CORE OPERATING LIMITS REPORT (Continued)

11. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.

12. Standby Makeup Pump water supply boron concentration limits of Specification 4.7.13.3.

13. Spent Fuel Pool boron concentration limit of Specification 3/4.9.12.

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

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   (Methodology for Specification 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-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," SER dated January 1991, (B&W Proprietary).
   REV 2, SER DATED ______; REV 3, SER DATED JUNE 15, 1994

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

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

CORE OPERATING LIMITS REPORT (Continued)

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-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985

   (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, and Specification 3.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002A, THROUGH REV 2, "FSAR Chapter 15 System Transient Analysis Methodology," ~~November 1991.~~ SER DATED APRIL 26, 1996

   (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.~~ SER DATED DECEMBER 27, 1995

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, REV 1, "Design Methodology Using CASMO-3/Simulate-3P," ~~November 1992.~~ SER DATED APRIL 26, 1996

    (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-NE-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)

CORE OPERATING LIMITS REPORT (Continued)

11. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.

12. Standby Makeup Pump water supply boron concentration limit of Specification 4.7.13.3.

13. Spent Fuel Pool boron concentration limit of Specification 3/4.9.12.

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

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   (Methodology for Specification 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-10168P, Rev. 1, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," SER dated January 1991, (B&W Proprietary). (REV. 2, SER DATED ———— ; REV. 3, SER DATED JUNE 15, 1994)

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

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

CORE OPERATING LIMITS REPORT (Continued)

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-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985

   (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, and Specification 3.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002A, THROUGH REV 2, "FSAR Chapter 15 System Transient Analysis Methodology," ~~November 1991~~. SER DATED APRIL 26, 1996

   (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~~. SER DATED DECEMBER 27, 1995.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, REV. 1 "Design Methodology Using CASMO-3/Simulate-3P," ~~November 1992~~. SER DATED APRIL 26, 1996.

    (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-NE-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)

Attachment I
New Original Pages
Catawba

CORE OPERATING LIMITS REPORT (Continued)

11. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.

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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-10168P, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," Rev. 1, SER dated January 1991; Rev. 2, SER Dated ____________; Rev. 3, SER Dated June 15, 1994 (B&W Proprietary).

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

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).

CORE OPERATING LIMITS REPORT (Continued)

(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-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985

   (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, and Specification 3.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002A, Through Rev. 2, "FSAR Chapter 15 System Transient Analysis Methodology," SER Dated April 26, 1996.

   (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," SER Dated December 27, 1995.

   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, Rev. 1, "Design Methodology Using CASMO-3/Simulate-3P," SER Dated April 26, 1996.

    (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-NE-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)

CORE OPERATING LIMITS REPORT (Continued)

11. Reactor Coolant System and refueling canal boron concentration limits for Specification 3/4.9.1.

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   (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).

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

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).

CORE OPERATING LIMITS REPORT (Continued)

(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-2010A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design," June 1985

   (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, and Specification 3.9.12 - Spent Fuel Pool Boron Concentration.)

8. DPC-NE-3002A, Through Rev. 2, "FSAR Chapter 15 System Transient Analysis Methodology," SER Dated April 26, 1996.

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

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   (Modeling used in the system thermal-hydraulic analyses)

10. DPC-NE-1004A, Rev. 1, "Design Methodology Using CASMO-3/Simulate-3P," SER Dated April 26, 1996.

    (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-NE-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)

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 without prior Commission approval.

**Justification**

The proposed changes incorporate NRC-approved revisions to previously-approved methodologies.

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.

This Technical Specification change will not result in a change to the station as described in the UFSAR.

Note that in the case of BAW-10168, Revisions 2 and 3 (revisions to different portions of the Topical Report) were pursued simultaneously. This resulted in Revision 3 being approved by the NRC before Revision 2. For completeness, the SER date for each revision is listed.

**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.