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

March 8, 2016

Mr. Joseph M. Frisco, Jr.
Vice President of Nuclear Engineering
Duke Energy Corporation
Mail Code EC07H
P.O. Box 1006
Charlotte, NC 28201-1006

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 REPORT DPC-NE-2005-P, REVISION 5, "THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY" (CAC NOS. MF5872 AND MF5873)

Dear Mr. Frisco:

By letter dated March 5, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15075A211), as supplemented by letters dated August 10, 2015, and December 17, 2015, and February 1, 2016 (ADAMS Accession Nos. ML15253A680, ML15356A315, and ML16032A004), Duke Energy Progress, Inc. (Duke Energy, the licensee), submitted a request, in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.90, for H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson) and Shearon Harris Nuclear Power Plant, Unit 1 (Harris) to amend their technical specifications (TSs) to adopt the methodology report DPC-NE-2005-P, Revision 5, "Thermal-Hydraulic Statistical Core Design Methodology," for application specific to Harris and Robinson. In its submittal, Duke Energy requested review and approval of DPC-NE-2005-P, Revision 5, which adds Appendix H and Appendix I to DPC-NE-2005-P, allowing the methodology be used at Harris and Robinson to perform the subject analysis in-house.

The Nuclear Regulatory Commission (NRC) has issued Amendment No. 148 to Renewed Facility Operating License No. NPF-63 for Harris and Amendment No. 244 to Renewed Facility Operating License No. DPR-23 for Robinson. The amendments change the TSs in response to your submittal dated March 5, 2015, as supplemented by letters dated August 10, 2015, December 17, 2015, and February 1, 2016.

The NRC staff has completed its review of the information provided by the licensee. The NRC staff's safety evaluation (SE) is enclosed. The NRC staff has determined that the enclosed SE (Enclosure 3) does not contain proprietary information or other sensitive information pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." However, the NRC will delay placing the enclosed SE in the public document room for a period of 10 working days from the date of this letter to provide Duke Energy the opportunity to comment on any sensitive aspects of the SE. If you believe that any information in Enclosure 3 contains sensitive information, please identify such information line-by-line and define the basis for withholding pursuant to the criteria of 10 CFR 2.390. After 10 working days, the enclosed SE will be made publicly available, unless we hear from you.

The 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,

/RA/

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. 148 to NPF-63
2. Amendment No. 244 to DPR-23
3. Safety Evaluation

cc w/enclosures:
Mr. Benjamin C. Waldrep
Site Vice President
Shearon Harris Nuclear Power Plant
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Additional Distribution w/enclosures: via Listserv **10 working days after issuance**

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

DUKE ENERGY PROGRESS, INC.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No.148
Renewed License No. NPF-63

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

   A. The application for amendment by Duke Energy Progress, Inc. (the licensee), dated March 5, 2015, as supplemented by letters dated August 10, 2015, December 17, 2015 and February 1, 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. 148, are hereby incorporated into this license. Duke Energy Progress, Inc. 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

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: March 8, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 148

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following page of the renewed facility operating license with the revised page. The revised page is identified by amendment number and contains a line in the margin indicating the area of change.

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Page 4	Page 4

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.

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6-24c	6-24c

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, Inc. 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. 148, are hereby incorporated into this license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, Inc. 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.

[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.

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)

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,

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DUKE ENERGY PROGRESS, INC.

DOCKET NO. 50-261

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

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 244
Renewed License No. DPR-23

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

    A. The application for amendment by Duke Energy Progress, Inc. (the licensee), dated March 5, 2015, as supplemented by letters dated August 10, 2015, December 17, 2015 and February 1, 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. 244 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

Benjamin 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: March 8, 2016

ATTACHMENT TO LICENSE AMENDMENT NO. 244

RENEWED FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Replace the following page of the renewed facility operating license with the revised page. The revised page is identified by amendment number and contains a line in the margin indicating the area of change.

Remove	Insert
Page 3	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 Page	Insert Page
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. <u>Maximum Power Level</u>

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

B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 244 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.

