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U. S. Nuclear Regulatory Commission
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Subject: McGuire Nuclear Station,
Docket Nos. 50-370
Unit 2, Cycle 17, Revision 25
Core Operating Limits Report (COLR)

Pursuant to McGuire Technical Specification 5.6.5.d, please find enclosed Revision 25 to the McGuire Unit 2 Cycle 17 Core Operating Limits Report (COLR).

Questions regarding this submittal should be directed to Kay Crane, McGuire Regulatory Compliance at (704) 875-4306.

Gary R. Peterson

Attachment

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McGuire Unit 2 Cycle 17

Core Operating Limits Report
Revision 25

February 2005

Calculation Number: MCC-1553.05-00-0418 (Rev. 0)

Duke Power Company

		Date
Prepared By:	David P. Botts	2/25/05
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Checked By:	[illegible] (Sections 2.2 and 2.10 – 2.17)	3/2/05
Approved By:	Stephen P. Schultz	3/2/05

QA Condition 1

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

McGuire 2 Cycle 17 Core Operating Limits Report

INSPECTION OF ENGINEERING INSTRUCTIONS

Inspection Waived By: Stephen P. Schultz (Sponsor)   Date: 3/02/05

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:	______

McGuire 2 Cycle 17 Core Operating Limits Report

Implementation Instructions For Revision 25

Revision 25 to the McGuire Unit 2 COLR contains limits specific to the McGuire Unit 2 Cycle 17 core and may become effective any time after no-mode is reached between Cycles 16 and 17. This revision must become effective prior to entering Mode 6 that starts Cycle 17.

McGuire 2 Cycle 17 Core Operating Limits Report

REVISION LOG

Revision	Issuance Date	Effective Pages	COLR
Revisions 0-2	Superseded	N/A	M2C09
Revisions 3-6	Superseded	N/A	M2C10
Revisions 7-12	Superseded	N/A	M2C11
Revision 13-15	Superseded	N/A	M2C12
Revision 16-17	Superseded	N/A	M2C13
Revision 18-20	Superseded	N/A	M2C14
Revision 21-22	Superseded	N/A	M2C15
Revision 23	September 3, 2003	N/A	M2C16 (Orig. Issue)
Revision 24	January 26, 2003	N/A	M2C16 (Rev. 1)
Revision 25	February 22, 2004	1-33	M2C17 (Orig. Issue)

McGuire 2 Cycle 17 Core Operating Limits Report

Insertion Sheet For Revision 25

Remove pages	Insert Rev. 25 pages
Pages 1 – 33	Pages 1 – 33
Appendix A*	Appendix A*

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

McGuire 2 Cycle 17 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 the COLR are summarized below.

TS Number	Technical Specifications	COLR Parameter	COLR Section	EI Page
1.1	Requirements for Operational Mode 6	Mode 6 Definition	2.1	9
2.1.1	Reactor Core Safety Limits	RCS Temperature and Pressure Safety Limits	2.2	9
3.1.1	Shutdown Margin	Shutdown Margin	2.3	9
3.1.3	Moderator Temperature Coefficient	MTC	2.4	11
3.1.4	Rod Group Alignment Limits	Shutdown Margin	2.3	9
3.1.5	Shutdown Bank Insertion Limits	Shutdown Margin	2.3	9
3.1.5	Shutdown Bank Insertion Limits	Shutdown Bank Insertion Limit	2.5	11
3.1.6	Control Bank Insertion Limits	Shutdown Margin	2.3	9
3.1.6	Control Bank Insertion Limits	Control Bank Insertion Limit	2.6	11
3.1.8	Physics Test Exceptions	Shutdown Margin	2.3	9
3.2.1	Heat Flux Hot Channel Factor	Fq, AFD, OTΔT and Penalty Factors	2.7	16
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor	FΔH, AFD and Penalty Factors	2.8	21
3.2.3	Axial Flux Difference	AFD	2.9	22
3.3.1	Reactor Trip System Instrumentation Setpoint	OTΔT and OPΔT Constants	2.10	25
3.4.1	RCS Pressure, Temperature and Flow limits for DNB	RCS Pressure, Temperature and Flow	2.11	27
3.5.1	Accumulators	Max and Min Boron Conc.	2.12	27
3.5.4	Refueling Water Storage Tank	Max and Min Boron Conc.	2.13	27
3.7.14	Spent Fuel Pool Boron Concentration	Min Boron Concentration	2.14	29
3.9.1	Refueling Operations – Boron Concentration	Min Boron Concentration	2.15	29
5.6.5	Core Operating Limits Report (COLR)	Analytical Methods	1.1	7

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

SLC Number	Selected Licensing Commitment	COLR Parameter	COLR Section	EI Page
16.9.14	Borated Water Source – Shutdown	Borated Water Volume and Conc. for BAT/RWST	2.16	30
16.9.11	Borated Water Source – Operating	Borated Water Volume and Conc. for BAT/RWST	2.17	31

McGuire 2 Cycle 17 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 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 M2C17

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 M2C17

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 M2C17

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

   Revision 3
   SER Date: September 24, 2003

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1.1 Analytical Methods (continued)

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

   Revision 0
   Report Date: November 15, 1991 (Republished December 2000)

8. DPC-NE-3002A, "FSAR 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 M2C17

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

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

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1.1 Analytical Methods (continued)

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

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 the NRC approved methodologies specified in Section 1.1.

