6.9 CORE OPERATING LIMITS REPORT

Specification

6.9.1 Core operating limits shall be established prior to each reload cycle or, prior to any remaining part of a reload cycle, for the following:

(1) Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits for Specification 2.1.

(2) Reactor Protective System Trip Setting limits for the Flux/Flow/Imbalance and Variable Low Reactor Coolant System Pressure trip functions in Specification 2.3.

(3) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.1.11, 3.5.2.1b, 3.5.2.2.d.2.c, 3.5.2.3, and 3.5.2.5.c.

(4) Concentrated Boric Acid Storage Tank volume and boron concentration for Specification 3.2.2.

(5) Core Flood Tank boron concentration for Specification 3.3.3.

(6) Borated Water Storage Tank boron concentration for Specification 3.3.4.

(7) Spent Fuel Pool boron concentration for Specification 3.8.15.

(8) Quadrant Power Tilt Limits for Specification 3.5.2.4.a, 3.5.2.4.b, 3.5.2.4.d, 3.5.2.4.e, and 3.5.2.4.f.

(9) Power Imbalance Limits for Specification 3.5.2.6.

and shall be documented in the CORE OPERATING LIMITS REPORTS.

6.9.2 The approved methods used to determine the core operating limits given in the Technical Specification 6.9.1 are specified in the CORE OPERATING LIMITS REPORT. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically:

(1) DPC-NE-1002A, Reload Design Methodology II, October, 1985.

(2) NFS-1001A, Reload design Methodology, April, 1984

(3) DPC-NE-2003A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, July 1989.

(4) DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992.

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6.9.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

Stuff:

(5) BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November, 1989

(6) BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, May, 1994

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, August 1994

(~~9~~ 8) DPC-NE-2005P-A, Statistical Core Design Methodology, February, 1995 THERMAL-HYDRAULIC

Attachment I.a2
New Original Pages
Oconee

6.9 CORE OPERATING LIMITS REPORT

Specification

6.9.1 Core operating limits shall be established prior to each reload cycle, or prior to any remaining part of a reload cycle, for the following:

(1) Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits for Specification 2.1.

(2) Reactor Protective System Trip Setting Limits for the Flux/Flow/Imbalance and Variable Low Reactor Coolant System Pressure trip functions in Specification 2.3.

(3) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.1.11, 3.5.2.1.b, 3.5.2.2.d.2.c, 3.5.2.3, and 3.5.2.5.c.

(4) Concentrated Boric Acid Storage Tank volume and boron concentration for Specification 3.2.2.

(5) Core Flood Tank boron concentration for Specification 3.3.3.

(6) Borated Water Storage Tank boron concentration for Specification 3.3.4.

(7) Spent Fuel Pool boron concentration for Specification 3.8.15.

(8) Quadrant Power Tilt Limits for Specification 3.5.2.4.a, 3.5.2.4.b, 3.5.2.4.d, 3.5.2.4.e, and 3.5.2.4.f.

(9) Power Imbalance Limits for Specification 3.5.2.6

and shall be documented in the CORE OPERATING LIMITS REPORTS.

6.9.2 The approved methods used to determine the core operating limits given in Technical Specification 6.9.1 are specified in the CORE OPERATING LIMITS REPORT. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically:

(1) DPC-NE-1002A, Reload Design Methodology II, October 1985.

(2) NFS-1001A, Reload Design Methodology, April 1984.

(3) DPC-NE-2003A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, July 1989.

(4) DPC-NE-1004A, Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992.

(5) BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November, 1989.

(6) BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, May, 1994.

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, August, 1994.

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, February, 1995.

6.9.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Attachment I.b1
Marked-up Technical Specifications
McGuire

CORE OPERATING LIMITS REPORT

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION," June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for FQ Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987 (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.) INS BLANK LINE

4. BAW-10168~~P~~PA, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," ~~September 1989~~ JANUARY, 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

5. DPC-NE-2011~~P~~PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March 1990 (DPC Proprietary).

   (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001~~P~~PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," ~~March 1991~~ NOVEMBER 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NE-2010~~P~~PA, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," ~~April 1984~~ JUNE, 1985 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3/4.9.12 - Spent Fuel Pool Boron Concentration.)

CORE OPERATING LIMITS REPORT

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," ~~August~~ NOVEMBER 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A, ~~Rev. 1~~ "Thermal-Hydraulic Transient Analysis Methodology," ~~May 1989.~~ AUGUST, 1994

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

INSERT ATT. B

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

Attachment B:

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor FΔH(X,Y).)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specification 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation setpoints).

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, May 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints).

Attachment I.b2
New Original Pages
McGuire

CORE OPERATING LIMITS REPORT

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in:

1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

   (Methodology for Specifications 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

2. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL FQ SURVEILLANCE TECHNICAL SPECIFICATION", June 1983 (W Proprietary).

   (Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (W(Z) surveillance requirements for F_Q Methodology.)

3. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

4. BAW-10168PA, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," January 1991 (B&W Proprietary).

   (Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor.)

5. DPC-NE-2011PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," March 1990 (DPC Proprietary).

   (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NE-2010PA, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3/4.9.12 - Spent Fuel Pool Boron Concentration.)

