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Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

Subject: Cycle 23 Core Operating Limits Report, Revision 0

Ladies and Gentlemen:

Pursuant to Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Technical Specification 6.9.1.6.4, please find enclosed Revision 0 of the HNP Cycle 23 Core Operating Limits Report.

This document contains no regulatory commitments. Please refer any questions regarding this submittal to Kevin Riley, Manager – Nuclear Support Services, at (919) 362-2124.

Sincerely,

Tanya M. Hamilton

Tanya M. Hamilton

Enclosure: Harris Unit 1 Cycle 23 Core Operating Limits Report (COLR), Revision 0

cc: J. Zeiler, NRC Senior Resident Inspector, HNP
T. Hood, NRC Project Manager, HNP
L. Dudes, NRC Regional Administrator, Region II

RA-19-0408

Enclosure

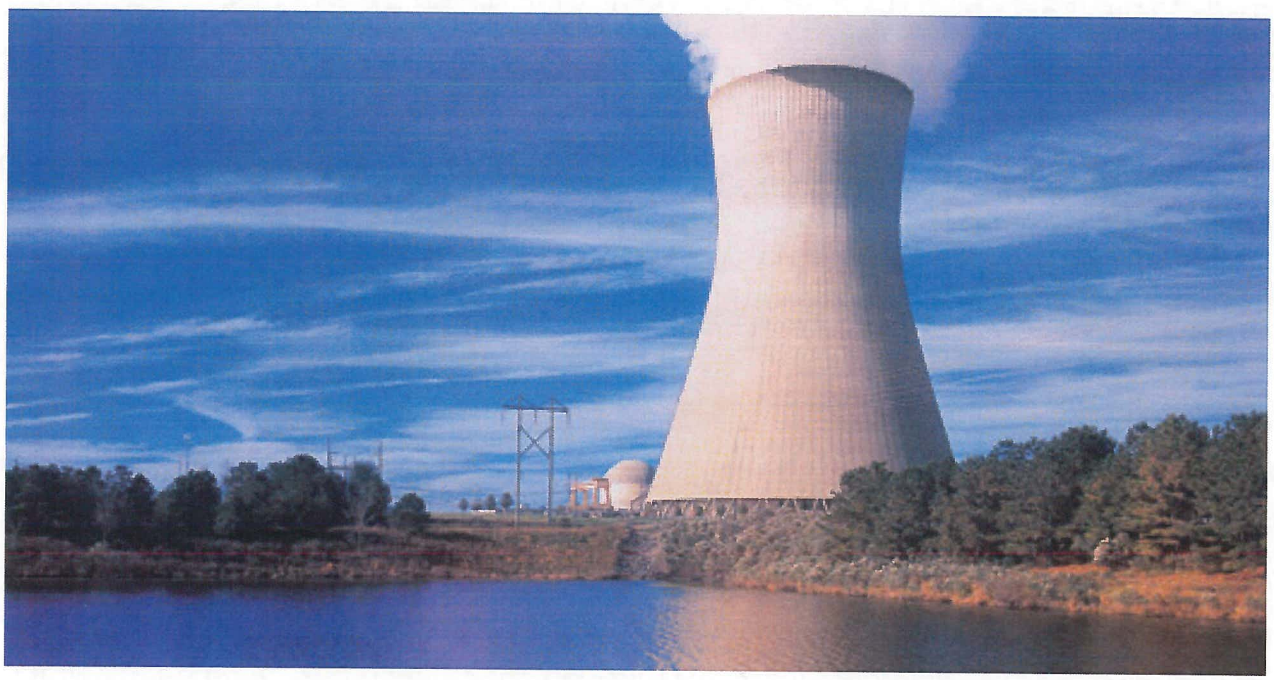
Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400 / Renewed License No. NPF-63

Harris Unit 1 Cycle 23
Core Operating Limits Report (COLR)
Revision 0

Harris Unit 1 Cycle 23
Core Operating Limits Report (COLR)

HNEI-0400-0016
Revision 0

References:
HNP-F/NFSA-0351, Revision 0
HNP-F/NFSA-0372, Revision 0



Quality Class A

The information presented in this report has been prepared and issued in accordance with Harris Technical Specification 6.9.1.6. Changes to the COLR are submitted to NRC per Technical Specification 6.9.1.6.4.

Harris Unit 1 Cycle 23 Core Operating Limits Report

Implementation Instructions for Revision 0

Revision Description and AR Tracking

Revision 0 is the original issue of the Harris Unit 1 Cycle 23 Core Operating Limits Report (COLR) contains limits specific to the reload core based on the information obtained from HNP-F/NFSA-0351, Revision 0. Implementation of this document is controlled by the normal cycle transition process. No ARs are associated with this revision.

Implementation Schedule

The Harris Unit 1 Cycle 23 COLR requires the cycle Reload Safety Evaluation (RSE), HNP-F/NFSA-0372, be approved prior to the COLR being issued. The RSE supports and references the reload 50.59 (AR# 02240406), which must be approved prior to the reload implementation and fuel loading.

Revision 0 may become effective after no-mode is reached between Cycle 22 and 23, but prior to entering Mode 6 that starts Harris Unit 1 Cycle 23 refueling. The Harris Unit 1 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.

Additional Information

CDR was performed by Safety Analysis for COLR Sections 2.1-2.3, 2.5-2.6, and 2.12 – 2.15.
HNP Reactor Engineering performed site inspection in accordance with AD-NF-ALL-0807 and AD-NF-NGO-0214.

Revision Log

Revision	Effective Date	Pages Affected
0	October 2019	Original Issue, pages 1-26, Appendix A*

* 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.

1.0 CORE OPERATING LIMITS REPORT

This Core Operating Limits Report (COLR) for Shearon Harris Unit 1 Cycle 23 has been prepared in accordance with the requirements of Technical Specification 6.9.1.6.

