Duke Energy
Carolinas

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September 29, 2010

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station Unit 1
Docket No. 50-413
Core Operating Limits Report (COLR)
Catawba Nuclear Station (CNS) Unit 2 Cycle 18, Revision 0

Attached, pursuant to Catawba Technical Specification 5.6.5, is an information copy and electronic copy of the Core Operating Limits Report for Catawba Unit 2 Cycle 18. This COLR is being submitted to update the limits of the Cycle 18 reload core.

The electronic copy of this COLR is included with the letter. The electronic copy includes the power distribution monitoring factors.

This letter, attached COLR, and computer disk do not contain any new commitments.

Please direct any questions or concerns to Toni K. Pasour at (803) 701-3566.

Sincerely,

James R. Morris

Attachments

A001
NRR

xc w/attachment only:

U.S. Nuclear Regulatory Commission

Mr. Luis A. Reyes, Regional Administrator
U.S. Nuclear Regulatory Commission
Region II
Marquis One Tower
245 Peachtree Center Ave., NE Suite 1200
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Mr. Jon H. Thompson, NRC Project Manager
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
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Washington, D.C. 20555

Mr. G.A. Hutto, NRC Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

Catawba Unit 2 Cycle 18

Core Operating Limits Report
Revision 0

August 2010

Calculation Number: CNC-1553.05-00-0535

Duke Energy

		Date
Prepared By:	Nicholas L Hagar	8/5/10
Checked By:	Sandra L Abbey	8/5/10
Checked By:	Stephen G. Siry (Sections 2.1 and 2.9 – 2.18)	8/11/10
Approved By:	RC Harvey	8/16/10

QA Condition 1

The information presented in this report has been prepared and issued in accordance with Catawba Technical Specification 5.6.5.

INSPECTION OF ENGINEERING INSTRUCTIONS

Inspection Waived By: RC Harvey (Sponsor) Date: 8/16/2010

CATAWBA

	Inspection Waived	
MCE (Mechanical & Civil)	☑	Inspected By/Date: ______
RES (Electrical Only)	☑	Inspected By/Date: ______
RES (Reactor)	☑	Inspected By/Date: ______
MOD	☑	Inspected By/Date: ______
Other (______)	☐	Inspected By/Date: ______

OCONEE

	Inspection Waived	
MCE (Mechanical & Civil)	☐	Inspected By/Date: ______
RES (Electrical Only)	☐	Inspected By/Date: ______
RES (Reactor)	☐	Inspected By/Date: ______
MOD	☐	Inspected By/Date: ______
Other (______)	☐	Inspected By/Date: ______

MCGUIRE

	Inspection Waived	
MCE (Mechanical & Civil)	☐	Inspected By/Date: ______
RES (Electrical Only)	☐	Inspected By/Date: ______
RES (Reactor)	☐	Inspected By/Date: ______
MOD	☐	Inspected By/Date: ______
Other (______)	☐	Inspected By/Date: ______

Catawba 2 Cycle 18 Core Operating Limits Report

Implementation Instructions for Revision 0

Revision Description and PIP Tracking

Revision 0 of the Catawba Unit 2 Cycle 18 COLR contains limits specific to the reload core. There is no PIP associated with this revision.

Implementation Schedule

Revision 0 may become effective any time during No MODE between Cycles 17 and 18 but must become effective prior to entering MODE 6 which starts Cycle 18. The Catawba Unit 2 Cycle 18 COLR will cease to be effective during No MODE between Cycle 18 and 19.

Data files to be Implemented

No data files are transmitted as part of this document.

Catawba 2 Cycle 18 Core Operating Limits Report

REVISION LOG

Revision	**Effective Date**	**Pages Affected**	**COLR**
0	August 2010	1-32, Appendix A*	C2C18 COLR, Rev. 0

* Appendix A contains power distribution monitoring factors used in Technical Specification Surveillance. Appendix A is included only in the electronic COLR copy sent to the NRC.

Catawba 2 Cycle 18 Core Operating Limits Report

1.0 Core Operating Limits Report

This Core Operating Limits Report (COLR) has been prepared in accordance with requirements of Technical Specification 5.6.5. Technical Specifications that reference this report are listed below:

TS Section	Technical Specifications	COLR Parameter	COLR Section	COLR Page
2.1.1	Reactor Core Safety Limits	RCS Temperature and Pressure Safety Limits	2.1	9
3.1.1	Shutdown Margin	Shutdown Margin	2.2	9
3.1.3	Moderator Temperature Coefficient	MTC	2.3	11
3.1.4	Rod Group Alignment Limits	Shutdown Margin	2.2	9
3.1.5	Shutdown Bank Insertion Limit	Shutdown Margin Rod Insertion Limits	2.2 2.4	9 11
3.1.6	Control Bank Insertion Limit	Shutdown Margin Rod Insertion Limits	2.2 2.5	9 15
3.1.8	Physics Tests Exceptions	Shutdown Margin	2.2	9
3.2.1	Heat Flux Hot Channel Factor	F_Q AFD OTΔT Penalty Factors	2.6 2.8 2.9 2.6	15 21 24 17
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor	FΔH Penalty Factors	2.7 2.7	20 21
3.2.3	Axial Flux Difference	AFD	2.8	21
3.3.1	Reactor Trip System Instrumentation	OTΔT OPΔT	2.9 2.9	24 25
3.3.9	Boron Dilution Mitigation System	Reactor Makeup Water Flow Rate	2.10	26
3.4.1	RCS Pressure, Temperature and Flow limits for DNB	RCS Pressure, Temperature and Flow	2.11	26
3.5.1	Accumulators	Max and Min Boron Conc.	2.12	26
3.5.4	Refueling Water Storage Tank	Max and Min Boron Conc.	2.13	26
3.7.15	Spent Fuel Pool Boron Concentration	Min Boron Concentration	2.14	28
3.9.1	Refueling Operations - Boron Concentration	Min Boron Concentration	2.15	28
5.6.5	Core Operating Limits Report (COLR)	Analytical Methods	1.1	6

The Selected License Commitments that reference this report are listed below:

SLC Section	Selected Licensing Commitment	COLR Parameter	COLR Section	COLR Page
16.7-9	Standby Shutdown System	Standby Makeup Pump Water Supply	2.16	29
16.9-11	Boration Systems – Borated Water Source – Shutdown	Borated Water Volume and Conc. for BAT/RWST	2.17	29
16.9-12	Boration Systems – Borated Water Source – Operating	Borated Water Volume and Conc. for BAT/RWST	2.18	30

Catawba 2 Cycle 18 Core Operating Limits Report

1.1 Analytical Methods

Analytical methods used to determine core operating limits for parameters identified in Technical Specifications and previously reviewed and approved by the NRC as specified in Technical Specification 5.6.5 are as follows.

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

   Revision 0
   Report Date: July 1985
   Not Used for C2C18

2. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, " (W Proprietary).

