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

Subject: Duke Energy Carolinas, LLC
McGuire Nuclear Station
Docket No. 50-369
Unit 1, Cycle 25, Revision 0
Core Operating Limits Report

Pursuant to McGuire Technical Specification 5.6.5.d, please find enclosed the McGuire Unit 1 Cycle 25, Revision 0, Core Operating Limits Report (COLR).

Questions regarding this submittal should be directed to P.T. Vu, Regulatory Affairs at (980) 875-4302.

SD Capps

Steven D. Capps

Attachment

A001
NRR

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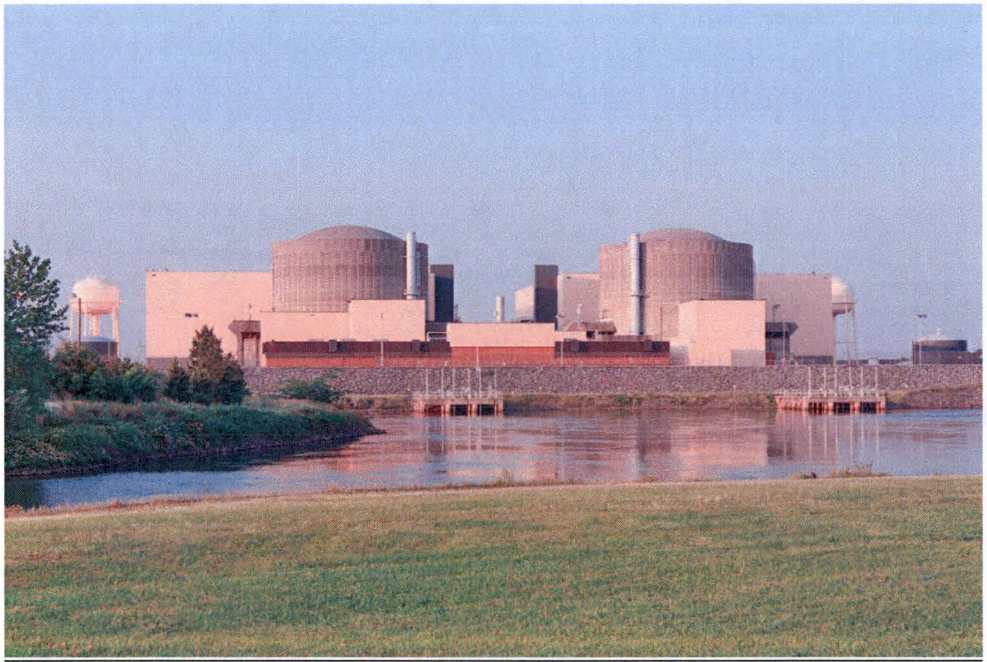
McGuire Unit 1 Cycle 25

Core Operating Limits Report
Revision 0

March 2016

Calculation Number: MCC-1553.05-00-0627



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 25 Core Operating Limits Report

Implementation Instructions for Revision 0

Revision Description and CR Tracking

Revision 0 of the McGuire Unit 1 Cycle 25 COLR contains limits specific to the reload core.

This is no CR associated with this revision.

Implementation Schedule

The McGuire Unit 1 Cycle 25 COLR requires the reload 50.59 to be approved prior to implementation and fuel loading.

Revision 0 may become effective any time during No MODE between Cycle 24 and 25 but must become effective prior to entering MODE 6 which starts Cycle 25.

The McGuire Unit 1 Cycle 25 COLR will cease to be effective during No MODE between Cycles 25 and 26.

Data files to be Implemented

No data files are transmitted as part of this document.

McGuire 1 Cycle 25 Core Operating Limits Report

REVISION LOG

Revision	Effective Date	Pages Affected	COLR
0	March 2016	1-31, Appendix A*	M1C25 COLR, Rev. 0

*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 1 Cycle 25 Core Operating Limits Report

1.0 Core Operating Limits Report

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

TS Section	Technical Specifications	COLR Parameter	COLR Section	NRC Approved Methodology (Section 1.1 Number)
2.1.1	Reactor Core Safety Limits	RCS Temperature and Pressure Safety Limits	2.1	6, 7, 8, 9, 10, 12, 15, 16
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 Limit	Shutdown Margin Shutdown Bank Insertion Limits	2.2 2.4	2, 4, 6, 7, 8, 9, 10, 12, 14, 15, 16
3.1.6	Control Bank Insertion Limit	Shutdown Margin Control Bank Insertion Limits	2.2 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	F_Q AFD OTΔT Penalty Factors	2.6 2.8 2.9 2.6	2, 4, 6, 7, 8, 9, 10, 12, 15, 16
3.2.2	Nuclear Enthalpy Rise Hot Channel Factor	FΔH Penalty Factors	2.7 2.7	2, 4, 6, 7, 8, 9, 10, 12, 15, 16
3.2.3	Axial Flux Difference	AFD	2.8	2, 4, 6, 7, 8, 15, 16
3.3.1	Reactor Trip System Instrumentation	OTΔT OPΔT	2.9 2.9	6, 7, 8, 9, 10, 12 15, 16
3.4.1	RCS Pressure, Temperature and Flow Limits for DNB	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 Section	Selected Licensee Commitment	COLR Parameter	COLR Section	NRC Approved Methodology (Section 1.1 Number)
16.9.14	Boration Systems – Borated Water Source – Shutdown	Borated Water Volume and Conc. for BAT/RWST	2.15	6, 7, 8, 12, 14, 16
16.9.11	Boration Systems – 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 1 Cycle 25 Core Operating Limits Report

1.1 Analytical Methods

Analytical methods used to determine core operating limits for parameters identified in Technical Specifications and previously reviewed and approved by the NRC as specified in Technical Specification 5.6.5 are as follows.

