Duke Energy Carolinas
Thermal-Hydraulic Statistical Core Design Methodology

DPND-DPC-NE-2005-A
Revision 4a
Editorial

December 2008

Nuclear Engineering Division
Nuclear Generation Department
Duke Energy Carolinas, LLC

NON-PROPRIETARY

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Revision 4

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

October 29, 2008

Mr. Dave Baxter
Vice President, Oconee Site
Duke Power Company LLC
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: 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)

Dear Mr. Baxter:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 362, 364, and 363 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 22, 2007, supplemented July 14, September 17, and October 27, 2008.

These amendments revise TSs to allow the accommodation of AREVA NP Mark-B-HTP fuel.

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

Sincerely,

Leonard N. Olshan, Sr. Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. Amendment No. 362 to DPR-38
2. Amendment No. 364 to DPR-47
3. Amendment No. 363 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 362
Renewed License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. DPR-38 filed by the Duke Energy Carolinas, LLC (the licensee), dated October 22, 2007, and supplemented July 14, September 17, and October 27, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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-38 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No.362, 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 its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John Stang For

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-38
and the Technical Specifications

Date of Issuance: October 29, 2008

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 364
Renewed License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility), Renewed Facility Operating License No. DPR-47 filed by the Duke Energy Carolinas, LLC (the licensee), dated October 22, 2007, and supplemented July 14, September 17, and October 27, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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-47 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 364, 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 its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

For

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-47
and the Technical Specifications

Date of Issuance: October 29, 2008

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 363
Renewed License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility), Renewed Facility Operating License No. DPR-55 filed by the Duke Energy Carolinas, LLC (the licensee), dated October 22, 2007, and supplemented July 14, September 17, and October 27, 2008, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

   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 hereby amended by page 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-55 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 363, 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 its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

For

Melanie C. Wong, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. DPR-55 and the Technical Specifications

Date of Issuance: October 29, 2008

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 362 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 364 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AND

AMENDMENT NO. 363 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By application dated October 22, 2007, to the U.S. Nuclear Regulatory Commission (NRC) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML 072990298), as supplemented by letters dated July 14, 2008 (ADAMS Accession No. ML082000134), September 17, 2008 (ADAMS Accession No. ML082700552), and October 27, 2008 (ADAMS Accession No. ML083020297), Duke Energy Carolinas, LLC (Duke, the licensee), requested changes to the Technical Specifications (TSs) for the Oconee Nuclear Station, Units 1, 2, and 3. DPC-NE-2015-P, "Oconee Nuclear Station, Mark-B-HTP Fuel Transition Methodology," was provided as Attachment 3 to the October 22, 2007, application; a non-proprietary version of DPC-NE-2015-P is available under ADAMS Accession No. ML082690091.

The supplements dated July 14, September 17, and October 27, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff original proposed no significant hazards consideration determination as published in the *Federal Register* on November 20, 2007 (72 FR 65365).

The proposed changes would revise the TSs to accommodate use of AREVA NP Mark-B-HTP fuel at Oconee.

The licensee plans to transition to the Mark-B-HTP fuel assemblies from the current Mark-B11 fuel assemblies, both AREVA NP fuel designs, for the core reloads beginning in 2008. The Mark-B-HTP fuel design is an evolution of the standard Mark-B fuel product line. The Mark-B-HTP fuel assembly is a 15x15 array design with M5 fuel rods, instrument tube, and guide tubes. The Mark-B-HTP fuel is more resistant to grid-to-rod fretting and uses the AREVA NP BHTP critical heat flux (CHF) correlation. The M5 material was approved in the topical report BAW-10227P-A,

entitled "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel." Introduction of the Mark-B-HTP fuel design requires revision to seven of the approved analytical methodology reports in the reload design process. The licensee consolidated all revisions to these previously approved reports into one reload report, DPC-NE-2015-P, which provides supporting information for the license amendment request. This report describes revisions to the methodologies for performing the nuclear design, mechanical design, thermal-hydraulic design, and the Chapter 15 non-loss-of-coolant-accident (LOCA) transient and accident analyses that are needed to use Mark-B-HTP fuel at Oconee. Some of the revisions are not associated with the change in fuel design, but are included for improvements, error corrections, and editorial clarification. The proposed license amendments will revise the TSs and associated Bases, which is necessary for the methodology revisions.

2.0 REGULATORY EVALUATION

The licensee requested license amendments to revise the TSs to transition to the Mark-B-HTP fuel for Oconee.

The regulations in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.90, "Application for amendment of license, construction permit, or early site permit" states that the holder of a license that desires to amend the license may file the application for an amendment with the NRC. 10 CFR 50.92, "Issuance of amendment," specifies that the NRC staff will be guided by the considerations that govern the issuance of initial licenses to the extent applicable and appropriate in determining whether an amendment will be issued to the applicant.

The following criteria from Appendix A of Part 50, "General Design Criteria for Nuclear Power Plants," apply: Criterion 10 – Reactor design, Criterion 11- Reactor inherent protection, and Criterion 28 – Reactor limits.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed DPC-NE-2015 in its entirety. The following technical evaluation addresses the revisions in DPC-NE-2015-P that are essential to the NRC staff's findings necessary to grant the requested license amendments.

3.1 Mechanical Design

3.1.1 Burnup Limit

The Mark-B-HTP fuel assembly to be used at the Oconee is an AREVA NP 15x15 fuel design with M5 cladding, instrument, and guide tubes. The intermediate and top spacer grids are also made of M5. The bottom spacer grid and upper and lower end fittings are made of Inconel 718. The M5 material was approved in the topical report, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," BAW-10227P-A, Revision 1, June 2003. The Mark-B-HTP fuel is an evolution of the standard Mark-B fuel product. The M5 material and Mark-B fuel design were approved to a rod average burnup limit of 62 GWd/MTU. The licensee indicated that fuel rod mechanical analyses were performed with the TACO3 fuel performance code. The TACO3 code was approved for the licensee's licensing applications in the methodology report, "Duke Power

Company Fuel Mechanical Reload Analysis Methodology Using TACO3," DPC-NE-2008P-A, Revision 0, April 1995, to a rod average burnup limit of 62 GWd/MTU.

Based on the approved reports, the NRC staff concludes that the Mark-B-HTP fuel design is approved to the rod average burnup limit of 62 GWd/MTU for Oconee.

3.1.2 Cladding Corrosion

Previously, the NRC staff approved AREVA NP high burnup applications in the topical report, "Extended Burnup Evaluation," BAW-10186P-A, Revision 2, June 2003. BAW-10186P-A, Revision 2 includes a best-estimate cladding corrosion model, COROS02, for zircaloy-4 fuel rods with a design limit of 100 microns in the corrosion analysis. The licensee will perform cladding corrosion analysis based on the AREVA NP methodology approved by the NRC, including the COROS02 corrosion model. However, the licensee had indicated that the COROS02 calculated results are reduced by certain amount to determine the best-estimate oxide thickness. Since the NRC staff had no knowledge of this provision in determining the best-estimate results, the NRC staff informed the licensee that the NRC staff did not approve such a reduction, which amounted to certain credit, in previous safety evaluations. Since the Mark-B-HTP fuel assembly uses the M5 cladding, the NRC staff recognizes that the approval of M5 in BAW-10227P-A does not encompass such a reduction for the best-estimate corrosion calculation. In addition, the amount of corrosion in the M5 cladding is generally much less than the amount in the zircaloy-4 cladding; a reduction could render unrealistically low corrosion for the M5 cladding. In fact, the NRC staff considers that the corrosion model in BAW-10227P-A is a best-estimate model. The best-estimate results are directly calculated from the cladding corrosion model without any reduction. Therefore, the NRC staff informed the licensee that the use of the reduction in determining the best-estimate results in the cladding corrosion model was not acceptable. Therefore, by letter dated October 27, 2008, the licensee stated that it will take no reduction in the COROS02 corrosion model for the calculated oxide thickness.

The NRC staff concludes that the approved model and the design limit of 100 microns is acceptable for analyzing cladding corrosion for the Mark-B-HTP fuel design for Oconee.

3.1.3 LOCA and Seismic Loading

Earthquake and postulated pipe breaks in the reactor coolant system would result in extreme forces on the fuel assembly. Section 4.2 of Appendix A to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," states that the fuel system coolable geometry shall be maintained and damage should not be so severe as to prevent control rod insertion during seismic and LOCA events.

In its letter dated September 17, 2008, the licensee analyzed the worst case loading on fuel assemblies due to LOCA and seismic events for the beginning of life (BOL) and end of life (EOL). The loads include core flood and decay heat LOCA loads and safe-shutdown earthquake seismic loads. Mixed and all Mark-BHTP fuel assemblies were evaluated. The licensee uses the square-root-of-sum-of-squares (SRSS) method to combine the two loads. The results show that the maximum impact load on fuel assemblies remains below the grid crushing load for the worst case core configuration. Thus, the fuel rod fragmentation does not occur and fuel coolability is maintained for the LOCA and seismic events. The NRC staff reviewed the results and concludes that the LOCA and seismic analyses are acceptable.

The NRC staff concludes that, based on these analyses, the mechanical design of the Mark-B-HTP fuel for Oconee is acceptable.

3.2 Nuclear and Reload Design

Three approved methodology reports; "Oconee Nuclear Station Reload Design Methodology," NFS-1001A, Revision 5, January 2001; "Oconee Nuclear Station Reload Design Methodology II," DPC-NE-1002-A, Revision 2, October 1985; and "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," DPC-NE-1004-A, Revision 1, December 1997; form the basis of nuclear and reload design for Oconee.

The revisions in the nuclear and reload design include the effect of fuel assembly bow, nuclear uncertainty factors, and the fuel densification power spike factor. The effect of fuel assembly bow on the pin power distribution is accounted for by a penalty factor in the analyses of fuel melting, clad strain, departure from nucleate boiling ratio (DNBR) transient, and LOCA. In its letter dated September 17, 2008, the licensee demonstrated that the fuel assembly bow peaking factor was statistically combined with other factors to form a single uncertainty factor using the SRSS method. The NRC staff reviewed the response and concludes that the analysis of assembly bow on the pin power distribution is acceptable.

The nuclear uncertainty factors were revised using the CASMO-3/SIMULATE-3P code as described in the approved report DPC-NE-1004-A. The fuel densification power spike factor was revised with an axially-dependent factor based on the approved report NFS-1001A. Based on the approved methodology reports, the NRC staff reviewed the nuclear uncertainty and densification factors and concludes that the analyses using these factors are acceptable for Oconee.

The NRC staff concludes that the revisions to the approved methodology reports and the analyses performed by the licensee are acceptable for the nuclear and reload design of Mark-B-HTP fuel at Oconee.

3.3 Thermal-Hydraulic Design

Two approved methodology reports; "Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2003P-A, Revision 1, September 2000, and "Thermal-Hydraulic Statistical Core Design Methodology," DPC-NE-2005-PA, Revision 3, September 2002; form the basis of thermal-hydraulic design for Oconee.

The licensee used the VIPRE-01 code for steady-state core thermal-hydraulic analyses. The licensee modeled the Mark-B-HTP fuel using the methodology described in DPC-NE-2003P-A for the analysis. Two CHF correlations, BHTP and BWU-N, were used. The BHTP correlation, which was approved in "BHTP DNB Correlation Applied with LYNXT," BAW-10241(P)(A), Revision 1, September 2004, was used for the fuel above the first intermediate grid spacer. The BWU-N correlation, which was approved in "The BWU Critical Heat Flux Correlations," BAW-10199P-A, August 1996 (and including Addendum, December 2000), was used for the fuel below the first intermediate grid spacer. Based on the BHTP correlation, the licensee determined the DNBR safety limit of 1.132 for Modes 1 and 2. Based on the approved correlation, the NRC staff concludes that the DNBR limit is acceptable for Oconee.

A new Appendix F, "Application of BHTP CHF Correlation to the Mark-B-HTP Fuel Design," is added to DPC-NE-2005-PA. Appendix F describes a methodology of DNB statistical design limits, parameters and uncertainties, and transition cores using VIPRE-01 with the BHTP correlation for the Mark-B-HTP fuel design. Appendix F is essentially similar to the approved Appendix D in DPC-NE-2005-PA for the current Mark-B11 fuel design. The licensee will continue to conform to the Limitations and Conditions described in the safety evaluation of BAW-10241(P)(A), Revision 1 for the use of BHTP CHF correlation. The NRC staff concludes that Appendix F is acceptable to incorporate into the approved methodology report DPC-NE-2005-PA.

For mixed core of the Mark-B-HTP and Mark-B11 fuel assemblies, the licensee analyzed the transition core penalty using the approved VIPRE-01 code. The licensee also analyzed the statistical DNB results using the approved methodology in DPC-NE-2005-PA. Based on the licensee's analysis using approved methodologies with appropriate revisions, the NRC staff concludes that the mixed core analysis is acceptable for Oconee.

The NRC staff concludes that the revisions to the approved methodology reports are acceptable for analyzing the thermal-hydraulic design of Mark-B-HTP fuel at Oconee.

3.4 Non-LOCA Transient and Accident Analyses

Two approved methodology reports; "Thermal-Hydraulic Transient Analysis Methodology," DPC-NE-3000-PA, Revision 3, September 2004, and "UFSAR Chapter 15 Transient Analysis Methodology," DPC-NE-3005-PA, Revision 2, May 2005; form the basis for non-LOCA transient and accident analyses. DPC-NE-3000-PA includes two codes, RETRAN-3D and VIPRE-01. DPC-NE-3005-PA encompasses three different codes, RETRAN-3D, CASMO-3/SIMULATE-3 and SIMULATE-3K. DPC-NE-3005-PA was revised to provide initial conditions and boundary conditions for Mark-B-HTP fuel in the Updated Final Safety Analysis Chapter 15 analysis.

A new Appendix D, "Methodology Revisions for Mark-B-HTP Fuel," is added to DPC-NE-3000-PA. Appendix D provides a description of design parameters in developing the RETRAN-3D and the VIPRE-01 models for the Mark-B-HTP fuel design. The BHTP and BWU CHF correlations are used, as indicated in Section 3.3, for most of the DNBR analyses. For the main steam line break analysis, the approved Modified-Barnett correlation, as described in "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," EMF-2310(P)(A), Revision 1, May 2004, was used in the low pressure regime. The NRC staff concludes that Appendix D is acceptable to incorporate into approved methodology report DPC-NE-3000-PA.

The licensee included two new features in the Appendix E, "Expanded Oconee VIPRE-01 Methodology," to DPC-NE-3000-PA. The first feature is a larger nodalization for VIPRE that enables modeling of most of the hot assembly and parts of three adjacent fuel assemblies. The second feature is a revised core power distribution. The current core power distribution is very conservative based on a cosine power distribution provided by the fuel vendor. The licensee developed a revised core power distribution using the approved SIMULATE-3 model, which reflects the current reload core design of a flattened power distribution. The licensee reasons that a flattened power distribution will cause less cross flow in the sub-channels and increase the number of limiting sub-channels. Thus, a flattened power distribution is considered conservative for the DNB analysis. The revised core power distribution will be used in the same manner as the current vendor-supplied power distribution. The licensee will perform analyses to confirm that the revised core power distribution remains conservative for future reload cores, or a new revised core

power distribution will be developed using the same process. Based on the adequate conservatism in the approved methodology report DPC-NE-3000-PA, the NRC staff concludes that the revised Appendix E is acceptable to incorporate into DPC-NE-3000-PA.

The licensee developed two approaches of the mixed core analysis to account for DNBR penalty. The first approach modeled the hot Mark-B-HTP fuel assembly surrounded by a lumped channel representing many co-resident Mark-B11 fuel assemblies. This approach maximizes the flow diversion out of the hot assembly, resulting in a very conservative mixed core penalty as described in the thermal-hydraulic design. The second approach explicitly modeled the actual mixed core loading of the Mark-B-HTP and Mark-B11 fuel assemblies. The second approach depicts realistically the flow diversion out of the hot assembly resulting in a conservative, though less conservative than the first approach, mixed core penalty. In its September 17, 2008, the licensee indicated that the second approach will be adopted for non-LOCA transient and accident analyses, because the first approach is too restrictive for the analysis. Based on the adequate conservatism in the approved methodology reports DPC-NE-3000-PA and DPC-NE-3005-PA, the NRC staff approves the second approach for the mixed core analysis.

Based on these approved methodology reports, the NRC staff concludes that the analysis methodology of non-LOCA transients and accidents is acceptable for Mark-B-HTP fuel at Oconee.

3.5 LOCA Analysis

The licensee will perform the LOCA analysis using the approved LOCA Evaluation Model (EM), as described in "BWNT LOCA – BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," BAW-10192P-A, June 1998. The approved "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," BAW-10164P-A, Revision 6, June 2007, describes the RELAP5 code that is used for simulating the LOCA conditions. The NRC staff has approved Revision 6 to BAW-10164P-A for analyzing the Mark-B-HTP fuel design.

The NRC staff concludes that the approved topical reports BAW-10192P-A and BAW-10164P-A are acceptable for LOCA analysis of Mark-B-HTP fuel at Oconee.

3.6 Technical Specification (TS) Revisions

3.6.1 Section 2.1.1.2, Reactor Core Safety Limits

The licensee will add the BHTP CHF correlation in TS Section 2.1.1.2 for the Mark-B-HTP fuel design. TS 2.1.1.2 will be revised as follows:

"In MODES 1 and 2, ...1.19 for the BWU correlation, and 1.132 for the BHTP correlation. Operation within these limits..."

Based on the preceding technical evaluation, the NRC staff finds this revision acceptable.

3.6.2 Section 5.6.5.b, Core Operating Limits Report (COLR)

The proposed revision to the COLR will add the BHTP CHF correlation, which is applicable to the Mark-B-HTP fuel design, to the RELAP5 code described in the approved topical report BAW-10164-PA. Based on the preceding technical evaluation, this revision is acceptable.

3.6.3 Bases, Section B 2.1.1, Reactor Core SLs

The licensee will revise TS Bases, Section B 2.1.1 to include the BHTP CHF correlation for the Mark-B-HTP fuel design and to add BAW-10241(P)(A). Based on the preceding technical evaluation, the revision is acceptable.

3.6.4 Bases, Section B 3.4.1, RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

The licensee will add the BHTP CHF correlation for the Mark-B-HTP fuel design to the TS Bases, Section B 3.4.1. Based on the preceding technical evaluation, the revision is acceptable.

4.0 SUMMARY

In summary, the NRC staff has reviewed the licensee's license amendment request for TS revisions and concludes that the TS revisions are acceptable. The NRC staff has also reviewed DPC-NE-2015-P; which discusses changes to the following previously approved methodology reports: NFS-1001A, DPC-NE-1002-A, DPC-NE-2008P-A, DPC-NE-2003P-A, DPC-NE-2005P-A, DPC-NE-3000-PA, and DPC-NE-3005-PA. Based on the NRC staff's review of DPC-NE-2015-P, including the preceding technical evaluation, the NRC staff approves the methodology revisions in DPC-NE-2015-P for use of Mark-B-HTP fuel at Oconee.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (72 FR 65365, November 20, 2007). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Shih-Liang Wu, NRR/SNPB

Date: October 29, 2008

ELL

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

December 18, 2002

Mr. M. S. Tuckman
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church St
Charlotte, NC 28202

SUBJECT: 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)

Dear Mr. Tuckman:

The Nuclear Regulatory Commission staff has completed its review of the revision to the topical report "Duke Power Company Westinghouse Fuel Transition Report, DPC-NE-2009P, Revision 2," submitted by the Duke Power Company (DPC) in a letter dated February 28, 2002, as supplemented by letter dated September 9, 2002. The report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the enclosed NRC Safety Evaluation. The safety evaluation defines the basis for acceptance of the report.

The staff does not intend to repeat its review of the matters described in the report and found acceptable when the report is referenced in future license applications, except to ensure that the material presented is applicable to the specific plant involved. Staff acceptance applies only to the matters described in the report.

We request that DPC publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed safety evaluation between the title page and the abstract. The accepted versions should include an " -A" (designating accepted) following the report identification symbol.

Should NRC criteria or regulations change so that staff conclusions regarding the acceptability of the report are invalidated, DPC will be expected to revise and resubmit its documentation,

or to submit justification for continued effective applicability of the topical report without revision of its documentation.

Should you have questions or comments, please contact Mr. Robert Martin of my staff at (301) 415-1493.

Sincerely,

H E Eaton for

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

Enclosure: As stated

cc w/encl: See next page

McGuire Nuclear Station
Catawba Nuclear Station

cc:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT DPC-NE-2009P, REVISION 2

DUKE POWER COMPANY WESTINGHOUSE FUEL TRANSITION REPORT

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 and 2

DUKE ENERGY CORPORATION

DOCKET NOS. 50-413, 50-414, 50-369, AND 50-370

1.0 INTRODUCTION

By letter dated February 28, 2002, (Reference 1), as supplemented by letter dated September 9, 2002 (Reference 2), Duke Power Company (DPC), a subsidiary of Duke Energy Company and the licensee for the operation of Catawba Nuclear Station (CNS), Units 1 and 2, and McGuire Nuclear Station (MNS), Units 1 and 2, submitted for NRC review and approval, the report DPC-NE-2009-P, Revision 2, "Duke Power Company Westinghouse Fuel Transition Report," dated February 2002.

The initial topical report DPC-NE-2009-P-A described the methodologies used for reload design analyses to support the licensing basis for the transition from Framatome Mark-BW fuel assemblies to the Westinghouse 17x17 Robust Fuel Assembly (RFA) design in the CNS and MNS reload cores. These methodologies include the core design, fuel rod design, thermal-hydraulic analysis, and accident analysis methodologies. The Nuclear Regulatory Commission (NRC) staff approved the report in September 1999 (References 2 and 3). In its letter of October 7, 2001 (Reference 4), as amended by its letter of August 7, 2002 (Reference 5), the licensee submitted Revision 1 of DPC-NE-2009-P for NRC staff review. Revision 1 consisted of changes to Chapter 6, "Updated Final Safety Analysis Report (UFSAR) Accident Analysis." The NRC staff approved Revision 1 of DPC-NE-2009 on October 1, 2002 (References 6 and 7).

Revision 2 of DPC-NE-2009 contains changes to Chapters 5, "Thermal-Hydraulic Analysis," to increase the reference peaking values for the Westinghouse RFA fuel. The licensee stated that this increase is due to additional departure from nucleate boiling (DNB) performance margin inherent in the fuel design. There are also some administrative updates in sections 2 and 4 of the topical report.

2.0 EVALUATION

Since the NRC has approved topical report DPC-2009-P-A, as well as Revision 1, the staff's review of Revision 2 was limited to those issues identified in Revision 2. The staff review of this

revision is based on evaluation of technical merit and compliance with applicable regulatory requirements.

General Design Criterion (GDC) 10, "Reactor Design" in Appendix A to 10 CFR Part 50, specifies that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOO). Standard Review Plan Section 4.4 describes a specific criterion to meet the requirement of GDC 10, which is to provide assurance of at least a 95 percent probability at a 95 percent confidence level that the hot fuel rod in the core does not experience a DNB during normal operation or AOO. The acceptance criterion is that the minimum departure from nucleate boiling ratio (DNBR) in the hot channel in the core calculated with an approved critical heat flux correlation for all AOOs is higher than the minimum DNBR limit established for the correlation. The staff evaluated the revisions related to the thermal-hydraulic analysis methodology for compliance with the minimum DNBR acceptance criterion.

2.1 Changes to Section 2, Fuel Design:

Section 2.0, "Fuel Design," of the topical report describes the RFA design features, such as the features initially licensed with the VANTAGE+ fuel design, the features that help mitigate debris failures and incomplete rod insertion, and other features. A discussion is also included of the Quick Release Top Nozzle (QRTN), as addressed using the Westinghouse Fuel Criteria Evaluation Process (FCEP) described in WCAP-12488-P-A (Reference 8).

Revision 2 of the topical report makes the following revisions to Section 2:

- Adds Reference 2-6 [Westinghouse FCEP notification letter (Reference 9)] to Section 2.1, "References."
- Adds a sentence to Section 2.0 stating that "Westinghouse sent notification per Reference 2-2 [WCAP-12488-P-A] to the NRC in Reference 2-6 confirming batch implementation of the QRTN at McGuire and Catawba."

The staff has reviewed the Westinghouse FCEP notification letter of Reference 2-6, and found it consistent with the fuel criteria evaluation process. The change to Section 2.0 of the topical report to mention the transmittal of the FCEP notification letter to NRC is an administrative change for completeness, and is, therefore acceptable.

2.2 Changes to Section 4.0, Fuel Rod Analysis:

Section 4.0, "Fuel Rod Analysis," of the topical report describes the fuel rod mechanical reload analysis methodology for the Westinghouse RFA fuel. In particular, the PAD code described in topical report WCAP-10851-P-A (Reference 10) is used for detailed fuel rod design analyses. Subsequent to the approval of DPC-NE-2009, the NRC staff approved the PAD 4.0 code described in WCAP-15063-P-A (Reference 11) in July 2000.

Revision 2 of the topical report makes the following revisions to Section 4:

- Adds WCAP-15063-P-A to Section 4.3, "References," as Reference 4-14.
- Revises the last paragraph in Section 4.1, "Computer Code," to state that "In July of 2000, Westinghouse received approval for PAD 4.0 (Reference 4-14). This newest version of the code includes a revised cladding creep model and irradiation growth model as well as updated cladding and oxide thermal conductivity values. Duke Power is implementing PAD 4.0 in the same forward fit approach as outlined in Reference 4-14."
- Adds Reference 4-14 in various places in Sections 4.1 and 4.2 where the PAD code is mentioned.

Since WCAP-15063-P-A has been approved by the NRC staff, its reference for licensing application and the use of PAD 4.0 for fuel rod analysis are acceptable.

2.3 Changes to Section 5.2, Thermal-Hydraulic Code and Model:

The thermal-hydraulic analyses for the MNS and CNS cores with the Mark-BW fuel design were performed with the VIPRE-01 code (Reference 12) using the core thermal-hydraulic models described in DPC-NE-2004P-A (Reference 13) and the statistical core design (SCD) methodology described in DPC-NE-2005P-A (Reference 14). Section 5.2, "Thermal-Hydraulic Code and Model," of the topical report describes the use of VIPRE-01 for the analysis of the Westinghouse RFA design. This includes: (1) use of the RFA design fuel geometry and form loss coefficients for the core models, (2) use of the WRB-2M critical heat flux (CHF) correlation, and (3) use of the Electric Power Research Institute (EPRI) subcooled boiling model and the EPRI bulk void model for the two-phase flow calculations.

Revision 2 makes the following changes to Section 5.2 in the use of the VIPRE-01 models:

- Increases the reference pin peaking factor from 1.60 to 1.67, and the associated pin power distributions were updated based on the higher reference peaking factor.
- Increases the reference axial power profile peak-to-average value from 1.55 to 1.60.
- Adds Figures 5-1, 5-2, and 5-3, for the 8, 12, and 75-Channel Models, respectively, with the new reference power distributions corresponding to the reference pin peaking factor of 1.67.
- Adds a new paragraph that explains the reasons for the increased referenced peaking factors.

The reference pin and axial peaking factors and power distributions are used to determine the core DNB limits, which are the combinations of power and coolant inlet temperature and pressure at which the minimum DNBR equals the design DNBR limit. The design DNBR limit maintains a margin to the statistical DNBR limit, which is determined from the SCD. The DNB margin allows for mechanisms that could adversely impact DNB, such as the reactor coolant system flow anomaly and transition core effects. The licensee stated that the new higher

reference radial and axial peaking factors are a result of applying, in core design space, the significant DNB margin realized from the intermediate flow mixing grids of the RFA design. With respect to radial peaking, all three models (8, 12, and 75-channel models) described in DPC-NE-2004P-A are based on the maximum pin power value, and therefore, Figures 5-1, 5-2, and 5-3 for these three models are updated to reflect the new peak pin value of 1.67.

During a reload analysis, the core DNB limits will be developed based on the new reference peaking factors and power distributions. The maximum allowable peaking limits will be determined and maneuvering analyses will be performed, and the rod insertion limits or the axial flux difference limits will be revised, if necessary, to ensure that the design DNBR limit is met during normal operation and AOOs. Therefore the adequacy of these new higher reference pin and axial peaking factors and power distributions, with respect to the DNBR limit, is demonstrated during the reload analyses with the RFA design. Therefore, the staff finds the above changes to be acceptable.

2.4 Change in Critical Heat Flux Correlation:

Section 5.3, "Critical Heat Flux Correlation," of the topical report describes the use of the WRB-2M CHF correlation with the VIPRE-01 core thermal-hydraulic analysis code for all statepoint DNBR calculations, with the exception of the steam line break transient.

Revision 2 revises Section 5.3 by adding one more exception, in addition to the steam line break, to the use of the WRB-2M correlation. The exception is to use the BWU-N CHF correlation, rather than the WRB-2M correlation, for the non-mixing vane span of the RFA fuel (located below the first mixing vane zircaloy grid). In addition, topical report BAW-10199P-A (Reference 15), that documents the BWU CHF correlation, including the non-mixing vane BWU-N correlation, is added to Section 5.8, "References."

In response to a staff question (Reference 16), the licensee provided justification for the applicability of the BWU-N correlation to the RFA fuel non-mixing vane span. The determination of the applicability is based on the comparison of the BWU-N correlation data base to the RFA geometric design parameters, and to the thermal-hydraulic conditions of the RFA fuel at MNS and CNS.

The staff reviewed the BWU-N correlation and the non-mixing vane CHF test data base described in topical report BAW-10199P-A. The BWU-N correlation consists of (1) the uniform heat flux base correlation, which is correlated with the thermal-hydraulic local conditions of pressure, mass velocity, and quality, and (2) the non-uniform heat flux F-factor, which is correlated with the rod average heat flux, axial power shape, and local heat flux, and the CHF axial location. (The heated length and spacer grid spacing correction factor is irrelevant as it is set to a value of 1.0 for the non-mixing vane correlation.) The applicability ranges of the parameters within the base correlation and the F-factor cover the thermal-hydraulic operating ranges of the RFA fuel at MNS and CNS. The heated length and grid spacing of the RFA fuel are also within the CHF test data base. However, other RFA fuel geometric parameters are slightly outside the BWU-N correlation data base. The RFA rod diameter and pitch are about 1.3 percent outside the correlation data base, and the pitch to diameter ratio and hydraulic diameter are about 0.3 percent outside the data base.

The licensee contends that only very small extrapolations are necessary to apply BWU-N to the RFA fuel. The licensee further states that the use of BWU-N is based on the similarity of the design, the fact that the geometric variables are not included in the base BWU correlation, and the fact that BWU-N results in conservative levels of CHF compared to mixing vane correlations.

Although the fuel geometric parameters are not included in the BWU-N correlation, the staff considers them important in the applicability of the correlation. The correlation was developed based on the fuel design with specific geometric characteristics. The staff would be concerned with the application of a CHF correlation to the full axial length of a fuel design that was not covered by the correlation data base. Even though the differences between the RFA geometric variable and the BWU-N correlation data base are very small (less than 1.3 percent), the acceptability of extrapolating the correlation applicable ranges would be questionable in such a case. However, since the licensee will only apply the BWU-N correlation to the non-mixing vane portion at the very bottom span portion (lower 21 inches of the heated length) of the RFA fuel design, where the coolant condition is such that the minimum DNBR generally does not occur, the staff concludes that the use of BWU-N in this span would have no impact on the minimum DNBR calculations, and is therefore acceptable.

2.5 Changes to Section 5.7, Transition Cores:

Section 5.7 of the topical report describes the transition core model used to determine the impact on DNBR of the geometric and hydraulic differences between the resident Mark-BW fuel and the RFA design. The analysis uses the 8-channel model to evaluate the impact or penalty for transition cores.

In Revision 2 of the topical report, the paragraph that states "[a] transition core DNBR penalty is determined for the RFA design using the 8-channel RFA/Mark-BW transition core model," is replaced with a new paragraph. The new paragraph is as follows:

> For initial transition reload cycles, a transition core DNBR penalty is determined for the RFA design using the 8 channel RFA/Mark-BW transition core model. For subsequent cycles where the RFA fuel composes greater than 80 percent of the assemblies incore, the 75- channel model shown in Figure 5-3 and described in Reference 5-1 [DPC-NE-2004P-A] is used to determine a transition core penalty. In either case, a conservative penalty is applied for all DNBR analyses in transition cycles to bound the effects of mixed cores.

The licensee, in response to a staff question (Reference 16), explained the need to use the 75-channel model for the calculation of the mixed core penalty when the RFA design composes more than 80 percent of the transition cores. Specifically, the RFA design contains 3 extra mixing-vane grids in the upper span compared to the Mark-BW fuel and the higher hydraulic resistance of the RFA assemblies forces flow out of the RFA assemblies into the surrounding Mark-BW assemblies during a transition mixed core. In the 8-channel model, the core is conservatively assumed to be one RFA assembly surrounded by 192 Mark-BW assemblies, where the single RFA hot assembly is modeled by the first 7 channels with the remainder of the core lumped into one single channel. This model maximizes the hydraulic difference in the transition cores and creates a bounding penalty for the RFAs. This penalty becomes more conservative as more RFA fuel assemblies are used in the transition. When the RFA fuel

constitutes more than 80 percent of the core, it is appropriate to use the more detailed 75-channel core model to better represent the hydraulic effects, and to determine a more realistic mixed core penalty than the 8-channel model would provide.

Since the 75-channel core model has also been approved by the NRC as described in DPC-NE-2004P-A, the staff finds the use of the 75-channel core model to be acceptable for the determination of the transition core penalty when the RFA fuel constitutes more than 80 percent of the core.

2.6 Typographic Error Corrections:

Revision 2 of the topical report also corrects two typographical errors. They are "Imtermediate" in Table 2-1, and "chararcteristic" in the last sentence of the sixth paragraph under Section 4.0. They are corrected to "intermediate" and "characteristic," respectively. These editorial changes are acceptable.

3.0 CONCLUSION

The staff has reviewed the Duke Energy Corporation's topical report DPC-NE-2009, Revision 2. The main revisions are related to the thermal-hydraulic analysis methodology for the use of higher reference peaking factors for the RFA fuel, the use of the BWU-N CHF correlation for the very bottom span of the RFA fuel, and the use of the 75-channel core model for the analysis of the transition core penalty when the RFA fuel constitutes more than 80 percent of the fuel in the core. Based on the evaluation described in Sections 2.3, 2.4 and 2.5 above, the staff concludes that these revisions are acceptable. Other revisions include administrative updates for completeness related to a an FCEP notification letter, an approved topical report, and editorial changes, as described in Sections 2.1, 2.2, and 2.6, respectively, of this report.

In summary, the staff concludes that DPC-NE-2009, Revision 2, is acceptable.

4.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation, McGuire Nuclear Station - Units 1 and 2, Docket Nos. 50-369 and 50-370; Catawba Nuclear Station - Units 1 and 2, Docket Nos. 50-413 and 50-414; Topical Report DPC-NE-2009 (TAC Nos. MA2359, MA2361, MA2411, MA2412), Revision 2 - Updates to Chapters 2, 4, and 5)," February 28, 2002.

2. Letter from Frank Rinaldi, NRC, to H. B. Barron, McGuire Site, Duke Energy Corporation, "McGuire Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MA2411 and MA2412)," September 22, 1999.

3. Letter from Peter Tam, NRC, to G. R. Peterson, Catawba Nuclear Station, "Catawba Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MA2359 and MA2361)," September 22, 1999.

4. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; Catawba Nuclear Station - Units 1 and 2, Docket Nos. 50-413 and 50-414; McGuire Nuclear Station - Units 1 and 2, Docket Nos.

50-369 and 50-370; License Amendment Request Applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to Bases 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003," October 7, 2001.

5. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369 and 370; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413 and 414; Response to NRC Request for Additional Information - TAC nos. MB3222, MB3223, MB3343 and MB3344) and License Amendment Request Supplement," August 7, 2002.

6. Letter from R. E. Martin, USNRC, to H. B. Barron, Duke Energy Corporation, "McGuire Nuclear Station, Units 1 and 2 Re: Issuance of Amendments (TAC Nos. MB3222 and MB3223)," October 1, 2002.

7. Letter from C.P. Patel, USNRC, to G. R. Peterson, Duke Energy Corporation, "Catawba Nuclear Station, Units 1 and 2 Re: Issuance of Amendments Re: (TAC Nos. MB3343 and MB3344)," October 1, 2002.

8. WCAP-12488-P-A, "Westinghouse Fuel Criteria Evaluation Process," October 1994.

9. Letter from Henry A. Sepp, Westinghouse, to J. S. Wermiel, USNRC, "Fuel Criterion Evaluation Process (FCEP) Notification of Quick Release Top Nozzle (QRTN) Design," January 15, 2002, LTR-NRC-02-2.

10. WCAP-10851-P-A, "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," August 1988.

11. WCAP-15063-P-A, Revision 1, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," July 2000.

12. EPRI NP-2511-CCm-A, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," August 1989.

13. DPC-NE-2004P-A, Rev. 1, "McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," Battelle Pacific Northwest Laboratories, February 1997.

14. DPC-NE-2005P-A, Rev. 2, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," June 1999.

15. BAW-10199P-A, "The BWU Critical Heat Flux Correlations," Framatome Cogema Fuels, April 1996.

16. Letter from M. S. Tuckman, Duke Energy Corporation, to U.S. Nuclear Regulatory Commission, "Duke Energy Corporation, McGuire Nuclear Station - Units 1 and 2, Docket Nos. 50-369 and 50-370, Catawba Nuclear Station - Units 1 and 2, Docket Nos. 40-413 and 50-414, Topical Report DPC-NE-2009, Revisions 2 - Updates to Chapters 2, 4, and 5 (TAC Nos. MB4502, MB4503, MB4504, MB4505), Response to NRC Request for Additional Information," September 9, 2002.

Principal Contributor: Y. Hsii, DSSA/SRXB

Date: December 18, 2002

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

September 22, 1999

Mr. G. R. Peterson
Site Vice President
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Duke Energy Corporation
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SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA2359 AND MA2361)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 180 to Facility Operating License NPF-35 and Amendment No. 172 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated July 22, 1998, and supplemented by letters dated October 22, 1998, January 28, May 6, June 24, August 17 and September 15, 1999.

The amendments revise various sections of the Technical Specifications (Appendix A of the Catawba operating licenses) to permit use of Westinghouse's Robust Fuel Assemblies for future core reloads. We will publish a Notice of Issuance in the Commission's biweekly *Federal Register* notice.

Concurrent with issuance of these amendments we have also approved topical report DPC-NE-2009, "Duke Power Company Westinghouse Fuel Transition Report." The Safety Evaluation (enclosed) provides details of our review of DPC-NE-2009P in support of the subject amendments. In accordance with procedures established in NUREG-0390, we request Duke Energy Corporation to publish an accepted version of DPC-NE-2009, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed Safety Evaluation after the title page. The accepted versions shall include an "A" (designating accepted) following the report identification symbol. Please include our request for additional information and Duke's response as an appendix to the report.

Sincerely,

Peter S. Tam

Peter S. Tam, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 180 to NPF-35
2. Amendment No. 172 to NPF-52
3. Safety Evaluation

cc w/encls: See next page

Catawba Nuclear Station

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION AND BACKGROUND

By letter dated July 22, 1998 (Ref. 1), and supplemented by a letter of October 22, 1998 (Ref 2), Duke Energy Corporation* (DEC, the licensee), the licensee for operation of McGuire and Catawba Nuclear Stations, proposed changes to the Technical Specifications (TS) of these plants in anticipation of a reactor core reload design using Westinghouse fuel. Accompanying the July 22, 1998, letter is a topical report DPC-NE-2009, "Duke Power Company" Westinghouse Fuel Transition Report," (Ref. 3) for NRC review and approval. When approved, this topical report will be listed in Section 5.6.5 of the Catawba and McGuire TSs as an approved methodology for the determination of the core operating limits.

The reactors of McGuire and Catawba Nuclear Stations are currently using Framatome Cogema Fuels (FCF) Mark-BW fuel assemblies (Ref. 4). The proposed amendment to the TSs would permit transition to the 17x17 Westinghouse Robust Fuel Assembly (RFA) design.

The RFA design is based on the VANTAGE+ fuel assembly design, which has been approved by NRC as described in WCAP-12610-P-A (Ref. 5). The RFA design to be used at McGuire and Catawba, as described in Section 2.0 of DPC-NE-2009, will incorporate the following features in addition to the VANTAGE+ design features:

- increased guide thimble and instrumentation tube outside diameter
- modified low pressure drop structural mid-grids
- modified intermediate flow mixing grids
- pre-oxide coating on the bottom of the fuel rods
- protective bottom grid with longer fuel rod end-plugs
- fuel rods positioned on the bottom nozzle
- a quick release top nozzle

The first three design features listed above were licensed via the Wolf Creek Fuel design (Ref. 6) using the NRC-approved Westinghouse Fuel Criteria Evaluation Process (Ref. 7). The next three features are included to help mitigate debris failures and incomplete rod insertion.

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* The official name of the licensee is Duke Energy Corporation, as is stated in the Catawba and McGuire operating licenses. "Duke Power Company" is a component of Duke Energy Corporation; however, for historical reasons, the licensee used "Duke Energy Corporation" and "Duke Power Company" interchangeably. This safety evaluation follows the licensee's practice.

The licensee states that these three features will be evaluated using the 10 CFR 50.59 process. The quick release top nozzle design is similar to the Reconstitutable Top Nozzle design with modifications for easier removal. This design will be licensed by Westinghouse using the fuel criteria evaluation process.

2.0 EVALUATION

Topical report DPC-NE-2009 provides general information about the RFA design and describes methodologies to be used for reload design analyses to support the licensing basis for the use of the RFA design in the McGuire and Catawba reload cores. These methodologies include DEC's fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The report does not provide the analyses of the core design, thermal-hydraulics and transients and accidents associated with the RFA design. Therefore, this safety evaluation will only address the acceptability of the methodologies described in DPC-NE-2009 for referencing in the analyses for operations with the reactor cores having a mix of Mark-BW and RFA fuel design or a full core of RFA design.

2.1 Fuel Rod Analysis Methodology

During transition periods, the reactor cores in the McGuire and Catawba plants will have both the FCF Mark-BW fuel and the Westinghouse RFA fuel. Section 4 of DPC-NE-2009 describes the fuel rod mechanical reload analysis methodology for the RFA design. While the fuel rod mechanical analyses for Mark-BW fuel will continue to be performed using the licensee's methodology described in DPC-NE-2008P-A (Ref. 8), the Westinghouse RFA fuel thermal-mechanical analyses will be performed using the NRC-approved Westinghouse fuel performance code, PAD 3.4 Code (Ref. 9). The fuel rod design bases for the RFA design are identical to those described in WCAP-12610-P-A (Ref. 5) for the VANTAGE+ fuel.

The staff's review of fuel rod analysis methodology was performed with technical assistance provided by Pacific Northwest National Laboratory (PNNL). PNNL's review findings and conclusion, with which the staff concurs, are described in its technical evaluation report (attached to this safety evaluation). Thus, the staff has found that the DEC design limits and thermal-mechanical analysis methodologies discussed in Section 4.0 of DPC-NE-2009 are acceptable for application by DEC to the RFA fuel design up to the currently approved (Ref. 41, 42, 43) rod average burnup limit of 62 GWd/mtU. The staff has previously performed an environmental assessment for fuel burnup up to 60 GWd/mtU (53 FR 30355, August 11, 1988). Consequently, due to this limitation from the environmental perspective, the licensee proposed (Ref. 44) a license condition. The staff will impose the license condition as proposed by the licensee to read: "The maximum rod average burnup for any rod shall be limited to 60,000 MWd/mtU [60 GWd/mtU] until the completion of an NRC environmental assessment supporting an increased limit."

2.2 Reload Core Design Methodology

For the RFA design, the core model, core operational imbalance limits, and key core physics parameters used to confirm the acceptability of Updated Final Safety Analysis Report (UFSAR) Chapter 15 safety analyses of transients and accidents will be developed with the methodologies described in DPC-NE-1004-A (Ref. 10), DPC-NE-2011P-A (Ref. 11), DPC-NF-2010A (Ref. 12), and DPC-NE-3001-PA (Ref 13). DPC-NE-2011P-A describes the nuclear design methodology for core operating limits of McGuire and Catawba plants. DPC-NF-2010A describes McGuire and Catawba nuclear physics methodology using

two-dimensional PDQ07 and 3-D EPRI-NODE-P models as reactor simulators. DPC-NE-1004A describes an alternative methodology for calculating nuclear physics data using the CASMO-3 fuel assembly depletion code and the SIMULATE-3P 3-D core simulator code for steady-state core physics calculations, substituting for CASMO-2, PDQ07 and EPRI-NODE-P used in DPC-NE-2010A. DPC-NE-3001-PA describes the methodologies, which expand on the reload design methods of DPC-NF-2010A, for systematically verifying that key physics parameters calculated for a reload core, such as control rod worth, reactivity coefficients, and kinetics parameters, are bounded by values assumed in the Chapter 15 licensing analyses. These topical reports have been approved for performing reload analyses for the B&W 177-assembly and/or Westinghouse 193-assembly cores, subject to the conditions specified in the staff's safety evaluations. Because of the similarity between the RFA design and the Mark-BW fuel design with respect to the dimensional characteristics of the fuel pellet, fuel rod and cladding, as well as nuclear characteristics, as shown in Table 2-1 of DPC-NE-2009, the staff concludes that these approved methodologies and core models currently employed in reload design analyses for McGuire and Catawba can be used to perform transition and full-core analyses of the RFA design.

Section 3.2 of DPC-NE-2009 states that conceptual transition core designs using the RFA design have been evaluated and results show that current reload limits remain bounding with respect to key physics parameters. As described in DPC's response to a staff question (Question 1, Ref. 14, January 28, 1999), the conceptual RFA transition core designs were evaluated for the effects of partial and full cores using NRC-approved codes and methods to determine the acceptability of the current licensing bases transient analyses. Key safety parameters, such as Doppler temperature coefficients, moderator temperature coefficients, control bank worth, individual rod worths, boron concentrations, differential boron worths and kinetics data, were calculated for the conceptual core designs and compared against reference values assumed in the UFSAR Chapter 15 accident analyses. The evaluation demonstrated the expected neutronic similarities between reactor cores loaded with RFA fuel and with Mark-BW fuel and the acceptability of key safety parameters assumed in the Chapter 15 accident analyses. Key physics parameters are calculated for each reload core and each new core design. If a key physics parameter is not bounded by the reference value in the UFSAR accident analyses, the affected accidents will be re-analyzed using the new key physics parameter, or the core will be re-designed to produce an acceptable result. The staff agrees that this is an acceptable approach.

The safety evaluation for DPC-NE-1004-A requires additional code validation to ensure that the methodology and nuclear uncertainties remain appropriate for application of CASMO-3 and SIMULATE-3P to fuel designs that differ significantly from those included in the topical report data base. Though the RFA design is not expected to change the magnitude of the nuclear uncertainty factors in DPC-NE-1004, the use of zirconium diboride integral fuel burnable absorber (IFBA) in the RFA is a design change from the burnable absorber types modeled in DEC's current benchmarking data base. DEC has re-evaluated and confirmed the nuclear uncertainties in DPC-NE-1004 to be bounding. This is done by explicitly modeling Sequoyah Unit 2, Cycles 5, 6, and 7, and by performing statistical analysis of the nuclear uncertainty factors. These cores were chosen because they are very similar to McGuire and Catawba and contained both IFBA and wet annular burnable absorber (WABA) fuel. The results, listed in Table 3-1 of DPC-NE-2009, showed that the current licensed nuclear uncertainty factors for the $F_{\Delta H}$, F_Z, and F_Q bound those for the Westinghouse fuel with IFBA and/or WABA burnable absorbers. Boron concentrations, rod worth, and isothermal temperature coefficients were also predicted and found to agree well with the measured data. In response to a staff question (Question 2, Ref. 14) regarding the applicability of the analysis of the Sequoyah core to the

McGuire and Catawba cores, DEC provided comparisons of the analysis results and the measured data of the Sequoyah cores and a list of the differences between the Westinghouse Vantage-5H fuel design used in Sequoyah and the RFA fuel design. The differences are primarily mechanical and do not impact the nuclear performance of the fuel assembly. Design features that do impact the neutronics (i.e., mid-span mixing grids) are specifically accounted for in the nuclear models. Therefore, the results and conclusions reached based on the analysis of Sequoyah core designs are applicable to the RFA fuel design. In addition, the licensee performed a 10 CFR 50.59 evaluation for unreviewed safety question (USQ). Results are as described in response to Question 2c of Ref.14, which demonstrates that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties are applicable to the RFA design. Therefore, DPC-NE-1004A nuclear physics calculation methodology is applicable to the RFA design.

In all nuclear design analyses, both the RFA and the Mark-BW fuel are explicitly modeled in the transition cores. The mixed core model for nuclear design analyses and the use of fuel-specific limits, described in response to a staff's question (Question 3, Ref. 14), are based on the same methodology that is used to set up a nuclear model for a reactor core containing a single fuel type. When establishing operating and reactor protection system limits (i.e., LOCA linear heat rate limit, departure from nucleate boiling (DNB), central fuel melt, transient strain), the fuel-specific limits or a conservative overlay of the limits are used. The staff concludes that the nuclear design analyses for the transition cores are acceptable.

2.3 Thermal-Hydraulic Analysis

Section 5 of DPC-NE-2009 describes the thermal-hydraulic analysis methodologies to be used for the RFA design. The thermal-hydraulic analyses for the existing Mark-BW fuel design are performed with NRC approved methodology using the VIPRE-01 core thermal-hydraulic code (Ref. 15), the BWU-Z critical heat flux (CHF) correlation (Ref. 16), and the thermal-hydraulic statistical core design methodology described in DPC-NE-2004P-A (Ref. 17) and DPC-NE-2005P-A (Ref. 18). As discussed in the ensuing sections of this report, these same methodologies will be used for the analyses of the RFA design with the exception that (1) the WRB-2M CHF correlation (Ref. 19) will be used in place of the BWU-Z correlation, and (2) the EPRI bulk void fraction model will be used in place of the Zuber-Findlay model.

2.3.1 VIPRE-01 Core Thermal Hydraulic Code:

The core thermal hydraulic analysis methodology using the VIPRE-01 code for McGuire and Catawba licensing calculations is described in DPC-NE-2004P-A. The VIPRE-01 models, which have been approved for the Mark-BW fuel, are also applicable to the RFA design with appropriate input of fuel geometry and form loss coefficients consistent with the RFA design. The reference pin power distribution based on an enthalpy rise factor, $F_{\Delta H}{}^{N}$, of 1.60 peak pin from DPC-NE-2004P-A will continue to be used to analyze the RFA design.

VIPRE-01 contains various void-quality relation models for two-phase flow calculation, in addition to the homogeneous equilibrium model. Either the Levy model or the EPRI model can be chosen for subcooled boiling, and the Zuber-Findlay or EPRI void models for bulk boiling. The combination of Levy subcooled boiling correlation and Zuber-Findlay bulk boiling model gives reasonable results for void fraction. This combination is currently used for McGuire/Catawba cores with the Mark-BW fuel. However, the Zuber-Findlay correlation is applicable only to qualities below approximately 0.7, and there is a discontinuity at a quality of 1.0. The licensee proposes to replace this combination with the combination of EPRI

subcooled and bulk void models. The use of the EPRI bulk void model, which is essentially the same as the Zuber-Findlay model except for the equation used to calculate the drift velocity, is to eliminate a discontinuity at qualities about 1.0. Also, the use of the EPRI subcooled void model is for overall model compatibility to have the EPRI models cover the full range of void fraction required for performing departure-from-nucleate-boiling calculations. To evaluate the impact of these model changes, the licensee performed an analysis of 51 RFA CHF test data points using both Levy/Zuber-Findlay and EPRI models in VIPRE-01. The results show a negligible 0.1 percent difference in the minimum departure-from-nucleate-boiling ratios (DNBRs). Therefore, the staff finds that the use of the EPRI subcooled and bulk void correlations for the analysis of the RFA design is acceptable. The acceptability of this revision remains subject to the limitations set forth in the safety evaluation on VIPRE-01 (EPRI NP-2511-CCM-A), DPC-NE-2004P-A and attendant revisions.

2.3.2 Critical Heat Flux (CHF) Correlation:

The licensee stated that the WRB-2M CHF correlation, described in the Westinghouse topical report WCAP-15025-P-A (Ref. 19), will be used for the RFA design. The WRB-2M correlation was developed by Westinghouse for application to new fuel designs such as the Modified Vantage 5H and Modified Vantage 5H/IFM. The WRB-2M correlation was programmed into the Westinghouse thermal hydraulic code THINC-IV or the VIPRE-01 thermal-hydraulic code for the calculation of the local conditions within the rod bundles. The staff has reviewed and approved the WRB-2M correlation with both THINC-IV and VIPRE-01 codes as described in References 20 and 21. The WRB-2M correlation is also applicable to the RFA design because of its similarity to the Vantage 5H fuel design. The staff concludes DEC's use of the WRB-2M along with VIPRE-01 in the DNBR calculations for the RFA design to be acceptable within the ranges of applicability of important thermal hydraulic parameters specified in the staff's safety evaluation on WCAP-15025-P-A (Ref. 20).

2.3.3 Thermal-Hydraulic Statistical Core Design Methodology:

The thermal-hydraulic analysis for the RFA design will be performed with the statistical core design (SCD) analysis method described in DPC-NE-2005P-A, Rev. 1 (Ref. 18). The SCD analysis technique differs from the deterministic thermal hydraulic method in that the effects on the DNB limit of the uncertainties of key parameters are treated statistically. The SCD methodology involves selection of key DNBR parameters, determination of their associated uncertainties, and propagation of uncertainties and their impacts to determine a statistical DNBR limit that provides an assurance with 95% probability at 95% confidence level that DNB will not occur when the nominal values of the key parameters are input in the safety analysis. The SCD methodology described in DPC-NE-2005P-A is identical to the SCD methodology described in DPC-NE-2004P-A (Ref. 17) with the exception that the intermediate step of using a response surface model to evaluate the impact of uncertainties of key DNBR parameters about a statepoint is eliminated and replaced with the VIPRE-01 code to directly calculate the DNBR values for each set of reactor conditions. The staff has approved the SCD methodology with restrictions that: (1) its use of specific uncertainties and distributions will be justified on a plant-specific basis, and its selection of statepoints used for generating the statistical design limit will be justified to be appropriate; and (2) only the single, most conservative DNBR limit of two limits proposed by DPC for separate axial power distribution regions is acceptable. The licensee subsequently submitted Appendix C to DPC-NE-2005P-A containing the plant-specific data and limits with Mark-BW 17x17 type fuel using the BWU-Z CHF correlation, the VIPRE-01 thermal-hydraulic computer code, and DEC SCD methodology to support McGuire and Catawba reload

analyses. The staff previously found the BWU-Z correlation and the statistical DNBR design limit to be acceptable for the Mark-BW 17x17 fuel (Ref. 16).

Table 5.3 of DPC-NE-2009 provides McGuire/Catawba plant-specific data on the uncertainties and distributions, as well as the justifications, of the SCD parameters, the WRB-2M CHF correlation, and the VIPRE-01 code/model. Table 5-4 provides the McGuire/Catawba statepoint statistical results with the WRB-2M CHF correlation for the RFA core. The statistical design limit of DNBR of 1.30 for the RFA core is chosen to bound the all statistical DNBRs. The staff finds them acceptable for the RFA design.

2.3.4 Transition Cores:

The licensee stated that for operation with transitional mixed cores having both the Mark-BW fuel and RFA designs, the impact on the thermal hydraulic behavior of the geometric and hydraulic differences between these two fuel designs will be evaluated with an 8-channel core model. This is done by placing the RFA design in the channels representing the limiting hot assembly and the Mark-BW fuel assemblies in the eighth channel representing the rest of the assemblies. The transition core analysis models each fuel type in its respective location with correct geometry and the form loss coefficients. A transition core DNBR penalty is determined for the RFA design, and a conservative DNBR penalty is applied for all DNBR analyses for the RFA/Mark-BW transition cores.

To determine the transition mixed core DNBR penalty, the licensee has re-analyzed the most limiting full core statepoint used in the SCD analysis using the 8-channel transition core model. The result of the transition core DNBR showed an increase of statistical DNBR by less than 0.2%, and the DNBR value is still less than the statistical design limit of 1.30 for the full core of RFA design with the WRB-2M CHF correlation. Therefore, the staff concludes that the statistical design limit of 1.30 can be used for both transition and full core analyses.

2.4 UFSAR Accident Analyses

To support operation with transitional Mark-BW/RFA mixed core and full RFA cores, the UFSAR Chapter 15 transients and accidents analyses will be performed. The LOCA analyses will be performed by Westinghouse using approved LOCA evaluation models. Non-LOCA transients and accidents will be performed by the licensee using previously approved methodologies.

2.4.1 LOCA Analyses:

Westinghouse will perform the large- and small-break LOCA analyses for operation with transition and full cores of the RFA design using approved versions of the Westinghouse Appendix K LOCA evaluation models (EM). The small-break LOCA EM (Ref. 22, 23) includes the NOTRUMP code for the reactor coolant system transient depressurization and the LOCTA-IV code for the peak cladding temperature calculation. The large-break LOCA EM (Ref. 24) includes BASH and other interfacing codes such as SATAN-VI, REFILL, and LOCBART, for various phases. For operation of the transition Mark-BW/RFA cores, explicit analyses will be performed simulating the cross-flow effects due to any hydraulic mismatch between the Mark-BW and the RFA design. The licensee stated that if it determined a transition core penalty is required during the mixed core cycles it will be applied as an adder to the LOCA results for a full core of the RFA design. Since the Westinghouse LOCA EMs, both

the large- and small-break, are approved methodologies for PWR fuel designs, the staff concludes they are acceptable for performing LOCA analyses for the RFA design.

