December 8, 2000

Mr. M. S. Tuckman
Executive Vice President
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SUBJECT: DUKE ENERGY CORPORATION RE: RELOAD DESIGN METHODOLOGY TECHNICAL REPORT (NFS-1001, REVISION 5), OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. MA7752, MA7753 AND MA7754)

Dear Mr. Tuckman:

By letter dated December 22, 1999, and amended by a letter dated August 23, 2000, Duke Energy Corporation submitted Technical Report NFS-1001, Revision 5. The report describes the reactor core reload design methodology for the Oconee Nuclear Station, Units 1, 2, and 3. This submittal is an update of NFS-1001, Revision 4 that was approved by the staff in 1981, to reflect several methodologies documented in other approved topical reports. As explained in the enclosed safety evaluation, no unreviewed or unapproved technical changes are involved. Therefore, the staff concludes that Revision 5 is acceptable. Our safety evaluation is enclosed.

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 NFS-1001, REVISION 5

RELOAD DESIGN METHODOLOGY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

DUKE ENERGY CORPORATION

1.0 INTRODUCTION

By letter dated December 22, 1999 (Reference 1), as amended by letter dated August 23, 2000 (Reference 2), Duke Energy Corporation, the licensee for the Oconee Nuclear Station, Units 1, 2, and 3, submitted Topical Report, NFS-1001, Revision 5, “Reload Design Methodology,” for NRC review and approval. The topical report describes the licensee’s reload methodology for the Oconee Nuclear Station. It contains information related to fuel mechanical and core physics designs, thermal-hydraulic designs, technical specifications, and accident analyses. The reload methodology is used by the licensee to ensure that the Oconee reactors with reload cores can be operated to a specific power level for a specific number of days within the acceptable safety criteria. The original NFS-1001 report and its four revisions were reviewed and approved by the NRC and used as references in licensing applications in 1981. Revision 5 of NFS-1001 provides an update to reflect several methodologies documented in other NRC-approved topical reports. The licensee stated that there were no unreviewed changes in analytical methods included in Revision 5.

2.0 EVALUATION

NFS-1001 describes the methodology used by the licensee for Oconee reload analyses. Revision 5 incorporates several reload methodologies documented in other NRC-approved topical reports. The methodologies newly referenced in Revision 5 are described in the following reports:

> The nuclear design methodology using CASMO-3/SIMULATE-3P is described in DPC-NE-1004-A (Reference 3).
>
> The fuel mechanical analysis methodology using TACO3 is described in DPC-NE-2008P-A (Reference 4).
>
> The extended fuel burnup calculations are described in BAW-10186P-A, Rev. 1 (Reference 5).

The core thermal-hydraulic methodology using VIPRE-01 is described in DPC-NE-2003, Rev. 1 (References 6 and 7).

The statistical core design (SCD) methodology is described in DPC-NE-2005-A, Rev. 2 (Reference 8).

The analyses of the Updated Final; Safety Analysis Report (UFSAR) Chapter 15 non-loss-of-coolant accident (LOCA) transients and accidents are described in DPC-NE-3005-PA, Rev. 1 (Reference 9).

The licensee stated that Revision 5 did not make any changes with respect to the analytical models and empirical correlations, but merely consolidated methodologies that have already been approved by the NRC. The staff review of Revision 5 is described below.

2.1 Fuel Design Features and Mechanical Analyses

Descriptions of the reload aspects of the fuel design are included in two sections of NFS-1001; namely, Section 2 describes fuel design features and Section 4 describes fuel mechanical analyses.