   Revision 0
   Report Date: August 1985

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

   Revision 2
   Report Date: March 1987
   Not Used for C2C18

4. WCAP-12945-P-A, Volume 1 and Volumes 2-5, "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," (W Proprietary).

   Revision: Volume 1 (Revision 2) and Volumes 2-5 (Revision 1)
   Report Date: March 1998

5. BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," (B&W Proprietary).

   Revision 1
   SER Date: January 22, 1991
   Revision 2
   SER Dates: August 22, 1996 and November 26, 1996.
   Revision 3
   SER Date: June 15, 1994.
   Not Used for C2C18

Catawba 2 Cycle 18 Core Operating Limits Report

1.1 Analytical Methods (continued)

6. DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology," (DPC Proprietary).

   Revision 4a
   Report Date: July 2009

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

   Revision 0a
   Report Date: May 2009

8. DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology".

   Revision 4a
   Report Date: April 2009

9. DPC-NE-2004-PA, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," (DPC Proprietary).

   Revision 2a
   Report Date: December 2008

10. DPC-NE-2005-P-A, "Thermal Hydraulic Statistical Core Design Methodology," (DPC Proprietary).

    Revision 4a
    Report Date: December 2008

11. DPC-NE-2008-PA, "Fuel Mechanical Reload Analysis Methodology Using TACO3," (DPC Proprietary).

    Revision 1a
    Report Date: December 2008
    Not Used for C2C18

12. DPC-NE-2009-P-A, "Westinghouse Fuel Transition Report," (DPC Proprietary).

    Revision 2a
    Report Date: July 2009

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

    Revision 1a
    Report Date: January 2009
    Not Used for C2C18

Catawba 2 Cycle 18 Core Operating Limits Report

1.1 Analytical Methods (continued)

14. DPC-NF-2010-A, "Duke Power Company McGuire Nuclear Station Catawba Nuclear Station Nuclear Physics Methodology for Reload Design."

    Revision 2a
    Report Date: December 2009

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

    Revision 1a
    Report Date: June 2009

16. DPC-NE-1005P-A, "Duke Power Nuclear Design Methodology Using CASMO-4 / SIMULATE-3 MOX", (DPC Proprietary).

    Revision 1
    Report Date: November 12, 2008

17. BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code" (Framatome ANP Proprietary)

    Revision 1
    SER Date: January 14, 2004
    Not Used for C2C18

Catawba 2 Cycle 18 Core Operating Limits Report

2.0 Operating Limits

Cycle-specific parameter limits for specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using NRC approved methodologies specified in Section 1.1.

2.1 Reactor Core Safety Limits (TS 2.1.1)

Reactor Core Safety Limits are shown in **Figure 1**.

2.2 Shutdown Margin - SDM (TS 3.1.1, TS 3.1.4, TS 3.1.5, TS 3.1.6, TS 3.1.8)

2.2.1 For TS 3.1.1, SDM shall be greater than or equal to 1.3% ΔK/K in MODE 2 with Keff < 1.0 and in MODES 3 and 4.

2.2.2 For TS 3.1.1, SDM shall be greater than or equal to 1.0% ΔK/K in MODE 5.

2.2.3 For TS 3.1.4, SDM shall be greater than or equal to 1.3% ΔK/K in MODE 1 and MODE 2.

2.2.4 For TS 3.1.5, SDM shall be greater than or equal to 1.3% ΔK/K in MODE 1 and MODE 2 with any control bank not fully inserted.

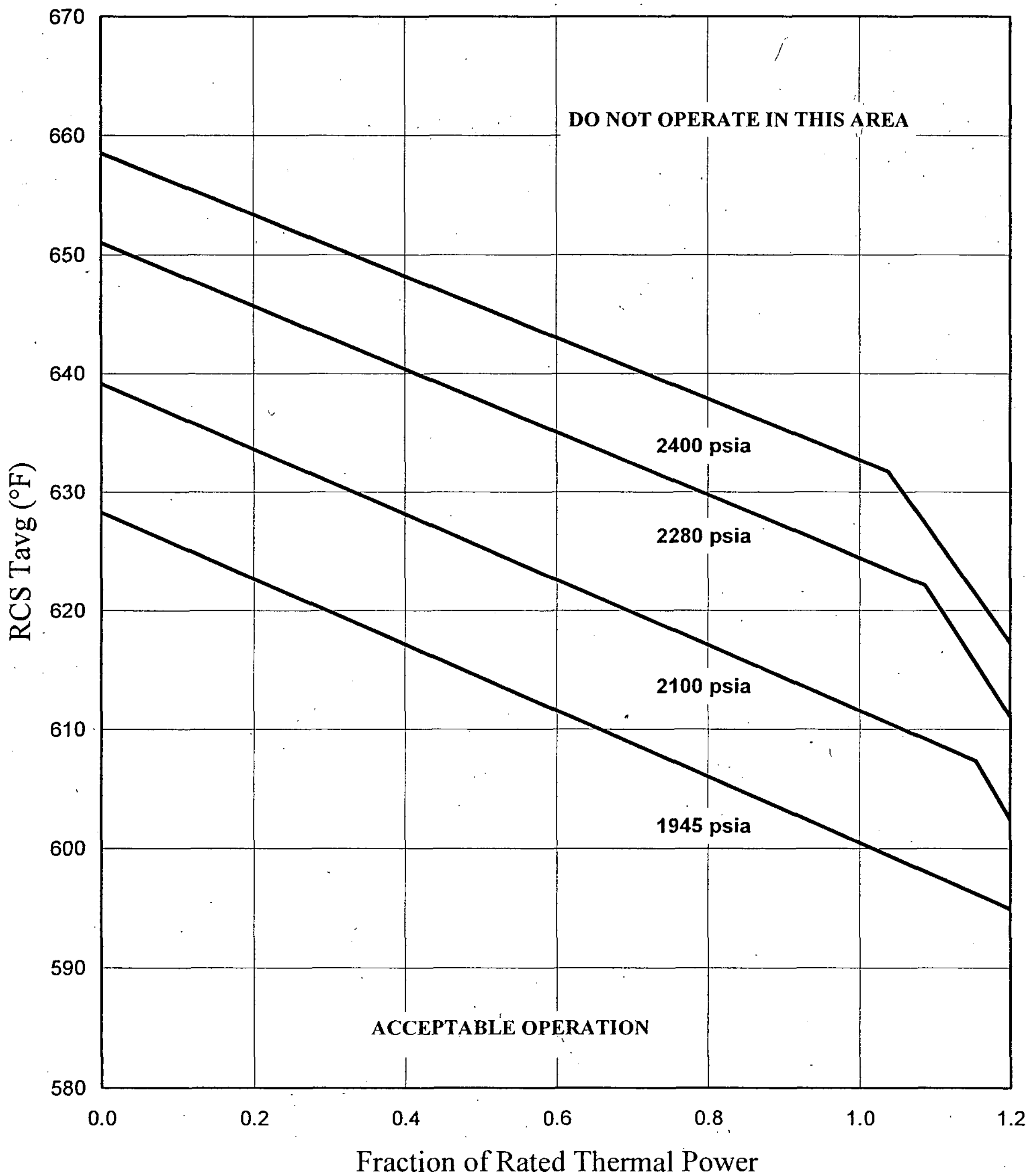
2.2.5 For TS 3.1.6, SDM shall be greater than or equal to 1.3% ΔK/K in MODE 1 and MODE 2 with Keff ≥ 1.0.