CORE OPERATING LIMITS REPORT

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A, Rev. 1, "Thermal-Hydraulic Transient Analysis Methodology," August 1994.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P," November 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F\Delta H(X,Y)$.)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specification 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation setpoints).

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, May 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10CFR50.4.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the NRC Regional Office within the time period specified for each report.

Attachment I.c1
Marked-up Technical Specifications
Catawba

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

   (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-~~NF~~ E-2010P-A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 4.7.13.3 - Standby Makeup Pump Water Supply Boron Concentration, and Specification 3.9.1 - RCS and Refueling Canal Boron Concentration.)

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A~~, Rev. 1~~, "Thermal-Hydraulic Transient Analysis Methodology," November 1991 August 1994

   (Modeling used in the system thermal-hydraulic analyses)

CORE OPERATING LIMITS REPORT (Continued)

10. DPC-NE-1004A, "Design Methodology Using CASMO-3/Simulate-3P," November 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F\Delta H$ (X,Y).)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10 CFR 50.4.

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specifications 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints)

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, May 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints)

Attachment I.c2
New Original Pages
Catawba

CORE OPERATING LIMITS REPORT (Continued)

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

   (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

6. DPC-NE-3001P-A, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," November 1991 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

7. DPC-NE-2010P-A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985 (DPC Proprietary).

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, Specification 4.7.13.3 - Standby Makeup Pump Water Supply Boron Concentration, and Specification 3.9.1 - RCS and Refueling Canal Boron Concentration.)

8. DPC-NE-3002A, "FSAR Chapter 15 System Transient Analysis Methodology," November 1991.

   (Methodology used in the system thermal-hydraulic analyses which determine the core operating limits)

9. DPC-NE-3000P-A, "Thermal-Hydraulic Transient Analysis Methodology," August 1994.

   (Modeling used in the system thermal-hydraulic analyses)

CORE OPERATING LIMITS REPORT (Continued)

10. DPC-NE-1004A, "Design Methodology Using CASMO-3/Simulate-3P," November 1992.

    (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient.)

11. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," December 1991 (DPC Proprietary).

    (Methodology for Specifications 2.2.1 - Reactor Trip System Instrumentation Setpoints, 3.2.1 - Axial Flux Difference (AFD), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$ (X,Y).)

12. DPC-NE-2001P-A, Rev. 1, "Fuel Mechanical Reload Analysis Methodology for Mark-BW Fuel," October 1990 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints.)

13. DPC-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," February 1995 (DPC Proprietary).

    (Methodology for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints, Specifications 3.2.1 - Axial Flux Difference, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)

14. BAW-10162P-A, TACO3 Fuel Pin Thermal Analysis Computer Code, B&W Fuel Company, November 1989.

    (Methodology used for Specification 2.2.1 - Reactor Trip System Instrumentation Setpoints)

15. BAW-10183P, Fuel Rod Gas Pressure Criterion, B&W Fuel Company, May 1994.

    (Used for Specification 2.2.1, Reactor Trip System Instrumentation Setpoints)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC in accordance with 10 CFR 50.4.

Attachment II
Justification and Statement of No Significant Hazards

Introduction

Generic Letter 88-16 provided guidance on removing cycle-specific parameters which are calculated using NRC-approved methodologies from Technical Specifications. The parameters are replaced in Tech Specs with a reference to a named report which contains the parameters, and a requirement that the parameters remain within the limits specified in the report. The report, unlike the Tech Specs, may be changed by the licensee.

Justification

The proposed changes incorporate NRC-approved methodologies, approved revisions to previously-approved methodologies, or republished versions of previously-approved methodologies. Newly-approved methodologies include BAW-10162P-A, BAW-10183P, and DPC-NE-2005. For McGuire and Catawba, the limits to which these methodologies are applied are explicitly listed in the Technical Specifications. For Oconee, these methodologies are used to calculate (numbers refer to items listed in Specification 6.9.1):

1) Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits for Specification 2.1;

2) Reactor Protective System Trip Setting Limits for the Flux/Flow/Imbalance and (DPC-NE-2005 only) Variable Low Reactor Coolant System Pressure trip functions in Specification 2.3; and

9) Power Imbalance Limits for Specification 3.5.2.6.

Since the proposed changes only incorporate NRC-approved methodologies into Technical Specifications, the changes are administrative in nature and can be assumed to have no impact, or potential impact, on the health and safety of the public or Duke employees.

No Significant Hazards Consideration

The proposed changes will not create a significant hazards consideration, as defined by 10 CFR 50.92, because:

1) The proposed changes will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative in nature, and do not affect any system, procedure, or manipulation of any equipment which could affect the probability or consequences of any accident.

2) **The proposed changes will not create the possibility of any new or different kind of accident from any accident previously evaluated.**

The proposed changes are administrative in nature, and cannot introduce any new failure mode or transient which could create any accident.

3) **The proposed changes will not involve a significant reduction in a margin of safety.**

The proposed changes are administrative in nature, and will not affect any operating parameters or limits which could result in a reduction in a margin of safety.

In addition, due to the administrative nature of the amendments, there will be no impact on the environment.