Fuel Design Features

Section 2 of Revision 4 to NFS-1001 provides brief descriptions of the fuel pellets, fuel rods, and fuel assembly designs including information related to material selection, rod dimensions, cladding type and dimensions, and design criteria. In Revision 5, the original descriptions of fuel designs are removed and replaced with references to the descriptions of fuel and core component designs provided in Chapter 4 of the Oconee UFSAR and the design requirements of fuel assembly and control assembly specified in Technical Specifications (TSs) 4.2.1 and 4.2.2. The staff finds that (1) the changes in Section 2 do not involve changes in methods used for the fuel design, and (2) the information related to mechanical characteristics of the fuel assembly, fuel rod, and associated structures included in the original Section 2 of NFS-1001 is adequately provided in UFSAR Chapter 4 and TSs 4.2.1 and 4.2.2. Therefore, the staff has determined that the licensee's approach referencing the fuel design related sections of the UFSAR and TSs for descriptions of fuel design features does not invalidate the acceptance of NFS-1001 and is acceptable.

Fuel Rod Mechanical Analyses

The fuel mechanical analyses are performed to ensure that the centerline temperature is maintained below fuel melt limits and the end of life pin pressure is maintained below the value that would cause clad lift-off. Section 4 of Revision 4 to NFS-1001 provides descriptions of fuel mechanical analyses addressing cladding collapse, cladding strain and stress analysis, and fuel pin pressure calculations. Revision 5 removes the original descriptions of fuel mechanical analyses and replaces them with references to the NRC-approved methodologies described in DPC-NE-2008 (Reference 4) for analyzing fuel rod internal pressure, centerline fuel melt and cladding strain, and BAW-10186, Revision 1 (Reference 5) for analyzing cladding stress and fatigue and cladding corrosion.

Topical Report DPC-NE-2008 describes the methodologies used by the licensee to perform fuel mechanical analyses with the TACO3 computer code. The TACO3 code was originally developed by Framatome Cogema Fuels Company (FCFC) based on an earlier NRC-approved TACO2 code. The licensee adopted TACO3 for Oconee reload analyses that included analyses for fuel rod pressure, linear heat rate to melt, cladding strain, and verification of FCFC loss-of-coolant accident (LOCA) initialization. The staff has determined that DPC-NE-2008 was previously approved by the NRC for use for Oconee reload licensing applications (Reference 4.).

BAW-10186 is a topical report that describes a fuel design methodology for FCFC fuels at high burnups. The licensee adopted the methods described in BAW-10186 for reload analyses that included cladding stress and fatigue analysis and cladding corrosion analysis. The staff has determined that Revision 1 of BAW-10186 was reviewed and approved by NRC for use in the Oconee reload analysis in March 1999 (Reference 10).

2.2 <u>Core Physics Analyses</u>

The core physics analyses are performed to ensure that the core has sufficient reactivity to produce the design power level and lifetime without exceeding the control capability or shutdown margin. During the process, the fuel cycle specific physics data including reactivity feedback coefficients, control rod worths, shutdown margin, and differential boron worths are calculated as input for safety analyses. The descriptions of the core physics analyses are provided in five sections of NFS-1001: (1) Section 3 describes the preliminary and final fuel cycle designs; (2) Section 5 documents the procedures used for maneuvering analyses; (3) Section 7 provides information related to the core safety limits and limiting safety system settings specified in the Oconee TS and Core Operating Limits Reports (COLR); (4) Section 8 characterizes core physics parameters used as input for safety analyses; and (5) Section 9 summarizes procedures used to determine the core physics parameters for startup, test predictions, core monitoring, and power controlling throughout the cycle.

The descriptions of the core physics analyses described in Revision 5 of NFS-1001 for the Oconee reload applications remain essentially the same as the original descriptions provided in Revision 4, except for changes resulting from referencing Topical Report DPC-NE-1004 (Reference 3) for nuclear design methodology using CASMO-3/SIMULATE-3P and DPC-NE-3005 for analysis of the control rod ejection accidents.

