also reviewed the models and methodologies discussed in the DPC-NE-1008-P report, which uses CASMO-5 as part of a code system for performing core-wide neutronics calculations with SIMULATE-3. The code system was found to have adequate performance, with some improvements over the previously-approved CASMO-4/SIMULATE-3 code system in use at other Duke Energy sites. The fuel assembly peaking factor ONRFs and SCUFs developed in the report for both LEU and gadolinia fuel were determined by the NRC staff to be acceptable, for up to 5% uranium enrichment and up to 8% gadolinia (by weight).

Overall, the NRC staff determined that the CASMO-5/SIMULATE-3 methodology described in DPC-NE-1008-P is acceptable for use in performing reactivity and power distribution

[9] Reference 5, Table 5-1.
[10] Reference 6, Table B5-1.

calculations as inputs to safety-related reload design analysis for reactors containing all-LEU fuel or mixtures of LEU and gadolinia fuel. The codes and methodology proposed for use by the licensee were found by the NRC staff to provide sufficiently accurate results for the fuel types in use at Harris and Robinson to form an adequate basis for ensuring that the reactor core is designed with appropriate margin to assure that SAFDLs are met during normal operation and AOOS, per draft GDC 6 and final GDC 10 requirements. Thus, the DPC-NE-1008-P methodology is approved for use on a plant-specific basis at Harris and Robinson.

The NRC staff does not consider approval of DPC-NE-1008-P to extend approval of the DRWM technique to Harris and Robinson. The use of benchmarking data from McGuire that was reduced from nuclear instrumentation signals using SIMULATE-3K does not imply approval of the use of SIMULATE-3K for Harris and Robinson. The NRC staff did not review SIMULATE-3K for this application, and any future use of SIMULATE-3K at Harris and Robinson in licensing methodologies would require NRC review and approval.

3.7 Evaluation of Technical Specification Changes

The TS changes originally requested by the licensee in Reference 1 stated that the approval date and revision number of the DPC-NE-1008-P methodology report would be listed in the COLR. In the May 4, 2016, letter (Reference 2), the requested TS changes were revised. The final set of proposed TS changes related to DPC-NE-1008-P are:

- Harris TS 6.9.1.6.2 is amended to include the following COLR reference:

  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).

- Robinson TS 5.6.5.b is amended to include the following COLR reference:

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

The proposed TS changes meet 10 CFR 50.36 and are consistent with Generic Letter 88-16 on the use of plant-specific methodologies in generating core operating limits, in that they appropriately identify the NRC staff's SE for each COLR reference. As such, and because DPC-NE-1008-P was determined to be acceptable on a plant-specific basis for Harris and Robinson as discussed above, the NRC staff determined that the proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official and the South Carolina State official were notified of the proposed issuance of the amendments on April 11, 2017. The State officials had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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, and there has been no public comment on such finding (81 FR 50746). Accordingly, the amendment meets 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 be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCE LIST

1. R. T. Repko, Duke Energy Progress, Inc. (Duke Energy), letter to Document Control Desk (DCD), U.S. Nuclear Regulatory Commission (NRC), "Application to Revise Technical Specifications for Methodology Report DPC-NE-1008-P Revision 0, 'Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors'," RA-15-0031, August 19, 2015, Agencywide Documents Access and Management System (ADAMS) Accession No. ML15236A044.

2. J. Elnitsky, Duke Energy, letter to DCD, U.S. NRC, "Supplemental Information for License Amendment Request Regarding Methodology Report DPC-NE-1008-P," RA-16-0023, May 4, 2016, ADAMS Accession No. ML16125A420.

3. J. Elnitsky, Duke Energy, letter to DCD, U.S. NRC, "Response to Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-1008-P, Revision 0," RA-16-0036, October 3, 2016, ADAMS Accession No. ML16278A090.

4. K. Henderson, Duke Energy, letter to DCD, USNRC, "Response for Additional Information (RAI) Regarding Application to Revise Technical Specifications for

Methodology Reports DPC-NF-2010, Revision 3 and DPC-NE-2011-P, Revision 2," RA-16-0042, November 17, 2016, ADAMS Accession No. ML16323A102.

5. Duke Energy, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," DPC-NE-1008-P, Revision 0, May 2015, Attachment 6, ADAMS Accession No. ML15236A044.

6. Duke Energy, "Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX," DPC-NE-1005-P-A, Revision 1, August 2004, ADAMS Accession No. ML051010321.

7. Duke Energy, "Oconee Nuclear Design Methodology Using CASMO-4 / SIMULATE-3," DPC-NE-1006-P, Revision 0, June 10, 2009, ADAMS Accession No. ML091630712.

8. J. Stang, USNRC, letter to P. Gillespie, Oconee Nuclear Station, "Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding the Use of CASMO-4/SIMULATE-3 Methodology for Reactor Cores Containing Gadolinia Bearing Fuel (TAC Nos. ME4646, ME4647, and ME4648)," August 2, 2011, ADAMS Accession No. ML101580106.

9. M. Barillas, USNRC, letter to R .T. Repko, Duke Energy, "Shearon Harris Nuclear Power Plant, Unit 1, and H. B. Robinson Steam Electric Plant, Unit No. 2 – Acceptance of Requested Licensing Action Re: Duke Energy Progress, Inc.'s Application to Revise Technical Specifications to Adopt Methodology Report DPC-NE-1008-P, Revision 0 (CAC Nos. MF6648 and MF6649)," December 1, 2015, ADAMS Accession No. ML15309A728.

10. M. Barillas, USNRC, e-mail to A. Zaremba, Duke Energy, "Duke Energy Harris Robinson LAR CASMO5_SIMULATE3 Audit Plan.docx," June 24, 2016, ADAMS Accession No. ML16180A098.

11. M. Barillas, USNRC, to Duke Energy Progress LLC, "Summary of the July 12 - 13, 2016 Audit Supporting the U.S. Nuclear Regulatory Commission Review of DPC-NE-1008-P, 'Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors' (CAC Nos. MF6648 and MF6649)," October 17, 2016, ADAMS Accession No. ML16235A238.

12. D. Galvin, USNRC, e-mail to A. Zaremba, Duke Energy, "Harris and Robinson RAIs – LAR to Adopt DPC-NE-1008-P, Revision 0 (MF6648 and MF6649)," September 7, 2016, ADAMS Accession No. ML16256A001.

13. Duke Energy, "Nuclear Physics Methodology For Reload Design," DPC-NF-2010, Revision 3, January 2016, Attachment 6, ADAMS Accession No. ML16125A420.

14. Duke Energy, "Nuclear Design Methodology Report For Core Operating Limits of Westinghouse Reactors," DPC-NE-2011-P, Revision 2, January 2016, Attachment 8, ADAMS Accession No. ML16125A420.

15. USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," NUREG-0800, Section 4.3, "Nuclear Design," Revision 3, March 2007, ADAMS Accession No. ML070740003.

16. USNRC, "Removal of Cycle-Specific Parameter Limits from Technical Specifications" Generic Letter 88-16, October 4, 1988, ADAMS Accession No. ML031130447.

17. M. Barillas, USNRC, letter to J. M. Frisco, Duke Energy, "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 Report DPC-NE-2005-P, Revision 5, 'Thermal Hydraulic Statistical Core Design Methodology' (CAC Nos. MF5872 and MF5873)," March 8, 2016, ADAMS Accession No. ML16049A630.

