Duke Power
A Duke Energy Company

M. S. Tuckman
Executive Vice President
Nuclear Generation

Duke Power
526 South Church St. EC07H
Charlotte, NC 28202
P. O. Box 1006 EC07H
Charlotte, NC 28201-1006
(704) 382-2200 OFFICE
(704) 382-4360 FAX

October 7, 2001

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Duke Energy Corporation

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

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

Pursuant to 10CFR50.90 and 10CFR2.790, attached is a Duke Energy Corporation (Duke) submittal package which contains a license amendment request (LAR) and four Duke topical reports for McGuire Nuclear Station (MNS) and Catawba Nuclear Station (CNS). This LAR applies to MNS Technical Specification (TS) 5.6.5 and CNS TS 5.6.5. These TS contain requirements for the Core Operating Limits Report (COLR). As detailed below, two of the attached topical reports, for which Duke is requesting NRC review and approval, contain information that Duke has determined to be proprietary.

Regarding TS 5.6.5.a, this LAR proposes additions to the list of other existing TS that refer to the COLR for the applicable operating limit. This is considered an editorial change since TS 5.6.5.a only contains references to other approved TS.

Regarding TS 5.6.5.b, this LAR changes the revision number and the NRC approval date for three Duke topical reports

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that are listed as reference documents. Specifically, these topical reports as listed are:

1) DPC-NE-2009-P-A, *Duke Power Company Westinghouse Fuel Transition Report*, which is being changed to Revision 1;

2) DPC-NF-2010-A, *Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design*, which is being changed to Revision 1; and

3) DPC-NE-2011-P-A, *Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*, which is being changed to Revision 1.

This submittal package also includes a revision to a fourth Duke topical report, which is not referenced in the McGuire or Catawba Technical Specifications, and is consequently not part of the LAR portion of this submittal package. This topical report is:

DPC-NE-1003, McGuire Nuclear Station and Catawba Nuclear Station *Rod Swap Methodology Report for Startup Physics Testing*, Revision 1.

Duke is also requesting that the NRC review and approve this topical report in addition to the three others listed above and the attached LAR. This submittal package also contains changes to the reference documents listed in Bases 3.2.1 and 3.2.3. These bases changes are consistent with the topical report revisions described above.

In addition, two other administrative changes are being made to TS 5.6.5.b. The SER date of November 15, 1991 is being specified for Topical Report DPC-NE-3001-P-A as listed in Item 5 of TS 5.6.5.b. Also, Topical Report DPC-NE-3002, Revision 4, was approved by the NRC in an SER dated April 6, 2001. Therefore, the revision number for this topical report is being changed in Item 7 of TS 5.6.5.b.

The contents of this LAR submittal package are as follows:

- An Affidavit for the LAR is provided within this cover letter.

- Attachments 1a and 1b provide a marked copy of the existing Technical Specifications and Bases for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively. These marked copies show the proposed changes.

- Attachments 2a and 2b provide the reprinted Technical Specifications and Bases pages for McGuire Units 1 and 2 and Catawba Units 1 and 2, respectively.

- Attachment 3 provides a Description of the Proposed Changes and Technical Justification.

- Pursuant to 10CFR50.92, Attachment 4 documents the determination that this LAR contains No Significant Hazards Consideration.

- Pursuant to 10CFR51.22(c)(9), Attachment 5 provides the basis for the categorical exclusion from performing an Environmental Assessment/Impact Statement.

- Attachment 6a provides a listing of changes to Topical Report DPC-NE-2009-P-A; Attachment 6b provides Topical Report DPC-NE-2009-P, Revision 1, showing the proposed changes and an affidavit that attests to the proprietary nature of this document; and Attachment 6c provides Topical Report DPC-NE-2009, Revision 1 (Non-Proprietary).

- Attachment 7a provides a detailed listing of changes to Topical Report DPC-NF-2010-A, and Attachment 7b provides Topical Report DPC-NF-2010, Revision 1.

- Attachment 8a provides a detailed listing of changes to Topical Report DPC-NE-2011-P-A, Attachment 8b provides Topical Report DPC-NE-2011-P, Revision 1, and an affidavit that attests to the proprietary nature of this document, and Attachment 8c provides Topical Report DPC-NE-2011, Revision 1 (Non-Proprietary).

- Attachment 9a provides a detailed listing of changes to Topical Report DPC-NE-1003-A, and Attachment 9b provides Topical Report DPC-NE-1003, Revision 1 which is not part of the LAR.

Implementation of this LAR in the Facility Operating Licenses and Technical Specifications will impact the McGuire and Catawba Updated Final Safety Analysis Reports (UFSAR). The necessary changes are discussed in Attachment 3 and these will be submitted in accordance with 10CFR50.71(e).

Duke is requesting NRC review and approval of this LAR and the enclosed four topical report revisions by October 7, 2002. It has been determined that the NRC's standard 30-day implementation period is acceptable for this LAR.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, the site-specific changes contained in this LAR have been reviewed and approved by the respective McGuire and Catawba Plant Operations Review Committee. This LAR has also been reviewed and approved on an overall basis by the Duke Nuclear Safety Review Board. Pursuant to 10CFR50.91, a copy of this LAR is being sent to the designated official of the State of North Carolina and the designated official of the State of South Carolina.

This submittal package contains information that Duke considers proprietary. This information is contained within the proprietary version of Topical Report DPC-NE-2009 (designated DPC-NE-2009-P) and Topical Report DPC-NE-2011 (designated DPC-NE-2011-P). These documents are provided respectively as Attachments 6b and 8b to this letter. In accordance with 10CFR2.790, Duke requests that this information be withheld from public disclosure. Affidavits that attest to the proprietary nature of this information are included within these attachments to this letter.

Inquiries on this matter should be directed to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc w/All Attachments:

C. P. Patel (Addressee Only)
NRC Senior Project Manager (CNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

R. E. Martin (Addressee Only)
NRC Senior Project Manager (MNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

xc w/Non-Proprietary Attachments Only:

L. A. Reyes
U. S. Nuclear Regulatory Commission
Regional Administrator, Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

D. J. Roberts
Senior Resident Inspector (CNS)
U. S. Nuclear Regulatory Commission
Catawba Nuclear Site

S. M. Shaeffer
Senior Resident Inspector (MNS)
U. S. Nuclear Regulatory Commission
McGuire Nuclear Site

M. Frye
Division of Radiation Protection
3825 Barrett Drive
Raleigh, NC 27609-7221

R. Wingard, Director
Division of Radioactive Waste Management
South Carolina Bureau of Land and Waste Management
2600 Bull Street
Columbia, SC 29201

AFFIDAVIT

M. S. Tuckman, being duly sworn, states that he is Executive Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission these amendments to the McGuire Nuclear Station Facility Operating Licenses Nos. NPF-9 and NPF-17 and the Catawba Nuclear Station Facility Operating Licenses Nos. NPF-35 and NPF-52 and associated Technical Specifications; and that all statements and matters set forth within this submittal are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to me: 10-7-01
Date

Mary P. Nehrs, Notary Public

My commission expires: JAN 22, 2001

SEAL

bxc w/o Attachments:

C. J. Thomas
M. R. Wilder
G. D. Gilbert
L. E. Nicholson
K. L. Crane
K. E. Nicholson
T. K. Pasour (2)
L. J. Rudy
N. T. Simms
R. M. Gribble
D. R. Koontz
G. A. Copp
R. L. Gill
MNS Master File - MG01DM
Catawba Master File - CN04DM
NRIA/ELL

Catawba Owners:
Saluda River Electric Corporation
P. O. Box 929
Laurens, SC 29360-0929

NC Municipal Power Agency No. 1
P. O. Box 29513
Raleigh, NC 27626-0513

T. R. Puryear
NC Electric Membership Corporation
CN03G

Piedmont Municipal Power Agency
121 Village Drive
Greer, SC 29651

bxc w/All Attachments:

P. M. Abraham
G. G. Pihl

Attachment 1a

McGuire Units 1 and 2 Technical Specifications

Marked Copy

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of the analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

---NOTE---
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit in the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in Chapter 16 of the UFSAR and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following: and 60 ppm

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3.1.3,
S

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor limits for Specification 3.2.2,

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

8. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

9. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

10. Spent fuel pool boron concentration limits for Specification 3.7.14,

11. SHUTDOWN MARGIN for Specification 3.1.1~~.~~, and

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

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

2. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).

3. BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Rev. 1, SER dated January 22, 1991; Rev. 2, SERs dated August 22, 1996 and November 26, 1996; Rev. 3, SER dated June 15, 1994 (B&W Proprietary).

12. 31 EFPD Surveillance Penalty Factors for Specifications 3.2.1 and 3.2.2.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

, Rev. 1

4. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," ~~March, 1990~~ (DPC Proprietary).

SER dated ______

SER dated

5. DPC-NE-3001PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November, 1991 (DPC Proprietary). 15

, Rev. 1

6. DPC-NF-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," ~~June, 1985~~

SER dated ______

4

7. DPC-NE-3002A, Rev. ~~3~~ "FSAR Chapter 15 System Transient Analysis Methodology," SER dated ~~February 5, 1999~~ April 6, 2001

8. DPC-NE-3000PA, Rev. 2 "Thermal-Hydraulic Transient Analysis Methodology," SER dated October 14, 1998. (DPC Proprietary).

9. DPC-NE-1004A, Rev. 1, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," SER dated April 26, 1996.

10. DPC-NE-2004P-A, Rev. 1, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," SER dated February 20, 1997 (DPC Proprietary).

11. DPC-NE-2005P-A, Rev. 1, "Thermal Hydraulic Statistical Core Design Methodology," SER dated November 7, 1996 (DPC Proprietary).

12. DPC-NE-2008P-A, "Fuel Mechanical Reload Analysis Methodology Using TACO3," SER dated April 3, 1995 (DPC Proprietary).

13. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code," August 1985 (W Proprietary).

, Rev. 1

14. DPC-NE-2009-P-A, "Westinghouse Fuel Transition Report, "SER dated ~~September 22, 1999~~ (DPC Proprietary).

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15. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2-5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998, (W Proprietary).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA, Rev. 1 "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", ~~March 1990~~.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. DPC-NE-2011PA, Rev. 1 "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", ~~March 1990.~~

2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).

3. UFSAR, Chapter 7.

Attachment 1b

Catawba Units 1 and 2 Technical Specifications

Marked Copy

5.6 Reporting Requirements

and 60 ppm

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. Moderator Temperature Coefficient BOL and EOL limits and 300 ppm surveillance limit for Specification 3.1.3,

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2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3.2.2,

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

8. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

9. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

10. Spent fuel pool boron concentration limits for Specification 3.7.15,

11. SHUTDOWN MARGIN for Specification 3.1.~~1.~~ ,

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

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

2. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).

12. 31 EFPD Surveillance Penalty Factors for Specifications 3.2.1 and 3.2.2, and

13. Reactor Makeup Water Pumps Combined Flow Rates limit for Specifications 3.3.9 and 3.9.2.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Rev. 1, SER dated January 22, 1991; Rev. 2, SERs Dated August 22, 1996 and November 26, 1996; Rev. 3, SER Dated June 15, 1994 (B&W Proprietary).

4. DPC-NE-2011P-A, Rev. 1, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," SER dated ~~March 1990~~ (DPC Proprietary).

5. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," SER dated November 15, 1991 (DPC Proprietary).

6. DPC-NF-2010A, Rev. 1, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," SER dated ~~June 1985~~.

7. DPC-NE-3002-A, Rev. ~~3~~ 4 "FSAR Chapter 15 System Transient Analysis Methodology," SER dated ~~February 5, 1999~~ April 6, 2001

8. DPC-NE-3000PA, Rev. 2 "Thermal-Hydraulic Transient Analysis Methodology," SER Dated October 14, 1998 (DPC Proprietary).

9. DPC-NE-1004A, Rev. 1, "Design Methodology Using CASMO-3/SIMULATE-3P," SER Dated April 26, 1996.

10. DPC-NE-2004P-A, Rev. 1, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," SER dated February 20, 1997 (DPC Proprietary).

11. DPC-NE-2005P-A, Rev. 1, "Thermal Hydraulic Statistical Core Design Methodology," SER dated November 7, 1996 (DPC Proprietary).

12. DPC-NE-2008P-A, "Fuel Mechanical Reload Analysis Methodology Using TACO3," SER dated April 3, 1995 (DPC Proprietary).

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

13. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

14. DPC-NE-2009P-A, Rev. 1, "Westinghouse Fuel Transition Report," SER dated September 22, 1999 (DPC Proprietary).

15. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2-5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998, (W Proprietary).

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

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

5.6.6 Ventilation Systems Heater Report

When a report is required by LCO 3.6.10, "Annulus Ventilation System (AVS)," LCO 3.7.10, "Control Room Area Ventilation System (CRAVS)," LCO 3.7.12, Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)," LCO 3.7.13, "Fuel Handling Ventilation Exhaust System (FHVES)," or LCO 3.9.3, "Containment Penetrations," a report shall be submitted within the following 30 days. The report shall outline the reason for the inoperability and the planned actions to return the systems to OPERABLE status.

5.6.7 PAM Report

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

5.6.8 Steam Generator Tube Inspection Report

a. The number of tubes plugged in each steam generator shall be reported to the NRC within 15 days following completion of the program;

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

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3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
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BASES

SURVEILLANCE REQUIREMENTS (continued)

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REFERENCES

1. DPC-NE-2011PA, Rev. 1 "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", ~~March 1990~~.
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Chapter 7.

Attachment 2a

McGuire Units 1 and 2 Technical Specifications

Reprinted Pages

Remove	Insert
5.6-2	5.6-2
5.6-3	5.6-3
5.6-4	5.6-4
B 3.2.1-11	B 3.2.1-11
B.3.2.3-4	B 3.2.3-4

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

The Annual Radiological Environmental Operating Report shall include summarized and tabulated results of the analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

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5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

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

   1. Moderator Temperature Coefficient BOL and EOL limits and 60 ppm and 300 ppm surveillance limits for Specification 3.1.3,

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor limits for Specification 3.2.2,

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

8. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

9. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

10. Spent fuel pool boron concentration limits for Specification 3.7.14,

11. SHUTDOWN MARGIN for Specification 3.1.1, and

12. 31 EFPD Surveillance Penalty Factors for Specifications 3.2.1 and 3.2.2.

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

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

2. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).

3. BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Rev. 1, SER dated January 22, 1991; Rev. 2, SERs dated August 22, 1996; and November 26, 1996; Rev. 3, SER dated June 15, 1994 (B&W Proprietary).

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. DPC-NE-2011PA, Rev.1, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," SER dated________ (DPC Proprietary).

5. DPC-NE-3001PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," SER dated November 15, 1991 (DPC Proprietary).

6. DPC-NF-2010A, Rev 1, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," SER dated ______.

7. DPC-NE-3002A, Rev. 4, "FSAR Chapter 15 System Transient Analysis Methodology," SER dated April 6, 2001.

8. DPC-NE-3000PA, Rev. 2 "Thermal-Hydraulic Transient Analysis Methodology," SER dated October 14, 1998. (DPC Proprietary).

9. DPC-NE-1004A, Rev. 1, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," SER dated April 26, 1996.

10. DPC-NE-2004P-A, Rev. 1, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," SER dated February 20, 1997 (DPC Proprietary).

11. DPC-NE-2005P-A, Rev. 1, "Thermal Hydraulic Statistical Core Design Methodology," SER dated November 7, 1996 (DPC Proprietary).

12. DPC-NE-2008P-A, "Fuel Mechanical Reload Analysis Methodology Using TACO3," SER dated April 3, 1995 (DPC Proprietary).

13. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code, " August 1985 (W Proprietary).

14. DPC-NE-2009-P-A, Rev. 1, "Westinghouse Fuel Transition Report, " SER dated ________, (DPC Proprietary).

15. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2-5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998, (W Proprietary).

(continued)

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA , Rev. 1 "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. DPC-NE-2011PA, Rev. 1, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Chapter 7.

Attachment 2b

Catawba Units 1 and 2 Technical Specifications

Reprinted Pages

Remove	Insert
5.6-3	5.6-3
5.6-4	5.6-4
5.6-5	5.6-5
B 3.2.1-11	B 3.2.1-11
B.3.2.3-4	B 3.2.3-4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. Moderator Temperature Coefficient BOL and EOL limits and 60 ppm and 300 ppm surveillance limits for Specification 3.1.3,

2. Shutdown Bank Insertion Limit for Specification 3.1.5,

3. Control Bank Insertion Limits for Specification 3.1.6,

4. Axial Flux Difference limits for Specification 3.2.3,

5. Heat Flux Hot Channel Factor for Specification 3.2.1,

6. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3.2.2,

7. Overtemperature and Overpower Delta T setpoint parameter values for Specification 3.3.1,

8. Accumulator and Refueling Water Storage Tank boron concentration limits for Specification 3.5.1 and 3.5.4,

9. Reactor Coolant System and refueling canal boron concentration limits for Specification 3.9.1,

10. Spent fuel pool boron concentration limits for Specification 3.7.15,

11. SHUTDOWN MARGIN for Specification 3.1.1,

12. 31 EFPD Surveillance Penalty Factors for Specifications 3.2.1 and 3.2.2, and

13. Reactor Makeup Water Pumps Combined Flow Rates limit for Specifications 3.3.9 and 3.9.2.

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

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

2. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Rev. 1, SER dated January 22, 1991; Rev. 2, SERs Dated August 22, 1996 and November 26, 1996; Rev. 3, SER Dated June 15, 1994 (B&W Proprietary).

4. DPC-NE-2011P-A, Rev. 1, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," SER dated __________(DPC Proprietary).

5. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," SER dated November 15, 1991 (DPC Proprietary).

6. DPC-NF-2010A, Rev. 1, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design", SER dated __________.

7. DPC-NE-3002A, Rev. 4 "FSAR Chapter 15 System Transient Analysis Methodology," SER dated April 6, 2001.

8. DPC-NE-3000PA, Rev. 2 "Thermal-Hydraulic Transient Analysis Methodology," SER Dated October 14, 1998 (DPC Proprietary).

9. DPC-NE-1004A, Rev. 1, "Design Methodology Using CASMO-3/SIMULATE-3P," SER Dated April 26, 1996.

10. DPC-NE-2004P-A, Rev. 1, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," SER dated February 20, 1997 (DPC Proprietary).

11. DPC-NE-2005P-A, Rev. 1, "Thermal Hydraulic Statistical Core Design Methodology," SER dated November 7, 1996 (DPC Proprietary).

12. DPC-NE-2008P-A, "Fuel Mechanical Reload Analysis Methodology Using TACO3," SER dated April 3, 1995 (DPC Proprietary).

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

13. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

14. DPC-NE-2009P-A, Rev. 1,"Westinghouse Fuel Transition Report," SER dated ___________ (DPC Proprietary).

15. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2-5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998, (W Proprietary).

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

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

5.6.6 Ventilation Systems Heater Report

When a report is required by LCO 3.6.10, "Annulus Ventilation System (AVS)," LCO 3.7.10, "Control Room Area Ventilation System (CRAVS)," LCO 3.7.12, Auxiliary Building Filtered Ventilation Exhaust System (ABFVES)," LCO 3.7.13, "Fuel Handling Ventilation Exhaust System (FHVES)," or LCO 3.9.3, "Containment Penetrations," a report shall be submitted within the following 30 days. The report shall outline the reason for the inoperability and the planned actions to return the systems to OPERABLE status.

5.6.7 PAM Report

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

5.6.8 Steam Generator Tube Inspection Report

a. The number of tubes plugged in each steam generator shall be reported to the NRC within 15 days following completion of the program;

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F_Q(X,Y,Z)$ limit with the last $F^M_Q(X,Y,Z)$ increased by the appropriate factor specified in the COLR or to evaluate $F_Q(X,Y,Z)$ prior to the projected point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F_Q(X,Y,Z)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_Q(X,Y,Z)$ is not required to be extrapolated for the initial flux map taken after reaching equilibrium conditions since the initial flux map establishes the baseline measurement for future trending. Also, extrapolation of $F^M_Q(X,Y,Z)$ limits are not valid for core locations that were previously rodded, or for core locations that were previously within ±2% of the core height about the demand position of the rod tip.

$F_Q(X,Y,Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(X,Y,Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(X,Y,Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46.
2. UFSAR Section 15.4.8.
3. 10 CFR 50, Appendix A, GDC 26.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2011PA, Rev. 1, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".

BASES

SURVEILLANCE REQUIREMENTS (continued)

This Surveillance verifies that the AFD, as indicated by the NIS excore channel, is within its specified limits and is consistent with the status of the AFD monitor alarm. With the AFD monitor alarm inoperable, the AFD is monitored every hour to detect operation outside its limit. The Frequency of 1 hour is based on operating experience regarding the amount of time required to vary the AFD, and the fact that the AFD is closely monitored. With the AFD monitor alarm OPERABLE, the Surveillance Frequency of 7 days is adequate considering that the AFD is monitored by a computer and any deviation from requirements is alarmed.

REFERENCES

1. DPC-NE-2011PA, Rev. 1, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors".
2. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
3. UFSAR, Chapter 7.

Attachment 3

Description of Proposed Changes and Technical Justification

Discussion

The changes proposed in this license amendment request (LAR) apply to Technical Specification (TS) 5.6.5, Core Operating Limits Report (COLR); Bases 3.2.1, Heat Flux Hot Channel Factor ($F_Q(X,Y,Z)$; and Bases 3.2.3, Axial Flux Difference for McGuire Nuclear Station (MNS) and Catawba Nuclear Station (CNS). The proposed changes are discussed below.

MNS - Proposed Changes to TS 5.6.5.a

This TS requires that core operating limits be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and lists various TS requirements which shall be documented in the COLR. MNS is proposing to include other existing TS in the list contained in TS 5.6.5.a.

The moderator temperature coefficient 60 ppm surveillance limit for Specification 3.1.3 is being added to TS 5.6.5.a, Item 1. This surveillance limit was relocated to the COLR in the MNS conversion to the Improved Technical Specifications (ITS) amendment (Facility Operating License (FOL) Amendments 184/166, NRC SER dated September 30, 1998). However, reference to this surveillance was not included in TS 5.6.5.a at that time.

The following new item is being added to TS 5.6.5.a:

12. 31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2.

New Item 12, as shown above, is being added to TS 5.6.5.a. This new item contains two additional MNS TS that reference the COLR. The surveillance penalty factors were relocated to the COLR by MNS FOL Amendments 188/169 (NRC SER dated September 22, 1999), but reference to these surveillances was not included in TS 5.6.5.a at that time.

Attachment 3

Description of Proposed Changes and Technical Justification

CNS - Proposed Changes to TS 5.6.5.a

This TS requires that core operating limits be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and lists various TS requirements which shall be documented in the COLR. CNS is proposing to include other existing TS in the list contained in TS 5.6.5.a.

The moderator temperature coefficient 60 ppm surveillance limit for Specifications 3.1.3 is being added to the current TS 5.6.5.a, Item 1. This surveillance limit was relocated to the COLR in the CNS conversion to ITS amendment (FOL Amendments 173/165, NRC SER dated September 30, 1998). However, reference to this surveillance was not included in TS 5.6.5.a at that time.

The following new items are being added to TS 5.6.5.a:

12. 31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2.

13. Reactor makeup water pumps combined flow rates limit for Specifications 3.3.9 and 3.9.2.

New Items 12 and 13, as shown above, are being added to TS 5.6.5.a. These two new items contain additional CNS TS that reference the COLR. The surveillance penalty factors for TS 3.2.1 and 3.2.2 shown above in new Item 12 were relocated to the COLR in CNS FOL Amendments 180/172 (NRC SER dated September 22, 1999), but reference to these surveillances was not included in TS 5.6.5.a at that time. The reactor makeup water pumps flow rate limits were relocated to the COLR in CNS FOL Amendments 115/109 (NRC SER dated March 25, 1994); however, reference to these flow rate limits was omitted from TS 5.6.5.a at that time.

Attachment 3

Description of Proposed Changes and Technical Justification

MNS and CNS Nuclear Stations - Proposed Changes to TS 5.6.5.b

The proposed changes to MNS TS 5.6.5.b and CNS TS 5.6.5.b are the same. Therefore, the discussion of these proposed changes is presented in a consolidated manner. This TS references various methods used to develop the COLR. MNS and CNS are proposing changes to support the implementation of the Topical Report revisions listed below.

