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DPC-NE-2009-A, Rev. 2

DUKE POWER COMPANY WESTINGHOUSE FUEL TRANSITION REPORT

Original Version Submitted : July 1998
Original Version Approved : September 22, 1999
Revision 1 Approved : October 1, 2002
Revision 2 Approved : December 18, 2002

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

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Revision 2 NRC Acceptance Letter and SER, Catawba and McGuire Nuclear Station

ECL

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

December 18, 2002

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

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: ACCEPTANCE FOR REFERENCING OF THE MODIFIED LICENSING TOPICAL REPORT, DPC-NE-2009P, REVISION 2 (TAC NOS. MB4502, MB4503, MB4504, AND MB4505)

Dear Mr. Tuckman:

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

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

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

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

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

Sincerely,

H E Edison for

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

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

Enclosure: As stated

cc w/encl: See next page

McGuire Nuclear Station
Catawba Nuclear Station

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

DUKE POWER COMPANY WESTINGHOUSE FUEL TRANSITION REPORT

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 and 2

DUKE ENERGY CORPORATION

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

1.0 INTRODUCTION

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

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

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

2.0 EVALUATION

Since the NRC has approved topical report DPC-2009-P-A, as well as Revision 1, the staff's review of Revision 2 was limited to those issues identified in Revision 2. The staff review of this revision is based on evaluation of technical merit and compliance with applicable regulatory requirements.

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

2.1 Changes to Section 2, Fuel Design:

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

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

- Adds Reference 2-6 [Westinghouse FCEP notification letter (Reference 9)] to Section 2.1, "References."

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

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

2.2 Changes to Section 4.0, Fuel Rod Analysis:

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

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

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

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

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

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

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

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

The reference pin and axial peaking factors and power distributions are used to determine the core DNB limits, which are the combinations of power and coolant inlet temperature and pressure at which the minimum DNBR equals the design DNBR limit. The design DNBR limit maintains a margin to the statistical DNBR limit, which is determined from the SCD. The DNB margin allows for mechanisms that could adversely impact DNB, such as the reactor coolant system flow anomaly and transition core effects. The licensee stated that the new higher reference radial and axial peaking factors are a result of applying, in core design space, the significant DNB margin realized from the intermediate flow mixing grids of the RFA design. With respect to radial peaking, all three models (8, 12, and 75-channel models) described in DPC-NE-2004P-A are based on the maximum pin power value, and therefore, Figures 5-1, 5-2, and 5-3 for these three models are updated to reflect the new peak pin value of 1.67.

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

2.4 Change in Critical Heat Flux Correlation:

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

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

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

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

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

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

2.5 Changes to Section 5.7, Transition Cores:

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

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

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

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

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

2.6 Typographic Error Corrections:

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

3.0 CONCLUSION

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

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

4.0 REFERENCES

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Principal Contributor: Y. Hsii, DSSA/SRXB

Date: December 18, 2002

Section B

Revision 1 NRC Acceptance Letter and SER, Catawba and McGuire Nuclear Station

EC 086 ~~[illegible]~~

D. R. Koontz

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

RECEIVED
OCT - 8 2002
REGULATORY COMPLIANCE

October 1, 2002

Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
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SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB3343 AND MB3344)

Dear Mr. Peterson:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 202 to Facility Operating License NPF-35 and Amendment No.195 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated October 7, 2001, as supplemented by letter dated August 7, 2002.

The amendments revise TS 5.6.5.a by adding a few parameter limits currently included in the Core Operating Limits Report. In addition to the license amendment request, you also submitted revisions to four previously approved topical reports for the Nuclear Regulatory Commission staff review and approval. The enclosed Safety Evaluation also address these topical reports.

A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Chandu P. Patel

Chandu P. Patel, Project Manager, Section 1
Project Directorate II /RA/
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosures:
1. Amendment No. 202 to NPF-35
2. Amendment No. 195 to NPF-52
3. Safety Evaluation

cc w/encls: See next page

Catawba Nuclear Station

cc:

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NPF-52

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letter dated October 7, 2001, as supplemented by letter dated August 7, 2002, Duke Energy Corporation, et al. (DEC, the licensee), submitted a request for changes to the Catawba Nuclear Station, Units 1 and 2, Technical Specifications (TS).

Revisions were proposed for TS 5.6.5.a, Item 1, to add the moderator temperature coefficient (MTC) 60 parts per million (ppm) surveillance limit. The specific value of the surveillance limit was previously relocated to the Core Operating Limits Report (COLR). Two new items were also proposed to be added to TS 5.6.5.a. These two items are (1) Item 12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," and (2) Item 13, "Reactor makeup water pumps combined flow rates limit for Specifications 3.3.9 and 3.9 2."

The initial submittal, dated October 7, 2001, proposed to change the dates and revision numbers for three of the Nuclear Regulatory Commission (NRC) approved analytical methods previously listed in TS 5.6.5.b, as listed below. The changes would reflect later versions of these topical reports that were also submitted with the October 7, 2001, submittal for NRC review and approval. As required by TS 5.6.5.b, only those methods listed within the TS as having been reviewed and approved by the NRC, can be used to determine the subject core operating limits. The subject core operating limits are listed in TS 5.6.5.a and their values are located in the COLR. A revision to a fourth report, DPC-NE-1003, was also submitted for NRC review and approval.

- DPC-NE-2009, Revision 1, "Duke Power Company Westinghouse Fuel Transition Report," August 2001.

- DPC-NF-2010, Revision 1, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," August 2001.

- DPC-NE-2011, Revision 1, "Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," August 2001.

- DPC-NE-1003, Revision 1, "McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing," August 2001.

The licensee in its letter of October 7, 2001, stated that, once approved, the approved topical report revisions, except for DPC-1003, Revision 1, will be listed in Section 5.6.5.b of the Catawba TS, to replace their respective original versions, and that the approved version of DPC-NE-2011-P, Revision 1, will also be listed in the references for TS Bases 3.2.1 and 3.2.3 to replace the existing reference to the original version, DPC-NE-2011-P-A.

However, on July 2, 2002, the NRC issued amendments numbered 199 and 192 to the Catawba Unit 1 and 2 operating licenses that effectively relocated the topical report revision numbers and dates from the TS 5.6.5.b list of approved methodologies to the COLR. Amendments 199 and 192 were consistent with the NRC Technical Specification Task Force (TSTF) Standard TS Traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR." Accordingly, since this portion of its request is no longer needed in view of amendments 199 and 192, the licensee's letter dated August 7, 2002, eliminated the requests to change TS 5.6.5.b and proposed revisions to BASES 3.2.1 and 3.2.3 to make its submittal consistent with the implementation of amendments 199 and 192 at the Catawba Nuclear Station. Nonetheless, this Safety Evaluation sets forth the NRC staff's evaluation of the licensee's proposed changes to the topical reports listed above.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulation* (10 CFR) Section 50.36 (c)(2)(ii)(B), Criterion 2 specifies that a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier must be included in the TS limiting conditions for operation (LCO). Accordingly, the reactor operating parameters, which are the initial conditions for the safety analyses of the design basis transients and accidents, are included in the TS LCO.