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," (W Proprietary).

   Revision 0
   Report Date: July 1985
   Not Used

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

   Revision 0
   Report Date: August 1985

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

   Revision 1
   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 1 Cycle 25 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" (DPC Proprietary).

    Revision 0
    Report Date: April 1995
    Not Used

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

    Revision 3a
    Report Date: September 2011

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

    Revision 1a
    Report Date: January 2009
    Not Used

McGuire 1 Cycle 25 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 1 Cycle 25 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 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, TS 3.1.8)

2.2.1 For TS 3.1.1, SDM shall be greater than or equal to 1.3% $\Delta K/K$ in MODE 2 with Keff < 1.0 and in MODES 3 and 4.

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

2.2.3 For TS 3.1.4, SDM shall be greater than or equal to 1.3% $\Delta K/K$ in MODE 1 and MODE 2.

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

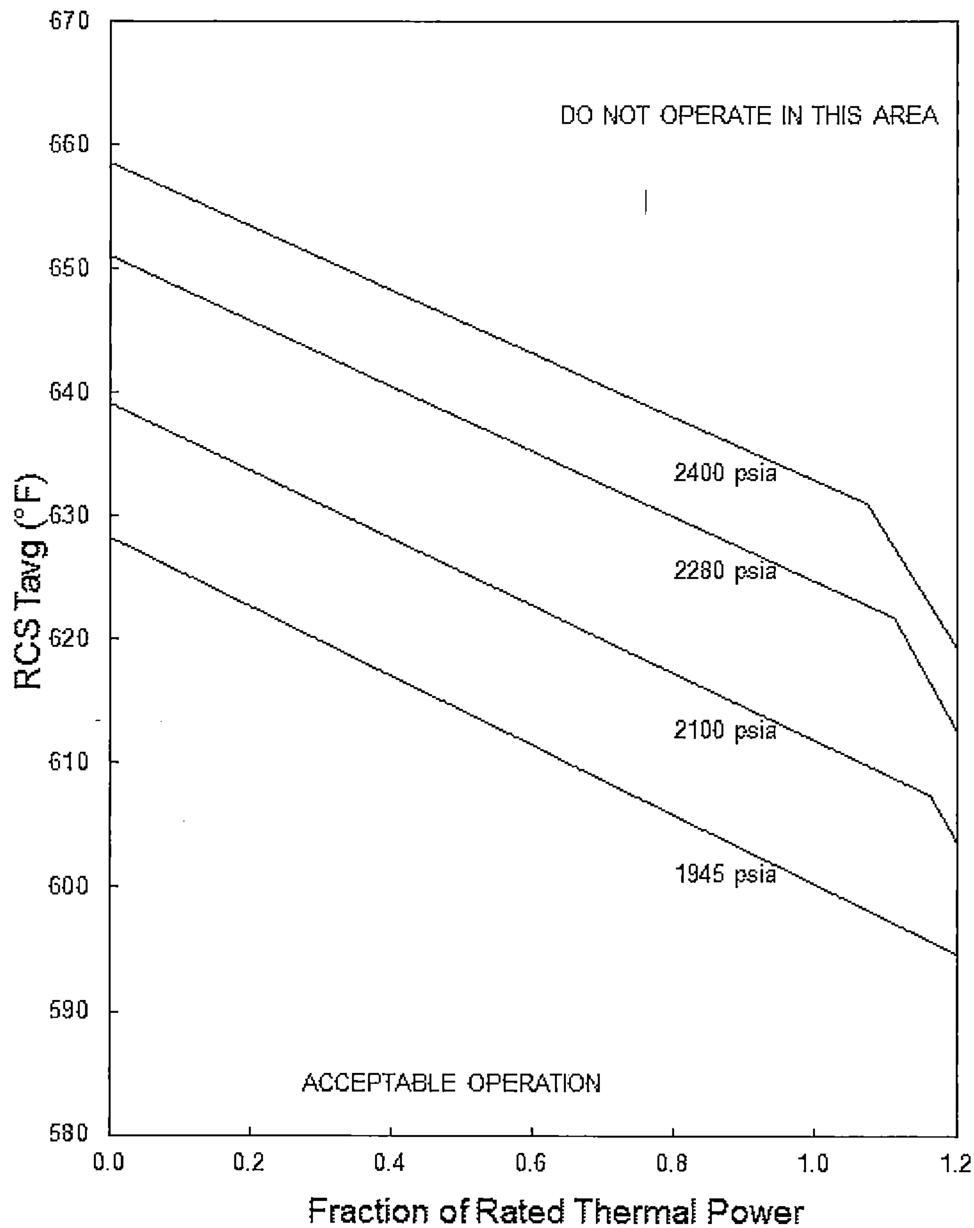
2.2.5 For TS 3.1.6, SDM shall be greater than or equal to 1.3% $\Delta K/K$ in MODE 1 and MODE 2 with Keff $\geq$ 1.0.

