



REGIS T. REPKO
Vice President
McGuire Nuclear Station

Duke Energy
MG01VP / 12700 Hagers Ferry Rd.
Huntersville, NC 28078

980-875-4111
980-875-4809 fax
regis.repko@duke-energy.com

September 29, 2011

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Subject: Duke Energy Carolinas, LLC (Duke)
McGuire Nuclear Station
Docket Nos. 50-370
Unit 1, Cycle 22, Revision 1
Core Operating Limits Report

Pursuant to McGuire Technical Specification (TS) 5.6.5.d, please find enclosed Revision 1 of the McGuire Unit 1 Cycle 22 Core Operating Limits Report (COLR).

Revision 0 was not required to be submitted in accordance with TS 5.6.5.d because it was superseded prior to its effective date (No Mode).

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

Regis T. Repko

Attachment

A001
NRR

U. S. Nuclear Regulatory Commission
September 29, 2011
Page 2

cc: Mr. J.H. Thompson (addressee only)
NRC Senior Project Manager (McGuire)
U.S. Nuclear Regulatory Commission
Mail Stop O-8 G9A
Washington, D.C. 20555-0001

Mr. V.M McCree,
Regional Administrator
U. S. Nuclear Regulatory Commission, Region II
Marquis One Tower
245 Peachtree Center Ave., NE Suite 1200
Atlanta, Georgia 30303-1257

Mr. John Zeiler
Senior Resident Inspector
McGuire Nuclear Station

McGuire Unit 1 Cycle 22
Core Operating Limits Report
Revision 1

September 2011

Calculation Number: MCC-1553.05-00-0549, Rev. 1

Duke Energy

		Date
Prepared By:	<u>ML Elder</u>	<u>9/23/11</u>
Checked By:	<u>Nicholas R Hager</u>	<u>9/23/11</u>
Checked By:	<u>Christy B. Loy</u> (Sections 2.2 and 2.10 - 2.18)	<u>9/23/11</u>
Approved By:	<u>RC Harney</u>	<u>9/26/2011</u>

QA Condition 1

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

McGuire 1 Cycle 22 Core Operating Limits Report

INSPECTION OF ENGINEERING INSTRUCTIONS

Inspection Waived By: RC Hawley
(Sponsor)

Date: 9/26/2011

<u>CATAWBA</u>		
	Inspection Waived	
MCE (Mechanical & Civil)	<input type="checkbox"/>	Inspected By/Date: _____
RES (Electrical Only)	<input type="checkbox"/>	Inspected By/Date: _____
RES (Reactor)	<input type="checkbox"/>	Inspected By/Date: _____
MOD	<input type="checkbox"/>	Inspected By/Date: _____
Other (_____)	<input type="checkbox"/>	Inspected By/Date: _____

<u>OCONEE</u>		
	Inspection Waived	
MCE (Mechanical & Civil)	<input type="checkbox"/>	Inspected By/Date: _____
RES (Electrical Only)	<input type="checkbox"/>	Inspected By/Date: _____
RES (Reactor)	<input type="checkbox"/>	Inspected By/Date: _____
MOD	<input type="checkbox"/>	Inspected By/Date: _____
Other (_____)	<input type="checkbox"/>	Inspected By/Date: _____

<u>MCGUIRE</u>		
	Inspection Waived	
MCE (Mechanical & Civil)	<input checked="" type="checkbox"/>	Inspected By/Date: _____
RES (Electrical Only)	<input checked="" type="checkbox"/>	Inspected By/Date: _____
RES (Reactor)	<input checked="" type="checkbox"/>	Inspected By/Date: _____
MOD	<input checked="" type="checkbox"/>	Inspected By/Date: _____
Other (_____)	<input type="checkbox"/>	Inspected By/Date: _____

McGuire 1 Cycle 22 Core Operating Limits Report

Implementation Instructions for Revision 1

Revision Description and PIP Tracking

Revision 1 of the McGuire Unit 1 Cycle 22 COLR contains limits specific to the reload core and was revised to update an Analytical Method reference in Section 1.1. The PIP associated with this revision is: M-11-03805.

Implementation Schedule

Revision 1 may become effective any time during No MODE between cycles 21 and 22 but must become effective prior to entering MODE 6 which starts cycle 22. The McGuire Unit 1 Cycle 22 COLR will cease to be effective during No MODE between cycle 22 and 23.

Data files to be Implemented

No data files are transmitted as part of this document.

McGuire 1 Cycle 22 Core Operating Limits Report

REVISION LOG

<u>Revision</u>	<u>Effective Date</u>	<u>Pages Affected</u>	<u>COLR</u>
0	August 2011	1-32, Appendix A*	M1C22 COLR, Rev. 0
1	September 2011	1-32	M1C22 COLR, Rev. 1

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

McGuire 1 Cycle 22 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 the Technical Specification 5.6.5. The Technical Specifications that reference the COLR are summarized below.

