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June 2, 2005

U.S. Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, D.C. 20555-0001

Subject: Duke Energy Corporation
Catawba Nuclear Station Unit 1
Docket No.: 50-413
Core Operating Limits Report (COLR)
Catawba Unit 1 Cycle 16, Revision 28

Attached, pursuant to Catawba Technical Specification 5.6.5, is an information copy of the Core Operating Limits Report for Catawba Unit 1 Cycle 16. The Unit 1 COLR is being revised to update the limits of the Catawba 1 Cycle 16 reload core.

An electronic copy of the COLR Appendix A is included with the letter to the NRC Document Control Desk on an enclosed computer disk. The COLR Appendix A contains 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 George Strickland at (803) 831-3585.

Sincerely,

D. M. Jamil

A001

Attachments

xc w/att: W. D. Travers, Regional, Administrator
USNRC, Region II

S. E. Peters, NRR Project Manager (CNS)
USNRC, ONRR

E. F. Guthrie
Senior Resident Inspector (CNS)

Catawba Unit 1 Cycle 16

Core Operating Limits Report
Revision 28

April 2005

Duke Power Company

		Date
Prepared By:	Nicholas R Hagar	4/26/05
Checked By:	Geff A Rw	4/26/2005
Checked By:	R J Al-Hfat	4/26/2005
Approved By:	Scott B. Th	4/26/2005

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: Scott B. Thomas (Sponsor) Date: 4/26/05

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 1 Cycle 16 Core Operating Limits Report

Implementation Instructions for Revision 28

Revision Description and PIP Tracking

Revision 28 of the Catawba Unit 2 COLR contains limits specific to the Catawba 1 Cycle 16 reload core. There is no PIP associated with this revision.

Implementation Schedule

Revision 28 may become effective any time during No MODE between Cycles 15 and 16 but must become effective prior to entering MODE 6 which starts Cycle 16. This revision replaces the current revisions (CNEI-0400-24, Rev. 27).

Data files to be Implemented

No data files are transmitted as part of this document.

Insertion/Deletion Instructions

Remove	Insert
pages 1- 34, of rev 27 (including Appendix A*)	pages 1- 35 of rev 28 (including Appendix A*)

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

Catawba 1 Cycle 16 Core Operating Limits Report

REVISION LOG

Revision	EI Date	Pages Affected	COLR
0 – 1	Superceded	N/A	C1C07
2 – 5	Superceded	N/A	C1C08
6 – 8	Superceded	N/A	C1C09
9 – 11	Superceded	N/A	C1C10
12 - 14	Superceded	N/A	C1C11
15 – 17	Superceded	N/A	C1C12
18 - 21	Superceded	N/A	C1C13
22 - 23	Superceded	N/A	C1C14
24	November 2003	All	C1C15 (Orig. Issue)
25	March 2004	1-34	C1C15 (Revision 1)
26	September 2004	1-34	C1C15 (Revision 2)
27	March 2005	1-34	C1C15 (Revision 3)
28	April 2005	1-35	C1C16 (Orig. Issue)

Catawba 1 Cycle 16 Core Operating Limits Report

1.0 Core Operating Limits Report

This Core Operating Limits Report (COLR) has been prepared in accordance with the requirements of Technical Specification 5.6.5. The 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 11
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 24 27 17
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor	FΔH Penalty Factors	2.7 2.7	23 24
3.2.3	Axial Flux Difference	AFD	2.8	24
3.3.1	Reactor Trip System Instrumentation	OTΔT OPΔT	2.9 2.9	27 28
3.3.9	Boron Dilution Mitigation System	Reactor Makeup Water Flow Rate	2.10	29
3.4.1	RCS Pressure, Temperature and Flow limits for DNB	RCS Pressure, Temperature and Flow	2.11	29
3.5.1	Accumulators	Max and Min Boron Conc.	2.12	29
3.5.4	Refueling Water Storage Tank	Max and Min Boron Conc.	2.13	29
3.7.15	Spent Fuel Pool Boron Concentration	Min Boron Concentration	2.14	31
3.9.1	Refueling Operations - Boron Concentration	Min Boron Concentration	2.15	31
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
16.7-9.3	Standby Shutdown System	Standby Makeup Pump Water Supply	2.16	32
16.9-11	Boration Systems – Borated Water Source – Shutdown	Borated Water Volume and Conc. for BAT/RWST	2.17	32
16.9-12	Boration Systems – Borated Water Source – Operating	Borated Water Volume and Conc. for BAT/RWST	2.18	33

Catawba 1 Cycle 16 Core Operating Limits Report

1.1 Analytical Methods

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

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

   Revision 0
   Report Date: July 1985
   Not Used for C1C16

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 C1C16

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 C1C16

Catawba 1 Cycle 16 Core Operating Limits Report

1.1 Analytical Methods (continued)

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

   Revision 3
   SER Date: September 24, 2003

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

   Revision 0
   Report Date: November, 1991, republished December 2000

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

   Revision 4
   SER Date: April 6, 2001

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

   Revision 1
   SER Date: February 20, 1997

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

    Revision 3
    SER Date: September 16, 2002

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

    Revision 0
    SER Date: April 3, 1995
    Not Used for C1C16

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

    Revision 2
    SER Date: December 18, 2002

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

    Revision 1
    SER Date: April 26, 1996
    Not Used for C1C16

Catawba 1 Cycle 16 Core Operating Limits Report

1.1 Analytical Methods (continued)

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

    Revision 2
    SER Date: June 24, 2003

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

    Revision 1
    SER Date: October 1, 2002

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

    Revision 0
    SER Date: August 20, 2004

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

    Revision 1
    SER Date: January 14, 2004

Catawba 1 Cycle 16 Core Operating Limits Report

2.0 Operating Limits

The cycle-specific parameter limits for the 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)

The 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, shutdown margin shall be greater than or equal to 1.3% $\Delta K/K$ in mode 2 with Keff < 1.0 and in modes 3 and 4.