2.2.6 For TS 3.1.8, SDM shall be greater than or equal to 1.3% ΔK/K in MODE 2 during PHYSICS TESTS.

Catawba 2 Cycle 18 Core Operating Limits Report

Figure 1

**Reactor Core Safety Limits
Four Loops in Operation**



Catawba 2 Cycle 18 Core Operating Limits Report

2.3 Moderator Temperature Coefficient - MTC (TS 3.1.3)

2.3.1 Moderator Temperature Coefficient (MTC) Limits are:

MTC shall be less positive than the upper limits shown in Figure 2. BOC, ARO, HZP MTC shall be less positive than 0.7E-04 ΔK/K/°F.

EOC, ARO, RTP MTC shall be less negative than the -4.3E-04 ΔK/K/°F lower MTC limit.

2.3.2 300 ppm MTC Surveillance Limit is:

Measured 300 PPM ARO, equilibrium RTP MTC shall be less negative than or equal to -3.65E-04 ΔK/K/°F.

2.3.3 60 PPM MTC Surveillance Limit is:

60 PPM ARO, equilibrium RTP MTC shall be less negative than or equal to -4.125E-04 ΔK/K/°F.

Where:
BOC = Beginning of Cycle (burnup corresponding to most positive MTC)
EOC = End of Cycle
ARO = All Rods Out
HZP = Hot Zero Thermal Power
RTP = Rated Thermal Power
PPM = Parts per million (Boron)

2.4 Shutdown Bank Insertion Limit (TS 3.1.5)

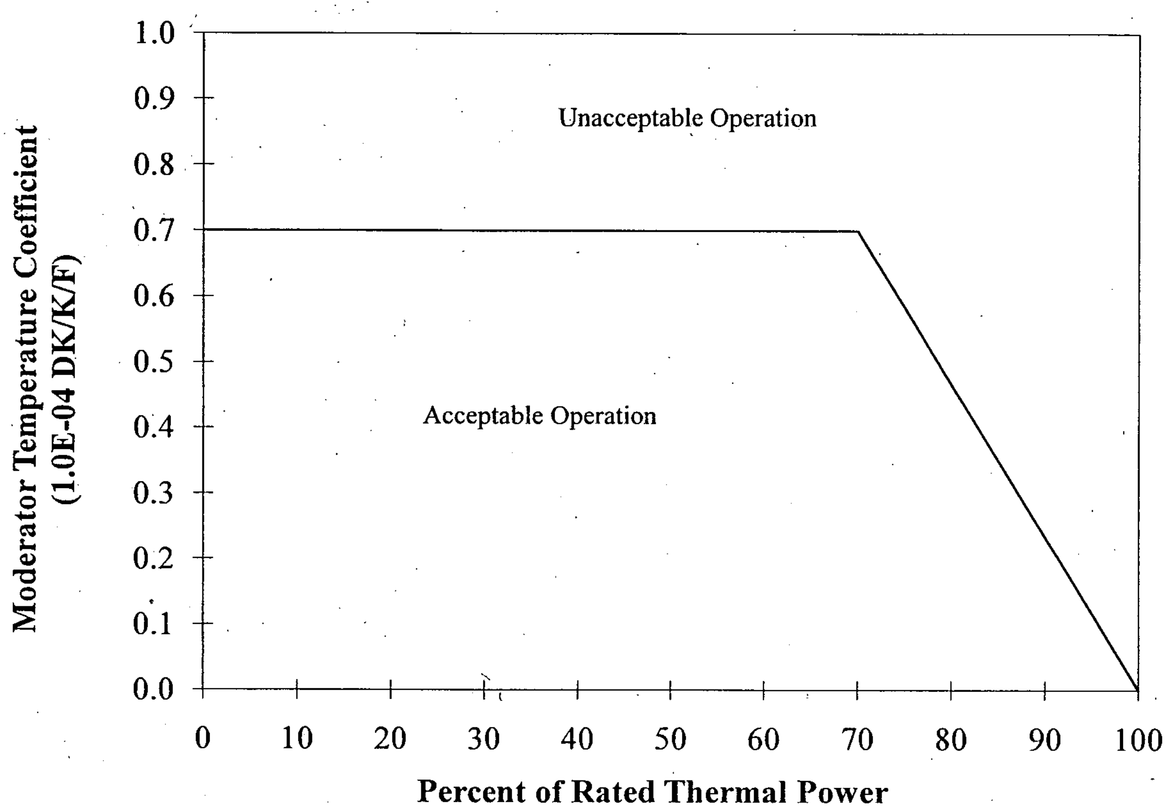
2.4.1 Each shutdown bank shall be withdrawn to at least 222 steps except under conditions listed in Section 2.4.2. Shutdown banks are withdrawn in sequence and with no overlap.

2.4.2 Shutdown banks may be inserted to 219 steps withdrawn individually for up to 48 hours provided the plant was operated in steady state conditions near 100% FP prior to and during this exception.

Catawba 2 Cycle 18 Core Operating Limits Report

Figure 2

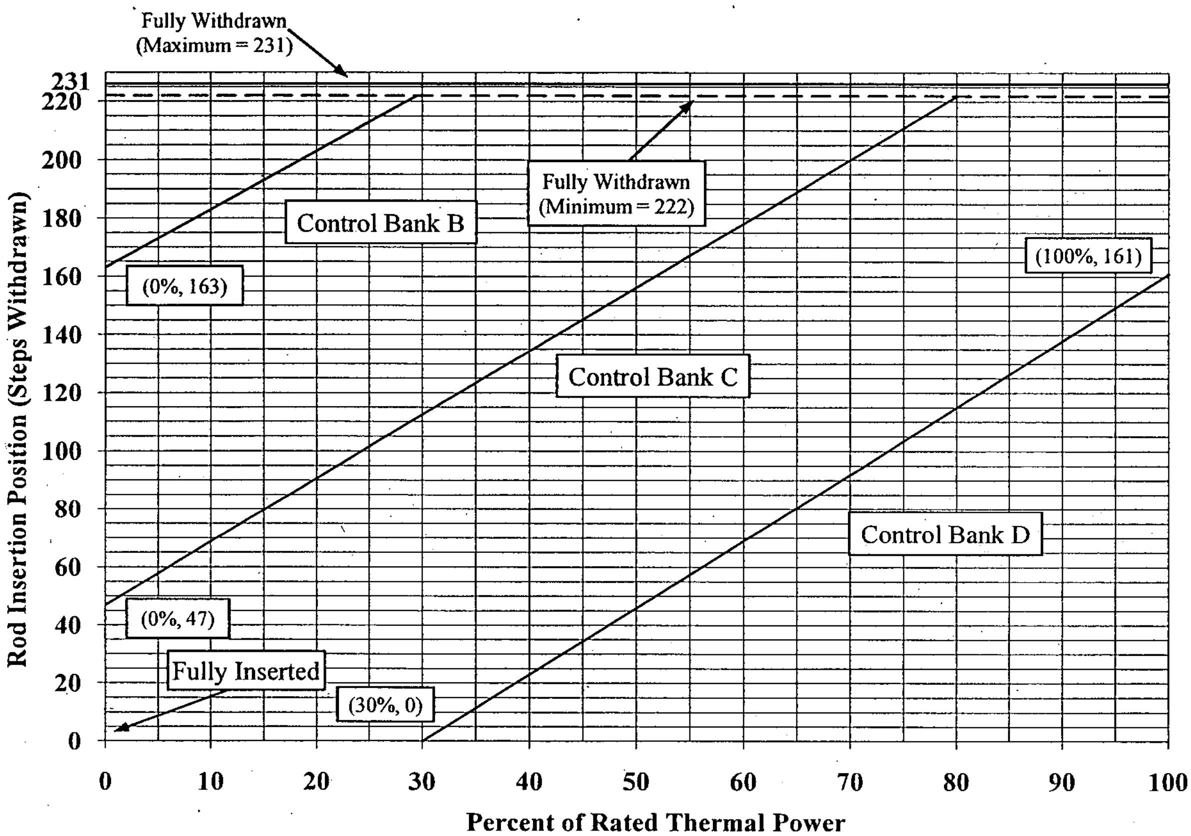
Moderator Temperature Coefficient Upper Limit Versus Power Level