2.1 Requirements for Operational Mode 6

The following condition is required for operational mode 6.

2.1.1 The Reactivity Condition requirement for operational mode 6 is that k_{eff} must be less than, or equal to 0.95.

2.2 Reactor Core Safety Limits (TS 2.1.1)

2.2.1 The Reactor Core Safety Limits are shown in Figure 1.

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

2.3.1 For TS 3.1.1, SDM shall be $\geq$ 1.3% $\Delta K/K$ in mode 2 with k-eff < 1.0 and in modes 3 and 4.

2.3.2 For TS 3.1.1, SDM shall be $\geq$ 1.0% $\Delta K/K$ in mode 5.

2.3.3 For TS 3.1.4, SDM shall be $\geq$ 1.3% $\Delta K/K$ in modes 1 and 2.

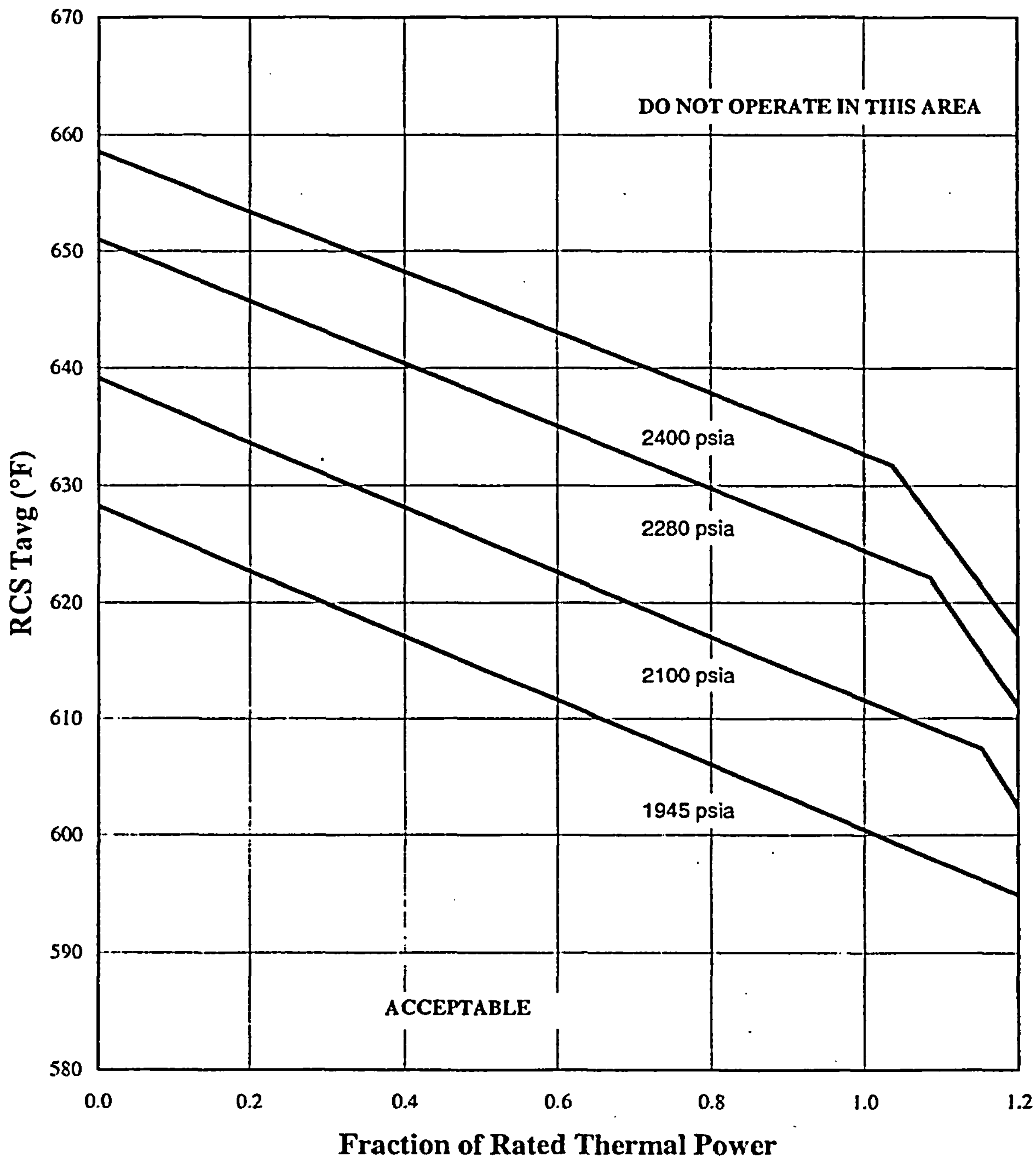
2.3.4 For TS 3.1.5, SDM shall be $\geq$ 1.3% $\Delta K/K$ in mode 1 and mode 2 with any control bank not fully inserted.

