



Tom Simril
Vice President
Catawba Nuclear Station

Duke Energy
CN01VP | 4800 Concord Road
York, SC 29745

o: 803.701.3340
f: 803.701.3221
tom.simril@duke-energy.com

RA-19-0365

September 12, 2019

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

Subject: Duke Energy Carolinas, LLC (Duke Energy)
Catawba Nuclear Station, Unit 2
Facility Operating License Number NPF-52
Docket Number 50-414
Core Operating Limits Report (COLR) for Unit 2 Cycle 23 Reload Core, Rev. 1

Pursuant to Catawba Technical Specification 5.6.5.d, please find attached an information copy of the subject COLR. This COLR revision is being submitted to update the limits of the Unit 2 Cycle 23 reload core.

The power distribution monitoring factors from Appendix A of Revision 0 remain valid and are not transmitted as part of Revision 1.

This letter and enclosed COLR do not contain any regulatory commitments.

Please direct any questions or concerns to Douglas Brunson, Engineer III, at (803) 701-3865.

Sincerely,

A handwritten signature in black ink that reads "Tom Simril". The signature is fluid and cursive, with the first name "Tom" and last name "Simril" clearly legible.

Tom Simril
Vice President, Catawba Nuclear Station

Enclosure (Catawba Unit 2 Cycle 23 COLR, Rev. 1)

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xc (with enclosures):

L. Dudes, Region II Administrator
U.S. Nuclear Regulatory Commission
Marquis One Tower
245 Peachtree Center Avenue NE, Suite 1200
Atlanta, GA 30303-1257

J. D. Austin, Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Catawba Nuclear Station

M. Mahoney, Project Manager
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
Mailstop O-8B1A
Rockville, MD 20852

Enclosure

**Catawba Unit 2 Cycle 23 COLR, Rev. 1
(paper COLR excluding Attachment A)**



Facility Code :	CN	
Applicable Facilities :		
Document Number :	CNEI-0400-332	
Document Revision Number :	001	
Document EC Number :		
Change Reason :	CR #02278974. EC not required per AD-NF-ALL-0807 Reload Design Process	
Document Title :	Catawba 2 Cycle 23 Core Operating Limits Report	
Hager, Nicholas R	Originator	7/11/2019
Klein, Casey	Verifier	7/11/2019
Hight, Randy A	Safety Analysis Verifier	7/11/2019
Murphy, William J	Site Acceptance Review	7/11/2019
Blom, Michael A	Approver	7/11/2019
Notes :		

Catawba 2 Cycle 23

Core Operating Limits Report
Revision 1

July 2019

Reference: CNC-1553.05-00-0666, Rev. 1
Reload 50.59 #: 02181851



QA CONDITION 1

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

Catawba 2 Cycle 23 Core Operating Limits Report

Implementation Instructions for Revision 1

Revision Description and CR Tracking

Revision 1 of the Catawba Unit 2 Cycle 23 COLR contains limits specific to the reload core.

This revision was initiated by CR #02278974 which describes the vendor Loss of Coolant Accident (LOCA) analysis update which incorporated a required surveillance of the initial FΔH assumption including measurement uncertainty. This design criterion requires surveillance of TS 3.2.2 against FΔH LOCA limit in addition to Loss of Flow Accident (LOFA) Maximum Allowable Radial Peak (MARP) limits. Table 3 of the COLR is revised to reflect the new limits for this surveillance of the limiting value between LOFA MARPs and FΔH LOCA peaking limits.

The power distribution monitoring factors from Appendix A of Revision 0 remain valid and are not transmitted as part of Revision 1.

Implementation Schedule

The Catawba Unit 2 Cycle 23 COLR requires the reload 50.59 (AR #02181851) be approved prior to implementation and fuel loading.

Revision 1 may become effective immediately up receipt. The Catawba Unit 2 Cycle 23 COLR will cease to be effective during No MODE between cycle 23 and 24.

Data files to be Implemented

No data files are transmitted as part of this document.

Additional Information

CDR was performed by Safety Analysis for COLR Sections 1.1, 2.1, and 2.9 – 2.18.
CNS Reactor Engineering performed site inspection in accordance with AD-NF-ALL-0807 and AD-NF-NGO-0214.

Catawba 2 Cycle 23 Core Operating Limits Report

REVISION LOG

<u>Revision</u>	<u>Effective Date</u>	<u>Pages Affected</u>	<u>COLR</u>
0	February 2018	1-31, Appendix A*	C2C23 COLR, Rev. 0
1	July 2019	1-3, 21	C2C23 COLR, Rev. 1

* 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 23 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 along with the NRC approved analytical methods used to develop and/or determine COLR parameters identified in Technical Specifications.