1) DPC-NE-2009-P-A, *Duke Power Company Westinghouse Fuel Transition Report*, is being changed to Revision 1;

2) DPC-NF-2010-A, *Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design*, is being changed to Revision 1; and

3) DPC-NE-2011-P-A, *Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*, is being changed to Revision 1.

Subsequent to the initial NRC approval of the above topical reports, the NRC has approved additional computational methods and computer codes, Technical Specification changes, and UFSAR changes. These topical reports are being revised to be consistent with these newer NRC approved methods and documents as applicable to the topical reports listed above. These revisions also incorporate some editorial changes, references, and changes in the descriptions of computational processes to avoid difficulties in literal interpretation. The proposed revisions are contained in Attachments 6, 7, and 8 of this submittal package.

Implementation of these revised topical reports will impact the MNS and CNS Updated Final Safety Analysis Reports (UFSAR). For MNS, UFSAR Chapters 1.6, 4.1, 4.2, 4.3, 4.4, 15.0, and 15.4 discuss or reference one or more of these topical reports. For CNS, UFSAR Chapters 1.5, 4.1, 4.2, 4.3, 4.4, and 15.4 discuss or reference one or more of these topical reports. Both of these UFSARs will be updated as appropriate in accordance with 10CFR50.71(e).

Two additional changes are being made to MNS and CNS TS 5.6.5.b. TS 5.6.5.b, Item 5, lists the reference document Topical Report DPC-NE-3001-P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC proprietary). The correct SER date of November 15, 1991 is being specified for this topical report. TS 5.6.5.b, Item 7, lists the reference document Topical Report DPC-NE-3002-A, Rev. 3 "FSAR Chapter 15 System Transient Analysis Methodology," SER dated February 5, 1999. The revision number is being changed to 4 and the SER date is being changed to April 6, 2001. These changes to Item 7 are consistent with the NRC's recently issued approval of this topical report revision.

MNS and CNS - Changes to Bases 3.2.1 and 3.2.3

The proposed changes to MNS Bases 3.2.1 and 3.2.3 and CNS Bases 3.2.1 and 3.2.3 are the same. Therefore, the discussion of these proposed changes is presented in a consolidated manner. The list of reference documents contained in Bases 3.2.1 and 3.2.3 are being changed to be consistent with the proposed revision to Topical Report DPC-NE-2011-P discussed above. "Rev. 1" is being added.

Conclusion

The proposed changes to the MNS and CNS TS and Bases, as described above, have been determined to be acceptable since the changes only add references to TS that are NRC-approved and already contained elsewhere in the current MNS and CNS TS, or update topical report references that, upon issuance of this LAR, will have been reviewed and approved by the NRC.

Attachment 4
No Significant Hazards Consideration Determination

Duke Energy Corporation (Duke) has made the determination that this license amendment request (LAR) involves No Significant Hazards Consideration by applying the standards established by the NRC's regulations in 10CFR50.92. These three standards are discussed below.

1. Would implementation of the changes proposed in this LAR involve a significant increase in the probability or consequences of an accident previously evaluated?

   No. This LAR makes conservative changes and/or additions to the list of referenced Technical Specifications (TS) and to five Duke topical reports listed in McGuire Nuclear Station TS 5.6.5 and Catawba Nuclear Station TS 5.6.5, Core Operating Limits Report (COLR). The topical reports are: 1) DPC-NE-2009-P-A, *Duke Power Company Westinghouse Fuel Transition Report*; 2) DPC-NF-2010-A, *Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design*; 3) DPC-NE-2011-P-A, *Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors*; 4) DPC-NE-3001-P-A, *Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology*; and 5) DPC-NE-3002-A, *FSAR Chapter 15 System Transient Analysis Methodology*. The changes proposed to these topical reports are consistent with the applicable McGuire Nuclear Station and Catawba Nuclear Station licensing bases transient analyses. Additionally, all applicable acceptance criteria continue to be met. The additions to the list of referenced TS are solely editorial in nature. Therefore, the proposed changes have no impact on any accident probabilities or consequences.

2. Would implementation of the changes proposed in this LAR create the possibility of a new or different kind of accident from any accident previously evaluated?

   No. The proposed changes contained in this LAR only make additions or clarifications that are consistent with the McGuire Nuclear Station and Catawba Nuclear Station licensing bases and established plant operating practices. Therefore, no new or different kinds of accidents are being created.

Attachment 4

No Significant Hazards Consideration Determination

3. Would implementation of the changes proposed in this LAR Involve a significant reduction in a margin of safety?

   No. Margin of safety is related to the confidence in the ability of the fission product barriers to perform their design functions during and following an accident situation. These barriers include the fuel cladding, the reactor coolant system, and the containment system. These barriers are unaffected by the changes proposed in this LAR. The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event and thereby protect the fission product barriers. The changes proposed in this LAR make editorial additions to a list of referenced TS that are currently approved for use at McGuire Nuclear Station and Catawba Nuclear Station. Additionally, this LAR revises the list of topical reports used as reference documents for the McGuire Nuclear Station and Catawba Nuclear Station COLR. The changes proposed to these topical reports are consistent with the applicable McGuire Nuclear Station and Catawba Nuclear Station licensing bases transient analyses such that all applicable acceptance criteria will continue to be met. Consequently, no margin of safety will be significantly impacted by this LAR.

Attachment 5

Environmental Assessment/Impact Statement

The proposed Technical Specification amendment has been reviewed against the criteria of 10 CFR 51.22 for environmental considerations. The proposed amendment does not involve a significant hazards consideration, nor increase the types and amounts of effluents that may be released offsite, nor increase individual or cumulative occupational radiation exposures. Therefore, the proposed amendment meets the criteria given in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirement for performing an Environmental Assessment/Impact Statement.

ATTACHMENT 6a

Listing of Changes to DPC-NE-2009-P-A

Page	
6-5	Added referral to references 6-27 and 6-39
6-25	Updated reference 6-25 to Rev. 1, July 1997
6-26	For reference 6-35, corrected proprietary topical report number and designated the 2nd report as a non-proprietary report
6-27	Added reference 6-39, an approved WCAP which was mistakenly left out of the original reference list

ATTACHMENT 6c

DPC-NE-2009, Revision 1

Duke Power®

A Duke Energy Company

ATTACHMENT 6c

DPC-NE-2009, Revision 1

NON-PROPRIETARY
Duke Power Company

DPC-NE-2009, Rev. 1

DUKE POWER COMPANY WESTINGHOUSE FUEL TRANSITION REPORT

Original Version: July 1998

Approved Version: December 1999

Revision 1: August 2001

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

TABLE OF CONTENTS

Section A

NRC Acceptance Letter and SER, McGuire Nuclear Station

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
Huntersville, NC 28078-8985

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

cc:
Ms. Lisa F. Vaughn
Legal Department (PBO5E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

County Manager of
Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Michael T. Cash
Regulatory Compliance Manager
Duke Energy Corporation
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Senior Resident Inspector
c/o U.S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Dr. John M. Barry
Mecklenberg County
Department of Environmental
Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Steven P. Shaver
Senior Sales Engineer
Westinshouse Electric Company
5929 Carnegie Blvd.
Suite 500
Charlotte, North Carolina 28209

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of
Justice
P. O. Box 629
Raleigh, North Carolina 27602

L. A. Keller
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

Mr. Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health and Natural
Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

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 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 Assembles," 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.

22. WCAP-10054-P-A (Proprietary), WCAP-10081 (Non-Proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

23. WCAP-10054-P-A Addendum 2 (Proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and Cosi Condensation Model," August 1994.

24. WCAP-10266-P-A Revision 2 with Addenda (Proprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

25. DPC-NE-3002-A, Rev. 2, "UFSAR Chapter 15 System Transient Analysis Methodology," December 1997.

26. DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology," Rev. 1, December 1997.

27. EPRI NP-7375-CCML, Vol. 1, "ARROTTA-01 - An Advanced Rapid Reactor Operational Transient Analysis Computer Code," August 1991.

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.

30. K. S. Smith, "QUANDRY: An Analytical Nodal Method for Solving the Two-Group, Multidimensional, Static and Transient Nodal Diffusion Equations," Massachusetts, 1979.

31. B. R. Bandini, "NEM: A Three Dimensional Transient Neutronics Routine for the TRAC-PF1 Reactor Thermal Hydraulic Computer Code," Pennsylvania State University, 1990.

32. S. Langenbuch, W. Maurer, and W. Werner, "CUBBOX: Coarse-Mesh Nodal Diffusion Method for the Analysis of Space-Time Effects in Large Light Water Reactors," Nuclear Sci. Eng. 63-437,1977.

33. H. Finneman and A. Galati, "NEACRP-3-D LWR Core Transient Benchmark," NEACRP-L-335, January 1992.

34. DPC-NE-3005-P, "Oconee UFSAR Chapter 15 Transient Analysis Methodology," July 1997.

35. 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," April 7, 1999.

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.

ATTACHMENT 1

TECHNICAL EVALUATION REPORT OF SECTION 4.0 OF TOPICAL REPORT DPC-NE-2009
"DUKE POWER COMPANY WESTINGHOUSE FUEL TRANSITION REPORT"

PREPARED BY
PACIFIC NORTHWEST NATIONAL LABORATORY

Technical Evaluation Report of Section 4.0 of Topical Report DPC-NE-2009P

"Duke Power Company Westinghouse Fuel Transition Report"

1.0 INTRODUCTION

This technical evaluation report (TER) only addresses Section 4.0 of DPC-NE-2009P (Reference 1) which describes Duke Power Company's (DPC) application of the Westinghouse (W) developed Performance Analysis and Design (PAD) code, Version 3.4 (PAD 3.4) fuel performance code and other W analysis methods. DPC will apply PAD 3.4 for reload thermal-mechanical licensing analyses for Westinghouse fuel in their PWR plants. The PAD 3.4 code has been approved by the U. S. Nuclear Regulatory Commission (Reference 2). DPC's quality assurance procedures to verify that the code performs as developed by W, and controls to prevent the code from being altered without adequate review and approval, are reviewed in this TER.

DPC intends to use the PAD 3.4 fuel performance code for the following licensing reload analyses:

1) fuel rod cladding stresses;
2) fuel rod cladding strain;
3) fuel rod cladding strain fatigue;
4) fuel rod internal pressure;
5) fuel temperature (melting); and
6) fuel rod cladding corrosion and hydriding.

Another W analysis method used is:

7) W developed correlations for fuel rod and assembly axial growth.

Pacific Northwest National Laboratory (PNNL) has acted as a consultant to the NRC in this review. The NRC staff and their PNNL consultants performed the review of the subject topical report and writing of this TER. The review was based on those licensing requirements identified in Section 4.2 of the Standard Review Plan (SRP) (Reference 3) for thermal-mechanical analyses. The objectives of this review of fuel design criteria, as described in Section 4.2 of the SRP, are to provide assurance that 1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), 2) the fuel system damage is never so severe as to prevent control rod insertion when it is required, 3) the number of fuel rod failures is not underestimated for postulated accidents, and 4) the coolability is always maintained. A "not damaged" fuel system is defined as fuel rods that do not fail, fuel system dimensions that remain within operational tolerances, and functional capabilities that are not reduced below those assumed in the safety analyses. Objective 1, above, is consistent with General Design Criterion (GDC) 10 [10 Code of Federal Regulations (CFR) 50, Appendix A] (Reference 4), and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR 100 (Reference 5) for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident (LOCA) are given in 10 CFR 50, Section 50.46.

In order to assure that the above stated objectives are met, this review addresses the thermal- mechanical issues identified in Section 4.2 of the SRP. DPC has addressed the major issues applicable to the fuel thermal-mechanical licensing analyses in Section 4 of DPC-NE-2009P. Section 4.2 of the SRP breaks the thermal-mechanical issues into two major categories; 1) Fuel System Damage Mechanisms, which are most applicable to normal operation and AOOs, and 2) Fuel Rod Failure Mechanisms, which apply to normal operation, AOOs, and postulated accidents. The SRP category of Fuel Coolability which is applied to postulated accidents is not addressed in Section 4.0 of the subject topical and is not reviewed in this TER. The TER utilizes the same format structure as provided in the subject topical report with the exception that each application is subdivided into Bases/Criteria and Evaluation subsections which loosely follows the SRP.

2.0 DPC APPLICATION OF PAD 3.4 CODE AND OTHER WESTINGHOUSE ANALYSIS METHODS

As noted in Section 1.0, DPC intends to use the PAD 3.4 fuel performance code for fuel rod cladding stress, fuel rod cladding strain, fuel rod cladding strain fatigue, fuel rod internal pressure, fuel temperature analyses and fuel rod cladding oxidation. The DPC fuel rod axial growth analysis uses the W models (correlations) for rod and assembly growth. Each of these analyses will be discussed separately below, which are subdivided into Bases/Criteria and Evaluation subsections. Each of the DPC Bases/Criteria given below is the same as those defined by W in their NRC approved Fuel Criteria Evaluation Process, FCEP (Reference 6).

2.1 Fuel Rod Cladding Stress

Basis/Criteria - The stress design limit requires that the volume averaged effective stress calculated with the Von Mises equation, considering interference due to uniform cylindrical pellet-to-cladding contact (caused by pellet thermal expansion and swelling, uniform cladding creep, and fuel rod/coolant system pressure differences), be less than the Zircaloy-4 and ZIRLO 0.2 percent offset yield stress with consideration of temperature and irradiation effects. The DPC design limit for fuel rod cladding stress under normal operation and AOOs is the same as defined by W in their NRC approved Fuel Criteria Evaluation Process, FCEP (Reference 6). PNNL concludes that this criterion is acceptable for application by DPC to W fuel reload applications.

Evaluation - The PAD 3.4 fuel performance code (Reference 2) is used by DPC to assure that the stress criterion is met. This code has been verified against fuel rod data with rod-average burn-up levels up to 62 GWd/MTU. This code takes into account those parameters important for determining cladding stresses and strains at extended burn-ups, such as pellet thermal expansion and swelling, cladding creep, and fuel rod/coolant system pressure differences. DPC has provided an example stress analysis for W reloads in the McGuire and Catawba plants (Reference 7). These analyses were reviewed and were found to be consistent with W analysis methodology.

One of the more important input parameters for the stress analysis is the power history with the higher rod power generally giving the more conservative value. Several possible bounding power histories are chosen by DPC to bound possible rod powers for each cycle of operation for the stress analyses. These are used as input to PAD 3.4 to determine those that are limiting in regards to the stress criterion. DPC determines the maximum possible bounding power histories using DPC neutronics codes and methodology approved by the NRC rather than Westinghouse codes. Also, AOOs are superimposed on these bounding power histories. This DPC methodology for determining bounding power histories is comparable to the W methodology. PNNL concludes that the PAD 3.4 code and DPC analysis methodology are acceptable for determining stress for W fuel reload applications.

2.2 Fuel Rod Cladding Strain

Bases/Criteria - The DPC design limit for cladding strain during steady-state operation is that the total plastic tensile creep due to uniform cylindrical fuel pellet expansion from fuel swelling and thermal expansion be less than 1 percent from the unirradiated condition. For AOO transients, the design limit for cladding strain is that the total tensile strain due to uniform cylindrical pellet thermal expansion during the transient be less than 1 percent of the pretransient value. These design limits are intended to preclude excessive cladding deformation during normal operation and AOOs. These limits are the same as used in Section 4.2 of the SRP.

It is noted, however, that the material property that could have a significant impact on the cladding strain limit at burn-up levels beyond those currently approved is cladding ductility. The strain criterion could be impacted if cladding ductility were decreased, as a result of extended burn-up operation, to a level that would allow cladding failure without the normal operation and AOOs cladding strain criteria being exceeded in the DPC analyses. This issue will be addressed when further burn-up extensions are requested beyond the currently approved burn-up limit of 62 GWd/MTU (rod-average). PNNL concludes that the DPC strain limits are acceptable for application to W fuel reload applications.

Evaluation - The PAD 3.4 fuel performance code (Reference 2) is used by DPC to assure that W fuel reloads meet the above criteria for steady-state and transient induced strains. As noted in the Design Stress section, this code has been verified against fuel rod data with rod-average burn-up levels up to 62 GWd/MTU and takes into account those parameters important for determining cladding stresses and strains at extended burn-up limits. DPC has provided an example strain analysis for W reloads in the McGuire and Catawba plants (Reference 8) and these were reviewed.

Similar to the stress analysis, several possible bounding power histories are chosen by DPC to bound possible rod powers and for the steady-state strain analysis. The limiting power histories are typically those rods with the maximum power and burn-up history, and the maximum power near the end-of-life (EOL). DPC determines the maximum possible bounding power histories using DPC neutronics codes and methodology previously approved by the NRC rather than Westinghouse codes. In order to further assure that the analysis is bounding, DPC performs a best estimate strain calculation using the bounding power history and then adds an uncertainly that is equal to the square root of the sum of the squares of those uncertainties introduced from fabrication and model uncertainties that are important to the strain analysis. This DPC methodology for determining boundary power histories for cladding strain is comparable to the W methodology.

DPC was questioned on the analysis for transient strain due to normal operating transients and AOOs. DPC responded that W had performed generic bounding analyses for current W fuel designs and concluded that the stress analysis is always bounding for a given delta power (kW/ft) increase (Reference 8). Therefore, DPC's position is the same as W in that the stress analysis is bounding for transient strain analyses. PNNL concludes that the PAD 3.4 code and DPC analysis methodology are acceptable for determining cladding strains for W fuel reload applications.

2.3 Fuel Rod Cladding Strain Fatigue

Bases/Criteria - The DPC design limit for strain fatigue is that the fatigue life usage factor be less than 1.0. That is, for a given strain range, the number of strain fatigue cycles are less than those required for failure when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is the more conservative, is imposed. This criteria is essentially the same as that described in Section 4.2 of the SRP. PNNL concludes that this criterion is acceptable for application by DPC to W fuel reload applications.

Evaluation - The PAD 3.4 fuel performance code (Reference 2) is used by DPC to assure that the strain fatigue criterion is met. This code has been verified against fuel rod data with rod-average burnup levels up to 62 GWd/MTU. This code takes into account those parameters important for determining cladding stresses and strains at extended burnups, such as pellet thermal expansion and swelling, cladding creep, and fuel rod/coolant system pressure differences. DPC has provided an example strain fatigue analysis for W reloads in the McGuire and Catawba plants (Reference 7). This analysis was reviewed and found to be consistent with W analysis methodologies.

One of the more important input parameters for the strain fatigue analysis is the power history with the higher rod power for a given cycle of operation generally giving the more conservative value for that cycle. Several possible bounding power histories are chosen by DPC to bound possible rod powers for each cycle of operation for the stress analyses and these are also applied to the fatigue analysis. These are used as input to PAD 3.4 to determine those that are limiting in regards to the strain fatigue criterion. DPC determines the maximum possible bounding power histories using DPC neutronics codes and methodology approved by the NRC rather than Westinghouse codes. The DPC methodology takes into account daily load follow operation and the additional fatigue load cycles that may result from extended burnup operation. This methodology for determining the power history for strain fatigue is conservative and comparable to the W methodology.

The Langer-O'Donnell fatigue model (Reference 9), with the empirical factors in the model modified in order to conservatively bound the W Zircaloy-4 data (also applicable to ZIRLO), is used with the strains from PAD 3.4 to assure that the above criterion is met. A description of this methodology and the W data base is presented in WCAP-9500 (Reference 10), which has been approved by the NRC. This strain fatigue methodology has also been found to be acceptable by NRC for ZIRLO clad fuel (Reference 11). PNNL concludes that the PAD 3.4 code and DPC analysis methodology are acceptable for determining strain fatigue for W fuel reload applications.

2.4 Fuel Rod Internal Pressure

Bases/Criteria - The DPC design limits are that the internal pressure of the lead rod (in terms of rod pressure) in the reactor will be limited to a value below which could result in 1) the diametral gap to increase due to outward cladding creep during steady-state operation, or 2) extensive departure from nucleate boiling (DNB) propagation to occur during normal operation or AOOs. The design limits have previously been found acceptable by the NRC up to 62 GWd/MTU (Reference 6). PNNL concludes they are also acceptable for application by DPC to W fuel reload applications.

Evaluation - The PAD 3.4 code (Reference 2) is used by DPC to assure that the diametral gap between the fuel and cladding does not open due to cladding creep (item 1 in Bases/Criteria above). This code has been verified against fuel rod data with rod-average burnup levels up to 62 GWd/MTU. This code models those phenomena important for evaluating rod pressure such as fission gas release, fuel swelling, and cladding creep. DPC uses the W analysis methodology to assure that extensive DNB propagation does not occur for normal operation or AOOs (item 2 in Bases/Criteria above) and that fuel failure and dose are not underestimated for accidents. DPC provided example DPC rod pressure analyses for both item 1 and 2 types of analyses for W reloads in the McGuire and Catawba plants (References 12 and 13, respectively). These analyses were reviewed and found to be consistent with W analysis methodology.

One of the more important input parameters for the rod internal pressure analysis in regards to item 1 is the power history with the higher rod power in a cycle giving the more conservative value for rod pressure for this cycle. DPC selects several possible bounding power histories to bound the rod powers for each cycle of operation for the rod pressure analysis. Also, power increases due to normal operating transients and AOOs are superimposed on these bounding power histories. These are used as input to PAD 3.4 to determine those rods that are limiting in regards to the rod pressure limit. DPC determines the maximum possible bounding power histories using DPC neutronics codes and methodology previously approved by the NRC rather than Westinghouse codes. DPC has utilized generic axial power shapes for their rod pressure analysis in Reference 12. It is noted that the rod pressure analysis can be dependent on the axial power shape. DPC was questioned on whether these axial shapes change from cycle to cycle. DPC replied that, in examining axial shapes for several past cycles of operation, they changed very little from the assumed generic axial shapes and the small change had little impact on the analysis. DPC has stated that they will continue to confirm that the generic axial shapes remain applicable to the operation of each future fuel reload for the rod pressure analysis.

Similar to the cladding strain analysis (Section 3.2), DPC performs a best estimate rod pressure calculation with PAD 3.4 using the bounding power history as input. In addition, DPC calculates the uncertainty in terms of rod pressure introduced by the uncertainty in each fabrication/design variable and also introduced by the model uncertainties that are important to the rod pressure analysis. The square root of the sum of squares of the individual rod pressure uncertainties are added to the best estimate rod pressure to obtain a bounding estimate of rod pressure for a 95% probability at a 95% confidence level. DPC will continue to confirm that the axial power shapes used for this analysis remains applicable to the specific fuel reload under evaluation. The DPC application of the PAD 3.4 fuel performance code for the rod pressure analysis to assure that the diametral gap does not open due to cladding creep was found to be consistent with W methodology and, therefore, is acceptable for W reload application.

DPC utilizes the W methodology for assuring that DNB propogation does not occur for normal operation and AOOs (item 2 above) and that fuel failures (and dose) are not underestimated for accidents. PNNL has reviewed the example DPC DNB propagation analysis for rod pressure for W reloads in the McGuire and Catawba plants (Reference 13). This analysis methodology was found to be consistent with W analysis methodology and, therefore, is acceptable for W reload applications..

PNNL concludes that the PAD 3.4 code and DPC analysis methodology are acceptable for evaluating rod internal pressures for W fuel reload applications.

2.5 Fuel Temperature

Bases/Criteria - The DPC fuel temperature limit precludes centerline pellet melting during normal operation and AOOs. This design limit is the same as given in the SRP and has been approved for application for W fuel designs up to a rod-average burnup level of 62 GWd/MTU (Reference 6). In order to ensure that this basis is met, DPC imposes a design limit on fuel temperatures such that there is at least a 95% probability at a 95% confidence level that during normal operation and AOO events the peak linear heat generation rate rod will not exceed the fuel melting temperature. W and DPC have placed a temperature limit on fuel melting at extended fuel burnup levels that have previously been approved for burnups up to 62 GWd/MTU. Therefore, PNNL concludes that DPC's design limit for fuel melting is acceptable for application to W fuel reload applications.

