DUKE ENERGY.

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~~PROPRIETARY INFORMATION - WITHHOLD UNDER 10 CFR 2.390~~
~~UPON REMOVAL OF ATTACHMENTS 6 AND 7 THIS LETTER IS UNCONTROLLED~~

Serial: RA-15-0004
March 5, 2015

10 CFR 50.90

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

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

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

SUBJECT: APPLICATION TO REVISE TECHNICAL SPECIFICATIONS FOR METHODOLOGY REPORT DPC-NE-2005-P, REVISION 5, "THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY"

REFERENCES:

1. NRC letter, *Catawba Nuclear Station, Units 1 and 2 and McGuire Nuclear Station Units 1 and 2 RE: Acceptance for Referencing of the Modified Licensing Topical Report DPC-NE-2009P, Revision 2 (TAC Nos. MB4502, MB4503, MB4504, and MB4505)*, dated December 18, 2002 (ADAMS Accession No. ML023520616)
2. NRC letter, *Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Use of AREVA NP Mark-B-HTP Fuel (TAC Nos. MD7050, MD7051, and MD7052)*, dated October 29, 2008 (ADAMS Accession No. ML082800408)

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, Inc., referred to henceforth as "Duke Energy", is submitting a request for an amendment to the Technical Specifications (TS) for Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP). Specifically, Duke Energy requests NRC review and approval of DPC-NE-2005-P, "Thermal-Hydraulic Statistical Core Design Methodology," Revision 5, and adoption of the methodology into the TS for SHNPP and HBRSEP. This change will allow Duke Energy to perform the subject analysis, as opposed to utilizing contract services.

A 001
NRR

The current Duke Energy Statistical Core Design (SCD) methodology report DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology," Revision 4a, contains NRC-approved appendices which currently apply the methodology to McGuire, Catawba, and Oconee nuclear stations. The most recent Appendices F and G were approved by the NRC in References 1 and 2. This proposed change extends applicability of the DPC-NE-2005-P-A methodology to SHNPP and HBRSEP. As a result, Appendices H and I are added to DPC-NE-2005-P-A for HBRSEP and SHNPP, respectively. Duke Energy and NRC staff participated in a pre-submittal meeting on November 12, 2014, regarding these changes.

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that the proposed changes involve no significant hazards consideration. The bases for these determinations are included in Attachment 3. Attachment 3 provides an evaluation of the proposed change. Attachment 4 provides the existing TS pages marked up to show the proposed change. Attachment 5 provides the retyped TS pages.

Attachments 6 and 7 contain the new Appendices H and I, which include information that is proprietary to Duke Energy and AREVA NP. In accordance with 10 CFR 2.390, Duke Energy, on behalf of itself and AREVA NP, requests that Attachments 6 and 7 be withheld from public disclosure. Affidavits are included from each organization (Attachments 1 and 2) attesting to the proprietary nature of the information. Non-proprietary versions of the attachments are included in Attachments 8 and 9.

Approval of the proposed amendment is requested by December 31, 2016 in order to support the core design of SHNPP Cycle 22, which is expected to commence operation Spring 2018. The requested approval date allows sufficient time to establish the appropriate contract services to perform the analysis, if the amendment is not approved. An implementation period of 120 days is requested to allow for updating the TS and Facility Operating License.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on
3/5/15.

Sincerely,

Joseph Frisco, Jr.
Vice President – Nuclear Engineering

JBD/MKL

Attachments:
1. Affidavit of Joseph Frisco
2. Affidavit of Gayle Elliott
3. Evaluation of the Proposed Change
4. Proposed Technical Specification Changes (Mark-Up)
5. Retyped Technical Specification Pages
6. DPC-NE-2005-P Appendix H Robinson Plant Specific Data (Proprietary)
7. DPC-NE-2005-P Appendix I Harris Plant Specific Data (Proprietary)
8. DPC-NE-2005 Appendix H Robinson Plant Specific Data (Redacted)
9. DPC-NE-2005 Appendix I Harris Plant Specific Data (Redacted)

cc: USNRC Region II
USNRC Senior Resident Inspector – SHNPP
USNRC Senior Resident Inspector – HBRSEP
M. C. Barillas, NRR Project Manager – SHNPP & HBRSEP
W. L. Cox, III, Section Chief, NC DHSR (Without Attachments 6 and 7)
S. E. Jenkins, Manager, Radioactive and Infectious Waste Management Section (SC) (Without Attachments 6 and 7)
Attorney General (SC) (Without Attachments 6 and 7)
A. Gantt, Chief, Bureau of Radiological Health (SC) (without Attachments 6 and 7)

Attachment 1

Affidavit of Joseph Frisco

AFFIDAVIT of Joseph Michael Frisco, Jr.