NOTE: Compliance with Technical Specification 3.1.3 may require rod withdrawal limits. Refer to the Unit 2 ROD manual for details.

Catawba 2 Cycle 18 Core Operating Limits Report

Figure 3
Control Bank Insertion Limits Versus Percent Rated Thermal Power



The Rod Insertion Limits (RIL) for Control Bank D (CD), Control Bank C (CC), and Control Bank B (CB) can be calculated by:

Bank CD RIL = 2.3(P) – 69 $\{30 \leq P \leq 100\}$
Bank CC RIL = 2.3(P) +47 $\{0 \leq P \leq 76.1\}$ *for CC RIL = 222* $\{76.1 < P \leq 100\}$
Bank CB RIL = 2.3(P) +163 $\{0 \leq P \leq 25.7\}$ *for CB RIL = 222* $\{25.7 < P \leq 100\}$

where P = %Rated Thermal Power

NOTES: (1) Compliance with Technical Specification 3.1.3 may require rod withdrawal limits. Refer to the Unit 2 ROD manual for details.

(2) Anytime any shutdown bank or control banks A, B, or C are inserted below 222 steps withdrawn, control bank D insertion is limited to ≥ 200 steps withdrawn (see Sections 2.4.2 and 2.5.2)

Catawba 2 Cycle 18 Core Operating Limits Report

Table 1
Control Bank Withdrawal Steps and Sequence

Fully Withdrawn at 222 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
222 Stop	106	0	0
222	116	0 Start	0
222	222 Stop	106	0
222	222	116	0 Start
222	222	222 Stop	106

Fully Withdrawn at 223 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
223 Stop	107	0	0
223	116	0 Start	0
223	223 Stop	107	0
223	223	116	0 Start
223	223	223 Stop	107

Fully Withdrawn at 224 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
224 Stop	108	0	0
224	116	0 Start	0
224	224 Stop	108	0
224	224	116	0 Start
224	224	224 Stop	108

Fully Withdrawn at 225 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
225 Stop	109	0	0
225	116	0 Start	0
225	225 Stop	109	0
225	225	116	0 Start
225	225	225 Stop	109

Fully Withdrawn at 226 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
226 Stop	110	0	0
226	116	0 Start	0
226	226 Stop	110	0
226	226	116	0 Start
226	226	226 Stop	110

Fully Withdrawn at 227 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
227 Stop	111	0	0
227	116	0 Start	0
227	227 Stop	111	0
227	227	116	0 Start
227	227	227 Stop	111

Fully Withdrawn at 228 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
228 Stop	112	0	0
228	116	0 Start	0
228	228 Stop	112	0
228	228	116	0 Start
228	228	228 Stop	112

Fully Withdrawn at 229 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
229 Stop	113	0	0
229	116	0 Start	0
229	229 Stop	113	0
229	229	116	0 Start
229	229	229 Stop	113

Fully Withdrawn at 230 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
230 Stop	114	0	0
230	116	0 Start	0
230	230 Stop	114	0
230	230	116	0 Start
230	230	230 Stop	114

Fully Withdrawn at 231 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
231 Stop	115	0	0
231	116	0 Start	0
231	231 Stop	115	0
231	231	116	0 Start
231	231	231 Stop	115

Catawba 2 Cycle 18 Core Operating Limits Report

2.5 Control Bank Insertion Limits (TS 3.1.6)

2.5.1 Control banks shall be within the insertion, sequence, and overlap limits shown in Figure 3 except under conditions listed in Section 2.5.2. Specific control bank withdrawal and overlap limits as a function of the fully withdrawn position are shown in Table 1.

2.5.2 Control banks A, B, or C may be inserted to 219 steps withdrawn individually for up to 48 hours provided the plant was operated in steady state conditions near 100% FP prior to and during this exception.

2.6 Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$ (TS 3.2.1)

2.6.1 $F_Q(X,Y,Z)$ steady-state limits are defined by the following relationships:

$$F_Q^{RTP} * K(Z)/P \qquad \text{for } P > 0.5$$

$$F_Q^{RTP} * K(Z)/0.5 \qquad \text{for } P \leq 0.5$$

where,

$$P = (\text{Thermal Power})/(\text{Rated Power})$$

Note: Measured $F_Q(X,Y,Z)$ shall be increased by 3% to account for manufacturing tolerances and 5% to account for measurement uncertainty when comparing against the LCO limits. The manufacturing tolerance and measurement uncertainty are implicitly included in the F_Q surveillance limits as defined in COLR Sections 2.6.5 and 2.6.6.

2.6.2 $F_Q^{RTP} = 2.60 \times K(BU)$

2.6.3 K(Z) is the normalized $F_Q(X,Y,Z)$ as a function of core height. K(Z) for Westinghouse RFA fuel is provided in Figure 4.

2.6.4 K(BU) is the normalized $F_Q(X,Y,Z)$ as a function of burnup. K(BU) for Westinghouse RFA fuel is 1.0 at all burnups.

The following parameters are required for core monitoring per the Surveillance Requirements of Technical Specification 3.2.1:

2.6.5 $$[F_Q^L(X,Y,Z)]^{OP} = \frac{F_Q^D(X,Y,Z) * M_Q(X,Y,Z)}{UMT * MT * TILT}$$

Catawba 2 Cycle 18 Core Operating Limits Report

where:

$[F_Q^L(X,Y,Z)]^{OP}$ = Cycle dependent maximum allowable design peaking factor that ensures $F_Q(X,Y,Z)$ LOCA limit is not exceeded for operation within AFD, RIL, and QPTR limits. $F_Q^L(X,Y,Z)^{OP}$ includes allowances for calculational and measurement uncertainties.

$F_Q^D(X,Y,Z)$ = Design power distribution for F_Q. $F_Q^D(X,Y,Z)$ is provided in Appendix Table A-1 for normal operating conditions and in Appendix Table A-4 for power escalation testing during initial startup operation.