Evaluation - The PAD 3.4 fuel performance code (Reference 2) is used by DPC to assure that the fuel melting criterion is met. This code has been verified against fuel rod data with rod-average burnup levels up to approximately 62 GWd/MTU. DPC provided an example fuel melting analysis for W reloads in the McGuire and Catawba plants (Reference 14). These example DPC analyses are consistent with W analysis methodology.

There has been recent evidence of a decrease in fuel thermal conductivity with burnup; however, there remains a considerable uncertainty in this data and the NRC is still examining the implications for the fuel melting analysis. In addition, W states (Reference 14) that maximum fuel temperatures occur near beginning-of-life (BOL). Because NRC and industry are still evaluating the decrease in thermal conductivity with burnup, the current fuel thermal conductivity model in PAD 3.4 remains acceptable. Therefore, PNNL concludes that DPC's use of the PAD 3.4 code for the fuel melting analysis is acceptable for application to W fuel reload applications.

2.6 Fuel Clad Oxidation and Hydriding

Bases/Criteria - In order to preclude a condition of accelerated oxidation and cladding degradation, DPC imposes the W temperature limits on the cladding and a limit on hydrogen pickup in the cladding due to corrosion. The temperature limits applied to cladding oxidation are that calculated cladding temperatures (at the oxide-to-metal interface) shall be less than a specific (proprietary) value during steady-state operation and AOOs transients (a higher temperature limit is applied for AOOs transients). In addition, W has a limit on hydrogen pickup for the cladding. These criteria have been approved by NRC (Reference 10) up to a rod-average burnup limit of 62 GWd/MTU. Therefore, PNNL concludes that the DPC design criteria for oxidation and hydriding are acceptable for W reload applications.

Evaluation - The corrosion model in PAD 3.4 is used by DPC to assure that the W limits on cladding corrosion are met. DPC has provided an example cladding corrosion analysis for the cladding and assembly structural members for W reloads in the McGuire and Catawba plants (Reference 15). Similar to the rod internal pressure analysis, DPC uses a generic axial power shape for cladding corrosion. It is noted that cladding corrosion can also be sensitive to the axial power shape and, therefore, DPC will continue to confirm that the generic axial shapes remain applicable to the operation of each future fuel reload for corrosion analyses. The example DPC oxidation analysis has been reviewed and found to be consistent with the W analysis methodology. PNNL concludes that DPC's use of the PAD 3.4 code corrosion model is acceptable for evaluating corrosion for W fuel reload applications.

2.7 Fuel Rod Axial Growth

Bases/Criteria - Failure to adequately design for axial growth of the fuel rods can lead to fuel rod-to-nozzle gap closure resulting in fuel rod bowing and possible rod failure or failure of the thimble tubes. The DPC design limit is that the space between the rod end plug-to-end plug outer dimension and the lower nozzle-to-top adapter plate inner dimension shall be sufficient to preclude interference of these members.

This design limit has been accepted by the NRC for current W fuel designs up to a rod-average burnup limit of 62 GWd/MTU (Reference 6). Therefore, PNNL concludes that the DPC design limit for axial growth is acceptable for application to W fuel reload applications.

Evaluation - DPC uses the W correlations for rod and assembly growth and the W analysis methodology to evaluate the rod-to-nozzle clearance. The analysis methodology conservatively uses the upper-bound rod growth and lower bound assembly growth correlations along with the minimum rod-to-nozzle clearance based on a statistical combination of fabrication tolerances. The W rod and assembly growth correlations and analysis methodology have been approved by the NRC up to a rod-average burnup limit of 62 GWd/MTU.

DPC has provided an example rod-to-nozzle clearance analysis for W reloads in the McGuire and Catawba plants (Reference 16). This example DPC growth analysis is consistent with W analysis methodology. PNNL concludes that the DPC application of the W fuel rod and assembly growth correlations and analysis methods are acceptable for evaluating axial growth for W fuel reload applications.

3.0 CONCLUSIONS

PNNL concludes that the DPC design limits and thermal-mechanical analyses discussed in Section 4.0 of DPC-NE-2009P are acceptable for application by DPC to W fuel reloads up to the currently approved rod-average burnup limit of 62 GWd/MTU. In addition, the use of W growth models and analysis methodology discussed in the subject submittal are acceptable for application by DPC to W fuel reload applications up to currently approved burnups.

4.0 REFERENCES

1. Duke Power Company, July 1998. Duke Power Company Westinghouse Fuel Transition Report, DPC-NE-2009P (Proprietary), Duke Power Company, Charlotte, North Carolina.

2. Weiner, R. A., et al. August 1988. Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations. WCAP-10851-P-A (Proprietary), Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

3. U.S. Nuclear Regulatory Commission. July 1981. "Section 4.2, Fuel System Design." In Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition. NUREG-0800, Revision 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

4. U.S. Federal Register. "Appendix A, General Design Criteria for Nuclear Power Plants." In 10 Code of Federal Regulations, Part 50. U.S. Printing Office, Washington, D.C.

5. U.S. Federal Register. "Reactor Site Criteria." In 10 Code of Federal Regulations, Part 100. U.S. Printing Office, Washington, D.C.

6. Davidson, S. L. (Editor). April 1990. Westinghouse Fuel Criteria Evaluation Process. WCAP-12488, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

7. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (All Units) -- Generic Stress and Fatigue Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0138 (Proprietary), Duke Power Company, Charlotte, North Carolina.

8. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (All Units) -- Generic Steady State Strain Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0135 (Proprietary), Duke Power Company, Charlotte, North Carolina.

9. O'Donnell, W. J., and B. F. Langer. 1964. "Fatigue Design Basis for Zircaloy Components." In Nuclear Science Engineering, Vol. 20, pp. 1.

10. Davidson, D. L., and J. A. Iorii. May 1982. Reference Core Report 17x17 Optimized Fuel Assembly. WCAP-9500-A (Proprietary), Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

11. Davidson, S. L., and D. L. Nuhfer (Editors). June 1990. VANTAGE+ Fuel Assembly Reference Core Report. WCAP-12610 (Proprietary), Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

12. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (Units 1 & 2) -- Generic Rod Internal Pressure Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0130 (Proprietary), Duke Power Company, Charlotte, North Carolina.

13. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (Units 1 & 2) -- Generic DNBRIP Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0140 (Proprietary), Duke Power Company, Charlotte, North Carolina.

14. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (Units 1 & 2) -- Generic Fuel Melt Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0133 (Proprietary), Duke Power Company, Charlotte, North Carolina.

15. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (All Units) – Generic Westinghouse 17 x 17 Corrosion Analysis, DPC-1553.26-00-0125 (Proprietary), Duke Power Company, Charlotte, North Carolina.

16. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (Units 1 & 2) -- Generic Rod Growth Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0130 (Proprietary), Duke Power Company, Charlotte, North Carolina.

Section B

NRC Acceptance Letter and SER, Catawba Nuclear Station

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

September 22, 1999

Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

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

cc:

Mr. Gary Gilbert
Regulatory Compliance Manager
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Ms. Lisa F. Vaughn
Legal Department (PB05E)
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

Anne Cottington, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626

County Manager of York County
York County Courthouse
York, South Carolina 29745

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

North Carolina Electric Membership
Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
4830 Concord Road
York, South Carolina 29745

Virgil R. Autry, Director
Division of Radioactive Waste Management
Bureau of Land and Waste Management
Department of Health and Environmental
Control
2600 Bull Street
Columbia, South Carolina 29201-1708

L. A. Keller
Manager - Nuclear Regulatory
Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Saluda River Electric
P. O. Box 929
Laurens, South Carolina 29360

Mr. Steven P. Shaver
Senior Sales Engineer
Westinghouse Electric Company
5929 Carnegie Blvd.
Suite 500
Charlotte, North Carolina 28209

Catawba Nuclear Station

cc:

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Richard M. Fry, Director
Division of Radiation Protection
North Carolina Department of
Environment, Health, and
Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

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.

* 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 <u>UFSAR Accident Analyses</u>

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 Assembles," 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.

22. WCAP-10054-P-A (Proprietary), WCAP-10081 (Non-Proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

23. WCAP-10054-P-A Addendum 2 (Proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and Cosi Condensation Model," August 1994.

24. WCAP-10266-P-A Revision 2 with Addenda (Proprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," March 1987.

25. DPC-NE-3002-A, Rev. 2, "UFSAR Chapter 15 System Transient Analysis Methodology," December 1997.

26. DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology," Rev. 1, December 1997.

27. EPRI NP-7375-CCML, Vol. 1, "ARROTTA-01 - An Advanced Rapid Reactor Operational Transient Analysis Computer Code," August 1991.

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.

30. K. S. Smith, "QUANDRY: An Analytical Nodal Method for Solving the Two-Group, Multidimensional, Static and Transient Nodal Diffusion Equations," Massachusetts, 1979.

31. B. R. Bandini, "NEM: A Three Dimensional Transient Neutronics Routine for the TRAC-PF1 Reactor Thermal Hydraulic Computer Code," Pennsylvania State University, 1990.

32. S. Langenbuch, W. Maurer, and W. Werner, "CUBBOX: Coarse-Mesh Nodal Diffusion Method for the Analysis of Space-Time Effects in Large Light Water Reactors," Nuclear Sci. Eng. 63-437,1977.

33. H. Finneman and A. Galati, "NEACRP-3-D LWR Core Transient Benchmark," NEACRP-L-335, January 1992.

34. DPC-NE-3005-P, "Oconee UFSAR Chapter 15 Transient Analysis Methodology," July 1997.

35. 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," April 7, 1999.

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.

ATTACHMENT 1

TECHNICAL EVALUATION REPORT OF SECTION 4.0 OF TOPICAL REPORT DPC-NE-2009 "DUKE POWER COMPANY WESTINGHOUSE FUEL TRANSITION REPORT"

PREPARED BY
PACIFIC NORTHWEST NATIONAL LABORATORY

Technical Evaluation Report of Section 4.0 of Topical Report DPC-NE-2009P

"Duke Power Company Westinghouse Fuel Transition Report"

1.0 INTRODUCTION

This technical evaluation report (TER) only addresses Section 4.0 of DPC-NE-2009P (Reference 1) which describes Duke Power Company's (DPC) application of the Westinghouse (W) developed Performance Analysis and Design (PAD) code, Version 3.4 (PAD 3.4) fuel performance code and other W analysis methods. DPC will apply PAD 3.4 for reload thermal-mechanical licensing analyses for Westinghouse fuel in their PWR plants. The PAD 3.4 code has been approved by the U. S. Nuclear Regulatory Commission (Reference 2). DPC's quality assurance procedures to verify that the code performs as developed by W, and controls to prevent the code from being altered without adequate review and approval, are reviewed in this TER.

DPC intends to use the PAD 3.4 fuel performance code for the following licensing reload analyses:

1) fuel rod cladding stresses;
2) fuel rod cladding strain;
3) fuel rod cladding strain fatigue;
4) fuel rod internal pressure;
5) fuel temperature (melting); and
6) fuel rod cladding corrosion and hydriding.

Another W analysis method used is:

7) W developed correlations for fuel rod and assembly axial growth.

Pacific Northwest National Laboratory (PNNL) has acted as a consultant to the NRC in this review. The NRC staff and their PNNL consultants performed the review of the subject topical report and writing of this TER. The review was based on those licensing requirements identified in Section 4.2 of the Standard Review Plan (SRP) (Reference 3) for thermal-mechanical analyses. The objectives of this review of fuel design criteria, as described in Section 4.2 of the SRP, are to provide assurance that 1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), 2) the fuel system damage is never so severe as to prevent control rod insertion when it is required, 3) the number of fuel rod failures is not underestimated for postulated accidents, and 4) the coolability is always maintained. A "not damaged" fuel system is defined as fuel rods that do not fail, fuel system dimensions that remain within operational tolerances, and functional capabilities that are not reduced below those assumed in the safety analyses. Objective 1, above, is consistent with General Design Criterion (GDC) 10 [10 Code of Federal Regulations (CFR) 50, Appendix A] (Reference 4), and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR 100 (Reference 5) for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC (e.g., GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accident (LOCA) are given in 10 CFR 50, Section 50.46.

In order to assure that the above stated objectives are met, this review addresses the thermal- mechanical issues identified in Section 4.2 of the SRP. DPC has addressed the major issues applicable to the fuel thermal-mechanical licensing analyses in Section 4 of DPC-NE-2009P. Section 4.2 of the SRP breaks the thermal-mechanical issues into two major categories; 1) Fuel System Damage Mechanisms, which are most applicable to normal operation and AOOs, and 2) Fuel Rod Failure Mechanisms, which apply to normal operation, AOOs, and postulated accidents. The SRP category of Fuel Coolability which is applied to postulated accidents is not addressed in Section 4.0 of the subject topical and is not reviewed in this TER. The TER utilizes the same format structure as provided in the subject topical report with the exception that each application is subdivided into Bases/Criteria and Evaluation subsections which loosely follows the SRP.

2.0 DPC APPLICATION OF PAD 3.4 CODE AND OTHER WESTINGHOUSE ANALYSIS METHODS

As noted in Section 1.0, DPC intends to use the PAD 3.4 fuel performance code for fuel rod cladding stress, fuel rod cladding strain, fuel rod cladding strain fatigue, fuel rod internal pressure, fuel temperature analyses and fuel rod cladding oxidation. The DPC fuel rod axial growth analysis uses the W models (correlations) for rod and assembly growth. Each of these analyses will be discussed separately below, which are subdivided into Bases/Criteria and Evaluation subsections. Each of the DPC Bases/Criteria given below is the same as those defined by W in their NRC approved Fuel Criteria Evaluation Process, FCEP (Reference 6).

2.1 Fuel Rod Cladding Stress

Basis/Criteria - The stress design limit requires that the volume averaged effective stress calculated with the Von Mises equation, considering interference due to uniform cylindrical pellet-to-cladding contact (caused by pellet thermal expansion and swelling, uniform cladding creep, and fuel rod/coolant system pressure differences), be less than the Zircaloy-4 and ZIRLO 0.2 percent offset yield stress with consideration of temperature and irradiation effects. The DPC design limit for fuel rod cladding stress under normal operation and AOOs is the same as defined by W in their NRC approved Fuel Criteria Evaluation Process, FCEP (Reference 6). PNNL concludes that this criterion is acceptable for application by DPC to W fuel reload applications.

Evaluation - The PAD 3.4 fuel performance code (Reference 2) is used by DPC to assure that the stress criterion is met. This code has been verified against fuel rod data with rod-average burn-up levels up to 62 GWd/MTU. This code takes into account those parameters important for determining cladding stresses and strains at extended burn-ups, such as pellet thermal expansion and swelling, cladding creep, and fuel rod/coolant system pressure differences. DPC has provided an example stress analysis for W reloads in the McGuire and Catawba plants (Reference 7). These analyses were reviewed and were found to be consistent with W analysis methodology.

One of the more important input parameters for the stress analysis is the power history with the higher rod power generally giving the more conservative value. Several possible bounding power histories are chosen by DPC to bound possible rod powers for each cycle of operation for the stress analyses. These are used as input to PAD 3.4 to determine those that are limiting in regards to the stress criterion. DPC determines the maximum possible bounding power histories using DPC neutronics codes and methodology approved by the NRC rather than Westinghouse codes. Also, AOOs are superimposed on these bounding power histories. This DPC methodology for determining bounding power histories is comparable to the W methodology. PNNL concludes that the PAD 3.4 code and DPC analysis methodology are acceptable for determining stress for W fuel reload applications.

2.2 Fuel Rod Cladding Strain

Bases/Criteria - The DPC design limit for cladding strain during steady-state operation is that the total plastic tensile creep due to uniform cylindrical fuel pellet expansion from fuel swelling and thermal expansion be less than 1 percent from the unirradiated condition. For AOO transients, the design limit for cladding strain is that the total tensile strain due to uniform cylindrical pellet thermal expansion during the transient be less than 1 percent of the pretransient value. These design limits are intended to preclude excessive cladding deformation during normal operation and AOOs. These limits are the same as used in Section 4.2 of the SRP.

It is noted, however, that the material property that could have a significant impact on the cladding strain limit at burn-up levels beyond those currently approved is cladding ductility. The strain criterion could be impacted if cladding ductility were decreased, as a result of extended burn-up operation, to a level that would allow cladding failure without the normal operation and AOOs cladding strain criteria being exceeded in the DPC analyses. This issue will be addressed when further burn-up extensions are requested beyond the currently approved burn-up limit of 62 GWd/MTU (rod-average). PNNL concludes that the DPC strain limits are acceptable for application to W fuel reload applications.

Evaluation - The PAD 3.4 fuel performance code (Reference 2) is used by DPC to assure that W fuel reloads meet the above criteria for steady-state and transient induced strains. As noted in the Design Stress section, this code has been verified against fuel rod data with rod-average burn-up levels up to 62 GWd/MTU and takes into account those parameters important for determining cladding stresses and strains at extended burn-up limits. DPC has provided an example strain analysis for W reloads in the McGuire and Catawba plants (Reference 8) and these were reviewed.

Similar to the stress analysis, several possible bounding power histories are chosen by DPC to bound possible rod powers and for the steady-state strain analysis. The limiting power histories are typically those rods with the maximum power and burn-up history, and the maximum power near the end-of-life (EOL). DPC determines the maximum possible bounding power histories using DPC neutronics codes and methodology previously approved by the NRC rather than Westinghouse codes. In order to further assure that the analysis is bounding, DPC performs a best estimate strain calculation using the bounding power history and then adds an uncertainly that is equal to the square root of the sum of the squares of those uncertainties introduced from fabrication and model uncertainties that are important to the strain analysis. This DPC methodology for determining boundary power histories for cladding strain is comparable to the W methodology.

DPC was questioned on the analysis for transient strain due to normal operating transients and AOOs. DPC responded that W had performed generic bounding analyses for current W fuel designs and concluded that the stress analysis is always bounding for a given delta power (kW/ft) increase (Reference 8). Therefore, DPC's position is the same as W in that the stress analysis is bounding for transient strain analyses. PNNL concludes that the PAD 3.4 code and DPC analysis methodology are acceptable for determining cladding strains for W fuel reload applications.

2.3 Fuel Rod Cladding Strain Fatigue

Bases/Criteria - The DPC design limit for strain fatigue is that the fatigue life usage factor be less than 1.0. That is, for a given strain range, the number of strain fatigue cycles are less than those required for failure when a minimum safety factor of 2 on the stress amplitude or a minimum safety factor of 20 on the number of cycles, whichever is the more conservative, is imposed. This criteria is essentially the same as that described in Section 4.2 of the SRP. PNNL concludes that this criterion is acceptable for application by DPC to W fuel reload applications.

Evaluation - The PAD 3.4 fuel performance code (Reference 2) is used by DPC to assure that the strain fatigue criterion is met. This code has been verified against fuel rod data with rod-average burnup levels up to 62 GWd/MTU. This code takes into account those parameters important for determining cladding stresses and strains at extended burnups, such as pellet thermal expansion and swelling, cladding creep, and fuel rod/coolant system pressure differences. DPC has provided an example strain fatigue analysis for W reloads in the McGuire and Catawba plants (Reference 7). This analysis was reviewed and found to be consistent with W analysis methodologies.

One of the more important input parameters for the strain fatigue analysis is the power history with the higher rod power for a given cycle of operation generally giving the more conservative value for that cycle. Several possible bounding power histories are chosen by DPC to bound possible rod powers for each cycle of operation for the stress analyses and these are also applied to the fatigue analysis. These are used as input to PAD 3.4 to determine those that are limiting in regards to the strain fatigue criterion. DPC determines the maximum possible bounding power histories using DPC neutronics codes and methodology approved by the NRC rather than Westinghouse codes. The DPC methodology takes into account daily load follow operation and the additional fatigue load cycles that may result from extended burnup operation. This methodology for determining the power history for strain fatigue is conservative and comparable to the W methodology.

The Langer-O'Donnell fatigue model (Reference 9), with the empirical factors in the model modified in order to conservatively bound the W Zircaloy-4 data (also applicable to ZIRLO), is used with the strains from PAD 3.4 to assure that the above criterion is met. A description of this methodology and the W data base is presented in WCAP-9500 (Reference 10), which has been approved by the NRC. This strain fatigue methodology has also been found to be acceptable by NRC for ZIRLO clad fuel (Reference 11). PNNL concludes that the PAD 3.4 code and DPC analysis methodology are acceptable for determining strain fatigue for W fuel reload applications.

2.4 Fuel Rod Internal Pressure

Bases/Criteria - The DPC design limits are that the internal pressure of the lead rod (in terms of rod pressure) in the reactor will be limited to a value below which could result in 1) the diametral gap to increase due to outward cladding creep during steady-state operation, or 2) extensive departure from nucleate boiling (DNB) propagation to occur during normal operation or AOOs. The design limits have previously been found acceptable by the NRC up to 62 GWd/MTU (Reference 6). PNNL concludes they are also acceptable for application by DPC to W fuel reload applications.

Evaluation - The PAD 3.4 code (Reference 2) is used by DPC to assure that the diametral gap between the fuel and cladding does not open due to cladding creep (item 1 in Bases/Criteria above). This code has been verified against fuel rod data with rod-average burnup levels up to 62 GWd/MTU. This code models those phenomena important for evaluating rod pressure such as fission gas release, fuel swelling, and cladding creep. DPC uses the W analysis methodology to assure that extensive DNB propagation does not occur for normal operation or AOOs (item 2 in Bases/Criteria above) and that fuel failure and dose are not underestimated for accidents. DPC provided example DPC rod pressure analyses for both item 1 and 2 types of analyses for W reloads in the McGuire and Catawba plants (References 12 and 13, respectively). These analyses were reviewed and found to be consistent with W analysis methodology.

One of the more important input parameters for the rod internal pressure analysis in regards to item 1 is the power history with the higher rod power in a cycle giving the more conservative value for rod pressure for this cycle. DPC selects several possible bounding power histories to bound the rod powers for each cycle of operation for the rod pressure analysis. Also, power increases due to normal operating transients and AOOs are superimposed on these bounding power histories. These are used as input to PAD 3.4 to determine those rods that are limiting in regards to the rod pressure limit. DPC determines the maximum possible bounding power histories using DPC neutronics codes and methodology previously approved by the NRC rather than Westinghouse codes. DPC has utilized generic axial power shapes for their rod pressure analysis in Reference 12. It is noted that the rod pressure analysis can be dependent on the axial power shape. DPC was questioned on whether these axial shapes change from cycle to cycle. DPC replied that, in examining axial shapes for several past cycles of operation, they changed very little from the assumed generic axial shapes and the small change had little impact on the analysis. DPC has stated that they will continue to confirm that the generic axial shapes remain applicable to the operation of each future fuel reload for the rod pressure analysis.

Similar to the cladding strain analysis (Section 3.2), DPC performs a best estimate rod pressure calculation with PAD 3.4 using the bounding power history as input. In addition, DPC calculates the uncertainty in terms of rod pressure introduced by the uncertainty in each fabrication/design variable and also introduced by the model uncertainties that are important to the rod pressure analysis. The square root of the sum of squares of the individual rod pressure uncertainties are added to the best estimate rod pressure to obtain a bounding estimate of rod pressure for a 95% probability at a 95% confidence level. DPC will continue to confirm that the axial power shapes used for this analysis remains applicable to the specific fuel reload under evaluation. The DPC application of the PAD 3.4 fuel performance code for the rod pressure analysis to assure that the diametral gap does not open due to cladding creep was found to be consistent with W methodology and, therefore, is acceptable for W reload application.

DPC utilizes the W methodology for assuring that DNB propogation does not occur for normal operation and AOOs (item 2 above) and that fuel failures (and dose) are not underestimated for accidents. PNNL has reviewed the example DPC DNB propagation analysis for rod pressure for W reloads in the McGuire and Catawba plants (Reference 13). This analysis methodology was found to be consistent with W analysis methodology and, therefore, is acceptable for W reload applications..

PNNL concludes that the PAD 3.4 code and DPC analysis methodology are acceptable for evaluating rod internal pressures for W fuel reload applications.

2.5 Fuel Temperature

Bases/Criteria - The DPC fuel temperature limit precludes centerline pellet melting during normal operation and AOOs. This design limit is the same as given in the SRP and has been approved for application for W fuel designs up to a rod-average burnup level of 62 GWd/MTU (Reference 6). In order to ensure that this basis is met, DPC imposes a design limit on fuel temperatures such that there is at least a 95% probability at a 95% confidence level that during normal operation and AOO events the peak linear heat generation rate rod will not exceed the fuel melting temperature. W and DPC have placed a temperature limit on fuel melting at extended fuel burnup levels that have previously been approved for burnups up to 62 GWd/MTU. Therefore, PNNL concludes that DPC's design limit for fuel melting is acceptable for application to W fuel reload applications.