TS Section	Technical Specifications	COLR Parameter	COLR Section	NRC Approved Methodology (Section 1.1 Number)
2.1.1	Reactor Core Safety Limits	RCS Temperature and Pressure Safety Limits	2.1	6, 7, 8, 9, 10, 12, 15, 16, 19, 20
3.1.1	Shutdown Margin	Shutdown Margin	2.2	6, 7, 8, 12, 14, 15, 16, 19, 20
3.1.3	Moderator Temperature Coefficient	MTC	2.3	6, 7, 8, 14, 16, 18
3.1.4	Rod Group Alignment Limits	Shutdown Margin	2.2	6, 7, 8, 12, 14, 15, 16, 19, 20
3.1.5	Shutdown Bank Insertion Limit	Shutdown Margin Rod Insertion Limits	2.2 2.4	2, 4, 6, 7, 8, 9, 10 12, 14, 15, 16, 19, 20
3.1.6	Control Bank Insertion Limit	Shutdown Margin Rod Insertion Limits	2.2 2.5	2, 4, 6, 7, 8, 9, 10 12, 14, 15, 16, 19, 20
3.1.8	Physics Tests Exceptions	Shutdown Margin	2.2	6, 7, 8, 12, 14, 15, 16, 19, 20
3.2.1	Heat Flux Hot Channel Factor	F _Q AFD OTΔT Penalty Factors	2.6 2.8 2.9 2.6	2, 4, 6, 7, 8, 9, 10 12, 15, 16, 19, 20
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor	FΔH Penalty Factors	2.7 2.7	2, 4, 6, 7, 8, 9, 10 12, 15, 16, 19, 20
3.2.3	Axial Flux Difference	AFD	2.8	2, 4, 6, 7, 8, 15, 16
3.3.1	Reactor Trip System Instrumentation	OTΔT OPΔT	2.9 2.9	6, 7, 8, 9, 10, 12 15, 16, 19, 20
3.3.9	Boron Dilution Mitigation System	Reactor Makeup Water Flow Rate	2.10	6, 7, 8, 14, 16
3.4.1	RCS Pressure, Temperature and Flow limits for DNB	RCS Pressure, Temperature and Flow	2.11	6, 7, 8, 9, 10, 12, 19, 20
3.5.1	Accumulators	Max and Min Boron Conc.	2.12	6, 7, 8, 14, 16
3.5.4	Refueling Water Storage Tank	Max and Min Boron Conc.	2.13	6, 7, 8, 14, 16
3.7.15	Spent Fuel Pool Boron Concentration	Min Boron Concentration	2.14	6, 7, 8, 14, 16
3.9.1	Refueling Operations - Boron Concentration	Min Boron Concentration	2.15	6, 7, 8, 14, 16
5.6.5	Core Operating Limits Report (COLR)	Analytical Methods	1.1	None

The Selected Licensee Commitments that reference this report are listed below

SLC Section	Selected Licensing Commitment	COLR Parameter	COLR Section	NRC Approved Methodology (Section 1.1 Number)
16.7-9	Standby Shutdown System	Standby Makeup Pump Water Supply	2.16	6, 7, 8, 14, 16
16.9-11	Boration Systems – Borated Water Source – Shutdown	Borated Water Volume and Conc. for BAT/RWST	2.17	6, 7, 8, 14, 16
16.9-12	Boration Systems – Borated Water Source – Operating	Borated Water Volume and Conc. for BAT/RWST	2.18	6, 7, 8, 14, 16

Catawba 2 Cycle 23 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 C2C23

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

Revision 0

Report Date: August 1985

Addendum 2, "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," (W Proprietary). (Referenced in Duke Letter DPC-06-101)

Revision 1

Report Date: July 1997

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 C2C23

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 C2C23

Catawba 2 Cycle 23 Core Operating Limits Report

1.1 Analytical Methods (continued)

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

Revision 5a
Report Date: October 2012
7. DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," (DPC Proprietary).

Revision 1
Report Date: March 2015
8. DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology".

Revision 4b
Report Date: September 2010
9. DPC-NE-2004P-A, "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-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," (DPC Proprietary).

Revision 5
Report Date: October 2016
11. DPC-NE-2008-PA, "Fuel Mechanical Reload Analysis Methodology Using TACO3," (DPC Proprietary).

Revision 0
Report Date: April 1995
Not Used for C2C23
12. DPC-NE-2009-P-A, "Westinghouse Fuel Transition Report," (DPC Proprietary).

Revision 3c
Report Date: March 2017
13. DPC-NE-1004-A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P."

Revision 1a
Report Date: January 2009
Not Used for C2C23

Catawba 2 Cycle 23 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-PA, "Duke Power Company Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," (DPC Proprietary).

Revision 1a

Report Date: June 2009

16. DPC-NE-1005-PA, "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 C2C23

18. DPC-NE-1007-PA, "Conditional Exemption of the EOC MTC Measurement Methodology", (DPC and W Proprietary)

Revision 0

Report Date: April 2015

19. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," (W Proprietary).

Revision 0

Report Date: April 1995

20. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," (W Proprietary).

Revision 0

Report Date: July 2006

Catawba 2 Cycle 23 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% $\Delta K/K$ in MODE 2 with $K_{eff} < 1.0$ and in MODES 3 and 4.