2.3.5 For TS 3.1.6, SDM shall be $\geq$ 1.3% $\Delta K/K$ in mode 1 and mode 2 with K-eff $\geq$ 1.0.

2.3.6 For TS 3.1.8, SDM shall be $\geq$ 1.3% $\Delta K/K$ in mode 2 during Physics Testing.

McGuire 2 Cycle 17 Core Operating Limits Report

Figure 1
Reactor Core Safety Limits
Four Loops in Operation



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2.4 Moderator Temperature Coefficient - MTC (TS 3.1.3)

2.4.1 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.

2.4.2 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.

2.4.3 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 the most positive MTC)
EOC = End of Cycle
ARO = All Rods Out
HZP = Hot Zero Power
RTP = Rated Thermal Power
PPM = Parts per million (Boron)

2.5 Shutdown Bank Insertion Limit (TS 3.1.5)

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

2.6 Control Bank Insertion Limits (TS 3.1.6)

2.6.1 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.

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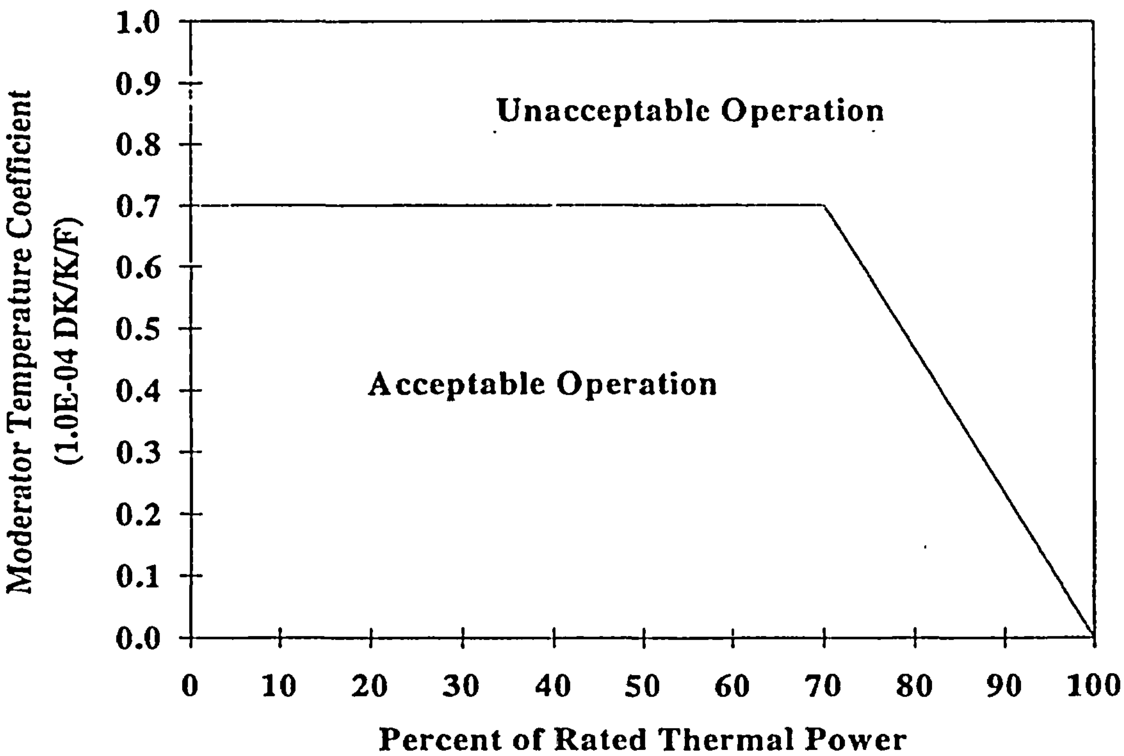
Figure 2

Moderator Temperature Coefficient Upper Limit Versus Power Level



NOTE: Compliance with Technical Specification 3.1.3 may require rod withdrawal limits. Refer to OP/2/A/6100/22 Unit 2 Data Book for details.

