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U. S. Nuclear Regulatory Commission
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Subject: Duke Energy Carolinas, LLC
McGuire Nuclear Station
Docket No. 50-370
Unit 2, Cycle 23, Revision 1
Core Operating Limits Report

Pursuant to McGuire Technical Specification (TS) 5.6.5.d, please find enclosed the McGuire Unit 2 Cycle 23, Revision 1, Core Operating Limits Report (COLR).

Questions regarding this submittal should be directed to Kay Crane, Regulatory Affairs at (980) 875-4306.

A handwritten signature in black ink, appearing to read 'Steven D. Capps', with the word 'for' written below it.

Steven D. Capps

Attachment

A001
NRK

U. S. Nuclear Regulatory Commission
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cc: Mr. G.E. Miller, Project Manager
U.S. Nuclear Regulatory Commission
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Mr. Victor M. McCree
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Senior Resident Inspector
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McGuire Unit 2 Cycle 23**Core Operating Limits Report
Revision 1****May 2015**

Calculation Number: MCC-1553.05-00-0595, Revision 1

Duke Energy Carolinas, LLC

		Date
Prepared By:	C. L. Klein <i>C. L. Klein</i>	<u>5/21/15</u>
Checked By:	N. D. Stehle <i>N. D. Stehle</i>	<u>5/21/15</u>
Checked By:	T. P. Phelps <i>T. P. Phelps</i> (Sections 2.1 and 2.9 – 2.17)	<u>5/21/15</u>
Approved By:	M.A. Blom <i>M.A. Blom</i>	<u>5/21/15</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 2 Cycle 23 Core Operating Limits Report

Implementation Instructions For Revision 1

Revision Description and PIP Tracking

Revision 1 of the McGuire Unit 2 Cycle 23 COLR contains limits specific to the reload core and is revised to include a conditional exemption for the MTC measurement described in SR 3.1.3.2. The power distribution monitoring factors from Appendix A of Revision 0 remain valid and are not transmitted as part of Revision 1.

This revision supports CA#4 of PIP M-15-02948.

Implementation Schedule

Revision 1 may become effective upon receipt. The McGuire Unit 2 Cycle 23 COLR will cease to be effective during No MODE between cycles 23 and 24.

Data files to be Implemented

No data files are transmitted as part of this document.

Engineering Instruction Inspection Waiver

Per EDM-130 "Engineering Drawings", the Engineering Instruction (EI) has been waived per Reference "MC-1438.88".

McGuire 2 Cycle 23 Core Operating Limits Report**REVISION LOG**

<u>Revision</u>	<u>Effective Date</u>	<u>Pages Affected</u>	<u>COLR</u>
0	February 2014	1-32, Appendix A*	M2C23 COLR, Rev. 0
1	May 2015	1-31	M2C23 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.

McGuire 2 Cycle 23 Core Operating Limits Report

1.0 Core Operating Limits Report

This Core Operating Limits Report (COLR) has been prepared in accordance with the requirements of Technical Specification 5.6.5. The Technical Specifications that reference this report are listed below along with the NRC approved analytical methods used to develop and/or determine COLR parameters in Technical Specifications.

TS Number	Technical Specifications	COLR Parameter	COLR Section	NRC Approved Methodology (Section 1.1 Number)
2.1.1	Reactor Core Safety Limits	RCS Temperature and Pressure Safety Limits	2.1	6,7,8,9,10,12,15,16
3.1.1	Shutdown Margin	Shutdown Margin	2.2	6,7,8,12,14,15,16
3.1.3	Moderator Temperature Coefficient	MTC	2.3	6,7,8,12,14,16, 17
3.1.4	Rod Group Alignment Limits	Shutdown Margin	2.2	6,7,8,12,14,15,16
3.1.5	Shutdown Bank Insertion Limits	Shutdown Margin	2.2	6,7,8,12,14,15,16
3.1.5	Shutdown Bank Insertion Limits	Shutdown Bank Insertion Limit	2.4	2,4,6,7,8,9,10,12,14,15, 16
3.1.6	Control Bank Insertion Limits	Shutdown Margin	2.2	6,7,8,12,14,15,16
3.1.6	Control Bank Insertion Limits	Control Bank Insertion Limit	2.5	2,4,6,7,8,9,10,12,14,15, 16
3.1.8	Physics Tests Exceptions	Shutdown Margin	2.2	6,7,8,12,14,15,16
3.2.1	Heat Flux Hot Channel Factor	Fq, AFD, OTΔT and Penalty Factors	2.6	2,4,6,7,8,9,10,12,15,16
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor	FΔH, AFD and Penalty Factors	2.7	2,4,6,7,8,9,10,12,15,16
3.2.3	Axial Flux Difference	AFD	2.8	2,4,6,7,8,15,16
3.3.1	Reactor Trip System Instrumentation Setpoints	OTΔT and OPΔT Constants	2.9	6,7,8,9,10,12,15,16
3.4.1	RCS Pressure, Temperature, and Flow DNB limits	RCS Pressure, Temperature and Flow	2.10	6,7,8,9,10,12
3.5.1	Accumulators	Max and Min Boron Conc.	2.11	6,7,8,12,14,16
3.5.4	Refueling Water Storage Tank	Max and Min Boron Conc.	2.12	6,7,8,12,14,16
3.7.14	Spent Fuel Pool Boron Concentration	Min Boron Concentration	2.13	6,7,8,12,14,16
3.9.1	Refueling Operations – Boron Concentration	Min Boron Concentration	2.14	6,7,8,12,14,16
5.6.5	Core Operating Limits Report (COLR)	Analytical Methods	1.1	None