18. B. Haugh, Studsvik Scanpower, Inc., "CASMO5 PWR Methods and Validation Report," SSP-14-P01/012-R, Revision 1, December 18, 2015, Attachment 3, ADAMS Accession No. ML15355A283/ML15355A290.

19. K. Smith and J. Rhodes, "Full-Core, 2-D, LWR Core Calculations with CASMO-4E," PHYSOR 2002, Seoul, 2002.

20. Duke Energy, "Dynamic Rod Worth Measurement Using CASMO/SIMULATE," DPC-NE-2012A, August 1999, ADAMS Accession No. ML073100085.

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Date: May 18, 2017

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.157 TO RENEWED FACILITY OPERATING LICENSE

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

AND AMENDMENT NO. 253 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-23

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2, DOCKET NO. 50-261

DUKE ENERGY PROGRESS, LLC

1.0 INTRODUCTION

By supplement letter dated May 4, 2016 (Reference 1), as supplemented by letter dated November 17, 2016 (Reference 2), Duke Energy Progress, LLC (Duke Energy), the licensee for Shearon Harris Nuclear Power Plant, Unit 1 (Harris), and H. B. Robinson Steam Electric Plant, Unit 2 (Robinson), requested changes to the Harris and Robinson Technical Specifications (TSs). Specifically, Duke Energy requested U.S. Nuclear Regulatory Commission (NRC) review and approval of DPC-NE-1008-P, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," and DPC-NE-2011-P, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors" (all in Reference 1) and adoption of all three methodologies into the Harris TS 6.9.1.6.2, and Robinson TS 5.6.5.b, respectively.

This license amendment request (LAR) superseded two prior LARs, an August 19, 2015, submittal (Reference 3) and a February 3, 2016, submittal (Reference 4), which had sought separate approval of the DPC-NE-1008-P methodology and the revisions to DPC-NF-2010 and DPC-NE-2011-P. However, during the acceptance review of the February 3, 2016, submittal, the NRC staff found that the DPC-NF-2010 and DPC-NE-2011-P reports referenced the DPC-NE-1008-P report, thus creating a link between the two LARs. Following communication with the licensee (References 5 and 6), the February 3, 2016, submittal was withdrawn, and the May 4, 2016, submittal was used to supplement the August 19, 2015, submittal with the TS changes and methodology reports originally requested in the February 3, 2016, submittal. Because the NRC staff had already started reviewing DPC-NE-1008-P by the time the original submittal was supplemented, this safety evaluation (SE) only addresses the changes related to DPC-NF-2010, Revision 3, and DPC-NE-2011-P, Revision 2. The NRC staff's evaluation of DPC-NE-1008-P, Revision 0 methodology report and its adoption into the Harris and Robinson TSs is presented in a separate safety evaluation (SE) as Enclosure 3 (PROPRIETARY SE) and Enclosure 4 (nonproprietary SE) of the transmittal letter for this SE.

The DPC-NF-2010 methodology was previously approved by the NRC, most recently in 2003 (Reference 7). The DPC-NE-2011-P methodology was previously approved by the NRC in Reference 8, and was subsequently updated in Reference 9 (approved in References 10 and

Enclosure 6

11). These methodology reports and accompanying TS revisions would allow Duke Energy to perform core reload design analyses, confirm safety analysis assumptions, and develop power distribution limits and associated reactor protection system (RPS) trip functions. While the new revisions to the methodology reports reference all of the Duke Energy facilities with Westinghouse-designed nuclear steam supply systems (NSSSs), the NRC staff reviewed these reports on a plant-specific basis for application to Harris and Robinson only, for site-specific approval as requested in the licensee's application. Because the information was provided as a supplement to the LAR in Reference 3, the NRC staff's acceptance letter dated December 1, 2015 (Reference 12), constitutes the totality of the NRC staff's acceptance review for this requested licensing action. Subsequently, the NRC staff issued a request for additional information (RAI) by letter dated October 26, 2016 (Reference 13). Duke responded to these RAIs in a letter dated November 17, 2016 (Reference 2).

On February 2, 2016, the NRC staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (81 FR 5492) for the proposed amendment. Subsequently, by letter dated May 4, 2016, the licensee provided additional information that expanded the scope of the amendment request as originally noticed in the *Federal Register*. Accordingly, the NRC published a second proposed NSHC determination in the *Federal Register* on August 2, 2016 (81 FR 50746), which superseded the original notice in its entirety. The supplemental letters dated October 3, and November 17, 2016, provided additional information that clarified the application, did not expand the scope beyond the second notice, and did not change the staff's proposed no significant hazards consideration determination as published in the *Federal Register*.

2.0 REGULATORY EVALUATION

Duke Energy requested NRC review and approval of methodology reports DPC-NF-2010, Revision 3, and DPC-NE-2011-P, Revision 2, and their incorporation into the Harris and Robinson TS list of references for use in the core operating limits report (COLR). These methodologies for which NRC approval is requested are for site-specific use at Harris and Robinson only at this time. The methodologies described in the two reports will be used to determine operating limits that will be provided in the respective plants' COLRs. Specifically, DPC-NF-2010, Revision 3, describes a methodology for performing nuclear reload design calculations and determinations of the values of certain key reload design physics parameters and core power distributions. The results of this analysis are then used within DPC-NE-2011-P, Revision 2, to determine the power-dependent axial flux difference (AFD) limits, the rod insertion limits, and the power distribution inputs (f(ΔI) function) to the over-power delta-temperature and over-temperature delta-temperature (OPΔT and OTΔT) RPS trips.

The use of the methodologies described in DPC-NF-2010, Revision 3, and DPC-NE-2011-P, Revision 2, for determining core operating limits requires NRC staff review and approval prior to implementation at Harris and Robinson. These methodologies will be used to define limits on plant operating parameters, as discussed in the previous paragraph, which will be used to ensure compliance with the General Design Criteria (GDC) listed below from Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A.

Part 50 of 10 CFR, Appendix A, GDC 10, Reactor design, states the reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during normal operation, including the effects of anticipated operational occurrences (AOOs).

GDC 11, Reactor inherent protection, states, the reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

GDC 20, Protection System Functions, states the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Though Harris is licensed to the final GDC in 10 CFR Part 50, Appendix A, Robinson was licensed to the draft GDC published in the *Federal Register* on July 11, 1967. The NRC staff reviewed the Robinson Updated Final Safety Analysis Report (UFSAR) and determined that the plant specific design criteria listed in Section 3 therein are equivalent to the final GDC, for the purposes of this review.

The NRC staff used NUREG-0800, Standard Review Plan, Section 4.3, "Nuclear Design," (Reference 14) in performing the review of the subject methodology reports and associated TS changes.

Section 50.36 of 10 CFR, "Technical specifications," states, "each applicant for a license authorizing operation of a production or utilization facility shall include in his application proposed technical specifications in accordance with the requirements section." This regulation requires that the TS include items in the following specific categories: (1) Safety limits, limiting safety system settings, and limiting control settings; (2) Limiting conditions for operation (LCO); (3) Surveillance requirements; (4) Design features; and (5) Administrative controls.

The TS changes requested by Duke Energy modify the references of the COLR in the administrative controls section of TSs. The concept of the COLR was developed based on guidance of NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications" (Reference 15), which indicates it is acceptable for licensees to control certain operating parameters by specifying an NRC-approved calculation methodology. These parameter limits may be removed from the TSs and placed in an administratively-controlled document known as the COLR, which is defined in the TSs and required to be submitted to the NRC each operating cycle or when revised. The list of referenced NRC-approved methodologies used to determine these core operating limits is maintained in Harris TS 6.9.1.6.2 and Robinson TS 5.6.5.b. The proposed TS changes would add DPC-NF-2010, Revision 3 and DPC-NE-2011-P, Revision 2, to each of these TSs.

