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RA-24-0082

March 21, 2024

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 27 Reload Core

Pursuant to Catawba Technical Specification 5.6.5.d, enclosed is the subject COLR. This COLR revision is being submitted to update the limits of the Catawba Unit 2 Cycle 27 reload core.

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

Please direct any questions or concerns to Ari Tuckman, Regulatory Affairs, at (803) 701-3771.

Sincerely,

Nicole Flippin

Nicole Flippin
Vice President, Catawba Nuclear Station

Enclosure: Catawba Unit 2, Cycle 27, Revision 0, Core Operating Limits Report xc (with enclosure; with attachment):

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Enclosure

Catawba Unit 2, Cycle 27, Revision 0, Core Operating Limits Report

Catawba 2 Cycle 27

Core Operating Limits Report
Revision 0

February 2024

Reference: CNC-1553.05-00-0749, Rev. 0
Reload 50.59 #: 02504941

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 27 Core Operating Limits Report

Implementation Instructions for Revision 0

Revision Description and CR Tracking

Revision 0 of the Catawba 2 Cycle 27 COLR contains limits specific to the reload core. There is no CR associated with this revision.

Implementation Schedule

The Catawba 2 Cycle 27 COLR requires the reload 50.59 (AR #02504941) be approved prior to implementation and fuel loading.

Revision 0 may become effective any time during NO MODE between cycles 26 and 27 but must become effective prior to entering MODE 6 which starts cycle 27. The Catawba 2 Cycle 27 COLR will cease to be effective during NO MODE between cycles 27 and 28.

Data files to be Implemented

No data files are transmitted as part of this document.

Additional Information

A review was performed by Safety Analysis for COLR Sections 1.1, 2.1, 2.9 – 2.11, 2.13, and 2.16 – 2.18.

CNS Reactor Engineering performed a site impact review in accordance with AD-NF-ALL-0807 and AD-NF-NGO-0214.

Catawba 2 Cycle 27 Core Operating Limits Report

REVISION LOG

Revision	Effective Date	Pages Affected	COLR
0	February 2024	1-31, Appendix A*	C2C27 COLR, Rev. 0

* Appendix A contains power distribution monitoring factors used in Technical Specification Surveillance and is not uploaded as part of the EI body. However, Appendix A (.pdf) is archived electronically for ease of transmittal to the NRC, upon request.

Catawba 2 Cycle 27 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 Licensee 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 27 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

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

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

Catawba 2 Cycle 27 Core Operating Limits Report

1.1 Analytical Methods (continued)

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

   Revision 5a
   Report Date: October 2012

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

   Revision 1
   Report Date: March 2015

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

   Revision 4c
   Report Date: January 2019

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

   Revision 2a
   Report Date: December 2008

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

    Revision 6
    Report Date: September 2020

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

    Revision 0
    Report Date: April 1995
    Not Used

12. DPC-NE-2009-P-A, "Westinghouse Fuel Transition Report," (Duke Energy 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

Catawba 2 Cycle 27 Core Operating Limits Report

Analytical Methods (continued)

14. DPC-NF-2010-A, "Nuclear Physics Methodology for Reload Design."

    Revision 2a
    Report Date: December 2009

15. DPC-NE-2011-PA, "Nuclear Design Methodology for Core Operating Limits of Westinghouse Reactors," (Duke Energy Proprietary).

    Revision 1a
    Report Date: June 2009

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

    Revision 1
    Report Date: November 2008

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

    Revision 1
    SER Date: January 14, 2004
    Not Used

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

    Revision 1
    Report Date: December 2022

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 27 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 ≥ 1.3% ΔK/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 ≥ 1.0% ΔK/K in MODE 5.

