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Docket Number 50-346

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United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: Core Operating Limits Report and Reload Report for Cycle 13

Ladies and Gentlemen:

Enclosed is a copy of the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Core Operating Limits Report (COLR) for Cycle 13 operation (Enclosure 1), which is scheduled to commence in May, 2000. The COLR is being submitted in accordance with DBNPS Technical Specification 6.9.1.7. In addition, an information-only copy of the DBNPS Cycle 13 Reload Report is provided as Enclosure 2.

Should you have any questions or require additional information, please contact Mr. James L. Freels, Manager - Regulatory Affairs, at (419) 321-8466.

Very truly yours,

Guy G. Campbell

MKL

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, NRC/NRR Project Manager
K. S. Zellers, NRC Region III, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

A001

Docket Number 50-346
License Number NPF-3
Serial Number 2653
Enclosure 1

Davis-Besse Nuclear Power Station Unit Number 1
Core Operating Limits Report
Cycle 13

(25 pages follow)

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE UNIT 1

CYCLE 13

CORE OPERATING LIMITS REPORT

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FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE UNIT 1

CYCLE 13

CORE OPERATING LIMITS REPORT

1.0 Core Operating Limits

This CORE OPERATING LIMITS REPORT for DB-1 Cycle 13 has been prepared in accordance with the requirements of Technical Specification 6.9.1.7. The core Operating Limits have been developed using the methodology provided in reference 2.0 (1). The licensed length of Cycle 13 is 683 EFPDs.

The following cycle-specific core Operating Limits, Protective Limit and Flux -Δ Flux/Flow

Reactor Protection System Allowable Values are included in this report:

1) Regulating Group Position Alarm Setpoints (error adjusted Operating Limits) and Xenon reactivity "power level cutoff"

2) Rod program group positions (Control Rod Core locations and group assignments)

3) Axial Power Shaping Rod Alarm Setpoints (error adjusted Operating Limits)

4) AXIAL POWER IMBALANCE Alarm Setpoints (error adjusted Operating Limits)

5) AXIAL POWER IMBALANCE Protective Limits

6) Flux-ΔFlux/Flow (or Power/Imbalance/Flow) Allowable Values

7) QUADRANT POWER TILT limits

8) Negative Moderator Temperature Coefficient limit

9) Nuclear Heat Flux Hot Channel Factor, F_Q and

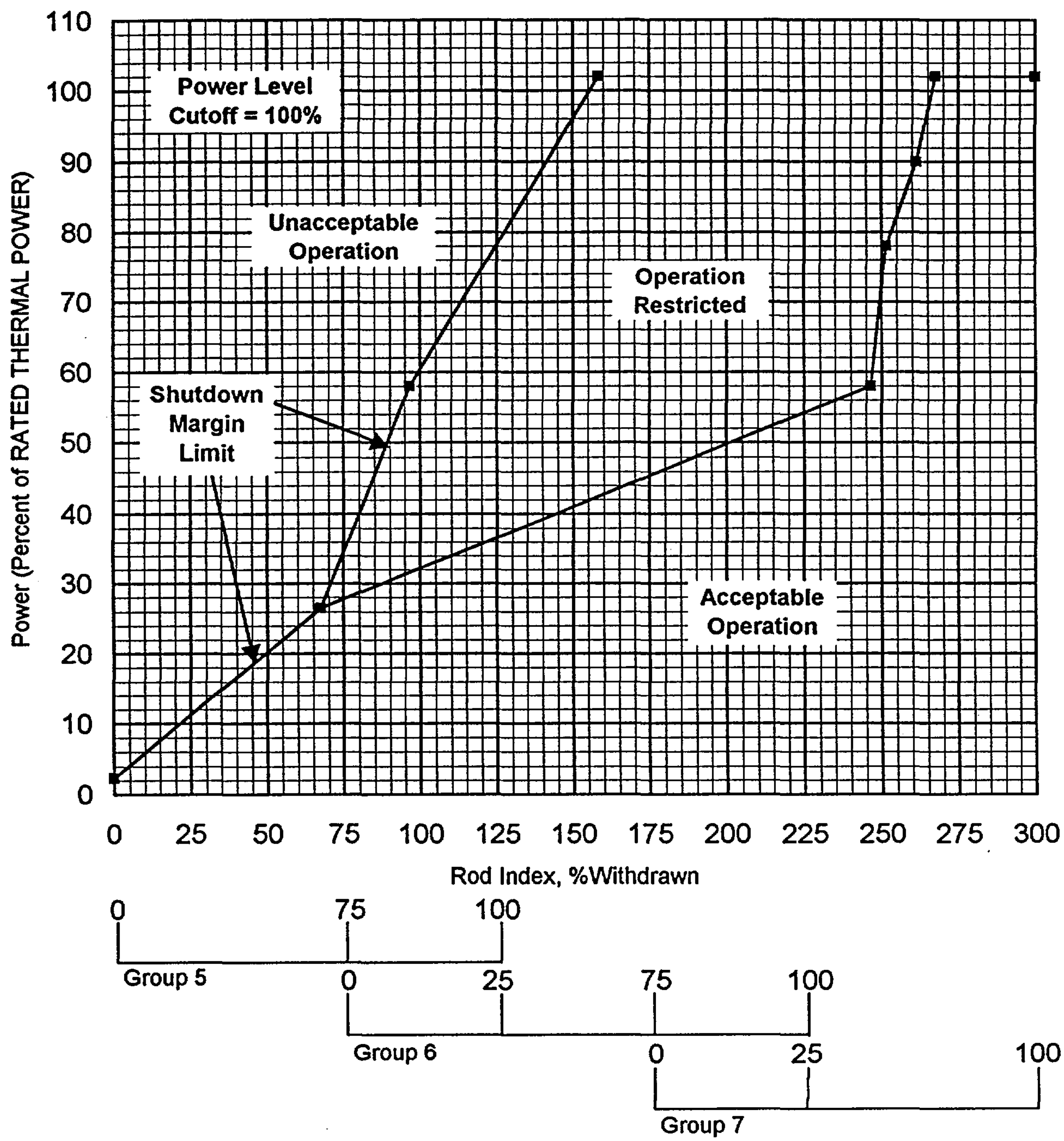
10) Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$

2.0 References

(1) BAW-10179P-A, Revision 3, "Safety Criteria and Methodology of Acceptable Cycle Reload Analysis.", dated October 1999.
(2) BAW-10227P-A, Revision 0, " Evaluation of Advanced Cladding and Structural Material (M5™) in PWR Reactor Fuel, dated February 2000.

Figure 1a Regulating Group Position Operating Limits
0 to 300 ±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 13

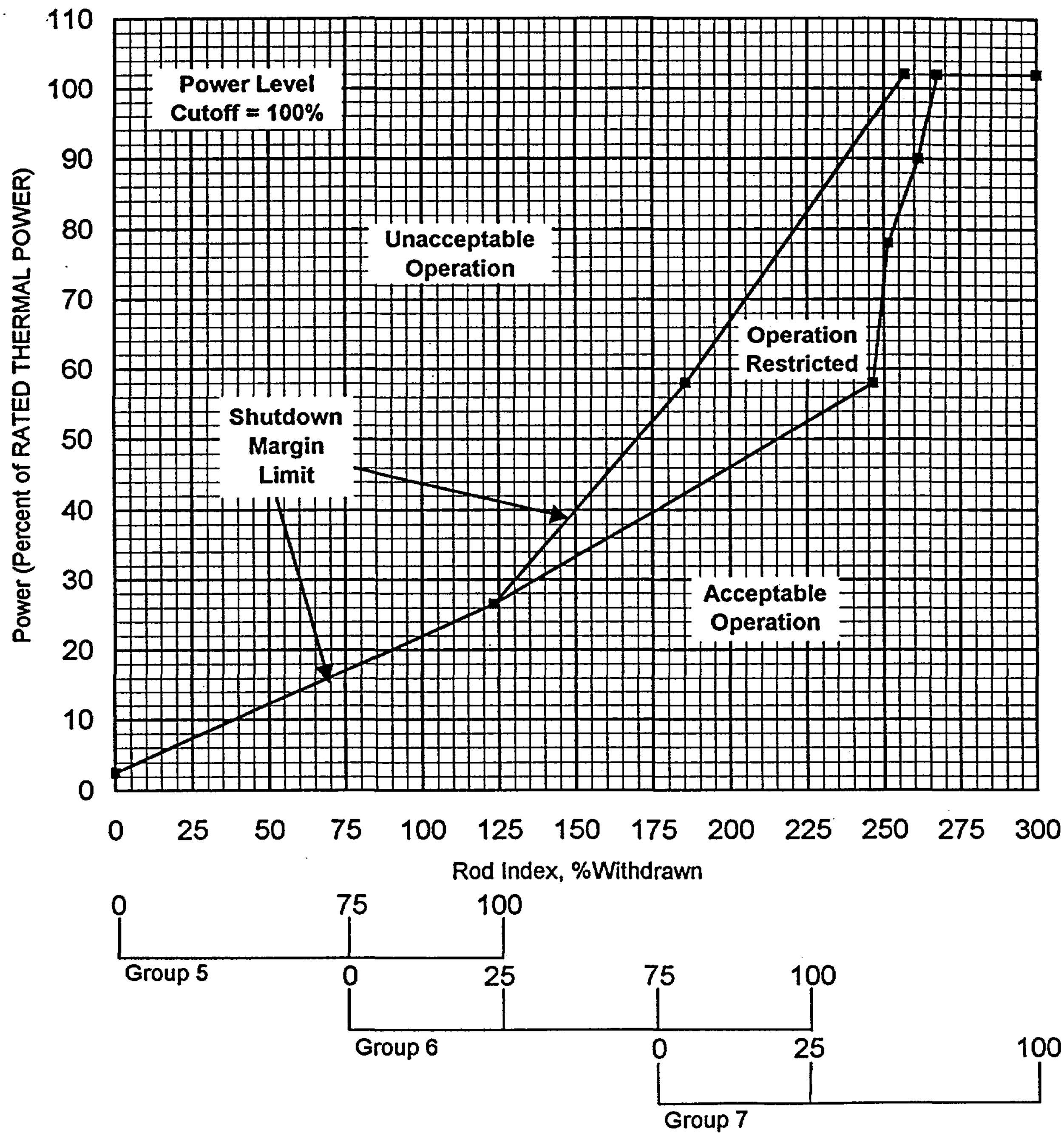
This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 1b Regulating Group Position Operating Limits
After 300 ±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 13

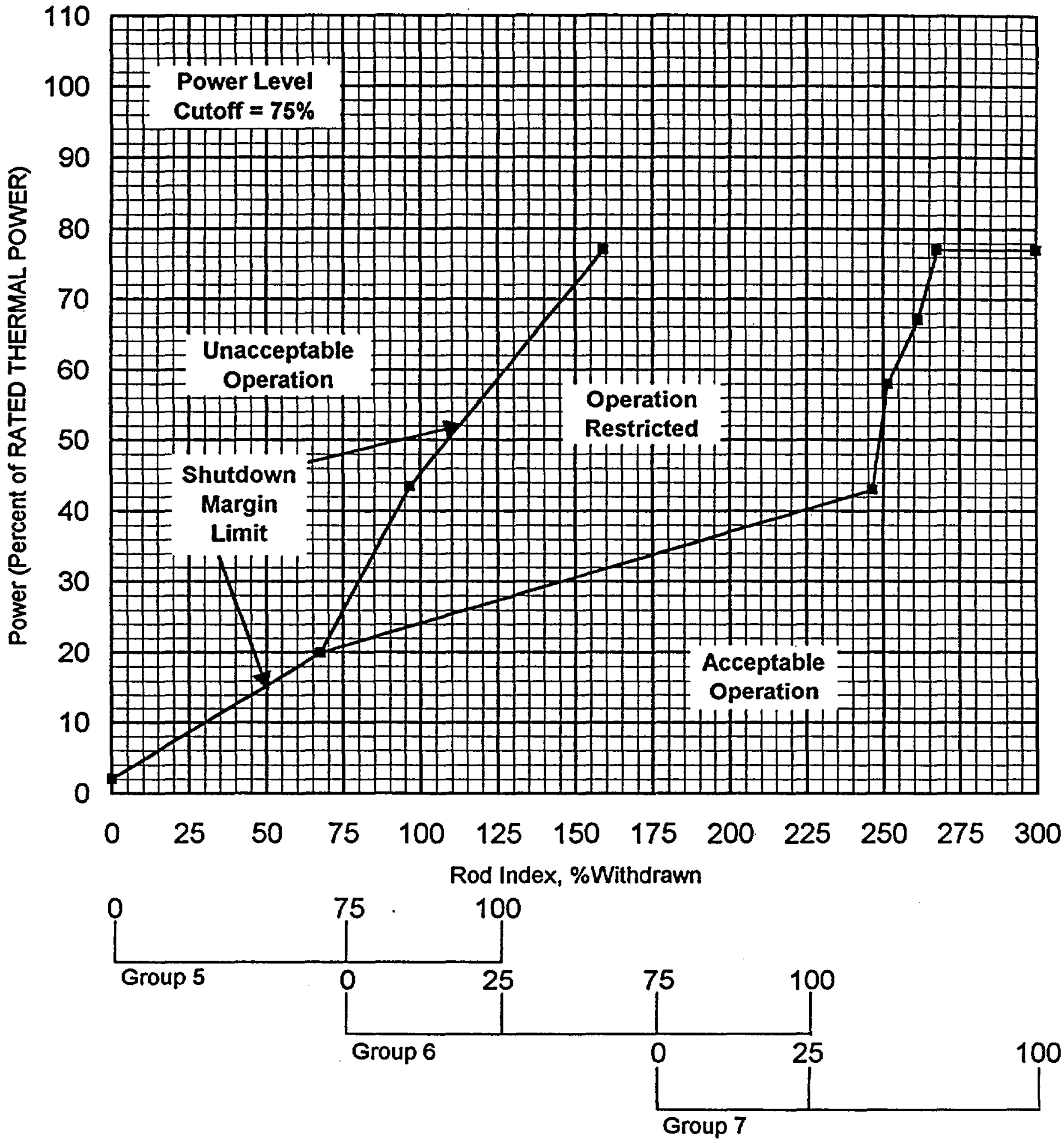
This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 1c Regulating Group Position Operating Limits
0 to 300 ±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 13

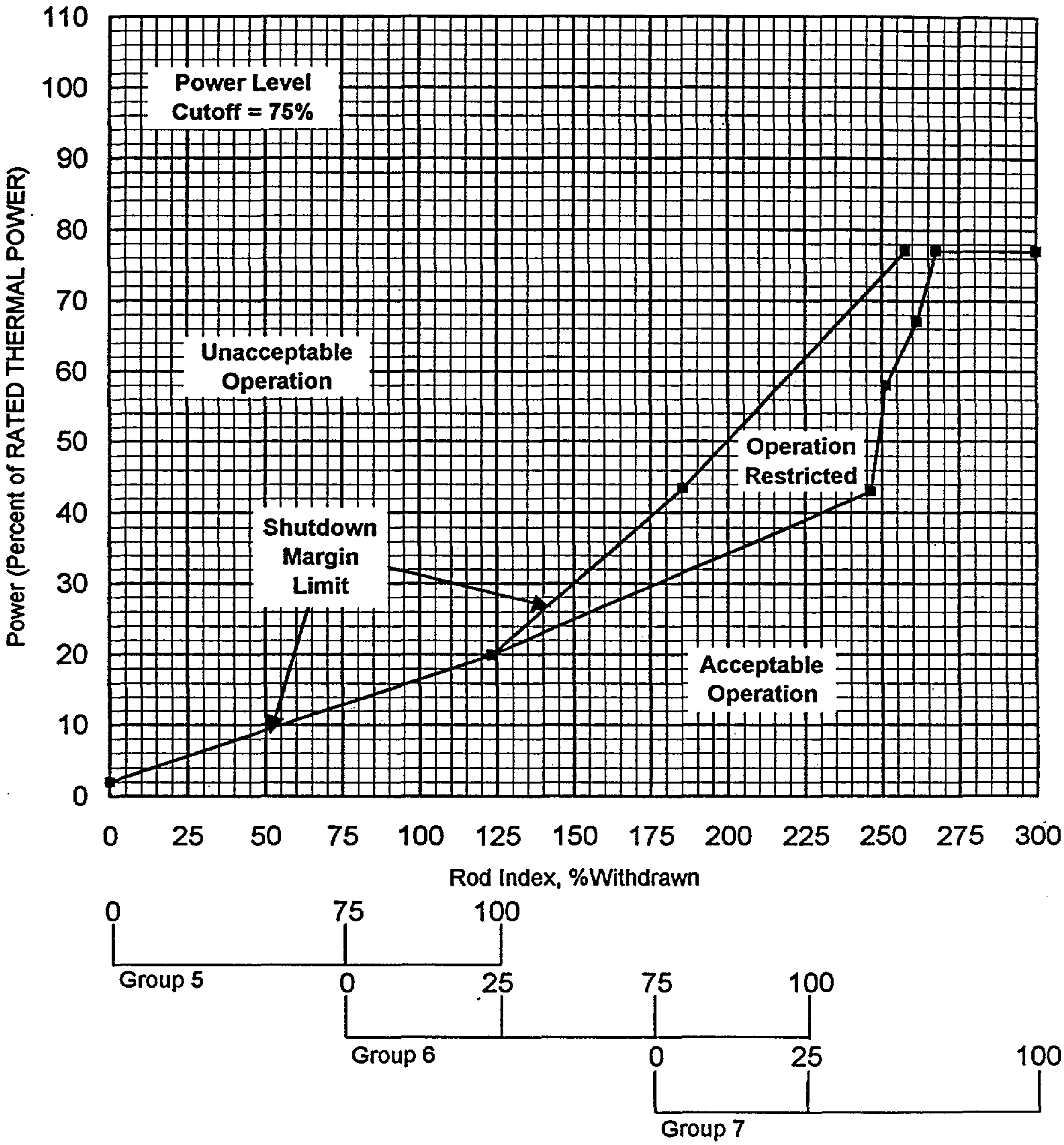
This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 1d Regulating Group Position Operating Limits After 300 ±10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 13

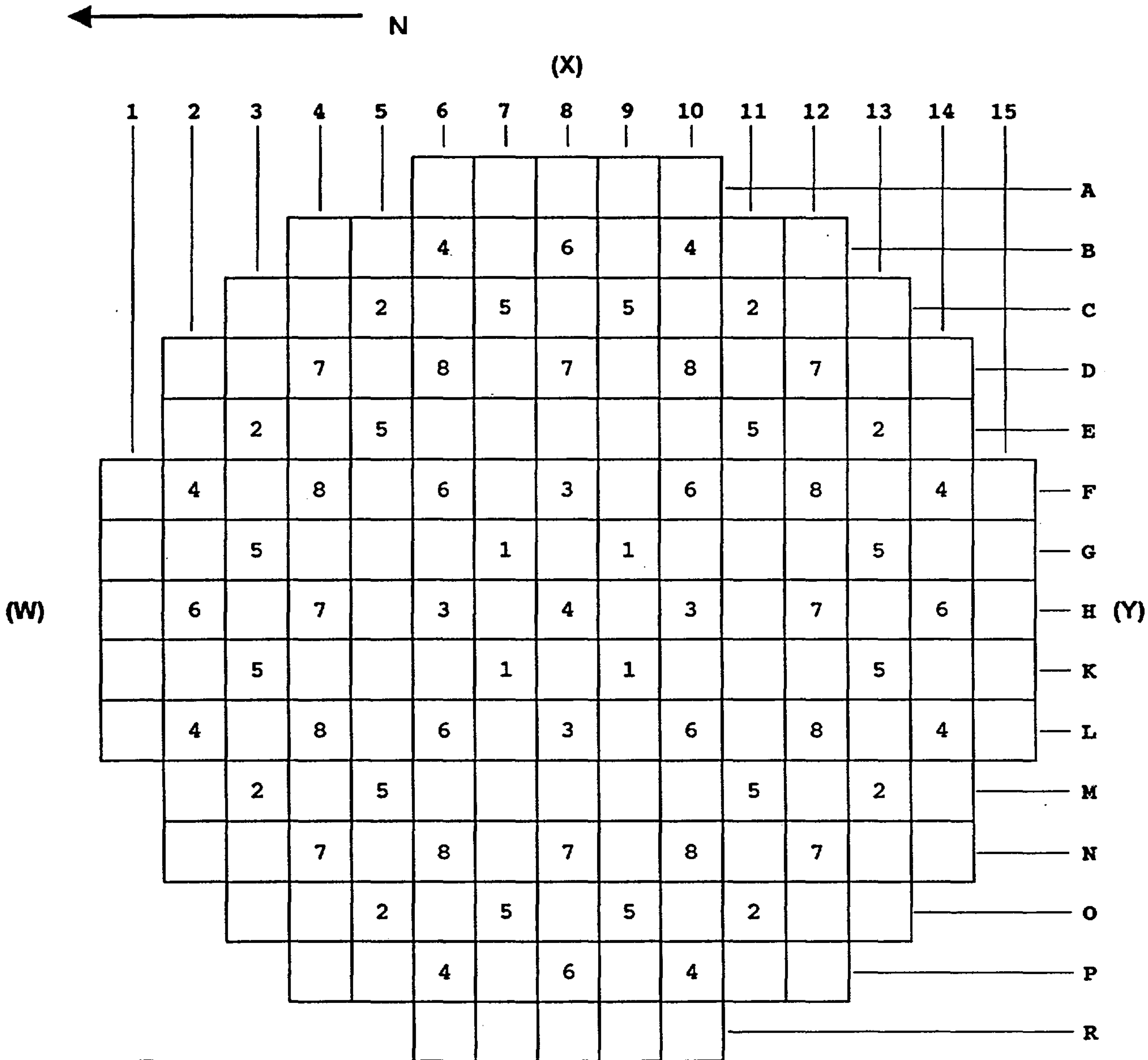
This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

**Figure 2 Control Rod Core Locations
and Group Assignments
Davis-Besse 1, Cycle 13**

This Figure is referred to by
Technical Specification 3.1.3.7



X Group Number

Group	No. of Rods	Function	Group	No. of Rods	Function
1	4	Safety	5	12	Control
2	8	Safety	6	8	Control
3	4	Safety	7	8	Control
4	9	Safety	8	8	APSRs
			Total	61	