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

SLC Number	Selected Licensing Commitment	COLR Parameter	COLR Section	NRC Approved Methodology (Section 1.1 Number)
16.9.14	Borated Water Source – Shutdown	Borated Water Volume and Conc. for BAT/RWST	2.15	6,7,8,12,14,16
16.9.11	Borated Water Source – Operating	Borated Water Volume and Conc. for BAT/RWST	2.16	6,7,8,12,14,16
16.9.7	Standby Shutdown System	Standby Makeup Pump Water Supply	2.17	6,7,8,12,14,16

McGuire 2 Cycle 23 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

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

Revision 0

Report Date: August 1985

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

Revision 1

July 1997

3. WCAP-10266-P-A, "The 1981 Version Of Westinghouse Evaluation Model Using BASH Code", (W Proprietary).

Revision 2

Report Date: March 1987

Not Used

4. WCAP-12945-P-A, Volume 1 and Volumes 2-5, "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," (W Proprietary).

Revision: Volume 1 (Revision 2) and Volumes 2-5 (Revision 1)

Report Date: March 1998

5. BAW-10168P-A, "B&W Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," (B&W Proprietary).

Revision 1

SER Date: January 22, 1991

Revision 2

SER Dates: August 22, 1996 and November 26, 1996

Revision 3

SER Date: June 15, 1994

Not Used

McGuire 2 Cycle 23 Core Operating Limits Report

1.1 Analytical Methods (continued)

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

Revision 5a

Report Date: October 2012

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

Revision 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 and GDTACO," (DPC Proprietary).

Revision 2

Report Date: August 2012

Not Used

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

McGuire 2 Cycle 23 Core Operating Limits Report

1.1 Analytical Methods (continued)

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

Revision 2a

Report Date: December 2009

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

Revision 1a

Report Date: June 2009

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

Revision 1

Report Date: November 12, 2008

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

Revision 0

Report Date: April 2015

McGuire 2 Cycle 23 Core Operating Limits Report

2.0 Operating Limits

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 Reactor Core Safety Limits (TS 2.1.1)

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 and TS 3.1.8)

2.2.1 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.