Since many parameters limits, such as core physics parameters, generally change with each reload core, licensees need to request TS amendments to update these parameters for each refueling cycle. NRC Generic Letter (GL) 88-16 (Ref. 4) provides guidance for relocating the values of the cycle-specific core operating parameter limits from TS to the COLR, and thus eliminates the unnecessary burden on the licensees and the NRC to update these limits in the TS each fuel cycle. The guidance includes adding the COLR in the TS administrative reporting requirement that also specifies (1) the cycle-specific parameters included in the COLR, and (2) the analytical methods that the NRC has previously reviewed and approved to be used to determine the core operating parameters limits.

The Catawba TS 5.6.5, "Core Operating Limits Report (COLR)," conforms to the GL 88-16 guidance. TS 5.6.5.a lists a set of parameters, including the reference to the actual TS number for each specified parameter. TS 5.6.5.b specifies the topical reports that are used for the determination of the core operating limits.

The proposed TS changes in this license amendment request are to revise the parameters listed in TS 5.6.5.a. These revisions are based on the guidance of GL 88-16.

3.0 STAFF EVALUATION

In this section, the staff will discuss the review of the revised versions of the four previously approved topical reports submitted for staff review, and the proposed TS changes.

3.1 Topical Reports Revisions

The licensee requested the NRC to review revisions of four topical reports that were previously approved and listed in TS 5.6.5.b as the approved methodologies used for the determination of the parameter limits in the COLR. Since the staff has reviewed and approved the original versions of these topical reports, the staff review of these revised versions will concentrate on the revisions made to the approved reports.

3.1.1 DPC-NE-2009, Revision 1

Topical report, DPC-NE-2009-P-A, (Ref. 5), provides general information about the Robust Fuel Assembly (RFA) design and describes methodologies used for reload design analyses to support the licensing basis for use of the RFA design in the McGuire and Catawba reload cores. These methodologies include fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The NRC approved the report in September 1999.

Revision 1 of DPC-NE-2009-A, as amended by the August 7, 2002, letter (Ref. 2), consists of the following minor changes to Chapter 6, "UFSAR Accident Analyses:"

(A) Update of the reference list in Section 6.7 as follows:

- Update reference 6-25, WCAP-10054-P-A Addendum 2, to Revision 1, dated July 1997.
- Correct reference 6-35, WCAP-8354, with proprietary topical report number, and designate the second report as a non-proprietary report.
- Add reference 6-39 a Westinghouse letter NSD-NRC-99-5839, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," dated July 15, 1999 (Ref. 6).

(B) Addition of a paragraph to Section 6.5.1, "Small Break LOCA," to explain that the Westinghouse small break LOCA NOTRUMP Evaluation Model includes the error corrections and model enhancements described in a few Westinghouse annual notifications required by 10 CFR 50.46, including the 1998 annual notification referenced in Reference 39.

The first two changes in the reference list are editorial and merely provide the latest version of the approved topical report or identify the proprietary and non-proprietary versions of a topical report. Reference 6-39, the Westinghouse letter NSD-NRC-99-5839, is the annual notification of the changes to the LOCA evaluation models during 1998. This notification documented the following error corrections or model enhancements to the NOTRUMP small break LOCA Evaluation Model:

- A programming error correction on the SBLOCTA rod-to-rod radiation model that is not modeled in licensing basis analyses and therefore, has no impact on the small break LOCA results.

- A logic simplification to the NOTRUMP droplet fall model that produces insignificant differences in results.

- A change in the reactor coolant pump heat in NOTRUMP that is not used in the evaluation model and therefore, has no impact on the small break LOCA results.

- A modification of NOTRUMP steam generator tube condensation heat transfer logic to a foreign plant that does not affect standard Westinghouse Pressurized Water Reactor calculations.

- An extension of reactor coolant conditions to allow for the NOTRUMP point kinetics calculations to be performed for cases that experience core uncovery conditions prior to reactor trip. For typical small break LOCA analyses, the reactor trips long before any threat of core uncovery and therefore, the change has no impact on peak cladding temperature calculations.

- A programming change in SBLOCTA code to allow for modeling of variable length blankets on either ends of the rod that involves no changes to the thermal-hydraulic fuel rod model, nor the solution technique.

Since the changes documented in the Westinghouse annual notice have insignificant impact on the small break LOCA analyses, the staff concludes the addition of Reference 6-39 is acceptable. Therefore, Revision 1 of DPC-NE-2009-P-A, as modified in the August 7, 2002, letter, is acceptable.

3.1.2 DPC-NF-2010A, Revision 1

Topical Report DPC-NF-2010A, (Ref. 7), describes Duke Power Company's Nuclear Design Methodology for McGuire and Catawba Nuclear Stations. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology the licensee intends to use to perform design calculations and to provide operational support, and the development of statistical factors.

Revision 1 of DPC-NF-2010A, updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as shutdown margin, boron and control rod worth, axial and radial peaking factors, and cycle length, as well as numerous editorial changes.

During the review, the staff also identified a few discrepancies associated with administrative changes. In response to the staff's request for additional information (Ref. 2), the licensee provided further changes to Revision 1 of the Topical report. These modifications include clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NF-2010A and the responses to the requests for additional information pertaining to these changes. The staff has concluded that the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report and that there are no unreviewed methodology or regulatory issues. Therefore, the staff finds the changes acceptable.

3.1.3 DPC-NE-2011, Revision 1

Topical Report DPC-NE-2011, (Ref. 9), describes the methodology for performing a maneuvering analysis for four-loop plants, such as McGuire and Catawba Nuclear Station. The licensee has developed this methodology as an alternate to the existing Relaxed Axial Offset Control Methodology. The licensee pointed out that this maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of the licensee's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower delta-temperature and over-temperature delta-temperature reactor protection system trip functions.

Revision 1 of DPC-NE-2011, updates the report to include editorial changes, and to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/SIMULATE-3P methodology (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as power peaking factors, axial and radial power distributions, and cycle length, as well as numerous editorial changes.

In response to the NRC staff's request for additional information (Ref. 2), the licensee provided additional information to the staff regarding cycle depletion times to clarify issues associated with power peaking versus burnup as a function of cycle time. The licensee's amendment request also included clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NE-2011-A and the responses to the requests for additional information pertaining to the requested changes. Since the changes to this topical report consists mostly of administrative changes and clarifications to the original NRC approved topical report, the staff find the changes acceptable.