Figure 3 APSR Position Operating Limits

This Figure is referred to by Technical Specification 3.1.3.9

Before APSR Pull: 0 EFPD to 626 ±10 EFPD,
Three or Four RC pumps operation*

Lower Limit: 0 %WD

Upper Limit: 100 %WD

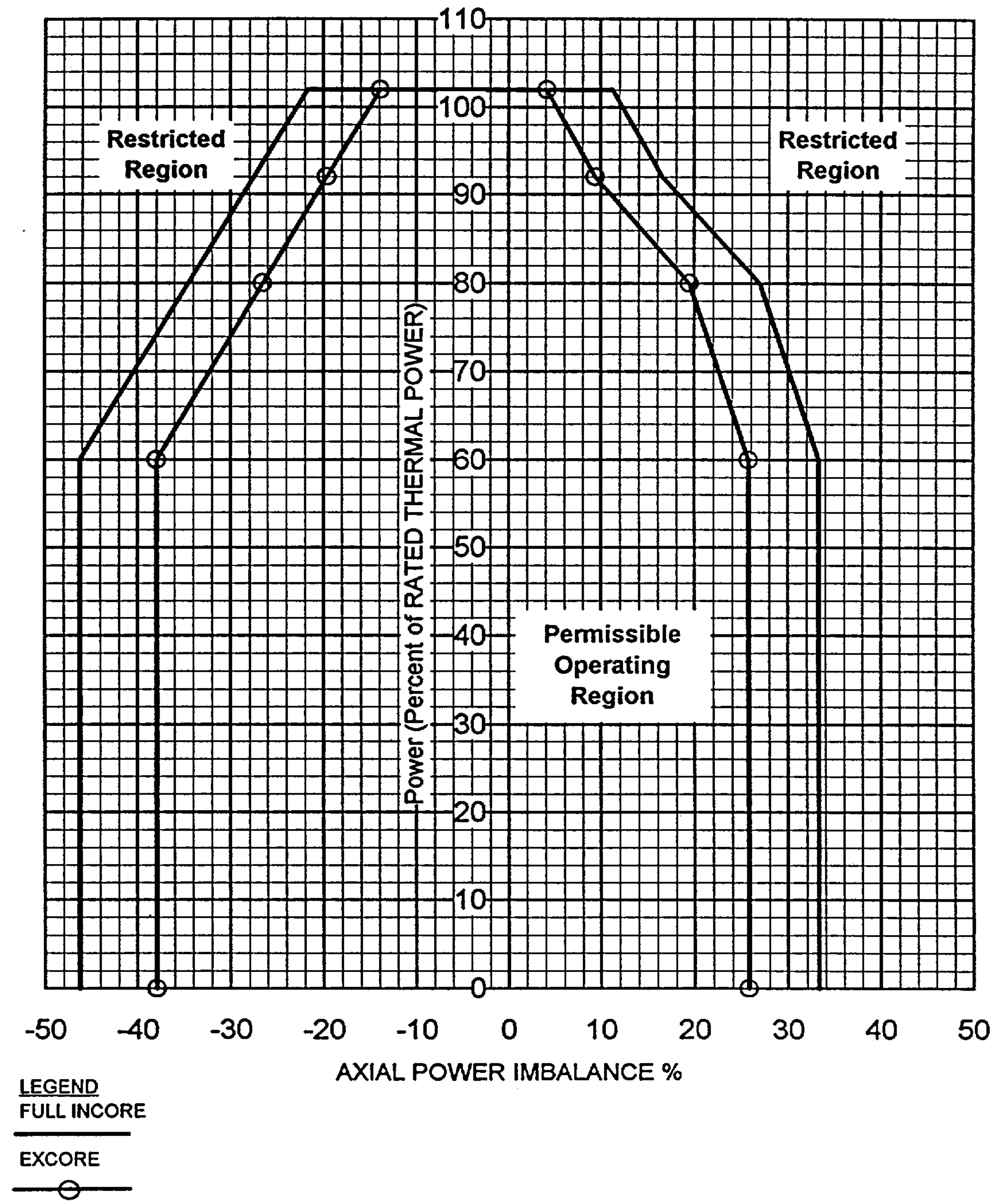
After APSR Pull: 626 ±10 EFPD to End-of-Cycle
Three or Four RC pumps operation*

Insertion Prohibited (maintain ≥99 %WD)

* Power restricted to 77% for 3 pump operation

Figure 4a AXIAL POWER IMBALANCE Operating Limits
0 to 300 ±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 13

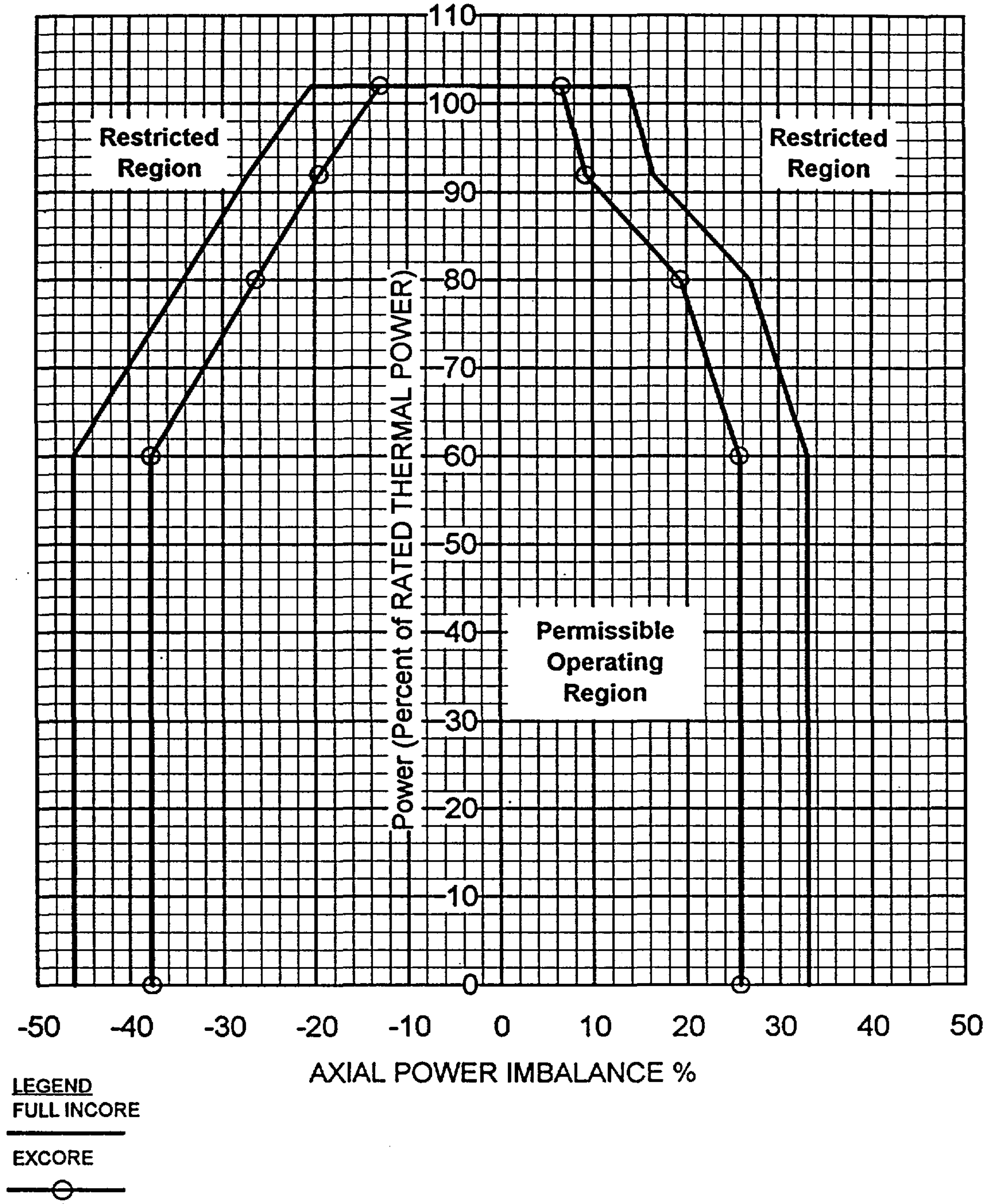
This Figure is referred to by Technical Specification 3.2.1



Note 1: Instrument error is accounted for in these Operating Limits.

Figure 4b AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 626 ±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 13

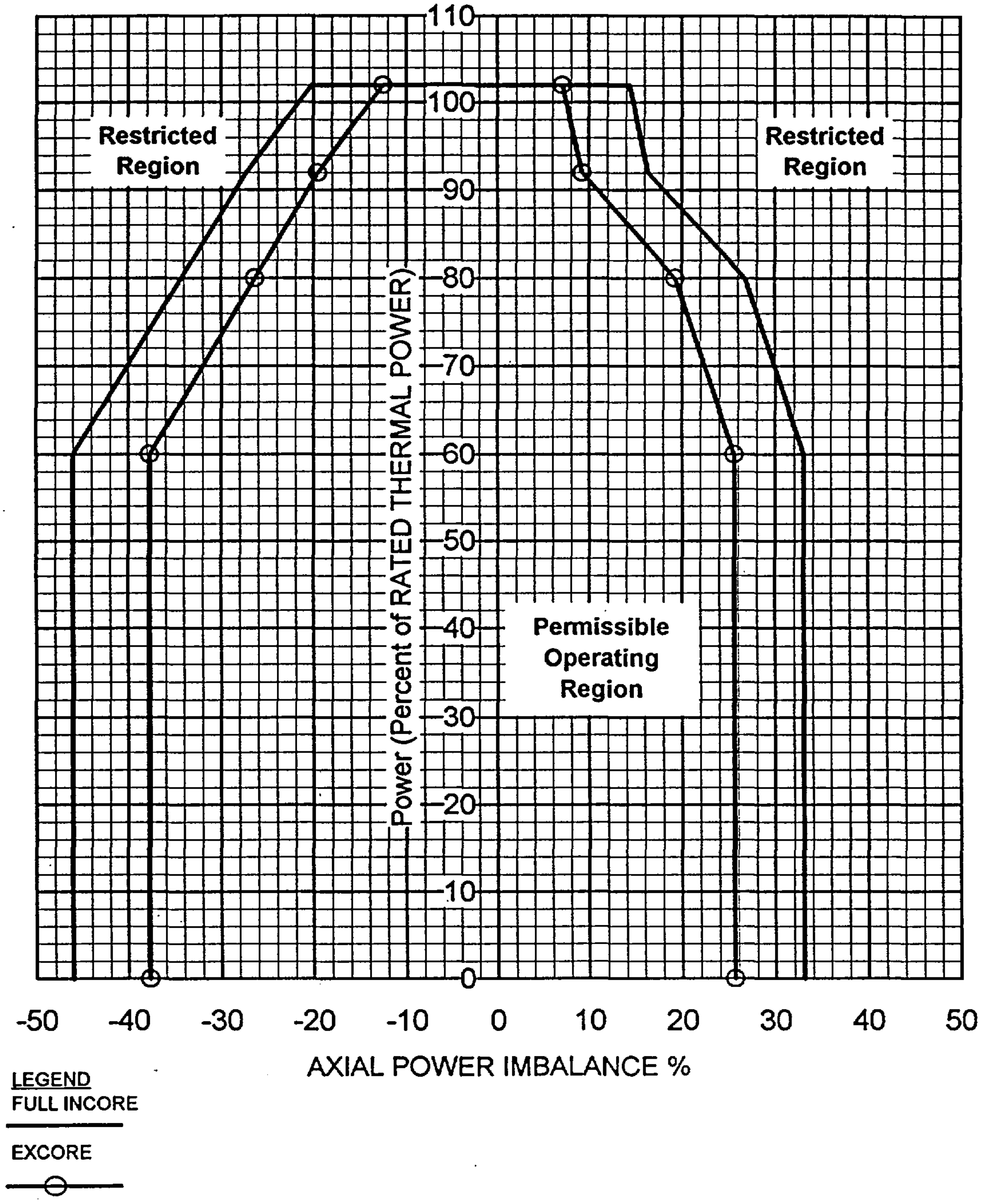
This Figure is referred to by
Technical Specification 3.2.1



Note 1: Instrument error is accounted for in these Operating Limits.

Figure 4c AXIAL POWER IMBALANCE Operating Limits After 626 ±10 EFPD, Four RC Pumps -- Davis-Besse 1, Cycle 13

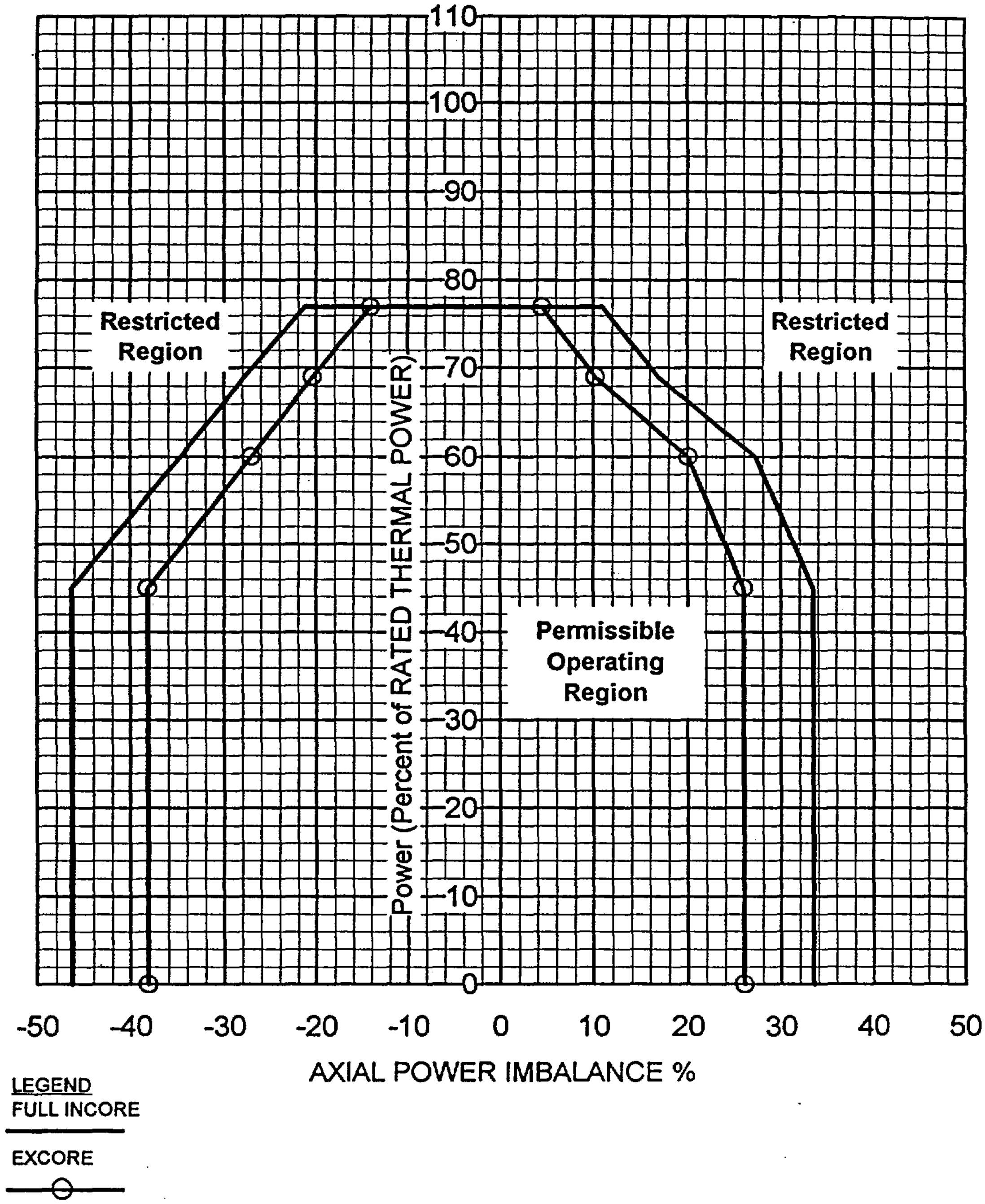
This Figure is referred to by Technical Specification 3.2.1



Note 1: Instrument error is accounted for in these Operating Limits.

Figure 4d AXIAL POWER IMBALANCE Operating Limits
0 to 300 ±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 13

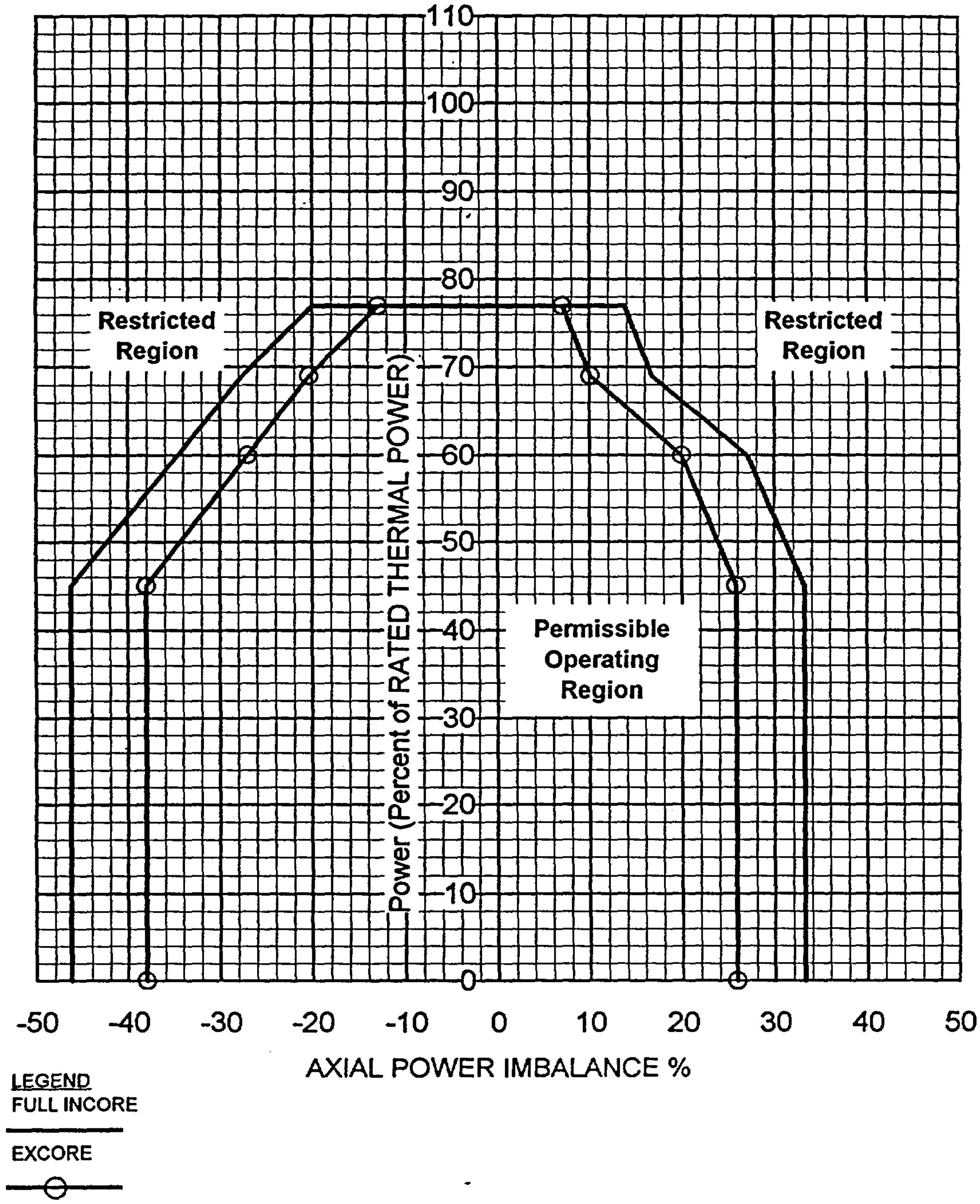
This Figure is referred to by
Technical Specification 3.2.1



Note 1: Instrument error is accounted for in these Operating Limits.

Figure 4e AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 626 ±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 13

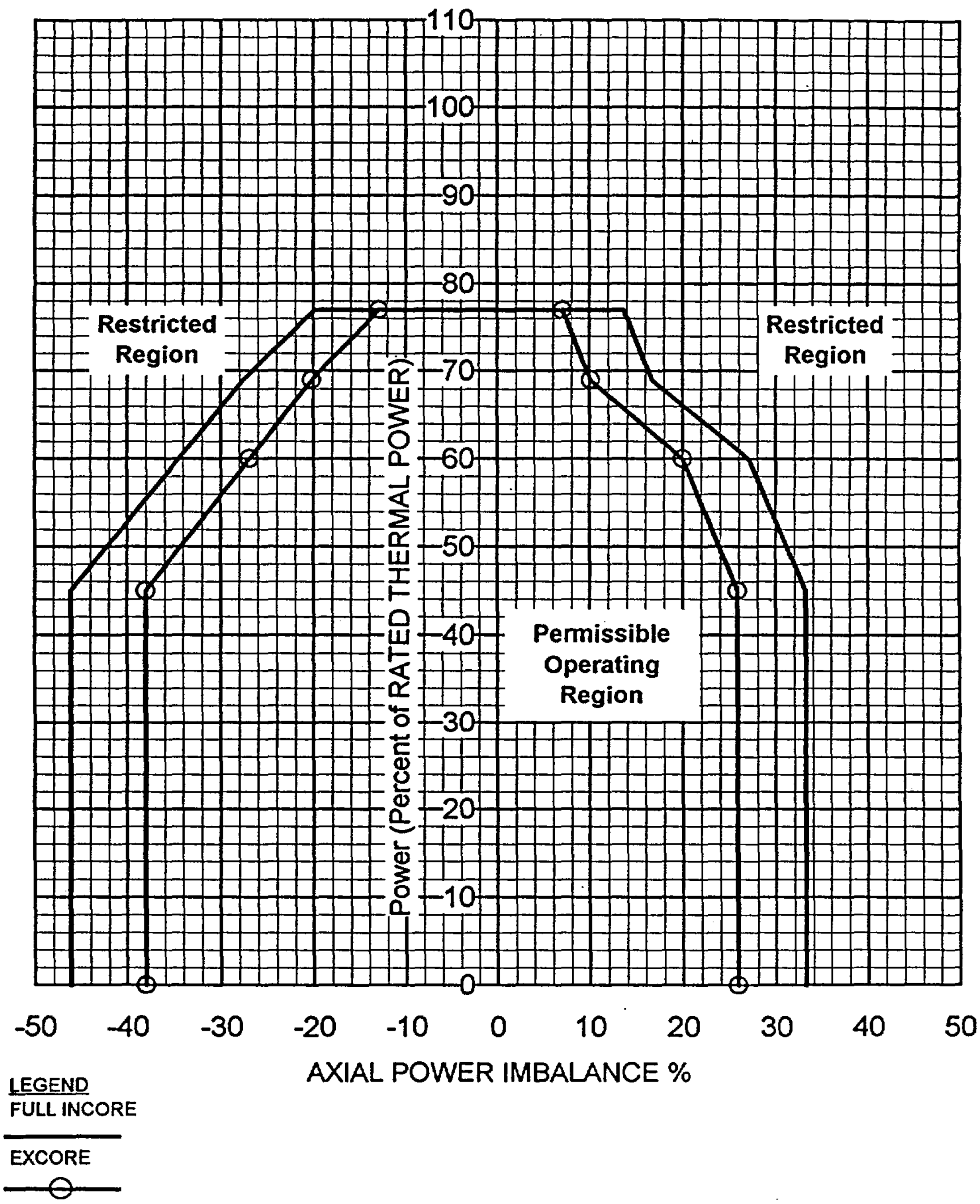
This Figure is referred to by
Technical Specification 3.2.1



Note 1: Instrument error is accounted for in these Operating Limits.