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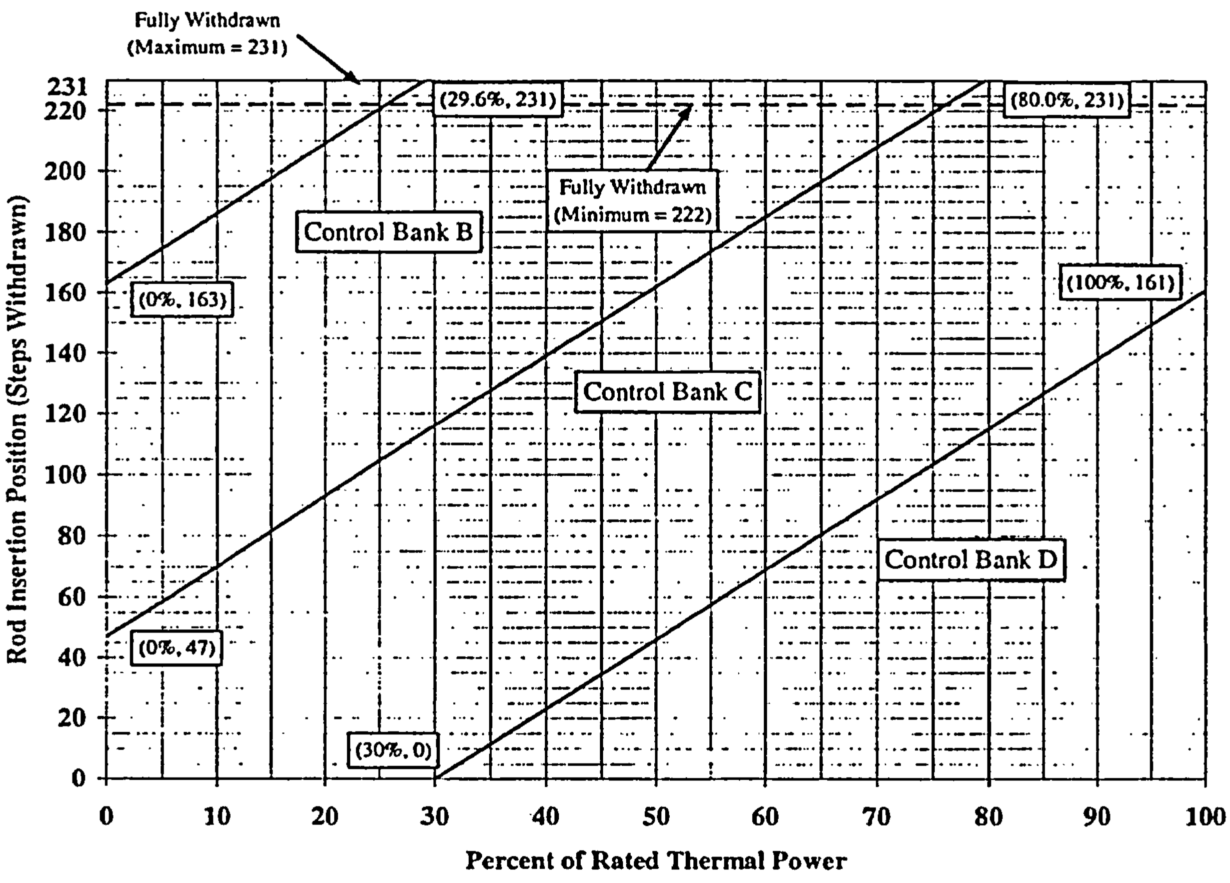
Figure 3

Control Bank Insertion Limits Versus Percent Rated Thermal Power



NOTE: Compliance with Technical Specification 3.1.3 may require rod withdrawal limits. Refer to OP/2/A/6100/22 Unit 2 Data Book for details.

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Table 1
RCCA Withdrawal Steps and Sequence

RCCAs Fully Withdrawn at 222 SWD

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

RCCAs Fully Withdrawn at 223 SWD

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

RCCAs Fully Withdrawn at 224 SWD

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

RCCAs Fully Withdrawn at 225 SWD

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

RCCAs Fully Withdrawn at 226 SWD

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

RCCAs Fully Withdrawn at 227 SWD

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

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Table 1 Continued
RCCA Withdrawal Steps and Sequence

RCCAs Fully Withdrawn at 228 SWD			
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

RCCAs Fully Withdrawn at 229 SWD			
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

RCCAs Fully Withdrawn at 230 SWD			
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

RCCAs Fully Withdrawn at 231 SWD			
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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2.7 Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$ (TS 3.2.1)

2.7.1 $F_Q(X,Y,Z)$ 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 = (Thermal Power)/(Rated Power)

Note: The 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 2.7.5 and 2.7.6.

2.7.2 $F_Q^{RTP} = 2.60 \times K(BU)$

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

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

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

2.7.5 $$F_Q^L(X,Y,Z)^{OP} = \frac{F_Q^D(X,Y,Z) * M_Q(X,Y,Z)}{UMT * MT * TILT}$$

where:

$F_Q^L(X,Y,Z)^{OP}$ = Cycle dependent maximum allowable design peaking factor that ensures that the $F_Q(X,Y,Z)$ LOCA limit will be preserved for operation within the LCO 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 A Table A-1 for normal operating conditions, and in

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Appendix A Table A-2 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 A Table A-1 for normal operating conditions, and in Appendix A Table A-2 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 that accounts for the peaking increase from an allowable quadrant power tilt ratio of 1.02. (TILT = 1.035)

2.7.6 $$F_Q^L(X,Y,Z)^{RPS} = \frac{F_Q^D(X,Y,Z) * M_C(X,Y,Z)}{UMT * MT * TILT}$$

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 will be preserved for operation within the LCO limits. $F_Q^L(X,Y,Z)^{RPS}$ includes allowances for calculation 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 A Table A-1 for normal operating conditions, and in Appendix A Table A-2 for power escalation testing during initial startup operation.

$M_C(X,Y,Z)$ = 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 A Table A-3 for normal operating conditions, and in Appendix A Table A-4 for power escalation testing during initial startup operation.

