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Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

AUG 0 8 2005

L-PI-05-071
TS 5.6.5.d

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

Prairie Island Nuclear Generating Plant Unit 1
Docket 50-282
License No. DPR-42

Core Operating Limits Report (COLR) for Prairie Island Unit 1 Cycle 23, Revision 1

Pursuant to the requirements of Technical Specification 5.6.5.d, the COLR for Prairie Island Unit 1 Cycle 23, Revision 1 is attached. The limits specified in the attached COLR have been established using Nuclear Regulatory Commission (NRC) approved methodologies.

The Unit 1 COLR has been revised for Cycle 23 to incorporate the following changes:

- Revised Isothermal Temperature Coefficient (ITC) upper limit from <0 pcm/°F for power levels >70% Rated Thermal Power (RTP) to less than a line that slopes linearly from 0 pcm/°F at 70% RTP to –2.9 pcm/°F at 100% RTP.
- Revised the title of Figure 3 to reference Technical Specification 3.1.4 Condition B.
- Revised the title of Figure 4 to reference Technical Specification 3.1.4 Condition A.
- Added references 24 and 25 to include the 50.59 screenings written to issue Revision 1.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

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Enclosure (1)

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ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
CORE OPERATING LIMITS REPORT
UNIT 1 – CYCLE 23
REVISION 1

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PRAIRIE ISLAND NUCLEAR GENERATING PLANT

CORE OPERATING LIMITS REPORT

UNIT 1 – CYCLE 23

REVISION 1

Reviewed By: Jon Kapitz Date: 7/7/05
Jon Kapitz
Supervisor, Nuclear Engineering

Reviewed By: Ed Mercier Date: 6/22/05
Ed Mercier
Supervisor, PWR Analysis

Approved By: T. Silverberg Date: 7/11/05
Terry Silverberg
Director, Engineering

Note: This report is not part of the Technical Specifications
This report is referenced in the Technical Specifications

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

CORE OPERATING LIMITS REPORT

UNIT 1 - CYCLE 23

REVISION 1

This report provides the values of the limits for Unit 1 Cycle 23 as required by Technical Specification Section 5.6.5. These values have been established using NRC approved methodology and are established such that all applicable limits of the plant safety analysis are met. The Technical Specifications affected by this report are listed below:

1. 2.1.1 Reactor Core SLs
2. 3.1.1 Shutdown Margin (SDM)
3. 3.1.3 Isothermal Temperature Coefficient (ITC)
4. 3.1.5 Shutdown Bank Insertion Limits
5. 3.1.6 Control Bank Insertion Limits
6. 3.1.8 Physics Tests Exceptions - MODE 2
7. 3.2.1 Heat Flux Hot Channel Factor ($F_Q(z)$)
8. 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor $\left(F_{\Delta H}^{N}\right)$
9. 3.2.3 Axial Flux Difference (AFD)
10. 3.3.1 Reactor Trip System (RTS) Instrumentation
    Overtemperature ΔT and Overpower ΔT Parameter Values for Table 3.3.1-1
11. 3.4.1 RCS Pressure, Temperature, and Flow - Departure from Nucleate
    Boiling (DNB) Limits
12. 3.9.1 Boron Concentration

1. 2.1.1 Reactor Core Safety Limits

Reactor Core Safety Limits are shown in Figure 1.

Reference Technical Specification section 2.1.1.

2. 3.1.1 Shutdown Margin Requirements

Minimum Shutdown Margin requirements are shown in Table 1.

Reference Technical Specification section 3.1.1.

3. 3.1.3 Isothermal Temperature Coefficient (ITC)

ITC Upper limit:

a. < 5 pcm/°F for power levels < 70% RTP; and
b. less than a line which slopes linearly from
   i. 0 pcm/°F at power level = 70% RTP to
   ii. –2.9 pcm/°F at power level = 100% RTP

ITC Lower limit:

a. -32.7 pcm/°F

Reference Technical Specification section 3.1.3.

4. 3.1.5 Shutdown Bank Insertion Limits

The shutdown rods shall be fully withdrawn.

Reference Technical Specification section 3.1.5.

5. 3.1.6 Control Bank Insertion Limits

The control rod banks shall be limited in physical insertion as shown in Figures 2, 3, and 4.