DPC-NE-1004 describes the licensee’s methodologies using CASMO-3 and SIMULATE-3P for steady-state physics calculations. This topical report describes the calculational methods and benchmark comparisons for critical boron concentrations, control rod worths, isothermal temperature coefficients, and core power distributions. DPC-NE-3005 describes the licensee’s methodologies used for non-LOCA analyses. It also describes, in part, methods for using the computer code ARROTTA or SIMULATE-3K to calculate the core power response during a control rod ejection event. The staff has determined that both DPC-NE-1004 and DPC-NE-3005 were previously reviewed and approved by NRC for referencing in the Oconee reload applications (References 3, 5, and 9.).

The licensee has also revised the descriptions to include previous relocation of some TS parameters representative to specific fuel cycles to the COLR. The staff finds that the approach to include the fuel cycle dependent parameters in the COLR is consistent with the licensee's current licensing practice. Therefore, the staff concludes that the approach is acceptable.

2.3 Thermal-hydraulic Analyses

The thermal-hydraulic analyses are performed by the licensee to establish the maximum allowable power distribution limits to maintain the required margin to departure from nucleate boiling (DNB) at various coolant flows, temperatures, and pressures. In Revision 4 of NFS-1001, Section 6 contains information pertaining to the thermal-hydraulic analysis for reload applications. Revision 5 removes the original descriptions of thermal-hydraulic analysis and replaces it with a reference to the NRC-approved methodologies described in DPC-NE-2003 for performing thermal-hydraulic analyses using the VIPRE-01 code, DPC-NE-2005 for performing the statistical core design analysis, and BAW-10186 for analyzing the effects of rod bow on DNB ratio (DNBR) calculations.

DPC-NE-2003 - Thermal-hydraulic Analyses Using the VIPRE-01 Code

Topical Report DPC-NE-2003 describes the licensee's methodology for using the VIPRE-01 code to perform steady-state thermal-hydraulic analysis of the reload cores for Oconee. The original version of DPC-NE-2003 (Reference 6) was approved by NRC in 1989. The latest version (Revision 1) incorporates several core thermal-hydraulic methodologies documented in other topical reports that have subsequently been approved by NRC. The newly referenced reports are: (1) DPC-NE-2005, Revision 2 that describes the statistical core design (SCD) methodology; (2) DPC-NE-3005, Revision 1 that describes the analyses of the UFSAR Chapter 15 non-LOCA events; and (3) BAW-10229P describes FCFC's Mark-B11 fuel assembly design. The staff has determined that Revision 1 of DPC-NE-2003 was reviewed and approved by NRC in June 2000 (Reference 7).

DPC-NE-2005 - Statistical Core Design Methodology

Topical Report DPC-NE-2005 describes the licensee's SCD methodology for performing statistical core thermal hydraulic analyses. Unlike the deterministic method, in which the uncertainties of various plant and operating parameters are assumed simultaneously at their worst uncertainty limits in the safety analyses, the SCD methodology statistically accounts for the uncertainties of the key thermal hydraulic parameters such as reactor average power, core power distribution reactor coolant system temperature, and core flow that affect DNB. The SCD methodology establishes an SCD DNBR limit that statistically accounts for the effects on DNB of the key parameters. Therefore, when the SCD methodology is used to perform thermal hydraulic analyses, initial condition uncertainties are allowed to exclude the plant parameters that are sensitive to the DNBR calculations since they are already included in the SCD DNBR limit. The staff has determined that Revision 2 of DPC-NE-2005 was approved for Oconee reload analyses by the NRC in April 2000 (Reference 8).

BAW-10186 - Rod Bow Penalty Analysis

Topical Report BAW-10186 describes methodologies for extended burnup evaluation including the analysis of rod bow penalty effects on the minimum DNBR. Rod bowing data presented in BAW-10186 for assembly average burnups up to 58,300 megwatt days per metric ton uranium (MWd/mtU) show that rod bowing maximizes at 30,000 MWd/mtU and does not increase between 30,000 to 58,300 MWd/mtU. The fuel vendor, FCFC, used an “observed limit” on rod bowing that bounded all of their data for use in its DNBR analyses at assembly average burnups above 29,000 MWd/mtU and demonstrated that the rod bow penalty effects were insignificant and could be neglected for DNBR calculations. The staff has determined that the applications of topical report BAW-10186 were approved by NRC for use in Oconee reload analysis in March 1999 (References 5 and 10).