Figure 4f AXIAL POWER IMBALANCE Operating Limits
After 626 ±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 13

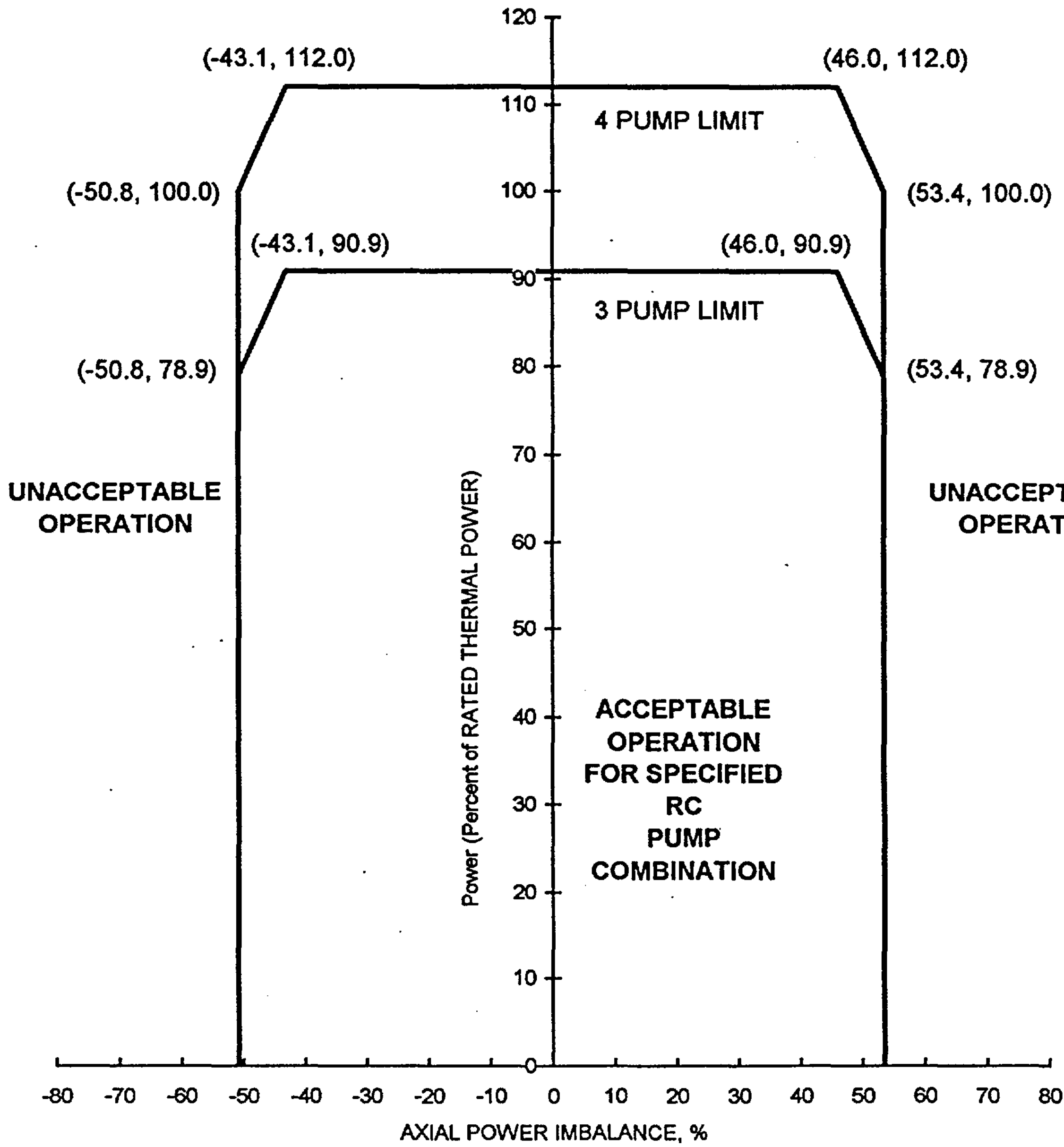
This Figure is referred to by
Technical Specification 3.2.1



Note 1: Instrument error is accounted for in these Operating Limits.

Figure 5 AXIAL POWER IMBALANCE Protective Limits

This Figure is referred to by Technical Specification 2.1.2



Pumps Operating	Reactor Coolant Flow, gpm	Required Measured Flow to Ensure Compliance, gpm
4	380,000	389,500
3	283,860	290,957

Figure 6 **Flux--ΔFlux/Flow (or Power/Imbalance/Flow) Allowable Values**

This Figure is referred to by Technical Specification 2.2.1

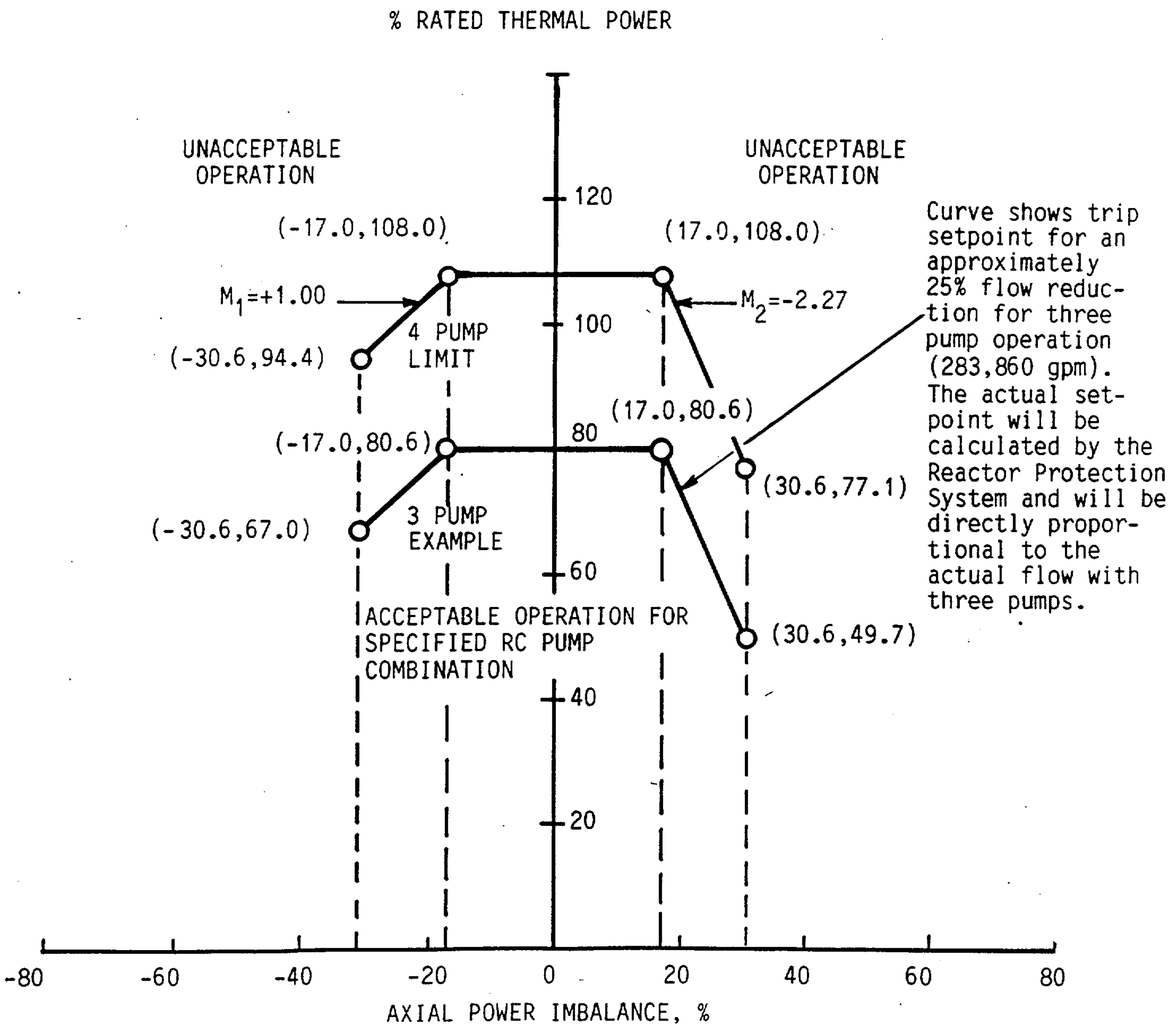


Table 1 QUADRANT POWER TILT Limits

This Table is referred to by Technical Specification 3.2.4

QUADRANT POWER TILT as measured by:	From 0 EFPD to EOC-13			
	Steady-state Limit for THERMAL POWER ≤ 60% (%)	Steady-state Limit for THERMAL POWER > 60% (%)	Transient Limit (%)	Maximum Limit (%)
Symmetrical Incore detector system	7.90	4.00	10.03	20.0

Table 2 Negative Moderator Temperature Coefficient Limit

This Table is referred to by Technical Specification 3.1.1.3c

Negative Moderator Temperature Coefficient Limit (at RATED THERMAL POWER)	-4.00×10^{-4} Δk/k/°F

Table 3 Power to Melt Limits

This Table is referred to by Technical Specification Bases B2.1

	Batch 9G	Batch 10A2	Batch 13	Batch 14	Batch 15
Fuel Assembly Type	Mark-B8A	Mark-B8A	Mark-B10A	Mark-B10M	Mark-B10K
Minimum linear heat rate to melt, kW/ft	20.5	20.5	22.3	22.3 (20.8)[a] (20.8)[b]	22.1 (21.1)[c] (20.7)[d] (20.3)[e]

[a] Limit for 3 wt% Gd rods - Batch 14
[b] Limit for 6 wt% Gd rods - Batch 14
[c] Limit for 2 wt% Gd rods - Batch 15
[d] Limit for 3 wt% Gd rods - Batch 15
[e] Limit for 8 wt% Gd rods - Batch 15

Table 4a Nuclear Heat Flux Hot Channel Factor - F_Q (NAS)

This Table is referred to by Technical Specification 3.2.2

Nuclear Heat Flux Hot Channel Factor - F_Q

F_Q shall be limited by the following relationships:

$F_Q \leq LHR^{ALLOW}(Bu)/[LHR^{AVG} * P]$ (for $P \leq 1.0$)

$LHR^{ALLOW}(Bu)$: See Tables below
LHR^{AVG} = 6.139 kW/ft for Mark-B8A fuel
LHR^{AVG} = 6.426 kW/ft for Mark-B10A fuel
LHR^{AVG} = 6.420 kW/ft for Mark-B10M fuel
LHR^{AVG} = 6.318 kW/ft for Mark-B10K fuel
P = ratio of THERMAL POWER/RATED THERMAL POWER
Bu = Fuel Burnup (MWd/mtU)

Batch 9G (Mark-B8A) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU	60,000 MWd/mtU
1	15.6	15.6	11.8	10.3
2	15.3	15.3	11.8	10.3
3	14.5	14.5	11.8	10.3
4	14.5	14.5	11.8	10.3
5	14.9	14.9	11.8	10.3
6	14.9	14.9	11.8	10.3
7	14.2	14.2	11.4	9.9
8	13.9	13.9	11.2	9.7

Batch 10A2 (Mark-B8A) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU
1	15.6	15.6	11.8
2	15.3	15.3	11.8
3	14.5	14.5	11.8
4	14.5	14.5	11.8
5	14.9	14.9	11.8
6	14.9	14.9	11.8
7	14.2	14.2	11.4
8	13.9	13.9	11.2

TABLE 4a continued

Batch 13 (Mark-B10A) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	12.8
2	17.5	16.7	12.8
3	17.0	15.6	12.8
4	16.6	15.3	12.8
5	16.0	15.3	12.8
6	15.3	15.3	12.8
7	14.7	14.7	12.8
8	14.5	14.5	12.8

Batch 14 (Mark-B10M) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	12.8
2	17.5	16.7	12.8
3	17.0	15.6	12.8
4	16.6	15.3	12.8
5	16.0	15.3	12.8
6	15.3	15.3	12.8
7	14.7	14.7	12.8
8	14.5	14.5	12.8

Batch 15 (Mark-B10K) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	35,000 MWd/mtU
1	17.6	16.8
2	17.5	16.7
3	17.0	15.6
4	16.6	15.3
5	16.0	15.3
6	15.3	15.3
7	14.7	14.7
8	14.5	14.5

[a] Linear interpolation for allowable linear heat rate between specified burnup points is valid for these tables.

Table 4b Nuclear Heat Flux Hot Channel Factor - F_Q (FIDMS)

This Table is referred to by Technical Specification 3.2.2

Nuclear Heat Flux Hot Channel Factor - F_Q

F_Q shall be limited by the following relationships:

$F_Q \leq LHR^{ALLOW}(Bu)/[LHR^{AVG} * P]$ (for $P \leq 1.0$)

$LHR^{ALLOW}(Bu)$: See Tables below
$LHR^{AVG} = 6.377$ kW/ft
P = ratio of THERMAL POWER/RATED THERMAL POWER
Bu = Fuel Burnup (MWd/mtU)

Batch 9G (Mark-B8A) LHR^{ALLOW} kW/ft[(a)]

Core Elevation ft.	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU	60,000 MWd/mtU
0.000	16.2	16.2	12.1	10.6
2.506	15.8	15.8	12.1	10.6
4.264	15.0	15.0	12.1	10.6
6.021	15.4	15.4	12.1	10.6
7.779	15.9	15.9	12.1	10.6
9.536	15.3	15.3	12.1	10.6
12.000	14.3	14.3	11.5	10.0

Batch 10A2 (Mark-B8A) LHR^{ALLOW} kW/ft[(a)]

Core Elevation ft.	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU
0.000	16.2	16.2	12.1
2.506	15.8	15.8	12.1
4.264	15.0	15.0	12.1
6.021	15.4	15.4	12.1
7.779	15.9	15.9	12.1
9.536	15.3	15.3	12.1
12.000	14.3	14.3	11.5

TABLE 4b continued

Batch 13 (Mark-B10A) LHR^{ALLOW} kW/ft[a]

Core Elevation ft.	0 MWd/mtU	35,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	12.8
2.506	17.6	16.8	12.8
4.264	17.1	15.7	12.8
6.021	16.6	15.3	12.8
7.779	16.0	15.8	12.8
9.536	15.3	15.3	12.8
12.000	14.5	14.5	12.8

Batch 14 (Mark-B10M) LHR^{ALLOW} kW/ft[a]

Core Elevation ft.	0 MWd/mtU	35,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	12.8
2.506	17.6	16.8	12.8
4.264	17.1	15.7	12.8
6.021	16.6	15.3	12.8
7.779	16.0	15.8	12.8
9.536	15.3	15.3	12.8
12.000	14.5	14.5	12.8

Batch 15 (Mark-B10K) LHR^{ALLOW} kW/ft[a]

Core Elevation ft.	0 MWd/mtU	35,000 MWd/mtU
0.000	17.6	16.8
2.506	17.6	16.8
4.264	17.1	15.7
6.021	16.6	15.3
7.779	16.0	15.8
9.536	15.3	15.3
12.000	14.5	14.5

(a) Linear interpolation for allowable linear heat rate between specified burnup points is valid for these tables.

Table 5 Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$

This Table is referred to by Technical Specification 3.2.3

Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$

$F^N_{\Delta H} \leq$ ARP $[1 + 0.3(1 - P/P_m)]$

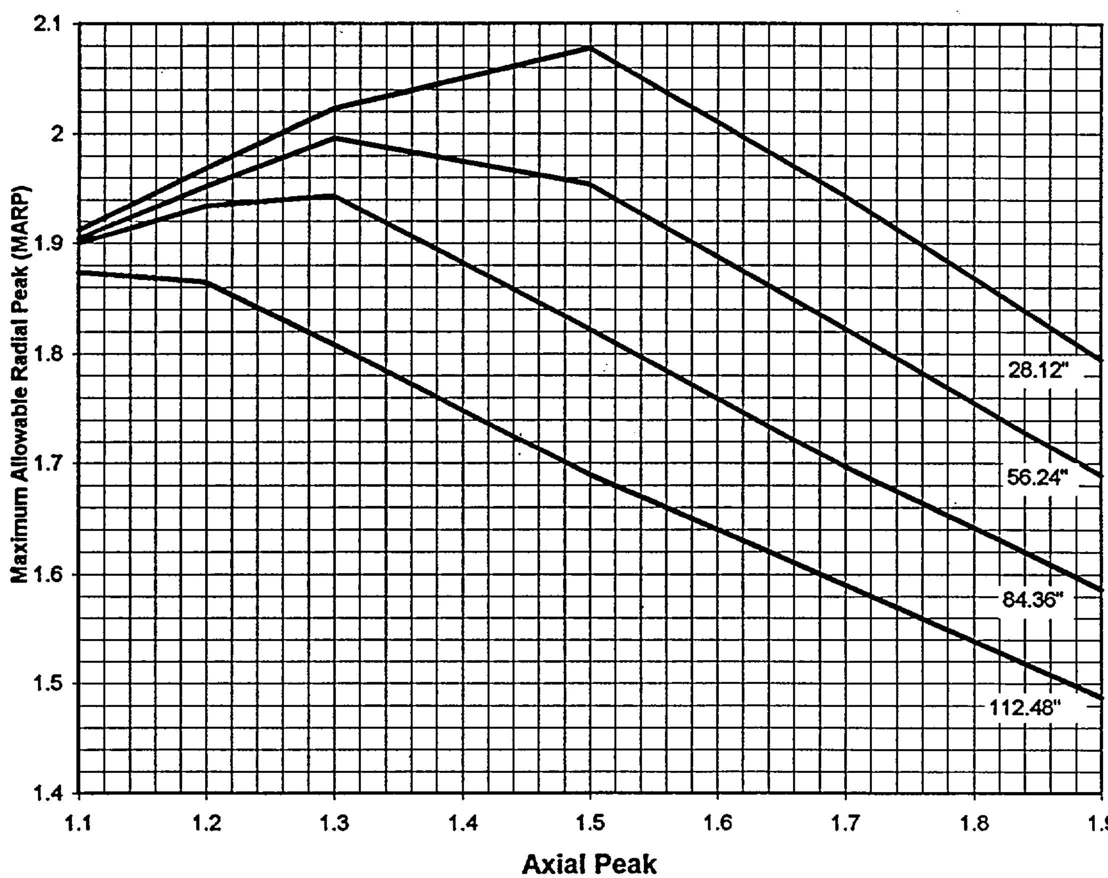
ARP = Allowable Radial Peak, see Figure

P = THERMAL POWER/RATED THERMAL POWER and $P \leq 1.0$

P_m = 1.0 for 4-RCP operation

P_m = 0.75 for 3-RCP operation

Figure 7* Allowable Radial Peak for $F^N_{\Delta H}$



* This figure is applicable to all fuel in the core. Linear interpolation and extrapolation above 112.48 inches are acceptable. For axial heights <28.12 inches, the value at 28.12 inches will be used.

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Davis-Besse Nuclear Power Station Unit Number 1
Reload Report
Cycle 13

(69 pages follow)

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DAVIS-BESSE NUCLEAR POWER STATION

UNIT 1, CYCLE 13 -- RELOAD REPORT

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1. INTRODUCTION AND SUMMARY

The analyses described in this report justify cycle 13 operation of the Davis-Besse Nuclear Power Station Unit 1 at a rated core power of 2772 MWt. The analyses are similar to those outlined in the Nuclear Regulatory Commission (NRC) document, "Guidance for Proposed License Amendments Relating to Refueling," June 1975. The analytical techniques and design bases utilized by the analyses described in this report have been approved by the NRC.

Cycle 13 reactor and fuel parameters related to full power capability are summarized in this report and compared to those for cycle 12. All accidents analyzed in the Davis-Besse Updated Safety Analysis Report (USAR, Reference 1) have been reviewed for cycle 13 operation. In all cases, the initial conditions of the transients in cycle 13 are bounded by the initial conditions of previous analyses.

The cycle 13 design incorporates an end-of-cycle (EOC) HFP extension maneuver which reduces the moderator average temperature (T_{avg}) by a maximum of 7°F (actual). The effects of the EOC T_{avg} reduction on the RCS structural, RCS operation, core mechanical (fuel), radiological dose consequences, nuclear (design-peaking), and thermal-hydraulic parameters as well as any potential effects and/or consequences on LOCA and non-LOCA safety analyses were evaluated and found to be acceptable. The analyses also verified that the operational maneuver at EOC is bounded by the safety analyses assumptions and will be accommodated by the core protective and operating limits.

Cycle 13 is the initial implementation of the Mark-B10K fuel assembly which features: M5™ fuel rod cladding, a highly loaded fuel rod, and the Trapper™ debris resistant lower end fitting. M5™ cladding has high corrosion resistance and improved mechanical performance. The batch 15 B10K fuel rod has higher uranium loading than the batch 14 fuel rod. The Trapper™ debris resistant lower end fitting will allow the Mark-B10K fuel assembly to resist fuel rod debris failures while incorporating a high performance fuel rod design.

Batch 15 will also contain four M5™ structural assemblies which will use the M5™ material for the guide tubes and the upper two intermediate spacer grids as well as for fuel rods. These four assemblies will demonstrate the improved mechanical performance of the M5™ material for structural components relative to Zircaloy-4.

The Technical Specifications have been reviewed and verified to require no changes for cycle 13 operation. Based on the reload report analyses performed and taking into account the emergency core cooling system (ECCS) Final Acceptance Criteria and postulated fuel densification effects, it is concluded that Davis-Besse Unit 1, cycle 13 can be operated safely at its licensed core power level of 2772 MWt. The Core Operating Limits Report (COLR) changes for cycle 13 are included in Section 8 of this report.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Davis-Besse Unit 1 is the currently operating cycle 12 (Reference 2), which achieved criticality on May 21, 1998. Power escalation began on May 22, 1998 and achieved approximately 90 %FP on May 25, 1998. Due to high OTSG levels, full power was not reached until July 8, 1998.

During cycle 12 operation, no operating anomalies have occurred that would adversely affect fuel performance during cycle 13. Cycle 13 was analyzed to 683 effective full power days (EFPD) based on cycle 12 operation of 620 ± 15 EFPD with an APSR pull, end-of-cycle (EOC) T_{avg} reduction, and CRG 7 withdrawal to 97%WD. The cycle 13 design includes an APSR pull, EOC T_{avg} reduction, CRG 7 withdrawal to 97%WD, and power coastdown.

3. GENERAL DESCRIPTION

The cycle 13 core consists of 177 fuel assemblies (FAs), each of which is a 15x15 array normally containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel consists of dished-end cylindrical pellets of uranium dioxide. The 72 batch 15 fuel assemblies are clad in M5^{TM} cladding and the remaining 105 assemblies in the cycle 13 core are clad in cold-worked Zircaloy-4. In batch 15, nine hundred twenty-eight fuel rods contain UO_2/Gd_2O_3 pellets in the central 123.20 inches of the fuel stack. The nominal fuel loadings for all fuel assemblies in cycle 13 are listed in Table 3-1. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are provided in Table 4-1.

Figure 3-1 is the core loading diagram for Davis-Besse Unit 1, cycle 13. Batch 15 is the second batch of fuel for Davis-Besse containing gadolinia (Gd_2O_3) and axial blankets. The initial enrichments in wt% ^{235}U and gadolinia concentrations in wt% Gd_2O_3 of all the cycle 13 batches are listed in Table 3-1. All batch 15 fuel rods except those bearing gadolinia have an upper and lower 6.05 inch blanket of 2.50 wt% ^{235}U pellets. Those fuel rods that contain gadolinia as a burnable absorber in a matrix of urania (UO_2), i.e. "Gd rods," have an upper and lower 9.90 inch blanket of 2.50 wt% ^{235}U pellets.

One batch 9F assembly, 4 batch 12A assemblies, 24 batch 12C assemblies, 1 batch 13A assembly, and 49 batch 13B assemblies will be discharged at the end of cycle 12. The remaining batch 13A and batch 13B FAs, along with batch 14A, 14B, 14C, and 14D FAs will be shuffled to their cycle 13 locations. All batch 13A FAs are on the core periphery. Batch 13A and 13B differ in fuel pin pre-pressure. Four batch 13B assemblies, discharged at the end of cycle 11, and two batch 10A assemblies, discharged at the end of cycle 9, will be reinserted in cycle 13. One batch 9G assembly, discharged at the end of cycle 8, will be reinserted in cycle 13 as the center FA.

The feed batch consists of 8 batch 15A, 8 batch 15B, 8 batch 15C, 44 batch 15D, and 4 batch 15E assemblies. The 4 batch 15E assemblies, residing in locations H14, P08, H02, and B08 contain M5^{TM} guide tubes and two M5^{TM} intermediate spacer grids, and have been pre-characterized. Four of the batch 15D fuel assemblies, located in L13, O06, F03, and C10 have also been pre-characterized. The feed batch will be loaded in a symmetric checkerboard pattern throughout the core. The cycle 13 shuffle scheme is a very low leakage (VLL) core loading. The VLL reload fuel shuffle scheme for cycle 13 will have a negligible effect on nuclear instrumentation response for all aspects of reactor startup and subsequent power operation. The cycle 13 design minimizes the number of same-quadrant shuffles into control rod positions to reduce the potential for incomplete rod insertion and excessive control rod assembly drag. The reduction in same-quadrant shuffles results in several cross-core shuffles despite past practices to avoid such shuffles. Nevertheless, the design maintains the number of cross-core shuffles as low as practical to reduce the potential for quadrant tilt amplification. Figure 3-2 is a quarter-core map showing each assembly's burnup at the beginning-of-cycle (BOC) 13 and its initial base enrichment.

Cycle 13 is operated in a feed-and-bleed mode. Fifty-three full-length Ag-In-Cd control rod assemblies, 928 Gd rods in the feed batch, and soluble boron control the core reactivity. There are no burnable poison rod assemblies (BPRAs) in cycle 13. In addition to the full-length control rods, eight Inconel-600 axial power shaping rods (gray APSRs) are provided for additional control of the axial power distribution. The gray APSR design lifetime was justified for an extension from 10 EFPY to 15 EFPY. The core locations and the rod group designations of the 61 control rods in cycles 12 and 13 are the same. Figure 3-3 shows the distribution of the Gd rods. The number of gadolinia rods per fuel assembly and initial Gd_2O_3 concentrations are also shown in Figure 3-3.