3.0 TECHNICAL EVALUATION

Duke Energy requested NRC site-specific review of methodology reports DPC-NF-2010, Revision 3, and DPC-NE-2011-P, Revision 2, and approval for their use at Harris and Robinson. The licensee also requested the changes to the TSs necessary to incorporate these reports into the two sites' lists of COLR references. This report, therefore, evaluates the revisions of the methodologies, in Sections 3.1 and 3.2, and the changes to the TSs, in Section 3.3.

3.1 Review of DPC-NF-2010, Revision 3

DPC-NF-2010, Revision 3, includes both editorial changes and technical changes. Primarily, the methodology had been updated to extend its applicability beyond the Catawba and McGuire Nuclear Stations to Harris and Robinson. However, other changes, both technical and editorial, were requested by the licensee.

3.1.1 Overview of DPC-NF-2010, Revision 3, Methodology

DPC-NF-2010, Revision 3, "Nuclear Physics Methodology for Reload Design," describes Duke Energy's core reload design process and covers a very wide range of subjects.

The early sections of the report provide high-level overviews of nuclear design in general and the tools used in Duke Energy's process. Chapter 1 of the report provides a general overview of the nuclear design process and provides some key design criteria that the nuclear designer must meet in the process of generating a core design. Chapter 2 provides a brief description of the fuel system designs in use at the plants where the methodology is to be employed. Chapter 3 provides a high-level overview of the nuclear code system, which is broken into descriptions of how input data is determined, how cross sections are prepared for reference by the core simulator model (i.e., using a lattice physics code), and a brief description of the core simulator model in SIMULATE-3.

Chapter 4 discusses the fuel cycle design process, including the process initialization and objectives. Ultimately, the final fuel cycle design is expected to meet a variety of limits that are verified in the fuel thermal-mechanical analysis, thermal-hydraulic analysis, the key safety analysis physics parameters, and maneuvering analyses. This section discusses how certain key fuel cycle parameters are calculated, including control rod worth, fuel burnup, reactivity coefficients and defects, and so on.

Chapter 5 provides more detail on the parameters calculated using the nodal core simulator code, many of which are also discussed in Chapter 4. These parameters include rod worths, shutdown margin, boron concentrations and worths, reactivity coefficients, and neutron poison worths. The discussion in this section focuses on the important considerations that must be taken into account when calculating these parameters. Chapter 6 gives an overview of the calculation of safety-related physics parameters, including core power distributions and power peaking factors, and how these are impacted by new core reload designs. The discussion in this section largely refers to other NRC-approved methodology reports for details.

Chapter 7 provides a very high-level overview of the three-dimensional (3-D) power peaking analysis, and refers to DPC-NE-2011-P for specific details on how this analysis is performed. Chapter 8 discusses how SIMULATE-3 calculates pin-power distributions and provides appropriate references to the pin-power uncertainties.

Chapter 9 discusses the development of operational core physics parameters once the final fuel cycle design has been completed. These parameters include predictions for startup physics testing, critical boron concentrations and worths, xenon and samarium worth and defect as a function of burnup, rod worths, reactivity coefficients, power distributions, and kinetics parameters. Chapter 10 details how the predicted startup physics test parameters discussed in Chapter 9 are to be compared to the actual startup physics test measurements. Chapter 11 details how the predicted power distributions discussed in Chapter 9 are compared to power

distribution measurements from the plants, and includes a discussion both of the measurement systems at the various sites and the statistical basis for comparing the prediction to the measurement. The development of statistically combined power distribution uncertainty factors is also briefly discussed.

The NRC staff reviewed both the previously-approved revision of DPC-NF-2010 (Revision 2) and the new Revision 3. The staff determined that though the report had previously been approved for specific use at the Catawba Nuclear Station (Catawba) and McGuire Nuclear Station (McGuire) in Revision 2, the core reload design process provided in the report would be broadly applicable to other plants, provided certain technical and editorial changes were made. These changes, made in Revision 3 to extend the applicability of the report to Harris and Robinson, are discussed in detail in the following sections.

3.1.2 Technical Changes in DPC-NF-2010, Revision 3

The NRC approval of DPC-NE-1008-P, Revision 0 was requested in Reference 3 to allow core physics simulations of Harris and Robinson using CASMO-5/SIMULATE-3. DPC-NE-1008-P, Revision 0 was found acceptable by the NRC staff, as documented in a separate SE of this amendment package. In Revision 3 of DPC-NF-2010, Duke Energy has requested incorporation of this CASMO-5/SIMULATE-3 methodology as an alternative to the existing CASMO-4/SIMULATE-3-based methodology (described in DPC-NE-1005, Reference 16) for core physics calculations and power distribution analyses. Since DPC-NE-1008-P has been found acceptable by the NRC staff for use at Harris and Robinson, and because approval for DPC-NF-2010, Revision 3, is sought on a plant-specific basis for Harris and Robinson, NRC staff determined that this change request was acceptable.

A technical change needed to extend DPC-NF-2010 to Harris and Robinson relates to the determination of the isothermal temperature coefficient (ITC) from plant measurements. The licensee stated that an alternative method for determining the ITC was added in order to reflect differences in the established test methodologies at McGuire/Catawba and Harris/Robinson. However, the NRC staff could not determine how the new method was different from the original one as presented in the methodology report. Additionally, while NRC staff could find discussions of the measurement methodology in the McGuire and Catawba, such discussions were not found in the Harris and Robinson Final Safety Analysis Reports (FSARs). Duke Energy clarified the distinction between the measurement techniques in response to RAI 1 (Reference 2). The "slope method," used at McGuire and Catawba, uses the reactivity computer to continuously calculate the derivative of the reactivity change resulting from a temperature change. As discussed in the RAI response, the slope is taken during a cooldown and a heatup and then averaged. This is distinguished from the "end-point method," used at Harris and Robinson and added to DPC-NF-2010 in Revision 3, where the total reactivity change resulting from a temperature change is divided by the temperature change to obtain the slope. Again, the slope is calculated for both a cooldown and a heatup and then averaged. The NRC staff concluded that the RAI response provided an adequate description of the new method added to Revision 3 of DPC-NF-2010 to measure the ITC. Because both measurement techniques ultimately result in an ITC that has been averaged over both positive and negative reactivity insertions, the NRC staff understands them to produce equivalent results. The NRC staff, therefore, determined the change was acceptable.

Several other technical changes were made to DPC-NF-2010 that are unrelated to the extension of the methodology to Harris and Robinson. Discussions of two-dimension (2-D)

integral rod worth and power distribution calculations were removed entirely. All of Duke Energy's modern core design and physics calculation methodologies use 3-D core simulators; thus, discussion of issues relevant to 2-D physics codes are no longer needed in DPC-NF-2010. Duke Energy also added discussions of the methodology used to generate core power distributions for initial conditions, the philosophy on confirming the acceptability of transient power distribution analyses relative to fuel thermal and power distribution limits, and the method by which power peaking, AFD, and rod insertion limits are developed. These discussions, contained in Chapters 6 and 7 of DPC-NF-2010, Revision 3, essentially parallel the methodology discussed in DPC-NE-2011-P, which will be discussed later in this SE. The NRC staff, therefore, determined that these changes are acceptable.

