Serial: RA-19-0315

Enclosure 2

McGuire Unit 2, Cycle 26, Core Operating Limits Report, Revision 1

DUKE ENERGY®

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Change Reason :	CR #02278984. No EC required per AD-NF-ALL-0807, Reload Design Process
Document Title :	MCGUIRE 2 CYCLE 26 CORE OPERATING LIMITS REPORT

Hager, Nicholas R	Originator	7/9/2019
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Siry, Steve	Safety Analysis DV	7/9/2019
Elkins, Jason R	Site Impact Review	7/9/2019
Blom, Michael A	Approver	7/10/2019

Notes :

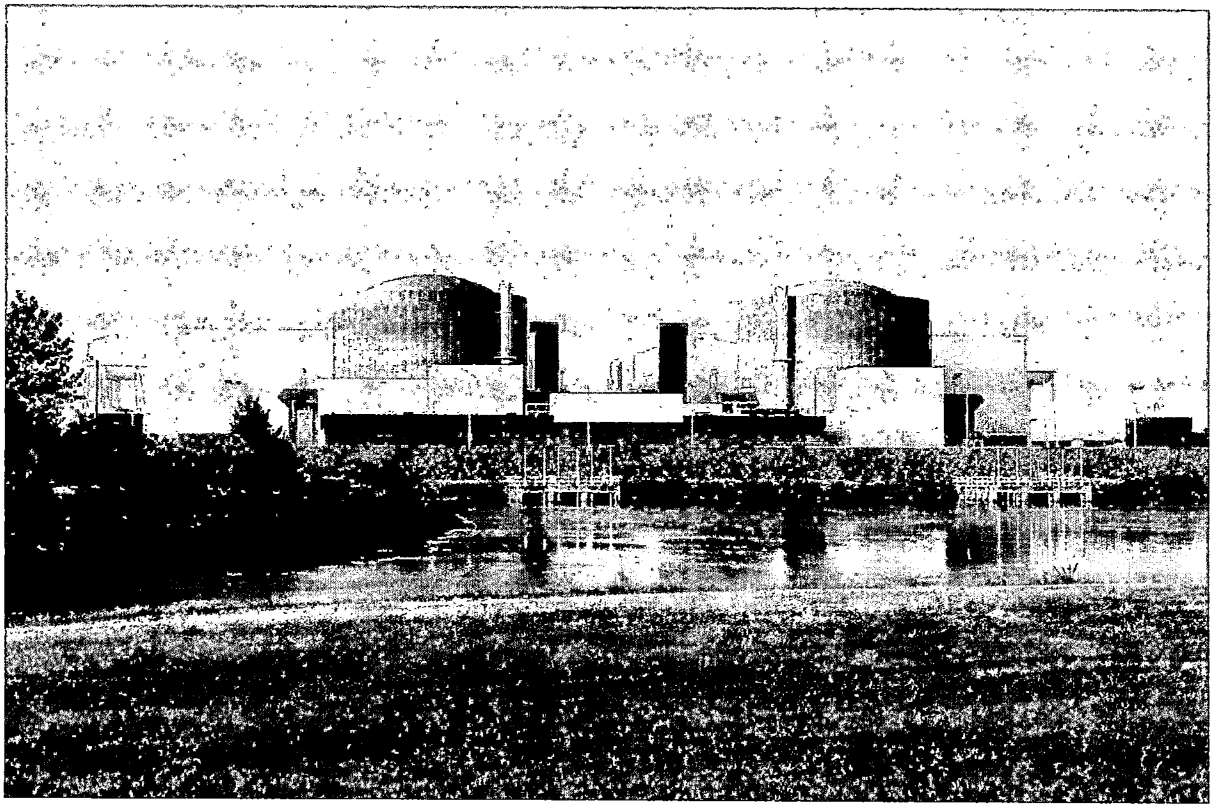
McGuire Unit 2 Cycle 26

Core Operating Limits Report
Revision 1

July 2019

Calculation Number: MCC-1553.05-00-0659, Revision 1

Duke Energy Carolinas, LLC



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 26 Core Operating Limits Report

Implementation Instructions for Revision 1

Revision Description and CR Tracking

Revision 1 of the McGuire Unit 2 Cycle 26 COLR contains limits specific to the reload core.

This revision was initiated by CR #02278984 which describes the vendor Loss of Coolant Accident (LOCA) analysis update which incorporated a required surveillance of the initial FΔH assumption including measurement uncertainty. This design criterion requires surveillance of TS 3.2.2 against FΔH LOCA limit in addition to Loss of Flow Accident (LOFA) Maximum Allowable Radial Peak (MARP) limits. Table 3 of the COLR is revised to reflect the new limits for this surveillance of the limiting value between LOFA MARPs and FΔH LOCA peaking limits.