3.1 4 DPC-NE-1003, Revision 1

Topical Report DPC-NE-1003 (Ref. 10) describes the measurement procedure used to determine the inferred bank worth and the calculation procedures used to develop the rod swap correction factor that accounts for the effect of a test bank on the partial integral worth of the reference bank. The NRC approved the report in May 1987 (Ref. 11) for rod worth measurement of reload cores for McGuire and Catawba Stations, Units 1 and 2.

Revision 1 of DPC-NE-1003 updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect the revision of the rod swap measurement procedures, and various editorial changes. In response to staff questions, the licensee, in its letter of August 7, 2002, provided the current version of the control rod worth measurement rod swap procedures, PT/0/A/4150/11A, dated January 19, 1996. The staff review of this current control rod worth measurement procedure has found it acceptable. The licensee in the August 7, 2002, letter also modified the equation in Section 3 of the topical report for the calculation of the inferred rod bank worth from the measured reference bank worth and bank height. This change is consistent with the equation described in step 12.12.5 of the current measurement procedures of January 19, 1996. Therefore, Revision 1 of DPC-NE-1003, as modified in the August 7, 2002, letter, is acceptable.

3.2 Proposed TS Changes

This section addresses the staff's evaluation of the proposed changes to TS 5.6.5.a regarding the cycle-specific operating parameters specified in the COLR. The staff review of these TS changes are based on the guidance of GL 88-16.

TS 5.6.5.a provides a list of core operating limits that are established prior to each reload cycle, or prior to any remaining portion of a reload cycle. The valves of the limits are in the COLR. For Catawba Units 1 and 2, the licensee proposed to revise the list by:

(1) adding "60 ppm" to Item 5.6.5.a.1 regarding the moderator temperature coefficient (MTC) surveillance limit for Specification 3.1.3,

(2) adding Item 5.6.5.a.12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," and

(3) adding Item 5.6.5.a.13, "Reactor makeup water pumps combined flow rates limit for Specifications 3.3.9 and 3.9.2."

These changes are evaluated below.

3.2.1 MTC 60 ppm Surveillance Limit

Catawba TS LCO 3.1.3 specifies that the MTC be maintained within the LCO limits, which are based on the safety analysis assumptions. For verification that these LCO limits are met, the Surveillance Requirements of TS 3.1.3 also places surveillance limits for conducting the end of cycle MTC measurement at 300 ppm and 60 ppm boron concentration. The LCO limits and the 300-ppm and 60-ppm surveillance limits are specified in the COLR. However, TS Item 5.6.5.a.1 operating limits does not currently identify the 60-ppm surveillance limit.

The proposed change to the Catawba TS would add the 60-ppm surveillance limit in Item 5.6.5.a.1. The new TS would read "Moderator Temperature Coefficients BOL and EOL limits and 60 ppm and 300 ppm surveillance limit for Specification 3.1.3." The NRC approved incorporating the 60-ppm surveillance limits into the COLR during the Improved Technical Specifications conversion in 1998 (Ref. 12 and 13); however, reference to this surveillance was not included in TS Item 5.6.5.a.1 at that time. The proposed TS change to include the 60-ppm surveillance limit in TS Item 5.6.5.a.1 provides consistency with previously approved requirements and, therefore, it is acceptable.

3.2.2 Relocation of Hot Channel Factors Surveillance Penalty Factors to COLR

Surveillance Requirements in TS 3.2.1 and 3.2.2, respectively, require that the heat flux hot channel factor, F_q (x,y,z), and the enthalpy rise hot channel factor, $F_{\Delta h}$ (x,y), be measured every 31 effective full power days (EFPD) during equilibrium conditions using the incore detector system to verify they are within the respective limits. To address the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, penalty factors are applied to these hot channel factors if their margins to the respective limits have decreased since the previous surveillance. These margin-decrease penalty factors are calculated by projecting the limiting hot channel factors over the 31 EFPD surveillance intervals with the maximum changes at the limiting core location, and are based on reload core design. In Section 8, "Improved Technical Specification Changes," of DPC-NE-2009, the licensee proposed to replace the penalty factors with tables of penalty value as functions of burnup in the COLR to facilitate cycle-specific updates. TS Item 5.6.5.b.14 lists topical report DPC-NE-2009-P-A that includes (in response to a staff question during the review of DPC-NE-2009) the approved methodology used to calculate these burnup-dependent penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin decrease penalty factors in the COLR acceptable as stated in the staff's safety evaluation supporting license amendment Nos. 180 and 172, respectively for Catawba Units 1 and 2 (Ref. 15).

The proposed changes to the Catawba TS would add Item 5.6.5.a.12, that reads: "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2." The addition of TS Item 5.6.5.a.12 would make it consistent with the previous staff approval of including these surveillance penalty factors in the COLR and, therefore, this proposed change is acceptable.

3.2.3 Reactor Makeup Water Pumps Combined Flow Rates Limit

The relocation of the reactor makeup water pumps combined flow rates limit for the boron dilution mitigation system from Catawba TS 3.3.9 and 3.9.2 to the COLR was approved by the NRC as described in a letter dated March 25, 1994 (Ref. 16). The reactor makeup water pumps flow rate limit is included in the Catawba COLR.

The proposed changes to the Catawba TS would add Item 5.6.5.a.13, "Reactor makeup water pumps combined flow rates limit for Specification 3.3.9 and 3.9.2," to TS 5.6.5.a. The addition of this item would make the TS 5 6 5.a list consistent with the core operating limits included in the Catawba COLR and is therefore, acceptable.

4.0 <u>SUMMARY</u>

The staff has reviewed the revisions of four previously approved topical reports described in Section 1.0 of this Safety Evaluation, and the proposed changes to Catawba Nuclear Station, Units 1 and 2, TS 5.6.5.a related to the COLR. Based on our evaluation described in Section 3 of this Safety Evaluation, the staff concludes that the these topical report revisions, as amended by the August 7, 2002, letter, and the TS changes are acceptable.

5.0 <u>STATE CONSULTATION</u>

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding [67 FR 54680]. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370; License Amendment Request Applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to BASES 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003," October 7, 2001.

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

3. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report - Oconee, McGuire, and Catawba Technical Specifications 5.6.5," December 20, 2001.

4. Letter from Dennis Crutchfield, USNRC, to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988.

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

6. Letter from J. S. Galembush, Westinghouse Electric Company, to US Nuclear Regulatory Commission, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," NSD-NRC-99-5839, July 15, 1999.

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

8. DPC-NE-1004A, Revision 1, "Nuclear Design Methodology Using CASMO-3/ SIMULATE-3P," SER dated April 26, 1997.

9. DPC-NE-2011, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," March 1990.

10. DPC-NE-1003, "Rod Swap Methodology Report for Startup Physics Testing," December 1986.

11. Letter from Darl Hood, USNRC, to H. B. Tucker, Duke Power Company, "Rod Swap Methodology Report for Startup Physics Testing, McGuire and Catawba Nuclear Stations, Units 1 and 2 (TACs 62981, 62982, 62983, 62984)," May 22, 1987.

12. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "Issuance of Amendments - McGuire Nuclear Station, Units 1 and 2, (TAC Nos. M98964 and M98965)," September 30, 1998.

13. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M95298 and M95299)," September 30, 1998.

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

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

16. Letter from Robert F. Martin, USNRC, to David L. Rehn, Catawba Site, Duke Power Company, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473)," March 25, 1994.

Principal Contributor: Y. Hsii
A Attard

Date: October 1, 2002

DRKoontz

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001
October 1, 2002

Mr. H. B. Barron
Vice President, McGuire Site
Duke Energy Corporation
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

SUBJECT: McGUIRE NUCLEAR STATION, UNITS 1AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MB3222 AND MB3223)

Dear Mr. Barron:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 208 to Facility Operating License NPF-9 and Amendment No. 189 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated October 7, 2001, as supplemented by letter dated August 7, 2002.

The amendments revise TS 5.6.5.a by adding a few parameter limits currently included in the Core Operating Limits Report. In addition to the license amendment request, you also submitted revisions to four previously approved topical reports for the Nuclear Regulatory Commission staff review and approval. The enclosed Safety Evaluation also addresses these topical reports.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Robert Martin

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

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. 208 to NPF-9
2. Amendment No. 189 to NPF-17
3. Safety Evaluation

cc w/encls. See next page

McGuire Nuclear Station

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CORPORATION

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated October 7, 2001, as supplemented by letter dated August 7, 2002, Duke Power Company, et al. (DPC, the licensee), submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS).

Revisions were proposed for TS 5.6.5.a, Item 1, to add the moderator temperature coefficient (MTC) 60 parts per million (ppm) surveillance limit. The specific value of the surveillance limit was previously relocated to the Core Operating Limits Report (COLR). A new item 12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2," is also proposed to be added to TS 5.6.5.a.

The initial submittal, dated October 7, 2001, proposed to change the dates and revision numbers for three of the Nuclear Regulatory Commission (NRC) approved analytical methods previously listed in TS 5.6.5.b, as listed below. The changes would reflect later versions of these topical reports that were also submitted with the October 7, 2001, submittal for NRC review and approval. As required by TS 5.6.5.b, only those methods listed within the TS as having been reviewed and approved by the NRC, can be used to determine the subject core operating limits. The subject core operating limits are listed in TS 5.6.5.a and their values are located in the COLR. A revision to a fourth report, DPC-NE-1003, was also submitted for NRC review and approval.

- DPC-NE-2009, Revision 1, "Duke Power Company Westinghouse Fuel Transition Report," August 2001.
- DPC-NF-2010, Revision 1, "Duke Power Company McGuire Nuclear Station and Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," August 2001.
- DPC-NE-2011, Revision 1, "Duke Power Company Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," August 2001.
- DPC-NE-1003, Revision 1, "McGuire Nuclear Station and Catawba Nuclear Station Rod Swap Methodology Report for Startup Physics Testing," August 2001

The licensee in its letter of October 7, 2001, stated that, once approved, the approved topical report revisions, except for DPC-1003, Revision 1, will be listed in Section 5.6.5.b of the McGuire TS, to replace their respective original versions, and that the approved version of DPC-NE-2011-P, Revision 1, will also be listed in the references for TS Bases 3.2.1 and 3.2.3 to replace the existing reference to the original version, DPC-NE-2011-P-A.

However, on July 10, 2002, the NRC issued amendments numbered 203 and 184 to the McGuire Unit 1 and 2 operating licenses that effectively relocated the topical report revision numbers and dates from the TS 5.6.5.b list of approved methodologies to the COLR. Amendments 203 and 184 were consistent with the NRC Technical Specification Task Force (TSTF) Standard TS Traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5 COLR." Accordingly, since this portion of its request is no longer needed in view of amendments 203 and 184, the licensee's letter dated August 7, 2002, eliminated the requests to change TS 5.6.5.b and proposed revisions to BASES 3.2.1 and 3.2.3 to make its submittal consistent with the implementation of amendments 203 and 184 at the McGuire Nuclear Station. Nonetheless, this Safety Evaluation sets forth the NRC staff's evaluation of the licensee's proposed changes to the topical reports listed above.

2.0 BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36 (c)(2)(ii)(B), Criterion 2, specifies that a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier must be included in the TS limiting conditions for operation (LCO). Accordingly, the reactor operating parameters, which are the initial conditions for the safety analyses of the design basis transients and accidents, are included in the TS LCOs.

Since many parameter limits, such as core physics parameters, generally change with each reload core, licensees previously needed to request TS amendments to update these parameters for each refueling cycle. NRC Generic Letter (GL) 88-16 (Ref. 4) provides guidance for relocating the values of the cycle-specific core operating parameter limits from TS to the COLR, thus eliminating unnecessary burden on the licensees and the NRC to update these limits in the TS for each fuel cycle. The guidance includes adding the COLR in the TS administrative reporting requirement that also specifies (1) the cycle-specific parameters included in the COLR, and (2) the analytical methods that the NRC has previously reviewed and approved to be used to determine the core operating parameters limits.

The McGuire TS 5.6.5, "Core Operating Limits Report (COLR)," conforms to GL 88-16 guidance. TS 5.6.5.a lists a set of parameters, including the reference to the actual TS number for each specified parameter. TS 5.6 5.b specifies the topical reports that are used for the determination of the core operating limits.

The proposed TS changes in this license amendment request are to revise the parameters listed in TS 5 6 5.a. These revisions are based on the guidance of GL 88-16.

3.0 STAFF EVALUATION

In this section, the staff will discuss the review of the revised versions of the four previously approved topical reports submitted for staff review, and the proposed TS changes.

3.1 Topical Reports Revisions

The licensee requested the NRC to review revisions to four topical reports that were previously approved and listed in TS 5.6.5.b as the approved methodologies used for the determination of the parameter limits in the COLR. Since the staff has reviewed and approved the original versions of these topical reports, the staff review of these revised versions concentrated on the revisions made to the approved reports.

3.1.1 DPC-NE-2009, Revision 1

Topical report, DPC-NE-2009-P-A, (Ref. 5), provides general information about the Robust Fuel Assembly (RFA) design and describes methodologies used for reload design analyses to support the licensing basis for use of RFAs in the McGuire and Catawba reload cores. These methodologies include fuel rod mechanical reload analysis methodology and the core design, thermal-hydraulic analysis, and accident analysis methodologies. The NRC approved the report in September 1999.

