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Docket Number 50-346

License Number NPF-3

Serial Number 2991

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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 14

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 14 operation (Enclosure 1). 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 14 Reload Report is provided as Enclosure 2.

Should you have any questions or require additional information, please contact Mr. Kevin L. Ostrowski, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

Lew W Myers

MKL

Enclosures

cc: Regional Administrator, NRC Region III
J. B. Hopkins, NRC/NRR Senior Project Manager
C. S. Thomas, NRC Region III, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

A001

Docket Number 50-346
License Number NPF-3
Serial Number 2991
Enclosure 1

Davis-Besse Nuclear Power Station Unit Number 1
Core Operating Limits Report
Cycle 14

(25 pages follow)

FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE UNIT 1

CYCLE 14

CORE OPERATING LIMITS REPORT

Prepared by S. M. Hopper 10/17/03
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Reviewed by Charles A. Alm 10-17-03
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Approved by David B. Kelley 10/20/03
D. B. Kelley

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FIRSTENERGY NUCLEAR OPERATING COMPANY

DAVIS-BESSE UNIT 1

CYCLE 14

CORE OPERATING LIMITS REPORT

1.0 Core Operating Limits

This CORE OPERATING LIMITS REPORT for DB-1 Cycle 14 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 14 is 736.8 EFPDs (based on a reactor thermal rating of 2772 MWt).

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

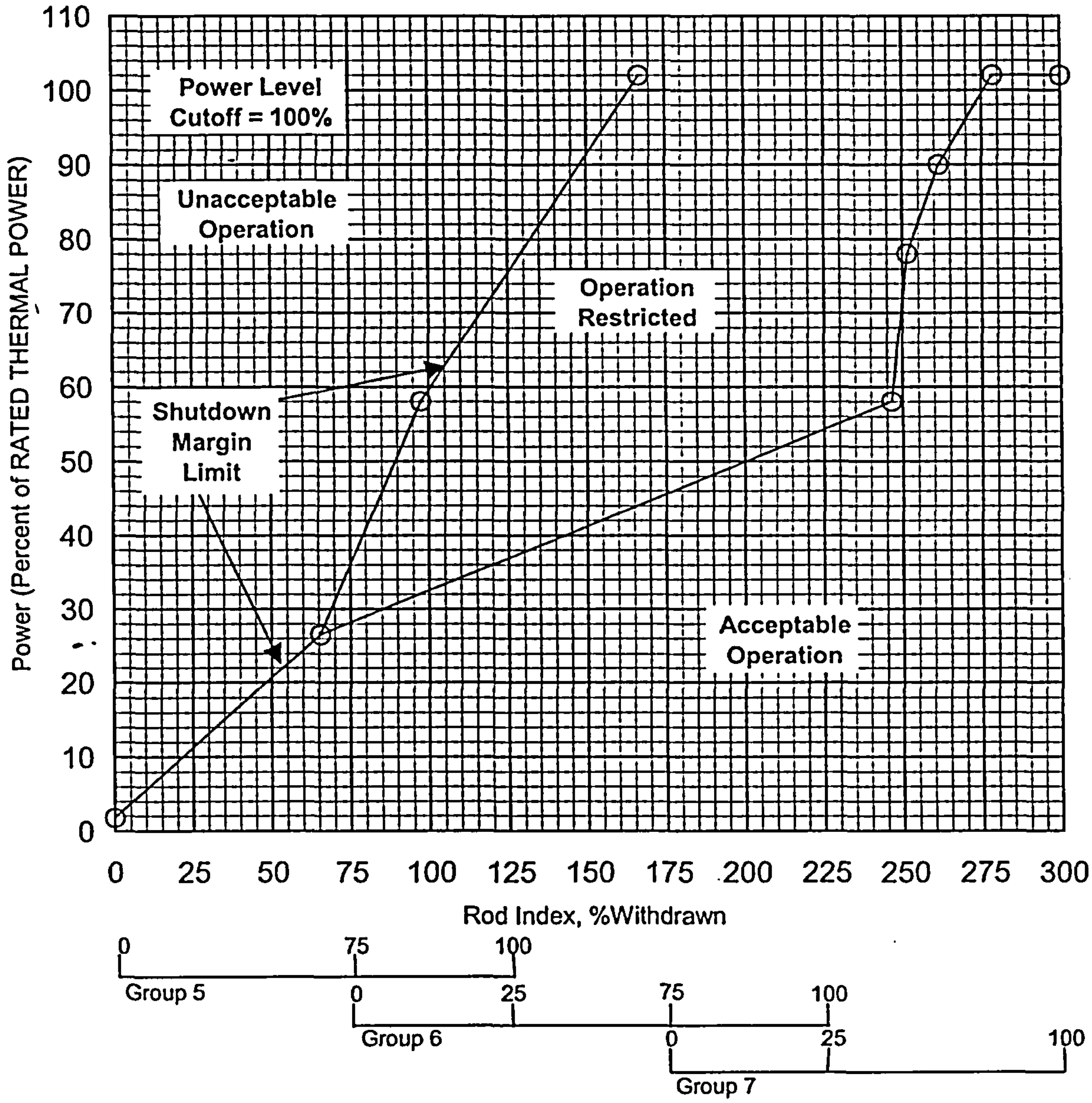
10) Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$

2.0 References

(1) BAW-10179P-A, Rev. 4, "Safety Criteria and Methodology For Acceptable Cycle Reload Analyses.", August, 2001.
(2) BAW-10164P-A, Rev. 4, " RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," November, 2002.

Figure 1a Regulating Group Position Operating Limits
0 to 400±10 EFPD, Four RC Pumps -- 2772 MWt
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

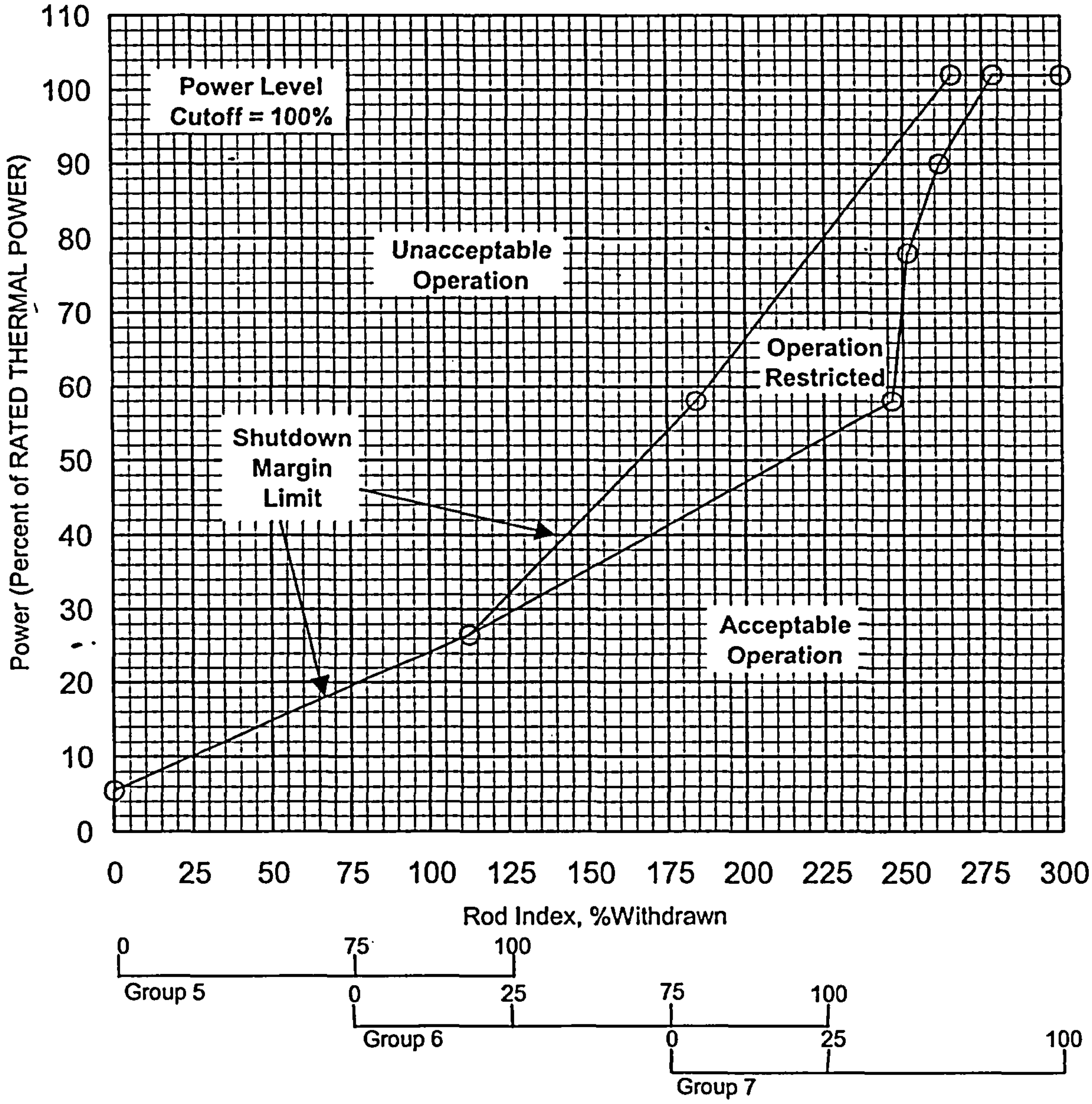


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.

Note 2: Instrument error is accounted for in these Operating Limits.

Figure 1b Regulating Group Position Operating Limits
After 400±10 EFPD, Four RC Pumps -- 2772 MWt
Davis-Besse 1, Cycle 14

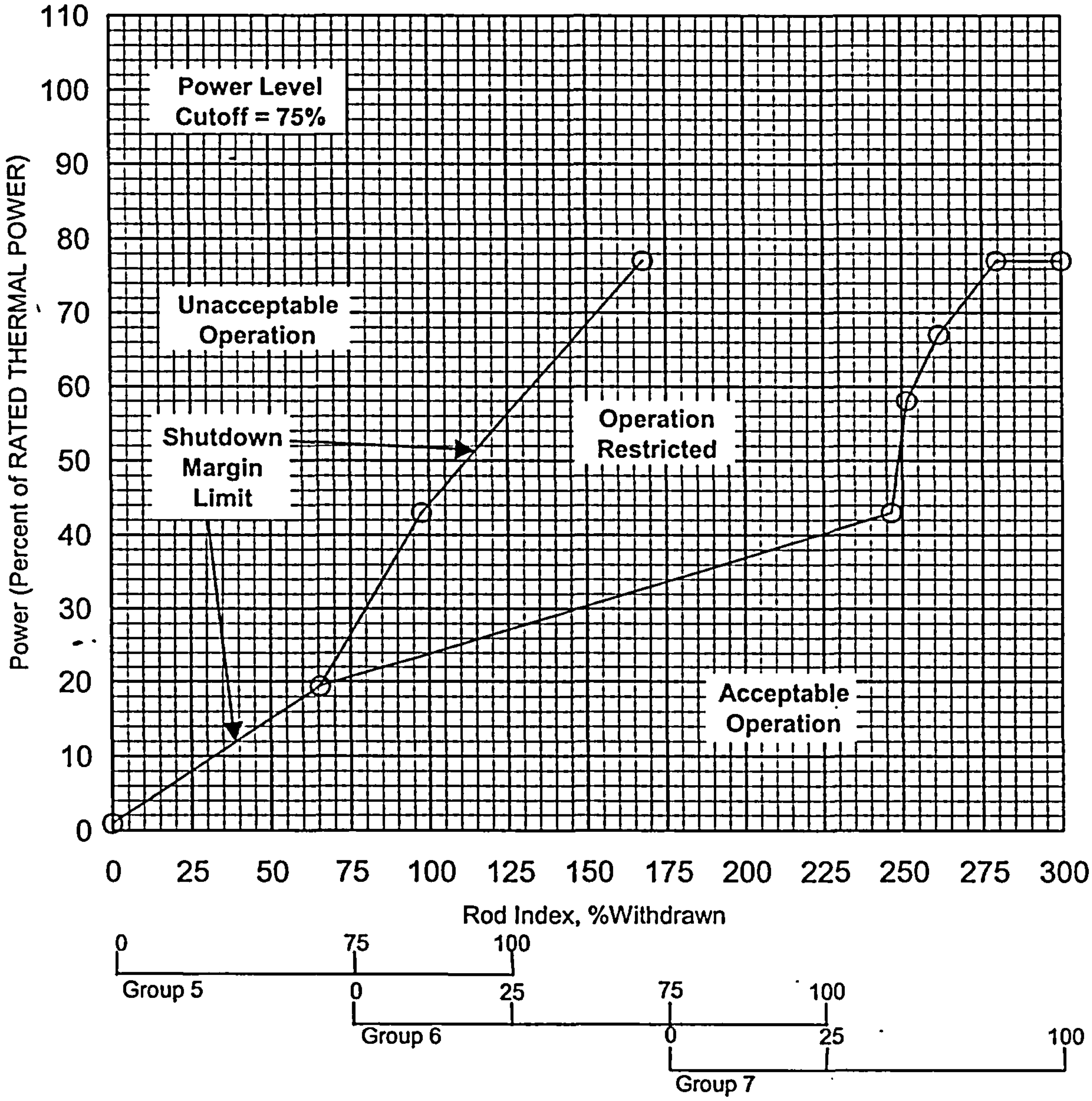
This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