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

2.2.3 For TS 3.1.4, shutdown margin 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, shutdown margin 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.

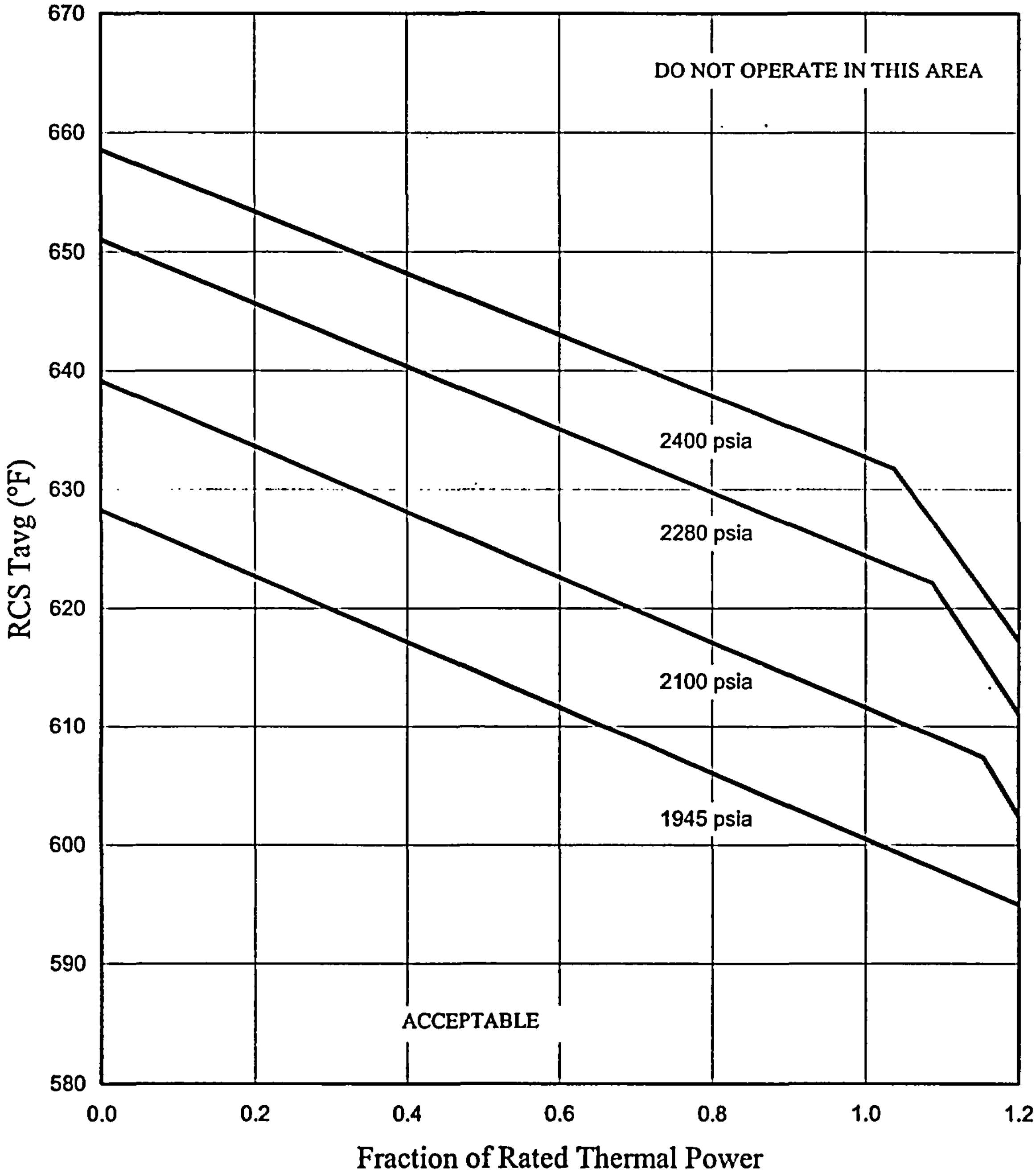
2.2.5 For TS 3.1.6, shutdown margin shall be greater than or equal to 1.3% $\Delta K/K$ in mode 1 and mode 2 with Keff ≥ 1.0.

2.2.6 For TS 3.1.8, shutdown margin shall be greater than or equal to 1.3% $\Delta K/K$ in mode 2 during Physics Testing.

Catawba 1 Cycle 16 Core Operating Limits Report

Figure 1

Reactor Core Safety Limits
Four Loops in Operation



Catawba 1 Cycle 16 Core Operating Limits Report

2.3 Moderator Temperature Coefficient - MTC (TS 3.1.3)

2.3.1 The Moderator Temperature Coefficient (MTC) Limits are:

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

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

2.3.2 The 300 ppm MTC Surveillance Limit is:

The 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 60 PPM MTC Surveillance Limit is:

The 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.