Revision 1 of DPC-NE-2009, as amended by the August 7, 2002, letter (Ref. 2), consists of the following minor changes to its Chapter 6, "UFSAR Accident Analyses."

(A) Update of the reference list in Section 6.7 as follows:

- Update reference 6-25, WCAP-10054-P-A Addendum 2, to Revision 1, dated July 1997.
- Correct reference 6-35, WCAP-8354, with proprietary topical report number, and designate the second report as a non-proprietary report.
- Add reference 6-39, Westinghouse letter NSD-NRC-99-5839, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," dated July 15, 1999 (Ref. 6).

(B) Addition of a paragraph to Section 6.5.1, "Small Break LOCA," to explain that the Westinghouse small break LOCA NOTRUMP Evaluation Model includes the error corrections and model enhancements described in a few Westinghouse annual notifications required by 10 CFR 50.46, including the 1998 annual notification referenced in Reference 39.

The first two changes in the reference list are editorial and merely provide the latest version of the approved topical report or identify the proprietary and non-proprietary versions of a topical report. Reference 6-39, Westinghouse letter NSD-NRC-99-5839, is the annual notification of the changes to the LOCA evaluation models during 1998. This notification documented the following error corrections or model enhancements to the NOTRUMP small break LOCA Evaluation Model:

- A programming error correction on the SBLOCTA rod-to-rod radiation model, that is not modeled in licensing basis analyses and therefore, has no impact on the small break LOCA results.

- A logic simplification to the NOTRUMP droplet fall model that produces insignificant differences in results.

- A change in the reactor coolant pump heat in NOTRUMP that is not used in the evaluation model and therefore, has no impact on the small break LOCA results.

- A modification of NOTRUMP steam generator tube condensation heat transfer logic for a foreign plant that does not affect standard Westinghouse Pressurized Water Reactor calculations.

- An extension of reactor coolant conditions to allow for the NOTRUMP point kinetics calculations to be performed for cases that experience core uncovery conditions prior to reactor trip. For typical small break LOCA analyses, the reactor trips long before any threat of core uncovery and therefore, the change has no impact on peak cladding temperature calculations.

- A programming change in SBLOCTA code to allow for modeling of variable length blankets on either ends of the rod that involves no changes to the thermal-hydraulic fuel rod model, nor the solution technique.

Since the changes documented in the Westinghouse annual notice have insignificant impact on the small break LOCA analyses, the staff concludes the addition of Reference 6-39 is acceptable. Therefore, Revision 1 of DPC-NE-2009-P-A, as modified in the August 7, 2002, letter, is acceptable.

3.1.2 DPC-NF-2010, Revision 1

Topical Report DPC-NF-2010, (Ref. 7), describes DPC's Nuclear Design Methodology for McGuire and Catawba Nuclear Stations. The nuclear design process consists of mechanical properties used as nuclear design input, the nuclear code system and methodology that DPC intends to use to perform design calculations and to provide operational support, and the development of statistical factors.

Revision 1 of DPC-NF-2010, updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/ SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as shutdown margin, boron and control rod worth, axial and radial peaking factors, and cycle length, as well as numerous editorial changes.

During the review, the staff also identified a few discrepancies associated with administrative changes. In response to the staff's request for additional information (Ref. 2), the licensee provided further changes to Revision 1 of the topical report These modifications include clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NF-2010 and the responses to the requests for additional information pertaining to these changes. The staff has concluded that the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report and that there are no unreviewed methodology or regulatory issues. Therefore, the staff finds the changes to be acceptable.

3.1.3 DPC-NE-2011, Revision 1

Topical Report DPC-NF-2011, (Ref. 9), describes the methodology for performing a maneuvering analysis for four-loop plants, such as the McGuire and Catawba Nuclear Stations. The licensee has developed this methodology as an alternate to the existing Relaxed Axial Offset Control (RAOC) Methodology. The licensee pointed out that this maneuvering analysis results in several advantages: more flexible and prompt engineering support for the operating stations, consistency with the methods of the licensee's nuclear design process, and potential increases in available margin through the use of three-dimensional monitoring techniques. The increase in margin occurs in limits on power distribution, control rod insertion, and power distribution inputs to the overpower delta-temperature and over-temperature delta-temperature reactor protection system (RPS) trip functions.

Revision 1 of DPC-NE-2011, updates the report to include editorial changes, and to permit the use of certain methods approved subsequent to the implementation of the original version, such as the CASMO-3/SIMULATE-3P methodology (Ref. 8). Other changes are made to reflect revisions to the core design parameters such as power peaking factors, axial and radial power distributions, and cycle length, as well as numerous editorial changes.

In response to the NRC staff's request for additional information (Ref. 2), the licensee provided additional information regarding cycle depletion times to clarify issues associated with power peaking versus burnup as a function of cycle time. The licensee's amendment request also included clarifications to revised sections and minor changes to equations. The NRC staff has reviewed the analyses associated with the changes to Topical Report DPC-NE-2011-A and the responses to the requests for additional information pertaining to the requested changes. Since the changes to this topical report consist mostly of administrative changes and clarifications to the original NRC approved topical report, the staff finds the changes to be acceptable.

3.1.4 DPC-NE-1003, Revision 1

Topical Report DPC-NE-1003 (Ref. 10), describes the measurement procedure used to determine the inferred bank worth and the calculation procedures used to develop the rod swap correction factor that accounts for the effect of a test bank on the partial integral worth of the reference bank. The NRC approved the report in May 1987 (Ref. 11) for rod worth measurement of reload cores for McGuire and Catawba Stations, Units 1 and 2.

Revision 1 of DPC-NE-1003 updates the report to permit the use of certain methods approved subsequent to the implementation of the original version, such as the use of CASMO-3/SIMULATE-3P reactor physics methods (Ref. 8). Other changes are made to reflect the revision of the rod swap measurement procedures, and various editorial changes. In response to staff questions, the licensee, in its letter of August 7, 2002, provided the current version of the control rod worth measurement rod swap procedures, PT/0/A/4150/11A, dated January 19, 1996. The staff review of this current control rod worth measurement procedure has found it to be acceptable The licensee, in the August 7, 2002, letter also modified the equation in Section 3 of the topical report for the calculation of the inferred rod bank worth from the measured reference bank worth and bank height. This change is consistent with the equation described in step 12.12.5 of the current measurement procedures of January 19, 1996. Therefore, Revision 1 of DPC-NE-1003, as modified in the August 7, 2002, letter, is acceptable.

3.2 Proposed TS Changes

This section addresses the staff's evaluation of the proposed changes to TS 5.6.5.a regarding the cycle-specific operating parameters specified in the COLR. The staff review of these TS changes are based on the guidance of GL 88-16.