The Technical Specifications affected by 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 Specification	COLR Parameter	COLR Section	NRC Approved Methodology (Section 3.0 Number)
2.1.1	Reactor Core Safety Limits	RCS Temperature and Pressure Safety Limits	2.1	4, 10, 11, 22
2.2.1	Reactor Trip System Instrumentation Setpoints	OTΔT OPΔT	2.2	4, 10, 11
3/4.1.1.1	Shutdown Margin – Modes 1 and 2	Shutdown Margin	2.3	11, 17, 18
3/4.1.1.2	Shutdown Margin – Modes 3,4, and 5	Shutdown Margin	2.3	11, 17, 18
3/4.1.1.3	Moderator Temperature Coefficient	MTC	2.4	11, 17, 18, 20, 21
3/4.1.2.5	Borated Water Source – Shutdown	Max, Min Boron Conc.	2.5	18
3/4.1.2.6	Borated Water Source - Operating	Max, Min Boron Conc.	2.6	18
3/4.1.3.5	Shutdown Rod Insertion Limit	Shutdown Margin, Rod Insertion Limits	2.7	11, 17, 18, 19, 20, 21
3/4.1.3.6	Control Rod Insertion Limits	Shutdown Margin, Rod Insertion Limits	2.8	11, 17, 18, 19, 20, 21
3/4.2.1	Axial Flux Difference	AFD	2.9	6, 11, 13, 15, 17, 19, 20, 21
3/4.2.2	Heat Flux Hot Channel Factor $F_Q(X,Y,Z)$	F_Q, AFD, OTΔT, Penalty Factors	2.10	4, 6, 11, 13, 15, 17, 19, 20, 21
3/4.2.3	Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}(X,Y)$	FΔH, Penalty Factors	2.11	4, 6, 10, 11, 13, 15, 16, 17, 19, 20, 21, 22
3/4.2.5	Reactor Coolant System DNB Parameters	RCS Pressure, Temperature, and Flow	2.13	4, 10, 11
3/4.5.1	Accumulators - Max and Min Boron Concentration	Max, Min Boron Conc.	2.14	18
3/4.5.4	Refueling Water Storage Tank - Max and Min Boron Concentration	Max, Min Boron Conc.	2.15	18
3/4.9.1.a	Boron Concentration During Refueling Operations	Min Boron Conc.	2.12	11, 17, 18

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. These limits have been developed using NRC approved methodologies specified in Technical Specification 6.9.1.6 and are detailed in Section 3.0.

2.1 Reactor Core Safety Limits (Specification 2.1.1)

The Reactor Core Safety Limits are shown in Figure 5.

2.2 Reactor Trip System Instrumentation Setpoints (Specification 2.2.1)

The Reactor Trip System Instrumentation Setpoints are shown in Tables 1 and 2.

2.3 Boration Control - Shutdown Margin (Specification 3/4.1.1)

1. Shutdown Margin – Modes 1 and 2 (Specification 3/4.1.1.1)
   a. Mode 1 Requirement: ≥ 1770 pcm
   b. Mode 2 Requirement: ≥ 1770 pcm
2. Shutdown Margin – Modes 3,4, and 5 (Specification 3/4.1.1.2)
   a. Mode 3 Requirement: ≥ 1770 pcm
   b. Mode 4 Requirement: ≥ 1770 pcm
   c. Mode 5 Requirement: Specified in Figure 1.

2.4 Moderator Temperature Coefficient (Specification 3/4.1.1.3)

1. The Moderator Temperature Coefficient (MTC) limits are:

   The Positive MTC Limit (ARO/HZP) shall be less positive than +4.0 pcm/°F with a linear ramp to 0.0 pcm/°F at 50% RTP. Then a constant MTC limit of 0.0 pcm/°F up to 100% RTP.

   The Negative MTC Limit (ARO/RTP) shall be less negative than -50 pcm/°F.

2. The MTC Surveillance limit is:

   The 300 ppm/ARO/RTP MTC should be less negative than or equal to -43.4191pcm/°F.

where:

a. ARO stands for All Rods Out

b. HZP stands for Hot Zero THERMAL POWER

c. RTP stands for RATED THERMAL POWER

d. ppm stands for Parts per million (Boron)

2.5 Borated Water Source - Shutdown (Specification 3/4.1.2.5)

1. The Boric Acid Tank (BAT) boron concentration limits at Shutdown in MODES 5 and 6 are:

   BAT minimum boron concentration = 7000 ppm
   BAT maximum boron concentration = 7750 ppm

2. The Refueling Water Storage Tank (RWST) boron concentration limits at Shutdown in MODES 5 and 6 are:

   RWST minimum boron concentration = 2424 ppm *
   RWST maximum boron concentration = 2574 ppm *

   * These Boron Concentrations include a 1% Boron Measurement Uncertainty

2.6 Borated Water Source - Operating (Specification 3/4.1.2.6)

1. The Boric Acid Tank (BAT) boron concentration limits at Operation in MODES 1, 2,3, and 4 is:

   BAT minimum boron concentration = 7000 ppm
   BAT maximum boron concentration = 7750 ppm

2. The Refueling Water Storage Tank (RWST) boron concentration limits at Operation in MODES 1, 2, 3, and 4 is:

   RWST minimum boron concentration = 2424 ppm *
   RWST maximum boron concentration = 2574 ppm *

   * These Boron Concentrations include a 1% Boron Measurement Uncertainty

2.7 Shutdown Rod Insertion Limit (Specification 3/4.1.3.5)

Fully withdrawn for all shutdown rods shall be greater than or equal to 225 steps.

2.8 Control Rod Insertion Limit (Specification 3/4.1.3.6)

The control rod banks shall be limited in physical insertion as specified in Figure 2. Fully withdrawn for all control rods shall be greater than or equal to 225 steps.

2.9 Axial Flux Difference (Specification 3/4.2.1)

The AXIAL FLUX DIFFERENCE (AFD) limits are specified in Figure 3.

2.10 Heat Flux Hot Channel Factor $F_Q^M(X,Y,Z)$ (Specification 3/4.2.2)

1. The $F_Q^M(X,Y,Z)$ Steady-State Limit as referenced in TS 3/4.2.2 is:

   $F_Q^M(X,Y,Z) \leq \frac{F_Q^{RTP}}{P}$ K(Z) * K(BU) for P > 0.5

   $F_Q^M(X,Y,Z) \leq \frac{F_Q^{RTP}}{0.5}$ K(Z) * K(BU) for P ≤ 0.5

   where:

   a. P = $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

   b. F_Q^{RTP} = 2.52 for HTP fuel.

   c. K(Z) = the normalized $F_Q\ (X,Y,Z)$ as a function of core height, as specified in Figure 4. K(Z) is set equal to 1.0 for all axial elevations.

   d. K(BU) = is the normalized $F_Q\ (X,Y,Z)$ as a function of burnup. K(BU) is set to 1.0 at all burnups.