$M_Q(X,Y,Z)$ = Margin remaining in core location X,Y,Z to the LOCA limit in the transient power distribution. $M_Q(X,Y,Z)$ is provided in Appendix Table A-1 for normal operating conditions and in Appendix Table A-4 for power escalation testing during initial startup operation.

UMT = Total Peak Measurement Uncertainty. (UMT = 1.05)

MT = Engineering Hot Channel Factor. (MT = 1.03).

TILT = Peaking penalty to account for allowable quadrant power tilt ratio of 1.02. (TILT = 1.035)

2.6.6 $$[F_Q^L(X,Y,Z)]^{RPS} = \frac{F_Q^D(X,Y,Z) * M_C(X,Y,Z)}{UMT * MT * TILT}$$

where:

$[F_Q^L(X,Y,Z)]^{RPS}$ = Cycle dependent maximum allowable design peaking factor that ensures $F_Q(X,Y,Z)$ Centerline Fuel Melt (CFM) limit is not exceeded for operation within AFD, RIL, and QPTR limits. $[F_Q^L(X,Y,Z)]^{RPS}$ includes allowances for calculational and measurement uncertainties.

$F_Q^D(X,Y,Z)$ = Design power distributions for F_Q. $F_Q^D(X,Y,Z)$ is provided in Appendix Table A-1 for normal operating conditions and in Appendix Table A-4 for power escalation testing during initial startup operations.

Catawba 2 Cycle 18 Core Operating Limits Report

$M_C(X,Y,Z)$ = Margin remaining to the CFM limit in core location X,Y,Z from the transient power distribution. $M_C(X,Y,Z)$ is provided in Appendix Table A-2 for normal operating conditions and in Appendix Table A-5 for power escalation testing during initial startup operations.

UMT = Measurement Uncertainty (UMT = 1.05)

MT = Engineering Hot Channel Factor. (MT = 1.03).

TILT = Peaking penalty to account for allowable quadrant power tilt ratio of 1.02. (TILT = 1.035)

2.6.7 KSLOPE = 0.0725

where:

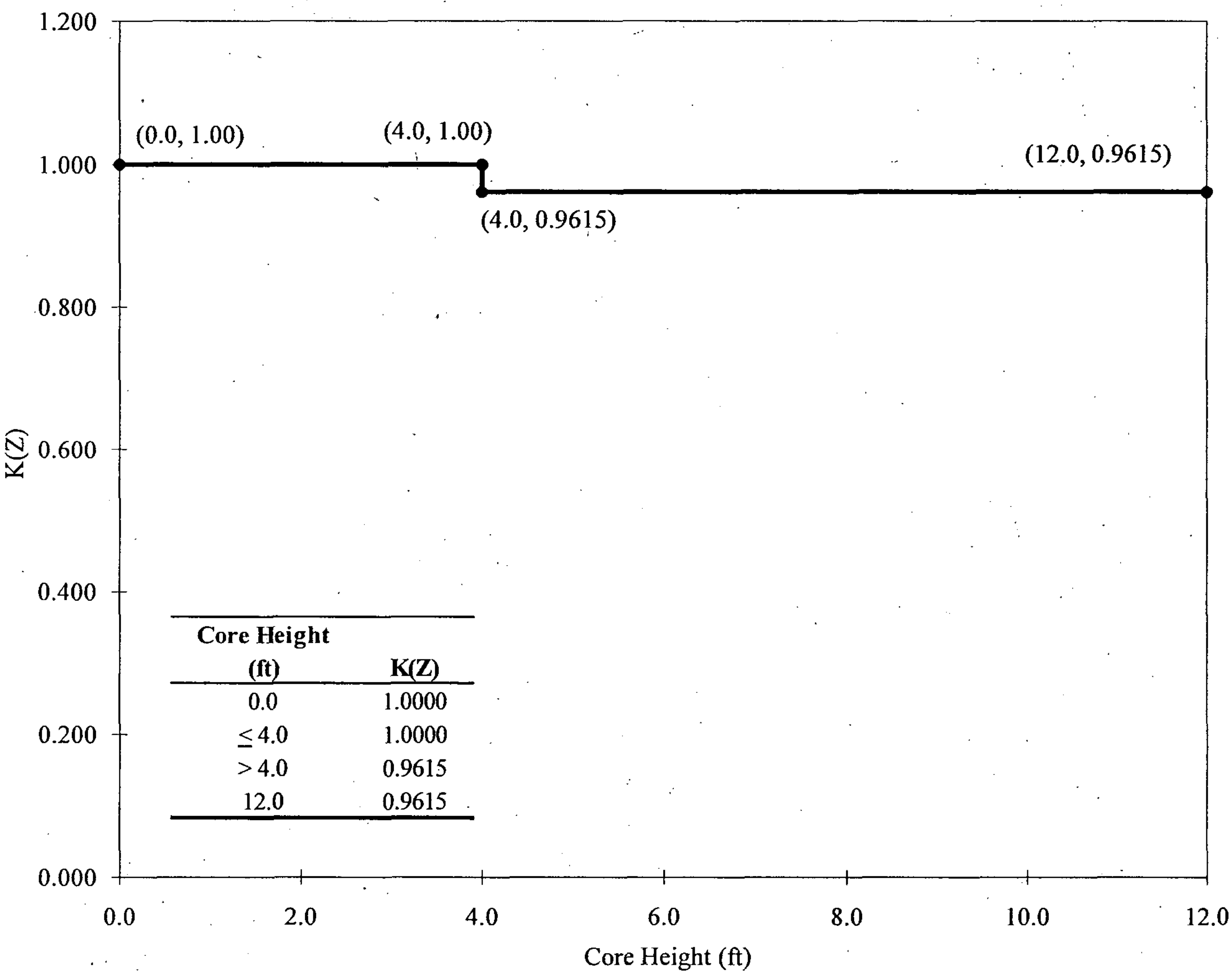
KSLOPE = adjustment to K_1 value from OTΔT trip setpoint required to compensate for each 1% F_Q^M (X,Y,Z) exceeds $[F_Q^L(X,Y,Z)]^{RPS}$.

2.6.8 $F_Q(X,Y,Z)$ Penalty Factors for Technical Specification Surveillances 3.2.1.2 and 3.2.1.3 are provided in Table 2.

Catawba 2 Cycle 18 Core Operating Limits Report

Figure 4

K(Z), Normalized $F_Q(X,Y,Z)$ as a Function of Core Height for RFA Fuel



Core Height (ft)	K(Z)
0.0	1.0000
≤ 4.0	1.0000
> 4.0	0.9615
12.0	0.9615

Catawba 2 Cycle 18 Core Operating Limits Report

Table 2

$F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ Penalty Factors For Tech Spec Surveillances 3.2.1.2, 3.2.1.3 and 3.2.2.2

Burnup (EFPD)	$F_Q(X,Y,Z)$ Penalty Factor(%)	$F_{\Delta H}(X,Y)$ Penalty Factor (%)
4	2.00	2.00
12	2.00	2.00
25	2.00	2.00
50	2.00	2.00
75	2.00	2.00
100	2.00	2.00
125	2.00	2.00
150	2.00	2.00
175	2.00	2.00
200	2.00	2.00
225	2.00	2.00
250	2.00	2.00
275	2.00	2.00
300	2.00	2.00
325	2.00	2.00
350	2.00	2.00
375	2.00	2.00
400	2.00	2.00
425	2.00	2.00
450	2.00	2.00
475	2.00	2.00
490	2.00	2.00
495	2.00	2.00
505	2.00	2.00
520	2.00	2.00

Note: Linear interpolation is adequate for intermediate cycle burnups. All cycle burnups outside the range of the table shall use a 2% penalty factor for both $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ for compliance with Tech Spec Surveillances 3.2.1.2, 3.2.1.3 and 3.2.2.2.