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 400±10 EFPD, Three RC Pumps -- 2772 MWt
Davis-Besse 1, Cycle 14

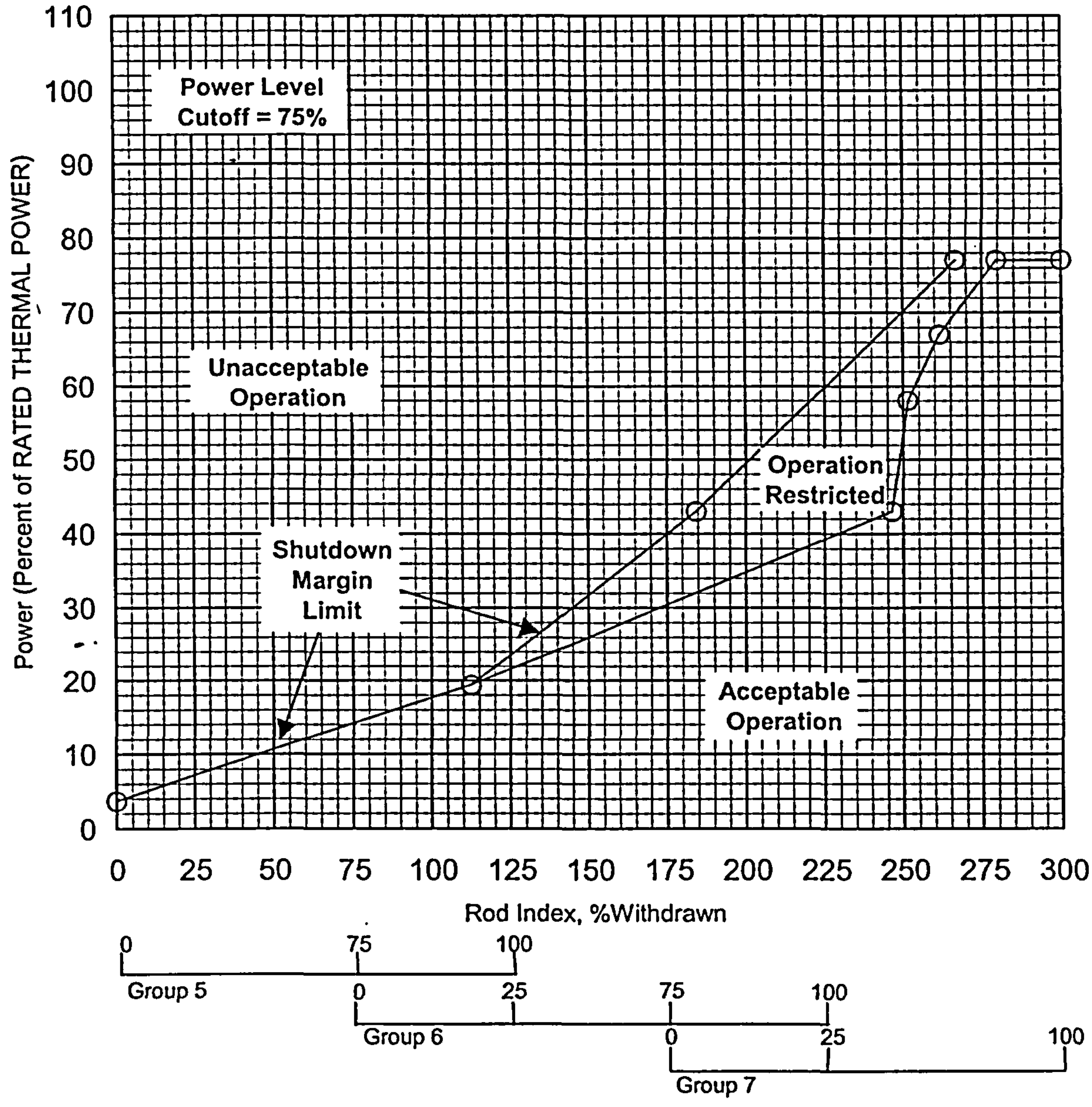
This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8



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 400±10 EFPD, Three RC Pumps -- 2772 MWt
Davis-Besse 1, Cycle 14

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 14

This Figure is referred to by Technical Specification 3.1.3.7

← N

(X)

	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		
(W) H (Y)		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)

$\boxed{X}$ 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

2772 MWt RTP

This Figure is referred to by Technical Specification 3.1.3.9

Before APSR Pull: 0 EFPD to 654 ±10 EFPD,
Three or Four RC pumps operation*

Lower Limit: 0 %WD

Upper Limit: 100 %WD

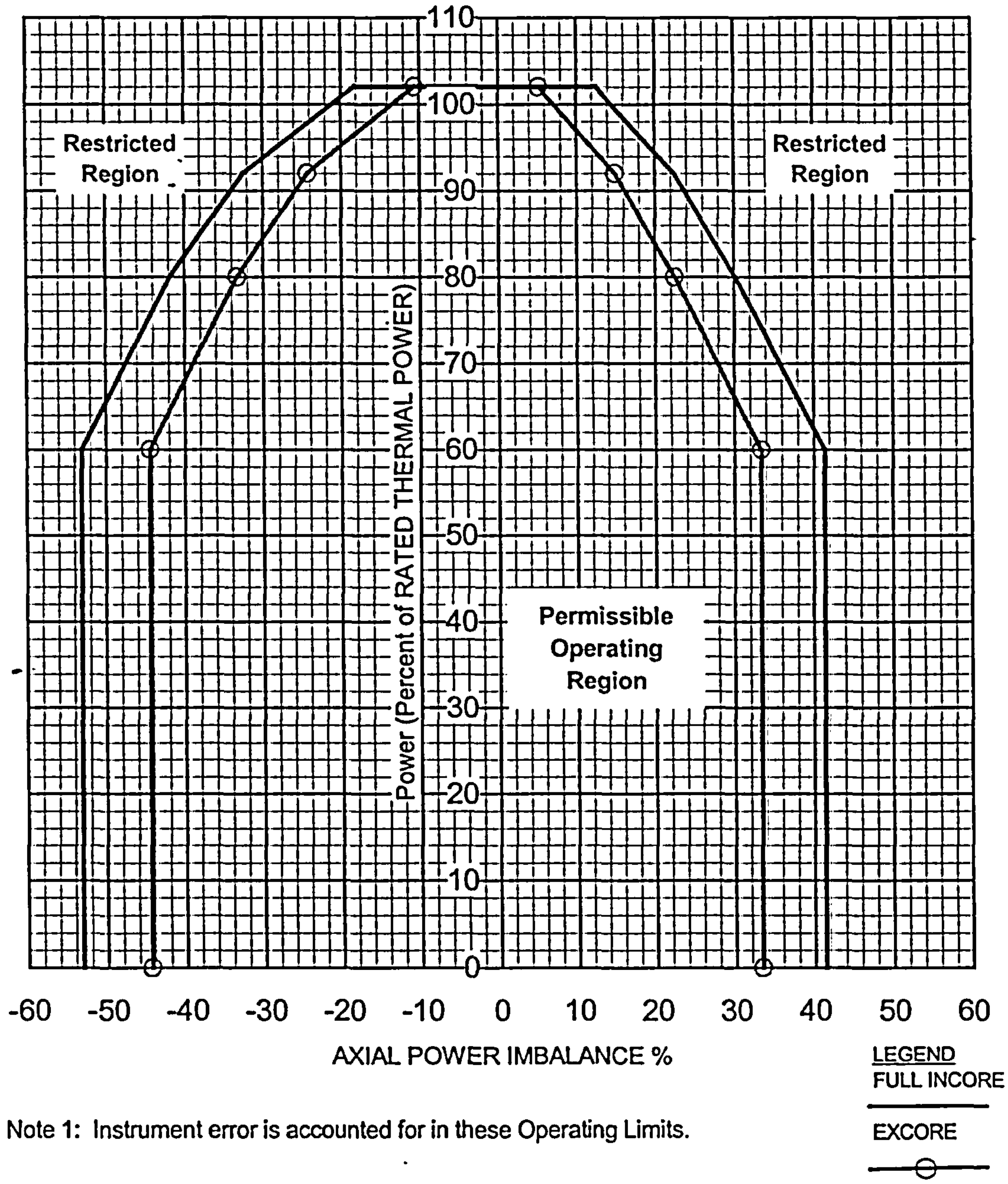
After APSR Pull: 654 ± 10 EFPD to End-of-Cycle
Three or Four RC pumps operation*

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 – 2772 MWt RTP
Davis-Besse 1, Cycle 14

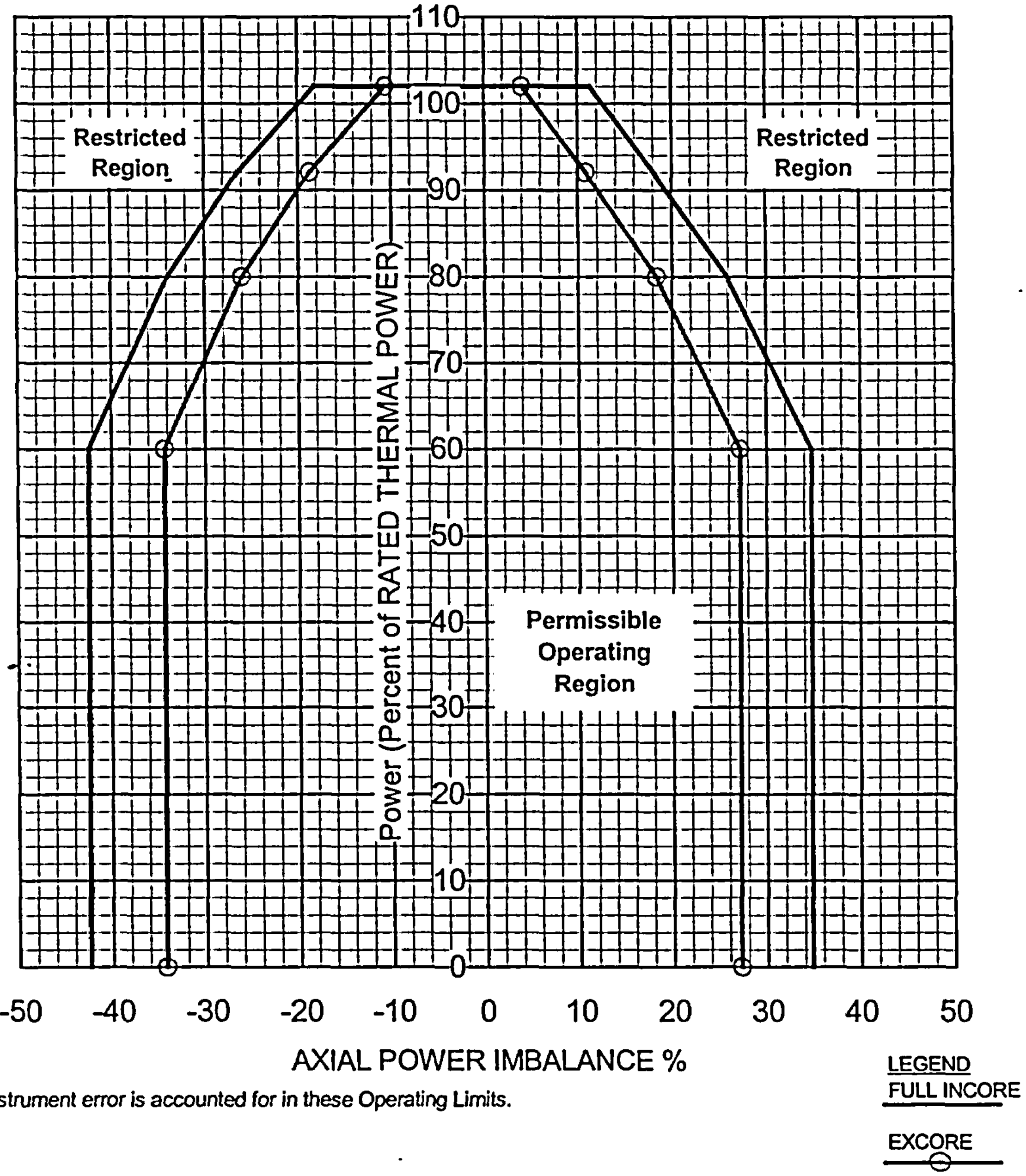
This Figure is referred to by Technical Specification 3.2.1