The control rod banks withdrawal sequence shall be Bank A, Bank B, Bank C, and finally Bank D.

The control rod banks shall be withdrawn maintaining 128 step tip-to-tip distance.

Reference Technical Specification section 3.1.6.

6. 3.1.8 Physics Tests Exceptions - MODE 2

Minimum Shutdown Margin requirements during physics testing are shown in Table 1.

Reference Technical Specification section 3.1.8.

7. 3.2.1 Heat Flux Hot Channel Factor ($F_Q(Z)$)

The Heat Flux Hot Channel Factor shall be within the following limits:

CFQ = 2.50

K(Z) is a constant value = 1.0 at all elevations.

W(Z) values are provided in Table 2.

$F^W_Q(Z)$ Penalty Factors are provided in Table 3.

Applicability: MODE 1.

Reference Technical Specification section 3.2.1

8. 3.2.2 Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$)

The Nuclear Enthalpy Rise Hot Channel Factor shall be within the following limit:

$$F_{\Delta H} \leq 1.77 \times [1 + 0.3(1 - P)]$$

where: P is the fraction of RATED THERMAL POWER at which the core is operating.

Applicability: MODE 1.

Reference Technical Specification section 3.2.2

9. 3.2.3 Axial Flux Difference (AFD)

The indicated axial flux difference, in % flux difference units, shall be maintained within the allowed operational space defined by Figure 5.

Applicability: MODE 1 with RATED THERMAL POWER $\geq$ 50% RTP.

Reference Technical Specification sections 3.2.3.

10. 3.3.1 Reactor Trip System (RTS) Instrumentation

Overtemperature ΔT and Overpower ΔT Parameter Values for Table 3.3.1-1;

Overtemperature ΔT Setpoint

Overtemperature ΔT setpoint parameter values:

ΔT_0	=	Indicated ΔT at RATED THERMAL POWER, %
T	=	Average temperature, °F
T′	=	560.0 °F
P	=	Pressurizer Pressure, psig
P′	=	2235 psig
K_1	$\leq$	1.17
K_2	=	0.014 /°F
K_3	=	0.00100 /psi
τ_1	=	30 seconds
τ_2	=	4 seconds
$f(\Delta I)$	=	A function of the indicated difference between top and bottom detectors of the power range nuclear ion chambers. Selected gains are based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of RATED THERMAL POWER, such that (a) For q_t - q_b within –13, +8 % $f(\Delta I) = 0$ (b) For each percent that the magnitude of q_t - q_b exceeds +8% the ΔT trip setpoint shall be automatically reduced by an equivalent of 1.73 % of RATED THERMAL POWER. (c) For each percent that the magnitude of q_t - q_b exceeds -13 % the ΔT trip setpoint shall be automatically reduced by an equivalent of 3.846 % of RATED THERMAL POWER.

Overpower ΔT Setpoint

Overpower ΔT setpoint parameter values:

ΔT_0	=	Indicated ΔT at RATED THERMAL POWER, %
T	=	Average temperature, °F
T′	=	560.0 °F
K_4	$\leq$	1.11
K_5	=	0.0275/°F for increasing T; 0 for decreasing T
K_6	=	0.002/°F for T > T′ ; 0 for T $\leq$ T′
τ_3	=	10 seconds

11. 3.4.1 RCS Pressure, Temperature, and Flow - Departure from Nucleate Boiling (DNB) Limits

Pressurizer pressure limit = 2205 psia
RCS average temperature limit = 564°F
RCS total flow rate limit = 178,000 gpm

Reference Technical Specification section 3.4.1.

12. 3.9.1, Boron Concentration.

The boron concentration of the reactor coolant system and the refueling cavity shall be sufficient to ensure that the more restrictive of the following conditions is met:

a) $K_{eff} \leq 0.95$
b) 2000 ppm
c) The Shutdown Margin specified in Table 1

Reference Technical Specification section 3.9.1.

REFERENCES

1. NSPNAD-8101-A, "Qualification of Reactor Physics Methods for Application to Prairie Island," Revision 2, October 2000.

2. NSPNAD-8102-PA, "Prairie Island Nuclear Power Plant Reload Safety Evaluation Methods for Application to PI Units," Revision 7, July 1999.