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)

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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 148 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-63, SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, DOCKET NO. 50-400, AND AMENDMENT NO. 244 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-23, H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2, DOCKET NO. 50-261

DUKE ENERGY PROGRESS, INC.

1.0 INTRODUCTION

By application dated March 5, 2015, (References 1, 2, and 3), as supplemented by letters dated August 10, 2015, December 17, 2015, and February 1, 2016 (References 4, 5, and 6) Duke Energy Progress, Inc. (Duke Energy or Duke), the licensee for the Shearon Harris Nuclear Power Plant, Unit 1 (Harris), and H. B. Robinson Steam Electric Plant, Unit 2 (Robinson), submitted a request for changes to the Harris and Robinson Technical Specifications (TSs) Core Operating Limits Report (COLR) reference lists, located in TS 6.9.1.6.2 and TS 5.6.5.b, respectively. The requested changes would add Duke Topical Report DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology," as a COLR reference. In the submittal, the licensee also requested U.S. Nuclear Regulatory Commission (NRC) review and approval of DPC-NE-2005-P, Revision 5 (References 2 and 3), which adds Appendix H, Robinson Plant Specific Data to the Westinghouse 15 x 15 high thermal performance (HTP) fuel design, and Appendix I, Harris Plant Specific Data to the Westinghouse 17 x 17 HTP fuel design, to DPC-NE-2005-P, approving the statistical core design methodology be used at Harris and Robinson to perform the subject analyses in-house.

The supplements dated August 10, 2015, December 17, 2015, and February 1, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's initial proposed no significant hazards consideration determination as published in the *Federal Register* on August 4, 2015 (80 FR 46342).

Duke's statistical core design (SCD) methodology is described in the main body of the DPC-NE-2005-P-A report, which was previously reviewed and approved by the NRC in 1995 (Reference 7[1]). This methodology describes a means of determining a departure from nucleate

[1] Though the reference is to Revision 3 of DPC-NE-2005, the NRC's safety evaluation (SE) for Revision 0 is available from pages 33 to 46 of the referenced report.

boiling ratio (DNBR) limit that statistically accounts for the effects of uncertainties in certain key parameters. Subsequent revisions have added appendices containing plant-specific data that apply the SCD methodology to individual Duke plants and particular fuel types. The proposed Revision 5 to DPC-NE-2005-P consists of Appendices H and I (References 2 and 3) that apply the SCD methodology to Robinson and Harris, respectively.

Each new appendix to DPC-NE-2005-P provides a variety of information, including data on the plant and fuel design; the thermal-hydraulic codes and models used in applying the SCD methodology; the critical heat flux (CHF) correlation to be applied; and the statepoints, parameters, and uncertainties used in the SCD analysis. Each appendix also includes a selection of the statistical design limit to be used in safety analyses. The proposed Appendices H and I state that the SCD analyses for Robinson and Harris will be performed with the VIPRE-01 code and the HTP CHF correlation, both of which have been reviewed and approved by the NRC. VIPRE-01 is described in Electric Power Research Institute (EPRI) document EPRI NP-2511-CC-A, "VIPRE-01: A Thermal Hydraulic Code for Reactor Cores" (Reference 8) and the HTP correlation is described in EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Revision 1 (Reference 9).

2.0 <u>REGULATORY EVALUATION</u>

The DPC-NE-2005-P-A, Statistical Core Design (SCD) methodology proposed for use at Harris and Robinson describes a means of calculating a statistical limit on the DNBR, below which fuel failure may occur.

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, General Design Criterion (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 any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

The licensee's application states Harris is licensed to final GDC 10 and adoption of DPC-NE-2005-P-A at Harris will not affect Harris complying with final GDC 10 requirements.

The Robinson plant was evaluated against the draft GDC published in the *Federal Register* on July 11, 1967. Criterion 6 of these draft GDC requires that:

> *The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.*

The licensee's application states adoption of DPC-NE-2005-P-A at Robinson will not affect Robinson complying with Criterion 6 of the draft 1967 Appendix A GDC, as stated in its current licensing basis.