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

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

3. I have knowledge of the criteria used by Duke Energy in designating information as proprietary or confidential.

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

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

   (ii) The information is of a type that would customarily be held in confidence by Duke Energy. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that provides a competitive advantage to Duke Energy.

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

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

   (v) The proprietary information sought to be withheld in the submittal is that which is marked in the proprietary versions of Appendix H (dated November 2014) and Appendix I (dated November 2014) of Duke methodology report DPC-NE-2005, *Thermal-Hydraulic Statistical Core Design Methodology*. This information enables Duke Energy to:

      (a) Support license amendment requests for its Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP).

      (b) Perform transient and accident analysis calculations for SHNPP and HBRSEP.

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

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

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

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

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

J M Frisco Jr
Joseph Michael Frisco, Jr.

Subscribed and sworn to me: March 5, 2015
Date

D M Reese Debra Reese
Notary Public

My commission expires: September 6, 2015

SEAL DEBRA REESE NOTARY PUBLIC MECKLENBURG COUNTY, NC

Attachment 2

Affidavits of Gayle Elliott
(1 each for SHNPP and HBRSEP)

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle Elliott. I am Manager, Product Licensing, for AREVA Inc. (AREVA) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in "DPC-NE-2005-P, Duke Energy Thermal Hydraulic Statistical Core Design Methodology, Appendix H, Robinson Plant Specific Data, Advanced W 15x15 HTP Fuel, Application of HTP CHF Correlation to the Advanced W 15x15 HTP Fuel Design," dated November 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in

accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(c), 6(d) and 6(e) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 17th

day of December, 2014.

309873

ELLA F. CARR-PAYNE
NOTARY PUBLIC
REG # 309873
MY COMMISSION EXPIRES 8/31/2017
COMMONWEALTH OF VIRGINIA

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

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2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in "DPC-NE-2005-P, Duke Energy Thermal Hydraulic Statistical Core Design Methodology, Appendix I, Harris Plant Specific Data, Advanced W 17x17 HTP Fuel, Application of HTP CHF Correlation to the Advanced W 17x17 HTP Fuel Design," dated November 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in

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SUBSCRIBED before me this 17th

day of December, 2014.

309873

ELLA F. CARR-PAYNE
NOTARY PUBLIC
REG # 309873
MY COMMISSION EXPIRES 8/31/2017
COMMONWEALTH OF VIRGINIA

Attachment 3

EVALUATION OF THE PROPOSED CHANGE

Subject: APPLICATION TO REVISE TECHNICAL SPECIFICATIONS FOR METHODOLOGY REPORT DPC-NE-2005-P, REVISION 5, "THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY"

1.0 SUMMARY DESCRIPTION

Duke Energy requests amendments to Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP) Technical Specifications (TSs) pursuant to 10 CFR 50.90, to support the allowance of Duke Energy to perform the analyses of record for its reload cores. The proposed change requests review and approval of DPC-NE-2005-P, Revision 5, "Thermal-Hydraulic Statistical Core Design Methodology," and subsequent inclusion of DPC-NE-2005-P-A into the SHNPP and HBRSEP Technical Specifications (the "-A" is added to indicate an approved report, in accordance with the NRC process for topical reports).

2.0 DETAILED DESCRIPTION

DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology," describes Duke Energy's methodology for determining the statistical Departure from Nucleate Boiling (DNB) Ratio (DNBR) limit for DNB analyses at Duke Energy plants. Revision 4a is the current revision of DPC-NE-2005-P-A, which includes appendices applying the method to McGuire, Catawba, and Oconee nuclear stations. Those appendices were approved by the NRC in References 1 and 2. This proposed change extends applicability of the DPC-NE-2005-P-A methodology to HBRSEP and SHNPP. As a result, Appendix H and I are added to DPC-NE-2005-P-A for HBRSEP and SHNPP, respectively. The existing approved Revision 4a, with the addition of Appendices H and I constitute DPC-NE-2005-P, Revision 5, for which NRC approval is requested in this submittal. In addition, the DPC-NE-2005-P-A report is added to HBRSEP TS Section 5.6.5.b and SHNPP TS Section 6.9.1.6.2, as shown in Attachments 4 and 5. Because the current HBRSEP and SHNPP TSs are consistent with TSTF-363, "Revise Topical Report References in ITS 5.6.5, COLR [Core Operating Limits Report]" (References 3 and 4), inclusion of revision dates for the topical report in the TS is not required, which is also consistent with NUREG 1431, Revision 4.