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

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

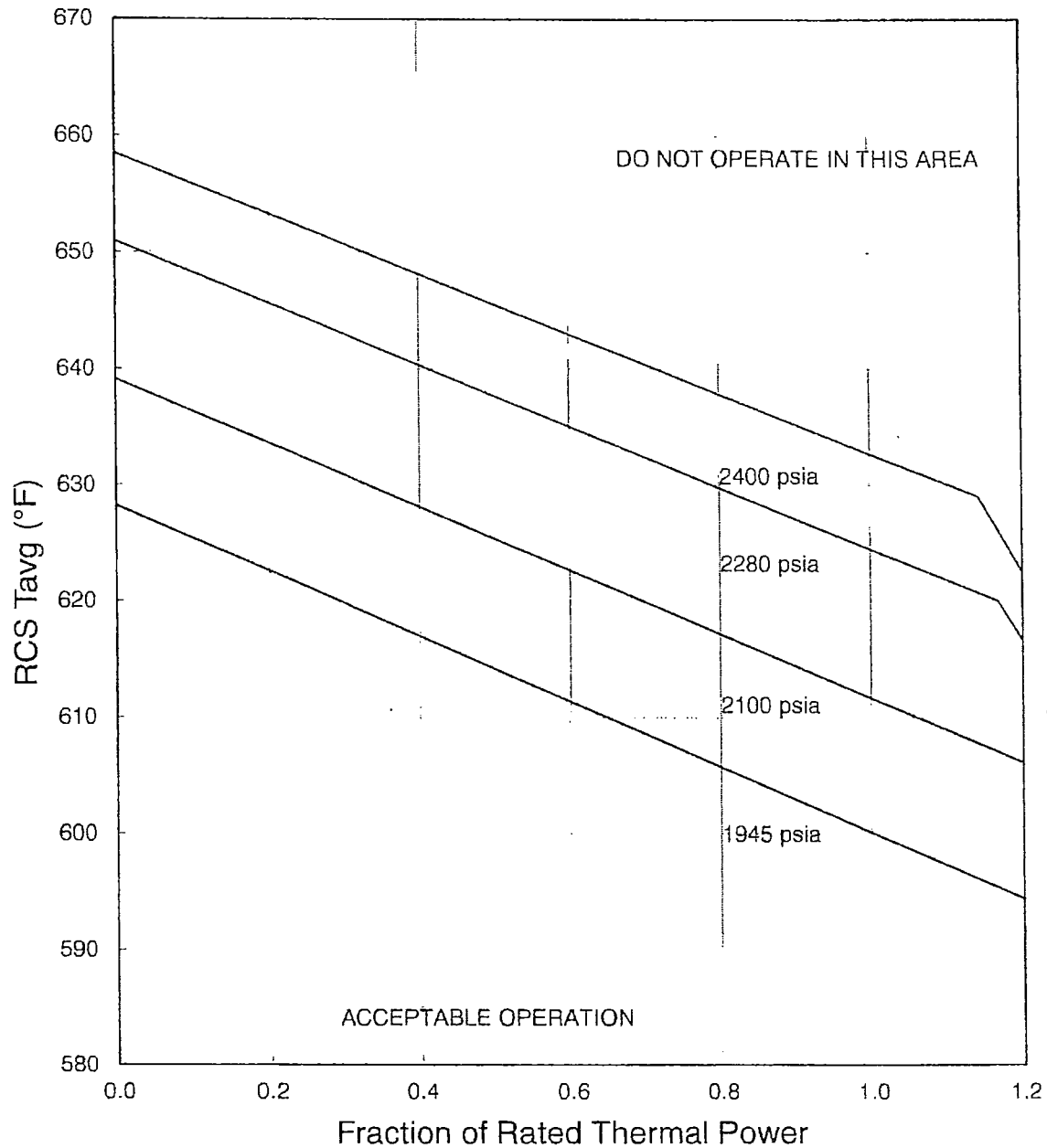
2.2.4 For TS 3.1.5, SDM shall be $\geq 1.3\% \Delta K/K$ in MODE 1 and MODE 2 with any control bank not fully inserted.

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

2.2.6 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
3469 MWth Up-rated Thermal Power



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2.3 Moderator Temperature Coefficient - MTC (TS 3.1.3)

2.3.1 The Moderator Temperature Coefficient (MTC) Limits are:

MTC shall be less positive than the upper limits shown in Figure 2. BOC, ARO, HZP MTC shall be less positive than $0.7\text{E-}04 \Delta\text{K/K/}^\circ\text{F}$.

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

2.3.2 300 PPM MTC Surveillance Limit is:

Measured 300 PPM ARO, equilibrium RTP MTC shall be less negative than or equal to $-3.65\text{E-}04 \Delta\text{K/K/}^\circ\text{F}$.

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

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

2.3.4 60 PPM MTC Surveillance Limit is:

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

Where:

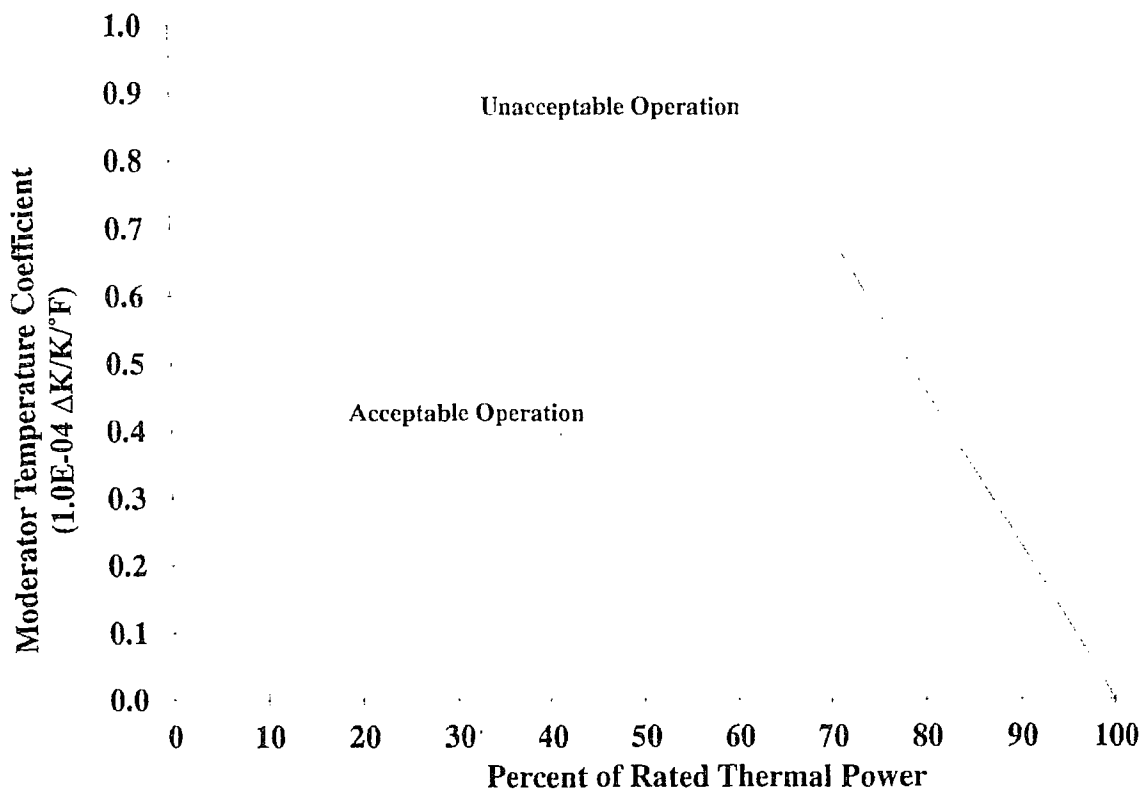
- BOC = Beginning of Cycle (burnup corresponding to the most positive MTC)
- EOC = End of Cycle
- ARO = All Rods Out
- HZP = Hot Zero Power
- RTP = Rated Thermal Power
- PPM = Parts per million (Boron)