2.2.3 For TS 3.1.4, SDM shall be ≥ 1.3% ΔK/K in MODE 1 and MODE 2.

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

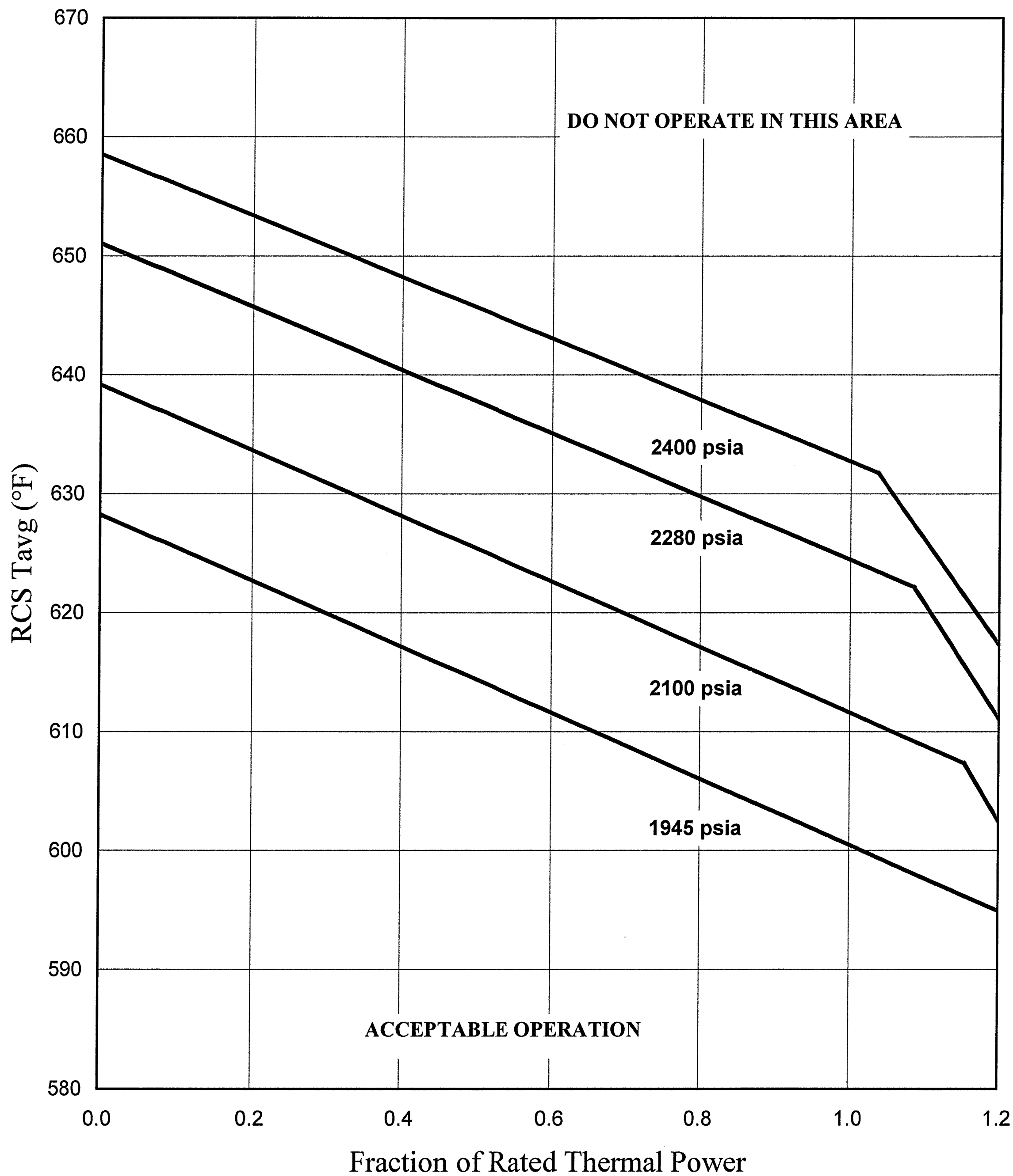
2.2.5 For TS 3.1.6, SDM shall be ≥ 1.3% ΔK/K in MODE 1 and MODE 2 with $K_{eff} \geq 1.0$.

2.2.6 For TS 3.1.8, SDM shall be ≥ 1.3% ΔK/K in MODE 2 during PHYSICS TESTS.

Catawba 2 Cycle 27 Core Operating Limits Report

Figure 1

Reactor Core Safety Limits
Four Loops in Operation



Catawba 2 Cycle 27 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 The Revised Predicted near-EOC 300 ppm 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 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. Shutdown banks are withdrawn in sequence and with no overlap.