2.4 Accident Analyses

Section 8 of NFS-1001 provides a description of the methodologies used to perform accident analyses for Oconee reload applications. For each reload application, the licensee reviews the reference analyses of all LOCAs and non-LOCA transients. In the review, The licensee evaluates the effects of plant control parameters, fuel, neutronic and thermal-hydraulic parameters, and engineering safety features on plant transients and accidents. For the cases that are bounded by the corresponding cases in reference calculations, the licensee determines that a reanalysis of the transients is not needed. For cases that are more limiting than the corresponding reference cases, the licensee performs a reanalysis of the affected cases using the NRC-approved methods described in DPC-NE-3005 for the non-LOCA transient analysis and the methods described in the UFSAR Section 15.14 for the LOCA analysis.

Non-LOCA Transient Analysis Methodology

Topical Report DPC-NE-3005 (References 9 and 11) provides descriptions of methodologies used for the non-LOCA transient analysis. The transient analysis uses the following computer codes:

RETRAN-02: This code is used to simulate the system response and calculate system parameters such as core power, reactor coolant system flow, and primary and secondary temperatures and pressures during a transient. An application of RETRAN-3D limited to the “RETRAN-02 mode" is also used to simulate the system response for transient analyses. The licensee’s use of RETRAN-3D does not include any of the non-equilibrium or three-dimensional (3-D) core modeling techniques.

VIPRE-01/MOD2: This code provides a simulation of the hot channel thermal-hydraulic analysis and determines the minimum DNBR using the approved critical heat flux (CHF) correlations. With the use of the VIPRE-01/MOD2 code and the approved CHF correlations, the DNBR safety limits are established to provide 95 percent probability of precluding DNB, and thus avoiding fuel failures, at a 95 percent confident level.

ARROTTA and SIMULATE: ARROTTA and SIMULATE-3P are 3-D, 2-energy group diffusion theory core simulator programs. SIMULATE-3K is a 3-D transient neutronic version of the SIMULATE-3P code. ARROTTA or SIMULATE-3K is used to calculate the core power response during a control rod ejection event. SIMULATE-3P is used to calculate nuclear parameters and core power distributions for the transient analysis.

CASMO-3: This computer code is a multigroup, 2-D transport theory code that is used for burnup calculations. The code produces two energy group edits of cross sections, assembly discontinuity factors, fission product data, and pin power data as input to the ARROTTA and SIMULATE codes.

The set of licensing basis transients considered for the reload analysis includes: (1) startup accident, (2) control rod withdrawal at power, (3) moderator dilution event, (4) cold water event, (5) control rod misalignment, (6) control rod ejection accident, (7) loss of coolant flow, (8) locked rotor, (9) turbine trip, (10) steam generator tube rupture, and (11) large and small steamline breaks. The transient analysis is performed to ensure that the analytical results for each transient meet the applicable acceptance criteria specified in the UFSAR Chapter 15 with respect to the offsite dose, reactor coolant system pressure, and fuel performance.

LOCA Analysis Methodology

The NRC-approved methodologies used for LOCA analysis are described in UFSAR Section 15.14. For the large break LOCA analysis, the computer codes used are (1) RELAP/MOD2-B&W that is used for determining the system response during the blowdown phase, (2) REFLOOD3B that is used for calculating the flooding rate, (3) CONTEMPT that is used for calculating the reactor building pressure response, and (4) BEACH that is used for calculating the peak cladding temperature (PCT) during core reflood.

For the small break LOCA analysis, the computer codes used are (1) CRAFT2 that is used for determining the system response, (2) THETA1-B that is used for calculating the fuel thermal and mechanical response and the PCT, and (3) FOAM that is used for calculating the core mixing level. The LOCA events are analyzed to ensure that the analytical results meet the 10 CFR 50.46 requirements with respect to the PCT, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling.