2.2.6 For TS 3.1.8, SDM shall be greater than or equal to 1.3% $\Delta K/K$ in MODE 2 during PHYSICS TESTS.

McGuire 1 Cycle 25 Core Operating Limits Report

Figure 1

Reactor Core Safety Limits
Four Loops in Operation



McGuire 1 Cycle 25 Core Operating Limits Report

2.3 Moderator Temperature Coefficient - MTC (TS 3.1.3)

2.3.1 Moderator Temperature Coefficient (MTC) Limits are:

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

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

2.3.2 300 ppm MTC Surveillance Limit is:

Measured 300 PPM ARO, equilibrium RTP MTC shall be less negative than or equal to -3.65E-04 ΔK/K/°F.

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

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

2.3.4 60 PPM MTC Surveillance Limit is:

Measured 60 PPM ARO, equilibrium RTP MTC shall be less negative than or equal to -4.125E-04 ΔK/K/°F.

Where:
- BOC = Beginning of Cycle (burnup corresponding to most positive MTC)
- EOC = End of Cycle
- ARO = All Rods Out
- HZP = Hot Zero 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.

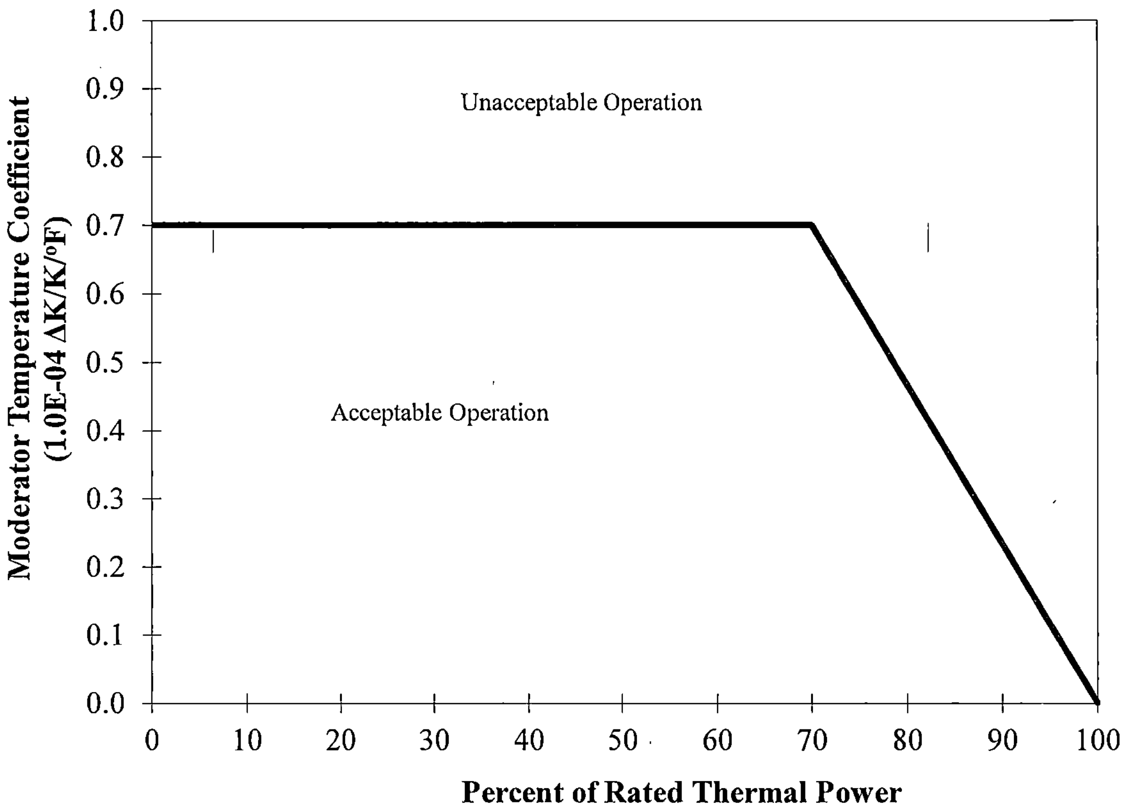
2.5 Control Bank Insertion Limits (TS 3.1.6)

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 1 Cycle 25 Core Operating Limits Report