UMT = Total Peak Measurement Uncertainty (UMT = 1.05)

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MT = Engineering Hot Channel Factor (MT = 1.03)

TILT = Peaking penalty that accounts for the peaking increase from an allowable quadrant power tilt ratio of 1.02. (TILT = 1.035)

2.7.7 KSLOPE = 0.0725

where:

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

2.7.8 F_Q(X,Y,Z) penalty factors for Technical Specification Surveillance's 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 Westinghouse RFA Fuel

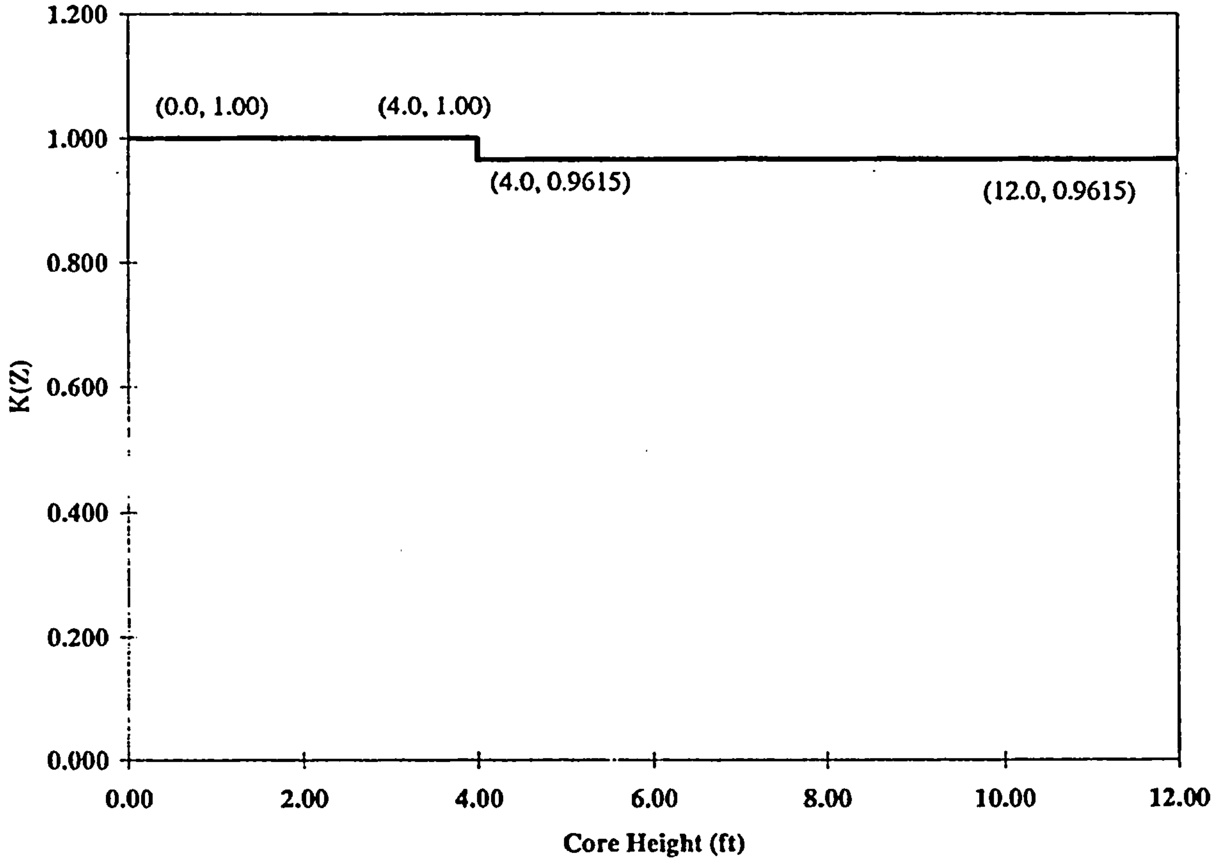

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Table 2

$F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ Penalty Factors

For Technical Specification Surveillance's 3.2.1.2, 3.2.1.3 and 3.2.2.2

Burnup (EFPD)	$F_Q(X,Y,Z)$ Penalty Factor (%)	$F_{\Delta H}(X,Y,Z)$ Penalty Factor (%)
0	2.00	2.00
4	2.00	2.00
12	2.00	2.00
25	2.63	2.00
50	2.83	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
536	2.00	2.00

Note: Linear interpolation is adequate for intermediate cycle burnups. All cycle burnups outside of 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 Technical Specification Surveillances 3.2.1.2, 3.2.1.3 and 3.2.2.2.

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2.8 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 is defined by the following relationship.

2.8.1 $$F^{L}_{\Delta H}(X,Y)^{LCO} = \text{MARP}(X,Y) * \left[1.0 + \frac{1}{\text{RRH}} * (1.0 - P)\right]$$

where:

$F^{L}_{\Delta H}(X,Y)^{LCO}$ is defined as the steady-state, maximum allowed radial peak. $F^{L}_{\Delta H}(X,Y)^{LCO}$ 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 its limit. RRH also is used to scale the MARP limits as a function of power per the $[F^{L}_{\Delta H}(X,Y)]^{LCO}$ equation. (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.

2.8.2 $$F^{L}_{\Delta H}(X,Y)^{SURV} = \frac{F^{D}_{\Delta H}(X,Y) \times M_{\Delta H}(X,Y)}{\text{UMR} \times \text{TILT}}$$

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 will be preserved for operation within the LCO limits. $F^{L}_{\Delta H}(X,Y)^{SURV}$ includes allowances for calculation-measurement uncertainty.