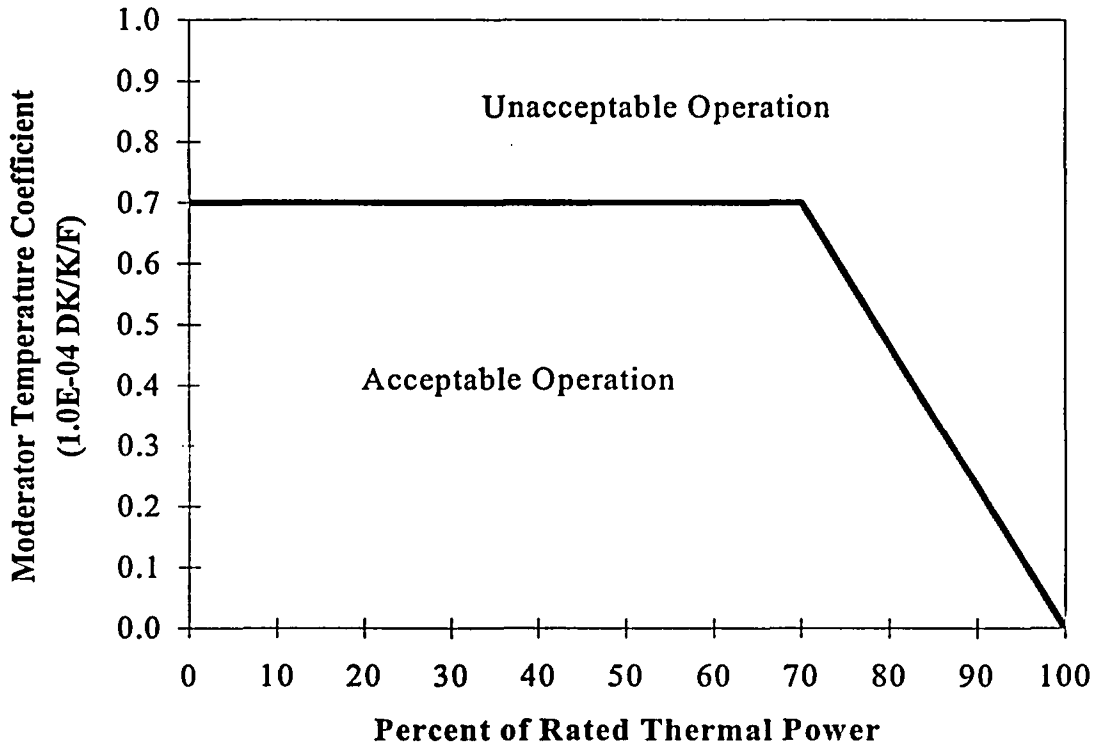
2.5 Control Bank Insertion Limits (TS 3.1.6)

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 1 Cycle 16 Core Operating Limits Report

Figure 2

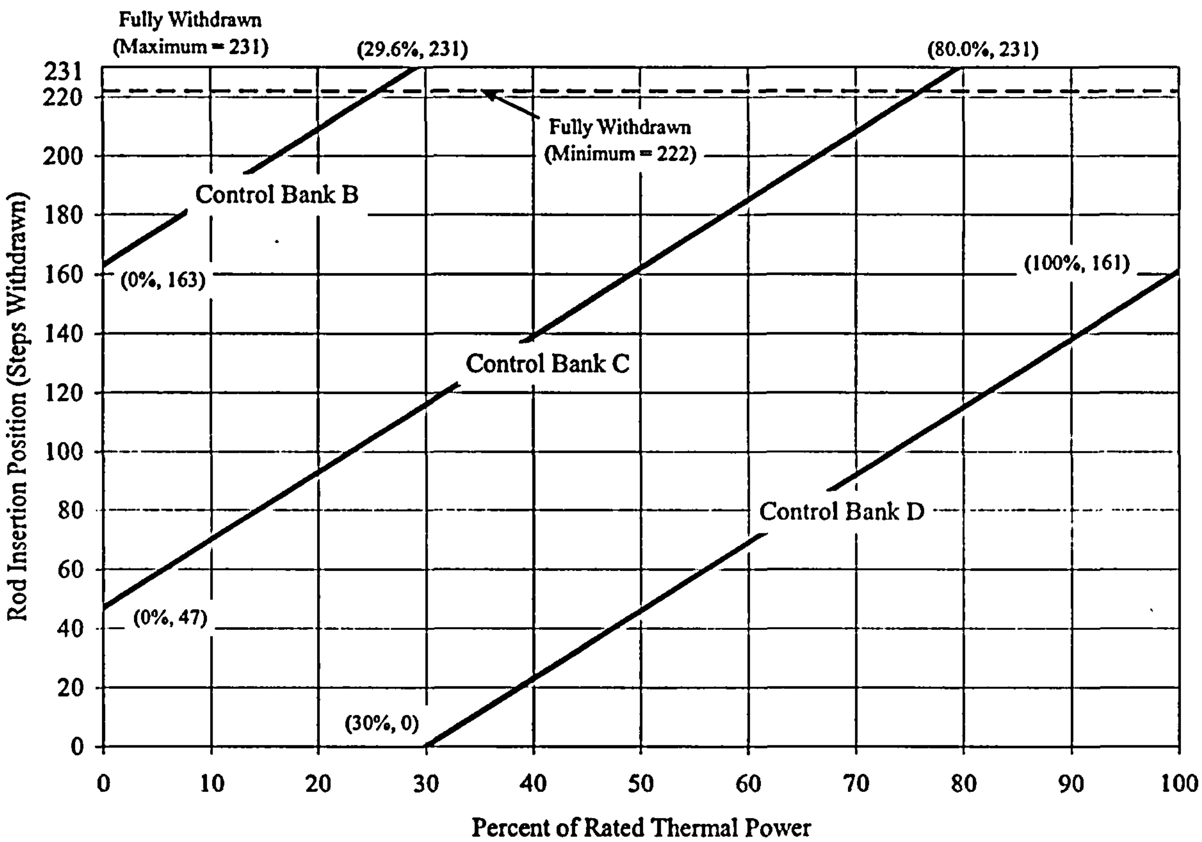
Moderator Temperature Coefficient Upper Limit Versus Power Level



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

Catawba 1 Cycle 16 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:

$$Bank\ CD\ RIL = 2.3(P) - 69\ \{30 \le P \le 100\}$$
$$Bank\ CC\ RIL = 2.3(P) + 47\ \{0 \le P \le 80\}$$
$$Bank\ CB\ RIL = 2.3(P) + 163\ \{0 \le P \le 29.6\}$$

$$where\ P = \%Rated\ Thermal\ Power$$

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

Catawba 1 Cycle 16 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

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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:

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

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

where,

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

Note: The measured $F_Q(X,Y,Z)$ shall be increased by 3.1% to account for manufacturing tolerances and 5% to account for measurement uncertainty when comparing against the LCO limit. The more conservative of the manufacturing tolerances for RFA/NGF fuel (3%) or MOX fuel (3.1%) is used when comparing against the LCO limit due to input limitations in monitoring software. The manufacturing tolerance and measurement uncertainty are implicitly included in the F_Q surveillance limits as defined below for COLR Sections 2.6.5 and 2.6.6.