Figure 2

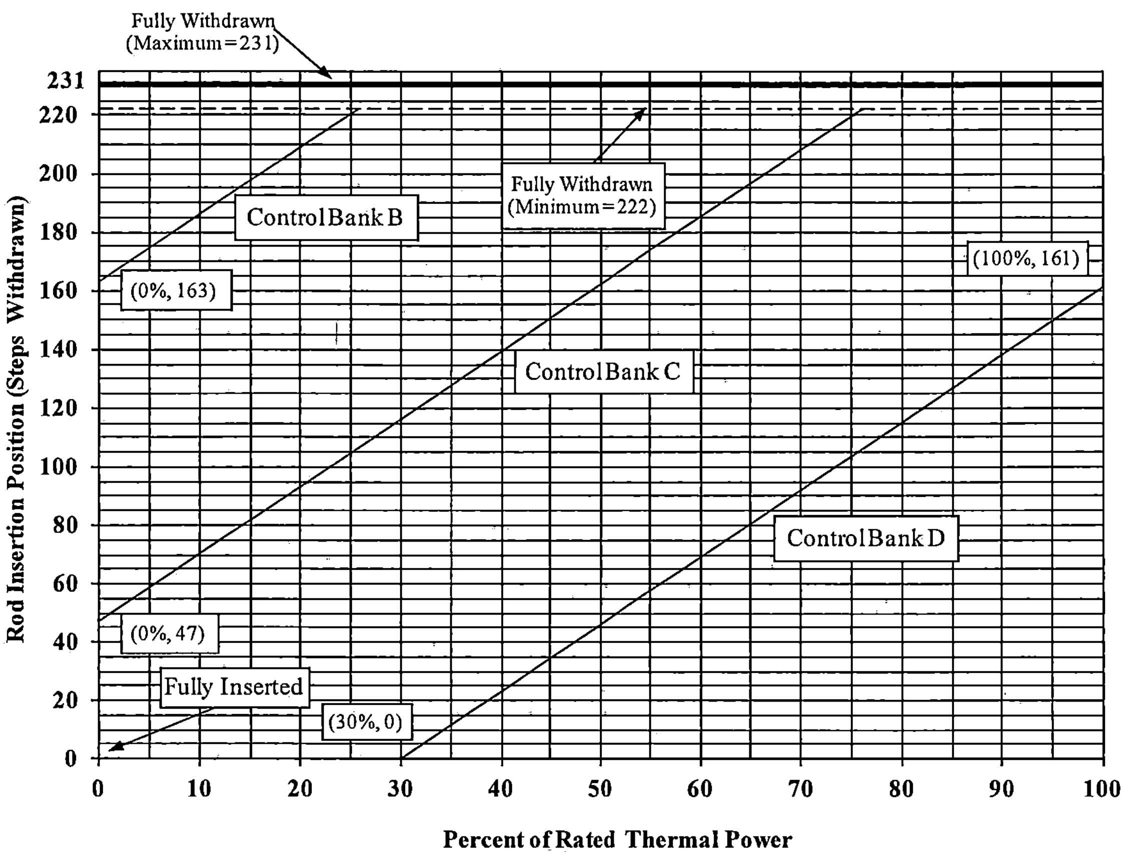
Moderator Temperature Coefficient Upper Limit Versus Power Level



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.

McGuire 1 Cycle 25 Core Operating Limits Report

Figure 3
Control Bank Insertion Limits Versus Percent Rated Thermal Power



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

Bank CD RIL = 2.3(P) – 69 $\{30 \le P \le 100\}$
Bank CC RIL = 2.3(P) +47 $\{0 \le P \le 76.1\}$ *for CC RIL = 222* $\{76.1 < P \le 100\}$
Bank CB RIL = 2.3(P) +163 $\{0 \le P \le 25.7\}$ *for CB RIL = 222* $\{25.7 < P \le 100\}$

where P = %Rated Thermal Power

NOTES: 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.

McGuire 1 Cycle 25 Core Operating Limits Report

Table 1
Control Bank Withdrawal Steps and Sequence

Fully Withdrawn at 222 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
222 Stop	106	0	0
222	116	0 Start	0
222	222 Stop	106	0
222	222	116	0 Start
222	222	222 Stop	106

Fully Withdrawn at 223 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
223 Stop	107	0	0
223	116	0 Start	0
223	223 Stop	107	0
223	223	116	0 Start
223	223	223 Stop	107

Fully Withdrawn at 224 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
224 Stop	108	0	0
224	116	0 Start	0
224	224 Stop	108	0
224	224	116	0 Start
224	224	224 Stop	108

Fully Withdrawn at 225 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
225 Stop	109	0	0
225	116	0 Start	0
225	225 Stop	109	0
225	225	116	0 Start
225	225	225 Stop	109

Fully Withdrawn at 226 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
226 Stop	110	0	0
226	116	0 Start	0
226	226 Stop	110	0
226	226	116	0 Start
226	226	226 Stop	110

Fully Withdrawn at 227 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
227 Stop	111	0	0
227	116	0 Start	0
227	227 Stop	111	0
227	227	116	0 Start
227	227	227 Stop	111

Fully Withdrawn at 228 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
228 Stop	112	0	0
228	116	0 Start	0
228	228 Stop	112	0
228	228	116	0 Start
228	228	228 Stop	112

Fully Withdrawn at 229 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
229 Stop	113	0	0
229	116	0 Start	0
229	229 Stop	113	0
229	229	116	0 Start
229	229	229 Stop	113

Fully Withdrawn at 230 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
230 Stop	114	0	0
230	116	0 Start	0
230	230 Stop	114	0
230	230	116	0 Start
230	230	230 Stop	114

Fully Withdrawn at 231 Steps

Control Bank A	Control Bank B	Control Bank C	Control Bank D
0 Start	0	0	0
116	0 Start	0	0
231 Stop	115	0	0
231	116	0 Start	0
231	231 Stop	115	0
231	231	116	0 Start
231	231	231 Stop	115

McGuire 1 Cycle 25 Core Operating Limits Report

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

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

F_Q^{RTP} *K(Z)/P	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 limit. The manufacturing tolerance and measurement uncertainty are implicitly included in the F_Q surveillance limits as defined for COLR Sections 2.6.5 and 2.6.6.