<u>TS Number</u>	<u>Technical Specifications</u>	<u>COLR Parameter</u>	<u>COLR Section</u>	<u>EI Page</u>
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	15
3.1.8	Physics Test Exceptions	Shutdown Margin	2.3	9
3.2.1	Heat Flux Hot Channel Factor	F _q , AFD, OTΔT and Penalty Factors	2.7	15
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor	FΔH, AFD and Penalty Factors	2.8	20
3.2.3	Axial Flux Difference	AFD	2.9	21
3.3.1	Reactor Trip System Instrumentation Setpoint	OTΔT and OPΔT Constants	2.10	24
3.4.1	RCS Pressure, Temperature and Flow limits for DNB	RCS Pressure, Temperature and Flow	2.11	26
3.5.1	Accumulators	Max and Min Boron Conc.	2.12	26
3.5.4	Refueling Water Storage Tank	Max and Min Boron Conc.	2.13	26
3.7.14	Spent Fuel Pool Boron Concentration	Min Boron Concentration	2.14	28
3.9.1	Refueling Operations – Boron Concentration	Min Boron Concentration	2.15	28
5.6.5	Core Operating Limits Report (COLR)	Analytical Methods	1.1	6

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

<u>SLC Number</u>	<u>Selected Licensing Commitment</u>	<u>COLR Parameter</u>	<u>COLR Section</u>	<u>EI Page</u>
16.9.14	Borated Water Source – Shutdown	Borated Water Volume and Conc. for BAT/RWST	2.16	29
16.9.11	Borated Water Source – Operating	Borated Water Volume and Conc. for BAT/RWST	2.17	30
16.9.7	Standby Shutdown System	Standby Makeup Pump Water Supply	2.18	30

McGuire 1 Cycle 22 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 M1C22

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 M1C22

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 M1C22

McGuire 1 Cycle 22 Core Operating Limits Report

1.1 Analytical Methods (continued)

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

Revision 4a
Report Date: July 2009
7. DPC-NE-3001-PA, "Multidimensional Reactor Transients and Safety Analysis Physics Parameter Methodology," (DPC Proprietary).

Revision 0a
Report Date: May 2009
8. DPC-NE-3002-A, "UFSAR Chapter 15 System Transient Analysis Methodology".

Revision 4b
Report Date: September 2010
9. DPC-NE-2004P-A, "Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01," (DPC Proprietary).

Revision 2a
Report Date: December 2008
10. DPC-NE-2005P-A, "Thermal Hydraulic Statistical Core Design Methodology," (DPC Proprietary).

Revision 4a
Report Date: December 2008
11. DPC-NE-2008P-A, "Fuel Mechanical Reload Analysis Methodology Using TACO3," (DPC Proprietary).

Revision 1a
Report Date: December 2008
Not Used for M1C22
12. DPC-NE-2009-P-A, "Westinghouse Fuel Transition Report," (DPC Proprietary).

Revision 3a
Report Date: September 2011
13. DPC-NE-1004A, "Nuclear Design Methodology Using CASMO-3/SIMULATE-3P."

Revision 1a
Report Date: January 2009
Not Used for M1C22

McGuire 1 Cycle 22 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 2a
Report Date: December 2009

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

Revision 1a
Report Date: June 2009

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

Revision 1
Report Date: November 2008

McGuire 1 Cycle 22 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 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 $k_{\text{eff}} < 1.0$ 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.

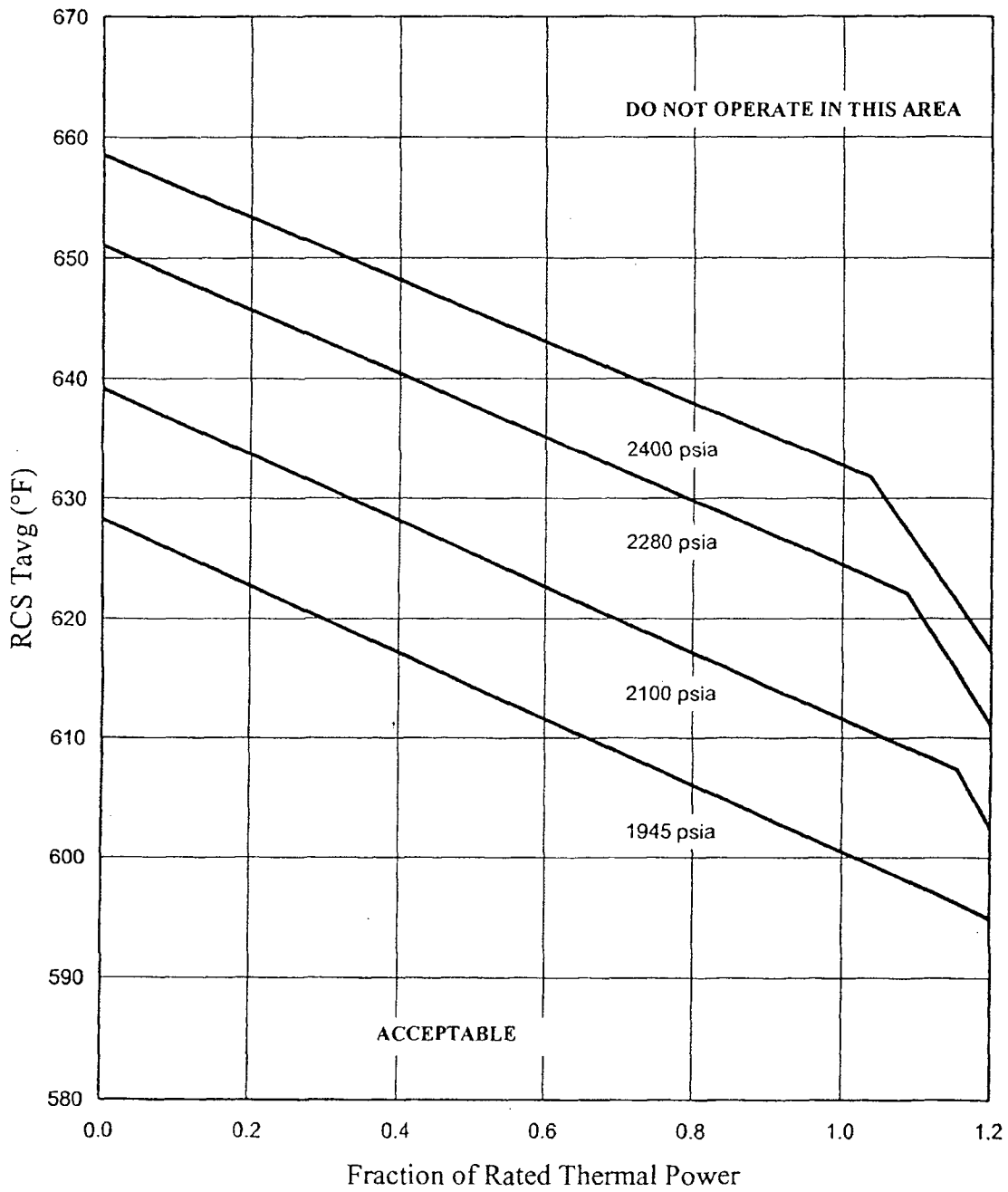
2.3.4 For TS 3.1.5, SDM shall be $\geq 1.3\% \Delta K/K$ in MODE 1 and MODE 2 with any control bank not fully inserted.