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

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

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

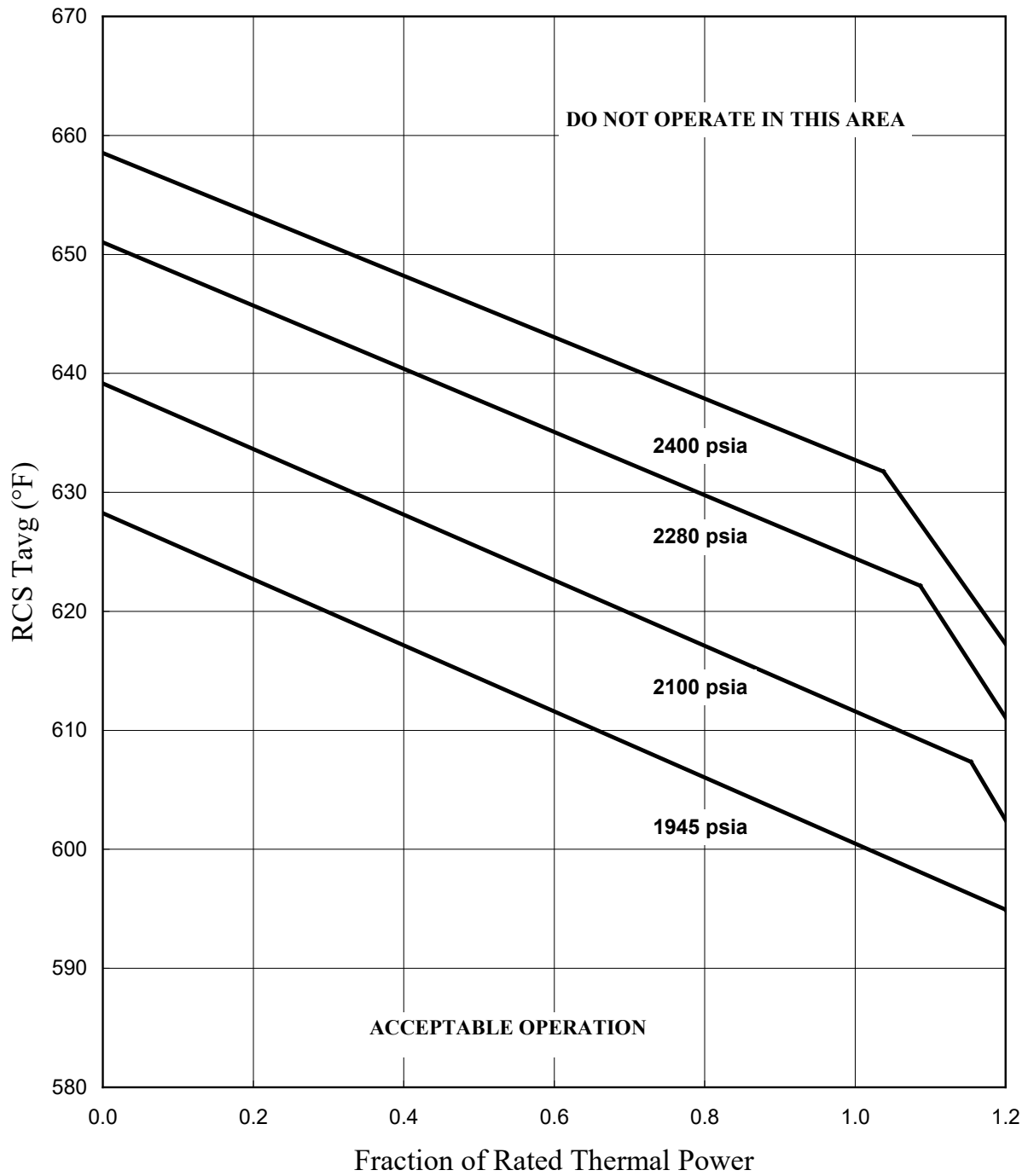
2.2.5 For TS 3.1.6, SDM shall be greater than or equal to 1.3% $\Delta K/K$ in MODE 1 and MODE 2 with $K_{eff} \geq 1.0$.

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

Catawba 2 Cycle 23 Core Operating Limits Report

Figure 1

Reactor Core Safety Limits Four Loops in Operation



Catawba 2 Cycle 23 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 $0.7E-04 \Delta K/K/^{\circ}F$.

EOC, ARO, RTP MTC shall be less negative than the $-4.3E-04 \Delta K/K/^{\circ}F$ 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 $-3.65E-04 \Delta K/K/^{\circ}F$.

2.3.3 The Revised Predicted near-EOC 300ppm ARO RTP MTC shall be calculated using the procedure contained in DPC-NE-1007-PA.

If the Revised Predicted MTC is less negative than or equal to the 300 ppm SR 3.1.3.2 Surveillance Limit, and all benchmark data contained in the surveillance procedure is satisfied, then an MTC measurement in accordance with SR 3.1.3.2 is not required to be performed.

2.3.4 60 ppm MTC Surveillance Limit is:

Measured 60 ppm ARO, equilibrium RTP MTC shall be less negative than or equal to $-4.125E-04 \Delta K/K/^{\circ}F$.

Where:

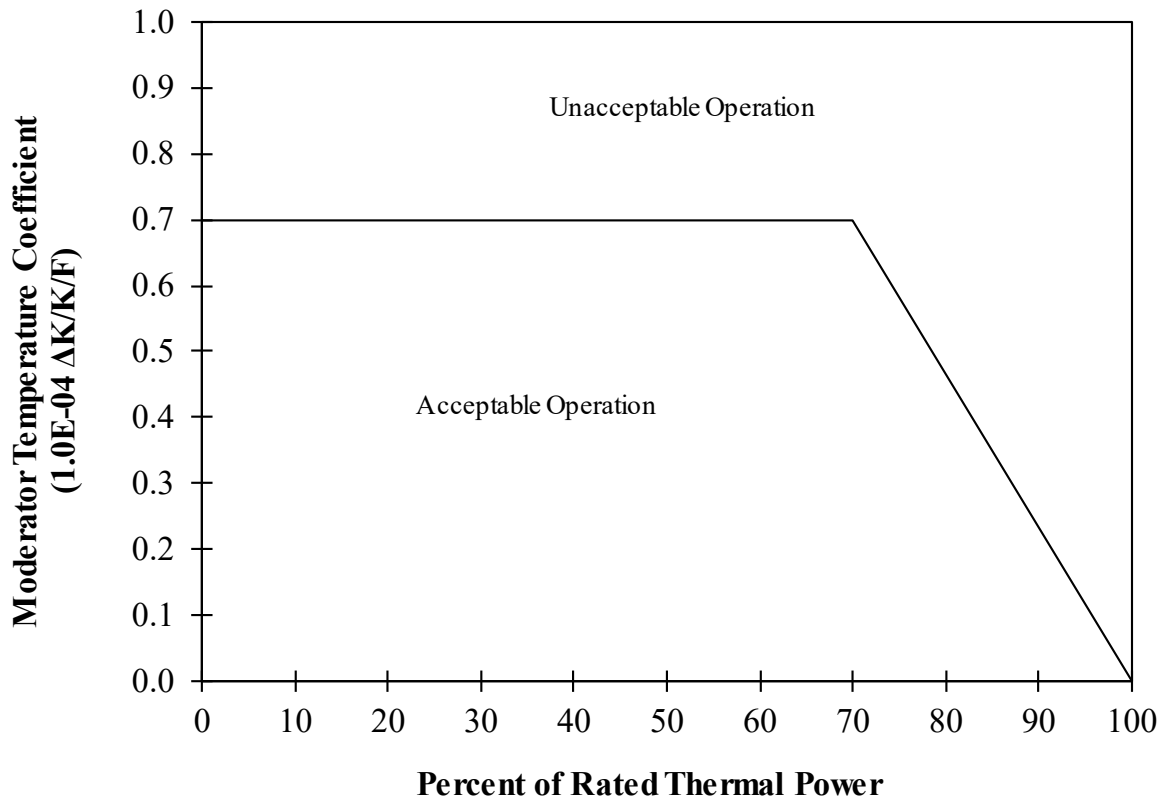
- BOC = Beginning of Cycle (burnup corresponding to most positive MTC)
- EOC = End of Cycle
- ARO = All Rods Out
- HZP = Hot Zero Thermal Power
- RTP = Rated Thermal Power
- ppm = Parts per million (Boron)

2.4 Shutdown Bank Insertion Limit (TS 3.1.5)

2.4.1 Each shutdown bank shall be withdrawn to at least 222 steps. Shutdown banks are withdrawn in sequence and with no overlap.

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. Specific control bank withdrawal and overlap limits as a function of the fully withdrawn position are shown in Table 1.

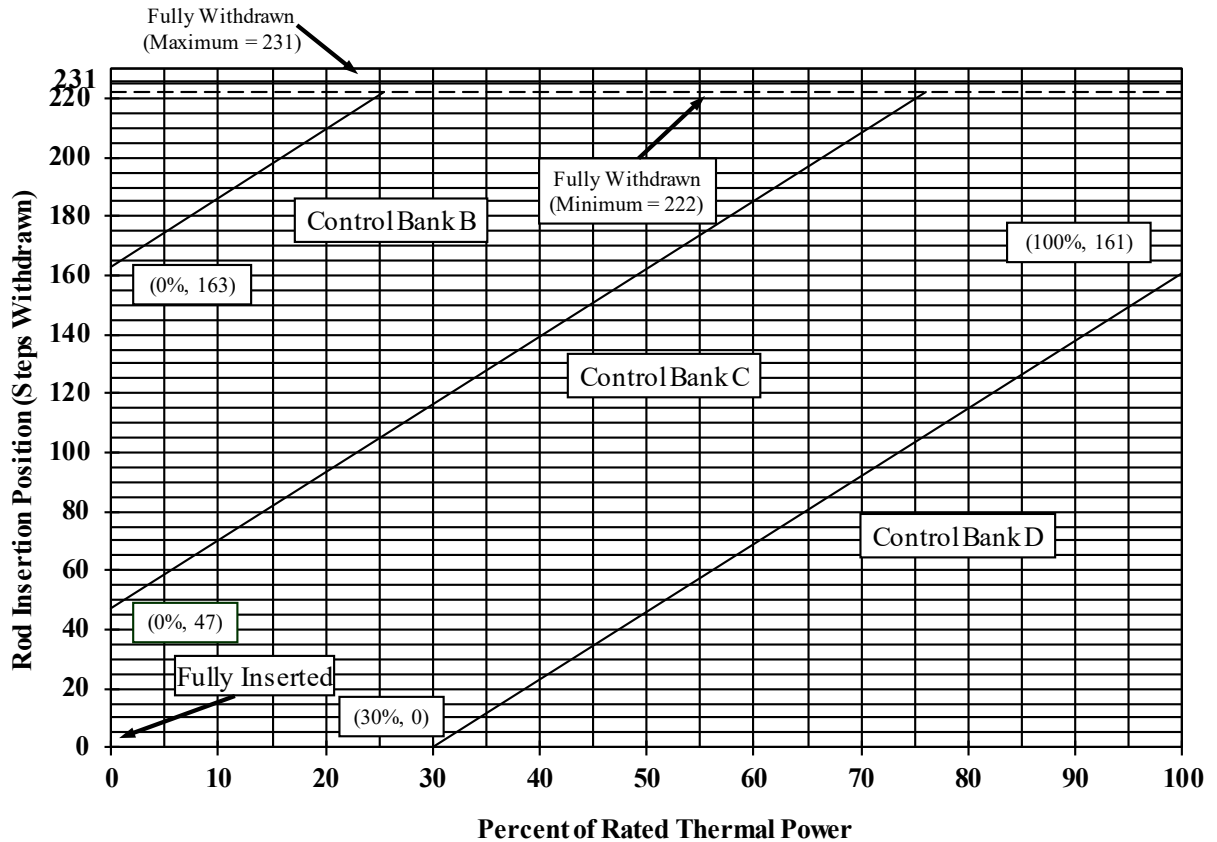
Catawba 2 Cycle 23 Core Operating Limits Report**Figure 2****Moderator Temperature Coefficient Upper Limit Versus Power Level**