Evaluation - The PAD 3.4 fuel performance code (Reference 2) is used by DPC to assure that the fuel melting criterion is met. This code has been verified against fuel rod data with rod-average burnup levels up to approximately 62 GWd/MTU. DPC provided an example fuel melting analysis for W reloads in the McGuire and Catawba plants (Reference 14). These example DPC analyses are consistent with W analysis methodology.

There has been recent evidence of a decrease in fuel thermal conductivity with burnup; however, there remains a considerable uncertainty in this data and the NRC is still examining the implications for the fuel melting analysis. In addition, W states (Reference 14) that maximum fuel temperatures occur near beginning-of-life (BOL). Because NRC and industry are still evaluating the decrease in thermal conductivity with burnup, the current fuel thermal conductivity model in PAD 3.4 remains acceptable. Therefore, PNNL concludes that DPC's use of the PAD 3.4 code for the fuel melting analysis is acceptable for application to W fuel reload applications.

2.6 Fuel Clad Oxidation and Hydriding

Bases/Criteria - In order to preclude a condition of accelerated oxidation and cladding degradation, DPC imposes the W temperature limits on the cladding and a limit on hydrogen pickup in the cladding due to corrosion. The temperature limits applied to cladding oxidation are that calculated cladding temperatures (at the oxide-to-metal interface) shall be less than a specific (proprietary) value during steady-state operation and AOOs transients (a higher temperature limit is applied for AOOs transients). In addition, W has a limit on hydrogen pickup for the cladding. These criteria have been approved by NRC (Reference 10) up to a rod-average burnup limit of 62 GWd/MTU. Therefore, PNNL concludes that the DPC design criteria for oxidation and hydriding are acceptable for W reload applications.

Evaluation - The corrosion model in PAD 3.4 is used by DPC to assure that the W limits on cladding corrosion are met. DPC has provided an example cladding corrosion analysis for the cladding and assembly structural members for W reloads in the McGuire and Catawba plants (Reference 15). Similar to the rod internal pressure analysis, DPC uses a generic axial power shape for cladding corrosion. It is noted that cladding corrosion can also be sensitive to the axial power shape and, therefore, DPC will continue to confirm that the generic axial shapes remain applicable to the operation of each future fuel reload for corrosion analyses. The example DPC oxidation analysis has been reviewed and found to be consistent with the W analysis methodology. PNNL concludes that DPC's use of the PAD 3.4 code corrosion model is acceptable for evaluating corrosion for W fuel reload applications.

2.7 Fuel Rod Axial Growth

Bases/Criteria - Failure to adequately design for axial growth of the fuel rods can lead to fuel rod-to-nozzle gap closure resulting in fuel rod bowing and possible rod failure or failure of the thimble tubes. The DPC design limit is that the space between the rod end plug-to-end plug outer dimension and the lower nozzle-to-top adapter plate inner dimension shall be sufficient to preclude interference of these members.

This design limit has been accepted by the NRC for current W fuel designs up to a rod-average burnup limit of 62 GWd/MTU (Reference 6). Therefore, PNNL concludes that the DPC design limit for axial growth is acceptable for application to W fuel reload applications.

Evaluation - DPC uses the W correlations for rod and assembly growth and the W analysis methodology to evaluate the rod-to-nozzle clearance. The analysis methodology conservatively uses the upper-bound rod growth and lower bound assembly growth correlations along with the minimum rod-to-nozzle clearance based on a statistical combination of fabrication tolerances. The W rod and assembly growth correlations and analysis methodology have been approved by the NRC up to a rod-average burnup limit of 62 GWd/MTU.

DPC has provided an example rod-to-nozzle clearance analysis for W reloads in the McGuire and Catawba plants (Reference 16). This example DPC growth analysis is consistent with W analysis methodology. PNNL concludes that the DPC application of the W fuel rod and assembly growth correlations and analysis methods are acceptable for evaluating axial growth for W fuel reload applications.

3.0 CONCLUSIONS

PNNL concludes that the DPC design limits and thermal-mechanical analyses discussed in Section 4.0 of DPC-NE-2009P are acceptable for application by DPC to W fuel reloads up to the currently approved rod-average burnup limit of 62 GWd/MTU. In addition, the use of W growth models and analysis methodology discussed in the subject submittal are acceptable for application by DPC to W fuel reload applications up to currently approved burnups.

4.0 REFERENCES

1. Duke Power Company, July 1998. Duke Power Company Westinghouse Fuel Transition Report, DPC-NE-2009P (Proprietary), Duke Power Company, Charlotte, North Carolina.

2. Weiner, R. A., et al. August 1988. Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations. WCAP-10851-P-A (Proprietary), Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

3. U.S. Nuclear Regulatory Commission. July 1981. "Section 4.2, Fuel System Design." In Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition. NUREG-0800, Revision 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

4. U.S. Federal Register. "Appendix A, General Design Criteria for Nuclear Power Plants." In 10 Code of Federal Regulations, Part 50. U.S. Printing Office, Washington, D.C.

5. U.S. Federal Register. "Reactor Site Criteria." In 10 Code of Federal Regulations, Part 100. U.S. Printing Office, Washington, D.C.

6. Davidson, S. L. (Editor). April 1990. Westinghouse Fuel Criteria Evaluation Process. WCAP-12488, Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

7. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (All Units) -- Generic Stress and Fatigue Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0138 (Proprietary), Duke Power Company, Charlotte, North Carolina.

8. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (All Units) -- Generic Steady State Strain Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0135 (Proprietary), Duke Power Company, Charlotte, North Carolina.

9. O'Donnell, W. J., and B. F. Langer. 1964. "Fatigue Design Basis for Zircaloy Components." In Nuclear Science Engineering, Vol. 20, pp. 1.

10. Davidson, D. L., and J. A. Iorii. May 1982. Reference Core Report 17x17 Optimized Fuel Assembly. WCAP-9500-A (Proprietary), Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

11. Davidson, S. L., and D. L. Nuhfer (Editors). June 1990. VANTAGE+ Fuel Assembly Reference Core Report. WCAP-12610 (Proprietary), Westinghouse Electric Corporation, Pittsburgh, Pennsylvania.

12. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (Units 1 & 2) -- Generic Rod Internal Pressure Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0130 (Proprietary), Duke Power Company, Charlotte, North Carolina.

13. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (Units 1 & 2) -- Generic DNBRIP Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0140 (Proprietary), Duke Power Company, Charlotte, North Carolina.

14. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (Units 1 & 2) -- Generic Fuel Melt Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0133 (Proprietary), Duke Power Company, Charlotte, North Carolina.

15. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (All Units) -- Generic Westinghouse 17 x 17 Corrosion Analysis, DPC-1553.26-00-0125 (Proprietary), Duke Power Company, Charlotte, North Carolina.

16. Duke Power Company, Certification of Engineering Calculation -- McGuire and Catawba Nuclear Stations (Units 1 & 2) -- Generic Rod Growth Analysis for Westinghouse 17 x 17 Fuel, DPC-1553.26-00-0130 (Proprietary), Duke Power Company, Charlotte, North Carolina.

Section C

Duke Power Company Clarifications

1. August 17, 1999, letter from M. S. Tuckman to NRC
2. December 13, 1999, letter from M. S. Tuckman to NRC

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
(704) 382-4360 FAX

August 17, 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 and 50-370

Catawba Nuclear Station - Units 1 & 2
Docket Nos. 50-413 and 50-414

Update of Fuel Design Section of Topical Report DPC-NE-2009 (TAC MA2359, MA2361, MA2411, MA2412)

REFERENCE: 1. WCAP-12610-P-A, VANTAGE+ Fuel Assembly Reference Core Report, April 1995.

Attached are three updated pages for DPC-NE-2009, submitted July 22, 1998. These pages modify the Fuel Design and Thermal-Hydraulic Analysis sections of DPC-NE-2009 to reflect the use of the standard length Westinghouse fuel assembly design at McGuire and Catawba. Duke Power has decided, at the recommendation of Westinghouse, to use the standard length fuel design versus a reduced length assembly. This change was pursued to mitigate the recent (spring of 1999) problems identified with broken holddown screws on some Westinghouse fuel designs.

The change reverts to a previously approved fuel assembly design for overall dimensions shown in Reference 1. Therefore, this does not constitute a design change and no 10CFR50.59 evaluation is required. The net fuel assembly holddown forces are the same for both assembly lengths. Consequently, the robustness of the fuel design with respect to Incomplete Rod Insertion is identical.

The attached pages of DPC-NE-2009 were revised to reflect this change. The discussion of a shorter fuel assembly as a design feature was removed from Page 2-2. Additionally, the RFA length dimension on Table 2-1 (Page 2-4) and Table 5-1 (Page 5-6) was updated to the correct value.

The attached Page 2-4 of Topical Report DPC-NE-2009 contains information that Duke considers PROPRIETARY. In accordance with 10CFR 2.790, Duke requests that this information be withheld from public disclosure. A non-proprietary version of this page is included in the attachment. An affidavit which attests to the proprietary nature of the affected information is also included with this letter.

Any questions regarding these updates should be directed to J. S. Warren at (704) 382-4986.

Very truly yours,

M.S. Tuckman

M. S. Tuckman

MST/JSW

Attachment

xc w/Proprietary Attachment:

F. Rinaldi, NRC Project Manager (MNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

P. S. Tam, NRC Project Manager (CNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

xc w/o Proprietary Attachment:

L. A. Reyes, Regional Administrator
U.S. Nuclear Regulatory Commission, Region II
Atlanta Federal Center
61 Forsyth St., SWW, Suite 23T85
Atlanta, GA 30303

S. M. Shaeffer, NRC Senior Resident Inspector (MNS)

D. J. Roberts, NRC Senior Resident Inspector (CNS)

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.

M. S. Tuck

M. S. Tuckman

(Continued)

(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 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 Topical Report designated DPC-NE-2009P, *Duke Power Company Westinghouse Fuel Transition Report*, and omitted from the non-proprietary version. This topical report was submitted to the NRC by Duke letter dated July 22, 1998 and revised by Duke letter dated August 17, 1999. This information enables Duke to:

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

M. S. Tuck

M. S. Tuckman

(Continued)

(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.

(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.

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M. S. Tuck

M. S. Tuckman

(Continued)

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. Tuck

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. Tuck

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 23RD day of

August, 1998

Mary P. Nelms

Notary Public

My Commission Expires:

JAN 22, 2001

SEAL

bxc w/att:

L. B. Jones
R. M. Gribble
J. E. Smith
M. T. Cash
K. L. Crane
G. D. Gilbert
K. E. Nicholson
T. K. Pasour (2)
J. S. Warren
NRIA File/ELL

The RFA design used at McGuire and Catawba will include the following additional features to help mitigate debris failures:

- Pre-oxide coating on the bottom of the fuel rods and
- Protective bottom grid with longer fuel rod end-plugs.

The RFA design used at McGuire and Catawba will include the following feature to help mitigate Incomplete Rod Insertion (IRI):

- fuel rods positioned on the bottom nozzle

The three features listed above will be evaluated using the 10CFR50.59 process.

One new feature that will be added to the McGuire and Catawba RFA design is a Quick Release Top Nozzle (QRTN). This top nozzle design is similar to the Reconstitutable Top Nozzle (RTN) design, but has been modified for easier removal. This design change will be licensed by Westinghouse using the Fuel Criteria Evaluation Process (Reference 2-2) and notification will be made to the NRC.

The Westinghouse RFA is designed to be mechanically and hydraulically compatible with the FCF Mark-BW fuel (Reference 2-4) that is currently used at McGuire and Catawba. The basic design parameters of the RFA are compared to those of the Mark-BW fuel assembly in Table 2-1.

The IFM grids are non-structural members whose primary function is to promote mid-span flow mixing. Therefore, the design bases for the IFM grids are to avoid cladding wear and interactive damage with grids of the neighboring fuel assemblies during fuel handling. Westinghouse fuel with IFM grids has been flow tested both adjacent to another assembly with IFM grids and adjacent to an assembly without IFM grids. There was no indication of adverse fretting wear of the fuel rods by the standard structural or IFM grids (Reference 2-5). No adverse fretting wear is

Table 2-1

Comparison of Robust Fuel Assembly and Mark-BW Fuel Assembly Design Parameters

	17x17 Robust Fuel Assembly Design	17x17 Mark-BW Fuel Assembly Design
Fuel Assembly Length, in.		
Assembly Envelope, in.		
Fuel Rod Pitch, in.		
Fuel Rod Material		
Fuel Rod Clad OD, in.		
Fuel Rod Clad Thickness, in.		
Fuel/Clad Gap, mils		
Fuel Pellet Diameter, in.		
Fuel Stack Height, in.		
Guide Thimble Material		
Outer Diameter of Guide Thimbles, in. (upper part)		
Inner Diameter of Guide Thimbles, in. (upper part)		
Outer Diameter of Guide Thimbles, in. (lower part)		
Inner Diameter of Guide Thimbles, in. (lower part)		
Outer Diameter of Instrument Guide Thimbles, in.		
Inner Diameter of Instrument Guide Thimbles, in.		
End Grid Material		
Intermediate Grid Material		
Imtermediate Flow Mixing Grid Material		

Table 5-1

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)	160.0

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 Mixing (Non-structural)
Nozzles	304SS	1	Debris Filtering Bottom
	304SS	1	Removable Top

Duke Power
A Duke Energy Company

Duke Power Company
A Duke Energy Company
EC07H
526 South Church Street
P.O. Box 1006
Charlotte, NC 28201-1006

(704) 382-2200 OFFICE
(704) 382-4360 FAX

M. S. Tuckman
Executive Vice President
Nuclear Generation

December 13, 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 and 50-370

Catawba Nuclear Station - Units 1 & 2
Docket Nos. 50-413 and 50-414

Topical Report DPC-NE-2009 (TAC Nos. MA2359, MA2361, MA2411, MA2412), Update of Chapter 6.0, UFSAR Analyses

Topical Report DPC-NE-2009-P, *Duke Power Company Westinghouse Fuel Transition Report*, was approved by the NRC in an SER issued September 22, 1999. This report was originally submitted for NRC review on July 22, 1998. The approved version of this topical report is being edited and assembled for publication and submittal to the NRC. During the review of the report, several minor updates have been identified as being necessary for accuracy. This letter describes these updates and includes revised pages that will be incorporated in the final approved version of DPC-NE-2009-PA that will be submitted to the NRC. These updates are considered by Duke to not require NRC review and approval. They are being submitted for information only prior to publication. No response to this letter is requested.

Item #1: Section 6.2.2, Steam Line Break
The void models used in the VIPRE-01 code for the steam line break analysis methodology have been changed for both steam line break analyses, rather than just for the case with loss of offsite power. The justification for the

change in void models, presented in Chapter 5 of the report remains valid. This update is necessary due to additional analysis experience gained since the submittal of the report.

Item #2: Section 6.5.2, Large Break LOCA
The words " . . . typically with Moody break discharge coefficients, CD, of 0.4, 0.6 , and 0.8." have been deleted since this information is not required in the context of the paragraph.

Item #3: Section 6.5.2, Large Break LOCA
The words "Explicit analyses will be performed simulating . . ." have been replaced with "An evaluation will be performed to address . . ." since it has been determined that an evaluation rather than an explicit analysis is sufficient.

Future Transition to Westinghouse Best-Estimate LOCA Methodology
In addition, Duke will be making a future transition from the Westinghouse LOCA Evaluation Model (described in Chapter 6 of DPC-NE-2009) to Westinghouse's Best-Estimate LOCA Evaluation Methodology. This transition will not occur until after several reloads are analyzed with the LOCA methods as described in DPC-NE-2009. Duke will notify the NRC concerning the future application of the best-estimate LOCA methods. The DPC-NE-2009 topical will not be revised in the future to include the best-estimate LOCA methods since implementation of those methods will occur subsequent to the initial transition to Westinghouse fuel, which is the subject of the topical report.

Attachment A provides the proprietary version of the updates to DPC-NE-2009, and Attachment B provides the non-proprietary version. The updates will be included in the published versions of DPC-NE-2009-PA and DPC-NE-2009-A.

The attached pages of Topical Report DPC-NE-2009 contain information that Duke considers PROPRIETARY. In accordance with 10CFR 2.790, Duke requests that this information be withheld from public disclosure. A non-proprietary version of the affected pages is included in the attachment. An affidavit which attests to the proprietary nature of the applicable information is also included with this letter.

Any questions regarding these updates should be directed to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

MST/JSW

Attachment

xc w/Proprietary and Non-proprietary Attachments:

F. Rinaldi, NRC Project Manager (MNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

P. S. Tam, NRC Project Manager (CNS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

xc w/Non-proprietary Attachment:

L. A. Reyes, Regional Administrator
U.S. Nuclear Regulatory Commission, Region II
Atlanta Federal Center
61 Forsyth St., SWW, Suite 23T85
Atlanta, GA 30303

S. M. Shaeffer, NRC Senior Resident Inspector (MNS)

D. J. Roberts, NRC Senior Resident Inspector (CNS)

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.

M. S. Tuckman

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(Continued)

(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.

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

M. S. Tuckman

M. S. Tuckman

(Continued)

(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.

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M S Tuckman

M. S. Tuckman

(Continued)

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M. S. Tuck

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. Tuck
M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 14TH day of

December, 1998

Mary P. Nelms
Notary Public

My Commission Expires:

JAN 22, 2001

SEAL

bxc w/att:

L. A. Keller
G. B. Swindlehurst
R. M. Gribble
D. R. Koontz
M. T. Cash
K. L. Crane
G. D. Gilbert
K. E. Nicholson
T. K. Pasour (2)
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NRIA File/ELL

Attachment A

Updates to DPC-NE-2009-P Chapter 6
Proprietary Version

PROPRIETARY

The remainder of the steam line break thermal-hydraulic methodology presented in Reference 6-2 remains unchanged, except for the selection of subcooled and bulk void models for offsite power lost (OSPL) cases for reasons described in Chapter 5. The [EPRI/EPRI subcooled and bulk void combination will replace the Levy/Zuber-Findlay combination] for steam line break cases ~~for which offsite power is lost~~. This is acceptable since the [EPRI/EPRI combination] gives more conservative DNBR results for steady-state cases (according to Reference 6-1), and preliminary studies of steam line break cases show no difference in results.

6.2.3 Dropped Rod

The changes presented in Section 6.1 also apply to the dropped rod transient. The remainder of the dropped rod thermal-hydraulic methodology presented in Reference 6-2 remains unchanged.

6.3 UFSAR Chapter 15 System Transient Analysis Methodology (DPC-NE-3002)

DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology" (Reference 6-3) documents the conservative modeling assumptions used by Duke Power Company in performing the NSSS primary and secondary system analyses of UFSAR Chapter 15 accidents. It covers all applicable non-LOCA accidents in UFSAR Sections 15.1-15.6, except those already discussed in Reference 6-2. There are no changes to Reference 6-3 with respect to analyzing the RFA design.

6.4 Mass and Energy Release and Containment Response Methodology (DPC-NE-3004)

DPC-NE-3004-PA, "Mass and Energy Release and Containment Response Methodology" (Reference 6-4), describes the Duke Power Company methodology for simulating the mass and energy release from high energy line breaks (LOCA and steam line break) and the resulting containment response to demonstrate that the containment peak pressure and temperature limits are not exceeded. Since the fuel stored energy for the RFA design is similar to that for the Mark-BW fuel, there are no changes anticipated for Reference 6-4 with respect to the RFA design except the RETRAN related changes described in Section 6.1 of this report. Similar changes to

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BASH Evaluation Model as reported in Reference 6-37. In addition, the LOTIC code has been coupled with the BASH code so that the codes run interactively. The BASH Evaluation Model now utilizes the SATAN code for the blowdown calculations, the BASH code for the refill and reflood phases with interactive LOTIC calculations for containment backpressure, and the LOCBART code for the fuel rod heatup calculations. The most recent version of the LOCBART code employs an improved grid heat transfer model which has been approved by the by NRC in Reference 6-38.

An input parameter that affects LOCA analysis results is the assumed axial power shape at the beginning of the accident. The methodology employed by Westinghouse is termed ESHAPE (Explicit SHape Analysis for Pct Effects). The ESHAPE methodology is based upon explicit analysis of the LBLOCA transient with a set of bounding skewed axial power shapes to supplement the base analysis performed with the chopped cosine power shape. The limiting case break, as demonstrated with a chopped cosine, will be reanalyzed using skewed power shapes and typically demonstrate that the chopped cosine power shape is limiting.

As required in Appendix K to 10 CFR 50, a minimum of a three break spectrum will be analyzed. ~~typically with Moody break discharge coefficients, CD, of 0.4, 0.6, and 0.8.~~ In addition, as required in the NRC Safety Evaluation Report (SER) for the BASH Evaluation Model, a maximum Safety Injection flow case will be analyzed.

When assessing the effect of transition cores on the LBLOCA analysis, it must be determined whether the transition core can have a greater calculated peak cladding temperature (PCT) than a complete core of the RFA design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. Hydraulic resistance mismatch will exist only for a transition core and is the only unique difference between a complete core of either fuel type and the transition core. ~~Explicit analyses will be performed simulating~~ the cross-flow effects due to any hydraulic mismatch between the current fuel and the Westinghouse fuel. If it is determined that a transition core penalty is required during the cycles that both fuels reside in the core, it will be applied as an adder to the LOCA results for a full core of the RFA design.

An evaluation will be performed to address

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Attachment B

Updates to DPC-NE-2009 Chapter 6
Non-Proprietary Version

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DPC-NE-2009, Rev. 1

Section D

Topical Report

(includes updated pages given in Section C)

NON-PROPRIETARY
Duke Power Company

DPC-NE-2009, Rev. 1

DUKE POWER COMPANY WESTINGHOUSE FUEL TRANSITION REPORT

Original Version: July 1998

Approved Version: December 1999

Revision 1: April 2001

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

Duke Power Company Westinghouse Fuel Transition Report

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1.0 INTRODUCTION

Duke Power Company is currently using Framatome Cogema Fuels (FCF) Mark-BW fuel assemblies in the McGuire and Catawba reactors. Duke Power will transition to the 17x17 Westinghouse 0.374 Robust Fuel Assembly (RFA) design described in Chapter 2 of this report. This topical report presents the information required to support the licensing basis for the use of the RFA design in McGuire and Catawba reload cores.

This report describes the core design, fuel rod design, and thermal-hydraulic analyses that are performed to show that all licensing criteria are met for each reload core. This report also discusses the UFSAR Chapter 15 transient and accident analyses methodology that is applicable to each reload design. Previously approved methodologies used by Duke Power Company to perform core design, thermal-hydraulic design, and UFSAR Chapter 15 Non-LOCA analyses for the Mark-BW fuel will be used to analyze the RFA design with the revisions described in Chapters 3, 5, and 6, respectively.

Chapter 4 describes the fuel rod design analysis methodology that will be used to analyze the RFA design. Although the fuel rod analysis methodology is new for Duke Power, the methods are essentially identical to the NRC-approved Westinghouse methods. The Westinghouse LOCA analysis methodology is described in Section 6.5. Section 6.6 presents an improved methodology that will be used to perform the nuclear analysis portion of the rod ejection accident (REA) analysis for McGuire and Catawba. The new methodology is based on the SIMULATE-3K computer code.

Chapter 7 discusses the licensing and analysis approach Duke Power will use for reconstitution of the RFA design. Chapter 8 describes the Technical Specification changes that will be made due to the transition to the RFA design and the analysis methodology described in this report.

2.0 FUEL DESIGN

Duke Power is transitioning to the Westinghouse 17x17 0.374 robust fuel assembly design for the McGuire and Catawba reactors. For the remainder of this report the fuel design will be referred to as simply the RFA design. The RFA design is based on the VANTAGE + fuel assembly design, licensed by the NRC in Reference 2-1. The RFA design used at McGuire and Catawba will include the following features initially licensed with the VANTAGE + fuel design:

- ZIRLO™ clad fuel rods,
- ZIRLO™ guide thimbles, instrumentation tubes and mid-grids (both structural and Intermediate Flow Mixing (IFM) grids),
- 0.374 inch fuel rod OD,
- Zirconium diboride Integral Fuel Burnable Absorbers (IFBAs),
- Mid-enriched annular axial blanket pellets,
- High burnup fuel skeleton, and
- Debris Filter Bottom Nozzle (DFBN).