3. NSPNAD-97002-PA, "Northern States Power Company's "Steam Line Break Methodology," Revision 1, October 2000.

4. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July, 1985.

5.a WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code," August, 1985.

5.b WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code," Addendum 2 Revision 1, July 1997.

6.a WCAP-10924-P-A, "Westinghouse Large Break LOCA Best Estimate Methodology," Revision 1, Volume 1 Addendum 1,2,3, December 1988.

6.b WCAP-10924-P-A, "Westinghouse Large Break LOCA Best Estimate Methodology," Revision 2 , Volume 2 Addendum 1, December 1988.

6.c WCAP-10924-P-A, "Westinghouse Large Break LOCA Best Estimate Methodology," Revision 1, Volume 1 Addendum 4, March 1991.

7. XN-NF-77-57-(A), XN-NF-77-57, Supplement 1 (A), "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase II," May 1981.

8. WCAP-13677-P-A, "10 CFR 50.46 Evaluation Model Report: W-COBRA/TRAC 2-Loop Upper Plenum Injection Model Update to Support ZIRLO™ Cladding Options," February 1994.

9. NSPNAD-93003-A, "Prairie Island Units 1 and 2 Transient Power Distribution Methodology," Revision 0, April 1993.

10. WCAP-10216-P-A, Revision 1A, "Relaxation of Constant Axial Offset Control/ FQ Surveillance Technical Specification," February 1994.

11. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

12. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

13. WCAP-14483-A, "Generic Methodology for Expanded Core Operating Limits Report," January 1999.

14. WCAP-7588 Rev. 1-A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," January 1975.

15. WCAP-7908-A, "FACTRAN – A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," December 1989.

16. WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.

17. WCAP-7979-P-A, "TWINKLE – A Multidimensional Neutron Kinetics Computer Code," January 1975.

18. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," December 1985.

19. WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," January 1990.

20 WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988.

21. WCAP-12910 Rev. 1-A, "Pressurizer Safety Valve Set Pressure Shift," May 1993.

22. WCAP-14565-P-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

23. WCAP-14882-P-A, "RETRAN-02 Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," April 1999.

24. 50.59 Screening 2347, Rev. 0, "Unit 1 Cycle 23 COLR Revision to Figures 3 and 4 Titles."

25. 50.59 Screening 2429, Rev. 0, "Unit 1 Cycle 23 COLR Revision to Section 3.1.3 Isothermal Temperature Coefficient."

Table 1

Minimum Required Shutdown Margin

Plant Conditions	Number of Charging Pumps Running**		
	0-1 Pump	2 Pumps	3 Pumps
Mode 1*	-	-	-
Mode 2*	2.0%	2.0%	2.0%
Mode 3, $T_{ave} \geq 520°F$	2.0%	2.0%	2.0%
Mode 3, $350°F \leq T_{ave} < 520°F$	2.0%	2.0%	2.5%
Mode 4	2.5%	4.5%	7.0%
Mode 5***, $T_{ave} \leq 200°F$	2.5%	5.0%	7.5%
Mode 6, ARI***, $T_{ave} \geq 68°F$	5.129%	5.129%	7.0%
Mode 6, ARO***, $T_{ave} \geq 68°F$	5.129%	6.0%	9.0%
Physics Testing in Mode 2	0.5%	0.5%	0.5%

Operational Mode Definitions, as per TS Table 1.1-1.

* For Mode 1 and Mode 2 with Keff $\geq$ 1.0, the minimum shutdown margin requirements are provided by the Rod Insertion Limits.

** Charging pump(s) in service only pertains to steady state operations. It does not include transitory operations. For example, operations such as starting a second charging pump in order to secure the operating pump would fall under the one pump in service column.