Note: $F_Q^M(X,Y,Z)$ 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 FQ surveillance limits as defined in COLR Sections 2.10.2 and 2.10.3.

2. The $F_Q^L(X,Y,Z)^{OP}$ Transient Operational Limit as referenced in TS 3/4.2.2 is:

   $F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{OP}$

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

   where:

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

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

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

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

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

3. The $F_Q^L(X,Y,Z)^{RPS}$ Transient Reactor Protection System Limit as referenced in TS 3/4.2.2 is:

$$F_Q^M(X,Y,Z) \leq F_Q^L(X,Y,Z)^{RPS}$$

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

where:

a. $F_Q^L(X,Y,Z)^{RPS}$ = 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 LCO limits. $F_Q^L(X,Y,Z)^{RPS}$ includes allowances for calculation and measurement uncertainties.

b. $F_Q^D(X,Y,Z)$ = Defined above in 2.b

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

d. UMT = Defined above in 2.d

e. MT = Defined above in 2.e

4. THERMAL POWER and AFD limit reductions required when $F_Q^L(X,Y,Z)^{OP}$ limit is exceeded are identified in Table 3.

5. KSLOPE = 1.70 ΔI / %F_Q

where:

KSLOPE = reduction to the OPΔT $f_2(\Delta I)$ breakpoints (Specification 2.2.1) required to compensate for each 1% measured $F_Q^M(X,Y,Z)$ exceeds $F_Q^L(X,Y,Z)^{RPS}$ limit.

6. $F_Q^M(X,Y,Z)$ Penalty Factors for Technical Specification Surveillances 3/4.2.2 is provided in Table 6.

2.11 Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}^{M}(X,Y)$ (Specification 3/4.2.3)

1. The $F_{\Delta H}^{L}(X,Y)$ Steady-State Limit as referenced in TS 3/4.2.3 is:

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

   where:

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

   b. $MARP\ (X,Y)$ = Cycle-specific operating limit Maximum Allowable Radial Peaks. $MARP\ (X,Y)$ radial peaking limits are provided in Table 5.

   c. RRH = 2.857, (0.0 < P < 1.0) RRH is the Thermal Power reduction required to compensate for each 1% measured radial peak, $F_{\Delta H}^{M}(X,Y)$, exceeds the limit.

   d. P = $\frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

2. The $[F_{\Delta H}^{L}(X,Y)]^{SURV}$ Transient Operational Limit as referenced in TS 3/4.2.3 is:

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

   where:

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

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

   c. $M_{\Delta H}(X,Y)$ = Margin remaining in core location X,Y relative to the Operational DNB limits in the transient power distribution. $M_{\Delta H}(X,Y)$ is provided in Appendix A Table A-3 for normal operation and in Appendix A Table A-6 for power escalation testing during initial startup operation.

   d. UMR = 1.0 (Uncertainty value for measured radial peaks). UMR is 1.0 since a factor of 1.04 is implicitly included in the variable $M_{\Delta H}(X,Y)$.

3. TRH = 0.02

   where:

   TRH is the OTΔT K_1 setpoint (Specification 2.2.1) reduction required to compensate for each 1% measured radial peak, $F_{\Delta H}^{M}(X,Y)$ exceeds its limit.

4. $F_{\Delta H}^{M}(X,Y)$ Penalty Factors for Technical Specification Surveillances 3/4.2.3 is provided in Table 6.

2.12 Boron Concentration During Refueling Operations (Specification 3/4.9.1.a)

Through the end of Cycle 23, the boron concentration required to maintain K_{eff} less than or equal to 0.95 is equal to 2178 ppm. Boron concentration must be maintained greater than or equal to 2178 ppm during refueling operations.

2.13 Reactor Coolant System DNB Parameters (Specification 3/4.2.5)

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

2.14 Accumulators - Max and Min Boron Concentration (Specification 3/4.5.1)

The Accumulators boron concentration limits in MODES 1, 2, and 3 is:

Accumulators minimum boron concentration = 2424 ppm *
Accumulators maximum boron concentration = 2574 ppm *

* These Boron Concentrations include a 1% Boron Measurement Uncertainty

2.15 Refueling Water Storage Tank - Max and Min Boron Concentration (Specification 3/4.5.4)

The Refueling Water Storage Tank (RWST) boron concentration limits in MODES 1, 2, 3, and 4 is:

RWST minimum boron concentration = 2424 ppm *
RWST maximum boron concentration = 2574 ppm *

* These Boron Concentrations include a 1% Boron Measurement Uncertainty

Table 1 - Overtemperature ΔT Setpoint Parameter Values (Specification 2.2.1)

Parameter	**Nominal Value**
Reference Tavg at RTP	T' ≤ 588.8 °F
Normal RCS Operating Pressure	P' = 2,235 psig
Overtemperature ΔT reactor trip setpoint coefficient	K1 ≤ 1.175
Overtemperature ΔT reactor trip heatup setpoint penalty coefficient	K2 = 0.0224 / °F
Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient	K3 = 0.001 / psig
Time constants utilized in lead-lag compensator for ΔT	τ1 = 0.0 sec τ2 = 0.0 sec
Time constant utilized in the lag compensator for ΔT	τ3 ≤ 4.0 sec
Time constants utilized in the lead-lag compensator for Tavg	τ4 ≥ 22.0 sec τ5 ≤ 4.0 sec
Time constant utilized in the measured Tavg lag compensator	τ6 = 0.0 sec
f1(ΔI) "positive" breakpoint	9 %ΔI
f1(ΔI) "negative" breakpoint	-21 %ΔI
f1(ΔI) "positive" slope	1.712 %ΔT / %ΔI
f1(ΔI) "negative" slope	3.18 %ΔT / %ΔI

Table 2 - Overpower ΔT Setpoint Parameter Values (Specification 2.2.1)

Parameter	**Nominal Value**
Reference T_{avg} at RTP	T" ≤ 588.8 °F
Overpower ΔT reactor trip setpoint coefficient	K4 ≤ 1.10
Overpower ΔT reactor trip penalty coefficient	K5 = 0.02 / °F for increasing Tavg K5 = 0.0 / °F for decreasing Tavg
Overpower ΔT reactor trip heatup setpoint penalty coefficient	K6 = 0.002 / °F for T > T" K6 = 0.0 / °F for T ≤ T"
Time constant utilized in the rate-lag compensator for T_{avg}	τ7 ≥ 13.0 sec
f2(ΔI) "positive" breakpoint	21 %ΔI
f2(ΔI) "negative" breakpoint	-21 %ΔI
f2(ΔI) "positive" slope	3.5 %ΔT / %ΔI
f2(ΔI) "negative" slope	3.5 %ΔT / %ΔI