2.4.2 Non-LOCA Transient and Accident Analyses:

The safety analyses of McGuire and Catawba UFSAR Chapter 15 non-LOCA transients and accidents are performed with the RETRAN-02 system transient code and the VIPRE-01 core thermal-hydraulic code. The non-LOCA transient analysis methodologies are described in several topical reports. DPC-NE-3002-A, Rev. 1 (Ref. 25) describes the system transient analysis methodology including the RETRAN model nodalization, initial and boundary conditions, and input assumptions regarding control, protection, and safeguard system functions used in the safety analyses of all Chapter 15 non-LOCA transients and accidents, except for those involving significant asymmetric core power peaking. DPC-NE-3001-PA describes the methodologies for systematically confirming that reload key physics parameters are bounded by values assumed in the Chapter 15 safety analyses and for analyses of the control rod ejection, steam line break, and dropped rod events which involve significant asymmetric core power peaking and require evaluation of multi-dimensional simulations of the core responses. DPC-NE-2004P-A and DPC-NE-2005P-A describe the procedure used to apply the VIPRE-01 code for the reactor core thermal-hydraulic analyses and the SCD methodologies for the derivation of the statistical DNBR limit. DPC-NE-3000-PA (Ref. 26) documents the development of thermal-hydraulic simulation models using RETRAN-02 and VIPRE-01 codes, including detailed descriptions of the plant nodalizations, control system models, code models, and the selected code options for McGuire and Catawba plants.

These methodologies have been previously approved by NRC for the analyses of non-LOCA transients and accidents for McGuire and Catawba with the Mark-BW fuel design. A change of reactor core fuel from Mark-BW to the RFA design does not affect the conclusion of the analytical capabilities of RETRAN-02 and VIPRE-01, except for the need to change the inputs to reflect the RFA design in the safety analyses. The licensee performed a review of DPC-NE-3000-PA and identified the necessary changes in the existing transient analyses methods for performance of safety analyses in support of the RFA design. Minor changes are required to the volume and associated junction and heat conductor calculations in the reactor core region of the RETRAN primary system nodalization model to reflect the dimensional changes to the RFA design. Input changes to the VIPRE model are required in core thermal hydraulic analysis to reflect the RFA design geometry and form loss coefficients. In addition, as discussed in Sections 2.3.2 and 2.4.3, respectively, of this safety evaluation, the WRB-2M CHF correlation will be used for the DNBR calculation, and the SIMULATE-3K code will be used in place of ARROTTA for the nuclear portion of the control rod ejection accident analysis. The staff concludes the non-LOCA safety analysis methodologies are acceptable for the RFA design.

2.4.3 Rod Ejection Accident Analysis Using SIMULATE-3K:

The rod ejection accident (REA) analysis methodology described in DPC-NE-3001-PA includes the use of the three-dimensional space-time transient neutronics nodal code ARROTTA (Ref. 27) to perform the nuclear analysis portion of transient response; the VIPRE-01 code to model the core thermal response including peak fuel enthalpy, a core-wide DNBR evaluation, and transient core coolant expansion; and the RETRAN-02 code to simulate the reactor coolant system pressure response to the core power excursion. This methodology will continue to be used for the REA analysis except for the use of the SIMULATE-3K code (Ref. 28) to replace ARROTTA to perform the nuclear analysis of the response of the reactor core to the rapid reactivity insertion resulting from a control rod being ejected out of the core.

Section 6.6 of DPC-NE-2009 describes the REA analysis methodology using SIMULATE-3K, including a brief description of the code and models, code verification and benchmark, and the REA analysis application of SIMULATE-3K. SIMULATE-3K is a three-dimensional transient neutronic version of the NRC approved SIMULATE-3P computer code (Ref. 29) and uses the same neutron cross section library. It uses a fully-implicit time integration of the neutron flux, delayed neutron precursors, and heat conduction models. The average beta for the time-varying neutron flux is determined by performing a calculation of the adjoint flux solution. The code user has the option of running the code with a fixed time step or a variable time step depending on the sensitivity to changes in the neutronics. The SIMULATE-3K code has incorporated additional capability to model reactor trips at user-specified times in the transient or following a specified excore detector response, which allows the user to specify the response of individual detectors as required to initiate the trip, as well as the time delay prior to release of the control rods based on the excore detector response model. The code also permits the user input to control the velocity of the control rod movement, providing a different perspective for each velocity chosen.

The SIMULATE-3K code vendor, Studsvik of America, Inc., had performed the code verification and validation during its development to verify correctness of the coding and to validate the applicability of the code to specified analyses and ensure compatibility with existing methodology. The validation included benchmarks of the fuel conduction and thermal hydraulic models, the transient neutronics model, and the coupled performance of the transient neutronics and thermal-hydraulic models. The fuel and thermal hydraulic models were validated against the TRAC code, while the neutronic model was benchmarked against the solutions of the industry standard light water reactor problems generated by QUANDRY, NEM, and CUBBOX (Ref. 30, 31, 32). Benchmarking of the coupled performance of the thermal hydraulic and transient neutronics models was carried out against the results from a standard NEACRP [Nuclear Energy Agency Control Rod Problem] rod ejection problem to the PANTHER code (Ref. 33). Steady state comparison of S3K was performed against the NRC approved CASMO-3/SIMULATE-3P. In addition, DPC performed comparisons of the SIMULATE-3K and ARROTTA calculations for the reference REA analysis for the Oconee Nuclear Station showing very good agreement for core power versus time for the ejection occurring at the end-of-cycle from the maximum allowable power level with 3 and 4 RCPs operating and from both beginning-of-cycle and end-of-cycle at hot zero power and hot full power conditions. These SIMULATE-3K validation benchmarks were presented in DPC-NE-3005-P (Ref. 34), which the staff has reviewed for approval of using SIMULATE-3K for the analysis of the REA for the Oconee plants.

Section 6.6.1.3.3 of DPC-NE-2009 provides an additional benchmark of SIMULATE-3K by comparing the SIMULATE-3K and ARROTTA calculations for the reference REA analyses performed for beginning of life (BOC) and end of life (EOC) at hot-full-power (HFP) and hot-zero-power (HZP) conditions for McGuire and Catawba Nuclear Stations. The reference core used in the benchmark calculations was a hypothetical Catawba 1 Cycle 15 core, which represents typical fuel management strategies currently being developed for reload core designs at McGuire and Catawba. The comparison between the SIMULATE-3K and ARROTTA calculations of the core power level and nodal power distribution as functions of time during the REA transient demonstrate the acceptability of the physical and numerical models of SIMULATE-3K for application in the REA analyses for McGuire and Catawba Nuclear Station.

Section 6.6.2.2 of DPC-NE-2009 describes the use of the SIMULATE-3K code to perform license analysis of the design basis REA. The basic methodology as described in

DPC-NE-3001PA remains unchanged with the exception of minor differences between SIMULATE-3K and ARROTTA. The core power levels and nodal power distributions calculated by SIMULATE-3K are used by VIPRE to determine the fuel enthalpy, the percentage of fuel pins exceeding the DNB limit, and the coolant expansion rate. All inputs to VIPRE, once supplied by the NRC approved-code ARROTTA, are now supplied by SIMULATE-3K.

In the SIMULATE-3K nuclear analysis of an REA, a fuel assembly is typically geometrically modeled by several radial nodes. Axial nodalization and the number of nodes are chosen to accurately describe the axial characteristics of the fuel. For current fuel designs, a typical axial nodalization of 24 equal length fuel nodes in the axial direction is used. SIMULATE-3K explicitly calculates neutron leakage from the core by use of reflector nodes in the radial direction beyond the fuel region and in the axial direction above and below the fuel column stack. The fuel and reflector cross sections are developed in accordance with the methodology described in the approved topical report DPC-NE-1004A for SIMULATE-3P.

The SIMULATE-3K REA analysis is performed at four statepoints: BOC and EOC at HZP and HFP conditions for the determination of three-dimensional steady-state and transient power distributions, as well as individual pin powers. Conservative input parameters are used to ensure that the rod ejection analysis produces limiting results that bound future reload cycles. Sections 6.6.2.2.1 and 6.6.2.2.2 describe the methods to ensure conservatism in the analysis of transient response by increasing the fission cross sections in the ejected rod locations and in each assembly and by applying the "factors of conservatism" to the reactivity feedback for moderator and fuel temperatures, control rod worths for withdrawal and insertion, effective delayed neuron, and ejected rod worth, etc. In response to a staff question (No. 9, Ref. 14), the licensee provided a description of the method of determining the "factors of conservatism." The staff has reviewed the overall SIMULATE-3K methodology, and found it to be acceptable for application to the REA analyses for McGuire and Catawba.

2.4.4 Compliance with Safety Evaluation Conditions:

As discussed above, licensing analyses of reload cores with the RFA design use the methodologies described in various topical reports for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current McGuire/Catawba cores. These methodologies may have inherent limitations, or conditions or restrictions imposed by the associated NRC safety evaluations in their applications. The acceptability of the licensing analyses is subject to the application being within the limitations of the methodologies used and the conditions or restrictions imposed in the respective safety evaluations. In response to a staff question regarding the resolutions of these limitations, conditions, and restrictions in the RFA reload safety analyses, the licensee provided (Response to Question 11, Ref. 14) a list of restrictions imposed by NRC safety evaluations and the corresponding resolutions in the application of the licensee's methodologies used for the safety analyses of the non-LOCA transients and accidents. In addition, for the LOCA analyses to be performed by Westinghouse, the licensee provided a Westinghouse response (Ref. 35) regarding the safety evaluation restrictions and corresponding compliance for the 1985 SBLOCA Evaluation Model with NOTRUMP and the 1981 Evaluation Model with BASH. The resolutions or compliance with the conditions or restrictions provided in these responses provide guidance for the licensee referencing DPC-NE-2009 in the RFA reload licensing analyses. The staff concludes that the safety evaluation conditions have been properly addressed.

2.5 Fuel Assembly Repair and Reconstitution

Section 7.0 of DPC-NE-2009 describes the evaluation of the reconstitution or repair of fuel assemblies having failed fuel rods during refueling outages in an effort to achieve the zero fuel defect goal during cycle operation. The primary replacement candidate for use in reconstitution of failed fuel rods is a fuel rod that contains pellets of natural uranium dioxide, but solid filler rods made of stainless steel, zircaloy, or ZIRLO would be used if local grid structural damage exists. The reconstitution of the RFA assembly with filler rods will be analyzed with NRC-approved methodology and guidelines described in DPC-NE-2007P-A (Ref. 36), along with other licensed codes and correlations, to ensure acceptable nuclear, mechanical, and thermal-hydraulic performance of reconstituted fuel assemblies.

For a reload core using reconstituted Westinghouse fuel, Westinghouse has reviewed the effects of the reconstituted fuel with the criteria specified in Standard Review Plan 4.2 and determined that the only fuel assembly mechanical criteria impacted by reconstitution are fuel assembly holddown force and assembly structural response to seismic/LOCA loads. Westinghouse has evaluated these effects on the LOCA analyses using the approved methodology WCAP-13060-P-A (Ref. 37), and concluded that the reconstituted fuel assembly designs are acceptable for both normal and faulted condition operations.

2.6 Technical Specifications Changes

The licensee's July 22 and October 22, 1998, letters proposed changes to the Technical Specifications with the technical justifications for these changes described in Chapter 8 of DPC-NE-2009. The licensee's January 28, May 6 and June 24, 1999, letters provided revisions to some of the proposed changes. The staff's evaluation follows.

2.6.1 Proposed Change to TS Figure 2.1.1-1:

The licensee proposed to modify Figure 2.1.1-1, "Reactor Core Safety Limits - Four Loops in Operation," by (1) deleting the 2455 psia safety limit line, which is the current upper bound pressure allowed for power operation; (2) combining separate Unit 1 and Unit 2 figures into only one figure; and (3) revising the other safety limit lines (see following paragraph). The resulting Figure 2.1.1-1 was submitted by a letter, M. Tuckman to NRC, dated June 24, 1999 (Ref. 39).

The 2455 psia bounding pressure is based on the pressure range of the CHF correlation used in DNBR analyses of the Mark-BW fuel. Since the upper range of applicability of the WRB-2M CHF correlation for the RFA design is 2425 psia, the 2455 psia safety limit line is deleted, and the remaining safety limit lines with 2400 psia as the upper bound safety limit line are within the range of the CHF correlations for the Mark-BW and RFA fuel designs. As described in its response to a staff's question (No. 12, Ref. 14), the licensee has performed an evaluation to ensure the remaining safety limit lines of Figure 2.1.1-1, which were based on the CHF correlation for the Mark-BW fuel design and the hot leg boiling limit, bound the safety limit for the DNBR limit of the WRB-2M correlation for the RFA design. Both the full RFA core and the transition RFA/Mark-BW cores were evaluated to ensure that the established limits were conservative. The DNBR values were greater than the design DNBR limit for all the cases in both evaluation. Therefore, the safety limit lines in Figure 2.1.1-1, with the deletion of the 2455 psia safety limit line, are acceptable.

2.6.2 Proposed Changes to Surveillance Requirements 3.2.1.2, 3.2.1.3, and 3.2.2.2:

TS Surveillance Requirements (SRs) 3.2.1.2, 3.2.1.3, and 3.2.2.2, respectively, require the heat flux hot channel factor F_q (x,y,z) and the enthalpy rise hot channel factor $F_{\Delta h}$ (x,y) to be measured periodically (once within 12 hours after achieving equilibrium conditions after a power change exceeding 10% rated thermal power and every 31 effective full power days thereafter) using the incore detector system to ensure the values of the total peaking factor and the enthalpy rise factor assumed in the accident analyses and the reactor protection system limit are not violated. To avoid the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, these SRs currently specify a penalty factor of 1.02 for the heat flux and enthalpy rise hot channel factors if the margin to the F_q (x,y,z) or $F_{\Delta h}$ (x,y) has decreased since the previous surveillance. The 2% margin-decrease penalty was based on the current reload cores.

For the reactor core containing the RFA fuel design with integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. The licensee proposed to remove the 2% penalty value from these SRs and replace them with tables of penalty values as functions of burnup in the Core Operating Limits Report (COLR) to facilitate cycle-specific updates. Tables 8-1 and 8-2, respectively, provide typical values for the burnup-dependent margin-decrease penalty factors for the heat flux and enthalpy rise hot channel factors. The actual values for the transitional core can not be provided until the final design for the core is complete. In response to a staff question (No. 13, Ref. 14), the licensee provided the methodology for calculating the burnup-dependent penalty factors. In addition, Technical Specification 5.6.5 will reference topical report DPC-NE-2009, which includes this response to the staff's question for the approved methodology used to calculate these penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin-decrease penalty factors in the COLR acceptable.

2.6.3 Proposed Change to TS 4.2.1:

TS 4.2.1, "Fuel Assemblies," which specifies the design features for fuel assemblies, will be revised to add ZIRLO cladding to the fuel assembly description.

2.6.4 Proposed Changes to Section 5.6.5b:

By a letter dated May 6, 1999 (Ref. 38), the licensee expanded the original amendment request by proposing more changes in Section 5.6.5. The section lists all the topical reports previously approved by the staff. Thus these proposed changes are administrative or editorial. The staff finds them all acceptable as follows:

WCAP-10216P-A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification" -- This is deleted since it had been previously replaced by Item 5 (re-numbered Item 4), DPC-NE-2011P-A.

BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants" -- The dates of the various staff safety evaluations have been updated.

DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology" -- The Revision number has been changed from "2" to "3". The staff's safety evaluation date is also updated.

DPC-NE-3000P-A, "Thermal-Hydraulic Transient Analysis Methodology" -- The Revision number is changed from "1" to "2". The staff's safety evaluation date is also updated.

DPC-NE-2001P-A "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel" -- This is deleted, and is replaced by DPC-NE-2008P-A.

BAW-10183P-A, "Fuel Rod Gas Pressure Criterion" -- This is deleted. DPC-NE-2008P-A references this report, and therefore there is no need for an individual listing.

WCAP-10054P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code" -- This report is applicable to the Westinghouse fuel.

DPC-NE-2009P-A, "Westinghouse Fuel Transition Report" -- This report has been evaluated in the above sections of this safety evaluation and found acceptable.

2.6.5 Proposed Changes to the Technical Specifications Bases Document:

The TS Bases is a licensee-controlled document and is not part of the Technical Specifications (10 CFR 50.36(a)). However, the staff reviewed the licensee's proposed changes as supplemental information for the TS changes evaluated above. The Bases sections for SR 3.2.1.2, 3.2.1.3 and 3.2.2.2 will be revised to reflect the corresponding TS changes. The staff finds the proposed changes to the Bases acceptable.

3.0 REVIEW SUMMARY OF TOPICAL REPORT

The staff has reviewed the licensee's Topical Report DPC-NE-2009P and found it acceptable for referencing for analysis of reloads with Westinghouse RFA design. The topical report references many topical reports, which provide methodologies for various aspects of the RFA reload licensing analyses. Acceptability of DPC-NE-2009P remains subject to the limitations set forth in the SERs on these topical reports.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, North Carolina State official Mr. Johnny James was notified of the proposed issuance of the amendments. The official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and change surveillance requirements. The NRC staff has determined that the amendments involve 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 staff has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 69338, dated December 16, 1998; 64 FR 35202, dated June 30, 1999, and 64 FR 43771, dated August 11, 1999) . The licensee's September 15, 1999, letter (Ref. 44) provided clarifying information that did not change the scope of the application and the initial proposed no significant hazards consideration determination. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in

10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Attachment: Technical Evaluation Report

Principal Contributor: Yi-Hsiung Hsii
Anthony Attard
Shih-Liang Wu
Peter Tam

Date: September 22, 1999

7.0 REFERENCES

1. Letter, M. S. Tuckman (Duke Energy Corporation) to NRC, "Duke Energy Corporation, McGuire Nuclear Station, Units 1 and 2, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Revisions to Improved Technical Specifications for Implementation of Westinghouse Fuel as Described in Topical Report Described in Topical Report DPC-NE-2009/DPC-NE-2009P, Duke Power Company Westinghouse Fuel Transition Report," July 22, 1998.

2. Letter, M. S. Tuckman (Duke Energy Corporation) to NRC, "Duke Energy Corporation, McGuire Nuclear Station, Units 1 and 2, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Supplement to License Amendment Request for Revisions to Improved Technical Specifications for Implementation of Westinghouse Fuel as Described in Topical Report Described in Topical Report DPC-NE-2009/DPC-NE-2009P, Duke Power Company Westinghouse Fuel Transition Report," October 22, 1998.

3. Duke Power Company, DPC-NE-2009/DPC-NE-2009P, "Duke Power Company Westinghouse Fuel Transition Report," July 1998.

4. "Mark-BW Mechanical Design Report," BAW-10172P-A, December 1989.

5. Davison, S. L., T. L. Ryan, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995, WCAP-12610-P-A.

6. Letter from N. J. Liparulo (Westinghouse) to J. E. Lyons (USNRC), "Transmittal of Response to NRC Request for Information on Wolf Creek Fuel design Modifications," June 30, 1997, NSD-NRC-97-5189.

7. Davison, S. L., "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-P-A, October 1994.

8. "Duke Power Company Fuel Rod Mechanical Reload Analysis Methodology Using TACO3," DPC-NE-2008P-A, April 1995.

9. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

10. DPC-NE-1004A, Rev. 1, "Design Methodology Using CASMO-3/SIMULATE-3P," April 1996.

11. DPC-NE-2011P-A, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March 1990.

12. DPC-NF-2010A, "Duke Power Company McGuire Station, Catawba Nuclear Station Nuclear Physics Methodology," June 1985.

13. DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," November 1991.

14. Letter, M. S. Tuckman (Duke Energy Corporation) to NRC, "Duke Energy Corporation, McGuire Nuclear Station, Units 1 and 2, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Response to NRC Requests for Additional Information on License Amendment Requests for McGuire and Catawba Nuclear Stations," January 28, 1999.

15. EPRI NP-2511-CCM-A, "VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores," August 1989.

16. Letter from H. N. Berkow (USNRC) to M. S. Tuckman (DPC), "Safety Evaluation on the Use of the BWU-Z Critical Heat Flux Correlation for McGuire Nuclear Station, Units 1 and 2; and Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M95267, M95268, and M95333, M95334)," November 7, 1996.

17. DPC-NE-2004P-A, Rev. 1, "McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," February, 1997.

18. DPC-NE-2005P-A, Rev. 1, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," November, 1996.

19. WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," Westinghouse Energy Systems, April 1999.

20. Letter from T. H. Essig (NRC) to H. Sepp (Westinghouse Electric Corporation), "Acceptance for Referencing of Licensing Topical Report WCAP-15025-P, 'Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids'," December 1, 1998.

21. Letter from T. H. Essig (NRC) to H. Sepp (Westinghouse Electric Corporation), "Acceptance for Referencing of Licensing Topical Report WCAP-14565-P, 'VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal/Hydraulic Safety Analysis (TAC No. M98666)'," January 19, 1999.

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28. "SIMULATE-3 Kinetics Theory and Model Description," SOA-96/26, Studsvik of America, April 1996.

29. "SIMULATE-3: Advanced Three-Dimensional Two-Group Reactor Analysis Code," Studsvik/SOA-92/01, Studsvik of America, April 1992.

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36. DPC-NE-2007P-A, "Duke Power Company Fuel Reconstitution Analysis Methodology," October 1995.

37. WCAP-13060-P-A, "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology," July 1993.

38. Letter, M. S. Tuckman (Duke Energy Corporation) to NRC, "Duke Energy Corporation, McGuire Nuclear Station, Units 1 and 2, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Supplement to License Amendment Request for Revisions to Improved Technical Specifications for Implementation of Westinghouse Fuel as Described in Topical Report Described in Topical Report DPC-NE-2009/DPC-NE-2009P, Duke Power Company Westinghouse Fuel Transition Report," May 6, 1999.

39. Letter, M. S. Tuckman to NRC, proposing amendments to McGuire and Catawba Technical Specifications regarding reactor coolant systems flow rate, June 24, 1999.

40. Letter, M. S. Tuckman to NRC, providing revised pages for topic report DPC-NE-2009, August 17, 1999.

41. Letter, F. Rinaldi (NRC) to H. B. Barron (McGuire Nuclear Station), finding use of high burnup methodology described in topical report BAW-10186P-A acceptable for reload licensing application at McGuire, March 3, 1999.

42. Letter, P. S. Tam (NRC) to G. R. Peterson (Catawba Nuclear Station), finding use of high burnup methodology described in topical report BAW-10186P-A acceptable for reload licensing application at Catawba, March 3, 1999.

43. Letter, R. Martin (NRC) to W. R. Mccollum (Catawba), transmitting operating license amendments 134 (Unit 1) and 128 (Unit 2), August 31, 1995.

44. Letter, M. S. Tuckman to NRC, proposing a license condition limiting fuel burnup up to 60 Gwd/mtU, September 15, 1999.

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

September 22, 1999

Mr. H. B. Barron
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
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SUBJECT: MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA2411 AND MA2412)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 188 to Facility Operating License NPF-9 and Amendment No. 169 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated July 22, 1998, and supplemented by letters dated October 22, 1998, and January 28, May 6, June 24, August 17 and September 15, 1999.

The amendments revise various sections of the Technical Specifications (Appendix A of the McGuire operating licenses) to permit use of Westinghouse's Robust Fuel Assemblies for future core reloads. We will publish a Notice of Issuance in the Commission's biweekly *Federal Register* notice.

Concurrent with issuance of these amendments we have also approved topical report DPC-NE-2009, "Duke Power Company Westinghouse Fuel Transition Report." The Safety Evaluation (enclosed) provides details of our review of DPC-NE-2009P in support of the subject amendments. In accordance with procedures established in NUREG-0390, we request Duke Energy Corporation to publish an accepted version of DPC-NE-2009, proprietary and nonproprietary, within 3 months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed Safety Evaluation after the title page. The accepted versions shall include an "A" (designating accepted) following the report identification symbol. Please include our request for additional information and Duke's response as an appendix to the report.

Sincerely,

Frank Rinaldi

Frank Rinaldi, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 188 to NPF-9
2. Amendment No. 169 to NPF-17
3. Safety Evaluation

cc w/encl: See next page

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CORPORATION, ET AL.

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION AND BACKGROUND

By letter dated July 22, 1998 (Ref. 1), and supplemented by a letter of October 22, 1998 (Ref 2), Duke Energy Corporation* (DEC, the licensee), the licensee for operation of McGuire and Catawba Nuclear Stations, proposed changes to the Technical Specifications (TS) of these plants in anticipation of a reactor core reload design using Westinghouse fuel. Accompanying the July 22, 1998, letter is a topical report DPC-NE-2009, "Duke Power Company" Westinghouse Fuel Transition Report," (Ref. 3) for NRC review and approval. When approved, this topical report will be listed in Section 5.6.5 of the Catawba and McGuire TSs as an approved methodology for the determination of the core operating limits.

The reactors of McGuire and Catawba Nuclear Stations are currently using Framatome Cogema Fuels (FCF) Mark-BW fuel assemblies (Ref. 4). The proposed amendment to the TSs would permit transition to the 17x17 Westinghouse Robust Fuel Assembly (RFA) design.

The RFA design is based on the VANTAGE+ fuel assembly design, which has been approved by NRC as described in WCAP-12610-P-A (Ref. 5). The RFA design to be used at McGuire and Catawba, as described in Section 2.0 of DPC-NE-2009, will incorporate the following features in addition to the VANTAGE+ design features:

- increased guide thimble and instrumentation tube outside diameter
- modified low pressure drop structural mid-grids
- modified intermediate flow mixing grids
- pre-oxide coating on the bottom of the fuel rods
- protective bottom grid with longer fuel rod end-plugs
- fuel rods positioned on the bottom nozzle
- a quick release top nozzle

The first three design features listed above were licensed via the Wolf Creek Fuel design (Ref. 6) using the NRC-approved Westinghouse Fuel Criteria Evaluation Process (Ref. 7). The next three features are included to help mitigate debris failures and incomplete rod insertion.

* The official name of the licensee is Duke Energy Corporation, as is stated in the Catawba and McGuire operating licenses. "Duke Power Company" is a component of Duke Energy Corporation; however, for historical reasons, the licensee used "Duke Energy Corporation" and "Duke Power Company" interchangeably. This safety evaluation follows the licensee's practice.

The licensee states that these three features will be evaluated using the 10 CFR 50.59 process. The quick release top nozzle design is similar to the Reconstitutable Top Nozzle design with modifications for easier removal. This design will be licensed by Westinghouse using the fuel criteria evaluation process.

2.0 EVALUATION

Topical report DPC-NE-2009 provides general information about the RFA design and describes methodologies to be used for reload design analyses to support the licensing basis for the use of the RFA design in the McGuire and Catawba reload cores. These methodologies include DEC's fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The report does not provide the analyses of the core design, thermal-hydraulics and transients and accidents associated with the RFA design. Therefore, this safety evaluation will only address the acceptability of the methodologies described in DPC-NE-2009 for referencing in the analyses for operations with the reactor cores having a mix of Mark-BW and RFA fuel design or a full core of RFA design.

2.1 Fuel Rod Analysis Methodology

During transition periods, the reactor cores in the McGuire and Catawba plants will have both the FCF Mark-BW fuel and the Westinghouse RFA fuel. Section 4 of DPC-NE-2009 describes the fuel rod mechanical reload analysis methodology for the RFA design. While the fuel rod mechanical analyses for Mark-BW fuel will continue to be performed using the licensee's methodology described in DPC-NE-2008P-A (Ref. 8), the Westinghouse RFA fuel thermal-mechanical analyses will be performed using the NRC-approved Westinghouse fuel performance code, PAD 3.4 Code (Ref. 9). The fuel rod design bases for the RFA design are identical to those described in WCAP-12610-P-A (Ref. 5) for the VANTAGE+ fuel.

The staff's review of fuel rod analysis methodology was performed with technical assistance provided by Pacific Northwest National Laboratory (PNNL). PNNL's review findings and conclusion, with which the staff concurs, are described in its technical evaluation report (attached to this safety evaluation). Thus, the staff has found that the DEC design limits and thermal-mechanical analysis methodologies discussed in Section 4.0 of DPC-NE-2009 are acceptable for application by DEC to the RFA fuel design up to the currently approved (Ref. 41, 42, 43) rod average burnup limit of 62 GWd/mtU. The staff has previously performed an environmental assessment for fuel burnup up to 60 GWd/mtU (53 FR 30355, August 11, 1988). Consequently, due to this limitation from the environmental perspective, the licensee proposed (Ref. 44) a license condition. The staff will impose the license condition as proposed by the licensee to read: "The maximum rod average burnup for any rod shall be limited to 60,000 MWd/mtU [60 GWd/mtU] until the completion of an NRC environmental assessment supporting an increased limit."

2.2 Reload Core Design Methodology

For the RFA design, the core model, core operational imbalance limits, and key core physics parameters used to confirm the acceptability of Updated Final Safety Analysis Report (UFSAR) Chapter 15 safety analyses of transients and accidents will be developed with the methodologies described in DPC-NE-1004-A (Ref. 10), DPC-NE-2011P-A (Ref. 11), DPC-NF-2010A (Ref. 12), and DPC-NE-3001-PA (Ref 13). DPC-NE-2011P-A describes the nuclear design methodology for core operating limits of McGuire and Catawba plants. DPC-NF-2010A describes McGuire and Catawba nuclear physics methodology using

two-dimensional PDQ07 and 3-D EPRI-NODE-P models as reactor simulators. DPC-NE-1004A describes an alternative methodology for calculating nuclear physics data using the CASMO-3 fuel assembly depletion code and the SIMULATE-3P 3-D core simulator code for steady-state core physics calculations, substituting for CASMO-2, PDQ07 and EPRI-NODE-P used in DPC-NE-2010A. DPC-NE-3001-PA describes the methodologies, which expand on the reload design methods of DPC-NF-2010A, for systematically verifying that key physics parameters calculated for a reload core, such as control rod worth, reactivity coefficients, and kinetics parameters, are bounded by values assumed in the Chapter 15 licensing analyses. These topical reports have been approved for performing reload analyses for the B&W 177-assembly and/or Westinghouse 193-assembly cores, subject to the conditions specified in the staff's safety evaluations. Because of the similarity between the RFA design and the Mark-BW fuel design with respect to the dimensional characteristics of the fuel pellet, fuel rod and cladding, as well as nuclear characteristics, as shown in Table 2-1 of DPC-NE-2009, the staff concludes that these approved methodologies and core models currently employed in reload design analyses for McGuire and Catawba can be used to perform transition and full-core analyses of the RFA design.

Section 3.2 of DPC-NE-2009 states that conceptual transition core designs using the RFA design have been evaluated and results show that current reload limits remain bounding with respect to key physics parameters. As described in DPC's response to a staff question (Question 1, Ref. 14, January 28, 1999), the conceptual RFA transition core designs were evaluated for the effects of partial and full cores using NRC-approved codes and methods to determine the acceptability of the current licensing bases transient analyses. Key safety parameters, such as Doppler temperature coefficients, moderator temperature coefficients, control bank worth, individual rod worths, boron concentrations, differential boron worths and kinetics data, were calculated for the conceptual core designs and compared against reference values assumed in the UFSAR Chapter 15 accident analyses. The evaluation demonstrated the expected neutronic similarities between reactor cores loaded with RFA fuel and with Mark-BW fuel and the acceptability of key safety parameters assumed in the Chapter 15 accident analyses. Key physics parameters are calculated for each reload core and each new core design. If a key physics parameter is not bounded by the reference value in the UFSAR accident analyses, the affected accidents will be re-analyzed using the new key physics parameter, or the core will be re-designed to produce an acceptable result. The staff agrees that this is an acceptable approach.

The safety evaluation for DPC-NE-1004-A requires additional code validation to ensure that the methodology and nuclear uncertainties remain appropriate for application of CASMO-3 and SIMULATE-3P to fuel designs that differ significantly from those included in the topical report data base. Though the RFA design is not expected to change the magnitude of the nuclear uncertainty factors in DPC-NE-1004, the use of zirconium diboride integral fuel burnable absorber (IFBA) in the RFA is a design change from the burnable absorber types modeled in DEC's current benchmarking data base. DEC has re-evaluated and confirmed the nuclear uncertainties in DPC-NE-1004 to be bounding. This is done by explicitly modeling Sequoyah Unit 2, Cycles 5, 6, and 7, and by performing statistical analysis of the nuclear uncertainty factors. These cores were chosen because they are very similar to McGuire and Catawba and contained both IFBA and wet annular burnable absorber (WABA) fuel. The results, listed in Table 3-1 of DPC-NE-2009, showed that the current licensed nuclear uncertainty factors for the $F_{\Delta H}$, F_Z, and F_Q bound those for the Westinghouse fuel with IFBA and/or WABA burnable absorbers. Boron concentrations, rod worth, and isothermal temperature coefficients were also predicted and found to agree well with the measured data. In response to a staff question (Question 2, Ref. 14) regarding the applicability of the analysis of the Sequoyah core to the

McGuire and Catawba cores, DEC provided comparisons of the analysis results and the measured data of the Sequoyah cores and a list of the differences between the Westinghouse Vantage-5H fuel design used in Sequoyah and the RFA fuel design. The differences are primarily mechanical and do not impact the nuclear performance of the fuel assembly. Design features that do impact the neutronics (i.e., mid-span mixing grids) are specifically accounted for in the nuclear models. Therefore, the results and conclusions reached based on the analysis of Sequoyah core designs are applicable to the RFA fuel design. In addition, the licensee performed a 10 CFR 50.59 evaluation for unreviewed safety question (USQ). Results are as described in response to Question 2c of Ref.14, which demonstrates that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties are applicable to the RFA design. Therefore, DPC-NE-1004A nuclear physics calculation methodology is applicable to the RFA design.

In all nuclear design analyses, both the RFA and the Mark-BW fuel are explicitly modeled in the transition cores. The mixed core model for nuclear design analyses and the use of fuel-specific limits, described in response to a staff's question (Question 3, Ref. 14), are based on the same methodology that is used to set up a nuclear model for a reactor core containing a single fuel type. When establishing operating and reactor protection system limits (i.e., LOCA linear heat rate limit, departure from nucleate boiling (DNB), central fuel melt, transient strain), the fuel-specific limits or a conservative overlay of the limits are used. The staff concludes that the nuclear design analyses for the transition cores are acceptable.

2.3 Thermal-Hydraulic Analysis

Section 5 of DPC-NE-2009 describes the thermal-hydraulic analysis methodologies to be used for the RFA design. The thermal-hydraulic analyses for the existing Mark-BW fuel design are performed with NRC approved methodology using the VIPRE-01 core thermal-hydraulic code (Ref. 15), the BWU-Z critical heat flux (CHF) correlation (Ref. 16), and the thermal-hydraulic statistical core design methodology described in DPC-NE-2004P-A (Ref. 17) and DPC-NE-2005P-A (Ref. 18). As discussed in the ensuing sections of this report, these same methodologies will be used for the analyses of the RFA design with the exception that (1) the WRB-2M CHF correlation (Ref. 19) will be used in place of the BWU-Z correlation, and (2) the EPRI bulk void fraction model will be used in place of the Zuber-Findlay model.

2.3.1 VIPRE-01 Core Thermal Hydraulic Code:

The core thermal hydraulic analysis methodology using the VIPRE-01 code for McGuire and Catawba licensing calculations is described in DPC-NE-2004P-A. The VIPRE-01 models, which have been approved for the Mark-BW fuel, are also applicable to the RFA design with appropriate input of fuel geometry and form loss coefficients consistent with the RFA design. The reference pin power distribution based on an enthalpy rise factor, $F_{\Delta H}{}^{N}$, of 1.60 peak pin from DPC-NE-2004P-A will continue to be used to analyze the RFA design.

VIPRE-01 contains various void-quality relation models for two-phase flow calculation, in addition to the homogeneous equilibrium model. Either the Levy model or the EPRI model can be chosen for subcooled boiling, and the Zuber-Findlay or EPRI void models for bulk boiling. The combination of Levy subcooled boiling correlation and Zuber-Findlay bulk boiling model gives reasonable results for void fraction. This combination is currently used for McGuire/Catawba cores with the Mark-BW fuel. However, the Zuber-Findlay correlation is applicable only to qualities below approximately 0.7, and there is a discontinuity at a quality of 1.0. The licensee proposes to replace this combination with the combination of EPRI

subcooled and bulk void models. The use of the EPRI bulk void model, which is essentially the same as the Zuber-Findlay model except for the equation used to calculate the drift velocity, is to eliminate a discontinuity at qualities about 1.0. Also, the use of the EPRI subcooled void model is for overall model compatibility to have the EPRI models cover the full range of void fraction required for performing departure-from-nucleate-boiling calculations. To evaluate the impact of these model changes, the licensee performed an analysis of 51 RFA CHF test data points using both Levy/Zuber-Findlay and EPRI models in VIPRE-01. The results show a negligible 0.1 percent difference in the minimum departure-from-nucleate-boiling ratios (DNBRs). Therefore, the staff finds that the use of the EPRI subcooled and bulk void correlations for the analysis of the RFA design is acceptable. The acceptability of this revision remains subject to the limitations set forth in the safety evaluation on VIPRE-01 (EPRI NP-2511-CCM-A), DPC-NE-2004P-A and attendant revisions.

2.3.2 Critical Heat Flux (CHF) Correlation:

The licensee stated that the WRB-2M CHF correlation, described in the Westinghouse topical report WCAP-15025-P-A (Ref. 19), will be used for the RFA design. The WRB-2M correlation was developed by Westinghouse for application to new fuel designs such as the Modified Vantage 5H and Modified Vantage 5H/IFM. The WRB-2M correlation was programmed into the Westinghouse thermal hydraulic code THINC-IV or the VIPRE-01 thermal-hydraulic code for the calculation of the local conditions within the rod bundles. The staff has reviewed and approved the WRB-2M correlation with both THINC-IV and VIPRE-01 codes as described in References 20 and 21. The WRB-2M correlation is also applicable to the RFA design because of its similarity to the Vantage 5H fuel design. The staff concludes DEC's use of the WRB-2M along with VIPRE-01 in the DNBR calculations for the RFA design to be acceptable within the ranges of applicability of important thermal hydraulic parameters specified in the staff's safety evaluation on WCAP-15025-P-A (Ref. 20).

2.3.3 Thermal-Hydraulic Statistical Core Design Methodology:

The thermal-hydraulic analysis for the RFA design will be performed with the statistical core design (SCD) analysis method described in DPC-NE-2005P-A, Rev. 1 (Ref. 18). The SCD analysis technique differs from the deterministic thermal hydraulic method in that the effects on the DNB limit of the uncertainties of key parameters are treated statistically. The SCD methodology involves selection of key DNBR parameters, determination of their associated uncertainties, and propagation of uncertainties and their impacts to determine a statistical DNBR limit that provides an assurance with 95% probability at 95% confidence level that DNB will not occur when the nominal values of the key parameters are input in the safety analysis. The SCD methodology described in DPC-NE-2005P-A is identical to the SCD methodology described in DPC-NE-2004P-A (Ref. 17) with the exception that the intermediate step of using a response surface model to evaluate the impact of uncertainties of key DNBR parameters about a statepoint is eliminated and replaced with the VIPRE-01 code to directly calculate the DNBR values for each set of reactor conditions. The staff has approved the SCD methodology with restrictions that: (1) its use of specific uncertainties and distributions will be justified on a plant-specific basis, and its selection of statepoints used for generating the statistical design limit will be justified to be appropriate; and (2) only the single, most conservative DNBR limit of two limits proposed by DPC for separate axial power distribution regions is acceptable. The licensee subsequently submitted Appendix C to DPC-NE-2005P-A containing the plant-specific data and limits with Mark-BW 17x17 type fuel using the BWU-Z CHF correlation, the VIPRE-01 thermal-hydraulic computer code, and DEC SCD methodology to support McGuire and Catawba reload

analyses. The staff previously found the BWU-Z correlation and the statistical DNBR design limit to be acceptable for the Mark-BW 17x17 fuel (Ref. 16).

Table 5.3 of DPC-NE-2009 provides McGuire/Catawba plant-specific data on the uncertainties and distributions, as well as the justifications, of the SCD parameters, the WRB-2M CHF correlation, and the VIPRE-01 code/model. Table 5-4 provides the McGuire/Catawba statepoint statistical results with the WRB-2M CHF correlation for the RFA core. The statistical design limit of DNBR of 1.30 for the RFA core is chosen to bound the all statistical DNBRs. The staff finds them acceptable for the RFA design.

2.3.4 Transition Cores:

The licensee stated that for operation with transitional mixed cores having both the Mark-BW fuel and RFA designs, the impact on the thermal hydraulic behavior of the geometric and hydraulic differences between these two fuel designs will be evaluated with an 8-channel core model. This is done by placing the RFA design in the channels representing the limiting hot assembly and the Mark-BW fuel assemblies in the eighth channel representing the rest of the assemblies. The transition core analysis models each fuel type in its respective location with correct geometry and the form loss coefficients. A transition core DNBR penalty is determined for the RFA design, and a conservative DNBR penalty is applied for all DNBR analyses for the RFA/Mark-BW transition cores.

To determine the transition mixed core DNBR penalty, the licensee has re-analyzed the most limiting full core statepoint used in the SCD analysis using the 8-channel transition core model. The result of the transition core DNBR showed an increase of statistical DNBR by less than 0.2%, and the DNBR value is still less than the statistical design limit of 1.30 for the full core of RFA design with the WRB-2M CHF correlation. Therefore, the staff concludes that the statistical design limit of 1.30 can be used for both transition and full core analyses.

2.4 UFSAR Accident Analyses

To support operation with transitional Mark-BW/RFA mixed core and full RFA cores, the UFSAR Chapter 15 transients and accidents analyses will be performed. The LOCA analyses will be performed by Westinghouse using approved LOCA evaluation models. Non-LOCA transients and accidents will be performed by the licensee using previously approved methodologies.

2.4.1 LOCA Analyses:

Westinghouse will perform the large- and small-break LOCA analyses for operation with transition and full cores of the RFA design using approved versions of the Westinghouse Appendix K LOCA evaluation models (EM). The small-break LOCA EM (Ref. 22, 23) includes the NOTRUMP code for the reactor coolant system transient depressurization and the LOCTA-IV code for the peak cladding temperature calculation. The large-break LOCA EM (Ref. 24) includes BASH and other interfacing codes such as SATAN-VI, REFILL, and LOCBART, for various phases. For operation of the transition Mark-BW/RFA cores, explicit analyses will be performed simulating the cross-flow effects due to any hydraulic mismatch between the Mark-BW and the RFA design. The licensee stated that if it determined a transition core penalty is required during the mixed core cycles it will be applied as an adder to the LOCA results for a full core of the RFA design. Since the Westinghouse LOCA EMs, both

the large- and small-break, are approved methodologies for PWR fuel designs, the staff concludes they are acceptable for performing LOCA analyses for the RFA design.

2.4.2 Non-LOCA Transient and Accident Analyses:

The safety analyses of McGuire and Catawba UFSAR Chapter 15 non-LOCA transients and accidents are performed with the RETRAN-02 system transient code and the VIPRE-01 core thermal-hydraulic code. The non-LOCA transient analysis methodologies are described in several topical reports. DPC-NE-3002-A, Rev. 1 (Ref. 25) describes the system transient analysis methodology including the RETRAN model nodalization, initial and boundary conditions, and input assumptions regarding control, protection, and safeguard system functions used in the safety analyses of all Chapter 15 non-LOCA transients and accidents, except for those involving significant asymmetric core power peaking. DPC-NE-3001-PA describes the methodologies for systematically confirming that reload key physics parameters are bounded by values assumed in the Chapter 15 safety analyses and for analyses of the control rod ejection, steam line break, and dropped rod events which involve significant asymmetric core power peaking and require evaluation of multi-dimensional simulations of the core responses. DPC-NE-2004P-A and DPC-NE-2005P-A describe the procedure used to apply the VIPRE-01 code for the reactor core thermal-hydraulic analyses and the SCD methodologies for the derivation of the statistical DNBR limit. DPC-NE-3000-PA (Ref. 26) documents the development of thermal-hydraulic simulation models using RETRAN-02 and VIPRE-01 codes, including detailed descriptions of the plant nodalizations, control system models, code models, and the selected code options for McGuire and Catawba plants.

These methodologies have been previously approved by NRC for the analyses of non-LOCA transients and accidents for McGuire and Catawba with the Mark-BW fuel design. A change of reactor core fuel from Mark-BW to the RFA design does not affect the conclusion of the analytical capabilities of RETRAN-02 and VIPRE-01, except for the need to change the inputs to reflect the RFA design in the safety analyses. The licensees performed a review of DPC-NE-3000-PA and identified the necessary changes in the existing transient analyses methods for performance of safety analyses in support of the RFA design. Minor changes are required to the volume and associated junction and heat conductor calculations in the reactor core region of the RETRAN primary system nodalization model to reflect the dimensional changes to the RFA design. Input changes to the VIPRE model are required in core thermal hydraulic analysis to reflect the RFA design geometry and form loss coefficients. In addition, as discussed in Sections 2.3.2 and 2.4.3, respectively, of this safety evaluation, the WRB-2M CHF correlation will be used for the DNBR calculation, and the SIMULATE-3K code will be used in place of ARROTTA for the nuclear portion of the control rod ejection accident analysis. The staff concludes the non-LOCA safety analysis methodologies are acceptable for the RFA design.

2.4.3 Rod Ejection Accident Analysis Using SIMULATE-3K:

The rod ejection accident (REA) analysis methodology described in DPC-NE-3001-PA includes the use of the three-dimensional space-time transient neutronics nodal code ARROTTA (Ref. 27) to perform the nuclear analysis portion of transient response; the VIPRE-01 code to model the core thermal response including peak fuel enthalpy, a core-wide DNBR evaluation, and transient core coolant expansion; and the RETRAN-02 code to simulate the reactor coolant system pressure response to the core power excursion. This methodology will continue to be used for the REA analysis except for the use of the SIMULATE-3K code (Ref. 28) to replace ARROTTA to perform the nuclear analysis of the response of the reactor core to the rapid reactivity insertion resulting from a control rod being ejected out of the core.

Section 6.6 of DPC-NE-2009 describes the REA analysis methodology using SIMULATE-3K, including a brief description of the code and models, code verification and benchmark, and the REA analysis application of SIMULATE-3K. SIMULATE-3K is a three-dimensional transient neutronic version of the NRC approved SIMULATE-3P computer code (Ref. 29) and uses the same neutron cross section library. It uses a fully-implicit time integration of the neutron flux, delayed neutron precursors, and heat conduction models. The average beta for the time-varying neutron flux is determined by performing a calculation of the adjoint flux solution. The code user has the option of running the code with a fixed time step or a variable time step depending on the sensitivity to changes in the neutronics. The SIMULATE-3K code has incorporated additional capability to model reactor trips at user-specified times in the transient or following a specified excore detector response, which allows the user to specify the response of individual detectors as required to initiate the trip, as well as the time delay prior to release of the control rods based on the excore detector response model. The code also permits the user input to control the velocity of the control rod movement, providing a different perspective for each velocity chosen.

The SIMULATE-3K code vendor, Studsvik of America, Inc., had performed the code verification and validation during its development to verify correctness of the coding and to validate the applicability of the code to specified analyses and ensure compatibility with existing methodology. The validation included benchmarks of the fuel conduction and thermal hydraulic models, the transient neutronics model, and the coupled performance of the transient neutronics and thermal-hydraulic models. The fuel and thermal hydraulic models were validated against the TRAC code, while the neutronic model was benchmarked against the solutions of the industry standard light water reactor problems generated by QUANDRY, NEM, and CUBBOX (Ref. 30, 31, 32). Benchmarking of the coupled performance of the thermal hydraulic and transient neutronics models was carried out against the results from a standard NEACRP [Nuclear Energy Agency Control Rod Problem] rod ejection problem to the PANTHER code (Ref. 33). Steady state comparison of S3K was performed against the NRC approved CASMO-3/SIMULATE-3P. In addition, DPC performed comparisons of the SIMULATE-3K and ARROTTA calculations for the reference REA analysis for the Oconee Nuclear Station showing very good agreement for core power versus time for the ejection occurring at the end-of-cycle from the maximum allowable power level with 3 and 4 RCPs operating and from both beginning-of-cycle and end-of-cycle at hot zero power and hot full power conditions. These SIMULATE-3K validation benchmarks were presented in DPC-NE-3005-P (Ref. 34), which the staff has reviewed for approval of using SIMULATE-3K for the analysis of the REA for the Oconee plants.

Section 6.6.1.3.3 of DPC-NE-2009 provides an additional benchmark of SIMULATE-3K by comparing the SIMULATE-3K and ARROTTA calculations for the reference REA analyses performed for beginning of life (BOC) and end of life (EOC) at hot-full-power (HFP) and hot-zero-power (HZP) conditions for McGuire and Catawba Nuclear Stations. The reference core used in the benchmark calculations was a hypothetical Catawba 1 Cycle 15 core, which represents typical fuel management strategies currently being developed for reload core designs at McGuire and Catawba. The comparison between the SIMULATE-3K and ARROTTA calculations of the core power level and nodal power distribution as functions of time during the REA transient demonstrate the acceptability of the physical and numerical models of SIMULATE-3K for application in the REA analyses for McGuire and Catawba Nuclear Station.

Section 6.6.2.2 of DPC-NE-2009 describes the use of the SIMULATE-3K code to perform license analysis of the design basis REA. The basic methodology as described in

DPC-NE-3001PA remains unchanged with the exception of minor differences between SIMULATE-3K and ARROTTA. The core power levels and nodal power distributions calculated by SIMULATE-3K are used by VIPRE to determine the fuel enthalpy, the percentage of fuel pins exceeding the DNB limit, and the coolant expansion rate. All inputs to VIPRE, once supplied by the NRC approved-code ARROTTA, are now supplied by SIMULATE-3K.

In the SIMULATE-3K nuclear analysis of an REA, a fuel assembly is typically geometrically modeled by several radial nodes. Axial nodalization and the number of nodes are chosen to accurately describe the axial characteristics of the fuel. For current fuel designs, a typical axial nodalization of 24 equal length fuel nodes in the axial direction is used. SIMULATE-3K explicitly calculates neutron leakage from the core by use of reflector nodes in the radial direction beyond the fuel region and in the axial direction above and below the fuel column stack. The fuel and reflector cross sections are developed in accordance with the methodology described in the approved topical report DPC-NE-1004A for SIMULATE-3P.

The SIMULATE-3K REA analysis is performed at four statepoints: BOC and EOC at HZP and HFP conditions for the determination of three-dimensional steady-state and transient power distributions, as well as individual pin powers. Conservative input parameters are used to ensure that the rod ejection analysis produces limiting results that bound future reload cycles. Sections 6.6.2.2.1 and 6.6.2.2.2 describe the methods to ensure conservatism in the analysis of transient response by increasing the fission cross sections in the ejected rod locations and in each assembly and by applying the "factors of conservatism" to the reactivity feedback for moderator and fuel temperatures, control rod worths for withdrawal and insertion, effective delayed neuron, and ejected rod worth, etc. In response to a staff question (No. 9, Ref. 14), the licensee provided a description of the method of determining the "factors of conservatism." The staff has reviewed the overall SIMULATE-3K methodology, and found it to be acceptable for application to the REA analyses for McGuire and Catawba.

2.4.4 Compliance with Safety Evaluation Conditions:

As discussed above, licensing analyses of reload cores with the RFA design use the methodologies described in various topical reports for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current McGuire/Catawba cores. These methodologies may have inherent limitations, or conditions or restrictions imposed by the associated NRC safety evaluations in their applications. The acceptability of the licensing analyses is subject to the application being within the limitations of the methodologies used and the conditions or restrictions imposed in the respective safety evaluations. In response to a staff question regarding the resolutions of these limitations, conditions, and restrictions in the RFA reload safety analyses, the licensee provided (Response to Question 11, Ref. 14) a list of restrictions imposed by NRC safety evaluations and the corresponding resolutions in the application of the licensee's methodologies used for the safety analyses of the non-LOCA transients and accidents. In addition, for the LOCA analyses to be performed by Westinghouse, the licensee provided a Westinghouse response (Ref. 35) regarding the safety evaluation restrictions and corresponding compliance for the 1985 SBLOCA Evaluation Model with NOTRUMP and the 1981 Evaluation Model with BASH. The resolutions or compliance with the conditions or restrictions provided in these responses provide guidance for the licensee referencing DPC-NE-2009 in the RFA reload licensing analyses. The staff concludes that the safety evaluation conditions have been properly addressed.

2.5 Fuel Assembly Repair and Reconstitution

Section 7.0 of DPC-NE-2009 describes the evaluation of the reconstitution or repair of fuel assemblies having failed fuel rods during refueling outages in an effort to achieve the zero fuel defect goal during cycle operation. The primary replacement candidate for use in reconstitution of failed fuel rods is a fuel rod that contains pellets of natural uranium dioxide, but solid filler rods made of stainless steel, zircaloy, or ZIRLO would be used if local grid structural damage exists. The reconstitution of the RFA assembly with filler rods will be analyzed with NRC-approved methodology and guidelines described in DPC-NE-2007P-A (Ref. 36), along with other licensed codes and correlations, to ensure acceptable nuclear, mechanical, and thermal-hydraulic performance of reconstituted fuel assemblies.

For a reload core using reconstituted Westinghouse fuel, Westinghouse has reviewed the effects of the reconstituted fuel with the criteria specified in Standard Review Plan 4.2 and determined that the only fuel assembly mechanical criteria impacted by reconstitution are fuel assembly holddown force and assembly structural response to seismic/LOCA loads. Westinghouse has evaluated these effects on the LOCA analyses using the approved methodology WCAP-13060-P-A (Ref. 37), and concluded that the reconstituted fuel assembly designs are acceptable for both normal and faulted condition operations.

2.6 Technical Specifications Changes

The licensee's July 22 and October 22, 1998, letters proposed changes to the Technical Specifications with the technical justifications for these changes described in Chapter 8 of DPC-NE-2009. The licensee's January 28, May 6 and June 24, 1999, letters provided revisions to some of the proposed changes. The staff's evaluation follows.

2.6.1 Proposed Change to TS Figure 2.1.1-1:

The licensee proposed to modify Figure 2.1.1-1, "Reactor Core Safety Limits - Four Loops in Operation," by (1) deleting the 2455 psia safety limit line, which is the current upper bound pressure allowed for power operation; (2) combining separate Unit 1 and Unit 2 figures into only one figure; and (3) revising the other safety limit lines (see following paragraph). The resulting Figure 2.1.1-1 was submitted by a letter, M. Tuckman to NRC, dated June 24, 1999 (Ref. 39).

The 2455 psia bounding pressure is based on the pressure range of the CHF correlation used in DNBR analyses of the Mark-BW fuel. Since the upper range of applicability of the WRB-2M CHF correlation for the RFA design is 2425 psia, the 2455 psia safety limit line is deleted, and the remaining safety limit lines with 2400 psia as the upper bound safety limit line are within the range of the CHF correlations for the Mark-BW and RFA fuel designs. As described in its response to a staff's question (No. 12, Ref. 14), the licensee has performed an evaluation to ensure the remaining safety limit lines of Figure 2.1.1-1, which were based on the CHF correlation for the Mark-BW fuel design and the hot leg boiling limit, bound the safety limit for the DNBR limit of the WRB-2M correlation for the RFA design. Both the full RFA core and the transition RFA/Mark-BW cores were evaluated to ensure that the established limits were conservative. The DNBR values were greater than the design DNBR limit for all the cases in both evaluation. Therefore, the safety limit lines in Figure 2.1.1-1, with the deletion of the 2455 psia safety limit line, are acceptable.

2.6.2 Proposed Changes to Surveillance Requirements 3.2.1.2, 3.2.1.3, and 3.2.2.2:

TS Surveillance Requirements (SRs) 3.2.1.2, 3.2.1.3, and 3.2.2.2, respectively, require the heat flux hot channel factor F_q (x,y,z) and the enthalpy rise hot channel factor $F_{\Delta h}$ (x,y) to be measured periodically (once within 12 hours after achieving equilibrium conditions after a power change exceeding 10% rated thermal power and every 31 effective full power days thereafter) using the incore detector system to ensure the values of the total peaking factor and the enthalpy rise factor assumed in the accident analyses and the reactor protection system limit are not violated. To avoid the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, these SRs currently specify a penalty factor of 1.02 for the heat flux and enthalpy rise hot channel factors if the margin to the F_q (x,y,z) or $F_{\Delta h}$ (x,y) has decreased since the previous surveillance. The 2% margin-decrease penalty was based on the current reload cores.

For the reactor core containing the RFA fuel design with integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. The licensee proposed to remove the 2% penalty value from these SRs and replace them with tables of penalty values as functions of burnup in the Core Operating Limits Report (COLR) to facilitate cycle-specific updates. Tables 8-1 and 8-2, respectively, provide typical values for the burnup-dependent margin-decrease penalty factors for the heat flux and enthalpy rise hot channel factors. The actual values for the transitional core can not be provided until the final design for the core is complete. In response to a staff question (No. 13, Ref. 14), the licensee provided the methodology for calculating the burnup-dependent penalty factors. In addition, Technical Specification 5.6.5 will reference topical report DPC-NE-2009, which includes this response to the staff's question for the approved methodology used to calculate these penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin-decrease penalty factors in the COLR acceptable.