*** These values are also applicable for the Unit 1 Cycle 22 end of cycle

Table 2 - W(z) Values(Top 10% and Bottom 8% excluded)

		Exposure (MWD/MTU)				
		150	4000	12000	16000	18000
	Height ft)	AO = 1.81	AO = -0.94	AO = -3.03	AO = -0.87	AO = -1.14
[Bottom] 1	0.0000	1.0000	1.0000	1.0000	1.0000	1.0000
2	0.2000	1.0000	1.0000	1.0000	1.0000	1.0000
3	0.4000	1.0000	1.0000	1.0000	1.0000	1.0000
4	0.6000	1.0000	1.0000	1.0000	1.0000	1.0000
5	0.8000	1.0000	1.0000	1.0000	1.0000	1.0000
6	1.0000	1.3308	1.2125	1.1540	1.2022	1.2085
7	1.2000	1.3205	1.2042	1.1468	1.1931	1.1983
8	1.4000	1.3095	1.1948	1.1390	1.1835	1.1874
9	1.6000	1.2970	1.1847	1.1312	1.1740	1.1767
10	1.8000	1.2836	1.1741	1.1234	1.1647	1.1662
11	2.0000	1.2693	1.1632	1.1158	1.1556	1.1558
12	2.2000	1.2543	1.1519	1.1085	1.1466	1.1455
13	2.4000	1.2389	1.1409	1.1014	1.1380	1.1354
14	2.6000	1.2231	1.1327	1.0947	1.1297	1.1255
15	2.8000	1.2074	1.1318	1.0880	1.1205	1.1146
16	3.0000	1.1903	1.1317	1.0825	1.1173	1.1102
17	3.2000	1.1768	1.1307	1.0812	1.1187	1.1121
18	3.4000	1.1724	1.1293	1.0852	1.1251	1.1247
19	3.6000	1.1715	1.1274	1.0901	1.1305	1.1370
20	3.8000	1.1700	1.1274	1.0954	1.1343	1.1482
21	4.0000	1.1674	1.1277	1.1002	1.1398	1.1585
22	4.2000	1.1641	1.1269	1.1042	1.1467	1.1673
23	4.4000	1.1598	1.1254	1.1075	1.1520	1.1746
24	4.6000	1.1547	1.1232	1.1101	1.1561	1.1804
25	4.8000	1.1489	1.1203	1.1120	1.1588	1.1847
26	5.0000	1.1422	1.1167	1.1130	1.1608	1.1874
27	5.2000	1.1352	1.1130	1.1140	1.1633	1.1888
28	5.4000	1.1275	1.1120	1.1169	1.1661	1.1887
29	5.6000	1.1180	1.1159	1.1224	1.1675	1.1871
30	5.8000	1.1185	1.1223	1.1310	1.1694	1.1878
31	6.0000	1.1257	1.1309	1.1431	1.1770	1.1964
32	6.2000	1.1325	1.1408	1.1570	1.1894	1.2104
33	6.4000	1.1399	1.1498	1.1698	1.1995	1.2219
34	6.6000	1.1464	1.1580	1.1817	1.2085	1.2320
35	6.8000	1.1519	1.1653	1.1927	1.2160	1.2407
36	7.0000	1.1565	1.1714	1.2022	1.2220	1.2475
37	7.2000	1.1598	1.1765	1.2106	1.2262	1.2524
38	7.4000	1.1618	1.1805	1.2185	1.2284	1.2551
39	7.6000	1.1623	1.1840	1.2250	1.2286	1.2554
40	7.8000	1.1613	1.1860	1.2295	1.2265	1.2533
41	8.0000	1.1586	1.1862	1.2320	1.2219	1.2484
42	8.2000	1.1541	1.1845	1.2322	1.2151	1.2408
43	8.4000	1.1477	1.1809	1.2301	1.2059	1.2305
44	8.6000	1.1391	1.1756	1.2254	1.1929	1.2182
45	8.8000	1.1308	1.1666	1.2198	1.1874	1.2057
46	9.0000	1.1316	1.1631	1.2118	1.1857	1.1896
47	9.2000	1.1446	1.1717	1.2051	1.1811	1.1752
48	9.4000	1.1599	1.1816	1.2083	1.1796	1.1751
49	9.6000	1.1740	1.1907	1.2117	1.1753	1.1735
50	9.8000	1.1892	1.1977	1.2140	1.1759	1.1733
51	10.0000	1.2022	1.2051	1.2195	1.1823	1.1764