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 23 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:

$$\text{Bank CD RIL} = 2.3(P) - 69 \quad \{30 \leq P \leq 100\}$$

$$\text{Bank CC RIL} = 2.3(P) + 47 \quad \{0 \leq P \leq 76.1\} \quad \text{for CC RIL} = 222 \quad \{76.1 < P \leq 100\}$$

$$\text{Bank CB RIL} = 2.3(P) + 163 \quad \{0 \leq P \leq 25.7\} \quad \text{for CB RIL} = 222 \quad \{25.7 < P \leq 100\}$$

where $P = \% \text{ of Rated Thermal Power}$

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 23 Core Operating Limits Report

Table 1
Control Bank Withdrawal Steps and Sequence

Fully Withdrawn at 222 Steps				Fully Withdrawn at 223 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
222 Stop	106	0	0	223 Stop	107	0	0
222	116	0 Start	0	223	116	0 Start	0
222	222 Stop	106	0	223	223 Stop	107	0
222	222	116	0 Start	223	223	116	0 Start
222	222	222 Stop	106	223	223	223 Stop	107

Fully Withdrawn at 224 Steps				Fully Withdrawn at 225 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
224 Stop	108	0	0	225 Stop	109	0	0
224	116	0 Start	0	225	116	0 Start	0
224	224 Stop	108	0	225	225 Stop	109	0
224	224	116	0 Start	225	225	116	0 Start
224	224	224 Stop	108	225	225	225 Stop	109

Fully Withdrawn at 226 Steps				Fully Withdrawn at 227 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
226 Stop	110	0	0	227 Stop	111	0	0
226	116	0 Start	0	227	116	0 Start	0
226	226 Stop	110	0	227	227 Stop	111	0
226	226	116	0 Start	227	227	116	0 Start
226	226	226 Stop	110	227	227	227 Stop	111

Fully Withdrawn at 228 Steps				Fully Withdrawn at 229 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
228 Stop	112	0	0	229 Stop	113	0	0
228	116	0 Start	0	229	116	0 Start	0
228	228 Stop	112	0	229	229 Stop	113	0
228	228	116	0 Start	229	229	116	0 Start
228	228	228 Stop	112	229	229	229 Stop	113

Fully Withdrawn at 230 Steps				Fully Withdrawn at 231 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
230 Stop	114	0	0	231 Stop	115	0	0
230	116	0 Start	0	231	116	0 Start	0
230	230 Stop	114	0	231	231 Stop	115	0
230	230	116	0 Start	231	231	116	0 Start
230	230	230 Stop	114	231	231	231 Stop	115

Catawba 2 Cycle 23 Core Operating Limits Report

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:

$$\begin{aligned} F_Q^{RTP} * K(Z)/P & \quad \text{for } P > 0.5 \\ F_Q^{RTP} * K(Z)/0.5 & \quad \text{for } P \leq 0.5 \end{aligned}$$

where,

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

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 $F_Q^{RTP} = 2.70 \times K(\text{BU})$

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 $K(\text{BU})$ is the normalized $F_Q(X,Y,Z)$ as a function of burnup. F_Q^{RTP} with the $K(\text{BU})$ penalty for Westinghouse RFA fuel is analytically confirmed in cycle-specific reload calculation. $K(\text{BU})$ is set to 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 $[F_Q^L(X,Y,Z)]^{OP} = \frac{F_Q^D(X,Y,Z) * M_Q(X,Y,Z)}{UMT * MT * TILT}$

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 calculation 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.

Catawba 2 Cycle 23 Core Operating Limits Report

$M_Q(X,Y,Z)$ = 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)

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

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 calculation and measurement uncertainties.

$F_Q^D(X,Y,Z)$ = Defined in Section 2.6.5.

$M_C(X,Y,Z)$ = 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 = Defined in Section 2.6.5.

MT = Defined in Section 2.6.5.

TILT = Defined in Section 2.6.5.