2.6.3 Proposed Change to TS 4.2.1:

TS 4.2.1, "Fuel Assemblies," which specifies the design features for fuel assemblies, will be revised to add ZIRLO cladding to the fuel assembly description.

2.6.4 Proposed Changes to Section 5.6.5b:

By a letter dated May 6, 1999 (Ref. 38), the licensee expanded the original amendment request by proposing more changes in Section 5.6.5. The section lists all the topical reports previously approved by the staff. Thus these proposed changes are administrative or editorial. The staff finds them all acceptable as follows:

WCAP-10216P-A, "Relaxation of Constant Axial Offset Control FQ Surveillance Technical Specification" -- This is deleted since it had been previously replaced by Item 5 (re-numbered Item 4), DPC-NE-2011P-A.

BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants" -- The dates of the various staff safety evaluations have been updated.

DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology" -- The Revision number has been changed from "2" to "3". The staff's safety evaluation date is also updated.

DPC-NE-3000P-A, "Thermal-Hydraulic Transient Analysis Methodology" -- The Revision number is changed from "1" to "2". The staff's safety evaluation date is also updated.

DPC-NE-2001P-A "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel" -- This is deleted, and is replaced by DPC-NE-2008P-A.

BAW-10183P-A, "Fuel Rod Gas Pressure Criterion" -- This is deleted. DPC-NE-2008P-A references this report, and therefore there is no need for an individual listing.

WCAP-10054P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code" -- This report is applicable to the Westinghouse fuel.

DPC-NE-2009P-A, "Westinghouse Fuel Transition Report" -- This report has been evaluated in the above sections of this safety evaluation and found acceptable.

2.6.5 Proposed Changes to the Technical Specifications Bases Document:

The TS Bases is a licensee-controlled document and is not part of the Technical Specifications (10 CFR 50.36(a)). However, the staff reviewed the licensee's proposed changes as supplemental information for the TS changes evaluated above. The Bases sections for SR 3.2.1.2, 3.2.1.3 and 3.2.2.2 will be revised to reflect the corresponding TS changes. The staff finds the proposed changes to the Bases acceptable.

3.0 REVIEW SUMMARY OF TOPICAL REPORT

The staff has reviewed the licensee's Topical Report DPC-NE-2009P and found it acceptable for referencing for analysis of reloads with Westinghouse RFA design. The topical report references many topical reports, which provide methodologies for various aspects of the RFA reload licensing analyses. Acceptability of DPC-NE-2009P remains subject to the limitations set forth in the SERs on these topical reports.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, North Carolina State official Mr. Johnny James was notified of the proposed issuance of the amendments. The official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20, and change surveillance requirements. The NRC staff has determined that the amendments involve 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 staff has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 69338, dated December 16, 1998; 64 FR 35202, dated June 30, 1999, and 64 FR 43771, dated August 11, 1999) . The licensee's September 15, 1999, letter (Ref. 44) provided clarifying information that did not change the scope of the application and the initial proposed no significant hazards consideration determination. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in

10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Attachment: Technical Evaluation Report

Principal Contributor: Yi-Hsiung Hsii
Anthony Attard
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Peter Tam

Date: September 22, 1999

7.0 <u>REFERENCES</u>

1. Letter, M. S. Tuckman (Duke Energy Corporation) to NRC, "Duke Energy Corporation, McGuire Nuclear Station, Units 1 and 2, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Revisions to Improved Technical Specifications for Implementation of Westinghouse Fuel as Described in Topical Report Described in Topical Report DPC-NE-2009/DPC-NE-2009P, Duke Power Company Westinghouse Fuel Transition Report," July 22, 1998.

2. Letter, M. S. Tuckman (Duke Energy Corporation) to NRC, "Duke Energy Corporation, McGuire Nuclear Station, Units 1 and 2, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station, Units 1 and 2, Docket Numbers 50-413 and 50-414, Supplement to License Amendment Request for Revisions to Improved Technical Specifications for Implementation of Westinghouse Fuel as Described in Topical Report Described in Topical Report DPC-NE-2009/DPC-NE-2009P, Duke Power Company Westinghouse Fuel Transition Report," October 22, 1998.

3. Duke Power Company, DPC-NE-2009/DPC-NE-2009P, "Duke Power Company Westinghouse Fuel Transition Report," July 1998.

4. "Mark-BW Mechanical Design Report," BAW-10172P-A, December 1989.

5. Davison, S. L., T. L. Ryan, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995, WCAP-12610-P-A.

6. Letter from N. J. Liparulo (Westinghouse) to J. E. Lyons (USNRC), "Transmittal of Response to NRC Request for Information on Wolf Creek Fuel design Modifications," June 30, 1997, NSD-NRC-97-5189.

7. Davison, S. L., "Westinghouse Fuel Criteria Evaluation Process," WCAP-12488-P-A, October 1994.

8. "Duke Power Company Fuel Rod Mechanical Reload Analysis Methodology Using TACO3," DPC-NE-2008P-A, April 1995.

9. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A, August 1988.

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42. Letter, P. S. Tam (NRC) to G. R. Peterson (Catawba Nuclear Station), finding use of high burnup methodology described in topical report BAW-10186P-A acceptable for reload licensing application at Catawba, March 3, 1999.

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SER

Revision 3

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

September 16, 2002

Mr. M. S. Tuckman
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church St
Charlotte, NC 28202

SUBJECT: 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-2005P (TAC NOS. MB3105, MB3106, MB3173, AND MB3175)

Dear Mr. Tuckman:

The Nuclear Regulatory Commission staff has completed its review of the revision to the topical report submitted by the Duke Power Company's (DPC) letters dated September 13, 2001, as supplemented by letter dated August 14, 2002, entitled "Appendix E to Topical Report DPC-NE-2005P, Duke Power Thermal-Hydraulic Statistical Core Design Methodology (Proprietary)". The report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the enclosed NRC evaluation. The evaluation defines the basis for acceptance of the report.

The staff does not intend to repeat its review of the matters described in the report and found acceptable when the report is referenced in a license application, except to ensure that the material presented is applicable to the specific plant involved. Staff acceptance applies only to the matters described in the report.

We request that DPC publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions should include an " -A" (designating accepted) following the report identification symbol.

Should NRC criteria or regulations change so that staff conclusions regarding the acceptability of the report are invalidated, DPC will be expected to revise and resubmit their documentation, or to submit justification for continued effective applicability of the topical report without revision of their documentation.

Sincerely,

J. A. Nakoski for

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

Enclosure: Safety Evaluation

cc w/encl: See next page

McGuire Nuclear Station
Catawba Nuclear Station

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UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO APPENDIX E TO TOPICAL REPORT DPC-NE-2005P

DUKE POWER THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 and 2

DUKE ENERGY CORPORATION

DOCKET NOS. 50-413, 50-414, 50-369, AND 50-370

1.0 INTRODUCTION

By letter dated September 13, 2001 (Reference 1), as supplemented by letter dated August 14, 2002 (Reference 2), Duke Power Company (DPC), a subsidiary of Duke Energy Company, submitted for NRC review and approval, an Appendix E, "McGuire/Catawba Plant Specific Data, Advanced Mark-BW Fuel, BWU-Z CHF Correlation," to the report, DPC-NE-2005P, "Duke Power Thermal-Hydraulic Statistical Core Design Methodology" (Reference 3).

The approval of DPC-NE-2005P in the staff's Safety Evaluation Report, as included in reference 3, acknowledged that the statistical core design (SCD) methodology is direct and general enough that it could be applicable to other pressurized-water reactors (PWRs), however, it was approved with the following restrictions:

(1) The VIPRE-01 methodology for thermal-hydraulic analysis must be approved for use with the core model.

(2) All correlations, including the critical heat flux (CHF) correlation, are subject to the conditions in the VIPRE safety evaluation report (Reference 4).

(3) The methodology was approved for use in DPC plants only.

In addition to the above restrictions, DPC is required to justify on a plant-specific basis the uncertainties and distributions used for each application. The selection of state points used for generating the statistical design limit must also be justified to be appropriate on a plant-specific basis.

2.0 EVALUATION

The submittals contain the plant-specific data and statistical departure from nucleate boiling ratio (DNBR) limits for the McGuire and Catawba Nuclear Stations with the Advanced Mark-BW

fuel design using the BWU-Z CHF correlation and provide details of the fuel assembly structural and thermal-hydraulic features unique to the Advanced Mark-BW fuel design.

DPC's August 14, 2002, submittal describes the two separate fuel pellet materials that can be used in this fuel design structure. When used with uranium fuel pellets, the fuel assembly is called Advanced Mark-BW. If used with mixed oxide fuel pellets, the fuel assembly is called Mark-BW/MOX1. Since the issues in this report are applicable to these fuel designs, the term Advanced Mark-BW in this report means both the Mark-BW/MOX1 and the Advanced Mark-BW design. DPC states that the Advanced Mark-BW fuel design is an evolutionary improvement of the successful Mark-BW17 fuel assembly design. The only thermal-hydraulic difference between the Mark-BW17 fuel and the Advanced Mark-BW fuel is the addition of three mid-span mixing grids for the Advanced Mark-BW design. Since the thermal-hydraulic features are the same, the only impact the different fuel rod designs could have on the statistical DNBR limit is in the radial and axial nuclear uncertainties of $F^N_{\Delta H}$ and F_z in Table E-4 of the submittal (Reference 1).

The analysis is for the McGuire and Catawba Plants (four-loop Westinghouse PWR's) with the Advanced Mark-BW fuel. Approved methodologies including the VIPRE-01 thermal-hydraulic computer code (Reference 5) and the McGuire/Catawba eight-channel model (Reference 6) are used in this analysis.

The SCD analysis described in Reference 1 includes: (1) state points which represent the range of conditions to which the statistical DNB analyses limit will be applied; (2) uncertainties that were selected to bound the values calculated for each parameter at McGuire and Catawba; (3) the transition core model which determines the impact of the geometric and hydraulic difference between the resident 17x17 Westinghouse robust fuel assembly fuel and the new Advanced Mark-BW design; and (4) the statistical DNBR design limit for each state point evaluated that was determined based on the 500 and 6,000 case runs.

The staff's concerns with respect to the SCD analysis in the areas of the applicability of the approved methodologies (References 5,6, 7, and 8) for the Advanced Mark-BW fuel design, the supporting data bases, and the mixed core application, were responded to by DPC's submittal dated August 14, 2002 (Reference 2), and in discussions on the September 13, 2001, submittal held with DPC on August 28, 2002. DPC indicated in those discussions that: (1) the results of the SCD analyses in Table E-5 are used for selection of a conservative DNBR value for McGuire and Catawba if the statepoints are within the range in Table E-6, otherwise, the DNBR values in Table E-2 from non-SCD analyses will be used; (2) the mixed core flow mismatch can be confirmed from the reactor core monitoring system; and (3) the analyses in Tables E-2 and E-5 were performed as a mixed core to reflect the McGuire and Catawba core designs.

Based on the NRC staff's review of Appendix E to topical report DPC-NE-2005P, and the response to the staff's request for additional information (Reference 2), the staff finds Appendix E, "Use of BWU-Z CHF Correlation with the Advanced Mark-BW Fuel Assembly," to be acceptable because of the following:

(1) NRC-approved methodologies (Thermal-Hydraulic SCD, the VIPRE-01 code, the mixed core model, and the BWU-Z CHF correlation) are used.

(2) The larger of the two correlation limits produced by VIPRE-01 or LYNXT will be used for non-SCD analyses. This DNBR value is 1.19, as shown in Table E-2.

(3) The conservative DNBR value of 1.36 from the 6,000 case runs will be used for SCD analyses.

The staff may audit the data bases used to support this application and the mixed core calculation record file as part of the application review for the first plant that uses this methodology.

3.0 CONCLUSION

Based on the above discussions, the staff concludes that the proposed use of the BWU-Z CHF correlation with the Advanced Mark-BW fuel Assembly for McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2, as described in DPC's submittals dated September 13, 2002, and August 14, 2002, is acceptable.

4.0 REFERENCES

1. Letter from K. S. Canady, DPC, to USNRC, "Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414, McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369,50-370, Appendix E to Topical Report DPC-NE-2005P, Duke Power Thermal-Hydraulic Statistical Core Design Methodology," September 13, 2001.

2. Letter from K. S. Canady, DPC, to USNRC, "McGuire and Catawba Nuclear Stations, Units 1 and 2, Docket Nos. 50-369, 50-370, 50-413, 50-414, Topical Report DPC-NE-2005P, Thermal-Hydraulic Statistical Core Design Methodology, Revision 3 (Appendix E); Request for Additional Information," August 14, 2002.

3. Letter from M. S. Tuckman, DPC, to USNRC, "Issuance of Approved Version of DPC-NE-2005P (DPC-NE-2005P-A)," dated August 8, 1995.

4. "Safety Evaluation Report on the VIPRE-01 Code, " May 1986, and "Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to VIPRE-01 Mod-02 for PWR and BWR Applications, EPRI-NP-2511-CCMA, Revision 3," October 30,1993, USNRC.

5. VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

6. DPC-NE-2004P-A, McGuire and Catawba Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, Revision 1, February 1997.

7. BAW-10199P-A, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," March 2002.

8. DPC-NE-2009P-A, "Duke Power Company Westinghouse Fuel Transition Report," December 1999.

Principal Contributor: T. Huang, DSSA/SRXB

Date: September 16,2002

SER, TER, and Correspondence
Revision 2

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

June 8, 1999

RECEIVED
JUN 1 5 1999
W. R. McCollum, Jr.

Mr. W. R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
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SUBJECT: OCONEE NUCLEAR STATION UNITS 1, 2 AND 3 RE: TOPICAL REPORT NUMBER DPC-NE-2005P USE OF BWU-Z CRITICAL HEAT FLUX CORRELATION FOR MARK-B11 FUEL (TAC NOS. M98660, M98661, AND M98662)

Dear Mr. McCollum:

By letter dated April 22, 1997, and supplemented by letters dated September 21, 1998, and May 13, 1999, Duke Energy Corporation requested NRC review and approval of the use of the BWU-Z Critical Heat Flux Correlation for Mark-B11 Fuel, which is described in Appendix D to Topical Report DPC-NE-2005P, "Duke Power Company Thermal - Hydraulic Statistical Core Design Methodology." The submittal contains analyses of the Mark-B11 fuel assemblies (which were analyzed using the BWU-Z critical heat flux correlation) and justifications for the specific uncertainties and distributions used in the application, and for the selected statepoints used to generate the statistical design limit.

The NRC staff was assisted in this review by its consultant, Pacific Northwest National Laboratory (PNNL). Based on the information provided and the analysis and recommendations provided by PNNL, we find the proposed Appendix D to DPC-NE-2005P to be acceptable. However, this approval is subject to the conditions described in the Safety Evaluation, which are also the commitments made in your letter dated May 13, 1999.

Sincerely,

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270 and 50-287

Enclosure: Safety Evaluation

cc w/encl: See next page

Oconee Nuclear Station

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT DPC-NE-2005P

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated April 22, 1997 (Reference 1), as supplemented September 21, 1998, and May 13, 1999, (References 2 and 3 respectively), Duke Energy Corporation (DEC), licensee for the Oconee Nuclear Station, Units 1, 2, and 3, requested NRC staff review and approval of Appendix D, "Oconee Plant Specific Data, Mark-B11 Fuel, Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design" (Reference 1), to DPC-NE-2005P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology" (Reference 4). The submittal contains analyses of the Mark-B11 fuel assemblies, analyzed using the BWU-Z critical heat flux correlation, and provides the required justifications for the specific uncertainties and distributions used in the application, and for the selected statepoints used to generate the statistical design limit.

The staff was assisted in this review by its consultant, Pacific Northwest National Laboratory (PNNL). The staff's evaluation includes the licensee's submittal (Reference 1), the licensee's response to the staff's request for additional information (RAI) dated September 21, 1998 (Reference 2), and the licensee's clarification dated May 13, 1999 (Reference 3). The staff has adopted the findings recommended in our consultant's attached technical evaluation report.

2.0 EVALUATION

This review considered Appendix D "Oconee Plant Specific Data, Mark-B11 Fuel, Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design" to DPC-NE-2005(P) "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology". The details of the evaluation are provided in the attachment.

This appendix contains plant-specific data for two-loop Babcock and Wilcox pressurized water reactors and specific limits for the Oconee Nuclear Station with Mark-B11 fuel using the BWU-Z

form of the BWU critical heat flux (CHF) correlation. Approved methodologies, including the VIPRE-01 thermal-hydraulic computer code (EPRI NP-2511-CCM-A, Vol. 1-4) and the Oconee eight and nine channel models (DPC-NE-2003P-A), are used in this analysis.

The statistical core design (SCD) analysis includes: (1) statepoints that represent the range of conditions to which the statistical DNB analyses limit will be applied; (2) uncertainties that were selected to bound the values calculated for each parameter at Oconee and have not changed except for the rod power hot channel factor (Fq), core flow measurement, and departure from nuclear boiling ratio (DNBR) correlation; (3) the statistical DNBR design limit for each statepoint evaluated that was determined based on the 500 and 5000 case runs; and (4) the transition core model that determines the impact of the geometric and hydraulic difference between the resident Mark-B10 series fuel and the new Mark-B11 design. The staff's concerns with respect to the statistical core design analysis in the areas of the applicable range of conditions, the uncertainties for core flow, the hot channel factor Fq and DNBR correlation, and the mixed core penalty were clarified in the licensee's response to the staff RAI (Reference 2).

Based on our review of Appendix D to Topical Report DPC-NE-2005P and the response to the staff's RAI (Reference 2), the staff finds Appendix D, "Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design" to be acceptable. However, this approval is subject to the following conditions that were committed to by DEC in Reference 3:

(1) Omission of the parameter "Fq" from the SCD analysis of the Oconee plant with a new fuel design must be justified for each particular case. Acceptance of its omission in the case of Mark-B11 fuel does not constitute a general approval of its removal from the parameters to be considered in this methodology.

(2) The applicability of a CHF correlation to mixed core geometries is an issue that must be examined for each transition to new fuel to determine if the mixed core non-uniformities take the local hot channel conditions outside the range of applicability of CHF correlation.

(3) The SCD analysis shall be reviewed and revised as needed if the Mark-B11 CHF correlation range of applicability is changed.

3.0 CONCLUSION

Based on our review of Appendix D to the topical report DPC-NE-2005P and supplemental information supplied by DEC, the staff concludes that Appendix D, "Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design" is acceptable. However, actions should be taken whenever a new fuel design is introduced, as follows:

1. Omission of the parameter "Fq" from the SCD analysis of the Oconee plant with a new fuel design must be justified for each particular case. Acceptance of its omission in the case of Mark-B11 fuel does not constitute a general approval of its removal from the parameters to be considered in this methodology.

2. The applicability of a CHF correlation to mixed core geometries is an issue that must be examined for each transition to new fuel to determine if the mixed core non-uniformities take the local hot channel conditions outside the range of applicability of CHF correlation.

3. The SCD analysis shall be reviewed and revised as needed if the Mark-B11 CHF correlation range of applicability is changed.

Attachment: Technical Report

Principal Contributor: Tai Huang

Date: June 8, 1999

REFERENCES

1. Letter from M. S. Tuckman to USNRC, Oconee Nuclear Station Docket Nos. 50-269, 50-270, and 50-287, Use of BWU-Z Critical Heat Flux Correlation for Mark-B11 Fuel, April 22, 1997 (Proprietary and Non-Proprietary Information Available).

2. Letter from M. S. Tuckman to USNRC, Response to NRC Request for Additional Information on Appendix D to Topical Report DPC-NE-2005-P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," September 21, 1998 (Proprietary and Non-Proprietary Information Available).

3. Letter from M. S. Tuckman to USNRC, Oconee Nuclear Station Docket Nos. 50-269, 50-270, and 50-287, Duke Commitment to Conditions of SER and Clarification of Topical Report DPC-NE-2005 Revision Level, May 13, 1999.

4. DPC-NE-2005P-A, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, February 1995.

5. BAW-10199P-A, The BWU Critical Heat Flux Correlations, August 1996.

TECHNICAL EVALUATION REPORT for

DUKE POWER COMPANY THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY; APPENDIX D: OCONEE PLANT SPECIFIC DATA
(DPC-NE-2005(P), Appendix D)

Judith M. Cuta

April 1999

Prepared for
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
under Contract DE-AC06-76RLO 1830
NRC FIN I2009

Pacific Northwest National Laboratory
Richland, WA 99352

SUMMARY

With the corrections to Table 0-2 and 0-4 provided in the OPC response to the RAI (see Reference 1), the plant specific data for Oconee presented in Appendix 0 of OPC-NE-2005P are appropriate for use in the SCO analysis. The parameters are for cores containing Mark-B11 fuel, and transition cores containing both Mark-B10 and Mark-B11 fuel assemblies.

BACKGROUND

The Duke Power Company (DPC) statistical core design (SCD) methodology as documented in Topical Report DPC-NE-2005P-A was granted approval by the Nuclear Regulatory Commission on February 24, 1995. 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: "

(1) The VIPRE-01 methodology for thermal-hydraulic analysis must be approved for use with the core model.

(2) All correlations, including the critical heat flux (CHF) correlation, are subject to the conditions in the VIPRE safety evaluation report (See Reference 2).

(3) The methodology is approved for use in Duke Power Company plants only.

In addition to the above restrictions, DPC is required to justify on a plant-specific basis the uncertainties and distributions used for each application. The selection of statepoints used for generating the statistical design limit must also be justified to be appropriate, on a plant specific basis.

The Topical Report DPC-NE-2005P-A includes an Appendix A with plant specific data for the Oconee plant with Mark-B10 fuel assemblies (B&W fuel), using the BWC critical heat flux correlation to determine the MDNBR limit. The current submittal, Appendix D, is for Mark-B11 fuel assemblies, analyzed using the BWU-Z critical heat flux correlation. The purpose of Appendix D is to provide the required justifications for the specific uncertainties and distributions used in the application, and for the selected statepoints used to generate the statistical design limit.

EVALUATIONS

The presentation of the plant-specific data for Oconee with Mark-B11 fuel is exceedingly terse in Appendix D. This made it extremely difficult to evaluate the justification for any changes in the plant specific data, or the selection of the set of statepoints used in the analysis. In response to a request for additional information (see Reference 1), DPC provided additional documentation of the means of justifying the specific uncertainties for the Mark B-11 core.

There are four major changes in the plant specific parameters used in the analysis of the Oconee core with Mark-B11 fuel (compare Table D-4 of the submittal with Table A-2 of DPC-NE-2005P-A, Appendix A). The core flow uncertainty is larger, the Fq parameter is also larger, the hot channel factor area uncertainty is unchanged (despite significant changes in the geometry), and the parameter Fq" has been omitted entirely from the analysis. These changes were merely reported in the original submittal, and no justification was given. However, additional information supplied in response to the RAI provided adequate justification for the changes. The change in the flow uncertainty is the result of re-calculating the Chapter 15 transients for Oconee with Mark-B11 fuel using the BWU-Z correlation for combinations of 4, 3, and 2 pump operation. The parameter Fq is based on the rod power hot channel factor

supplied by the fuel vendor and the approved value of the radial peak uncertainty. The hot channel factor area uncertainty is based on information supplied by the fuel vendor and will be verified by inspection of the final fuel assemblies and components. The specific value of 3.00% for Oconee bounds particular acceptance criteria for the fuel, and cannot be exceeded without invoking additional analyses to determine the effect on the statistical design limit (see Table 7 of DPC-NE-2005P-A).

Omitting the parameter Fq" from the analysis was justified by DPC based on work by other fuel vendors (specifically, in WCAP-8202 and CENPD-207) showing that local heat flux spikes as great as 20% above the local nominal heat flux do not have any noticeable effect on the minimum DNBR. DPC believes that this effect is generic to PWR fuel, and states that it was confirmed to be applicable to Mark-B11 fuel by the fuel vendor. In addition, the parameter Fq" calculated by the vendor is much smaller for this fuel than for Mark-B10 fuel (a value of 1.41% for Mark-B11, compared to 2.08% for Mark-B10.)

The additional information supplied by DPC shows that it is justifiable to omit the parameter Fq" from the SCD analysis of the Oconee plant with Mark-B11 fuel. However, the assertion that the parameter can be omitted in analysis of PWR fuel in general is too broad. Fuel designs developed in future might conceivably have a different sensitivity to this parameter, and DPC should be required to evaluate its applicability to each new fuel design. If it can be omitted for a particular fuel design, DPC must provide justification for such omission, as required by the SER for DPC-NE-2005P-A.

The discussion in Appendix D of the treatment of transition fuel cycles. when Mark-B1 0 and Mark-B11 fuel would be co-resident in the core, is extremely vague and incomplete. In response to the RAI, however, OPC provided additional details to clarify the method of determining the transition core penalty and implementation of the options for its application. It appears that the methodology used will capture the largest penalty applicable to a specific core design.

Appendix D contains no discussion of the applicability of the BWU-Z CHF correlation to Mark-B11 fuel in mixed cores. This is a serious oversight, since there are marked local pressure drop differences between the Mark-B10 and Mark-B11 fuel assembly designs, even though the overall pressure drop is essentially the same. The local differences (due to differences in the grid design) will result in subchannel flow distributions in the vicinity of the spacer grids that are significantly different from the distributions in a uniform core of Mark-B11 fuel only. Since the BWU-Z CHF correlation is based on data for Mark-B11 fuel only, it is not obvious that the correlation is applicable to cores containing both B-10 and B-11 fuel. Additional information supplied by DPC referenced CHF testing by the fuel vendor[1] with a 5x5 test assembly simulating mixed core conditions (BAW-10143P-A). For the conditions tested, there was no significant change in the accuracy of the BWC correlation for the bundle modeling a mixed core, compared to results obtained for bundles modeling a uniform core.

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[1]Framatome Cogema Fuels, formerly Babcock & Wilcox Fuel Company.

This is evidence that the mixed core conditions do not result in local conditions in the subchannel that are outside the range of the CHF correlation. Thermal-hydraulic calculations with the VIPRE code show that the geometry corresponding to a mixed core of Mark-B10 and Mark-B11 fuel produces local velocity distributions that differ from a uniform core by only about one-fifth as much as the most severe conditions encountered in test data reported in BAW-10143P-A. Based on these factors, DPC concludes that the BWU-Z correlation is also applicable to mixed cores.

This argument has several weaknesses. It is based on data for Mark-B10 fuel, not Mark-B11, and the correlation used to evaluate the data was the BWC correlation, not the BWU-Z correlation. Because CHF correlations are *ad hoc* fits to data sets rather than models based on the physical behavior of the system, there is no reason to suppose as a general rule that what is true of one fuel design and CHF correlation will be true of another fuel design and its CHF correlation. In this particular case, however, it can be argued that there are two main reasons to expect the BWU-Z correlation to behave in essentially the same manner as the BWC correlation for a mixed core of B-10 and B-11 fuel. First, the fuel designs are from the same vendor, and have similar physical geometry. Second, the CHF correlations share a common developmental path, have similar form, and show similar fit to their respective databases. In addition, DPC reports that thermal-hydraulic calculations show the non-uniformities for mixed B-10/B-11 cores will in general be much smaller than the conditions tested using the BWC correlation in the bundle modeling a mixed core for Mark-B10 fuel.

For this case, DPC has shown that the BWU-Z CHF correlation can be expected to be applicable to mixed cores of Mark-B10 and Mark-B11 fuel. However, this conclusion should not be interpreted as laying to rest the generic issue of the applicability of CHF correlations to mixed core geometries. This issue must be examined for each transition to new fuel, to determine if the mixed core non-uniformities result in local hot channel conditions outside the range of applicability of the CHF correlation. At a minimum, subchannel thermal-hydraulic calculations are needed to determine the magnitude of the most severe local velocity depression in the hot channel. Test data obtained in bundles modeling a mixed core may be necessary in some cases to fully resolve the issue.

The description of the range of applicability of the BWU-Z CHF correlation for system pressure was not presented appropriately in the original submittal. Additional information supplied by DPC corrected this deficiency, and a revised version of Table D-2 is included in the response to the RAI (see Reference 1). This table shows that the design limit DNBR of 1.199 is applicable to conditions between 700 and 1000 psia. Below 700 psia, the design limit DNBR is 1.59. In addition, the response states that if a statepoint with pressure below 1000 psia is encountered in an SCD analysis for Oconee, the applicable design limit DNBR will be used and the impact of the higher correlation standard deviation on the statistical design limit will be calculated. If the SDL for the new statepoint is greater than the licensing limit, the higher SDL will be used when analyzing the lower pressure conditions. This procedure is in accordance with the approved methodology, as described in DPC-NE-2005P-A.

RECOMMENDATIONS

The plant specific data for Oconee with Mark-B11 fuel and for transition cores containing Mark-B10 and Mark-B11 fuel is appropriate for use in the SCD analysis, based on the justifications provided in the DPC response to the RAI (see Reference 1). This includes the corrections to Table D-2 and D-4 provided in the DPC response to the RAI. However, approval of these parameters for Oconee with Mark-B11 fuel does not constitute generic approval of all matters in Appendix D pertaining to the SCD analysis. Specifically,

- Omission of the parameter Fq" from the SCD analysis of the Oconee plant with a new fuel design must be justified for each particular case. Acceptance of its omission in the case of Mark-B11 fuel does not constitute a general approval of its removal from the parameters to be considered in this methodology.

- The applicability of a particular CHF correlation to mixed core geometries is an issue that must be examined for each transition to new fuel, to determine if the mixed core non-uniformities take the local hot channel conditions outside the range of applicability of the CHF correlation.

The methodology requires that the approved CHF correlation for a given fuel design must be used in the SCD analysis for the Oconee plant. As of this writing, the proposed CHF correlation for Mark B-11 fuel (the BWU-Z correlation, submitted as Appendix E in Addendum 1 of BAW-10199P) is under review and has not yet been approved by the NRC. Any changes to the CHF correlation or restrictions in its application as a result of the NRC review must be evaluated for effects on the application of the correlation to Mark-B11 fuel. If the CHF correlation range of applicability is changed, the SCD analysis must be revised in accordance with the modification, and the correlation must not be used outside the parameter range specified in the safety evaluation report (SER) for application to Mark-B11 fuel.

Duke Energy

M. S. Tuckman
Executive Vice President
Nuclear Generation

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May 13, 1999

U. S. Nuclear Regulatory Commission
Washington D. C. 20555-0001

ATTENTION: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Duke Commitment to Conditions of SER and
Clarification of Topical Report DPC-NE-2005
Revision Level

Duke Energy Corporation Topical Report DPC-NE-2005, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," was submitted to NRC in September 1992; approval was granted in February 1995. This initial revision included Appendices A and B, which contained Oconee and McGuire/Catawba plant specific data. Subsequent to Rev 0, Duke submitted Appendix C on April 26, 1996 requesting approval for applying the BWU-Z CHF correlation for analyses of the McGuire and Catawba reactor cores with MkBW fuel. Appendix C contained McGuire/Catawba plant specific data for MkBW fuel using the new CHF correlation, BWU-Z. Appendix C was approved on November 7, 1996. Duke placed the November 7, 1996 NRC Safety Evaluation and Appendix C in the back of DPC-NE-2005 and entitled this report DPC-NE-2005P-A, Rev 1. Rev 1 contains no unreviewed technical information. It simply places previously NRC approved documents DPC-NE-2005, Rev. 0 and Appendix C into the same report.

Within this letter, Duke makes the following commitment:

Following NRC's approval of Appendix D (which was submitted in a Duke letter to the NRC dated April 22, 1997) Duke will incorporate the NRC's Safety Evaluation and Appendix D into DPC-NE-2005P-A, Rev. 1 and at this time change the revision level to DPC-NE-2005P-A, Rev 2. No other technical changes will be made.

Further, Duke Energy Corporation accepts the following NRC specified conditions applicable to use of the BWU-Z critical heat flux correlation for Mark B11 fuel in the Oconee reactors:

(1) Omission of the parameter "Fq" from the SCD analysis of the Oconee plant with a new fuel design must be justified for each particular case. Acceptance of its omission in the case of Mark-B11 fuel does not constitute a general approval of its removal from the parameters to be considered in this methodology.

(2) The applicability of a CHF correlation to mixed core geometries is an issue that must be examined for each transition to new fuel to determine if the mixed core non-uniformities take the local hot channel conditions outside the range of applicability of CHF correlation.

(3) The SCD analysis should be revised as needed to reflect the modification if Mark-B11 CHF correlation range of applicability is changed.

If there are any questions, or additional information required, please call R. M. Gribble at (704) 382-6160 or K. R. Epperson at (704) 382-6785.

M. S. Tuckman

M. S. Tuckman

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SER

Revision 1

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

November 7, 1996

RECEIVED
NOV 14 1996
REGULATORY COMPLIANCE

Mr. M. S. Tuckman
Senior Vice President
Nuclear Generation
Duke Power Company
P. O. Box 1006
Charlotte, NC 28201

SUBJECT: SAFETY EVALUATION ON THE USE OF THE BWU-Z CRITICAL HEAT FLUX CORRELATION FOR MCGUIRE NUCLEAR STATION, UNITS 1 AND 2; AND CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M95267, M95268 AND M95333, M95334)

Dear Mr. Tuckman:

By letters dated October 13 and December 4, 1995, as supplemented by letters dated April 26 and September 5, 1996, Duke Power Company requested approval for applying the BWU-Z critical heat flux (CHF) correlation for analyses of the McGuire and Catawba reactor cores with Mark-BW 17x17 type fuel. The BWU-Z CHF correlation for the Mark-BW 17x17 type fuel is one of the three applications stated in Babcock and Wilcox Fuel Company's (BWFC's) (now Framatome Cogema Fuels) Topical Report BAW-10199P, "The BWU CHF Correlations." This topical report was reviewed and approved by the NRC by letter dated April 5, 1996.

Based on its review, the staff finds the proposed application of the BWU-Z CHF correlation for the McGuire and Catawba Mark-BW 17x17 type fuel acceptable. Our safety evaluation, which provides the results of the review, is enclosed.

Sincerely,

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369, 50-370
50-413, and 50-414

Enclosure: Safety Evaluation

cc w/encl: See next page

Duke Power Company

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letters dated October 13, 1995 (Reference 1) and December 4, 1995 (Reference 2), as supplemented by letters dated April 26, 1996 (Reference 3) and September 5, 1996 (Reference 4), Duke Power Company (DPC or the licensee) requested the use of the BWU-Z critical heat flux (CHF) correlation for analyses of the McGuire and Catawba reactor cores, which consist of a full core of Mark-BW 17x17 type fuel assemblies.

2.0 DISCUSSION/EVALUATION

The licensee submitted Appendix C to DPC-NE-2005P-A to support plant-specific applications to the reload analyses for the McGuire and Catawba plants. Specifically, Appendix C contains the plant-specific data and limits with Mark-BW 17x17 type fuel using the BWU-Z form of the BWU CHF correlation, the VIPRE-01 thermal-hydraulic computer code (Reference 6), and Duke Power Company thermal-hydraulic (T-H) statistical core design (SCD) methodology (Reference 7). The licensee stated that the BWU-Z form of the BWU correlation used in the analyses for the McGuire and Catawba units is exactly the same as the correlation used in BAW-10199P (Reference 5).

In addition, the licensee used the approved method as described in Reference 7 regarding the statepoint propagation. In its calculation of the statistical limit, the licensee increased the number of cases from 3,000 to 5,000 per statepoint. The licensee stated that increasing the number of cases provided higher confidence of defining the bounding behavior and reducing the multipliers. The 5,000-case number was selected due to a balance between computer resources required for the calculation and the reduction in statistical uncertainty to determine a conservative Statistical Design Limit (SDL).

The maximum statepoint statistical value for departure from nucleate boiling ratio (DNBR) for the 5,000-case propagation is given in Table C-4 of Reference 3. This table also contains the values where case propagation is

less than the 5,000-case propagation. The 5,000-case value will be used in analyses with the BWU-Z form of the BWU CHF correlation for Mark-BW 17xI7 type fuel at McGuire and Catawba.

The statistical design limit given in Table C-4 is applicable to this analysis only when all statepoint parameters fall within the McGuire/Catawba key parameter ranges given in Table C-5 of Reference 3.

DPC has also used the VIPRE-01 thermal-hydraulic computer code (Reference 6) to calculate the measured-to-predicted (M/P) CHF ratios with respect to mass velocity, pressure, or thermodynamic quality. The results show that the average M/P value and the data standard deviation are within 1% of the values reported in BWU CHF correlation (Reference 5).

A comparison between the BWU-Z ranges of applicability for Mark-BW 17xI7 type fuel database given in Table 4-1 of Reference 5 and the parameter ranges provided in Table C-1 of Reference 3 shows a 0.01 difference in design limit DNBR using the LYNX and the VIPRE-01 code (1.19 design limit DNBR resulted from the LYNX code versus 1.18 design limit DNBR resulted from VIPRE-01 code). However, DPC will use the larger of the two non-statistical correlation limits.

The staff reviewed the submitals provided by DPC (Reference 1 through Reference 4), and found that the proposed use of BWU-Z CHF correlation is acceptable for use at the McGuire and Catawba plants. This conclusion is based on core analyses that (1) both plants have a full homogeneous core of Mark-BW 17 x 17 type fuel assemblies for upcoming reloads, (2) NRC-approved methodologies (T-H SCD, VIPRE-01, and BWU-Z CHF) are used, (3) the larger of the two correlation limits (VIPRE-01 or LYNX) will be used for non-SCD analyses, and (4) the conservative result from the 5,000-case propagation will be used for SCD analyses.

3.0 CONCLUSION

Based on the above discussions, the staff concludes that the proposed use of BWU-Z critical heat flux correlation for McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2, acceptable.

4.0 REFERENCES

1. Letter from M. S. Tuckman to USNRC requesting review the use of the BWU-Z critical heat flux correlation, dated October 13, 1995.

2. Letter from M. S. Tuckman to USNRC discussing Duke Power Company intent to use of the BWU-Z critical heat flux correlation, dated December 4, 1995.

3. Letter from M. S. Tuckman to USNRC submitting the Appendix to DPC-NE-2005P-A, "McGuire/Catawba Plant Specific Data, Mark-BW Fuel BWU-Z Critical Heat Flux Correlation," dated April 26, 1996.

4. Letter from M. S. Tuckman to USNRC responding to the USNRC's Request for Additional Information regarding Appendix C to DPC-NE-2005P-A, dated September 5, 1996.

5. BAW-I0199P, The BWU Critical Heat Flux Correlations, BWFC, November 1994 (Approved by letter from R. C. Jones to J. H. Taylor, dated April 5, 1996).

6. DPC-NE-2004P-A, Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, December 1991.

7. DPC-NE-2005P-A, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, February 1995.

Principal Contributor: T. Huang

Date: November 7, 1996

SER and TER

Revision 0

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

February 24, 1995

Mr. H. B. Tucker
Senior Vice President
Duke Power Company
P.O. Box 1006
Charlotte, NC 28201-1006

Dear Mr. Tucker:

SUBJECT: ACCEPTANCE FOR REFERENCING OF THE MODIFIED LICENSING TOPICAL REPORT, DPC-NE-2005P, "THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY" (TAC NO. M85181)

The staff has completed its review of the subject topical report submitted by the Duke Power Company (DPC) by letter dated September 28, 1992. The report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

The staff does not intend to repeat its review of the matters described in the report and found acceptable when the report is referenced in a license application, except to ensure that the material presented is applicable to the specific plant involved. Staff acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, DPC must publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should NRC criteria or regulations change so that staff conclusions regarding the acceptability of the report are invalidated, DPC will be expected to revise and resubmit their documentation, or to submit justification for continued effective applicability of the topical report without revision of their documentation.

Sincerely,

Robert Jones for

Gary M. Holahan, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure: NRC Evaluation

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

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY

TOPICAL REPORT DPC-NE-2005P

FOR

THE DUKE POWER COMPANY

1.0 INTRODUCTION

By letter dated September 28, 1992 (Ref. 1), the Duke Power Company (DPC) submitted for staff review and approval a report for use in core thermal-hydraulic analysis. DPC submitted additional information on September 29, 1993 (Ref. 2) and again on February 15, 1994 (Ref. 3). This topical report and the supplemental submittals document the development of core thermal-hydraulic analysis based upon the statistical core design (SCD) methodology using the VIPRE-01 computer code (Ref.4) for the DPC plants: McGuire, Catawba, and Oconee nuclear stations.

The SCD method is a thermal-hydraulic analysis technique which computes departure from nucleate boiling (DNB) margin by statistically combining core and fuel bundle uncertainties. The submittal provides a description and justification for applying uncertainties to the DPC DNB ratio (DNBR) limit calculations using a statistical rather than a deterministic (traditional) method. In 1991, NRC, as part of a reload review, approved limited application of the SCD methodology described in References 5 and 6 for McGuire and Catawba applications. By submitting this topical report, DPC proposes to extend the use of this methodology to all DPC plants for the DNB analysis.

The objective of the subject topical report, therefore, is twofold: (1) to formally present a description of the DPC SCD methodology and (2) to justify its use for all DPC plants. In addition, DPC presented its rationale for setting two separate statistical design limits. The underlying core thermal-

hydraulic methodology based upon the use of VIPRE-01 was approved for all DPC plants (Refs. 5 and 7).

The SCD methodology presented in the topical report is discussed in generic terms in this review. Where applicable, plant-specific features are discussed separately.

2.0 STAFF EVALUATION

The review of "Thermal/Hydraulic Statistical Core Design Methodology," report DPC-NE-2005P, was performed with technical assistance from International Technical Services (ITS). The ITS review findings are contained in the Technical Evaluation Report (TER) which is attached to this safety evaluation report. The staff has reviewed the TER and has concurred with all its findings.

The traditional method for accounting for design and modeling uncertainties that enter into the determination of a DNBR assumes that key input parameters to the core thermal-hydraulic code are simultaneously at their worst level of uncertainty. The currently licensed SCD methodology for McGuire and Catawba assumes that, while the input parameters are occasionally at their worst case values, the input uncertainties are independent and it is highly unlikely that all the input parameters will take on their worst-case values simultaneously. Therefore, the application of the SCD method differs from the deterministic techniques in that the DNBR limit is obtained from statistical analysis of a series of computations as a result of propagation of uncertainties about a statepoint and associated distribution of the DNBR values. DPC has applied the SCD method to simulate the direct computation of DNBR with VIPRE-01.

The TER discusses the DPC VIPRE methodology, the current and revised SDC methodology, the selection of key parameters and uncertainties, propagation of uncertainties, calculation of the statistical design limit (SDL) for DNBR, flexibility of the methodology, and the statistical DNB behavior and use of two DNBR limits.

During the review of DPC-NE-2005P, questions were raised on the use of two DNBR limits. However, the responses were not detailed enough to supply the information needed for resolving the possible use of two DNBR limits. Therefore, the use of the SCD methodology is approved now for only the single, most-conservative DNBR limit.

3.0 CONCLUSION

The staff has reviewed the subject topical report together with the DPC responses and has found them to be acceptable with respect to documentation of the statistical core design methodology using the VIPRE computer code subject to the following restrictions:

1. The statistical core design (SCD) methodology developed by DPC, 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 committed in their topical report that its use of specific uncertainties and distributions will be justified on a plant specific basis, and also that its selection of statepoints used for generating the statistical design limit will be justified to be appropriate. This methodology is approved only for use in DPC plants.

2. Of the two DNBR limits, only the use of the single, most-Conservative DNBR limit is approved.

4.0 REFERENCES

1. Letter from H. B. Tucker (DPC) to USNRC, submitting "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology! II DPC-NE-2005P, September 28, 1992.

2. Letter from H. B. Tucker (DPC) to USNRC, submitting "Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005," September 29, 1993.

3. Letter from H. B. Tucker (DPC) to USNRC, submitting "Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005," February 19, 1994.

4. Electric Power Research Institute, "Acceptance for Referencing of Licensing Topical Report VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM Vols. 1-4," May 1, 1986.

5. Duke Power Company, "McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2004, December 1988.

6. Letter from H. B. Tucker (DPC) to USNRC, submitting "Handouts Presented in the October 7 & 8, 1991 Meeting with NRC Staff and Contract Reviewers," October 16, 1991.

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ITS/NRC/94-3

TECHNICAL EVALUATION:
THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY
DPC-NE-2005P
FOR
DUKE POWER COMPANY

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Prepared for
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under NRC Contract No. NRC-03-90-027
FIN No. L1318

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TOPICAL REPORT DPC-NE-2005P
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1.0 INTRODUCTION

DPC-NE-2005P, dated September 1992 (Ref. 1) was submitted by Duke Power Company (DPC) for NRC review and approval. Additional information was submitted on September 29, 1993 (Ref. 2) and on February 19, 1994 (Ref. 3). This topical report and the supplemental submittals document the development of core thermal-hydraulic analysis based upon statistical core design (SCD) methodology using the VIPRE-01 computer code (Ref. 4) for the DPC plants; McGuire, Catawba (M/C) and Oconee Nuclear Stations.

The SCD method is a thermal-hydraulic analysis technique which computes DNB margin by statistically combining core and fuel bundle uncertainties. The submittal provides a description and justification for applying uncertainties to the DPC DNBR limit calculations using a statistical rather than a deterministic (traditional) method. In 1991 NRC, as part of a reload review, approved limited application of the SCD methodology described in References 5 and 6 for M/C applications. By submitting this topical report DPC proposes to extend the use of this methodology for the ONB analysis of all DPC plants.

The objective of the subject topical report, therefore, is twofold: (i) to formally present a description of the DPC SCD methodology; and (ii) to justify its use for all DPC plants. In addition, DPC presented their rationale for setting two separate statistical design limits. The underlying core thermal-hydraulic methodology based upon the use of VIPRE-01 was approved for all DPC plants (Refs. 5 and 7).

The SCD methodology presented in the topical report is discussed in generic terms in this review. Where applicable, plant specific features are discussed separately.

2.0 SUMMARY OF TOPICAL REPORT and SUPPLEMENTS

The topical report DPC-NE-2005 and its associated submittal document descriptions of DPC's VIPRE-01 based statistical core design (SCD) methodology for all of DPC's nuclear stations. The SCD methodology described in the topical has been approved as part of another review in a limited scope. The submittal formalizes the documentation of methodology description and justification for applying uncertainties to the DPC DNBR limit calculations using a statistical rather than a deterministic (traditional) method, since DPC proposes to extend the application of this methodology to

DNB analysis of all DPC plants.

In addition, DPC presented its rationale, and limited justification, for setting two separate statistical design limits due to sensitivity of DNB to the axial power distributions.

3.0 DPC Statistical Core Design Methodology

The traditional method for accounting for design and modeling uncertainties that enter into the determination of a DNBR assumes that key input parameters to the core thermal-hydraulic code are simultaneously at their worst level of uncertainty. The currently licensed SCD methodology for McGuire and Catawba assumes that, while the input parameters are occasionally at their worst case values, the input uncertainties are independent and it is highly unlikely that all the input parameters will take on their worst case values simultaneously. Therefore, the application of the SCD method differs from the deterministic techniques in that the thermal-hydraulic limit analyses are performed by statistical analysis of a series of computations as a result of propagation of uncertainties about a statepoint and associated distribution of the DNBR values. DPC has applied the SCD method to simulate the direct computation of DNBR with VIPRE-01.

3.1 DPC's VIPRE Methodology

DPC has, in place, NRC approved DNB methodology using the VIPRE-01 computer code for all DPC plants. Both the current and revised SCD methodologies are based on use of such NRC approved VIPRE methodology.

3.2 Current Methodology

The current DPC SCD methodology, based upon the B&W SCD method, relies upon the use of the response surface model (RSM) to evaluate the impact of uncertainties associated with each of the key parameters upon the DNB behavior. Therefore, the range of applicability of the SCD method (therefore the RSM) is limited by the range of values from which the composite design points, used to determine the RSM equation, are selected.

In order to overcome the main limitation of the current SCD methodology with respect to the statepoints which fall outside of the SCD range but which must nevertheless be analyzed for certain transients, DPC developed a simplified method which used VIPRE-01 directly and avoided use of the RSM.

The simplified method bypasses the RSM by directly computing DNBR with the VIPRE-01 code based on the values for the key variables generated by the propagation of uncertainties through the use of the Monte Carlo method. An SCD limit is determined for each case as before and compared against the SDL.

3.3 Revised Methodology

The revised methodology, an extension of the simplified method, is similar to other SCD methodologies in that (1) key parameters are selected, (2) their associated uncertainties are propagated about a statepoint and (3) a large

number of DNBR's are calculated. However, with this methodology the intermediate step of developing the RSM is eliminated. Instead, statistical behavior at a statepoint is evaluated by observing the distribution of the DNBR values and the mean and standard deviation of DNB for the given conditions computed by use of a Monte Carlo method for selection of values of the independent variable. All DNBR calculations are performed directly by the use of the thermal-hydraulic code VIPRE.

This is an advantage since the applicability issue with the previous method is eliminated. Further, if an assumed uncertainty should become non-bounding, the limiting statepoint can be re-evaluated to determine the impact of the changed parameter on the SDL.

3.3.1 Selection of Key Parameters

The key parameters (including reactor power, core flowrate, core exit pressure, core inlet temperature, radial power distribution and axial peak magnitude and location) which significantly impact the calculation of DNBR used in the revised methodology are the same as those used in the previous SCD methodology.

As DPC stated, these key parameters associated with DNBR are generic to US PWRs and are independent of reactor design. Plant specific information determines the uncertainties associated with each parameter.

3.3.2 Selection of Uncertainties

As in the previous SCD formulation, in order to statistically combine the effect of the uncertainties of the parameters, DPC determined the uncertainties, uncertainty distributions and the uncertainty standard deviations. An uncertainty distribution is established for each of the seven variables with the nominal state conditions as the center. DPC's rationale for assignment of uncertainty distribution was that a normal distribution was assumed when the uncertainty was due either to measurement uncertainty or a known statistical uncertainty distribution. Whenever such assumption could not be reasonably made, DPC chose the conservative approach of assuming a uniform distribution with estimated reasonable upper and lower bounds.

In addition to the seven variables related to the core and fuel conditions, two other variables related to the analysis method are assumed to impact computation of DNBR; code/model uncertainty and CHF correlation uncertainty. The uncertainties associated with the code/model allows for uncertainties due to the thermal hydraulic code and VIPRE core models.

The licensee stated in the topical report that the uncertainties and distributions will be justified on a plant-specific basis in the reload report for the first application of this methodology.

3.3.3 Propagation of Uncertainties

In order to combine the uncertainties to compute an overall DNBR uncertainty, a Monte Carlo method analysis is performed using the distribution of

uncertainties defined with each variables. A Monte-Carlo computation is used to select sets of values at random (weighted by the distribution functions) about a statepoint of interest selected from a list of statepoints which form the basis for the statistical design limit.

3.3.4 Calculation of Statistical Design Limit (SDL) for DNBR

Using the Monte-Carlo generated input for the DNB computation, VIPRE-01 is run to calculate the DNBR for each case in a statepoint. The statistical DNB evaluations are performed at two levels. The first level of evaluation taking 500 propagated cases per statepoint is used to determine the DNB behavior over the entire analysis space. The second group of statepoints have 3000 cases each and contains a selected subset of the first group used to evaluate the statistical DNBR values and to improve the associated variance. Statistical analysis is then performed on the set of MDNBRs so generated to determine the statistical design limit (SDL) to replace the traditional DNB limit.

The statistical design limit is determined from the largest coefficient of variation based on the DNBRs computed by the Monte Carlo computations referred to above which avoid DNB at a 95% probability/95% confidence level.

3.3.5 Flexibility of the Methodology

DPC selected a few cases to demonstrate the flexibility of the methodology to changes in any of the key parameter uncertainty distribution, fuel designs or statepoint conditions.

The methodology is direct and general enough to be widely applicable to any fuel or reactor provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the CHF correlation and the uncertainties and associated distributions are reasonable.

3.3.6 Statistical DNB Behavior and Use of Two DNBR Limits

From the cases run with the above described method, DPC observed a dependence of statistically determined DNBRs on the axial location of the power peak, the magnitude of that peak and DNB locat ion. When examining a series of calculations with the peak located in the lower 2/3 of the core and what DPC characterized as "flatter" power profiles, DPC observed a non-linear relationship between the DNBR responses and the axial power peak location and magnitude of the power peak. The study also indicated, for those cases, that the predicted limiting SDLs involved the DNBR occurring at the end of channel. DPC concluded that higher sensitivity of DNBR to certain key parameters was accompanied by higher SDL for the statepoints selected.

Observing the ONBR sensitivity and coupling that to the fact that Chapter-15 type analyses are all performed with power profiles which do not yield DNBRs near the limit, DPC proposed to divide the continuous DNBR space into two regions by the degree of predicted DNBR sensitivity to the axial power distribution: (1) One region contains the DNBRs predicted using the end-of-hannel MONBR limited axial power distributions (flatter and bottom-peaked)

and (2) the other region contains all others. Correspondingly, DPC calculated separate statistical design limits for these regions. In the region associated with the flat and bottom-peaked power distributions, the predicted SDL was higher and a lower SDL value was computed in the other region. The higher limit is the one which would be used if the traditional one-limit methodology is to be approved.

The net result of DPC's proposed double limit would be that the lower limit would be used in all Chapter-15 type analyses. The method for determining the line separating the two areas has a fundamental impact since use of the lower limit results in a large gain in the margin.

However, the space to be divided is a continuous space, and there is a gradual transition from the region of higher DNBR limit to the region of *lower DNB limits*. DPC presented no definitive analytical method for dividing this space. Furthermore, any subdivision could result in reduction of the SDL from the current bounding DNBR limit and the result would be non-conservative.

4.0 <u>CONCLUSION</u>

The subject topical report together with OPC responses were reviewed and found to be acceptable with respect to documentation of the statistical core design methodology using the VIPRE computer code subject to the following limitations and restrictions:

1. The DPC developed statistical core design (SCD) methodology, as described in the submittal, is direct and general enough to be widely applicable to any PWR core, provided that 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, 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.

2. This methodology is approved only for use in DPC plants.

3. For the reasons set forth in Section 3.3.6, it is recommended that (i) of the two DNBR limits, only the use of the single most conservative DNBR limit be approved and (ii) the use of two SDLs not be approved at this time and that it be handled as a separate issue to be resolved in the future.

5.0 <u>REFERENCES</u>

1. "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," DPC-NE-2005P, September 1992.

2. Letter from H.B. Tucker (DPC) to USNRC, Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005," September 29, 1993.

3. Letter from H.B. Tucker (DPC) to USNRC, "Thermal/Hydraulic Statistical

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4. "Acceptance for Referencing of Licensing Topical Report VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM Vols. 1-4," May 1, 1986.

5. "McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2004P-A, December 1991.

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7. "Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE01," DPC-NE-2003P-A, August 1988.

Topical Report Main Body

ABSTRACT

This report presents Duke Energy Carolinas's methodology for performing statistical core thermal-hydraulic analyses. This method uses the models and thermal-hydraulic code currently approved for the Oconee and the McGuire/Catawba Nuclear Stations. The analyses method is based on DNBR limits that statistically account for the effects on DNB of key parameters such as reactor power, temperature, flow, and core power distribution. This report details the methodology development, the application to Duke plants, and the process for future technical enhancements.

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Definitions

Case - A unique set of conditions analyzed by the thermal-hydraulic computer code. These conditions are based on a statepoint and include individual statistical variations of each key parameter.

Design DNBR Limit (DDL) - A numerical DNBR value that includes margin above the statistical design limit and is used for DNBR analyses. The DDL is calculated by multiplying the SDL by a fixed factor such as 1.10.

Key Parameter - A physical parameter that is important to the calculation of DNBR.

Statepoint - A unique set of fluid and reactor conditions evaluated for DNBR performance. These conditions include reactor power, pressure, temperature, coolant flow rate, and a three dimensional nuclear power distribution.

Statistical Core Design (SCD) - An analysis method that statistically combines the effects of all key parameter uncertainties associated with DNB predictions.

Statistical Design Limit (SDL) - A numerical DNBR value resulting from a SCD analysis that ensures, with a 95% probability at a 95% confidence level, DNB will not occur.

Statistical DNBR - The numerical value calculated by the SDL equation for a specific statepoint.

Revision History

Revision 0 is comprised of the main body of the report and Appendix A for Oconee plant specific data (AREVA NP Mark-B10 fuel using the BWC CHF correlation) and Appendix B for McGuire/Catawba plant specific data (Westinghouse OFA or AREVA NP Mark-BW fuel using the BWCMV CHF correlation). The statistical DNBR limits (SDLs) are provided in the respective appendices.

Revision 1 added Appendix C, which documents the BWU-Z CHF correlation and resultant SDL for use with AREVA NP Mark-BW fuel in McGuire/Catawba. In addition to the new CHF correlation, this revision increased the number of cases documented in Revision 0 from 3000 to 5000 cases.

Revision 2 added Appendix D, which documents the BWU-Z CHF correlation and resultant SDL for use with AREVA NP Mark-B11 fuel in Oconee. In addition to the new fuel and CHF correlation, this revision increased the number of cases from 3000 to 5000, increased the turbulent mixing factor, and changed the bulk void and subcooled void models from Levy/Zuber-Findlay to EPRI/EPRI.