This revision also reflects an increase in analyzed burnup window to 524 EFPD. The McGuire 2 Cycle 26 burnup is tracked by NTM #02245502-08.

The power distribution monitoring factors from Appendix A of Revision 0 remain valid and are not transmitted as part of Revision 1.

The 50.59 AR is 02224883.

Implementation Schedule

The McGuire Unit 2 Cycle 26 COLR requires the reload 50.59 be approved prior to implementation and fuel loading.

Revision 1 may become effective immediately upon receipt.

The McGuire Unit 2 Cycle 26 COLR will cease to be effective during No MODE between cycles 26 and 27.

Data Files to be Implemented

No data files are transmitted as part of this document.

McGuire 2 Cycle 26 Core Operating Limits Report

REVISION LOG

Revision	Effective Date	Pages Affected	COLR
0	August 2018	1-31, Appendix A*	M2C26 COLR, Rev. 0
1	July 2019	1-3, 18, 21, 25	M2C26 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 26 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	6,7,8,9,10,12,15,16,18, 19
3.1.1	Shutdown Margin	Shutdown Margin	2.2	6,7,8,12,14,15,16,18,19
3.1.3	Moderator Temperature Coefficient	MTC	2.3	6,7,8, 14,16, 17
3.1.4	Rod Group Alignment Limits	Shutdown Margin	2.2	6,7,8,12,14,15,16,18,19
3.1.5	Shutdown Bank Insertion Limits	Shutdown Margin	2.2	6,7,8,12,14,15,16,18,19
3.1.5	Shutdown Bank Insertion Limits	Shutdown Bank Insertion Limit	2.4	2,4,6,7,8,9,10,12,14,15, 16,18,19
3.1.6	Control Bank Insertion Limits	Shutdown Margin	2.2	6,7,8,12,14,15,16,18,19
3.1.6	Control Bank Insertion Limits	Control Bank Insertion Limit	2.5	2,4,6,7,8,9,10,12,14,15, 16,18,19
3.1.8	Physics Tests Exceptions	Shutdown Margin	2.2	6,7,8,12,14,15,16,18,19
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, 18,19
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, 18,19
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	6,7,8,9,10,12,15,16,18, 19
3.4.1	RCS Pressure, Temperature, and Flow DNB limits	RCS Pressure, Temperature and Flow	2.10	6,7,8,9,10,12,18,19
3.5.1	Accumulators	Max and Min Boron Conc.	2.11	6,7,8,14,16
3.5.4	Refueling Water Storage Tank	Max and Min Boron Conc.	2.12	6,7,8,14,16
3.7.14	Spent Fuel Pool Boron Concentration	Min Boron Concentration	2.13	6,7,8,14,16
3.9.1	Refueling Operations – Boron Concentration	Min Boron Concentration	2.14	6,7,8,14,16
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	6,7,8,14,16
16.9.11	Borated Water Source – Operating	Borated Water Volume and Conc. for BAT/RWST	2.16	6,7,8,14,16
16.9.7	Standby Shutdown System	Standby Makeup Pump Water Supply	2.17	6,7,8,14,16

McGuire 2 Cycle 26 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 26 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 1
   Report Date: March 2015

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 5
    Report Date: March 2016

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

    Revision 0
    Report Date: April 3, 1995
    Not Used

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

    Revision 3c
    Report Date: March 2017

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

    Revision 1a
    Report Date: January 2009
    Not Used

McGuire 2 Cycle 26 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

18. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," (W Proprietary).

    Revision 0
    Report Date: April 1995

19. WCAP-12610-P-A & CENPD-404-P-A, Addendum 1-A, "Optimized ZIRLO™," (W Proprietary).

    Revision 0
    Report Date: July 2006

McGuire 2 Cycle 26 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% ΔK/K in MODE 2 with k-eff < 1.0 and in MODES 3 and 4.

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

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

2.2.4 For TS 3.1.5, SDM shall be $\geq$ 1.3% ΔK/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% ΔK/K in MODE 1 and MODE 2 with K-eff $\geq$ 1.0.