52	10.2000	1.2113	1.2143	1.2324	1.1887	1.1786
53	10.4000	1.2230	1.2219	1.2512	1.1949	1.1802
54	10.6000	1.2312	1.2314	1.2691	1.2025	1.1829
55	10.8000	1.2375	1.2377	1.2822	1.2118	1.1894
56	11.0000	1.0000	1.0000	1.0000	1.0000	1.0000
57	11.2000	1.0000	1.0000	1.0000	1.0000	1.0000
58	11.4000	1.0000	1.0000	1.0000	1.0000	1.0000
59	11.6000	1.0000	1.0000	1.0000	1.0000	1.0000
60	11.8000	1.0000	1.0000	1.0000	1.0000	1.0000
[Top] 61	12.0000	1.0000	1.0000	1.0000	1.0000	1.0000

Table 3

$F^{W}_{Q}(Z)$ Penalty Factor

Exposure Range	$F^{W}_{Q}(Z)$ Penalty Factor
BOC – EOC	1.02

Figure 1

Reactor Core Safety Limits

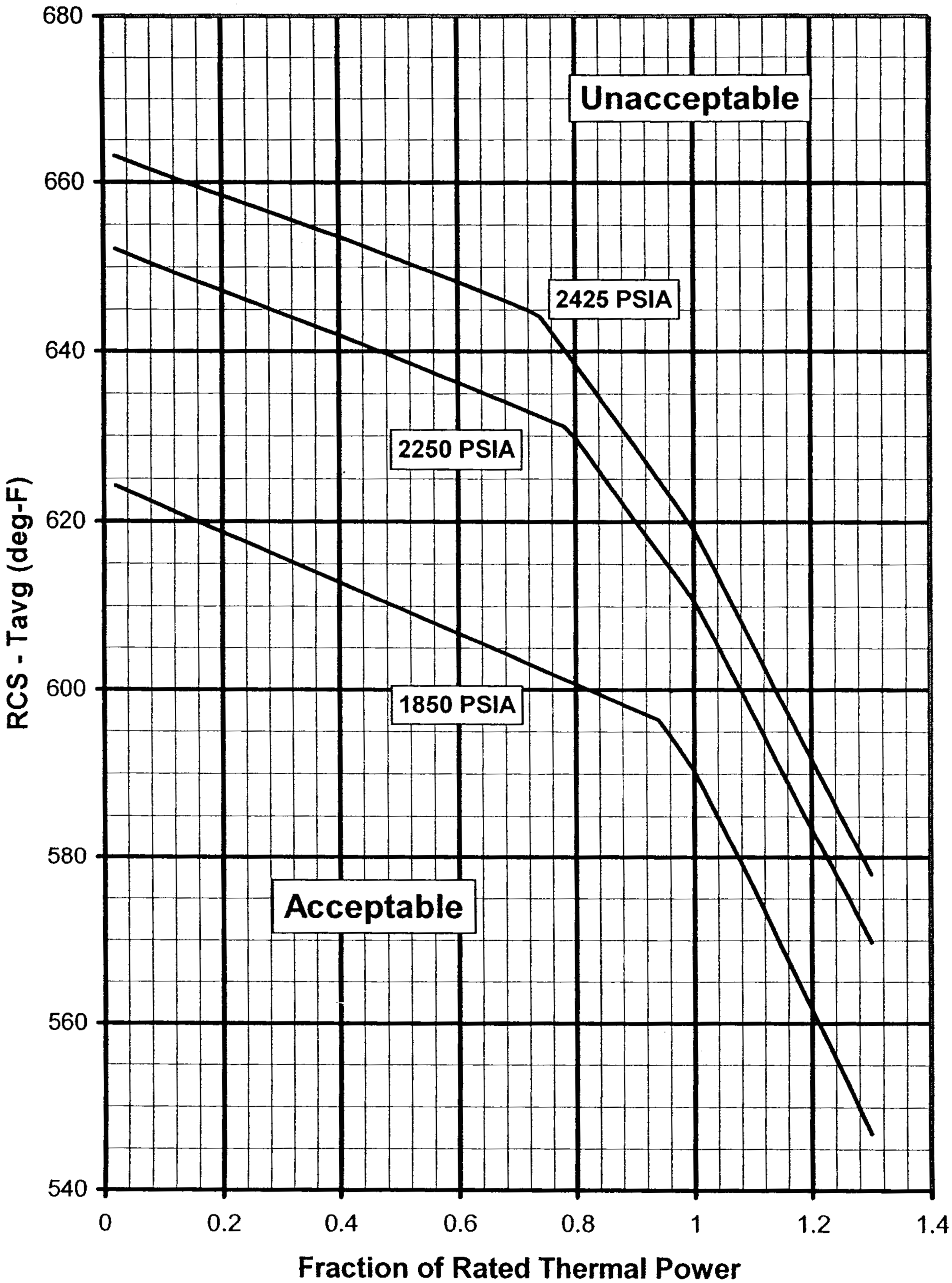
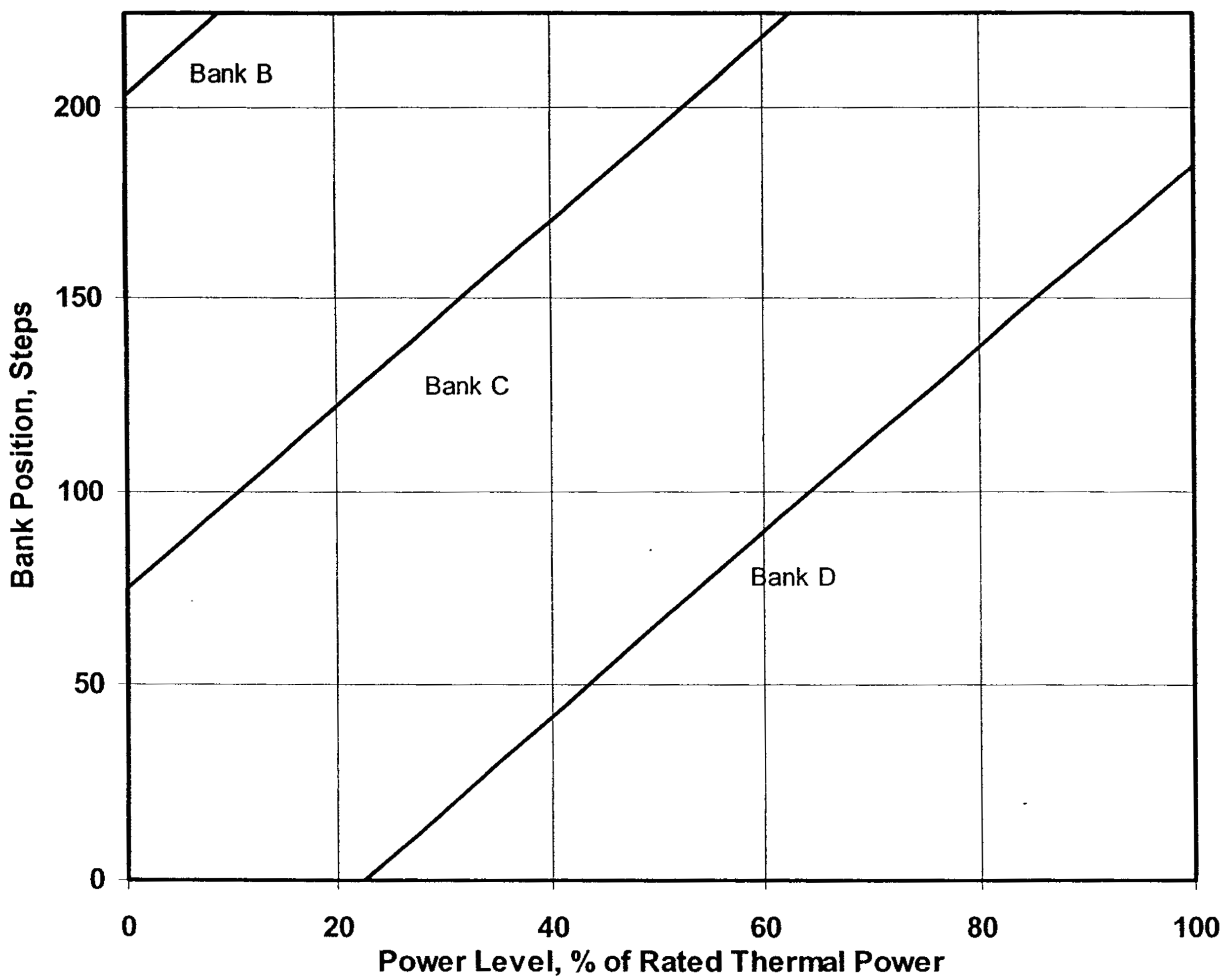