The NRC staff used NUREG-0800, "Standard Review Plan" (SRP), Section 4.4, "Thermal and Hydraulic Design" in its review of the proposed TS change. SRP 4.4 provides criteria for ensuring that the requirements of draft GDC 6 and final GDC 10 are met. In particular, SRP Acceptance Criteria 1, 4, and 5 are applicable for this license amendment review.

Section 50.36, "Technical specifications," of 10 CFR Part 50 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 of this section." This regulation requires that the TSs include items in five 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. The licensee's proposed changes to Harris TS 6.9.1.6 and Robinson TS 5.6.5 are within the administrative controls category.

The TS changes requested by the licensee modify the references of the Core Operating Limits Report (COLR). 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," which indicates that 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 COLR, which is defined in the TSs and required to be submitted to the NRC each operating cycle or when revised. As recommended by GL 88-16, the Harris and Robinson TSs include lists of references for the NRC-approved calculation methodologies used to generate the cycle-specific operating limits, in TS 6.9.1.6.2 and TS 5.6.5.b, respectively. The proposed changes would add DPC-NE-2005-P-A, Revision 5, to these COLR reference lists.

3.0 TECHNICAL EVALUATION

3.1 Review of DPC-NE-2005-P, Revision 5

Previous revisions of the DPC-NE-2005-P-A methodology, the VIPRE-01 thermal-hydraulic code, and HTP departure from nucleate boiling (DNB) correlation have already been reviewed and approved by the NRC staff. The NRC staff's review will focus on the applicability of the DPC-NE-2005-P-A to Harris and Robinson.

3.1.1 Description of DPC-NE-2005 Methodology

The SCD methodology described in DPC-NE-2005-P-A allows for computation of a statistical DNBR limit – known in the Duke methodology as the statistical design limit (SDL) – that explicitly accounts for the uncertainties of the key plant and fuel parameters, including the CHF correlation. The thermal-hydraulic analyses are then performed at nominal plant conditions and compared to the SDL, resulting in a more precise quantification of the margin to DNB.

The NRC staff's SE on Revision 0 to DPC-NE-2005-P-A contained the following restrictions:

- The statistical core design (SCD) methodology developed by DPC (Duke Power Company), as described in the submittal, is direct and general enough to be widely applicable to any pressurized-water reactor (PWR) fuel or reactor, provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the critical heat flux (CHF) correlation subject to the conditions in the VIPRE safety evaluation report (SER).
- DPC must demonstrate that DPC's use of specific uncertainties and distributions based upon plant data and its selection of statepoints used for generating the statistical design limit are appropriate.
- This methodology is approved only for use in Duke Energy plants.

The last restriction, which requires that the methodology be used only at Duke plants, is satisfied for Harris and Robinson. The first two restrictions, on the use of VIPRE-01 and the uncertainties, distributions, and statepoints used in generating the Harris and Robinson design limits, is discussed below.

3.1.2 Thermal-Hydraulic Code and Subchannel Modeling

For both Robinson and Harris, Duke plans to use the VIPRE-01 thermal-hydraulic computer code, as described in Reference 8. The restriction discussed in the staff's SE for DPC-NE-2005-P-A states that VIPRE-01 must be approved with the use of the core model and correlations subject to the conditions of the VIPRE-01 SER. In particular, the conditions of importance are Condition 3, which requires discussion and justification of modeling choices for each licensee's application of VIPRE-01, and Condition 2, which allows the use of previously approved CHF correlations within VIPRE-01 provided that the analysis results in a correlation limit that is conservative or the same. Condition 3 will be addressed in this section, while Condition 2 will be addressed in Section 3.1.3.

The core models developed for Robinson and Harris are both based on the Oconee model described in DPC-NE-3000-PA, Revision 5a (Reference 4) with some differences as described in the submitted Appendices H and I that form DPC-NE-2005-P, Revision 5 (References 2 and 3). Though previous versions of this report were available to the NRC staff, the version referenced by the licensee in the methodology report was not. Revision 5a of DPC-NE-3000 was submitted in response to the NRC's request for additional information (RAI) in Reference 4. The NRC staff reviewed this document and determined that the Oconee core model was unchanged from previous approved versions of the document.