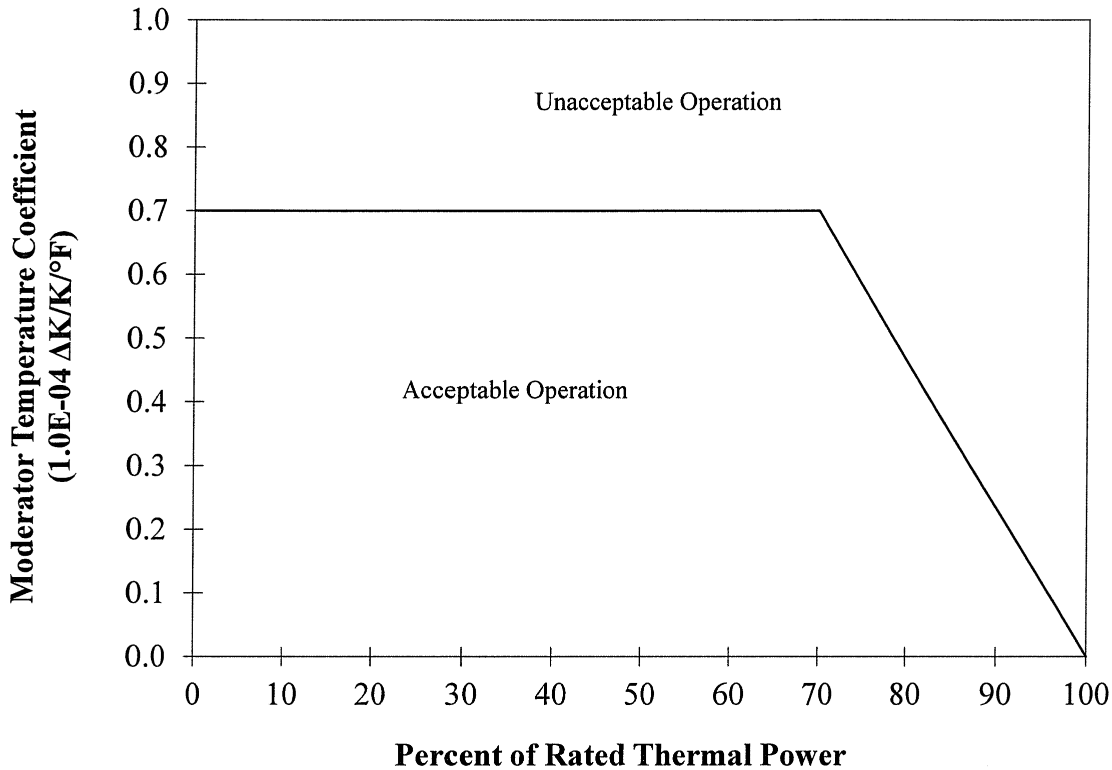
2.5 Control Bank Insertion Limits (TS 3.1.6)

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 27 Core Operating Limits Report

Figure 2

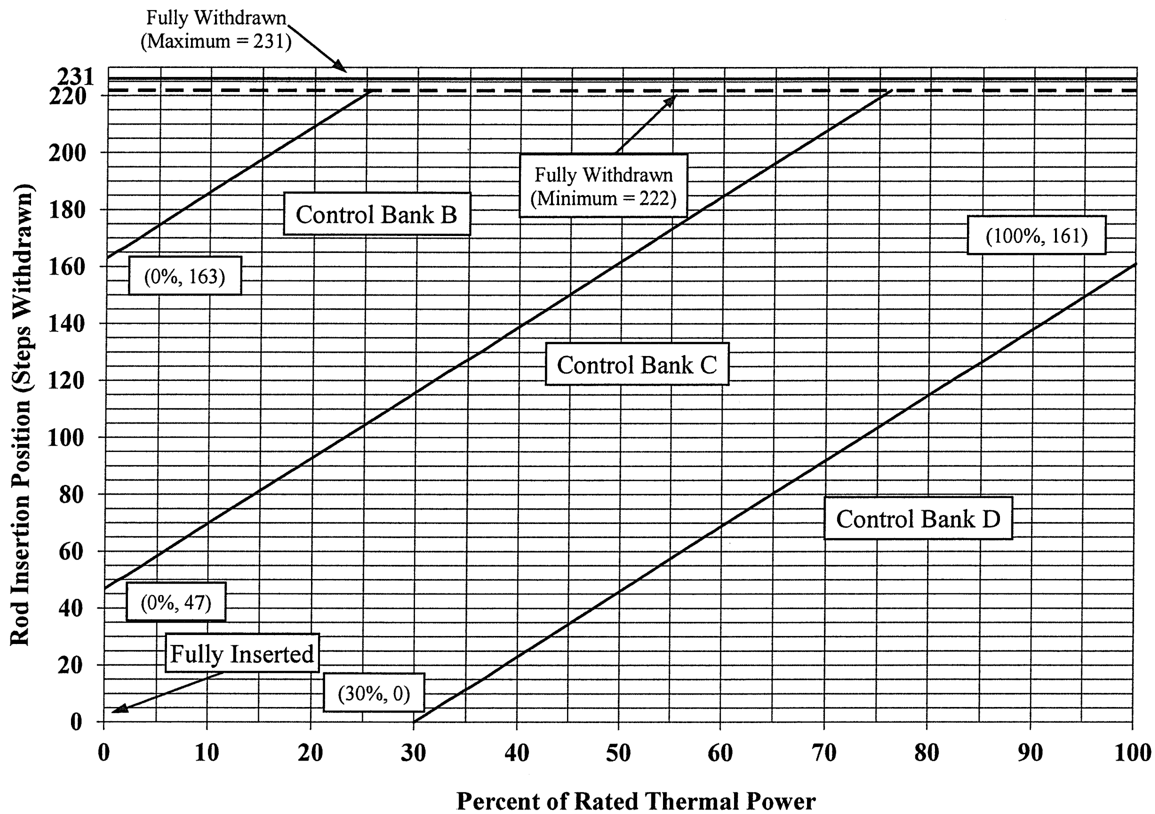
Moderator Temperature Coefficient Upper Limit Versus Power Level