TS 5.6.5.a provides a list of core operating limits that are established prior to each reload cycle, or prior to any remaining portion of a reload cycle. The values of the limits are located in the COLR. For McGuire Nuclear Station, Units 1 and 2, the licensee proposed to revise the list by:

(1) adding "60 ppm" to Item 5.6.5.a.1 regarding the moderator temperature coefficient (MTC) surveillance limit for Specification 3.1.3, and

(2) adding Item 5.6.5.a.12, "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2."

These changes are evaluated below.

3 2.1 MTC 60 ppm Surveillance Limit

McGuire TS LCO 3.1.3 specifies that the MTC be maintained within the LCO limits, which are based on the safety analysis assumptions. For verification that these LCO limits are met, the Surveillance Requirements of TS 3.1.3 also place surveillance limits for conducting the end of cycle MTC measurement at boron concentrations of 300 ppm and 60 ppm. The LCO limits and the 300 ppm and 60 ppm surveillance limits are specified in the COLR. However, TS Item 5.6.5.a.1 operating limits does not currently identify the 60-ppm surveillance limit.

The proposed change to the McGuire TS would add the 60 ppm surveillance limit in Item 5.6 5 a.1. The new TS would read "Moderator Temperature Coefficients BOL and EOL limits and 60 ppm and 300 ppm surveillance limit for Specification 3.1.3." The NRC approved incorporating the 60-ppm surveillance limits into the COLR during the Improved Technical Specifications conversion in 1998 (Ref. 12 and 13); however, reference to this surveillance was not included in TS Item 5.6.5 a.1 at that time. The proposed TS change to include the 60 ppm surveillance limit in TS Item 5.6.5.a 1 provides consistency with previously approved requirements and, therefore, it is acceptable.

3 2 2 Relocation of Hot Channel Factors Surveillance Penalty Factors to COLR

Surveillance Requirements in TS 3.2.1 and 3.2.2, respectively, require that the heat flux hot channel factor, F_q (x,y,z), and the enthalpy rise hot channel factor, $F_{\Delta h}$ (x,y), be measured every 31 effective full power days (EFPD) during equilibrium conditions using the incore detector system to verify they are within the respective limits. To address the possibility that these hot channel factors may increase and exceed their allowable limits between surveillances, penalty factors are applied to these hot channel factors if their margins to the respective limits have decreased since the previous surveillance These margin-decrease penalty factors are calculated by projecting the limiting hot channel factors over the 31 EFPD surveillance intervals with the maximum changes at the limiting core location, and are based on reload core design. In Section 8, "Improved Technical Specification Changes," of DPC-NE-2009, the licensee proposed to replace the penalty factors with tables of penalty value as a function of burnup in the COLR to facilitate cycle-specific updates. TS Item 5.6.5.b.14 lists topical report DPC-NE-2009-P-A that includes (in response to a staff question during the review of DPC-NE-2009) the approved methodology used to calculate these burnup-dependent penalty factors. The staff found the methodology and the inclusion of the burnup-dependent margin decrease penalty factors in the COLR acceptable, as stated in the staff's Safety Evaluation supporting license Amendment Nos. 188 and 169, respectively, for McGuire Nuclear Station, Units 1 and 2 (Ref. 14).

The proposed changes to the McGuire TS would add Item 5.6.5.a.12 that reads: "31 EFPD surveillance penalty factors for Specifications 3.2.1 and 3.2.2." The addition of TS Item 5.6.5.a.12 would make it consistent with the previous staff approval of including these surveillance penalty factors in the COLR and, therefore, this proposed change is acceptable.

4.0 SUMMARY

The staff has reviewed the revisions to four previously approved topical reports described in Section 1.0 of this Safety Evaluation, and the proposed changes to McGuire Nuclear Station, Units 1 and 2, TS 5.6.5.a related to the COLR. Based on our evaluation, described in Section 3 of this Safety Evaluation, the staff concludes that the these topical report revisions, as amended by the August 7, 2002, letter, and the TS changes are acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change recordkeeping, reporting, or administrative procedure 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 Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (67FR 54680). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

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

8.0 REFERENCES

1. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "Duke Energy Corporation; Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414; McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370; License Amendment Request Applicable to Technical Specifications 5.6.5, Core Operating Limits Report; Revisions to BASES 3.2.1 and 3.2.3; and Revisions to Topical Reports DPC-NE-2009-P, DPC-NF-2010, DPC-NE-2011-P, and DPC-NE-1003," October 7, 2001.

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

3. Letter from M. S. Tuckman, Duke Energy Corporation, to US Nuclear Regulatory Commission, "License Amendment Request Applicable to the Technical Specifications Requirements for the Core Operating Limits Report - Oconee, McGuire, and Catawba Technical Specifications 5.6.5," December 20, 2001.

4. Letter from Dennis Crutchfield, USNRC, to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988.

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

6 Letter from J. S. Galembush, Westinghouse Electric Company, to US Nuclear Regulatory Commission, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10 CFR 50.46(a)(3)(ii)," NSD-NRC-99-5839, July 15, 1999.

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

8. DPC-NE-1004A, Revision 1, "Nuclear Design Methodology Using CASMO-3/ SIMULATE-3P," SER dated April 26, 1997.

9 DPC-NE-2011, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," March 1990

10. DPC-NE-1003, "Rod Swap Methodology Report for Startup Physics Testing," December 1986.

11. Letter from Darl Hood, USNRC, to H. B. Tucker, Duke Power Company, "Rod Swap Methodology Report for Startup Physics Testing, McGuire and Catawba Nuclear Stations, Units 1 and 2 (TACs 62981, 62982, 62983, 62984)," May 22, 1987.

12. Letter from Frank Rinaldi, USNRC, to H. B. Brown, McGuire Site, Duke Energy Corporation, "Issuance of Amendments - McGuire Nuclear Station, Units 1 and 2, (TAC Nos. M98964 and M98965)," September 30, 1998.

13. Letter from Peter Tam, USNRC, to G. R. Peterson, Catawba Nuclear Station, Duke Energy Corporation, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 (TAC Nos. M95298 and M95299)," September 30, 1998.

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

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

16. Letter from Robert F. Martin, USNRC, to David L. Rehn, Catawba Site, Duke Power Company, "Issuance of Amendments - Catawba Nuclear Station, Units 1 and 2 Cycle Specific Parameters to the Core Operating Limits Report (TAC Nos. M85472 and M85473)," March 25, 1994.

Principal Contributor: Y. Hsii
A. Attard

Date: October 1, 2002

Section C

Original Report NRC Acceptance Letter and SER, Catawba and McGuire Nuclear Station

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

September 22, 1999

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

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA2359 AND MA2361)

Dear Mr. Peterson:

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

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

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

Sincerely,

Peter S. Tam

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

Docket Nos. 50-413 and 50-414

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

cc w/encls: See next page

Catawba Nuclear Station

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

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

DUKE ENERGY CORPORATION, ET AL.