The Robinson and Harris core models are shown graphically in Figures H-1 and I-1 of References 2 and 3, and have the same basic fourteen-channel design as the approved Oconee model. However, the fuel assembly geometries, radial power distributions, and axial nodalization have been modified from the Oconee model to reflect the Robinson and Harris fuel designs. The geometries and axial nodalizations for the Robinson and Harris core models, presented in Figures H-1 and I-1 and Tables H-1 and I-1, accurately reflect the fuel designs in use at the two sites. The radial power distributions, presented in Figures H-2 and I-2, reproduce the limiting radial peaking factors from the two plants' COLRs. The NRC staff determined that the VIPRE-01 core models proposed for Robinson and Harris are acceptable because they accurately reflect the geometries and limiting radial power distributions of the fuel described in the each of the plants' COLRs.

Additionally, Duke modified the approved base Oconee VIPRE-01 model for use at Robinson and Harris by changing from the Zuber-Findlay bulk void model and LEVY subcooled void model to the corresponding EPRI models. Duke has previously made the same modeling choices for different fuel products at Catawba, McGuire, and Oconee, and has received NRC approval in each case. The NRC staff's SE for the use of VIPRE-01 states that the EPRI bulk and subcooled void models are acceptable for licensing applications. These modeling choices, therefore, comply with the NRC's restriction on the use of DPC-NE-2005-P-A that requires the correlations used within VIPRE-01 to be approved subject to the conditions of the VIPRE SER.

Because the NRC staff determined that the fourteen-channel VIPRE-01 core models created by Duke are appropriate for the fuel designs at Harris and Robinson, and that the correlations for bulk and subcooled void have been approved for licensing applications, the NRC staff concluded that the use of the VIPRE-01 thermal-hydraulic code and the associated subchannel modeling are acceptable for Harris and Robinson for the purpose of performing SCD analyses.

3.1.3 Critical Heat Flux Correlation

Duke has proposed AREVA's HTP CHF correlation for use in the SCD methodology at Robinson and Harris. The NRC staff's approval of HTP contained a number of conditions and limitations. First, the correlation is only considered to be applicable to fuels in the correlation database presented in the HTP topical report. Second, use of the HTP correlation is restricted to a certain range of operating conditions listed in the SE. Finally, the correlation limit provided in the HTP topical report is to be used in licensing applications.

The acceptable range of fuel design parameters from the HTP correlation database is presented below in Table 1. The NRC staff examined the Robinson and Harris Final Safety Analysis Reports (FSARs) to determine the corresponding parameters for the Advanced W 15x15 HTP and Advanced W 17x17 HTP fuel assemblies in use at the two plants to ensure they were appropriately covered by the database. These fuel design parameters are also presented in Table 1, which demonstrates that both plants' fuel types are within the HTP database.

Table 1: Fuel design parameters.

Parameter	**HTP Database Range**		**Robinson Fuel**	**Harris Fuel**
	Min	**Max**		
Fuel Rod Diameter, in.	0.360	0.440	0.424	0.376
Fuel Rod Pitch, in.	0.496	0.580	0.563	0.496
Axial Spacer Span, in.	10.5	26.2	26.2	20.55
Hydraulic Diameter, in.	0.4571	0.5334	0.5278	0.4571
Heated Length, ft.	8.0	14.0	12.0	12.0

The range of allowable operating conditions for the HTP correlation is presented below in Table 2. The range presented below is based on Revision 1 of EMF-92-153(P)(A) (Reference 9), which allowed the HTP correlation to be used at lower pressures and mass fluxes, and higher local qualities. The revision also removed the minimum quality restriction. The NRC staff analyzed the proposed statepoints for Robinson and Harris, presented in Tables H-4 and I-4, respectively, and determined that the mass fluxes, inlet enthalpies, and pressures fall within the approved HTP range. Duke also stated (in References 2 and 3) that Tables H-3 and I-3 represent the "correlation allowable parameter range," and the values presented in those tables are within the approved range of the HTP correlation presented below.