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 27 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 $= 2.3(P) - 69 \quad \{30 \leq P \leq 100\}$
Bank CC RIL $= 2.3(P) + 47 \quad \{0 \leq P \leq 76.1\}$ *for CC RIL* $= 222\ \{76.1 < P \leq 100\}$
Bank CB RIL $= 2.3(P) + 163 \quad \{0 \leq P \leq 25.7\}$ *for CB RIL* $= 222\ \{25.7 < P \leq 100\}$

where P = % 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 27 Core Operating Limits Report

Table 1

Control Bank Withdrawal Sequence Equation

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
CBA Stop	CBA - 116	0	0
CBA	116	0 Start	0
CBA	CBB Stop	CBB - 116	0
CBA	CBB	116	0 Start
CBA	CBB	CBC	CBC - 116

Where:
CBA = Fully withdrawn position of Control Bank A
CBB = Fully withdrawn position of Control Bank B
CBC = Fully withdrawn position of Control Bank C

Allowed Control Bank Fully Withdrawn Positions Range are provided in CNEI-0400-091, "RCCA Axial Repositioning Schedule for Catawba Nuclear Station."

Catawba 2 Cycle 27 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:

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

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

where,

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

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

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. F_Q^{RTP} with the K(BU) penalty for Westinghouse RFA fuel is analytically confirmed in cycle-specific reload calculation. K(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 27 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 $$[F_Q^L(X,Y,Z)]^{RPS} = \frac{F_Q^D(X,Y,Z) * M_C(X,Y,Z)}{UMT * MT * TILT}$$

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 27 Core Operating Limits Report

2.6.7 KSLOPE = 0.0725

where:

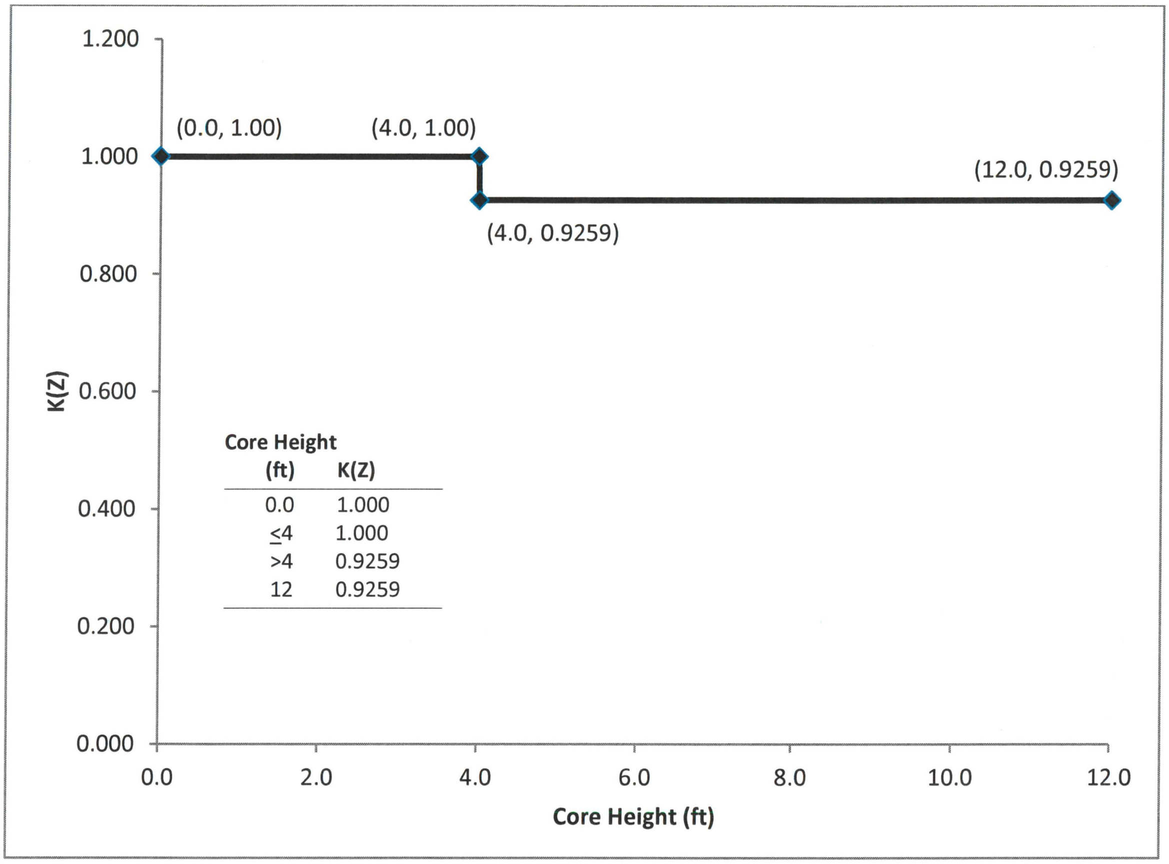
KSLOPE = adjustment to K_1 value from OTΔT trip setpoint required to compensate for each 1% measured F_Q^M (X,Y,Z) exceeds [F_Q^L (X,Y,Z)]^{RPS}.

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 27 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 27 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	2.85	2.00
125	2.68	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
480	2.00	2.00
493	2.00	2.00
496	2.00	2.00
506	2.00	2.00
516	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 27 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 $$[F^{L}_{\Delta H}(X,Y)]^{LCO} = \text{MARP}(X,Y) * \left[1.0 + \frac{1}{RRH} * (1.0 - P)\right]$$

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 calculation 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.

Catawba 2 Cycle 27 Core Operating Limits Report

$M_{\Delta H}(X,Y)$ = 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 OTΔT K_1 setpoint required to compensate for each 1% that the 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.

