

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_0 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).
(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 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-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-NF-2010A, "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.)

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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)
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, as approved by SER dated February, 1994.

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

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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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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-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 3.9.1 - RCS and Refueling Canal Boron Concentration, and Specification 3/4.9.12 - Spent Fuel Pool Boron Concentration.)

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(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, TAC03 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, as approved by SER dated February 1994.
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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.

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