Table 3-1. Fuel Assembly Composition Data for Davis-Besse Cycle 13

Fuel Batch Number	Number of FAs	wt% ^{235}U Std./Gd Rod	wt% Gd_2O_3	Number of Gd Rods	Nominal Loading, KgU
9G	1	3.38	--	--	468.25
10A2	2	3.69	--	--	468.25
13A2	11	4.46	--	--	468.56
13B2	11	4.46	--	--	468.56
13B3	4	4.46	--	--	468.56
14A	16	4.47/3.80*	3.0	4	468.80
14B	8	4.47/3.80*	3.0	8	468.48
14C	16	4.47/3.13*	6.0	8	467.87
14D	36	4.47/3.13*	6.0	12	467.25
15A	8	4.88/4.15**	2.0	4	489.35
15B	8	4.88/4.15**	2.0	8	489.12
15C	8	4.88/2.93**	8.0	12	486.99
15D	44	4.88/4.15**	8@3.0	16	487.18
		4.88/2.93**	8@8.0		
15E	4	4.88/4.15**	2.0	8	489.12

* Uranium fuel rods have 5.984 inch top and bottom blankets of 2.50 wt% ^{235}U. Gd rods have 9.792 inch ends of 2.50 wt% ^{235}U.

** Uranium fuel rods have 6.050 inch top and bottom blankets of 2.50 wt% ^{235}U. Gd rods have 9.90 inch ends of 2.50 wt% ^{235}U.

Figure 3-1. Davis-Besse Cycle 13 Core Loading Diagram

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						13B2 P11	14C F07	13A2 R08	14C F09	10A2 011 9					
B				13A2 M13	14B P09	15B F	15C F	15E F	15C F	15B F	14B P07	13A2 M03			
C			13B2 013	15A F	15D F	15D F	14D E12	14C E08	14D E04	15D F	15D F	15A F	13B2 003		
D		13A2 011	15A F	14A P10	14A N03	14D L05	15D F	14D N09	15D F	14D L11	14A N13	14A L02	15A F	13A2 005	
E		14B K14	15D F	14A C12	13B3 K04 11	15D F	14D F13	15D F	14D F03	15D F	13B3 D07 11	14A C04	15D F	14B K02	
F	13B2 M14	15B F	15D F	14D E10	15D F	14D D09	15D F	14D H13	15D F	14D K12	15D F	14D E06	15D F	15B F	13B2 M02
G	14C G06	15C F	14D N05	15D F	14D 006	15D F	14C H09	14A P06	14C K08	15D F	14D 010	15D F	14D N11	15C F	14C G10
H	13A2 H15	15E F	14C H05	14D G12	15D F	14D C08	14A L14	9G D04 8	14A F02	14D 008	15D F	14D K04	14C H11	15E F	13A2 H01
K	14C K06	15C F	14D D05	15D F	14D C06	15D F	14C G08	14A B10	14C H07	15D F	14D C10	15D F	14D D11	15C F	14C K10
L	13B2 E14	15B F	15D F	14D M10	15D F	14D G04	15D F	14D H03	15D F	14D N07	15D F	14D M06	15D F	15B F	13B2 E02
M		14B G14	15D F	14A 012	13B3 N09 11	15D F	14D L13	15D F	14D L03	15D F	13B3 G12 11	14A 004	15D F	14B G02	
N		10A2 E13 9	15A F	14A F14	14A D03	14D F05	15D F	14D D07	15D F	14D F11	14A D13	14A B06	15A F	13A2 C05	
O			13B2 C13	15A F	15D F	15D F	14D M12	14C M08	14D M04	15D F	15D F	15A F	13B2 C03		
P				13A2 E13	14B B09	15B F	15C F	15E F	15C F	15B F	14B B07	13A2 E03			
R						13B2 B11	14C L07	13A2 A08	14C L09	13B2 B05					

Key

XXX YY Z

XXX - Batch ID
YY - Previous cycle location
Z - Previous cycle if reinsert
Note: "F" denotes fresh fuel assembly

Figure 3-2. Davis-Besse Cycle 13 Enrichment and BOC Burnup Distribution

	8	9	10	11	12	13	14	15
H	3.38 34,540	4.47 21,532	4.47 27,203	4.88 0	4.47 27,967	4.47 28,583	4.88 0	4.46 38.377
K	4.47 21,532	4.47 26,468	4.88 0	4.47 26,907	4.88 0	4.47 27,225	4.88 0	4.47 28,537
L	4.47 27,203	4.88 0	4.47 27,772	4.88 0	4.47 27,999	4.88 0	4.88 0	4.46 41,636
M	4.88 0	4.47 27,155	4.88 0	4.46 28,926	4.47 19,990	4.88 0	4.47 23,046	
N	4.47 27,967	4.88 0	4.47 28,170	4.47 20,053	4.47 21,609	4.88 0	4.46 39,699	
O	4.47 28,583	4.47 27,458	4.88 0	4.88 0	4.88 0	4.46 38,850		
P	4.88 0	4.88 0	4.88 0	4.47 23,096	4.46 39,567			
R	4.46 38,377	4.47 28,503	4.46 41,665					

x.xx	Initial Base Enrichment (not weighted for Gd)
xx,xxx	Burnup, MWd/mtU

Figure 3-3. Davis-Besse Cycle 13 Gadolinia Concentrations in Fresh Assemblies

	8	9	10	11	12	13	14	15
H				8×8.0 8×3.0			8×2.0	
K			8×8.0 8×3.0		8×8.0 8×3.0		12×8.0	
L		8×8.0 8×3.0		8×8.0 8×3.0		8×8.0 8×3.0	8×2.0	
M	8×8.0 8×3.0		8×8.0 8×3.0			8×8.0 8×3.0		
N		8×8.0 8×3.0				4×2.0		
O			8×8.0 8×3.0	8×8.0 8×3.0	4×2.0			
P	8×2.0	12×8.0	8×2.0					
R								

Key

NxZ.Z	Number of Gd Rods @ wt% Gd_2O_3

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

Table 4-1 lists the types of fuel assemblies and pertinent fuel parameters for Davis-Besse cycle 13. Batch 15 fuel, the Mark-B10K design, incorporates design modifications from batch 14 including the use of M5^{TM} advanced, low corrosion cladding and the introduction of the Trapper^{TM} debris resistant lower end fitting.

The implementation of these two features resulted in a redesign of the fuel rod for batch 15 relative to the fuel rod design of batch 14. The new batch 15 fuel rod is referred to as the B10K and features the previously mentioned low corrosion M5^{TM} cladding, M5^{TM} end-caps, a short lower end-cap, a larger diameter pellet (relative to the B9A rod of batch 14), a longer fuel stack, and a redesigned fuel rod plenum spring system.

Cycle 13 will also contain four M5^{TM} structural assemblies (batch 15E). In addition to the design changes mentioned above, these assemblies will have M5^{TM} guide tubes and M5^{TM} grids for the upper two intermediate grid locations. The fuel rod design for these four assemblies is identical to the B10K fuel and Gd rod design for the rest of the batch; therefore, they were treated the same as the rest of the batch 15 rods in the fuel rod mechanical analyses.

Mark-B10K fuel assemblies contain Gd rods in select locations of the 15x15 fuel rod array. The Gd rods are designed similar to the uranium fuel rods and are pressurized and seal welded. Both rod types contain axial blanket pellets with a 2.50 wt% ^{235}U enrichment. The batch 15 uranium and Gd rods are pressurized to the same pressure used in the batch 14 fuel rods.

Eight gray APSRAs and 53 Ag-In-Cd CRAs will be used in cycle 13. All of the CRAs are of the extended life design (ELCRA). No BPRAs will be used in cycle 13.

4.2. Fuel Rod Design

The fuel rod design and mechanical evaluation are discussed below.

4.2.1. Cladding Collapse

The computer code TACO3 (Reference 3) is used to provide conservative values of cladding temperature and pin pressure to the computer code CROV (Reference 4), which determines whether or not cladding collapse is predicted during the cycle.

B8A Fuel Rods (Batch 9G, 10A2)

The most limiting power history for batches 9G and 10A2 were determined. This history was enveloped by the power history used in the cycle 10 B8A fuel rod TACO3 analysis. Therefore, the results of the cycle 10 cladding collapse analysis apply. No creep collapse is predicted to occur through a burnup of at least 60 GWd/mtU, which exceeds the cycle 13 in-core life of these fuel rods.

B9A Fuel Rods (Batches 13 and 14)

The most limiting power histories for batches 13 and 14 were determined. The cycle 13 power histories for the two batches containing B9A fuel rods were shown to be enveloped by the power history used in the cycle 11 B9A fuel rod creep collapse analysis. Other analysis inputs such as rod prepressure and plenum volume are conservative when applied to the cycle 13 B9A rods. Results of the cycle 11 analysis show that no creep collapse is predicted to occur through a burnup of 60 GWd/mtU, which exceeds the cycle 13 in-core life of these fuel rods. This result also applies to the Gd rods since the power history in the B9A fuel rod analysis bounds their operation.

B10K Fuel Rods (Batch 15)

The most limiting power history for batch 15 was determined. This power history is enveloped by the power history used in the B10K creep collapse analysis. The fuel rod cladding creep collapse analysis for the batch 15 fuel rods showed that these rods have creep collapse lifetimes that exceed 65 GWd/mtU. The analysis applies to both the UO_2 and Gd rods. The batch 15 rods will not reach burnups in this range during cycle 13; therefore, all batch 15 rods are acceptable for resistance to creep collapse.

4.2.2. Cladding Stress

The stress parameters for the fuel rod designs are enveloped by conservative generic fuel rod stress analyses. The analysis method for M5^{TM} cladding differs somewhat from that for Zircaloy-4 cladding. For design evaluation, certain stress intensity limits for all Condition I and II events must be met. Limits are based on ASME criteria. Stress intensities are calculated in accordance with the ASME Code, which includes both normal and shear stress effects. These stress intensities are compared to S_m. The definition of S_m for M5[TM] differs from that for Zircaloy-4 cladding, as described in the following discussion.

Batches 9G, 10A2, 13 and 14 (Zircaloy-4)

S_m is equal to two-thirds of the minimum specified unirradiated yield strength of the material at the operating temperature (650°F). The stress intensity limits are as follows:

Primary general membrane stress intensities (P_m) shall not exceed S_m.

Local primary membrane stress intensities (P_l) shall not exceed 1.5 S_m. These include the contact stresses from the spacer grid stop and the fuel rod.

Primary membrane + bending stress intensities ($P_l + P_b$) shall not exceed 1.5 S_m.

Primary membrane + bending + secondary stress intensities ($P_l + P_b + Q$) shall not exceed 3.0 S_m.

where

P_m = General primary membrane stress intensity

P_l = Local primary membrane stress intensity

P_b = Primary bending stress intensity

Q = Secondary stress intensity

Stress intensity calculations combine stresses so that the resulting stress intensity is maximized.

For both the B8A and B9A UO_2 fuel rod designs, the margins are in excess of 12%. The following sources of conservatism were used in the stress analyses to ensure that all Condition I and II operating parameters were enveloped:

1. Low post-densification internal pressure, or as-built prepressure;

2. High system pressure;

3. High thermal gradient across the cladding;

4. Minimum specified cladding thickness.

For the Gd rods, the minimum margin is 6.4%. This number is lower than the margin of the B9A fuel rod due to a difference in the required fuel rod weld strength.

Batch 15 (M5^{TM})

The methodology that governs the stress analysis of the B10K M5^{TM} fuel rod is described in FCF's advanced cladding topical report (Reference 5). The major differences in the stress analysis methodology for M5^{TM} cladding are as follows: S_m for the M5^{TM} cladding material is equal to two-thirds of the lower bound yield strength in the hoop direction at operating temperature. The stress intensity limit for primary general membrane stress intensity (P_m) is S_m in tension and 1.5 S_m in compression. The remainder of the methodology is similar to the methodology for the Zircaloy-4 cladding material outlined above.

The minimum margin for the B10K stress analysis is 1.3%. The margins for the B10K Gd rod are the same as those for the UO_2 rod due to the similarity of the designs.

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% plastic tensile circumferential strain of the cladding. Cladding transient strain linear heat rate (LHR) limits were generated for each of the five fuel rod types in cycle 13 (B8A, B9A, B9A Gd, B10K, B10K Gd). Operation within these LHR limits ensures that the fuel rod cladding will not exceed the 1.0% transient strain limit. Table 4-2 lists limits for the B8A UO_2 rods of batches 9G and 10A2, Table 4-3 lists limits for the B9A UO_2 rods of batches 13 and 14, Table 4-4 lists limits for the B10K UO_2 rods of batch 15, Table 4-5 lists limits for the B9A Gd rods of batch 14, and Table 4-6 lists limits for the B10K Gd rods of batch 15.

4.2.4. Cladding Fatigue

The predicted fatigue factor must be less than or equal to 0.90 for the expected life of each fuel rod. The table below shows the maximum incore time for each batch at EOC-13 and the time limit resulting from the fatigue analysis for each rod type. Results in the table show that all the fuel rods meet the cladding fatigue criterion for cycle 13.

Rod design (batch)	Maximum time	Fatigue limit	Fatigue factor at limit
B8A (batches 9G&10A2)	4.5 years	5.79 years	0.90
B9A (batches 13&14)	5.3 years	10 years	0.574
B10K (batch 15)	1.9 years	10 years	>0.1

4.2.5. Cladding Oxide

Cladding waterside oxide thickness for FCF fuel is limited to 100 microns. FCF's COROSO2 model (Reference 6) generates oxide predictions at each input time step and each input axial node. Each of the five cycle 13 batches was evaluated for oxide using cycle 13 power histories to the maximum pin burnup for each batch. Acceptable oxide predictions were obtained for all five batches.

4.3. Thermal Design

All fuel assemblies in the cycle 13 core are thermally similar. The fresh batch 15 fuel are of the Mark-B10K design with axial blankets of slightly enriched ^{235}U fuel pellets and Gd fuel rods. Fuel performance for the Mark-B8, Mark-B10A, Mark-B10M, and Mark-B10K UO_2 fuel was evaluated with TACO3 (Reference 3). The Mark-B10K fuel assembly has the following features which differ from those of the Mark-B10 fuel assembly: debris-trapping lower end fitting, M5^{TM} fuel rod cladding, and longer fuel length. Nominal undensified input parameters used in the analyses are presented in Table 4-1. The GDTACO code (Reference 7) was used for predicting the fuel performance of the Gd rods. Densification effects were accounted for in the TACO3 and GDTACO code densification models.

The results of the thermal design evaluation of the cycle 13 core are summarized in Table 4-1. Cycle 13 core protection limits were based on linear heat rate (LHR) to centerline fuel melt (CFM) limits determined by the TACO3 and GDTACO codes.

The maximum fuel pin burnup at EOC-13 is predicted to be less than 56,000 MWd/mtU (batch 13B3). The fuel rod internal pressures have been evaluated with TACO3 and GDTACO for the highest burnup of each fuel rod type. The predicted internal pressures for all cycle 13 fuel were justified with the approved fuel rod gas pressure criterion methodology described in Reference 8.

4.4. Spacer Grid Deformation

The structural integrity of the fuel assembly spacer grids under faulted conditions was evaluated based on leak-before-break (LBB) methodology described in Reference 9. LBB methodology is consistent with FCF LOCA evaluations for B&W-designed raised and lowered loop plants, which includes Davis-Besse. Application of the LBB methodology confirmed that the requirement to maintain a coolable geometry is met for all faulted loading cases and for all fuel assemblies in the core.

4.5. Material Compatibility

The compatibility of all possible fuel-cladding-coolant-assembly interactions for the Mark-B10K fuel assemblies, containing M5™ material, was considered in the advanced cladding topical report (Reference 5), and was shown to be acceptable.

4.6. Operating Experience

Framatome Cogema Fuels operating experience with the Mark-B 15x15 assembly has verified the adequacy of its design. Mark-B fuel assemblies have operated successfully in over 100 fuel cycles at eight nuclear power plant facilities. Axial blanket fuel has operated successfully in eight cycles at four B&W units, and Gd rods have operated successfully in four cycles at three B&W units.

M5™ cladding material has been used in demonstration assemblies for two cycles at one B&W unit and for multiple cycles at two Westinghouse units. The TRAPPER™ debris resistant lower end fitting has operated at two Westinghouse units in a total of nearly 1000 fuel assemblies. Davis-Besse cycle 13 will be the first use of the TRAPPER™ at a B&W unit.

Table 4-1. Fuel Design Parameters

	Batch 9G	Batch 10A2	Batch 13A2/13B2/ 13B3	Batch 14A/14B	Batch 14C/14D
Fuel assembly type	Mark-B8A	Mark-B8A	Mark-B10A	Mark-B10M	Mark-B10M
No. of assemblies	1	2	11/11/4	16/8	16/36
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377	0.377
Undensified active fuel length, in.	143.2	143.2	140.6	140.733 140.634 (Gd)	140.733 140.634 (Gd)
Pellet OD, in.	.3686	.3686	.3700	.3700	.3700
Fuel pellet initial density, %TD mean	95.0	95.0	96.0	96.0	96.0
Initial fuel batch enrichment, wt% ^{235}U	3.38	3.69	4.46	4.47 (3.80 for 3 wt% Gd) w/2.50 axial blanket	4.47 (3.13 for 6 wt% Gd) w/2.50 axial blanket
Average burnup BOC, MWd/mtU	34,540	35,289	38,218	25,751	25,751
Cladding collapse burnup, MWd/mtU[a]	>60,000	>60,000	>60,000	>60,000	>60,000
Maximum pin burnup, MWd/mtU	52,852	48,323	55,238	50,421	54,846
Nom. linear heat rate at 2772 MWt, kW/ft[b]	6.14	6.14	6.25	6.25	6.25
Minimum linear heat rate to melt, kW/ft	20.5	20.5	22.3	22.3 (20.8 Gd)	22.3 (20.8 Gd)

(a) Calculated using method from Reference 4.
(b) LHR calculations include a 0.973 energy deposition factor.

Table 4-1. Fuel Design Parameters (cont.)

	Batch 15A	Batch 15B	Batch 15C	Batch 15D	Batch 15E
Fuel assembly type	Mark-B10K	Mark-B10K	Mark-B10K	Mark-B10K	Mark-B10K M5[TM] structural assemblies
No. of assemblies	8	8	8	44	4
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.380	0.380	0.380	0.380	0.380
Undensified active fuel length, in.	143.0	143.0	143.0	143.0	143.0
Pellet OD, in.	.3735	.3735	.3735	.3735	.3735
Fuel pellet initial density, %TD mean	96.0	96.0	96.0	96.0	96.0
Initial fuel batch enrichment, wt% ^{235}U	4.88 (4.15 for 2 wt% Gd)	4.88 (4.15 for 2 wt% Gd)	4.88 (2.93 for 8 wt% Gd)	4.88 (4.15 for 3 wt% Gd) (2.93 for 8 wt% Gd)	4.88 (4.15 for 2 wt% Gd)
Average burnup BOC, MWd/mtU	0	0	0	0	0
Cladding collapse burnup, MWd/mtU[(a)]	>60,000	>60,000	>60,000	>60,000	>60,000
Maximum pin burnup, MWd/mtU	29,296	31,744	31,195	32,299	30,579
Nom linear heat rate at 2772 MWt, kW/ft[(b)]	6.15	6.15	6.15	6.15	6.15
Minimum linear heat rate to melt, kW/ft	22.1 (21.1 for 2 wt% Gd)	22.1 (21.1 for 2 wt% Gd)	22.1 (20.3 for 8 wt% Gd)	22.1 (20.7 for 3 wt% Gd) (20.3 for 8 wt% Gd)	22.1 (21.1 for 2 wt% Gd)

(a) Calculated using method from Reference 4.
(b) LHR calculations include a 0.973 energy deposition factor.

Table 4-2. B8A Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
12,000	45.8
20,000	33.3
32,000	27.0
40,000	24.8
52,000	21.6
60,000	19.8

Table 4-3. B9A UO_2 Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
13,000	28.9
21,000	27.6
33,000	26.8
41,000	28.0
53,000	26.0
61,000	21.2

Table 4-4. B10K UO_2 Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
20,000	28.9
30,000	27.3
40,000	24.6
50,000	23.7
60,000	21.3
65,000	20.3

Table 4-5. B9A Gd Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
30,000	23.5
40,000	23.3
50,000	22.8
60,000	20.5

Table 4-6. B10K Gd Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
20,000	27.4
30,000	25.2
40,000	24.9
50,000	24.3
60,000	19.0
65,000	18.3

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters for the cycle 12 and 13 designs. The values for cycles 12 and 13 were generated with the NEMO code (Reference 10). Differences in core physics parameters are to be expected between the cycles due to the changes in fuel and burnable poison types and concentrations that create changes in flux and burnup distributions. A design with increased initial ^{235}U enrichments and differences in the shuffle pattern, BPRA loading, and the gadolinia burnable poison create the differences in the physics parameters between cycles 12 and 13.

Figure 5-1 illustrates a representative relative power distribution for BOC 13 at full power with equilibrium xenon, group 7 inserted to nominal HFP position, and gray APSRs partially inserted. The ejected rod worths in Table 5-1 are the maximum calculated values. Calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod position limits presented in Section 8. The adequacy of the shutdown margin with cycle 13 rod worths is shown in Table 5-2. The following conservatisms were applied to the shutdown calculations:

1. 6% uncertainty on net rod worth (Reference 11).

2. Off-nominal flux distribution (e.g. xenon transient allowance).

The off-nominal flux distribution allowance was taken into account to ensure that the effects of operational maneuvering transients were included in the shutdown analysis. In previous cycles a specific allowance was taken for the poison material depletion allowance. Current calculations have determined that the depletion allowance is adequately bounded by the off-nominal flux distribution allowance. Furthermore, improvements to the NEMO model remove the necessity to adjust the power deficit.

5.2. Changes in Nuclear Design

The design changes for cycle 13 include increased enrichment, no BPRA poison, the introduction of M5™ cladding, and a longer fuel and gadolinia stack height for batch 15. There are also asymmetries in the full core loading by batch (enrichment) and burnup in core locations N02 and A10, compared with the other FAs in symmetric locations. These changes were incorporated in the physics model where significant. Reference 10 illustrates the calculational accuracy obtainable with NEMO for gadolinia cores.

No significant operational or procedural changes exist with regard to axial or radial power shape, xenon, or tilt control. The stability and control of the core with APSRs withdrawn was analyzed. The operating limits (COLR changes) for the reload cycle are given in Section 8.

Table 5-1. Davis-Besse Unit 1, Cycle 13 Physics Parameters[a]

	Cycle 12[b]	Cycle 13[c]
Cycle length, EFPD	684	683
Cycle burnup, MWd/mtU	22,877	22,469
Average core burnup - 683 EFPD[b], MWd/mtU	39,568	39,444
Initial core loading, mtU	82.9	84.3
Critical boron[d] - 0 EFPD, ppm		
HZP	2,323	2315
HFP	2,109	2095
Critical boron[d] - 683 EFPD[b], ppm		
HZP	198	237
HFP	5[e]	5[e]
Control rod worths - HFP, 4 EFPD, %Δk/k		
Group 6	0.89	1.00
Group 7	0.93	0.89
Group 8	0.12	0.11
Control rod worths - HFP, 683 EFPD[b], %Δk/k		
Group 7	1.02	0.92
Max ejected rod worth - HZP, %Δk/k		
0 EFPD, Groups 5-8 inserted (N-12)	0.27	0.45
683 EFPD[b], Groups 5-7 inserted (N-12)	0.28	0.45
Max stuck rod worth - HZP, %Δk/k		
0 EFPD (N-12)	0.43	0.46
683 EFPD[b] (M-13)	0.56	0.69
Power deficit[f] - HZP to HFP, %Δk/k		
4 EFPD	-1.50	-1.49
683 EFPD[b]	-3.00	-3.01
Doppler coeff[f,g] - HFP, 10^{-3} %Δk/k/°F		
0 EFPD[h]	-1.59	-1.58
683 EFPD[b], 0 ppm	-1.79	-1.78
Moderator coeff[f] - HFP, 10^{-2} %Δk/k/°F		
0 EFPD[h]	-0.18	-0.21
683 EFPD[b], 0 ppm[i]	-3.51	-3.52
Temperature coeff[f] - HZP, 10^{-2} %Δk/k/°F		
683 EFPD[b], Groups 1-7 Inserted, M13 out, 0 ppm	-2.59	-2.56

Table 5-1. Davis-Besse Unit 1, Cycle 13 Physics Parameters [a] (cont.)

