June 23, 2000

Mr. M.S. Tuckman
Executive Vice President
Nuclear Generation
Duke Energy Corporation
526 South Church Street
Charlotte, NC 28201-1006

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 RE: TOPICAL REPORT
             DPC-NE-2003, REVISION 1 (TAC NOS. MA8234, MA8235, AND MA8236)

Dear Mr. Tuckman:

By letter dated February 10, 2000, Duke Energy Corporation submitted Topical Report

DPC-NE-2003, Revision 1, "Core Thermal-Hydraulic Methodology Using VIPRE-01." This

information was amended by letter dated June 7, 2000. The report describes the methodology

for using the VIPRE-01 computer code to perform steady-state thermal-hydraulic analyses of

the reload cores for Oconee Nuclear Station, Units 1, 2, and 3. The original report,

DPC-NE-2003, was approved by the NRC in 1989. The current submittal, Revision 1, updates

this report to reflect several methodologies that have been documented in other approved

topical reports. Since no unreviewed or unapproved technical changes are involved, the staff

concludes Revision 1 is acceptable as described in the attached safety evaluation.

Sincerely,

/RA/

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

Docket Nos. 50-269, 50-270, and 50-287

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT DPC-NE-2003, REVISION 1,

CORE THERMAL-HYDRAULIC METHODOLOGY USING VIPRE-01

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

DUKE ENERGY CORPORATION[1]

1.0 INTRODUCTION

By letter dated February 10, 2000 (Reference 1), as amended by letter dated June 7, 2000 (Reference 2), Duke Energy Corporation, licensee for the Oconee Nuclear Station, Units 1, 2, and 3 (Oconee), submitted Topical Report, DPC-NE-2003, Revision 1, "Core Thermal-Hydraulic Methodology using VIPRE-01," for Nuclear Regulatory Commission (NRC) review and approval.

The VIPRE-01 computer code (Reference 3), developed by Electric Power Research Institute, was approved by the NRC for steady-state and transient reactor core thermal-hydraulic analyses. The NRC acceptance of VIPRE-01 included conditions that each VIPRE-01 user documents and submits for NRC approval its procedure for using VIPRE-01, by providing justifications for its specific modeling assumptions, choices of particular two-phase flow models and correlations, heat transfer correlations, critical heat flux (CHF) correlation and its departure from nucleate boiling ratio (DNBR) limit, and input values of plant specific data such as turbulent mixing coefficient and grid loss coefficients. To address these conditions, Topical Report DPC-NE-2003 (Reference 4) described the licensee's methodology for using the VIPRE-01 code to perform steady-state thermal-hydraulic analyses of the reload cores for Oconee. The original version of DPC-NE-2003, Revision 0, was approved by the NRC in 1989.

Revision 1 of DPC-NE-2003 provides an update to reflect several methodologies documented in other topical reports that have been subsequently approved by the NRC. The licensee stated that no unreviewed technical changes have been included in Revision 1.

_____

[1]On September 16, 1997, the NRC approved the licensee's name change from Duke Power Company to Duke Energy Corporation.

2.0 EVALUATION

DPC-NE-2003 describes the methodology for using the VIPRE-01 code to perform steady-state core thermal-hydraulic analyses for Oconee. Revision 1 incorporates several core thermal hydraulic methodologies documented in other topical reports that have subsequently been approved by the NRC; namely, statistical core design (SCD) methodology described in DPC-NE-2005, Revision 2 (Reference 5), the analyses of the Updated Final Safety Analysis Report (UFSAR) Chapter 15 non-loss-of-coolant accident transients and accidents described in DPC-NE-3005, Revision 1 (Reference 6), and Cogema's Mark-B11 fuel assembly design described in BAW-10229P-A (Reference 7). Revision 1 does not make any change with respect to the core model and correlations, but merely reflects the additional approved methodologies. The staff review of Revision 1 is described below.

2.1 SCD Methodology

DPC-NE-2005-A describes the licensee's SCD methodology for performing statistical core thermal hydraulic analyses. The SCD methodology statistically accounts for the uncertainties of key thermal hydraulic parameters such as reactor power, core power distributions, reactor coolant system temperature and flow that affect departure from nucleate boiling (DNB). This differs from the deterministic method where the uncertainties of various plant and operating parameters are assumed simultaneously at their worst uncertainty limits in the safety analyses. The SCD methodology establishes an SCD DNBR limit that statistically accounts for the effects on DNB of the key parameters and, therefore, allows for the use of the nominal values of these parameters in the safety analyses.