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

2.3.6 For TS 3.1.8, SDM shall be $\geq 1.3\% \Delta K/K$ in MODE 2 during PHYSICS TESTS.

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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 $0.7E-04 \Delta K/K/^{\circ}F$.

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

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 $-4.125E-04 \Delta K/K/^{\circ}F$.

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)

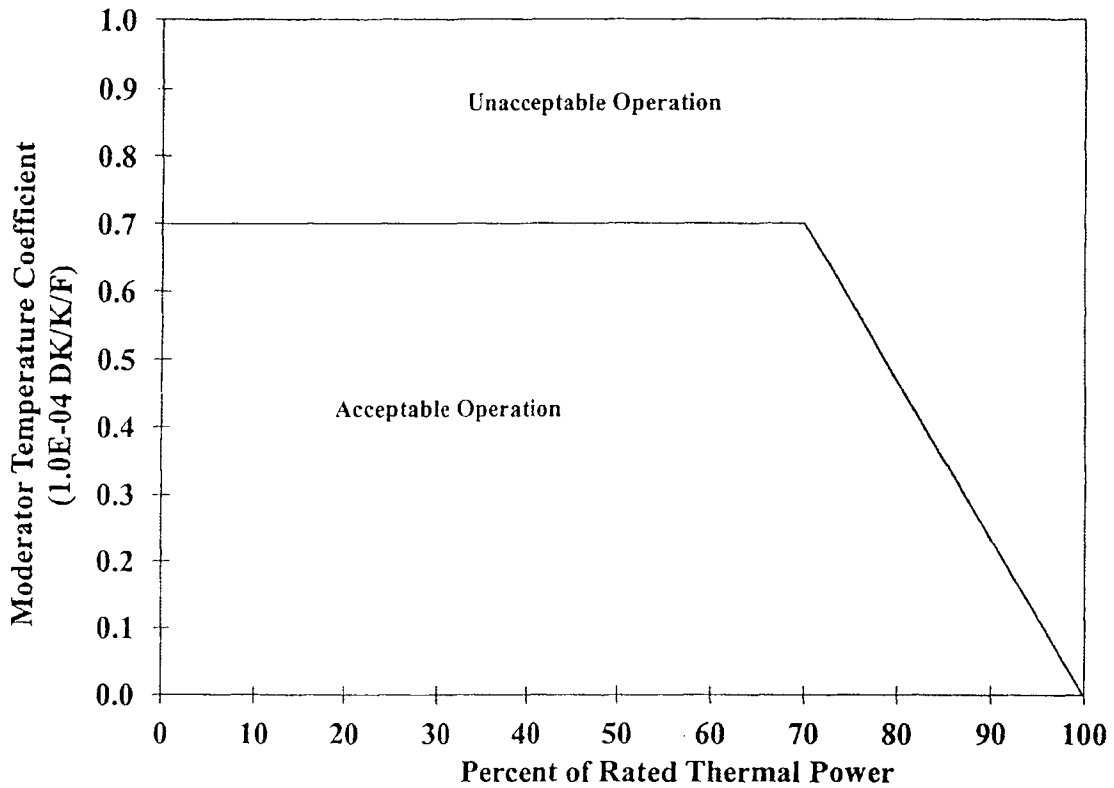
2.5.1 Each shutdown bank shall be withdrawn to at least 222 steps except under the conditions listed in Section 2.5.2. Shutdown banks are withdrawn in sequence and with no overlap.