Catawba 2 Cycle 18 Core Operating Limits Report

2.7 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(X,Y)$ (TS 3.2.2)

$F_{\Delta H}$ steady-state limits referred to in Technical Specification 3.2.2 are defined by the following relationship.

2.7.1 $$[F^{L}_{\Delta H}(X,Y)]^{LCO} = MARP(X,Y) * \left[1.0 + \frac{1}{RRH} * (1.0 - P)\right]$$

where:

$[F^{L}_{\Delta H}(X,Y)]^{LCO}$ is the steady-state, maximum allowed radial peak and includes allowances for calculation/measurement uncertainty.

MARP(X,Y) = Cycle-specific operating limit Maximum Allowable Radial Peaks. MARP(X,Y) radial peaking limits are provided in Table 3.

$$P = \frac{\text{Thermal Power}}{\text{Rated Thermal Power}}$$

RRH = Thermal Power reduction required to compensate for each 1% measured radial peak, $F^{M}_{\Delta H}(X,Y)$, exceeds the limit.
(RRH = 3.34, $0.0 < P \leq 1.0$)

The following parameters are required for core monitoring per surveillance requirements of Technical Specification 3.2.2.

2.7.2 $$[F^{L}_{\Delta H}(X,Y)]^{SURV} = \frac{F^{D}_{\Delta H}(X,Y) * M_{\Delta H}(X,Y)}{UMR * TILT}$$

where:

$[F^{L}_{\Delta H}(X,Y)]^{SURV}$ = Cycle dependent maximum allowable design peaking factor that ensures $F_{\Delta H}(X,Y)$ limit is not exceeded for operation within AFD, RIL, and QPTR limits. $F^{L}_{\Delta H}(X,Y)^{SURV}$ includes allowances for calculational and measurement uncertainty.

$F^{D}_{\Delta H}(X,Y)$ = Design power distribution for $F_{\Delta H}$. $F^{D}_{\Delta H}(X,Y)$ is provided in Appendix Table A-3 for normal operation and in Appendix Table A-6 for power escalation testing during initial startup operation.

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$M_{\Delta H}(X,Y)$ = Margin remaining in core location X,Y relative to Operational DNB limits in the transient power distribution. $M_{\Delta H}(X,Y)$ is provided in Appendix Table A-3 for normal operation and in Appendix Table A-6 for power escalation testing during initial startup operation.

UMR = Uncertainty value for measured radial peaks (UMR = 1.0). UMR is 1.0 since a factor of 1.04 is implicitly included in the variable $M_{\Delta H}(X,Y)$.

TILT = Peaking penalty to account for allowable quadrant power tilt ratio of 1.02. (TILT = 1.035)

2.7.3 RRH = 3.34

where:

RRH = Thermal Power reduction required to compensate for each 1% measured radial peak, $F^{M}_{\Delta H}$ (X,Y) exceeds its limit. ($0 < P \leq 1.0$)

2.7.4 TRH = 0.04

where:

TRH = Reduction in OTΔT K_1 setpoint required to compensate for each 1% measured radial peak, $F^{M}_{\Delta H}$ (X,Y) exceeds its limit.

2.7.5 $F_{\Delta H}(X,Y)$ Penalty Factors for Technical Specification Surveillance 3.2.2.2 are provided in Table 2.

2.8 Axial Flux Difference – AFD (TS 3.2.3)

2.8.1 Axial Flux Difference (AFD) Limits are provided in Figure 5.

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Table 3
Maximum Allowable Radial Peaks (MARPS)

RFA Fuel MARPs
100% Full Power

Core Height (ft)	Axial Peak												
	1.05	1.1	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	2.1	3.0	3.25
0.12	1.8092	1.8553	1.9489	1.9953	1.9741	2.1073	2.0498	2.009	1.9333	1.8625	1.778	1.3151	1.2461
1.20	1.8102	1.854	1.9401	1.9953	1.9741	2.1073	2.0191	1.9775	1.9009	1.8306	1.7852	1.3007	1.2235
2.40	1.8093	1.8525	1.9312	1.9779	1.9741	2.0735	1.9953	1.9519	1.876	1.8054	1.732	1.4633	1.4616
3.60	1.8098	1.8514	1.9204	1.9641	1.9741	2.0495	1.9656	1.9258	1.8524	1.7855	1.6996	1.4675	1.3874
4.80	1.8097	1.8514	1.9058	1.9449	1.9741	2.0059	1.9441	1.9233	1.8538	1.7836	1.6714	1.2987	1.2579
6.00	1.8097	1.8514	1.8921	1.9212	1.9455	1.9336	1.8798	1.8625	1.8024	1.7472	1.6705	1.3293	1.2602
7.20	1.807	1.8438	1.8716	1.893	1.8872	1.8723	1.8094	1.7866	1.7332	1.6812	1.5982	1.2871	1.2195
8.40	1.8073	1.8319	1.8452	1.8571	1.8156	1.795	1.7359	1.7089	1.6544	1.601	1.5127	1.2182	1.1578
9.60	1.8072	1.8102	1.8093	1.7913	1.7375	1.7182	1.6572	1.6347	1.5808	1.5301	1.4444	1.1431	1.0914
10.80	1.798	1.7868	1.7611	1.7163	1.6538	1.6315	1.5743	1.5573	1.5088	1.4624	1.3832	1.1009	1.047
11.40	1.7892	1.7652	1.725	1.6645	1.6057	1.5826	1.5289	1.5098	1.4637	1.4218	1.3458	1.067	1.0142