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION AND BACKGROUND

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

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

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

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

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

* 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 UFSAR Accident Analyses

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

2.4.1 LOCA Analyses:

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

2.4.2 Non-LOCA Transient and Accident Analyses:

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

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

2.4.3 Rod Ejection Accident Analysis Using SIMULATE-3K:

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

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

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

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

Section 6.6.2 2 of DPC-NE-2009 describes the use of the SIMULATE-3K code to perform license analysis of the design basis REA. The basic methodology as described in

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

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

The SIMULATE-3K REA analysis is performed at four statepoints: BOC and EOC at HZP and HFP conditions for the determination of three-dimensional steady-state and transient power distributions, as well as individual pin powers. Conservative input parameters are used to ensure that the rod ejection analysis produces limiting results that bound future reload cycles. Sections 6.6.2.2.1 and 6 6.2.2.2 describe the methods to ensure conservatism in the analysis of transient response by increasing the fission cross sections in the ejected rod locations and in each assembly and by applying the "factors of conservatism" to the reactivity feedback for moderator and fuel temperatures, control rod worths for withdrawal and insertion, effective delayed neuron, and ejected rod worth, etc. In response to a staff question (No. 9, Ref. 14), the licensee provided a description of the method of determining the "factors of conservatism." The staff has reviewed the overall SIMULATE-3K methodology, and found it to be acceptable for application to the REA analyses for McGuire and Catawba

2.4 4 Compliance with Safety Evaluation Conditions

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

2.5 Fuel Assembly Repair and Reconstitution

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

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

2.6 Technical Specifications Changes

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

2.6.1 Proposed Change to TS Figure 2.1.1-1:

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

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

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

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

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

2.6.3 Proposed Change to TS 4.2.1:

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

2.6.4 Proposed Changes to Section 5.6.5b:

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

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

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

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

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

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

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

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

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

2.6.5 Proposed Changes to the Technical Specifications Bases Document:

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

3.0 REVIEW SUMMARY OF TOPICAL REPORT

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

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

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

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

6.0 CONCLUSION

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

Attachment: Technical Evaluation Report

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Date: September 22, 1999

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ATTACHMENT 1

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

PREPARED BY
PACIFIC NORTHWEST NATIONAL LABORATORY

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

"Duke Power Company Westinghouse Fuel Transition Report"

1.0 INTRODUCTION

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

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

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

Another W analysis method used is:

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

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

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

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

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

2.1 Fuel Rod Cladding Stress

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

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

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

2.2 Fuel Rod Cladding Strain

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

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

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

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

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

2.3 Fuel Rod Cladding Strain Fatigue

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

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

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

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

2.4 Fuel Rod Internal Pressure

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

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

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

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

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

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

2.5 Fuel Temperature

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

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

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

2.6 Fuel Clad Oxidation and Hydriding

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

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

2.7 Fuel Rod Axial Growth

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

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

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

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

3.0 CONCLUSIONS

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

4.0 REFERENCES

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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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

AND AMENDMENT NO 169 TO FACILITY OPERATING LICENSE NPF-17

DUKE ENERGY CORPORATION, ET AL.

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS 50-369 AND 50-370

1.0 INTRODUCTION AND BACKGROUND

By letter dated July 22, 1998 (Ref 1), and supplemented by a letter of October 22, 1998 (Ref 2), Duke Energy Corporation* (DEC, the licensee), the licensee for operation of McGuire and Catawba Nuclear Stations, proposed changes to the Technical Specifications (TS) of these plants in anticipation of a reactor core reload design using Westinghouse fuel. Accompanying the July 22, 1998, letter is a topical report DPC-NE-2009, "Duke Power Company* Westinghouse Fuel Transition Report," (Ref. 3) for NRC review and approval When approved, this topical report will be listed in Section 5 6.5 of the Catawba and McGuire TSs as an approved methodology for the determination of the core operating limits.

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

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

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

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

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

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

2.0 EVALUATION

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

2.1 Fuel Rod Analysis Methodology

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

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

2.2 Reload Core Design Methodology

For the RFA design, the core model, core operational imbalance limits, and key core physics parameters used to confirm the acceptability of Updated Final Safety Analysis Report (UFSAR) Chapter 15 safety analyses of transients and accidents will be developed with the methodologies described in DPC-NE-1004-A (Ref. 10), DPC-NE-2011P-A (Ref. 11), DPC-NF-2010A (Ref. 12), and DPC-NE-3001-PA (Ref 13). DPC-NE-2011P-A describes the nuclear design methodology for core operating limits of McGuire and Catawba plants. DPC-NF-2010A describes McGuire and Catawba nuclear physics methodology using two-dimensional PDQ07 and 3-D EPRI-NODE-P models as reactor simulators DPC-NE-1004A describes an alternative methodology for calculating nuclear physics data using the CASMO-3 fuel assembly depletion code and the SIMULATE-3P 3-D core simulator code for steady-state core physics calculations, substituting for CASMO-2, PDQ07 and EPRI-NODE-P used in DPC-NE-2010A. DPC-NE-3001-PA describes the methodologies, which expand on the reload design methods of DPC-NF-2010A, for systematically verifying that key physics parameters calculated for a reload core, such as control rod worth, reactivity coefficients, and kinetics parameters, are bounded by values assumed in the Chapter 15 licensing analyses. These topical reports have been approved for performing reload analyses for the B&W 177-assembly and/or Westinghouse 193-assembly cores, subject to the conditions specified in the staff's safety evaluations. Because of the similarity between the RFA design and the Mark-BW fuel design with respect to the dimensional characteristics of the fuel pellet, fuel rod and cladding, as well as nuclear characteristics, as shown in Table 2-1 of DPC-NE-2009, the staff concludes that these approved methodologies and core models currently employed in reload design analyses for McGuire and Catawba can be used to perform transition and full-core analyses of the RFA design.

Section 3 2 of DPC-NE-2009 states that conceptual transition core designs using the RFA design have been evaluated and results show that current reload limits remain bounding with respect to key physics parameters As described in DPC's response to a staff question (Question 1, Ref. 14, January 28, 1999), the conceptual RFA transition core designs were evaluated for the effects of partial and full cores using NRC-approved codes and methods to determine the acceptability of the current licensing bases transient analyses. Key safety parameters, such as Doppler temperature coefficients, moderator temperature coefficients, control bank worth, individual rod worths, boron concentrations, differential boron worths and kinetics data, were calculated for the conceptual core designs and compared against reference values assumed in the UFSAR Chapter 15 accident analyses. The evaluation demonstrated the expected neutronic similarities between reactor cores loaded with RFA fuel and with Mark-BW fuel and the acceptability of key safety parameters assumed in the Chapter 15 accident analyses. Key physics parameters are calculated for each reload core and each new core design If a key physics parameter is not bounded by the reference value in the UFSAR accident analyses, the affected accidents will be re-analyzed using the new key physics parameter, or the core will be re-designed to produce an acceptable result. The staff agrees that this is an acceptable approach.