Revision 3 added Appendix E, which documents the BWU-Z CHF correlation and resultant SDL for use with AREVA NP Advanced Mark-BW and Mark-BW/MOX1 fuel in McGuire/Catawba. It is noted that the Advanced Mark-BW assembly described in this revision is design used for the MOX lead test assembly (LTA) program. In addition to the new fuel and CHF correlation, this revision increased the number of cases from 5000 to 6000, changed the bulk void and subcooled void models from Levy/Zuber-Findlay to EPRI/EPRI, and used the 8 channel mixed core model to represent the MOX LTA surrounded by Westinghouse Robust Fuel Assembly (RFA) fuel.

Revision 4 added Appendices F and G. Appendix F documents the B-HTP CHF correlation and resultant SDL for use with AREVA NP Mark-B-HTP fuel in Oconee. In addition to the new fuel and CHF correlation, Appendix F changed the turbulent mixing factor to be grid dependent, changed the bulk void and subcooled void models from Levy/Zuber-Findlay to EPRI/EPRI, and used the 9 channel mixed core model to represent Mark-B-HTP fuel surrounded by Mark-B11 and Mark-B-HTP fuel. Also, Appendix F introduced company specific proprietary designations, with "D" being proprietary to Duke Energy and "A" being proprietary to AREVA NP.

Appendix G documents the WRB-2M CHF correlation and resultant SDL for use with Westinghouse Robust Fuel Assembly (RFA) fuel in McGuire/Catawba. Original approval for applying the SCD methodology to RFA fuel and WRB-2M was obtained upon approval of DPC-NE-2009, which was the methodology for transitioning to RFA fuel. The content in Appendix G was consequently approved several years prior to the inclusion of Appendices E and F in this report. The content is subsequently moved to this report for completeness. Several additional changes, relative to the body of this report, were also made. The number of cases was increased to 5000, the LEVY/Zuber-Findlay bulk void and subcooled void models were replaced with the EPRI/EPRI models, the $F\Delta h$ was increased as was the axial peak. Furthermore, the AREVA NP BWU-N CHF correlation was used in the first non-mixing vane grid span instead of the WRB-2M CHF correlation.

Revision 4a includes editorial updates that do not change the technical content of the report.

1.0 INTRODUCTION

The thermal-hydraulic design methodology accounts for the effects on DNB of the uncertainties of key parameters such as power, pressure, temperature and flow. Statistically combining these effects yields a better quantification of the DNB margin which, in turn, enhances core reload design flexibility. This report details the thermal-hydraulic statistical core design methodology developed by Duke Energy Carolinas for application to pressurized water reactors.

Several different statistical DNB analysis methods have been approved and are currently in use by various vendors and utilities. All the methods have slight differences but the major similarity is the basic concept that statistical behavior is defined by the sensitivity of DNB to key parameters and their associated uncertainties. When this relationship is well defined, a high degree of confidence in the applicability of the statistical DNB limit is assured.

1.1 METHODOLOGY

Duke Energy has developed a method to evaluate the statistical behavior of DNBR that uses the VIPRE-01 thermal-hydraulic computer code (Reference 1) to calculate the DNBR values for each set of reactor conditions. Figure 2 shows the flowchart for this approach.

The SCD methodology is identical in most respects to other statistical thermal-hydraulic analysis methodologies. Key DNBR parameters are selected, their associated uncertainties are propagated about a statepoint, and a large number of DNBR's are calculated. The

statistical behavior at that statepoint is evaluated by observing the distribution of the DNBR values and the mean and standard deviation of DNB for the given conditions. This same approach is repeated over a range of statepoints. The Statistical Design Limit (SDL) is based on the largest coefficient of variation and therefore the largest statistical DNBR value for the statepoints considered.

The statistical analysis method described in this report is applied to both the Oconee (Babcock and Wilcox) and McGuire/Catawba (Westinghouse) plant designs. The main body of this report details the specifics of the method and gives typical results. Two Appendices are included that contain plant specific information and results. This is necessary due to the differences in CHF correlations, fuel design, and specific uncertainties for each plant design. Appendix A contains the specific information for Oconee and Appendix B contains the same information for McGuire/Catawba. The plant specific thermal-hydraulic models and computer code configurations described in Reference 2 (DPC-NE-2003P-A) and Reference 3 (DPC-NE-2004P-A) are used in this analysis without modification.

This method of developing an SCD limit provides an accurate representation of statistical DNB behavior because the thermal-hydraulic code is used directly to perform all DNBR calculations. This methodology consists of over 151,000 individual VIPRE-01 cases at various statepoints.

1.2 FUTURE USES

One benefit of this thermal-hydraulic analysis method is the ability to analyze factors outside of the original scope of analysis for a particular plant. This is due to the fact that the thermal-hydraulic code is used directly to determine statistical behavior. For example, if an assumed uncertainty should become non-bounding, the limiting statepoint can be re-evaluated to determine the impact of the changed parameter on the SDL. This method can also be used to evaluate a statepoint outside the range of the original key parameters assumed. If the statepoint statistical DNBR does not exceed the SDL, the statepoint can apply the licensed limit.

If the statepoint statistical DNBR does exceed the limit, appropriate measures, such as increasing the design DNBR limit (DDL) for that statepoint's analyses, can be used to ensure conservative DNBR limits are used. (The design DNBR limit approach is discussed in Section 2.5 of this report). This higher design limit will mean lower allowable radial power distributions for the affected statepoint. The higher limit would apply to all the subsequent analyses performed on that set of conditions. Another alternative to increasing the design DNBR limit is to use the available margin between the existing SDL and design DNBR limits to account for the change.

Secondly, this statistical analysis method shows generic DNB behavior that extends across fuel designs and plant types. The limiting SDL value is primarily affected by the particular Critical Heat Flux (CHF) Correlation used, the fuel assembly design, and the key parameter uncertainties. This allows the methodology to be applied to new or

revised CHF correlations, or new fuel assembly designs, requiring only the submittal of an additional Appendix that provides the same information as included in the two attached.

2.0 STATISTICAL CORE DESIGN METHODOLOGY

The procedure for determining the statistical DNBR limit (SDL) contains four steps:

1. Selection of key parameters
2. Selection of uncertainties
3. Propagation of uncertainties
4. Calculation of the statistical DNBR limit (SDL).

The key parameters associated with DNBR are generic to pressurized water reactors and are independent of reactor design. The important plant specific information is the uncertainties associated with each parameter.

2.1 SELECTION OF KEY PARAMETERS

The key parameters used in this analysis are the parameters which significantly impact the calculation of DNBR and include:

Reactor Power

Core Flow Rate (including effects of core bypass flow)

Core Exit Pressure

Core Inlet Temperature

Radial Power Distribution (including Hot Channel Factors)

Axial Peak Magnitude

Axial Peak Location

These seven parameters are used to set limits when performing reload thermal-hydraulic analyses. A statepoint in this analysis is a defined by a combination of all seven of these parameters.

The range of individual key parameter values in this analysis are based on statepoints that are using or will use the SCD DNB methodology. A majority of the statepoints analyzed have mean Minimum DNBR (MDNBR) values close to the statistical design limit itself. Table 1 shows typical statepoints that form the basis for the statistical design limit {Table 1 in Appendices A & B shows the statepoints analyzed for each plant}. Table 4 in Appendices A & B contains the range of values for each key parameter represented by the analyzed statepoints. The same information is provided in subsequent Appendices, but not necessarily in Tables 1 and 4.

Since this method mechanistically evaluates each statepoint, new or revised statepoints can be easily evaluated in the same manner. If, for example, the plant is uprated to a higher licensed power level or the pressure/temperature points change or a new transient statepoint is calculated, a propagation of the revised conditions about the limiting point would be performed. If the licensed SDL is conservative, no further action would be required. If the statistical DNBR value is higher, appropriate compensatory measures will be applied to ensure the allowable DNB behavior for the statepoint is conservatively bounded.

Duke Energy's reload methodology, described in References 4 and 5, gives special attention to the axial power distribution (axial peak location and magnitude) in determining acceptable DNB performance. The axial peak location and magnitudes evaluated in this analysis are concentrated about a selected region. The axial power distribution area of interest is based on the peak magnitudes and locations that are typically predicted during the standard cycle design process. Figures 3A and 3B show a graphic representation of typical axial peak values (Fz) and locations (Z) calculated by the physics codes. Figure 3A is for Oconee and Figure 3B shows the same data for McGuire and Catawba.

2.2 <u>SELECTION OF UNCERTAINTIES</u>

A statistical core design analysis combines the effects of individual key parameter uncertainties that significantly affect DNB. Typical uncertainties for a reactor design are shown in Table 2 {Table 2 in Appendices A & B shows the plant specific values}. The same information is also included in the subsequent Appendices if not always in Table 2 of those Appendices. Distributions for the uncertainties are assumed to be either normal or uniform. The basis for the type of distribution assumed for each key parameter is included in the Appendices. Two additional uncertainties are included, one for the CHF correlation and one for code/model conservatism. The CHF correlation uncertainty is based on the standard deviation of the correlation data base and accounts for the correlation's uncertainty in DNB predictions. The code/model uncertainty allows for thermal-hydraulic code uncertainties and simplified versus detailed core model differences.

2.3 PROPAGATION OF UNCERTAINTIES

Multiple random cases are generated for each statepoint by independently varying all key parameters according to their associated uncertainty value and distribution. The SAS (Reference 6) statistical computer package random number function generators are used to create the necessary distributions. The key parameter distributions are calculated individually based on the type of uncertainty distribution and uncertainty magnitude.

There are two different types of uncertainties analyzed. The first type, denoted additive, is an uncertainty that has a fixed value. An example of this is the RCS temperature uncertainty of +/- 4 degrees F (see Table 2). The value is the same number of degrees F everywhere it is applied. The second type of uncertainty is called multiplicative and is based on a percentage of the parameter. An example of this is the radial power distribution uncertainty (3.25% in Table 2). Here, the radial peak used in each statepoint has an impact on the magnitude of the uncertainty. This statistical method of application accounts for both the uncertainty magnitude and distribution type (normal or uniform).

A total of either 500 or 3000 (or more) propagated cases (one case being a set of the seven key parameters) are generated for each statepoint. The different propagation sizes are compared to verify that the statistical behavior is consistent between the two levels of analysis and to be confident that the most limiting SDL is determined. Table 3 contains an example of key parameter propagations that together

make up ten DNB cases for a given statepoint. The values were extracted from a typical 500 case propagation.

As stated previously, this analysis method allows for direct evaluation of the impact of increased uncertainties. If an uncertainty value assumed in the original analysis is exceeded in the future, the limiting statepoint can be re-analyzed with the changed value. If the statepoint statistical DNBR does not increase above the licensed limit, no further action is required. If it does, proper compensatory measures can be applied.

2.4 CALCULATION OF THE STATISTICAL DNBR LIMIT

After the VIPRE-01 code is used to calculate the MDNBR's for each case in a statepoint, the code/model and CHF correlation uncertainties are applied and the coefficient of variation (CV) is calculated. Cases that yield either a MDNBR value of less than 1.0 or that exceed the quality limit of the CHF correlation used are excluded from the data base prior to calculating the coefficient of variation. The distribution of MDNBR's is checked for normality by performing the D'Agostino (or D Prime) test on the final set of MDNBR values for each statepoint. The most limiting CV calculated from all of the statepoints is then used in the calculation of the SDL (see the SDL equation below).

The appropriate Chi Square (Chi^2) and K factor (K) multipliers are determined based on the final number of MDNBR's for each statepoint. The statistical DNBR value for each statepoint is then calculated by the following equation,

$$SDL = 1.0 / \{1.0 - (K * Chi^2 * CV)\}$$

Table 4 contains example results of the mean, standard deviation, coefficient of variation, and the statistical DNBR values calculated for the Table 1 statepoints. {Table 3 in Appendices A & B contains the plant specific data. Subsequent Appendices include the same information but in different table numbers.}

Table 4 contains two groups of statepoints in separate sections. This is because the statistical DNB evaluations in this analysis were completed at two levels. The first level of evaluation (500 propagated cases/statepoint) is used to determine the DNB behavior over the entire analysis space. The intent of the 500 case runs is to determine DNB behavior with respect to axial and radial peaking conditions, core power level, and changes in fluid conditions.

The second group of statepoints have 3000 cases each and are a selected subset of the first group (denoted by -T after the statepoint number). This group is used to determine the SDL of DNB analyses for each reactor type. Figures 4A (500 cases) and 5A (3000 cases) graphically show the results for Oconee at a selected set of fluid conditions. Figure 6A shows the comparisons of the same axial peak locations and magnitudes for different fluid conditions. Figures 4B, 5B, and 6B show the corresponding graphs for the McGuire/Catawba statepoints.

2.4.1 VARIANCE OF STATISTICAL DNB BEHAVIOR

Comparing all these Figures showing the statistical DNBR for [ ] across a range of fluid conditions and for different fuel/reactor types, a significant dependency [ ] is observed. [ ] show a more limiting statistical DNBR behavior than the remaining points. To evaluate this, the sensitivity of DNBR [ ] was evaluated in two manners.

First, the sensitivity of DNB [ ] was determined. This was done by [ ] constant and analyzing [ ] Figure 7A shows the sensitivity of DNB [ ] for the BWC correlation (Oconee). Figure 7B shows the sensitivity for the BWCMV correlation (McGuire/ Catawba).

Two items of interest are displayed in this representation. The first fact is that the slope [ ] Secondly, the slope in this area [ ] on the remainder of the graph. The absolute value of the slope is the important factor in determining the statistical response of a key parameter (slope is the sensitivity of DNBR [ ] This indicates that [

] will have a different statistical behavior than the area where the slope is less steep. Note the agreement between Figures 7A and 7B (different fuel assembly designs and CHF correlations). This consistency continues to affirm that this observation is a mechanistic DNB behavior.

The second sensitivity evaluation varied all key DNB parameters of a statepoint by their uncertainty magnitude and calculated the slope for each (Δ DNBR / Δ parameter). These results are shown in Table 5. This type of analysis shows [

]

Additionally, there is another phenomenon that is also present with [

]

This more limiting statistical behavior has been evaluated for generic applicability and was found to occur for each reactor type and CHF correlation as shown by Figures 4A and 4B. The statistical behavior [

] All these factors point to the conclusion that this more limiting statistical variance [ ] is a generic, mechanistic DNB behavior and as such is applicable to any CHF correlation and core model (Oconee, McGuire, Catawba, etc).

2.4.2 FLEXIBILITY OF THE ANALYSIS METHOD FOR MODIFIED PARAMETER EVALUATIONS

Several different comparisons are included to demonstrate the ability of this method to address changes in core models or uncertainty distributions. Table 6 shows the results of three different evaluations. The first section includes two points that show the results of changing a single key parameter's uncertainty distribution from normal to uniform. Statepoints 33 and 34 from the McGuire/Catawba evaluation were identical in all respects except for the RCS flow distribution. In Statepoint 33, the distribution was normal (same for all other statepoints) and in Statepoint 34 the distribution was changed to uniform. The affects of this single parameter distribution change is readily calculated and shown to be negligible.

The section has two points that show the impact of a VIPRE-01 model change. Statepoints 37 and 38 both have identical conditions and

uncertainties. Statepoint 37 used the eight channel McGuire/Catawba model from Reference 3 while Statepoint 38 used the fourteen channel model from Reference 8. Again, the comparison is easily accomplished and Table 6 shows the difference in the statistical DNBR values.

The third section contains a group of points that shows the comparison between Westinghouse OFA and Babcock Wilcox Mark-BW 17x17 mixing vane fuel. Four statepoints were run with both fuel types at the same fluid and power distribution conditions. The difference between the models is the changed subchannel flow areas, wetted and heated perimeters, gap connections, and grid form loss coefficients to correctly reflect each fuel type. The comparison shows that the OFA fuel model's behavior is the same as the Mark-BW model and the Mark-BW SDL conservatively bounds OFA fuel for McGuire and Catawba analyses.

2.4.3 FUTURE APPLICATIONS OF SCD METHODOLOGY

The fact that this analysis method is direct allows this statistical approach to be applied to any fuel type or reactor using an NRC approved thermal-hydraulic model and CHF correlation. Even if DNB behavior showed a stronger or weaker functionality for a different core design or CHF correlation, this method would correctly reflect this behavior in the statistical design limit or limits determined. If a new CHF correlation is used by Duke or if a different plant is analyzed, an additional Appendix will be submitted to the NRC detailing the model, CHF correlation, uncertainties, and statepoints used to determine the SDL for the plant specified.

2.5 APPLICATION OF THE SCD LIMIT

Since the statistical DNBR behavior demonstrated in this analysis shows [

]

The method for applying [

] The NRC Safety Evaluation Report dated February 24, 1995 did not approve the use of two separate Statistical DNBR limits. Consequently, the largest SDL is applied to all peaking space. Additionally, DNB analyses may be performed using a design DNBR limit (DDL) which includes margin above the statistical design limit.

Should an analysis be performed that uses a new CHF correlation, or for a new fuel design, statepoints will be analyzed to determine the SDL. This information will be reported to the NRC by submitting a new Appendix similar to Appendix A and B.

3.0 CONCLUSIONS

The methodology described in this report shows the major factors affecting statistical DNB behavior are [

]

This analysis method can be used to evaluate new fluid statepoints or revised uncertainties directly to determine the statistical limit. As long as the SDL is not exceeded, the established limits can be applied unmodified. If the statistical DNBR value for the new conditions is higher than the current limit, appropriate compensation measures such as increasing the design DNBR limit for the statepoint or using available margin between the design and statistical limits can be used. These actions penalize the statepoint by reducing the allowable radial peaking to ensure acceptable DNB behavior.

Since Duke's statistical thermal-hydraulic design methodology relies solely on DNB behavior, any PWR facility can be analyzed using this approach with an appropriate core model and bounding uncertainties. Also, new fuel designs or critical heat flux correlations can be evaluated to determine the appropriate SDL. The results of such an analysis would be submitted to the NRC for approval in the form of an additional Appendix that would contain the following:

1) Identification of the plant, fuel type, and CHF correlation with appropriate references to the approved fuel design and CHF correlation topicals.

2) Statement of the thermal-hydraulic code and model used with appropriate references to the approved code topical report.

3) A list of the key parameters, their uncertainty values, and distributions.

4) A list of the statepoints analyzed.

5) The Statistical Design Limits and how they are applied.

Table 7 contains a listing of some anticipated conditions and the corresponding actions.

4.0 SUMMARY

This report describes the analysis method used to determine the statistical core design DNB limit for reactor core thermal-hydraulic analyses. This methodology is used to account for the impact on DNB of the uncertainties of key parameters such as power, pressure, temperature, and core peaking. The methodology determines the statistical behavior of DNBR with respect to all these key parameters for many different statepoints and provides a method of applying the SCD DNB limits derived.

The specific SCD DNB limits for the Oconee and McGuire/Catawba units are stated in the Conclusions section of the attached Appendices.

5.0 REFERENCES

1. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

2. Oconee Nuclear Station Core Thermal-Hydraulic Methodology using VIPRE-01, DPC-NE-2003-PA, Revision 2a, Duke Energy Carolinas, Charlotte, North Carolina, December 2008.

3. McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2004-PA, Revision 2a, Duke Energy Carolinas, Charlotte, North Carolina, December 2008.

4. Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors, DPC-NE-2011-PA, Revision 1a, Duke Energy Carolinas, Charlotte, North Carolina, December 2008.

5. Oconee Nuclear Station Reload Design Methodology Technical Report, NFS-1001A, Revision 5, January 2001.

6. SAS Language Reference, Version 6, First Edition, SAS Institute Incorporated, Cary, North Carolina, 1990.

7. Deleted

8. Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000P-A, Revision 3, Duke Power Company, Charlotte North Carolina, September 2004.

9. BWC Correlation Of Critical Heat Flux, BAW-10143P-A, AREVA NP, Lynchburg, Virginia, April 1985.

10. BWCMV Correlation Of Critical Heat Flux In Mixing Vane Grid Fuel Assemblies, BAW-10159P-A, AREVA NP, Lynchburg, Virginia, May 1986.

TABLE 1. Typical Reactor SCD Statepoints

Stpt #	Power	Pressure	Temperature	Flow	Axial Peak	FΔh
DNB Limit Line Statepoints						
1						
3						
4						
12						
14						
17						
26						
Loss Of RCS Flow Transient Statepoints						
21						
24						
29						
Uncontrolled Bank Withdrawal Transient Statepoint						
33						
Nominal Operating Statepoints						
16						
27						

TABLE 2. Typical Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type of Distribution
Reactor Power	+/- 2% / +/- 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / +/- 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^N_{\Delta H}$		
Measurement	+/- 3.25% / 1.98%	Normal
$F^E_{\Delta H}$	+/- 3.0% / 1.82%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.41% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	+/- 16.78% / 10.2%	Normal
Code/Model	[ ]	Normal

TABLE 3. Typical Monte Carlo Propagation Statepoint Values

(Values After Uncertainty Propagation of Stpt. # 1 from TABLE 1)

Base Statepoint

Case#	Power	Press	Temp	Flow	Fz	Z	FΔh
0	[						]

Propagation

Case#	Power	Press	Temp	Flow	Fz	Z	FΔh
1	[						
50							
100							
150							
200							
250							
300							
350							
400							
450							
500							]

TABLE 4. Example of Typical Statepoint Statistical Results

Section 1 - 500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
DNB Limit Line Statepoints				
1	[			
3				
4				
12				
14				
17				
26				]
Loss Of RCS Flow Transient Statepoints				
21	[			
24				
29				]
Uncontrolled Bank Withdrawal Transient Statepoint				
33	[			]
Nominal Operating Statepoints				
16	[			
27				]

TABLE 4 - continued Example of Typical Statepoint Statistical Results

Section 2 - 3000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
DNB Limit Line Statepoints				
3-T				
4-T				
12-T				
14-T				
Nominal Operating Statepoint				
16-T				

TABLE 5. Individual Key Parameter Slopes At Statepoint Conditions

Key Parameter*	Stpt 6	Stpt 25	Stpt 9	Stpt 21
Power	3.54%	3.51%	2.11%	2.37%
Pressure	0.16%	0.11%	0.08%	0.07%
Temperature	2.83%	2.02%	1.56%	1.42%
Flow	3.41%	2.68%	1.79%	1.46%
FΔH	4.19%	3.37%	2.46%	2.15%
Fz	1.24%	1.00%	0.91%	1.14%
Z	1.04%	0.63%	3.80%	2.39%
Statepoint SDL =	1.385	1.388	1.325	1.323
Axial Power Dist. (Fz @ Z)	1.3@0.2	1.2@0.3	1.3@0.8	1.8@0.2

The statepoints listed above are from the McGuire/Catawba 500 case runs. [

] Statepoints 6 and 25 [

]

Statepoints 9 and 21 [

]

* All values shown are in %DNBR per unit of parameter (Δ DNBR / Δ parameter). For example, the first entry in the table of [ ] means a [ ] DNBR change for every 1% power change.)

Table 6. Uncertainty and Model Changes - Impact On Statistical DNBR Behavior

Uncertainty Distribution Change

The following two statepoints show the change in the statistical behavior for a fixed set of conditions if the RCS flow uncertainty distribution is changed from normal to uniform.

Statepoint #	RCS Flow Uncertainty Dist.	Coefficient Of Variation	Statistical DNBR
33	Normal	0.1244	1.301
34	Uniform	0.1226	1.295

Thermal-Hydraulic Model Detail Change

The following two statepoints show the change in the statistical behavior for a fixed set of conditions using two different VIPRE-01 models.

Statepoint #	McGuire/Catawba VIPRE-01 Model	Coefficient Of Variation	Statistical DNBR
37	8 Channel	0.1244	1.301
38	14 Channel	0.1256	1.305

Minor Fuel Geometry and Design Changes

The following eight statepoints show the change in the statistical behavior for the geometry and form loss coefficient changes between Mark-BW and OFA fuel assemblies for the same fluid and peaking conditions.

MARK-BW			OFA		
Statepoint #	Coefficient Of Variation	Stat. DNBR	Statepoint #	Coefficient Of Variation	Stat. DNBR
6	0.1489	1.385	40	0.1518	1.397
12	0.1279	1.312	41	0.1258	1.306
14	0.1222	1.294	42	0.1212	1.291
16	0.1546	1.404	43	0.1512	1.391

TABLE 7. SDL Evaluation And Re-Submittal Criteria

The following table lists different events or conditions that would require an evaluation of the applicability of an approved SDL and the subsequent actions based on the results of the analysis.

CONDITION	ACTION
Revised uncertainty larger than the limiting value used in the original analysis.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
Revised uncertainty distribution.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
New statepoint.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
Minor modifications to the current fuel design.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
A modified CHF correlation.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
Change to a new fuel design/fuel type.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
A new CHF correlation.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
New Thermal-Hydraulic Code.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.

FIGURE 1

Deleted

FIGURE 2

SCD FLOWCHART

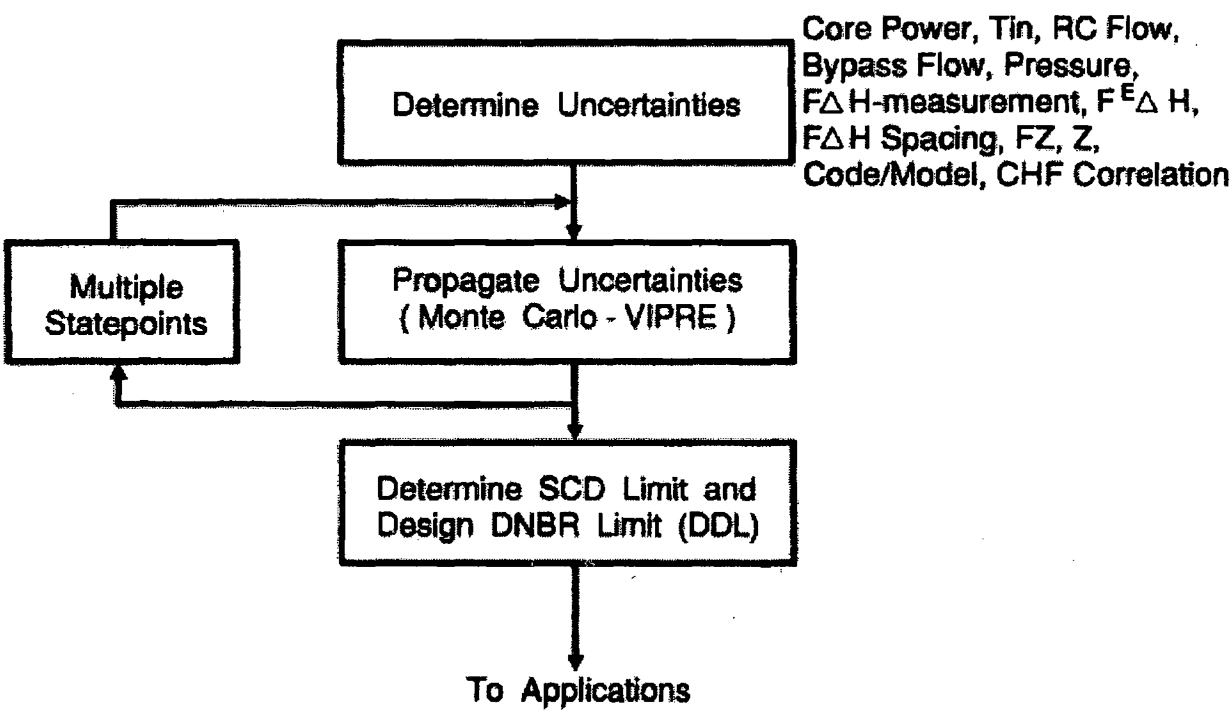

FIGURE 3A Oconee Physics Code Axial Power Distributions (Peak Magnitude and Locations)

FIGURE 3B M/C Physics Code Axial Power Distributions (Peak Magnitude and Locations)

FIGURE 4A
Oconee SDL Distribution At Constant Conditions, BWC
500 Case Propagation

FIGURE 4B
M/C SDL Distribution At Constant Conditions, BWCMV)
500 Case Propagation

FIGURE 5A
Oconee SDL's For 3000 Case Statepoints, BWC

FIGURE 5B
M/C SDL's For 3000 Case Statepoints, BWCMV

FIGURE 6A
Oconee SDL's For Various Conditions

FIGURE 6B
M/C SDL's For Various Conditions

FIGURE 7A Sensitivity of DNBR [
] BWC

FIGURE 7B Sensitivity of DNBR [

] BWCMV

FIGURE 8A [

] BWC

FIGURE 8B [

] BWCMV

FIGURE 9

Example [ ]

Application

APPENDIX A

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APPENDIX A

Oconee Plant Specific Data

This Appendix contains the plant specific data and limits for the Oconee Nuclear Station. The thermal hydraulic statistical core design was performed as described in the main body of this report.

Plant Specific Data

This analysis is for the Oconee plant (two loop B&W PWR) with Mark-B fuel assemblies detailed in Reference 2. The BWC critical heat flux correlation described in Reference 9 is used.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal hydraulic computer code (Reference 1) and the Oconee eight channel model approved in Reference 2 are used in this analysis.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table A-1.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table A-2. The range of key parameter values is listed on Table A-4.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table A-3. Section 1 of Table A-3 contains the 500 case runs and Section 2 contains the 3000 case runs. All statepoint SDL values reported in this analysis are normally distributed. The statistical design limit using the BWC CHF correlation for Oconee was determined to be [

] Figure A-1 graphically depicts the application

[ ]

TABLE A-1. Oconee SCD Statepoints

TABLE A-1 Continued Oconee SCD Statepoints

NOTES:

- 100% design flow is equal to four times 88,000 gpm/pump or 352,000 gpm total system flow.

- 100% Full Power (FP) is equal to 2568 MWth.

(1) Outlet temperature equals 581.0 °F.

(2)

TABLE A-2. Oconee Statistically Treated Uncertainties

Parameter	Type of Uncertainty	Type of Distribution	Uncertainty	Standard Deviation
Reactor System				
Power	Measurement	Normal	± 2.0 %FP	± 1.0 %FP
Temperature	Measurement	Normal	± 2.0 °F	± 1.0 °F
Pressure	Measurement	Normal	± 30.0 psi	± 15.0 psi
Core Flow	Measurement	Normal	± 2.0 %design	± 1.0 %design
Nuclear				
FΔh	Calculation	Normal	-----------	± 2.84 %
F_z	Calculation	Normal	-----------	± 2.91 %
z	Calculation	Uniform	± 6.0 in.	----------
Fq"	Calculation	Normal	[	
Fq	Calculation	Normal		]
Hot Channel Flow Area	Measurement	Uniform	[ ]	----------
DNBR	Correlation	Normal	-----------	± 8.88 %
DNBR	Code	Normal	[	]

TABLE A-2 Continued Oconee Statistically Treated Uncertainties

Parameter	Justification
System Pressure	This uncertainty accounts for random uncertainties in various instrumentation components. Since the random uncertainties are normally distributed, the square root of the sum of the squares (SRSS) that results in the pressure uncertainty is also normally distributed.
Inlet Temperature	Same approach as System Pressure uncertainty.
Core Power	The core power uncertainty was calculated by statistically combining the various random uncertainties associated with the measurement of core power. Since the random uncertainties are normally distributed, the srss of them that results in the core power uncertainty is also normally distributed.
Core Flow	Same approach as Core Power uncertainty.
Radial Power, $F\Delta h$	This uncertainty accounts for the error associated in the physics code's calculation of radial assembly power and the measurement of the assembly power. This uncertainty distribution is normal.
Axial Peak Power, Fz	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty is normally distributed.
Axial Peak Location, z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.

TABLE A-2 Continued Oconee Statistically Treated Uncertainties

Parameter	Justification
Local Heat Flux HCF, Fq"	This uncertainty accounts for the decrease in DNBR at the point of MDNBR due to engineering tolerances. This uncertainty is also increased to account for flux depression at the spacer grids. The uncertainty is normally distributed and conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
Rod Power HCF, Fq	This uncertainty accounts for the increase in rod power due to manufacturing tolerances. The uncertainty in calculating the peak pin from assembly radial peak is also statistically combined with the manufacturing tolerance uncertainty to arrive at the correct value. The uncertainty is normally distributed and conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
Hot Channel Flow Area	This uncertainty accounts for manufacturing variations in the instrument guide tube sub-channel flow area. This uncertainty is uniformly distributed and is conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
DNBR - Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty is normally distributed.
DNBR - Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatism's. This uncertainty also accounts for the small DNB prediction differences between various model sizes. This uncertainty is normally distributed.

TABLE A-3. Oconee Statepoint Statistical Results

500 Case Runs

Statepoint #
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
32
33
34
35
36
37
38
39
40
41

TABLE A-3 Continued Oconee Statepoint Statistical Results

500 Case Runs

Statepoint #
42
43
44
45
46
47
48
49
50
52
53
54
55
56
57
58
59
60
61
62
63
64
65
66
67
68
69
70
71
72
73
74
75
76
77
78
79
80
81
82

TABLE A-3 continued Oconee Statepoint Statistical Results

3000 Case Runs

Statepoint #
2-T
3-T
6-T
20-T
24-T
26-T
29-T
34-T
39-T
41-T
44-T
53-T
54-T
59-T
62-T
63-T
68-T
72-T
78-T

TABLE A-4 Oconee Key Parameter Ranges

Parameter	Maximum	Minimum

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

APPENDIX B

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APPENDIX B

McGuire/Catawba Plant Specific Data

This Appendix contains the plant specific data and limits for the McGuire and Catawba Nuclear Stations. The thermal hydraulic statistical core design was performed as described in the main body of this report.

Plant Specific Data

This analysis is for the McGuire and Catawba plants (four loop Westinghouse PWR's) with either Mark-BW or Optimized Fuel Assemblies as described in Reference 3. The BWCMV critical heat flux correlation described in Reference 10 is used for analyzing both fuel types.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal hydraulic computer code (Reference 1) and the McGuire/Catawba eight channel model approved in Reference 3 are used in this analysis.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table B-1.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table B-2. The range of key parameter values is listed on Table B-4.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table B-3. Section 1 of Table B-3 contains the 500 case runs and Section 2 contains the 3000 case runs. All statepoint SDL values listed in this analysis are normally distributed. The statistical design limit using the BWCMV CHF correlation for McGuire/Catawba was determined to be [

] Figure B-1 graphically depicts the application [ ]

TABLE B-1. McGuire/Catawba SCD Statepoints

Stpt #	Power	Pressure	Temperature	Flow	Axial Peak	Radial Peak
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						

TABLE B-1 - Continued McGuire/Catawba SCD Statepoints

Stpt #	Power	Pressure	Temperature	Flow	Axial Peak	Radial Peak
24						
25						
26						
27						
28						
29						
30						
31						
32						
33						
34						
35						
36						
37						
38						
39						
40						
41						
42						
43						

TABLE B-2. McGuire/Catawba Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type of Distribution
Core Power	+/- 2% / +/- 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / +/- 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^N_{\Delta H}$		
Measurement	+/- 3.25% / 1.98%	Normal
$F^E_{\Delta H}$	+/- 3.0% / 1.82%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.41% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	+/- 16.78% / 10.2%	Normal
Code/Model	[ ]	Normal

TABLE B-2 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
Core Power	The core power uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from normally distributed random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc combined by the square root sum of squares method (SRSS). Since the uncertainty is calculated from normally distributed values, the parameter distribution is also normal.
Core Flow Measurement	Same approach as Core Power.
Bypass Flow	The core bypass flow is the parallel core flow paths in the reactor vessel (guide thimble cooling flow, head cooling flow, fuel assembly/baffle gap leakage, and hot leg outlet nozzle gap leakage) and is dependent on the driving pressure drop. Parameterizations of the key factors that control ΔP, dimensions, loss coefficient correlations, and the effect of the uncertainty in the driving ΔP on the flow rate in each flow path, was performed. The dimensional tolerance changes were combined with the SRSS method and the loss coefficient and driving ΔP uncertainties were conservatively added to obtain the combined uncertainty. This uncertainty was conservatively applied with a uniform distribution.
Pressure	The pressure uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc combined by the square root sum of squares method. The uncertainty distribution was conservatively applied as uniform.
Temperature	Same approach as Pressure.

TABLE B-2 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
$F^N_{\Delta H}$ Measurement	This uncertainty is the measurement uncertainty for the movable incore instruments. A measurement uncertainty can arise from instrumentation drift or reproducibility error, integration and location error, error associated with the burnup history of the core, and the error associated with the conversion of instrument readings to rod power. The uncertainty distribution is normal.
$F^E_{\Delta H}$	This uncertainty accounts for the manufacturing variations in the variables affecting the heat generation rate along the flow channel. This conservatively accounts for possible variations in the pellet diameter, density, and U_{235} enrichment. This uncertainty distribution is normal and was conservatively applied as one-sided in the analysis to ensure the MDNBR channel location was consistent for all cases.
Spacing	This uncertainty accounts for the effect on peaking of reduced hot channel flow area and spacing between assemblies. The power peaking gradient becomes steeper across the assembly due to reduced flow area and spacing. This uncertainty distribution is normal and was conservatively applied as one-sided to ensure consistent MDNBR channel location.
F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty distribution is applied as normal.
Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.
DNBR Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normal.

TABLE B-2 Continued   McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
DNBR	
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is applied as normal.

TABLE B-3. McGuire/Catawba Statepoint Statistical Results

500 Case Runs

Statepoint #
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
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25
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29
30
31
32
33
34
35
36
37
38
39
40
41
42
43

TABLE B-3 Continued McGuire/Catawba Statepoint Statistical Results

3000 Case Runs

Statepoint #
2-T
3-T
4-T
12-T
13-T
14-T
16-T
20-T
37-T
38-T
39-T

TABLE B-4 McGuire/Catawba Key Parameter Ranges

Parameter	Maximum	Minimum

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

Attachment 1
Response To Request for
Additional Information For
Main Body, Appendix A, and
Appendix B

Revision 0

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DUKE POWER

September 29, 1993

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
McGuire Nuclear Station
Docket Nos. 50-369, -370
Catawba Nuclear Station
Docket Nos. 50-413, -414
Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005

By letter dated September 28, 1992, Duke Power Company submitted Topical Report DPC-NE-2005, "Thermal/Hydraulic Statistical Core Design Methodology." The NRC staff issued a request for additional information (RAI) dated July 27, 1993. Attached are the responses to the questions contained in the RAI.

In accordance with 10 CFR 2.790, Duke Power Company requests that the attached information relating to DPC-NE-2005 be considered proprietary. Information supporting this request is included in the affidavit which appears as Attachment I.

If we can be of assistance in your review please call Scott Gewehr at (704) 382-7581.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

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File: GS-801.01

ATTACHMENT I

AFFIDAVIT OF M. S. TUCKMAN

1. I am Senior Vice President, Nuclear Generation Department, Duke Power Company ("Duke"), 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.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission ("NRC") and in conjunction with Duke's application for withholding which accompanies this affidavit.

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

4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.790, 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 and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. 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.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, 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 this submittal is that which is marked in the proprietary version of the report DPC-NE-2005, "Thermal/Hydraulic Statistical Core Design Methodology" and supporting documentation, and omitted from the non-proprietary versions.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN

This information enables Duke to:

(a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

(b) Respond to NRC requests for information regarding the transient response of Babcock & Wilcox and Westinghouse reactors.

(c) Support license amendment and Technical Specification revision request for Babcock & Wilcox and Westinghouse reactors.

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

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

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

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

5. Public disclosure of this information is likely to cause harm to Duke 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 to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN (Page 3)

M. S. Tuckman, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

M. S. Tuckman

M. S. Tuckman

Sworn to and subscribed before me this 29th day of September, 1993. Witness my hand and official seal.

Linda L. Kessler

Notary Public

My commission expires May 7, 1994.

LINDA L. KESSLER
NOTARY
PUBLIC

Request for Additional Information and Responses To Topical Report DPC-NE-2005P

The questions are shown in italics and the responses immediately follow.

1. *Explain DPC's intent for this topical report. Does DPC seek its review with respect to its plants or generic PWR application? How does DPC plan to deal with the restrictions and requirements imposed by the VIPRE-01 code SER?*

The intent of this submittal is to outline a statistical Departure from Nucleate Boiling methodology. In DPC-NE-2005, DPC has outlined a statistical analysis method that is based on inherent behavior of the DNBR phenomena in pressurized water reactors. The numerical value of the Statistical Design Limit (SDL) will vary, depending on the CHF correlation used and parameter uncertainties assumed. However, direct use of the VIPRE-01 thermal hydraulic code (rather than the RSM) to calculate the phenomenological statistical variance of DNBR insures the direct applicability of this method to many varying fuel designs and parameter conditions.

DPC seeks the following approval from the NRC regarding this report:

1) Review and approval of the methodology <u>and</u> the stated statistical DNB limits for use at Oconee, McGuire, and Catawba based on the information in the body of the report and the site specific information in the Appendices.
2) Review and approval of the use of the methodology for future analyses of non-DPC reactors consistent with the commitments made in Section 1.3 and 2.5 of the report. This involves development or justification of the models and uncertainties used for any other site. If DPC were to extend this method to another PWR facility, a separate submittal will be made detailing the intent and justification for specific modeling assumptions, choice of flow models and correlations, and plant specific input data, as well as the resulting statistical DNB limits. The form of this submittal would be an additional Appendix to this report. This meets item (3) of Section 3 of the VIPRE-01 SER. The SDL would be calculated using the methodology outlined in the body of the report.

2. *DPC previously submitted two sets of DNB models for each type of plant. One was approved for use in steady-state type calculations and the other for use in transient type calculations. Since there are differences between these models on the basic level of model/input selection, discuss the impact of these differences on SDL determined. The SCD is developed based upon a series of steady state calculations. Explain how the SDL is used for transient analysis.*

Both models used by DPC were included in the statistical propagations detailed in the report. This is explained on page 14 of the report. Statepoints 37 and 38 in Appendix B are identical in fluid and peaking conditions. Statepoint 37 was propagated with the eight channel M/C model from Reference 5 and Statepoint 38 used the fourteen channel model

from Reference 6. Table 6 of the report (page 27) shows the results of this comparison. The difference in Statistical DNBR's is negligible.

The determination of whether the SCD limit can be used for a transient is based on the fluid conditions at the point of minimum DNBR (MDNBR) during the transient. If the power, pressure, temperature, and flow rate of this statepoint fall within the parameter range listed in Table 4 of the appropriate Appendix, the SDL can be used. All the statepoint statistical propagations are made from a single set of fluid and peaking conditions.

3. *Discuss how "appropriate compensatory measures" will be applied to ensure the allowable DNB behavior for the statepoint is conservatively bounded.*

Please refer to the Definitions page (Page V) of the report for the following definitions:

Design DNBR Limit, DDL
Statistical Design Limit, SDL
Statistical DNBR

The term statistical DNBR applies to a specific statepoint, the SDL is the licensed limit, and the DDL is the MDNBR value used in steady-state and transient DNB analyses.

As stated in the report, new statepoints or revised uncertainties can be evaluated directly with this method. As long as the statistical DNBR value is less than the SDL, the statepoint is conservatively bounded by the SDL. If, however, the statistical DNBR value is larger than the SDL, actions can be taken to ensure that DNB predictions for the condition meet the required 95/95 acceptance criteria. These compensatory measures include either

1) Increasing the design DNBR limit for the statepoint.
2) Using available margin present between the statistical and design DNB limits (between the SDL and the DDL).

Increasing the design DNBR limit (DDL) will increase the minimum DNBR that is allowed in the analysis of the statepoint This requires another key transient analysis input (such as maximum allowable peaking) to be reduced. Penalizing a statepoint in this manner will ensure the required DNBR protection is maintained.

Another equally valid method is to apply any unused margin already available between the statistical (SDL) and design DNBR limit (DDL). This margin is inherently retained in all analyses by using the DDL in design calculations which includes margin above the SDL. A portion of this margin is currently used to account for such things as reactor vessel flow anomalies, instrumentation biases that cannot be statistically compensated for, and physical changes to the fuel assembly not accounted for in standard models (such as rod bow). The margin remaining after all of the DNBR penalties are accounted for can be used to compensate for the increase in SDL required for a particular statepoint. Either of these methods will conservatively adjust the MDNBR limit that must be met in the analysis to ensure adequate protection is maintained.

4. *Explain why RCS flow is varied only between 100 and 106.5% and not below 100% (see Table A-1 on p. A-3) even for low flow cases.*

The percent flow listed for the low flow cases in Table A-2 is in error. The flow rate used for all the statepoints identified as Low Flow in the Comments column was [79.8%.] Additionally, the Minimum flow value listed on Table A-4 for percent design RCS flow should be [79.8%.] The corrected pages are included with this response.

Additionally, the flow chart in Figure 2 also contains a typo. The Propagate Uncertainties box should have the words (Monte Carlo - VIPRE) underneath. The RSM is not used at all in the revised method described by the report. A corrected page 30 is also included.

5. *Explain thoroughly how ranges of uncertainties and their associated standard deviations are determined.*

The numerical range of each uncertainty is selected to bound the value calculated for the parameter. This ensures that conservative statistical behavior is calculated and allows for changes in the uncertainty value without requiring re-analysis of the SDL.

(a) *Explain how uncertainties in instrumentation are accounted for. What is meant by the term "random uncertainty" (see Table A-2)? Explain how it is related to instrument error uncertainty.*

The term "random uncertainty" used in Table A-2 of the report means the instrument uncertainties such as sensor calibration accuracy, rack drift, sensor drift, etc., that are combined by the SRSS method. The term was used because the biases which are constant in sign (either positive or negative) are not included in the propagation of an uncertainty and must be accounted for by another means, such as a DNB penalty.

(b) *Identify the sources of the quantitative ranges of uncertainties and their associated standard deviations (for both types of plants).*

The source of the quantitative ranges and the standard deviations are provided on Table 1 of this response for each plant. The statistical propagations for each normally distributed parameter are based on the standard deviation numerical values. Uniform uncertainty propagations are based on the uncertainty numerical magnitude.

6. Explain thoroughly the mechanistic DNB behavior observed in Figures 7A and B.

Figures 7A and 7B in the report show the sensitivity of DNBR to axial peak location and magnitude. This sensitivity was calculated by holding all other parameters (power, pressure, temperature, flow, and radial peaking) constant. Both the BWC (7A) and BWCMV (7B) CHF correlation results are shown. These graphs show that the response of DNBR varies with axial peak conditions.

(a) *Discuss why the sensitivity to the axial peaks and locations is significantly stronger for Oconee than it is for M/C.*

The evaluations contained in the report indicate that the numerical value of the SDL is dependent on the CHF correlation used in the analysis. Table 5 in the report contained individual parameter sensitivities to DNB for the BWCMV CHF correlation in both axial peak areas defined. Table 2 in this response contains an identical sensitivity evaluation for the BWC and DCHF-1 CHF correlations in both axial peak areas.

For the region of higher statistical behavior, comparison of the BWC and BWCMV sensitivities shows the sensitivity calculated for each key parameter with the BWC correlation has slightly higher sensitivity to DNBR. This results in a higher final calculated SDL. The sensitivity values are more consistent when the same evaluation is made in the lower SDL area and the corresponding statistical DNBR's for the two correlations are almost identical. Correspondingly, the DCHF-1 correlation has lower sensitivities in both areas and has the lowest statistical DNBR in both cases.

Again, Table 2 in this response as well as Table 5 in the report (page 26) shows that the behavior is remarkably consistent between Oconee and McGuire/Catawba and is linked to axial power distribution. There is a difference in the numerical value of the statistical DNBR, and the key to this is the CHF correlation being used. DPC's conclusion is that the general behavior is mechanistic and this is proven by the consistent behavior when the sensitivity is calculated for different fuel types (15x15 non-mixing vane and 17x17 mixing vane), different fuel vendors (Westinghouse and Babcock & Wilcox), and even different CHF correlations (BWC, BWCMV, and DCHF-1).

(b) *DPC's conclusion based upon Figure 6A and B on p. 13 is not clear. Explain further.*

The discussion on page 13 and Figures 4A, 4B, 4C, 5A, 5B, 5C, 6A and 6B of the report show how the statistical DNB behavior is much more dependent on axial peak location than on the fluid parameter values for a particular statepoint, the fuel type, or the CHF correlation. The Figure 4 and 5 series show how the statistical DNB behavior changes with shifts in the axial power distribution. The axial peak location has a large impact on the statistical DNB value. By contrast, Figures 6A and 6B show how little the statistical DNB behavior changes with

large changes in the statepoint pressure, temperature, flow rate, and core power variables. This means that if the SDL is determined in either of the axial power distribution areas for one set of fluid conditions, this SDL value would be consistent even if the fluid conditions changed dramatically.

7. *Provide a table which identifies which DNB methodology is used for each transient and explain each such selection.*

The McGuire/Catawba DNB transients currently analyzed using the SCD methodology are listed in Table 3 of this response. No transients are currently analyzed for Oconee with the SCD methodology. All of the transients analyzed with the SCD methodology were selected based on the values of the individual parameters at the point of MDNBR during the transient as explained in the response to Question 2. If these values are within the range for each parameter defined on Table 4 of the appropriate Appendix, the SCD limit can be applied to the transient.

As discussed by the note below Table 4-A and 4-B, this parameter list is subject to change. One of the advantages of the explicit evaluation method describe in the report is the ability to specifically evaluate new conditions for SCD limit applicability. If a new statepoint has a parameter(s) outside the given range, it would be analyzed and if the current SCD limit is conservative, the table would be updated to show the expanded range. The transient that generated the statepoint would then be included on the internal DPC list (Table 3 of this response). This increased parameter range would not be reported directly to NRC.

8. *Explain the last two paragraphs of Section 2.4. Discuss the need to perform statistical DNB analysis in two levels and with two different sample sizes.*

The two different sample sizes were used to minimize the total number of cases propagated for each set of fluid conditions analyzed. The first level of 500 cases per statepoint is used to quickly evaluate the behavior of a statepoint with respect to the two axial peak areas. This shows the statistical DNB behavior and approximate numerical SDL value for the fluid conditions being evaluated.

The second group of 3000 case statepoints are selected to calculate the limiting SDL value for the reactor type being analyzed. The increase in number of cases to 3000 provides a more thorough evaluation of the statistical DNB response and improves statistically the Chi Square and K factor multipliers used to conservatively increase the coefficient of variation in the final SDL calculation. The licensed statistical design limit is greater than the largest value calculated in all the 3000 case propagations for each axial peak area.

DPC may increase the number of cases at a particular statepoint for future evaluations to take advantage of the improved effect on the statistical multipliers. This increase in the number of cases is consistent with the methodology as presented and does not in any way reduce the conservatism of the SDL limit calculated. Increasing the number of cases

simply reduces the statistical uncertainty associated with calculation of the coefficient of variation.

9. Explain the rational for and appropriateness of selection of certain sets of statepoints to determine the impact of changes on statistical DNBR behavior (see Table 6).

The evaluations in Table 6 of the report show how little the statistical DNB behavior is affected by small modifications in the analysis. The first section shows the change for identical conditions and models with a change in one parameter uncertainty distribution (normal versus uniform). Section 2 shows the change if a different VIPRE-01 model is used with the same fluid conditions, peaking conditions, and uncertainty distributions. The last section shows the change with the same VIPRE-01 model, fluid conditions, and uncertainties but with a different fuel design. As discussed in the response to question 6b, Figures 6A and 6B demonstrate the there is very little change in statistical DNB behavior for large changes in the statepoint pressure, temperature, flow rate, or core power variables. Thus, the sensitivity of the SDL to other changes can be evaluated using a single statepoint.

All of these evaluations were included to further demonstrate that the statistical DNB behavior and SDL are more closely related to the CHF correlation and axial power distribution than to small perturbations in individual uncertainties, VIPRE-01 models, or fuel type. This evaluations also provide the basis for the criteria for re-submittal or in-house evaluation detailed on Table 7 (as explained in the response to Question 10).

10. Explain Table 7.

Table 7 in the report is intended as a guide for use by DPC in evaluating what action must be taken for anticipated changes (a revised uncertainty, new fuel type, etc.). In all cases, the evaluations will use the methodology detailed in the report. Basically, changes that are anticipated to have a negligible or very small impact on the SDL will require internal DPC evaluation. Only changes that have a significant impact on the calculated SDL number will be submitted to the NRC for approval.

An example of the kind of anticipated events is a change in an uncertainty magnitude. For this instance, limiting SCD statepoints in each axial power distribution area will be evaluated to determine the impact on the SCD limit. If the statistical DNBR value is the same or smaller than the SDL, no additional work is required. If the value is larger, appropriate compensation measures will be used to conservatively compensate for the change (as described in the answer to Question 3). This same approach will be used for different uncertainty distributions, new fluid or peaking condition statepoints, or minor modifications to the fuel assembly design.

For changes that will have a much bigger impact on the statistical DNB behavior, the impact of the change will be evaluated and a new Appendix to this report submitted for NRC approval. This additional Appendix will have the same format and content as the

two already included in the report. Examples of when this approach would be used are a completely new fuel assembly design, a new thermal hydraulic code, a new CHF correlation, or DPC analysis of a third party's reactor.

A slight change to Table 7 is also included in the response to this question. The original table required that a modified CHF correlation would require submittal of a new Appendix. This has been changed to require an evaluation only. The term modified means the form of the CHF correlation is the same, just a single factor or multiplier has been changed or added. This change is because a modified correlation will not impact the statistical DNB behavior and will not significantly change the SDL compared to the original correlation. A modified correlation will still require a separate CHF correlation topical submittal to the NRC. Any other changes that affect the correlation form will be considered a new CHF correlation.

11. Provide the SDL if no distinctions are made of axial power distributions.

The results of the entire analysis completed in the report show how mechanistic the statistical DNB response is to axial power distribution. This mechanistic behavior was determined by direct use of the thermal hydraulic codes, models, and correlations used in DNB predictions. This behavior is consistent with different fluid conditions, fuel geometries, and CHF correlations. The one consistent fact is the larger statistical variation for a specific set of axial peaks. The use of two statistical DNB limits to address this behavior is a straight forward application. Use of a single limit would be unnecessarily conservative. However, if the appropriate distinctions are not made for the generic DNB behavior with axial power distributions, the SDL for all cases will be the largest value calculated for all the conditions evaluated. If this restriction were imposed, the SDL would be 1.43 for Oconee and 1.40 for McGuire/Catawba.

TABLE 1

Uncertainty Ranges And Standard Deviations

The following table shows the source of the quantitative range of each uncertainty and its associated standard deviation. Section 1 of the table contains the Oconee information and Section 2 the sources for the McGuire/Catawba values.

SECTION 1 - Oconee

Parameter	Source
Power	Standard deviation of 1.0% based on DPC calculations. Uncertainty value is a 2σ value (2%).
Pressure	Standard deviation of 15 psi based on DPC calculations. Uncertainty value is a 2σ value (30 psi).
Temperature	Standard deviation of 1.0 degrees Fahrenheit based on DPC calculations. Uncertainty value is a 2σ value (2 deg F).
Flow	Standard deviation of 1.0% design flow based on DPC calculations. Uncertainty value is listed as a 2σ value (2%).
$F\Delta H$	Standard deviation of 2.84% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for Oconee (Reference 1).
F_Z	Standard deviation of 2.91% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for Oconee (Reference 1).
Z	Uncertainty range of +/- 6 inches. Selected based on the noding size of nuclear codes. No standard deviation (uniform uncertainty).
Local Heat Flux HCF	Uncertainty range of [.] Based on calculated values from the nuclear fuel vendor. Standard deviation is calculated from [ ] uncertainty value [ ]

Rod Power HCF	Uncertainty range of [ ] Based on calculated values from the nuclear fuel vendor. Standard deviation is calculated from [ ] uncertainty value [ ]
Hot Channel Flow Area	Uncertainty range of [ ]. Based on calculated values from the nuclear fuel vendor. No standard deviation (uniform uncertainty).
CHF Correlation	Standard deviation of 8.88% calculated from the BWC CHF test data base (Reference 2).
Thermal Hydraulic Code / Model	Uncertainty range of [ ] Value used in Reference 3. Standard deviation is calculated from the [ ] uncertainty value [ ].

SECTION 2 - McGuire/Catawba

Parameter	Source
Power	Uncertainty Range of 2%. Selected from Reference 5. Kept at 2% to bound specific uncertainties calculated for M/C. Standard deviation is calculated from 2% uncertainty value (2/1.64 = 1.22%).
Pressure	Uncertainty Range of 30 psi. Selected from Reference 5. Kept at 30 psi to bound specific uncertainties calculated for M/C. No standard deviation (uniform uncertainty).
Temperature	Uncertainty Range of 4 degrees Fahrenheit. Selected from Reference 5. Kept at 4 degrees to bound specific uncertainties calculated for M/C. No standard deviation (uniform uncertainty).
Flow	Uncertainty Range of 2.2%. Selected from Reference 5. Kept at 2.2% to bound specific uncertainties calculated for M/C. Standard deviation is calculated from 2.2% uncertainty value (2.2/1.64 = 1.34%).
FΔH Measurement	Standard deviation of 1.98% calculated based on the nuclear code packages used for core design and analysis. This standard

	deviation bounds the highest value from the code package combination used for M/C (Reference 1).
Engineering HCF	Uncertainty range of 3.0%. Selected based on the value in Technical Specifications. Standard deviation is calculated from the 3% uncertainty value (3/1.64 = 1.82%).
Spacing	Uncertainty range of 2.0%. Selected from Reference 5. Standard deviation is calculated from the 2% uncertainty value (2/1.64 = 1.22%).
F_Z	Standard deviation of 2.68% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for M/C (Reference 1).
Z	Uncertainty range of +/- 6 inches. Selected based on the noding size of nuclear codes. No standard deviation (uniform uncertainty).
CHF Correlation	Standard deviation of 10.2% calculated from the BWCMV CHF test data base (Reference 4).
Thermal Hydraulic Code / Model	Uncertainty range of [ ] Value used in Reference 5. Standard deviation is calculated from the [ ] uncertainty value [ ].