2.4 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.

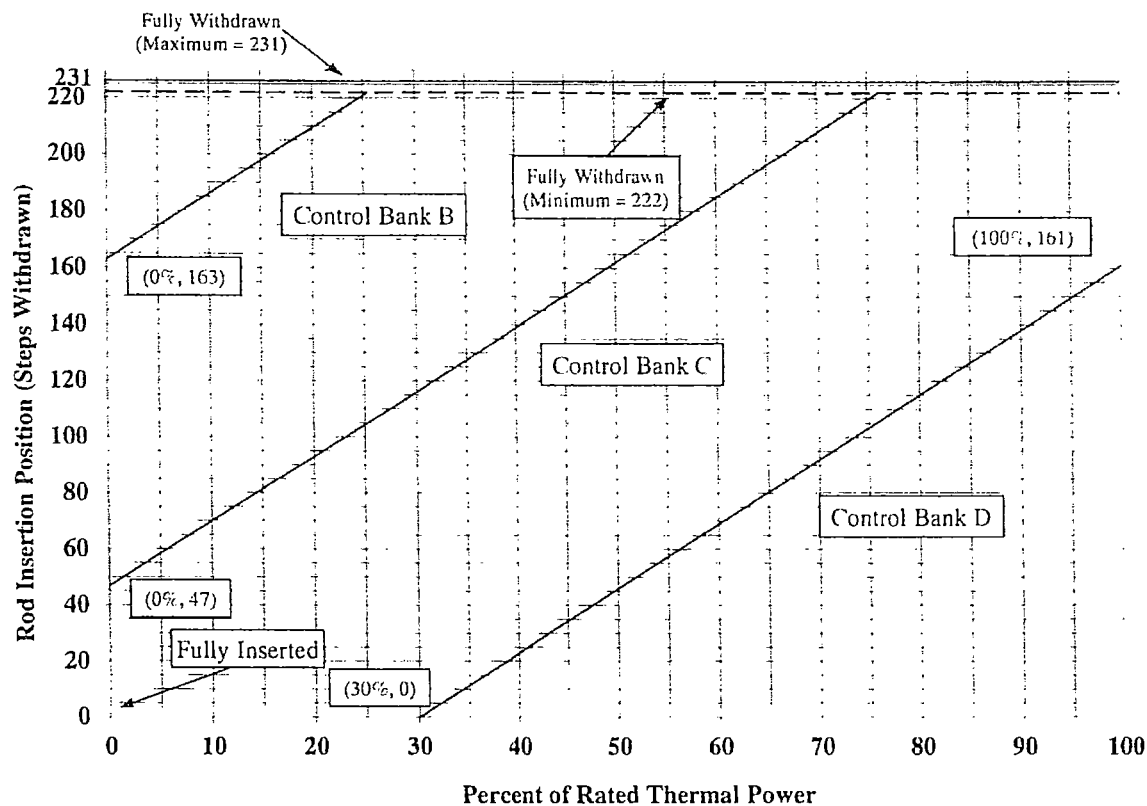
McGuire 2 Cycle 23 Core Operating Limits Report**Figure 2****Moderator Temperature Coefficient Upper Limit Versus Power Level**

NOTE: Compliance with Technical Specification 3.1.3 may require rod withdrawal limits.
Refer to 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



The Rod Insertion Limits (RIL) for Control Bank D (CD), Control Bank C (CC), and Control Bank B (CB) can be calculated by:

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

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

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

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

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

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

McGuire 2 Cycle 23 Core Operating Limits Report

2.6 Heat Flux Hot Channel Factor - $F_Q(X,Y,Z)$ (TS 3.2.1)

2.6.1 $F_Q(X,Y,Z)$ steady-state limits are defined by the following relationships:

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

where,

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

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 the LCO limits. The manufacturing tolerance and measurement uncertainty are implicitly included in the F_Q surveillance limits as defined in Sections 2.6.5 and 2.6.6.