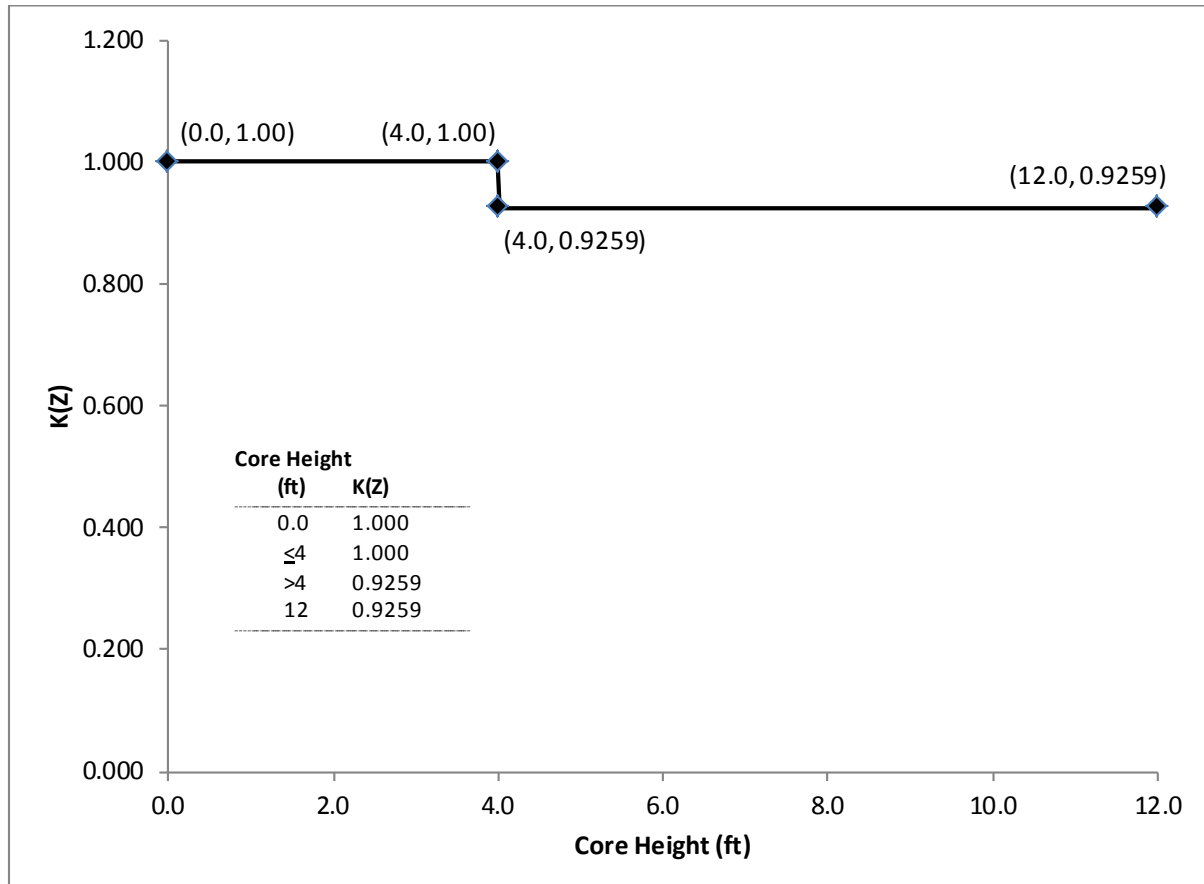
Catawba 2 Cycle 23 Core Operating Limits Report

2.6.7 KSLOPE = 0.0725

where:

KSLOPE = adjustment to K_1 value from OTΔT trip setpoint required to compensate for each 1% measured $F_Q^M(X,Y,Z)$ 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 23 Core Operating Limits Report**Figure 4** **$K(Z)$, Normalized $F_Q(X,Y,Z)$ as a Function of Core Height
for RFA Fuel**

Catawba 2 Cycle 23 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
492	2.00	2.00
497	2.00	2.00
507	2.00	2.00
517	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.

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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.

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

where:

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

$\text{MARP}(X,Y)$ = Cycle-specific operating limit Maximum Allowable Radial Peaks. $\text{MARP}(X,Y)$ 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, $F_{\Delta H}^M(X,Y)$, exceeds the limit.

$$(\text{RRH} = 3.34, \quad 0.0 < P \leq 1.0)$$

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

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

where:

$[F_{\Delta H}^L(X,Y)]^{SURV}$ = 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. $F_{\Delta H}^L(X,Y)^{SURV}$ includes allowances for calculation and measurement uncertainty.

$F_{\Delta H}^D(X,Y)$ = Design power distribution for $F_{\Delta H}$. $F_{\Delta H}^D(X,Y)$ 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 the 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 = Defined in Section 2.6.5.

2.7.3 RRH is defined in Section 2.7.1.

2.7.4 TRH = 0.04

where:

TRH = Reduction in OTAT K_1 setpoint required to compensate for each 1% measured radial peak, $F_{\Delta H}^M(X,Y)$ 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.

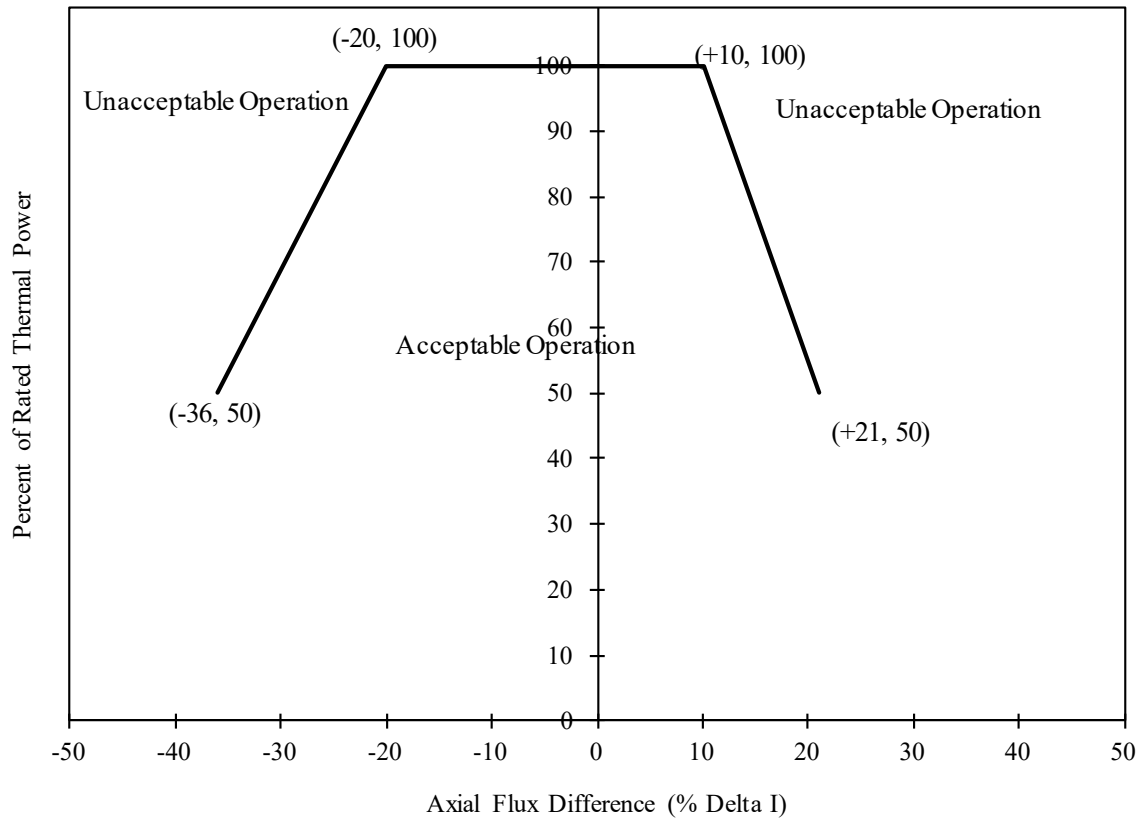
2.8 Axial Flux Difference – AFD (TS 3.2.3)