The statistical thermal-hydraulic design methodology accounts for the effects on DNB of the uncertainties of key parameters. Statistically combining these effects yields a better quantification of the DNB margin which, in turn, enhances core reload design flexibility. The main body of DPC-NE-2005-P-A describes the Duke Energy approach for calculating the DNBR limit for a plant with a specific fuel design and associated Critical Heat Flux (CHF) correlation. The method includes determination of plant specific parameter uncertainties and statepoint conditions and also describes a process for applying the approved methodology to different plants, new fuel designs, and/or new or revised CHF correlations.

Duke Energy has provided five appendices subsequent to Revision 0 of DPC-NE-2005-P-A (Revision 0 included Appendices A and B) for either fuel transitions and/or CHF correlation changes. The change in this submittal adds two new appendices to DPC-NE-2005-P-A to support the application of the Duke Energy methodology to the HBRSEP (Appendix H) and SHNPP (Appendix I). The required information in each appendix as per the methodology is provided, including:

a. Identification of the plant, fuel type, and CHF correlation with appropriate references to the approved fuel design and CHF correlation topical reports.
b. Statement of the thermal-hydraulic code and model used with appropriate references to the approved methods reports.
c. A list of the key parameters, their uncertainty values, and distributions.

d. A list of the statepoints analyzed.
e. The Statistical Design Limit (SDL).

The SDL is used in DNB analyses for the plants applying Duke Energy methodology for cycle reload safety analyses. There are additional methodology reports and analyses in development related to the application of the approved SDL; therefore, specific impact on general DNB descriptions and Updated Final Safety Analysis Report (UFSAR) accident analyses are not available. The appropriate SHNPP FSAR and HBRSEP UFSAR changes will be processed once core designs using the methodology addressed by this LAR (and the methodologies which will be the subject of subsequent LARs) are implemented.

3.0 TECHNICAL EVALUATION

The Duke Energy statistical core design (SCD) methodology as documented in methodology report DPC-NE-2005-P-A was granted approval by the NRC in Reference 5. This approval acknowledged that the statistical core design methodology is direct and general enough to be widely applicable to any pressurized-water reactor (PWR), with the following restrictions that are applicable to this amendment:

(1) The VIPRE-01 methodology is approved with the use of the core model and correlations including the CHF correlation subject to the VIPRE SER conditions. Furthermore, Duke Energy must demonstrate that Duke Energy'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.

(2) This methodology is approved only for use in Duke Energy plants.

The technical justification supporting this amendment request (including the required information above regarding the VIPRE-01 model description, the applied CHF correlation, the assumed uncertainty values and distributions, and the statepoints currently analyzed) is included in the attached methodology report appendices (Attachment 6 - Appendix H for HBRSEP and Attachment 7 - Appendix I for SHNPP) per the methodology report.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix A, General Design Criterion (GDC) 10, "Reactor Design," requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. SHNPP is licensed to GDC 10 and this proposed change will not affect the SHNPP conformance to GDC 10.

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

> "The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall

provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power."

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

4.2 Precedent

The use of the methodology in DPC-NE-2005-P-A was approved for use at McGuire, Catawba, and Oconee nuclear stations in References 1 and 2.

4.3 No Significant Hazards Consideration Determination

Duke Energy Progress, Inc., referred to henceforth as "Duke Energy", requests NRC review and approval of a reactor core design methodology report DPC-NE-2005-P, "Thermal-Hydraulic Statistical Core Design Methodology," Revision 5, and adoption of the methodology into the Technical Specifications (TS) for Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP) and H. B. Robinson Steam Electric Plant, Unit No. 2 (HBRSEP).

Duke Energy has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change extends use of DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology" to Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP). The NRC has previously reviewed and approved use of this methodology, stating it is direct and general enough to be widely applicable to any Pressurized Water Reactor (PWR) core. The methodology will be applied to SHNPP and HBRSEP after approval by the NRC. The proposed methodology does not affect the performance of any equipment used to mitigate the consequences of an analyzed accident. There is no impact on the source term or pathways assumed in accidents previously assumed. No analysis assumptions are violated and there are no adverse effects on the factors that contribute to offsite or onsite dose as the result of an accident. No accident probabilities or consequences will be impacted by this LAR.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