In Revision 3 of DPC-NF-2010, Duke Energy recognized that boron-10 depletion is an important source of uncertainty in measured critical boron concentrations and made appropriate changes to the methodology report. This is consistent with both DPC-NE-1008-P and DPC-NE-1005, the core physics methodologies, where measured values of critical boron concentration are corrected to the calculated value for benchmarking and development of measurement uncertainties. Because the change adds consistency with the approved core physics methodologies, the NRC staff determined that it was acceptable.

3.1.3 Editorial Changes in DPC-NF-2010, Revision 3

As with the technical changes, the most significant editorial changes made in Revision 3 of DPC-NF-2010 deal with the extension of the methodology to Harris and Robinson. The report is updated to state that it applies to Harris and Robinson in addition to Catawba and McGuire. Harris and Robinson are added to lists of Duke Energy plants with Westinghouse NSSS designs, and brief descriptions of their reactor design, fuel design, and incore instrumentation layouts are also added. Where references are made to plant TSs or FSARs, appropriate references to the Harris and Robinson TSs and FSARs are added. The use of two separate nuclear analysis methodologies is also clarified, with statements added to indicate that different pin and assembly power uncertainty factors are calculated for CASMO-4/SIMULATE-3 and CASMO-5/SIMULATE-3.

Additionally, a number of editorial changes serve to make the methodology appear less plant-specific. Specific references to Catawba and McGuire are removed, such that statements made in the report are generically applicable to all of the plants covered by the methodology. Control rod operation is clarified and references to specific banks of control rods are removed to accommodate design differences among McGuire, Catawba, Harris, and Robinson. Specific values of certain key core design parameters, such as shutdown margin and hot zero power moderator temperature, are removed and replaced with references to plant TS limits. The revision to the report also removes references to specific methodologies for rod ejection accident analyses, dropped rod accident analyses, and determination of safety analysis physics parameters and core operating limits. These are replaced with statements that an NRC-approved methodology must be used.

Other editorial changes to DPC-NF-2010 do not directly relate to the addition of Harris and Robinson to the methodology. These include modifications and deletion of some definitions, as well as the addition of a definition for relative power difference. Clarifications are made to fuel shuffle optimization terms, the rod swap technique for rod worth measurements, and the one-sided upper tolerance limits used to determine power distribution uncertainties. Coastdown

modes and durations, discussed throughout the methodology as important design considerations, are added to a list containing information needed for nuclear design.

All of the editorial changes discussed in this section of the SE are either directly related to the extension of Harris and Robinson to the DPC-NF-2010 methodology as requested in the LAR or are minor changes that do not substantively affect the technical content. The NRC staff, therefore, determined that these editorial changes were acceptable.

3.1.4 Conclusion Regarding DPC-NF-2010, Revision 3

The NRC staff reviewed the changes made to DPC-NF-2010 in Revision 3 and determined that they were acceptable, as detailed in Sections 3.1.2 and 3.1.3 above. To summarize, updates to the method include changes needed to apply it to Harris and Robinson as well as miscellaneous changes to the report to make it more compatible with modern codes and methods, clarify the report, and make the report more technology-independent. Because the NRC staff determined that the changes were acceptable in the context of this plant-specific review at Harris and Robinson, the NRC staff determined that DPC-NF-2010, Revision 3, is acceptable for use at Harris and Robinson.

3.2 Changes to DPC-NE-2011-P Revision 2

DPC-NE-2011-P, Revision 2, includes both editorial changes and technical changes. Primarily, the methodology has been updated to extend its applicability beyond Catawba and McGuire to Harris and Robinson. However, other changes, both technical and editorial, were requested by the licensee.

3.2.1 Overview of DPC-NE-2011-P, Revision 2, Methodology

DPC-NE-2011-P describes Duke Energy's methodology for performing maneuvering analysis to determine plant AFD operating space, rod insertion limits, and power distribution inputs to the OPΔT and OTΔT RPS trip functions. The maneuvering analysis is performed using the SIMULATE-3 nodal core simulator code to calculate power distributions based on a variety of input parameters.

Chapter 1 provides a brief overview of the method and its objectives. Chapter 2 discusses the methodology used for generating power distributions. These power distributions are based in part on the generation of abnormal xenon distributions, which are used to push axial peaking higher or lower in the core. Chapter 3 describes the uncertainty factors applied to the power distributions generated during the analysis discussed in Chapter 2. These uncertainties include the pin-power peaking and assembly peaking factors developed in DPC-NE-1005-P and DPC-NE-1008-P for CASMO-4/SIMULATE-3 and CASMO-5/SIMULATE-3 methods.

Chapter 4 describes the methodology used to determine the TS LCO and RPS limits. The margin to each proposed limit is calculated using equations found in this chapter. Separate calculations are performed for the loss-of-coolant accident (LOCA) peaking factor margin, the loss of flow accident (LOFA) departure from nucleate boiling (DNB) margin, the RPS limit DNB margin, and the centerline fuel melt (CFM) margin. This chapter also describes the procedure for determining the AFD-power level limits and the rod insertion limits.

Chapter 5 describes Duke Energy's "base load operation" methodology, which is similar to Westinghouse's constant axial offset control (CAOC) methodology. The option for base load operation is removed in the new revision to the topical report, as will be discussed later in this SE.

Chapter 6 describes the power distribution surveillance methodology. When comparing measurements to the limits, the limits on the power distribution are reduced by pre-calculated factors that account for perturbations from steady state conditions and uncertainty in the measurements and calculations. The chapter also includes a section discussing how plant measured parameters are monitored.

DPC-NE-2011-P has two appendices. Appendix A of the methodology report describes the computer programs used in computing the thermal limits. Appendix B of the report describes the methodology used to calculate the OTΔT and OPΔT setpoints.

The NRC staff reviewed both the previously-approved revision of DPC-NE-2011-P (Revision 1) and the new Revision 2. The staff concluded that though the report had previously been approved for specific use at the Catawba and McGuire, the maneuvering analysis concepts expressed in the report would be broadly applicable to other plants provided certain technical and editorial changes were made to the report.

The DPC-NE-2011-P report also refers the concept of Maximum Allowed Total Peaking (MATP) as the means of determining margin to DNB in Section 4.3, referring to the Duke Energy methodology report DPC-NE-2004-PA, Revision 1, "Core Thermal-Hydraulic Methodology Using VIPRE-01" (Reference 17), which discusses the concept in more detail. This methodology was approved for use at McGuire and Catawba, but was not requested to be added to the TS for Harris and Robinson in this LAR. Duke Energy stated in Attachment 9 to the LAR in Reference 1 that the MATP concept and methodology for determining margin to thermal limits described in DPC-NE-2004-PA, Revision 1, is applicable to Harris and Robinson without revision to the methodology report or plant TSs. The NRC staff's evaluation of this aspect of the LAR is presented in the next section.

3.2.2 Applicability of the Maximum Allowed Total Peaking Methodology to Harris and Robinson

The core DNB limits and OTΔT RPS trip function provide protection against DNB in the core, while accounting for core power and coolant temperature and pressure. These limits, as discussed in DPC-NE-2005-PA, Revision 5 for Harris and Robinson, are based on reference axial power distributions. To ensure DNB protection for axially skewed power shapes, there is a need to develop an f(ΔI) function that modifies the OTΔT trip setpoint as a function of the power imbalance between the top and bottom of the core.