	Cycle 12[b]	Cycle 13[c]
Boron worth[f] - HFP, ppm/%Δk/k		
0 EFPD	165	169
683 EFPD[b]	124	126
Xenon worth[f] - HFP, %Δk/k		
4 EFPD	2.48	2.41
683 EFPD[b]	2.74	2.69
Effective delayed neutron fraction[f] - HFP		
4 EFPD	0.00648	0.00643
683 EFPD[b]	0.00525	0.00530

(a) Calculations at 0 EFPD are done with No Xenon. All other calculations are at 100%FP Eq Xe.
(b) Cycle 12 values are from Reference 2. EOC values calculated at 684 EFPD for cycle 12.
(c) Based on cycle 11 length of 645.3 EFPD (actual) and cycle 12 length of 620 EFPD.
(d) Control rod group 8 is inserted for calculation at 0 EFPD and withdrawn for calculation at 683 EFPD.
(e) Power coastdown to 684 EFPD at 5 ppm for cycle 12 and to 683 EFPD at 5 ppm for cycle 13.
(f) All calculations done with control rod groups 1-7 at 100% WD and control rod group 8 at nominal HFP position, unless otherwise noted.
(g) Doppler temperature coefficient calculated using a distributed fuel temperature.
(h) Cycle 13 values were calculated at 2207 ppm (includes allowances for reactivity anomalies and shutdown window flexibility); cycle 12 values were calculated at 2221 ppm.
(i) These values were calculated with the control rods at rod index 260% WD.

Table 5-2. Shutdown Margin Calculation for Davis-Besse, Cycle 13

	BOC, %Δk/k	EOC, %Δk/k	
	4 EFPD	636 EFPD	683 EFPD
	Group 8 in	Group 8 in	Group 8 out
Available Rod Worth			
Total rod worth, HZP	6.03	6.69	6.73
Maximum stuck rod worth, HZP	-0.50	-0.66	-0.69
Net Worth	5.53	6.03	6.04
Less 6% Uncertainty	-0.33	-0.36	-0.36
Total available worth	5.20	5.67	5.68
Required Rod Worth			
Power deficit, HFP to HZP	1.49	2.98	3.01
Off-nominal flux distribution allowance	0.30	0.30	0.30
Max allowable inserted rod worth at RI = 260% WD	0.31	0.49	0.49
Total required worth	2.10	3.77	3.80
Shutdown Margin			
Total available minus total required	3.10	1.90	1.88

Note: Required shutdown margin is 1.00%Δk/k.

Figure 5-1. Davis-Besse Cycle 13 Relative Power Distribution at BOC (4 EFPD), Full Power, Equilibrium Xenon, Group 7 at 90% WD, Group 8 at 30% WD

	8	9	10	11	12	13	14	15
H	0.646	0.949	1.037	1.328	7 1.039	1.010	1.265	0.437
K		0.994	1.300	1.111	1.307	1.054	1.167	0.449
L			1.111	1.341	8 1.051	1.330	1.157	0.310
M				1.107	1.174	1.261	0.662	
N					7 1.125	1.159	0.341	
O						0.432		
P								
R								

x	Inserted Rod Group Number
x.xxx	Relative Power Density

6.0 THERMAL-HYDRAULIC DESIGN

The cycle 13 core is composed of several Mark-B assembly designs including the latest design, the Mark-B10K (batch 15). The Mark-B10K design contains a slightly higher hydraulic resistance for the lower end fitting than other fuel designs in the core (Mark-B8, Mark-B10A, and Mark-B10M designs). There are also four Mark-B10K assemblies that contain M5^{TM} guide tubes and two M5^{TM} intermediate spacer grids. Evaluations have shown there is no DNB transition core penalty for the Mark-B10K during cycle 13 since the benefit of a longer fuel stack for the Mark-B10K fuel offsets the DNB effect of the higher hydraulic resistance of the lower end fitting. Therefore, the reference core analysis for cycle 13 remains the same as that used for cycles 10, 11, and 12. The approved analysis methods described in Reference 12 and the statistical core design (SCD) methodology, Reference 13, were utilized in the analysis. The four Mark-B10K assemblies that contain M5^{TM} guide tubes and two M5^{TM} intermediate spacer grids were shown to have acceptable operation within thermal-hydraulic limits based on the results of the reference core analysis.

The Mark-B10A, Mark-B10M, and Mark-B10K fuel designs contain optimized guide tubes that minimize the control rod guide tube core bypass flow. The cycle 13 specific core bypass flow rate of 5.8% exceeds the 5.3% value used in the reference core analysis. The effect of this increased bypass flow rate is offset by retained DNB margin.

The DNB-based thermal-hydraulic analyses for cycle 13 are applicable for UO_2 and gadolinia fuel rods. The applicability of the DNBR results to the assemblies containing axial blanket fuel rods was further verified in the evaluation of power distribution check cases where the DNB peaking margin for the cycle-specific axial flux shapes was confirmed.

An improved spacer grid restraint system was initially incorporated in the batch 14 fuel design. The modification results in an increase in the instrument guide tube subchannel hydraulic resistance. The effect of the modification is offset by retained DNB margin.

Table 6-1 provides a summary comparison of the DNB analysis parameters for cycles 12 and 13.

Table 6-1. Limiting Thermal-Hydraulic Design Conditions, Cycles 12 and 13

	Cycle 12	Cycle 13
Design power level, MWt	2772	2772
Nominal core exit pressure, psia	2200	2200
Minimum core exit pressure, psia	2135	2135
Reactor coolant flow, gpm	380,000	380,000
Core bypass flow, %	5.3[a]	5.3[a]
DNBR modeling	SCD	SCD
Reference design (radial x local) power peaking factor	1.795	1.795
Reference design axial flux shape	1.65 chopped cosine	1.65 chopped cosine
Hot channel factors		
Enthalpy rise	1.015	1.015
Heat flux	N/A[b]	N/A[b]
Flow Area	0.97	0.97
Active fuel length, in.	140.6[c]	140.6[c]
Avg heat flux at 100% power, 10^5 Btu/h-ft^2	1.89	1.89
Max heat flux at 100% power, 10^5 Btu/h-ft^2	5.60	5.60
CHF Correlation	BWC	BWC
CHF Correlation DNB limit	1.40 TDL[d]	1.40 TDL[d]
Minimum DNBR		
at 102% power	2.02	2.02
at 112% power	1.79	1.79

(a) Used in the analysis
(b) The hot channel factor for heat flux is no longer applicable in DNB calculations as allowed by Reference 12
(c) Value used is conservative for DNB analysis relative to the 143.0 in. batch 15 active fuel length
(d) Thermal design limit

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each USAR accident analysis has been examined with respect to changes in the cycle 13 parameters to determine the effects of the cycle 13 reload and to ensure that thermal performance during hypothetical transients is not degraded.

The radiological dose consequences of the USAR Chapter 15 accidents were evaluated using conservative radionuclide source terms that bound the cycle specific source terms for Davis-Besse cycle 13. With the exception of the fuel handling accidents, which are discussed below, the results of the dose evaluation show that the offsite radiological doses for each accident did not increase relative to cycle 12 results. Thus the doses are not adversely impacted by the cycle 13 design and remain below the respective acceptance criteria values as documented in the USAR.

The dose consequences for the fuel handling accidents, inside and outside of containment, were re-calculated using the cycle 13-specific radial power peaking factor. The results are presented below in Table 7-1. The cycle 13 doses remain below the NUREG-0800 acceptance criteria.

7.2. Accident Evaluation

The key parameters that have the greatest effect on determining the outcome of a transient can typically be classified in three major areas. These areas are: (1) core thermal, (2) thermal-hydraulic and (3) kinetics parameters, including the reactivity feedback coefficients and control rod worths.

Fuel thermal analysis parameters from each batch in cycle 13 are given in Table 4-1. The cycle 13 thermal-hydraulic maximum design conditions are presented in Table 6-1. A comparison of the key kinetics parameters from the USAR and cycle 13 is provided in Table 7-2.

The EOC moderator temperature coefficient listed in Table 7-2 for cycle 13 is the 3-D, hot full power (HFP) temperature coefficient. An evaluation was performed to verify the acceptability of the cycle 13 moderator temperature coefficients for all USAR accidents excluding steam line breaks. The results of the evaluation were acceptable for all USAR accidents, excluding steam line breaks.

The steam line break accident was evaluated based on the total reactivity change from 532°F to a minimum temperature of 510°F. The temperature coefficient used in safety analysis of the steam line break is -3.10×10^{-2} %Δk/k/°F. This value is based on the sum of the moderator density, control rod worth degradation and Doppler reactivity, over the temperature range from 532°F to 510°F. The combined EOC temperature coefficient for cycle 13 is shown in Tables 5-1 and 7-2 as -2.56×10^{-2} %Δk/k/°F. Since the safety analysis value for the EOC temperature coefficient is more negative than the cycle 13 value, the steam line break analysis remains bounding for cycle 13.

Loss-of-coolant accident (LOCA) analyses for the B&W 177-FA raised-loop nuclear steam system (NSS) have been performed to calculate allowable LOCA linear heat rate (LHR) limits that are applicable to the Mark-B8A, Mark-B10A, Mark-B10M, and Mark-B10K fuel types. The RELAP5/MOD2-B&W ECCS evaluation model techniques and assumptions as described in BAW-10192PA (Reference 14), were used in the Mark-B10A, Mark-B10M, and Mark-B10K analyses. These analyses were performed at 2966 MWt to support a future power uprate. The CRAFT2-based ECCS evaluation model as described in BAW-10104P, Rev. 5 (Reference 15), was used in the Mark-B8A analyses. These analyses were performed at 2772 MWt. Since the Mark-B8A fuel was not reanalyzed with the RELAP5/MOD2-B&W ECCS evaluation model, the Mark-B8A LHR limits were adjusted to account for the change to the RELAP5/MOD2-B&W LOCA methodology as well as changes to plant boundary conditions.

Table 7-3 shows the maximum allowable LOCA linear heat rate limits for the different types of fuel in the Davis-Besse Unit 1 cycle 13 core as functions of burnup. The LHR limits for Mark-B10A, Mark-B10M, and Mark-B10K were determined at 1.02 * 1.07 * 2772 MWt (i.e. 1.02 * 2966 MWt). Sensitivity studies performed at lower power levels using the uprated power LHR limits produced more severe results (i.e. increased PCT, hydrogen generation, and peak local oxidation). Therefore, a reduction of up to 0.2 kW/ft on the Mark-B10A, Mark-B10M, and Mark-B10K LHR limits is necessary for application in the maneuvering analyses for core power levels other than 2966 MWt to ensure that the LHRs determined at 2966 MWt remain limiting. The Mark-B8A LHR limits are based on an initial core power level of 2772 MWt and do not require further adjustment based on the current rated thermal power level.

For batches 9G and 10A2, linear interpolation between the elevation-specific linear heat rate limits at 24,500 MWd/mtU and the linear heat rate limit of 12.0 kW/ft at 52,000 MWd/mtU was justified for cycle 13. For batch 9G, the LHR limit for any burnup beyond 52,000 MWd/mtU can be interpolated between 12 kW/ft at 52,000 MWd/mtU and 10.5 kW/ft at 60,000 MWd/mtU.

For the batch 13 and 14 UO_2 fuel, the cycle-specific fuel rod performance data and predicted radial peaks for cycle 13 were found to be bounded by the fuel data used in the Mark-B10A/Mark-B10M (B9A fuel rods) LOCA analyses, which were calculated using the TACO3 fuel performance code (Reference 3). At high fuel burnups, the limits for batches 13 and 14 are reduced in order to maintain the internal fuel rod pressure less than or equal to the limit based on the NRC-approved fuel rod gas pressure criterion (Reference 8).

The maximum allowable LOCA linear heat rate limits for the fresh batch 15 Mark-B10K UO_2 fuel were determined using material properties for M5™ cladding (Reference 5). The effect of loading Batch 15E assemblies with M5™ spacer grids occupying the top two grid locations below the upper grid support into a Davis-Besse core was evaluated using the RELAP5/MOD2-B&W evaluation model. The evaluation determined that the M5™ grids have no adverse impact on the LOCA LHR limits.

The linear heat rate limits for batch 14 and 15 fuel rods containing gadolinia, which are based on fuel rod performance data from the GDTACO (Reference 7) fuel rod performance code, were determined for evaluation in the subsequent power distribution analysis (as discussed in Section 8). The Gd rod linear heat rates were shown to be non-limiting with respect to the UO_2 linear heat rate limits.

LBLOCA analyses for the Davis-Besse plant do not currently support a moderator temperature coefficient (MTC) of $+0.9 \times 10^{-2}$ %Δk/k/°F for core power levels at or below 95 percent full power. LOCA analyses were performed at various partial power levels to define a maximum permissible (most positive) MTC versus power level. The predicted MTC curve for cycle 13 was compared to the resulting allowable MTC to confirm that the cycle design is sufficiently bounded.

An analysis was performed using the RELAP5/MOD2-B&W ECCS evaluation model to assess the conditions under which an end of cycle (EOC) T_{avg} reduction maneuver could be performed. The results of the analysis showed that operation for an EOC T_{avg} reduction of 10°F, with ± 2°F uncertainty, and an MTC more negative than -10 pcm/°F provides LOCA results that are bounded by the nominal T_{avg} LOCA results. The 10°F reduction bounds the currently justified T_{avg} reduction allowance of 7°F. The EOC MTC values are significantly more negative than the -10 pcm/°F allowed.

The continued validity of the non-LOCA USAR analyses was assessed for a withdrawal of the APSRs and a reduction in T_{avg} near the end of cycle 13. It was determined that the non-LOCA USAR analyses remain valid for the APSR withdrawal and T_{avg} reduction near EOC.

It is concluded by the examination of cycle 13 core thermal, thermal-hydraulic and kinetics properties, with respect to acceptable previous cycle values, that the cycle 13 core reload will not adversely affect the ability to safely operate the Davis-Besse plant during cycle 13. Considering the previously accepted design basis used in the USAR and subsequent cycles, the transient evaluation of cycle 13 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 13 are bounded by the USAR and/or subsequent cycle analyses.

Table 7-1. Fuel Handling Accident Dose Consequences

Accident/Dose Type	USAR	SRP	Cycle 13
Fuel Handling Accident Inside Containment:	(Rem)	(Rem)	(Rem)
2 HR. Thyroid at EAB	62.6	75	73.4
2 HR. Whole-Body at EAB	0.55	6	0.168
30 Day Thyroid at LPZ	3.26	75	3.82
30 Day Whole-Body at LPZ	0.03	6	0.009
Fuel Handling Accident Outside Containment:			
2 HR. Thyroid at EAB	0.85	75	0.988
2 HR. Whole-Body at EAB	0.15	6	0.032
30 Day Thyroid at LPZ	0.044	75	0.052
30 Day Whole-Body at LPZ	0.008	6	0.002

Table 7-2. Comparison of Key Parameters for Accident Analysis

Parameter	USAR Value	Cycle 13 Value	Bounding Value is:
BOL[a] Doppler coeff, 10^{-3} %Δk/k/°F	-1.28	-1.58	Less Negative
EOL[a] Doppler coeff, 10^{-3} %Δk/k/°F	-1.45[b]	-1.78	More Negative
BOL HFP moderator coeff, 10^{-2} %Δk/k/°F	+0.13	-0.21	Less Negative/ More Positive
BOL HZP moderator coeff, 10^{-2} %Δk/k/°F	+0.90	+0.39	Less Negative/ More Positive
EOL HFP moderator coeff, 10^{-2} %Δk/k/°F	-4.0	-3.52	More Negative
EOL temperature coeff (532 to 510°F) 10^{-2} %Δk/k/°F	-3.10	-2.56	More Negative
BOL All rod group worth (HZP), %Δk/k	10.0	5.977	Larger[f]
Boron reactivity worth (HFP), ppm/%Δk/k	100	171	Note (g)
Max ejected rod worth (HFP), %Δk/k	0.65	<0.65[c]	Larger
Max dropped rod worth (HFP), %Δk/k	0.65[h]	<0.20	Larger
Initial boron conc (HFP), ppm	1407	2207[d]	Note (g)

(a) BOL denotes beginning of life; EOL denotes end of life.

(b) -1.77×10^{-3} %Δk/k/°F was used for steam line failure analysis (also see Note e).

(c) Calculational uncertainty (15%) is applied to the limit in the design analysis when determining cycle-specific regulating group position limits.

(d) Includes allowances for ^{10}B atom variations and reactivity anomalies.

(e) The EOL Doppler coefficient value used in the steam line break analysis is less negative than, and therefore not bounding for, the cycle 13 Doppler coefficient. However the steam line break is evaluated based on the EOL temperature coefficient, which considers the combined effects of the temperature decrease on the moderator temperature coefficient, Doppler coefficient, control rod worth, boron concentration and moderator density. The analysis value for the EOL temperature coefficient is greater than, and therefore bounding for, the cycle 13 temperature coefficient.

(f) For the analysis to remain bounding, the cycle-specific value must be ≤ 10.0 %Δk/k

(g) For the analysis to remain bounding, the ratio of the critical boron concentration to the boron reactivity worth for the safety analysis must be greater than the corresponding ratio for the cycle-specific values.

(h) Dropped rod accident analyses performed subsequent to the issuance of the Davis-Besse USAR, which determined that middle-of-life moderator coefficients are limiting, considered a dropped rod worth of 0.33 %Δk/k, which also bounds the cycle 13 value.

Table 7-3. Bounding Values for Allowable LOCA Peak Linear Heat Rates

Mark-B8A Fuel Type as Determined at 2772 MWt
Allowable Peak LHR for Specified Burnup, kW/ft

Core Elevation, ft	Batch 9G/10A2 24,500 MWd/mtU	Batch 9G/10A2 52,000 MWd/mtU	Batch 9G 60,000 MWd/mtU
0	16.0	12.0	10.5
2	16.0	12.0	10.5
4	14.8	12.0	10.5
6	15.2	12.0	10.5
8	15.8	12.0	10.5
10	14.9	12.0	10.5
12	14.1	12.0	10.5

Mark-B10A and Mark-B10M Fuel Types as Analyzed at 2966 MWt*
Allowable Peak LHR for Specified Burnup, kW/ft

Core Elevation, ft	0 MWd/mtU	35,000 MWd/mtU	62,000 MWd/mtU
0.0	17.8	17.0	13.0
2.506	17.8	17.0	13.0
4.264	17.3	15.9	13.0
6.021	16.8	15.5	13.0
7.779	16.2	16.0	13.0
9.536	15.5	15.5	13.0
12.0	14.7	14.7	13.0

Mark-B10K Fuel Type as Analyzed at 2966 MWt*
Allowable Peak LHR for Specified Burnup, kW/ft

Core Elevation, Ft	0 MWd/mtU	35,000 MWd/mtU	LOCA LHR Limit at pin pressure of 3000 psia and indicated burnup	62,000 MWd/mtU
0.0	17.8	17.0	14.9 @ 58 GWd/mtU	13.7
2.506	17.8	17.0	14.9 @ 58 GWd/mtU	13.7
4.264	17.3	15.9	14.9 @ 58 GWd/mtU	13.7
6.021	16.8	15.5	14.6 @ 59 GWd/mtU	13.7
7.779	16.2	16.0	14.3 @ 60 GWd/mtU	13.7
9.536	15.5	15.5	13.7 @ 62 GWd/mtU	13.7
12.0	14.7	14.7	13.0 @ 62 GWd/mtU	13.0

Linear interpolation between burnup points to calculate the Allowable LHR is permitted.

* These LHR limits must be reduced by up to 0.2 kW/ft for power levels less than 2966 MWt.

8. PROPOSED CORE OPERATING LIMITS REPORT

The Core Operating Limits Report (COLR) has been revised for cycle 13 operation to accommodate the influence of the cycle 13 core design on power peaking, reactivity, and control rod worths. Revisions to the cycle-specific parameters were made in accordance with the requirements of NRC Generic Letter 88-16 and Technical Specification 6.9.1.7. The core protective and operating limits were determined from a cycle 13 specific power distribution analysis using NRC approved methodology provided in the references of Technical Specification 6.9.1.7.

A cycle 13 specific analysis was conducted to generate the axial power imbalance protective limits, corresponding power/imbalance/flow trip allowable values, and the Limiting Conditions for Operation (rod index, axial power imbalance, and quadrant tilt), based on the NRC-approved methodology described in Reference 12. The analysis incorporates DNB peaking limits based on the allowable increase in design (radial x local) peaking provided by the statistical core design methodology described in Reference 13. The effects of control rod group 7 and gray APSR repositioning were included explicitly in the analysis. The analysis also determined that the cycle 13 core operating limits provide protection for the overpower condition that could occur during an overcooling transient because of nuclear instrumentation errors.

The capability to perform the end of cycle (EOC) hot full power maneuver is included in the rod index and axial power imbalance operating limits in the COLR. The maneuver consists of an APSR withdrawal designed to occur at 626 $\pm$ 10 EFPD and a T_{avg} reduction of up to 7°F (actual) to extend HFP operation. The xenon stability index after APSR withdrawal was determined to be -0.0805 h^{-1}, which demonstrates the axial stability of the core during operation with the APSRs fully withdrawn. An additional evaluation of power peaking margins was performed to verify the acceptability of the core limits because the cycle 13 APSR withdrawal window occurs outside the 60 EFPD operating window assumption in the B&W Owners Group generic EOC T_{avg} reduction maneuver analyses. The evaluation results are included in the rod index and axial power imbalance operating limits in the COLR. The analysis verified that the operational maneuver at EOC is bounded by the safety analysis assumptions and will be accommodated by the core protective and operating limits.

The maximum allowable LOCA linear heat rate limits used in the analysis are based on the ECCS analysis described in Section 7.2. Table 8-4 provides the burnup- and elevation-dependent LOCA linear heat rate limits for each incore segment for input to the Nuclear Applications Software (NAS). Table 8-5 provides the burnup- and elevation-dependent LOCA linear heat rate limits with elevation in units of feet for input to the Fixed Incore Detector Monitoring System (FIDMS) software. The linear heat rate limits in Tables 8-4 and 8-5 are reduced by 0.2 kW/ft compared to those provided in Section 7.2 (Table 7-3). The reduction is reflected in the maneuvering analysis (by up to 0.2 kW/ft) and was made in order to account for the power level dependence of the LOCA kW/ft limits calculated for cycle 13 operation. The linear heat rate limits provided in Tables 8-4 and 8-5 are the basis of the F_Q power peaking surveillance limits required by Technical Specification 3/4.2.2.

As part of determining the core protective and operating limits, an evaluation of margin to the DNB, LOCA, cladding strain, and centerline fuel melt limits for the individual gadolinia fuel rods and the $M5^{TM}$ structural fuel assemblies was performed. The gadolinia rods and the $M5^{TM}$ structural assemblies were determined to be non-limiting during the entire cycle.