The proposed change extends use of DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology" to Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP). It does not change any system functions or maintenance activities. The change does not involve physical alteration of the plant, that is, no new or different type of equipment will be installed. The change does not alter assumptions made in the safety analyses but ensures that the core will operate within safe limits. This change does not create new failure modes or mechanisms which are not identifiable during testing, and no new accident precursors are generated.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change extends use of DPC-NE-2005-P-A, "Thermal-Hydraulic Statistical Core Design Methodology" to Shearon Harris Nuclear Power Plant (SHNPP) and H. B. Robinson Steam Electric Plant (HBRSEP). The NRC has previously reviewed and approved use of this methodology, stating it is direct and general enough to be widely applicable to any PWR core. The methodology will be applied to SHNPP and HBRSEP after approval by the NRC. Consistent with the existing methodology, the use of the proposed methodology will continue to ensure that all applicable design and safety limits are satisfied such that the fission product barriers will continue to perform their design functions.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6.0 REFERENCES

1. NRC letter, *Catawba Nuclear Station, Units 1 and 2 and McGuire Nuclear Station Units 1 and 2 RE: Acceptance for Referencing of the Modified Licensing Topical Report DPC-NE-2009P, Revision 2 (TAC Nos. MB4502, MB4503, MB4504, and MB4505)*, dated December 18, 2002 (ADAMS Accession No. ML023520616)
2. NRC letter, *Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Use of AREVA NP Mark-B-HTP Fuel (TAC Nos. MD7050, MD7051, and MD7052)*, dated October 29, 2008 (ADAMS Accession No. ML082800408)
3. Carolina Power & Light Company letter, *Request for Technical Specification Change Revision to Core Operating Limits Report (COLR) References*, dated June 14, 2000 (ADAMS Accession No. ML003725331)
4. Carolina Power & Light Company letter, *Revised Technical Specification Pages for License Amendment Request – Addition of Methodology References to Core Operating Limits Report*, dated January 11, 2000 (ADAMS Accession No. ML003676878)
5. NRC letter, *Acceptance for Referencing of the Modified Licensing Topical Report, DPC-NE-2005P, "Thermal-Hydraulic Statistical Core Design Methodology" (TAC No. M85181)*, dated February 24, 1995

Attachment 4

Proposed Technical Specification Changes (Mark-up)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

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

5.6.3 Radioactive Effluent Release Report

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

5.6.4 DELETED

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

1. Shutdown Margin (SDM) for Specification 3.1.1;

2. Moderator Temperature Coefficient limits for Specification 3.1.3;

3. Shutdown Bank Insertion Limits for Specification 3.1.5;

4. Control Bank Insertion Limits for Specification 3.1.6;

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

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

(continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

8. Boron Concentration limit for Specification 3.9.1.

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

1. Deleted

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

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

4. Deleted

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

6. Deleted.

7. Deleted

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

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

10. Deleted

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

12. Deleted

13. Deleted.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

14. Deleted
15. Deleted
16. ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.
17. ANF-88-133 (P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," approved version as specified in the COLR.
18. ANF-89-151(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.
19. EMF-92-081(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.
20. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.
21. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.
22. EMF-96-029(P)(A), "Reactor Analysis System for PWRs," approved version as specified in the COLR.
23. EMF-92-116, "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.
24. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," approved version as specified in the COLR.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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

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

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

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,

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

6.9.1.6 CORE OPERATING LIMITS REPORT

6.9.1.6.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR), plant procedure PLP-106, prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. SHUTDOWN MARGIN limits for Specification 3/4.1.1.2.

b. Moderator Temperature Coefficient Positive and Negative Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3.

c. Shutdown Bank Insertion Limits for Specification 3/4.1.3.5.

d. Control Bank Insertion Limits for Specification 3/4.1.3.6.

e. Axial Flux Difference Limits for Specification 3/4.2.1.

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

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

h. Boron Concentration for Specification 3/4.9.1.

6.9.1.6.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC at the time the reload analyses are performed, and the approved revision number shall be identified in the COLR.

a. XN-75-27(P)(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," approved version as specified in the COLR.

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

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

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

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

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

No changes to this page. Included for information only.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

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

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

e. EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

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

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

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

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

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

No changes to this page. Included for information only.

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

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

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

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

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

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

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

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

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

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

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

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

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

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

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

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

o. Mechanical Design Methodologies

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

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

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

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

Insert 1
(see next page)

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).

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, Insert 1:

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)

Attachment 5

Retyped Technical Specification Pages

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

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(continued)

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).

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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.

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c. Nondestructive examination techniques utilized for each degradation mechanism,