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

Figure 4b AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 654 ±10 EFPD, Four RC Pumps – 2772 MWt RTP
Davis-Besse 1, Cycle 14

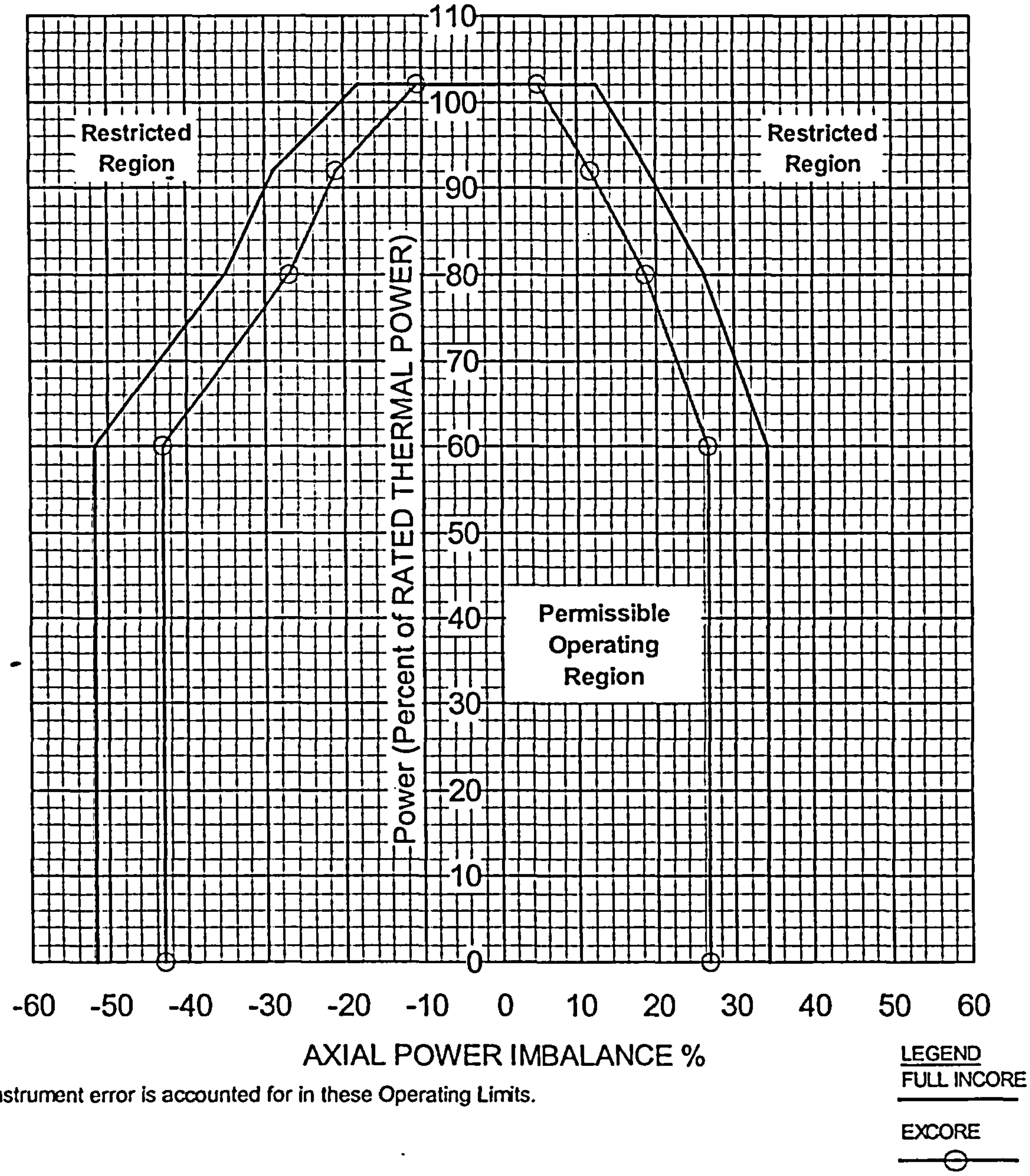
This Figure is referred to by
Technical Specification 3.2.1



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

Figure 4c AXIAL POWER IMBALANCE Operating Limits
After 654 ±10 EFPD, Four RC Pumps – 2772 MWt RTP
Davis-Besse 1, Cycle 14

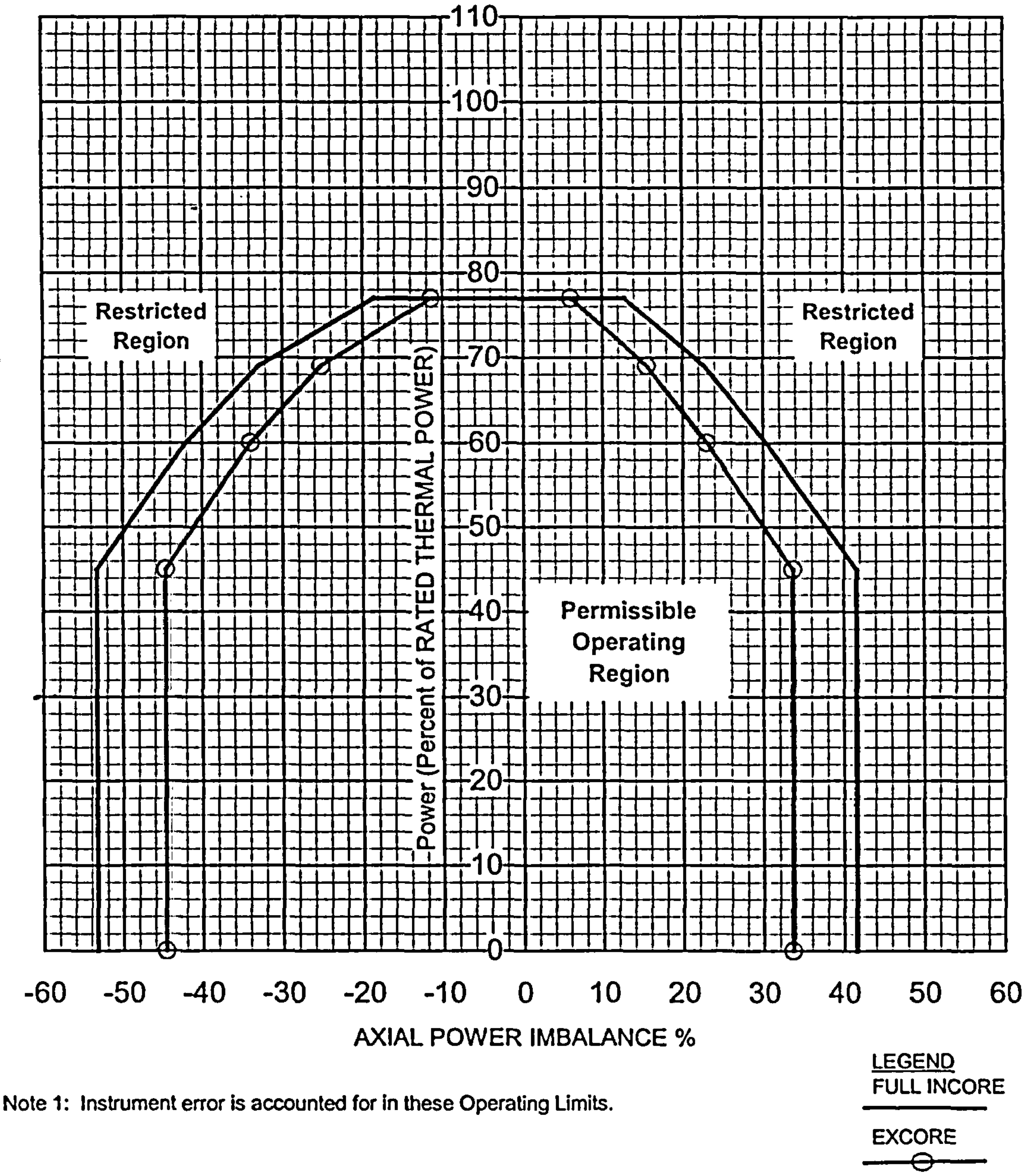
This Figure is referred to by Technical Specification 3.2.1



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 -- 2772 MWt RTP
Davis-Besse 1, Cycle 14

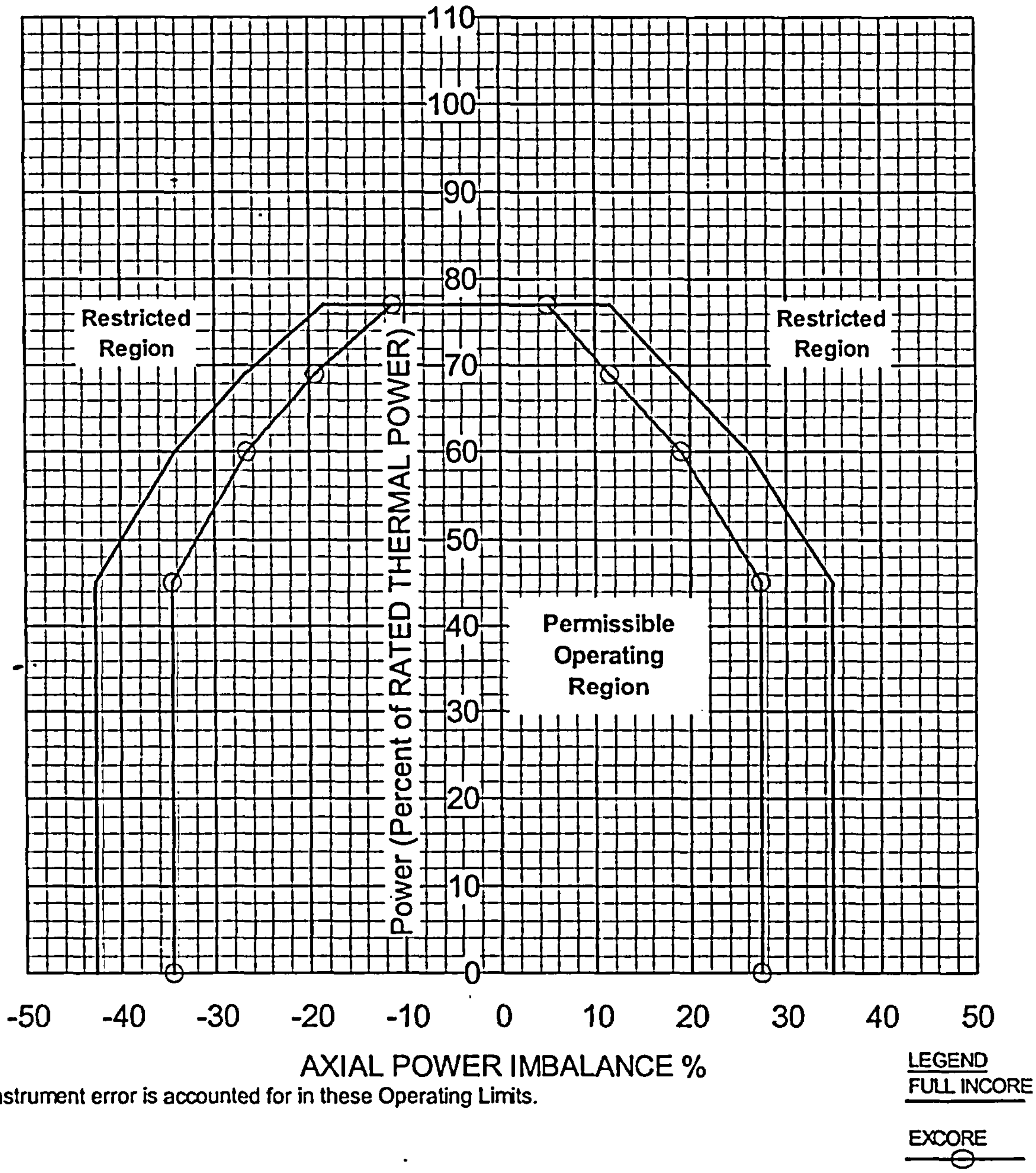
This Figure is referred to by Technical Specification 3.2.1