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

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

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

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

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

where:

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

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

McGuire 1 Cycle 25 Core Operating Limits Report

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

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

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

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

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

where:

$[F_Q^L(X,Y,Z)]^{RPS}$ = Cycle dependent maximum allowable design peaking factor that ensures $F_Q(X,Y,Z)$ 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 calculation and measurement uncertainties.

$F_Q^D(X,Y,Z)$ = Defined in Section 2.6.5.

$M_C(X,Y,Z)$ = Margin remaining to the CFM limit in core location X,Y,Z from the transient power distribution. $M_C(X,Y,Z)$ is provided in Appendix Table A-2 for normal operating conditions and in Appendix Table A-5 for power escalation testing during initial startup operations.

UMT = Defined in Section 2.6.5.

MT = Defined in Section 2.6.5.

TILT = Defined in Section 2.6.5.

McGuire 1 Cycle 25 Core Operating Limits Report

2.6.7 KSLOPE = 0.0725

where:

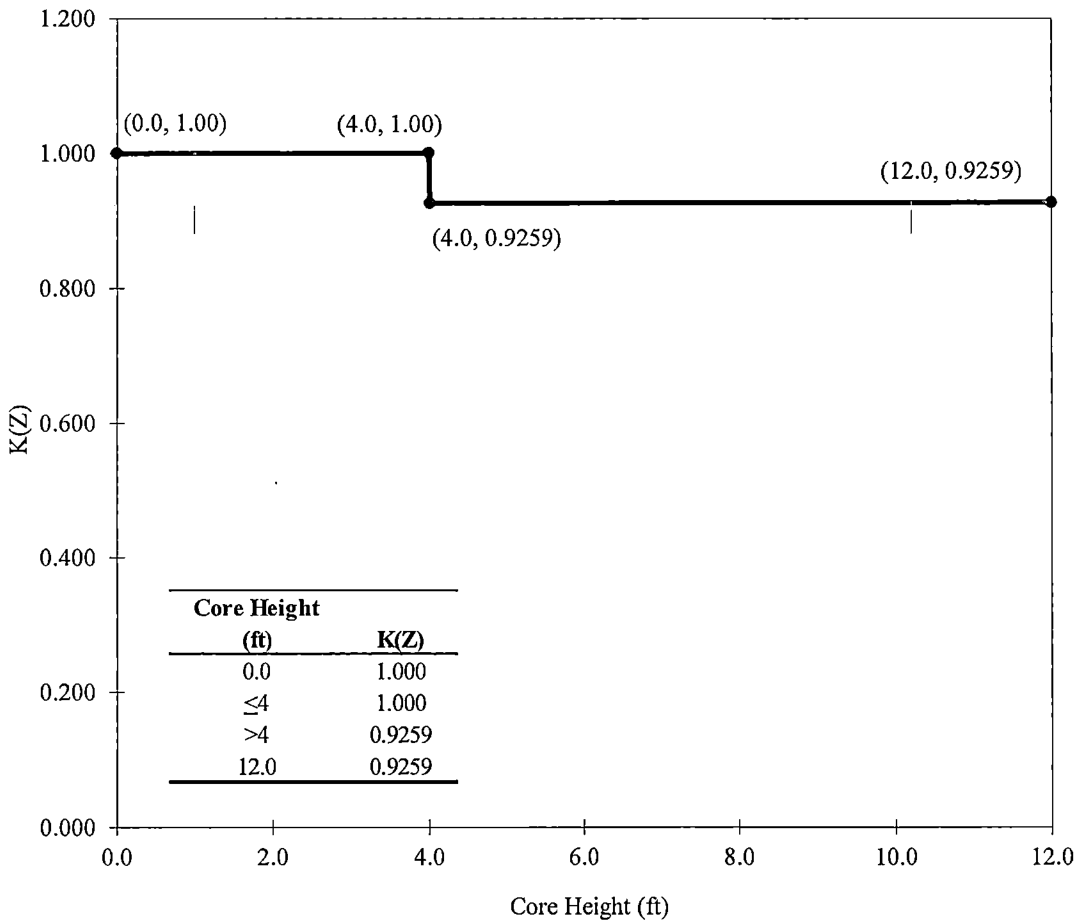
KSLOPE = Adjustment to K_1 value from OTΔT trip setpoint required to compensate for each 1% F_Q^M (X,Y,Z) exceeds F_Q^L (X,Y,Z)^{RPS}.

2.6.8 F_Q(X,Y,Z) Penalty Factors for Technical Specification Surveillances 3.2.1.2 and 3.2.1.3 are provided in Table 2.

McGuire 1 Cycle 25 Core Operating Limits Report

Figure 4

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



Core Height (ft)	K(Z)
0.0	1.000
≤4	1.000
>4	0.9259
12.0	0.9259

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Table 2

$F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ Penalty Factors For Tech Spec Surveillances 3.2.1.2, 3.2.1.3 and 3.2.2.2

Burnup (EFPD)	$F_Q(X,Y,Z)$ Penalty Factor(%)	$F_{\Delta H}(X,Y)$ Penalty Factor (%)
4	2.00	2.00
12	2.00	2.00
25	2.00	2.00
50	2.00	2.00
75	2.00	2.00
100	2.00	2.00
125	2.00	2.00
150	2.00	2.00
175	2.00	2.00
200	2.00	2.00
225	2.00	2.00
250	2.00	2.00
275	2.00	2.00
300	2.00	2.00
325	2.00	2.00
350	2.00	2.00
375	2.00	2.00
400	2.00	2.00
425	2.00	2.00
450	2.00	2.00
465	2.00	2.00
475	2.00	2.00
497	2.00	2.00
502	2.00	2.00
507	2.00	2.00
517	2.00	2.00
527	2.00	2.00

Note: Linear interpolation is adequate for intermediate cycle burnups. All cycle burnups outside the range of the table shall use a 2% penalty factor for both $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ for compliance with the Technical Specification Surveillances 3.2.1.2, 3.2.1.3 and 3.2.2.2.