TABLE 2

Comparison of the DNB Parameter Sensitivity of Different CHF Correlations With Consistent Axial Power Distributions

The following table shows the DNB sensitivity of each key parameter for the BWC CHF correlation (Oconee), the BWCMV CHF correlation (McGuire/Catawba), and the DCHF-1 CHF correlation (McGuire/Catawba). The first comparison is of a statepoint in the higher SDL area and the second is in the lower SDL area. The fluid and radial peaking conditions for each statepoint are given in the Appendices.

CHF Correlation	1.3 Peak @ 0.2 Z	1.3 Peak @ 0.8 Z
BWC	Statepoint 63	Statepoint 75
BWCMV	Statepoint 6	Statepoint 9
DCHF-1	Statepoint 6	Statepoint 9

1.3 Axial Peak, 0.2 Z

Parameter	BWC	BWCMV	DCHF-1
Power (%)	[		
Pressure (psi)			
Temperature (Deg F)			
Flow (%)			
FΔH (%)			
F_Z (%)			
Z (per 6 inches)			
SDL			]

1.3 Axial Peak, 0.8 Z

Parameter	BWC	BWCMV	DCHF-1
Power (% RTP)	[		
Pressure (psi)			
Temperature (Deg F)			
Flow (%)			
FΔH (%)			
F_Z (%)			
Z (per 6 inches)			
SDL			]

All values shown are in terms of % DNB per unit of parameter.

TABLE 3
SCD Transient Limiting Statepoints

The following table shows all the M/C transients currently evaluated with the SCD methodology. The determination of whether the transient uses the SCD approach is the value of all the key parameters (power, pressure, temperature, flow, peaking) at the point of MDNBR during the transient. All values listed are from the MDNBR point of the transient.

Transient	Core Power	Core Inlet Flow (Kgpm)	Core Inlet Temperature	Pressure	FΔH	F_Z	Z
Feed Line Break	[						
Partial Loss of RCS flow							
Total Loss of RCS Flow							
Uncontrolled RCCA Withdrawal / Subcritical							
*Uncontrolled RCCA Withdrawal / 100%							
*Uncontrolled RCCA Withdrawal / 100%							
Uncontrolled RCCA Withdrawal / 50%							
*Uncontrolled RCCA Withdrawal / 10%							
*Uncontrolled RCCA Withdrawal / 10%							
Single RCCA Withdrawal							
Statically Misaligned RCCA							
Dropped RCCA							]

* This accident was analyzed with two different reactivity insertion rates.

\# This accident was analyzed with a FΔH range of [

]

REFERENCES

1) Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power Company, Charlotte, North Carolina, November 1992.

2) BWC Correlation for Critical Heat Flux, BAW-10143P-A, Babcock And Wilcox , Lynchburg, Virginia, April 1985.

3) Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003P-A, Duke Power Company, Charlotte, North Carolina, August 1988.

4) BWCMV Correlation Of Critical Heat Flux In Mixing Vane Grid Fuel Assemblies, BAW-10159P-A, Babcock And Wilcox, Lynchburg, Virginia, February, 1989.

5) McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2004P-A, Duke Power Company, Charlotte, North Carolina, December 1991.

6) Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000P-A, Revision 1, Duke Power Company, Charlotte, North Carolina, December 1991.

Revised Pages For Topical Report DPC-NE-2005P

The bar in the right hand margin notes revised lines.

TABLE 7. SDL Evaluation And Re-Submittal Criteria

The following table lists different events or conditions that would require an evaluation of the applicability of an approved SDL and the subsequent actions based on the results of the analysis.

CONDITION	ACTION
Revised uncertainty larger than the limiting value used in the original analysis.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
Revised uncertainty distribution.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
New statepoint.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
Minor modifications to the current fuel design.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
A modified CHF correlation.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
Change to a new fuel design/fuel type.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
A new CHF correlation.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
Duke analysis of a non-Duke reactor.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
New Thermal-Hydraulic Code.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.

FIGURE 2

REVISED SCD FLOWCHART

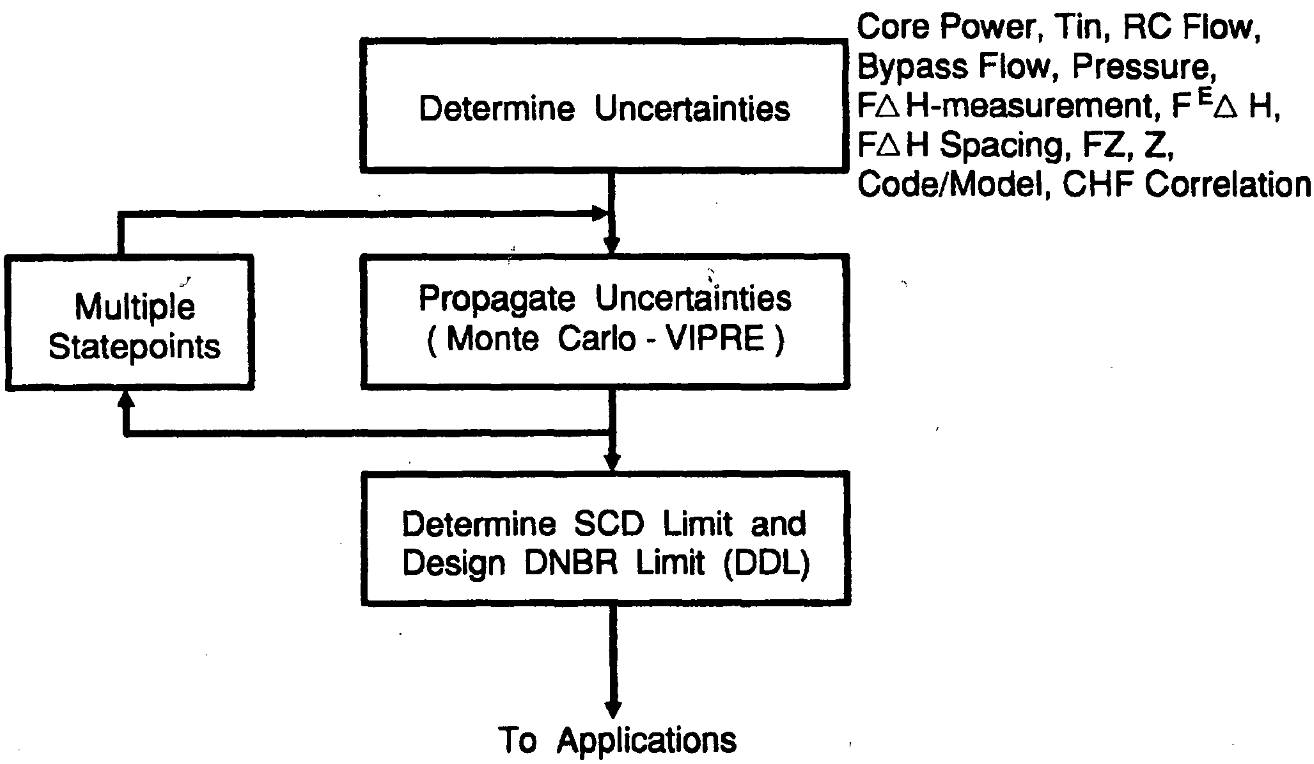

TABLE A-1. Oconee SCD Statepoints

TABLE A-4 Oconee Key Parameter Ranges

Parameter	Maximum	Minimum

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

Appendix C and Response To Request for Additional Information

Revision 1

DPC-NE-2005-PA

Duke Energy Carolinas
Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX C

McGuire/Catawba Plant Specific Data

Mark-BW Fuel

BWU-Z CHF Correlation

Submitted April 1996

Approved November 1996

This Appendix contains the plant specific data and limits for the McGuire and Catawba Nuclear Stations with Mark-BW fuel using the BWU-Z form of the BWU critical heat flux correlation. The thermal hydraulic statistical core design analysis was performed as described in the main body of this report.

Plant Specific Data

This analysis is for the McGuire and Catawba plants (four loop Westinghouse PWR's) with Mark-BW fuel assemblies as described in Reference C-1. The parameter uncertainties and statepoint ranges were selected to bound the unit and cycle specific values of the McGuire and Catawba stations.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference C-3 and the McGuire/Catawba eight channel code model approved in Reference C-1 are used in this analysis.

Critical Heat Flux Correlation

The BWU-Z form of the BWU critical heat flux correlation described in Reference C-2 is used for all statepoint analyses. This correlation was developed by Areva NP for application to the Mark-BW fuel design. Reference C-2 was performed with the LYNXT thermal-hydraulic computer codes. The correlation was programmed into the VIPRE-01 thermal-hydraulic computer code by Duke Energy and the BWU-Z CHF data base analyzed in its entirety. The results of this analysis are shown in Table C-1. The resulting Average M/P value and data standard deviation are within 1% of the values reported in Reference C-2.

Figures C-1 through C-5 graphically show the results of this evaluation. Figure C-1 shows there is no bias of measured CHF values to VIPRE-01 predicted values for the data base. Figure C-2 shows a histogram of the VIPRE-01 M/P ratios for the 530 point data base. Figures C-3 through C-5 show there is no bias with the VIPRE-01 calculated M/P ratios with respect to mass velocity, pressure, or thermodynamic quality. These figures compare closely with the same parameter representations in Reference C-2.

Based on the results shown in Table C-1 and Figures C-1 through C-5, the BWU-Z form of the BWU CHF correlation licensed in Reference C-2 can be used in DNBR calculations with VIPRE-01 for Mark-BW fuel.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table C-2. These statepoints represent the range of conditions to which the statistical DNB analyses limit will be applied.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table C-3. The uncertainties were selected to bound the values calculated for each parameter at McGuire and Catawba. The resulting range of key parameter values generated in this analysis is listed on Table C-5.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table C-4. Section 1 of Table C-4 contains the 500 case runs and Section 2 contains the 5000 case runs. The number of cases was increased from 3000 to 5000 as described in Attachment 1 of the main body of the report. All statepoint SDL values listed in this analysis are normally distributed. The maximum statepoint statistical DNBR value in Table C-4 for the 5000 case propagations was [ ]. Therefore, the statistical design limit using the BWU-Z form of the BWU CHF correlation for Mark-BW fuel at McGuire/Catawba was conservatively determined to be [ ].

FIGURE C-1

Measured CHF Versus Predicted CHF

Mark-BW Data Base

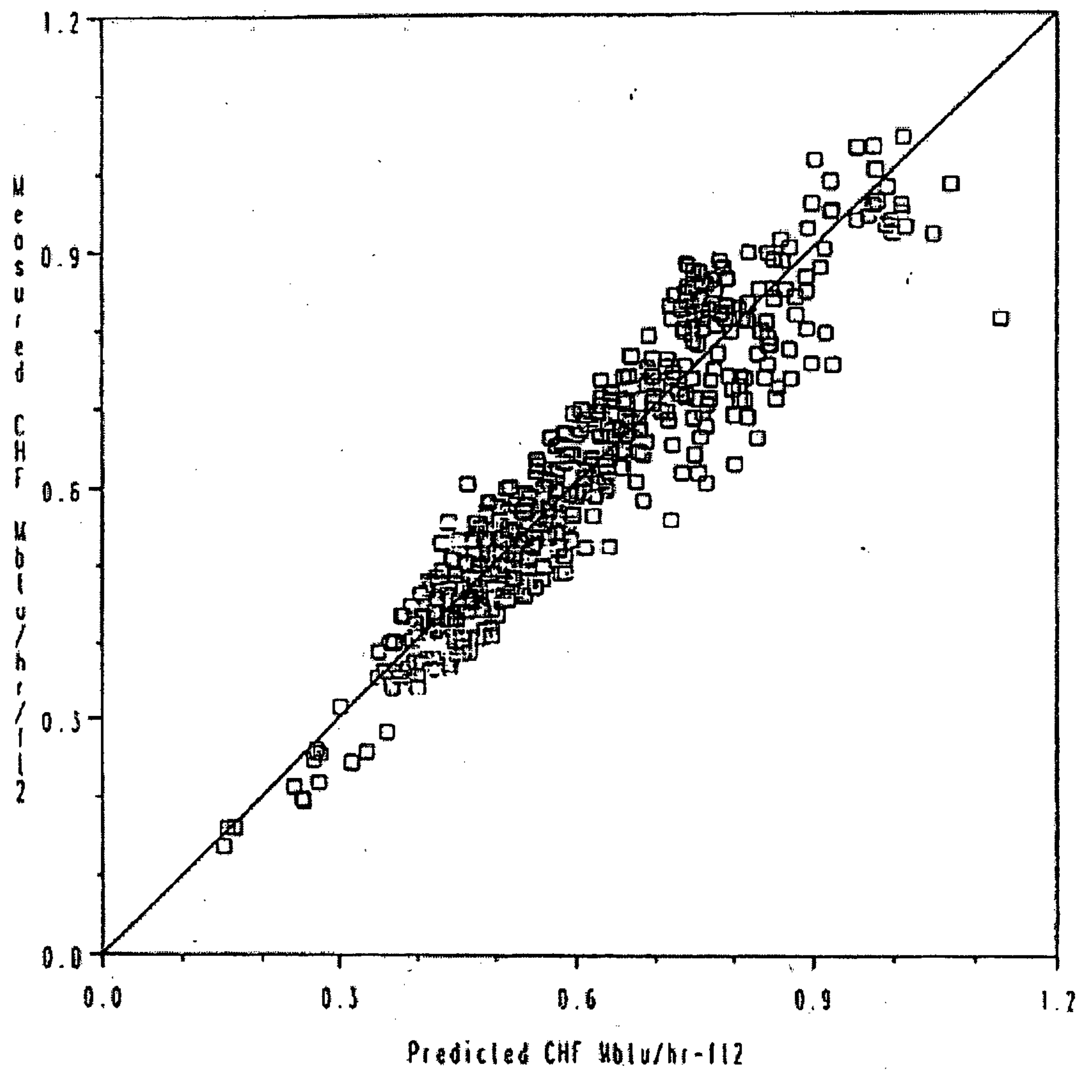

FIGURE C-2

Distribution of CHF Ratios

Mark-BW Data Base

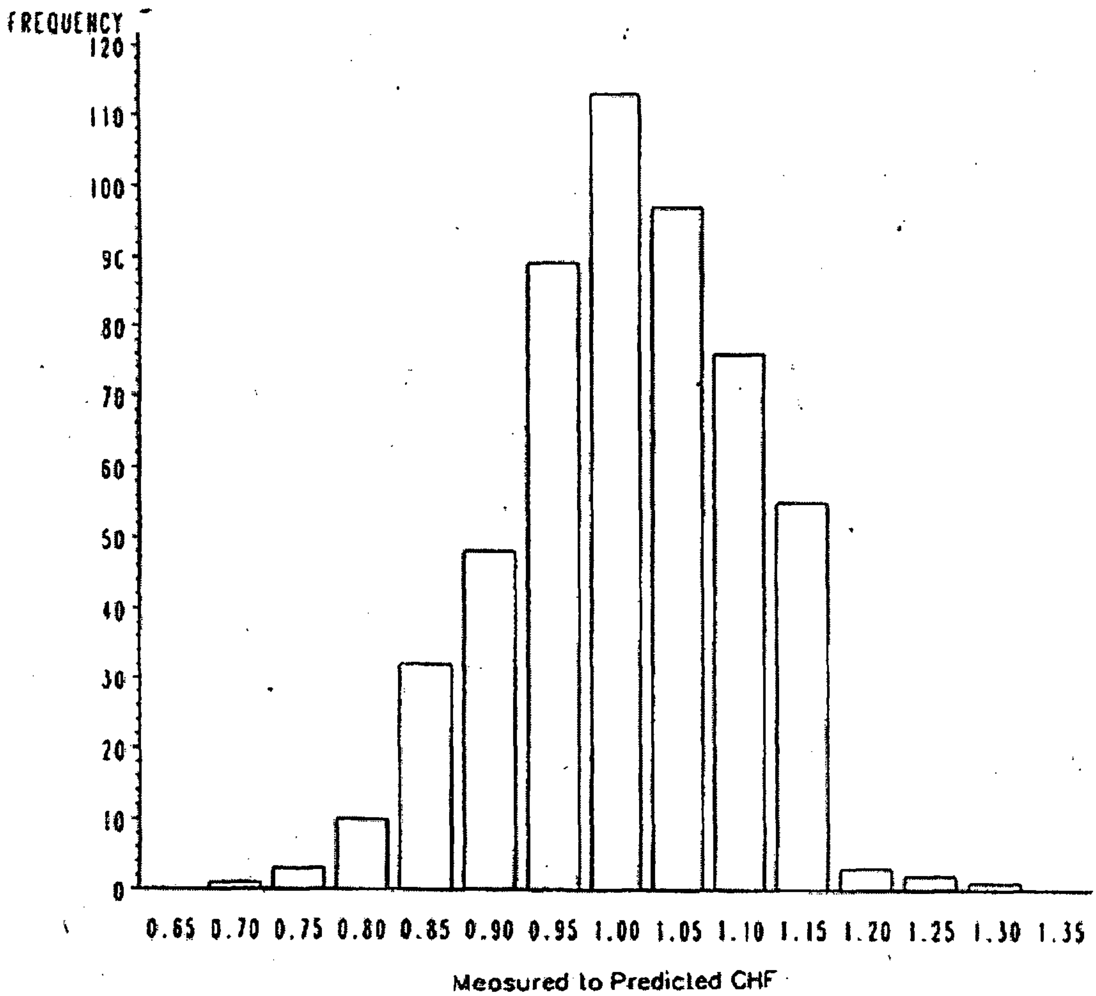

FIGURE C-3

Measured to Predicted CHF Versus Mass Velocity

Mark-BW Data Base

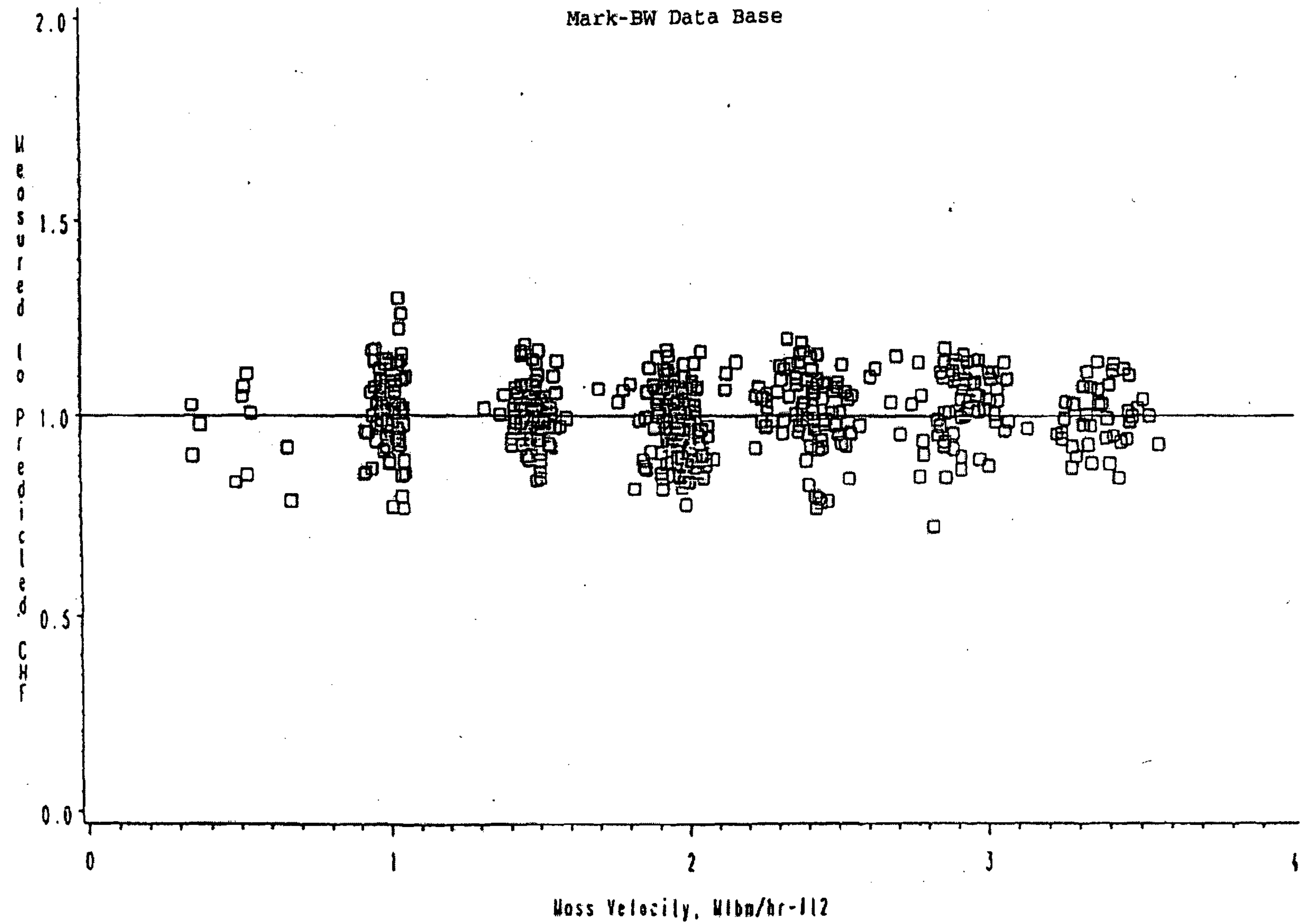

FIGURE C-4

Measured to Predicted CHF Versus Pressure

Mark-BW Data Base

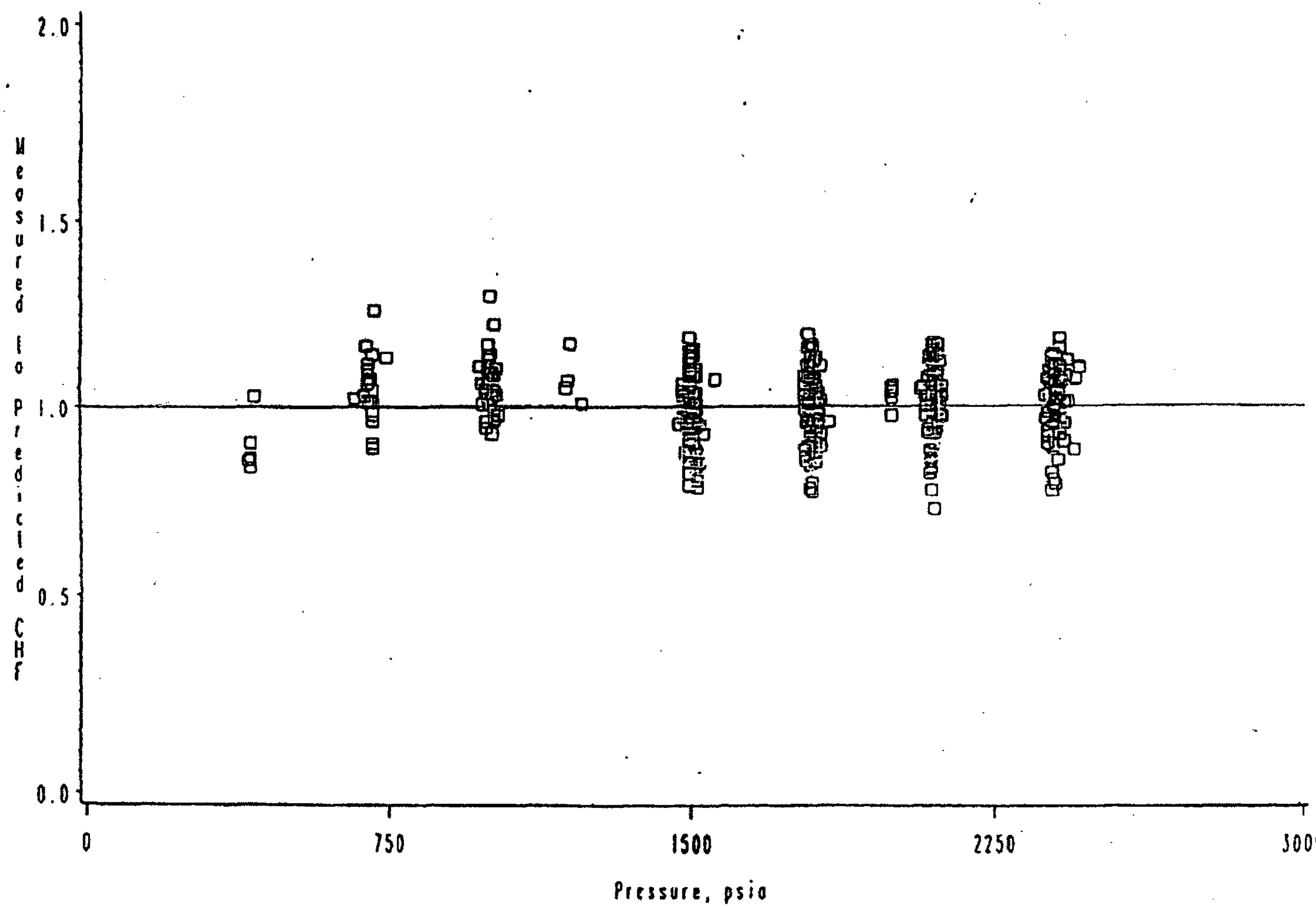

FIGURE C-5

Measured to Predicted CHF Versus Quality

Mark-BW Data Base

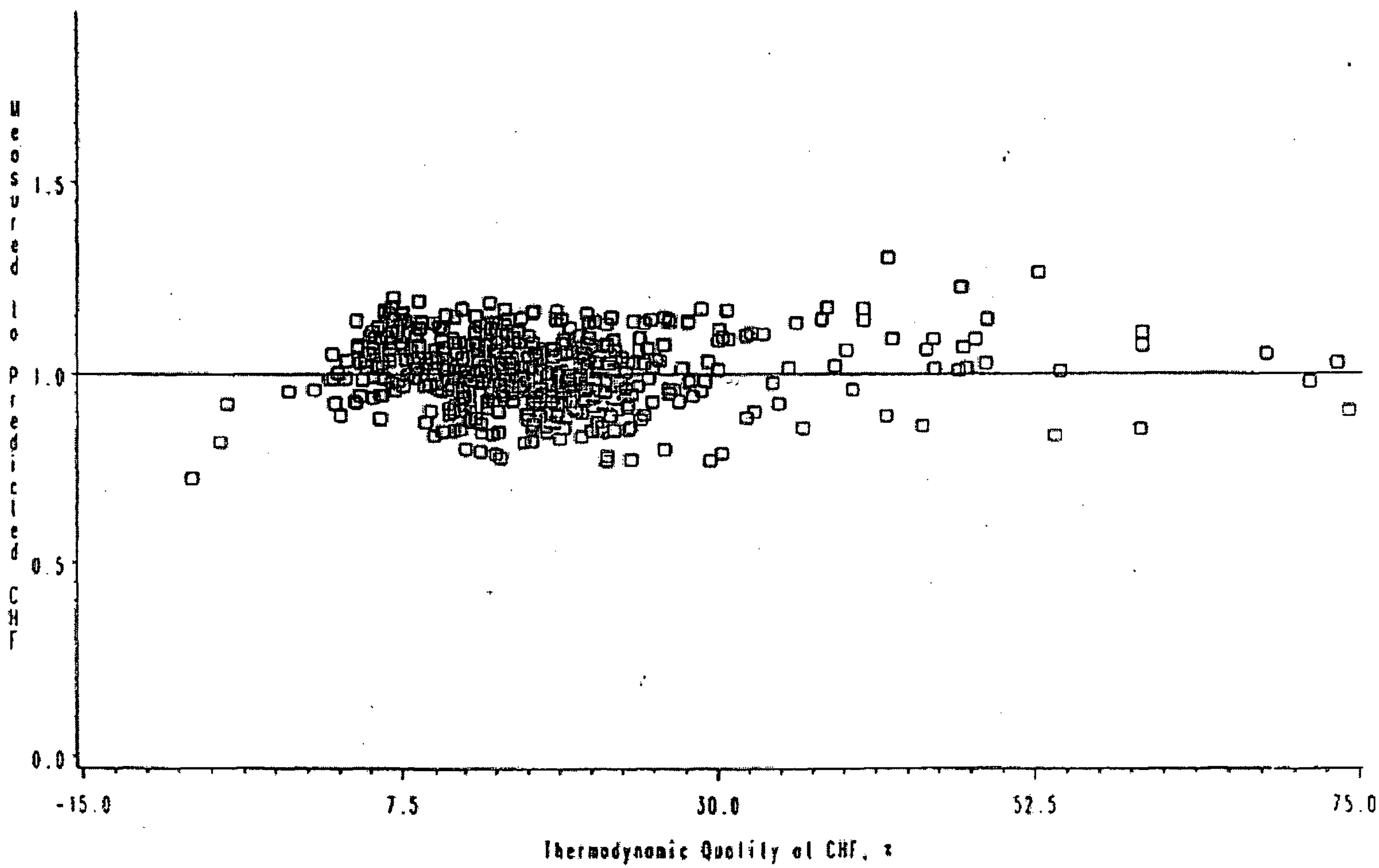

TABLE C-1 VIPRE-01 BWU-Z Correlation Verification

CHF Test Database Analysis Results

VIPRE-01 Statistical Results

Number Of Data Points	530
Average M/P	1.00850
Standard Deviation	0.09217
Upper D Prime	3469.0
Lower D Prime	3407.0
D Prime Value	3453.68
Accept Normality at 5% Level	

Parameter Ranges

Pressure, psia	400 to 2465
Mass Velocity, Mlbm/hr-ft^2	0.36 to 3.55
Thermodynamic Quality at CHF	less than 0.74
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Mark-BW 17x17
Design Limit DNBR, VIPRE-01	1.18

TABLE C-2 McGuire/Catawba SCD Statepoints

Stpt No.	Power* (% RTP)	RCS Flow (K gpm)	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak ($F\Delta H$)
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						

* 100% RTP = 3411 Megawatts Thermal

TABLE C-3 McGuire/Catawba Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type of Distribution
Core Power*	+/- 2% / +/- 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / +/- 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^N_{\Delta H}$		
Measurement	+/- 4.0% / 2.43%	Normal
$F^E_{\Delta H}$	+/- 3.0% / 1.82%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.41% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	+/- 15.25% / 9.27%	Normal
Code/Model	[ ]	Normal

* - Percentage of 100% RTP (68.22 MWth wherever applied).

TABLE C-3 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
Core Power	The core power uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from normally distributed random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method (SRSS). Since the uncertainty is calculated from normally distributed values, the parameter distribution is also normal.
Core Flow	
Measurement	Same approach as Core Power.
Bypass Flow	The core bypass flow is the parallel core flow paths in the reactor vessel (guide thimble cooling flow, head cooling flow, fuel assembly/baffle gap leakage, and hot leg outlet nozzle gap leakage) and is dependent on the driving pressure drop. Parameterizations of the key factors that control ΔP, dimensions, loss coefficient correlations, and the effect of the uncertainty in the driving ΔP on the flow rate in each flow path, was performed. The dimensional tolerance changes were combined with the SRSS method and the loss coefficient and driving ΔP uncertainties were conservatively added to obtain the combined uncertainty. This uncertainty was conservatively applied with a uniform distribution.
Pressure	The pressure uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method. The uncertainty distribution was conservatively applied as uniform.
Temperature	Same approach as Pressure.

TABLE C-3 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
$F^N_{\Delta H}$	
Measurement	This uncertainty is the measurement uncertainty for the movable incore instruments. A measurement uncertainty can arise from instrumentation drift or reproducibility error, integration and location error, error associated with the burnup history of the core, and the error associated with the conversion of instrument readings to rod power. The uncertainty distribution is normal.
$F^E_{\Delta H}$	This uncertainty accounts for the manufacturing variations in the variables affecting the heat generation rate along the flow channel. This conservatively accounts for possible variations in the pellet diameter, density, and U_{235} enrichment. This uncertainty distribution is normal and was conservatively applied as one-sided in the analysis to ensure the MDNBR channel location was consistent for all cases.
Spacing	This uncertainty accounts for the effect on peaking of reduced hot channel flow area and spacing between assemblies. The power peaking gradient becomes steeper across the assembly due to reduced flow area and spacing. This uncertainty distribution is normal and was conservatively applied as one-sided to ensure consistent MDNBR channel location.
F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty distribution is applied as normal.
Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.

TABLE C-3 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
DNBR	
Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normal.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is applied as normal.

TABLE C-4 McGuire/Catawba Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation

500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				

TABLE C-4 Continued McGuire/Catawba Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
7				
9				
12				

TABLE C-5 McGuire/Catawba Key Parameter Ranges

Parameter	Maximum	Minimum
Core Power (% RTP)		
Pressure (psia)		
T inlet (deg. F)		
RCS Flow (Thousand GPM)		
$F_{\Delta H}$, Fz, Z		

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

REFERENCES

C-1. DPC-NE-2004-PA, Revision 2a, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, December 2008.

C-2. The BWU Critical Heat Flux Correlations, BAW-10199-P, AREVA NP, Lynchburg, Virginia, December 1994 (SER received April 5, 1996).

C-3. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

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DUKE POWER

September 5, 1996

U. S. Nuclear Regulatory Commission
Washington, D. D. 20555

Attention: Document Control Desk

Subject: McGuire Nuclear Station
Docket Numbers 50-369 and -370
Catawba Nuclear Station
Docket Numbers 50-413 and -414
Use of BWU-Z Correlation by Duke Power;
Supplemental Information

By letter dated April 26, 1996, Duke Power requested NRC approval for use of the BWU-Z correlation at its McGuire and Catawba nuclear stations. A supplement was provided by letter dated December 4, 1995. The December 4, 1996 letter (paragraph 4) stated that the better thermal performance of the fuel can be used to reduce cycle fuel costs. This is due to fact the licensed BWU-Z correlation conservatively quantifies the inherent thermal margin of the Mark-BW17 fuel. This margin can be used in fuel cycle analyses to raise peaking, thereby saving fuel costs. Additionally, the December 4, 1996 letter contained a typographical error in the last sentence of Paragraph 5. The references identified should be 5 and 6, not 6 and 7 as the letter stated.

During telcons on August 21 and 27, 1996, between the NRC staff and Duke, additional information/clarification was requested by the Staff. Attached are the questions and associated responses.

Note that upon approval of the new Appendix C (to topical report DPC-NE-2005), which was transmitted by the April 26, 1996 letter and contains the technical basis for the use of BWU-Z, the topical report will be republished, including the new Appendix C, as DPC-NE-2005, Revision 1.

Printed on recycled paper

If there are any questions or additional information is required, please call Scott Gewehr at (704) 382-7581.

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Request for Additional Information To Topical Report DPC-NE-2005P, Appendix C

The questions are shown in italics and the responses immediately follow.

1) What fuel type and core configuration are currently operating at McGuire and Catawba?

McGuire and Catawba are both operating with a full (homogenous) core of Mark-BW17 fuel assemblies, also called Mark-BW 17x17. This will be the fuel type until a transition, beginning in the year 2000, to Westinghouse 17x17 mixing vane fuel. Transition to Westinghouse fuel will require licensing of a different critical heat flux correlation and corresponding statistical design limit applicable to that fuel type.

2) Table C-1 lists the statistical results of the CHF test data base analysis with the VIPRE-01 thermal-hydraulic computer code. Explain the differences in values between this table and the table in BAW-10199P-A which documents the same data analysis with the LYNXT or LYNX 2 code.

The information provided for the Mark-BW17 data base using the BWU-Z correlation on the top of page 4-3 in BAW-10199P-A shows the average M/P, Standard Deviation (corrected for N), and Design Limit DNBR (denoted DNBR(L)) for the test data when analyzed with LYNXT or LYNX 2. Table C-1 of the DPC-NE-2005 Appendix C is a direct comparison of the same analysis and the same test data with VIPRE-01 code. The VIPRE-01 code has a slightly higher average M/P and slightly lower standard deviation for the entire test data base when compared to LYNXT or LYNX 2. The combination of these two parameters gives the VIPRE-01 code a slightly lower Design Limit DNBR for the test data base.

The more conservative value for the parameter is selected by Duke Power Company (DPC) when performing an analyses. For example, the standard deviation listed in Table C-3 (DPC-NE-2005 Appendix C) for the correlation uncertainty is the higher of the LYNX and VIPRE-01 values rounded to two significant figures. The Design Limit DNBR calculated with VIPRE-01 is presented in Table C-1 for comparison only. The standard deviation is the only value that impacts the SCD calculation. If the BWU-Z form of the BWU correlation is used by DPC in non-SCD analyses, the larger of the two non-statistical correlation limits (the LYNX value listed on page 4-3 of BAW-10199P-A) will be used.

3) Explain the method used to calculated the 500 and 5000 case statistical DNBR values for each statepoint and how the statistical limit is used.

The method used to evaluated the BWU-Z form of the BWU correlation in Appendix C is identical to the procedure outlined in the main body of the DPC-NE-2005 report. This procedure is outlined starting in Section 2.0 on page 5. The key parameters listed in Section 2.1, page 6, are identical in Appendix C. The statepoints in Table C-2 of Appendix C were selected to bound the range of key parameters where the SCD analyses with BWU-Z will be applied.

The selection of uncertainties is discussed in Section 2.2, page 7, of DPC-NE-2005. Table C-3 lists the values used in the BWU-Z SCD analyses. These are identical to the values used in the BWCMV analysis (Appendix B of DPC-NE-2005) except for the correlation standard deviation (as explained in Question 2 above) and the $F\Delta H$ measurement uncertainty which was increased slightly for the BWU-Z analysis.

The method for statepoint propagation is explained in Section 2.3, page 8, of DPC-NE-2005. The calculation of the statepoint statistical limit is explained in Section 2.4, page 9 through 11, of DPC-NE-2005. The equation for the SDL calculation is shown on page 10. Included in the equation are Chi Square and K factor multipliers to ensure a conservative limit based on the number of cases calculated. The mean and standard deviation values for a statepoint fluctuate slightly as the number of cases increase. Increasing the number of cases gives higher confidence that the data analyzed defines bounding behavior, therefore the multipliers are reduced. This ensures the SDL limit is equally conservative even though the final statistical DNBR value is smaller as the number of cases gets larger. An example of the way the values change with an increasing number of cases is shown on Table 1.

The main body of the report lists the number of cases as either 500 or 3000 per statepoint. The propagation method is identical regardless of the number of cases generated. In the response to Question 8 of Attachment II, Request For Additional Information, in DPC-NE-2005, DPC stated that the number of cases may be increased. This number was increased to 5000 for the BWU-Z analysis in Appendix C. As explained in the response to Question 8, this increase is consistent with the methodology and does not in any way reduce the conservatism of the SDL limit calculated.

The 5000 case number was selected as a balance between computer resources required for the calculations and the reduction in statistical uncertainty. For example, increasing the number of cases by two thirds from 3,000 to 5,000 reduces the K factor by 0.011 (from 1.692 at 3,000 to 1.681 at 5,000). Further increasing the number of cases to 10,000 would require another doubling of resources for the same K factor reduction (from 1.681 at 5000 to 1.670 at 10,000).

The 500 and 5000 case results for the BWU-Z analysis are listed in Table C-4 of Appendix C. As described in DPC-NE-2005, the 5000 case statepoints are selected based on the results of the 500 case statepoint propagations. The 5000 case runs are used to determine a conservative Statistical Design Limit (SDL) for the correlation SCD analyses. A value larger than the largest 5000 case statepoint statistical DNBR value is listed on page C-4. This is the statistical design limit that will be used in analyses with the BWU-Z form of the BWU correlation for Mark-BW fuel at McGuire and Catawba. The statistical design limit listed on page C-4 will be applicable to an analysis as long as all statepoint parameters fall between the Maximum and Minimum ranges listed on Table C-5.

TABLE 1

Statepoint 1 Values

Number Of Cases	Coefficient Of Variation*	Chi Square Multiplier	K Factor Multiplier	Statistical DNBR**
500	0.1514	1.05549	1.763	1.392
1000	0.1541	1.03848	1.727	1.382
1500	0.1528	1.03115	1.712	1.369
2000	0.1537	1.02684	1.703	1.368
2500	0.1534	1.02393	1.698	1.364
3000	0.1539	1.02179	1.692	1.363
3500	0.1538	1.02013	1.689	1.362
4000	0.1543	1.01880	1.686	1.361
4500	0.1540	1.01771	1.683	1.358
5000	0.1539	1.01681	1.681	1.357

* **Coefficient of Variation = Standard Deviation / Mean for the number of cases**

****Statistical DNBR** = $\dfrac{1}{[\,1 - \{(\text{Coeff. of Variation}) * (\text{Chi Square Mult.}) * (\text{K Factor Mult.})\}]}$

Appendix D and Response To Request for Additional Information

Revision 2

DPC-NE-2005-PA

Duke Energy Carolinas
Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX D

Oconee Plant Specific Data

Mark-B11 Fuel

Application of BWU-Z CHF Correlation to Mark-B11
Mixing Vane Spacer Grid Fuel Design

April 1997

This Appendix contains the plant specific data and limits for the Oconee Nuclear Station with Mark-B11 fuel using the BWU-Z form of the BWU critical heat flux correlation. The thermal hydraulic statistical core design analysis was performed as described in the main body of this report.

Plant Specific Data

This analysis is for the Oconee plant (two loop Babcock and Wilcox PWR's) as described in Reference D-1. The parameter uncertainties and statepoint ranges were selected to bound the unit and cycle specific values of the Oconee station. This analysis models the improved, small diameter, mixing vane grid, Mark-B fuel assembly denoted as the Mark-B11 design.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference D-3 and the Oconee eight and nine channel models approved in Reference D-1 are used in this analysis. Due to the fuel assembly design change, some specific data supplementary to Table 3-1 in Reference D-1 requires updating. This data is listed in Table

D-1. Table D-1 includes fuel rod, control rod, and instrument guide tube diameters, the number of mixing and non-mixing vane grids, and the fuel rod length. The following section compares Mark-B design fuel assemblies with the Mk-B11 fuel assemblies.

Previous Mark-B design fuel assemblies consisted of 0.430 inch diameter fuel rods with 2 inconel and 6 intermediate non-mixing vane zircaloy grids. The Mark-B11 fuel assembly design is composed of fuel pins with a 0.416 inch outside diameter and two inconel grids and six intermediate zircaloy grids, one non-mixing grid and five mixing vane grids. The higher pressure drop and higher cladding surface heat flux of the Mark-B11 design is offset by the larger flow area and the presence of the mixing vane grids to result in improved assembly thermal performance.

The VIPRE-01 models approved in Reference D-1 are used to analyze the Mark-B11 fuel with the following exceptions:

1) The Mark-B11 fuel assembly geometry information is listed in Table D-1.

2) The turbulent mixing factor has been changed from 0.01 to 0.038 for the Mark-B11 fuel assembly design due to the presence of mixing vane grids. The numerical value was determined and provided by the fuel supplier. This is consistent with Areva NP's 17x17 Mark-BW fuel assembly product and has been confirmed by Mark-B11 LDV test data.

3) The bulk void fraction model was changed from the Zuber-Findlay model to the EPRI. The Zuber-Findlay bulk void model is applicable only to qualities below approximately 0.7 and is discontinuous at a quality equal to 1.0 (Reference D-3). The EPRI bulk void model is essentially the same as the Zuber-Findlay bulk void model except for the equation used to calculate the drift velocity (Reference D-3). This eliminates the discontinuity at a quality equal to 1.0. Therefore, the EPRI model provides a full range (i.e., void fraction range, 0 - 1.0) of applicability required for performing DNB calculations. Also, for overall model compatibility, the subcooled void model was changed from LEVY, as specified in Reference D-1, to the EPRI correlation for the Mark-B11 fuel.

To evaluate the impact of changing bulk void models on DNB prediction, forty-four Mark-B11 CHF test data points (Reference D-2) were compared using both the Levy/Zuber-Findlay and EPRI/EPRI subcooled void/bulk void combinations in VIPRE-01. These data points cover a pressure range of 1005 to 2425 psia and an inlet temperature range 361.3 to 604.3°F. The mass flux at the MDNBR location varied from 0.542 to 2.963 Mlbm/hr-ft^2. The void fraction at the MDNBR location varied from 0.106 to 0.711. The equilibrium quality at the MDNBR location varied from -0.104 to 0.198. The results of this comparison are as follows:

	Levy/Zuber-Findlay	EPRI/EPRI
Minimum DNBR (Avg)	0.991	0.996

The minimum DNBR results show a minimal difference of 0.54% (0.005 in DNB). Therefore, the EPRI bulk void model and EPRI subcooled void correlation will be used in Mark-B11 analysis.

Critical Heat Flux Correlation

The NRC approved BWU-Z form of the BWU critical heat flux correlation with the Mark B11V multiplier described in Reference D-2 is used for all Mark-B11 analyses. This correlation was developed by AREVA NP for application to the Mark-B11 fuel design. The analysis in Reference D-2 was performed with the LYNXT thermal-hydraulic computer codes. This correlation was programmed into the VIPRE-01 thermal-hydraulic computer code by Duke Energy and the Mark B11V data base analyzed in its entirety. The results of this analysis are shown in Table D-2. The resulting Average M/P value, data standard deviation, and CHF correlation limit are within 1% of the values reported in Reference D-2, page E-4 (also shown on Table D-2 under LYNXT column).

Figures D-1 through D-4 graphically show the results of this evaluation. Figure D-1 shows there is no bias of measured CHF values to VIPRE-01 predicted values for the data base. Figures D-2 through D-4 show there is no bias with the VIPRE-01 calculated M/P

ratios with respect to mass velocity, pressure, or thermodynamic quality. These figures compare closely with the same parameter representations in Reference D-2.

Based on the results shown in Table D-2 and Figures D-1 through D-4, the BWU-Z form of the BWU CHF application correlation with the Mark-B11V multiplier, licensed in Reference D-2, can be used in DNBR calculations with VIPRE-01 for Mark-B11 fuel.

Statistical Core Design Analysis

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table D-3. These statepoints represent the range of conditions to which the statistical DNB analyses limit will be applied. The range of key parameter values analyzed is listed on Table D-6.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table D-4. The uncertainties were selected to bound the values calculated for each

parameter at Oconee. The uncertainties have not changed except for the rod power hot channel factor (Fq), core flow measurement, and DNBR correlation. The uncertainty for Fq has changed due to fuel design changes. The core flow measurement uncertainty was increased to ensure that it is bounding. This results in a more conservative SDL. The DNBR correlation uncertainty is the same as that stated in Reference D-5, page 4-3.

DNB Statistical Design Limits

The statistical DNBR limit for each statepoint evaluated is listed on Table D-5. Section 1 of Table D-5 contains the 500 case runs and Section 2 contains the 5000 case runs. The number of cases was increased from 3000 to 5000 as described in Attachment 1 of the main body of this report (DPC-NE-2005) and Appendix C (Reference D-4). All of the DNBR distributions are normally distributed. The maximum statistical DNBR value in Table D-5 (full core of Mark-B11 fuel) for 5000 propagations is [ ]. Therefore, the statistical design limit, using the BWU-Z form of the BWU CHF correlation with the Mark-B11V multiplier for Mark-B11 fuel at Oconee, is [ ] for the range of parameters given in Table D-6.

Transition Cores

The transition core model determines the impact of the geometric and hydraulic differences between the resident Mark-B10 series fuel and the new Mark-B11 design. The 9 channel model described in Reference D-1 is

used to evaluate the impact of transition cores containing Mark-B11 fuel. In Figure 4-5 in Reference D-1, Mark-B11 fuel is used instead of Mark-B6/7 and Mark-B10F/G fuel instead of Mark-B5. Therefore, channels 1 - 7 are modeled as Mark-B11 fuel, Channel 8 is modeled as Mark-B10F/G fuel, and Channel 9 is modeled as Mark-B11 fuel. The transition core analysis models each fuel type in those respective locations with the correct geometry. The form loss coefficients for each fuel design are input so the effect of crossflow out of the higher pressure drop mixing vane grid (Mark-B11) fuel is calculated.

A transition core penalty is evaluated by determining the DNBR impact on a Mark-B11 limiting assembly when analyzed with the 9 channel model. Once determined, several methods are available to conservatively compensate for the penalty. One method of compensating for the reduction in DNB performance due to the hydraulic effects of the conservatively modeled transition core is to explicitly apply a penalty to the Mark-B11 fuel generic peaking limits based on a full Mark-B11 core. Another option is to calculate maximum allowable peaking limits specifically modeling the transition core loading pattern in the detailed 64 channel model approved in Reference D-1. These methods will be used, as necessary, to determine the DNB effect of transition cores.

To evaluate the statistical DNB impact of the transition core, the most limiting statistical DNB statepoint (Statepoint 22 on Table D-5) was evaluated using the 9 channel model. This statepoint is designated TR22

in Table D-5. At 5000 cases, the statistical DNBR for statepoint TR22 is slightly greater than the limit for statepoint 22, but less than the statistical design limit, [ ]. Therefore, the statistical design limit, [ ], is bounding for Mark-B10/B11 transition cores; as well as, full Mark-B11 cores.

Figure D-1 - Mark-B11 Vane Data Base
VIPRE-01 Measured Versus Predicted CHF

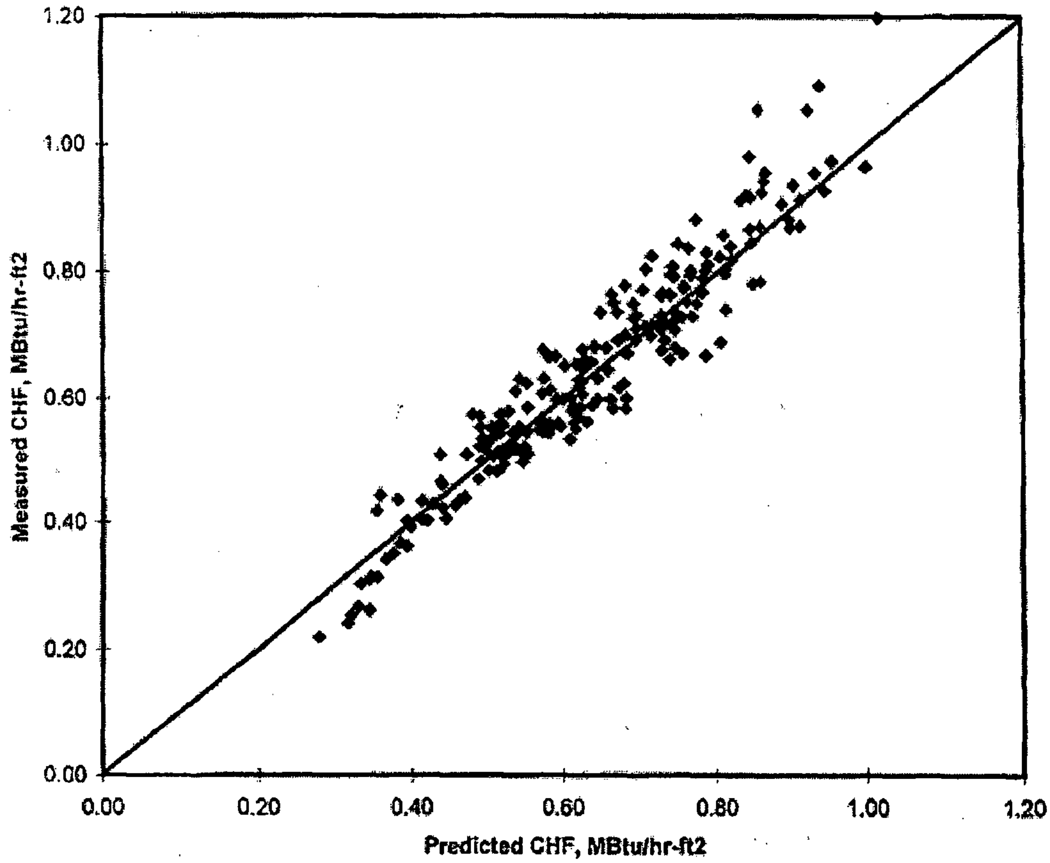

Figure D-2 - Mark-B11 Vane Data Base
VIPRE-01 Measured to Predicted CHF versus Mass Velocity

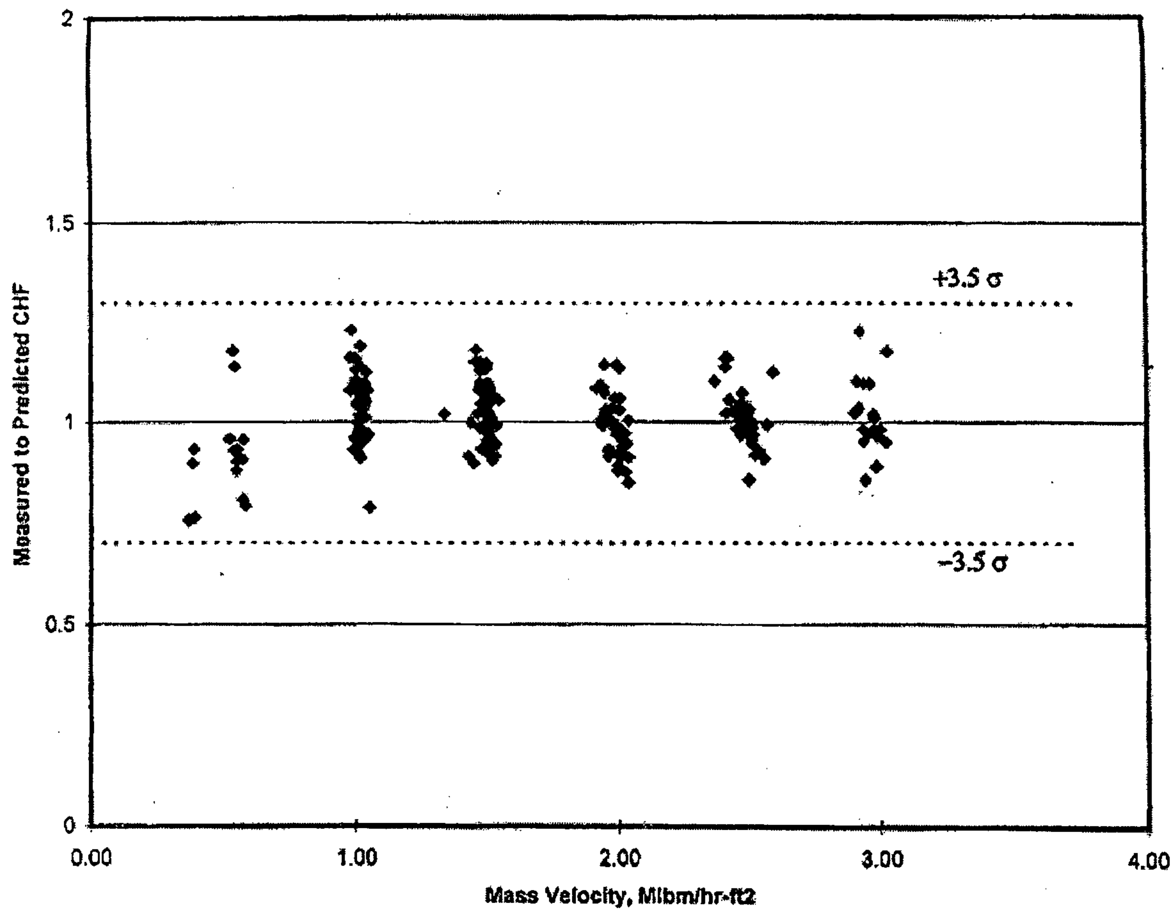

Figure D-3 - Mark-B11 Vane Data Base
VIPRE-01 Measured to Predicted CHF versus Pressure

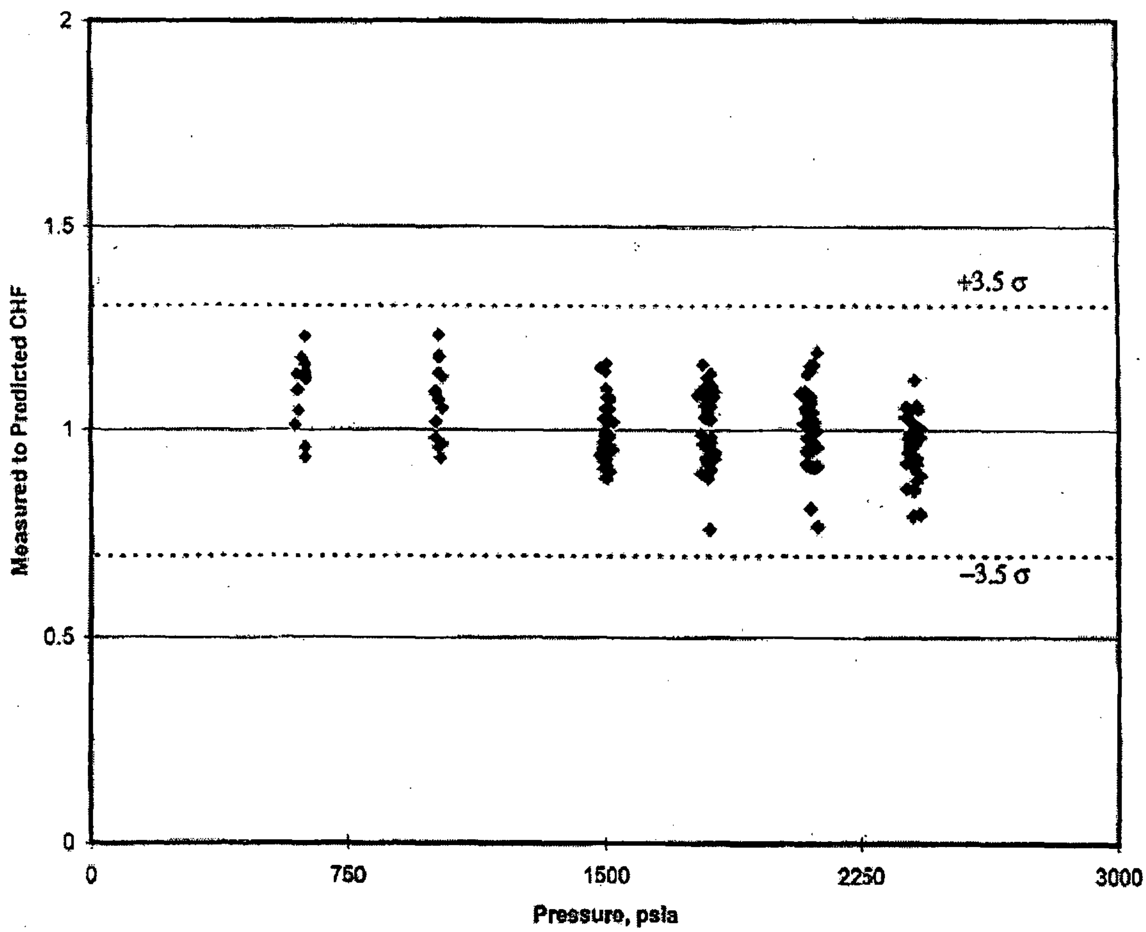

Figure D-4 - Mark-B11 Vane Data Base
VIPRE-01 Measured to Predicted CHF versus Quality

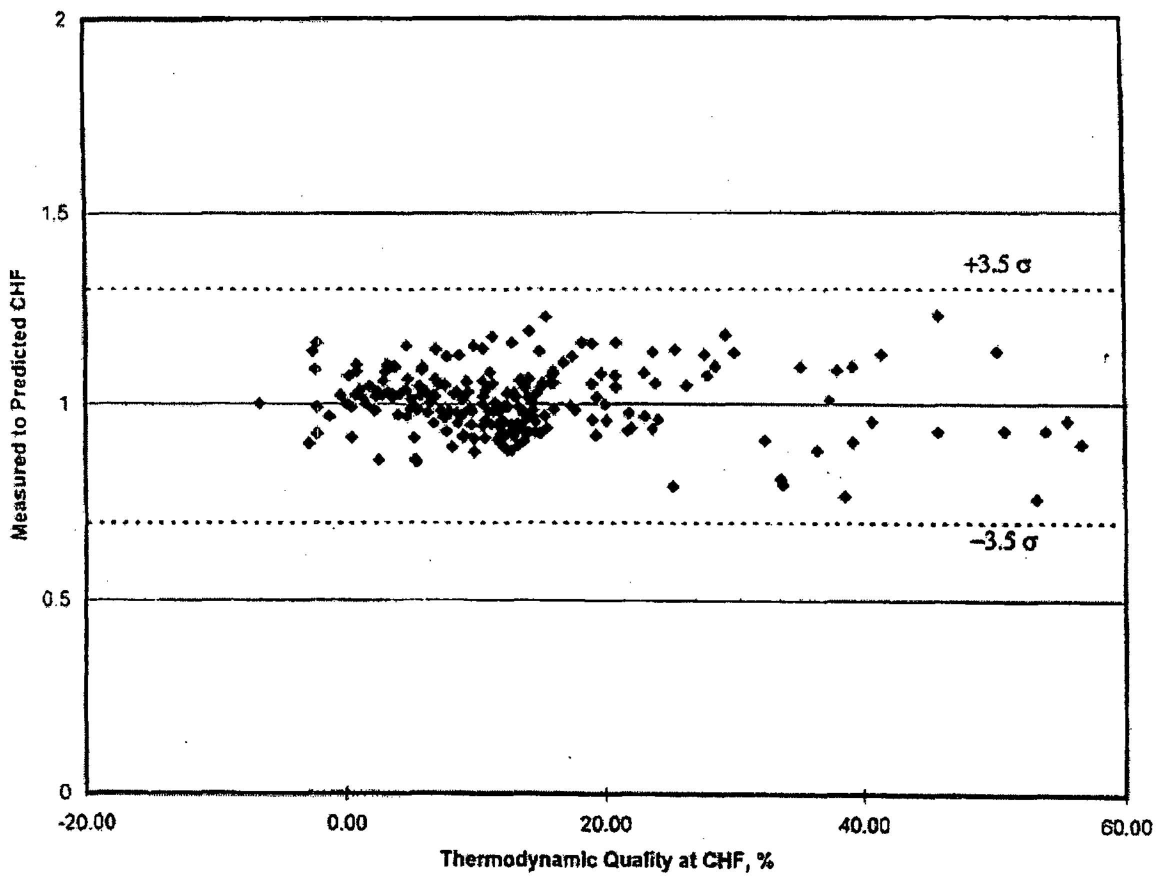

TABLE D-1 MARK-B11 FUEL ASSEMBLY DATA

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, in. (Nom.)	0.416
Thimble tube diameter, in. (Nom.)	0.530
Instrument guide tube diameter, in. (Nom.)	0.554(1)/0.567(2)
Fuel rod pitch, in (Nom.)	0.568
Fuel assembly pitch, in. (Nom.)	8.587
Fuel rod length, in. (Nom.)	154.16

(1) Above lowest mixing vane grid (MV) and between MV grids.