2.6.2 $F_Q^{RTP} = 2.70 \times K(\text{BU})$

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

2.6.4 $K(\text{BU})$ is the normalized $F_Q(X,Y,Z)$ as a function of burnup. F_Q^{RTP} with the $K(\text{BU})$ penalty for Westinghouse RFA fuel is analytically confirmed in cycle-specific reload calculations. $K(\text{BU})$ is set to 1.0 at all burnups.

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

2.6.5 $F_Q^L(X,Y,Z)^{OP} = \frac{F_Q^D(X,Y,Z) * M_Q(X,Y,Z)}{UMT * MT * TILT}$

McGuire 2 Cycle 23 Core Operating Limits Report

where:

$F_Q^L(X,Y,Z)^{OP}$ = Cycle dependent maximum allowable design peaking factor that ensures $F_Q(X,Y,Z)$ 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 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 = 1.03)

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

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

where:

$F_Q^L(X,Y,Z)^{RPS}$ = Cycle dependent maximum allowable design peaking factor that ensures 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 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)

2.6.7 KSLOPE = 0.0725

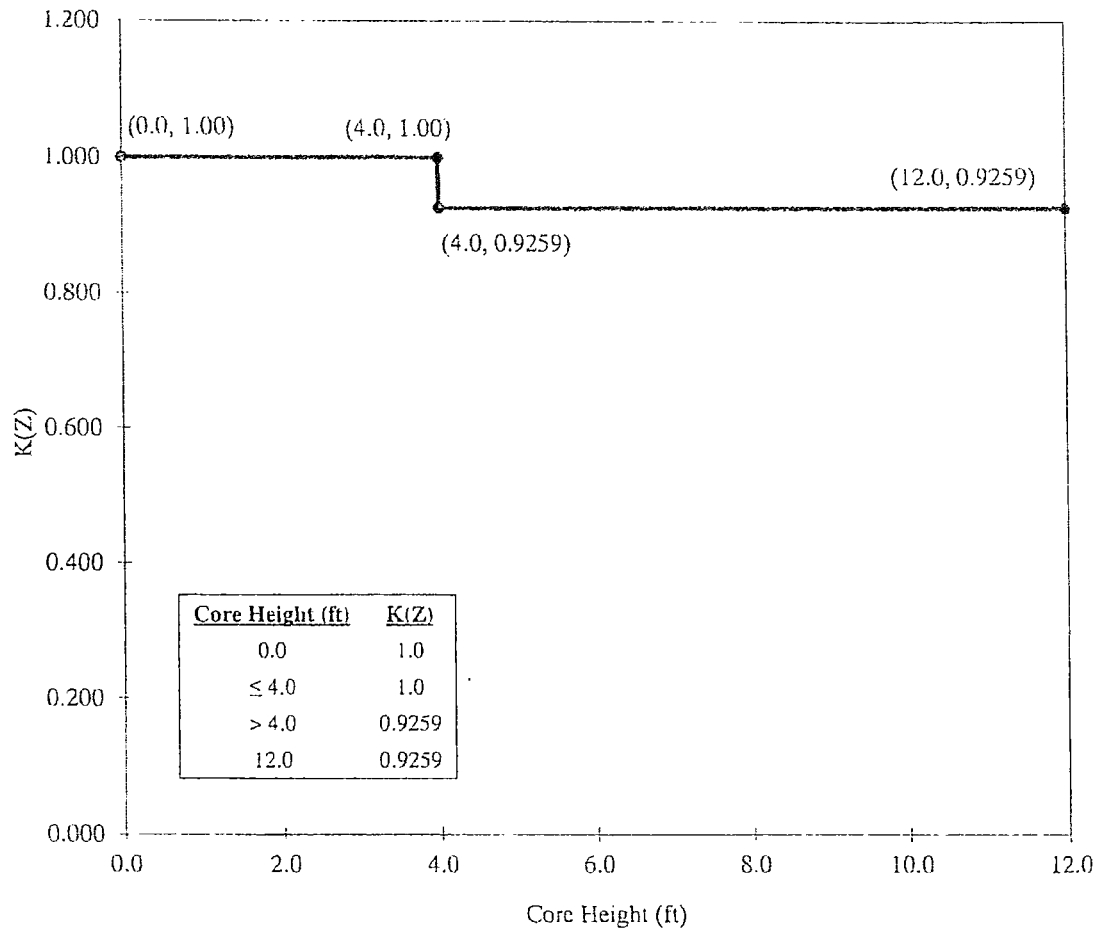
where:

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

2.6.8 $F_O(X,Y,Z)$ penalty factors for Technical Specification Surveillances 3.2.1.2 and 3.2.1.3 are provided in Table 2.