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

Figure 4e AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 654 ±10 EFPD, Three RC Pumps – 2772 MWt RTP
Davis-Besse 1, Cycle 14

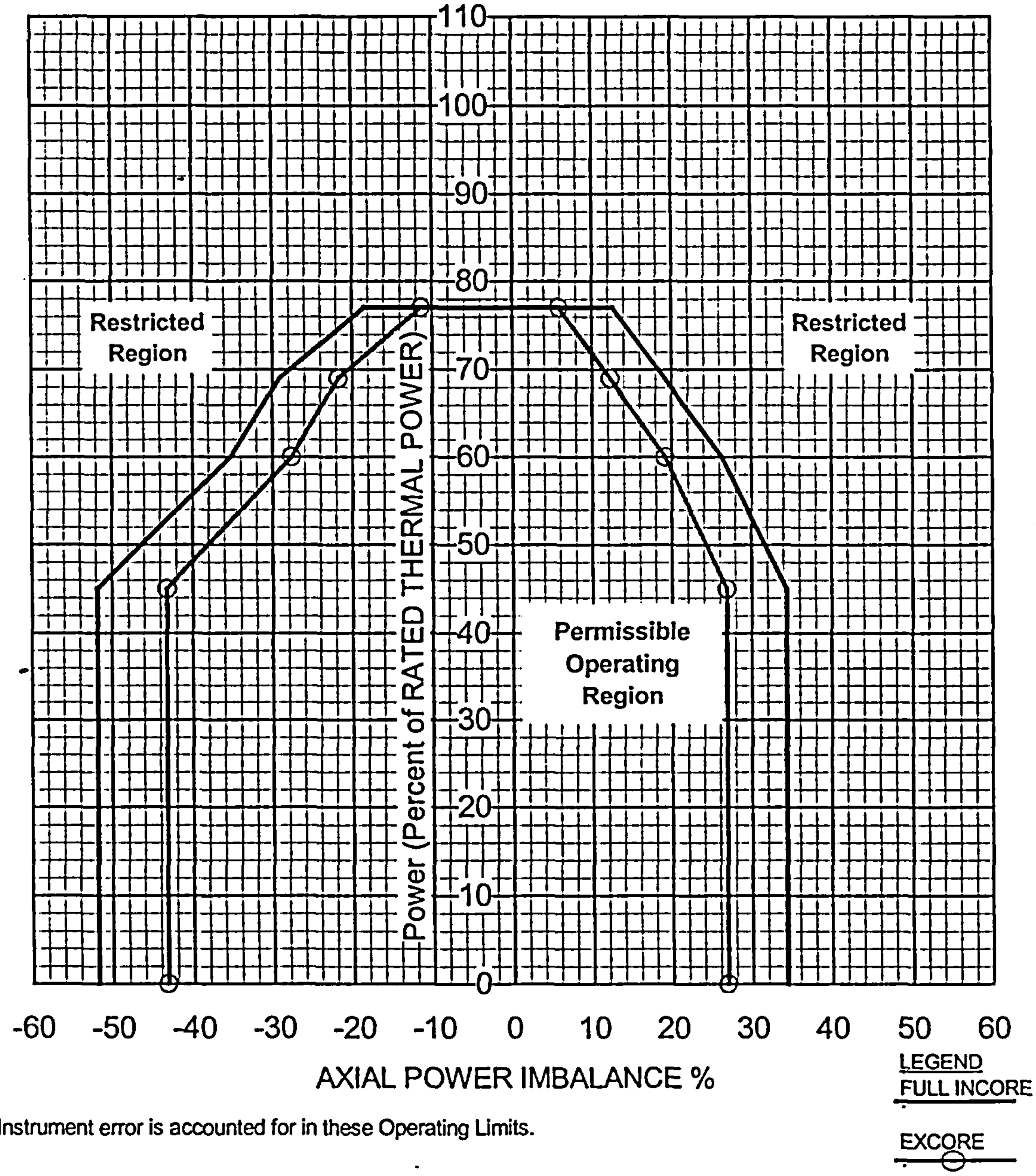
This Figure is referred to by Technical Specification 3.2.1



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

Figure 4f AXIAL POWER IMBALANCE Operating Limits
After 654 ±10 EFPD, Three RC Pumps – 2772 MWt RTP
Davis-Besse 1, Cycle 14

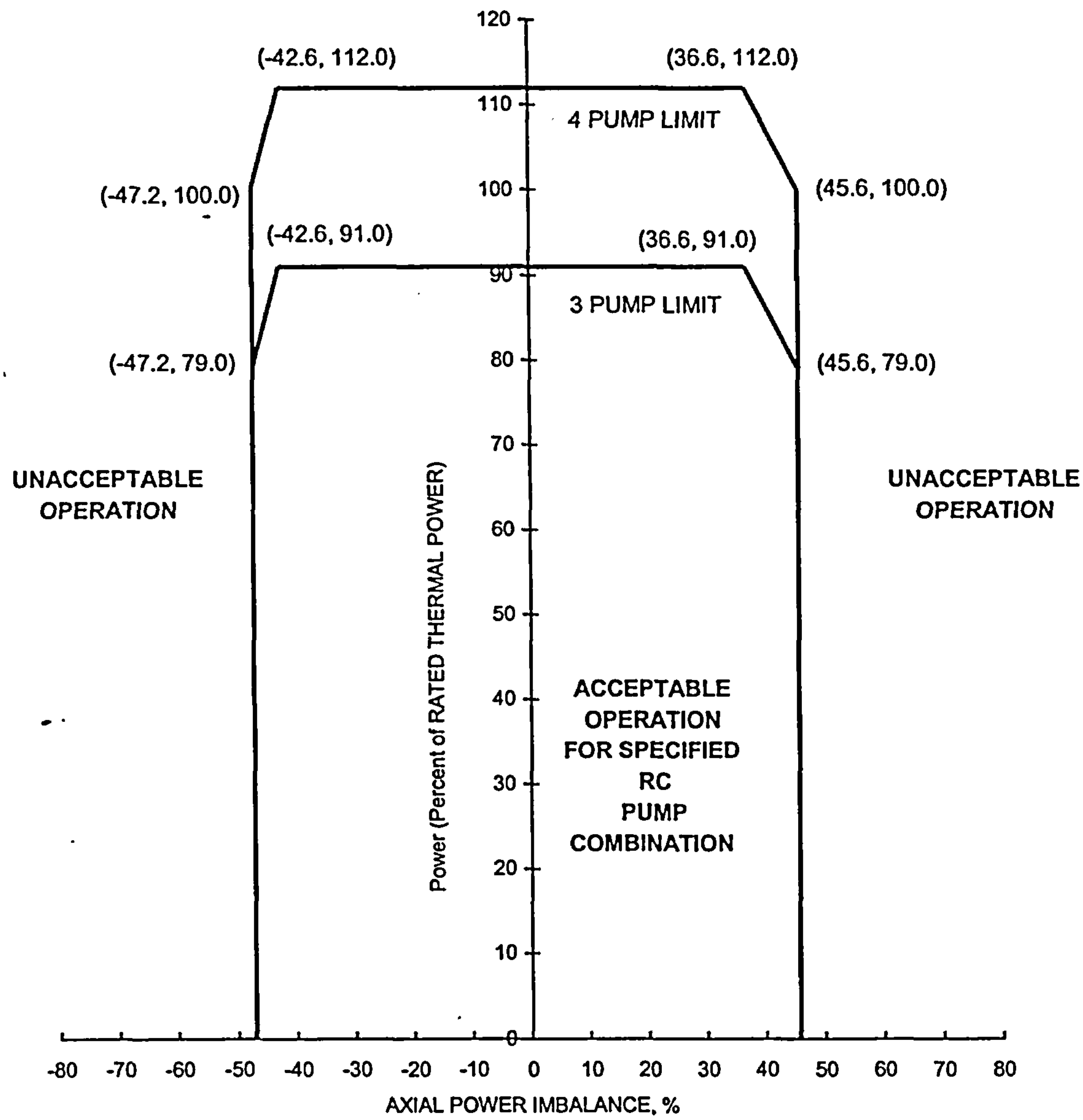
This Figure is referred to by Technical Specification 3.2.1



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

Figure 5 AXIAL POWER IMBALANCE Protective Limits
2772 MWt RTP

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

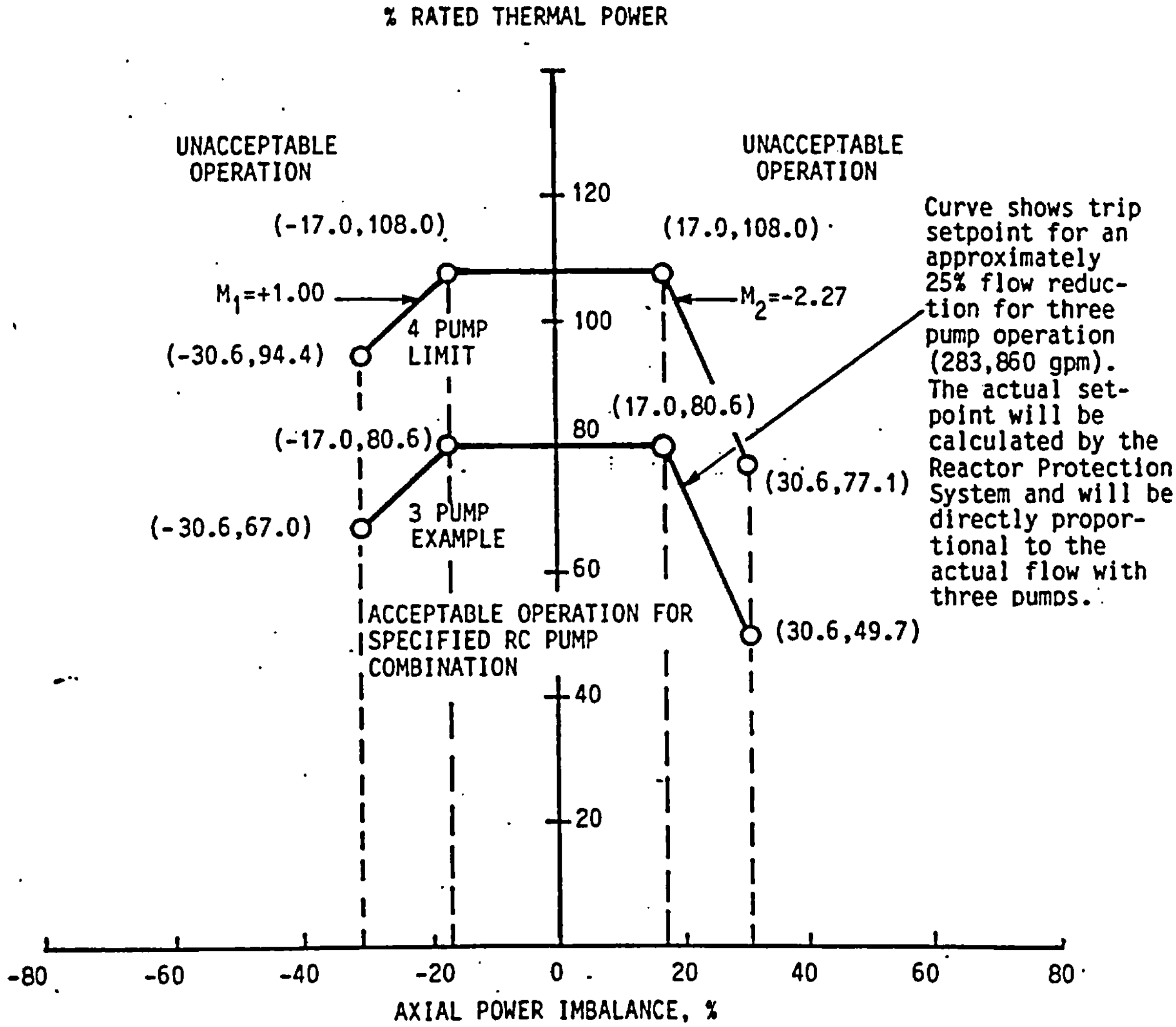


Table 1 QUADRANT POWER TILT Limits

This Table is referred to by Technical Specification 3.2.4

QUADRANT POWER TILT as measured by:	From 0 EFPD to EOC-14			
	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)	-3.83×10^{-4} Δk/k/°F