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

McGuire 1 Cycle 22 Core Operating Limits Report

Figure 2

Moderator Temperature Coefficient Upper Limit Versus Power Level

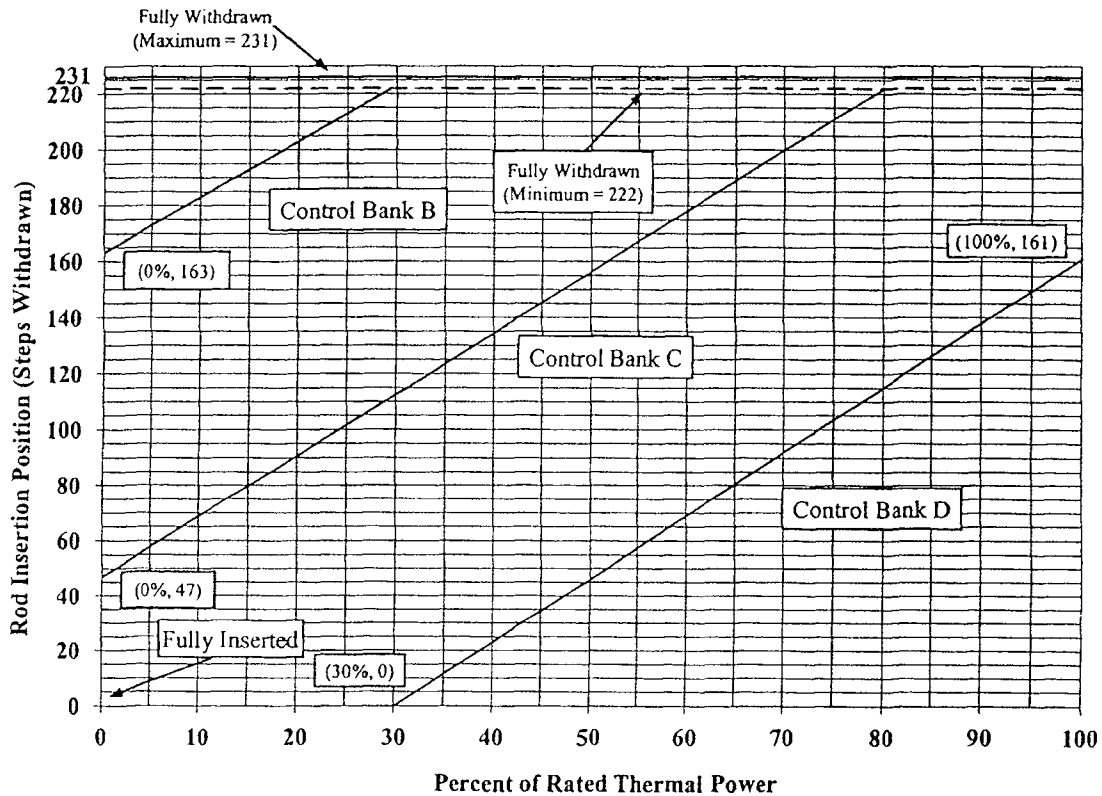


NOTE: Compliance with Technical Specification 3.1.3 may require rod withdrawal limits. Refer to OP/1/A/6100/22 Unit 1 Data Book for details.

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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:

$$\begin{aligned}
 \text{Bank CD RIL} &= 2.3(P) - 69 \quad \{30 \leq P \leq 100\} \\
 \text{Bank CC RIL} &= 2.3(P) + 47 \quad \{0 \leq P \leq 76.1\} \text{ for } CC \text{ RIL} = 222 \quad \{76.1 < P \leq 100\} \\
 \text{Bank CB RIL} &= 2.3(P) + 163 \quad \{0 \leq P \leq 25.7\} \text{ for } CB \text{ RIL} = 222 \quad \{25.7 < P \leq 100\}
 \end{aligned}$$

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

NOTES: (1) Compliance with Technical Specification 3.1.3 may require rod withdrawal limits. Refer to OP/1/A/6100/22 Unit 1 Data Book for details.

(2) Anytime any shutdown bank or control banks A, B, or C are inserted below 222 steps withdrawn, control bank D insertion is limited to ≥ 200 steps withdrawn (see Sections 2.5.2 and 2.6.2)

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Table 1
RCCA Withdrawal Steps and Sequence

Fully Withdrawn at 222 Steps				Fully Withdrawn at 223 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
222 Stop	106	0	0	223 Stop	107	0	0
222	116	0 Start	0	223	116	0 Start	0
222	222 Stop	106	0	223	223 Stop	107	0
222	222	116	0 Start	223	223	116	0 Start
222	222	222 Stop	106	223	223	223 Stop	107
Fully Withdrawn at 224 Steps				Fully Withdrawn at 225 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
224 Stop	108	0	0	225 Stop	109	0	0
224	116	0 Start	0	225	116	0 Start	0
224	224 Stop	108	0	225	225 Stop	109	0
224	224	116	0 Start	225	225	116	0 Start
224	224	224 Stop	108	225	225	225 Stop	109
Fully Withdrawn at 226 Steps				Fully Withdrawn at 227 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
226 Stop	110	0	0	227 Stop	111	0	0
226	116	0 Start	0	227	116	0 Start	0
226	226 Stop	110	0	227	227 Stop	111	0
226	226	116	0 Start	227	227	116	0 Start
226	226	226 Stop	110	227	227	227 Stop	111
Fully Withdrawn at 228 Steps				Fully Withdrawn at 229 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
228 Stop	112	0	0	229 Stop	113	0	0
228	116	0 Start	0	229	116	0 Start	0
228	228 Stop	112	0	229	229 Stop	113	0
228	228	116	0 Start	229	229	116	0 Start
228	228	228 Stop	112	229	229	229 Stop	113
Fully Withdrawn at 230 Steps				Fully Withdrawn at 231 Steps			
Control Bank A	Control Bank B	Control Bank C	Control Bank D	Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0	0 Start	0	0	0
116	0 Start	0	0	116	0 Start	0	0
230 Stop	114	0	0	231 Stop	115	0	0
230	116	0 Start	0	231	116	0 Start	0
230	230 Stop	114	0	231	231 Stop	115	0
230	230	116	0 Start	231	231	116	0 Start
230	230	230 Stop	114	231	231	231 Stop	115

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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 except under the conditions listed in Section 2.6.2. Specific control bank withdrawal and overlap limits as a function of the fully withdrawn position are shown in Table 1.
- 2.6.2 Control banks A, B, or C may be inserted to 219 steps withdrawn individually for up to 48 hours provided the plant was operated in steady state conditions near 100% FP prior to and during this exception.