McGuire 2 Cycle 23 Core Operating Limits Report

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 (%)	$F_{\Delta H}(X,Y)$ Penalty Factor (%)
0	2.00	2.00
4	2.00	2.00
12	2.00	2.00
25	2.00	2.00
50	2.73	2.00
75	2.06	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
495	2.00	2.00
505	2.00	2.00
515	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.

McGuire 2 Cycle 23 Core Operating Limits Report

2.7 Nuclear Enthalpy Rise Hot Channel Factor - $F_{\Delta H}(X,Y)$ (TS 3.2.2)

$F_{\Delta H}$ steady-state limits referred to in Technical Specification 3.2.2 is defined by the following relationship.

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

where:

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

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

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

RRH = Thermal Power reduction required to compensate for each 1% that the measured radial peak, $F_{\Delta H}^M(X, Y)$, exceeds its limit. RRH also is used to scale the MARP limits as a function of power per the $F_{\Delta H}^L(X, Y)^{LCO}$ equation. ($\text{RRH} = 3.34$ ($0.0 < P \leq 1.0$))

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

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

where:

$F_{\Delta H}^L(X, Y)^{SURV}$ = Cycle dependent maximum allowable design peaking factor that ensures the $F_{\Delta H}(X, Y)$ limit will be preserved for operation within the LCO limits. $F_{\Delta H}^L(X, Y)^{SURV}$ includes allowances for calculation/measurement uncertainty.

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$F_{\Delta H}^D(X,Y)$ = Design radial power distribution for $F_{\Delta H}^D$. $F_{\Delta H}^D(X,Y)$ 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).

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

2.7.5 $F_{\Delta H}(X,Y)$ penalty factors for Technical Specification Surveillance 3.2.2.2 are provided in Table 2.

2.8 Axial Flux Difference – AFD (TS 3.2.3)

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

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Table 3
Maximum Allowable Radial Peaks (MARPS)