Table 2: HTP approved range.

Variable	HTP Approved Range	
	Minimum	**Maximum**
Pressure (psia)	1385	2425
Local Mass Flux (Mlb/hr/ft^2)	0.498	3.573
Inlet Enthalpy (Btu/lb)	382.3	649.9
Local Quality	None	0.515

Finally, the correlation limit as stated in the HTP topical report is 1.141. Appendices H and I both present the correlation limit as 1.141 in Tables H-3 and I-3, and state that the higher of the two DNBR limits in Tables H-2 and I-2 (which in this case is the XCOBRA-IIIC based limit of 1.141) will be used for VIPRE-01 analyses.

The NRC staff's restriction on the use of DPC-NE-2005-P-A also requires that the CHF correlation be approved for use with VIPRE-01. The HTP correlation was originally implemented and approved for use with the XCOBRA-IIIC thermal-hydraulic code, and has not been previously approved for use with VIPRE-01. However, Condition 2 of the NRC staff's SE for VIPRE-01 makes provision for the implementation of new CHF correlations in VIPRE, stating:

> The use of a steady state CHF correlation which has been previously approved for application in connection with another thermal hydraulic code other than VIPRE-01 will require an analysis showing that, given the correlation data base, VIPRE-01 gives the same or a conservative safety limit, or a new higher DNBR limit must be used, based on the analysis results.

Duke performed analyses of the HTP test data from Reference 9 using VIPRE-01 to demonstrate the correlation's performance. Plots of predicted over-measured heat fluxes were provided in Figures H/I-4 through H/I-6 in References 2 and 3, and these plots generally show that the correlation as implemented in VIPRE-01 provides acceptable performance. To perform additional statistical analyses, the NRC staff requested that Duke provide the raw data used to generate these plots; and it was provided by Duke in Reference 4. The NRC staff determined that any trends with respect to the correlating parameters were either minimal or conservative, and that the correlation limit of 1.141 will apply appropriately throughout the application domain.

Both Duke's and the NRC's analyses of the data showed that the mean measured over predicted CHF was lower than the original implementation of HTP within XCOBRA-IIIC. However, the data had the same standard deviation, which is a key parameter in the statistical treatment of uncertainties within the SCD methodology, as it is a factor in the statistical treatment of uncertainties. Since the correlation limit of 1.141 will be used for VIPRE-01 analyses, as discussed previously, this meets the VIPRE-01 condition requiring the safety limit to be conservative or the same. Since the standard deviation is the same and the mean is more conservative than in the XCOBRA-IIIC implementation of HTP, the staff determined that the use of the HTP correlation in VIPRE-01, as described in References 2 and 3, is acceptable.

3.1.4 Statepoints Used in the Statistical Core Design Analyses

As discussed in Section 3.1.1, one of the conditions on the use of the DPC-NE-2005-P-A methodology is that selection of statepoints used for generating the design limit will be justified to be appropriate when it is applied to new plants. These statepoints are identified in Tables H-4 and I-4 of Reference 1, and, according to Duke, represent the conditions to which the statistical DNB analysis limit will be applied. The range of the key parameters analyzed in the SCD analysis is summarized in Tables H-7 and I-7. It is worth noting that Table 7 of Reference 8 states that new statepoints may be added. If the inclusion of a new statepoint in the analysis changes the SDL such that it is greater than the limit, Duke will have to take appropriate compensatory measures, such as increasing the design limit for the statepoint. The NRC staff requested Duke explain in further detail the selection of the statepoints. In its response (Reference 4), Duke stated that the statepoints were selected to conservatively bound the range of conditions of the transients for which the SCD methodology is intended to be applied, and that the statepoint conditions were obtained from the applicable Updated FSAR (UFSAR) Chapter 15 systems analyses performed by the fuel vendor. The NRC staff determined that this is an acceptable means of selecting statepoints for analysis, because the envelope of plant safety analyses performed under the Duke SCD methodology will be similar to that of the existing UFSAR analyses. Therefore, the NRC staff determined that the selected statepoints are adequately justified, in accordance with the NRC's restriction on the use of DPC-NE-2005-P-A discussed in Section 3.1.1 of this SE.