DPC-NE-2004-PA, Revision 1 provides a methodology by which a series of MATP limits are generated. These limits take the form of lines of constant minimum departure from nucleate boiling ratio for a range of axial peaks and axial peak locations varying from the bottom to the top of the core. Each point on the curve represents a total peak ($F_{\Delta H}$ times F_Z) that yields a minimum DNB ratio (DNBR) equal to the design limit DNBR for the given axial peak and axial peak location. Axial power shapes for the MATP analysis are determined using a module that Duke Energy has built into VIPRE-01. These shapes must satisfy several constraints: the normalized axial power shape and its derivative must be continuous from the beginning to the

end of heated length, and the integral of the shape over the heated length (and normalized by the heated length) must equal one.

Power distributions from the maneuvering analysis, as discussed in DPC-NE-2011-P, are compared to the MATP limits. The margin between the MATP curves and the power distributions generated in the maneuvering analysis is used to determine the acceptability of the proposed f(ΔI) function and operational AFD limits. The specific process used to determine margin to the limits is discussed in more detail in Sections 4.3 and 4.6 of DPC-NE-2011-P.

Duke Energy stated, in Attachment 9 to the Harris and Robinson LAR (Reference 1), that the MATP methodology described in Section 5 of DPC-NE-2004-PA, Revision 1, is applicable to Harris and Robinson as well as Catawba and McGuire. The basis for the licensee's reasoning is that the methodology can be utilized to determine operating limits for DNB protection by any plant with a Westinghouse NSSS control and protection system, provided that the plant has the following three elements in place:

- An NRC-approved VIPRE-01 model for the unit being evaluated
- An NRC-approved critical heat flux (CHF) correlation for the fuel type
- An NRC-approved statistical DNBR limit for the fuel type

The NRC staff agrees that a Westinghouse protection system and NRC-approved VIPRE-01 model, CHF correlation, and statistical DNBR limit are necessary for application of the MATP methodology described in DPC-NE-2004-PA. However, the NRC staff does not agree that the methodology is otherwise generically applicable. The NRC staff's SE for DPC-NE-2004-PA, Revision 0, states that the report is "acceptable for referencing in the McGuire and Catawba core thermal-hydraulic analysis, subject to the conditions delineated in Section 4.0 of the [technical evaluation report (TER)]." This limits applicability of the report to McGuire and Catawba.

The staff considered, in its present review, the purpose of the original methodology and the determination documented in the SE and TER. The primary purpose of the DPC-NE-2004-PA report was to justify the use of a statistical core design (SCD) methodology for computing the design DNBR limit. The NRC staff's SE and contractor's TER focus almost exclusively on this aspect of the methodology report. The SCD methodology presented in DPC-NE-2004-PA was further refined in the NRC-approved methodology report DPC-NE-2005-PA, "Duke Energy Thermal-Hydraulic Statistical Core Design Methodology," Revision 5 of which was determined to be acceptable for use at Harris and Robinson by the NRC staff (Reference 18). Duke Energy's justification for the use of DPC-NE-2004-PA at Harris and Robinson references this report for the NRC approval of the VIPRE-01 model, CHF correlation, and statistical DNBR limits for the two plants.

The NRC staff reviewed the design bases for Harris and Robinson as documented in the plant UFSARs and found them to be compatible with the thermal-hydraulic design basis described in DPC-NE-2004-PA on which the MATP methodology is developed. The MATP methodology is a means of generating limits for the power distribution that will satisfy the DNB correlation limits, and is used in the DPC-NE-2011-P methodology to ensure that the f(ΔI) function of the OTΔT and OPΔT RPS trips provides sufficient DNB protection for axially skewed power shapes and that the operational AFD limits provide adequate DNB margin for the LOFA. As discussed by the licensee in Attachment 9 to Reference 1, the applicability of the MATP concept to a plant is

based on the existence of an NRC-approved thermal-hydraulic model (using VIPRE-01), an NRC-approved CHF correlation for the fuel in use at the plant, and an NRC-approved statistical DNBR limit for the fuel and plant. As previously discussed, the SCD methodology associated with the MATP methodology has been approved for Harris and Robinson, and appropriate VIPRE-01 models, CHF correlations, and statistical DNBR limits are thus available and approved for use. Based on this primary consideration, the NRC staff determined that the use of the MATP methodology described in Section 5 of DPC-NE-2004-PA, as justified by Attachment 9 to Reference 1, is acceptable. However, the staff's acceptability determination for this report should not be interpreted as generic approval of the use of DPC-NE-2004-PA for any Westinghouse plant. Rather, it is merely that the MATP methodology is acceptable for reference, based on the justification provided by the licensee in this LAR.

3.2.3 Technical Changes in DPC-NE-2011-P, Revision 2

Technical changes made to DPC-NE-2011-P in Revision 2 can be separated into two groups: changes that directly support the extension of the methodology to Harris and Robinson and other technical changes to the methodology.

3.2.3.1 Changes Supporting Extension to Harris and Robinson

The addition of the CASMO-5/SIMULATE-3 code system, discussed in DPC-NE-1008, for performing nuclear physics and power distribution calculations is a significant technical change that supports extension of the methodology to Harris and Robinson. As discussed previously in Section 3.1.2, the addition of DPC-NE-1008 to this methodology is consistent with its usage at Harris and Robinson and is thus acceptable.

Another set of changes was made to clarify that the LOFA is not necessarily the limiting anticipated operational occurrence (AOO) where the initial power distribution does not change as a result of the event. This change is made because the current AREVA analyses for Harris and Robinson suggest that other AOOs may be more limiting than the LOFA. The method now states that the AFD operating limits must consider the limiting AOO where initial condition peaking does not change as a result of the transient, rather than just stating that they must consider the LOFA. Because the limiting AOO is considered, the NRC staff determined this change is acceptable.

Several changes were made to DPC-NE-2011-P to make it more consistent with the Harris and Robinson licensing bases. Since the new revision of the document is being reviewed on a plant-specific basis for Harris and Robinson, any changes do not affect the use of the report at other sites. The termination of the MODE 1 boron dilution transient was modified to be consistent with discussion of the event in the Harris and Robinson FSARs. The heat flux hot channel factor surveillance discussion was modified to state that limits are not imposed on the top and bottom 10-15% of the core. Harris, McGuire, and Catawba exclude the top and bottom 15% of the core from the surveillance, while Robinson excludes 10%, due to uncertainties in the measurement in these regions. The NRC staff determined because these changes are consistent with the existing licensing basis of Harris and Robinson, they are acceptable.

A new section is added to Appendix B describing the OPΔT and OTΔT setpoint methodology for Harris and Robinson. The new section refers back to the existing section on OP and OTΔT setpoints for McGuire and Catawba, though it states that NRC-approved models, rather than specific methodologies, are used to evaluate transients as well as the effects of different

dynamic compensation terms. The methodology for McGuire and Catawba is unchanged. The new section for Harris and Robinson eliminates specificity that limits the setpoint methodology to the specific accident and transient analysis methods in use at Catawba and McGuire. The transient and accident analysis methods discussed in the setpoint methodology are used to evaluate the acceptability of the proposed setpoints (i.e., to ensure that they provide acceptable transient analysis results). Though it is necessary to use an acceptable method for transient or accident analysis when performing this evaluation, the particular choice of method is not important. The NRC staff determined that the revision is acceptable and necessary to apply the setpoint methodology at Harris and Robinson, which use different transient and accident analysis methods from Catawba and McGuire.

