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,

Original signed by:

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

DISTRIBUTION:

Docket File	ACRS
PUBLIC	OGC
PDII-1 R/F	GHill(4)
WBeckner,TSB	LBerry
COgle,RII	
RScholl (e-mail SE only)	

DF01

CP-1

DOCUMENT NAME: G:\PDII-1\MCGUIRE\m2411amd.wpd *See Previous Concurrence

To receive a copy of this document, indicate in the box C=Copy w/o attachment/enclosure E=Copy with attachment/enclosure N = No copy

OFFICE	PDII-1/PM		PDII-1/LA	E	OGC*		PDII-1/SC		PDII/D
NAME	FRinaldi:cn		CHawes		AHodgton		REmch		HBerkow
DATE	9/20/99		9/20/99		09/09/99		9/22/99		9/22/99

OFFICIAL RECORD COPY

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 188
License No. NPF-9

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

   A The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility), Facility Operating License No. NPF-9 filed by the Duke Energy Corporation (licensee) dated July 22, 1998, and supplemented by letters dated October 22, 1998, and January 28, May 6, June 24 and August 17, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraphs 2.C.(2) and 2.c.(13) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 188 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

   (13) Additional Conditions

   The Additional Conditions contained in Appendix C, as revised through Amendment No. 188, are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to beginning the installation of the Westinghouse fuel, currently projected to be Fuel Cycle 15.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Operating License and Technical
Specification Changes

Date of Issuance: September 22, 1999

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

DUKE ENERGY CORPORATION

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility), Facility Operating License No. NPF-17 filed by the Duke Energy Corporation (licensee) 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, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraphs 2.C.(2) and 2.C.(13) of Facility Operating License No. NPF-17 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 169 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

   (13) Additional Conditions

   The Additional Conditions contained in Appendix D, as revised through Amendment No. 169, are hereby incorporated into this license. Duke Energy Corporation shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to beginning the installation of the Westinghouse fuel, currently projected to be Fuel Cycle 14.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard L. Emch, Jr.

Richard L. Emch, Jr., Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Operating License and Technical Specification Changes

Date of Issuance: September 22, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 188

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

ATTACHMENT TO LICENSE AMENDMENT NO. 169

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of Appendices C (for Unit 1) and D (for Unit 2), Additional Conditions, of the operating licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
--	2

Replace the following pages of Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
2.0-2	2.0-2
2.0-3	-----
3.2.1-4	3.2.1-4
3.2.1-5	3.2.1-5
3.2.2-4	3.2.2-4
4.0-1	4.0-1
5.6-3	5.6-3
5.6-4	5.6-4

Replace the following pages of the Technical Specifications Bases document with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
B 3.2.1-11	B 3.2.1-11
B 3.2.2-9	B 3.2.2-9

APPENDIX C

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-9 (Continued)

Amendment Number	Additional Condition	Implementation Date
	The maximum rod average burnup for any rod shall be limited to 60 GWd/mtU until the completion of an NRC environmental assessment supporting an increased limit.	Within 30 days of date of amendment

APPENDIX D

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. NPF-17 (Continued)

Amendment Number	Additional Condition	Implementation Date
	The maximum rod average burnup for any rod shall be limited to 60 GWd/mtU until the completion of an NRC environmental assessment supporting an increased limit.	Within 30 days of date of amendment