Table 3 - Thermal Power and AFD Limit Reductions Required When $F_Q^L(X,Y,Z)^{OP}$ is Exceeded (Specification 3/4.2.2)

Negative Margin (%)	Power (%)	AFD Limit Change (%)	
		Negative Limit	Positive Limit
< 2.0	100	≥ 3	≥ 4
≥ 2.0 and < 4.0	97	≥ 4	≥ 8
≥ 4.0 and < 6.0	94	≥ 6	≥ 8

Note – Confirm positive margin exists at the reduced AFD limits by recalculating margin using updated Monitor Factors. If the out-of-limit condition is not resolved, reduce THERMAL POWER by greater than 3% for each 1% of negative margin.

Table 4 - Reactor Coolant System DNB Parameters (Specification 3/4.2.5)

Parameter	Indication	No. Operable Channels	Limits
Indicated RCS Average Temperature	MCB	3	≤ 591.4 °F
	MCB	2	≤ 591.1 °F
	MCB	1	≤ 590.2 °F
	ERFIS	3	≤ 592.0 °F
	ERFIS	2	≤ 591.8 °F
	ERFIS	1	≤ 591.3 °F
Indicated Pressurizer Pressure	MCB	3	≥ 2205 psig
	MCB	2	≥ 2209 psig
	MCB	1	≥ 2219 psig
	ERFIS	3	≥ 2201 psig
	ERFIS	2	≥ 2205 psig
	ERFIS	1	≥ 2213 psig
RCS Total Flow Rate*			≥ 293,540 gpm

*After subtraction for instrument uncertainty

Table 5 - Maximum Allowable Radial Peaks (MARPs)

HTP Fuel, 100% RTP, Steady State Limits

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**	**3.5**
0.12	1.798	1.798	1.797	1.798	1.870	1.892	1.823	1.753	1.688	1.621	1.547	0.928
1.20	1.798	1.759	1.759	1.759	1.852	1.851	1.781	1.714	1.650	1.564	1.445	0.867
2.40	1.798	1.781	1.759	1.799	1.833	1.757	1.666	1.600	1.540	1.506	1.410	0.846
3.60	1.798	1.798	1.799	1.762	1.826	1.707	1.633	1.566	1.525	1.488	1.426	0.856
4.80	1.797	1.785	1.778	1.770	1.826	1.752	1.693	1.621	1.557	1.497	1.394	0.836
6.00	1.795	1.785	1.773	1.768	1.869	1.823	1.749	1.681	1.618	1.559	1.458	0.875
7.20	1.786	1.785	1.749	1.717	1.786	1.820	1.773	1.726	1.680	1.634	1.566	0.940
8.40	1.765	1.784	1.747	1.717	1.715	1.688	1.646	1.633	1.615	1.573	1.502	0.901
9.60	1.733	1.780	1.737	1.680	1.638	1.612	1.589	1.582	1.544	1.506	1.437	0.862
10.80	1.676	1.706	1.734	1.695	1.650	1.608	1.568	1.530	1.494	1.459	1.398	0.839
12.00	1.585	1.560	1.513	1.469	1.426	1.385	1.348	1.313	1.280	1.249	1.188	0.713

Table 6 - $F_Q(X,Y,Z)$ and $F_{\Delta H}(X,Y)$ Penalty Factors

For Technical Specification Surveillances 3/4.2.2 and 3/4.2.3

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	**2.43**	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
492	2.00	2.00
512	2.00	2.00
529	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 Tech Spec Surveillances 3/4.2.2 and 3/4.2.3.

3.0 METHODOLOGY REFERENCES

1. XN-75-27(P)(A) (June 1975) and Supplements 1 (September 1976), 2 (December 1977), 3 (November 1980), 4 (December 1985), and 5 (February 1987), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," Exxon Nuclear Company, Richland, WA 99352. **(Not used for Cycle 23)**

   (Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - Modes 3, 4, and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

2. ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, Richland, WA 99352, May 1992. **(Not used for Cycle 23)**

   (Methodology for Specification 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

3. XN-NF-82-21(P)(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Richland, WA 99352, September 1983. **(Not used for Cycle 23)**

   (Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

4. XN-75-32(P)(A), (April 1975) Supplements 1 (July 1979), 2 (July 1979), 3 (January 1980), and 4 (October 1983), "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company, Richland, WA 99352.

   (Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

5. EMF-84-093(P)(A), Revision 1, "Steam line Break Methodology for PWRs," Siemens Power Corporation, May 1999. **(Not used for Cycle 23)**

   (Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

6. ANP-3011(P), Revision 1, "Harris Nuclear Plant Unit 1 Realistic Large Break LOCA Analysis," as approved by NRC Safety Evaluation dated May 30, 2012, issued August 2011.

(Methodology for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

7. XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors," Exxon Nuclear Company, Richland, WA 99352, October 1983. **(Not used for Cycle 23)**

(Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, and 3.2.2 - Heat Flux Hot Channel Factor).

8. ANF-88-054(P)(A), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation, Richland, WA 99352, October 1990. **(Not used for Cycle 23)**

(Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor).

9. EMF-92-081(P)(A), Revision 1, "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," Siemens Power Corporation, July 2000. **(Not used for Cycle 23)**

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

10. EMF-92-153(P)(A), Revision 1, "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," Siemens Nuclear Power Corporation, Richland, WA 99352, January 2005.

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

11. BAW-10240 (P)(A),Revision 0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," Framatome ANP, May 2004

(Methodology for Specification 2.1.1 – Reactor Core Safety Limits, 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4 and 5, 3.1.1.3- Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6- Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.5 – DNB Parameters, and 3.9.1 - Boron Concentration).

12. EMF-96-029(P)(A), Volumes 1 and 2, "Reactor Analysis Systems for PWRs, Volume 1 - Methodology Description, Volume 2 - Benchmarking Results," Siemens Power Corporation, January 1997. **(Not used for Cycle 23)**

    (Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4, and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration).