3.1.5 Key Parameter Uncertainties Used in the Statistical Core Design Analyses

The final condition on the use of DPC-NE-2005-P-A requires the use of specific uncertainties and distributions to be justified on a plant-specific basis. Tables H-5 and I-5 in References 2 and 3 contain lists of the parameters considered to be uncertain, the value of the uncertainty used, the distribution associated with the uncertainty of each parameter, and a summary justification of each statistically treated parameter and associated uncertainty. The statistically treated parameters are: core power, coolant flow measurement, bypass flow, core exit pressure, core inlet temperature, radial power measurements, radial power engineering uncertainties, axial power peak prediction uncertainty (from the physics code), axial peak location uncertainty, DNBR correlation uncertainty, and thermal-hydraulic code uncertainties. For reference, these parameters and their uncertainties are reprinted below in Table 3 and Table 4; proprietary values of uncertainty for the DNBR correlation and code/model uncertainties may be found in Tables H-5 and I-5 of References 2 and 3.

Table 3: Robinson statistically treated uncertainties.

Parameter	Uncertainty	Standard Deviation	Distribution
Reactor Power	± 0.3%	0.18%	Normal
Coolant Flow Measurement	± 2.7%	1.64%	Normal
Coolant Bypass Flow	± 1.5%	-	Uniform
Pressure	± 40 psia	-	Uniform
Temperature	± 4°F	-	Uniform
Radial Peaking Measurement	± 4.0%	2.43%	Normal
$F^E_{\Delta H}$	± 3.0%	1.82%	Normal
F_Z	± 4.5%	2.75%	Normal
Z	± 3 inches	-	Uniform
DNBR Correlation	Proprietary	Proprietary	Normal
Code/Model Uncertainties	Proprietary	Proprietary	Normal

Table 4: Harris statistically treated uncertainties.

Parameter	Uncertainty	Standard Deviation	Distribution
Reactor Power	± 0.34%	0.21%	Normal
Coolant Flow Measurement	± 2.2%	1.34%	Normal
Coolant Bypass Flow	± 1.5%	-	Uniform
Pressure	± 35 psia	-	Uniform
Temperature	± 3.5°F	-	Uniform
Radial Peaking Measurement	± 4.0%	2.43%	Normal
$F^E_{\Delta H}$	± 3.0%	1.82%	Normal
F_Z	± 4.5%	2.75%	Normal
Z	± 3 inches	-	Uniform
DNBR Correlation	Proprietary	Proprietary	Normal
Code/Model Uncertainties	Proprietary	Proprietary	Normal

The NRC staff identified a need for additional information in the justifications for a number of these uncertainties. Duke's response, in Reference4, provided additional detail on how the coolant flow measurement uncertainty, coolant flow bypass uncertainty, core exit pressure uncertainty, axial peak uncertainty, axial peak location uncertainty, and code/model uncertainty were determined. In the response to the NRC staff's RAI, Duke also stated that the uncertainties will be verified to be bounding each cycle.

Uncertainties associated with measurements, such as core power, coolant flow, core exit pressure, and core inlet temperature, are determined by combining various component uncertainties. The radial power engineering uncertainties are similarly determined by statistically combining manufacturing tolerances. This approach is standard engineering practice, and the uncertainties for these parameters proposed in the table are consistent with the plant licensing bases as documented in the Harris and Robinson FSARs and TSs.