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

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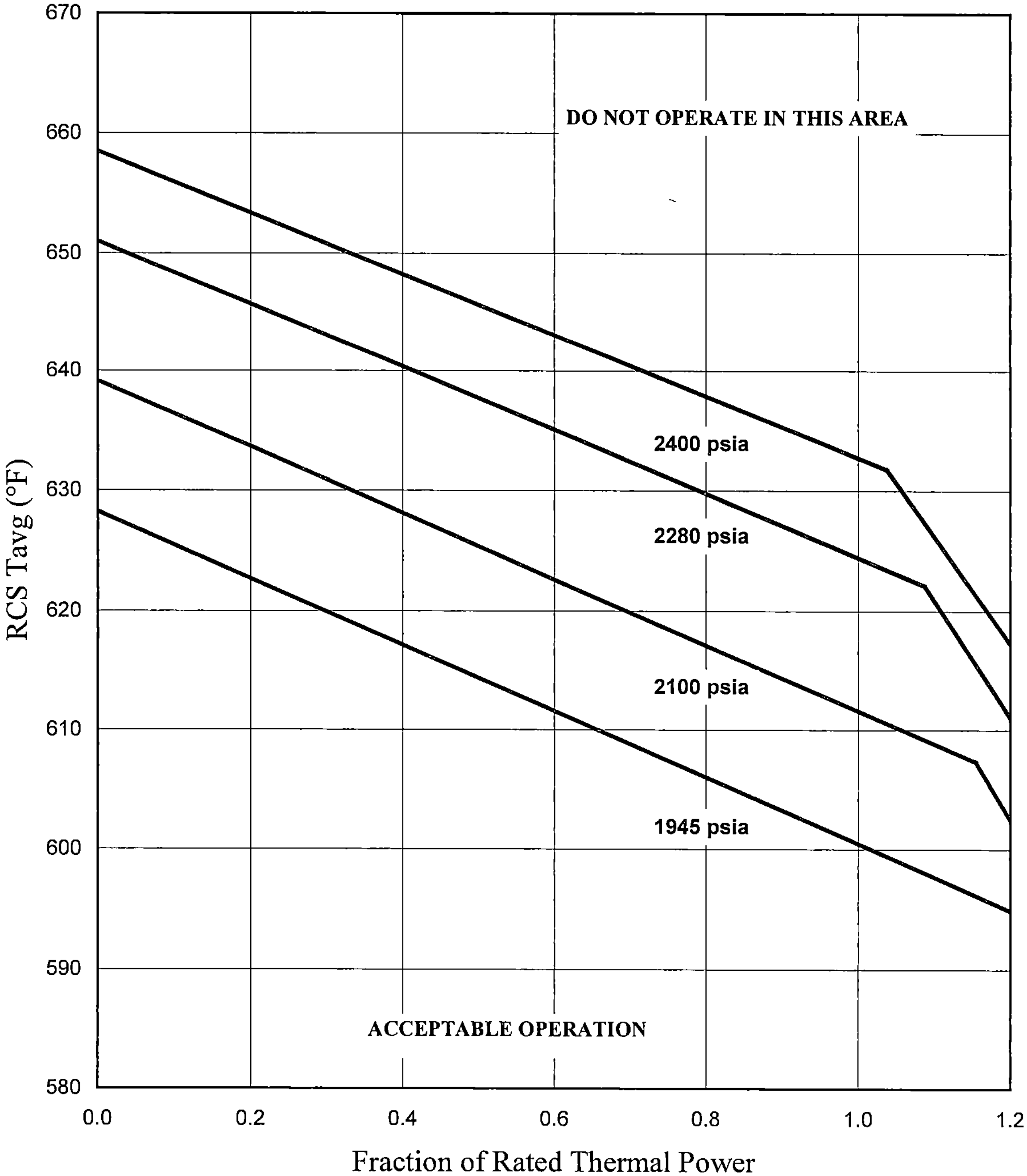
Figure 1

Reactor Core Safety Limits
Four Loops in Operation



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2.3 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 0.7E-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 a 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 -4.125E-04 ΔK/K/°F.

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

2.4 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.