13. EMF-2328(P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," Framatone ANP, May 2001, and Errata, January 2008.

    (Methodology for Specification 3.2.1 - Axial Flux Difference, and 3.2.2 - Heat Flux Hot Channel Factor), and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

14. EMF-2310 (P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP, May 2004. **(Not used for Cycle 23)**

    (Methodology for Specification 2.2.1 – Reactor Trip System Instrumentation Setpoints, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1- Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.5 – DNB Parameters).

15. Mechanical Design Methodologies
XN-NF-81-58(P)(A), Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, March 1984. **(Not used for Cycle 23)**

ANF-81-58(P)(A), Revision 2 and Supplements 3 and 4, "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," Advanced Nuclear Fuels Corporation, June 1990. **(Not used for Cycle 23)**

XN-NF-82-06(P)(A), Revision 1 and Supplements 2, 4, and 5, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986. **(Not used for Cycle 23)**

ANF-88-133(P)(A), and Supplement 1, "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," Advanced Nuclear Fuels Corporation, December 1991. **(Not used for Cycle 23)**

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986. **(Not used for Cycle 23)**

EMF-92-116(P)(A), Revision 0 and Supplement 1(P)(A)-000, "Generic Mechanical Design Criteria for PWR Fuel Designs," Siemens Power Corporation, February 1999 and May 2015. **(Not used for Cycle 23)**

BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code,” Framatome ANP, Inc, January 2004

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

16. DPC-NE-2005-P-A, Revision 5, "Thermal-Hydraulic Statistical Core Design Methodology," NRC Safety Evaluation: ML 16049A630

(Methodology for Specification 3.2.3 -Nuclear Enthalpy Rise Hot Channel Factor)

17. DPC-NE-1008-P-A, Revision 0, "Nuclear Design Methodology Using CASMO-5/SIMULATE-3 for Westinghouse Reactors," NRC Safety Evaluation: ML17102A923.

(Methodology for Specification 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4, and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor, and 3.9.1 - Boron Concentration)

18. DPC-NF-2010-A, Revision 3, "Nuclear Physics Methodology for Reload Design," NRC Safety Evaluation: ML17102A923.

    (Methodology for Specification 3.1.1.1 - SHUTDOWN MARGIN - MODES 1 and 2, 3.1.1.2 - SHUTDOWN MARGIN - MODES 3, 4, and 5, 3.1.1.3 - Moderator Temperature Coefficient, 3.1.2.5 - Borated Water Source – Shutdown, 3.1.2.6 – Borated Water Sources – Operating, 3.1.3.5 - Shutdown Bank Insertion Limits, 3.1.3.6 - Control Bank Insertion Limits, 3.5.1 – ECCS Accumulators – Cold Leg Injection, 3.5.4 – ECCS Refueling Water Storage Tank and 3.9.1 - Boron Concentration)

19. DPC-NE-2011-P-A, Revision 2, "Nuclear Design Methodology Report for Core Operating Limits of Westinghouse Reactors," NRC Safety Evaluation: ML17102A923.

    (Methodology for Specification 3.1.3.5 - Shutdown Bank Insertion Limits,3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor)

20. DPC-NE-3008-P-A, Revision 0, “Thermal-Hydraulic Models for Transient Analysis,” as approved by NRC Safety Evaluation dated April 10, 2018.

    (Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor)

21. DPC-NE-3009-P-A, Revision 0, “FSAR / UFSAR Chapter 15 Transient Analysis Methodology,” as approved by NRC Safety Evaluation dated April 10, 2018.

    (Methodology for Specification 3.1.1.3 – Moderator Temperature Coefficient, 3.1.3.5 – Shutdown Bank Insertion Limits, 3.1.3.6 – Control Bank Insertion Limits, 3.2.1 – Axial Flux Difference, 3.2.2 – Heat Flux Hot Channel Factor, and 3.2.3 – Nuclear Enthalpy Rise Hot Channel Factor)

22. DPC-NE-2004P-A, Revision 2a, “Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology using VIPRE-01,” NRC Safety Evaluation: ML17102A923.

    (Methodology approved for use at Harris Nuclear Plant per License Amendment No.157)

4.0 OTHER REQUIREMENTS

The following requirement is not identified per Technical Specification 6.9.1.6 to appear in the COLR but is presented in compliance with Amendment No. 65 issued July 24, 1996 which relocated the Movable Incore Detection System from Technical Specification 3.3.3.2 to the COLR.

4.1 Movable Incore Detection System

1. Functionality: The Movable Incore Detection System shall be functional with:
   a. At least 38 detector thimbles at the beginning of cycle (where the beginning of cycle is defined in this instance as a flux map determination that the core is loaded consistent with design),

   b. A minimum of 38 detector thimbles for the remainder of the operating cycle,

   c. A minimum of two detector thimbles per core quadrant, and

   d. Sufficient movable detectors, drive, and readout equipment to map these thimbles.

2. Applicability: When the Movable Incore Detection System is used for:

   a. Recalibration of the Excore Neutron Flux Detection System, or

   b. Monitoring the QUADRANT POWER TILT RATIO, or

   c. Measurement of $F_{\Delta H}^{M}(X,Y)$ and $F_{Q}^{M}(X,Y,Z)$

3. Surveillance Requirements: The Movable Incore Detection System shall be demonstrated functional, within 24 hours prior to use, by irradiating each detector used and determining the acceptability of its voltage curve when required for:

   a. Recalibration of the Excore Neutron Flux Detection System, or

   b. Monitoring the QUADRANT POWER TILT RATIO, or

   c. Measurement of $F_{\Delta H}^{M}(X,Y)$ and $F_{Q}^{M}(X,Y,Z)$

4.1 Movable Incore Detection System (continued)

4. Bases:

The functionality of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the core. The functionality of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_{\Delta H}^{M}(X,Y)$ and $F_{Q}^{M}(X,Y,Z)$, a full incore flux map is used.

Quarter core flux maps, as defined in WCAP 8648, June 1976, may be used in recalibration of the Excore Neutron Flux Detection System, and full incore flux maps or symmetric incore thimbles may be used for monitoring QUADRANT POWER TILT RATIO when one Power Range channel is INOPERABLE.