2.7 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:

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

where,

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

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 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 $F_Q^{RTP} = 2.70 \times K(\text{BU})$
- 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 $K(\text{BU})$ is the normalized $F_Q(X,Y,Z)$ as a function of burnup. $K(\text{BU})$ 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 $F_Q^L(X,Y,Z)^{OP} = \frac{F_Q^D(X,Y,Z) * M_Q(X,Y,Z)}{UMT * MT * TILT}$

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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 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 A-1 for normal operating conditions and in Appendix Table A-4 for power escalation testing during initial startup operation.

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

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

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

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

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

where:

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

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

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

2.7.7 KSLOPE = 0.0725

where:

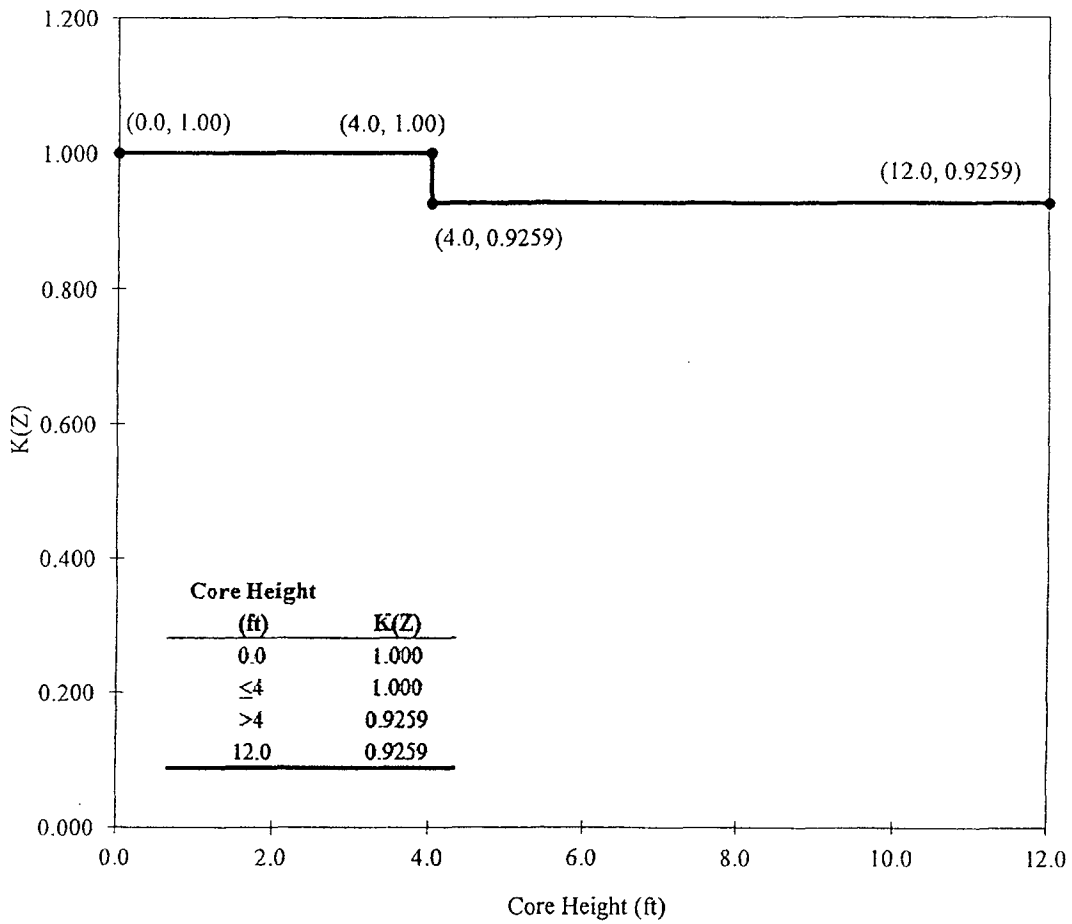
KSLOPE is the adjustment to the K_1 value from OT Δ T trip setpoint required to compensate for each 1% that $F_Q^M(X,Y,Z)$ exceeds $F_Q^L(X,Y,Z)^{RPS}$.

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



McGuire 1 Cycle 22 Core Operating Limits Report

Table 2
 $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ Penalty Factors
For Technical Specification Surveillances 3.2.1.2, 3.2.1.3 and 3.2.2.2

<u>Burnup (EFPD)</u>	<u>$F_Q(X,Y,Z)$ Penalty Factor (%)</u>	<u>$F_{\Delta H}(X,Y,Z)$ Penalty Factor (%)</u>
0	2.00	2.00
4	2.00	2.00
12	2.00	2.00
25	3.30	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
465	2.00	2.00
489	2.00	2.00
499	2.00	2.00
514	2.00	2.00
524	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.