In addition to the VANTAGE + fuel design features listed above, the RFA design used at McGuire and Catawba will incorporate the following features that were licensed using the Fuel Criteria Evaluation Process (Reference 2-2) via Reference 2-3:

- Increased guide thimble and instrumentation tube OD (0.482 inch),
- Modified Low Pressure Drop (MLPD) structural mid-grids, and
- Modified Intermediate Flow Mixing (MIFM) grids.

The RFA design used at McGuire and Catawba will include the following additional features to help mitigate debris failures:

- Pre-oxide coating on the bottom of the fuel rods and
- Protective bottom grid with longer fuel rod end-plugs.

The RFA design used at McGuire and Catawba will include the following feature to help mitigate Incomplete Rod Insertion (IRI):

- fuel rods positioned on the bottom nozzle

The three features listed above will be evaluated using the 10CFR50.59 process.

One new feature that will be added to the McGuire and Catawba RFA design is a Quick Release Top Nozzle (QRTN). This top nozzle design is similar to the Reconstitutable Top Nozzle (RTN) design, but has been modified for easier removal. This design change will be licensed by Westinghouse using the Fuel Criteria Evaluation Process (Reference 2-2) and notification will be made to the NRC.

The Westinghouse RFA is designed to be mechanically and hydraulically compatible with the FCF Mark-BW fuel (Reference 2-4) that is currently used at McGuire and Catawba. The basic design parameters of the RFA are compared to those of the Mark-BW fuel assembly in Table 2-1.

The IFM grids are non-structural members whose primary function is to promote mid-span flow mixing. Therefore, the design bases for the IFM grids are to avoid cladding wear and interactive damage with grids of the neighboring fuel assemblies during fuel handling. Westinghouse fuel with IFM grids has been flow tested both adjacent to another assembly with IFM grids and adjacent to an assembly without IFM grids. There was no indication of adverse fretting wear of the fuel rods by the standard structural or IFM grids (Reference 2-5). No adverse fretting wear is expected in transition cores with the Westinghouse RFA design and Mark-BW fuel since the Mark-BW fuel is very similar to Westinghouse fuel assembly designs without IFM grids.

2.1 References

2-1 S. L. Davidson & T. L. Ryan, "Vantage+ Fuel Assembly Reference Core Report", WCAP-12610-P-A, April 1995.

2-2 S. L. Davidson (Ed.), "Westinghouse Fuel Criteria Evaluation Process", WCAP-12488-P-A, October 1994.

2-3 NSD-NRC-97-5189, 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.

2-4 "Mark-BW Mechanical Design Report", BAW-10172P-A, December 1989.

2-5 S. L. Davidson (Ed.), "Reference Core Report Vantage 5 Fuel Assembly", WCAP-10444-P-A, September 1985.

Table 2-1

Comparison of Robust Fuel Assembly and Mark-BW Fuel Assembly Design Parameters

	17x17 Robust Fuel Assembly Design	17x17 Mark-BW Fuel Assembly Design
Fuel Assembly Length, in.		
Assembly Envelope, in.		
Fuel Rod Pitch, in.		
Fuel Rod Material		
Fuel Rod Clad OD, in.		
Fuel Rod Clad Thickness, in.		
Fuel/Clad Gap, mils		
Fuel Pellet Diameter, in.		
Fuel Stack Height, in.		
Guide Thimble Material		
Outer Diameter of Guide Thimbles, in. (upper part)		
Inner Diameter of Guide Thimbles, in. (upper part)		
Outer Diameter of Guide Thimbles, in. (lower part)		
Inner Diameter of Guide Thimbles, in. (lower part)		
Outer Diameter of Instrument Guide Thimbles, in.		
Inner Diameter of Instrument Guide Thimbles, in.		
End Grid Material		
Intermediate Grid Material		
Imtermediate Flow Mixing Grid Material		

3.0 CORE DESIGN

3.1 Introduction

The nuclear characteristics of the Westinghouse RFA design and the Mark-BW fuel design are almost identical due to similar dimensional characteristics of the fuel pellet, fuel rod and cladding. As a result, the methods and core models used to perform transition and full core analyses of the Westinghouse RFA design are the same as those currently licensed and employed in reload design analyses for McGuire and Catawba.

3.2 Reload Design Methodology

The development of core models, core operational imbalance limits and the evaluation of key physics parameters used to confirm the acceptability of UFSAR Chapter 15 accidents will be performed in compliance with the approved methodology defined in References 3-1 through 3-4. Conceptual transition core designs using the Westinghouse RFA design have been evaluated and show that current reload limits remain bounding with respect to key physics parameters. In the event that one of the key parameters is exceeded, the evaluation process described in Reference 3-3 would be performed.

The introduction of the Westinghouse RFA design is not expected to change the magnitude of the nuclear uncertainty factors described in Reference 3-1. However, the use of zirconium di-boride Integral Fuel Burnable Absorbers (IFBA) is a fuel design change which is different from the burnable absorber types modeled in Duke's current benchmarking database. The NRC SER for Reference 3-1 requires Duke to re-benchmark the nuclear code package and assure that the nuclear uncertainties remain appropriate for significant changes in fuel design. While the introduction of the IFBA burnable absorber is not considered significant, the nuclear uncertainties in Reference 3-1 were re-evaluated and confirmed to be bounding.

Duke explicitly modeled Seqouyah Unit 2 Cycles 5, 6, and 7 and performed statistical analysis of the nuclear uncertainty factors as described in Reference 3-1. 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 of the statistical analysis are shown in Table 3-1 and show that the current licensed nuclear uncertainty factors bound those for the Westinghouse fuel with a combination of IFBA and/or WABA burnable absorbers. Boron concentrations, rod worths, and isothermal temperature coefficients were also predicted and found to agree well with the measured data. A 10CFR50.59 USQ evaluation has been performed to demonstrate that the currently approved CASMO-3/SIMULATE-3P methods and nuclear uncertainties are applicable to the Westinghouse RFA design described in this report.

In all nuclear design analyses, both the Westinghouse RFA and the Mark-BW fuel are explicitly modeled in the transition cores. When establishing Operating and RPS limits (i.e. LOCA kw/ft, DNB, CFM, transient strain), the fuel specific limits or a conservative overlay of the limits are used.

The nuclear design related Technical Specification limits were reviewed for transition and full core reloads comprised of the Westinghouse RFA design. The only change required to the Technical Specifications is to replace the factor used to account for possible increases in $F\Delta H$ and Fq between flux maps with a burnup dependent factor (see Chapter 8 for additional details).

In summary, the steady-state physics codes, methodology and nuclear uncertainty factors remain unchanged for the transition to the Westinghouse RFA design. The evaluation of conceptual core designs with the RFA design indicate that key physics parameters assumed in the UFSAR Chapter 15 accident analyses remain bounding. The introduction of the IFBA burnable poison design will require that the factor used to account for the possible increase in peaking over a 31 EFPD surveillance period be replaced by a burnup dependent factor (see Chapter 8).

3.3 References

3-1 "Design Methodology Using CASMO-3/SIMULATE-3P," DPC-NE-1004A (Revision 1), SER dated April 26, 1996.

3-2. "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," DPC-NE-2011PA, March 1990.

3-3 "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology," DPC-NE-3001-PA, November 1991.

3-4 "Duke Power Company McGuire Nuclear Station, Catawba Nuclear Station Nuclear Physics Methodology," DPC-NF-2010A, June 1985.

Table 3-1

Nuclear Uncertainty Factors

(Statistically combined factors without Engineering Hot Channel Factor)

Parameter	Westinghouse Fuel with IFBA/WABA	DPC-NE-1004A
$F_{\Delta h}$	1.027	1.028
F_z	1.049	1.053
F_q	1.049	1.061

4.0 FUEL ROD ANALYSIS

This chapter describes Duke Power's fuel rod mechanical reload analysis methodology for Westinghouse fuel. The fuel rod analysis methodology discussed in this Chapter is essentially identical to Westinghouse's approved methodology. The analyses will be performed using the NRC approved Westinghouse fuel performance code, PAD, described in Section 4.1. Fuel rod mechanical analyses for Mark-BW fuel at McGuire and Catawba will continue to be performed using the NRC-approved methodology given in Reference 4-12.

The fuel rods are designed to meet the requirements of 10CFR50, Appendix A, "General Design Criteria" (Reference 4-1), specifically Criterion 10 "Reactor Design", which states: "The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation including the effects of anticipated operational occurrences."

To meet this requirement and the requirements of Section 4.2 of the Standard Review Plan (SRP) (Reference 4-2), Westinghouse has established specific fuel design criteria associated with Condition I and II operation (Reference 4-3). Section 4.2 of this report describes each of the fuel rod design criteria which are evaluated as required by SRP 4.2 for Condition I and II operation. A description of the fuel rod analysis methodology which is used to show that the design criteria are met each cycle is also provided.

Detailed fuel rod design analyses consider parameters such as the pellet/clad diametral gap, the size and density of the pellet, the gas plenum volume, and the helium prepressurization. Using the approved fuel performance models in PAD (Reference 4-4), the analyses also consider effects such as fuel densification and swelling, cladding creep, cladding corrosion, fission gas release and other physical properties which vary with burnup. The integrity of the fuel rods is ensured by designing the rods and operating the core to prevent excessive fuel temperatures, excessive fuel rod internal gas pressures, and excessive cladding stresses and strains. This is achieved by verifying that the conservative design criteria described in Section 4.2 are satisfied during Condition I and II events over the life of the fuel.

The fuel rod analyses must consider the uncertainties associated with design models and variations in as-built dimensions. Due to the empirical basis of the performance models used in the design codes (e.g., fission gas release, clad creep, etc.), there is variability in the data used for model validation. To have confidence that the extremes of the performance spectrum are covered, deviations from best estimate model projections must be accounted for. Each model which has a significant effect on fuel rod performance includes uncertainty bands defined to bound 95 % of the data. These uncertainty bands are used to define conservative upper bound uncertainty levels in the model predictions. These uncertainty levels are considered in the fuel rod analyses, assuring that all fuel rods in a core will satisfy the design criteria.

The fuel rod analyses also consider the variations in rod dimensions and fuel fabrication characteristics. Typically drawing tolerances which are assumed to represent at least a 2 sigma bound are used in fuel rod analyses. Actual as-built measurements and bounding values based on measured standard deviations may be used for critical fuel parameters. The typical method for including model, rod dimension, and fuel chararcteristic uncertainties is by statistical convolution.

The fuel rod for the RFA design is identical to the fuel rod for the VANTAGE+ design, thus the licensed pin burnup for the Westinghouse RFA design is 60,000 MWd/mtU (Reference 4-3). Using the Westinghouse Fuel Criteria Evaluation Process (FCEP) (Reference 4-13), the burnup limit can be increased to 62,000 MWd/mtU for specific reload cores.

Fuel rod analyses or evaluations to verify that a generic analysis is applicable must be performed for each reload cycle. Typically, generic analyses are completed that are expected to envelope the operation of future fuel cycles. The generic fuel rod analyses are then shown to be valid for each reload cycle design. This chapter describes the generic fuel rod analysis methods. In most cases, the generic analyses are bounding for each fuel cycle design and no new analyses are required. Cycle specific fuel rod analyses may be performed to obtain additional margin.

4.1 Computer Code

The PAD fuel performance code (Reference 4-4) is the main code used for evaluating fuel rod performance. PAD iteratively calculates the interrelated effects of temperature, pressure, cladding elastic and plastic behavior, cladding corrosion, fission gas release, and fuel densification and swelling as a function of time and power. PAD evaluates the power history of a rod as a series of steady-state power levels with instantaneous changes from one power level to another.

PAD divides the fuel rod into several axial segments and each segment is assumed to operate at a constant set of conditions over its length. Fuel densification and swelling, cladding stresses and strains, temperatures, burnup and fission gas release are calculated separately for each axial segment and the effects are integrated to obtain the overall fission gas release and rod internal pressure. The coolant temperature rise along the rod is calculated based on the flow rate and axial power distribution and the cladding surface temperature is calculated considering the effects of corrosion and the possibility of local boiling.

PAD considers the fuel pellet as a solid cylinder with allowances for dishing, chamfering, and pellet chipping. To calculate thermal expansion, fuel densification and swelling, and fission gas release, the pellet is divided into equal volume concentric rings and each ring is assumed to be at its average temperature during a given time step. Axial and radial thermal expansion, swelling and densification are determined for each ring and these effects are integrated over the entire fuel rod to calculate the length of the fuel column and the void volume to calculate the rod internal pressure.

The current version of the PAD code is PAD 3.4 (Reference 4-4). This version of the code includes an updated fission gas release model, fuel densification and swelling models, and cladding creep model. The PAD code has been certified for use in safety-related analyses according to Duke Power's Quality Assurance program. When any new versions of the PAD code are submitted to the NRC by Westinghouse, Duke Power plans to use the new version after it is approved for licensing analyses.

4.2 Fuel Rod Design Bases and Analyses

The design bases for the RFA design that will be used in McGuire and Catawba are identical to those given in Reference 4-3 for Vantage+ fuel. The fuel rod design bases and analysis methodologies are described below.

4.2.1 Fuel Rod Internal Pressure

The fuel rod internal pressure design basis is that the fuel system will not be damaged due to excessive fuel rod internal pressure (Reference 4-3 and 4-6). The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause (1) the diametral gap to increase due to outward clad creep during steady-state operation and (2) extensive DNB propagation to occur.

4.2.1.1 Analysis

Part 1 of this design basis precludes the cladding outward creep rate from exceeding the fuel solid swelling rate, and, thus, ensures that during steady-state operation the fuel-cladding gap will not re-open following contact, or increase in size. The PAD code is used to predict fuel rod internal pressures that are used to verify that the fuel rod internal pressure design basis is met. The rod average burnup at which the diametral gap begins to increase due to the outward cladding creep rate is calculated. This allowable rod burnup is compared to predicted rod burnups for each reload design to confirm that the rod internal pressure criterion is met for all of the fuel.

A bounding pin power history, similar to that shown in Fig. 4-1, is used to perform a generic rod internal pressure analysis. A cycle-specific rod internal pressure analysis may be performed using predicted limiting pin power histories if the bounding power history does not envelope the pin powers for a future core design. The transient gas release contribution to the rod internal pressure must be included in the rod internal pressure analyses. Both Condition I axial xenon oscillations and Condition II overpower transients are considered in calculating the rod internal pressure.

Sensitivity studies have been performed to determine the design parameters and PAD models which are the most significant contributors to the uncertainty in the rod internal pressure. An upper bound rod internal pressure is calculated to account for the impact of possible variations in design parameters or models. The bounding pressure is compared to a lower bound steady-state pressure limit.

Part 2 of the rod internal pressure design basis deals with DNB propagation, which is discussed in Reference 4-6. The current methodology for calculating the frequency and expected location of fuel rods experiencing both DNB and internal pressure greater than the reactor coolant system pressure is consistent with that used for the evaluations documented in Reference 4-6. For each rod that is both in DNB and above system pressure, the number of additional rods in DNB due to propagation effects are calculated based on whether the neighboring rods are in DNB or above system pressure. A fuel rod which is both in DNB and above system pressure is assumed to balloon at the location of DNB. When the ballooned clad contacts its neighboring rods, it is assumed that these rods will also experience DNB as a result of the flow blockage. If one of these rods is also above system pressure, it would also balloon to contact its neighboring rods. This process is assumed to continue if any of the neighbor rods are above system pressure. The total number of rods in DNB initially, rods above system pressure, rods both in DNB and above system pressure, and rods in DNB due to propagation are calculated.

4.2.2 Cladding Stress

The cladding stress design basis is the fuel system will not be damaged due to excessive fuel cladding stress (Reference 4-3 and 4-9). The volume average effective stress calculated with the Von Mises equation considering interference due to uniform cylindrical pellet cladding contact, caused by thermal expansion, pellet swelling and uniform cladding creep, and pressure differences, is less than the ZIRLO™ 0.2 % offset yield stress, with due consideration of temperature and irradiation effects under Condition I and II modes of operation. While the cladding has some capability for accommodating plastic strain, the yield stress has been established as a conservative design limit.

4.2.2.1 Analysis

Excessive clad stress can arise due to rapid local power increases such that clad creep cannot accommodate the pellet thermal expansion. The clad stress criterion is applied to the volume average effective stress which occurs as a result of a Condition II transient local power increase. The primary mechanism which increases the clad stresses during a Condition II transient, relative to the steady-state stresses, is the differential thermal expansion between the pellet and the cladding.

For each reload design, the allowable changes in local linear heat rate (Δ kw/ft) as a function of burnup are compared to predicted peaking changes that result from either Condition I or II events.

4.2.3 Cladding Strain

The cladding strain design basis is that the fuel system will not be damaged due to excessive fuel cladding strain (Reference 4-3 and 4-9). The design limit is that during steady-state operation, the total plastic tensile creep strain due to uniform cladding creep and uniform fuel pellet expansion associated with fuel swelling and thermal expansion is less than 1% from the unirradiated condition. The acceptance limit for fuel rod cladding strain during Condition II events is that the total tensile strain due to uniform cylindrical pellet thermal expansion is less than 1% from the pre-transient value (Reference 4-2).

4.2.3.1 Analysis

The intent of this criterion is to minimize the potential for clad failure due to excessive clad straining. This criterion addresses slow strain rate mechanisms where the effective clad stress never reaches the yield strength due to stress relaxation. Clad strain allowable local power limits (Δ kw/ft) are calculated using PAD and the methodology discussed above for calculating clad stress local power limits. Analyses have generally shown that the transient clad stress analyses are more limiting than the transient clad strain analyses (i.e., the clad stress Δ kw/ft limits are typically more restrictive than the clad strain Δ kw/ft limits).

4.2.4 Cladding Fatigue

The cladding fatigue design basis is that the fuel system will not be damaged due to excessive clad fatigue (Reference 4-3 and 4-9). The fatigue life usage factor is limited to less than 1.0 to prevent reaching the material fatigue limit.

4.2.4.1 Analysis

A cladding fatigue analysis is performed to consider the accumulated effects of short term, cyclic, cladding stress and strain resulting primarily from daily load follow operation. The accumulated effects of cyclic strains associated with normal plant shutdowns and returns to full power are also considered.

The fatigue model in PAD calculates the low cyclic fatigue and the fatigue life fraction of a fuel rod during load follow operation, as a function of time and irradiation history. The Langer-O'Donnell low cyclic fatigue model (Reference 4-7) constitutes the basic approach used in the fatigue analysis. The empirical factors used in the Langer-O'Donnell fatigue model have been modified to conservatively bound the results of Westinghouse test programs presented in Reference 4-8. The design equations follow the concepts of the fatigue design criterion given in the ASME Code, Section III:

The calculated pseudo-stress amplitude (S_a) is multiplied by 2 to obtain the allowable number of cycles (N_f)

The allowable cycles for a given S_a is five percent of N_f or a safety factor of 20 on the number of cycles.

The lower of the two allowable number of cycles is selected and the cumulative fatigue life fraction is then calculated as:

$$n_k/N_{fk} < 1.0$$

where:

N_k = number of cycles of mode k

N_{fk} = number of allowable cycles

PAD is used to analyze a spectrum of pin power histories to determine the fatigue life.

4.2.5 Fuel Clad Oxidation and Hydriding

The fuel clad oxidation and hydriding design basis is that fuel damage will not occur due to excessive clad oxidation or hydriding (Reference 4-3). To limit metal-oxide formation to acceptable values, the ZIRLO™ metal-oxide interface temperature is limited] (Reference 4-3). The clad and structural component hydrogen pickup is limited to [ ] (Reference 4-3) at end of life to preclude loss of ductility due to hydrogen embrittlement by the formation of zirconium hydride platelets.

4.2.5.1 Analysis

A spectrum of pin power histories, including a bounding power history similar to that shown in Fig. 4-1, are analyzed to verify that the cladding metal-oxide interface temperature limits are met during steady-state operation and during Condition II local power increases. For each steady-state power history, the temperature of the metal-oxide interface is calculated. The oxide layer on the fuel is calculated using the ZIRLO™ corrosion model described in Reference 4-3. At various times during the steady-state depletion, Condition II local power increases are simulated. The local power is increased until the cladding metal-oxide interface temperature is equal to the transient cladding temperature limit. An analysis is performed for each reload which verifies that the local power limit associated with the transient cladding temperature limit is not exceeded during Condition II events (Reference 4-11).

The methodology for calculating the hydrogen pickup of the cladding is the same as that described above for calculating the metal-oxide interface temperature. In addition to the zirc-oxide buildup on the cladding, the hydrogen pickup resulting from the corrosion process is calculated. Corrosion and percent metal wastage for the grids and thimbles is also calculated.

4.2.6 Fuel Temperature

The fuel temperature design basis is that fuel rod damage will not occur due to excessive fuel temperatures (Reference 4-3). The fuel system and protection system are designed to assure that for Condition I and II events, the calculated centerline fuel temperature does not exceed the fuel melting temperature. The melting temperature of unirradiated UO_2 is taken as 5080 °F, decreasing by 58 °F per 10,000 MWd/mtU of fuel burnup (Reference 4-3). A centerline fuel temperature of 4700 °F has been selected by Westinghouse as the design limit for fuel temperature analyses, References 4-9 and 4-10.

4.2.6.1 Analysis

The PAD 3.4 code (Reference 4-4) is used to verify that the fuel temperature design limit is met. Using a fuel centerline temperature limit of 4700 °F covers both the reduction in melt temperature with burnup and manufacturing and modeling uncertainties. PAD is used to calculate the fuel centerline temperature and the local linear heat rate to prevent fuel melting or linear heat rate to melt (LHRTM). As explained in Reference 4-11 an analysis is performed for each reload which verifies that this local power limit is not exceeded for Condition I and II events.

4.2.7 Fuel Clad Flattening

From Reference 4-3, the design basis for fuel clad flattening is that fuel rod failures will not occur due to clad flattening.

4.2.7.1 Analysis

Westinghouse demonstrated in Reference 4-5 that clad flattening will not occur for current Westinghouse fuel designs. Based on post irradiation examination and in-core flux data Westinghouse confirmed that significant axial gaps in the fuel column due to densification will not occur for current Westinghouse fuel. Therefore, it was concluded that clad flattening will not occur.

A new clad flattening evaluation is required only if any of the following fuel rod design parameters change: cladding creep properties, cladding thickness, fuel densification, rod prepressure, and as-fabricated pellet-clad gap. All of these parameters are related to the fuel design itself; they are not affected by a particular reload core design. For each new region of fuel; the cladding thickness, fuel rod prepressure, and as-fabricated pellet-clad gap will be verified to be within the range of parameters considered in Reference 4-5.

4.2.8 Fuel Rod Axial Growth

From Reference 4-3, the fuel rod growth design basis is that the fuel rods will be designed with adequate clearance between the fuel rod end plugs and the top and bottom nozzles to accommodate the difference in the growth of the fuel rods and the growth of the fuel assembly. The Westinghouse RFA was designed to assure that there is no interference between the fuel rods and the fuel assembly top and bottom nozzles during the design life of the fuel.

4.2.8.1 Analysis

The fuel rod growth model described in Reference 4-4 is used to show that the fuel rod growth criterion is met. The rod growth analysis assumes upper bound fuel rod growth, lower bound fuel assembly growth, minimum initial fuel rod to nozzle gap, upper bound rod fast fluence, and nominal differential thermal expansion between the fuel rod cladding and the fuel assembly structure. A generic analysis is performed to calculate the maximum allowable rod average burnup for which the rod to nozzle gap is zero. For the current RFA design, the allowable rod burnup with respect to the rod growth criterion is greater than the licensed burnup limit of 60,000 MWd/mtU. Using the Westinghouse Fuel Criteria Evaluation Process (FCEP) (Reference 4-13), the burnup limit can be increased to 62,000 MWd/mtU for specific reload cores.