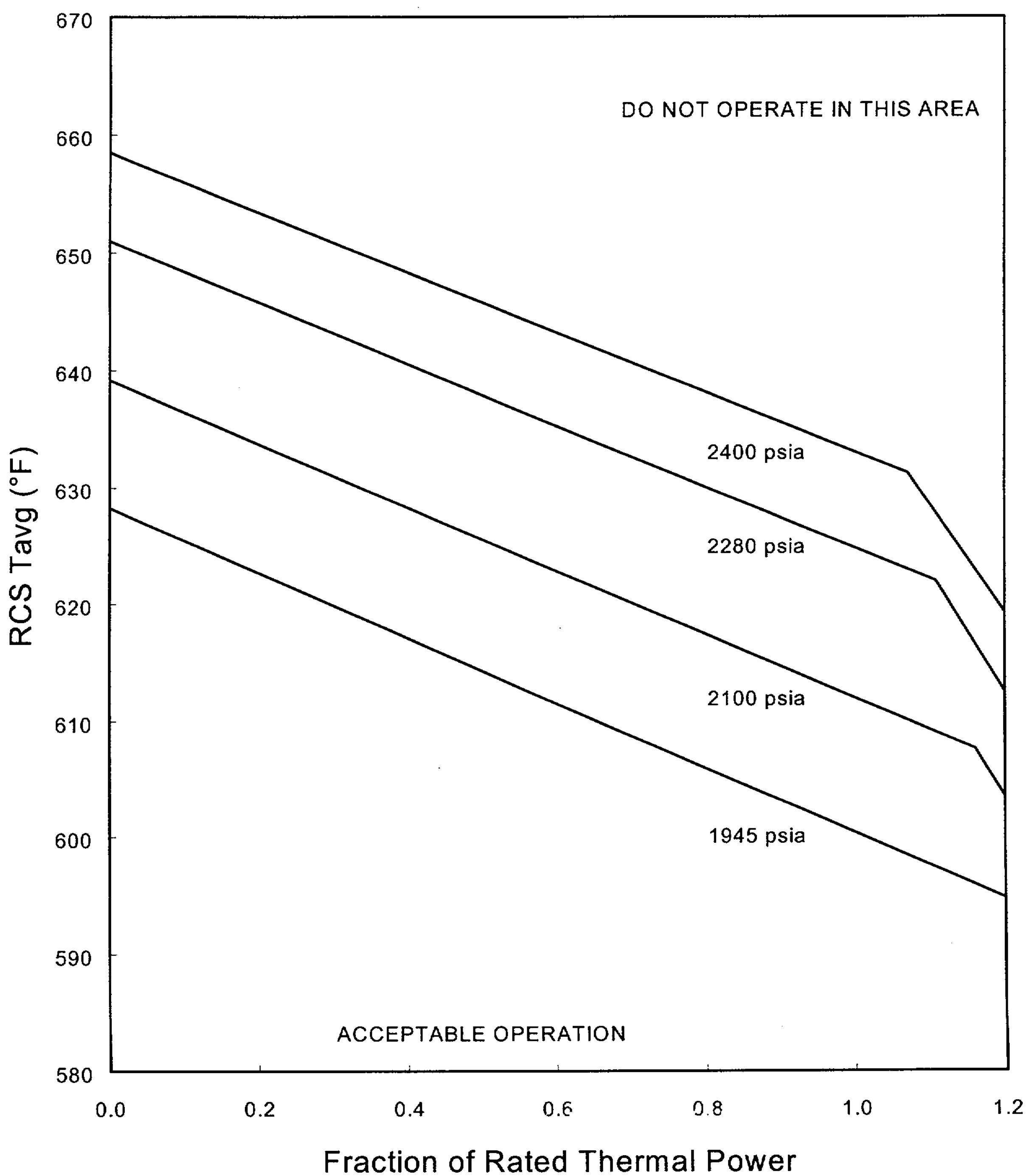

Figure 2.1.1-1

Reactor Core Safety Limits -
Four Loops in Operation

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	-----NOTE----- 1. Extrapolate $F^M_Q(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_Q(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_Q(X,Y,Z)_{EXTRAPOLATED} \geq F^L_Q(X,Y,Z)^{OP}{}_{EXTRAPOLATED}$, and $\frac{F^M_Q(X,Y,Z)_{EXTRAPOLATED}}{F^L_Q(X,Y,Z)^{OP}{}_{EXTRAPOLATED}} > \frac{F^M_Q(X,Y,Z)}{F^L_Q(X,Y,Z)^{OP}}$ then: a. Increase $F^M_Q(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$; or b. Repeat SR 3.2.1.2 prior to the time at which $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$ is extrapolated to not be met. 2. Extrapolation of $F^M_Q(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions. -----	
	Verify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{OP}$.	Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified <u>AND</u> 31 EFPD thereafter

(continued)

SURVEILLANCE REQUIREMENTS (continued)

<table>
<tr><th></th><th>SURVEILLANCE</th><th>FREQUENCY</th></tr>
<tr><td>SR 3.2.1.3</td><td>--------------------NOTES--------------------
1. Extrapolate $F^M_Q(X,Y,Z)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_Q(X,Y,Z)$ is within limits and the 31 EFPD extrapolation indicates:

$F^M_Q(X,Y,Z)_{EXTRAPOLATED} \geq F^L_Q(X,Y,Z)^{RPS}{}_{EXTRAPOLATED}$,

and

$\dfrac{F^M_Q(X,Y,Z)_{EXTRAPOLATED}}{F^L_Q(X,Y,Z)^{RPS}{}_{EXTRAPOLATED}} > \dfrac{F^M_Q(X,Y,Z)}{F^L_Q(X,Y,Z)^{RPS}}$

then:

a. Increase $F^M_Q(X,Y,Z)$ by the appropriate factor specified in the COLR and reverify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$; or

b. Repeat SR 3.2.1.3 prior to the time at which $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$ is extrapolated to not be met.