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

where:

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

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

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

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

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

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

where:

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

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

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

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

2.8.3 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. ($0 < P \leq 1.0$)

2.8.4 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.

2.8.5 $F_{\Delta H}(X,Y)$ 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)

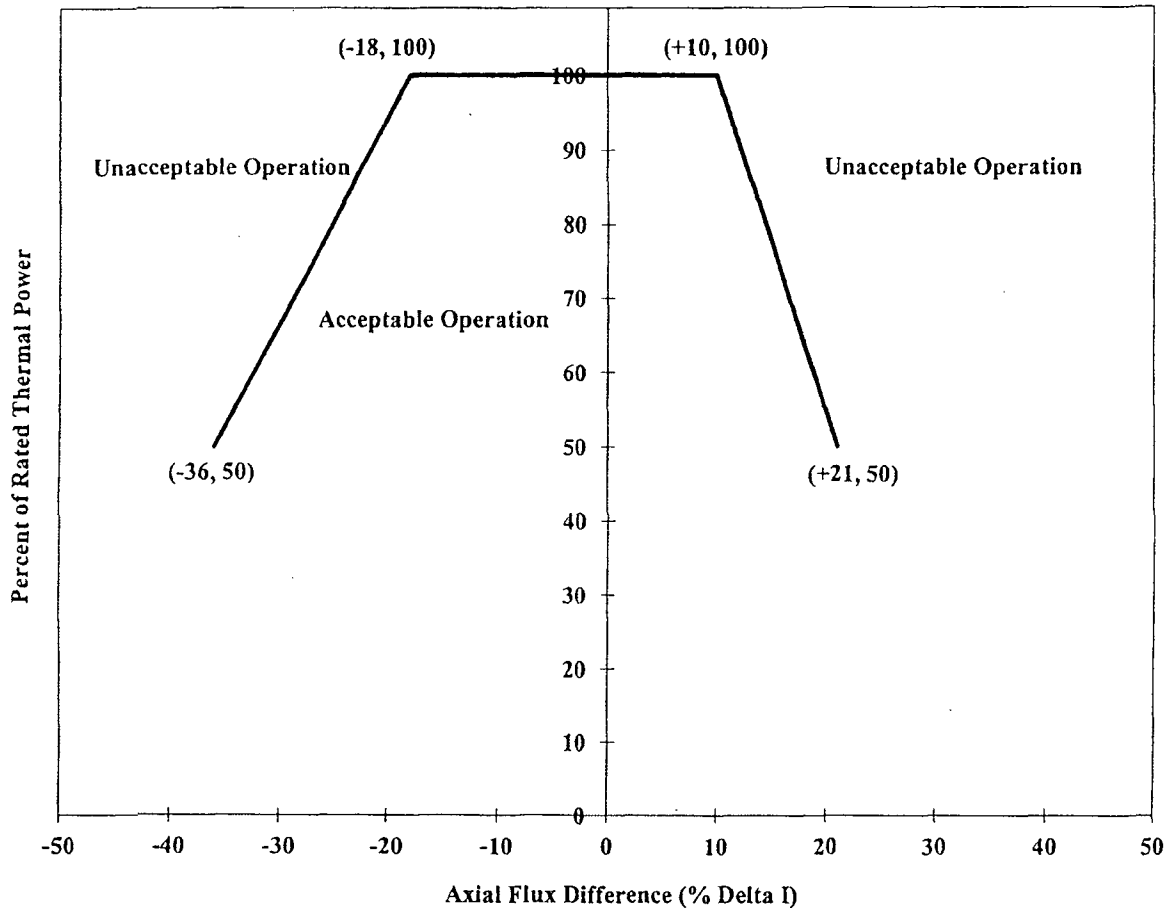
RFA Fuel

Core Ht (ft.)	Axial Peak												
	<u>1.05</u>	<u>1.1</u>	<u>1.2</u>	<u>1.3</u>	<u>1.4</u>	<u>1.5</u>	<u>1.6</u>	<u>1.7</u>	<u>1.8</u>	<u>1.9</u>	<u>2.1</u>	<u>3.0</u>	<u>3.25</u>
0.12	1.809	1.855	1.949	1.995	1.974	2.107	2.050	2.009	1.933	1.863	1.778	1.315	1.246
1.2	1.810	1.854	1.940	1.995	1.974	2.107	2.019	1.978	1.901	1.831	1.785	1.301	1.224
2.4	1.809	1.853	1.931	1.978	1.974	2.074	1.995	1.952	1.876	1.805	1.732	1.463	1.462
3.6	1.810	1.851	1.920	1.964	1.974	2.050	1.966	1.926	1.852	1.786	1.700	1.468	1.387
4.8	1.810	1.851	1.906	1.945	1.974	2.006	1.944	1.923	1.854	1.784	1.671	1.299	1.258
6.0	1.810	1.851	1.892	1.921	1.946	1.934	1.880	1.863	1.802	1.747	1.671	1.329	1.260
7.2	1.807	1.844	1.872	1.893	1.887	1.872	1.809	1.787	1.733	1.681	1.598	1.287	1.220
8.4	1.807	1.832	1.845	1.857	1.816	1.795	1.736	1.709	1.654	1.601	1.513	1.218	1.158
9.6	1.807	1.810	1.809	1.791	1.738	1.718	1.657	1.635	1.581	1.530	1.444	1.143	1.091
10.8	1.798	1.787	1.761	1.716	1.654	1.632	1.574	1.557	1.509	1.462	1.383	1.101	1.047
11.4	1.789	1.765	1.725	1.665	1.606	1.583	1.529	1.510	1.464	1.422	1.346	1.067	1.014