4.3 References

4-1 Title 10, Chapter 1, Code of Federal Regulations - Energy, Part 50, "Domestic Licensing of Production and Utilization Facilities", Appendix A, "General Design Criteria for Nuclear Power Plants".

4-2 "Section 4.2, Fuel System Design", Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, NUREG-0800, Rev. 2, US Nuclear Regulatory Commission, July 1981.

4-3 S. L. Davidson & T. L. Ryan, "Vantage+ Fuel Assembly Reference Core Report", WCAP-12610-P-A, April 1995.

4-4 Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations", WCAP-10851-P-A, August 1988.

4-5 P. J. Kersting, et al., "Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel", WCAP-13589-A, March 1995.

4-6 Risher, D., et al., "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis", WCAP-8963-P-A, August 1978.

4-7 W. J. O'Donnell and B. F. Langer, "Fatigue Design Basis of Zircaloy Components", Nuclear Science and Engineering, 20, 1-12, 1964.

4-8 S. L. Davidson and J. A. Iorii, "Reference Core Report 17x17 Optimized Fuel Assembly", WCAP-9500-P-A, May 1982.

4-9 S. L. Davidson (ed.), et al., "Extended Burnup Evaluation of Westinghouse Fuel", WCAP-10125-P-A, December 1985.

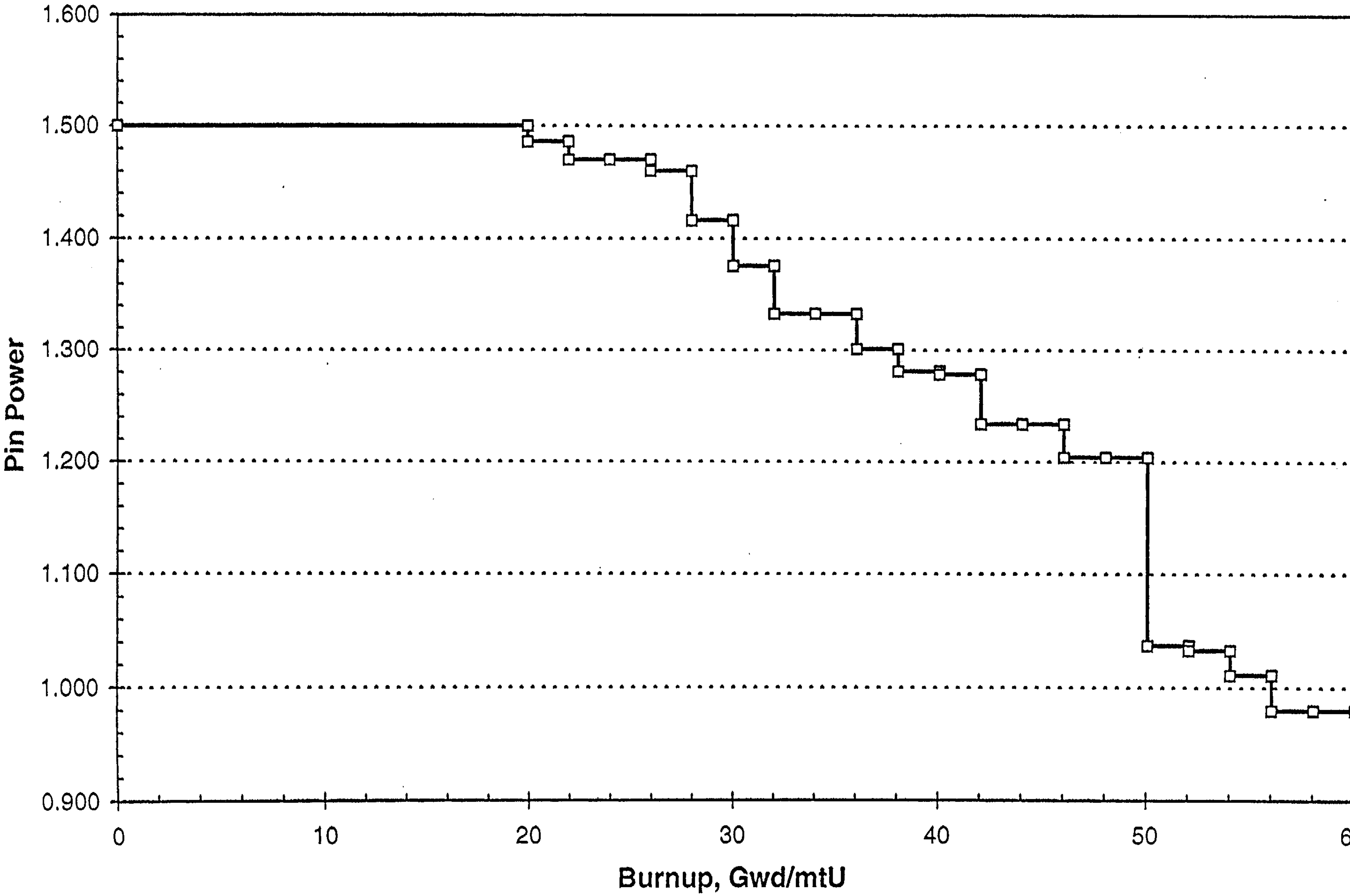
4-10 S. L. Ellenberger, et al., "Design Bases for the Thermal Overpower Delta-T and Thermal Overtemperature Delta-T Trip Functions", WCAP-8745-P-A, September 1986.

4-11 "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", DPC-NE-2011P-A, March 1990.

4-12 "Duke Power Company Fuel Rod Mechanical Reload Analysis Methodology Using TACO3", DPC-NE-2008P-A, SER dated April 3, 1995.

4-13 S. L. Davidson (Editor), "Westinghouse Fuel Criteria Evaluation Process", WCAP-12488P-A, October 1994.

Fig. 4-1 Typical Bounding Pin Power History



5.0 THERMAL-HYDRAULIC ANALYSIS

Steady-state thermal-hydraulic analyses for the Westinghouse RFA design will be performed using the NRC approved methodology given in References 5-1 and 5-4. Reference 5-1 describes the VIPRE-01 core thermal-hydraulic models used for steady state analyses at McGuire and Catawba. The only changes necessary to perform core thermal-hydraulic analyses for the Westinghouse RFA design are to specifically model the fuel (dimensions, form loss coefficients, etc.) and to use the WRB-2M critical heat flux (CHF) correlation (Reference 5-2). The RFA design, VIPRE-01 models, and the WRB-2M CHF correlation are discussed in Sections 5.1, 5.2, and 5.3, respectively.

DPC-NE-2005P-A (Reference 5-4) describes Duke Power's NRC-approved methodology for calculating a Statistical Core Design (SCD) DNBR limit for application to pressurized water reactors. Individual appendices to the report list information necessary to complete the calculations for specific plants and fuel types. This includes the fuel data for the VIPRE-01 model, parameter uncertainties, the CHF correlation, and the range of conditions analyzed. The remainder of Chapter 5 is written in the same format as an appendix to Reference 5-4. Sections 5.1 through 5.3 list the plant specific data, models, and CHF correlation. Section 5.4 lists the range of statepoint conditions analyzed and Section 5.5 describes the key parameters and associated uncertainties. The statistical design limit, or SDL, which will be used for licensing analyses for Westinghouse Robust fuel at McGuire and Catawba is discussed in Section 5.6. Section 5.7 discusses how the impact of the geometric and hydraulic differences between the resident Mark-BW fuel and the Westinghouse RFA design is addressed and determines the SDL for RFA/Mark-BW transition cores.

Unless otherwise noted, all VIPRE-01 modeling inputs listed in Reference 5-1 for the 17x17 fuel at McGuire and Catawba are unchanged. The thermal-hydraulic SCD analysis discussed in this chapter was performed using the approved methodology given in the main body of Reference 5-4.

5.1 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. See Chapter 2 of this report for a complete description of the fuel design.

The parameter uncertainties and statepoint ranges were selected to bound the McGuire and Catawba unit and cycle-specific values (see Sections 5.4 and 5.5).

5.2 Thermal-Hydraulic Code and Model

The VIPRE-01 thermal-hydraulic computer code described in Reference 5-3 and the McGuire/Catawba eight channel model approved in Reference 5-1 are used in this analysis. The reference pin power distribution based on a 1.60 peak pin from Reference 5-1 was used. The VIPRE-01 models approved in Reference 5-1 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 5-1. 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.

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 5-3). 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 5-3). 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 5-1, 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 5-2) 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.

5.3 Critical Heat Flux Correlation

The WRB-2M critical heat flux correlation described in Reference 5-2 is used for all statepoint analyses. This correlation was developed by Westinghouse for application to the RFA design. As discussed in Reference 5-2 the WRB-2M correlation was developed with the VIPRE-01 thermal-hydraulic computer code. This correlation was programmed into the Duke Power version of VIPRE-01 and will be used in all DNBR calculations for the RFA design, except for the steam line break transient (see Section 6.2.2).

5.4 Statepoints

The statepoint conditions evaluated in this analysis are listed in Table 5-2. 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 5-5.

5.5 Key Parameters and Uncertainties

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

5.6 DNB Statistical Design Limit

The statistical DNBR value for each statepoint evaluated is listed on Table 5-4. Section 1 of Table 5-4 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 Reference 5-4. The DNBRs calculated for all of the statepoints are normally distributed. As shown in Section 2 of Table 5-4 the maximum statepoint statistical DNBR value is [ ]. Therefore, the statistical design limit (SDL) using the WRB-2M CHF correlation for the RFA design at McGuire/Catawba is conservatively determined to be [ ].

5.7 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 5-1 is used to evaluate the impact of transition cores containing the RFA design. In Figure 5 of Reference 5-1, 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.

A transition core DNBR penalty is determined for the RFA design using the 8 channel RFA/Mark-BW transition core model. A conservative DNBR penalty is applied for all DNBR analyses for RFA/Mark-BW transition cores.

To evaluate the impact of the transition core on the statistical DNBR limit, the most limiting full core statepoint (Statepoint 12 on Table 5-4) was evaluated using the 8 channel transition core model. This case is designated as statepoint 12TR in Sections 1 and 2 of Table 5-4. 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 5-4, this value is still less than [ ]. Therefore, the statistical design limit of [ ] is bounding for RFA/Mark-BW transition cores as well as full RFA cores.

5.8 References

5-1 DPC-NE-2004P-A, McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, Rev 1, February 1997.

5-2 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.

5-3 VIPRE-01: A Thermal-Hydraulic Code For Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

5-4 DPC-NE-2005P-A, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, Rev 1, November 1996.

Table 5-1

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)	160.0

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 Mixing (Non-structural)
Nozzles	304SS	1	Debris Filtering Bottom
	304SS	1	Removable Top

Table 5-2

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 (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						
12TR***						

* 100% RTP = 3411 Megawatts Thermal

** Mass flow rate should be calculated using the given core inlet temp.

*** TR - transition core model

Table 5-3

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	[ ]	Normal

* Percentage of 100% RTP (3411 MWth)

Table 5-3 (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.
$F^N_{\Delta H}$	
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 5-3 (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 of the physics code's axial nodes. 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 5-4

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*				

* TR - transition core model

Table 5-4 (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*				

* TR - transition core model

Table 5-5

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		

* 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.

6.0 UFSAR ACCIDENT ANALYSES

DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology" (Reference 6-1), DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology" (Reference 6-2), and DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology" (Reference 6-3) describe the Duke Power NRC-approved models and methodology for analyzing UFSAR Chapter 15 Non-LOCA transients and accidents. DPC-NE-3004-PA, "Mass and Energy Release and Containment Response Methodology" (Reference 6-4), describes the Duke Power NRC-approved models and methodology for analyzing UFSAR Chapter 6.2 mass and energy release accidents and containment response.

UFSAR Chapter 15 non-LOCA analyses will continue to be performed according to the methodologies described previously in Reference 6-1, Reference 6-2, and Reference 6-3, except as noted in Sections 6.1-6.3, respectively. LOCA mass and energy release analyses (UFSAR Chapter 6.2) will continue to be performed according to the methodology described in Reference 6-4, except as noted in Section 6.4. LOCA analyses (UFSAR Chapter 15.6.5) will be performed by Westinghouse as described in Section 6.5.

6.1 Thermal-Hydraulic Transient Analysis Methodology (DPC-NE-3000)

DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology" (Reference 6-1), serves as the Duke Power Company response to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Action," which requires that licensees performing their own safety analyses demonstrate their analytical capabilities. Reference 6-1 describes the RETRAN-02 (Reference 6-21) system transient thermal-hydraulic models, and the VIPRE-01 (Reference 6-20) core thermal-hydraulic models developed for Oconee, McGuire, and Catawba Nuclear Stations. The previous comparisons of computer code results to experimental data, plant operational data, and other benchmarked analyses, continue to demonstrate the analytical capability to perform non-LOCA transient thermal-hydraulic analyses. Changing from Mark-BW to the RFA design does not affect this conclusion.

A review of Reference 6-1 indicates that only portions of Chapter 3 (McGuire/Catawba Transient Analyses) currently do not support the RFA design from a technical standpoint. Chapters 2 and 4 pertain to Oconee Nuclear Station only, and therefore remain unaffected. Chapter 5 pertains to McGuire/Catawba RETRAN benchmark analyses, which continue to demonstrate analytical capability to perform non-LOCA transient thermal-hydraulic analyses regardless of fuel type. Chapters 1 (Introduction) and 6 (Summary) are affected from an editorial standpoint only.

6.1.1 Plant Description (Section 3.1 in DPC-NE-3000)

The only difference with respect to the plant description will be the change from Mark-BW fuel to the RFA design. Chapter 2 of this report gives a complete description of the RFA design.

6.1.2 McGuire/Catawba RETRAN Model (Section 3.2 in DPC-NE-3000)

Volumes [ ] in the primary system nodalization scheme represent the reactor core region from the [.] Dimensional changes due to the change to the RFA design will require minor changes to these volume calculations, as well as associated junction and heat conductor calculations.

6.1.3 McGuire/Catawba VIPRE Model (Section 3.3 in DPC-NE-3000)

The McGuire/Catawba simplified [ ] channel model in Reference 6-1 is used for analyzing the RFA design. As described in Chapter 5, the reference radial pin power distribution remains unchanged, but the peak pin is increased from 1.50 to 1.60 and the WRB-2M CHF correlation (Reference 6-5) and the SCD limit developed in Chapter 5 are used. The axial node size is adjusted to be compatible with the WRB-2M CHF correlation. The RFA design geometry is listed in Table 5-1 and applicable form loss coefficients are used. The remaining code inputs and options remain identical to that originally approved in Reference 6-1.

No transition core transient analyses are performed as the results determined in Chapter 5 also apply for transient analyses. As discussed in Reference 6-1, the [ ] channel model 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. Should it be determined in the future that transition core transient analyses are warranted, they will be performed accordingly.

6.2 Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology (DPC-NE-3001)

DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology" (Reference 6-2), describes the Duke Power Company methodologies for simulating the UFSAR Chapter 15 events characterized by multidimensional reactor transients (rod ejection, steam line break, and dropped rod), and for systematically confirming that reload physics parameters important to Chapter 15 transients and accidents are bounded by values assumed in the licensing analyses (the Safety Analysis Physics Parameters (SAPP) methodology). The SAPP methodology remains unchanged when analyzing the RFA design. Thermal-hydraulic changes for analyzing the RFA design in rod ejection, steam line break, and dropped rod accidents are discussed in the sections that follow.

6.2.1 Rod Ejection

The changes presented in Section 6.1 also apply to the rod ejection accident. The nuclear analysis of the rod ejection accident using SIMULATE-3K is presented in Section 6.6. The remainder of the rod ejection thermal-hydraulic methodology presented in Reference 6-2 remains unchanged.

6.2.2 Steam Line Break

The changes presented in Section 6.1 also apply to steam line break, with the exception of the CHF correlation. Since the WRB-2M CHF correlation pressure range of applicability is not acceptable for steam line break analyses (see Chapter 5 of this report for ranges of applicability), the W3-S CHF correlation will continue to be used as originally documented in Reference 6-2.

The remainder of the steam line break thermal-hydraulic methodology presented in Reference 6-2 remains unchanged, except for the selection of subcooled and bulk void models for offsite power lost (OSPL) cases for reasons described in Chapter 5. The [ ] for steam line break cases for which offsite power is lost. This is acceptable since the [ ] gives more conservative DNBR results for steady-state cases (according to Reference 6-1), and preliminary studies of steam line break cases show no difference in results.

6.2.3 Dropped Rod

The changes presented in Section 6.1 also apply to the dropped rod transient. The remainder of the dropped rod thermal-hydraulic methodology presented in Reference 6-2 remains unchanged.

6.3 UFSAR Chapter 15 System Transient Analysis Methodology (DPC-NE-3002)

DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology" (Reference 6-3) documents the conservative modeling assumptions used by Duke Power Company in performing the NSSS primary and secondary system analyses of UFSAR Chapter 15 accidents. It covers all applicable non-LOCA accidents in UFSAR Sections 15.1-15.6, except those already discussed in Reference 6-2. There are no changes to Reference 6-3 with respect to analyzing the RFA design.

6.4 Mass and Energy Release and Containment Response Methodology (DPC-NE-3004)

DPC-NE-3004-PA, "Mass and Energy Release and Containment Response Methodology" (Reference 6-4), describes the Duke Power Company methodology for simulating the mass and energy release from high energy line breaks (LOCA and steam line break) and the resulting containment response to demonstrate that the containment peak pressure and temperature limits are not exceeded. Since the fuel stored energy for the RFA design is similar to that for the Mark-BW fuel, there are no changes anticipated for Reference 6-4 with respect to the RFA design except the RETRAN related changes described in Section 6.1 of this report. Similar changes to

the RELAP5 model, which is used to model the mass and energy release from LOCAs, are also anticipated. The RETRAN and RELAP5 model changes for the RFA design are not significant enough to require reanalyses. Future reanalyses will incorporate the RFA design model revisions.

6.5 LOCA Analyses

Large and small break LOCA analyses will be performed by Westinghouse using approved versions of the Westinghouse Appendix K LOCA evaluation models. All features employed have been approved by the NRC as required and annual model reports for the evaluation models have been supplied to the NRC, the most recent of which is found in Reference 6-22. Therefore, no NRC review of the evaluation model features is necessary, and only methodology with respect to analyzing McGuire/Catawba will be presented in this section. New LOCA analyses will be performed to support the licensing of McGuire/Catawba during the transition and full core operation of the RFA design.

6.5.1 Small Break LOCA

For small break LOCAs (SBLOCAs) due to breaks less than 1 ft^2, Westinghouse developed the NOTRUMP computer code (Reference 6-23) to calculate the transient depressurization of the reactor coolant system (RCS) as well as to describe the mass and enthalpy of flow through the break. The NOTRUMP Small Break LOCA Emergency Core Cooling System (ECCS) Evaluation Model (References 6-24, 6-25, 6-26, 6-27, and 6-39) was developed and licensed by Westinghouse to determine the RCS response to design basis SBLOCAs, and to address NRC concerns expressed in NUREG-0737, Item II.K.3.30.

The NRC approved noding scheme for the NOTRUMP Evaluation Model is shown in Reference 6-24, although minor noding changes to facilitate the modeling of broken loop ECCS were instituted and reported to the NRC in Reference 6-28. Peak cladding temperature (PCT) calculations are performed with the LOCTA-IV code (Reference 6-29) using the NOTRUMP calculated core pressure, fuel rod power history, uncovered core steam flow and mixture heights as boundary conditions. Additional modifications to the LOCTA-IV code to allow the modeling of annular fuel pellets in the axial blankets have been reviewed and approved by the NRC in Reference 6-27. The axial shape chosen for McGuire/Catawba SBLOCA will be based on the desired core operating limits and axial offset control strategy so as to bound all burnups and operating cycles.

Due to the nature of SBLOCA transients, the rod heatup and resulting calculated PCT is insensitive to transition core effects, and an evaluation is performed to demonstrate that this is a valid assumption. Therefore, SBLOCA will generally have no additional penalty for transition core effects.

6.5.2 Large Break LOCA

For the Westinghouse large break LOCA (LBLOCA) methodology, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. The most recent version of the 1981 Westinghouse Large Break LOCA ECCS Evaluation Model with BASH (Reference 6-30) will be used to perform the LBLOCA analysis for the transition of McGuire/Catawba to the RFA design. A description of the various aspects of the Westinghouse LOCA analysis methodology can be found in WCAP-8339 (Reference 6-31). This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the acceptance criteria. The SATAN-VI (Reference 6-32), WREFLOOD (Reference 6-33), BASH and LOCBART codes, which are used in the LOCA analysis, are described in detail in References 6-30 and 6-34. These codes assess the core heat transfer geometry and determine if the core remains amenable to cooling through and subsequent to the blowdown, refill, and reflood phases of the LOCA. The LOTIC computer code (Reference 6-35) calculates the minimum containment backpressure transient required for LBLOCA analyses in Appendix K to 10 CFR Part 50. Although there have been several updates to the original SATAN-VI code, the most notable upgrade is delineated in Reference 6-34.

The WREFLOOD code has been replaced by the REFILL code as reported in Reference 6-36. The REFILL code is identical to the section of the WREFLOOD code that modeled the refill phase of the transient. There has also been a recent change (the incorporation of the REFILL and LOCTA codes directly into the BASH code as subroutine modules) in the methodology for execution of the

BASH Evaluation Model as reported in Reference 6-37. In addition, the LOTIC code has been coupled with the BASH code so that the codes run interactively. The BASH Evaluation Model now utilizes the SATAN code for the blowdown calculations, the BASH code for the refill and reflood phases with interactive LOTIC calculations for containment backpressure, and the LOCBART code for the fuel rod heatup calculations. The most recent version of the LOCBART code employs an improved grid heat transfer model which has been approved by the by NRC in Reference 6-38.

An input parameter that affects LOCA analysis results is the assumed axial power shape at the beginning of the accident. The methodology employed by Westinghouse is termed ESHAPE (Explicit SHape Analysis for Pct Effects). The ESHAPE methodology is based upon explicit analysis of the LBLOCA transient with a set of bounding skewed axial power shapes to supplement the base analysis performed with the chopped cosine power shape. The limiting case break, as demonstrated with a chopped cosine, will be reanalyzed using skewed power shapes and typically demonstrate that the chopped cosine power shape is limiting.

As required in Appendix K to 10 CFR 50, a minimum of a three break spectrum will be analyzed. In addition, as required in the NRC Safety Evaluation Report (SER) for the BASH Evaluation Model, a maximum Safety Injection flow case will be analyzed.

When assessing the effect of transition cores on the LBLOCA analysis, it must be determined whether the transition core can have a greater calculated peak cladding temperature (PCT) than a complete core of the RFA design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. Hydraulic resistance mismatch will exist only for a transition core and is the only unique difference between a complete core of either fuel type and the transition core. An evaluation will be performed to address the cross-flow effects due to any hydraulic mismatch between the current fuel and the Westinghouse fuel. If it is determined that a transition core penalty is required during the cycles that both fuels reside in the core, it will be applied as an adder to the LOCA results for a full core of the RFA design.

6.6 Rod Ejection Analysis Using SIMULATE-3K

This section presents an improved methodology to be used by Duke Power to perform the nuclear analysis portion of the rod ejection accident (REA) analysis for the McGuire and Catawba Nuclear Stations. The current approved REA analysis methodology is described in the topical report titled, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters" (Reference 6-2) and uses the computer code ARROTTA to perform the nuclear analysis portion of the REA calculation. A Safety Evaluation Report (SER) for this topical was received on November 15, 1991 (Reference 6-6). The new methodology is based on the SIMULATE-3K (Reference 6-9) computer code which employs a three-dimensional neutron kinetics model based on the QPANDA two-group nodal model to calculate three-dimensional power distributions, core reactivity, or a core power level for both static and transient applications.

The SIMULATE-3K methodology affords compatibility with the current SIMULATE-3P nuclear design methodology (Reference 6-8) and will enhance the generation of forcing functions (transient core power distribution and hot assembly peak pin power distribution) at bounding physics parameter conditions for input into fuel enthalpy, peak RCS pressure, and DNB calculations. The SIMULATE-3K cross section model is also more robust than that used by ARROTTA. The transition from ARROTTA to SIMULATE-3K will reduce the engineering resources required to perform future REA analyses and enhance the transition from Mark-BW fuel to Westinghouse RFA or other fuel types in the future.