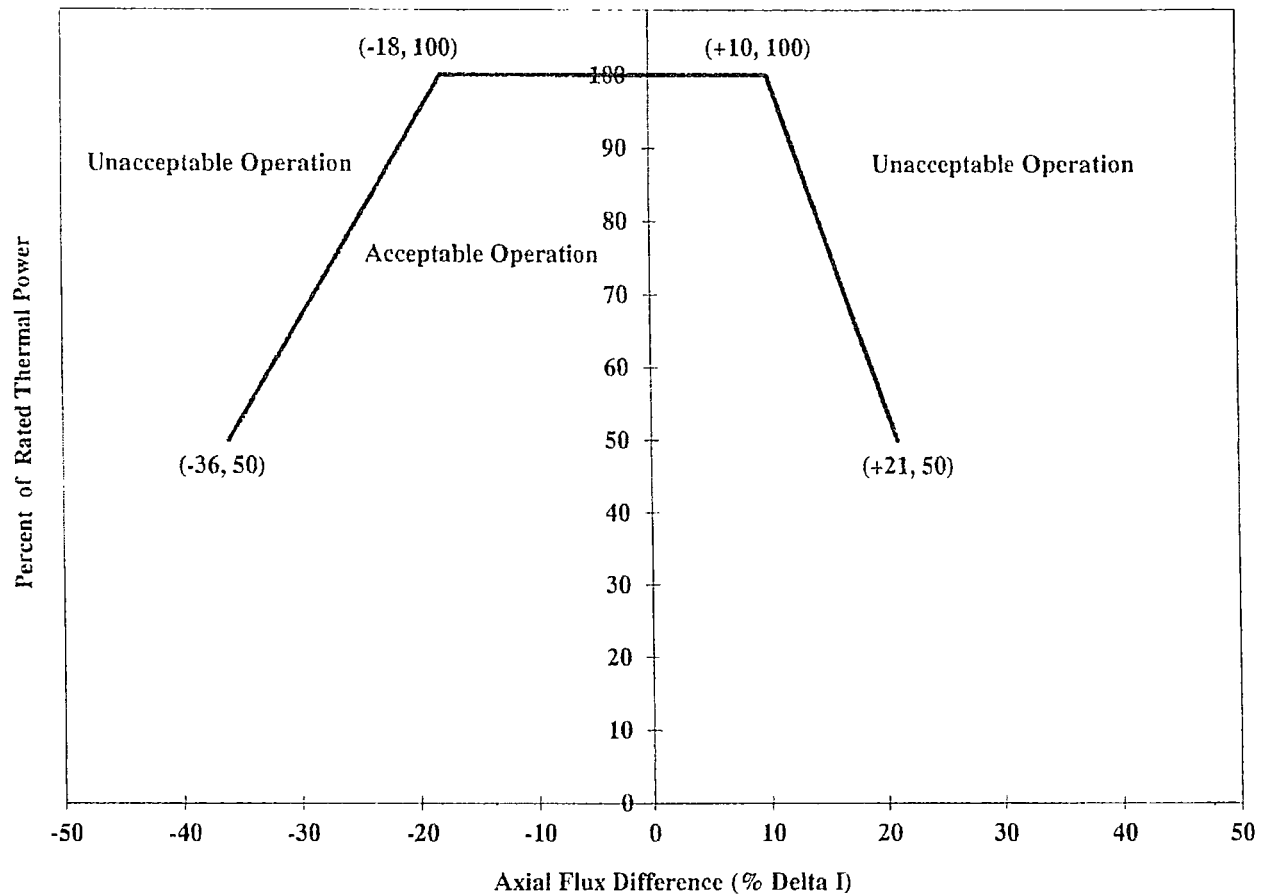
RFA MARPS

Core Ht (ft.)	Axial Peak												
	<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/2/A/6100/22 Unit 2 Data Book for more details.

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

2.9.1 Overtemperature ΔT Setpoint Parameter Values

<u>Parameter</u>	<u>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 the $f_1(\Delta I)$ "negative" slope 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 the OT ΔT $f_1(\Delta I)$ negative breakpoint and slope unnecessary.

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2.9.2 Overpower ΔT Setpoint Parameter Values

<u>Parameter</u>	<u>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	$\tau_1 \geq 8 \text{ sec.}$ $\tau_2 \leq 3 \text{ sec.}$
Time constant utilized in the lag compensator for ΔT	$\tau_3 \leq 2 \text{ sec.}$
Time constant utilized in the measured T_{avg} lag compensator	$\tau_6 \leq 2 \text{ sec.}$
Time constant utilized in the rate-lag controller for T_{avg}	$\tau_7 \geq 5 \text{ sec.}$
$f_2(\Delta I)$ "positive" breakpoint	$= 35.0 \% \Delta I$
$f_2(\Delta I)$ "negative" breakpoint	$= -35.0 \% \Delta I$
$f_2(\Delta I)$ "positive" slope	$= 7.0 \% \Delta T_0 / \% \Delta I$
$f_2(\Delta I)$ "negative" slope	$= 7.0 \% \Delta T_0 / \% \Delta I$

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

2.10.1 RCS pressure, temperature and flow limits for DNB are shown in Table 4.

2.11 Accumulators (TS 3.5.1)

2.11.1 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	2,475 ppm
Accumulator minimum boron concentration.	250.1 - 300 EFPD	2,389 ppm
Accumulator minimum boron concentration.	300.1 - 350 EFPD	2,282 ppm
Accumulator minimum boron concentration.	350.1 - 400 EFPD	2,208 ppm
Accumulator minimum boron concentration.	400.1 - 450 EFPD	2,140 ppm
Accumulator minimum boron concentration.	450.1 - 505 EFPD	2,072 ppm
Accumulator minimum boron concentration.	505.1 - 515 EFPD	1,994 ppm
Accumulator maximum boron concentration.	0 - 515 EFPD	2,875 ppm

2.12 Refueling Water Storage Tank - RWST (TS 3.5.4)

2.12.1 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

McGuire 2 Cycle 23 Core Operating Limits Report**Table 4****Reactor Coolant System DNB Parameters**

Parameter	Indication	No. Operable Channels	Limits
1. Indicated RCS Average Temperature	meter	4	$\leq 587.2^{\circ}\text{F}$
	meter	3	$\leq 586.9^{\circ}\text{F}$
	computer	4	$\leq 587.7^{\circ}\text{F}$
	computer	3	$\leq 587.5^{\circ}\text{F}$
2. Indicated Pressurizer Pressure	meter	4	$\geq 2219.8 \text{ psig}$
	meter	3	$\geq 2222.1 \text{ psig}$
	computer	4	$\geq 2215.8 \text{ psig}$
	computer	3	$\geq 2217.5 \text{ psig}$
3. RCS Total Flow Rate			$\geq 388,000 \text{ gpm}$

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2.13 Spent Fuel Pool Boron Concentration (TS 3.7.14)