2.6.2 $F_Q^{RTP} = 2.60 \times K(BU)$ for RFA and NGF fuel

$F_Q^{RTP} = 2.40 \times K(BU)$ for MOX fuel

2.6.3 K(Z) is the normalized $F_Q(X,Y,Z)$ as a function of core height. K(Z) for Westinghouse RFA and NGF fuel is provided in Figure 4, and the K(Z) for MOX fuel is provided in Figure 5.

2.6.4 K(BU) is the normalized $F_Q(X,Y,Z)$ as a function of burnup. K(BU) for Westinghouse RFA and NGF fuel is provided in Figure 6, and the K(BU) for MOX fuel is provided in Figure 7.

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 1 Cycle 16 Core Operating Limits Report

where:

$[F_Q^L(X,Y,Z)]^{OP}$ = Cycle dependent maximum allowable design peaking factor that ensures that the $F_Q(X,Y,Z)$ LOCA limit is not exceeded for operation within the 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 = 1.031). The more conservative of the manufacturing tolerances for RFA/NGF fuel (3%) or MOX fuel (3.1%) is used when comparing against the F_Q LOCA surveillance limit due to input limitations in monitoring software. The manufacturing tolerances for RFA/NGF fuel and MOX fuel is implicitly included in the FQ LOCA surveillance limits (Mq).

TILT = Peaking penalty that accounts 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 that the $F_Q(X,Y,Z)$ Centerline Fuel Melt (CFM) limit is not exceeded for operation within the AFD, RIL, and

Catawba 1 Cycle 16 Core Operating Limits Report

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.

$M_C(X,Y,Z)$ = 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.

MT = Engineering Hot Channel Factor. (MT = 1.031). The more conservative of the manufacturing tolerances for RFA/NGF fuel (3%) or MOX fuel (3.1%) is used when comparing against the F_Q RPS surveillance limit due to input limitations in monitoring software. The manufacturing tolerances for RFA/NGF fuel and MOX fuel is implicitly included in the FQ RPS surveillance limits (M_C).

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

2.6.7 KSLOPE = 0.0725

where:

KSLOPE = the adjustment to the K_1 value from OTΔT trip setpoint required to compensate for each 1% that 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.

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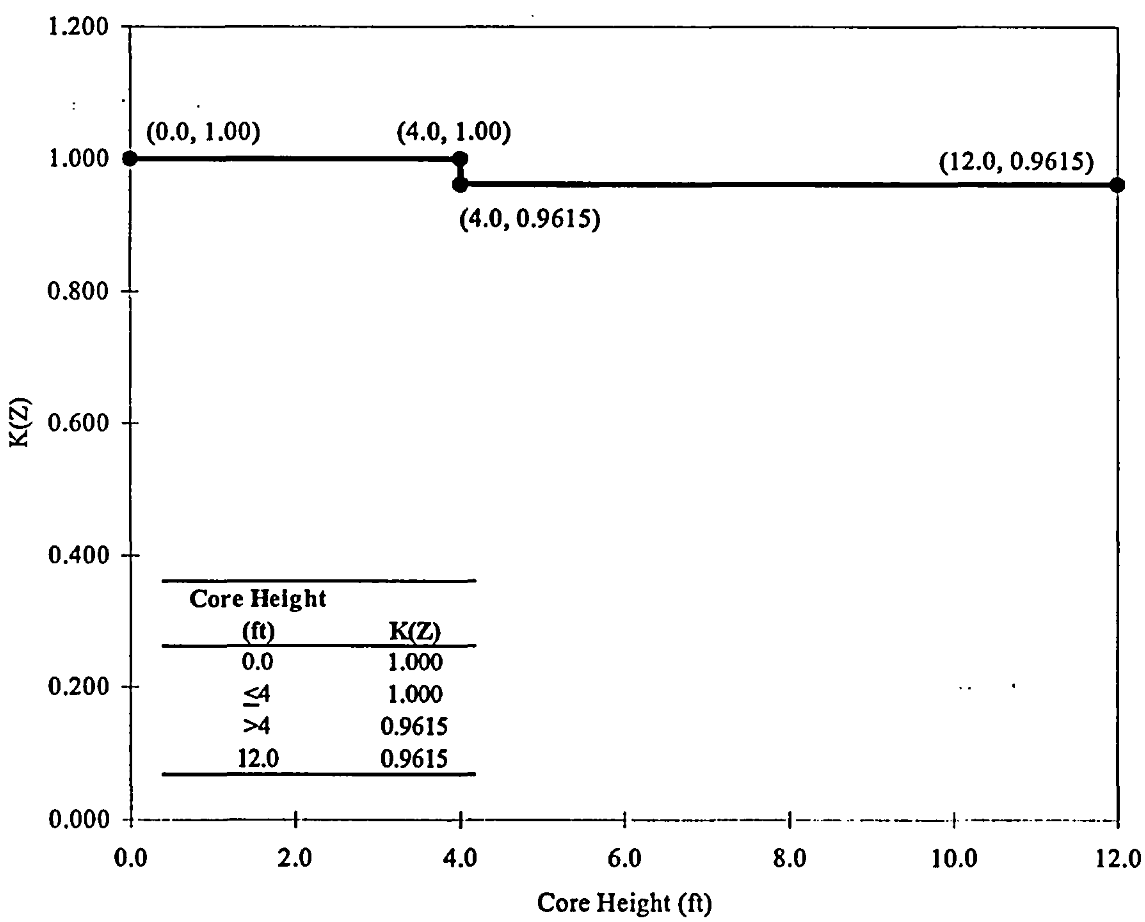
Figure 4