2. Extrapolation of $F^M_Q(X,Y,Z)$ is not required for the initial flux map taken after reaching equilibrium conditions.

Verify $F^M_Q(X,Y,Z) \leq F^L_Q(X,Y,Z)^{RPS}$.</td><td>Once within 12 hours after achieving equilibrium conditions after exceeding, by $\geq$ 10% RTP, the THERMAL POWER at which $F^M_Q(X,Y,Z)$ was last verified

<u>AND</u>

31 EFPD thereafter</td></tr>
</table>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	-----NOTES----- 1. Extrapolate $F^M_{\Delta H}(X,Y)$ using at least two measurements to 31 EFPD beyond the most recent measurement. If $F^M_{\Delta H}(X,Y)$ is within limits and the 31 EFPD extrapolation indicates: $F^M_{\Delta H}(X,Y)_{EXTRAPOLATED} \geq F^L_{\Delta H}(X,Y)^{SURV}{}_{EXTRAPOLATED}$ and $\dfrac{F^M_{\Delta H}(X,Y)_{EXTRAPOLATED}}{F^L_{\Delta H}(X,Y)^{SURV}{}_{EXTRAPOLATED}} > \dfrac{F^M_{\Delta H}(X,Y)}{F^L_{\Delta H}(X,Y)^{SURV}}$ then: a. Increase $F^M_{\Delta H}(X,Y)$ by the appropriate factor specified in the COLR and reverify $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$; or b. Repeat SR 3.2.2.2 prior to the time at which $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$ is extrapolated to not be met. 2. Extrapolation of $F^M_{\Delta H}(X,Y)$ is not required for the initial flux map taken after reaching equilibrium conditions. -----	
	Verify $F^M_{\Delta H}(X,Y) \leq F^L_{\Delta H}(X,Y)^{SURV}$.	Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F^M_{\Delta H}(X,Y)$ was last verified AND 31 EFPD thereafter

4.0 DESIGN FEATURES

4.1 Site Location

The McGuire Nuclear Station site is located at latitude 35 degrees, 25 minutes, 59 seconds north and longitude 80 degrees, 56 minutes, 55 seconds west. The Universal Transverse Mercator Grid Coordinates are E 504, 669, 256, and N 3, 920, 870, 471. The site is in northwestern Mecklenburg County, North Carolina, 17 miles north-northwest of Charlotte, North Carolina.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 193 fuel assemblies. Each assembly shall consist of a matrix of either ZIRLO™ or Zircalloy fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of ZIRLO™, zirconium alloy, or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 53 control rod assemblies. The control material shall be silver indium cadmium (Unit 1) silver indium cadmium and boron carbide (Unit 2) as approved by the NRC.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 4.75 weight percent;

b. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1 of the UFSAR;

(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.

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, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March, 1990 (DPC Proprietary).

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

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

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

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, "Westinghouse Fuel Transition Report, "SER dated September 22, 1999 (DPC Proprietary).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

channel factor to the surveillance limit is likely to decrease below the value of that ratio when the measurement was taken.

Each of these extrapolations is applied separately to the enthalpy rise hot channel factor surveillance limit. If both of the extrapolations are unfavorable, i.e., if the extrapolated factor is expected to exceed the extrapolated limit and the extrapolated factor is expected to become a larger fraction of the extrapolated limit than the measured factor is of the current limit, additional actions must be taken. These actions are to meet the $F^M_{\Delta H}(X,Y)$ limit with the last $F^M_{\Delta H}(X,Y)$ increased by the appropriate factor specified in the COLR or to evaluate $F^M_{\Delta H}(X,Y)$ prior to the point in time when the extrapolated values are expected to exceed the extrapolated limits. These alternative requirements attempt to prevent $F^M_{\Delta H}(X,Y)$ from exceeding its limit for any significant period of time without detection using the best available data. $F^M_{\Delta H}(X,Y)$ 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.

$F^M_{\Delta H}(X,Y)$ is verified at power levels 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F^M_{\Delta H}(X,Y)$ is within its limit at high 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 $F^M_{\Delta H}(X,Y)$ 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. UFSAR Section 15.4.8
2. 10 CFR 50, Appendix A, GDC 26.
3. 10 CFR 50.46.
4. 10 CFR 50.36, Technical Specifications, (c)(2)(ii).
5. DPC-NE-2005P, "Duke Power Company Thermal Hydraulic Statistical Core Design Methodology", September 1992.

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, 1974.
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 "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors", March 1990.

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 <u>Thermal-Hydraulic Analysis</u>

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 <u>REVIEW SUMMARY OF TOPICAL REPORT</u>

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 <u>STATE CONSULTATION</u>

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 <u>ENVIRONMENTAL CONSIDERATION</u>

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.

<u>ATTACHMENT 1</u>

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.