The DNBR correlation uncertainty was discussed in detail in the licensee's submittal and in Section 3.1.3 of this SE. As previously mentioned, the NRC staff analyzed the data provided by Duke and determined that the correlation standard deviation remained consistent between the original implementation in XCOBRA-IIIC and the new implementation in VIPRE-01. The correlation uncertainty for SCD analyses, which is based on this standard deviation, is therefore considered by the NRC staff to be acceptable because the thermal-hydraulic code uncertainty is the same as has been used in other implementations of the DPC-NE-2005-P-A methodology and is considered by the staff to be conservative.

The bypass flow uncertainty of 1.5 percent was selected to bound the difference between the minimum and maximum bypass flow calculated by the fuel vendor. The NRC staff found this to be acceptable because the uncertainty will be verified to be bounding each cycle.

As discussed in References 2, 3, and 4, the radial power measurement uncertainty, axial power peak prediction uncertainty, and axial peak location uncertainty are all selected to bound the uncertainties in the neutronics codes. The codes intended to be applied at Harris and Robinson and their associated uncertainties are documented in DPC-NE-1008-P "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors" (Reference 10), which is currently under review by the NRC staff. Because the new neutronics codes are currently under NRC review, it is possible that Harris or Robinson may go into an upcoming outage with DPC-NE-2005-P-A and not DPC-NE-1008-P.

The NRC staff considered the applicability of the proposed uncertainties to the Harris and Robinson current licensing basis neutronics codes. From a review of the plant FSARs, the current neutronics codes at Harris and Robinson have uncertainties on $F_{\Delta H}$ and z[2] that are bounded by those proposed by Duke. However, the NRC staff was concerned that these current codes do not have uncertainties in terms of F_Z, making it therefore unclear whether or not the proposed uncertainty would be bounding. The NRC staff asked, in an RAI, how Duke would verify whether or not the proposed uncertainties bounded the current neutronics codes. Duke stated in response (Reference 5) that the total peaking factor (F_Q) uncertainty from the current neutronics code would be conservatively used in place of the axial uncertainty (F_Z) in the DPC-NE-2005-P-A methodology. If the value of the F_Q uncertainty exceeded that of the F_Z uncertainty assumed in the SCD analyses, a penalty would be applied to the SDL, as provided for in Table 7 of Reference 7. The NRC staff determined that this approach is acceptable under the approved SCD methodology and assures that the peaking factor uncertainties will be applied conservatively regardless of the nuclear design methodology. This will ensure compliance with GDC 10 for Harris and draft GDC 6 for Robinson. Therefore the NRC staff determined that the radial power measurement uncertainty, axial power peak prediction uncertainty, and axial power peak location uncertainty are acceptable.

In consideration of the above, the NRC staff determined that the selections of key parameters and their associated uncertainty values and distributions are acceptable, fulfilling the restrictions stated in the DPC-NE-2005-P-A SE as provided in Section 3.1.1 of this SE.

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[2] *z* is the axial peak location; uncertainty in *z* accounts for differences in noding between the neutronic and thermal-hydraulic codes

3.1.6 Conclusion Regarding DPC-NE-2005, Revision 5

As discussed in the previous sections, the NRC staff determined that Revision 5 to DPC-NE-2005-P appropriately satisfies the conditions and limitations on the use of Duke's statistical core design methodology as laid out in the NRC's SE of DPC-NE-2005-P-A, Revision 0. As such, the NRC staff considers application of the SCD methodology to be appropriate for use at Harris and Robinson and concludes that DPC-NE-2005-P, Revision 5, is acceptable.

3.2 Technical Specification Changes

In Reference 1, Duke proposed to amend Robinson TS 5.6.5.b and Harris TS 6.9.1.6.2 to add DPC-NE-2005-P-A as a COLR reference. As discussed in the licensee's submittal, the Robinson and Harris TSs have both incorporated Technical Specification Task Force Traveler No. TSTF-363, which allows licensees to omit topical report revision numbers and dates from the TS list of COLR references. As such, the licensee has proposed to not include the revision number and date associated with the current methodology review.