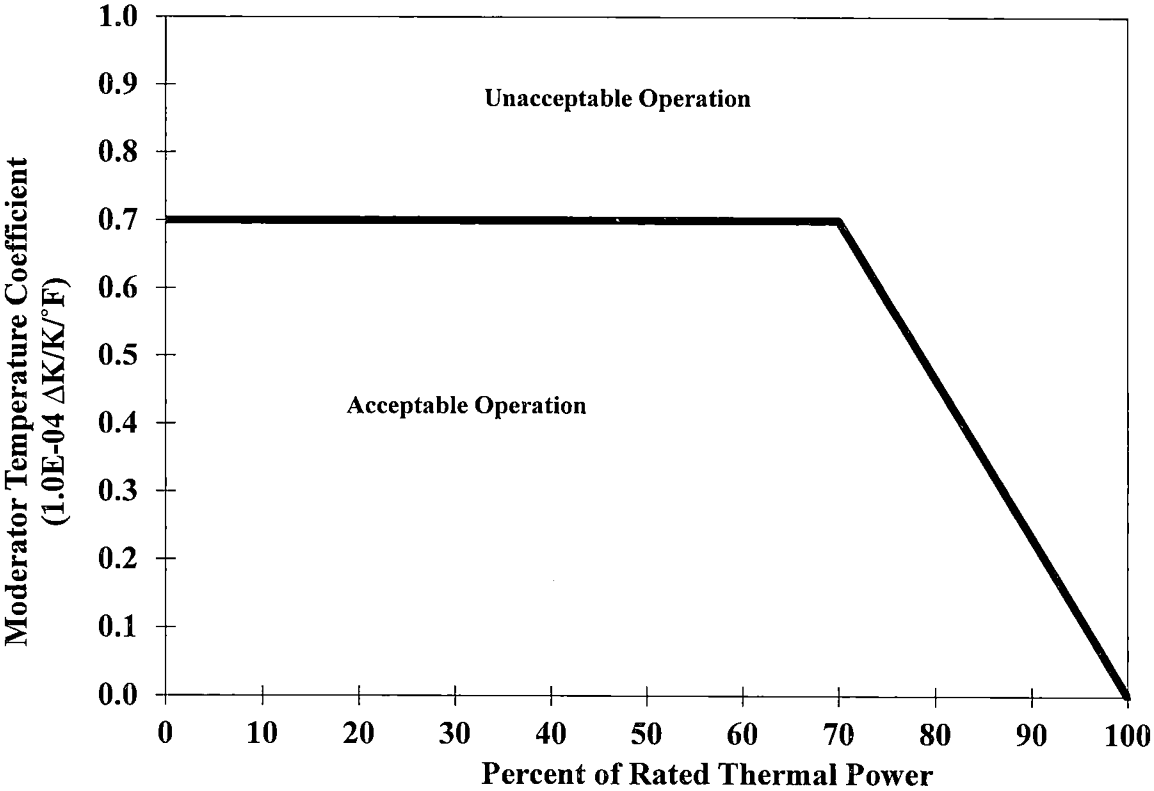
2.5 Control Bank Insertion Limits (TS 3.1.6)

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 26 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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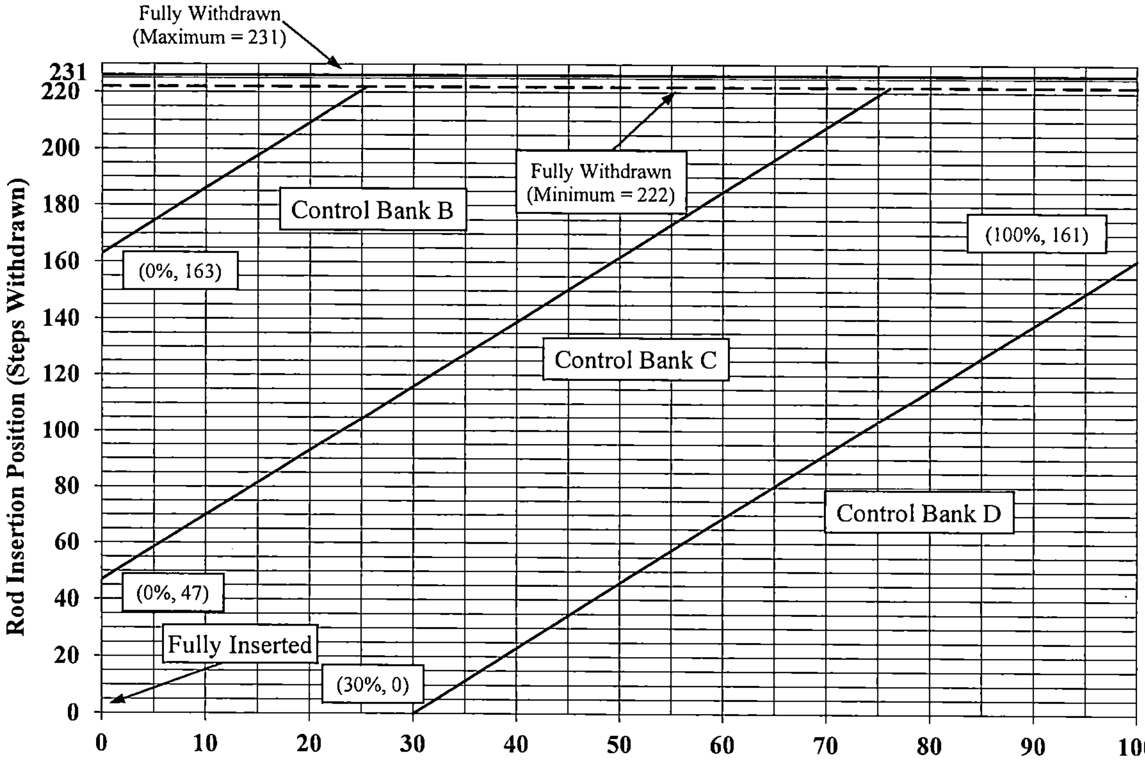
Figure 3