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



Core Height (ft)	K(Z)
0.0	1.000
≤4	1.000
>4	0.9615
12.0	0.9615

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Figure 5

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

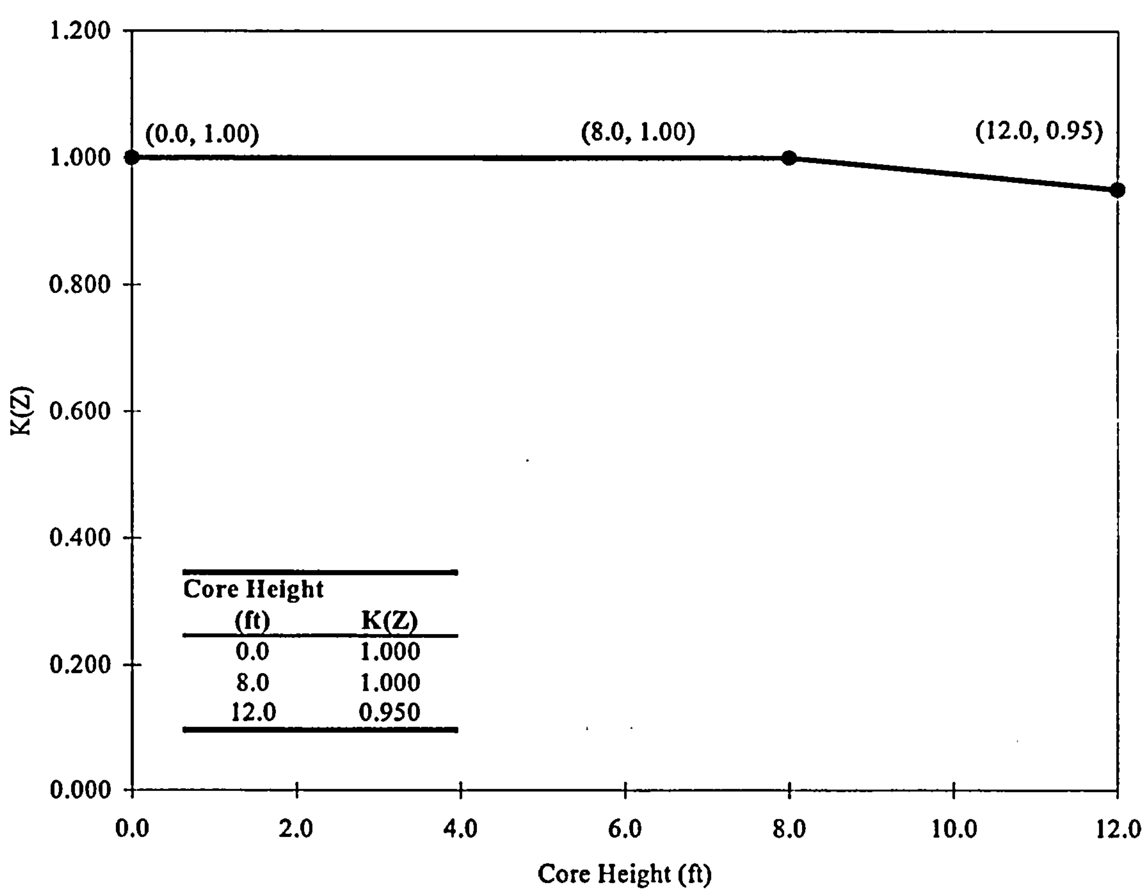

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Figure 6

K(BU), Normalized $F_Q(X,Y,Z)$ as a Function of Burnup for RFA and NGF Fuel

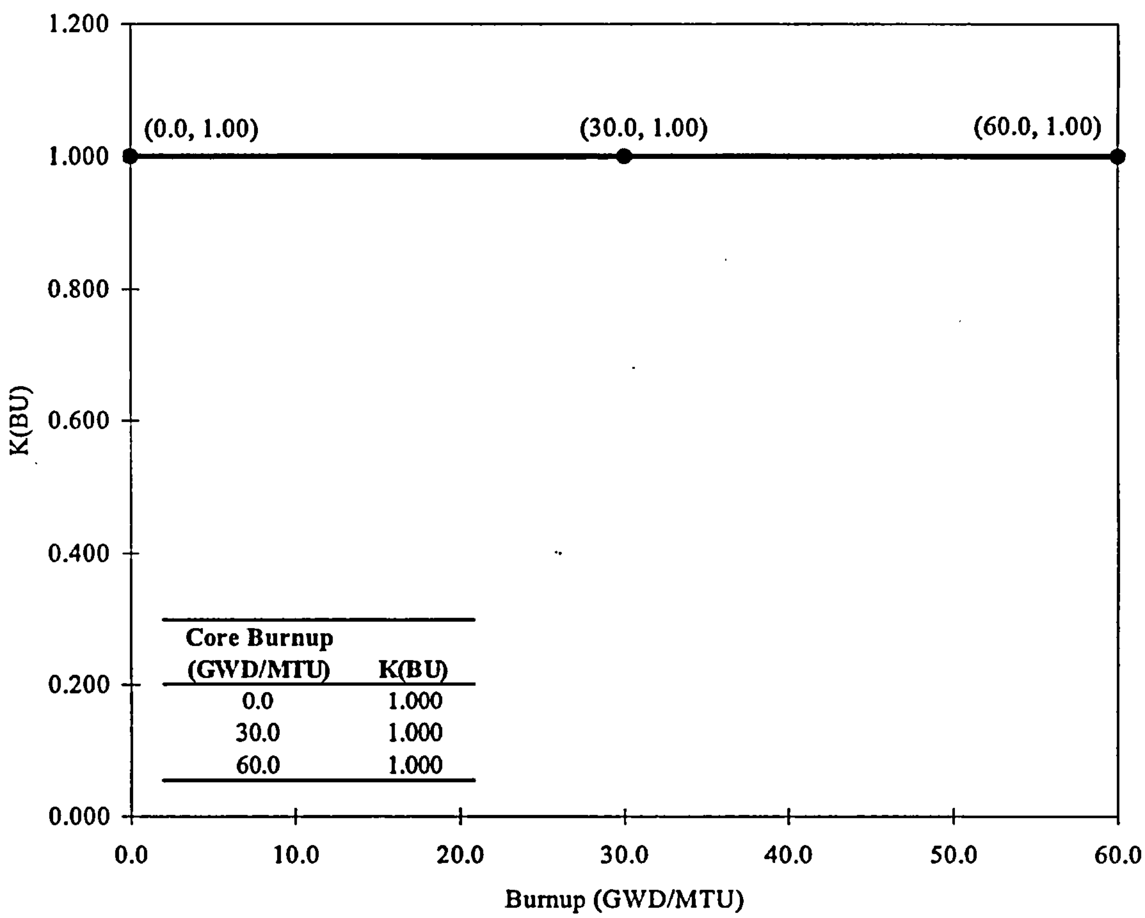

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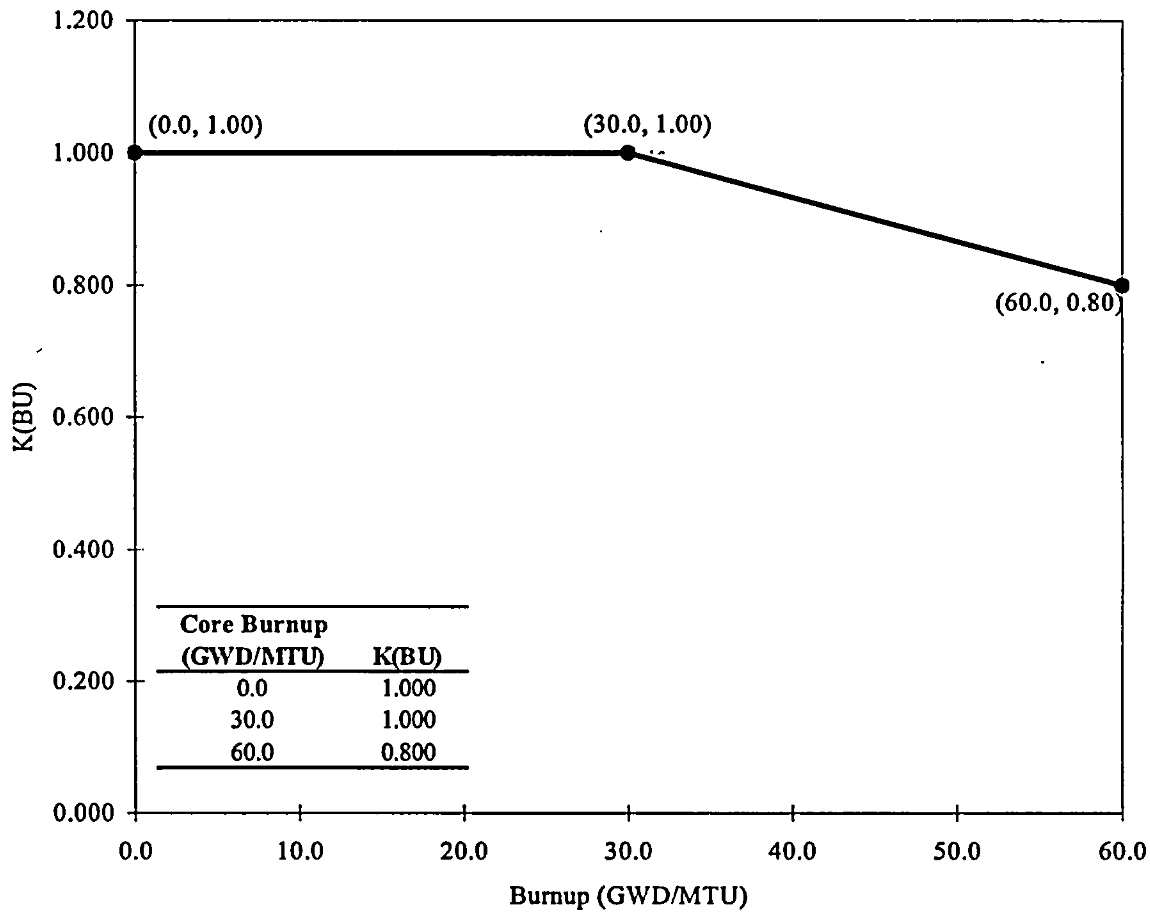
Figure 7

K(BU), Normalized $F_Q(X,Y,Z)$ as a Function of Burnup for MOX Fuel



Catawba 1 Cycle 16 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
480	2.00	2.00
505	2.00	2.00
509	2.00	2.00
524	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 the 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)

The $F_{\Delta H}$ 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 defined as 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% that the 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 the 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 that the $F_{\Delta H}(X,Y)$ limit is not exceeded for operation within the 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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$M_{\Delta H}(X,Y)$ = The 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 is set to 1.0 since a factor of 1.04 is implicitly included in the variable $M_{\Delta H}(X,Y)$.

TILT = Peaking penalty that accounts 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% that the measured radial peak, $F_{\Delta H}^{M}(X,Y)$ 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% that the measured radial peak, $F_{\Delta H}(X,Y)$ 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 The Axial Flux Difference (AFD) Limits are provided in Figure 8.