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

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

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

where:

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

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

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

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

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

2.7.2 $$[F^{L}_{\Delta H}(X,Y)]^{SURV} = \frac{F^{D}_{\Delta H}(X,Y) * M_{\Delta H}(X,Y)}{UMR * TILT}$$

where:

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

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

McGuire 1 Cycle 25 Core Operating Limits Report

$M_{\Delta H}(X,Y)$ = Margin remaining in core location X,Y relative to 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 set to 1.0 since a factor of 1.04 is implicitly included in the variable $M_{\Delta H}(X,Y)$.

TILT = Defined in Section 2.6.5.

2.7.3 RRH is defined in Section 2.7.1.

2.7.4 TRH = 0.04

where:

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

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

2.8 Axial Flux Difference – AFD (TS 3.2.3)

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

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Table 3
Maximum Allowable Radial Peaks (MARPs)
RFA Fuel MARPs
100% Full Power

Core Height (ft)	Axial Peak												
	1.05	**1.1**	**1.2**	**1.3**	**1.4**	**1.5**	**1.6**	**1.7**	**1.8**	**1.9**	**2.1**	**3.0**	**3.25**
0.12	1.8092	1.8553	1.9248	1.9146	1.9179	2.0621	2.0498	2.0090	1.9333	1.8625	1.7780	1.3151	1.2461
1.20	1.8102	1.8540	1.9248	1.9146	1.9179	2.1073	2.0191	1.9775	1.9009	1.8306	1.7852	1.3007	1.2235
2.40	1.8093	1.8525	1.9312	1.9146	1.9179	2.0735	1.9953	1.9519	1.8760	1.8054	1.7320	1.4633	1.4616
3.60	1.8098	1.8514	1.9204	1.9146	1.9179	2.0495	1.9656	1.9258	1.8524	1.7855	1.6996	1.4675	1.3874
4.80	1.8097	1.8514	1.9058	1.9146	1.9179	2.0059	1.9441	1.9233	1.8538	1.7836	1.6714	1.2987	1.2579
6.00	1.8097	1.8514	1.8921	1.9212	1.9179	1.9336	1.8798	1.8625	1.8024	1.7472	1.6705	1.3293	1.2602
7.20	1.8070	1.8438	1.8716	1.8930	1.8872	1.8723	1.8094	1.7866	1.7332	1.6812	1.5982	1.2871	1.2195
8.40	1.8073	1.8319	1.8452	1.8571	1.8156	1.7950	1.7359	1.7089	1.6544	1.6010	1.5127	1.2182	1.1578
9.60	1.8072	1.8102	1.8093	1.7913	1.7375	1.7182	1.6572	1.6347	1.5808	1.5301	1.4444	1.1431	1.0914
10.80	1.7980	1.7868	1.7611	1.7163	1.6538	1.6315	1.5743	1.5573	1.5088	1.4624	1.3832	1.1009	1.0470
11.40	1.7892	1.7652	1.7250	1.6645	1.6057	1.5826	1.5289	1.5098	1.4637	1.4218	1.3458	1.0670	1.0142

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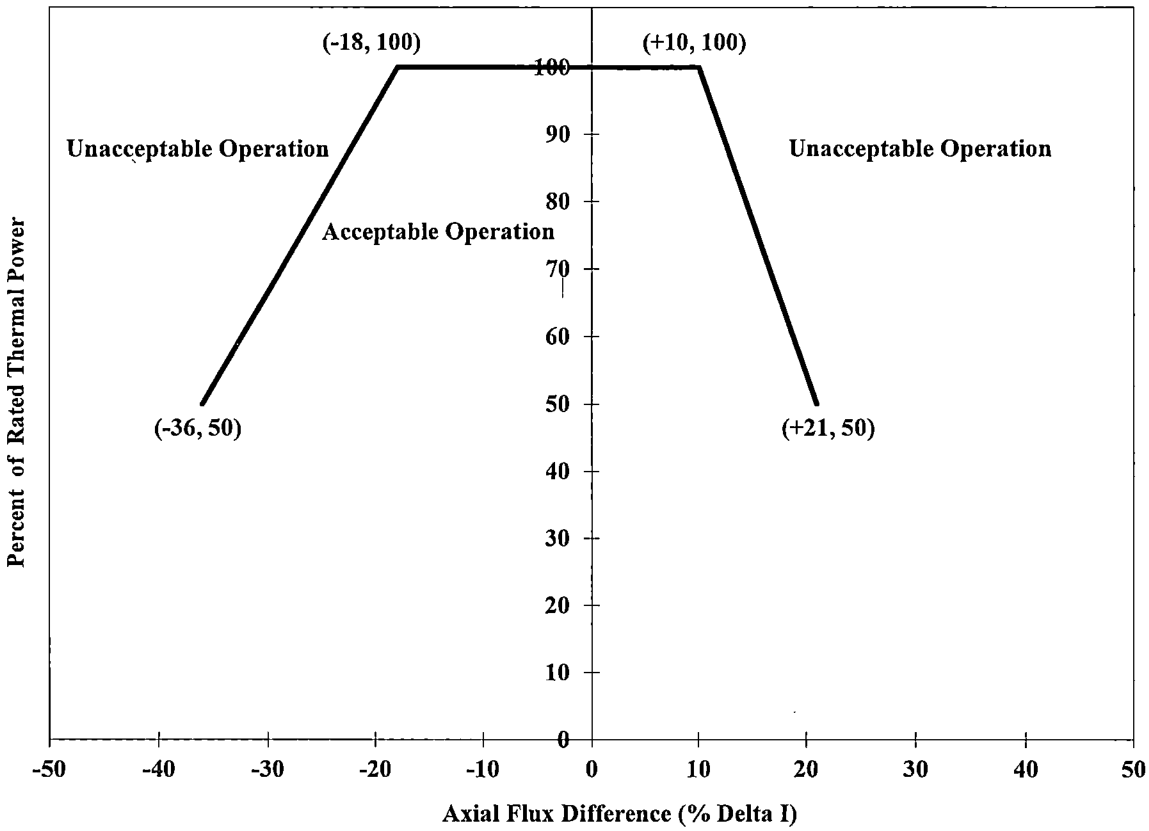
Figure 5