(2) Below the first mixing vane grid and above the top of the last mixing vane.

GENERAL FUEL CHARACTERISTICS

Grids:	Material	Quantity	Location	Type
	Inconel	2	Upper and Lower	Non-Mixing Vane
	Zircaloy	6	Intermediate	1 Non-Mixing Vane, 5 Mixing Vane

Fuel Rods:	Material	Quantity
	Zircaloy-4	208

Fuel Cycle Design Assembly Features

Fuel Assy. Mark

Designation: B11

Features: Smaller clad outside diameter and mixing vane grids.

TABLE D-2 VIPRE-01 BWU-Z Correlation with Mark-B11V Multiplier Verification

CHF Test Database Analysis Results

VIPRE-01/LYNXT Statistical Results

	VIPRE-01	LYNXT
n, # Of data	216	216
N, degrees of freedom (n-1)	215	215
M/P, Average measured to predicted CHF	1.0084	1.0040
σ (M/P/N)	0.0859	0.0868
K(215,0.95,0.95),one sided tolerance factor Ref. D-2)	1.830	1.830
DNBR(L) = $1/(M/P - K\sigma) = 1/[1.0040 - 1.830(0.0868)]$	1.175	1.183

Parameter Ranges

Pressure, psia	400 to 2465
Mass Velocity, Mlbm/hr-ft^2	0.36 to 3.55
Thermodynamic Quality at CHF	less than 0.74
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Mark-B11 15x15 Mixing Vane
Design Limit DNBR, VIPRE-01	1.19*

* The correlation design limit DNBR (1.19) applies only at or above a nominal pressure of 1000 psia (Reference D-2). In the low pressure region (below a nominal pressure of 1000 psia) the design limit DNBR in the following table will be used (Reference D-2):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.20

TABLE D-3 Oconee SCD Statepoints

Statepoint Number	Power[1] (% RTP)	RCS Flow[2] % DF	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak $F\Delta H$
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						
25						
26						
27						
TR22						

1) 100% RTP = 2568 Megawatts Thermal
2) 100% design flow is equal to 352,000gpm.

TABLE D-4 Oconee Statistically Treated Uncertainties

Parameter	Type	Type of Distribution	Uncertainty	Standard Deviation
Reactor System				
Core Power*	Measurement	Normal	+/-2.0%FP	+/-1.0%FP
Core Flow	Measurement	Normal	4 Pump: +/-2.0% 3 Pump: +/-3.2% 2 Pump: +/-4.2% design	+/-1.0% +/-1.6% +/-2.1% design
Pressure	Measurement	Normal	+/-30.0 psi	+/-15.0 psi
Temperature	Measurement	Normal	+/-2.0°F	+/-1.0°F
Nuclear				
FΔH	Calculation	Normal	---	+/-2.84%
Fz	Calculation	Normal	---	+/-2.91%
Z	Calculation	Uniform	+/-6 inches	---
Fq	Calculation	Normal	[	]
Hot Channel Flow Area	Measurement	Uniform	[ ]	---
DNBR	Correlation	Normal	---	9.268%
DNBR	Code	Normal	[	]

* Percentage of 100% RTP (69.75 MWth wherever applied).

TABLE D-4 Continued Oconee Statistically Treated Uncertainties

Parameter	Justification
System Pressure	This uncertainty accounts for random uncertainties in various instrumentation components. Since the random uncertainties are normally distributed, the square root of the sum of the squares (SRSS) that results in the pressure uncertainty is also normally distributed.
Inlet Temperature	Same approach as Pressure uncertainty.
Core Power	The core power uncertainty was calculated by statistically combining various random uncertainties associated with the measurement of core power. Since the random uncertainties are normally distributed, the SRSS that results in the core power uncertainty is also normally distributed.
Core Flow	Same approach as Core Power uncertainty.
Radial Power, $F_{\Delta H}$	This uncertainty accounts for the error associated in the physics code's calculation of radial assembly power and the measurement of the assembly power. This uncertainty distribution is normal.
Axial Peak Power, F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty is normally distributed.
Axial Peak Location, Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.

TABLE D-4 Continued   Oconee Statistically Treated Uncertainties

Parameter	Justification
Rod Power HCF, Fq	This uncertainty accounts for the increase in rod power due to manufacturing tolerances. The uncertainty in calculating the peak pin from assembly radial peak is also statistically combined with the manufacturing tolerance uncertainty to arrive at the correct value. The uncertainty is normally distributed and conservatively applied as one-sided in the analysis to assure the MDNBR channel location is consistent for all cases.
Hot Channel Flow Area	This uncertainty accounts for manufacturing variations in the instrument guide tube subchannel flow area. This uncertainty is uniformly distributed and is conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
DNBR - Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normally distributed.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is normally distributed.

TABLE D-5 Oconee Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation With Performance Factor

500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
25				
26				
27				
TR22				

TABLE D-5 Continued   Oconee Statepoint Statistical Results

BWU-Z Critical Heat Flux Correlation With Performance Factor

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1-T				
3-T				
17-T				
21-T				
22-T				
24-T				
TR22-T				

TABLE D-6 Oconee Key Parameter Ranges

Parameter	Maximum	Minimum
Core Power (%RTP)		
Pressure (psia)		
T inlet (deg F)		
RCS Flow (% Design)		
FΔH, Fz, Z		

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

REFERENCES

D-1. DPC-NE-2003-PA, Rev. 2a, Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, December 2008.

D-2. The BWU Critical Heat Flux Correlations, Addendum 1 to BAW-10199P-A, AREVA NP, Lynchburg, Virginia, April 6, 2000

D-3. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

D-4. DPC-NE-2005-PA, Rev. 4a, Duke Energy Carolinas Thermal-Hydraulic Statistical Core Design Methodology, Appendix C, December 2008.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
July 1, 1998

RECEIVED
JUL 7 1998
DUKE POWER CO.
NUCLEAR ENGINEERING

Mr. William R. McCollum
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
Seneca, South Carolina 29679

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - APPENDIX D, "OCONEE PLANT SPECIFIC DATA, MARK-B11 FUEL, APPLICATION OF BWU-Z CHF CORRELATION TO MARK-B11 MIXING VANE SPACER GRID FUEL DESIGN," TO DPC-NE-2005P, "DUKE POWER COMPANY THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY" (TAC NOS. 98660, 98661, AND 98662)

Dear Mr. McCollum:

By letter dated April 22, 1997, Duke Energy Corporation transmitted the subject topical report, Appendix D to DPC-NE-2005P, for staff review. In order to complete its review, the NRC staff has determined that additional information is needed. The staff's request for additional information is enclosed.

Sincerely,

David E. LaBarge, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure: Request for Additional Information

cc w/encl: See next page

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Request for Additional Information
Review of Duke Topical Report DPC-NE-2005P,
Appendix D, "Oconee Plant Specific Data, Mark-B11 Fuel,
and Application of BWU-Z CHF Correlation to Mark-B11
Mixing Vane Spacer Grid Fuel Design"
to DPC-NE-2005P, "Duke Power Company
Thermal-hydraulic Statistical Core Design Methodology"

1. The safety evaluation report (SER) for DPC-NE-2005P-A requires that in all applications of the statistical core design methodology, the uncertainties and distributions used in the analysis will be justified on a plant-specific basis. This has not been done in Appendix D, which presents plant-specific data for Oconee with Mark-B11 fuel. Comparing Table D-4 of Appendix D with Table A-2 of Appendix A (which contains the approved uncertainties and distributions for the Oconee units with Mark-B10 fuel), there are four major changes, none of which are explained adequately. Specifically:

(a) The core flow uncertainty has been increased from ±2.0 percent design (with standard deviation ±1.0 percent design) to ±4.2 percent design (with standard deviation ±2.1 percent design). The report says simply that the value was increased "to ensure that it is bounding." Bounding in what way? How was this determined? It appears to be an arbitrary adjustment to what should be a real indicator of the uncertainty in the measured core flow. What is the justification for this change?

(b) Table A-2 includes the parameter Fq" (local heat flux hot channel flow (HCF)). which is an uncertainty to account for the decrease in departure from nuclear boiling ratio (DNBR) at the point of minimum DNBR due to engineering tolerances. It also accounts for flux depression at a spacer grid, and has a value of +2.08 percent (with standard deviation +1.26 percent). This parameter has been omitted from Table D-4. What is the justification for this change?

(c) Table A-2 includes the parameter Fq (rod power HCF), which is an uncertainty to account for rod power increases due to manufacturing tolerances. This parameter also includes the uncertainty in calculating the pin peak from the assembly radial peak, and has a value of +2.27 percent (with standard deviation +1.38 percent) for Mark-B10 fuel. In Table D-4, this parameter has the value +2.40 percent (with standard deviation +1.46 percent) for Mark-B11 fuel. How was this uncertainty determined, and why is it larger for Mark-B11 fuel than for Mark-B10 fuel?

(d) The HCF area uncertainty is reported as -3.00 percent in Table D-4, unchanged from the value in Table A-2 for Mark-B10 fuel, even though there are significant differences in the assembly geometry of Mark-B11 fuel. In addition, the value reported for the parameter Fq indicates that there are significant differences in the manufacturing tolerances for the fuel rods in Mark-B11 fuel, which would seem to imply that there should also be significant differences in the flow channel geometry variations. What is the justification for using the value of -3.00 percent for this parameter?

Enclosure

2. The description of how transition cores will be treated is unclear. Please provide additional information, addressing the following points:

   (a) What is a "transition core penalty," and how is it determined?

   (b) The local pressure drop differences-between the Mark-B10 and Mark-B11 fuel assemblies mean that the local assembly flow distributions may be very different in a mixed core, due to differences in inter-assembly crossflow patterns. The departure from nucleate boiling (DNB) behavior of a mixed core may, therefore, be significantly different from that of a full core of Mark-B11 fuel only. Justify the assumption that the BWU-Z CHF correlation can be applied to Mark-B11 fuel in a core containing both Mark-B10 and Mark-B11 fuel.

   (c) Two options are described (see p. D-7) that will be used to "conservatively compensate" for the transition core penalty. The report states that they will be applied "as necessary" to determine the DNB effect of a transition core. What are the criteria for selecting one or the other of the two options? How will it be determined that the selected option is "conservative" for a given transition core?

   (d) One option of the two described on p. D-7 is to explicitly apply a penalty to the Mark-B11 fuel generic peaking limit based on a full Mark-B11 core. What is this penalty? How is it determined? How will it be determined that the penalty adequately accounts for the effects of a mixed core on DNB behavior?

3. Table D-2 (p. D-14) claims a pressure range of 400 to 2465 psia for the BWU-Z correlation with the Mark-B11V multiplier. The database supporting this form of the correlation includes tests only over the pressure range 695 to 2425 psia. In addition, there is a distinct nonconservative bias evident in the correlation's predictions with decreasing pressure (see Figure D-3, p. D-11). The BWU-Z correlation for Mark-BW17 fuel (as documented in BAW-10199-A) has a demonstrated bias with decreasing pressure, and the SER for this correlation specifies a separate design limit DNBR of 1.59 for pressures below 700 psia. If the BWU-Z correlation with the Mark-B11V multiplier is to be applied to conditions where the pressure is below 700 psia, what value will be used for the design limit DNBR and how will it be determined?

4. The SER for DPC-NE-2005P-A requires that the selected state points for an application of the SCD methodology shall be justified to be appropriate, on a plant-specific basis. Documentation of this justification in Appendix D consists only of the statement on p. D-1, "...state point ranges were selected to bound the unit and cycle-specific values of the Oconee Station." However, the document also notes that the values of the key parameter ranges used to define the state points (in Table D-6, p. D-21) are "based on the currently analyzed state points," and further notes that "ranges are subject to change based on future state point conditions." The procedure and justification for selecting state points is unclear, and additional information is needed. Specifically, please provide a more detailed description of how the state points are selected for the Oconee plant-specific data, with particular attention to how bounding values are to be determined for the B11 and mixed B10/B11 cores.

5. The calculations with the VIPRE-01 code using the BWU-Z correlation form for B11 fuel show essentially the same results as the those obtained with LYNX over the correlation's database (as documented in Addendum 1 of BAW-10199). However, the BWU-Z correlation as modified for analysis of B11 fuel has not yet been approved by the staff, and the topical report describing this correlation, Addendum 1 of BAW-10199, is still being reviewed. This means that the design limit DNBR for the parameter ranges stated in Table D-2 may not be the final approved value or range of applicability. Specifically, the database for the form of the correlation spans a pressure range of 700 to 2400 psia, not the 400 to 2400 psia range stated in Table D-2. Also, the plot in Figure D-3 (see p. D-11) shows a distinct nonconservative bias with decreasing pressure (which is identical to the trend shown for the correlation in the Addendum 1 submittal). There is also a nonconservative bias with increasing power, clearly shown by the plot of measured versus predicted Critical Heat Flux (CHF) in Figure D-1. What would be the effect on the thermal hydraulic statistical core design analysis for Oconee if the DNBR design limit of the CHF correlation for B11 fuel were to be increased, or if the range of applicability of the correlation were to be limited to pressures of 700 to 2400 psia?

Duke Energy

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Executive Vice President
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September 21, 1998

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001
Attention: Document Control Desk

Subject: Duke Energy Corporation

Oconee Nuclear Station, Units 1, 2 and 3
Docket Numbers 50-269, 50-270, and 50-287

Response to NRC Request for Additional Information on Appendix D to Topical Report DPC-NE-2005-P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology"

This submittal contains information that Duke Energy Corporation considers PROPRIETARY and is being made pursuant to 10CFR 2.790.

By letter dated July 1, 1998 the NRC requested additional information on Appendix D to Topical Report DPC-NE-2005P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology." This topical report had been previously submitted for NRC review by Duke letter dated April 22, 1997.

The questions contained in the July 1 NRC letter, and the corresponding Duke answers, are provided in the attachment to this letter. Additionally, Table D-1, which is also included in the attachment, has been revised to correct a typographical error.

Some of the information contained in the attachment is considered proprietary. In accordance with 10CFR 2.790, Duke Energy Corporation requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of the affected information is included with this letter. A non-proprietary version of the affected material is also included.

Please address any comments or questions regarding this matter to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc:

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L. A. Keller
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C. L. Naugle
ELL

AFFIDAVIT OF M. S. TUCKMAN

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

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

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, 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 and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. 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.

M. S. Tuckman

M. S. Tuckman

(Continued)

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, 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 this submittal is that which is marked in the attachment to Duke Energy Corporation letter dated September 21, 1998; SUBJECT: Response to NRC Request for Additional Information on Topical Report DPC-NE-2005P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methology." This information enables Duke to:

(a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

(b) Support Facility Operating Licenses/Technical Specifications amendment requests for Babcock & Wilcox PWRs.

(c) Perform safety evaluations per 10CFR50.59.

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

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

M. S. Tuckman

M. S. Tuckman

(Continued)

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

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

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

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M.S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 22nd day of September, 1998

Linda Case Smith

~~Mary P. Nelms~~ Notary Public
Linda Case Smith

My Commission Expires: ~~January 22, 2001~~ May 6, 2000

SEAL

NRC Questions On Mark-B11 SCD Submittal

Questions shown in italics, answers immediately follow.

1. *The safety evaluation report (SER)for DPC-NE-2005P-A requires that in all applications of the statistical core design methodology, the uncertainties and distributions used in the analysis will be justified on a plant-specific basis. This has not been done in Appendix D, which presents plant-specific data for Oconee with Mark-BII fuel. Comparing Table D-4 of Appendix D with Table A-2 of Appendix A (which contains the approved uncertainties and distributions for the Oconee units with Mark-BIO fuel), there are four major changes, none of which are explained adequately. Specifically:*

   a) *The core flow uncertainty has been increased from +/- 2.0 percent design (with standard deviation of +/-1.0 percent design) to +/-4.2 percent design (with standard deviation +/-2.1 percent design). The report says simply that the value was increased "to ensure that it is bounding." Bounding in what way? How was this determined? It appears to be an arbitrary adjustment to what should be a real indicator of the uncertainty in the measured flow. What is the justification for this change?*

The current Chapter 15 analyses for Oconee were performed by FCF. Duke Power has recently reanalyzed the Chapter 15 transients and submitted a topical report that is currently being reviewed (DPC-NE-3005). As part of this effort, Duke recalculated the flow uncertainties for combinations of 4,3, and 2 operating reactor coolant pumps. Table D-4 has been revised to show the flow uncertainties used in the BWU-Z SCD analyses (the original table only listed the maximum flow uncertainty for 2 pump operation). The statepoints listed in Table D-3 were propagated using the appropriate flow uncertainty. For example, statepoint 22 is the limiting statepoint for the 2 pump coastdown transient. Thus, this statepoint was propagated using a flow uncertainty of 4.2 % (std. deviation of 2.1 %).

Statepoints using the flow uncertainties for 3 and 2 operating reactor coolant pumps have also been propagated using the BWC correlation. The statistical DNB limit for all cases using the higher flow uncertainties was less than the SCD limit given in Appendix A. Thus, no NRC submittal was required based on the criteria given in Table 7 of DPC-NE2005.

b) *Table A-2 includes the parameter Fq" (local heat flux hot channel flow (HCF)), which is an uncertainty to account for the decrease in departure from nuclear boiling ratio (DNBR) at the point of minimum DNBR due to engineering tolerances. It also accounts for flux depression at a spacer grid, and has a value of [+2.08 percent] (with standard deviation [+1.26 percent]). This parameter has been omitted in Table D-4. What is the justification for this change?*

The local heat flux hot channel factor accounts for the effects on DNBR of local variations in pellet enrichment and weight on local (hot spot) power, and flux depressions at spacer grid. Small local heat flux spikes have been shown to have no effect on the critical heat flux (CHF) per the Oconee 1 Cycle 14 Reload Report, DPC-RD-2018.

DPC-RD-2018 was submitted as supplementary information in support of Technical Specification changes for Oconee Unit 1 (Amendment 191, TAC 80378), Unit 2 (Amendment 191, TAC 80379), and Unit 3 (Amendment 188, TAC 80380). NRC approved the Technical Specification change. Duke considered the NRC's implementation of the recommended Technical Specification changes to be implicit approval of DPC-RD-2018.

Removal of Fq" from DNBR analyses was justified in DPC-RD-2018 based on WCAP-8202 and CENPD-207. The WCAP-8202 evaluation concluded that the data and analysis clearly indicate no effect on the minimum DNBR due to large local heat flux spikes. The spikes tested were in the region of MDNBR and were 20% greater than the heat flux in the immediate vicinity. The conclusion of these reports is that the local heat flux spikes associated with fuel densification have no affect on DNBR. This effect is generic to PWR fuel types and was confirmed to be applicable by the fuel vendor. Additionally, the magnitude of Fq" calculated by the vendor for Mark-B11 fuel is much smaller, [ ]. Based on this information and the approved reload report submittal, the Fq" factor was omitted from the Mark-B11 analyses and, therefore, Table D-4.

c) *Table A-2 includes the parameter Fq (rod power HCF), which is an uncertainty to account for rod power increases due to manufacturing tolerances. This parameter also includes the uncertainty in calculating the pin peak from the assembly radial peak, and has a value off [+2.27 percent] (with standard deviation [+1.38 percent]) for Mark-B10 fuel. In Table D-4, this parameter has the value of [+2.40 percent] (with standard deviation of [+1.46 percent]) for Mark-B11 fuel. How was this uncertainty determined, and why is it larger for Mark-B11 fuel than for Mark-B10 fuel?*

A rod power hot channel factor of [ ] was specified in the original issue of DPC-NE-2005 for Mark-B10 fuel. This value has increased to [ ] for the Mark-B10 fuel (beginning with Oconee 1, 2, and 3 Batch 17) and [ ] for the Mark-B11 fuel to account for dry blending of UO_2 powder to achieve the desired enrichment. The Statistical Design Limit (SDL) given in DPC-NE-2005 was shown to still be valid for Mark-B10 fuel using the increased rod power hot channel factor evaluated as per the process described in Table 7 of DPC-NE-2005P-A.

The value for F_q used in the SCD analysis is calculated as follows based on the rod power hot channel factor of [ ] for Mark-B11 provided by the fuel manufacturer and the radial peak uncertainty of [ ] per DPC-NE-1004P-A.

$$F_q\% = [\quad]$$

The standard deviation is calculated as follows:

$$\sigma(F_q)\% = \frac{[\quad]}{1.645}$$

$$= [\quad]$$

The 1.645 is the one-sided 95/95 statistical K factor for an infinite number of points.

The rod power hot channel factor, Fq, is provided as part of the fuel fabrication process and is formally transmitted in batch specific design and fabrication data supplied by the fuel manufacturer for each reload batch. If the rod power hot channel factor were greater than [ ], then the fuel manufacturer would notify Duke Power and the impact on the SDL will be evaluated.

d) *The HCF area uncertainty is reported as [-3.00 percent] in Table D-4, unchanged from the value in Table A-2, for Mark-B10 fuel, even though there are significant differences in the assembly geometry of Mark-B11 fuel. In addition, the value reported for the parameter Fq indicates that there are significant differences in the manufacturing tolerances for the fuel rods in Mark-B11 fuel, which would seem to imply that there should also be significant differences in the flow channel geometry variations. What is the justification for using the value of [-3.00 percent] for this parameter?*

The value listed in Table D-4 was provided by the fuel manufacturer. The fuel fabricator will verify through inspection of the final fuel assemblies and components that the uncertainty on flow area assumed in the analysis is valid. The same inspection techniques employed in earlier designs will be used for the Mark-B11 fuel. Comparison to acceptance criteria for the Mark-B11 fuel will ensure compliance with the [ ] flow area uncertainty. Water channel data taken on the Mark-B11 lead test assemblies were evaluated and found to be acceptable. If the flow area uncertainty is greater than [ ], then the fuel manufacturer will notify Duke Power and the impact on the SDL will be evaluated as per the process described in Table 7 of DPC-NE-2005P-A.

2. *The description of how transition cores will be treated is unclear. Please provide additional information, addressing the following points:*

   a) What is a "transition core penalty," and how is it determined?

A generic transition core penalty is determined by comparing the DNBR results from a full core of Mark-B11 fuel with the DNBR results from a conservative Mark-B11/Mark-B10 transition core. The 9 channel transition core model licensed in DPC-NE-2003 and described in Appendix D of DPC-NE-2005 is used in this analysis. The MDNBR (or allowable radial peaking) is calculated with both models for a range of fluid conditions and axial peaking combinations. The largest penalty calculated from this matrix of conditions is used as the transition core penalty.

The process for determining a generic transition core penalty is as follows:

1. Develop an 8 channel Mark-B11 full core model and a 9 channel Mark-B11/Mark-B10 transition core model (per DPC-NE-2003P-A and described in Appendix D of DPC-NE-2005).

2. Evaluate each model for a range of fluid conditions (shown below) that is representative of fluid conditions for which the Maximum Allowable Peaking limits are developed using VIPRE-01. These fluid conditions are evaluated at the axial peaking conditions shown below.

Parameter	**Maximum**	**Minimum**
Core Power (% RTP)	110	80
RCS Flow (% Design Flow)	107.5	80.3
T inlet (deg F)	872.8	529.2
Pressure (psia)	2242	1830
Fz (normalized axial peak)	1.1,1.4,1.7,2.1	
z (location of axial peak)	0.2, 0.4, 0.6, 0.8	

The radial peaking results from VIPRE-01, for both the Mark-B11 full and transition core models at each fluid condition, were compared to determine the limiting fluid condition. Then a complete set of MAP curves were developed for both the full core Mark-B11 and the transition core models. These peaking results were compared, and a maximum transition core peaking penalty was determined. In addition, the axial dependency of the transition core penalty was determined. The response to question 2c) specifies the options for the application of the transition core penalty.

b) *The local pressure drop differences between the Mark-B10 and Mark-B11 fuel assemblies mean that the local assembly flow distributions may be very different in a mixed core, due to differences in inter-assembly crossflow patterns. The departure from nucleate boiling (DNB) behavior of a mixed core may, therefore, be significantly different from that of a full core of Mark-B11 fuel only. Justify the assumption that the BWU-Z CHF correlation can be applied to Mark-B11 fuel in a core containing both Mark-B10 and Mark-B11 fuel*

In mixed cores, the possibility of large axial velocity upsets at or around dissimilar grids exists. These upsets imply different local thermal-hydraulic conditions in surrounding subchannels. It has been questioned as to whether traditional steady state CHF correlations are applicable in this instance.

The FCF CHF correlation form (BWU) is composed of three parts: a uniform part dependent solely on the local thermal-hydraulic conditions of pressure, mass velocity and thermodynamic quality at the axial location of CHF, a non-uniform F factor modification dependent on the shape of the axial heat flux input, and a multiplicative geometric factor dependent on the overall fuel assembly grid spacing and heated length. It is with the uniform, local conditions part that the mixed core conditions question surfaces.

CHF correlations are developed from data from full length electrically heated bundles in 5-by-5 rod arrays. For each data point, the inlet conditions of coolant mass velocity, pressure and temperature are known, as is the power (heat flux) required to produce a DNB event. The local thermal-hydraulic conditions at the axial location of CHF must then be calculated with a computer code.

The proof of applicability of a CHF correlation, then, is how well it can predict the critical heat flux that was measured in the DNB event using the calculated local conditions. Thus, the applicability of a CHF correlation is dependent not only on its form and data base, but on the accuracy with which the local conditions can be calculated in any given situation. Because of the size of the test section (a 5-by-5 rod array) and the use of a series of single spacer grids (axially), normal CHF tests do not exhibit large hydraulic axial differences. FCF, however, has performed one test with widely varying subchannel axial resistances producing the large velocity upsets representative of mixed core conditions. This test was a 5-by-5 test of the Mark B zircaloy grid modeled as the corner intersection of four fuel assemblies. LDV testing of the intersection grid showed velocity depressions as large as 50% between the intersection subchannel and the surrounding unit cell subchannels. This test was conducted at the Babcock & Wilcox Alliance Research Center and is documented in BAW-10143P-A (BWC correlation of Critical Heat Flux, April, 1985). In the topical, the measured to predicted (M/P) CHF results were compared for two traditional test bundles and the intersection bundle. The guide tube bundle (B15) had an average M/P of 0.971, the unit cell bundle (B16) 0.985 and the intersection bundle (B17) 0.976. The difference in M/P results is statistically

insignificant. This qualified the BWC correlation for use with the Mark B fuel assembly design.

The local conditions necessary for the BWC correlation were calculated with a thermal-hydraulic computer code. The local conditions for the normal unit and guide tube bundles had very little axial upset, while the intersection bundle (which produces conditions representative of a mixed core) had severe upsets resulting from the two to one velocity upsets. The fact that the BWC correlation performed consistently on conditions representative of both homogeneous and mixed cores confirms that the FCF local conditions CHF correlations are valid for both homogeneous and mixed core applications as long as the local conditions can be accurately predicted by the subchannel thermal-hydraulic computer code.

In this particular application, the velocity upsets calculated in the Mark-B11 transition core analysis are on the order of 10%. These calculations assume the limiting geometry (a single Mark-B11 assembly surrounded by Mark-B10 fuel). Since the test data included local depressions as large as 50%, the FCF test results bound by a significant amount the transition core configuration.

c) *Two options are described (see p. D-7) that will be used to "conservatively compensate" for the transition core penalty. The report states that they will be applied "as necessary" to determine the DNB effect of a transition core. What are the criteria for selecting one or the other of the two options? How will it be determined that the selected option is "conservative" for a given transition core?*

The three methods for penalizing a transition core are to

1) Penalize the DNBR limit used in the analyses directly or
2) Penalize the Maximum Allowable Peaking Total (MATP) limits determined for a transition core
3) Use a combination of the two above.

The penalty applied using either method 1 or 2 is based on the most limiting transition core statepoint determined as described in the response to Question 2(a) above. This ensures either option is conservative for the transition core.

Option 3) listed above is the current one selected to provide a bounding, conservative transition core penalty while maximizing core design flexibility. As described in Question 2(a), the transition core penalty was evaluated with a subset of axial peak locations (Fz, Z) over a wide range of fluid conditions. Then, the fluid condition with the largest penalty was evaluated with a complete set of axial peaks (Fz from 1.1 to 2.1, Z from 0.01 to 1.0). This is the same set of axial peak locations used to generate the MATP limits and resulting curves described in DPC-NE-2003.

In this analysis, the transition core penalty shows axial shape dependence. Due to this relationship, it is reasonable to include part of the penalty directly in the applicable MATP limits. Based on the analysis described above, the transition core penalty is applied as follows:

1. A 0.5% peaking penalty (1.5% DNB penalty) is applied to the retained DNB margin available between the SDL and the DDL for Mark-B11 transition cores. This directly applies the penalty to all DNB calculations.

2. A 1% radial peaking penalty is applied to selected axial peak locations. These are generally the large axial peaks (Fz of ~1.4) in the top half of the core (at Z ~ 0.6). Again, these axial peak locations were determined by comparison of a complete set of MATP curves. This penalty will be applied to all Mark-B11 MATP's limits at the axial peaking locations necessary.

Duke will retain the option of applying any of the three methods described as a conservative transition core penalty such that cycle design impact is minimized.

d) *One option of the two described on p. D-7 is to explicitly apply a penalty to the Mark-B11 fuel generic peaking limit based on a full Mark-B11 core. What is the penalty? How is it determined? How will it be determined that the penalty adequately accounts for the effects of a mixed core on DNB behavior?*

The transition core penalty is determined as described in the response to Question 2(a) above. This adequately accounts for the mixed core effect as explained in the response to Question 2(b). As stated in the response to Question 2(c), the transition core penalty can be applied to the maximum allowable peaking (MAP) limits calculated for a full Mark-B11 core. This reduces the allowable peaking in the transition core to account for the hydraulic and geometry effects. This also ensures that the MDNBR in all transition core analyses is greater than the licensed SDL.

As with previously licensed transition core methods, the transition core geometry for a reload cycle can be specifically modeled using the 64 channel model described in DPC-NE-2003. This larger model allows analyses of the actual cycle loading pattern to determine the impact of a mixed core on the maximum allowable peaking limits for the transition cycle.

3. *Table D-2 (p. D-14) claims a pressure range of 400 to 2465 psia for the BWU-Z correlation with the Mark-B11V multiplier. The database supporting this form of the correlation includes tests only over the pressure range 695 to 2425 psia. In addition, there is a distinct nonconservative bias evident in the correlation's predictions with decreasing pressure (See Figure D-3, p. D-11). The BWU-Z correlation for Mark-BW17 fuel (as documented in BAW-10199-A) has demonstrated bias with decreasing pressure, and the SER for this correlation specifies a separate design limit DNBR of 1.59 for pressures below 700 psia. If the BWU-Z correlation with the Mark-B11V multiplier is to be applied to conditions where the pressure is below 700 psia, what value will be used for the design limit DNBR and how will it be determined?*

See Question 5 for response.

4. *The SER for DPC-NE-2005P-A requires that the selected state points for an application of the SCD methodology shall be justified to be appropriate, on a plant-specific basis. Documentation of this justification in Appendix D consists only of the statement on p. D-1 "... state point ranges were selected to bound the unit and cycle-specific values of the Oconee Station." However, the document also notes that the values of key parameter ranges used to define the state points (Table D-6, p. D-21) are "based on the currently analyzed state points," and further notes that "ranges are subject to change based on future state point conditions." The procedure and justification for selecting state points is unclear, and additional information is needed. Specifically, please provide a more detailed description of how the state points are selected for the Oconee plant-specific data, with particular attention to how bounding values are to be determined for B11 and mixed B10/B11 cores.*

The power/flow/pressure/temperature ranges for the SCD analyses are determined by the steady state and transient analyses for which DNBR is calculated. The Safety Analysis group provides the statepoint conditions to be evaluated in the SCD analysis. These statepoints represent expected ranges of operation in Chapter 15 transients. The statepoints shown in Table D-6 currently bound the range of conditions for Oconee where the SCD methodology is used to calculate DNBR. As necessary, additional statepoints from Safety Analysis are evaluated using the approved methodology in DPC-NE-2005 to verify that the Statistical DNB Limit determined is still bounding for the new set of conditions.

5. *The calculation with the VIPRE-01 code using BWU-Z correlation form for B11 fuel show essentially the same results as those obtained with LYNX over the correlation's database (as documented in Addendum 1 of BAW-10199). However, the BWU-Z correlation as modified for analysis of B11 fuel has not yet been approved by the staff, and the topical report describing this correlation, Addendum 1 of BAW-10199, is still being reviewed. This means that the design limit DNBR for the parameter ranges stated in Table D-2 may not be the final approved value or range of applicability. Specifically, the database for the form of the correlation spans a pressure range of 700 to 2400 psia, not 400 to 2465 psia range stated in Table D-2. Also, the plot in Figure D-3 (see p. D-11) shows a distinct nonconservative bias with decreasing pressure (which is identical to the trend shown for the correlation in the Addendum 1 submittal). There is also a nonconservative bias with the increasing power, clearly shown by plot of measured versus predicted Critical Heat Flux (CHF) in Figure D-1. What would be the effect on the thermal-hydraulic statistical core design analysis for Oconee if the DNBR design limit of the CHF correlation for B11 fuel were to be increased, or if the range of applicability of the correlation were to be limited to pressures of 700 to 2400 psia?*

The pressure range reported in Table D-2 is consistent with the conclusion made in Addendum 1 of BAW-10199. The Addendum 1 conclusion states that the correlation parameter range for the BWU-Z correlation with the Mark-B11V multiplier is the same as the BWU-Z correlation. The data base for the Mark-B11 fuel included a pressure range of 595 psia to 2425 psia as stated in Table E-7 of Addendum 1 to BAW-10199. Also, Figure D-3 shows a slight conservatism with decreasing pressure. Likewise, Figure D-1 shows a slight conservatism with increasing power.

The pressure range for the statepoints evaluated in Appendix D is 1600 psia to 2242 psia. The pressure/temperature conditions for these statepoints were selected to bound the range of fluid conditions at Oconee which will use the statistical DNBR methodology. Other DNB calculations are performed via the non-statistical DNB method. Non-statistical DNB calculations will use the applicable design limit DNBR (from the approved BWU-Z correlation, see table below). The correlation design limit DNBR (1.19) applies only at or above a nominal pressure of 1000 psia (Reference D-5 of Appendix D). In the lower pressure region (below a nominal pressure of 1000psia) the design limit DNBR in the following table will be used (Reference D-5):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.20

Attached is Table D-2 which has been updated to clarify the pressure dependency of the design limit DNBR. Also, references D-2 and D-5 have been updated to reflect the current revision of the approved topicals.

If a statepoint with pressure less than 1600 psia were identified, it would be propagated using the applicable CHF correlation standard deviation. A statepoint with pressure less than 1000 psia is not expected for Oconee SCD analyses. If a statepoint with a pressure less than 1000 psia were analyzed, the applicable design limit DNBR will be used and the impact of the higher correlation standard deviation on the statistical design limit would be directly calculated. This verifies the statistical design limit for the statepoint is bounded. If the SDL for the new statepoint is greater than the licensing limit, the higher SDL will be used when analyzing the lower pressure conditions. This is in accordance with the methodology as described in Table 7 of DPC-NE-2005.

Any changes to the CHF correlation or restrictions in its application resulting from the NRC review process will be communicated to Duke Power by the fuel vendor. If the Mark-B11 CHF correlation range of applicability is changed, the SCD analysis would be revised as needed to reflect the modification. The correlation will not be used for DNB calculations outside the parameter range stated in the approved correlation topical. If the correlation standard deviation increases above the value used in the analyses, the limiting statepoint will be re-propagated to verify the SDL given in Appendix D.

TABLE D-1 MARK-B11 FUEL ASSEMBLY DATA

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, in. (Nom.)	0.416
Thimble tube diameter, in. (Nom.)	0.530
Instrument guide tube diameter, in. (Nom.)	0.554(1)/0.567(2)
Fuel rod pitch, in (Nom.)	0.568
Fuel assembly pitch, in. (Nom.)	8.587
Fuel rod length, in. (Nom.)	154.16

(1) Above lowest mixing vane grid (MV) and between MV grids.

(2) Below the first mixing vane grid and above the top of the last mixing vane.

GENERAL FUEL CHARACTERISTICS

Grids:	Material	Quantity	Location	Type
	Inconel	2	Upper and Lower	Non-Mixing Vane
	Zircaloy	6	Intermediate	1 Non-Mixing Vane, 5 Mixing Vane

Fuel Rods:	Material	Quantity
	Zircaloy-4	208

Fuel Cycle Design Assembly Features

Fuel Assy. Mark

Designation: B11

Features: Smaller clad outside diameter and mixing vane grids.

TABLE D-2 VIPRE-01 BWU-Z Correlation with Mark-B11V Multiplier Verification

CHF Test Database Analysis Results

VIPRE-01/LYNXT Statistical Results

	VIPRE-01	LYNXT
n, # Of data	216	216
N, degrees of freedom (n-1)	215	215
M/P, Average measured to predicted CHF	1.0084	1.0040
σ (M/P/N)	0.0859	0.0868
K(215,0.95,0.95),one sided tolerance factor Ref. D-2)	1.830	1.830
DNBR(L)= 1/(M/P - Kσ)= 1/[1.0040 - 1.830(0.0868)]	1.175	1.183

Parameter Ranges

Pressure, psia	400 to 2465
Mass Velocity, Mlbm/hr-ft^2	0.36 to 3.55
Thermodynamic Quality at CHF	less than 0.74
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Mark-B11 15x15 Mixing Vane
Design Limit DNBR, VIPRE-01	1.19*

* The correlation design limit DNBR (1.19) applies only at or above a nominal pressure of 1000 psia (Reference D-2). In the low pressure region (below a nominal pressure of 1000 psia) the design limit DNBR in the following table will be used (Reference D-2):

Pressure	Design Limit DNBR
400 to 700 psia	1.59
700 to 1000 psia	1.20

TABLE D-4 Oconee Statistically Treated Uncertainties

Parameter	Type	Type of Distribution	Uncertainty	Standard Deviation
Reactor System				
Core Power*	Measurement	Normal	+/-2.0%FP	+/-1.0%FP
Core Flow	Measurement	Normal	4 Pump: +/-2.0% 3 Pump: +/-3.2% 2 Pump: +/-4.2% design	+/-1.0% +/-1.6% +/-2.1% design
Pressure	Measurement	Normal	+/-30.0 psi	+/-15.0 psi
Temperature	Measurement	Normal	+/-2.0°F	+/-1.0°F
Nuclear				
FΔH	Calculation	Normal	---	+/-2.84%
Fz	Calculation	Normal	---	+/-2.91%
Z	Calculation	Uniform	+/-6 inches	---
Fq	Calculation	Normal	[	]
Hot Channel Flow Area	Measurement	Uniform	[ ]	---
DNBR	Correlation	Normal	---	9.268%
DNBR	Code	Normal	[	]

* Percentage of 100% RTP (69.75 MWth wherever applied).

REFERENCES

D-1. DPC-NE-2003P-A, Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, October 1989.

D-2. The BWU Critical Heat Flux Correlations, Addendum 1 to BAW-10199P-A, Framatome Cogema Fuels, Lynchburg, Virginia, April 6, 2000

D-3. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

D-4. DPC-NE-2005P-A, Rev. 1, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, Appendix C, November 1996.

Appendix E and Response to

Request for Additional

Information

Revision 3

DPC-NE-2005-PA

Duke Energy Carolinas
Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX E

McGuire/Catawba Plant Specific Data

Advanced Mark-BW Fuel and Mark-BW/MOX1 Fuel,
BWU-Z/MSM CHF Correlation

Submitted: September 2001
Approved: September 2002

This Appendix contains the plant specific data and statistical DNB limits for the McGuire and Catawba Nuclear Stations with the Advanced Mark-BW fuel design using the BWU-Z/MSM critical heat flux correlation. The thermal-hydraulic statistical core design analysis was performed as described in the main body of this report (DPC-NE-2005).

This appendix details the fuel assembly structural and thermal-hydraulic features unique to the Advanced Mark-BW fuel design. Two separate fuel pellet materials can be used in this structure. When used with uranium fuel pellets, the fuel assembly is called Advanced Mark-BW. If used with mixed oxide fuel pellets, the fuel assembly is called Mark-BW/MOX1. The fuel mechanical structure and grids are identical in each case, therefore the same critical heat flux correlation is applicable to both designs. The nuclear uncertainties used in this analysis bound both uranium and mixed oxide fuel rods. Therefore, the SCD analysis documented here is applicable to and bounds both the Advanced Mark-BW and the Mark-BW/MOX1 fuel designs. For simplicity in this appendix, the term Advanced Mark-BW will be used.

Plant Specific Data

This analysis is for the McGuire and Catawba plants (four loop Westinghouse PWR's) with the Advanced Mark-BW fuel. This fuel design incorporates a 17x17 fuel lattice with 0.374 inch outside diameter (OD) fuel rods, M5™ cladding, and three additional non-structural Mid-

Span Mixing (MSM) grids in the upper fuel assembly spans to improve DNB performance. All the parameter uncertainties and statepoint ranges used in this analysis were selected to bound the unit and cycle specific system values at the McGuire and Catawba stations.

Thermal Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference E-3 and the McGuire/Catawba eight channel model approved in Reference E-1 are used in this analysis. The reference pin power distribution is the same as that used for the Westinghouse RFA fuel described in Appendix G of this report (DPC-NE-2005). The VIPRE-01 models, approved in Reference E-1 for the Mark-BW fuel, are used to analyze the Advanced Mark-BW fuel design with the following changes:

1) The Advanced Mark-BW fuel assembly geometry information is listed in Table E-1. Applicable form loss coefficients as per the vendor were used in the model.
2) The bulk void fraction model was changed from the Zuber-Findlay model to the EPRI model. Correspondingly, the subcooled void model was changed from LEVY to the EPRI model.

The Zuber-Findlay bulk void model is applicable only to qualities below approximately 0.7 (void fractions of 0.85) and is discontinuous at a quality equal to 1.0 (Reference E-3). The EPRI bulk void model is essentially the same as the Zuber-Findlay bulk void model except for the equation used to calculate the drift velocity (Reference E-3). This eliminates the discontinuity at high qualities and void fractions. Therefore, the EPRI model covers the full range (i.e.,

void fraction range, 0 - 1.0) of void fractions required for performing DNB calculations. Also, for overall void model compatibility, the subcooled void model was changed from the Levy model, as specified in Reference E-1, to the EPRI correlation. This change has been previously submitted and approved by the NRC for both the Westinghouse RFA fuel design (Appendix G of this report) and the Mark-B11 fuel design (Appendix D of this report).

Critical Heat Flux Correlation

The BWU-Z/MSM critical heat flux correlation described in Reference E-2 is used for all statepoint analyses. This correlation was developed by AREVA NP and is applicable to the Advanced Mark-BW fuel design. The analysis in Reference E-2 was performed with the LYNXT thermal-hydraulic computer code. This correlation was programmed into the VIPRE-01 thermal-hydraulic computer code and the Advanced Mark-BW fuel database was analyzed in its entirety. The results of this analysis are shown in Table E-2. The resulting average measured to predicted (M/P) value and data standard deviation are within 1% and the CHF correlation limit with VIPRE-01 is 2% lower than the values in Reference E-2, page F-5 (also shown on Table E-2 under the LYNXT column).

Figures E-1 through E-4 graphically show the results of this evaluation. Figure E-1 shows there is no bias of measured CHF values to VIPRE-01 predicted values for the database. Figures E-2 through E-4 show there is no bias with the VIPRE-01 calculated M/P ratios with respect to mass velocity, pressure, or thermodynamic quality. These

figures compare closely with the same parameter representations in Reference E-2.

Based on the results shown in Table E-2 and Figures E-1 through E-4, the BWU-Z form of the BWU CHF application correlation can be used in DNBR calculations with VIPRE-01 for Advanced Mark-BW fuel.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table E-3. These statepoints represent the range of conditions to which the statistical DNB analysis limit will be applied.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table E-4. The uncertainties were selected to bound the values calculated for each parameter at McGuire and Catawba. As noted in Table E-4, the nuclear uncertainties used in this analysis bound both uranium and mixed oxide fuel. The resulting range of key parameter values generated in this analysis is listed on Table E-6.

Mixed Core Application

The mixed core model determines the impact of the geometric and hydraulic differences between the resident 17x17 Westinghouse RFA fuel described in Appendix G of this report (DPC-NE-2005) and the new Advanced Mark-BW design. The 8 channel model described in

Reference E-1 is used to evaluate the impact of mixed cores containing Westinghouse RFA fuel and the Advanced Mark-BW fuel. In Figure 5 of Reference E-1, Advanced Mark-BW fuel is used instead of Mark-BW fuel. Therefore, the limiting assembly in Channels 1 through 7 are modeled as Advanced Mark-BW fuel and the remaining core, Channel 8, is modeled as Westinghouse RFA fuel. The mixed core analysis models each fuel type in those respective locations with the correct geometry. The form loss coefficients for each fuel design are input so the effect of crossflow between the different fuel types by elevation is calculated. This conservative mixed core model is used for all analyses since the equilibrium core reload cycles will contain both fuel types.

DNB Statistical Design Limits

The statistical design limit for each statepoint evaluated is listed on Table E-5. Section 1 of Table E-5 contains the 500 case runs and Section 2 contains the 6000 case runs. The number of cases was increased from 5000 to 6000 as described in Attachment 1 of Revision 0 of DPC-NE-2005. The DNBR distributions for all statepoints in this analysis were normally distributed. It is seen from Section 2 of Table E-5 that the maximum statepoint statistical DNBR value is [ ]. Therefore, the statistical design limit using the BWU-Z/MSM CHF correlation for Advanced Mark-BW fuel at McGuire/Catawba was conservatively determined to be 1.36. This limit also applies to mixed cores with Advanced Mark-BW and Westinghouse RFA fuel.