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$F^{D}_{\Delta H}(X,Y)$ = Design radial power distribution for $F_{\Delta H}$. $F^{D}_{\Delta H}(X,Y)$ is provided in Appendix A Table A-5 for normal operation, and in Appendix A Table A-6 for power escalation testing during initial startup operation.

$M_{\Delta H}(X,Y)$ = 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 A Table A-5 for normal operation, and in Appendix A Table A-6 for power escalation testing during initial startup operation.

UMR = Uncertainty value for measured radial peaks, (UMR= 1.04). 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 the peaking increase for an allowable quadrant power tilt ratio of 1.02, (TILT = 1.035).

2.8.3 RRH = 3.34

where:

RRH = Thermal power reduction required to compensate for each 1% that the measured radial peak, $F^{M}_{\Delta H}(X,Y)$ exceeds its limit.

2.8.4 TRH = 0.04

where:

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

2.8.5 $F_{\Delta H}(X,Y)$ penalty factors for Technical Specification Surveillance 3.2.2.2 are provided in Table 2.

2.9 Axial Flux Difference – AFD (TS 3.2.3)

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

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Table 3
Maximum Allowable Radial Peaks (MARPS)

RFA MARPS

Core Ht (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	2.068	2.090	2.049	1.972	1.900	1.778	1.315	1.246
1.2	1.843	1.879	1.938	1.992	1.974	2.068	2.054	2.012	1.935	1.862	1.785	1.301	1.224
2.4	1.846	1.876	1.931	1.981	1.974	2.068	2.025	1.981	1.903	1.832	1.757	1.468	1.456
3.6	1.843	1.869	1.920	1.964	1.974	2.068	2.005	1.968	1.892	1.820	1.716	1.471	1.431
4.8	1.838	1.868	1.906	1.945	1.974	2.006	1.945	1.925	1.862	1.802	1.725	1.326	1.285
6.0	1.834	1.856	1.891	1.921	1.946	1.934	1.878	1.863	1.802	1.747	1.673	1.384	1.317
7.2	1.828	1.845	1.871	1.893	1.887	1.872	1.809	1.787	1.732	1.681	1.618	1.316	1.277
8.4	1.823	1.829	1.847	1.857	1.816	1.795	1.739	1.722	1.675	1.630	1.551	1.247	1.211
9.6	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
10.8	1.798	1.784	1.761	1.738	1.697	1.682	1.626	1.605	1.558	1.512	1.430	1.149	1.097
11.4	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

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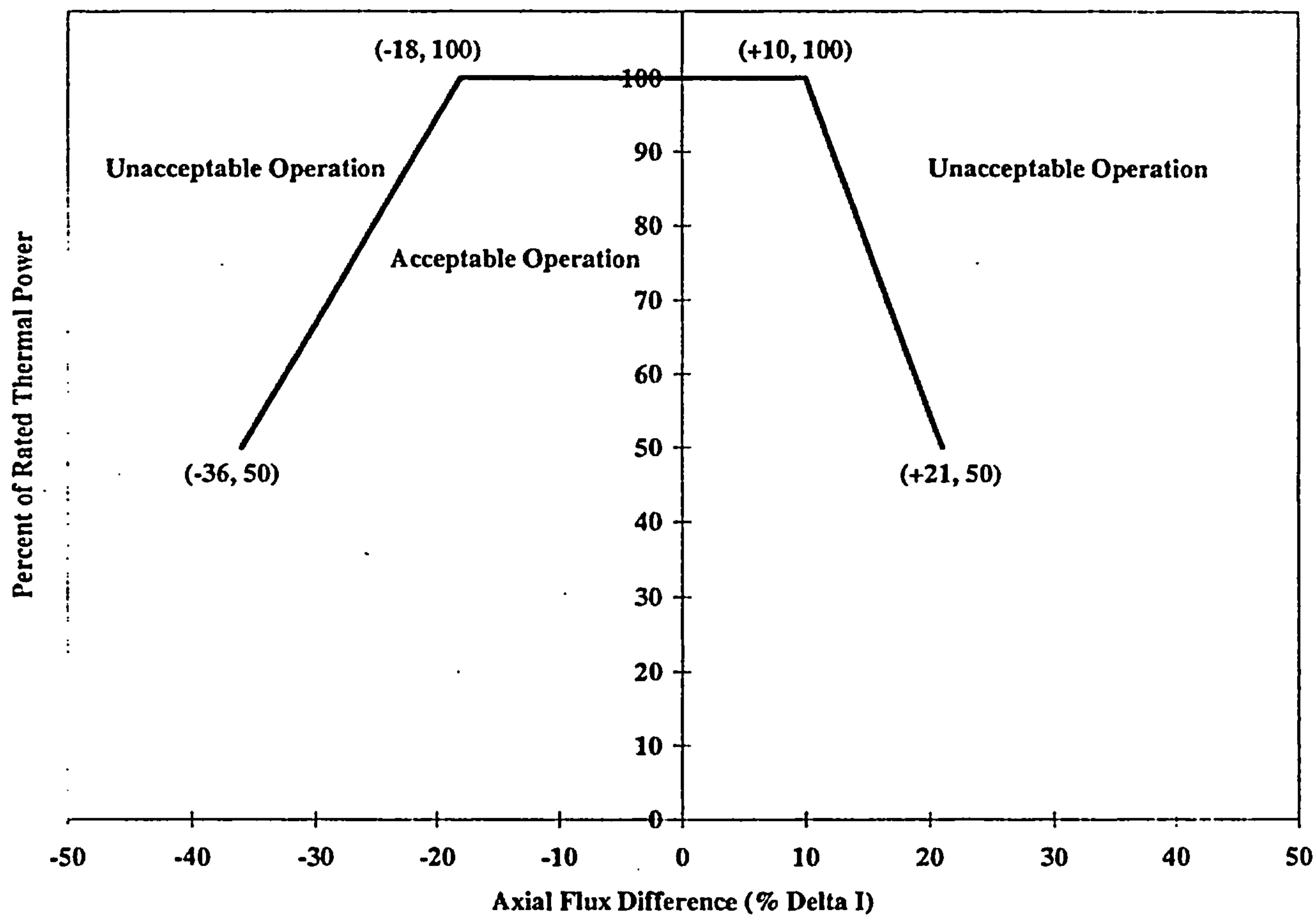
Figure 5