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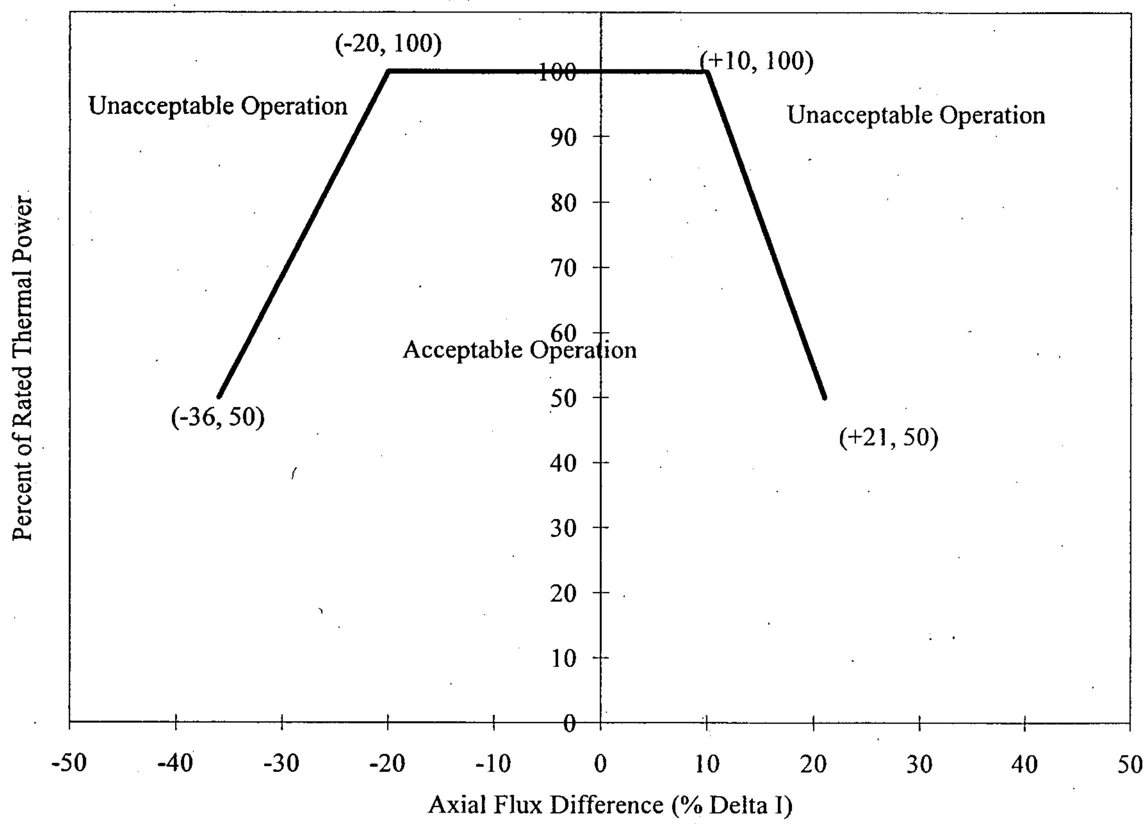
Figure 5

Percent of Rated Thermal Power Versus Percent Axial Flux Difference Limits



NOTE: Compliance with Technical Specification 3.2.1 may require more restrictive AFD limits. Refer to the Unit 2 ROD manual for operational AFD limits.

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2.9 Reactor Trip System Instrumentation Setpoints (TS 3.3.1) Table 3.3.1-1

2.9.1 Overtemperature ΔT Setpoint Parameter Values

Parameter	Nominal Value
Nominal Tavg at RTP	$T' \leq 590.8$ °F
Nominal RCS Operating Pressure	$P' = 2235$ psig
Overtemperature ΔT reactor trip setpoint	$K_1 = 1.1953$
Overtemperature ΔT reactor trip heatup setpoint penalty coefficient	$K_2 = 0.03163/°F$
Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient	$K_3 = 0.001414/psi$
Time constants utilized in the lead-lag compensator for ΔT	$\tau_1 = 8$ sec. $\tau_2 = 3$ sec.
Time constant utilized in the lag compensator for ΔT	$\tau_3 = 0$ sec.
Time constants utilized in the lead-lag compensator for T_{avg}	$\tau_4 = 22$ sec. $\tau_5 = 4$ sec.
Time constant utilized in the measured T_{avg} lag compensator	$\tau_6 = 0$ sec.
$f_1(\Delta I)$ "positive" breakpoint	= 3.0 %ΔI
$f_1(\Delta I)$ "negative" breakpoint	= N/A*
$f_1(\Delta I)$ "positive" slope	= 1.525 $\%\Delta T_0/\%\Delta I$
$f_1(\Delta I)$ "negative" slope	= N/A*

* $f_1(\Delta I)$ negative breakpoints and slopes for OTΔT are less restrictive than OPΔT $f_2(\Delta I)$ negative breakpoint and slope. Therefore, during a transient which challenges negative imbalance limits, OPΔT $f_2(\Delta I)$ limits will result in a reactor trip before OTΔT $f_1(\Delta I)$ limits are reached. This makes implementation of an OTΔT $f_1(\Delta I)$ negative breakpoint and slope unnecessary.

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2.9.2 Overpower ΔT Setpoint Parameter Values

Parameter	Nominal Value
Nominal Tavg at RTP	T" ≤ 590.8 °F
Overpower ΔT reactor trip setpoint	K_4 = 1.0819
Overpower ΔT reactor trip penalty	K_5 = 0.02 / °F for increasing Tavg K_5 = 0.00 / °F for decreasing Tavg
Overpower ΔT reactor trip heatup setpoint penalty coefficient	K_6 = 0.001291/°F for T > T" K_6 = 0.0 /°F for T ≤ T"
Time constants utilized in the lead-lag compensator for ΔT	τ_1 = 8 sec. τ_2 = 3 sec.
Time constant utilized in the lag compensator for ΔT	τ_3 = 0 sec.
Time constant utilized in the measured T_{avg} lag compensator	τ_6 = 0 sec.
Time constant utilized in the rate-lag controller for T_{avg}	τ_7 = 10 sec.
$f_2(\Delta I)$ "positive" breakpoint	= 35.0 %ΔI
$f_2(\Delta I)$ "negative" breakpoint	= -35.0 %ΔI
$f_2(\Delta I)$ "positive" slope	= 7.0 %ΔT_0/ %ΔI
$f_2(\Delta I)$ "negative" slope	= 7.0 %ΔT_0/ %ΔI

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2.10 Boron Dilution Mitigation System (TS 3.3.9)

2.10.1 Reactor Makeup Water Pump flow rate limits:

Applicable Mode	Limit
MODE 3	≤ 150 gpm
MODE 4 or 5	≤ 70 gpm

2.11 RCS Pressure, Temperature and Flow Limits for DNB (TS 3.4.1)

RCS pressure, temperature and flow limits for DNB are shown in Table 4.

2.12 Accumulators (TS 3.5.1)

2.12.1 Boron concentration limits during MODES 1 and 2, and MODE 3 with RCS pressure >1000 psi:

Parameter	Limit
Accumulator minimum boron concentration.	2,500 ppm
Accumulator maximum boron concentration.	3,075 ppm

2.13 Refueling Water Storage Tank - RWST (TS 3.5.4)

2.13.1 Boron concentration limits during MODES 1, 2, 3, and 4:

Parameter	Limit
RWST minimum boron concentration.	2,700 ppm
RWST maximum boron concentration.	3,075 ppm

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Table 4

Reactor Coolant System DNB Parameters

PARAMETER	INDICATION	No. Operable CHANNELS	LIMITS
1. Indicated RCS Average Temperature	meter	4	≤ 589.6 °F
	meter	3	≤ 589.3 °F
	computer	4	≤ 590.1 °F
	computer	3	≤ 589.9 °F
2. Indicated Pressurizer Pressure	meter	4	≥ 2219.8 psig
	meter	3	≥ 2222.1 psig
	computer	4	≥ 2215.8 psig
	computer	3	≥ 2217.5 psig
3. RCS Total Flow Rate			≥ 390,000 gpm

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2.14 Spent Fuel Pool Boron Concentration (TS 3.7.15)

2.14.1 Minimum boron concentration limit for the spent fuel pool. Applicable when fuel assemblies are stored in the spent fuel pool.

