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

DEC 26 2006

L-PI-06-099
TS 5.6.5.d

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

Prairie Island Nuclear Generating Plant Unit 2
Docket 50-306
License No. DPR-60

Core Operating Limits Report (COLR) for Prairie Island Unit 2 Cycle 24, Revision 0

Pursuant to the requirements of Technical Specification 5.6.5.d, the COLR for Prairie Island Nuclear Generating Plant Unit 2, Cycle 24, Revision 0 is attached. The limits specified in the attached COLR have been established using Nuclear Regulatory Commission (NRC) approved methodologies.

The Unit 2 COLR has been revised for Cycle 24, Modes 5 and 6, to incorporate the following changes:

- Revised Section 3.1.3.b.ii, Isothermal Temperature Coefficient (ITC) upper limit value from -2.9 pcm/°F at power level =100% Rated Thermal Power (RTP) to -1.5 pcm/°F at power level =100% RTP.
- Revised Section 3.2.1 to state the W(Z) values and $F^{W}_{Q}(Z)$ Penalty Factors are not applicable for this revision.
- Revised the Reference section to add the reference to WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," March 2005.
- Revised the References section to delete the reference to the previous 50.59 evaluation 1046, Rev. 1, "Unit 2, Cycle 23 Core Reload", and added a new reference to include the 50.59 Evaluation 1055, "Unit 2 Cycle 24 Core Reload".
- Revised the References section to add a reference to Westinghouse letter NF-NMC-06-142, "Cycle 24 Modes 5 and 6 Information in Support of Reconstitution and Redesign Efforts", November 22, 2006.
- Revised Table 1 to incorporate revised values for Modes 5 and 6 for Unit 2, Cycle 24 COLR, Revision 0.
- Deleted Tables 2 and 3 since they are not applicable to Modes 5 and 6.

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Revision 0 contains three typographical errors that were identified following issuance, and were corrected with issuance of Revision 1, which will be submitted shortly. Page 8 of 14, reference number 24, should read "March 2005", versus Marck 2005. Reference number 26, should read "Cycle 24 Modes 5 and 6..", versus Cycle 23. Page 9, Table 1 - the ** notation in the table heading labeled "Number of Charging Pumps Running**", did not get deleted when the Table was updated for Modes 5 and 6.

Summary of Commitments

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

Thomas J. Palmisano
Site Vice President, Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure (1)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC

ENCLOSURE 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
CORE OPERATING LIMITS REPORT
UNIT 2 – CYCLE 24
REVISION 0

14 pages to follow

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

CORE OPERATING LIMITS REPORT

UNIT 2 – CYCLE 24

REVISION 0

Reviewed By: RG for Date: 11-26-06
Lynn Johnson
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Reviewed By: Date: 11/26/06
Ed Mercier
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Approved By: Date: 11/26/06
Mike Carlson
Director, Site 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 2 - CYCLE 24
REVISION 0

This Revision is valid only for Modes 5 and 6.

This report provides the values of the limits for Unit 2 Cycle 24 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. <u>2.1.1 Reactor Core Safety Limits</u>

   Reactor Core Safety Limits are shown in Figure 1.

   Reference Technical Specification section 2.1.1.

2. <u>3.1.1 Shutdown Margin Requirements</u>

   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) -1.5pcm/°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 not applicable for this revision.

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 not applicable for this revision.

$F^W_Q(Z)$ Penalty Factors are not applicable for this revision.

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 ≥ 50% RTP.

Reference Technical Specification sections 3.2.3.

10. <u>3.3.1 Reactor Trip System (RTS) Instrumentation</u>

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

<u>Overtemperature ΔT Setpoint</u>

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.

<u>Overpower ΔT Setpoint</u>

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. <u>3.4.1 RCS Pressure, Temperature, and Flow - Departure from Nucleate</u>

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 Refueling 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. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," Marck 2005.

25. 50.59 Evaluation 1055, "Unit 2 Cycle 24 Core Reload."

26. Westinghouse letter NF-NMC-06-142, "Cycle 23 Modes 5 and 6 Information in Support of Reconstitution and Redesign Efforts", November 22, 2006.