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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	1.847	1.882	1.947	1.992	1.974	2.107	2.050	2.009	1.933	1.863	1.778	1.315	1.246
1.20	1.843	1.879	1.938	1.992	1.974	2.107	2.019	1.978	1.901	1.831	1.785	1.301	1.224
2.40	1.846	1.876	1.931	1.981	1.974	2.084	1.994	1.952	1.876	1.805	1.732	1.476	1.462
3.60	1.843	1.869	1.920	1.964	1.974	2.065	1.988	1.951	1.874	1.801	1.701	1.467	1.453
4.80	1.838	1.868	1.906	1.945	1.974	2.006	1.945	1.922	1.862	1.800	1.718	1.329	1.288
6.00	1.834	1.856	1.891	1.921	1.946	1.934	1.880	1.863	1.802	1.747	1.673	1.383	1.317
7.20	1.828	1.845	1.871	1.893	1.887	1.872	1.809	1.787	1.732	1.681	1.618	1.318	1.277
8.40	1.823	1.829	1.847	1.857	1.816	1.795	1.739	1.722	1.675	1.630	1.551	1.247	1.209
9.60	1.814	1.812	1.809	1.792	1.738	1.724	1.678	1.665	1.621	1.578	1.492	1.191	1.137
10.80	1.798	1.784	1.761	1.738	1.697	1.682	1.626	1.605	1.558	1.512	1.430	1.149	1.096
11.40	1.789	1.765	1.725	1.684	1.632	1.614	1.569	1.557	1.510	1.466	1.392	1.113	1.060

MOX 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	1.746	1.777	1.834	1.884	1.925	1.951	1.860	1.803	1.719	1.643	1.568	1.256	1.195
1.20	1.746	1.774	1.826	1.872	1.912	1.931	1.849	1.806	1.724	1.648	1.546	1.207	1.145
2.40	1.743	1.769	1.816	1.854	1.889	1.906	1.827	1.790	1.719	1.653	1.539	1.171	1.117
3.60	1.740	1.763	1.802	1.836	1.860	1.849	1.797	1.777	1.719	1.653	1.559	1.182	1.107
4.80	1.736	1.755	1.787	1.816	1.802	1.794	1.741	1.724	1.675	1.623	1.574	1.204	1.151
6.00	1.732	1.747	1.770	1.792	1.751	1.737	1.688	1.672	1.624	1.578	1.526	1.287	1.248
7.20	1.727	1.738	1.752	1.741	1.696	1.683	1.634	1.618	1.574	1.531	1.477	1.244	1.206
8.40	1.723	1.725	1.731	1.690	1.641	1.627	1.580	1.567	1.525	1.484	1.429	1.207	1.167
9.60	1.712	1.709	1.685	1.638	1.589	1.574	1.529	1.520	1.481	1.442	1.386	1.170	1.129
10.80	1.701	1.687	1.640	1.590	1.540	1.524	1.482	1.472	1.435	1.401	1.344	1.147	1.109
12.00	1.694	1.670	1.619	1.569	1.518	1.505	1.462	1.453	1.417	1.386	1.334	1.135	1.096

NGF Fuel MARPs
100% Full Power

Core Height (ft)	Axial Peak						
	1.05	1.2	1.4	1.6	1.8	2.1	3.25
0.12	1.771	1.871	1.805	2.049	1.931	1.786	1.266
2.40	1.760	1.853	1.805	1.993	1.870	1.724	1.442
4.80	1.757	1.824	1.805	1.884	1.809	1.693	1.261
7.20	1.745	1.784	1.805	1.735	1.659	1.553	1.227
9.60	1.729	1.723	1.652	1.587	1.527	1.402	1.059
11.40	1.707	1.642	1.550	1.477	1.416	1.304	1.003

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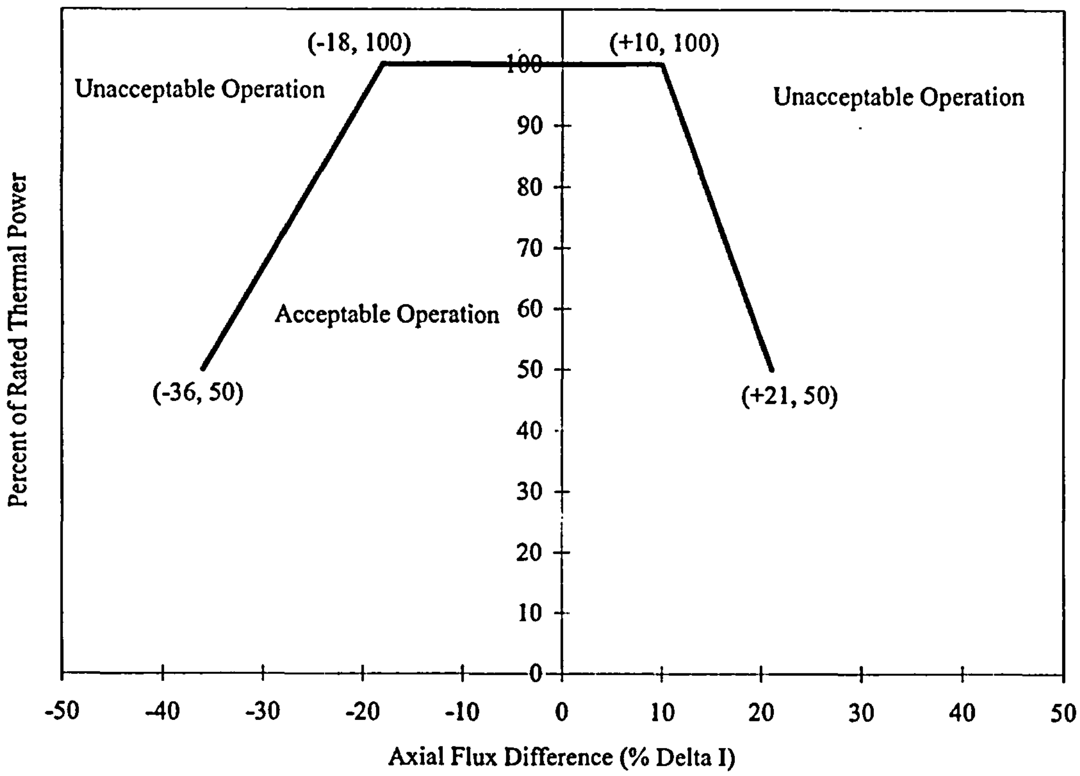
Figure 8

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 1 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	$T' \leq 585.1$ °F
Nominal RCS Operating Pressure	$P' = 2235$ psig
Overtemperature ΔT reactor trip setpoint	$K_1 = 1.1978$
Overtemperature ΔT reactor trip heatup setpoint penalty coefficient	$K_2 = 0.03340/^{o}F$
Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient	$K_3 = 0.001601$/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	= 19.0 %ΔI
$f_1(\Delta I)$ "negative" breakpoint	= N/A*
$f_1(\Delta I)$ "positive" slope	= 1.769 $\%\Delta T_0/\ \%\Delta I$
$f_1(\Delta I)$ "negative" slope	= N/A*