Another change to the method allows the $f_1(\Delta I)$ breakpoints for OTΔT and $f_2(\Delta I)$ breakpoints for the OPΔT functions to be modified if the measured F_Q exceeds the limit set by CFM analyses. Previously, the methodology only allowed the $f_2(\Delta I)$ breakpoints to be modified. This change was made explicitly because the $f_2(\Delta I)$ function for OPΔT is not active at Harris, and thus the same protection that would be provided by adjusting the $f_2(\Delta I)$ breakpoints at other sites must be provided by adjusting the $f_1(\Delta I)$ breakpoints at Harris. The OTΔT trip function is capable of providing this protection. Thus, the NRC staff considers this change to be acceptable.

3.2.3.2 Other Technical Changes

One of the primary changes to DPC-NE-2011-P, Revision 2, is to the generation of xenon transients, which are applied in setting the power distributions used in developing the operational AFD-power level limits. In the approved version of the methodology, the 3-D core simulator code is used to model an equilibrium xenon core at a selected initial condition. **[[**

]] Sample initial and transient conditions are provided in Table 1 of DPC-NE-2011-P, Revision 2. Once initiated, the transient is allowed to proceed for a period of time to obtain a range of xenon distributions. As discussed in the response to RAI 2.a (Reference 2), a variety of xenon transients are run to obtain xenon distributions initiated from a range of conditions.

In the approved methodology, the control rod positions used to initiate the xenon transients are specified to be "at or near the expected rod insertion limits" (Section 2.3.1 of Attachment 7 to Reference 1), to allow for some flexibility in **[[**

]] However, Duke Energy has found through experience in applying this methodology that **[[**

]] can produce extremely conservative xenon transients. When combined with possible control rod insertions and power levels to generate a set of 3-D power distributions for the maneuvering analysis, **[[**

]]. This is best demonstrated in Figure 1 of the justification to Change 2-3 in Attachment 7/8 to Reference 1, which shows the results of a sensitivity study on Control Bank D position.

As discussed further in the justification of Change 2-3 (Attachment 7/8 to Reference 1), the licensee does not believe that xenon distributions **[[**

]], and it is justifiable to exclude these xenon distributions in the development of the power distributions for the maneuvering analysis. The change to the methodology requested by Duke Energy specifies that, when generating a xenon distribution for this purpose, the control rods will be

inserted at most to the rod insertion limits [[

]].

The NRC staff reviewed the Harris TS 3/4.2.1 and Robinson TS 3.2.3 and found that both plants had LCOs and surveillance requirements associated with AFD. At both sites, an alarm will sound if the AFD is outside of the operational limits, and the TS requires that AFD be restored in the target band within 15 minutes or else power must be reduced. If the alarm is inoperable, a surveillance check must be logged every hour at Harris and every 15 minutes at Robinson. Because of the existence of the alarms and immediate actions required if AFD is found to be outside the TS limits, the NRC staff agrees with Duke Energy's position that [[

]] However, the NRC staff consider it important to understand the impact this change will have on the operating margins at the plants as well as how the licensee will ensure that the AFD space is appropriately spanned, at a sufficient variety of possible core conditions and with sufficient margin for uncertainties.

As demonstrated in Figure 2 of the justification to Change 2-3 in Attachment 7 to Reference 1, the primary impact of this methodology change is to [[

]]. As discussed in the licensee's response to RAI 2.b (Reference 2), this is because [[

]]. This effect is well illustrated in Figure A of Duke Energy's response to RAI 2.b (Reference 2), as well as in the LOCA F_Q limit margin plot provided in Figure 2 of Change 2-3.

Duke Energy stated that the new methodology maintains an appropriate span of the operational AFD space in response to RAIs 2.a and 2.c. The design xenon transients are selected to create power distributions that exceed the AFD allowed by the operational AFD limits in the TS. These AFD limits are specified such that there is sufficient margin to the thermal limits when the core reaches the AFD limit, [[

]]. Furthermore, as discussed previously, a variety of xenon transients are performed at different times in core life and are initiated [[

]] As indicated by the licensee in response to RAI 2.a and in Table 2 of the methodology report, the xenon distributions that result from these transients are combined with a [[

]] to generate the power distributions used in determining the margin to the thermal limits. In producing power distributions, the methodology considers combinations of control rod positions spanning the rod insertion limits and [[

]].

Duke Energy also noted, in response to RAI 2.a, that [[

]].

The NRC staff found the changes in margin, which occur [[ ]], to be acceptable based on Duke Energy's justifications, as discussed above. The NRC staff also found that, by analyzing power distributions exceeding the operational AFD limits, developing xenon and power distributions based on a wide variety of initial and transient conditions, and appropriately incorporating uncertainties in plant operating parameters, Duke Energy's proposed methodology provides adequate coverage of the operational AFD space. This is true even [[ ]], where the NRC staff found Duke Energy's approach discussed above to be acceptable, in that it provides acceptable margin between achievable AFDs and analyzed AFD conditions with positive thermal margin. The NRC staff therefore determined, based on the foregoing discussion, that this particular change to the methodology for developing xenon distributions is acceptable.

As part of the DPC-NE-2011-P methodology, Duke Energy determines the margin to the proposed core thermal limits as determined by the LOCA, LOFA,[1] and CFM analyses. In determining the margin to the limits, the calculated design axial and radial peaking factors and linear heat rates are adjusted to account for uncertainties,[2] power deposition in the fuel, additional reserved design margin, and quadrant tilt before comparison to the limits. The factor used to account for quadrant power tilt ratio (QPTR) is presented in the methodology as TILT. As a change in Revision 2 of DPC-NE-2011-P, the value of the TILT power peaking penalty was removed; in its place, the discussion in Section 3.2 of the methodology was revised to state that an appropriate allowance corresponding to a 2% excore tilt will be provided in the plant COLRs. This change is intended to make the methodology less plant-specific – since the core characteristics of Harris and Robinson are different from McGuire and Catawba, the value of TILT corresponding to a 2% excore tilt is expected to be different. Because the licensee stated that the value of TILT corresponding to a 2% excore QPTR will be used in the margin analysis and provided in the plant COLRs, the staff finds this change to be acceptable.

In the existing methodology, the heat flux hot channel factor (F_Q) and nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) surveillances discussed in Section 6 also include penalties for quadrant tilt. Duke Energy has proposed to remove the TILT penalty entirely from these surveillances in

[1] The LOFA analysis, as noted above, is not necessarily limiting. LOFA is used here, as in the revised DPC-NE-2011-P, to mean the limiting AOO where the initial condition power peaking does not change over the course of the transient.

[2] Uncertainties accounted for in the margin calculation include power distribution uncertainties from the nuclear analyses, engineering factors, and rod and assembly bow (if not considered in analyses).

Revision 2 of DPC-NE-2011-P. There are fundamentally two aspects of the licensee's rationale for this change. First, as discussed above, the core thermal limits are already verified to be appropriate with design power distributions that include an allowance for quadrant tilt. Thus, the limits themselves account for tilt. Second, Duke Energy argues that quadrant tilt was only included in the surveillance to account for the potential for a core power perturbation causing tilt between surveillances. This is because the measurement of the actual core power distribution during a surveillance implicitly includes any real core tilt. Duke Energy then argues that inclusion of a penalty for tilt between surveillances is unnecessary because:

- TILT is included in the verification and development of thermal limits,
- quadrant tilt is indicated and monitored in the control room,
- F_Q and $F_{\Delta H}$ are trended and extrapolated to see if they are expected to exceed limits between surveillances,
- severe changes in quadrant tilt are not expected during normal operation,
- conservatism in the analysis methodology provides additional assurance that the $F_{\Delta H}$ and F_Q limits will not be exceeded prior to the next surveillance interval, and
- the Westinghouse plant Standard Technical Specifications (as provided in NUREG-1431, "Standard Technical Specifications for Westinghouse Plants" (Reference 19), do not include QPTR penalties for F_Q and $F_{\Delta H}$ surveillances.

