December 18, 2002

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

/RA by GEdison 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

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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

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