Percent of Rated Thermal Power Versus Percent Axial Flux Difference Limits



NOTE: Compliance with Technical Specification 3.2.1 may require more restrictive AFD limits. Refer to OP/1/A/6100/22 Unit 1 Data Book for 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

Parameter	Nominal Value
Nominal Tavg at RTP	T' ≤ 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	$\tau_3 \leq 2$ 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	$\tau_6 \leq 2$ 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*

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

++ ΔT_0 is assumed to be renormalized to 100% RTP following the MUR power uprate.

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2.9.2 Overpower ΔT Setpoint Parameter Values

Parameter	Nominal Value
Nominal Tavg at RTP	$T'' \leq 585.1$ °F
Overpower ΔT reactor trip setpoint ++	$K_4 \leq 1.0864$
Overpower ΔT reactor trip penalty	$K_5 = 0.02$ / °F for increasing Tavg $K_5 = 0.00$ / °F for decreasing Tavg
Overpower ΔT reactor trip heatup setpoint penalty coefficient	$K_6 = 0.001179$/°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	$\tau_3 \leq 2$ 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 %ΔI
$f_2(\Delta I)$ "negative" breakpoint	= -35.0 %ΔI
$f_2(\Delta I)$ "positive" slope	= 7.0 $\%\Delta T_0 / \%\Delta I$
$f_2(\Delta I)$ "negative" slope	= 7.0 $\%\Delta T_0 / \%\Delta I$

++ ΔT_0 is assumed to be renormalized to 100% RTP following the MUR power uprate.

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

2.10.1 The 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:

Parameter	Applicable Burnup	Limit
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	2,475 ppm
Accumulator **minimum** boron concentration.	300.1 - 350 EFPD	2,404 ppm
Accumulator **minimum** boron concentration.	350.1 - 400 EFPD	2,294 ppm
Accumulator **minimum** boron concentration.	400.1 - 450 EFPD	2,219 ppm
Accumulator **minimum** boron concentration.	450.1 - 497 EFPD	2,148 ppm
Accumulator **minimum** boron concentration.	497.1 - 517 EFPD	2,079 ppm
Accumulator **minimum** boron concentration.	517.1 - 527 EFPD	2,046 ppm
Accumulator **maximum** boron concentration.	0 - 527 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:

Parameter	Limit
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	≥ 2212.3 psig
	meter	3	≥ 2215.0 psig
	computer	4	≥ 2209.1 psig
	computer	3	≥ 2211.3 psig
3. RCS Total Flow Rate			≥ 390,000 gpm*

*Note: The RCS minimum coolant flow rate assumed in the licensing analyses for the M1C25 core is 388,000 gpm. However, the flow is set at 390,000 gpm, which is conservative.

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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.

Parameter	Limit
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 filled portions of the Reactor Coolant System, refueling canal, and refueling cavity for MODE 6 conditions. The minimum boron concentration limit and plant refueling procedures ensure that core K_{eff} remains within the MODE 6 reactivity requirement of $K_{eff} \leq 0.95$.

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

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2.15 Boration Systems 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.