The NRC staff agrees with the licensee's justification that the purpose of the TILT penalty in surveillances is to cover the interval between surveillances, based on an assessment of the DPC-NE-2011-P methodology and its consistency with other surveillance formulations used at other facilities. Though this change represents a reduction in conservatism relative to the existing methodology, sufficient conservatism in the overall methodology will be maintained by accounting for quadrant tilt when developing and verifying the core thermal limits. The NRC staff also notes that the existing design and analysis methodologies currently show substantial margin to peaking factor limits and thus conservatism will be maintained even if it is reduced in the new methodologies. The NRC determined that this change was acceptable because adequate conservatism was retained in the methodology, both Harris and Robinson have TS limits for QPTR that prevent excessive tilt, and because the change is consistent with existing surveillance requirements in the rest of the Westinghouse pressurized-water reactor (PWR) fleet.

DPC-NE-2011-P, Revision 2, also defines the method used to determine the OPΔT f(ΔI) function. This section originally specified that the OPΔT f(ΔI) trip was usually zero, and a check was performed at the plant's upper power level limit to verify that a penalty was not required to assure that limits would be met. However, as fuel and core designs have changed since the methodology was originally produced, the OPΔT f(ΔI) has been put into regular use at the Duke Energy Westinghouse plants. Thus, the section was modified to allow the use of a non-zero OPΔT f(ΔI) function and make several clarifications about how additional margin to CFM limits can be incorporated into the OPΔT and OTΔT trip functions. Since these trips are standard parts of PWR protection systems, the NRC staff determined that the proposed changes to this section are acceptable. However, the staff also notes that additional changes to the Harris TSs, beyond those proposed in the LAR, are needed to implement an OPΔT f(ΔI) function at Harris. Such changes are outside the scope of the changes requested in this LAR.

In Revision 2 of DPC-NE-2011-P, discussion of and references to the post-processing codes SMARGINS and MARGINPLOT were removed. Duke Energy does not consider these codes to

be part of the DPC-NE-2011-P methodology and believes it is appropriate to remove them from the plant licensing bases. The NRC staff determined based on the descriptions provided in the previous revision to DPC-NE-2011-P that SMARGINS and MARGINPLOT are only used for simple post-processing of results from nodal physics codes. As such, NRC staff concurs that removal of these codes from DPC-NE-2011-P is appropriate and the changes are, thus, acceptable. As discussed elsewhere in this safety evaluation, because the new revision of the document is being reviewed on a plant-specific basis for Harris and Robinson, any changes do not affect the use of the report at other sites.

Base load operation was also removed as an option in the update to the methodology. This methodology was essentially analogous to Westinghouse's CAOC methodology, where operation was restricted to within a certain axial offset around a predicted target AFD. The current TSs at Catawba and McGuire do not currently support such an operational mode, and removing it from the topical report would make operation at Harris and Robinson more consistent with the existing Duke Energy plants. Since the base load mode of operation is not needed because the methodology maintains appropriate AFD limits using other operating modes, and because its removal will provide operational consistency between the Duke Energy Westinghouse plants, the NRC staff determined that this change was acceptable.

Revision 2 of DPC-NE-2011-P also includes several minor technical changes. One such change clarifies the times in core life in which the maneuvering analysis is performed. The modification retains the minimum amount of conservatism necessary in modeling time in core life, and can help in capturing reactivity peaks associated with burnout of burnable poisons; it is thus acceptable to the NRC staff. Another change specifies that failure to meet F_Q^M limits and complete the required actions should not result in a reactor trip, but instead an orderly shutdown to MODE 2. This is consistent with the requirements of TS 3/4.2.2 at Harris and TS 3.2.2 at Robinson, and thus the NRC staff determined the changes are acceptable.

3.2.4 Editorial Changes

Some editorial changes made in Revision 2 of DPC-NE-2011-P relate to the extension of the methodology to Harris and Robinson. The report is updated to state that it applies to Harris and Robinson in addition to Catawba and McGuire. Harris and Robinson are added to lists of Duke Energy plants with Westinghouse NSSS designs. Where references are made to plant TSs or FSARs, appropriate references to the Harris and Robinson TSs and FSARs are added. The new revision of the methodology also clarifies that implementation of DPC-NE-2011-P requires the use of an NRC-approved core simulator system, and replaces a statement referring to the list of computer codes in Appendix A with references to the CASMO-5/SIMULATE-3 and CASMO-4/SIMULATE-3 methodologies.

As with DPC-NF-2010, Revision 3, Revision 2 of DPC-NE-2011-P also contains a number of changes that make the methodology less plant-specific. Specific values of the percentage overlap between rod banks are removed, because the three-loop Westinghouse plants (Harris and Robinson) operate with a slightly different overlap than the four-loop plants (Catawba/McGuire). Additionally, the particular value of the maximum power level RPS trip is removed; while Catawba and McGuire use 118%, Harris and Robinson use 116%. The specific value is not as important as ensuring that the value assumed when developing the OTΔT and OPΔT trip functions is consistent with the actual RPS trip setpoint. Also, the specific values are retained in the TSs (or in the COLR, and based appropriately on the facility safety analyses).

Other, more general editorial changes were made as clarifications to the methodology. New definitions were provided and existing definitions were clarified. Several changes point to where discussions (definitions, development of uncertainties, etc.) are made elsewhere in the document. Clearer distinctions are made between the operational AFD limits and the CFM AFD–power level limits used to develop the OPΔT f(ΔI) penalty. Changes were made to clarify that the burnup dependency of F_Q^{RTP} is specified by the factor K(BU), which is presented in each plant's COLR. Further changes were made to clarify that operating AFD–power level limits, control rod insertion limits, and cycle-specific margin decrease penalty factors for F_Q and $F_{\Delta H}$ are specified in the COLR. Another clarification was made to clarify that the K_5 constant in the OPΔT trip function at Catawba and McGuire was moved from the TSs to the COLR; this change is editorial in nature because the move to the COLR was already approved by the NRC, as was verified by staff review of the current Catawba and McGuire TSs.

All of the editorial changes discussed in this section of the SE are either directly related to the extension of Harris and Robinson to the DPC-NE-2011-P methodology or were found by the NRC staff to be minor changes that do not substantively affect the technical content.

3.2.5 Conclusion Regarding DPC-NE-2011-P, Revision 2

The changes made in Revision 2 to DPC-NE-2011-P were discussed in detail above. Though the primary purpose of Revision 2 was to update the methodology to allow it to be applied to Harris and Robinson, several changes were made based on experience in applying the methodology to reduce unnecessary conservatisms. Several minor editorial changes were also made. The NRC staff reviewed the changes to the methodology report in the prior sections of this SE and found them to be acceptable within the context of their application to Harris and Robinson. Thus, the NRC staff determined that DPC-NE-2011-P, Revision 2, is acceptable for use at Harris and Robinson, as it provides an adequate basis for determining TS parameters that help to ensure compliance with the GDC discussed in Section 2 of this SE.