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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/1/A/6100/22 Unit 1 Data Book for 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

<u>Parameter</u>	<u>Value</u>
Nominal T_{avg} at RTP	$T' \leq 585.1^{\circ}\text{F}$
Nominal RCS Operating Pressure	$P' = 2235 \text{ psig}$
Overtemperature ΔT reactor trip setpoint	$K_1 \leq 1.1978$
Overtemperature ΔT reactor trip heatup setpoint penalty coefficient	$K_2 = 0.0334/^{\circ}\text{F}$
Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient	$K_3 = 0.001601/\text{psi}$
Time constants utilized in the lead-lag compensator for ΔT	$\tau_1 \geq 8 \text{ sec.}$ $\tau_2 \leq 3 \text{ sec.}$
Time constant utilized in the lag compensator for ΔT	$\tau_3 \leq 2 \text{ sec.}$
Time constants utilized in the lead-lag compensator for T_{avg}	$\tau_4 \geq 28 \text{ sec.}$ $\tau_5 \leq 4 \text{ sec.}$
Time constant utilized in the measured T_{avg} lag compensator	$\tau_6 \leq 2 \text{ sec.}$
$f_1(\Delta I)$ "positive" breakpoint	$= 19.0 \% \Delta I$
$f_1(\Delta I)$ "negative" breakpoint	$= \text{N/A}^*$
$f_1(\Delta I)$ "positive" slope	$= 1.769 \% \Delta T_0 / \% \Delta I$
$f_1(\Delta I)$ "negative" slope	$= \text{N/A}^*$

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

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2.10.2 Overpower ΔT Setpoint Parameter Values

<u>Parameter</u>	<u>Value</u>
Nominal T_{avg} at RTP	$T'' \leq 585.1^\circ\text{F}$
Overpower ΔT reactor trip setpoint	$K_4 \leq 1.0864$
Overpower ΔT reactor trip Penalty	$K_5 = 0.02/^\circ\text{F}$ for increasing T_{avg} $K_5 = 0.0$ for decreasing T_{avg}
Overpower ΔT reactor trip heatup setpoint penalty coefficient	$K_6 = 0.001179/^\circ\text{F}$ 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	$\tau_3 \leq 2$ sec.
Time constant utilized in the measured T_{avg} lag compensator	$\tau_6 \leq 2$ sec.
Time constant utilized in the rate-lag controller for T_{avg}	$\tau_7 \geq 5$ sec.
$f_2(\Delta I)$ "positive" breakpoint	$= 35.0 \% \Delta I$
$f_2(\Delta I)$ "negative" breakpoint	$= -35.0 \% \Delta I$
$f_2(\Delta I)$ "positive" slope	$= 7.0 \% \Delta T_\theta / \% \Delta I$
$f_2(\Delta I)$ "negative" slope	$= 7.0 \% \Delta T_\theta / \% \Delta I$

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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:

<u>Parameter</u>	<u>Applicable Burnup</u>	<u>Limit</u>
Accumulator minimum boron concentration.	0 - 200 EFPD	2,475 ppm
Accumulator minimum boron concentration.	200.1 - 250 EFPD	2,407 ppm
Accumulator minimum boron concentration.	250.1 - 300 EFPD	2,347 ppm
Accumulator minimum boron concentration.	300.1 - 350 EFPD	2,279 ppm
Accumulator minimum boron concentration.	350.1 - 400 EFPD	2,216 ppm
Accumulator minimum boron concentration.	400.1 - 450 EFPD	2,157 ppm
Accumulator minimum boron concentration.	450.1 - 500 EFPD	2,100 ppm
Accumulator minimum boron concentration.	500.1 - 514 EFPD	2,034 ppm
Accumulator minimum boron concentration.	514.1 - 524 EFPD	2,016 ppm
Accumulator maximum boron concentration.	0 - 524 EFPD	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:

<u>Parameter</u>	<u>Limit</u>
RWST minimum boron concentration.	2,675 ppm
RWST 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	≤ 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			$\geq 390,000$ gpm*

*Note: The RCS minimum coolant flow rate assumed in the licensing analyses for the M1C22 core is 388,000 gpm. However, the flow is set at 390,000 gpm, which is conservative.

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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.

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

<u>Parameter</u>	<u>Limit</u>
Minimum Boron concentration of the Reactor Coolant System, the refueling canal, and the refueling cavity.	2,675 ppm

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2.16 Borated Water Sources – 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 ≤ 300 °F and MODES 5 and 6.

<u>Parameter</u>	<u>Limit</u>
BAT minimum contained borated water volume	10,599 gallons 13.6% Level
Note: When cycle burnup is > 455 EFPD, Figure 6 may be used to determine the required BAT minimum level.	
BAT minimum boron concentration	7,000 ppm
BAT minimum water volume required to maintain SDM at 7,000 ppm	2,300 gallons
RWST minimum contained borated water volume	47,700 gallons 41 inches
RWST minimum boron concentration	2,675 ppm
RWST minimum water volume required to maintain SDM at 2,675 ppm	8,200 gallons

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2.17 Borated Water Sources - 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 temperatures > 300°F.