FIGURE E-1
Measured CHF versus Predicted CHF
Advanced Mark-BW Fuel Database

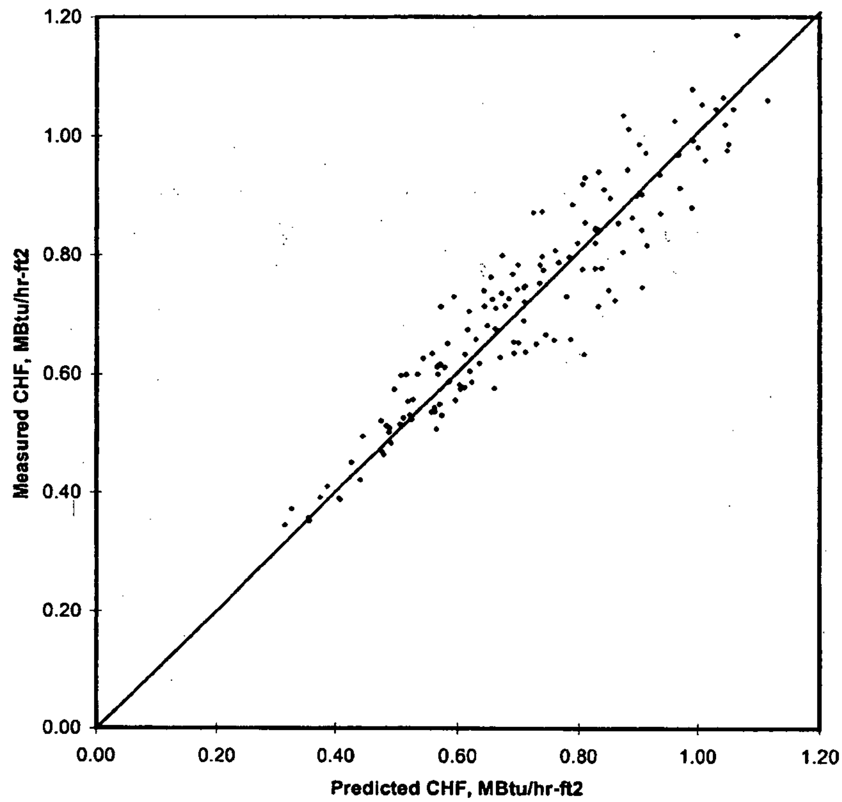

FIGURE E-2
Measured to Predicted CHF versus Mass Velocity
Advanced Mark-BW Fuel Data Base

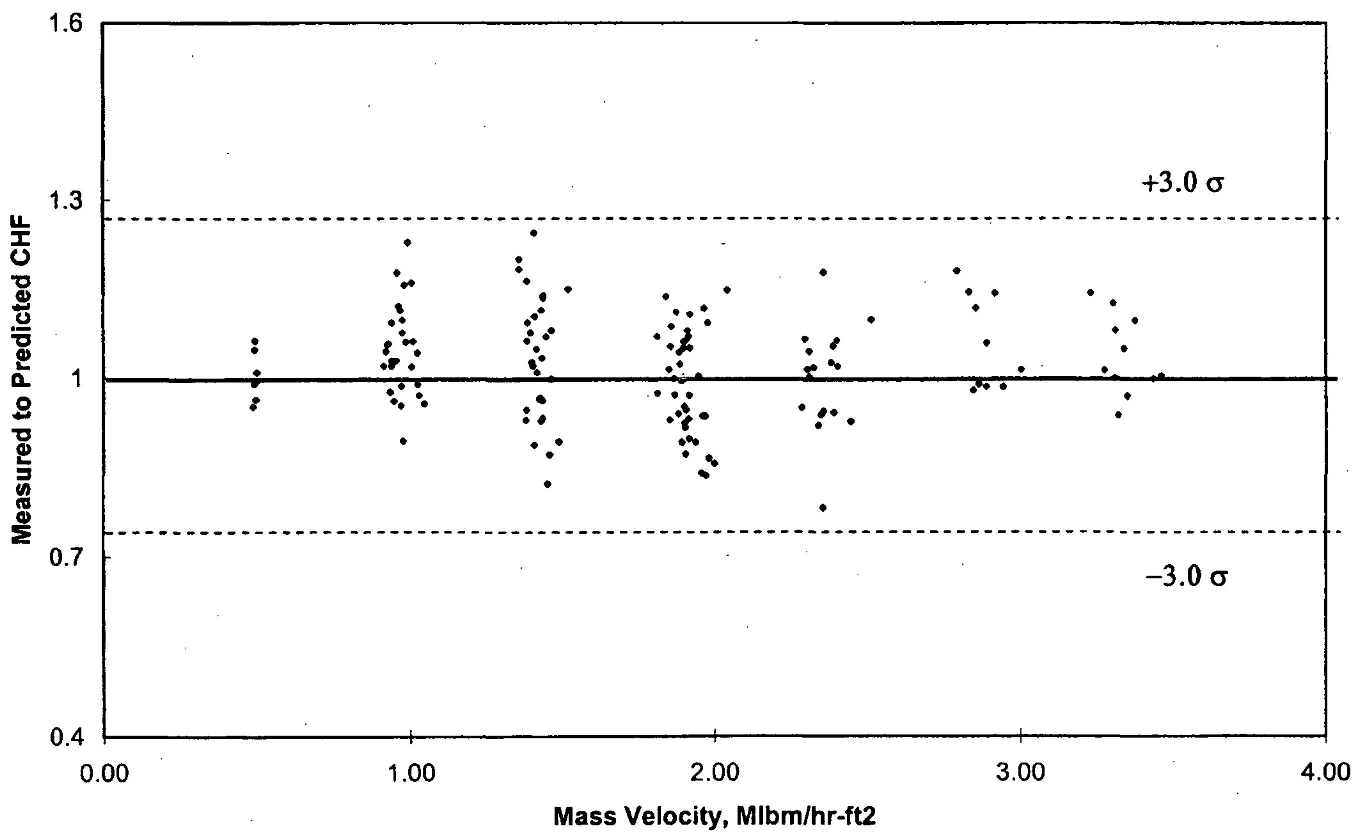

FIGURE E-3
Measured to Predicted CHF versus Pressure
Advanced Mark-BW Fuel Data Base

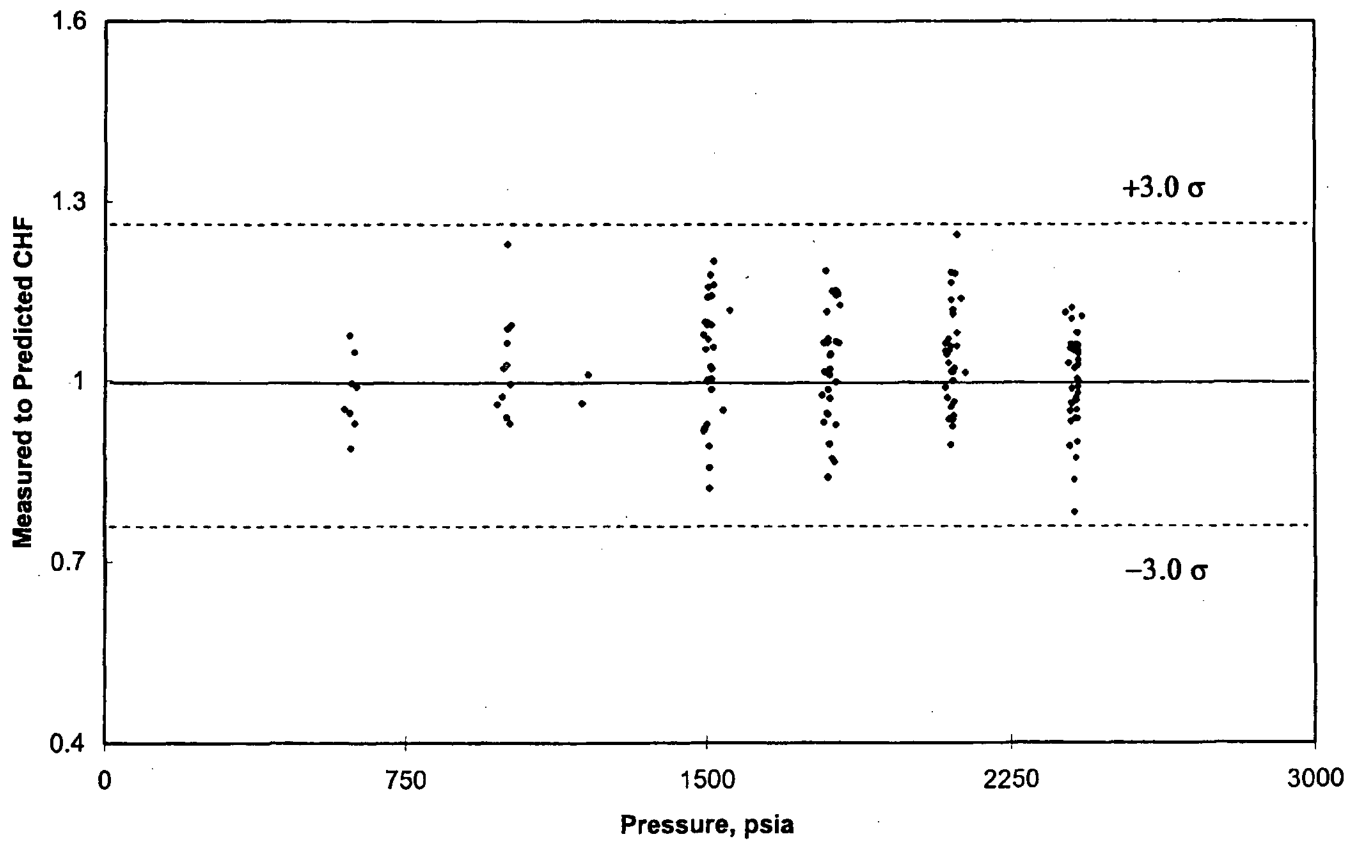

FIGURE E-4
Measured to Predicted CHF versus Quality
Advanced Mark-BW Fuel Data Base

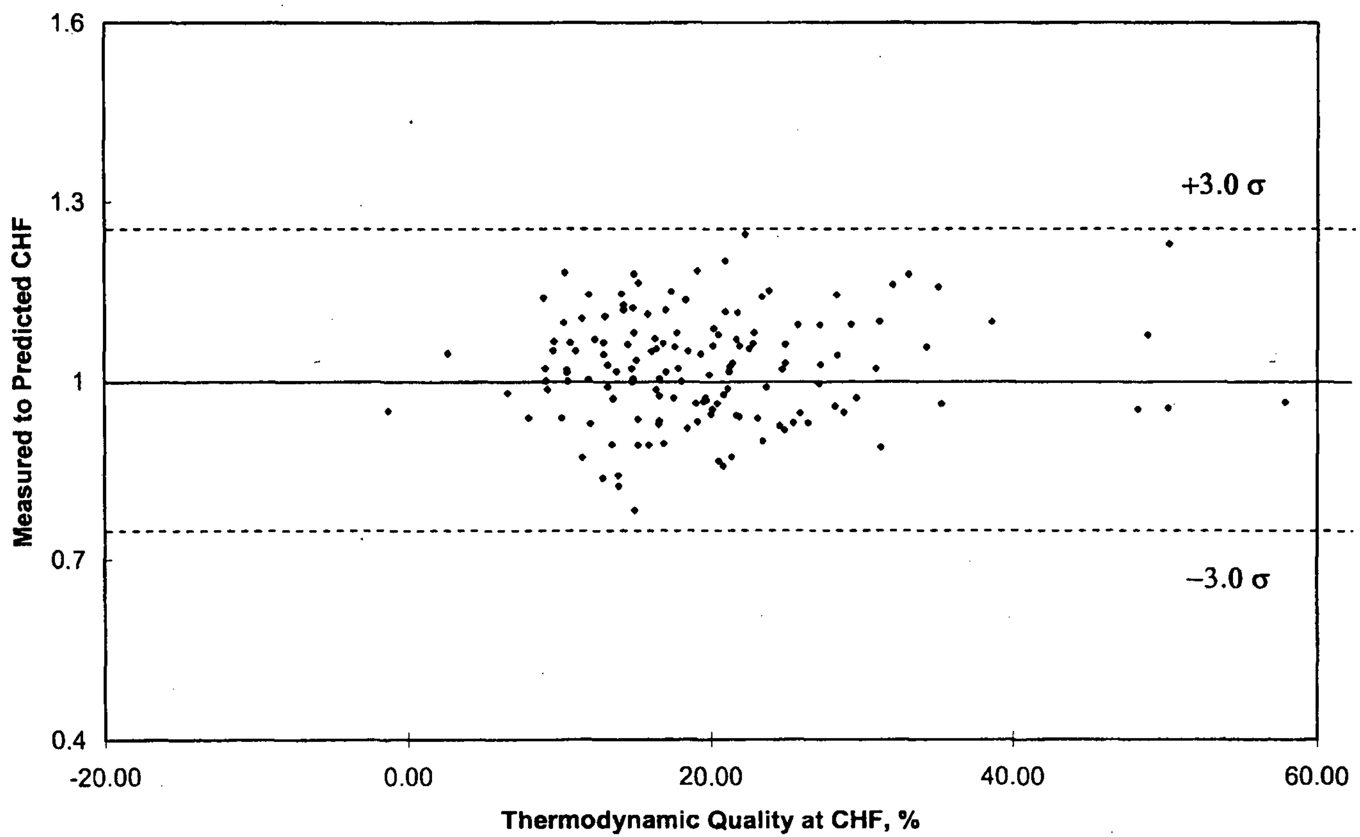

TABLE E-1 Advanced Mark-BW Fuel Assembly Data

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, inches (Nominal)	0.374
Guide tube diameter, inches (Nominal)	0.482
Fuel rod pitch, inches (Nominal)	0.496
Fuel Assembly pitch, inches (Nominal)	8.466
Fuel Assembly length, inches (Nominal)	160.0

GENERAL FUEL CHARACTERISTICS

Component	Material	Number	Location	Type
Grids	Inconel	2	Upper and Lower	Non-Mixing Vane
	Zircaloy ($M5^{TM}$)	6	Intermediate	5 Vaned, 1 Vaneless
	Zircaloy ($M5^{TM}$)	3	Intermediate	Mid-Span Mixing (Non-structural)
Nozzles	304SS	1	Bottom	Fine Mesh
	304SS	1	Top	Quick Disconnect

TABLE E-2 CHF Test Database Analysis Results With VIPRE-01 Advanced Mark-BW Fuel, BWU-Z/MSM CHF Correlation

VIPRE-01/LYNXT Statistical Results

	VIPRE-01	LYNXT
n, # Of data	148	148
N, degrees of freedom (n-1)	147	147
M/P, Average measured to predicted CHF	1.0214	1.0138
σ (M/P/N)	0.0883	0.0920
K(147,0.95,0.95),one sided tolerance factor Ref. E-2)	1.872	1.872
DNBR(L)= 1 /(M/P - K * σ)	1.168	1.188

BWU-Z Parameter Ranges

Pressure, psia	400 to 2465
Mass Velocity, Mlbm/hr-ft^2	0.47 to 3.55
Thermodynamic Quality at CHF	Less than 0.68
Thermal-Hydraulic Computer Code	VIPRE-01
Spacer Grid	Advanced Mark-BW, F_{MSM} = 1.18
Design Limit DNBR, VIPRE-01	1.19*

* The correlation design limit DNBR (1.19) applies only at or above the nominal pressure of 594 psia (Reference E-2). In the low pressure region (400 to < 594 psia) the design limit is 1.59 (Reference E-2).

TABLE E-3 McGuire/Catawba SCD Statepoints

Stpt No.	Power* (% RTP)	RCS Flow (K gpm)	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak# (FΔH)
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						
25						

* 100% RTP = 3411 Megawatts Thermal

\# FΔH is maximum pin peak

TABLE E-4 McGuire/Catawba Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type of Distribution
Core Power*	+/- 2% / +/- 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / +/- 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^{N}_{\Delta H}$		
Measurement	+/- 4.0% / 2.43%	Normal
$F^{E}_{\Delta H}$	+/- 3.5% / 2.13%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.4% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	15.3% / +/- 9.27%	Normal
Code/Model	[ ]	Normal

* - Percentage of 100% RTP (68.22 MWth wherever applied).

TABLE E-4 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
Core Power	The core power uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from normally distributed random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method (SRSS). Since the uncertainty is calculated from normally distributed values, the parameter distribution is also normal.
Core Flow Measurement	Same approach as Core Power.
Bypass Flow	The core bypass flow is the parallel core flow paths in the reactor vessel (guide thimble cooling flow, head cooling flow, fuel assembly/baffle gap leakage, and hot leg outlet nozzle gap leakage) and is dependent on the driving pressure drop. Parameterizations of the key factors that control ΔP, dimensions, loss coefficient correlations, and the effect of the uncertainty in the driving ΔP on the flow rate in each flow path, was performed. The dimensional tolerance changes were combined with the SRSS method and the loss coefficient and driving ΔP uncertainties were conservatively added to obtain the combined uncertainty. This uncertainty was conservatively applied with a uniform distribution.
Pressure	The pressure uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method. The uncertainty distribution was conservatively applied as uniform.
Temperature	Same approach as Pressure.

TABLE E-4 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
$F^{N}_{\Delta H}$	
Measurement	This uncertainty is the measurement uncertainty for the movable incore instruments. A measurement uncertainty can arise from instrumentation drift or reproducibility error, integration and location error, error associated with the burnup history of the core, and the error associated with the conversion of instrument readings to rod power. The uncertainty distribution is normal. This uncertainty is bounding for both uranium and mixed oxide fuel.
$F^{E}_{\Delta H}$	This uncertainty accounts for the manufacturing variations in the variables affecting the heat generation rate along the flow channel. This conservatively accounts for possible variations in the pellet diameter, density, and U_{235} enrichment. This uncertainty distribution is normal and was conservatively applied as one-sided in the analysis to ensure the MDNBR channel location was consistent for all cases. This uncertainty bounds both uranium and mixed oxide fuel pellets.
Spacing	This uncertainty accounts for the effect on peaking of reduced hot channel flow area and spacing between assemblies. The power peaking gradient becomes steeper across the assembly due to reduced flow area and spacing. This uncertainty distribution is normal and was conservatively applied as one-sided to ensure consistent MDNBR channel location.
F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty distribution is applied as normal.
Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one physics code axial node length. The uncertainty distribution is conservatively applied as uniform.

TABLE E-4 Continued McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
DNBR	
Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The LYNXT value was used since the VIPRE-01 value was smaller. The uncertainty distribution is applied as normal.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is applied as normal.

TABLE E-5 McGuire/Catawba Statepoint Statistical Results

SECTION 1

BWU-Z/MSM Critical Heat Flux Correlation

500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
25				

TABLE E-5 Continued McGuire/Catawba Statepoint Statistical Results

SECTION 2

BWU-Z/MSM Critical Heat Flux Correlation

6000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
6				
7				
10				
11				
12				

TABLE E-6 McGuire/Catawba Key Parameter Ranges

Parameter	Maximum	Minimum
Core Power* (% RTP)		
Pressure (psia)		
T inlet (deg. F)		
RCS Flow (Thousand GPM)		
FΔH, Fz, Z		

* 100% RTP = 3411 Megawatts Thermal

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.

REFERENCES

E-1. DPC-NE-2004-PA, McGuire and Catawba Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, Revision 2a, December 2008.

E-2. BAW-10199P-A, Addendum 2, Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids, June 2002.

E-3. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

Duke Power.
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August 14, 2002

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

Subject: McGuire and Catawba Nuclear Station, Units 1 and 2
Docket Nos. 50-369, 50-370, 50-413, 50-414
Topical Report DPC-NE-2005P, *Thermal-Hydraulic Statistical Core Design Methodology*, Revision 3 (Appendix E); Request for Additional Information

Reference: U.S. Nuclear Regulatory Commission letter dated August 1, 2002, Request for Additional Information - Review of Duke Topical Report DPC-NE-2005P, (TAC Nos. MB3105, MB3106, MB3173 and MB3175)

Duke Power Company's hereby submits its response to the Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) transmitted by the reference letter. This submittal consists of two attachments; Attachment 1 is a proprietary version of Duke's response to the RAI and Attachment 2 is a non-proprietary version. The proprietary information in Attachment 1 is enclosed in brackets [ ]. In accordance with 10 CFR 2.790, Duke requests that this proprietary information be withheld from public disclosure. An affidavit that attests to the proprietary nature of this information is included with this letter.

Also included in this submittal is a revised References page (E-20) for DPC-NE-2005P reflecting NRC approval of Framatome ANP Topical Report BAW-10199P-A, Addendum 2.

Inquiries on this matter should be directed to G.A Copp at (704) 373-5620.

Very truly yours,

K. S. Canady

Attachments

PROPRIETARY
Material Attached

xc w/attachments:

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AFFIDAVIT OF K.S. CANADY

1. My name is K. S. Canady. I am Vice President of 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, am authorized to apply for its withholding on behalf of Duke.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding which accompanies this affidavit.

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

4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.790, 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 and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. 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.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, 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 this submittal is that which is marked in the attachment to Duke Energy Corporation letter dated August 15, 2002; Response to Request for Additional Information Re: Topical Report DPC-NE-2005P, Revision 3, Duke Power Thermal-Hydraulic Statistical Core Design (SCD) Methodology. This information enables Duke to:

      (a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

K. S. Canady

(continued)

(b) Support license amendment and Technical Specification revision requests for Duke reactors to support the use of Advanced Mk-BW or Mk-BW/MOX1 fuel assemblies.

(c) Perform safety reviews per 10 CFR 50.59.

(d) Evaluate core thermal-hydraulic performance and support the establishment of core operating limits.

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

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

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

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

5. Public disclosure of this information is likely to cause harm to Duke 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 to recoup a portion of its expenditures or benefit from the sale of the information.

K. S. Canady, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

K. S. Canady

Sworn to and subscribed before me this 14TH day of August, 2002.
Witness my hand and official seal.

Mary P. Nehus
Notary Public

My commission expires: JAN 22, 2006

SEAL

Attachment 1
Duke Response to NRC Request for
Additional Information on DPC-NE-2005, Revision 3 (Appendix E)

The staff has reviewed Duke Power's submittal dated September 13, 2001, "Appendix E to DPC-NE-2005P, Duke Power Thermal-Hydraulic Statistical Core Design (SCD) Methodology (Proprietary)" and has identified a need for the following additional information.

1. The submittal states that Appendix E contains the plant specific data and statistical departure from nucleate boiling (DNB) limits for the McGuire and Catawba Nuclear Stations with the Advanced Mark-BW fuel design using the BWU-Z critical heat flux (CHF) correlation and provides the fuel assembly structural and thermal-hydraulic features unique to the Advanced Mark-BW fuel design. However, the submittal also states that its SCD analysis is applicable to and bounds both the Advanced Mark-BW and the Mark-BW/MOX1 fuel designs. It appears that the data provided in the submittal are only applicable to the Advanced Mark-BW fuel design. Please clarify whether the methodology described in Appendix E to DPC-NE-2005P will be applied to both the Advanced Mark-BW and the Mark-BW/MOX1 fuel designs. If it is applicable to both designs, then additional data sets for the Mark-BW/MOX1 should be provided. Also, please identify those differences between the Advanced Mark-BW and Mark-BW17 fuel design.

Response:

To clarify the terminology, a separate description of all three fuel types listed in the question is provided below.

Mark-BW17 Fuel

The Mark-BW17 fuel design is a 17x17 fuel assembly design for operation in Westinghouse NSSS systems by Framatome/ANP. The Mark-BW17 fuel assembly was introduced in 1991 and is currently operating in the McGuire, Catawba, and Sequoyah units. The Mark-BW17 includes the following features:

- 0.374 fuel rod outer diameter
- Debris resistant bottom nozzle
- Zircaloy mixing vane spacer grids
- Removable top nozzle

Advanced Mark-BW Fuel

The Advanced Mark-BW fuel design is an evolutionary improvement on the successful Mark-BW17 fuel assembly design. The only thermal-hydraulic difference between the Mark-BW17 fuel and the Advanced Mark-BW fuel is the addition of three mid-span mixing grids to the Advanced Mark-BW design. All other features listed above are the same between Mark-BW17 and the Advanced Mark-BW. Therefore, the Advanced Mark-BW design is the Mark-BW17 fuel with mid-span mixing grids.

Mark-BW/MOX1 Fuel Design

The Mark-BW/MOX1 design is virtually identical to the Advanced Mark-BW. The only difference between the designs other than the title is in the fuel rod design. There are two specific differences, namely:

1. The Mark-BW/MOX1 fuel design contains mixed oxide fuel pellets and Advanced Mark-BW fuel design contains UO_2 fuel pellets.
2. The Mark-BW/MOX1 overall fuel rod length is 152.40 inches, 0.24 inches longer than the Advanced Mark-BW fuel rod length of 152.16 inches.

All other fuel assembly hardware, features, and dimensions are the same. The extra fuel rod length described in Item 2 is at the top of the fuel assembly above the end of the heated length and does not impact the DNBR calculations. Therefore, from a thermal-hydraulic fuel feature perspective, there is no difference between the Advanced Mark-BW and the Mark-B/MOX1 fuel designs.

Since the thermal-hydraulic features are the same, the only impact the different fuel rod designs could have on the statistical DNBR limit is the radial and axial nuclear uncertainties. These uncertainties are listed under the headings $F^N_{\Delta H}$ and F_Z in Table E-4 of the submittal. The uncertainty values listed on Table E-4 and used in the analysis bound both mixed oxide and UO_2 fuel rods. Since a larger uncertainty is conservative for the SCD analyses and since both UO_2 and mixed oxide uncertainties are bounded, the methodology and the calculated SCD limit of 1.36 is equally valid for the Advanced Mark-BW and the Mark-BW/MOX1 fuel designs.

As stated in the last sentence of paragraph 2 on page E-1, the term Advanced Mark-BW was used throughout the report for simplicity. The term Advanced Mark-BW in the report means both the Mark-BW/MOX1 and the Advanced Mark-BW design.

2. Provide the Advanced Mark-BW fuel database used in Table E-2 of the September 13, 2001, submittal and describe the process used to obtain the 148 new data points in the two tests. Also, please demonstrate that the new data are duplicate and close to the old data used in the topical report, BAW-10199P, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids", and justify that the data base is sufficient for this application.

Response:

The 148 data points analyzed by Duke Energy with the BWU-Z CHF correlation in Appendix E of DPC-NE-2003 are exactly the same 148 data points analyzed in BAW-10199P-A, Addendum 2 by Framatome/ANP. Duke Energy analyzed the exact same data base with the VIPRE-01 thermal-hydraulic code. The resulting data in Table E-2 of the report shows how the correlation statistics with VIPRE-01 compared to the LYNXT analysis of the data shown in BAW-10199P-A. The VIPRE-01 code results had a smaller standard deviation compared to LYNXT for the same database. The SCD calculation in DPC-NE-2005, Appendix E, conservatively bounded the larger LYNXT value for the correlation uncertainty.

3. Provide details of the calculation procedure used to evaluate the effect of crossflow between the different fuel types as well as the form loss coefficients used as inputs for the mixed core analysis. Also, describe the real test data available for this application to McGuire and Catawba and justify that the DNB statistical design limit of 1.36 is sufficient for McGuire and Catawba using the BWU-Z CHF correlation for Advanced Mark-BW fuel mixed with Westinghouse robust fuel assembly fuel.

Response:

The procedure used to account for the effects of crossflow between different fuel types in the analysis was to use a conservative mixed core model for all calculations. As stated in Mixed Core Application on page E-4 of the submittal, the 8 channel model was used. The center hot or highest powered assembly, consisting of Channels 1 through 7, is modeled as an Advanced Mark-BW. The remaining core, 192 fuel assemblies, is modeled as Westinghouse RFA fuel in Channel 8. The mixed core analysis models each fuel type in those respective locations with the correct geometry and form losses. The forms loss coefficients by fuel design from top to bottom are as follows:

COMPONENT	ADVANCED MARK-BW	WESTINGHOUSE RFA
Top Nozzle	[	
Top Inconel		
Mixing Vane Zirc Grids		
IFM / MSMG Grid		
Non-Mixing Vane Zirc Grid		
Bottom Inconel		
Bottom Nozzle		]

(1) Inconel grid plus Protective grid

The process for evaluating crossflow is to model the core conservatively with the correct geometry and design data and allow VIPRE-01 to calculate the crossflow in each axial node based on the data. In this manner, the core is accurately modeled and the results represent a conservative mixed core.

With respect to real test data available for this application to McGuire and Catawba, the test data is just as applicable to mixed core configuration as a full core. The Framatome ANP CHF correlation form (BWU) is composed of three parts:

1) a uniform part dependent solely on the local thermal-hydraulic conditions of pressure, mass velocity and thermodynamic quality at the axial location of CHF,

2) a non-uniform F factor modification dependent on the shape of the axial heat flux input, and
3) a multiplicative geometric factor dependent on the overall fuel assembly grid spacing and heated length. It is with Item 1 (the local conditions thermal-hydraulic conditions) that the mixed core conditions question surfaces. The other two items are unaffected by mixed cores.

CHF correlations are developed from data from full length electrically heated bundles in 5-by-5 rod arrays. For each data point, the inlet conditions of coolant mass velocity, pressure and temperature are known, as is the power (heat flux) required to produce a DNB event. The local thermal-hydraulic conditions at the axial location of CHF must then be calculated with a computer code.

The proof of applicability of a CHF correlation, then, is how well it can predict the critical heat flux that was measured in the DNB event using the calculated local conditions. Thus, the applicability of a CHF correlation is dependent not only on its form and data base, but on the accuracy with which the thermal-hydraulic code calculates local conditions. Because of the size of the test section (a 5-by-5 rod array) and the use of the same spacer grids and elevations, normal CHF tests do not exhibit large hydraulic differences. Framatome ANP, however, has performed one test with widely varying subchannel resistances producing the large velocity upsets representative of mixed core conditions. This test was a 5-by-5 test of the Mark B zircaloy grid modeled as the corner intersection of four fuel assemblies. Laser Doppler Velocimetry (LDV) testing of the intersection grid showed velocity depressions as large as 50% between the intersection subchannel and the surrounding unit cell subchannels. This CHF test was conducted at the Babcock & Wilcox Alliance Research Center and is documented in BAW-10143P-A (BWC correlation of Critical Heat Flux, April, 1985). In the topical, the measured to predicted (M/P) CHF results were compared for two traditional test bundles and the intersection bundle. The guide tube bundle (B15) had an average M/P of 0.971, the unit cell bundle (B16) 0.985 and the intersection bundle (B17) 0.976. The difference in M/P results is statistically insignificant.

The predicted local conditions for the unit and guide tube bundles had very little hydraulic upset, while the intersection bundle (conditions representative of a mixed core) had severe predicted upsets, similar to the measured data. The fact that the CHF correlation performed consistently on conditions representative of both homogeneous and mixed cores confirms that the thermal-hydraulic codes predict the right conditions even with large velocity upsets. This confirms the Framatome ANP CHF correlations are valid for both homogeneous and mixed core applications.

Appendix F and Response to Request for Additional Information

Revision 4

Submitted: September 2007
Approved: December 2008

DPC-NE-2005-PA

Duke Energy Carolinas
Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX F

Oconee Nuclear Station Specific Data

Mark-B-HTP Fuel

Application of the BHTP CHF Correlation to the Mark-B-HTP Fuel Design

Submitted: September 2007
Approved: October 2008

This appendix contains the plant-specific data and limits for the Oconee Nuclear Station with Mark-B-HTP fuel using the BHTP critical heat flux correlation. Below the first intermediate grid the BWU-N critical heat flux correlation is used. The thermal-hydraulic statistical core design analysis was performed as described in the main body of this report.

Plant-Specific Data

This analysis is for the Oconee Nuclear Station (two-loop Babcock and Wilcox PWR) as described in Reference F-1. The parameter uncertainties and statepoint ranges were selected to bound the Oconee unit and cycle-specific values. This analysis models the 0.430 inch diameter fuel rod Mark-B-HTP fuel assembly design.

Thermal-Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference F-2 and the Oconee eight and nine channel models approved in Reference F-1 are used in this analysis. Due to the fuel assembly design change, some specific data supplementary to Table 3-1 in Reference F-1 requires updating. This data is listed in Table F-1 in this appendix. Table F-1 includes fuel rod, control rod, and instrument guide tube diameters, the number and design of the grids, and the fuel rod length.

The VIPRE-01 models approved in Reference F-1 are used to analyze the Mark-B-HTP fuel with the following exceptions:

1) The Mark-B-HTP fuel assembly dimensional information is listed in Table F-1.
2) The turbulent mixing factor has been changed to [ ] for the Mark-B-HTP fuel assembly design due to the presence of HTP grid mixing features, and [ ] for the HMP grid. The numerical value was determined and provided by the fuel supplier. A
3) The bulk void fraction model was changed from the Zuber-Findlay model to the EPRI model. The Zuber-Findlay bulk void model is applicable only to qualities below approximately 0.7 and is discontinuous at a quality equal to 1.0 (Reference F-2). The EPRI bulk void model is essentially the same as the Zuber-Findlay bulk void model except for the equation used to calculate the drift velocity (Reference F-2). This eliminates the discontinuity at a quality equal to 1.0. Therefore, the EPRI model

provides a full range (i.e., void fraction range, 0 - 1.0) of applicability required for performing DNB calculations. Also, for overall model compatibility, the subcooled void model was changed from LEVY, as specified in Reference F-1, to the EPRI correlation for the Mark-B-HTP fuel.

Critical Heat Flux Correlation

The NRC-approved BHTP critical heat flux correlation in Reference F-3 is used for all Mark-B-HTP analyses, with the exception that below the first intermediate grid the BWU-N critical heat flux correlation (Reference F-4) is used. The BHTP correlation was developed by AREVA NP for application to the Mark-B-HTP fuel design. The analysis in Reference F-3 was performed with the LYNXT thermal-hydraulic computer code (Reference F-5). This correlation was programmed into the VIPRE-01 thermal-hydraulic computer code and the Mark-B-HTP data base analyzed in its entirety. The results of this analysis are shown in Tables F-2 and F-3. The resulting average P/M value, data standard deviation, and CHF correlation limit are within 1% of the values reported in Reference F-3 (also shown in Table F-2 under LYNXT column).

Figures F-1 through F-4 graphically show the results of this evaluation. Figure F-1 shows there is no bias of VIPRE-01 predicted CHF values to measured values for the data base. Figures F-2 through F-4 show there is no bias with the VIPRE-01 calculated P/M ratios with respect to mass velocity, pressure, or thermodynamic quality. These figures compare closely with the same parameter representations in Reference F-3.

Based on the results shown in Tables F-2 and F-3 and Figures F-1 through F-4, the BHTP CHF correlation licensed in Reference F-3, can be used in DNBR calculations with VIPRE-01 for Mark-B-HTP fuel.

Statistical Core Design Analysis

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table F-4. These statepoints represent the range of conditions to which the statistical DNB analysis limit will be applied. The range of key parameter values analyzed is listed in Table F-7.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed in Table F-5. The uncertainties were selected to bound the values calculated for each parameter. The uncertainties have not changed from Reference F-1 except for the rod power hot channel factor (Fq), core flow measurement, and CHF correlation. The uncertainty for Fq has changed due to fuel design changes. The core flow measurement uncertainty was increased to ensure that it is bounding. This results in a more conservative SDL. The DNBR correlation uncertainty is a bounding value as compared to Reference F-3. It is noted that the Fz parameter uncertainty distribution in Table F-5 is treated as a normal distribution. The nuclear uncertainty database for Fz is actually characterized as nearly normal, with the data more skewed than normal. It is judged that the data can be treated as normal for the purposes of the SCD methodology.

DNB Statistical Design Limits

The statistical DNBR limit for each statepoint evaluated is listed in Table F-6. Section 1 of Table F-6 contains the 500 case runs and Section 2 contains the 5,000 case runs. All of the DNBR distributions are normally distributed. The maximum statistical DNBR value in Table F-6 (full core of Mark-B-HTP fuel) for 5,000 propagations is [ ]. Therefore, the statistical design limit, using the BHTP CHF correlation for Mark-B-HTP fuel at Oconee, is [ ] for the range of parameters given in Table F-7. This result is also valid below the first intermediate grid using the BWU-N correlation. D

Transition Cores

The transition core model determines the impact of the geometric and hydraulic differences between the resident Mark-B11 fuel and the new Mark-B-HTP design. The 9 channel model described in Reference F-1 is used to evaluate the impact of transition cores containing Mark-B-HTP fuel. Referring to Figure 4-5 in Reference F-1, Mark-B-HTP fuel is modeled instead of Mark-B6/7, and Mark-B11 fuel is modeled instead of Mark-B5. Therefore, Channels 1 - 7 are modeled as Mark-B-HTP fuel, Channel 8 is modeled as Mark-B11 fuel, and Channel 9 is modeled as Mark B-HTP fuel. The transition core analysis models each fuel type in those respective locations with the correct geometry. The form loss coefficients and geometry for each fuel design are input so the effect of crossflow between fuel designs is calculated.

A transition core penalty is evaluated by determining the DNBR impact on a Mark-B-HTP limiting assembly when analyzed with the 9 channel model. Once determined, several methods are available to conservatively account for the penalty. One method of accounting for the reduction in DNB performance due to the hydraulic effects of the conservatively modeled transition core is to explicitly apply a penalty to the Mark-B-HTP fuel generic peaking limits based on a full Mark B-HTP core. Another option is to calculate maximum allowable peaking limits specifically modeling the transition core loading pattern in the detailed 64 channel model approved in Reference F-1. These methods will be used, as necessary, to determine the DNB effect of transition cores.

To evaluate the impact of the transition core on the statistical DNB analysis, the most limiting statistical DNB statepoint (Statepoint 17 in Table F-6) was evaluated using the 9 channel model. This statepoint is designated 26-T in Table F-6. At 5,000 cases, the statistical DNBR for statepoint 26T is lower than the statistical design limit, [ ]. Therefore, the statistical design limit [ ] is bounding for Mark-B-HTP/Mark-B11 transition cores, as well as for full Mark-B-HTP cores. D

References

F-1 Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003-PA, Revision 2a, December 2008

F-2 VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Revision 4, February 2001

F-3 BHTP DNB Correlation Applied with LYNXT, BAW-10241(P)(A), Revision 1, AREVA NP, July 2005

F-4 The BWU Critical Heat Flux Correlations, BAW-10199P-A, AREVA NP, August 1996 (and including Addendum 1, December 2000)

F-5 LYNXT Core Transient Thermal-Hydraulic Program, BAW-10156-A, Revision 1, AREVA NP, August 1993

Table F-1

Mark-B-HTP Fuel Assembly Data

Dimensions (nominal inches)

Fuel rod diameter	0.430	
Guide tube diameter	[ ]	A
Instrument guide tube diameter	[ ]	
Fuel rod pitch	[ ]	
Fuel assembly pitch	8.587	
Fuel rod length	155.0	

Design Characteristics

	Material	Quantity	Location	Type
Grids	Inconel 718	1	Lower	HMP non-mixing
	M5®	7	Intermediate and upper	HTP non-mixing
Fuel Rods	M5®	208		
Guide Tubes	M5®	16		
Instrument Tube	M5®	1		

Table F-2

VIPRE-01 BHTP Correlation Verification

VIPRE-01/LYNXT Statistical Results

	VIPRE-01	LYNXT
n, # of data points	[ ] A,D	[ ] A
P/M, average predicted to measured CHF	[ ] A,D	[ ] A
σ (M/P/N)	[ ] A,D	[ ] A
DNBR correlation limit	1.120	1.132

Table F-3

CHF Test Database Analysis Results

Parameter Ranges

Pressure (psia)	1385 to 2425
Mass flux (Mlbm/hr-ft^2)	0.492 to 3.549
Thermodynamic quality at CHF	Less than 0.512
Thermal-hydraulic computer code	VIPRE-01
Spacer grid	Mark-B-HTP 15x15
Correlation design limit DNBR	1.132

Table F-4

Oconee SCD Statepoints

Statepoint Number	Power[1] (% RTP)	RCS Flow[2] % DF	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak FΔH
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						
25						
(3) 26						

D

Notes:

1) 100% RTP = 2568 MWt
2) 100% design flow = 352,000 gpm
3) Statepoint 17 was rerun with the mixed Mark-B-HTP/Mark-B11 core Oconee 9 channel model as per Reference F-1

Table F-5

Oconee Statistically Treated Uncertainties

Parameter	Type	Type of Distribution	Uncertainty	Standard Deviation	
Reactor System					
Core Power	Measurement	Normal	+/-2.0%FP	+/-1.0%FP	
Core Flow	Measurement	Normal	+/-4.2% design	+/-2.1% design	
Pressure	Measurement	Normal	+/-30.0 psi	+/-15.0 psi	
Temperature	Measurement	Normal	+/-2.0°F	+/-1.0°F	
Nuclear					
FΔH	Calculation	Normal	---	+/-2.84%	
Fz	Calculation	Normal	---	+/-2.91%	
Z	Calculation	Uniform	+/-6 inches	---	
Fq	Calculation	Normal	[	]	A
Hot Channel Flow Area	Measurement	Uniform	[ ]	---	A
DNBR	Correlation	Normal	---	[ ]	A, D
DNBR	Code	Normal	[	]	D

Table F-5 (cont.)

Oconee Statistically Treated Uncertainties

Parameter	Justification
System Pressure	This uncertainty accounts for random uncertainties in various instrumentation components. Since the random uncertainties are normally distributed, the square root of the sum of the squares (SRSS) that results in the pressure uncertainty is also normally distributed.
Inlet Temperature	Same approach as pressure uncertainty.
Core Power	The core power uncertainty was calculated by statistically combining various random uncertainties associated with the measurement of core power. Since the random uncertainties are normally distributed, the SRSS that results in the core power uncertainty is also normally distributed.
Core Flow	Same approach as core power uncertainty.
Radial Power	This uncertainty accounts for the error associated in the physics code's calculation of radial assembly power ($F\Delta h$) and the measurement of the assembly power. This uncertainty distribution is normal.
Axial Peak Power	This uncertainty accounts for the axial peak (Fz) prediction uncertainty of the physics codes. The uncertainty is assumed normally distributed.
Axial Peak Location	This uncertainty accounts for the possible error in interpolating on axial peak location (Z) in the maneuvering analysis. The uncertainty is based on the axial node length in the physics code. The uncertainty distribution is conservatively applied as uniform.
Rod Power HCF	This uncertainty accounts for the increase in rod power due to manufacturing tolerances. The uncertainty in calculating the peak pin from assembly radial peak is also statistically combined with the manufacturing tolerance uncertainty to arrive at the correct value. The uncertainty is normally distributed and conservatively applied as one-sided in the analysis to assure the MDNBR channel location is consistent for all cases.
Hot Channel Flow Area	This uncertainty accounts for manufacturing variations in the instrument guide tube subchannel flow area. This uncertainty is uniformly distributed and is conservatively applied as one-sided in the analysis to ensure the MDNBR channel location is consistent for all cases.
DNB Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normally distributed.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is normally distributed.

Table F-6

Oconee Statepoint Statistical Results
BHTP Critical Heat Flux Correlation

500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
25				
26				

D

Table F-6 (cont.)

Oconee Statepoint Statistical Results
BHTP Critical Heat Flux Correlation

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
9-T				
11-T				
15-T				
16-T				
17-T				
18-T				
20-T				
24-T				
25-T				
26-T				

D

Table F-7

Oconee Key Parameter Ranges

Parameter	Maximum	Minimum
Core power (%RTP)		
RCS flow (% Design)		
Pressure (psia)		
T inlet (°F)		
FΔH, Fz, Z		

D

Note: All values listed in this table are based on the currently analyzed statepoints. Ranges are subject to change based on future statepoint conditions.

Figure F-1

VIPRE-01 Predicted CHF Versus Measured CHF
Mark-B-HTP Data Base

A, D

Figure F-2

VIPRE-01 Predicted-to-Measured CHF vs. Mass Velocity
Mark-B-HTP Data Base

A, D

Figure F-3

VIPRE-01 Predicted to Measured CHF vs. Pressure
Mark-B-HTP Data Base

A, D

Figure F-4

VIPRE-01 Predicted to Measured CHF vs. Quality
Mark-B-HTP Data Base

A, D

The NRC request for additional information is provided on the following pages. However, the questions pertain to DPC-NE-2015, which is the Mk-B-HTP transition methodology report. Consequently, several questions do not pertain to DPC-NE-2005. In fact, only one question pertains to DPC-NE-2005 (question 8). The response to that question is provided herein while the responses to the other questions are not.

Duke Energy

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September 17, 2008

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Site, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Requests for Additional Information for Proposed License Amendment Request to Revise the Technical Specifications for AREVA NP Mark-B-HTP Fuel and for Methodology Report DPC-NE-2015-P "Mark-B-HTP Fuel Transition Methodology"
License Amendment Request No. 2007-12

Duke Energy Carolinas, LLC (Duke) submitted a license amendment request (LAR) dated October 22, 2007, for the Oconee Nuclear Station Renewed Facility Operating License (FOL) and Technical Specifications (TS) pursuant to 10 CFR 50.90. Specifically, Duke requested NRC review and approval of methodology report DPC-NE-2015-P, "Mark-B-HTP Fuel Transition Methodology" and revisions to Technical Specifications 2.1.1.2 and 5.6.5.b. Associated revisions to associated Technical Specification Bases B.2.1.1 and B.3.4.1 are provided. These revisions will allow the use of the AREVA NP Mark-B-HTP fuel design at the Oconee Nuclear Station beginning with Oconee Unit 2 Cycle 24 in December 2008. The Mark-B-HTP design is currently in use at several B&W design reactors.

Duke met with the NRC on March 3, 2008 to facilitate the LAR review. In emails dated May 8, 2008 and May 28, 2008, Duke received requests for additional information (RAI). Duke submitted responses to this RAI on July 14, 2008.

On August 27, 2008, following a conference call between Duke and the NRC, additional clarification was requested to the earlier responses to questions 6, 9, and 10. This submittal supersedes Duke's earlier RAI submittal dated July 14, 2008, and includes the revisions to these questions as well as a restatement of Duke's prior responses to the remaining questions.

Attachment 1 contains information that is proprietary to Duke and AREVA NP. In accordance with 10 CFR 2.390, Duke requests that this information be withheld from public disclosure. Affidavits are included (Enclosures 2 and 3) from each organization attesting to the proprietary nature of the information in the report. The specific information that is proprietary to each

organization is identified in the report. A non-proprietary version of this report is included in Attachment 2 that is suitable for public dissemination.

As communicated earlier, Duke requests approval of the LAR by September 30, 2008 with the amendment to become effective commencing with Oconee Unit 2 Cycle 24. This response is bounded by the initial review and approval of the Plant Operations Review Committee and Nuclear Safety Review Board; therefore, additional reviews were not required. Additionally, a copy of this response is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements.

Inquiries on this proposed amendment request should be directed to Reene' Gambrell of the Oconee Regulatory Compliance Group at (864) 885-3364.

Sincerely,

Dave Baxter, Vice President
Oconee Nuclear Site

Enclosures:
1. Notarized Affidavit of Dave Baxter
2. Notarized Affidavit of T. C. Geer
3. Notarized Affidavit of Gayle F. Elliott

Attachment:
1. Oconee Nuclear Station, Mark-B-HTP Fuel Transition Methodology, Response to NRC Request for Additional Information [Proprietary Version - Withhold from Public Disclosure]
2. Oconee Nuclear Station, Mark-B-HTP Fuel Transition Methodology, Response to NRC Request for Additional Information [Non-Proprietary Version].

bc w/enclosures and attachments:

Mr. Luis Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
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C. D. Fago
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S. P. Schultz
R. C. Harvey
R. R. St. Clair
R. L. Gill – NRI&IA
R. D. Hart – CNS
K. R. Ashe – MNS
G. B. Swindlehurst - NED
A. R. James - NED
NSRB, EC05N
ELL, EC050
File - T.S. Working
ONS Document Management

ENCLOSURE 1

AFFIDAVIT OF DAVE BAXTER

AFFIDAVIT

Dave Baxter, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Energy Carolinas, LLC, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this revision to the Renewed Facility Operating License Nos. DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

Dave Baxter, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 17 day of September 2008

Sheila A Smith
Notary Public

My Commission Expires:

6-12-2013
Date

SEAL

ENCLOSURE 2

AFFIDAVIT OF T. C. GEER

AFFIDAVIT OF THOMAS C. GEER

1. I am Vice President of 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.

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's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke 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 and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. 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.

   (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 version of the response to the Request for Additional Information from the Nuclear Regulatory Commission concerning Revision 0 to the Duke methodology report DPC-NE-2015-P, *Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology*. This information enables Duke to:

   (a) Support license amendment and Technical Specification revision requests for its Oconee reactors.

   (b) Perform nuclear design calculations on Oconee reactor cores containing low enriched uranium fuel.

Thomas C. Geer
T. C. Geer

(Continued)

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

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

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

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

5. Public disclosure of this information is likely to cause harm to Duke 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 to recoup a portion of its expenditures or benefit from the sale of the information.

Thomas C. Geer 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.

Thomas C. Geer
Thomas C. Geer

Subscribed and sworn to me: June 19, 2008
Date

Deborah S. Rome Deborah S. Rome
Notary Public

My Commission Expires: December 19, 2009

SEAL

DEBORAH S ROME
NOTARY
PUBLIC
CATAWBA COUNTY, NC

ENCLOSURE 3

AFFIDAVIT OF GAYLE F. ELLIOTT

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

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

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

3. I am familiar with the AREVA NP information contained in the Response to NRC RAI on DPC-NE-2015-P, Revision 0, "Oconee Nuclear Station Mark-B-HTP Fuel Transition Methodology," dated June 2008 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP 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 NP 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 NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP'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 NP.

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

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

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

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

8. AREVA NP 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 June, 2008.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/10
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010

RAI Question # 8
Provide justification that the use of BHTP CHF correlation meets the Limitations and Conditions described in the SE of BAW-10241P, Revision 1.

Response
As discussed on page 6-9 of DPC-NE-2015 (Appendix F of DPC-NE-2005-P), the complete BHTP CHF correlation database, including the additional [ ] data points used to justify A
extension of the
range of application of the correlation (Reference 1), was analyzed with the VIPRE-01 code. The predicted CHF to measured CHF (P/M) ratios are plotted against mass velocity, pressure and thermodynamic quality in attached Figures 1, 2 and 3. Figures 1-3 show (1) that there is no bias of VIPRE-01 predicted to measured CHF values with respect to mass velocity, pressure and thermodynamic quality, and (2) the BHTP correlation in VIPRE-01 conservatively predicts CHF for the extended range of independent parameters. Figures 1-3 compare closely with the same parameter representations in Reference 1 (Figures A.3, A.6 and A.7).

The BHTP CHF correlation will be used for DNBR analyses of Mark-B-HTP fuel using VIPRE-01 and the BHTP correlation will be applied within the range of independent variables given in Table F-3 (pg. 6-14) of DPC-NE-2015. The range of variables in Table F-3 is identical to the extended range of variables given in Table 1 of Reference 1. If operating conditions require extrapolations beyond the approved pressure or quality ranges, the limitations and conditions listed in Section 4 of Reference 1 will be adhered to:

- When pressure greater than the pressure limit of 2425 psia, but less than 2600 psia, is encountered, all of the local coolant conditions are calculated at the upper pressure limit of 2425 psia using the NRC-approved VIPRE-01 thermal-hydraulic code and then used in the calculation of the BHTP CHF.

- Extrapolation below the minimum quality range is performed with no lower limit, consistent with EMF-92-153(P)(A) Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel".

No other parameter extrapolations are performed.

<u>Reference</u>

Letter, H. N. Berkow (USNRC) to R. L. Gardner (Framatome ANP), Final Safety Evaluation for Framatome ANP (FANP), Appendix A to Topical Report (TR) BAW-10241(P), Revision

1, "Extension of the BHTP CHF (Critical Heat Flux) Correlation Ranges", TAC No. MC6374, July 25, 2005.

[ ]A,D

A,D

A,D

Appendix G and Response to Request for Additional Information

Revision 4

DPC-NE-2005-PA

Duke Energy Carolinas
Thermal-Hydraulic Statistical Core Design Methodology

APPENDIX G

McGuire/Catawba Plant Specific Data

Westinghouse RFA Fuel and WRB-2M CHF Correlation

Approved in Revision 2 of DPC-NE-2009: December 2002
Included in Revision 4 of DPC-NE-2005: December 2008

This Appendix contains the plant specific data and statistical DNB limits for the McGuire and Catawba Nuclear Stations with the Westinghouse RFA fuel design using the WRB-2M critical heat flux correlation. Unless otherwise noted, all previous modeling inputs in VIPRE-01 listed in Reference G-1 for the 17x17 fuel at McGuire and Catawba are unchanged. The thermal-hydraulic statistical core design analysis was performed as described in the main body of this report (DPC-NE-2005).

Plant Specific Data

This analysis is for the McGuire and Catawba plants (four-loop Westinghouse PWR's) with the RFA design. The Robust fuel design includes 0.374 OD fuel rods and non-structural Intermediate Flow Mixing (IFM) grids in the upper three spans to improve DNB performance. This design also includes the fuel reliability features of a debris filtering bottom and a protective grid between this nozzle and the first structural grid.

The parameter uncertainties and statepoint ranges were selected to bound the McGuire and Catawba unit and cycle-specific values (see Statepoints and Key Parameters and Uncertainties).

Thermal-Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference G-3 and the McGuire/Catawba eight channel model approved in Reference G-1 are used in this analysis. The VIPRE-01 models approved in Reference G-1 for the Mark-BW fuel are used to analyze the RFA design with the following changes:

1) The RFA design geometry information is listed in Table G-1. Applicable form loss coefficients as per the vendor were used in the models. Also, the axial noding was adjusted to be compatible with the Westinghouse WRB-2M CHF correlation.
2) The bulk void fraction model was changed from the Zuber-Findlay model to the EPRI model. Correspondingly, the subcooled void model was changed from the Levy to EPRI model.
3) The reference pin peak described in Reference G-1 was increased from 1.60 to 1.67. The associated pin power distribution was also updated based on this higher value.
4) The reference axial power profile (symmetric chopped cosine) peak to average value described in Reference G-1 was increased from 1.55 to 1.60.

With respect to Item 2), the Zuber-Findlay bulk void model is applicable only to qualities below approximately 0.7 (void fractions of 0.85) and is discontinuous at higher values (Reference G-3). The EPRI bulk void model is essentially the same as the Zuber-Findlay bulk void model except for the equation used to calculate the drift velocity (Reference G-3). This eliminates the discontinuity at high qualities and void fractions. Therefore, the EPRI model covers the full range (i.e., void fraction range, 0 - 1.0) of void fractions required for performing DNB calculations. Also, for overall void model compatibility, the subcooled void model was changed from the Levy model, as specified in Reference G-1, to the EPRI correlation.

To evaluate the impact of changing bulk void models on DNB predictions, fifty-one RFA critical heat flux test data points (Reference G-2) were compared using both the Levy/Zuber-Findlay and EPRI/EPRI subcooled void / bulk void model combinations in VIPRE-01. These data points cover a pressure range of 1519 to 2426 psia and an inlet temperature range 397.4 to 617.6°F. The mass flux at the MDNBR location varied from 1.48 to 3.02 Mlbm/hr-ft^2. The void fraction at the MDNBR location varied from 0.309 to 0.697. The equilibrium quality at the MDNBR location varied from 0.07 to 0.254. The results of this comparison are as follows:

	Levy/Zuber-Findlay	EPRI/EPRI
Minimum DNBR (Avg.)	1.029	1.028

The minimum DNBR results show a minimal difference of 0.1% (0.001 in DNB). Therefore, the EPRI bulk void model and EPRI subcooled void correlation will be used in RFA analyses.

The changes related to Items 3) and 4) above are due to the RFA fuel design containing significant DNBR margin due to the addition of the IFM grids. This DNB margin is applied in core design space by increasing the reference radial and axial peaking. With respect to radial peaking, all three models described in Reference G-1 (8, 12, and 75 channel models) are based on the maximum pin power value. Therefore, all three models were updated to the new peak pin value of 1.67. The resulting pin power distributions from this change are shown in Figures G-1, G-2, and G-3.

Critical Heat Flux Correlation

The WRB-2M critical heat flux correlation described in Reference G-2 is used for all statepoint analyses. This correlation was developed by Westinghouse for application to the RFA design. As discussed in Reference G-2 the WRB-2M correlation was developed with the VIPRE-01 thermal-hydraulic computer code. This correlation was programmed into the Duke Energy version of VIPRE-01 and will be used in all DNBR calculations for the RFA design, except for the following:

1. steam line break transient described in Reference G-5.
2. the non-mixing vane span of the RFA fuel (below the first mixing vane Zircaloy grid). For this region of the fuel, the BWU-N CHF correlation will be applied (Reference G-4) to the RFA fuel.

Statepoints

The statepoint conditions evaluated in this analysis are listed in Table G-2. These statepoints cover the range of conditions to which the statistical DNBR limit will be applied. The range of key parameter values evaluated in this analysis are listed on Table G-5.

Key Parameters and Uncertainties

The key parameters and their uncertainty magnitude and associated distribution used in this analysis are listed on Table G-3. The uncertainties were selected to bound the values calculated for each parameter at McGuire and Catawba.

DNB Statistical Design Limit

The statistical DNBR value for each statepoint evaluated is listed on Table G-4. Section 1 of Table G-4 contains the 500 case runs and Section 2 contains the 5000 case runs. The number of cases was increased from 3000 to 5000 as described in Attachment 1 of the main body of this report. The DNBRs calculated for all of the statepoints are normally distributed. As shown in Section 2 of Table G-4 the maximum statepoint statistical DNBR value is [ ]. Therefore, the D
statistical design limit (SDL) using the WRB-2M CHF correlation for the RFA design at McGuire/Catawba is conservatively determined to be 1.30.

Transition Cores

A transition core model is used to determine the impact of the geometric and hydraulic differences between the resident FCF Mark-BW fuel and the Westinghouse RFA design. The 8 channel model described in Reference G-1 is used to evaluate the impact of transition cores containing the RFA design. In Figure 5 of Reference G-1, the RFA design is used instead of Mark-BW fuel. Therefore, the limiting assembly (Channels 1 through 7) is modeled as the RFA design and the remainder of the core (Channel 8) is modeled as Mark-BW fuel. The transition core analysis models each fuel type in their respective locations with the correct geometry. The form loss coefficients for each fuel design are input so the effect of crossflow out of the IFM grid spans in the limiting channel is calculated.

To evaluate the impact of the transition core on the statistical DNBR limit, the most limiting full core statepoint (Statepoint 12 on Table G-4) was evaluated using the 8 channel transition core model. This case is designated as statepoint 12TR in Sections 1 and 2 of Table G-4. The statistical DNBR calculated using the transition core model (statepoint 12TR) is slightly greater than the Statistical DNBR value for the full RFA core (statepoint 12) at both the 500 and 5000 cases levels. As shown in Section 2 of Table G-4, this value is still less than 1.30. Therefore, the statistical design limit of 1.30 is bounding for RFA/Mark-BW transition cores as well as full RFA cores.

For initial transition reload cycles, a transition core DNBR penalty is determined for the RFA design using the 8 channel RFA/Mark-BW transition core model. For subsequent cycles where the RFA fuel composes greater than 80% of the assemblies incore, the 75 channel model shown in Figure G-3 and described in Reference G-1 is used to determine a transition core penalty. In either case, a conservative penalty is applied for all DNBR analyses in transition cycles to bound the effects of mixed cores.

References

G-1. DPC-NE-2004-PA, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, Rev 2a, December 2008.

G-2. WCAP-15025-P, Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids, Westinghouse Energy Systems, February 1998.

G-3. VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

G-4. BAW-10199P-A, The BWU Critical Heat Flux Correlations, AREVA NP, April 1996.

G-5. DPC-NE-3001P-A, McGuire Nuclear Station and Catawba Nuclear Station Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, Revision 0, December 2000.

TABLE G-1

RFA Design Data

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, inches (Nominal)	0.374
Guide tube diameter, inches (Nominal)	0.482
Fuel rod pitch, inches (Nominal)	0.496
Fuel Assembly pitch, inches (Nominal)	8.466
Fuel Assembly length, inches (Nominal)	159.8

GENERAL FUEL CHARACTERISTICS

Component	Material	Number	Location/Type
Grids	Inconel	1	Lower Protective
	Inconel	2	Upper and Lower Non-Mixing Vane
	ZIRLO™	6	Intermediate Mixing Vane
	ZIRLO™	3	Intermediate Flow Mixer (Non-structural)
Nozzles	304SS	1	Debris Filtering Bottom
	304SS	1	Removable Top

TABLE G-2

McGuire/Catawba SCD Statepoints, WRB-2M Correlation

Stpt No.	Power* (% RTP)	RCS Flow** (K gpm)	Pressure (psia)	Core Inlet Temperature (°F)	Axial Peak (F_z @ Z)	Radial Peak ($F\Delta H$)
1						
2						
3						
4						
5						
6						
7						
8						
9						
10						
11						
12						
13						
14						
15						
16						
17						
18						
19						
20						
21						
22						
23						
24						
12TR***						D

* 100% RTP = 3411 Megawatts Thermal

** Mass flow rate should be calculated using the given core inlet temp.

*** TR - transition core model

TABLE G-3

McGuire/Catawba Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type Of Distribution
Core Power*	+/- 2% / 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^N_{\Delta H}$		
Measurement	+/- 4.0% / 2.43%	Normal
$F^E_{\Delta H}$	+/- 3.0% / 1.82%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.41% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	+/- 10.73% / 6.52%	Normal
Code/Model	[ ] D	Normal

* Percentage of 100% RTP (3411 MWth).

TABLE G-3 (Continued)

McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
Core Power	The core power uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from normally distributed random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method (SRSS). Since the uncertainty is calculated from normally distributed values, the parameter distribution is also normal.
Core Flow	
Measurement	Same approach as core power.
Bypass Flow	The core bypass flow is the parallel core flow paths in the reactor vessel (guide thimble cooling flow, head cooling flow, fuel assembly/baffle gap leakage, and hot leg outlet nozzle gap leakage) and is dependent on the driving pressure drop. Parameterizations of the key factors that control ΔP, dimensions, loss coefficient correlations, and the effect of the uncertainty in the driving ΔP on the flow rate in each flow path, was performed. The dimensional tolerance changes were combined with the SRSS method and the loss coefficient and driving ΔP uncertainties were conservatively added to obtain the combined uncertainty. This uncertainty was conservatively applied with a uniform distribution.
Pressure	The pressure uncertainty was calculated by statistically combining the uncertainties of the process indication and control channels. The uncertainty is calculated from random error terms such as sensor calibration accuracy, rack drift, sensor drift, etc. combined by the square root sum of squares method. The uncertainty distribution was conservatively applied as uniform.
Temperature	Same approach as pressure.
$\mathbf{F^N_{\Delta H}}$	
Measurement	This uncertainty is the measurement uncertainty for the movable incore instruments. A measurement uncertainty can arise from instrumentation drift or reproducibility error, integration and location error, error associated with the burnup history of the core, and the error associated with the conversion of instrument readings to rod power. The uncertainty distribution is normal.