3.3 Evaluation of Technical Specifications Changes

As part of the LAR request for a site specific review of DPC-NF-2010, Revision 3, and DPC-NE-2011-P, Revision 2, Duke Energy requested incorporation of these reports into the Harris and Robinson TSs. As discussed in Sections 3.1 and 3.2, the NRC staff determined that both these methodology reports were acceptable for site-specific use at Harris and Robinson.

The proposed TS changes related to DPC-NF-2010, Revision 3, and DPC-NE-2011-P, Revision 2, are:

- Harris TS 6.9.1.6.2 is amended to include the following COLR references:

  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).

- Robinson TS 5.6.5.b is amended to include the following COLR references:

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

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

The proposed TS changes meet 10 CFR 50.36 and are consistent with Generic Letter 88-16 on the use of plant-specific methodologies in generating core operating limits, in that they appropriately identify the NRC staff's SE for each COLR reference. As such, and because DPC-NF-2010, Revision 3, and DPC-NE-2011-P, Revision 2, were determined to be acceptable on a plant-specific basis for Harris and Robinson, as discussed above, the NRC staff determined that the proposed TS changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official and the South Carolina State official were notified of the proposed issuance of the amendment on April 11, 2017. The State officials had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. 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, and there has been no public comment on such finding (81 FR 50746). Accordingly, the amendment meets 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 be prepared in connection with the issuance of the amendment.

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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCE LIST

1. J. Elnitsky, Duke Energy Progress, Inc. (Duke Energy), letter to Document Control Desk (DCD), U.S. Nuclear Regulatory Commission (USNRC), "Supplemental Information for License Amendment Request Regarding Methodology Report DPC-NE-1008-P," RA-16-0023, May 4, 2016, Agencywide Documents Access and Management System (ADAMS) Accession No. ML16125A420.

2. K. Henderson, Duke Energy, letter to DCD, USNRC, "Response to Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Reports DPC-NF-2010, Revision 3 and DPC-NE-2011-P, Revision 2," RA-16-0042, November 17, 2016, ADAMS Accession No. ML16323A102.

3. R. T. Repko, Duke Energy, letter to DCD, USNRC, "Application to Revise Technical Specifications for Methodology Report DPC-NE-1008-P Revision 0, 'Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors'," RA-15-0031, August 19, 2015, ADAMS Accession No. ML15236A044.

4. R. T. Repko, Duke Energy, letter to DCD, USNRC, "Application to Revise Technical Specifications to Adopt Methodology Reports DPC-NF-2010 Revision 3, 'Nuclear Physics Methodology for Reload Design,' and DPC-NE-2011-P Revision 2, 'Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors'," RA-16-0003, February 3, 2016, ADAMS Accession No. ML16034A610.

5. D. Galvin, USNRC, e-mail to R. T. Repko, Duke Energy, "Harris/Robinson Acceptance Review Draft Supplemental Information request – LAR to Adopt Methodology Reports DPC-NF-2010 Revision 3 and DPC-NE-2011-P Revision 2 (MF7337, MF73338)," March 16, 2016, ADAMS Accession No. ML16076A434.

6. J. Elnitsky, Duke Energy, letter to DCD, USNRC, "Withdrawal of License Amendment Request Regarding Methodology Reports DPC-NF-2010 and DPC-NE-2011-P," RA-16-0020, April 7, 2016, ADAMS Accession No. ML16098A317.

7. Duke Power Company, "Nuclear Physics Methodology for Reload Design," DPC-NF-2010-A, Revision 2, July 30, 2007, ADAMS Accession No. ML072150586.

8. Duke Power Company, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," DPC-NE-2011A, May 2, 1990, ADAMS Accession No. ML16250A237.

9. Duke Power Company, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," DPC-NE-2011-NP, Revision 1, August 2001. Part 3, ADAMS Accession No. ML012910680.

10. C. P. Patel, USNRC, letter to G. R. Peterson, Duke Energy, "Catawba Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB3343 and MB3344)," October 1, 2002, ADAMS Accession No. ML022740677.

11. R. E. Martin, USNRC, letter to H. B. Barron, Duke Energy, "McGuire Nuclear Station, Units 1and 2 Re: Issuance of Amendments (TAC Nos. MB3222 and MB3223)," October 1, 2002, ADAMS Accession No. ML022740527.

12. M. Barillas, USNRC, letter to R. T. Repko, Duke Energy, "Shearon Harris Nuclear Power Plant, Unit 1, and H. B. Robinson Steam Electric Plant, Unit No. 2 – Acceptance of Requested Licensing Action Re: Duke Energy Progress, Inc.'s Application to Revise Technical Specifications to Adopt Methodology Report DPC-NE-1008-P, Revision 0 (CAC Nos. MF6658 and MF6649)," December 1, 2015, ADAMS Accession No. ML15309A728.

13. M. Barillas, USNRC, letter to J. Elnitsky, Duke Energy, "Duke Energy Progress, LLC, for Shearon Harris Nuclear Power Plant, Unit 1, and H. B. Robinson Steam Electric Plant, Unit No. 2 – Request for Additional Information Regarding Application to Adopt DPC-NF-2010, Revision 3, 'Nuclear Physics Methodology for Reload Design,' and DPC-NE-2011-P, Revision 2, 'Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors' (CAC Nos. MF7693 and MF7694)," October 26, 2016, ADAMS Accession No. ML16288A078.

14. USNRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light-Water Reactor] Edition," NUREG-0800, Section 4.3, "Nuclear Design," Revision 3, March 2007, ADAMS Accession No. ML070740003.

15. USNRC, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," Generic Letter 88-16, October 4, 1988, ADAMS Accession No. ML031130447.

16. Duke Energy, "Nuclear Design Methodology Using CASMO-4/SIMULATE-3 MOX [mixed oxide]," DPC-NE-1005-P-A, Revision 1, August 2004, ADAMS Accession No. ML051010321.

17. Duke Power Company, "Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2004P-A, Revision 1, February 1997, ADAMS Accession No. ML17066A261.

18. M. Barillas, USNRC, letter to J. M. Frisco, Duke Energy, "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 Report DPC-NE-2005-P, Revision 5, 'Thermal Hydraulic Statistical Core Design Methodology' (CAC Nos. MF5872 and MF5873)," March 8, 2016, ADAMS Accession No. ML16049A630.

19. USNRC, "Standard Technical Specifications – Westinghouse Plants," NUREG-1431, Volume 1, "Specifications," Revision 4.0, April 2012, ADAMS Accession No. ML12100A222.

Principal Contributor: R. Anzalone

Date: May 18, 2017

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 AND H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 – ISSUANCE OF AMENDMENTS REVISING TECHNICAL SPECIFICATIONS FOR METHODOLOGY REPORTS DPC-NE-1008-P REVISION 0, "NUCLEAR DESIGN METHODOLOGY USING CASMO-5/SIMULATE-3 FOR WESTINGHOUSE REACTORS," DPC-NF-2010 REVISION 3, "NUCLEAR PHYSICS METHODOLOGY FOR RELOAD DESIGN," AND DPC-NE-2011-P REVISION 2, "NUCLEAR DESIGN METHODOLOGY REPORT FOR CORE OPERATING LIMITS OF WESTINGHOUSE REACTORS" (CAC NOS. MF6648/MF6649 AND MF7693/MF7694) DATED MAY 18, 2017

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