The measurement system-independent rod position and axial power imbalance limits determined by the cycle 13 analysis were error adjusted to generate operating limits for power operation. Figures 8-1 through 8-4 and Figures 8-6 through 8-12 are revisions to the operating limits contained in the COLR and have been adjusted for instrument error. Figure 8-5 provides the control rod core locations and group assignments for cycle 13. Figures 8-13 and 8-14 are the core protective limits and RPS imbalance trip allowable values. A nuclear instrumentation scaled difference amplifier gain of at least 2.0 was assumed in determining the RPS imbalance trip allowable values. Figure 8-15 provides the allowable radial peaking factors to be used in the calculation of the $F^N_{\Delta H}$ limits specified in Table 8-6. They are the basis of the $F^N_{\Delta H}$ power peaking surveillance limits required by Technical Specification 3/4.2.3. The values specified in Table 8-6 and Figure 8-15 are used by both the NAS and FIDMS software applications. The 3-RCP axial power imbalance operating limits provided in Figures 8-7 through 8-12 are based on the 4-RCP LOCA LHR limits, however, they include the power level dependence of the LOCA kW/ft limits calculated for cycle 13 operation. Table 8-1 presents the power- and burnup-dependent quadrant power tilt limits for cycle 13, Table 8-2 provides the negative moderator temperature coefficient limit for cycle 13, and Table 8-3 provides minimum linear heat rate to melt (kW/ft) limits. Tables 8-4 and 8-5 provide the F_Q limits and Table 8-6 provides the $F^N_{\Delta H}$ limits. These limits are preserved by the rod index and axial power imbalance operating limits required by Technical Specification 3/4.1.3.6 and 3/4.2.1. The F_Q limits for NAS application reflect the four different active fuel lengths and respective allowable linear heat rate limits as functions of incore segment (core elevation) and burnup. The F_Q limits for FIDMS application are defined in terms of a single core average linear heat rate. The $F^N_{\Delta H}$ relationship defined in Table 8-6 ensures acceptable DNBR performance using statistical core design methodology in the event of the limiting Condition I and II transient. The family of curves in Figure 8-15 preserves the initial condition DNBR limit in the form of equivalent allowable initial condition peaking. Allowable $F^N_{\Delta H}$ values can be determined based on particular axial peaks at a given axial elevation for either three or four RC pump operation.

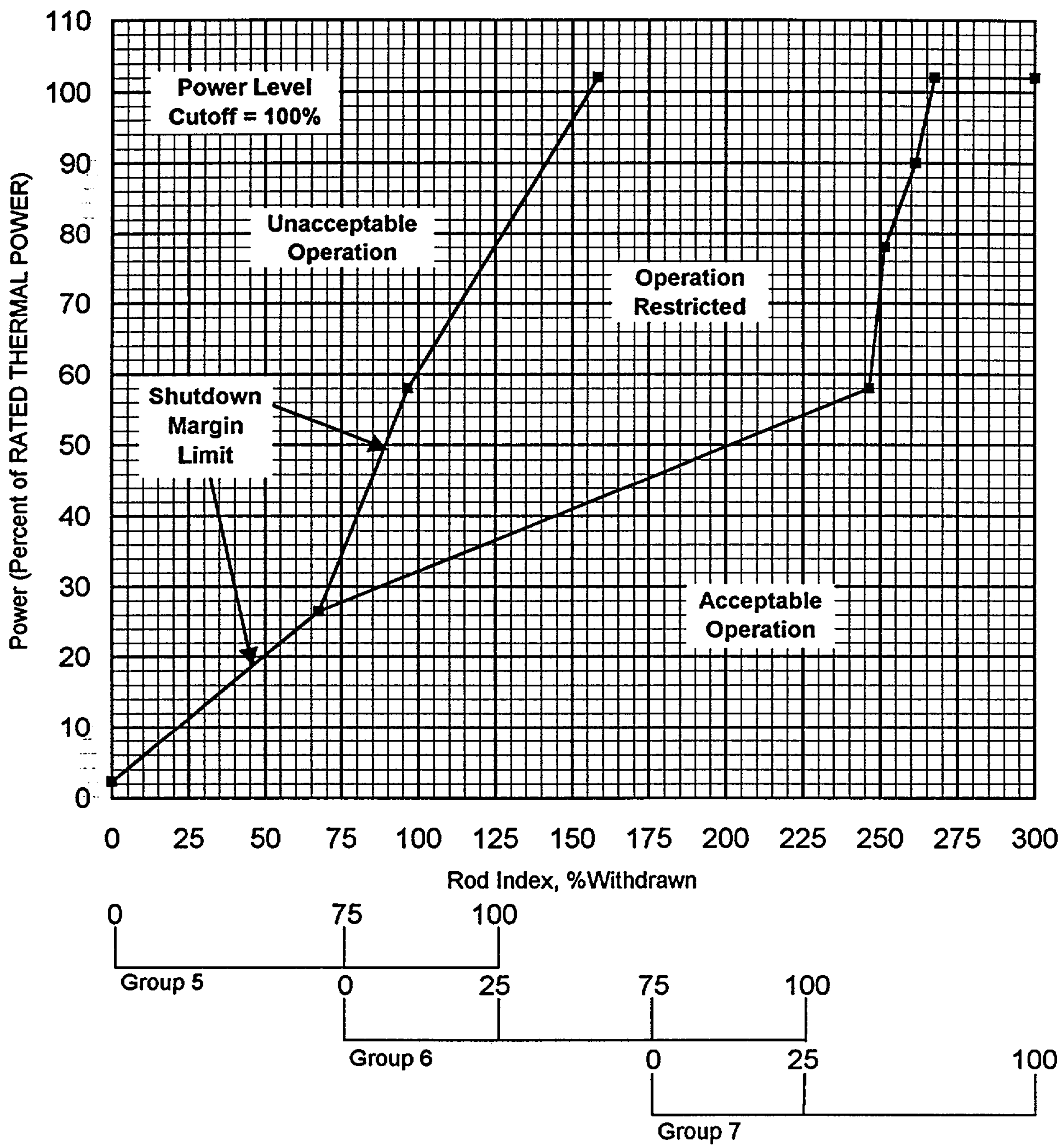
Boric acid volume storage for the boric acid addition system (BAAS) and the borated water storage tank (BWST) required by Technical Specifications 3/4.1.2.8, 3/4.1.2.9, and Figure 3.1-1 were verified to be acceptable for cycle 13. In addition, the minimum boron concentration requirements for the BWST given in Technical Specifications 3/4.1.2.8, 3/4.1.2.9, and 3/4.5.4 were verified to be acceptable for cycle 13 operation.

Based on the analyses and operating limit revisions described in this report, the Final Acceptance Criteria ECCS limits will not be exceeded, nor will the thermal design criteria be violated.

Figure 8-1

Figure Regulating Group Position Operating Limits
0 to 300 ±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 13

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

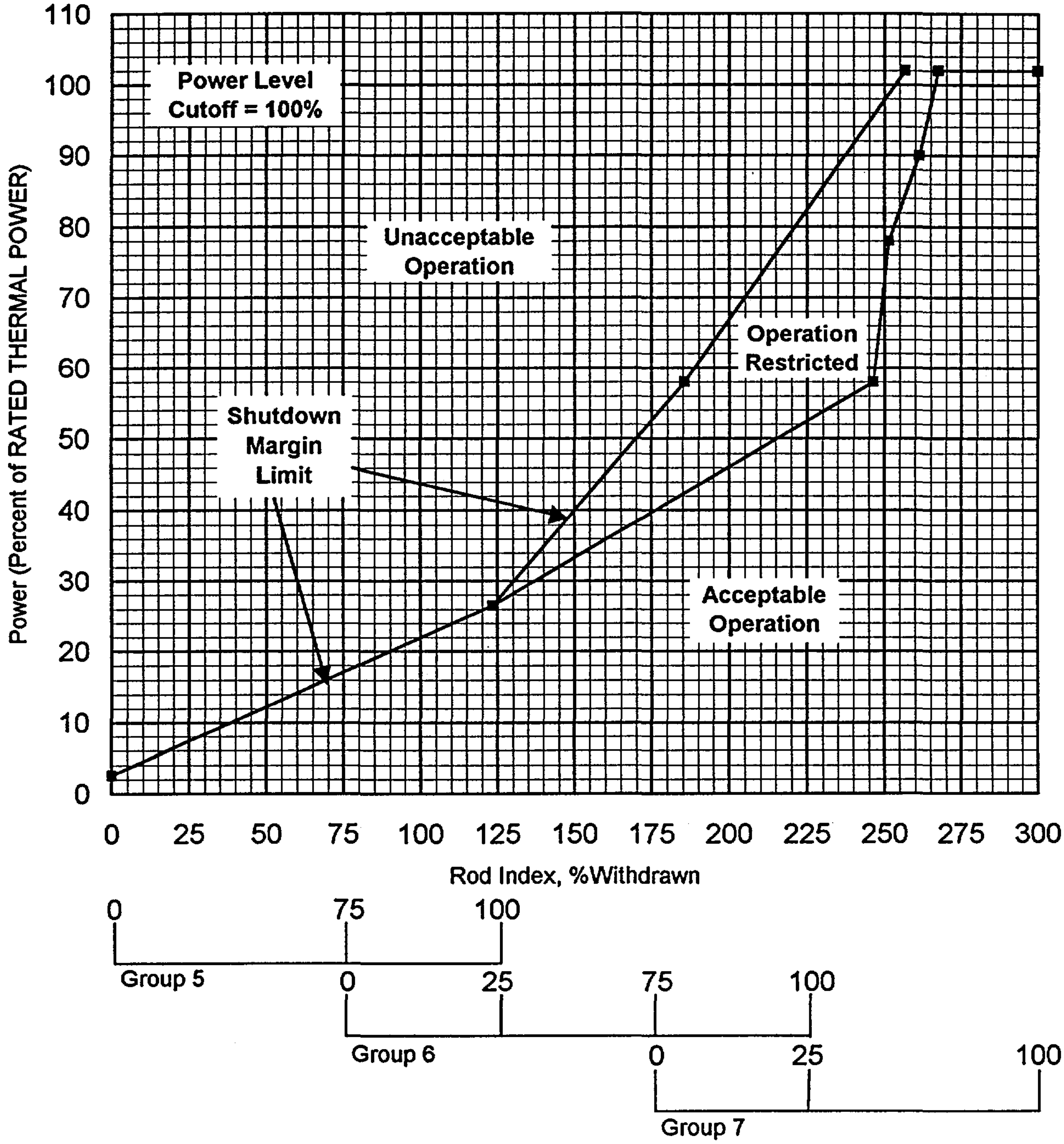


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-2

Figure Regulating Group Position Operating Limits
After 300 ±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 13

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

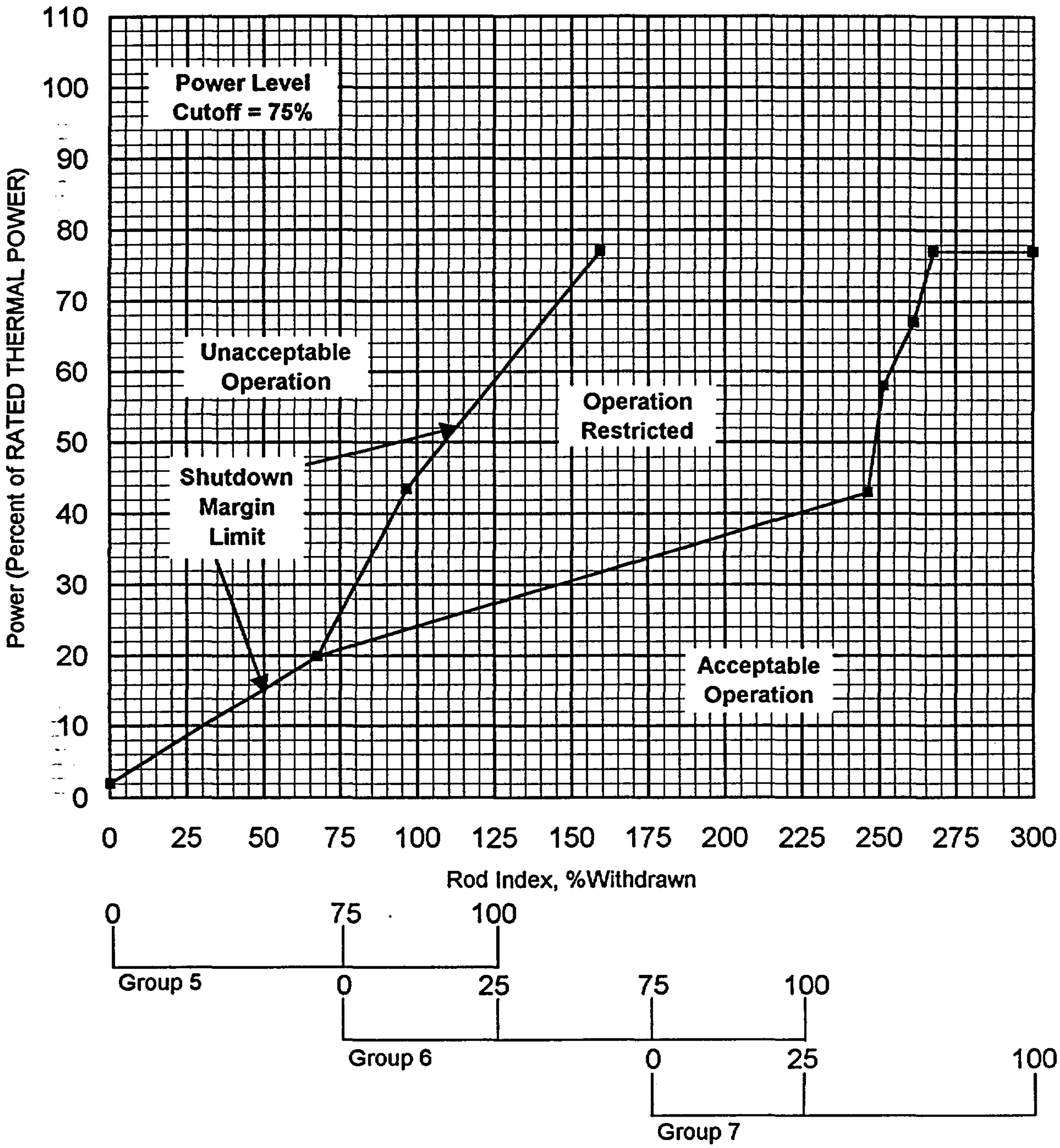


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-3

Figure Regulating Group Position Operating Limits
0 to 300 ±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 13

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

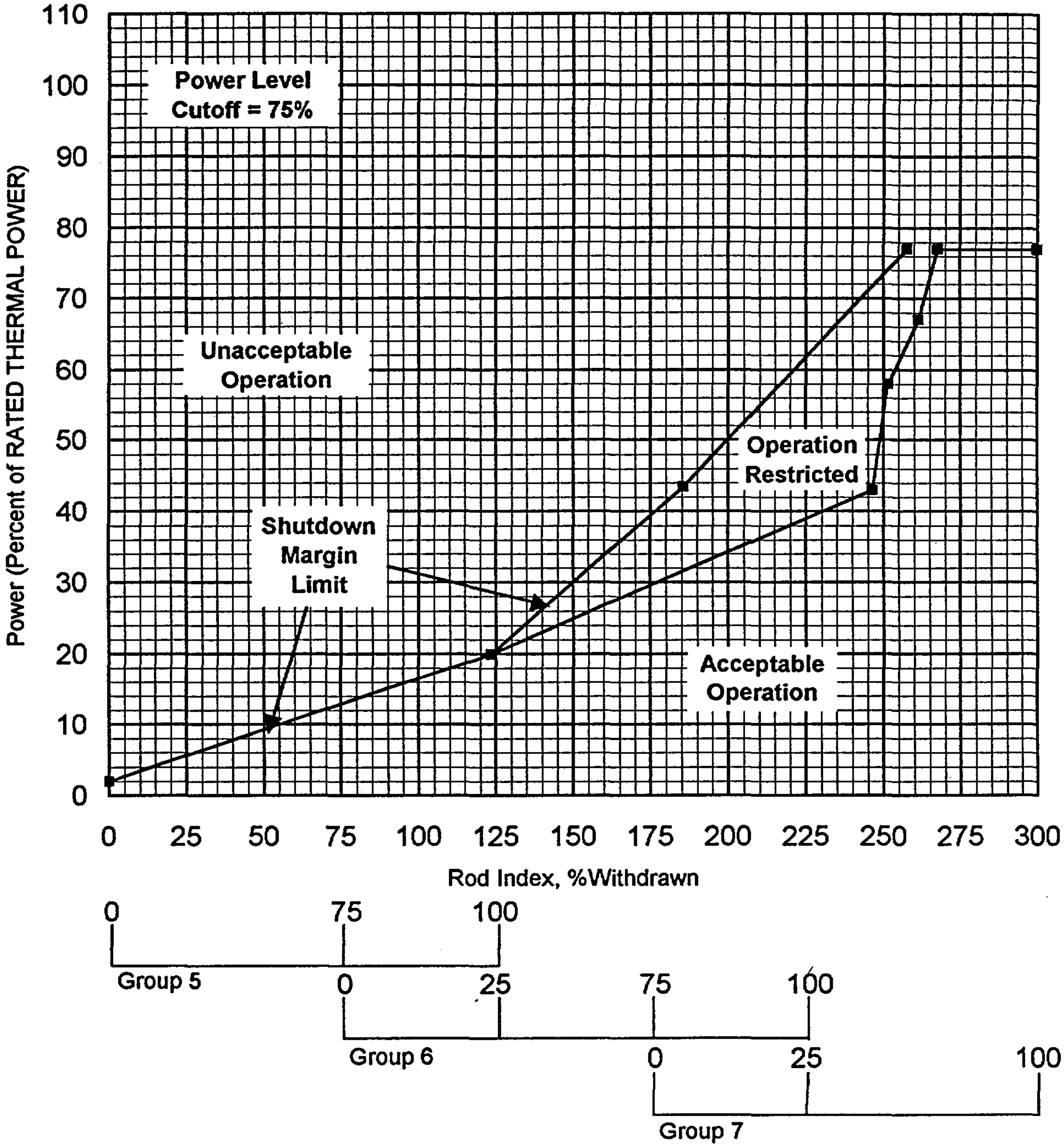


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-4

Figure Regulating Group Position Operating Limits After 300 $\pm$10 EFPD, Three RC Pumps -- Davis-Besse 1, Cycle 13

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



Note 1: A Rod Group overlap of 25 $\pm$5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-5

Figure Control Rod Core Locations
and Group Assignments
Davis-Besse 1, Cycle 13

This Figure is referred to by
Technical Specification 3.1.3.7

← N

(X)

(W) / (Y)

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B						4		6		4					
C					2		5		5		2				
D				7		8		7		8		7			
E			2		5						5		2		
F		4		8		6		3		6		8		4	
G			5				1		1				5		
H		6		7		3		4		3		7		6	
K			5				1		1				5		
L		4		8		6		3		6		8		4	
M			2		5						5		2		
N				7		8		7		8		7			
O					2		5		5		2				
P						4		6		4					
R															

(Z)

X Group Number

Group	No. of Rods	Function	Group	No. of Rods	Function
1	4	Safety	5	12	Control
2	8	Safety	6	8	Control
3	4	Safety	7	8	Control
4	9	Safety	8	8	APSRs
			Total	61	

Figure 8-6

<u>Figure APSR Position Operating Limits</u>

This Figure is referred to by Technical Specification 3.1.3.9

Before APSR Pull: 0 EFPD to 626 ±10 EFPD,
<u>Three or Four RC pumps operation</u>*

Lower Limit: 0 %WD

Upper Limit: 100 %WD

After APSR Pull: 626 ±10 EFPD to End-of-Cycle
<u>Three or Four RC pumps operation</u>*

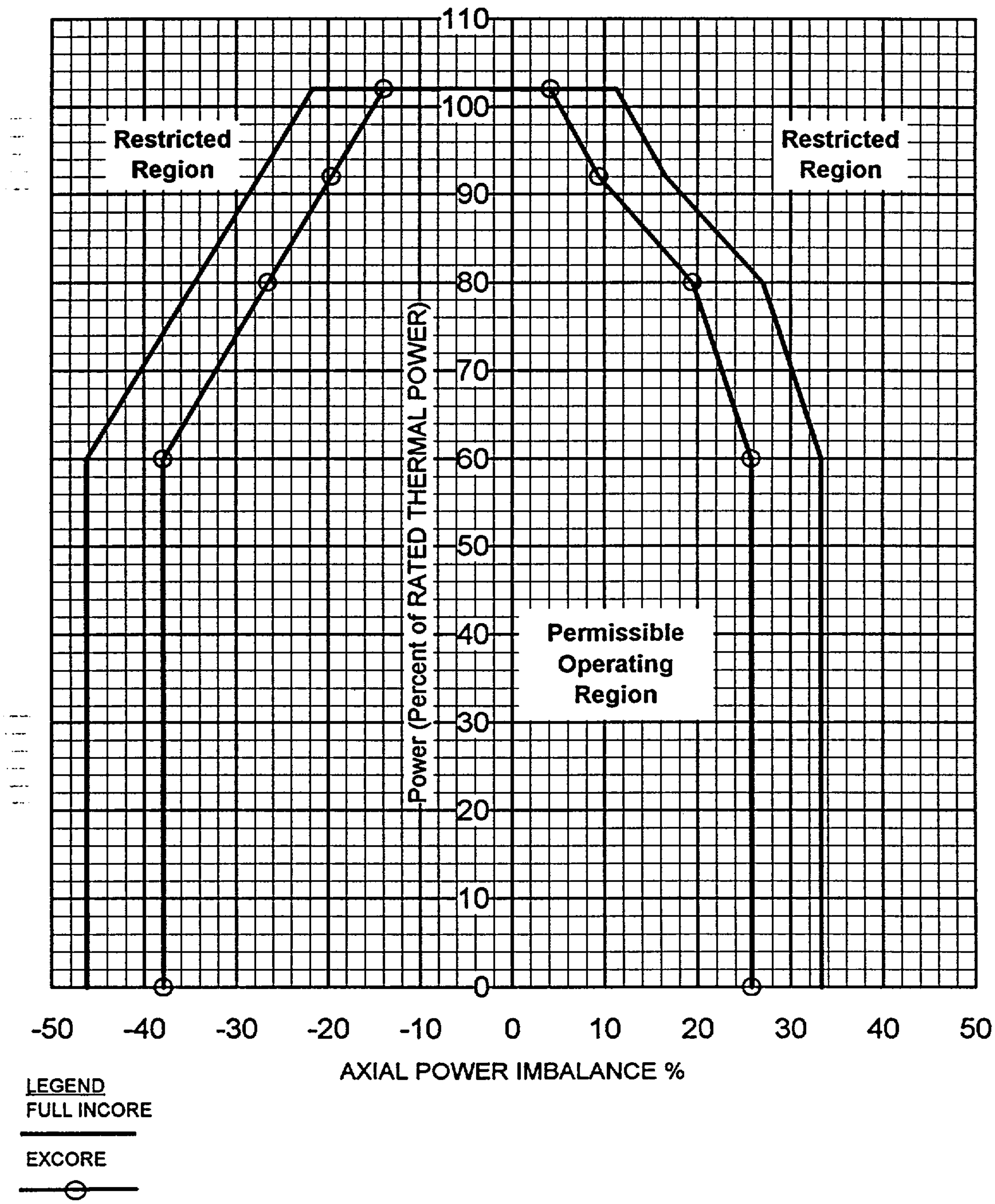
Insertion Prohibited (maintain ≥99 %WD)

* Power restricted to 77% for 3 pump operation

Figure 8-7

Figure AXIAL POWER IMBALANCE Operating Limits
0 to 300 ±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 13