Table 3 Power To Melt Limits

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

	Batch 9H	Batch 14	Batch 15	Batch 16
Fuel Assembly Type	Mark-B8A	Mark-B10M	Mark-B10K	Mark-B12
Minimum linear heat rate to melt, kW/ft	20.5	22.3 (20.8)[a] (19.9)[b]	22.1 (21.1)[c] (20.7)[d] (19.3)[e]	22.1 (20.3)[f] (19.3)[g]

(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
(f) Limit for 4 wt% Gd rods - Batch 16
(g) Limit for 8 wt% Gd rods - Batch 16

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

2772 MWt RTP

This Table is referred to by Technical Specification 3.2.2

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} = 6.3095 kW/ft for Mark-B8A fuel

LHR^{AVG} = 6.4201 kW/ft for Mark-B10M fuel

LHR^{AVG} = 6.3183 kW/ft for Mark-B10K fuel

LHR^{AVG} = 6.3183 kW/ft for Mark-B12 fuel

P = ratio of THERMAL POWER/RATED THERMAL POWER

Bu = Fuel Burnup (MWd/mtU)

Batch 9H (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	16.1	16.1	12.0	10.2
2	15.8	15.8	12.0	10.2
3	15.0	15.0	12.0	10.2
4	15.0	15.0	12.0	10.2
5	15.4	15.4	12.0	10.2
6	15.4	15.4	12.0	10.2
7	14.6	14.6	12.0	10.2
8	14.3	14.3	12.0	10.2

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

Table 4a, continued

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

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	14.7	14.4	14.1	13.5
2	17.5	16.7	14.7	14.4	14.1	13.5
3	17.0	15.6	14.6	14.4	14.1	13.5
4	16.6	15.3	14.4	14.4	14.1	13.5
5	16.0	15.3	14.2	14.2	14.1	13.5
6	15.3	15.3	13.8	13.7	13.6	13.5
7	14.7	14.7	13.3	13.2	13.1	13.0
8	14.5	14.5	13.1	13.0	12.9	12.8

Batch 16 (Mark-B12) LHR^{ALLOW} kW/ft[(a)]

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	14.7	14.4	14.1	13.5
2	17.5	16.7	14.7	14.4	14.1	13.5
3	17.0	15.6	14.6	14.4	14.1	13.5
4	16.6	15.3	14.4	14.4	14.1	13.5
5	16.0	15.3	14.2	14.2	14.1	13.5
6	15.3	15.3	13.8	13.7	13.6	13.5
7	14.7	14.7	13.3	13.2	13.1	13.0
8	14.5	14.5	13.1	13.0	12.9	12.8

(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)

2772 MWt RTP

This Table is referred
to by Technical Specification 3.2.2

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 the Tables below

LHR^{AVG} = 6.3095 kW/ft for Mark-B8A fuel

LHR^{AVG} = 6.4201 kW/ft for Mark-B10M fuel

LHR^{AVG} = 6.3183 kW/ft for Mark-B10K fuel

LHR^{AVG} = 6.3183 kW/ft for Mark-B12 fuel

P = ratio of THERMAL POWER/RATED THERMAL POWER

Bu = Fuel Burnup (MWd/mtU)

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

Core Elevation (feet)	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU	60,000 MWd/mtU
0.000	16.3	16.3	12.0	10.2
2.506	15.9	15.9	12.0	10.2
4.264	15.1	15.1	12.0	10.2
6.021	15.5	15.5	12.0	10.2
7.779	16.0	16.0	12.0	10.2
9.536	15.4	15.4	12.0	10.2
12.000	14.3	14.3	12.0	10.2

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

Core Elevation (feet)	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

Table 4b, continued

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

Core Elevation (feet)	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	14.7	14.4	14.1	13.5
2.506	17.6	16.8	14.7	14.4	14.1	13.5
4.264	17.1	15.7	14.7	14.4	14.1	13.5
6.021	16.6	15.3	14.4	14.4	14.1	13.5
7.779	16.0	15.8	14.2	14.2	14.1	13.5
9.536	15.3	15.3	13.8	13.7	13.6	13.5
12.000	14.5	14.5	13.1	13.0	12.9	12.8

Batch 16 (Mark-B12) LHR^{ALLOW} kW/ft [a]

Core Elevation (feet)	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	14.7	14.4	14.1	13.5
2.506	17.6	16.8	14.7	14.4	14.1	13.5
4.264	17.1	15.7	14.7	14.4	14.1	13.5
6.021	16.6	15.3	14.4	14.4	14.1	13.5
7.779	16.0	15.8	14.2	14.2	14.1	13.5
9.536	15.3	15.3	13.8	13.7	13.6	13.5
12.000	14.5	14.5	13.1	13.0	12.9	12.8

[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 $F^N_{\Delta H}$

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

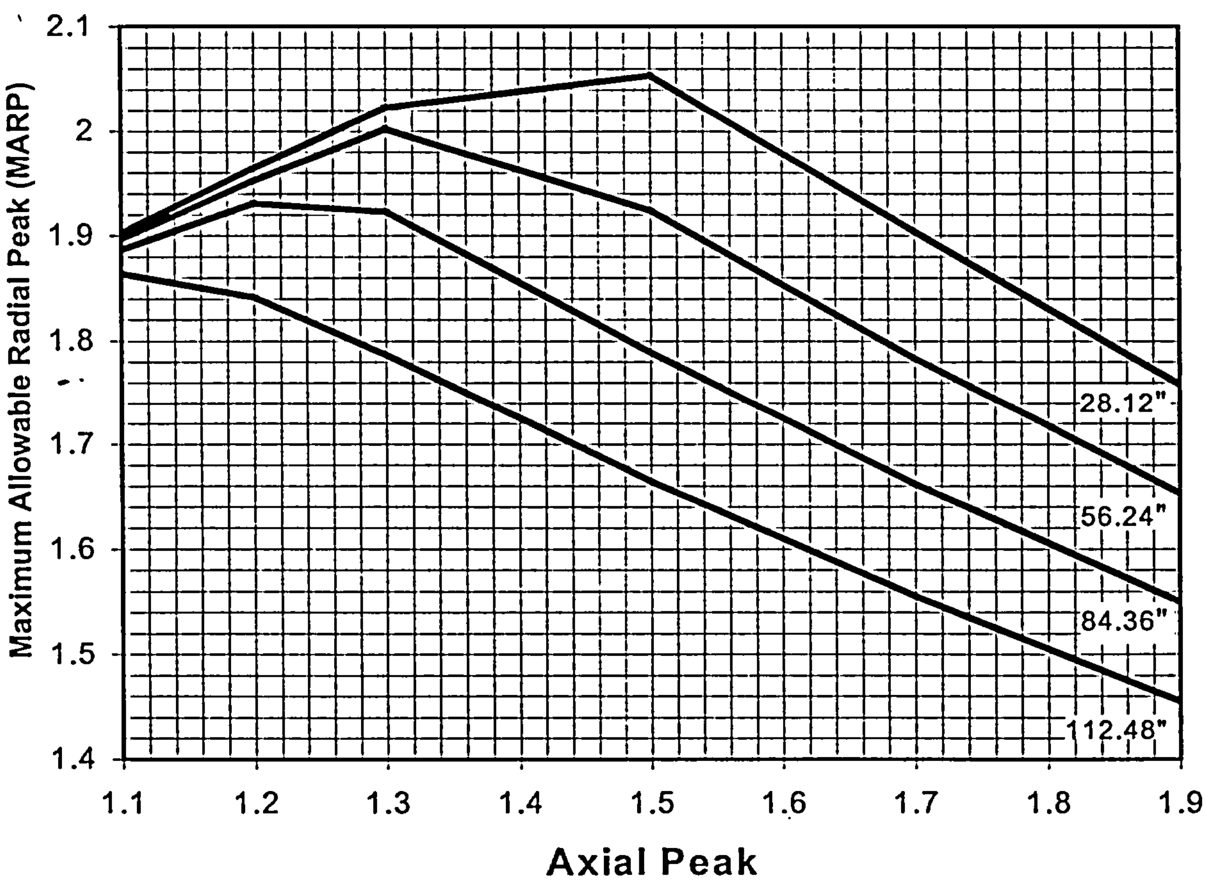
ARP = Allowable Radial Peak, see Figure

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 14

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DAVIS-BESSE NUCLEAR POWER STATION
UNIT 1, CYCLE 14 – RELOAD REPORT

FRAMATOME ANP

BAW-2417, Rev. 2
October 2003
Doc. ID 103-2417-02

DAVIS-BESSE NUCLEAR POWER STATION

UNIT 1, CYCLE 14 -- RELOAD REPORT

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1.0 INTRODUCTION AND SUMMARY

The analyses described in this report justify cycle 14 operation of the Davis-Besse Nuclear Power Station Unit 1 at a rated thermal power (RTP) of 2817 MWt. The analyses also support operation at 2772 MWt with implementation of the RTP uprate to 2817 MWt, as described below, at any time during the cycle. All analytical techniques and design bases utilized in the analyses summarized in this report have been approved by the NRC for the intended application. These methodologies, described in Reference 1, BAW-10179P-A, Rev. 4, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," and subsequently approved documents, represent no departure from methods of evaluation approved for Davis-Besse application.

Subsequent to completion of the reload licensing analyses summarized in this report, a further verification analysis was performed to confirm the continued validity and the conclusions of the reload licensing analyses for an extended cycle 13 refueling outage lasting until March 31, 2004.

The current RTP for Davis-Besse Unit 1 is 2772 MWt. With the implementation of the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ system, the RTP will be increased to 2817 MWt. FENOC has submitted a license amendment request (LAR) to the NRC to justify this increase.

Cycle 14 reactor and fuel parameters related to full power capability are summarized in this report and compared to those for cycle 13. All accidents analyzed in the Davis-Besse Updated Safety Analysis Report (USAR, Reference 2) have been reviewed for cycle 14 operation. In all cases, the initial conditions of the transients in cycle 14 are bounded by the initial conditions of previous analyses.

The cycle 14 design incorporates an end-of-cycle (EOC) HFP extension maneuver that reduces the moderator average temperature (T_{avg}) by a maximum of 12°F (analyzed). 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 14 is the initial implementation of the Mark-B12 fuel assembly. This design is similar to the Mark-B10K fuel assembly introduced in cycle 13 and includes the following design improvements: M5™ guide tubes identical to those used on the four batch 15E M5™ structural assemblies introduced in cycle 13, and a six-leaf holddown spring that optimizes the holddown load in order to reduce the anticipated magnitude of fuel assembly bow and twist relative to the 8-leaf spring used on previous Mark-B assembly designs. The Mark-B12 design also has a reduced fuel rod length to provide more shoulder gap margin. Cycle 14 also incorporates the MONOBLOC™ instrument tube, which is a single-piece instrument tube with a variable inner diameter for incore instrumentation guidance.

The cycle 14 core includes twelve reconstituted fuel assemblies that contain a total of twenty-five stainless steel replacement rods. The reactor vessel closure head is also being replaced in cycle 14. The effects of the stainless steel rods, RV head replacement, potential loose rods, and EOC operational maneuver have been considered in all applicable analyses including the mechanical, nuclear, thermal-hydraulic and power distribution analyses as well as the LOCA and non-LOCA safety analyses. The effects of the corner cell grid damage for the seventeen assemblies for D-B cycle 14 have been considered in the fuel assembly and fuel rod mechanical analyses. The corner cell grid damage is characterized as having a negligible effect on the other D-B cycle 14 reload licensing analyses including the mechanical, nuclear, thermal-hydraulic and power distribution analyses as well as the LOCA and non-LOCA safety analyses. All of the applicable analyses were performed with NRC approved methodologies. The results from the applicable analyses show that all design criteria are met and that there is no significant adverse impact on any USAR design function.