* The $f_1(\Delta I)$ negative breakpoints and slopes for OTΔT are less restrictive than the OPΔT $f_2(\Delta I)$ negative breakpoint and slope. Therefore, during a transient which challenges the negative imbalance limits the OPΔT $f_2(\Delta I)$ limits will result in a reactor trip before the 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	$T'' \leq 585.1$ °F
Overpower ΔT reactor trip setpoint	$K_4 = 1.0864$
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 (for T>T")	$K_6 = 0.001179$/°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	$\tau_3 = 0$ sec.
Time constant utilized in the measured T_{avg} lag compensator	$\tau_6 = 0$ sec.
Time constant utilized in the rate-lag controller for T_{avg}	$\tau_7 = 10$ 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	$\leq$ 150 gpm
Mode 4 or 5	$\leq$ 70 gpm

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

The 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
Cold Leg Accumulator minimum boron concentration.	2,500 ppm
Cold Leg Accumulator maximum boron concentration.	2,975 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
Refueling Water Storage Tank minimum boron concentration.	2,700 ppm
Refueling Water Storage Tank maximum boron concentration.	2,975 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	≤ 587.2 °F
	meter	3	≤ 586.9 °F
	computer	4	≤ 587.7 °F
	computer	3	≤ 587.5 °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			≥ 388,000 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 the 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 the Keff of the core will remain within the 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 surveillance SLC-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 $\leq 210^{O}$F, and Modes 5 and 6.

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

NOTE: When cycle burnup is > 500 EFPD, Figure 9 may be used to determine the required Boric Acid Tank Minimum Level.

Parameter	Limit
Refueling Water Storage Tank minimum boron concentration	2,700 ppm
Volume of 2,700 ppm boric acid solution required to maintain SDM at 68 °F	7,000 gallons
Refueling Water Storage Tank 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°F.

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

NOTE: When cycle burnup is > 500 EFPD, Figure 9 may be used to determine the required Boric Acid Tank Minimum Level.

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

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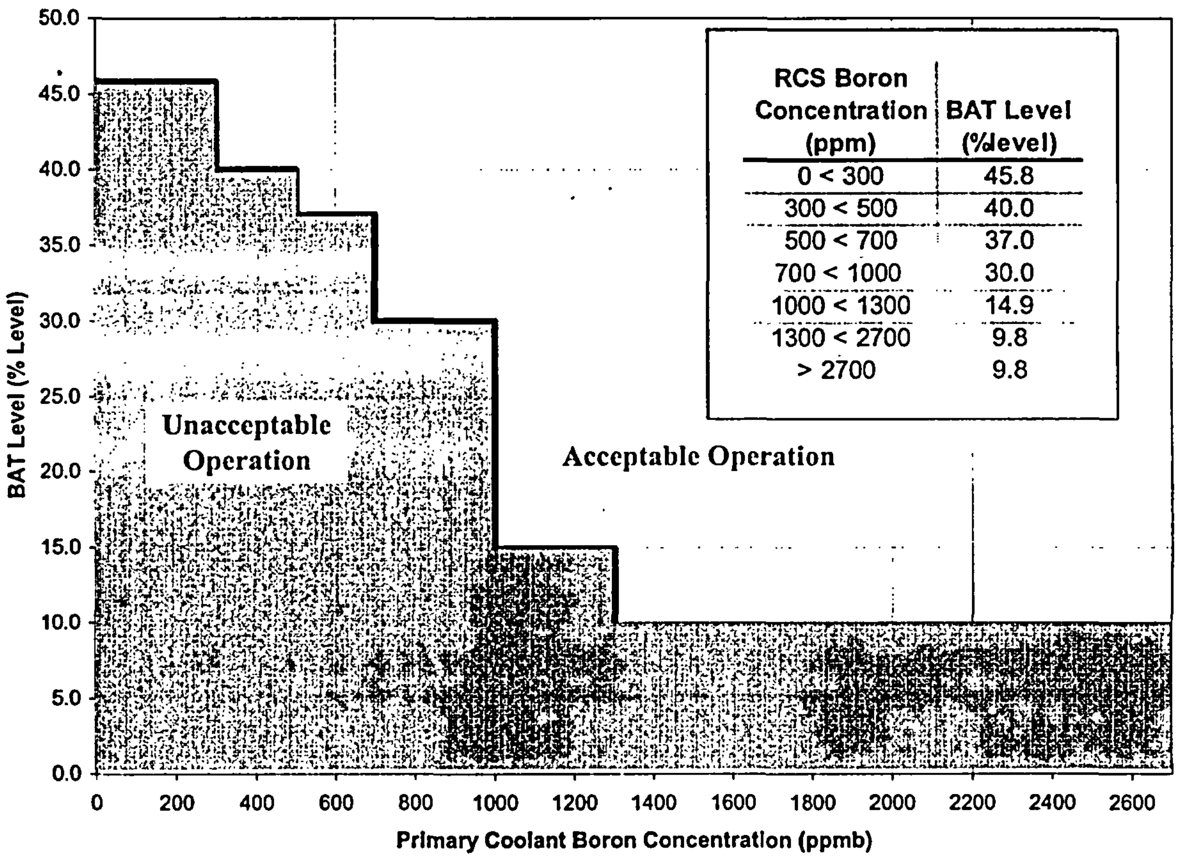
Figure 9

Boric Acid Storage Tank Indicated Level Versus Primary Coolant Boron Concentration

(Valid When Cycle Burnup is > 500 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. Due to the size of the monitoring factor data, Appendix A is controlled electronically within Duke and is not included in the Duke internal copies of the COLR. The Catawba Reactor and Electrical Systems Engineering Section controls this information via computer files and should be contacted if there is a need to access this information.

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