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 Steady State Limiting Value Between
Loss of Flow Accident (LOFA) MARPs and $F\Delta H_{LOCA}$

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	3.25
0.12	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.3151	1.2461
1.20	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.3007	1.2235
2.40	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.4633	1.4616
3.60	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.4675	1.3874
4.80	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.2987	1.2579
6.00	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.3293	1.2602
7.20	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.5982	1.2871	1.2195
8.40	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6010	1.5127	1.2182	1.1578
9.60	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.5808	1.5301	1.4444	1.1431	1.0914
10.80	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.5743	1.5573	1.5088	1.4624	1.3832	1.1009	1.0470
11.40	1.6058	1.6058	1.6058	1.6058	1.6057	1.5826	1.5289	1.5098	1.4637	1.4218	1.3458	1.0670	1.0142

Catawba 2 Cycle 23 Core Operating Limits Report**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

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

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

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2.9.2 Overpower ΔT Setpoint Parameter Values

<u>Parameter</u>	<u>Nominal Value</u>
Nominal T_{avg} at RTP	$T'' \leq 590.8 \text{ }^\circ\text{F}$
Overpower ΔT reactor trip setpoint	$K_4 = 1.0819$
Overpower ΔT reactor trip penalty	$K_5 = 0.02 \text{ } / \text{ }^\circ\text{F}$ for increasing T_{avg} $K_5 = 0.00 \text{ } / \text{ }^\circ\text{F}$ for decreasing T_{avg}
Overpower ΔT reactor trip heatup setpoint penalty coefficient	$K_6 = 0.001291 / \text{ }^\circ\text{F}$ for $T > T''$ $K_6 = 0.0 \text{ } / \text{ }^\circ\text{F}$ for $T \leq T''$
Time constants utilized in the lead-lag compensator for ΔT	$\tau_1 = 8 \text{ sec.}$ $\tau_2 = 3 \text{ sec.}$
Time constant utilized in the lag compensator for ΔT	$\tau_3 = 0 \text{ sec.}$
Time constant utilized in the measured T_{avg} lag compensator	$\tau_6 = 0 \text{ sec.}$
Time constant utilized in the rate-lag controller for T_{avg}	$\tau_7 = 10 \text{ sec.}$
$f_2(\Delta I)$ "positive" breakpoint	$= 35.0 \text{ } \%\Delta I$
$f_2(\Delta I)$ "negative" breakpoint	$= -35.0 \text{ } \%\Delta I$
$f_2(\Delta I)$ "positive" slope	$= 7.0 \text{ } \%\Delta T_0 / \%\Delta I$
$f_2(\Delta I)$ "negative" slope	$= 7.0 \text{ } \%\Delta T_0 / \%\Delta I$

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2.10 Boron Dilution Mitigation System (TS 3.3.9)

2.10.1 Reactor Makeup Water Pump flow rate limits:

<u>Applicable Mode</u>	<u>Limit</u>
MODE 3	≤ 80 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:

<u>Parameter</u>	<u>Applicable Burnup</u>	<u>Limit</u>
Accumulator minimum boron concentration.	0 - 200 EFPD	2,500 ppm
Accumulator minimum boron concentration.	200.1 - 250 EFPD	2,500 ppm
Accumulator minimum boron concentration.	250.1 - 300 EFPD	2,447 ppm
Accumulator minimum boron concentration.	300.1 - 350 EFPD	2,336 ppm
Accumulator minimum boron concentration.	350.1 - 400 EFPD	2,256 ppm
Accumulator minimum boron concentration.	400.1 - 450 EFPD	2,183 ppm
Accumulator minimum boron concentration.	450.1 - 475 EFPD	2,114 ppm
Accumulator minimum boron concentration.	475.1 - 507 EFPD	2,079 ppm
Accumulator minimum boron concentration.	507.1 - 517 EFPD	2,031 ppm
Accumulator maximum boron concentration.	0 - 517 EFPD	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	≥ 2209.8 psig
	meter	3	≥ 2212.1 psig
	computer	4	≥ 2205.8 psig
	computer	3	≥ 2207.5 psig
3. RCS Total Flow Rate			$\geq 390,000$ gpm

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2.13 Refueling Water Storage Tank - RWST (TS 3.5.4)

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

<u>Parameter</u>	<u>Limit</u>
RWST minimum boron concentration.	2,700 ppm
RWST maximum boron concentration.	3,075 ppm

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.