The Technical Specifications have been reviewed and verified to require no changes for cycle 14 operation at 2772 MWt RTP. The applicable Technical Specification changes necessary to operate at 2817 MWt RTP will be included in the NRC approval of the LAR. 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 14 can be operated safely at 2772 MWt, as well as at a licensed core power level of 2817 MWt after NRC approval of the LAR. The cycle 14 EFPDs quoted in this report are based on 2817 MWt RTP unless otherwise specifically stated.

2.0 OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Davis-Besse Unit 1 is cycle 13 (Reference 3), which achieved criticality on May 17, 2000. Power escalation began on May 18, 2000 and full power was reached on May 21, 2000.

During cycle 13 operation, no operating anomalies have occurred that would adversely affect fuel performance during cycle 14. Cycle 14 has a design length of 725 effective full power days (EFPD) based on cycle 13 operation of 620 ± 15 EFPD with an extended EOC Tavg reduction from approximately 566 EFPD to EOC. The actual cycle 13 length was 630.6 EFPD. The cycle 14 design includes an APSR pull, EOC Tavg reduction, CRG 7 withdrawal to 97%WD, and power coastdown.

3.0 GENERAL DESCRIPTION

The cycle 14 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 and 76 batch 16 fuel assemblies are clad in M5™ cladding and the remaining 29 assemblies in the cycle 14 core are clad in Zircaloy-4. In batch 16, one thousand two hundred forty-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 14 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 14. Batch 16 is the third 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 14 batches are listed in Table 3-1. All batch 16 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 9G assembly, 2 batch 10A assemblies, 11 batch 13A assemblies, 15 batch 13B assemblies, 4 batch 14A assemblies, 8 batch 14C assemblies, and 36 batch 14D assemblies will be discharged at the end of cycle 13. The remaining batch 14A and batch 14C FAs, along with batch 14B, 15A, 15B, 15C, 15D, and 15E FAs will be shuffled to their cycle 14 locations. The 4 batch 15E assemblies, residing in cycle 14 locations H03, C08, H13, and O08 contain M5™ guide tubes and two M5™ intermediate spacer grids, and have been pre-characterized. Four of the batch 15D fuel assemblies, in cycle 14 locations G07, G09, K07, and K09 have also been pre-characterized. All batch 14A, 14B, and 14C FAs are on the core periphery. Four of the batch 14A assemblies, 2 of the batch 14B assemblies, 3 of the batch 14C assemblies, 2 of the batch 15B assemblies, and 1 of the batch 15D assemblies have been reconstituted with 1, 2, 3, or 4 stainless steel replacement rods, as noted in Figure 3-1. One batch 9H assembly, discharged at the end of cycle 10 as batch 9A, will be reinserted in cycle 14 as the center FA.

The 76 Mark-B12 assemblies in the feed batch consist of 4 batch 16A, 20 batch 16B, 8 batch 16C, 20 batch 16D, and 24 batch 16E assemblies. The feed batch will be loaded in a symmetric checkerboard pattern throughout the core. The cycle 14 shuffle scheme is a very low leakage (VLL) core loading. The VLL reload fuel shuffle scheme for cycle 14 will have a negligible effect on nuclear instrumentation response for all aspects of reactor startup and subsequent power operation. The cycle 14 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 14 and its initial base enrichment.

Cycle 14 will be operated in a feed-and-bleed mode. Fifty-three full-length Ag-In-Cd control rod assemblies, 1248 Gd rods in the feed batch, and soluble boron control the core reactivity. There are no burnable poison rod assemblies (BPRAs) in cycle 14. 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 previously 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 13 and 14 are the same. Figure 3-3 shows the distribution of the Gd rods. The number of Gd rods per fuel assembly and initial Gd_2O_3 concentrations is also shown in Figure 3-3.

Table 3-1. Fuel Assembly Composition Data for Davis-Besse Cycle 14

Fuel Batch Number	Number of FAs	wt% ^{235}U Std./Gd Rod	wt% Gd_2O_3	Number of Gd Rods	Nominal Loading, KgU
9H	1	3.38	--	--	468.25
14A2	12	4.47/3.80*	3.0	4	468.80
14B	8	4.47/3.80*	3.0	8	468.48
14C2	8	4.47/3.13*	6.0	8	467.87
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**	3.0	8	487.18
		4.88/2.93**	8.0	8	
15E	4	4.88/4.15**	2.0	8	489.12
16A	4	4.59/3.67**	4.0	8	486.98
		4.59/2.75**	8.0	8	
16B	20	4.59/3.67**	4.0	12	486.55
		4.59/2.75**	8.0	8	
16C	8	4.94/3.95**	4.0	8	488.70
16D	20	4.94/3.95**	4.0	12	488.26
16E	24	4.94/3.95**	4.0	12	486.55
		4.94/2.96**	8.0	8	

* 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 14 Core Loading Diagram

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						14A2 M04	14B B05	14A2 G08	14B M14	14A2 M12					
B				14C2 R09	15B P10	16D F	16D F	16D F	16D F	16D F	15B P06	14C2 R07			
C			15D F07	16C F	16E F	16E F	15D L05	15E P08	15D L11	16E F	16E F	16C F	15D G10		
D		14C2 K15	16C F	15C B09	15A N03	15D G12	16E F	15D M08	16E F	15D G04	15A N13	15C K14	16C F	14C2 K01	
E		15B L14	16E F	15A C12	15C P09	16B F	15D E13	16B F	15D E03	16B F	15C K02	15A C04	16E F	15B L02	
F	14A2 D11	16D F	16E F	15D N07	16B F	15D O10	16B F	15D G06	16B F	15D L03	16B F	15D N09	16E F	16D F	14A2 D05
G	14B B11	16D F	15D E10	16E F	15D O05	16B F	15D L13	16A F	15D O06	16B F	15D O11	16E F	15D E06	16D F	14B E14
H	14A2 H07	16D F	15E H14	15D H11	16B F	15D L07	16A F	9H H09 10	16A F	15D F09	16B F	15D H05	15E H02	16D F	14A2 H09
K	14B M02	16D F	15D M10	16E F	15D C05	16B F	15D C10	16A F	15D F03	16B F	15D C11	16E F	15D M06	16D F	14B P05
L	14A2 N11	16D F	16E F	15D D07	16B F	15D F13	16B F	15D K10	16B F	15D C06	16B F	15D D09	16E F	16D F	14A2 N05
M		15B F14	16E F	15A O12	15C G14	16B F	15D M13	16B F	15D M03	16B F	15C B07	15A O04	16E F	15B F02	
N		14C2 G15	16C F	15C G02	15A D03	15D K12	16E F	15D E08	16E F	15D K04	15A D13	15C P07	16C F	14C2 G01	
O			15D K06	16C F	16E F	16E F	15D F05	15E B08	15D F11	16E F	16E F	16C F	15D L09		
P				14C2 A09	15B B10	16D F	16D F	16D F	16D F	16D F	15B B06	14C2 A07			
R						14A2 E04	14B E02	14A2 K08	14B P11	14A2 E12					

Key

XXX YY Z	XXX - Batch ID YY - Previous cycle location Z - Previous cycle if reinsert

Note: "F" denotes fresh fuel assembly

FAs in locations A06, B04, B12, G01, and P12 each have 1 stainless steel rod.
FAs in locations H01, H15, and L15 each have 2 stainless steel rods.
FAs in locations O03 and B11 each have 3 stainless steel rods.
FAs in locations K15 and P05 each have 4 stainless steel rods.

Figure 3-2. Davis-Besse Cycle 14 Enrichment and BOC Burnup Distribution

	8	9	10	11	12	13	14	15
H	3.38 33,662	4.59 0	4.88 26,868	4.59 0	4.88 27,558	4.88 24,923	4.94 0	4.47 40,899
K	4.59 0	4.88 27,638	4.59 0	4.88 25,717	4.94 0	4.88 27,302	4.94 0	4.47 37,683
L	4.88 26,868	4.59 0	4.88 27,572	4.59 0	4.88 27,005	4.94 0	4.94 0	4.47 42,524
M	4.59 0	4.88 25,630	4.59 0	4.88 23,963	4.88 21,574	4.94 0	4.88 23,619	
N	4.88 27,558	4.94 0	4.88 26,948	4.88 21,505	4.88 23,987	4.94 0	4.47 39,130	
O	4.88 24,923	4.88 27,252	4.94 0	4.94 0	4.94 0	4.88 26,825		
P	4.94 0	4.94 0	4.94 0	4.88 23,585	4.47 39,094			
R	4.47 40,899	4.47 37,697	4.47 42,492					

x.xx	Initial Base Enrichment (not weighted for Gd)
xx,xxx	Burnup, MWd/mtU off 620 EFPD (nominal) cycle 13

Figure 3-3. Davis-Besse Cycle 14 Gadolinia Concentrations in Fresh Assemblies

	8	9	10	11	12	13	14	15
H		8×4.0 8×8.0		12×4.0 8×8.0			12×4.0	
K	8×4.0 8×8.0		12×4.0 8×8.0		12×4.0 8×8.0		12×4.0	
L		12×4.0 8×8.0		12×4.0 8×8.0		12×4.0 8×8.0	12×4.0	
M	12×4.0 8×8.0		12×4.0 8×8.0			12×4.0 8×8.0		
N		12×4.0 8×8.0				8×4.0		
O			12×4.0 8×8.0	12×4.0 8×8.0	8×4.0			
P	12×4.0	12×4.0	12×4.0					
R								

Key

NxZ.Z	Number of Gd Rods @ wt% Gd_2O_3

4.0 FUEL SYSTEM DESIGN

Cycle 14 is the initial implementation of the Mark-B12 fuel assembly. This design is similar to the Mark-B10K fuel assembly introduced in cycle 13 and includes the following design improvements: M5™ guide tubes identical to those used on the four batch 15E M5™ structural assemblies introduced in cycle 13, and a six-leaf holddown spring that optimizes the holddown load in order to reduce the anticipated magnitude of fuel assembly bow and twist relative to the 8-leaf spring used on previous Mark-B assembly designs. The Mark-B12 design also has a reduced fuel rod length to provide more shoulder gap margin. These features improve the burnup capability of the fuel assembly and reduce the axial loads acting on the guide tubes during operation. Cycle 14 also incorporates the MONOBLOC™ instrument tube (IT), which is a single-piece instrument tube with a variable inner diameter for incore instrumentation guidance. The fuel system design-based analyses were performed with NRC approved methodologies and, where applicable show that all design criteria are met.

4.1 Fuel Assembly Mechanical Design

Table 4-1 lists the types of fuel assemblies and pertinent fuel parameters for Davis-Besse cycle 14. Batch 16 fuel, the Mark-B12 design, incorporates design modifications first implemented on the batch 15 Mark-B10K design including the use of M5™ advanced, low corrosion cladding and the introduction of the Trapper™ debris resistant lower end fitting.

The B12 fuel rod design is very similar to the batch 15 design with the exception of the rod length, which was reduced by 0.3 inches to increase the end-of-life shoulder gap margin. The shorter rod reduced fuel rod internal void volume, but did not impact the fuel rod nominal stack length, which remains equal to the batch 15 fuel rod nominal stack length.

Mark-B12 fuel assemblies, like the Mark-B10M and Mark-B10K fuel assemblies, contain Gd fuel 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 16 uranium and Gd rods are pressurized to the same pressure used in the batch 14 and batch 15 fuel rods.

Cycle 14 will also contain the four M5™ structural assemblies (batch 15E) first implemented in cycle 13. The assemblies were inspected during the end-of-cycle 13 refueling outage and all criteria, including the end-of-life shoulder gap margin, were within design limits. Eight gray APSRAs and 53 Ag-In-Cd CRAs will be used in cycle 14. All of the CRAs are of the extended life design (ELCRA). No BPRAs will be used in cycle 14.

4.2 Fuel Rod Design

The fuel rod design and mechanical evaluation are discussed in this section.

4.2.1 Cladding Collapse

The computer code TACO3 (Reference 4) is used to provide conservative values of cladding temperature and pin pressure (with no calculated fission gas release) to the computer code CROV (Reference 5), which determines whether or not cladding collapse is predicted during the cycle.