5. Evaluation Requirements:

In order to change the requirements concerning the number and location of functional detectors, the NRC staff deems that a rigorous evaluation and justification is required. The following is a list of elements that must be part of a 50.59 determination and available for audit if the licensee wishes to change the requirements:

a. How an inadvertent loading of a fuel assembly into an improper location will be detected,

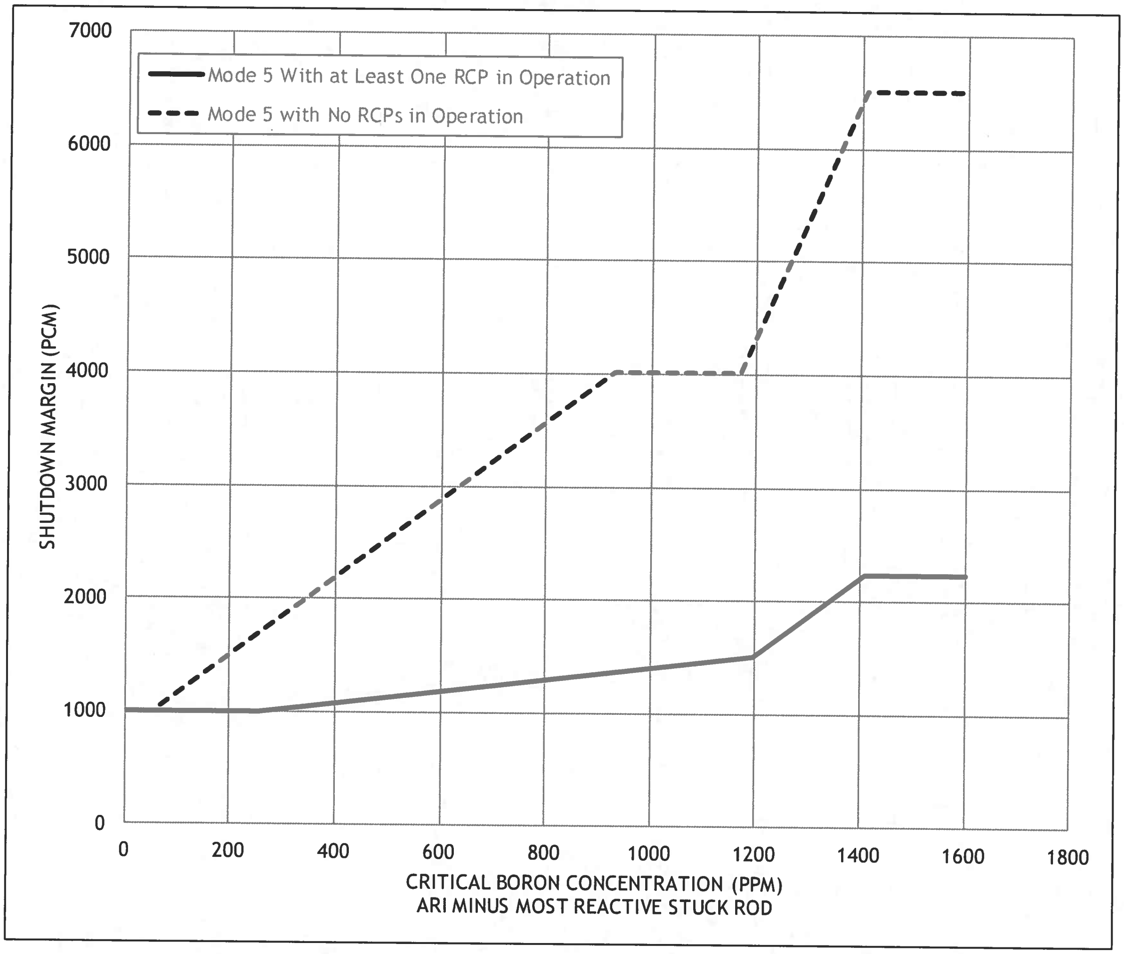
b. How the validity of the tilt estimates will be ensured,

c. How adequate core coverage will be maintained,

d. How the measurement uncertainties will be assured and why the added uncertainties are adequate to guarantee that measured nuclear heat flux hot channel factor, nuclear enthalpy rise hot channel factor, radial peaking factor and quadrant power tilt factor meet Technical Specification limits, and

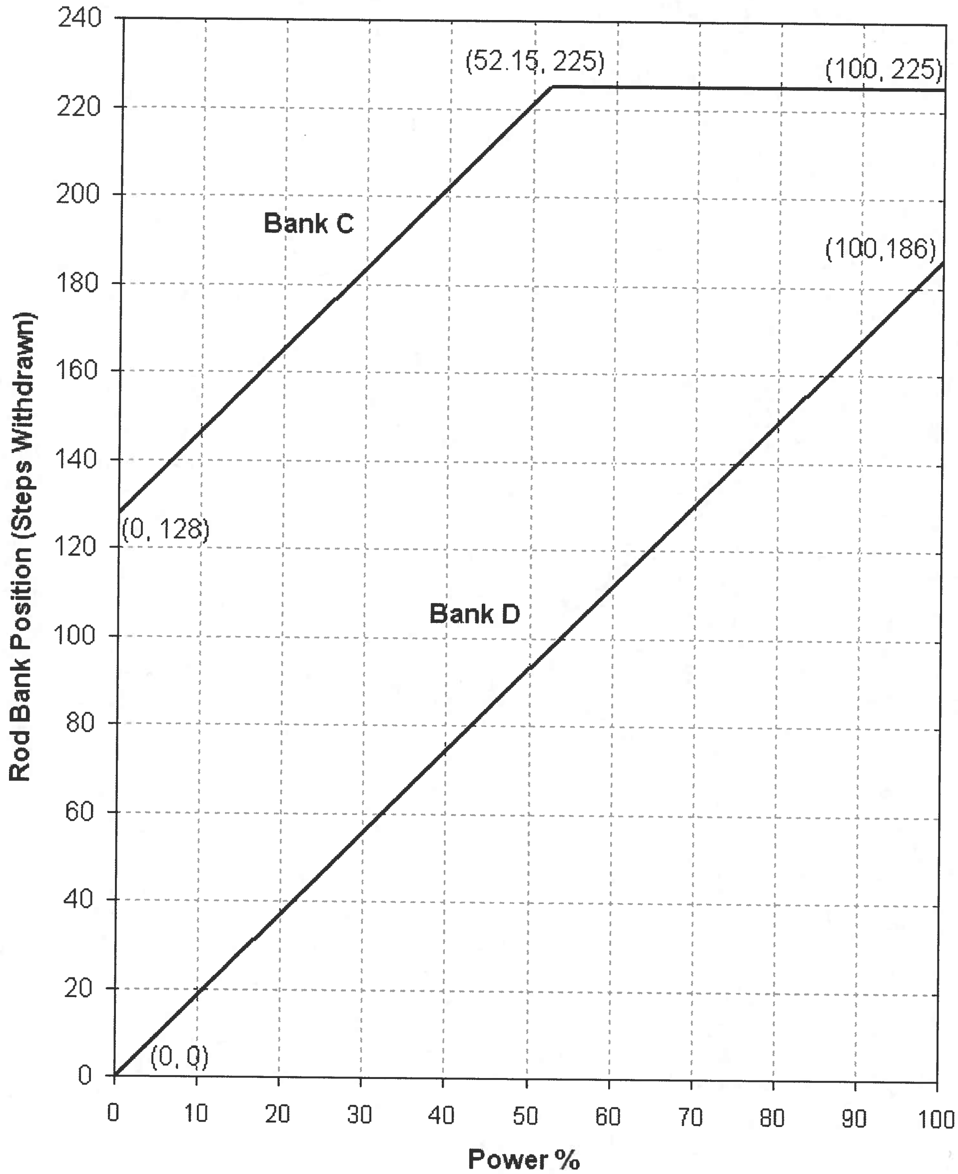
e. How the Movable Incore Detection System will be restored to full (or nearly full) service before the beginning of each cycle.