<u>Parameter</u>	<u>Limit</u>
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 K_{eff} remains within MODE 6 reactivity requirement of $K_{eff} \leq 0.95$.

<u>Parameter</u>	<u>Limit</u>
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 - (SLC-16.7-9)

2.16.1 Minimum boron concentration limit for the spent fuel pool required for Standby Makeup Pump Water Supply. Applicable for MODES 1, 2, and 3.

<u>Parameter</u>	<u>Limit</u>
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 $\leq 210^{\circ}\text{F}$, and MODES 5 and 6.

<u>Parameter</u>	<u>Limit</u>
BAT minimum boron concentration	7,000 ppm
Volume of 7,000 ppm boric acid solution required to maintain SDM at 68 °F	2000 gallons

NOTE: When cycle burnup is > 446 EFPD, Figure 6 may be used to determine required BAT minimum level.

BAT Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-11)	13,086 gallons (14.9% level)
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% level)

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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 °F *.

*** NOTE: The SLC 16.9-12 applicability is down to MODE 4 temperatures of > 210°F. The minimum volumes calculated support cooldown to 200°F to satisfy UFSAR Chapter 9 requirements.**

<u>Parameter</u>	<u>Limit</u>
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

NOTE: When cycle burnup is > 446 EFPD, Figure 6 may be used to determine required BAT minimum level.

BAT Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9-12)	25,200 gallons (45.8% level)
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% level)

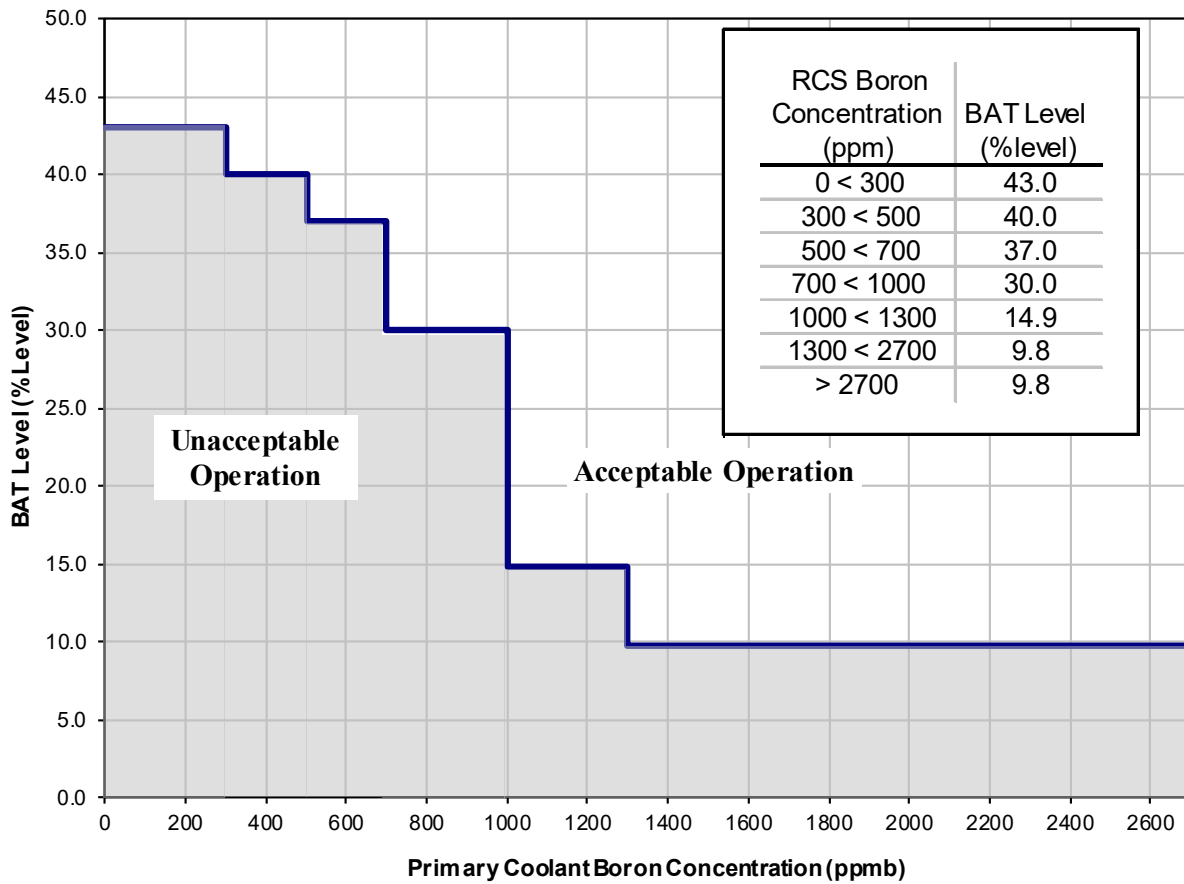
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Figure 6

**Boric Acid Storage Tank Indicated Level Versus
Primary Coolant Boron Concentration**

(Valid When Cycle Burnup is > 446 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 23 Maneuvering Analysis calculation file, CNC-1553.05-00-0660. 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 there is a need to access this information.

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