Since DPC-NE-2005-P-A is a generically approved Duke topical report, it is consistent with current NRC policy to allow the licensee to amend the TS list of COLR references to include the report without revision numbers or dates. Major changes to the methodology such as introductions of new fuel designs or changes of CHF correlations would require revisions[3] to the topical report as discussed in Table 7 of DPC-NE-2005-P-A. Since these revisions would require prior NRC approval before use at a site, the NRC staff concludes that the licensee's proposal provides an acceptable level of control over the methodology. The NRC staff therefore determined that the TS changes proposed by the licensee are acceptable.

The NRC staff reviewed the information provided by Duke and determined that DPC-NE-2005-P, Revision 5, is appropriate for both Robinson and Harris and acceptable for licensing applications. As discussed in previous sections, any changes to the plant specific models, parameters, and uncertainties provided in DPC-NE-2005-P, Revision 5, must be evaluated for their impact on the SDL and may require NRC approval per Table 7 of DPC-NE-2005-PA (Reference 7). The staff reviewed the licensee's proposed changes to Robinson TS 5.6.5.b and Harris TS 6.9.1.6.2 and determined that they were acceptable, based on the applicability of the DPC-NE-2005-P, Revision 5 methodology.

4.0 STATE CONSULTATION

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

[3] This process is what Duke followed in developing DPC-NE-2005-P, Revision 5, which was reviewed in Section 3 of this SE.

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 (80 FR 46342). 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 REFERENCES

1. J. M. Frisco, Jr., Duke Energy Corporation (Duke), letter to Document Control Desk (DCD), U. S. Nuclear Regulatory Commission (NRC), "Application to revise technical specifications for methodology report DPC-NE-2005-P, Revision 5, 'Thermal-Hydraulic Statistical Core Design Methodology'," RA-15-0004, March 5, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15075A211).

2. Duke, "Thermal-Hydraulic Statistical Core Design Methodology, Appendix H: Robinson Plant Specific Data," DPC-NE-2005-P, Revision 5, Appendix H, November 2014 (ADAMS Accession No. ML15075A212).

3. Duke, "Thermal-Hydraulic Statistical Core Design Methodology, Appendix I: Harris Plant Specific Data," DPC-NE-2005-P, Revision 5, Appendix I, November 2014 (ADAMS Accession No. ML15075A213).

4. R. T. Repko, Duke, letter to DCD, NRC, "Response to NRC Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-2005-P, Revision 5," RA-15-0037, September 9, 2015 (ADAMS Accession No. ML15253A680).

5. E. J. Kapopoulos, Jr., Duke, letter to DCD, NRC, "Response to Second NRC Request for Additional Information (RAI) Regarding Application to Revise Technical Specifications for Methodology Report DPC-NE-2005-P, Revision 5," RA-15-0058, December 21, 2015 (ADAMS Accession No. ML15356A315).

6. R. T. Repko, Duke, letter to DCD, NRC, "Supplement to Application to revise Technical Specifications for Methodology Report DPC-NE-2005-P, Revision 5, February 1, 2016 (ADAMS Accession No. ML16032A004).

7. Duke, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," DPC-NE-2005-PA, Revision 3, Charlotte, NC, September 2002 (ADAMS Accession Nos. ML023090299 and ML023090252).

8. Electric Power Research Institute (EPRI), "VIPRE-01: A Thermal Hydraulic Code for Reactor Cores," NP-2511-CCM-A, Revision 4, Palo Alto, CA, June 2007 (ADAMS Accession Nos. ML102090545, ML102090544, ML102090543, and ML102070202; Non-Publicly Available).

9. Framatome ANP, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," EMF-92-153(P)(A), Revision 1, January 2005 (ADAMS Accession No. ML051020017).

10. R. T. Repko, Duke, letter to DCD, 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 (ADAMS Accession No. ML15236A044).

Principal Contributor: Reed Anzalone

Date: March 8, 2016

pursuant to the criteria of 10 CFR 2.390. After 10 working days, the enclosed SE will be made publicly available, unless we hear from you.

The 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,

/RA/

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. 148 to NPF-63
2. Amendment No. 244 to DPR-23
3. Safety Evaluation

cc w/enclosures:
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