The basic methodology described in Reference 6-2 for the nuclear analysis portion of the REA remains intact with only minor differences which are outlined in this report. All other methods described in Reference 6-2 remain unchanged, i.e. core thermal-hydraulic and system thermal-hydraulic analysis. To demonstrate the transient capability of SIMULATE-3K, comparisons between SIMULATE-3K and ARROTTA reference REA analyses at beginning-of-cycle (BOC) and end-of-cycle (EOC), hot full power (HFP) and hot zero power (HZP) conditions were performed. These comparisons demonstrate the acceptability of the physical and numerical models within the SIMULATE-3K code as compared to the current licensed methodology.

A description of the models employed and the benchmark calculations performed in the verification of the SIMULATE-3K computer code are presented in Section 6.6.1. This section also includes a comparison of ARROTTA and SIMULATE-3K REA results applicable to the McGuire and Catawba Nuclear Stations at BOC and EOC, HFP and HZP conditions.

Section 6.6.2 describes the nuclear analysis methodology to be used in the evaluation of the UFSAR Chapter 15 REA using SIMULATE-3K.

6.6.1 SIMULATION CODES AND MODELS

6.6.1.1 <u>CASMO-3 & SIMULATE-3P</u>

CASMO-3 is used to produce two energy group edits of homogenized cross sections, assembly discontinuity factors, fission product data, and pin power data for input to ARROTTA, SIMULATE-3P, and SIMULATE-3K core models. CASMO-3 is a multigroup, two dimensional transport theory code for burnup calculations on PWR or BWR fuel assemblies. The code models a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with integral burnable absorber, lumped burnable absorber rods, clustered discrete control rods, incore instrument channels, assembly guide tubes, and intra-assembly water gaps. The program utilizes a cross section library based on ENDF/B-IV with some data taken from ENDF/B-V. Reference 6-11 provides a detailed description of the theory and equations solved by CASMO-3. The use of CASMO-3 in this report is consistent with the previously approved methodologies of References 6-8 and 6-2.

SIMULATE-3P is used to set up the cycle-specific model and conditions for the REA. It may also be used to generate pin-to-assembly factors for the conversion of nodal powers to pin powers for the REA analyses. SIMULATE-3P is a three-dimensional, two energy group, diffusion theory core simulator program which explicitly models the baffle and reflector regions of the reactor. Homogenized cross sections and discontinuity factors developed with CASMO-3 are used on a coarse mesh nodal basis to solve the two group diffusion equations using the QPANDA neutronics model. A nodal thermal hydraulics model is incorporated to provide both fuel and moderator temperature feedback effects. Inter- and intra-assembly information from the coarse mesh solution is then utilized along with the pinwise assembly lattice data from CASMO-3 to reconstitute pin-by-pin power distributions in two and three dimensions. The program performs a macroscopic depletion of fuel with microscopic depletion of iodine, xenon, promethium, and samarium fission products. Reference 6-10 provides a detailed description of the theory and equations solved by SIMULATE-3P. The use of SIMULATE-3P in this report is consistent with the previously approved methodologies of References 6-8 and 6-2.

6.6.1.2 ARROTTA

ARROTTA is a three-dimensional, two energy group diffusion theory core simulator applicable for both static and transient kinetics simulations. Homogenized cross sections, discontinuity factors, and six groups of delayed neutron precursor data are generated with CASMO-3 and used on a coarse mesh nodal basis to solve the two energy group diffusion equations using the QPANDA neutronics model. The thermal-hydraulic model is comprised of both fluid dynamics and heat transfer models. Reference 6-12 provides a detailed description of the theory and equations solved by ARROTTA. The use of ARROTTA for the benchmark calculations performed in this report is consistent with the previously approved methodology documented in Reference 6-2.

6.6.1.3 SIMULATE-3K

6.6.1.3.1 Code Description

The SIMULATE-3K code (Reference 6-9) is a three-dimensional transient neutronic version of the SIMULATE-3P code (Reference 6-10). SIMULATE-3K uses the QPANDA full two-group nodal spatial model developed in SIMULATE-3P, with the addition of six delayed neutron groups. The program employs a fully-implicit time integration of the neutron flux, delayed neutron precursor, and heat conduction models. Beta is fully functionalized similar to other cross sections to provide an accurate value of beta for the time-varying neutron flux. The control of time step size may be determined either as an automated feature of the program or by user input. Use of the automated feature allows the program to utilize larger time steps (which may be restricted to a maximum size based on user input) at times when the neutronics are changing slowly and smaller time steps when the neutronics are changing rapidly.

Additional capability is provided in the form of modeling a reactor trip. The trip may be initiated at a specific time in the transient or following a specified excore detector response. Use of the excore detector response model to initiate the trip 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. The velocity of the control rod movement is also controlled by user input.

The SIMULATE-3K thermal-hydraulic model includes a spatial heat conduction and a hydraulic channel model. The heat conduction model solves the conduction equation on a multi-region mesh in cylindrical coordinates. Temperature-dependent values may be employed for the heat capacity, thermal conductivity, and gap conductances. A single characteristic pin conduction calculation is performed consistent with the radial neutronic node geometry, with an optional calculation of the peak pin behavior available to monitor local maxima. A single characteristic hydraulic channel calculation is performed based on the radial neutronic node geometry. The model allows for direct moderator heating at the option of the user. This thermal-hydraulic model is used to determine fuel and moderator temperatures for updating the cross-sections, and may additionally be used to provide edits of fuel temperature throughout the transient.

The SIMULATE-3K program utilizes the same cross-section library and reads the same restart file (exposure and burnup-related information) as SIMULATE-3P. Executed in the static mode, SIMULATE-3K performs the same solution techniques, pin power reconstruction, and cross-section development as SIMULATE-3P. Additional features of SIMULATE-3K include the application of conservatism to key physics parameters through simple user input. Also, the inlet thermal-hydraulic conditions can be provided on a time dependent basis through user input.

6.6.1.3.2 SIMULATE-3K Code Verification

The SIMULATE-3K code has been benchmarked against many numerical steady state and transient benchmark problems by the code vendor, Studsvik of America, Inc. The results of these benchmarks are described in Reference 6-9 and show excellent agreement between

SIMULATE-3K and the reference solutions. Some of the SIMULATE-3K benchmarks which have been performed are: The fuel conduction and thermal-hydraulics model has been benchmarked against the TRAC code (Reference 6-13). The transient neutronics model has been benchmarked, using standard LWR problems, to reference solutions generated by QUANDRY (Reference 6-14), SPANDEX (Reference 6-15), NEM (Reference 6-16), and CUBBOX (Reference 6-17). Finally, a benchmark of the coupled performance of the transient neutronics and thermal-hydraulic models was provided by comparison of results from a standard NEACRP rod ejection problem to the PANTHER code (Reference 6-18). Steady-state components of the SIMULATE-3K model are implemented consistent with the CASMO-3/SIMULATE-3P methodology and performance benchmarks which were approved for use on all Duke Power reactors in Reference 6-8. In addition, a benchmark to ARROTTA for the Oconee REA analyses was performed in topical report DPC-NE-3005-P, "Oconee UFSAR Chapter 15 Transient Analysis Methodology (Reference 6-19).

6.6.1.3.3 SIMULATE-3K / ARROTTA REA Benchmark

The three dimensional neutron kinetics capability of the SIMULATE-3K code is demonstrated by comparing SIMULATE-3K and ARROTTA calculations for the reference rod ejection accident analyses performed at BOC and EOC, HFP and HZP conditions for McGuire and Catawba. For the REA benchmark, ARROTTA and SIMULATE-3K are used to calculate the core power level and nodal power distribution versus time during the rod ejection transient for the BOC and EOC, HFP and HZP REA cases. These comparisons demonstrate the acceptability of the physical and numerical models within SIMULATE-3K for application in the REA analyses for McGuire and Catawba Nuclear Station.

The reference core used in the benchmark calculations is a hypothetical Catawba 1 Cycle 15 core. This core represents typical fuel management strategies (i.e. core loadings and cycle lengths) currently being developed for reload core designs at McGuire and Catawba Nuclear Stations. The ARROTTA and SIMULATE-3K models for this core were then adjusted to produce a conservative initial condition Doppler and moderator temperature coefficient, ejected rod worth, Beta, and power distribution as described in the "Multidimensional Reactor Transients and Safety Analysis Physics Parameter" topical report DPC-NE-3001 (Reference 6-2).

The combination of these conservative input parameters produces conservative transient results. The assembly enrichments, burnable poison loading, and assembly exposures for the reference core are shown in Figure 6-1. The core consists of all Framatome Mark-BW fuel.

6.6.1.3.3.1 ARROTTA Analysis

The ARROTTA REA analysis is based on the methodology described in the "Multidimensional Reactor Transients and Safety Analysis Physics Parameters" topical report DPC-NE-3001 (Reference 6-2) with the exceptions that the initial power conditions have been increased to reflect a design pin FΔH of 1.6, and the ARROTTA model was updated to reflect the C1C15 reference core design.

The REA analyses of Reference 6-2 were made limiting by setting key physics parameters to conservative or bounding values. Utilizing this approach produces limiting results which are expected to bound future reload cycles. The ARROTTA model was adjusted to produce conservative MTC, DTC, Beta, and ejected rod worths as identified in Tables 6-3.

[ ]

6.6.1.3.3.2 SIMULATE-3K Analysis

The SIMULATE-3K analysis is performed as described in DPC-NE-3001, Reference 6-2. The SIMULATE-3K model employed in this analysis was adjusted to be functionally equivalent to the ARROTTA model to account for differences in the two codes cross section model. Since ARROTTA is restricted to one node per fuel assembly in the radial direction, the SIMULATE-3K model was set up to be consistent with this assumption. The axial nodalization depends on the fuel assembly design, such as whether or not axial blanket fuel is being modeled. For the analysis presented, an axial nodalization of 18 equal length fuel nodes is used.

[

] Additional model adjustments were performed to produce limiting values for the Doppler temperature coefficient, moderator temperature coefficient, ejected rod worth, and Beta. Table 6-3 provides a summary of initial condition values for each of these parameters for the SIMULATE-3K analyses. Trip times were input to be consistent with the ARROTTA analyses.

6.6.1.3.3.3 Results

ARROTTA results from each of the four cases evaluated are summarized in Table 6-1. Results from the SIMULATE-3K cases are provided in Table 6-2. Table 6-3 lists the REA initial condition kinetics parameters for both the ARROTTA and SIMULATE-3K benchmarks. Core power versus time for each case is shown in Figures 6-2 through 6-4.

For the HFP cases, which begin at 102% power, core power increases rapidly as the control rod is ejected. The ejected rod worth in these transients is not sufficient to achieve a prompt critical state. Power increases until Doppler feedback from increasing fuel temperature begins to turn the excursion around. Core power level continues to decrease as the fuel temperature approaches an equilibrium value. A reactor trip signal on high flux occurs very early in these transients but the conservative trip delay time prevents rod motion until after the peak core power occurs. Additional conservatisms applied to the rate of rod insertion and scram worth minimizes the effect of the reactor trip until the rods approach the bottom of the reactor core.

The transients initiated from HZP differ from the at-power initial conditions in that the ejected rod worth is large enough to achieve a prompt critical core. The power increase continues after the control rod is fully ejected until the fuel heats up enough for Doppler feedback to turn the excursion around. Conservatisms on trip delay time, rate of rod insertion, and scram worth minimize the impact of the reactor trip.

These results showed good agreement between SIMULATE-3K and ARROTTA for the reference analyses. The transient power response and time of peak power statepoint agreed well. The nodal peak powers agreed well with the exception of the EOC HZP case. This was due to the unique combination of adjustments which had to be made for this case to duplicate ARROTTA's initial conditions as specified in Table 6-3. In conclusion, these comparisons demonstrate the acceptability of the physical and numerical models within the SIMULATE-3K code for application in analyses of the REA for McGuire and Catawba Nuclear Station.

6.6.2 Rod Ejection Nuclear Analysis

The current approved methodology for the REA utilizes the computer code ARROTTA (Reference 6-12) to perform nuclear analysis calculations. This section describes the use of SIMULATE-3K for the nuclear analysis calculations for the REA analyses as described in topical report, "Multidimensional Reactor Transients and Safety Analysis Physics Parameters" DPC-NE-3001 (Reference 6-2).

6.6.2.1 REA Analytical Approach

The complexity of the core and system response to a rod ejection event requires the application of a sequence of computer codes. The rapid core power excursion is simulated with a three-dimensional transient neutronic and thermal-hydraulic model using the SIMULATE-3K code (Reference 6-9). [ ] The resulting transient core power distribution results are then input to VIPRE-01 (Reference 6-20) core thermal-hydraulic models. The VIPRE models calculate the fuel temperatures, the allowable power peaking to avoid exceeding the DNBR limit, and the core coolant expansion rate. The allowable power peaking is then used along with a post-ejected condition fuel pin census to determine the percentage of pins exceeding the DNB limit. The coolant expansion rate is input to a RETRAN-02 (Reference 6-21) model of the Reactor Coolant System to determine the peak pressure resulting from the core power excursion.

The remainder of this section will address how the nuclear analyses of the REA will be performed with SIMULATE-3K. The basic methodology, as described in Reference 6-2, remains unchanged with the exception of minor differences between SIMULATE-3K and ARROTTA which are discussed in the following section.

6.6.2.2 SIMULATE-3K Nuclear Analysis

The response of the reactor core to the rapid reactivity insertion from the control rod ejection is simulated with SIMULATE-3K code (Reference 6-9). SIMULATE-3K computes a three-dimensional power distribution (in rectangular coordinates) and reactivity or power level for both static and transient applications. SIMULATE-3K includes a prediction of individual pin powers. Modifications are made to the core model to ensure conservative results. These changes produce a rod ejection model which produces limiting results that are expected to bound future reload cycles. A complete description of the SIMULATE-3K code is discussed in Section 6.6.1.3 and Reference 6-9.

The SIMULATE-3K model geometry will typically be [ ] per fuel assembly in the radial direction. The axial nodalization depends on the fuel assembly design, such as whether or not axial blanket fuel is being modeled. The number of axial levels is 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. The SIMULATE-3K model 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. Required fuel and reflector cross sections are developed consistent with the methodology approved for SIMULATE-3P in topical report DPC-NE-1004A (Reference 6-8).

SIMULATE-3K is used to calculate the core power level and nodal power distribution versus time during the rod ejection transient. [ ] This information is used by VIPRE to determine the fuel enthalpy, the percentage of the fuel pins exceeding the DNB limit, and the coolant expansion rate.

6.6.2.2.1 Initial Conditions

The SIMULATE-3K rod ejection analysis is analyzed at four statepoints; beginning-of-cycle (BOC) at hot zero power (HZP) and hot full power (HFP) and end-of-cycle (EOC) at HZP and HFP. The conservatisms applied to the rod ejection analysis as described in Reference 6-2 are implemented based on the methodology described in Reference 6-9 and are expected to bound future reload cycles. Initial conditions for SIMULATE-3K different than those discussed in Reference 6-2 are described below.

[ ]

The moderator temperature coefficient (MTC) is also adjusted to conservative values at BOC or EOC which bounds the magnitude of the MTC expected in a reload core. The MTC is adjusted in SIMULATE-3K by [ ] This adjustment is made via the equation from SIMULATE-3K (Reference 6-9);

[

]

Similar adjustments are made to yield conservative rod worth for control rod withdrawal and rod worth for control rod insertion.

The Doppler (or fuel) temperature coefficient (DTC) is important to this transient because the negative reactivity from the increased fuel temperature is the only effect that limits the power excursion and starts to shut down the reactor. The DTC is adjusted to a conservative value which bounds the magnitude of the DTC expected in a reload core. The DTC is adjusted in SIMULATE-3K by [ ]

The effective delayed neutron fraction (β) and the ejected rod worth both determine the transient power response of the reactor. The peak power level obtained during the transient will increase for small values of β and larger values of the ejected rod worth. The ejected rod worth and β are adjusted to conservative values which bound values expected for a reload core. The ejected rod worth is adjusted in SIMULATE-3K by [ ] β can be adjusted in SIMULATE-3K by [

]

[ ]

or β can be adjusted by inputting a single set of delayed neutron parameters to be used for all fueled nodes.

The combined effect of all these changes to the SIMULATE-3K model is to produce a model that is expected to bound future reload cycles for both McGuire and Catawba Nuclear Stations.

6.6.2.2.2 Boundary Conditions

The fuel and core thermal-hydraulic boundary conditions are established using conservative assumptions. Boundary conditions for initial power, core flow, inlet temperature, reactor pressure, and fission power fraction in the coolant are selected to yield conservative results.

The reactor trip signal is generated when the third highest excore channel reaches either [for the HZP cases or [ ] for the HFP cases. This modeling is based on a single failure of the highest channel and a two-out-of-the-remaining-three trip coincidence logic. [

] in SIMULATE-3K (Reference 6-9) can be used. The remaining control rods fall into the reactor assuming a conservative trip delay after the trip signal is generated.

During the reactor trip, the ejected rod and a second rod with the highest worth are assumed not to fall into the reactor. To conservatively model the reactor trip, not all of the control rod banks are allowed to drop, and some of the banks that are dropped have their worth reduced by a cross section adjustment. The rod worth adjustment is made in SIMULATE-3K by [

] based on Eq. 6.1. Also, negative reactivity inserted due to the reactor trip is not allowed to exceed the conservative trip reactivity curve. The integral worth of the falling control rods is computed for several different axial positions of the rods at the initial conditions. [

]

6.6.2.3 Core Thermal-Hydraulic Analysis

The core thermal-hydraulic analyses use the VIPRE-01 code for the calculation of peak fuel enthalpy, DNBR, and the coolant expansion rates for various initial and boundary conditions postulated for the REA transient. All input to the core thermal-hydraulic analyses once supplied by ARROTTA can now be supplied by SIMULATE-3K. The nuclear analysis input boundary conditions supplied by SIMULATE-3K for the thermal-hydraulic analyses are [

]

6.6.2.3.1 Fuel Temperature and Peak Fuel Enthalpy

The calculation of the transient maximum hot spot average fuel temperature and the maximum radial average fuel enthalpy requires the following input boundary conditions to be supplied by SIMULATE-3K: [

] This information is consistent with that provided by ARROTTA in Reference 6-2.

6.6.2.3.2 DNBR Evaluation

The percentage of the core experiencing DNBR is calculated as explained in Reference 6-2 except SIMULATE-3K results are used instead of ARROTTA results. For the HFP REA cases,

[

] For a given axial power profile, the maximum pin radial peak can be determined such that DNB would not occur during the transient. These DNB limits are referred to as maximum allowable radial peaks (MARP) limits. A fuel pin census is then performed to determine the number of fuel pins in the core that exceed the power peaking limit.

[

]

6.6.2.3.3 Coolant Expansion Rate

The calculation of the coolant expansion rate requires the following input boundary conditions to be supplied by SIMULATE-3K: [

] This SIMULATE-3K information is input to VIPRE to calculate the flow rate in each channel during the transient. Using the VIPRE channel flow rates, the total coolant expansion rate can be calculated. This total coolant expansion rate is input to the RETRAN plant transient model for simulating the resulting pressure response. This SIMULATE-3K information is consistent with that provided by ARROTTA in Reference 6-2.

6.6.2.4 Cycle-Specific Evaluation

Due to the conservative assumptions and modeling used in the SIMULATE-3K model, it is anticipated that for reload cores, no new SIMULATE-3K cases will be necessary. The determination as to whether the existing SIMULATE-3K cases remain bounding will be made by performing a cycle-specific reload check of the key physics input parameters as described in Reference 6-2. These parameters will be calculated using steady-state neutronics codes approved by the NRC for reload design. If the key physics parameters remain bounded then no new SIMULATE-3K analyses are necessary; otherwise, an evaluation, reanalysis, or re-design of the reload core will be performed.

For the HFP REA cases, a DNB pin census will be performed for the reload cycle, as described in Section 4.7 of Reference 6-2, with the radial power information being calculated with an NRC approved steady-state neutronics code. The HZP REA cases are bounded by the HFP cases in the offsite dose analyses, and therefore, a pin census is not required. The ejected rod worth shall be calculated with the fuel and moderator temperatures frozen in the pre-ejected condition or uniform throughout the core (either method will generate conservative results). [

] The power distribution with the ejected rod out will be used for the DNB pin census. The calculated percent fuel failure due to DNB will be compared for each cycle to the fuel failure limit assumed in the dose calculation. If the cycle specific value is less than the limit, then the existing safety analysis is still valid. Otherwise, an evaluation, a new dose calculation, reanalysis, or new reload design will be performed as appropriate.

6.6.2.5 Mixed Cores

The Westinghouse fuel is expected to behave neutronically similar to that of the Framatome Cogema Fuels Mark-BW fuel. The steady-state cycle-specific checks will verify that all key physics parameters remain valid and the DNB census will use the appropriate CHF correlations for the various fuel types present in the core.

6.7 References

6-1 DPC-NE-3000-PA, Thermal-Hydraulic Transient Analysis Methodology, Rev. 1, December 1997.

6-2 DPC-NE-3001-PA, Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, Rev. 0, November 1991.

6-3 DPC-NE-3002-A, UFSAR Chapter 15 System Transient Analysis Methodology, Rev. 2, December 1997.

6-4 DPC-NE-3004-PA, Mass and Energy Release and Containment Response Methodology, Rev. 0, November 1997.

6-5 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.

6-6 Letter, T. A. Reed (NRC) to H. B. Tucker (Duke), November 15, 1991 (SER for DPC-NE-3001).

6-7 Letter from Jeffrey A. Borkowski (Studsvik of America) to Robert St.Clair (Duke Power Company), January 12, 1996.

6-8 Nuclear Design Methodology using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Revision 1, Duke Power Company, December 1997.

6-9 SIMULATE-3 Kinetics Theory And Model Description, SOA-96/26, Studsvik of America, April 1996

6-10 SIMULATE-3: Advanced Three-Dimensional Two-Group Reactor Analysis Code, Studsvik/SOA-92/01, Studsvik of America, April 1992

6-11 CASMO-3: A Fuel Assembly Burnup Program User's Manual, STUDSVIK/NFA-88/48, Studsvik of America, September 1988

6-12 ARROTTA: Advanced Rapid Reactor Operational Transient Analysis, EPRI, August 1993

6-13 TRAC-BF1/MOD1, Vol. 1, NUREG/CR-4356, J. Borkowski and N. Wade (Editors), May 1992

6-14 QUANDRY: An Analytical Nodal Method for Solving the Two-Group, Multidimensional, Static and Transient Nodal Diffusion Equations, K. S. Smith, Massachusetts Institute of Technology, 1979

6-15 Development of a Variable Time-Step Transient NEM Code: SPANDEX, Trans. Am. Nucl. Soc. 68- 425, B. N. Aviles, 1993

6-16 NEM: A Three Dimensional Transient Neutronics Routine for the TRAC-PF1 Reactor Thermal Hydraulic Computer Code, B. R. Bandini, Pennsylvania State University, 1990

6-17 CUBBOX: Coarse-Mesh Nodal Diffusion Method for the Analysis of Space-Time Effects In Large Light Water Reactors, S. Langenbuch, W. Maurer, and W. Werner, Nucl. Sci. Eng. 63-437, 1977

6-18 NEACRP 3-D LWR Core Transient Benchmark, NEACRP-L-335, H. Finneman and A. Galati, January 1992

6-19 UFSAR Chapter 15 Transient Analysis Methodology, DPC-NE-3005-P, Duke Power Company, July 1997.

6-20 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM, Revision 2, EPRI, July 1985

6-21 RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988.

6-22 NSD-NRC-98-5575, "1997 Annual Notification of Changes to Westinghouse Small Break LOCA ECCS Evaluation Model and Large Break LOCA ECCS Evaluation Model Pursuant to 10CFR50.46(a)(3)(ii)", April 9, 1998.

6-23 WCAP-10079-P-A (Proprietary), WCAP-10080-A (Non-Proprietary), Meyer, P.E., "NOTRUMP - A Nodal Transient Small Break And General Network Code", August 1985.

6-24 WCAP-10054-P-A (Proprietary), WCAP-10081 (Non-Proprietary), Lee, H., et al., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", August 1985.