The original version of DPC-NE-2003 only addressed the deterministic analysis of VIPRE-01 application. In Revision 1, Section 5.11, "Hot Channel Factor," has been updated to include the use of the SCD methodology, and clarify the distinction between the values of the local hot channel heat flux factor, $F_{q"}$, used in the deterministic (non-SCD) method and the SCD method. For non-SCD analysis, $F_{q"}$ for Mark-BZ fuel is obtained from DPC-NE-1004P-A (Reference 8). For SCD analysis, $F_{q"}$ is used to account for axial nuclear uncertainty only. It references other fuel vendors' topical reports, which show that local heat flux spikes have no effect on CHF results and, therefore, the effect of (1) variations in the pellet enrichment and weight on local power, and (2) power spikes occurring as a result of flux depressions at spacer grids are not accounted for. This position was presented in response to a staff request for additional information during its review of DPC-NE-2005P-A, Revision 2, and accepted by the staff.

Section 6.4 of DPC-NE-2003 describes the calculation of the pressure-temperature envelope, which defines a region of allowable operation in terms of reactor coolant system pressure and outlet temperature. In Revision 1, a change is made in the reactor coolant system flow rate used in the generic Oconee thermal-hydraulic analyses to determine the pressure-temperature envelope from the current 366,080 gallons per minute to the value consistent with the number specified in the plant cycle specific core operating limits reports.   For example, if the SCD is used, the reactor coolant system flow value is consistent with the value used in DPC-NE-2005, Revision 2.

Steady state core thermal-hydraulic analyses are still performed the same way as discussed in DPC-NE-2003, except for the use of nominal values for those parameters whose uncertainties are statistically treated in the SCD. Therefore, the use of SCD methodology does not invalidate the NRC approval of the DPC-NE-2003, and is acceptable.

2.2 MARK-B11 FUEL DESIGN

In the original DPC-NE-2003 the typical fuel design specification data and various sensitivity studies were based on the Mark-BZ fuel design. But the report also states that the VIPRE-01 models will be used to predict and evaluate the thermal-hydraulic effects of other fuel assembly design. Revision 1 of DPC-NE-2003 extends the application of VIPRE-01 to a new fuel design, Cogema's Mark-B11 fuel design, which is described in topical report BAW-10229P-A (Reference 7), "Mark-B11 Fuel Assembly Design Topical Report." Appendix D, "Oconee Plant Specific Data, Mark-B11 Fuel, Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design," to DPC-NE-2005P-A, Revision 2, provides the plant specific data and specific limits for Oconee with Mark-B11 fuel design using the BWU-Z CHF correlation and the VIPRE-01 thermal-hydraulic code. The NRC staff has previously reviewed and approved both BAW-10229P-A and DPC-NE-2005P-A, Revision 2, including the Mark-B11 fuel design and VIPRE-01 inputs required to model the Mark-B11 design. Therefore, VIPRE-01 is acceptable for analysis of cores with Mark-B11 fuel design.

2.3 Operational MAP Limits Calculation

In the original version of DPC-NE-2003 the two-reactor coolant pump coastdown transient was used for the determination of the operational maximum allowable peaking (MAP) limits. Section 6.5, "Generic Maximum Allowable Peaking Limit Curves," describes the methodology for generation of operational MAP limits, which are used to determine the DNB operational offset limits, based on the two-reactor coolant pump coastdown transient. The MAP limits are presented in the form of lines of constant Minimum DNBR for a range of axial peaks with the location of the peak varied from the bottom to the top of the core. The two-pump coastdown transient was analyzed to assure that the design DNBR limit is not violated after the loss of one or more pumps, and to determine the operational MAP limits. Section 6.6, "Two-Pump Coastdown Transient Analyses," describes the steady-state analyses method using VIPRE-01, based on the two-pump coastdown statepoints, that determine the Operational MAP limits, during the limiting DNBR transient. The method of analysis of two-pump coastdown transient includes the use of the heat conduction model in VIPRE-01 for the heat flux calculation. Section 6.6 also describes the conduction model inputs used to calculate the two-pump coastdown transient.

In Revision 1 of DPC-NE-2003, the determination of the operational MAP limits is not limited to the two-pump coastdown transient, but is based on the limiting DNB transient. Therefore, the term "two-pump coastdown transient" in Section 5.9, which provides a description of the reference power distribution used in the determination of the MAP limits, and Section 6.5, has been replaced with "Operational DNB transient." Section 6.6 has been revised and renamed "Operational MAP Limit Generation" from "Two-Pump Coastdown Transient Analyses" in the original report. This section discusses the limiting DNB transient that is the basis for the operational MAP limits. The limiting DNB transient is analyzed with the non-loss-of-coolant accident transient analysis methodology described in Topical Report DPC-NE-3005P-A,

Revision 1. The DNB transient statepoints are determined using the clad surface heat flux versus time calculated by RETRAN-02 (Reference 9). The VIPRE-01 conduction model is no longer used when calculating the Operational MAP limits. Therefore, the conduction model input used to calculate the Operational MAP limits described in the original report has been deleted.