Parameter	Limit
Spent fuel pool minimum boron concentration.	2,700 ppm

2.15 Refueling Operations - Boron Concentration (TS 3.9.1)

2.15.1 Minimum boron concentration limit for filled portions of the Reactor Coolant System, refueling canal, and refueling cavity for MODE 6 conditions. The minimum boron concentration limit and plant refueling procedures ensure that Core Keff remains within MODE 6 reactivity requirement of $Keff \leq 0.95$.

Parameter	Limit
Minimum boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity.	2,700 ppm

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2.16 Standby Shutdown System - Standby Makeup Pump Water Supply - (SLC-16.7-9.3)

2.16.1 Minimum boron concentration limit for the spent fuel pool. Applicable for MODES 1, 2, and 3.

Parameter	Limit
Spent fuel pool minimum boron concentration for TR 16.7-9.3.	2,700 ppm

2.17 Borated Water Source – Shutdown (SLC 16.9-11)

2.17.1 Volume and boron concentrations for the Boric Acid Tank (BAT) and the Refueling Water Storage Tank (RWST) during MODE 4 with any RCS cold leg temperature ≤ 210°F, and MODES 5 and 6.

Parameter	Limit
BAT minimum boron concentration	7,000 ppm
Volume of 7,000 ppm boric acid solution required to maintain SDM at 68°F	2000 gallons
BAT Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-11)	13,086 gallons (14.9%)

NOTE: When cycle burnup is > 450 EFPD, Figure 6 may be used to determine required BAT minimum level.

Parameter	Limit
RWST minimum boron concentration	2,700 ppm
Volume of 2,700 ppm boric acid solution required to maintain SDM at 68 °F	7,000 gallons
RWST Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-11)	48,500 gallons (8.7%)

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2.18 Borated Water Source - Operating (SLC 16.9-12)

2.18.1 Volume and boron concentrations for the Boric Acid Tank (BAT) and the Refueling Water Storage Tank (RWST) during MODES 1, 2, and 3 and MODE 4 with all RCS cold leg temperatures > 210^{O}F.

Parameter	Limit
BAT minimum boron concentration	7,000 ppm
Volume of 7,000 ppm boric acid solution required to maintain SDM at 210°F	13,500 gallons
BAT Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-12)	25,200 gallons (45.8%)

NOTE: When cycle burnup is > 450 EFPD, Figure 6 may be used to determine required BAT minimum level.

Parameter	Limit
RWST minimum boron concentration	2,700 ppm
Volume of 2,700 ppm boric acid solution required to maintain SDM at 210 °F	57,107 gallons
RWST Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-12)	98,607 gallons (22.0%)

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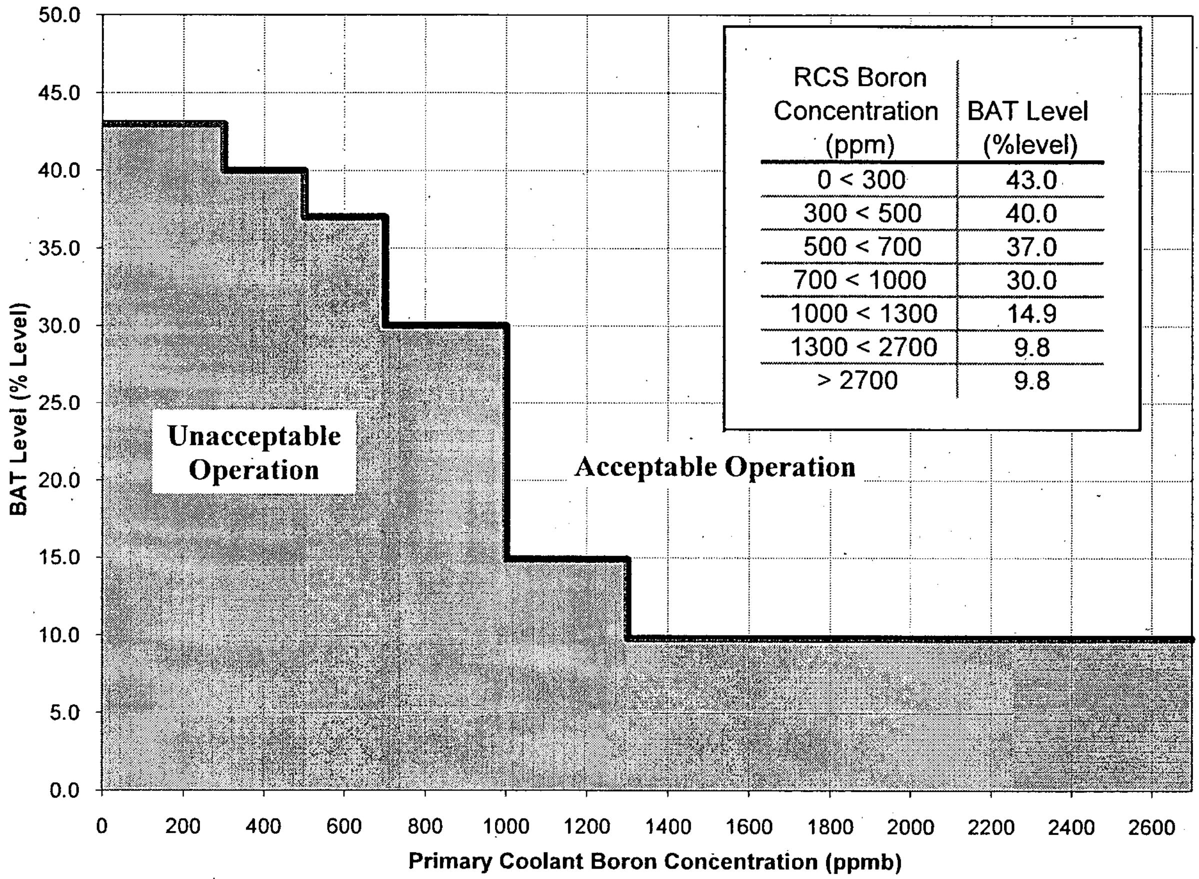
Figure 6

Boric Acid Storage Tank Indicated Level Versus Primary Coolant Boron Concentration

(Valid When Cycle Burnup is > 450 EFPD)

This figure includes additional volumes listed in SLC 16.9-11 and 16.9-12



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Appendix A

Power Distribution Monitoring Factors

Appendix A contains power distribution monitoring factors used in Technical Specification Surveillance. This data was generated in the Catawba 2 Cycle 18 Maneuvering Analysis calculation file, CNC-1553.05-00-0529. Due to the size of monitoring factor data, Appendix A is controlled electronically within Duke and is not included in Duke internal copies of the COLR. Catawba Reactor and Electrical Systems Engineering controls monitoring factor via computer files and should be contacted if questions concerning this information arise.

Appendix A is included in the COLR transmitted to the NRC.