Figure 2
Rod Insertion Limit, 128 Step Tip-to-Tip

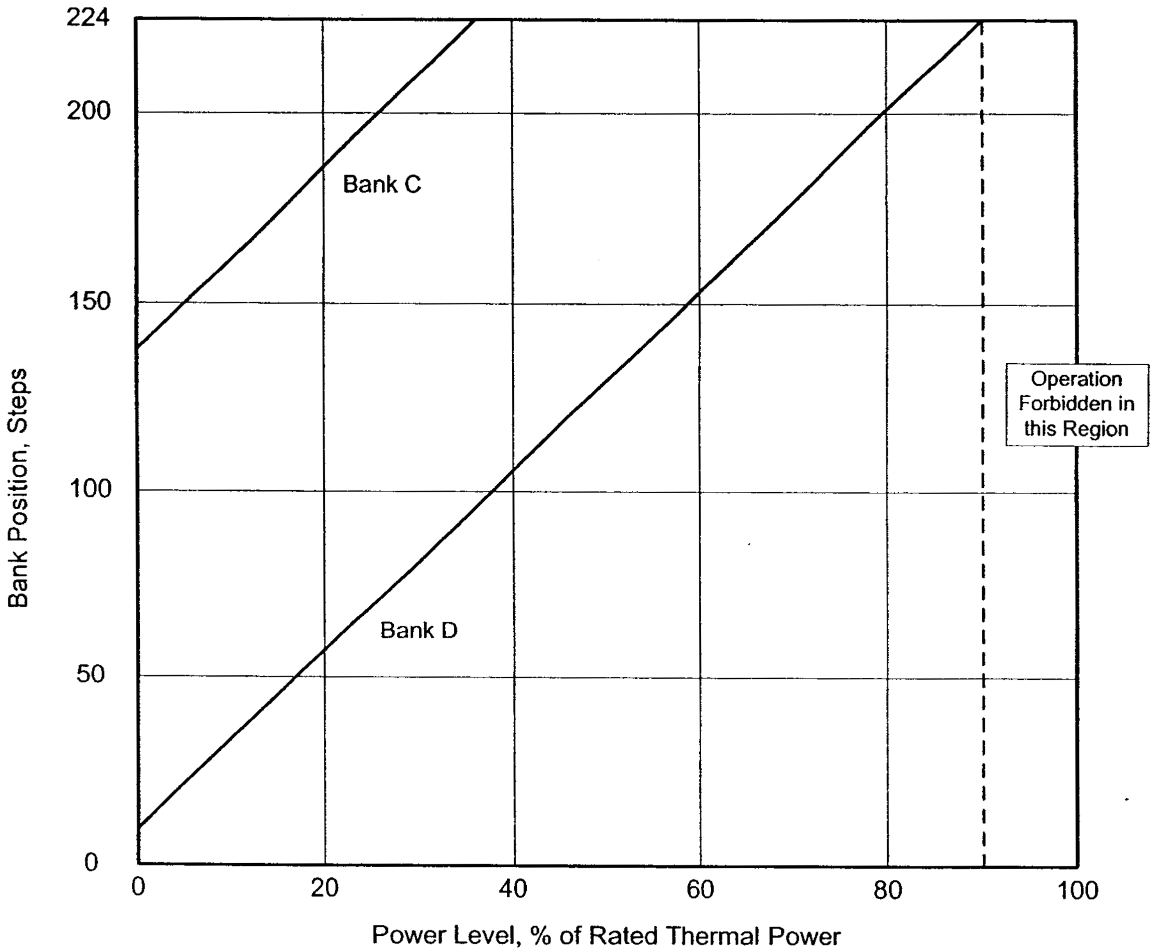


Bank Positions Given By:

- Bank D = (150 / 63) * (P – 100) + 185
- Bank C = (150 / 63) * (P – 100) + 185 + 128
- Bank B = (150 / 63) * (P – 100) + 185 + 128 + 128

NOTE: The top of the active fuel height corresponds to 224 steps. The ARO parking position may be any position above 224 steps.