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 OP/2/A/6100/22 Unit 2 Data Book of more details.

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

2.10.1 Overtemperature ΔT Setpoint Parameter Values

Parameter	Value
Nominal Tavg at RTP	$T' \leq 585.1°F$
Nominal RCS Operating Pressure	$P' = 2235$ psig
Overtemperature ΔT reactor trip setpoint	$K_1 \leq 1.1978$
Overtemperature ΔT reactor trip heatup setpoint penalty coefficient	$K_2 = 0.0334/°F$
Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient	$K_3 = 0.001601$/psi
Time constants utilized in the lead-lag compensator for ΔT	$\tau_1 \geq 8$ sec. $\tau_2 \leq 3$ sec.
Time constant utilized in the lag compensator for ΔT	$\tau_3 \leq 2.0$ sec.
Time constants utilized in the lead-lag compensator for T_{avg}	$\tau_4 \geq 28$ sec. $\tau_5 \leq 4$ sec.
Time constant utilized in the measured T_{avg} lag compensator	$\tau_6 \leq 2.0$ 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 the $f_1(\Delta I)$ "negative" slope 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 the OTΔT $f_1(\Delta I)$ negative breakpoint and slope unnecessary.

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2.10.2 Overpower ΔT Setpoint Parameter Values

Parameter	Value
Nominal Tavg at RTP	$T'' \leq 585.1°F$
Overpower ΔT reactor trip setpoint	$K_4 \leq 1.0864$
Overpower ΔT reactor trip Penalty	$K_5 = 0.02/°F$ for increasing Tavg $K_5 = 0.0$ for decreasing Tavg
Overpower ΔT reactor trip heatup setpoint penalty coefficient	$K_6 = 0.001179/°F$ for $T > T''$ $K_6 = 0.0$ for $T \leq T''$
Time constants utilized in the lead-lag compensator for ΔT	$\tau_1 \geq 8$ sec. $\tau_2 \leq 3$ sec.
Time constant utilized in the lag compensator for ΔT	$\tau_3 \leq 2.0$ sec.
Time constant utilized in the measured T_{avg} lag compensator	$\tau_6 \leq 2.0$ sec.
Time constant utilized in the rate-lag controller for T_{avg}	$\tau_7 \geq 5$ 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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2.11 RCS Pressure, Temperature and Flow Limits for DNB (TS 3.4.1)

2.11.1 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.	2,475 ppm
Cold Leg Accumulator maximum boron concentration.	2,875 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.	2,675 ppm
Refueling Water Storage Tank maximum boron concentration.	2,875 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	$\leq$ 587.2 °F
	meter	3	$\leq$ 586.9 °F
	computer	4	$\leq$ 587.7 °F
	computer	3	$\leq$ 587.5 °F
2. Indicated Pressurizer Pressure	meter	4	$\geq$ 2219.8 psig
	meter	3	$\geq$ 2222.1 psig
	computer	4	$\geq$ 2215.8 psig
	computer	3	$\geq$ 2217.5 psig
3. RCS Total Flow Rate			$\geq$ 388,000 gpm

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2.14 Spent Fuel Pool Boron Concentration (TS 3.7.14)

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.	2,675 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.	2,675 ppm

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2.16 Borated Water Source – Shutdown (SLC 16.9.14)

2.16.1 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 $\leq$ 300 °F and modes 5 and 6.

Parameter	Limit
Boric Acid Tank minimum contained borated water volume	10,599 gallons 13.6% Level
Note: When cycle burnup is > 460 EFPD, Figure 6 may be used to determine the required BAT minimum level.	
Boric Acid Tank minimum boron concentration	7,000 ppm
Boric Acid Tank minimum water volume required to maintain SDM at 7,000 ppm	2,300 gallons
Refueling Water Storage Tank minimum contained borated water volume	47,700 gallons 41 inches
Refueling Water Storage Tank minimum boron concentration	2,675 ppm
Refueling Water Storage Tank minimum water volume required to maintain SDM at 2,675 ppm	8,200 gallons

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2.17 Borated Water Source - Operating (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 modes 1, 2, 3, and mode 4 with all RCS cold leg temperature > 300 °F.