B8A Fuel Rods (Batch 9H)

The most limiting power history for batch 9H was determined. This history was enveloped by the power history used in a previous B8A fuel rod TACO3 analysis. Therefore, the results of the previous analysis bound cycle 14 operation. No creep collapse is predicted to occur through a burnup of at least 60 GWd/mtU, which exceeds the cycle 14 in-core life of these fuel rods.

B9A Fuel Rods (Batch 14)

The most limiting power history for batch 14 was determined. The cycle 14 power history for the B9A fuel rods was shown to be enveloped by the power history used in a previous B9A fuel rod creep collapse analysis. Other analysis inputs such as rod prepressure and plenum volume are conservative when applied to the cycle 14 B9A rods. Results of the previous analysis show that no creep collapse is predicted to occur through a burnup of 60 GWd/mtU, which exceeds the cycle 14 in-core life of these fuel rods. This result also applies to the batch 14 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 previous B10K fuel rod 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 $U0_2$ and Gd rods. The batch 15 rods will not reach burnups in this range during cycle 14; therefore, all batch 15 rods are acceptable for resistance to creep collapse.

B12 Fuel Rods (Batch 16)

The most limiting power history for batch 16 was determined. This power history is enveloped by the power history used in the B12 fuel rod creep collapse analysis. The fuel rod cladding creep collapse analysis for the batch 16 fuel rods showed that these rods have creep collapse lifetimes that exceed 65 GWd/mtU. The analysis applies to both the $U0_2$ and Gd rods. The batch 16 rods will not reach burnups in this range during cycle 14; therefore, all batch 16 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™ 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™ differs from that for Zircaloy-4 cladding, as described in the following discussion.

Batches 9 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 exceed 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 since differences in the required fuel rod weld strength for Gd fuel rods were conservatively taken into account.

Batches 15 and 16 ($M5^{TM}$)

The methodology that governs the stress analysis of the batch 15 and 16 $M5^{TM}$ fuel rods is described in FRA-ANP's advanced cladding topical report (Reference 6). 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 minimum 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 and B12 fuel rod stress analysis is 1.3%. The margins for the corresponding Gd fuel rods are the same as those for the UO_2 rod due to the similarity of their designs.

4.2.3 Cladding Strain

The fuel design criteria of Reference 7 specify a limit of 1.0% transient circumferential strain of the cladding. Cladding transient strain linear heat rate (LHR) limits were generated for each of the fuel rod types in cycle 14 (B8A, B9A, B9A Gd, B10K, B10K Gd, B12, and B12 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 batch 9H, Table 4-3 lists limits for the B9A UO_2 rods of batch 14, Table 4-4 lists limits for the B10K and B12 UO_2 rods of batches 15 and 16, Table 4-5 lists limits for the B9A Gd rods of batch 14, and Table 4-6 lists limits for the B10K and B12 Gd rods of batches 15 and 16.

4.2.4 Cladding Fatigue

Per Reference 7, the predicted total fatigue utilization factor must be less than or equal to 0.90 for the life of each fuel rod. The table below shows the maximum incore time for each batch at EOC-14 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 14.

Rod design (batch)	Maximum incore time	Fatigue limit	Fatigue factor at limit
B8A (batch 9H)	4.5 years	5.79 years	0.90
B9A (batch 14)	5.4 years	10 years	0.574
B10K (batch 15)	3.7 years	10 years	<0.1
B12 (batch 16)	2.0 years	10 years	<0.1

4.2.5 Cladding Oxide

Cladding waterside oxide thickness for FRA-ANP fuel is limited to 100 microns. FRA-ANP's COROS02 model (Reference 7) generates oxide predictions at each input time step and each input axial node. Per the licensed methodology, the high burnup fuel rod for each batch in cycle 14 was evaluated for oxide

using cycle 14 power histories to the maximum pin burnup for each batch. Acceptable oxide predictions were obtained for each batch.

4.3 Thermal Design

All fuel assemblies in the cycle 14 core are thermally similar. The fresh batch 16 fuel assemblies are of the Mark-B12 design with axial blankets of slightly enriched ^{235}U fuel pellets and Gd fuel rods. The Mark-B12 fuel assembly incorporates design modifications, including M5™ guide tubes, six-leaf holddown spring, MONOBLOC™ IT, and a shortened fuel rod length, as described in Sections 4.0 and 4.1. Fuel performance for the Mark-B8A, Mark-B10M, Mark-B10K, and Mark-B12 UO_2 fuel was evaluated with TACO3 (Reference 4). Nominal undensified input parameters used in the analyses are presented in Table 4-1. The GDTACO code (Reference 8) 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 presence of twenty-five stainless steel rods in nine batch 14 and three batch 15 fuel assemblies was considered in the thermal evaluation by appropriately increasing the affected local peaking. The results of the thermal design evaluation of the cycle 14 core are summarized in Table 4-1. The nominal linear heat rate (LHR) for each batch is shown at rated thermal powers of 2772 MWt and 2817 MWt. Cycle 14 core protection limits were based on LHR to centerline fuel melt (CFM) limits determined by the TACO3 and GDTACO codes.

The maximum fuel rod burnup at EOC-14 is predicted to be less than 58,000 MWd/mtU (batch 9H). The fuel rod internal pressures were evaluated with TACO3 and GDTACO for the highest burnup of each fuel rod type. The predicted internal pressures for all cycle 14 fuel were justified with the approved fuel rod gas pressure criterion methodology described in Reference 9.

4.4 Spacer Grid Deformation and Loose Rods

The structural integrity of the fuel assembly spacer grids under faulted conditions was evaluated based on leak-before-break (LBB) methodology described in Reference 10 and the latest NRC approved fuel assembly faulted methodology as described in Reference 11. LBB and FA faulted methodologies are consistent with FRA-ANP LOCA evaluations for B&W-designed raised and lowered loop plants, which includes Davis-Besse. Application of the LBB and FA faulted methodologies confirmed that the requirement to maintain a coolable geometry is met for all faulted loading cases and for all fuel assemblies in the core, including the effects of the Davis-Besse permanent seal plate installation.

Fuel assembly inspections during the cycle 13 refueling outage (13RFO) resulted in seventeen fuel assemblies having one or more spacer grids with corner damage. The seventeen assemblies are scheduled to go into the cycle 14 core. The grids were evaluated for the degree of damage present and the ability to continue to operate with leaker-free fuel rod assemblies. Nine of the fuel assemblies were repaired by replacing fuel rods with stainless steel rods (SSRs). One of the fuel assemblies was

completely re-caged, while the remaining seven assemblies were designated as acceptable for use (one had a grid corner slightly modified to ensure the grid envelope was retained).

The ability of the damaged spacer grids to meet the requirements of faulted analyses is also not compromised. Testing was done on grids with damaged corners. The conclusions made from the test results are that as long as the damage to the grid is limited to one corner cell and the adjacent cell, the damaged grids continue to have sufficient margin with respect to faulted condition criteria. Grids with similar and less damage will behave in a similar fashion.

In addition, for the seven assemblies recommended as acceptable for use, the rods are captured within the array of the spacer grid. For the assemblies in which capture of the rods is not certain, stainless steel rods were inserted. In either case, the damage is deemed minimal enough such that there is not significant local flow blockage. Also the intensity of the turbulence will not be significantly diminished by the material distortion. Any effect on the thermal-hydraulic performance is deemed negligible. The fuel assemblies continue to meet all design criteria.

Also during the cycle 13 refueling outage, sixteen fuel assemblies that operated in core peripheral locations during cycle 13 that are to be placed in core peripheral locations for cycle 14 were inspected for loose rods. The inspection was performed on the outer row of fuel rods that were adjacent to the baffle wall for cycle 13. The results of the loose rod inspections led to the reconstitution of 4 fuel assemblies with a total of 6 stainless steel rods (SSRs). Also, 8 additional fuel assemblies with spacer grid damage were reconstituted with a total of 12 additional SSRs (three of these assemblies also had a total of 6 additional SSRs inserted for preventative reasons since they will be located in peripheral core locations deemed highly susceptible to spacer grid fretting). One of the fuel assemblies with loose rods also had a damaged grid corner, which required an additional SSR. The result of the 13RFO fuel inspections and preventative measures led to the reconstitution of 12 fuel assemblies with a total of 25 SSRs.

Based on the 13RFO inspection results and the reconstitution with SSRs, there are no known loose rods in the cycle 14 core. Evaluations have shown that if any fuel rods become loose during cycle 14 operation their location is expected to be in low power fuel rods located on the core periphery facing the core baffle and will have an insignificant impact on design bases and the safe operation of the core.

4.5 Material Compatibility

The compatibility of all possible fuel-cladding-coolant-assembly interactions for all cycle 14 fuel assemblies, including those containing M5™ material as approved in the advanced cladding topical report (Reference 6), was demonstrated to be acceptable.

4.6 Operating Experience

FRA-ANP 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 multiple cycles at six of the seven operating B&W units, and Gd rods have operated successfully in eight cycles at three B&W units.

M5™ cladding material has been implemented on a batch basis in five B&W units and one Westinghouse unit. The TRAPPER™ debris resistant lower end fitting has operated at four Westinghouse units and is currently being used in two B&W units. All the operating B&W plants have implemented the MONOBLOC™ instrument tube.

Table 4-1. Fuel Design Parameters

Batch	9H	14A2	14B	14C2	15A	15B	15C
Fuel assembly type	Mark-B8A	Mark-B10M	Mark-B10M	Mark-B10M	Mark-B10K	Mark-B10K	Mark-B10K
No. of assemblies	1	12	8	8	8	8	8
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377	0.380	0.380	0.380
Undensified active fuel length, in.							
UO_2 rods	143.2	140.733	140.733	140.733	143.0	143.0	143.0
Gd rods	—	140.634	140.634	140.634	143.0	143.0	143.0
Pellet OD, in.	0.3686	0.3700	0.3700	0.3700	0.3735	0.3735	0.3735
Fuel pellet initial density, %TD mean	95.0	96.0	96.0	96.0	96.0	96.0	96.0
Average burnup BOC, MWd/mtU[(a)]	33,662	42,622	38,340	39,762	22,190	24,252	24,625
Cladding collapse burnup, MWd/mtU[(b)]	>60,000	>60,000	>60,000	>60,000	>60,000	>60,000	>60,000
Maximum pin burnup, MWd/mtU[(a)]	57,472	54,740	57,101	52,499	53,428	44,273	54,299
Initial fuel enrichment, wt% ^{235}U							
UO_2 rods	3.38	4.47	4.47	4.47	4.88	4.88	4.88
Gd rods	—	3.80	3.80	3.13	4.15	4.15	2.93
Nom. LHR at 2772 MWt, kW/ft[(c)]	6.14	6.25	6.25	6.25	6.15	6.15	6.15
Nom. LHR at 2817 MWt, kW/ft[(c)]	6.24	6.35	6.35	6.35	6.25	6.25	6.25
Minimum linear heat rate to melt, kW/ft							
UO_2 rods	20.5	22.3	22.3	22.3	22.1	22.1	22.1
2 wt% Gd rods	—	—	—	—	21.1	21.1	—
3 wt% Gd rods	—	20.8	20.8	—	—	—	—
4 wt% Gd rods	—	—	—	—	—	—	—
6 wt% Gd rods	—	—	—	19.9	—	—	—
8 wt% Gd rods	—	—	—	—	—	—	19.3

(a) Includes allowance for cycle 13 shutdown length flexibility.
(b) Calculated using method from Reference 5.
(c) LHR calculations include a 0.973 energy deposition factor.