Figure 3
Rod Insertion Limit, 128 Step Tip-to-Tip, One Bottomed Rod
(Technical Specification 3.1.4 Condition B)

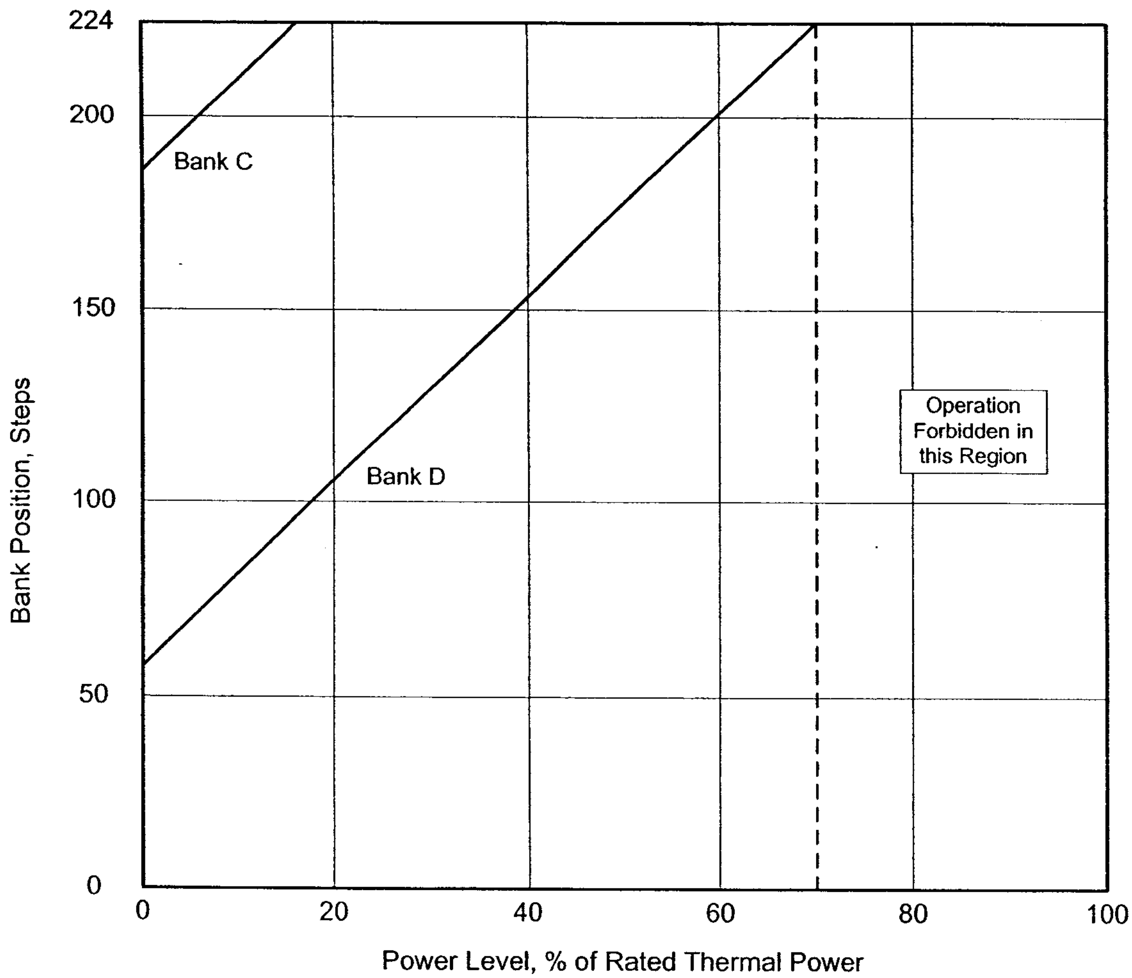


Bank Positions Given By:

- Bank D = (150 / 63) * (P – 90) + 224
- Bank C = (150 / 63) * (P – 90) + 224 + 128

NOTE: The top of the active fuel height corresponds to 224 steps. The ARO parking position may be any position above 224 steps.

Figure 4
Rod Insertion Limit, 128 Step Tip-to-Tip, One Inoperable Rod
(Technical Specification 3.1.4 Condition A)



Bank Positions Given By:

- Bank D = (150 / 63) * (P – 70) + 224
- Bank C = (150 / 63) * (P – 70) + 224 + 128

NOTE: The top of the active fuel height corresponds to 224 steps. The ARO parking position may be any position above 224 steps.

Figure 5
Flux Difference Operating Envelope