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:

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

where P = %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

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

Fully Withdrawn at 223 Steps

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

Fully Withdrawn at 224 Steps

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

Fully Withdrawn at 225 Steps

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

Fully Withdrawn at 226 Steps

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

Fully Withdrawn at 227 Steps

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

Fully Withdrawn at 228 Steps

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

Fully Withdrawn at 229 Steps

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

Fully Withdrawn at 230 Steps

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

Fully Withdrawn at 231 Steps

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

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

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

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

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

where,

P = (Thermal Power)/(Rated Power)

Note: The measured $F_Q(X,Y,Z)$ shall be increased by 3% to account for manufacturing tolerances and 5% to account for measurement uncertainty when comparing against the LCO limits. The manufacturing tolerance and measurement uncertainty are implicitly included in the F_Q surveillance limits as defined in Sections 2.6.5 and 2.6.6.

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

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

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

where:

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

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

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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 = Defined in Section 2.6.5.

MT = Defined in Section 2.6.5.

TILT = Defined in Section 2.6.5.

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

Figure 4
K(Z), Normalized $F_Q(X,Y,Z)$ as a Function of Core Height for Westinghouse RFA Fuel

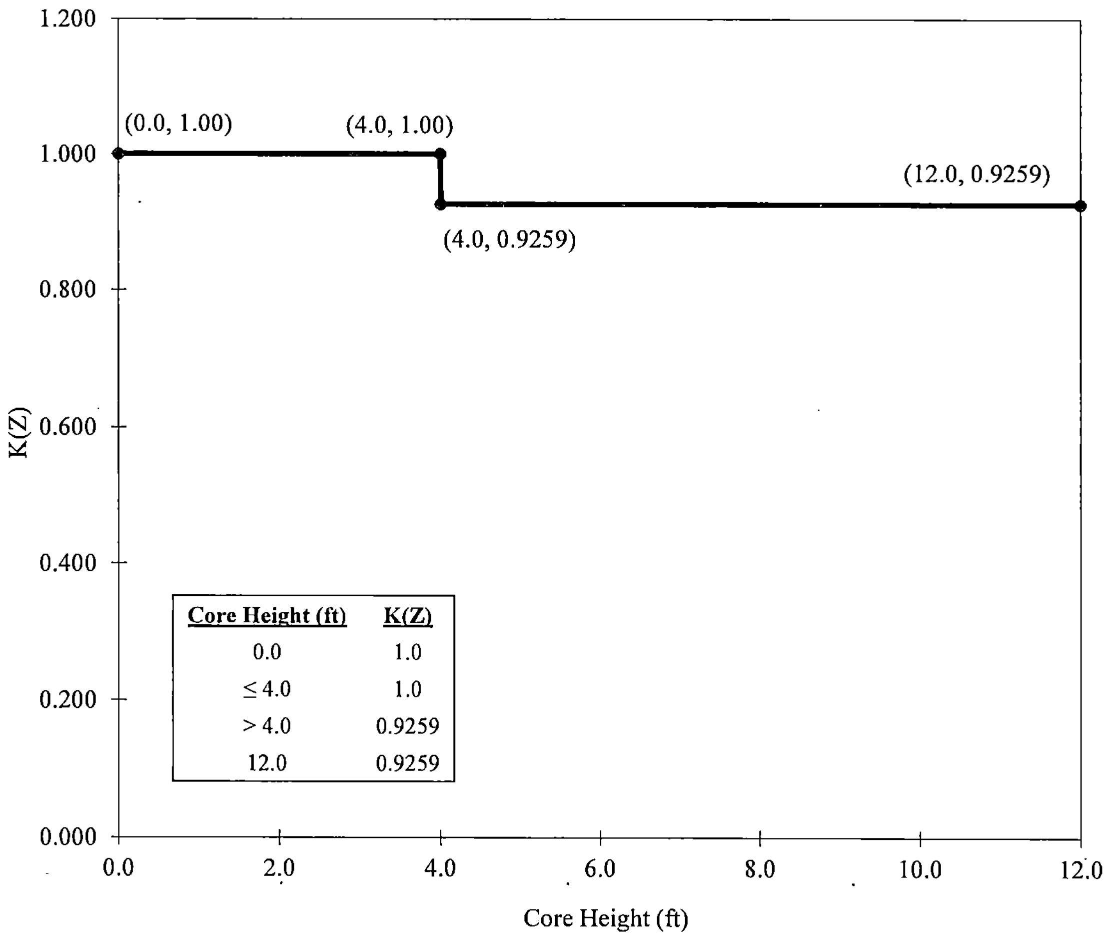

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Table 2

$F_Q(X,Y,Z)$ 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 (%)
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
460	2.00	2.00
475	2.00	2.00
489	2.00	2.00
494	2.00	2.00
499	2.00	2.00
509	2.00	2.00
524	2.00	2.00