Figure 1, Shutdown Margin Versus ARI-1 Critical Boron Concentration Mode 5



DATA POINTS FOR FIGURE 1		
CONFIGURATION	**ARI-1 CRITICAL BORON (PPM)**	**SHUTDOWN MARGIN (PCM)**
Mode 5 No RCPs in Operation	0	1000
	55	1000
	930	4010
	1170	4010
	1410	6510
	1600	6510
Mode 5 At Least 1 RCP in Operation	0	1000
	255	1000
	1195	1500
	1410	2230
	1600	2230

Figure 2, Rod Group Insertion Limits Versus Thermal Power (Three Loop Operation)



Notes:

1. Fully withdrawn position shall be greater than or equal to 225 steps.
2. Control Banks A and B must be withdrawn from the core prior to power operation.

Figure 3, Axial Flux Difference Limits as a Function of Rated Thermal Power

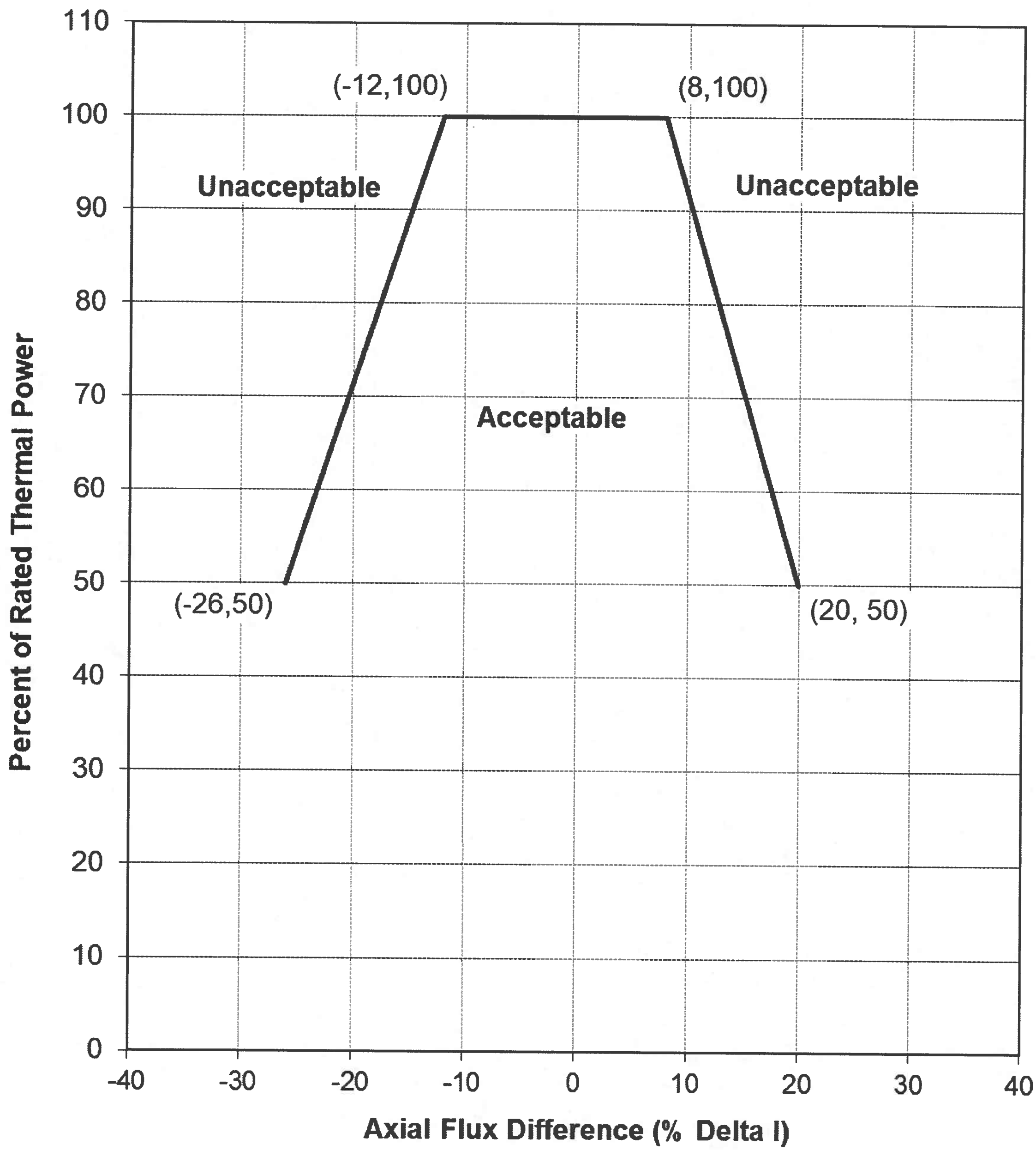


Figure 4, K(Z) Local Axial Penalty Function for F_Q (X,Y,Z)