TABLE G-3 (Continued)

McGuire/Catawba Statistically Treated Uncertainties

Parameter	Justification
$F^{E}_{\Delta H}$	This uncertainty accounts for the manufacturing variations in the variables affecting the heat generation rate along the flow channel. This conservatively accounts for possible variations in the pellet diameter, density, and U_{235} enrichment. This uncertainty distribution is normal and was conservatively applied as one-sided in the analysis to ensure the MDNBR channel location was consistent for all cases.
Spacing	This uncertainty accounts for the effect on peaking of reduced hot channel flow area and spacing between assemblies. The power peaking gradient becomes steeper across the assembly due to reduced flow area and spacing. This uncertainty distribution is normal and was conservatively applied as one-sided to ensure consistent MDNBR channel location.
F_Z	This uncertainty accounts for the axial peak prediction uncertainty of the physics codes. The uncertainty distribution is applied as normal.
Z	This uncertainty accounts for the possible error in interpolating on axial peak location in the maneuvering analysis. The uncertainty is one half of the physics code's axial node. The uncertainty distribution is conservatively applied as uniform.
DNBR	
Correlation	This uncertainty accounts for the CHF correlation's ability to predict DNB. The uncertainty distribution is applied as normal.
Code/Model	This uncertainty accounts for the thermal-hydraulic code uncertainties and offsetting conservatisms. This uncertainty also accounts for the small DNB prediction differences between the various model sizes. The uncertainty distribution is applied as normal.

Table G-4

McGuire/Catawba Statepoint Statistical Results

SECTION 1
WRB-2M Critical Heat Flux Correlation
500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
12TR*				

D

* TR - transition core model

Table G-4 (Continued)

McGuire/Catawba Statepoint Statistical Results

SECTION 2

WRB-2M Critical Heat Flux Correlation

5000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
7				
11				
12				
12TR*				

D

* TR - transition core model

Table G-5

McGuire/Catawba Key Parameter Ranges

WRB-2M CHF Correlation

Parameter	Maximum	Minimum
Core Power* (% RTP)	[	
Pressure (psia)		
T inlet (deg. F)		
RCS Flow (Thousand GPM)		
FΔH, Fz, Z		] D

* 100% RTP = 3411 Megawatts Thermal

All values listed in this table are based on the currently analyzed statepoints (Table 5-2). Ranges are subject to change based on future statepoint conditions.

FIGURE G-1
8 CHANNEL MODEL – GEOMETRY AND REFERENCE
POWER DISTRIBUTION

[ ] D

FIGURE G-2
12 CHANNEL MODEL – GEOMETRY AND REFERENCE
POWER DISTRIBUTION

D

FIGURE G-3
75 CHANNEL MODEL – GEOMETRY AND REFERENCE POWER DISTRIBUTION

D

Revision 0 of Appendix G
RAI Letters and Responses

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

December 9, 1998

Mr. Gary R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION - REQUEST FOR ADDITIONAL INFORMATION ON YOUR AMENDMENT REQUEST OF JULY 22, 1998 (TAC NOS. MA2359 AND MA2361)

Dear Mr. Peterson:

By letter dated July 22, 1998, Duke Energy Corporation (DEC) proposed to amend the Catawba Nuclear Station, Units 1 and 2, Technical Specifications to permit use of Westinghouse fuel. Topical Report DPC-NE-2009P/ DPC-NE-2009, "Duke Power Company Westinghouse Fuel Transition Report" was part of DEC's submittal. The original submittal was supplemented by letter dated October 22, 1998.

The staff is reviewing DEC's submittals, and has found that additional information is needed to complete the review (enclosed). We have discussed this request for additional information with Mr. Steve Warren of your staff, and agreed that the response would be due on or before January 31, 1999. We will be glad to discuss the questions with you upon your request.

Sincerely,

Peter S. Tam

Peter S. Tam, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosure: Request for Additional Information

cc w/encl: See next page

REQUEST FOR ADDITIONAL INFORMATION

DPC-NE-2009, "DUKE POWER COMPANY

WESTINGHOUSE FUEL TRANSITION REPORT"

(Reference: Letter, M. S. Tuckman to NRC, July 22, 1998)

1. Section 3.2 of DPC-NE-2009P states that conceptual transition core designs using the Robust Fuel Assembly (RFA) design have been evaluated and show that current reload limits remain bounding with respect to key physics parameters, and that in the event that one of the key parameters is exceeded, the evaluation process described in DPC-NE-3001-PA would be performed.

   (a) Describe the evaluation and the result of the conceptual transition core design.

   (b) Based on the statement, it appears that the evaluation process described in DPC-NE-3001-PA will not be performed unless one of the key parameters is exceeded. Without actual analysis of the RFA transitional or full cores, how is it determined that any of the key parameters is exceeded?

2. To demonstrate that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties in DPC-NE-1004-PA are applicable to the RFA design, Section 3.2 cites the analyses performed using Sequoyah Unit 2 Cycles 5, 6, and 7, as well as a 10 CFR 50.59 unreviewed safety question (USQ) evaluation. It is stated that the Sequoyah cores were chosen because they are similar to McGuire and Catawba and contained both Integral Fuel Burnable Absorber (IFBA) and Wet Annular Burnable Absorber fuel. Table 3-1 provides the statistical analysis results of nuclear uncertainty factors, which show they are bounded by the uncertainty factors of DPC-NE-1004A.

   (a) Describe any difference between the Catawba RFA cores and the Sequoyah cores analyzed. Describe why these differences would not affect the applicability of the analyses of the Sequoyah cores to Catawba.

   (b) Provide the comparison of the analysis results with measured data of boron concentrations, rod worths, and isothermal temperature coefficients.

   (c) Describe the details and results of the 10 CFR 50.59 USQ evaluation.

3. Section 3.2 states that (1) in all nuclear design analysis, both the RFA and the Mark-BW fuel are explicitly modeled in the transition cores, and (2) when establishing operating and reactor protection system limits (i.e., loss-of-coolant accident (LOCA) kw/ft, departure from nucleate boiling (DNB), containment failure mode, transient strain), the fuel specific limits or a conservative overlay of the limits are used. Please elaborate on the mixed core model for nuclear design analyses, and how fuel-specific limits are used.

Enclosure

4. Section 5.2 states that in using the VIPRE-01 code for the reactor core thermal-hydraulic analysis, the reference power distribution based on a 1.60 peak pin from DPC-NE-2004P-A, Revision 1, was used.

   (a) The report states that this reference pin power distribution "was" used. Will it be used for future RFA reload analyses?

   (b) Does the reference pin power distribution used in the core thermal-hydraulic analyses bound all power distribution for the RFA cores for future reload cycles?

5. Section 5.2 states that in the thermal-hydraulic analysis of the RFA design using VIPRE-01, the two-phase flow correlations will be changed from the Levy subcooled void correlation and the Zuber-Findlay bulk void correlation to the EPRI subcooled and bulk void correlations, respectively. While the sensitivity study provided in the report shows a minimal difference of 0.1 percent between the minimum DNB ratios (DNBRs) of 51 RFA critical heat flux (CHF) test data points calculated with both sets of correlations, it was stated in DPC-NE-2004 that the Levy/Zuber-Findlay combination compared most favorably with the Mark-BW test results as the DNBRs of the tests calculated with this combination yielded conservative results relative to the EPRI correlations.

   (a) Discuss whether the EPRI correlations will be used for the RFA design only, or if they will also be used for the Mark-BW design.

   (b) If the EPRI correlations will also be used for Mark-BW design, provide justification for their use.

   (c) If the Levy/Zuber-Findlay correlations will continue to be used for the Mark-BW fuel design, discuss how the VIPRE-01 code will be used to analyze transient mixed cores having both Mark-BW and RFA fuel designs.

6. Section 5.7 describes the use of a transition 8-channel RFA/Mark-BW core model to determine the impact of the geometric and hydraulic differences between the resident Mark-BW fuel and the RFA design, and determine a conservative DNBR penalty to be applied for the transition cores. Table 5-4 presented the statistical DNBRs for the 500 and 5000 case runs for various statepoints including the transition core case of the most limiting statepoint 12. The statistical design limit is chosen to bound both the full RFA cores and RFA/Mark-BW transition cores for the 5000 case runs.

   (a) Why is the statistical design limit value proprietary information?

   (b) With respect to the statistical core design methodology, describe how the uncertainties of the CHF correlation and the VIPRE code/model are propagated with the uncertainties of the selected parameters of each statepoint for the calculation of the statistical DNBR for each statepoint in Table 5-4.

   (c) With the statistical design limit specified in Section 5.7, is it your intention to use a full core of RFA in the thermal-hydraulic analysis for the transition core without the transition core DNBR penalty factor?

7. Section 2.0 states that the RFA is designed to be mechanically and hydraulically compatible with the Mark-BW fuel. Table 2.1 provides a comparison of the basic-design parameters of the two fuel designs, but does not provide a comparison of the hydraulic characteristics of spacer grids. Section 5.2 states that the VIPRE-01 core thermal-hydraulic analyses were performed with applicable form loss coefficients according to the vendor. Table 5.1 provides general RFA fuel specifications and characteristics without the hydraulic characteristics of the spacer grids.

   (a) Provide comparisons for the thickness, height, and form loss coefficients of the RFA and Mark-BW fuel spacer grids, including mixing-vane and nonmixing vane structural grids, and intermediate flow mixing grids.

   (b) Provide the form loss coefficients of the spacer grids used in the analyses and in the RFA CHF test assemblies if they are different from the values described in item (a).

   (c) Describe the procedures to ensure that the form loss coefficients of the RFA grids are comparable to those used in the statistical core design analysis and the CHF tests so that both the WRB-2M CHF correlation DNBR limit and the statistical core design limit are valid.

8. Section 6.1.3 states that the thermal-hydraulic methodology described in DPC-NE-3000-PA, Revision 1, with a simplified core model will be used for thermal-hydraulic analysis of the Updated Final Safety Analysis Report Chapter 15 non-LOCA transients and accidents for the RFA design. It also states that (1) no transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses, (2) the simplified core model of DPC-NE-3000-PA used for transient analyses was originally developed with additional conservatism over the 8-channel model used for steady-state analyses to specifically minimize the impact of changes in core reload design methods or fuel assembly design, and (3) should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly.

   (a) Explain what additional conservatism is provided in using the simplified core model of DPC-NE-3000-PA.

   (b) What is the criterion/criteria used to determine if transition core transient analyses are warranted? How would it be determined that the criteria have been exceeded without RFA transition core analyses?

9. Regarding rod ejection analysis using SIMULATE-3K, Section 6.6.2.2.1 states that the transient response is made more conservative by increasing the fission cross sections in the ejected rod location and in each assembly and by applying "factors of conservatism" in the moderator temperature coefficient, control rod worths for withdrawal and insertion, Doppler temperature coefficient, effective delay neutron fraction, and ejected rod worth, etc.

   (a) What are the values of the multiplication factors used for fission cross sections, and how are they determined?

(b) How are the input multipliers "VAL" in Equations 6.1 and 6.2 determined? Does "VAL" have a different value for different parameters, such as MTC or DTC? What are the values for these VALs?

(c) In Equation 6.1, the X's are described as "moderator temperatures." Should they be moderator temperature coefficients?

10. Regarding the SIMULATE-3K code, there is an optional "frequency transform" approach, under the "Temporal Integration Models," that can be chosen to separate the fluxes into exponential time varying and predominately spatial components, thus accelerating convergence of the transient neutronic solution and preserving accuracy on a coarser time mesh (see Page 5, Ref. 6-9).

   (a) What determines when the "frequency transform" approach should be used?

   (b) What are the consequences of exercising (or not exercising) this option? Please provide technical justification and comparisons of results.

11. The licensing analyses of reload cores with the RFA design will use the methodologies described in various topical reports and revisions for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current Catawba cores not having the RFA design. For example, DPC-NE-1004A, DPC-NE-2011-PA, DPC-NF-2010A, and DPC-NE-3001-PA are used for the nuclear design calculations. DPC-NE-2004-PA, DPC-NE-2005-PA, and the VIPRE-01 code are used for the core thermal-hydraulic analyses and statistical core design. DPC-NE-3000-PA, DPC-NE-3001-PA, DPC-NE-3002-A, and RETRAN-02 code are used for non-LOCA transient and accident analyses. Westinghouse small- and large-break LOCA evaluation models described in WCAP-10054-P-A and WCAP-10266-P-A, and related topical reports, are used for the small- and large-break LOCA analyses. Some of these methodologies have inherent limitations, and some have conditions or limitations imposed by the NRC safety evaluation reports in their applications. Provide a list of the inherent limitations, conditions, or restrictions applicable to the RFA core design from all the methodologies to be used for the RFA reload design analyses, and describe the resolutions of these limitations, conditions, and restrictions in the applications to the RFA cores and the transitional RFA/Mark-BW cores.

12. Section 8.0 states that TS Figure 2.1.1-1 for the reactor core safety limits will be modified by deleting the 2455 psia safety limit line and making the 2400 psia safety limit line as the upper bound pressure allowed for power operation. Since the upper range of applicability of the WRB-2M CHF correlation for the RFA design is 2425 psia, the 2400 psia safety limit line is within the range of the CHF correlations for the Mark-BW and RFA fuel designs.

    However, the safety limit lines in Figure 2.1.1-1 were based on the CHF correlation for the Mark-BW fuel design, in addition to the hot leg boiling limit. Has an analysis been performed to ensure these safety limit lines bound the safety limit for the DNBR limit of the WRB-2M correlation for the RFA design?

13. TS Surveillance Requirements (SRs) 3.2.1.2, 3.2.1.3, and 3.2.2.2, respectively, require the heat flux hot channel factor F_q (x,y,z) and the enthalpy rise hot channel factor $F_{\Delta h}$ (x,y) to be measured periodically using the incore detector system to ensure that the values of the total peaking factor and the enthalpy rise factor assumed in the accident analyses and the reactor protection system limits are not violated. To avoid the possibility that these hot channel factors may increase beyond their allowable limits between surveillances, these SRs currently specify a penalty factor of 1.02 for the heat flux and enthalpy rise hot channel factors if the margin to the F_q (x,y,z) or $F_{\Delta h}$ (x,y) has decreased since the previous surveillance. For the reactor core containing the RFA fuel design with integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. Section 8.1 proposes to remove the 2 percent penalty value from these surveillance requirements and replace them with tables of penalty values as functions of burnup in the Core Operating Limits Report (COLR) to facilitate cycle-specific updates. Tables 8-1 and 8-2, respectively, provide "typical values" for the burnup-dependent margin-decrease penalty factors for the heat flux and enthalpy rise hot channel factors.

(a) Provide the actual values of the margin-decrease penalty factors, as well as the bases, for these values.

(b) Provide references for the approved methodologies used to calculate these values, and to be included in TS 5.6.5 as a part of acceptability for COLR.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
January 5, 1999

Mr. H. B. Barron
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: MCGUIRE NUCLEAR STATION - REQUEST FOR ADDITIONAL INFORMATION ON YOUR AMENDMENT REQUEST OF JULY 22, 1998 (TAC NOS. MA2411 AND MA2412)

Dear Mr. Barron:

By letter dated July 22, 1998, Duke Energy Corporation (DEC) proposed to amend the McGuire Nuclear Station, Units 1 and 2, Technical Specifications to permit use of Westinghouse fuel. Topical Report DPC-NE-2009P/DPC-NE-2009, "Duke Power Company Westinghouse Fuel Transition Report" was part of DEC's submittal. The original submittal was supplemented by letter dated October 22, 1998.

The staff is reviewing DEC's submittals, and has found that additional information is needed to complete the review (enclosed). We have discussed this request for additional information with Mr. Steve Warren of your staff, and agreed that the response would be due on or before January 31, 1999. We will be glad to discuss the questions with you upon your request.

Sincerely,

Frank Rinaldi

Frank Rinaldi, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosure: Request for Additional Information

cc w/encl: See next page

REQUEST FOR ADDITIONAL INFORMATION

DPC-NE-2009, "DUKE POWER COMPANY

WESTINGHOUSE FUEL TRANSITION REPORT"

(Reference: Letter, M. S. Tuckman to NRC, July 22, 1998)

1. Section 3.2 of DPC-NE-2009P states that conceptual transition core designs using the Robust Fuel Assembly (RFA) design have been evaluated and show that current reload limits remain bounding with respect to key physics parameters, and that in the event that one of the key parameters is exceeded, the evaluation process described in DPC-NE-3001-PA would be performed.

   (a) Describe the evaluation and the result of the conceptual transition core design.

   (b) Based on the statement, it appears that the evaluation process described in DPC-NE-3001-PA will not be performed unless one of the key parameters is exceeded. Without actual analysis of the RFA transitional or full cores, how is it determined that any of the key parameters is exceeded?

2. To demonstrate that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties in DPC-NE-1004-PA are applicable to the RFA design, Section 3.2 cites the analyses performed using Sequoyah Unit 2 Cycles 5, 6, and 7, as well as a 10 CFR 50.59 unreviewed safety question (USQ) evaluation. It is stated that the Sequoyah cores were chosen because they are similar to McGuire and Catawba and contained both Integral Fuel Burnable Absorber (IFBA) and Wet Annular Burnable Absorber fuel. Table 3-1 provides the statistical analysis results of nuclear uncertainty factors, which show they are bounded by the uncertainty factors of DPC-NE-1004A.

   (a) Describe any difference between the Catawba RFA cores and the Sequoyah cores analyzed. Describe why these differences would not affect the applicability of the analyses of the Sequoyah cores to Catawba.

   (b) Provide the comparison of the analysis results with measured data of boron concentrations, rod worths, and isothermal temperature coefficients.

   (c) Describe the details and results of the 10 CFR 50.59 USQ evaluation.

3. Section 3.2 states that (1) in all nuclear design analysis, both the RFA and the Mark-BW fuel are explicitly modeled in the transition cores, and (2) when establishing operating and reactor protection system limits (i.e., loss-of-coolant accident (LOCA) kw/ft, departure from nucleate boiling (DNB), containment failure mode, transient strain), the fuel specific limits or a conservative overlay of the limits are used. Please elaborate on the mixed core model for nuclear design analyses, and how fuel-specific limits are used.

Enclosure

4. Section 5.2 states that in using the VIPRE-01 code for the reactor core thermal-hydraulic analysis, the reference power distribution based on a 1.60 peak pin from DPC-NE-2004P-A, Revision 1, was used.

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   (a) Why is the statistical design limit value proprietary information?

   (b) With respect to the statistical core design methodology, describe how the uncertainties of the CHF correlation and the VIPRE code/model are propagated with the uncertainties of the selected parameters of each statepoint for the calculation of the statistical DNBR for each statepoint in Table 5-4.

   (c) With the statistical design limit specified in Section 5.7, is it your intention to use a full core of RFA in the thermal-hydraulic analysis for the transition core without the transition core DNBR penalty factor?

7. Section 2.0 states that the RFA is designed to be mechanically and hydraulically compatible with the Mark-BW fuel. Table 2.1 provides a comparison of the basic-design parameters of the two fuel designs, but does not provide a comparison of the hydraulic characteristics of spacer grids. Section 5.2 states that the VIPRE-01 core thermal-hydraulic analyses were performed with applicable form loss coefficients according to the vendor. Table 5.1 provides general RFA fuel specifications and characteristics without the hydraulic characteristics of the spacer grids.

   (a) Provide comparisons for the thickness, height, and form loss coefficients of the RFA and Mark-BW fuel spacer grids, including mixing-vane and nonmixing vane structural grids, and intermediate flow mixing grids.

   (b) Provide the form loss coefficients of the spacer grids used in the analyses and in the RFA CHF test assemblies if they are different from the values described in item (a).

   (c) Describe the procedures to ensure that the form loss coefficients of the RFA grids are comparable to those used in the statistical core design analysis and the CHF tests so that both the WRB-2M CHF correlation DNBR limit and the statistical core design limit are valid.

8. Section 6.1.3 states that the thermal-hydraulic methodology described in DPC-NE-3000-PA, Revision 1, with a simplified core model will be used for thermal-hydraulic analysis of the Updated Final Safety Analysis Report Chapter 15 non-LOCA transients and accidents for the RFA design. It also states that (1) no transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses, (2) the simplified core model of DPC-NE-3000-PA used for transient analyses was originally developed with additional conservatism over the 8-channel model used for steady-state analyses to specifically minimize the impact of changes in core reload design methods or fuel assembly design, and (3) should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly.

   (a) Explain what additional conservatism is provided in using the simplified core model of DPC-NE-3000-PA.

   (b) What is the criterion/criteria used to determine if transition core transient analyses are warranted? How would it be determined that the criteria have been exceeded without RFA transition core analyses?

9. Regarding rod ejection analysis using SIMULATE-3K, Section 6.6.2.2.1 states that the transient response is made more conservative by increasing the fission cross sections in the ejected rod location and in each assembly and by applying "factors of conservatism" in the moderator temperature coefficient, control rod worths for withdrawal and insertion, Doppler temperature coefficient, effective delay neutron fraction, and ejected rod worth, etc.

   (a) What are the values of the multiplication factors used for fission cross sections, and how are they determined?

(b) How are the input multipliers "VAL" in Equations 6.1 and 6.2 determined? Does "VAL" have a different value for different parameters, such as MTC or DTC? What are the values for these VALs?

(c) In Equation 6.1, the X's are described as "moderator temperatures." Should they be moderator temperature coefficients?

10. Regarding the SIMULATE-3K code, there is an optional "frequency transform" approach, under the "Temporal Integration Models," that can be chosen to separate the fluxes into exponential time varying and predominately spatial components, thus accelerating convergence of the transient neutronic solution and preserving accuracy on a coarser time mesh (see Page 5, Ref. 6-9).

    (a) What determines when the "frequency transform" approach should be used?

    (b) What are the consequences of exercising (or not exercising) this option? Please provide technical justification and comparisons of results.

11. The licensing analyses of reload cores with the RFA design will use the methodologies described in various topical reports and revisions for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current Catawba cores not having the RFA design. For example, DPC-NE-1004A, DPC-NE-2011-PA, DPC-NF-2010A, and DPC-NE-3001-PA are used for the nuclear design calculations. DPC-NE-2004-PA, DPC-NE-2005-PA, and the VIPRE-01 code are used for the core thermal-hydraulic analyses and statistical core design. DPC-NE-3000-PA, DPC-NE-3001-PA, DPC-NE-3002-A, and RETRAN-02 code are used for non-LOCA transient and accident analyses. Westinghouse small- and large-break LOCA evaluation models described in WCAP-10054-P-A and WCAP-10266-P-A, and related topical reports, are used for the small- and large-break LOCA analyses. Some of these methodologies have inherent limitations, and some have conditions or limitations imposed by the NRC safety evaluation reports in their applications. Provide a list of the inherent limitations, conditions, or restrictions applicable to the RFA core design from all the methodologies to be used for the RFA reload design analyses, and describe the resolutions of these limitations, conditions, and restrictions in the applications to the RFA cores and the transitional RFA/Mark-BW cores.

12. Section 8.0 states that TS Figure 2.1.1-1 for the reactor core safety limits will be modified by deleting the 2455 psia safety limit line and making the 2400 psia safety limit line as the upper bound pressure allowed for power operation. Since the upper range of applicability of the WRB-2M CHF correlation for the RFA design is 2425 psia, the 2400 psia safety limit line is within the range of the CHF correlations for the Mark-BW and RFA fuel designs.

    However, the safety limit lines in Figure 2.1.1-1 were based on the CHF correlation for the Mark-BW fuel design, in addition to the hot leg boiling limit. Has an analysis been performed to ensure these safety limit lines bound the safety limit for the DNBR limit of the WRB-2M correlation for the RFA design?

13. TS Surveillance Requirements (SRs) 3.2.1.2, 3.2.1.3, and 3.2.2.2, respectively, require the heat flux hot channel factor F_q (x,y,z) and the enthalpy rise hot channel factor $F_{\Delta h}$ (x,y) to be measured periodically using the incore detector system to ensure that the values of the total peaking factor and the enthalpy rise factor assumed in the accident analyses and the reactor protection system limits are not violated. To avoid the possibility that these hot channel factors may increase beyond their allowable limits between surveillances, these SRs currently specify a penalty factor of 1.02 for the heat flux and enthalpy rise hot channel factors if the margin to the F_q (x,y,z) or $F_{\Delta h}$ (x,y) has decreased since the previous surveillance. For the reactor core containing the RFA fuel design with integral burnable absorbers, a larger penalty may be required over certain burnup ranges early in the cycle due to the rate of burnout of this poison. Section 8.1 proposes to remove the 2 percent penalty value from these surveillance requirements and replace them with tables of penalty values as functions of burnup in the Core Operating Limits Report (COLR) to facilitate cycle-specific updates. Tables 8-1 and 8-2, respectively, provide "typical values" for the burnup-dependent margin-decrease penalty factors for the heat flux and enthalpy rise hot channel factors.

(a) Provide the actual values of the margin-decrease penalty factors, as well as the bases, for these values.

(b) Provide references for the approved methodologies used to calculate these values, and to be included in TS 6.6.5 as a part of acceptability for COLR.

Duke Energy.

M. S. Tuckman
Executive Vice President
Nuclear Generation

Duke Energy Corporation
526 South Church Street
P.O. Box 1006 (EC07H)
Charlotte, NC 28201-1006
(704) 382-2200 OFFICE
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January 28, 1999

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

ATTENTION: Document Control Desk

Subject: Duke Energy Corporation

McGuire Nuclear Station Units 1 & 2
Docket Nos. 50-369, 50-370

Catawba Nuclear Station Units 1 & 2
Docket Nos. 50-413, 50-414

Response to NRC Requests for Additional Information on License Amendment Requests for McGuire and Catawba Nuclear Stations

This submittal contains information that Duke Energy Corporation considers PROPRIETARY and is being made pursuant to 10CFR 2.790.

By letters dated December 9, 1998 and January 5, 1999 the NRC requested additional information on Duke Energy Corporation's July 22, 1998 license amendment requests (LARs) for the McGuire Nuclear Station, Units 1 & 2; and the Catawba Nuclear Station, Units 1 & 2 Technical Specifications. These LARs would permit use of Westinghouse fuel at McGuire and Catawba. Topical Report DPC-NE-2000P/DPC-NE-2009 was also included in the July 22, 1998 Duke submittal.

The thirteen questions contained in the December 9, 1998 NRC letter, and the corresponding Duke answers, are provided in the attachments to this letter. A proprietary version and a non-proprietary version of the Duke response are attached to this letter.

Some of the information contained in Attachment 1 is considered proprietary. In accordance with 10CFR 2.790, Duke Energy Corporation requests that this information be withheld from public disclosure. An affidavit which attests to the

proprietary nature of the affected information is included with this letter. A non-proprietary version of the Duke response is included as Attachment 2 to this letter.

Please address any comments or questions regarding this matter to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc (w/o Attachment 1):

Mr. L. A. Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

Mr. F. Rinaldi, Senior Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-14H25
Washington, D. C. 20555-0001

Mr. P. S. Tam, Senior Project Manager
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Mr. S. M. Shaeffer
NRC Senior Resident Inspector
McGuire Nuclear Station

Mr. D. J. Roberts
NRC Senior Resident Inspector
Catawba Nuclear Station

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant ~~licensing;~~ and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

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

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, 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 and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. 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.

M. S. Tucker
M. S. Tuckman

(Continued)

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, 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 this submittal is that which is marked in the proprietary version of the Duke response to NRC requests for additional information dated December 9, 1998 and January 5, 1999. The subject of these requests for additional information is a Duke license amendment request dated July 22, 1998 and accompanying topical report designated DPC-NE-2009P, *Duke Power Company Westinghouse Fuel Transition Report*. The information of concern is omitted from the non-proprietary version of the Duke response. This information enables Duke to:

(a) Respond to Generic Letter 83-11, *Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions*.

(b) Perform core design, fuel rod design, and thermal-hydraulic analyses for the Westinghouse Robust Fuel Assembly design.

(c) Simulate UFSAR Chapter 15 transients and accidents for McGuire and Catawba Nuclear Stations.

(d) Perform safety evaluations per 10CFR50.59.

(e) Support Facility Operating Licenses/Technical Specifications amendments for McGuire and Catawba Nuclear Stations.

M. S. Tuckman

M. S. Tuckman

(Continued)

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

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

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

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

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

M. S. Tuckman

M. S. Tuckman

(Continued)

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 28TH day of

JANUARY, 1999

Mary P. Nelms

Notary Public

My Commission Expires:

JAN 22, 2001

SEAL

bxc (w/o Attachment 1):

L. A. Keller
M. T. Cash
G. D. Gilbert
K. L. Crane
K. E. Nicholson
R. H. Clark
G. B. Swindlehurst
D. E. Bortz
Catawba Owners: NCMPA-1, NCEMC, PMPA, SREC
Catawba Document Control File (T. K. Pasour)
Catawba RGC File 801.01 (T. K. Pasour)
ELL

Attachment 1

Response to NRC Requests for Additional Information Dated December 9, 1998 and January 5, 1999 Applicable to Duke Energy Carolinas License Amendment Requests Dated July 22, 1998

Note: Only the responses relating to DPC-NE-2005 are provided on the following pages, which pertain to questions 4, 5, 6, 7, and part of 11 of the NRC RAI. References to Sections in DPC-NE-2009 have been edited to reference the appropriate sections in DPC-NE-2005.

4. **Section 5.2 (*Duke edit: Section 5.2 is the Thermal-Hydraulic Code and Model section of Appendix G to DPC-NE-2005*) states that in using the VIPRE-01 code for the reactor core thermal-hydraulic analysis, the reference power distribution based on a 1.60 peak pin from DPC-NE-2004P-A, Rev.1, was used.**

   a. **The report states that this reference pin power distribution "was" used. Will it be used for future RFA reload analyses?**

   b. **Does the reference pin power distribution used in the core thermal-hydraulic analyses bound all power distribution for the RFA cores for future reload cycles?**

Response 4a:

The reference power distribution given in DPC-NE-2004P-A, Rev. 1 will be used in all future RFA analyses. This radial pin power distribution (the relationship of the peak pin to the remaining fuel pins in the highest power fuel assembly) used in DPC-NE-2009 and previous topical reports will not be modified. This maintains the relative radial power distribution the same as previously approved. There are no plans to change this distribution.

The peak pin value, however, could be increased in the future to utilize the increased thermal performance available in the RFA design. For DNB analyses using the Maximum Allowable Peaking (MAP) methodology described in DPC-NE-2004, Rev. 1, the key DNB parameter is the reference power distribution, not the peak pin power. The peak pin power is only meaningful when all other DNB parameters are specified (axial peak location and magnitude, core power level, RCS pressure, flow rate, and temperature). The reference power distribution is used to create the Maximum Allowable Peaking (MAP) limits that ensure the required level of DNBR protection is provided. The MAP limits define the maximum allowable peak pin as a function of axial peak. The reference power distribution is used consistently in all DNB analyses (core DNB limit lines, transient analyses, SCD statepoint determinations, etc.). Any change in the peak pin value will be evaluated in all DNB analyses and will be reflected in the Maximum Allowable Peaking limits provided in the COLR for each reload cycle.

The ability to increase the peak pin value is a result of a new fuel design, additional design features, a new or modified CHF correlation, or changes to the analysis conditions. If the performance improvement is related to fuel hardware of correlation change, a submittal is made to the NRC and approval require prior to use. If the change is to the analysis conditions and no methodology is modified, the change can be implemented through the 10CFR50.59 process. In either case, any increase in the peak pin value is not made unless all analyses and related licensing limits are verified to be conservatively satisfied.

Response 4b:

The reference power distribution used to create the Maximum Allowable Peaking (MAP) limits is used in all steady state generic analyses. This distribution is verified each reload by performing DNB calculations with cycle specific predicted radial pin power distributions. This specific pin distribution comparison between what is predicted for a particular cycle and the generic analysis reference power distribution verifies the conservatism of the reference distribution.

5. **Section 5.2 (*Duke edit: Section 5.2 is the Thermal-Hydraulic Code and Model section of Appendix G to DPC-NE-2005*) states that in the thermal-hydraulic analysis of the RFA design using VIPRE-01, the two-phase flow correlations will be changed from the Levy subcooled void correlation and the Zuber-Findlay bulk void correlation to the EPRI subcooled and bulk void correlations, respectively. While the sensitivity study provided in the report shows a minimal difference of 0.1% between the minimum DNBRs of 51 RFA CHF test data points calculated with both set of correlations, it was stated in DPC-NE-2004 that the Levy/Zuber-Findlay combination compared most favorably with the Mark-BW test results as the DNBRs of the tests calculated with this combination yielded conservative results relative to the EPRI correlations.**

   a. **Discuss whether the EPRI correlations will be used for the RFA design only, or they will also be used for the Mark-BW design.**

   b. **If the EPRI correlations will also be used for Mark-BW design, provide justification for their use.**

   c. **If the Levy/Zuber-Findlay correlations will continue to be used for Mark-BW fuel design, discuss how the VIPRE-01 code will be used to analyze transient mixed cores having both Mark-BW and RFA fuel designs.**

Response 5a:

The EPRI correlations will only be used in the RFA models in VIPRE-01. The Levy/ Zuber-Findlay combination will be used when modeling Mark-BW fuel.

Duke considers the selection of the two-phase flow correlations to be a very minor effect on DNBR analyses. Mark-BW CHF test data was analyzed with both Levy/Zuber-Findlay and EPRI/EPRI in the same manner as the RFA with comparable results.

Response 5b:

See 5(a) above.

Response 5c:

The transition core models use the simplified (8 Channel) models to maximize the impact of different fuel types. In the transition core model, the limiting assembly is modeled as an RFA and the rest of the core is modeled as Mark-BW fuel. Since the MDNBR occurs in the limiting assembly, the void correlations are input for the fuel type modeled as the limiting assembly. For the RFA/Mark-BW transition core model, the RFA design is the limiting assembly; thus the EPRI set of correlations are used.

The transition analyses covered a wide range of statepoint fluid conditions and 3-dimensional core power distributions. This matrix of conditions were analyzed using both the EPRI and Levy/Zuber-Findlay correlations with minimal difference in transition core results using either set of void correlations (average difference of < 1% in peaking).

6. **Section 5.7 (*Duke edit: Section 5.7 is the* <u>*Transition Cores*</u> *section of Appendix G to DPC-NE-2005*) describes the use of a transition 8-channel RFA/Mark-BW core model to determine the impact of the geometric and hydraulic differences between the resident Mark-BW fuel and the RFA design, and determine a conservative DNBR penalty to be applied for the transition cores. Table 5-4 (*Duke edit: Table G-4*) presented the statistical DNBRs for the 500 and 5000 case runs for various statepoints including the transition core case of the most limiting statepoint 12. The statistical design limit is chosen to bound both the full RFA cores and RFA/Mark-BW transition cores for the 5000 case runs.**

   a. **Why is the statistical design limit value proprietary information?**

   b. **With respect to the statistical core design methodology, describe how the uncertainties of the CHF correlation and the VIPRE code/model are propagated with the uncertainties of the selected parameters of each statepoint for the calculation of the statistical DNBR for each statepoint in Table 5-4.**

   c. **With the statistical design limit specified in Section 5.7, is it your intention to use a full core of RFA in the thermal hydraulic analysis for the transition core without the transition core DNBR penalty factor?**

Response 6a:

The Statistical Design Limit (SDL) will be changed to non-proprietary. This change will be included when the approved versions of the report are issued.

Response 6b:

When a statepoint is selected all key parameters, including CHF correlation and code/model uncertainties, are randomly varied based on the uncertainty distribution and magnitude. The resulting values of power, pressure, temperature, flow, and 3-D power distribution are used to create the VIPRE-01 input for the cases. After the code is executed and the DNBR calculated for each case, the DNBR value is multiplied by the propagated values for the CHF correlation uncertainty and the VIPRE code/model uncertainty. This final DNBR value for each case (500 or 5000 cases are run for each statepoint) is used to determine the statepoint's statistical DNBR value.

Response 6c:

The analysis discussed in the last paragraph of Section 5.7 verified that the statistical DNB limit developed with a full core RFA model is valid for transition RFA/Mark-BW cores. The limiting statepoint (12TR) was evaluated using the RFA/Mark-BW transition core model, confirming that the same SDL can be used for transition and full core analyses.

The transition core DNB penalty factor is determined separately using the RFA/Mark-BW transition core model described in Section 5.7. The DNB penalty is determined by evaluating the effect of the transition core hydraulic behavior on the Maximum Allowable Peaking (MAP) limits calculated for a full RFA core. The resulting DNB penalty is then accounted for in all RFA/Mark-BW transition core DNB analyses.

7. **Section 2.0 states that the RFA is designed to be mechanically and hydraulically compatible with the Mark-BW fuel. Table 2.1 provides a comparison of the basic design parameters of the two fuel designs, but does not provide a comparison of the hydraulic characteristics of spacer grids. Section 5.2 (*Duke edit: Section 5.2 is the <u>Thermal-Hydraulic Code and Model</u> section of Appendix G to DPC-NE-2005*) states that the VIPRE-01 core thermal-hydraulic analyses were performed with applicable form loss coefficients as per the vendor. Table 5.1 (*Duke edit: Table G-1*) provides general RFA fuel specifications and characteristics without the hydraulic characteristics of the spacer grids.**

   a. **Provide comparisons for the thickness, height, and form loss coefficients of the RFA and Mark-BW fuel spacer grids, including mixing-vane and non-mixing vane structural grids, and intermediate flow mixing grids.**

   b. **Provide the form loss coefficients of the spacer grids used in the analyses and in the RFA CHF test assemblies if they are different from the values described in item (a) above.**

   c. **Describe the procedures to ensure that the form loss coefficients of the RFA grids are comparable to those used in the SCD analysis and the CHF tests so that both the WRB-2M CHF correlation DNBR limit and the statistical core design limit are valid.**

Response 7a:

The grid data is shown in the following table:

[

]D

Response 7b:

The RFA CHF tests used Mixing Vane (MV) and intermediate flow mixing (IFM) grids representative of the production RFA design fuel assembly. The CHF test sections are a 5x5 rod bundle with either all typical (unit) cells or typical cells with a thimble (guide tube) cell in the center. The form loss coefficients for the CHF test section are calculated for these subchannels and are based on the total 5x5 bundle flow area. Likewise, the fuel assembly subchannel form loss coefficients are calculated based on the fuel assembly flow area. The ratio of thimble/typical cell form loss coefficients, to which DNBR is sensitive, is equivalent for the CHF test section and the production grid (for both MV and IFM grids). Therefore, the CHF test section and production RFA grids are identical with respect to DNBR analyses.

In comparing the test versus production geometry, the vanes and strap features of the respective grid types are consistent. There is one slight difference between one of the CHF test sections and the production fuel assemblies. The thimble OD was 0.474 inches for the thimble CHF rod bundle section tested. The production assembly will have thimbles with an OD of 0.482 inches. The difference in thimble tube OD has negligible impact on the correlation's predictive capability. This difference was addressed in WCAP-15025 and determined to be acceptable.

Response 7c:

The RFA analysis was completed with the form loss coefficients supplied in response to Question 7a. The transition core analysis used the RFA and Mark-BW values listed in the table in the respective model locations to accurately capture the hydraulic differences between the fuel types side-by-side incore.

For each batch of fuel manufactured, critical RFA grid dimensions and form loss coefficients are supplied by the vendor to Duke Power. This data, along with other critical reload analysis parameters, are transmitted to Duke, on a batch basis, in a QA document known as the Databook. Upon receipt of the Databook, the fuel design is frozen and may not be changed without Duke Power concurrence. This design notification process, including the process for changes occurring after the batch is frozen, is described in Duke Power Nuclear Engineering Workplace Procedure XSTP-101. The batch specific design information, transmitted in the Databook, will be used to ensure the validity of the Duke VIPRE-01 RFA models and associated SCD limit.

Any changes in the design data will be evaluated to verify that the generic analyses remain valid or the analyses will be revised using the new design data.

11. **The licensing analyses of reload cores with the RFA design will use the methodologies described in various topical reports and revisions for the analyses of fuel design, core reload design, physics, thermal-hydraulics, and transients and accidents, which were approved by NRC for analyses of current Catawba cores not having the RFA deign. For example, DPC-NE-1004A, DPC-NE-2011-PA, DPC-NF-2010A and DPE-NE-3001-PA are used for the nuclear design calculations. DPC-NE-2004-PA, DPC-NE-2005-PA, and the VIPRE-01 code are used for the core thermal-hydraulic analyses and statistical core design. DPC-NE-3000-PA, DPC-NE-3001-PA, DPC-NE-3002-A, and RETRAN-02 code are used for non-LOCA transient and accident analyses. Westinghouse small- and large-break LOCA evaluation model described in WCAP-10054-P-A and WCAP-10266-P-A, and related topical reports, are used for the small- and large-break LOCA analyses. Some of these methodologies have inherent limitations, and some have conditions or limitations imposed by the NRC safety evaluation reports in their applications. Provide a list of the inherent limitations, conditions, or restrictions applicable to the RFA core design from all the methodologies to be used for the RFA reload design analyses, and describe the resolutions of these limitations, conditions and restrictions in the applications to the RFA cores and the transitional RFA/Mark-BW cores.**

Response:

DPC-NE-2005P-A, Thermal-Hydraulic Statistical Core Desgin Methodology", Revision 1, November 1996.

The limitations, conditions or restrictions identified in the SER and TER for DPC-NE-2005P-A, are:

a. The statistical core design (SCD) methodology developed by DPC, as described in the submittal (DPC-NE-2005), 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 committed in their topical report that its use of specific uncertainties and distributions will be justified on a plant specific basis, and also that its selection of statepoints used for generating the statistical design limit will be justified to be appropriate. The methodology is approved only for use in DPC plants.

   Resolution: Addressed in Chapter 5 of DPC-NE-2009. The RFA analysis presented in DPC-NE-2009 is only for McGuire and Catawba. (*edit for inclusion to DPC-NE-2005: Chapter 5 of DPC-NE-2009 is now Appendix G of DPC-NE-2005*)

b. Of the two DNBR limits, only the use of the single, most-conservative DNBR limit is approved.

   Resolution: Use of two DNBR limits was not requested in this submittal. The single DNBR limit stated for the RFA design in full cores or transition cores will be used for all statepoints within the conditions listed in Table 5-5 (*edit for inclusion to DPC-NE-2005: Table G-5*).

Revision 1 of Appendix G
RAI Letters and Responses

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

July 26, 2002

Mr. M. S. Tuckman
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church St
Charlotte, NC 28202

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF DUKE TOPICAL REPORT DPC-NE-2009, REVISION 2 (TAC NOS. MB4502, MB4503, MB4504 AND MB4505)

Dear Mr. Tuckman:

The Nuclear Regulatory Commission is reviewing your application dated February 28, 2002, entitled "Topical Report DPC-NE-2009, Revision 2 - Updates to Chapters 2, 4, and 5" and has identified a need for additional information as identified in the Enclosure. These issues were discussed with your staff on July 24, 2002. Please provide a response to this request within 45 days of receipt of this letter so that we may complete our review.

Sincerely,

Robert Martin

Robert E. Martin, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

Enclosure: Request for Additional Information

cc w/encl: See next page

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST APPLICABLE TO

REVISIONS TO TOPICAL REPORT DPC-NE-2009, REVISION 2

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DUKE ENERGY CORPORATION

The staff has reviewed Duke Energy Corporations's submittal dated Febuary 28, 2002, "Topical Report DPC-NE-2009, Revision 2 - Updates to Chapters 2, 4, and 5" and has identified a need for the following additional information.

1. Section 5.3 of DPC-NE-2009, Revision 2, states that the WRB-2M critical heat flux (CHF) correlation will be used for the robust fuel assembly (RFA) design, whereas the BWU-N CHF correlation will be applied for the non-mixing vane span of the RFA fuel.

    A. Discuss the applicability of the BWU-N correlation to the RFA non-mixing vane span. The discussion should include whether the RFA fuel design is within the range of the test assemblies data base used to develop the BWU-N correlation. The test assemblies data base parameters include the fuel diameter, pitch, hydraulic diameters, grid design (grid thickness, height, and vane design), grid spacing, and heated length.

    B. The WRB-2M correlation described in WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17 x 17 Rod Bundles with Modified LPD Mixing Vane Grids," is applicable to the 17 x 17 fuel with 0.374 inch outer diameter rods and modified low pressure drop grids, with or without modified intermediate flow mixing grids. Is the WRB-2M correlation not applicable to the RFA non-mixing vane span? Why is the BWU-N correlation used?

    C. Discuss how two different correlations are applied to the different spans of a fuel assembly. Is the VIPRE-01 code programmed to automatically perform the switch in the correlations? Has verification and validation been done to ensure correctness of VIPRE-01 in the correlation switch?

2. For the transition cores with co-existence of the RFA and Mark-BW fuel designs, Section 5.7 of Revision 2 of the report, states that a transition core departure from nucleate boiling ratio penalty for the RFA design is determined using the 8 channel RFA/Mark-BW transition core model for initial transition reload cycles, and using the 75 channel model for subsequent cycles where the RFA fuel composes greater than 80 percent of the assemblies in the core.

    Explain why it is necessary to use different core models depending on whether the RFA fuel composes greater than 80 percent of the assemblies.

Duke Energy

M. S. Tuckman
Executive Vice President
Nuclear Generation

Duke Energy Corporation
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September 9, 2002

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Duke Energy Corporation
McGuire Nuclear Station - Units 1 and 2
Docket Nos. 50-369 and 50-370
Catawba Nuclear Station - Units 1 and 2
Docket Nos. 50-413 and 50-414
Topical Report DPC-NE-2009, Revision 2 - Updates to Chapters 2, 4, and 5 (TAC Nos. MB4502, MB4503, MB4504, MB4505)

Response to NRC Request for Additional Information

By letter dated July 26, 2002, the NRC requested additional information regarding Topical Report DPC-NE-2009, Revision 2, "Updates to Chapters 2, 4, and 5." The questions contained in the July 26, 2002 NRC letter, and the corresponding Duke answers, are provided in the attachment to this letter.

If there are any questions or additional information is needed on this matter, please call A. Jones-Young at (704) 382-3154.

Very truly yours,

M S Tuckman

M.S. Tuckman

ATTACHMENT

xc:

L.A. Reyes, Regional Administrator
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D.J. Roberts, NRC Senior Resident Inspector (CNS)

S.M. Shaeffer, NRC Senior Resident Inspector (MNS)

bxc:

M. T. Cash
R. M. Gribble
K. R. Epperson
G. D. Gilbert
C. J. Thomas
L. E. Nicholson
A. D. Jones-Young
ELL

ATTACHMENT

(reference to Sections in DPC-NE-2009 have been edited to reference the appropriate sections in DPC-NE-2005)

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST APPLICABLE TO

REVISIONS TO TOPICAL REPORT DPC-NE-2009, REVISION 2

CATAWBA NUCKEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DUKE ENERGY CORPORATION

The staff has reviewed Duke Energy Corporation's submittal dated February 28, 2002, "Topical Report DPC-NE-2009, Revision 2 - Updates to Chapters 2, 4, and 5" and has identified a need for the following information.

1. Section 5.3 of DPC-NE-2009, Revision 2, (*Duke edit: Section 5.3 is now the* _*Critical Heat Flux*_ *section in Appendix G of DPC-NE-2005*) states that the WRB2-M critical heat flux (CHF) correlation will be used for the robust fuel assembly (RFA) design, whereas the BWU-N CHF correlation will be applied for the non-mixing vane span of the RFA fuel.

   A. Discuss the applicability of the BWU-N correlation to the RFA non-mixing vane span. The discussion should include whether the RFA fuel design is within the range of test assemblies data base used to develop the BWU-N correlation. The test assemblies data base parameters include the fuel diameter, pitch, hydraulic diameters, grid design (grid thickness, height, and vane design), grid spacing, and heated length.

   B. The WRB-2M correlation described in WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux In 17x17 Rod Bundles with Modified LPD Mixing Vane Grids," is applicable to the 17x17 fuel with 0.374 inch outer diameter rods and modified low pressure drop grids, with or without modified intermediate flow mixing grids. Is the WRB-2M correlation not applicable to the RFA non-mixing vane span? Why is the BWU-N correlation used?

C. Discuss how two different correlations are applied to the different spans of a fuel assembly. Is the VIPRE-01 code programmed to automatically perform the switch in the correlations? Has verification and validation been done to ensure correctness of the VIPRE-01 in the correlation switch?

2. For the transition cores with co-existence of the RFA and Mark-BW fuel designs, Section 5.7 of Revision 2 of the report (*Duke edit: Section 5.7 is now the* <u>*Transition Cores*</u> *section in Appendix G of DPC-NE-2005*), states that a transition core departure from nucleate boiling ratio penalty for the RFA design is determined using the 8 channel RFA/Mark-BW transition core model for the initial transition reload cycles, and using the 75 channel model for subsequent cycles where RFA fuel composes greater than 80 percent of the assemblies in the core.

   Explain why it is necessary to use different core models depending on whether the RFA fuel composes greater than 80 percent of the assemblies.

1. *Section 5.3 of DPC-NE-2009, Revision 2 (Duke edit: Section 5.3 is now the <u>Critical Heat Flux</u> section in Appendix G of DPC-NE-2005), states that the WRB2-M critical heat flux (CHF) correlation will be used for the robust fuel assembly (RFA) design, whereas the BWU-N CHF correlation will be applied for the non-mixing vane span of the RFA fuel.*
    A. *Discuss the applicability of the BWU-N correlation to the RFA non-mixing vane span. The discussion should include whether the RFA fuel design is within the range of test assemblies data base used to develop the BWU-N correlation. The test assemblies data base parameters include the fuel diameter, pitch, hydraulic diameters, grid design (grid thickness, height, and vane design), grid spacing, and heated length.*

The BWU-N correlation is based on local conditions (pressure, mass flux, local quality) that bound the operation of the RFA fuel at McGuire and Catawba. The following table compares the geometry parameters for the RFA design against the BWU-N correlation:

Parameter	RFA Fuel	BWU-N Database
Fuel Diameter	0.374	0.379 - 0.430
Rod Pitch	0.496	0.501 - 0.590
Hydraulic Diameter	0.375 - 0.464	0.39 - 0.60
*Grid Spacing (inches)	20.5	21.0
Heated Length (feet)	12	6 - 12

* - In the span of interest

BWU-N is one of a series of CHF correlations developed to apply to PWR cores with mixing or non-mixing vane spacer grids. In each of the approved correlations, the correlated independent variables were the thermal-hydraulic local conditions (pressure, mass velocity and equilibrium thermodynamic quality at CHF), axial flux shape (via the F factor), heated length, and the grid axial spacing. The geometric independent variables such as rod diameter, pitch to diameter ratio, hydraulic or heated diameters were found to be non-correlated (that is, there was no sensitivity in CHF level for geometric independent variables) and thus these parameters were not needed as part of the correlation.

Even though the geometric variables were found to be non-correlated, it would be improper to make large extrapolations of these geometric variables. Only very small extrapolations are necessary to apply BWU-N to RFA fuel. This is shown in the following table:

Geometric Variable	**RFA Application**	**BWU-N Data Base**	**Difference, %**
Pin Pitch, in.	0.496	0.501	1.0
Rod Diameter, in.	0.374	0.379	1.3
Pitch to Diameter Ratio	0.496/0.374 = 1.326	0.501/0.379 = 1.322	0.3
Unit Hydraulic Diameter, in.	0.4635	0.4642	0.2

The grid design is the same in that BWU-N is being applied to the RFA fuel only above a non-mixing vane grid. There are no vanes present on the grid in question. The grid heights and thickness are within 0.026 and 0.003 inches respectively. As explained above, these parameters have no significant impact on the CHF performance in a non-mixing vane span.

Table 4-3 of Reference 1 limits BWU-N to Non-Mixing Grids. Thus the use of BWU-N is based on:

1. the geometric similarity of the designs
2. the fact that the geometric variables are not included (needed) in the base BWU correlations and
3. the fact that BWU-N results in conservative levels of CHF compared to the mixing vane correlations.

In summary, CHF performance is influenced by the presence or absence of mixing vanes and the local conditions. There are no specific grid features to enhance thermal performance in the span of interest and the local conditions are bounded. Therefore, BWU-N can be applied to the non-mixing vane span of the RFA assembly and will predict lower CHF (conservative) than the mixing vane grid correlations.

B. The WRB-2M correlation described in WCAP-15025-P-A, "Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux In 17x17 Rod Bundles

with Modified LPD Mixing Vane Grids," is applicable to the 17x17 fuel with 0.374 inch outer diameter rods and modified low pressure drop grids, with or without modified intermediate flow mixing grids. Is the WRB-2M correlation not applicable to the RFA non-mixing vane span? Why is the BWU-N correlation used?

The WRB-2M correlation was developed from fuel with mixing vane modified LPD mid-grids, modified LDP IFM grids, and non-vaned end grids. All the CHF data from the test program documented in WCAP-15025-P-A was in a region above one of the mixing vane grid types. Therefore, the WRB-2M correlation is directly applicable to regions of the fuel above a modified LPD mixing vane grid of either type. The very bottom span of the RFA fuel assembly (lower ~21 inches of the heated length) is above an Inconel grid without any type of mixing vane. For this region of the fuel assembly, Duke considers the use of the BWU-N non-mixing vane grid correlation to be appropriate and conservative as discussed in the answer to question 1 (A).

C. *Discuss how two different correlations are applied to the different spans of a fuel assembly. Is the VIPRE-01 code programmed to automatically perform the switch in the correlations? Has verification and validation been done to ensure correctness of the VIPRE-01 in the correlation switch?*

The VIPRE-01 computer code solves the sets of equations for the geometry modeled and the boundary conditions specified to determine a converged fluid solution. This converged fluid solution yields the local conditions at each node and elevation modeled. After the fluid solution is converged, all the inputs for the CHF correlation (local pressure, mass flux, enthalpy, etc.) are fixed and the DNBR calculation is performed. Therefore, the calculation of CHF and DNBR has no effect on the converged fluid solution.

Due to this, VIPRE-01 has the built-in capability to calculate DNBR with multiple CHF correlations. Each correlation is applied to all channels at all elevations. Since the switch in this case is based solely on grid type and elevation, two options are available to apply the BWU-N correlation:

- manually overlay the output of the code after selecting both correlations
- program the code to automatically switch based on grid elevation inputs

For the current application of BWU-N on the RFA fuel, the manual process was used. The automatic switching from WRB-2M to BWU-N was not programmed into VIPRE-01. However, the VIPRE-01 code has been programmed by Duke to automatically perform the switch in other applications such as the Mark-BW (BWU-N to BWU-Z) and the Advanced Mark-BW (BWU-N to BWU-Z/MSM) and may be added to this application (RFA) in the future.

The verification and validation of the manual overlay process is performed by the independent review of the calculation results in the standard quality assurance process. The verification and validation of an automatic switchover by elevation is performed in the code revision process by performing independent calculations of the correct critical heat flux value from the local fluid conditions in the channel. This independent calculation by elevation of the critical heat flux is compared against the code output for cases to confirm the switch is being performed correctly.

2. *For the transition cores with co-existence of the RFA and Mark-BW fuel designs, Section 5.7 of Revision 2 of the report (Duke edit: Section 5.7 is now the Transition Cores section in Appendix G of DPC-NE-2005), states that a transition core departure from nucleate boiling ratio penalty for the RFA design is determined using the 8 channel RFA/Mark-BW transition core model for the initial transition reload cycles, and using the 75 channel model for subsequent cycles where RFA fuel composes greater than 80 percent of the assemblies in the core.*

   Explain why it is necessary to use different core models depending on whether the RFA fuel composes greater than 80 percent of the assemblies.

   The RFA fuel assembly contains 3 extra grids, the IFM grids, compared to the Mark-BW assembly. These extra grids in the upper span force flow out of the RFA assemblies and into the surrounding Mark-BW assemblies.

In the 8 channel model, the single hot assembly (RFA) is modeled by the first 7 channels and the remainder of the core (Mark-BW fuel) is lumped into one single channel. Therefore, in the 8 channel transition model, there is one RFA assembly surrounded by 192 Mark-BW assemblies. This maximizes the hydraulic difference in transition cores and creates a very bounding penalty for the RFAs.

The loss of flow in the upper spans of the RFA is the major element of the DNB penalty. This hydraulic effect of flow reduction in the RFA is a direct function of the number of RFA and Mark-BW assemblies incore. As subsequent cores of RFA fuel are loaded, only a few Mark-BW assemblies remain. As fewer Mark-BWs are present, the simple 8 channel model becomes overly conservative for the RFAs in transition. The only option to better reflect the physical effects of the last transition cycles is to increase the detail in VIPRE-01 to the 75 channel model. This more detailed model better represents the hydraulic effects of cores where most of the fuel is RFA where a small fraction (less than 20% or fewer than 38 assemblies) of the core is Mark-BW. The 80% value was selected because it corresponds to approximately two batches of RFA fuel residing incore. With this more detailed 75 channel model, a conservative penalty is still determined in the same manner as with the 8 channel model.

This approach of using a more detailed transition core model was discussed previously in Reference 2 [response to Question 2] for Mark-BW/OFA transition at McGuire/Catawba and Reference 3 [response to Question 2(d)] for the Mark-B11/Mark-B10 transition at Oconee.

References

1) BAW-10199P-A, August, 1996, "The BWU CHF Correlations", D. A. Farnsworth and G. A. Meyer.
2) Letter from M.S. Tuckman to USNRC, Supplemental Information to Assist in Review of Topical Reports DPC-NE-3000 and DPC-NE-2004, August 29, 1991 (included in Attachment D of DPC-NE-2004P-A, Revision 1)
3) Letter from M.S Tuckman to USNRC, Response to NRC Request for Additional Information on Appendix D to Topical Report DPC-NE-2005-P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, September 21, 1998 (included in Appendix D of DPC-NE-2005P-A, Revision 2)