September 16, 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-2005P (TAC NOS. MB3105, MB3106, MB3173, AND MB3175)

Dear Mr. Tuckman:

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

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

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

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

Sincerely,
/**RA by Lenny Olshan**/

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

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

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

DUKE POWER THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 and 2

DUKE ENERGY CORPORATION

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

1.0 INTRODUCTION

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

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

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

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

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

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

2.0 EVALUATION

The submittals contain the plant-specific data and statistical departure from nucleate boiling ratio (DNBR) limits for the McGuire and Catawba Nuclear Stations with the Advanced Mark-BW fuel design using the BWU-Z CHF correlation and provide details of the fuel assembly structural and thermal-hydraulic features unique to the Advanced Mark-BW fuel design.

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

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

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

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

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

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

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

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

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

3.0 CONCLUSION

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

4.0 REFERENCES

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

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

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

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

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

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

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

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

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Date: September 16, 2002

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