Based on its review, the staff finds that the methodologies documented in DPC-NE-3005 for non-LOCA transient analysis were previously approved (References 6 through 9) by the NRC, and the methodologies for the LOCA analysis are the same as the approved methodologies documented in the UFSAR Section 15.14. Therefore, the staff determines that the methodologies for the non-LOCA transient and LOCA analyses continue to be acceptable for referencing in reload licensing applications.

3.0 CONCLUSION

The objective of NFS-1001 is to document the licensee’s reload methodology using approved computer codes for fuel and fuel cycle designs, as well as core thermal-hydraulic and accident analyses. The reload methodologies described in NFS-1001 are used by the licensee to ensure that the Oconee reactors with reload cores can be operated to a specific power level for a specific number of days within the acceptable safety criteria. Revision 4 of NFS-1001 was previously approved by NRC. Revision 5 of NFS-1001 merely updates its content by referencing methodologies documented in other topical reports that have been approved by NRC. Since the staff has verified that there is no unreviewed or unapproved technical changes involved, the staff concludes that Revision 5 of NFS-1001 is acceptable. The staff approval of Revision 5 does not remove or change the limitations stated in the staff’s safety evaluation reports for the topical reports referenced in Revision 5 of NFS-1001.

4.0 REFERENCES

1. Letter from M. S. Tuckman (Duke Energy Corporation) to U. S. Nuclear Regulatory Commission, “Oconee Nuclear Station, Docket Numbers 50-269, 50-270, and 50-287, Topical Report NFS-1001, Revision 5, Reload Design Methodology,” December 22, 1999.

2. Letter from M. S. Tuckman (Duke Energy Corporation) to U. S. Nuclear Regulatory Commission, “Oconee Nuclear Station, Docket Numbers 50-269, 50-270, and 50-287, Topical Report NFS- 1001, Revision 5, Reload Design Methodology,” August 23, 2000.

3. DPC-NE-1004-A, “Duke Power Company Nuclear Design Methodology Using CASMO-3/SIMULATE-3P,” February 24, 1993.

4. DPC-NE-2008P-A, Revision 0, “Duke Power Company’s Fuel Mechanical Reload Analysis Methodology Using TACO3,” November 13, 1997.

5. BAW-10186P-A, Revision 1, “Extended Burnup Evaluation,” April 7, 2000.

6. DPC-NE-2003P-A, “Duke Power Company, Oconee Nuclear Station, Core Thermal-Hydraulic Methodology Using VIPRE-01,” November 21, 1989.

7. Letter from D. E. LaBarge (NRC) to M. S. Tuckman (DEC), “ Oconee Nuclear Station, Units 1, 2, and 3, RE: Topical Report DPC-NE-2003, Rev. 1 (TAC Nos. MA8234, MA8235, MA8236),” June 23, 2000.

8. DPC-NE-2005-A, Revision 2, “Duke Power Company, Thermal-Hydraulic Statistical Core Design Methodology,” April 7, 2000.

9. DPC-NE-3005-PA, Rev. 1, “Duke Power Company, Oconee Nuclear Station, UFSAR Chapter 15 Transient Analysis Methodology,” September 1, 1999.

10. Letter form D. L. LaBarge (NRC) to W. R. McCollum, Jr. (ONS), “ Use of Framatome Cogema Fuels Topical Report on High Burnup - Oconee Nuclear Station, Units 1, 2, and 3 (TAC MA0405, MA0406, MA0407), Docket Nos. 50-269, 50-270, and 50-287," March 1 , 1999.

11. Letter from M. S. Tuckman (Duke Energy Corporation) to U. S. Nuclear Regulatory Commission, “Oconee Nuclear Station, Docket Numbers 50-269, 50-270, and 50-287, Topical Report NFS-1001, Reload Design Methodology, Revision 5,” October 30, 2000.

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