In Revision 1, the reference axial peaking factor in Section 5.10, "Axial Power Distribution," has been revised to a lower value than the original version. This lower value is consistent with the value used in the FSAR Chapter 15 transients to verify that the result are acceptable. As a part of the review of DPC-NE-2003, Revision 0, and DPC-NE-3000, the licensee in its letter of June 19, 1989 (Reference 10), stated that the higher reference value given in DPC-NE-2003, Revision 0 indicated the objective of using a higher value that would result in less limiting operational MAP limits, and that the reference axial peaking factor value used in the MAP methodology is the same value used in the UFSAR Chapter 15 transient analysis. Therefore, the lower axial peaking factor value specified in Revision 1 is consistent with UFSAR Chapter 15 analysis as well as the current value used in the MAP methodology.

Based on the above discussions, the staff concludes that the changes related to operation MAP calculation are based on approved methodologies and are acceptable.

2.4 Other Revisions

In DPC-NE-2003, Revision 1, the values of several trip setpoints in Table 6.1, "RPS Trip Functions," were revised. However, as stated in the footnote of the table, the RPS trip functions listed in the table are for information only, and the actual Reactor Protection System trip functions are specified in the Oconee Technical Specifications or the Core Operating Limits Report. Therefore, these changes do not invalidate the acceptance of DPC-NE-2003.

3.0 CONCLUSION

The objectives of DPC-NE-2003 is to document the licensee's procedure for using the VIPRE-01 code for core thermal hydraulic analyses, and provide justifications for its specific modeling assumptions, choice of particular two-phase flow models and correlations, heat transfer correlations, CHF correlations and its DNBR limit, and input values of plant-specific data. Revision 1 of DPC-NE-2003 merely expand the scope by incorporating other methodologies documented in other topical reports that have been approved by NRC. Since there is no unreviewed or unapproved technical changes involved, the staff concludes Revision 1 to be acceptable.

4.0 REFERENCES

1.   Letter from M. S. Tuckman (Duke Energy Corporation) to US Nuclear Regulatory Commission, "Oconee Nuclear Station, Docket Numbers 50-269, 50-270, and 50-287, Topical Report DPC-NE-2003, Revision 1, Core Thermal-Hydraulic Methodology Using VIPRE-01," February 10, 2000.

2.   Letter from M. S. Tuckman to US Nuclear Regulatory Commission, "Oconee Nuclear Station, Docket Numbers 50-269, 50-270, and 50-287, Topical Report DPC-NE-2003, Revision 1, Core Thermal-Hydraulic Methodology Using VIPRE-01," June 7, 2000.

3.   EPRI-NP-2511-CCM, Revision 1, "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Battelle Pacific Northwest Laboratories, September 1989.

4.   DPC-NE-2003-A, "Duke Power Company, Oconee Nuclear Station, Core Thermal-Hydraulic Methodology Using VIPRE-01," July 1989.

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

6.   DPC-NE-3005-A, Revision 1, "Duke Power Company, Oconee Nuclear Station, UFSAR Chapter 15 Transient Analysis Methodology," August 1999.

7.   BAW-10229P-A, "Mark-B11 Fuel assembly Design Topical Report," Framatome Cogema Fuels, October 1999.

8.   DPC-NE-1004P-A, "Duke Power Company Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November 1992.

9.   EPRI NP-1850-CCM, Revision 2, "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," November 1984.

10.   Letter from H. B. Tucker to US Nuclear Regulatory Commission, "Oconee Nuclear Station, Docket Numbers 50-269, 50-270, and 50-287, Response to Questions Regarding Differences Between Duke Topical Reports DPC-NE-2003 and DPC-NE-3000," June 19, 1989.

Principle Contributor: Yi-Hsiung Hsii

Date: June 23, 2000

Oconee Nuclear Station

cc:

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

Anne W. Cottingham, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

Mr. Rick N. Edwards
Framatome Technologies
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852-1631

Manager, LIS
NUS Corporation
2650 McCormick Drive, 3rd Floor
Clearwater, Florida 34619-1035

Senior Resident Inspector
U. S. Nuclear Regulatory
 Commission
7812B Rochester Highway
Seneca, South Carolina 29672

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

Mr. L. E. Nicholson
Compliance Manager
Duke Energy Corporation
Oconee Nuclear Site
7800 Rochester Highway
Seneca, South Carolina 29672

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

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory
 Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

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

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