Note: Linear interpolation is adequate for intermediate cycle burnups. All cycle burnups outside 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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2.7 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 $$F^{L}_{\Delta H}(X,Y)^{LCO} = \text{MARP}(X,Y) * \left[1.0 + \frac{1}{RRH} * (1.0 - P)\right]$$

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% that the measured radial peak, $F^{M}_{\Delta H}(X,Y)$, exceeds its limit. RRH also is used to scale the MARP limits as a function of power per the $F^{L}_{\Delta H}(X,Y)^{LCO}$ equation. (RRH = 3.34 $(0.0 < P \leq 1.0)$)

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

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

where:

$F^{L}_{\Delta H}(X,Y)^{SURV}$ = 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. $F^{L}_{\Delta H}(X,Y)^{SURV}$ includes allowances for calculation/measurement uncertainty.

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$F^D_{\Delta H}$ (X,Y) = Design radial power distribution for $F_{\Delta H}$. $F^D_{\Delta H}$ (X,Y) is provided in Appendix 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 is defined in Section 2.7.1.

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, $F^M_{\Delta H}$ (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 The Axial Flux Difference (AFD) Limits are provided in Figure 5.

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Table 3
Maximum Allowable Radial Peaks (MARPS)

RFA Steady State Limiting Value Between
Loss of Flow Accident (LOFA) MARPs and $F\Delta H_{LOCA}$

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**	**3.25**
0.12	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.3151	1.2461
1.20	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.3007	1.2235
2.40	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.4633	1.4616
3.60	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.4675	1.3874
4.80	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.2987	1.2579
6.00	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.3293	1.2602
7.20	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.5982	1.2871	1.2195
8.40	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6010	1.5127	1.2182	1.1578
9.60	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.5808	1.5301	1.4444	1.1431	1.0914
10.80	1.6058	1.6058	1.6058	1.6058	1.6058	1.6058	1.5743	1.5573	1.5088	1.4624	1.3832	1.1009	1.0470
11.40	1.6058	1.6058	1.6058	1.6058	1.6057	1.5826	1.5289	1.5098	1.4637	1.4218	1.3458	1.0670	1.0142

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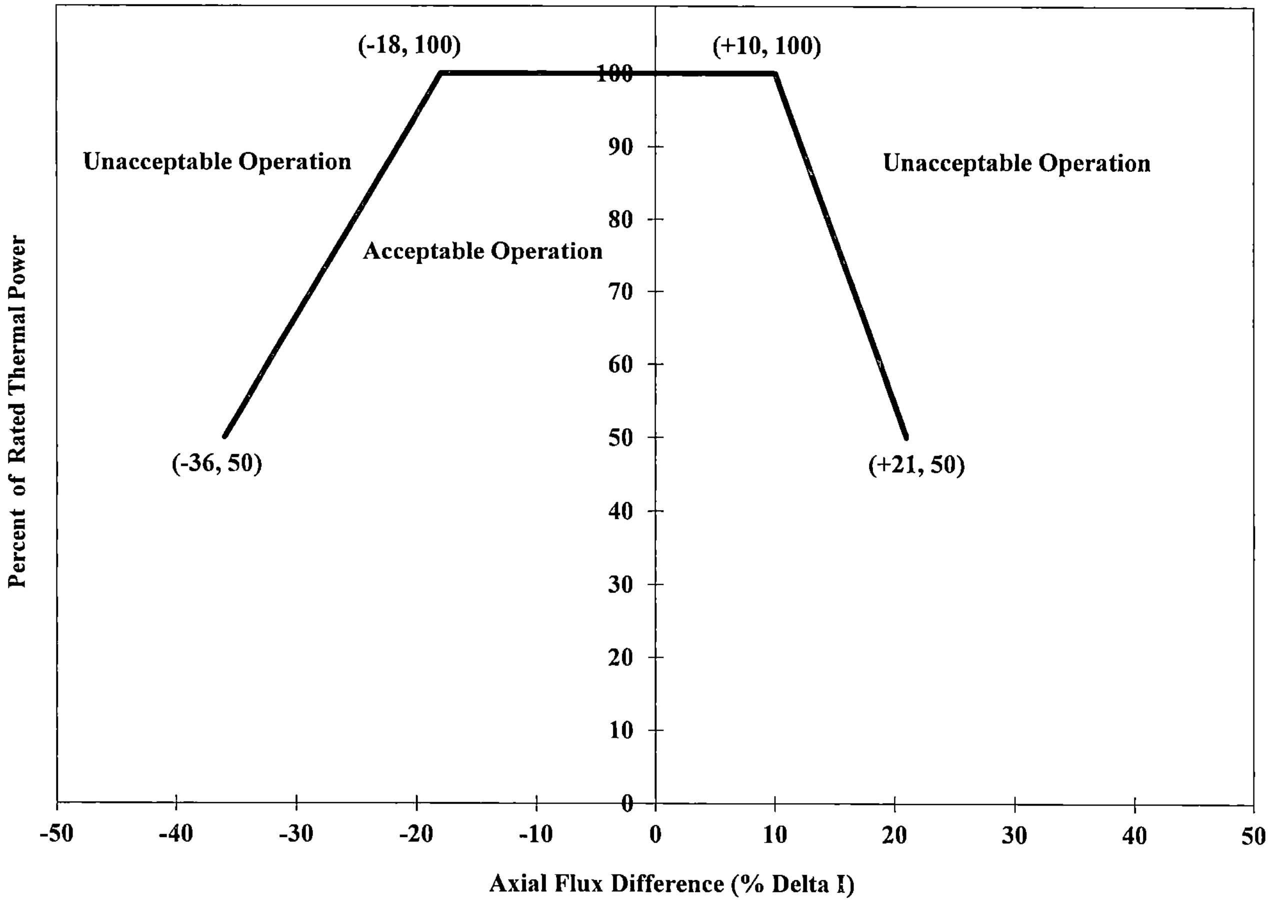
Figure 5