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

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

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

2.3 <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 REVIEW SUMMARY OF TOPICAL REPORT

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

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

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

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

6.0 CONCLUSION

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

Attachment: Technical Evaluation Report

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

Date: September 22, 1999

7.0 REFERENCES

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

35. Letter, M. S. Tuckman (Duke Energy Corporation) to NRC, "Duke Energy Corporation, McGuire Nuclear Station Units 1 and 2, Docket Numbers 50-369 and 50-370, Catawba Nuclear Station Units 1 and 2, Docket Numbers 50-413 and 50-414, Response to NRC Requests for Additional Information on License Amendment Requests for McGuire and Catawba Nuclear Stations," April 7, 1999.

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

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

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

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

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

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

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

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

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

ATTACHMENT 1

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

PREPARED BY
PACIFIC NORTHWEST NATIONAL LABORATORY

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

"Duke Power Company Westinghouse Fuel Transition Report"

1.0 INTRODUCTION

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

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

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

Another W analysis method used is:

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

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

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

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

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

2.1 Fuel Rod Cladding Stress

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

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

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

2.2 Fuel Rod Cladding Strain

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

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

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

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

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

2.3 Fuel Rod Cladding Strain Fatigue

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

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

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

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

2.4 Fuel Rod Internal Pressure

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

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

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

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

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

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

2.5 Fuel Temperature

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

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

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

2.6 Fuel Clad Oxidation and Hydriding

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

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

2.7 Fuel Rod Axial Growth

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

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

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

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

3.0 CONCLUSIONS

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

4.0 REFERENCES

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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.

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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.

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

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

Section D

Duke Power Company Clarifications, Original Report

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

Duke Energy.

M. S. Tuckman
Executive Vice President
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August 17, 1999

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

ATTENTION: Document Control Desk

SUBJECT: Duke Energy Corporation

McGuire Nuclear Station - Units 1 & 2
Docket Nos. 50-369 and 50-370

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

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

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

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

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

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

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

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

Very truly yours,

M. S. Tuckman

M. S. Tuckman

MST/JSW

Attachment

xc w/Proprietary Attachment:

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

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

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

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

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

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

M. S. Tuck

M. S. Tuckman

(Continued)

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

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

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

(v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the Duke Topical Report designated DPC-NE-2009P, *Duke Power Company Westinghouse Fuel Transition Report*, and omitted from the non-proprietary version. This topical report was submitted to the NRC by Duke letter dated July 22, 1998 and revised by Duke letter dated August 17, 1999. This information enables Duke to:

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

M. S. Tuckman

M. S. Tuckman

(Continued)

(b) Perform core design, fuel rod design, and thermal-hydraulic analyses for the Westinghouse Robust Fuel Assembly design.

(c) Simulate UFSAR Chapter 15 transients and accidents for McGuire and Catawba Nuclear Stations.

(d) Perform safety evaluations per 10CFR50.59.

(e) Support Facility Operating Licenses/Technical Specifications amendments for McGuire and Catawba Nuclear Stations.

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

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

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

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

M. S. Tuck

M. S. Tuckman

(Continued)

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

M. S. Tuck

M. S. Tuckman

(Continued)

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuck

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 23RD day of August, 1998

Mary P. Nelms
Notary Public

My Commission Expires:
JAN 22, 2001

SEAL

bxc w/att:

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

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

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

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

- fuel rods positioned on the bottom nozzle

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

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

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

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

Table 2-1

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

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

Table 5-1

RFA Design Data

(TYPICAL)

GENERAL FUEL SPECIFICATIONS

Fuel rod diameter, inches (Nominal)	0.374
Guide tube diameter, inches (Nominal)	0.482
Fuel rod pitch, inches (Nominal)	0.496
Fuel Assembly pitch, inches (Nominal)	8.466
Fuel Assembly length, inches (Nominal)	160.0

GENERAL FUEL CHARACTERISTICS

Component	Material	Number	Location/Type
Grids	Inconel	1	Lower Protective
	Inconel	2	Upper and Lower Non-Mixing Vane
	ZIRLO™	6	Intermediate Mixing Vane
	ZIRLO™	3	Intermediate Flow Mixing (Non-structural)
Nozzles	304SS	1	Debris Filtering Bottom
	304SS	1	Removable Top

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

ATTENTION: Document Control Desk

SUBJECT: Duke Energy Corporation

McGuire Nuclear Station - Units 1 & 2
Docket Nos. 50-369 and 50-370

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

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

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

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

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

Item #2: Section 6.5.2, Large Break LOCA

The words " . . . typically with Moody break discharge coefficients, CD, of 0.4, 0.6 , and 0.8." have been deleted since this information is not required in the context of the paragraph.

Item #3: Section 6.5.2, Large Break LOCA

The words "Explicit analyses will be performed simulating . . ." have been replaced with "An evaluation will be performed to address . . ." since it has been determined that an evaluation rather than an explicit analysis is sufficient.

Future Transition to Westinghouse Best-Estimate LOCA Methodology

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

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

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

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

Very truly yours,

M. S. Tuckman

M. S. Tuckman

MST/JSW

Attachment

xc w/Proprietary and Non-proprietary Attachments:

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

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

xc w/Non-proprietary Attachment:

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

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

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

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

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

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

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

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

M. S. Tuckman

M. S. Tuckman

(Continued)

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

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

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

(v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the Duke Topical Report designated DPC-NE-2009P, *Duke Power Company Westinghouse Fuel Transition Report*, and omitted from the non-proprietary version. This topical report was originally submitted to the NRC by Duke letter dated July 22, 1998 and revised by Duke letters dated August 17, 1999 and December 13, 1999. The NRC SER for this topical report was issued September 22, 1999. This information enables Duke to:

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

M. S. Tucker

M. S. Tuckman.

(Continued)

(b) Perform core design, fuel rod design, and thermal-hydraulic analyses for the Westinghouse Robust Fuel Assembly design.

(c) Simulate UFSAR Chapter 15 transients and accidents for McGuire and Catawba Nuclear Stations.

(d) Perform safety evaluations per 10CFR50.59.

(e) Support Facility Operating Licenses/Technical Specifications amendments for McGuire and Catawba Nuclear Stations.

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

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

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

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

M S Tuckman

M. S. Tuckman

(Continued)

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

M. S. Tuck

M. S. Tuckman

(Continued)

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuck

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 14TH day of

December, 1998

Mary P. Nelms

Notary Public

My Commission Expires:

JAN 22, 2001

SEAL

bxc w/att:

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

Attachment B

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

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

6.2.3 Dropped Rod

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

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

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

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

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

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

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

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

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

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