> **NOTE: When cycle burnup is ≥ 475 EFPD, Figure 6 may be used to determine the required BAT Minimum Level.**

Parameter	Limit
BAT minimum boron concentration	7,000 ppm
Volume of 7,000 ppm boric acid solution required to maintain SDM at 68°F	2,300 gallons
BAT Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9.14)	10,599 gallons (13.6%)
RWST minimum boron concentration	2,675 ppm
Volume of 2,675 ppm boric acid solution required to maintain SDM at 68 °F	8,200 gallons
RWST Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9.14)	47,700 gallons (41 inches)

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2.16 Boration Systems Borated Water Source - Operating (SLC 16.9.11)

2.16.1 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.*

***NOTE: The SLC 16.9.11 applicability is down to MODE 4 temperatures of > 300°F. The minimum volumes calculated support cooldown to 200°F to satisfy UFSAR Chapter 9 requirements.**

Parameter	Limit
NOTE: When cycle burnup is ≥ 475 EFPD, Figure 6 may be used to determine the required BAT Minimum Level.	
BAT minimum boron concentration	7,000 ppm
Volume of 7,000 ppm boric acid solution required to maintain SDM at 300°F	13,750 gallons
BAT Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9.11)	22,049 gallons (38.0%)
RWST minimum boron concentration	2,675 ppm
RWST maximum boron concentration (TS 3.5.4)	2,875 ppm
Volume of 2,675 ppm boric acid solution required to maintain SDM	57,107 gallons
RWST Minimum Shutdown Volume (Includes the additional volumes listed in SLC 16.9.11)	96,607 gallons (103.6 inches)

2.17 Standby Shutdown System - (SLC 16.9.7)

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

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

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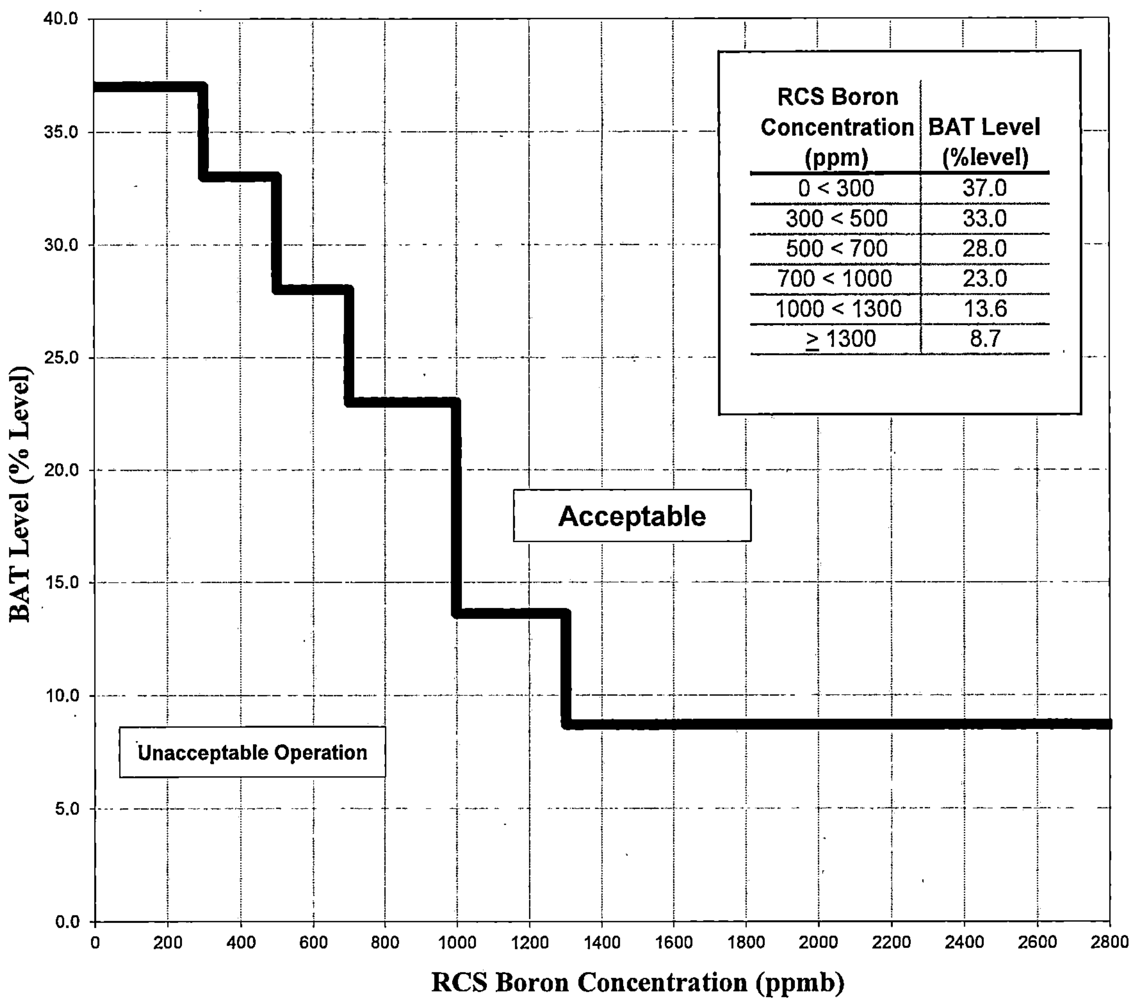
Figure 6

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

(Valid When Cycle Burnup is ≥ 475 EFPD)

This figure includes additional volumes listed in SLC 16.9.11 and 16.9.14



RCS Boron Concentration (ppm)	BAT Level (%level)
0 < 300	37.0
300 < 500	33.0
500 < 700	28.0
700 < 1000	23.0
1000 < 1300	13.6
≥ 1300	8.7

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Appendix A

Power Distribution Monitoring Factors

Appendix A contains power distribution monitoring factors used in Technical Specification Surveillance. This data was generated in the McGuire 1 Cycle 25 Maneuvering Analysis calculation file, MCC-1553.05-00-0622. 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 controls this information via computer files and should be contacted if there is a need to access this information.

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