Table 1

Minimum Required Shutdown Margin

Plant Conditions	Number of Charging Pumps Running**		
	0-1 Pump	2 Pumps	3 Pumps
Mode 5***, $T_{ave} \leq 200°F$	2.5%	5.5%	8.0%
Mode 6, ARI***, $T_{ave} \geq 68°F$	5.129%	5.5%	8.0%
Mode 6, ARO***, $T_{ave} \geq 68°F$	5.129%	6.5%	10.0%

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

*** These values are also applicable for the Unit 2 Cycle 23 end of cycle.

Figure 1

Reactor Core Safety Limits

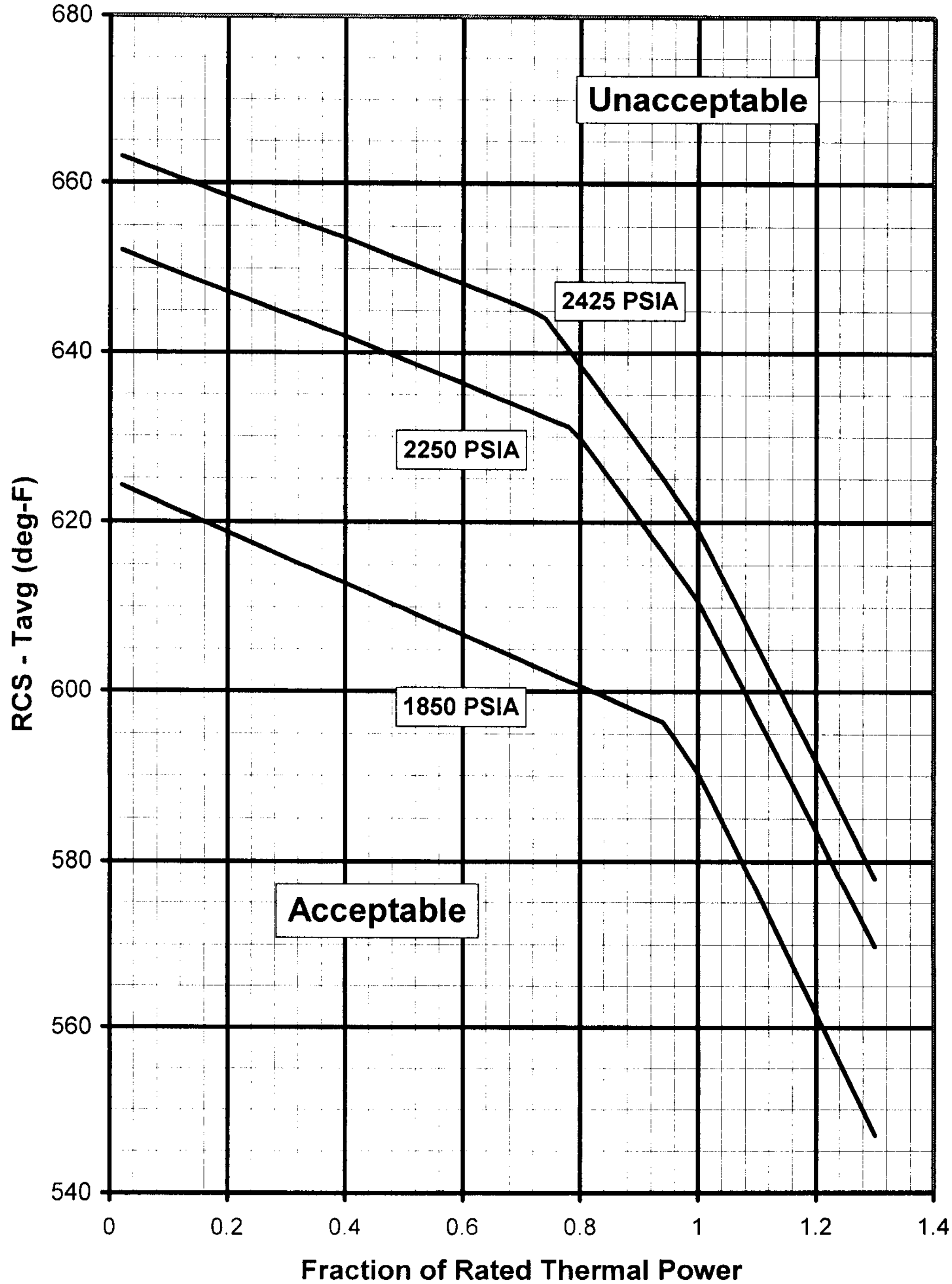
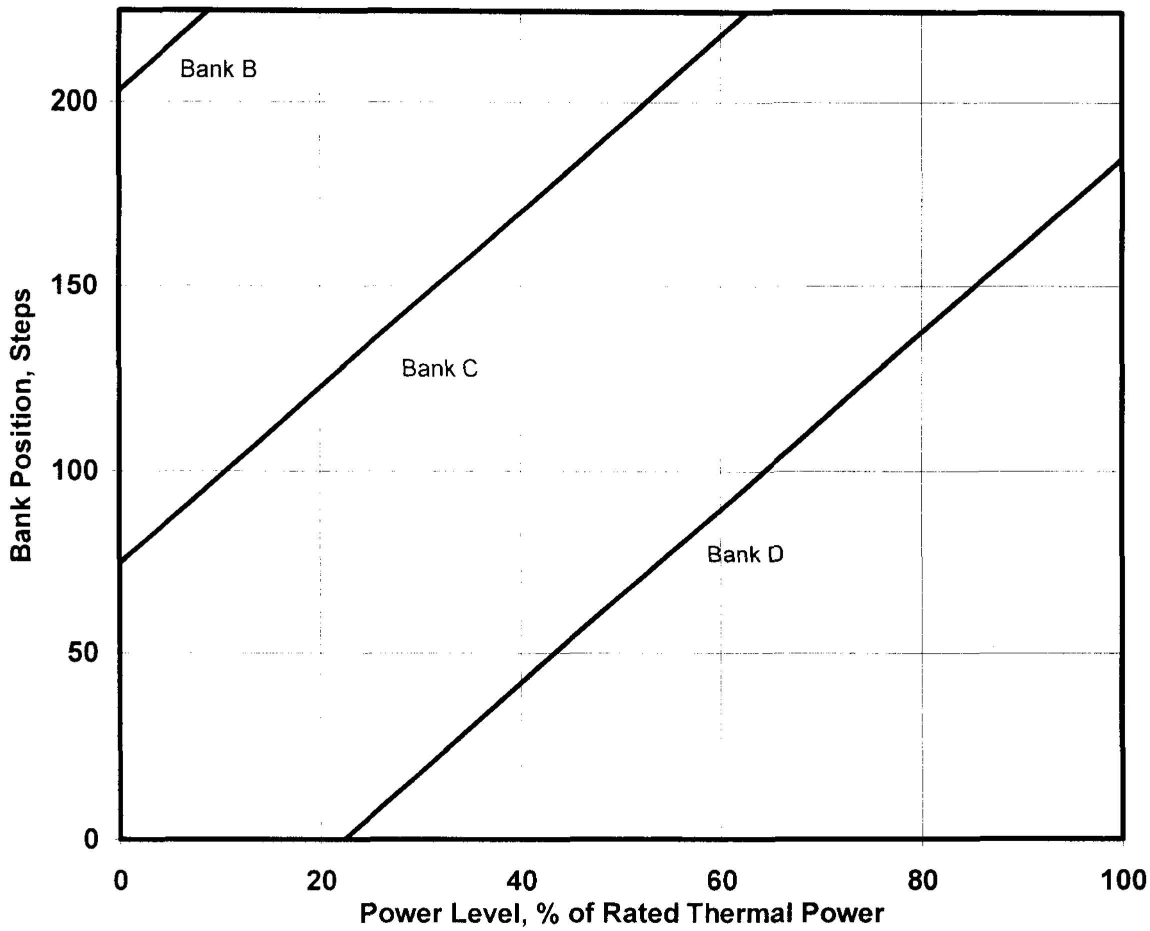