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/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

Parameter	Value
Nominal Tavg at RTP	$T' \leq 585.1°F$
Nominal RCS Operating Pressure	$P' = 2235$ psig
Overtemperature ΔT reactor trip setpoint	$K_1 \leq 1.1978$
Overtemperature ΔT reactor trip heatup setpoint penalty coefficient	$K_2 = 0.0334/°F$
Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient	$K_3 = 0.001601$/psi
Time constants utilized in the lead-lag compensator for ΔT	$\tau_1 \geq 8$ sec. $\tau_2 \leq 3$ sec.
Time constant utilized in the lag compensator for ΔT	$\tau_3 \leq 2$ 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*

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

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2.9.2 Overpower ΔT Setpoint Parameter Values

Parameter	Value
Nominal Tavg at RTP	$T'' \leq 585.1°F$
Overpower ΔT reactor trip setpoint	$K_4 \leq 1.0864$
Overpower ΔT reactor trip Penalty	$K_5 = 0.02/°F$ for increasing Tavg $K_5 = 0.0$ for decreasing Tavg
Overpower ΔT reactor trip heatup setpoint penalty coefficient	$K_6 = 0.001179/°F$ for $T > T''$ $K_6 = 0.0$ for $T \leq T''$
Time constants utilized in the lead-lag compensator for ΔT	$\tau_1 \geq 8$ sec. $\tau_2 \leq 3$ sec.
Time constant utilized in the lag compensator for ΔT	$\tau_3 \leq 2$ 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 $\%\Delta T_0 / \%\Delta I$

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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:

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	2,406 ppm
Accumulator **minimum** boron concentration.	350.1 - 400 EFPD	2,290 ppm
Accumulator **minimum** boron concentration.	400.1 - 450 EFPD	2,215 ppm
Accumulator **minimum** boron concentration.	450.1 – 494 EFPD	2,144 ppm
Accumulator **minimum** boron concentration.	494.1 - 509 EFPD	2,074 ppm
Accumulator **minimum** boron concentration.	509.1 - 524 EFPD	2,047 ppm
Accumulator **maximum** boron concentration.	0 - 524 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			≥ 388,000 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.

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 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 Keff $\leq$ 0.95.

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

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2.15 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.

Parameter	Limit
Note: When cycle burnup is > 464 EFPD, Figure 6 may be used to determine required BAT minimum level.	
BAT minimum contained borated water volume	10,599 gallons 13.6% Level
BAT minimum boron concentration	**7,150** ppm
BAT minimum water volume required to maintain SDM at **7,150** 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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2.16 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 temperature > 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.**

Note: When cycle burnup is > 464 EFPD, Figure 6 may be used to determine required BAT minimum level.