2.13.1 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

2.14 Refueling Operations - Boron Concentration (TS 3.9.1)

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

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

McGuire 2 Cycle 23 Core Operating Limits Report**2.15 Borated Water Source – Shutdown (SLC 16.9.14)**

2.15.1 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 > 460 EFPD, Figure 6 may be used to determine required BAT minimum level.
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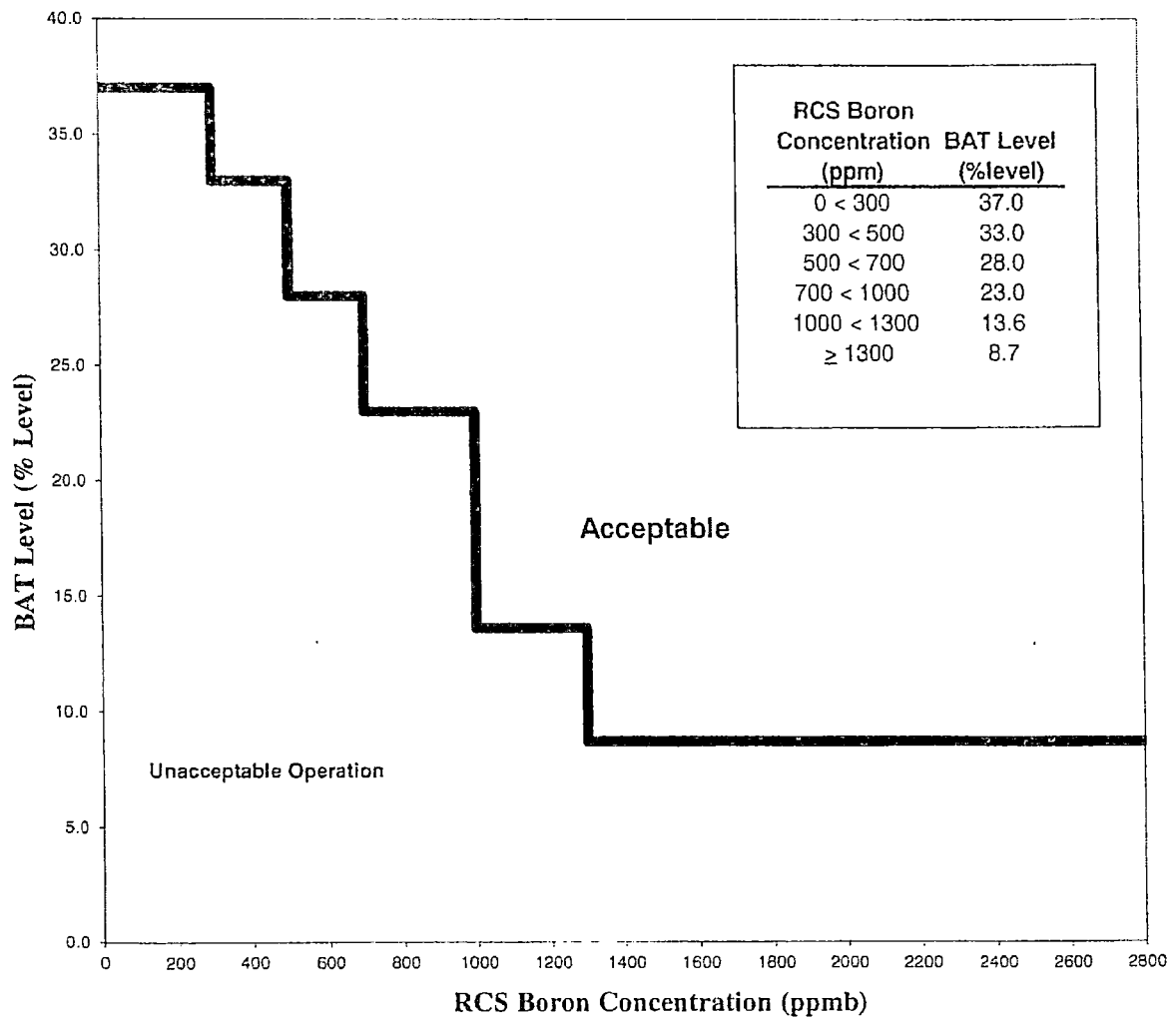
BAT minimum boron concentration	7,000 ppm
BAT minimum water volume required to maintain SDM at 7,000 ppm	2,300 gallons
RWST minimum contained borated water volume	47,700 gallons 41 inches
RWST minimum boron concentration	2,675 ppm
RWST minimum water volume required to maintain SDM at 2,675 ppm	8,200 gallons

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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: Appendix A contains power distribution monitoring factors used in Technical Specification Surveillance. This data was generated in the McGuire 2 Cycle 23 Maneuvering Analysis calculation file, MCC-1553.05-00-0590. 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.