This Figure is referred to by
Technical Specification 3.2.1

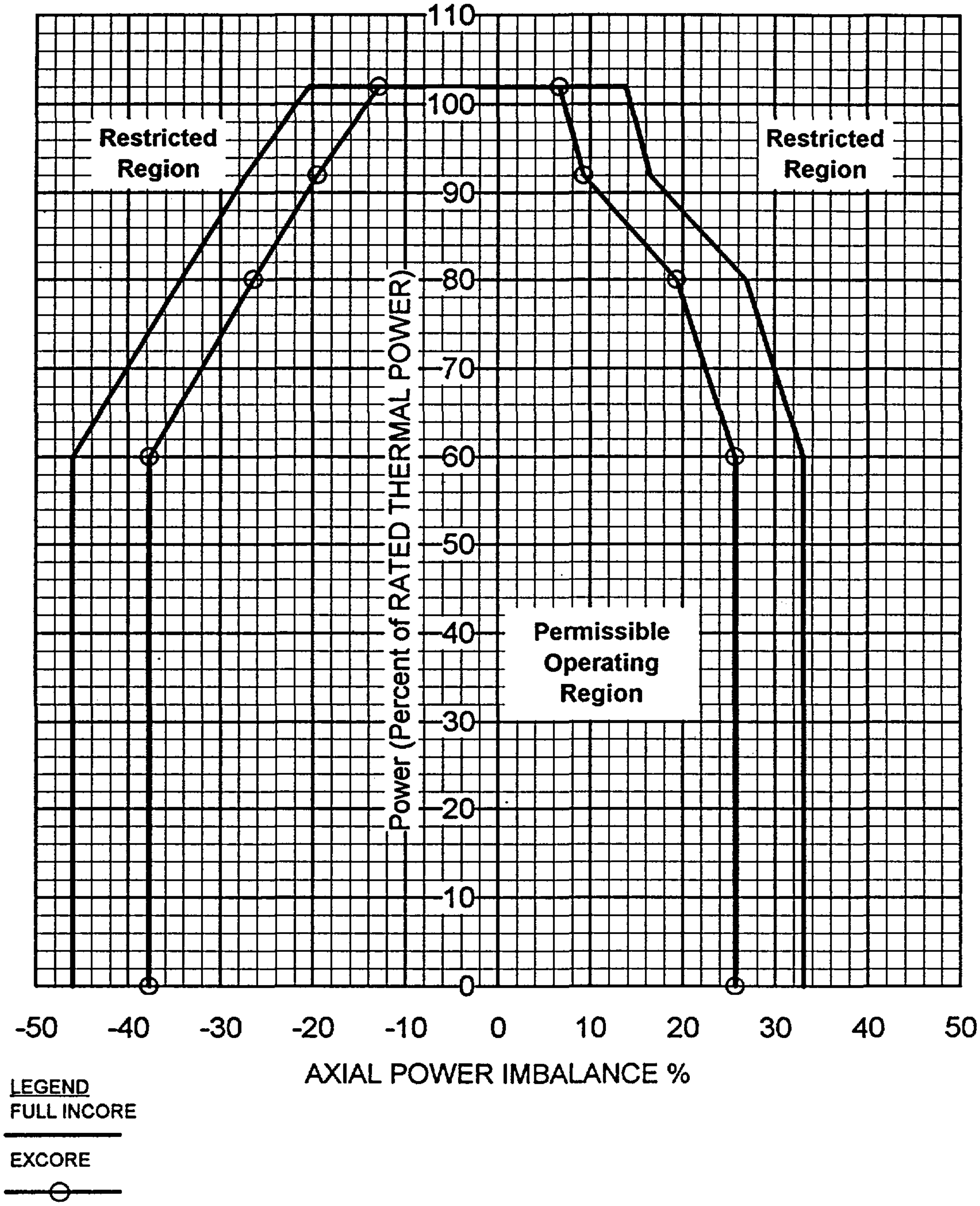


Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-8

Figure AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 626 ±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 13

This Figure is referred to by
Technical Specification 3.2.1

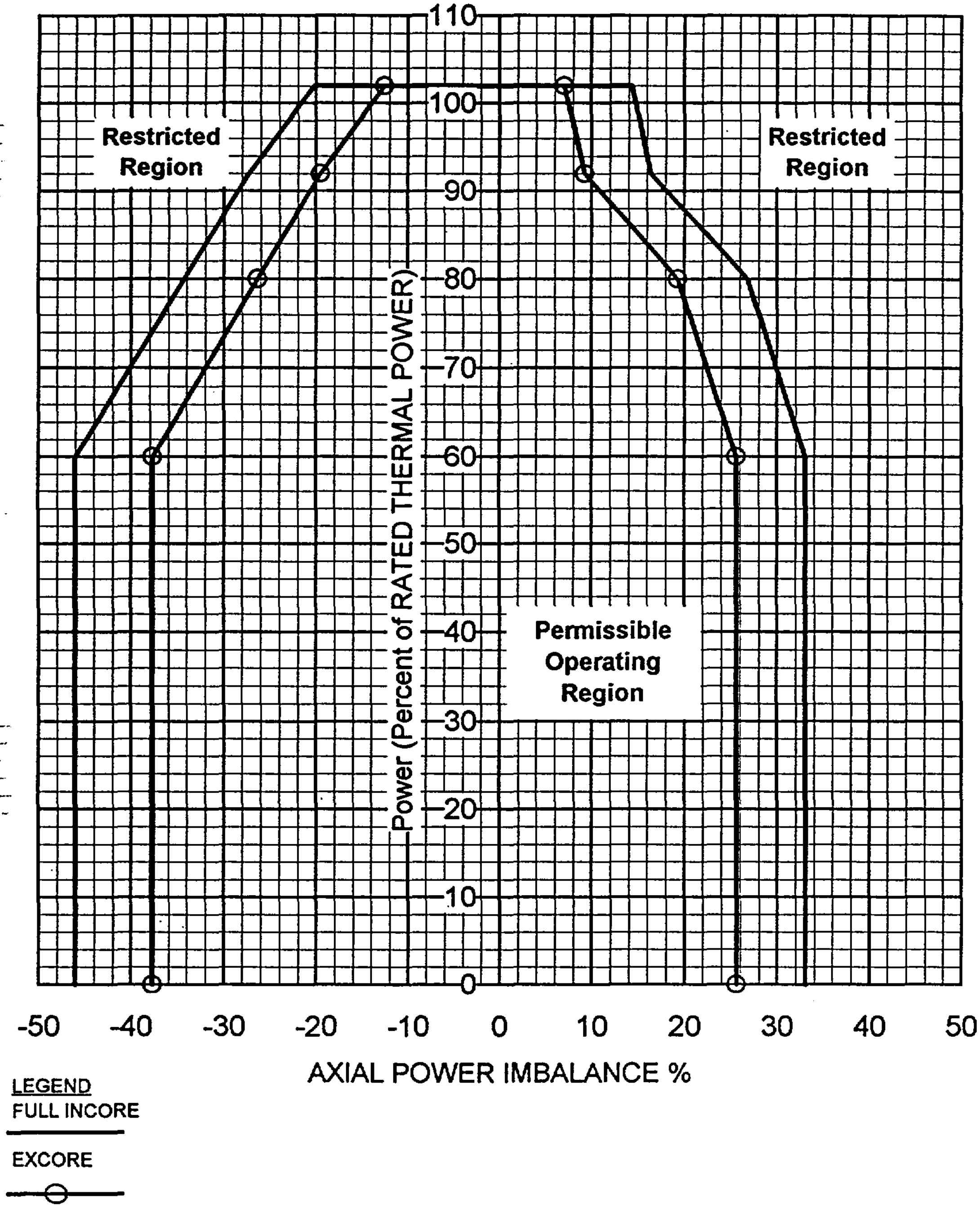


Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-9

Figure AXIAL POWER IMBALANCE Operating Limits
After 626 ±10 EFPD, Four RC Pumps --
Davis-Besse 1, Cycle 13

This Figure is referred to by
Technical Specification 3.2.1

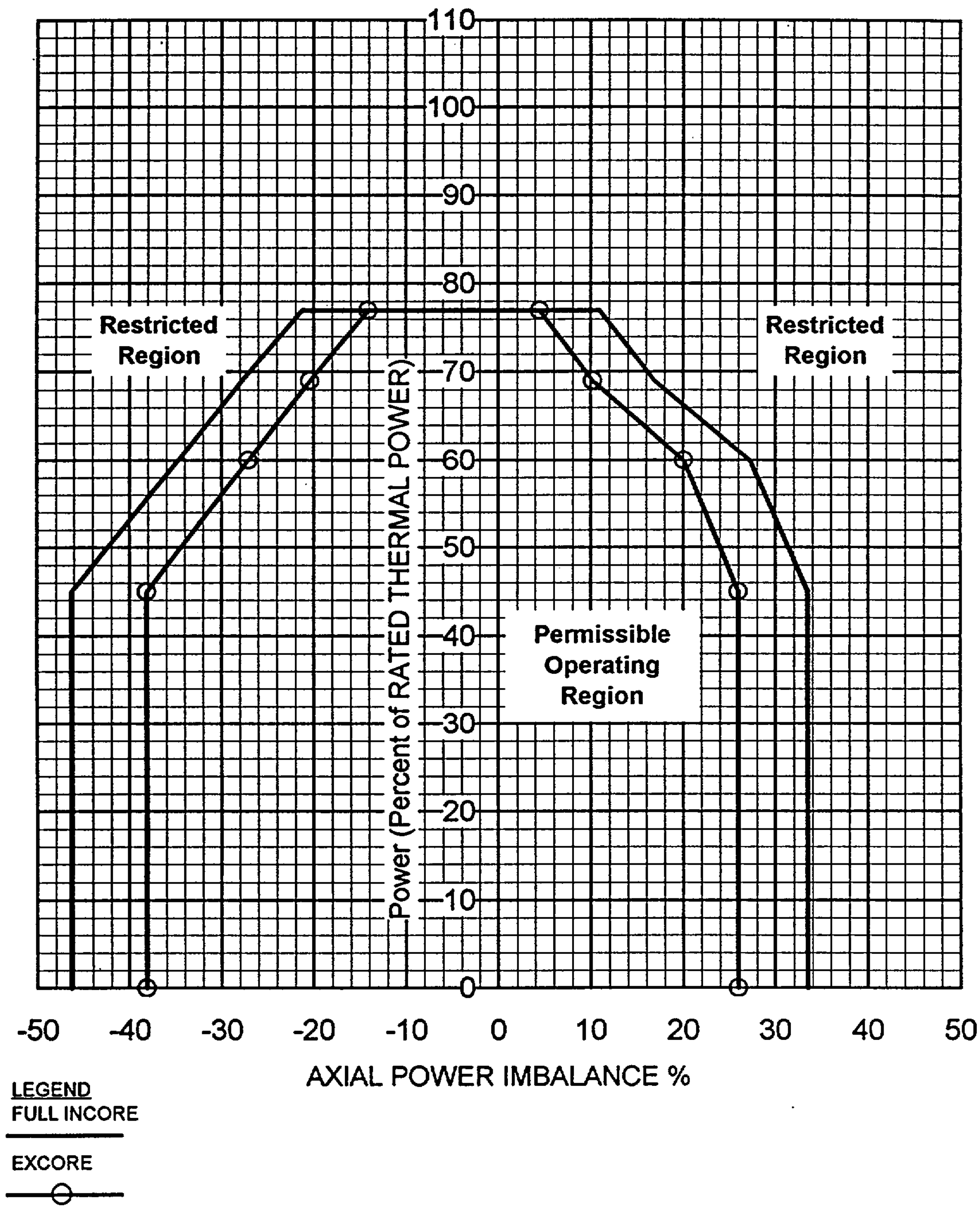


Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-10

Figure AXIAL POWER IMBALANCE Operating Limits
0 to 300 ±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 13

This Figure is referred to by
Technical Specification 3.2.1

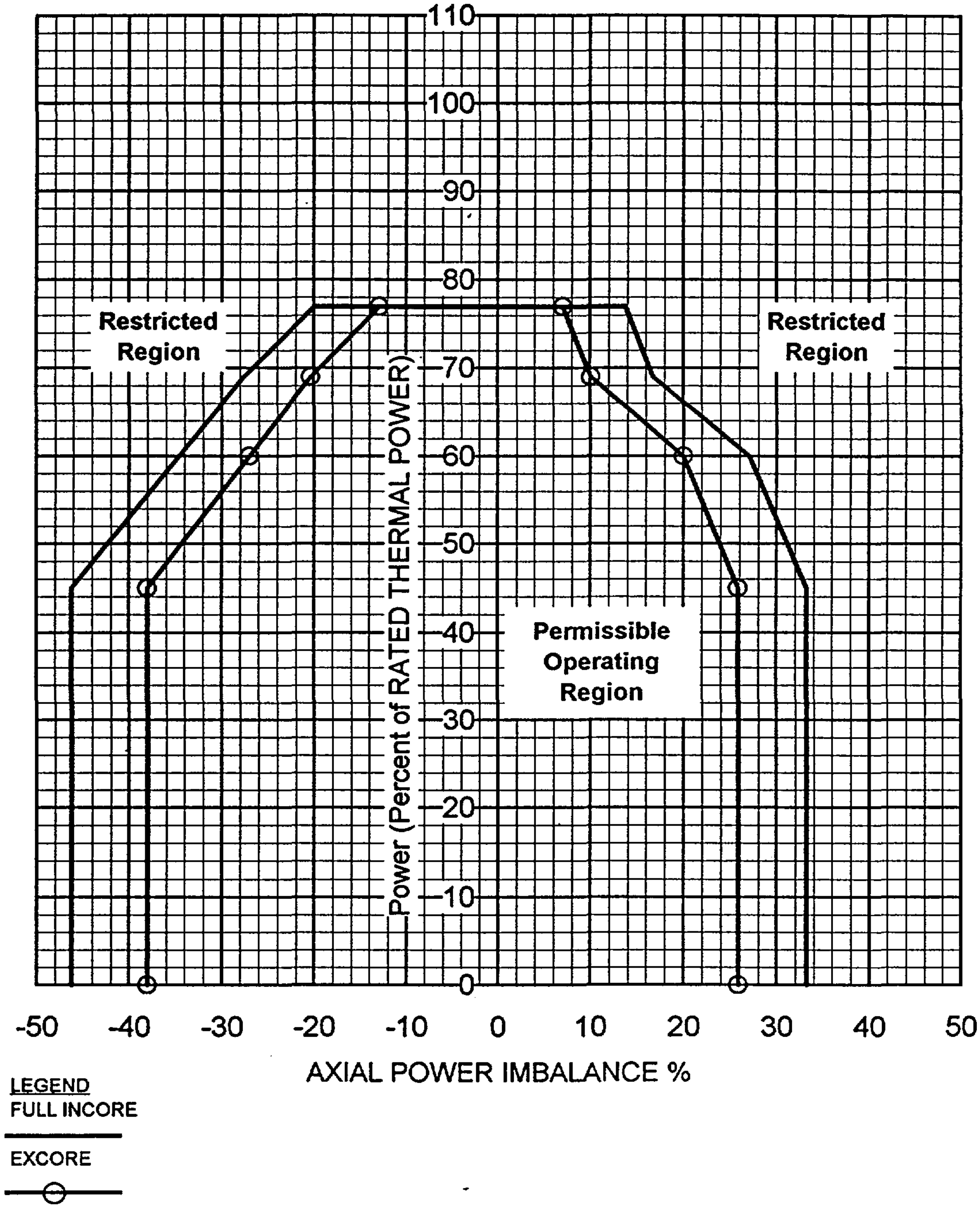


Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-11

Figure AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 626 ±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 13

This Figure is referred to by
Technical Specification 3.2.1

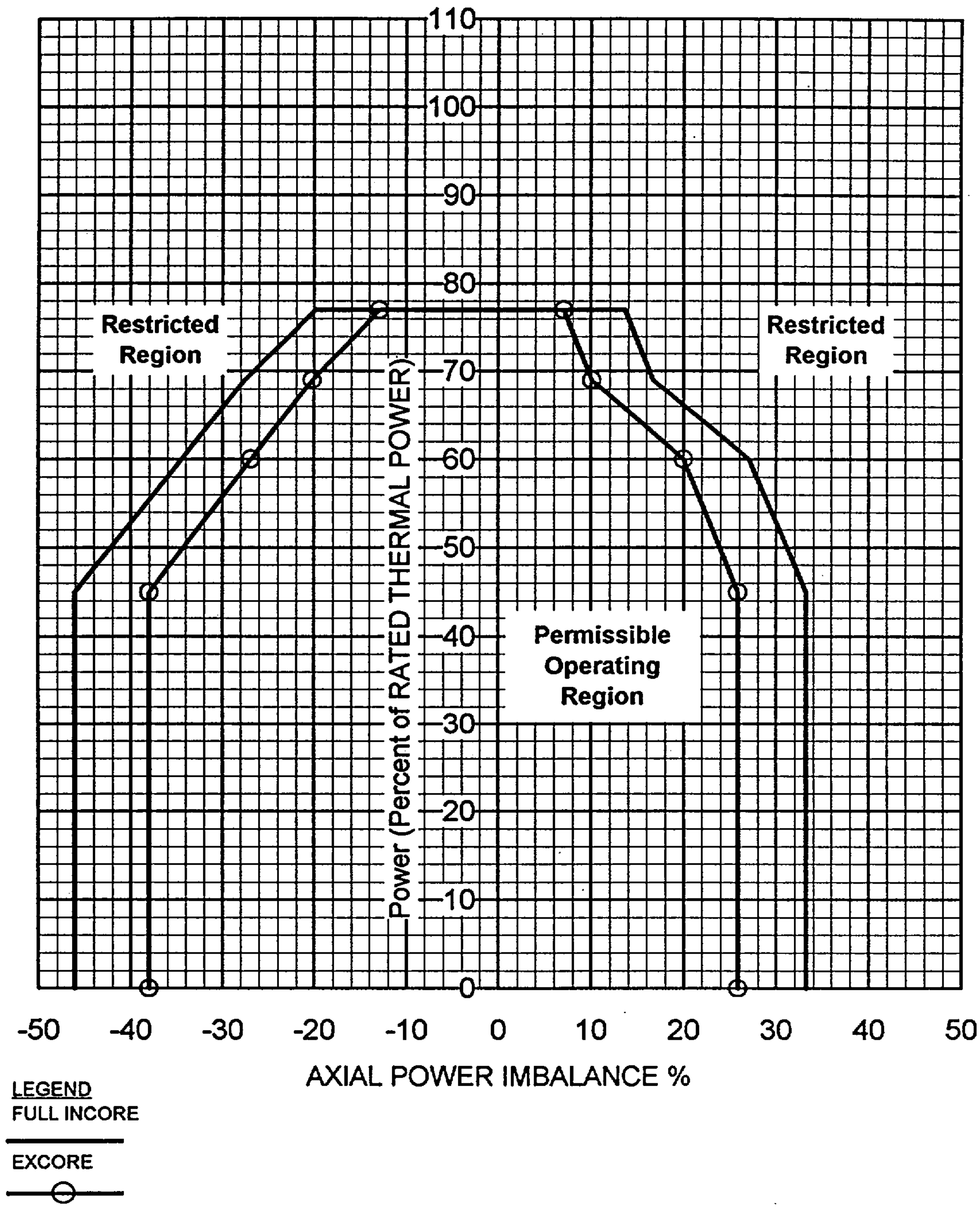


Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-12

Figure AXIAL POWER IMBALANCE Operating Limits
After 626 ±10 EFPD, Three RC Pumps --
Davis-Besse 1, Cycle 13

This Figure is referred to by Technical Specification 3.2.1

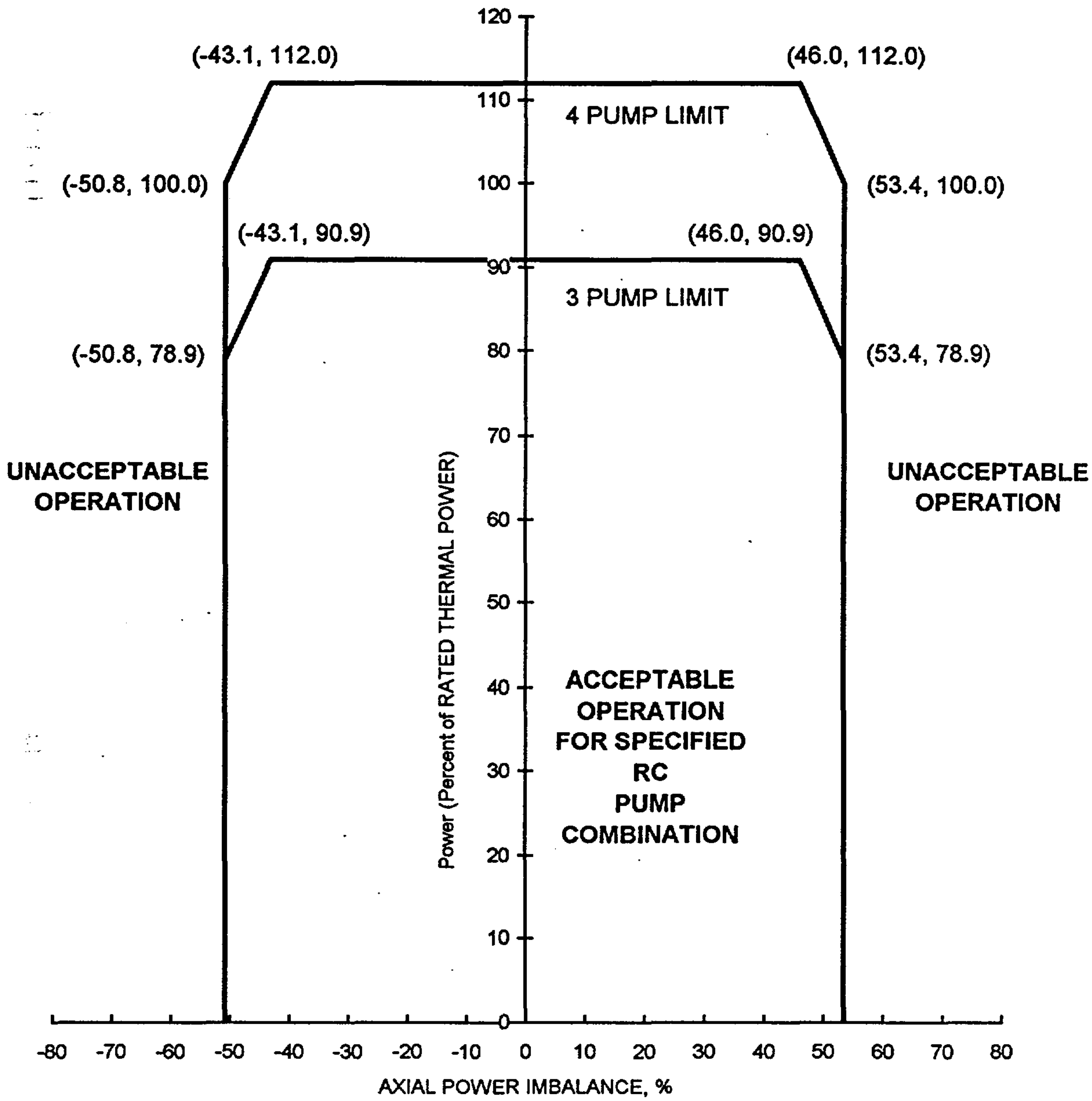


Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-13

Figure AXIAL POWER IMBALANCE Protective Limits

This Figure is referred to by
Technical Specification 2.1.2



Pumps Operating	Reactor Coolant Flow, gpm	Required Measured Flow to Ensure Compliance, gpm
4	380,000	389,500
3	283,860	290,957