Parameter	Limit
Boric Acid Tank minimum contained borated water volume	22,049 gallons 38.0% Level

Note: When cycle burnup is > 460 EFPD, Figure 6 may be used to determine the required BAT minimum level.

Parameter	Limit
Boric Acid Tank minimum boron concentration	7,000 ppm
Boric Acid Tank minimum water volume required to maintain SDM at 7,000 ppm	13,750 gallons
Refueling Water Storage Tank minimum contained borated water volume	96,607 gallons 103.6 inches
Refueling Water Storage Tank minimum boron concentration	2,675 ppm
Refueling Water Storage Tank maximum boron concentration (TS 3.5.4)	2,875 ppm
Refueling Water Storage Tank minimum water volume required to maintain SDM at 2,675 ppm	57,107 gallons

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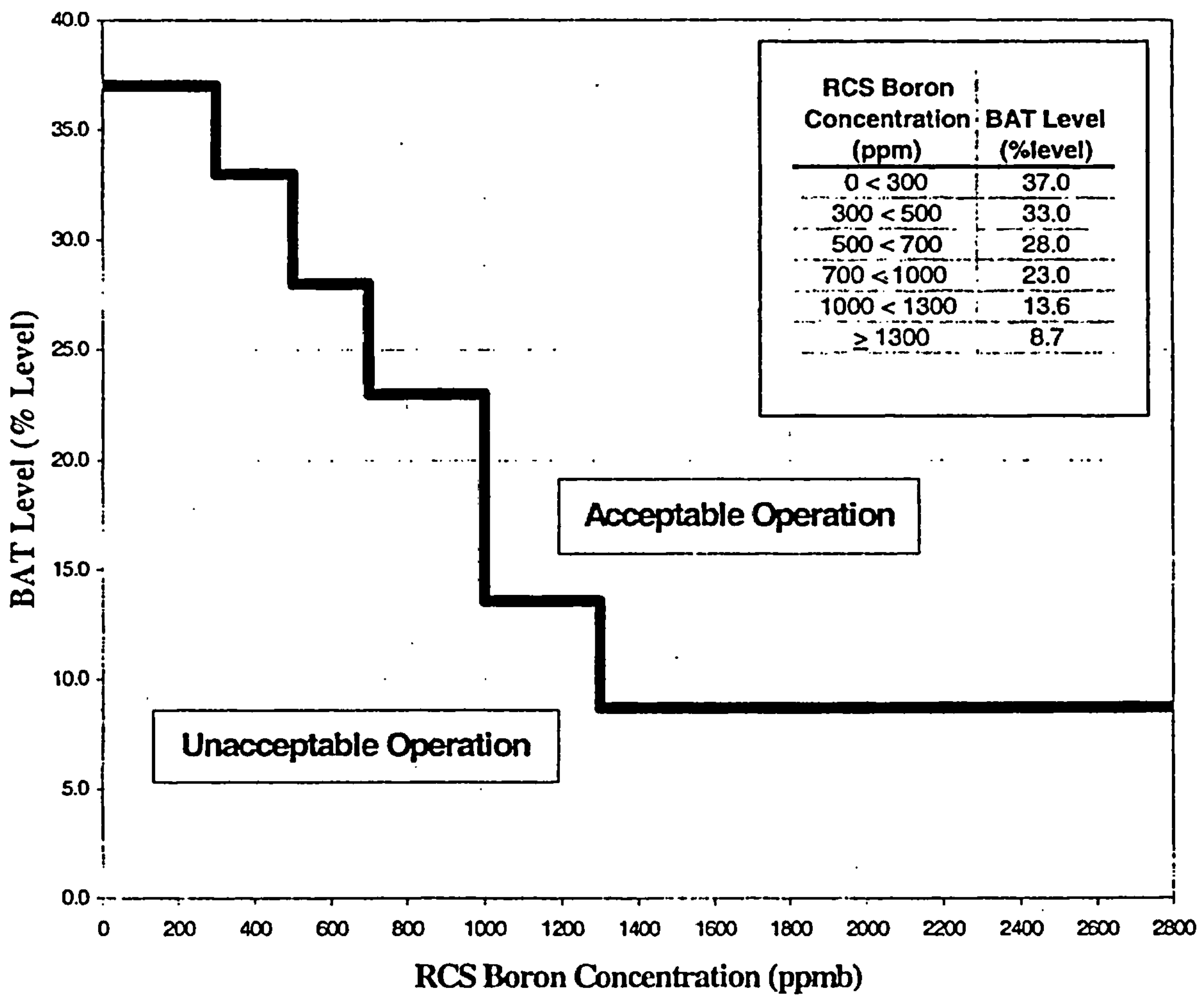
Figure 6
Boric Acid Storage Tank Indicated Level Versus RCS Boron Concentration

(Valid When Cycle Burnup is > 460 EFPD)

This figure includes additional volumes listed in SLC 16.9-14 and 16.9-11



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NOTE: Data contained in the Appendix to this document was generated in the McGuire 2 Cycle 17 Maneuvering Analysis calculation file, MCC-1553.05-00-0408. 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 McGuire Nuclear Engineering Section will control this information via computer file(s) and should be contacted if there is a need to access this information.