Catawba 2 Cycle 27 Core Operating Limits Report

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

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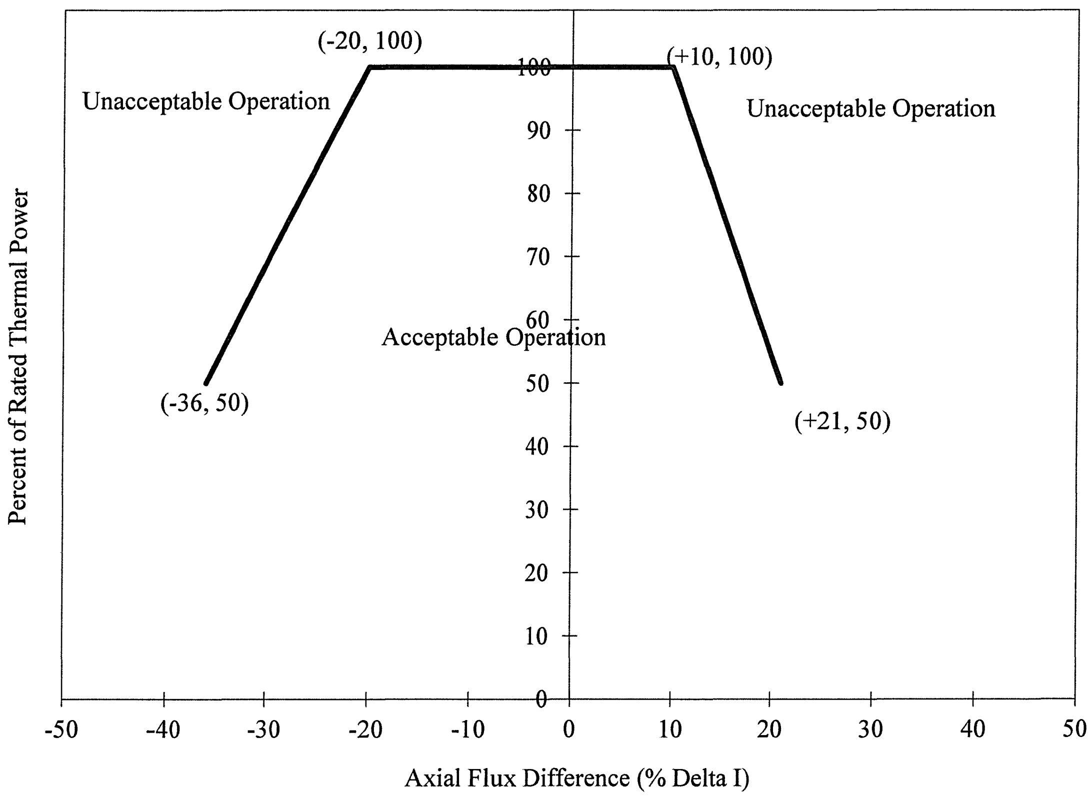
Figure 5

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	$\tau_3 \leq 1.8$ 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	$\tau_6 \leq 1.8$ 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 (ΔI) negative breakpoint and slope for OTΔT are less restrictive than OPΔT f_2 (ΔI) negative breakpoint and slope. Therefore, during a transient which challenges negative imbalance limits, OPΔT f_2 (ΔI) limits will result in a reactor trip before OTΔT f_1 (ΔI) limits are reached. This makes implementation of an OTΔT f_1 (ΔI) negative breakpoint and slope unnecessary.

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2.9.2 Overpower ΔT Setpoint Parameter Values

Parameter	Nominal Value
Nominal Tavg at RTP	$T'' \leq 590.8$ °F
Overpower ΔT reactor trip setpoint	$K_4 \leq 1.0909$
Overpower ΔT reactor trip penalty	$K_5 \geq 0.0$ / °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 $T \leq T''$
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	$\tau_3 \leq 1.8$ sec.
Time constant utilized in the measured T_{avg} lag compensator	$\tau_6 \leq 1.8$ sec.
Time constant utilized in the rate-lag controller for T_{avg}	$\tau_7 = 10$ sec.
$f_2(\Delta I)$ "positive" breakpoint	= 27.0 %ΔI
$f_2(\Delta I)$ "negative" breakpoint	= -27.0 %ΔI
$f_2(\Delta I)$ "positive" slope	= 7.0 $\%\Delta T_0 / \%\Delta I$
$f_2(\Delta I)$ "negative" slope	= 7.0 $\%\Delta T_0 / \%\Delta I$

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2.10 Boron Dilution Mitigation System (TS 3.3.9)

2.10.1 Reactor Makeup Water Pump flow rate limits:

Applicable Mode	Limit
MODE 3	$\leq$ 80 gpm
MODE 4 or 5	$\leq$ 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	Applicable Burnup	Limit
Accumulator **minimum** boron concentration.	0 - 200 EFPD	2,500 ppm
Accumulator **minimum** boron concentration.	200.1 - 300 EFPD	2,465 ppm
Accumulator **minimum** boron concentration.	300.1 - 400 EFPD	2,296 ppm
Accumulator **minimum** boron concentration.	400.1 - 506 EFPD	2,155 ppm
Accumulator **minimum** boron concentration.	506.1 - 516 EFPD	2,018 ppm
Accumulator **maximum** boron concentration.	0 - 516 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	≤ 591.3 °F
	meter	3	≤ 591.1 °F
	computer	4	≤ 591.8 °F
	computer	3	≤ 591.6 °F
2. Indicated Pressurizer Pressure	meter	4	≥ 2206.9 psig
	meter	3	≥ 2208.7 psig
	computer	4	≥ 2204.0 psig
	computer	3	≥ 2205.4 psig
3. RCS Total Flow Rate			≥ 387,000 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:

Parameter	Limit
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.