Figure 2
Rod Insertion Limit, 128 Step Tip-to-Tip

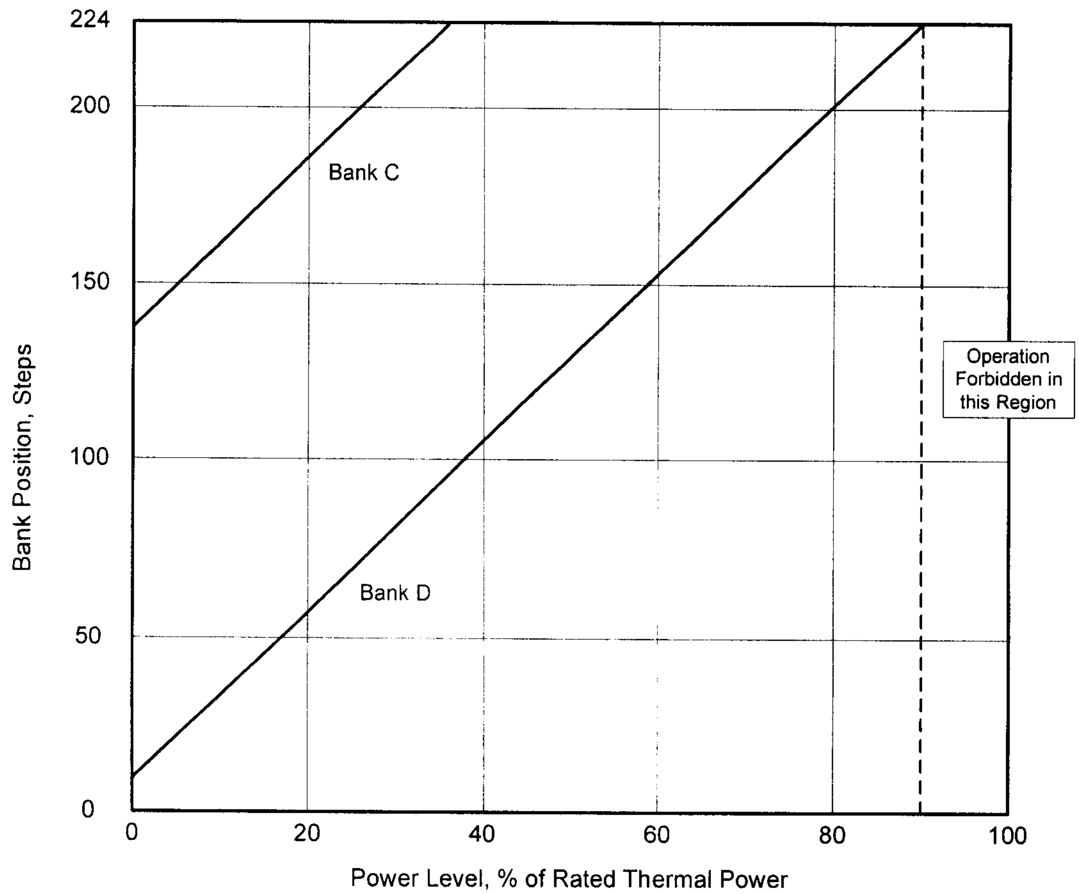


Bank Positions Given By:

- 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)

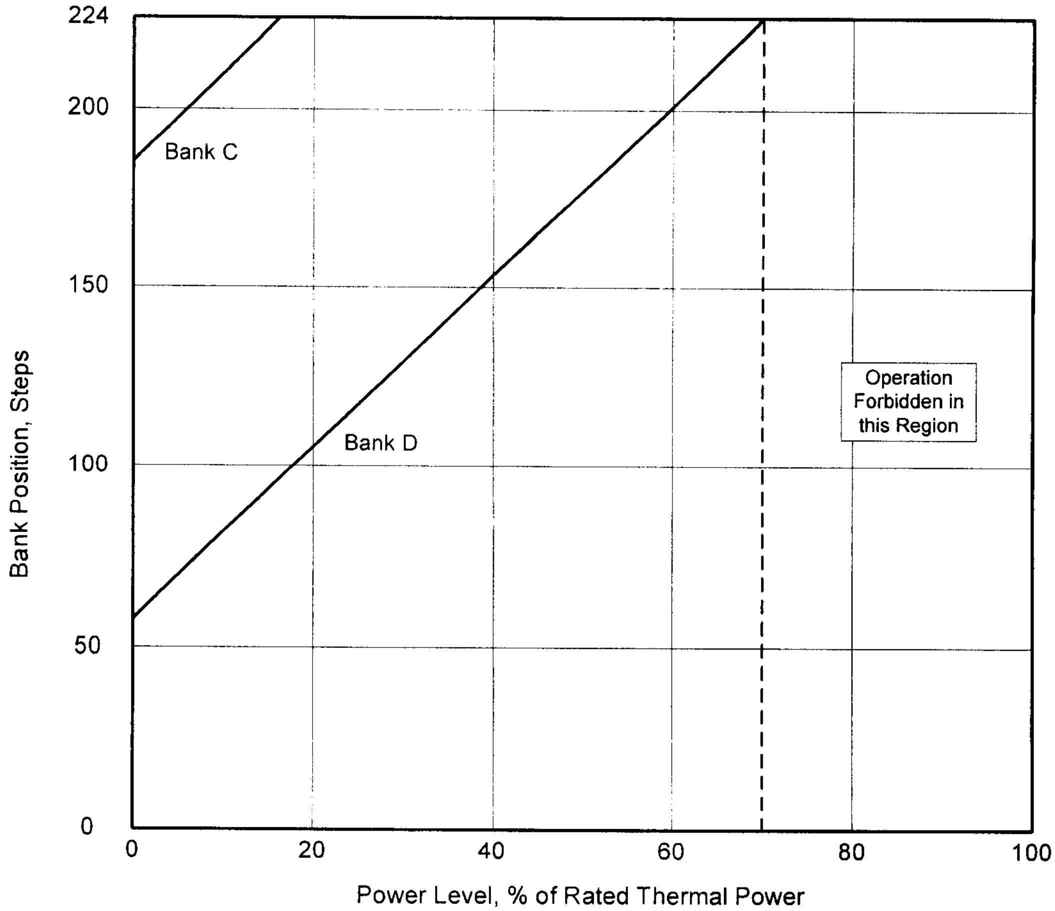


Bank Positions Given By:

- 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