6-25 WCAP-10054-P-A Addendum 2 (Proprietary), Rev. 1, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model", July 1997.

6-26 WCAP-11145-P-A (Proprietary), WCAP-11372 (Non-Proprietary), Rupprecht, S.D., et al., "Westinghouse Small Break LOCA ECCS Evaluation Model Generic Study with the NOTRUMP Code", October 1986.

6-27 WCAP-14710-P-A, D. J. Shimeck, "1-D Heat Conduction Model for Annular Fuel Pellets", May 1998.

6-28 NSD-NRC-96-4639, "1995 Annual Notification of Changes to the Westinghouse Small Break LOCA ECCS Evaluation Model and Large Break LOCA ECCS Evaluation Model Pursuant to 10 CFR 50.46 (a)(3)(ii)", February 9, 1996.

6-29 WCAP-8301 (Proprietary), WCAP-8305 (Non-Proprietary), Borderlon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," June 1974.

6-30 WCAP-10266-P-A Revision 2 with Addenda (Proprietary), Kabadi, J.N., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March 1987.

6-31 WCAP-8339 (Non-Proprietary), Bordelon, F. M., et al., "Westinghouse ECCS Evaluation Model - Summary", June 1974.

6-32 WCAP-8302 (Proprietary) and WCAP-8306 (non-Proprietary), F. M. Borderlon, et. al., "SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant", June 1974.

6-33 WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), Kelly, R. D. et al., "Calculation model for core Reflooding After a Loss-of-Coolant Accident (WREFLOOD Code)," June 1974.

6-34 WCAP-9220-P-A Revision 1 (Proprietary), WCAP-9221 (Non-Proprietary), Eicheldinger, C., "Westinghouse ECCS Evaluation Model - 1981 Version", February 1982.

6-35 WCAP-8354 (Proprietary), Hsieh, T., and Raymund, M., "Long-Term Ice Condenser Containment LOTIC Code Supplement 1," July 1974, WCAP-8355 (Non-Proprietary), Supplement 1, May 1975.

6-36 NTD-NRC-94-4143, Letter from N. J. Liparulo (Westinghouse) to W. T. Russel (USNRC), “Change in Methodology for Execution of BASH Evaluation Model”, May 23, 1994.

6-37 NTD-NRC-95-4520, Letter from N. J. Liparulo (Westinghouse) to W. T. Russel (USNRC), “Change in Methodology for Execution of BASH Evaluation Model”, August 29, 1995.

6-38 WCAP-10484-P-A Addendum 1, “Spacer Grid Heat Transfer Effects During Reflood”, September 1993.

6-39 WCAP-15085, "Model Changes to the Westinghouse Appendix K Small Break LOCA NOTRUMP Evaluation Model: 1988-1997", July 1998.

Table 6-1

Rod Ejection ARROTTA Results

Parameter	BOC HZP	BOC HFP	EOC HZP	EOC HFP
Time of peak power, sec	0.286	0.077	0.173	0.080
Peak power level, % of full power	1880	138	5139	155
Peak nodal power relative to core average	7.99	3.44	16.40	3.96
Time that trip setpoint reached, sec	0.246	0.061	0.155	0.057
Time for beginning of trip rod motion	0.746	0.561	0.655	0.557

Table 6-2

Rod Ejection SIMULATE-3K Results

Parameter	BOC HZP	BOC HFP	EOC HZP	EOC HFP
Time of peak power, sec	0.296	0.076	0.187	0.083
Peak power level, % of full power	1884	133	5280	154
Peak nodal power relative to core average	7.127	3.508	12.997	3.605
Time that trip setpoint reached, sec	0.246	0.061	0.155	0.057
Time for beginning of trip rod motion	0.746	0.561	0.655	0.557

Table 6-3

Rod Ejection Transient Kinetics Input Parameters

Parameter	Computer Code	BOC HZP	BOC HFP	EOC HZP	EOC HFP
Ejected Rod Worth, pcm	ARROTTA	720	201	900	196
MTC (pcm/°F)	ARROTTA	+7.06	+0.05	-9.45	-9.73
DTC (pcm/°F)	ARROTTA	-0.90	-0.90	-1.19	-1.19
Delayed Neutron Fraction, β	ARROTTA	0.0055	0.0055	0.0040	0.0040
Ejected Rod Worth, pcm	SIMULATE-3K	721	203	900	197
MTC (pcm/°F)	SIMULATE-3K	+7.00	+0.08	-10.09	-10.09
DTC (pcm/°F)	SIMULATE-3K	-0.90	-0.90	-1.20	-1.20
Delayed Neutron Fraction, β	SIMULATE-3K	0.0055	0.0055	0.0040	0.0040

Figure 6-1

Reference Core Loading Information

	H	G	F	E	D	C	B	A
8	14 4.10 24/2.5 P 43.165 58.124	17 4.40 24/2.5 0 21.143	15 4.15 24/3.0 P 33.53 51.68	17 4.15 24/3.0 0 22.606	15 4.15 24/3.0 P 33.452 51.893	16 4.40 0 15.245 36.447	16 4.40 24/2.5 P 21.428 40.26	17 4.40 0 0 15.204
9		14 4.40 0 36.418 53.838	17 4.15 24/3.0 0 22.308	16 4.15 24/3.0 P 22.295 43.375	17 4.15 24/3.0 0 22.364	16 4.15 24/2.5 P 22.495 42.515	17 4.40 24/3.0 0 19.812	16 4.15 24/3.0 P 22.41 33.733
10			16 4.15 24/3.0 P 22.583 43.611	17 4.15 24/3.0 0 22.605	15 4.15 24/3.0 P 32.304 51.037	17 4.15 24/3.0 0 21.883	16 4.40 12/2.0 P 18.423 36.579	16 4.15 24/3.0 P 22.588 32.353
11				15 4.15 24/3.0 P 35.265 53.466	17 4.15 24/2.5 0 22.485	16 4.40 24/2.5 P 19.774 40.535	17 4.40 12/2.0 0 18.483	15 4.40 24/3.0 P 38.018 44.405
12					16 4.15 24/3.0 P 21.824 41.811	17 4.40 24/2.5 0 19.727	16 4.40 24/3.0 P 19.812 32.032	
13						16 4.15 24/3.0 P 21.823 35.106	15 4.40 24/3.0 P 32.088 38.42	

Key:
Batch number
Enrichment
Number of BP fingers/ wt% of boron in BP (P means BPs pulled)
BOC exposure (GWD/MT)
EOC exposure (GWD/MT)

Batch 17 is fresh fuel
Batch 16 is starting its second burn
Batches 14 and 15 are starting their third burn

Figure 6-2

FSAR Section 15.4.8 - Control Rod Ejection

BOC HFP Core Power vs. Time

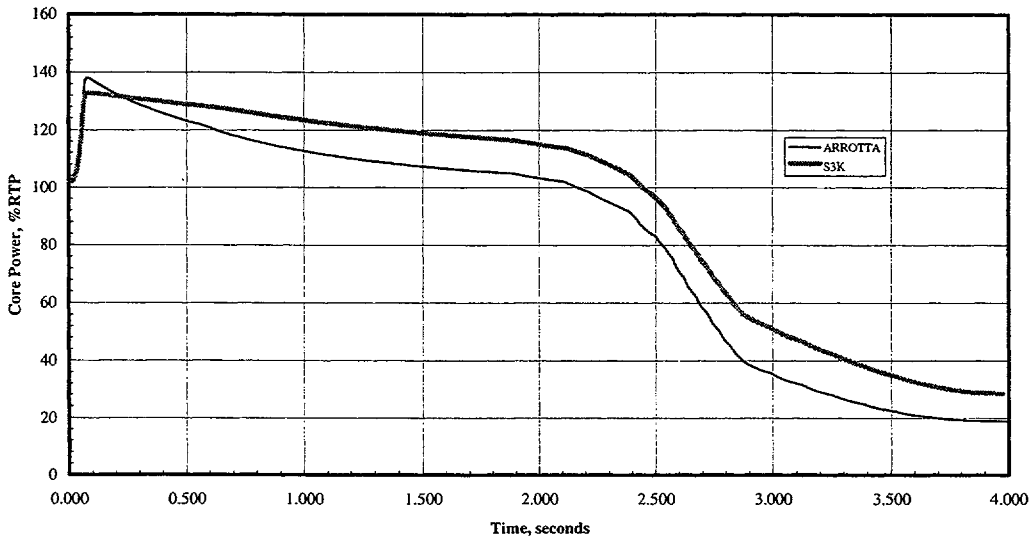


Figure 6-3

FSAR Section 15.4.8 - Control Rod Ejection

BOC HZP Core Power vs. Time

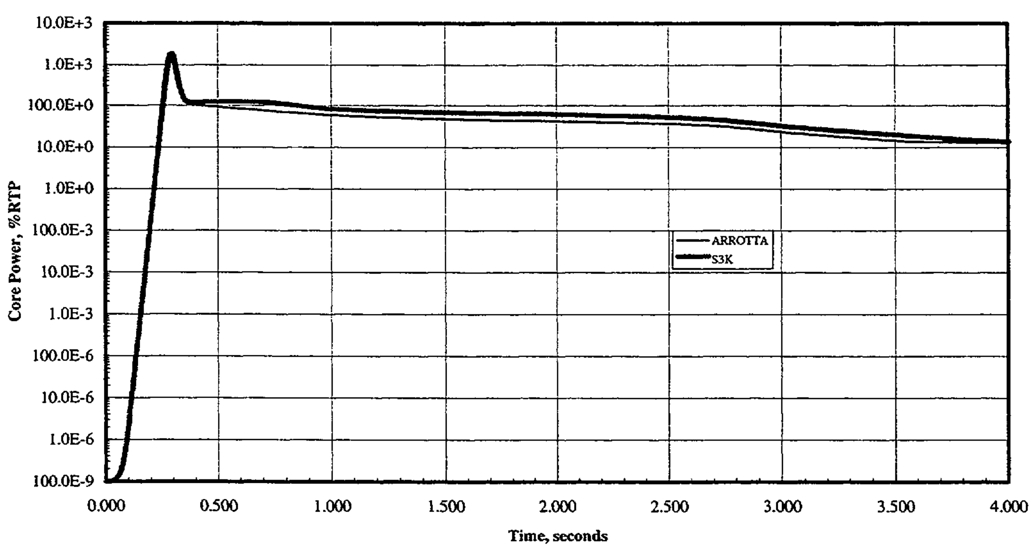


Figure 6-4

FSAR Section 15.4.8 - Control Rod Ejection

EOC HFP Core Power vs. Time

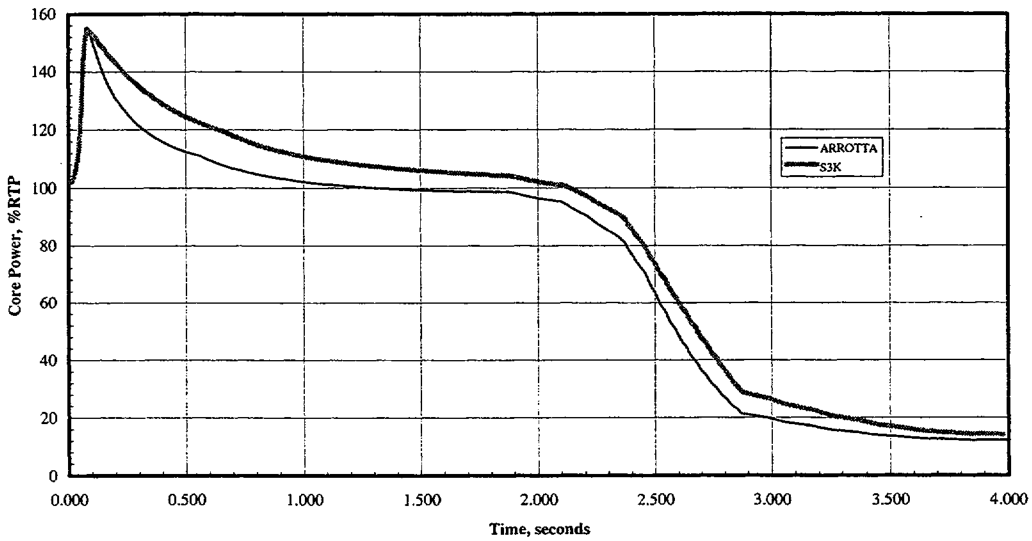
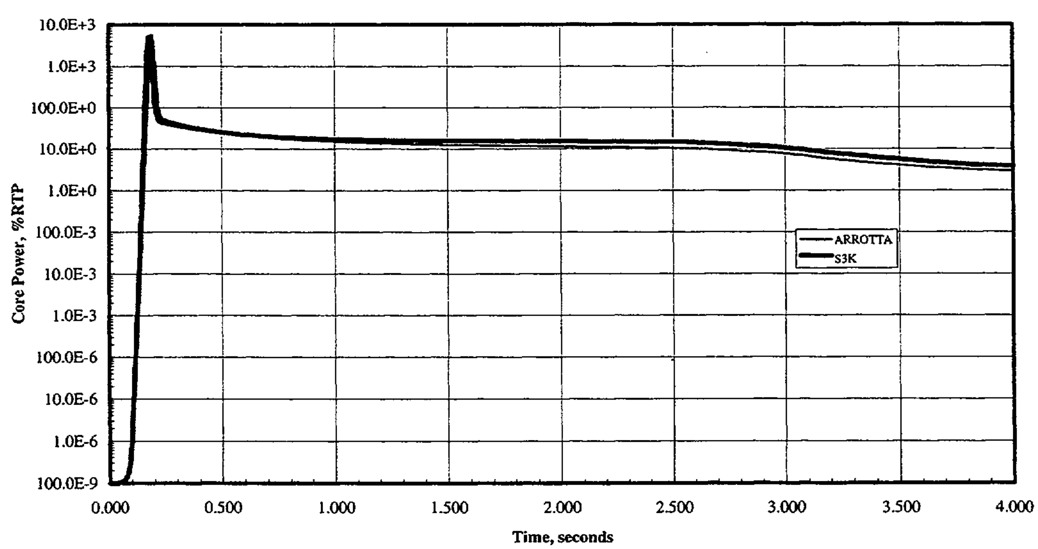


Figure 6-5

FSAR Section 15.4.8 - Control Rod Ejection

EOC HZP Core Power vs. Time



7.0 FUEL ASSEMBLY REPAIR AND RECONSTITUTION

The reconstitution of fuel assemblies is a routine occurrence during refueling outages in light water reactors. This is due to the concerted effort on the part of utilities to maintain zero fuel defects during cycle operation. This zero defect goal requires aggressive programs in two areas. First, all reasonable measures must be taken in the design and manufacturing of fuel assemblies to prevent any type of known failure mechanism. Secondly, failures that do occur during operation should be identified and the failed fuel rods removed before subsequent cycles.

Duke Power's primary replacement candidate for use in reconstitution is a fuel rod that contains pellets of natural uranium dioxide (UO_2). Aside from enrichment, this rod is the same in design and behavior as a standard fuel rod and is analyzed using standard approved methods. If local grid structural damage exists, the use of a natural UO_2 replacement rod is not the preferred alternative and solid filler rods made of stainless steel, zircaloy, or ZIRLO™ would be used.

The NRC-approved DPC-NE-2007 topical report, Reference 7-1, describes the methodology and guidelines Duke Power uses to support fuel assembly reconstitution with filler rods. The guidelines were developed to ensure acceptable nuclear, mechanical, and thermal-hydraulic performance of reconstituted fuel assemblies. Specific results were provided in the report for the Mark-B and Mark-BW fuel designs with licensed codes. As stated in DPC-NE-2007, the methodology would be applicable if different fuel designs or codes are licensed by Duke Power.

Duke Power will use the same licensing and analysis approach for reconstitution of the RFA design at McGuire and Catawba. The methodology described in Reference 7-1 will be used along with the licensed codes and correlations described in this report. These codes will be used to analyze reconstitution with filler rods for acceptable nuclear, mechanical, and thermal-hydraulic performance. For a reload core using reconstituted Westinghouse fuel, Westinghouse will evaluate the effects of the reconstitution on the LOCA analysis using the methodology given in Reference 7-2.

As discussed in Reference 7-2, Westinghouse has reviewed the criteria specified in Standard Review Plan 4.2 (Reference 7-3) and determined that the only fuel assembly mechanical criteria impacted by reconstitution are:

1) fuel assembly holddown force, and

2) fuel assembly structural response to Seismic/LOCA loads.

Westinghouse evaluated both of these criteria and concluded that the reconstituted fuel assembly designs are acceptable for both normal and faulted condition operations.

7.1 <u>References</u>

7-1 DPC-NE-2007P-A, Duke Power Company Fuel Reconstitution Analysis Methodology, October 1995.

7-2 W. H. Slagle (Ed.), "Westinghouse Fuel Assembly Reconstitution Evaluation Methodology", WCAP-13060-P-A, July 1993.

7-3 "Section 4.2, Fuel System Design", Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, NUREG-0800, Rev. 2, US Nuclear Regulatory Commission, July 1981.

8.0 IMPROVED TECHNICAL SPECIFICATION CHANGES

Since the RFA design will be first implemented for Catawba 2 Cycle 11, changes to the current McGuire and Catawba Technical Specifications are not necessary. However, the following changes to the Improved Technical Specifications (ITS), originally submitted to the NRC on May 27, 1997 with numerous supplements submitted thereafter, are necessary to license the RFA design.

Figure 2.1.1-1 (Reactor Core Safety Limits - Four Loops in Operation) will be modified to delete the 2455 psia safety limit line. This line is the current upper bound pressure at which power operation is permitted and is dependent on the pressure range of the critical heat flux (CHF) correlation used in DNBR analyses. The critical heat flux correlation of the resident Mark-BW fuel is applicable up to a pressure of 2455 psia. Deleting the 2455 psia safety limit line is necessary due to implementation of the WRB-2M CHF correlation for the RFA design, which has an upper range of 2425 psia (Reference 8-1). The 2400 psia safety limit line will remain as the upper bound safety limit line because it is within the range of the CHF correlations for the RFA and Mark-BW fuel designs.

ITS 4.2.1 will be revised to add ZIRLO™ cladding to the fuel assembly description. ITS 5.6.5 will be revised to add this topical report to the list of approved methodologies for McGuire and Catawba.

The nuclear design related Technical Specification limits were reviewed for transition and full core reloads comprised of the Westinghouse RFA design. The power distribution Technical Specifications for Fq and FΔH have a 2% factor in each specification's surveillances which is used to account for the possible increase in Fq and FΔH between flux maps. This factor for IFBA cores will have to be burnup dependent because of the increased burnout rate of the integral burnable absorber relative to the lumped burnable absorbers. The technical justification for this proposed change is given in Sections 8.1 and 8.2.

8.1 Technical Justification for Surveillance Requirement SR 3.2.1.2 and SR 3.2.1.3

$F_q(x,y,z)$ is measured periodically using the incore detector system to ensure that the value of the total peaking factor, F_q-RTP, assumed in the accident analysis is bounding. The frequency requirement for this measurement is 31 effective full power days (EFPD). To account for the possibility that $F_q(x,y,z)$ may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If the extrapolation of the measurement indicates that the $F_q(x,y,z)$ measurement would exceed the $F_q(x,y,z)$ limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval would be decreased based on available margin, or the $F_q(x,y,z)$ measurement would be increased by an appropriate penalty (currently 1.02) and compared against the $F_q(x,y,z)$ operational and RPS surveillance limits to ensure allowable total peaking limits are not exceeded.

Technical Specification surveillances SR 3.2.1.2 and SR 3.2.1.3 currently specify that the $F_q(x,y,z)$ measurement be increased by 1.02. This value was chosen because it bounded the maximum $F_q(x,y,z)$ increase in typical reload cores. However, for reactor cores containing 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. This penalty can be incorporated into either the $M_q(x,y,z)$ or $M_c(x,y,z)$ margin factors, or be provided in tabular form as a function of burnup.

It is proposed that this penalty factor be moved to the Core Operating Limits Report (COLR) in tabular form to facilitate cycle specific updates. Table 8-1 provides an example burnup dependent penalty factor that would replace the current 1.02 value. For burnup ranges where the increase in Fq over the 31 EFPD surveillance interval is less than 2.0%, the current 1.02 penalty factor will be maintained.

Relocation of this penalty factor to the Core Operating Limits Report (COLR) was included in TSTF-98 (Technical Specification Task Force), Revision 2. This generic change to NUREG-1431 was approved by the NRC in April 1998.

8.2 Technical Justification for Surveillance Requirement SR 3.2.2.2

The nuclear enthalpy rise hot channel factor, $F_{\Delta H}(x,y,)$, is measured periodically using the incore detector system to ensure that fuel design criteria are not violated and accident analysis assumptions are not violated. The frequency requirement for this measurement is 31 effective full power days (EFPD). To account for the possibility that $F_{\Delta H}(x,y)$ may increase between surveillances, a trend of the measurement is performed to determine the point where peaking would exceed allowable limits if the current trend continues. If the extrapolation of the measurement indicates that the $F_{\Delta H}(x,y)$ measurement would exceed the $F_{\Delta H}(x,y)$ surveillance limit prior to 31 EFPD beyond the most recent measurement, then either the surveillance interval would be decreased based on available margin, or the $F_{\Delta H}(x,y)$ measurement would be increased by an appropriate penalty (currently 1.02) and compared against the $F_{\Delta H}(x,y)$ surveillance limit to ensure allowable peaking limits are not exceeded.

Technical Specification surveillance SR 3.2.2.2 currently specifies that the $F_{\Delta H}(x,y)$ measurement be increased by 1.02. This value was chosen because it bounded the maximum $F_{\Delta H}(x,y)$ increase in typical reload cores. However, for reactor cores containing 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. This penalty can be incorporated into either the $F_{\Delta H}(x,y)$ surveillance limit or be provided in tabular form as a function of burnup.

It is proposed that this penalty factor be moved to the Core Operating Limits Report (COLR) in tabular form to facilitate cycle specific updates. Table 8-2 provides an example burnup dependent penalty factor that would replace the current 1.02 value. For burnup ranges where the increase in $F_{\Delta H}(x,y)$ over the 31 EFPD surveillance interval is less than 2.0%, the current 1.02 penalty factor will be maintained.

Relocation of this penalty factor to the Core Operating Limits Report (COLR) was included in TSTF-98 (Technical Specification Task Force), Revision 2. This generic change to NUREG-1431 was approved by the NRC in April 1998.

8.3 References

8-1 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.

Table 8-1

$F_q(x,y,z)$ Margin Decrease Over 31 EFPD Surveillance Interval

(Typical Values)

Burnup (EFPD)	$F_q(x,y,z)$ Margin Decrease Penalty Factor
4	2.00 %
12	2.28 %
25	3.31 %
50	3.45 %
100	3.24 %
200	2.00 %
EOC	2.00 %

Note: Linear interpolation of the penalty factors is adequate for surveillances performed at intermediate burnups.

Table 8-2

$F_{\Delta H}(x,y)$ Margin Decrease Over 31 EFPD Surveillance Interval

(Typical Values)

Burnup (EFPD)	$F_{\Delta H}(x,y)$ Margin Decrease Penalty Factor
4	2.00 %
12	2.40 %
25	2.50 %
50	2.60 %
100	2.15 %
200	2.00 %
EOC	2.00 %

Note: Linear interpolation of the penalty factors is adequate for surviellances performed at intermediate burnups.

DPC-NE-2009, Rev. 1 - List of Changes

<u>Page</u>

6-5	Added referral to references 6-27 and 6-39
6-25	Updated reference 6-25 to Rev. 1, July 1997
6-26	For reference 6-35, corrected proprietary topical report number and designated the 2nd report as a non-proprietary report
6-27	Added reference 6-39, an approved WCAP which was mistakenly left out of the original reference list

Section E

RAI Letters and Responses

1. December 9, 1998, NRC RAI letter (Catawba), P. S. Tam to G. R. Peterson
2. January 5, 1999, NRC RAI letter (McGuire), F. Rinaldi to H. B. Barron
3. January 28, 1999, letter responding to NRC RAI, M. S. Tuckman to NRC
4. April 7, 1999, letter responding to NRC RAI (Question 11), M. S. Tuckman to NRC

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.

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.

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.