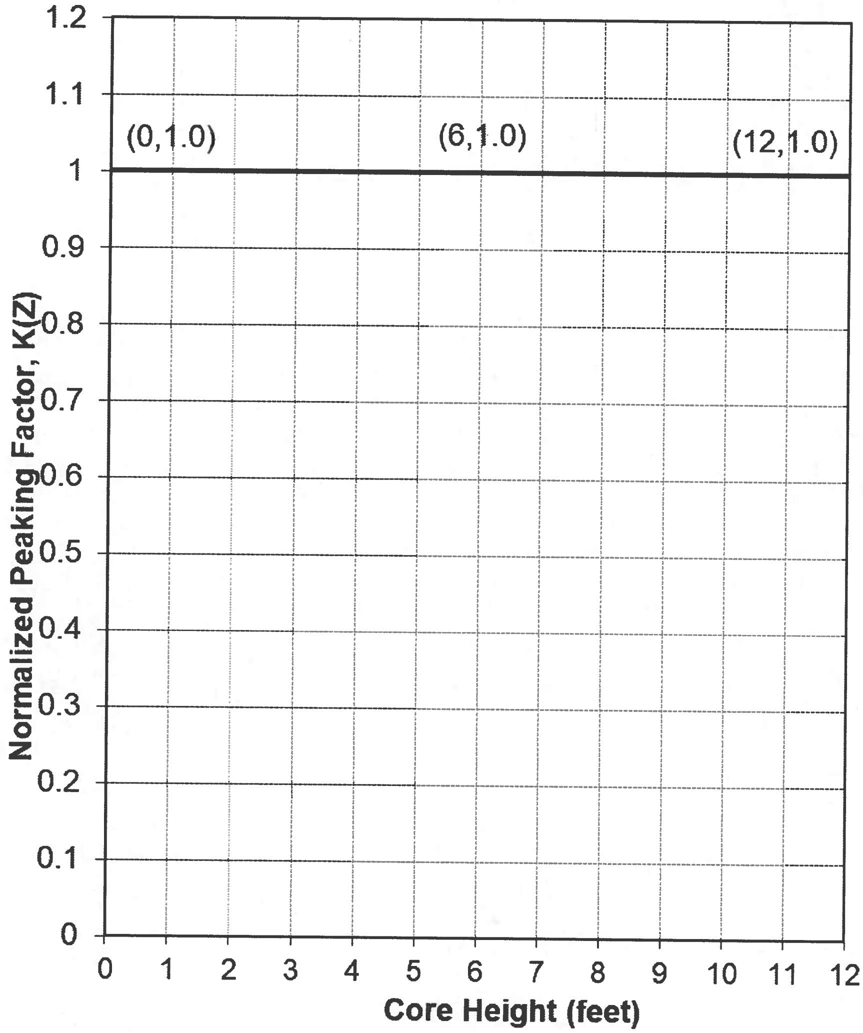
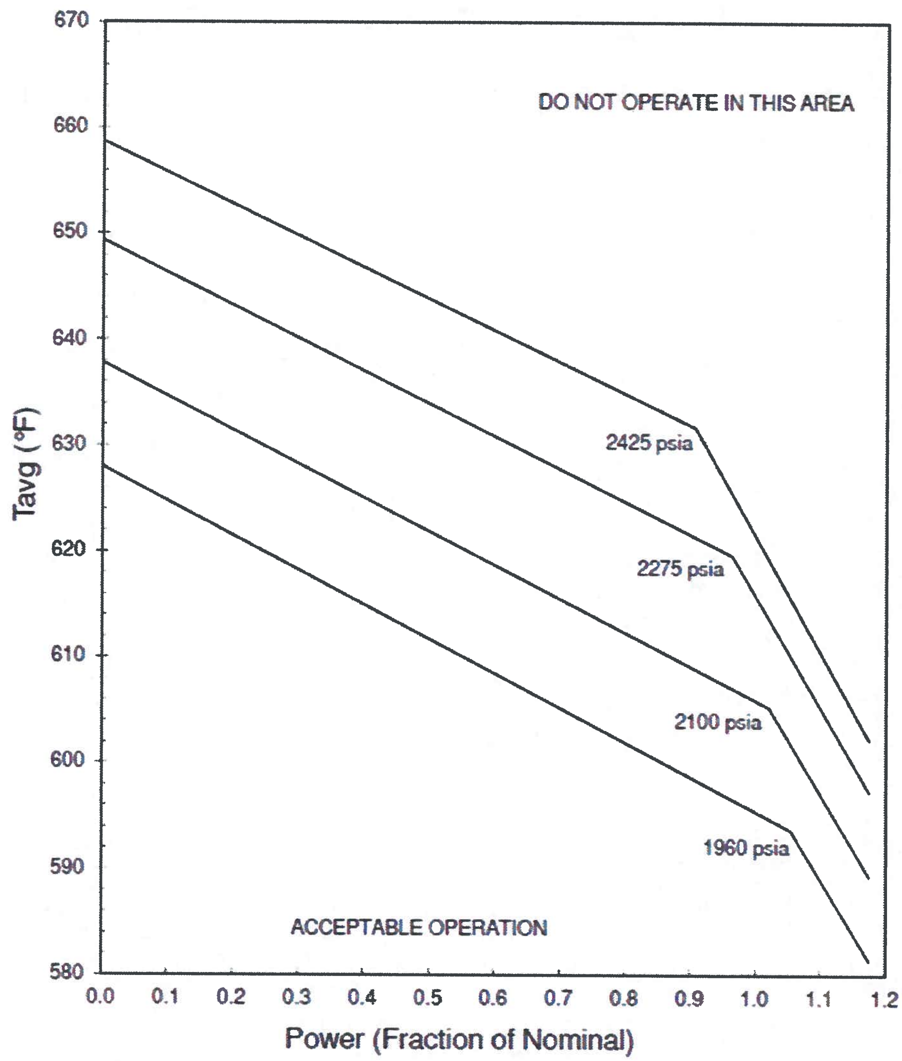


Figure 5, Reactor Core Safety Limits
Three Loops in Operation



Note: The safety limit lines correspond to the pressure at the core exit since DNBR calculations are performed using core exit pressure. To convert to pressurizer pressure, subtract 25 psi for the pressure loss between the core exit and the pressurizer pressure and subtract 15 psi to convert psia to psig.

Appendix A

Power Distribution Monitoring Factors

Appendix A contains power distribution monitoring factors used in Technical Specification Surveillance. This data was generated in the Harris Cycle 23 Maneuvering Analysis calculation file, HNP-F/NFSA-0349. 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 available to be transmitted to the NRC.