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 K_{eff} remains within MODE 6 reactivity requirement of $K_{eff} \leq 0.95$.

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 - (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.

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

NOTE: When cycle burnup is > 448 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.**

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

> **NOTE: When cycle burnup is > 448 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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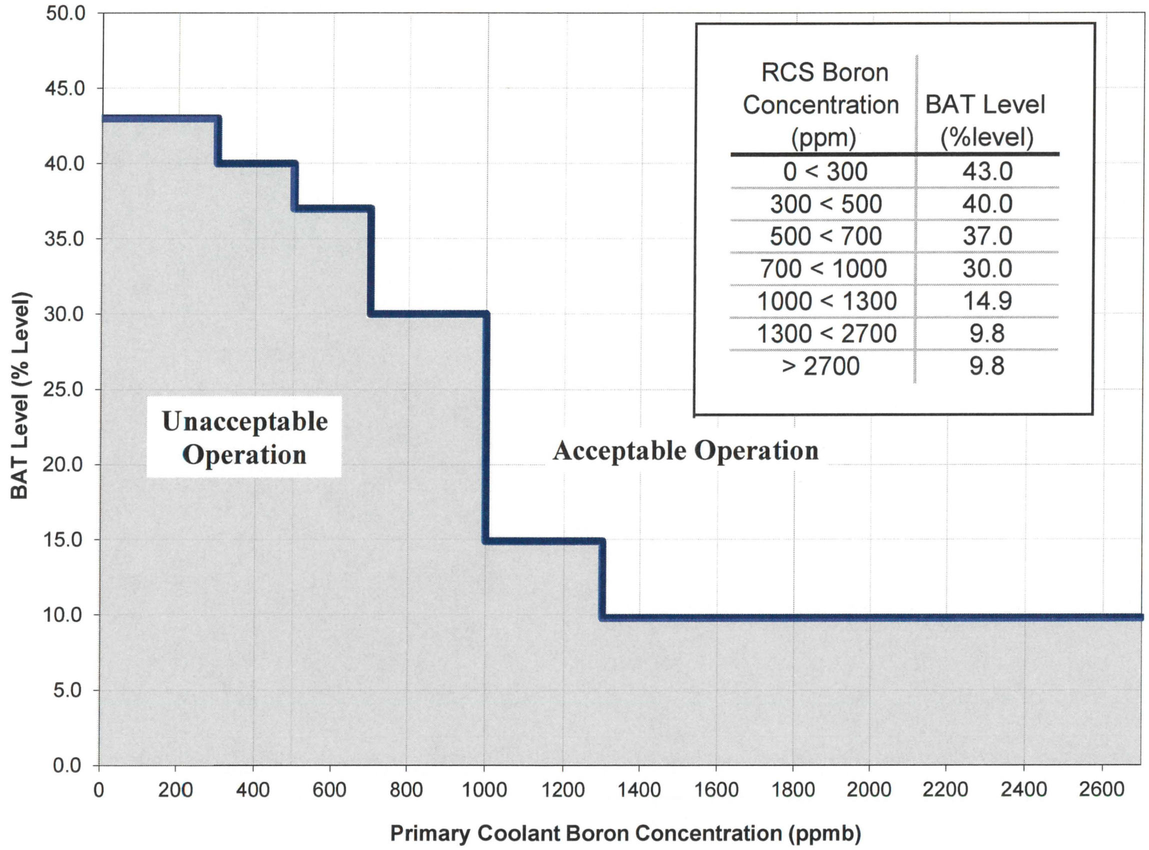
Figure 6

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

(Valid When Cycle Burnup is > 448 EFPD)

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



RCS Boron Concentration (ppm)	BAT Level (%level)
0 < 300	43.0
300 < 500	40.0
500 < 700	37.0
700 < 1000	30.0
1000 < 1300	14.9
1300 < 2700	9.8
> 2700	9.8

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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 27 Maneuvering Analysis calculation file, CNC-1553.05-00-0745, Rev 0. Due to the size of the monitoring factor data, Appendix A is controlled electronically within Duke Energy and therefore is not included in the Duke Energy internal copies of the COLR EI. Nuclear Fuels Engineering will control this information via computer file(s) and should be contacted if there is a need to access this information.

Appendix A is available to be transmitted to the NRC upon request.