<u>Parameter</u>	<u>Limit</u>
BAT minimum contained borated water volume	22,049 gallons 38.0% Level
Note: When cycle burnup is > 455 EFPD, Figure 6 may be used to determine the required BAT minimum level.	
BAT minimum boron concentration	7,000 ppm
BAT minimum water volume required to maintain SDM at 7,000 ppm	13,750 gallons
RWST minimum contained borated water volume	96,607 gallons 103.6 inches
RWST minimum boron concentration	2,675 ppm
RWST maximum boron concentration (TS 3.5.4)	2875 ppm
RWST minimum water volume required to maintain SDM at 2,675 ppm	57,107 gallons

2.18 Standby Shutdown System - (SLC-16.9.7)

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

<u>Parameter</u>	<u>Limit</u>
Spent fuel pool minimum boron concentration for TR 16.9.7.2.	2,675 ppm

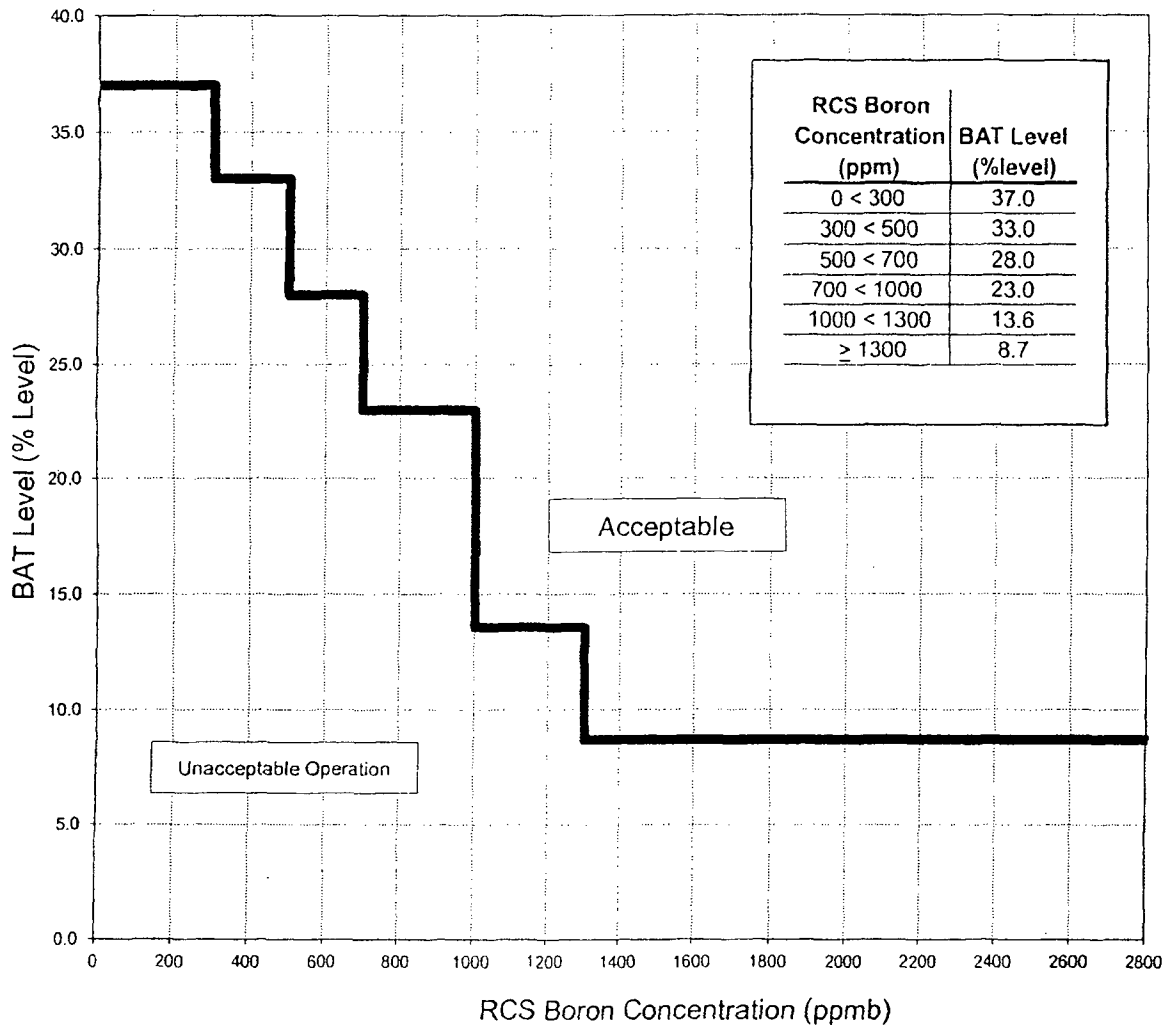
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Figure 6

Boric Acid Storage Tank Indicated Level Versus RCS Boron Concentration

(Valid When Cycle Burnup is > 455 EFPD)

This figure includes additional volumes listed in SLC 16.9.14 and 16.9.11



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NOTE: Appendix A contains power distribution monitoring factors used in Technical Specification Surveillance. This data was generated in the McGuire 1 Cycle 22 Maneuvering Analysis calculation file, MCC-1553.05-00-0540. 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 Plant Nuclear Engineering Section will control this information via computer file(s) 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, upon request.