Figure 8-14

Figure Flux--ΔFlux/Flow
(or Power/Imbalance/Flow)
Allowable Values

This Figure is referred to by
Technical Specification 2.2.1

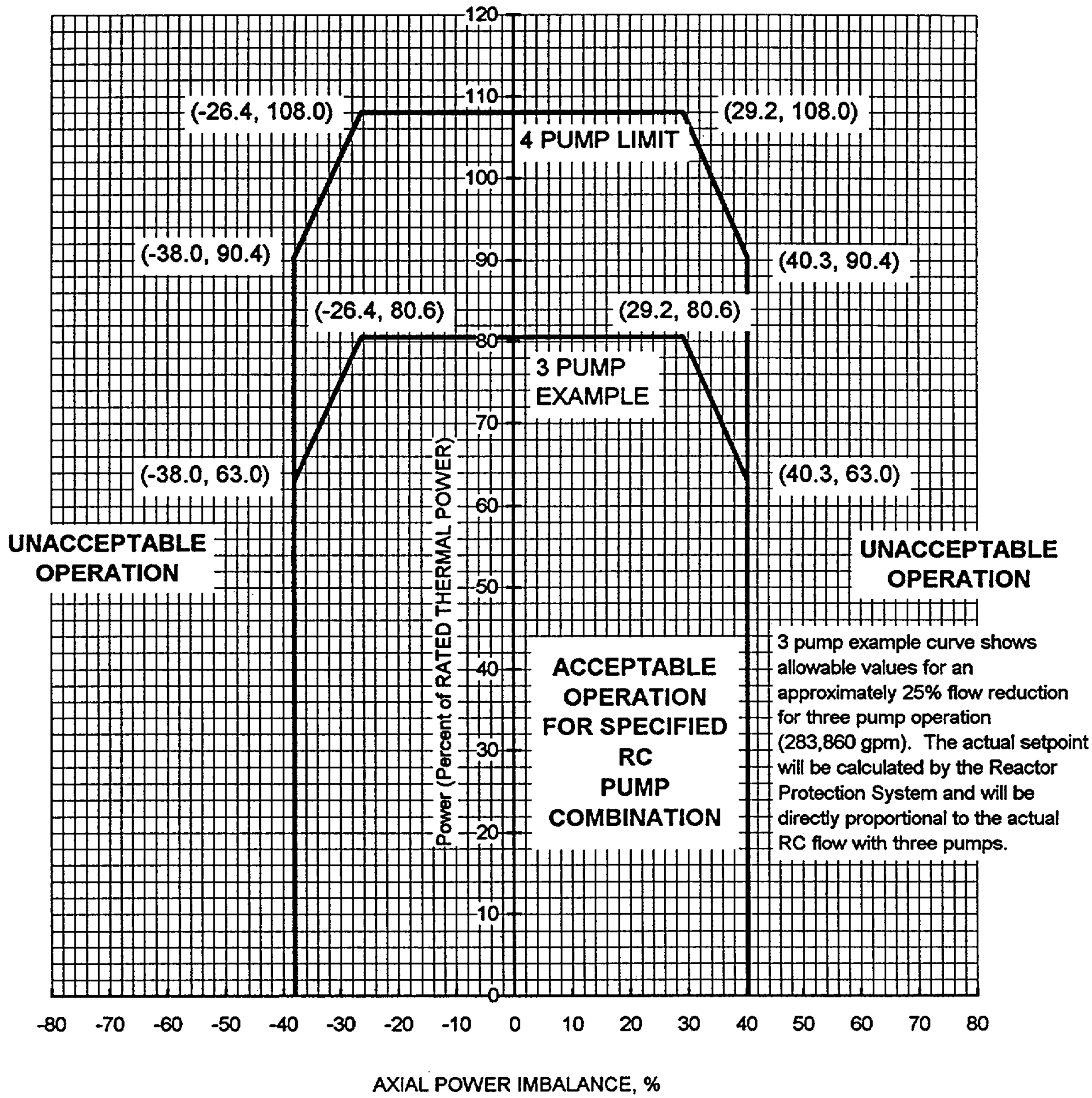


TABLE 8-1

Table QUADRANT POWER TILT Limits

This Table is referred to by Technical Specification 3.2.4

QUADRANT POWER TILT as measured by:	From 0 EFPD to EOC-13			
	Steady-state Limit for THERMAL POWER ≤ 60% (%)	Steady-state Limit for THERMAL POWER > 60% (%)	Transient Limit (%)	Maximum Limit (%)
Symmetrical Incore detector system	7.90	4.00	10.03	20.0

TABLE 8-2

Table Negative Moderator Temperature Coefficient Limit

This Table is referred to by Technical Specification 3.1.1.3c

Negative Moderator Temperature Coefficient Limit (at RATED THERMAL POWER)	-4.00×10^{-4} $\Delta k/k/^{\circ}F$

TABLE 8-3

Table Power to Melt Limits

This Table is referred to by Technical Specification Bases B2.1

	Batch 9G	Batch 10A2	Batch 13	Batch 14	Batch 15
Fuel Assembly Type	Mark-B8A	Mark-B8A	Mark-B10A	Mark-B10M	Mark-B10K
Minimum linear heat rate to melt, kW/ft	20.5	20.5	22.3	22.3 (20.8)[a] (20.8)[b]	22.1 (21.1)[c] (20.7)[d] (20.3)[e]

[a] Limit for 3 wt% Gd rods - Batch 14
[b] Limit for 6 wt% Gd rods - Batch 14
[c] Limit for 2 wt% Gd rods - Batch 15
[d] Limit for 3 wt% Gd rods - Batch 15
[e] Limit for 8 wt% Gd rods - Batch 15

TABLE 8-4

Table Nuclear Heat Flux Hot Channel Factor - F_Q (NAS)

This Table is referred to by Technical Specification 3.2.2

Nuclear Heat Flux Hot Channel Factor - F_Q

F_Q shall be limited by the following relationships:

$F_Q \leq LHR^{ALLOW}(Bu)/[LHR^{AVG} * P]$ (for $P \leq 1.0$)

$LHR^{ALLOW}(Bu)$: See Tables below
LHR^{AVG} = 6.139 kW/ft for Mark-B8A fuel
LHR^{AVG} = 6.426 kW/ft for Mark-B10A fuel
LHR^{AVG} = 6.420 kW/ft for Mark-B10M fuel
LHR^{AVG} = 6.318 kW/ft for Mark-B10K fuel
P = ratio of THERMAL POWER/RATED THERMAL POWER
Bu = Fuel Burnup (MWd/mtU)

Batch 9G (Mark-B8A) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU	60,000 MWd/mtU
1	15.6	15.6	11.8	10.3
2	15.3	15.3	11.8	10.3
3	14.5	14.5	11.8	10.3
4	14.5	14.5	11.8	10.3
5	14.9	14.9	11.8	10.3
6	14.9	14.9	11.8	10.3
7	14.2	14.2	11.4	9.9
8	13.9	13.9	11.2	9.7

Batch 10A2 (Mark-B8A) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU
1	15.6	15.6	11.8
2	15.3	15.3	11.8
3	14.5	14.5	11.8
4	14.5	14.5	11.8
5	14.9	14.9	11.8
6	14.9	14.9	11.8
7	14.2	14.2	11.4
8	13.9	13.9	11.2

TABLE 8-4, continued

Batch 13 (Mark-B10A) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	12.8
2	17.5	16.7	12.8
3	17.0	15.6	12.8
4	16.6	15.3	12.8
5	16.0	15.3	12.8
6	15.3	15.3	12.8
7	14.7	14.7	12.8
8	14.5	14.5	12.8

Batch 14 (Mark-B10M) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	12.8
2	17.5	16.7	12.8
3	17.0	15.6	12.8
4	16.6	15.3	12.8
5	16.0	15.3	12.8
6	15.3	15.3	12.8
7	14.7	14.7	12.8
8	14.5	14.5	12.8

Batch 15 (Mark-B10K) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	35,000 MWd/mtU
1	17.6	16.8
2	17.5	16.7
3	17.0	15.6
4	16.6	15.3
5	16.0	15.3
6	15.3	15.3
7	14.7	14.7
8	14.5	14.5

(a) Linear interpolation for allowable linear heat rate between specified burnup points is valid for these tables.

TABLE 8-5

Table Nuclear Heat Flux Hot Channel Factor - F_Q (FIDMS)

This Table is referred to by Technical Specification 3.2.2

Nuclear Heat Flux Hot Channel Factor - F_Q

F_Q shall be limited by the following relationships:

$F_Q \leq LHR^{ALLOW}(Bu)/[LHR^{AVG} * P]$ (for $P \leq 1.0$)

$LHR^{ALLOW}(Bu)$: See Tables below
$LHR^{AVG} = 6.377$ kW/ft
P = ratio of THERMAL POWER/RATED THERMAL POWER
Bu = Fuel Burnup (MWd/mtU)

Batch 9G (Mark-B8A) LHR^{ALLOW} kW/ft[a]

Core Elevation ft.	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU	60,000 MWd/mtU
0.000	16.2	16.2	12.1	10.6
2.506	15.8	15.8	12.1	10.6
4.264	15.0	15.0	12.1	10.6
6.021	15.4	15.4	12.1	10.6
7.779	15.9	15.9	12.1	10.6
9.536	15.3	15.3	12.1	10.6
12.000	14.3	14.3	11.5	10.0

Batch 10A2 (Mark-B8A) LHR^{ALLOW} kW/ft[a]

Core Elevation ft.	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU
0.000	16.2	16.2	12.1
2.506	15.8	15.8	12.1
4.264	15.0	15.0	12.1
6.021	15.4	15.4	12.1
7.779	15.9	15.9	12.1
9.536	15.3	15.3	12.1
12.000	14.3	14.3	11.5

TABLE 8-5, continued

Batch 13 (Mark-B10A) LHR^{ALLOW} kW/ft[a]

Core Elevation ft.	0 MWd/mtU	35,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	12.8
2.506	17.6	16.8	12.8
4.264	17.1	15.7	12.8
6.021	16.6	15.3	12.8
7.779	16.0	15.8	12.8
9.536	15.3	15.3	12.8
12.000	14.5	14.5	12.8

Batch 14 (Mark-B10M) LHR^{ALLOW} kW/ft[a]

Core Elevation ft.	0 MWd/mtU	35,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	12.8
2.506	17.6	16.8	12.8
4.264	17.1	15.7	12.8
6.021	16.6	15.3	12.8
7.779	16.0	15.8	12.8
9.536	15.3	15.3	12.8
12.000	14.5	14.5	12.8

Batch 15 (Mark-B10K) LHR^{ALLOW} kW/ft[a]

Core Elevation ft.	0 MWd/mtU	35,000 MWd/mtU
0.000	17.6	16.8
2.506	17.6	16.8
4.264	17.1	15.7
6.021	16.6	15.3
7.779	16.0	15.8
9.536	15.3	15.3
12.000	14.5	14.5

(a) Linear interpolation for allowable linear heat rate between specified burnup points is valid for these tables.

TABLE 8-6

Table Nuclear Enthalpy Rise Hot Channel Factor - $F^N_{\Delta H}$

This Table is referred to by Technical Specification 3.2.3

Enthalpy Rise Hot Channel Factor $F^N_{\Delta H}$

$F^N_{\Delta H} \leq$ ARP [1 + 0.3(1 - P/P_m)]

ARP = Allowable Radial Peak, see Figure

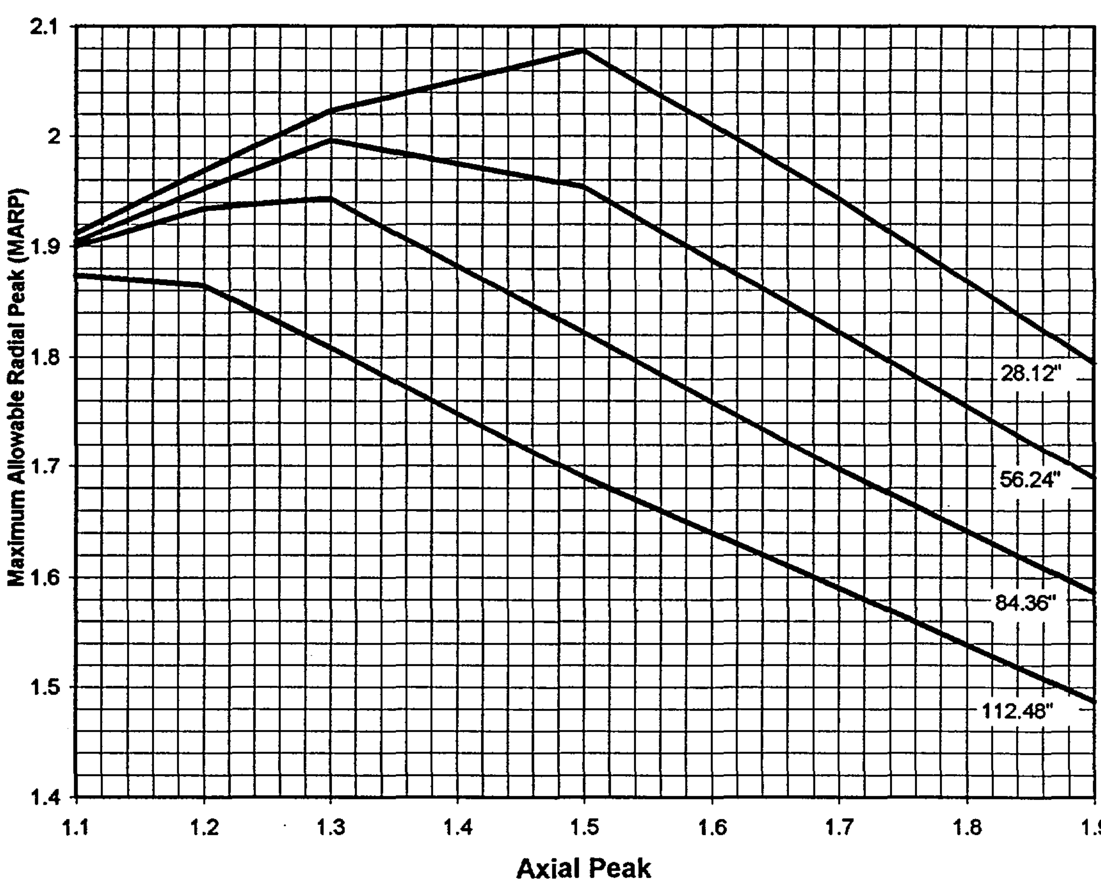
P = THERMAL POWER/RATED THERMAL POWER and $P \leq 1.0$

P_m = 1.0 for 4-RCP operation

P_m = 0.75 for 3-RCP operation

Figure 8-15*

Figure Allowable Radial Peak for $F^N_{\Delta H}$



* This figure is applicable to all fuel in the core. Linear interpolation and extrapolation above 112.48 inches are acceptable. For axial heights <28.12 inches, the value at 28.12 inches will be used.

9. STARTUP PROGRAM - PHYSICS TESTING

The planned startup test program associated with core performance is outlined below. These tests verify that core performance is within the assumptions of the safety analysis and provide information for continued safe operation of the unit.

9.1. Precritical Tests

9.1.1. Control Rod Trip Test

Precritical control rod drop times are recorded for all control rods at hot full-flow conditions before zero power physics testing begins. Acceptance criteria state that the rod drop time from fully withdrawn to 75% inserted shall be less than 1.58 seconds at the conditions above.

It should be noted that safety analysis calculations are based on a rod drop from fully withdrawn to two-thirds inserted. Since the most accurate position indication is obtained from the zone reference switch at the 75% inserted position, this position is used instead of the two-thirds inserted position for data gathering.

9.1.2. RC Flow

Reactor coolant flow with four RC pumps running will be measured at hot standby conditions. The measured flow shall be within allowable limits.

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

Once initial criticality is achieved, equilibrium boron is obtained and the critical boron concentration determined. The critical boron concentration is calculated by correcting for any rod withdrawal required to achieve the all rods out equilibrium boron and for ^{10}B depletion if data are available and applicable. The acceptance criterion placed on critical boron concentration is that the actual boron concentration shall be within $\pm$ 50 ppm boron of the predicted value.

9.2.2. Temperature Reactivity Coefficient

The isothermal HZP temperature coefficient is measured at approximately the all-rods-out configuration. During changes in temperature, reactivity feedback may be compensated by control rod movement. The change in reactivity is then calculated by the summation of reactivity associated with the temperature change. The acceptance criterion for the temperature coefficient is that the measured value shall not differ from the predicted value by more than $\pm$ 0.2 x 10^{-2} %Δk/k/°F.

The moderator temperature coefficient of reactivity is calculated in conjunction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is subtracted to obtain the moderator temperature coefficient (MTC). This value shall be less than +0.9 x 10^{-2} %Δk/k/°F. The MTC is also extrapolated to full power conditions, and is then compared to the appropriate HFP limit.

9.2.3. Control Rod Group/Boron Reactivity Worth

Individual control rod group reactivity worths (groups 5, 6, and 7) are measured at hot zero power conditions using the boron/rod swap method. This technique consists of deborating the reactor coolant system and compensating for the reactivity changes from this deboration by inserting individual control rod groups 7, 6, and 5 in incremental steps. The reactivity changes that occur during these measurements are calculated based on reactimeter data, and incremental rod worths are obtained from the measured reactivity worth versus the change in rod group position. The incremental rod worths of each of the controlling groups are then summed to obtain integral rod group worths. The acceptance criteria for the control rod group worths are as follows:

1. Individual group 5, 6, 7 worth:

$$\left|\frac{\text{predicted value - measured value}}{\text{predicted value}}\right| \text{ x 100\% shall be} \leq 15\%$$

2. Sums of groups 5, 6, and 7:

$$\left|\frac{\text{predicted value} - \text{measured value}}{\text{predicted value}}\right| \times 100\% \text{ shall be } \leq 6\%$$

The boron reactivity worth (differential boron worth) is measured by dividing the total inserted rod worth by the boron change made for the rod worth test. The acceptance criterion for measured differential boron worth is as follows:

$$\left|\frac{\text{predicted value} - \text{measured value}}{\text{predicted value}}\right| \times 100\% \text{ shall be } \leq 15\%$$

The predicted rod worths and differential boron worth are taken from the ATOM.

9.3. Power Escalation Tests

9.3.1. Core Symmetry Test

The purpose of this test is to evaluate the symmetry of the core at low power during the initial power escalation following a refueling. Symmetry evaluation is based on incore quadrant power tilts during escalation to the intermediate power level. The absolute values of the quadrant power tilts should be less than the COLR limit.

9.3.2. Core Power Distribution Verification at Intermediate Power Level (IPL) and ~100% FP

Core power distribution tests are performed at the IPL and approximately 100% full power (FP). Equilibrium xenon is established prior to the ~100% FP test. The test at the IPL (40-80 %FP) is essentially a check of the power distribution in the core to identify any abnormalities before escalating to the ~100% FP plateau. Peaking factor criteria are applied to the IPL core power distribution results to determine if additional tests or analyses are required prior to ~100% FP operation.

The following acceptance criteria are placed on the IPL and ~100% FP tests:

1. The maximum F_Q values shall not exceed the limits specified in the COLR.

2. The maximum $F^N_{\Delta H}$ value shall not exceed the limits specified in the COLR.

3. The measured radial (assembly) peaks for each 1/8 core fresh fuel location shall be within the following limits:

$$\frac{\text{predicted value} - \text{measured value}}{\text{predicted value}} \times 100\% \text{ more positive than } -3.8\%$$

4. The measured total (segment) peaks for each 1/8 core fresh fuel location shall be within the following limits:

$$\frac{\text{predicted value} - \text{measured value}}{\text{predicted value}} \times 100\% \text{ more positive than } -4.8\%$$

The following review criteria also apply to the core power distribution results at the IPL and at ~100% FP:

5. The 1/8 core RMS of the differences between predicted and measured radial (assembly) peaking factors should be less than 0.05.

6. For all 1/8 core locations, the (absolute) difference between predicted and measured radial (assembly) peaking factors should be less than 0.10.

Items 1 and 2 ensure that the initial condition limits are maintained at the IPL and ~100% FP.

Items 3 and 4 are established to determine if measured and predicted power distributions are within allowable tolerances assumed in the reload analysis.

Items 5 and 6 are review criteria, established to determine if measured and predicted power distributions are consistent.

9.3.3. Incore vs. Excore Detector Imbalance Correlation Verification

Imbalances, set up in the core by control rod positioning, are read simultaneously on the incore detectors and excore power range detectors. The excore detector offset versus incore detector offset slope shall be greater than 0.96 and the y-intercept (excore offset) shall be between -2.5% and 2.5%. If either of these criteria are not met, gain amplifiers on the excore detector signal processing equipment are adjusted to provide the required slope and/or intercept.

9.3.4. Hot Full Power All Rods Out Critical Boron Concentration

The hot full power (HFP) all rods out critical boron concentration (AROCBC) is determined at ~100% FP by first recording the RCS boron concentration during equilibrium, steady state conditions. Corrections to the measured RCS boron concentration are made for control rod group insertion and power deficit (if not at 100% FP) using predicted data for CRG worth, power Doppler coefficient, and differential boron worth. A correction for ^{10}B depletion may be made if data are available and applicable. A correction may also be made to account for the observed difference between the measured and predicted AROCBC at zero power. The review criterion placed on the HFP AROCBC is that the measured AROCBC should be within ± 50 ppm boron of the predicted value.

9.4. Procedure for Use if Acceptance/Review Criteria Not Met

If an acceptance criterion ("shall" as opposed to "should") for any test is not met, an evaluation is performed before continued testing at a higher power plateau is allowed. This evaluation is performed by site test personnel with participation by Framatome Technologies technical personnel as required. Further specific actions depend on evaluation results. These actions can include repeating the tests with more detailed test prerequisites and/or steps, added tests to search for anomalies, or design personnel performing detailed analyses of potential safety problems because of parameter deviation. Power is not escalated until evaluation shows that plant safety will not be compromised by such escalation.

If a review criterion ("should" as opposed to "shall") for any test is not met, an evaluation is performed before continued testing at a higher power plateau is recommended. This evaluation is similar to that performed to address failure of an acceptance criterion.

10. REFERENCES

1. Davis-Besse Nuclear Power Station No. 1, Updated Safety Analysis Report, Docket No. 50-346.

2. Davis-Besse Nuclear Power Station Unit 1, Cycle 12 Reload Report, BAW-2320 and Addendum 1, Framatome Cogema Fuels, Lynchburg, Virginia, dated May 1998.

3. TACO3: Fuel Pin Thermal Analysis Computer Code, BAW-10162P-A, Babcock & Wilcox, Lynchburg, Virginia, dated November 1989.

4. Program to Determine In-Reactor Performance of BWFC Fuel - Cladding Creep Collapse, BAW-10084P-A, Rev. 3, Babcock and Wilcox, Lynchburg, Virginia, dated July 1995.

5. Evaluation of Advanced Cladding and Structural Material (M5™) in PWR Reactor Fuel, BAW-10227P-A, Framatome Cogema Fuels, Lynchburg, Virginia, dated February 2000.

6. Extended Burnup Evaluation, BAW-10186P-A, Framatome Cogema Fuels, Lynchburg, Virginia, dated June 1997.

7. GDTACO - Urania-Gadolinia Fuel Pin Thermal Analysis Code, BAW-10184P-A, B&W Fuel Company, Lynchburg, Virginia, dated February 1995.

8. Fuel Rod Gas Pressure Criterion (FRGPC), BAW-10183P-A, B&W Fuel Company, Lynchburg, Virginia, dated July 1995.

9. Framatome Mark-B Spacer Grid Deformation in B&W Designed 177 Fuel Assembly Plants, BAW-2292P, Framatome Cogema Fuels, Lynchburg, Virginia, dated March 1997 (SER dated August 20, 1997).

10. NEMO- Nodal Expansion Method Optimized, BAW-10180-A, Rev. 1, B&W Fuel Company, Lynchburg, Virginia, dated March 1993.

11. Letter, Robert Jones (NRC) to J. H. Taylor (FTI), Subject: Acceptance of Revised Measurement Uncertainty for Control Rod Worth Calculations, dated January 26, 1996.

12. Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, BAW-10179P-A, Rev. 3, Framatome Cogema Fuels, Lynchburg, Virginia, dated October 1999.

13. Statistical Core Design for B&W-Designed 177-FA Plants, BAW-10187P-A, B&W Fuel Company, Lynchburg, Virginia, dated March 1994.

14. BWNT LOCA Evaluation Model for OTSG Plants, BAW-10192PA, Framatome Technologies Inc., Lynchburg, Virginia, dated June 1998.

15. B&W's ECCS Evaluation Model, BAW-10104P, Rev. 5, Babcock & Wilcox, Lynchburg, Virginia, dated April 1986.

Docket Number 50-346
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Enclosure 3

COMMITMENT LIST

THE FOLLOWING LIST IDENTIFIES THOSE ACTIONS COMMITTED TO BY THE DAVIS-BESSE NUCLEAR POWER STATION (DBNPS) IN THIS DOCUMENT. ANY OTHER ACTIONS DISCUSSED IN THE SUBMITTAL REPRESENT INTENDED OR PLANNED ACTIONS BY THE DBNPS. THEY ARE DESCRIBED ONLY FOR INFORMATION AND ARE NOT REGULATORY COMMITMENTS. PLEASE NOTIFY THE MANAGER – REGULATORY AFFAIRS (419-321-8466) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	**DUE DATE**
None	N/A