Parameter	Limit
BAT minimum contained borated water volume	22,049 gallons 38.0% Level
BAT minimum boron concentration	**7,150** ppm
BAT minimum water volume required to maintain SDM at **7,150** 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)	2,875 ppm
RWST minimum water volume required to maintain SDM at 2,675 ppm	57,107 gallons

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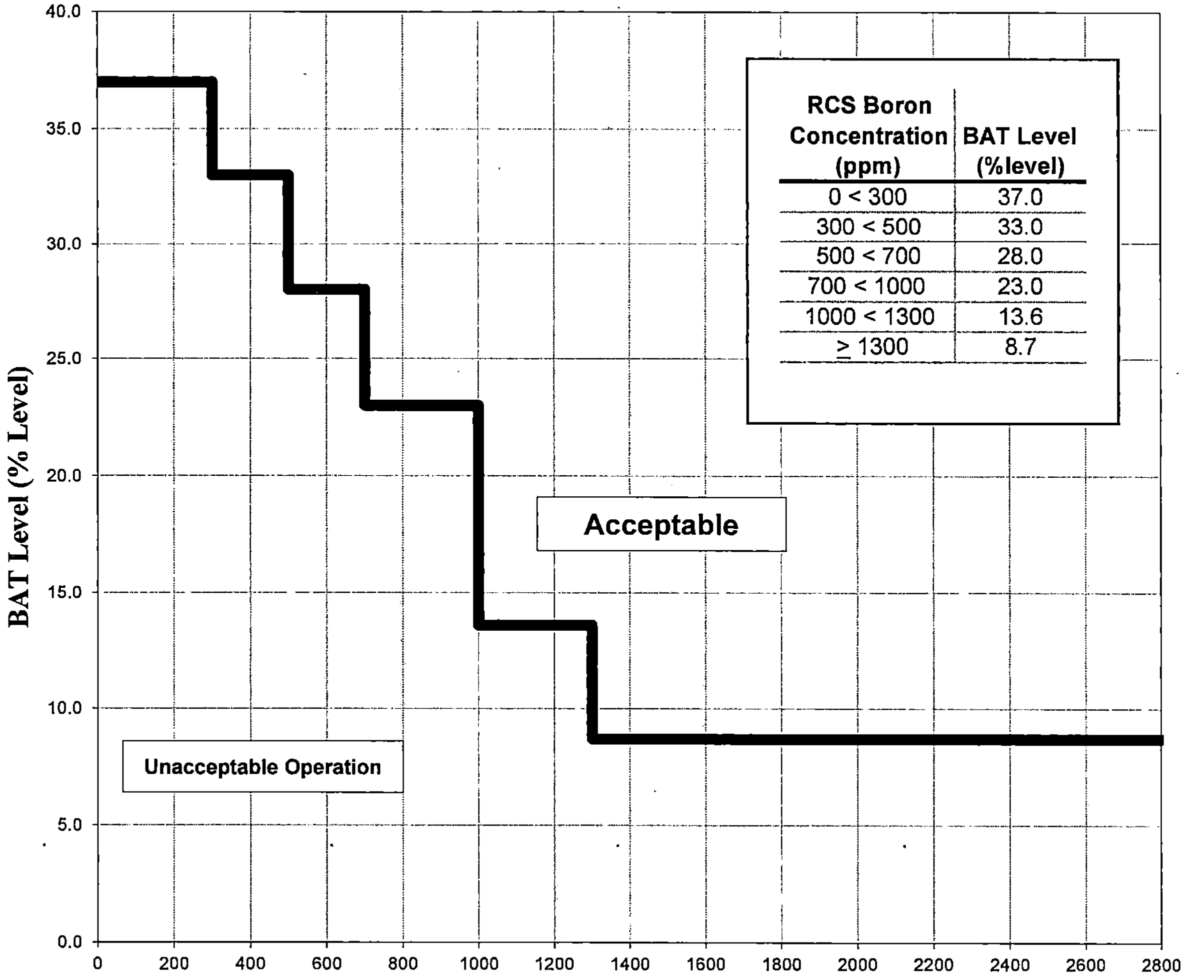
Figure 6
Boric Acid Storage Tank Indicated Level Versus RCS Boron Concentration

(Valid When Cycle Burnup is > 464 EFPD)

This figure includes additional volumes listed in SLC 16.9.14 and 16.9.11



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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NOTE: Appendix A contains power distribution monitoring factors used in Technical Specification Surveillance. This data was generated in the McGuire 2 Cycle 26 Maneuvering Analysis calculation file, MCC-1553.05-00-0652. Due to the size of the monitoring factor data, Appendix A is controlled electronically within the Duke document management system and is not included in the Duke internal copies of the COLR. The Plant Reactor Engineering and Support Systems 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.