Table 4-1. Fuel Design Parameters (Cont'd)

Batch	15D	15E	16A	16B	16C	16D	16E
Fuel assembly type	Mark-B10K	Mark-B10K	Mark-B12	Mark-B12	Mark-B12	Mark-B12	Mark-B12
No. of assemblies	44	4	4	20	8	20	24
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.380	0.380	0.380	0.380	0.380	0.380	0.380
Undensified active fuel length, in.							
UO_2 rods	143.0	143.0	143.0	143.0	143.0	143.0	143.0
Gd rods	143.0	143.0	143.0	143.0	143.0	143.0	143.0
Pellet OD, in.	0.3735	0.3735	0.3735	0.3735	0.3735	0.3735	0.3735
Fuel pellet initial density, %TD mean	96.0	96.0	96.0	96.0	96.0	96.0	96.0
Average burnup BOC, MWd/mtU[a]	27,588	25,573	0	0	0	0	0
Cladding collapse burnup, MWd/mtU[b]	>60,000	>60,000	>60,000	>60,000	>60,000	>60,000	>60,000
Maximum pin burnup, MWd/mtU[a]	57,356	54,110	32,599	33,343	30,061	32,305	33,821
Initial fuel enrichment, wt% ^{235}U							
UO_2 rods	4.88	4.88	4.59	4.59	4.94	4.94	4.94
Gd rods	2.93 & 4.15	4.15	2.75 & 3.67	2.75 & 3.67	3.95	3.95	2.96 & 3.95
Nom. LHR at 2772 MWt, kW/ft[c]	6.15	6.15	6.15	6.15	6.15	6.15	6.15
Nom. LHR at 2817 MWt, kW/ft[c]	6.25	6.25	6.25	6.25	6.25	6.25	6.25
Minimum linear heat rate to melt, kW/ft							
UO_2 rods	22.1	22.1	22.1	22.1	22.1	22.1	22.1
2 wt% Gd rods	—	21.1	—	—	—	—	—
3 wt% Gd rods	20.7	—	—	—	—	—	—
4 wt% Gd rods	—	—	20.3	20.3	20.3	20.3	20.3
6 wt% Gd rods	—	—	—	—	—	—	—
8 wt% Gd rods	19.3	—	19.3	19.3	—	—	19.3

(a) Includes allowance for cycle 13 shutdown length flexibility.

(b) Calculated using method from Reference 5.

(c) LHR calculations include a 0.973 energy deposition factor.

Table 4-2. B8A UO_2 Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
12,000	45.80
20,000	33.30
32,000	27.00
40,000	24.80
52,000	21.60
60,000	19.80

Table 4-3. B9A UO_2 Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
13,000	28.92
21,000	27.55
33,000	26.81
41,000	27.96
53,000	25.96
61,000	21.18

Table 4-4. B10K and B12 UO_2 Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
20,000	28.95
30,000	27.31
40,000	24.68
50,000	23.70
60,000	21.36
65,000	20.36

Table 4-5. B9A Gd Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
30,000	23.50
40,000	23.38
50,000	22.81
60,000	20.59

Table 4-6. B10K and B12 Gd Rod Transient Strain Limits

Burnup (MWd/mtU)	LHR at 1.0% Strain (kW/ft)
20,000	27.40
30,000	25.29
40,000	24.98
50,000	24.36
60,000	19.03
65,000	18.38

5.0 NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters for the cycle 13 and 14 designs. The values for cycles 13 and 14 were generated with the NRC approved NEMO code (Reference 12). Differences in core physics parameters are to be expected between the cycles due to the changes in fuel and burnable poison concentrations, which create changes in flux and burnup distributions. A 1.63% power level uprate and a longer design life with increased initial ^{235}U enrichments along with the differences in the shuffle pattern and the gadolinia burnable poison placement also contribute to the differences in the physics parameters between cycles 13 and 14. Figure 5-1 illustrates a representative relative power distribution for BOC 14 at full power with equilibrium xenon, group 7 inserted to nominal HFP position, and gray APSRs partially inserted.

All analyses of the nuclear physics parameters were performed with NRC approved methodologies. The physics characteristics of the cycle 14 design were evaluated with respect to the applicable design criteria for the accident and transient analysis as described in Section 7. The rod position limits presented in Section 8 considered the shutdown margin requirements and the calculated ejected rod worths and their adherence to design criteria at all times in life and at all power levels. The ejected rod worths in Table 5-1 are the maximum calculated values. The adequacy of the shutdown margin with cycle 14 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 13).
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.

5.2 Changes in Nuclear Design

The design changes for cycle 14 include increased enrichment, a larger feed batch size, a longer cycle length, a 12°F (analyzed) EOC T_{avg} reduction, a 1.63% RTP uprate, and a second batch of M5™ cladding. These changes were incorporated into the physics model. No changes were required to the physics model as a result of replacing the reactor vessel closure head. In addition, the impact of the stainless steel rods was evaluated and determined not to significantly impact core reactivity, stuck rod worth, or ejected rod worth. Reference 12 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 14 Physics Parameters[a]

	Cycle 13[b]	Cycle 14[c]
Cycle length, EFPD	683	725
Cycle burnup, MWd/mtU	22,469	23,822
Average core burnup - 725 EFPD[b], MWd/mtU	39,444	40,570
Initial core loading, mtU	84.3	85.7
Critical boron[d] - 0 EFPD, ppm		
HZP	2,315	2,266
HFP	2,095	2,022
Critical boron[d] - 725 EFPD[b], ppm		
HZP	237	208
HFP	5[e]	5[e]
Control rod worths - HFP, 4 EFPD, %Δk/k		
Group 6	1.00	0.92
Group 7	0.89	0.89
Group 8	0.11	0.11
Control rod worths - HFP, 725 EFPD[b], %Δk/k		
Group 7	0.92	0.90
Max ejected rod worth - HZP, %Δk/k		
4 EFPD[f], Groups 5-8 Inserted (N-12)	0.45	0.27
725 EFPD[b], Groups 5-7 inserted (N-12)	0.45	0.37
Max stuck rod worth - HZP, %Δk/k		
4 EFPD[f] (N-12)	0.46	0.47
725 EFPD[b] (M-13)	0.69	0.77
Power deficit[g] - HZP to HFP, %Δk/k		
4 EFPD	-1.49	-1.61
725 EFPD[b]	-3.01	-3.05
Doppler coeff[g,h] - HFP, 10^{-3} %Δk/k/°F		
0 EFPD[i]	-1.58	-1.61
725 EFPD[b], 0 ppm	-1.78	-1.78
Moderator coeff[g] - HFP, 10^{-2} %Δk/k/°F		
0 EFPD[i]	-0.21	-0.37
725 EFPD[b], 0 ppm[j]	-3.52	-3.57
Temperature coeff[g] - HZP, 10^{-2} %Δk/k/°F		
725 EFPD[b], Groups 1-7 Inserted, M13 out, 0 ppm	-2.56	-2.77

Table 5-1. Davis-Besse Unit 1, Cycle 14 Physics Parameters [a] (cont.)

	Cycle 13[b]	Cycle 14[c]
Boron worth[g] - HFP, ppm/%Δk/k		
0 EFPD	169	175
725 EFPD[b]	126	131
Xenon worth[g] - HFP, %Δk/k		
4 EFPD	2.41	2.38
725 EFPD[b]	2.69	2.68
Effective delayed neutron fraction[g] - HFP		
4 EFPD	0.00643	0.00650
725 EFPD[b]	0.00530	0.00531

(a) Calculations at 0 EFPD are done with No Xenon. All other calculations are at 100%FP Eq Xe.

(b) Cycle 13 values are from Reference 3. EOC values calculated at 683 EFPD for cycle 13.

(c) Based on cycle 12 length of 630.73 EFPD (actual) and cycle 13 length of 620 EFPD.

(d) Control rod group 8 is inserted for calculation at 0 EFPD and withdrawn for calculation at 725 EFPD.

(e) Power coastdown to 683 EFPD at 5 ppm for cycle 13 and to 725 EFPD at 5 ppm for cycle 14.

(f) The cycle 13 value is calculated at 0 EFPD.

(g) All calculations done with control rod groups 1-7 at 100% WD and control rod group 8 at nominal HFP position, unless otherwise noted.

(h) Doppler temperature coefficient calculated using a distributed fuel temperature.

(i) Cycle 14 values were calculated at 2133 ppm (includes allowances for reactivity anomalies and shutdown window flexibility); cycle 13 values were calculated at 2207 ppm.

(j) These values were calculated with the control rods at rod index 260% WD.

Table 5-2. Shutdown Margin Calculation for Davis-Besse, Cycle 14

	BOC, %Δk/k	EOC, %Δk/k		
	4 EFPD	600 EFPD	664 EFPD	725 EFPD*
Available Rod Worth				
Total rod worth, HZP	5.902	6.412	6.543	6.612
Maximum stuck rod worth, HZP	-0.467	-0.666	-0.729	-0.771
Net Worth	5.435	5.746	5.814	5.841
Less 6% Uncertainty	-0.326	-0.345	-0.349	-0.350
Total available worth	5.109	5.401	5.465	5.491
Required Rod Worth				
Power deficit, HFP to HZP	1.607	2.785	3.001	3.054
Off-nominal flux distribution allowance	0.350	0.200	0.200	0.200
Max allowable inserted rod worth at RI = 260% WD	0.321	0.439	0.467	0.480
Total required worth	2.278	3.424	3.668	3.734
Shutdown Margin				
Total available minus total required	2.831	1.977	1.797	1.757

Note: Required shutdown margin is 1.000%Δk/k.

* Group 8 out

Figure 5-1. Davis-Besse Cycle 14 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	0.790	1.214	1.152	1.266	7 1.142	1.172	1.187	0.368
K		1.118	1.244	1.184	1.307	1.141	1.159	0.362
L			1.150	1.265	8 1.112	1.252	1.024	0.264
M				1.209	1.215	1.159	0.650	
N					7 1.125	1.068	0.319	
O						0.503		
P								
R								

Legend	
x	Inserted Rod Group Number
x.xxx	Relative Power Density

6.0 THERMAL-HYDRAULIC DESIGN

The cycle 14 core is composed of several Mark-B assembly designs including the Mark-B10K (batch 15) and the Mark-B12 (batch 16). The Mark-B10K and Mark-B12 designs contain a slightly higher hydraulic resistance for the lower end fitting than other fuel designs in the core (Mark-B8A and Mark-B10M designs). The transition to M5™ guide tubes in the fresh batch 16 fuel has an insignificant impact on the core thermal-hydraulic performance. There are also four Mark-B10K assemblies that contain M5™ guide tubes and two M5™ intermediate spacer grids. Evaluations have shown there is no DNB transition core penalty for the Mark-B10K and Mark-B12 during cycle 14 since the benefit of a longer fuel stack for these designs offsets the DNB effect of the higher hydraulic resistance of the lower end fitting. A core analysis at a rated power of 2772 MWt and a core analysis at a thermal power of 2820 MWt analyzed for the Caldon power level uprate (PLU) were considered. The pressure-temperature limits from the reference core analysis for the Caldon PLU are more restrictive than those for the reference core analysis applicable for 2772 MWt. This requires a change to the Technical Specifications for operation at the Caldon PLU Rated Thermal Power (2817 MWt). This Technical Specification change will be implemented upon NRC approval of the LAR supporting the Caldon PLU. The approved analysis methods described in Reference 1 and the statistical core design (SCD) methodology, Reference 14, were utilized in the analysis. The four batch 15E Mark-B10K M5™ structural assemblies were shown to have acceptable operation within thermal-hydraulic limits for both rated power levels for cycle 14.

The effects of the twenty-five stainless steel rods (SSRs) in nine batch 14 and three batch 15 fuel assemblies were considered in all thermal-hydraulic analyses for both rated power levels for cycle 14. The effects of the SSRs were evaluated in accordance with Reference 15.

Two design modifications that have been implemented for the fresh batch 16 fuel were evaluated with respect to their impact on thermal-hydraulic performance: the MONOBLOC™ instrument tube and the six-leaf holddown spring. The MONOBLOC™ instrument tube was evaluated, and it was shown that the differences relative to the existing design are not significant. The holddown capability of the six-leaf holddown spring was also evaluated, and it was shown that the existing relationships between fourth pump startup temperature and maximum flow rate could still be maintained.

The Mark-B10M, Mark-B10K, and Mark-B12 fuel designs contain optimized guide tubes that minimize the control rod guide tube core bypass flow. The cycle 14 specific core bypass flow rate of 5.7% is equivalent to the value used in the reference core analysis for the Caldon PLU and exceeds the 5.3% value used in the reference core analysis for a rated thermal power level of 2772 MWt. For the reference core analysis at 2772 MWt, the effect of this increased bypass flow rate is offset by retained DNB margin.

The DNB-based thermal-hydraulic analyses for cycle 14 are applicable for UO_2 and Gd 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 increased resistance was incorporated into the reference core analysis for the Caldon PLU. For the reference core analysis at 2772 MWt, 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 13 and 14. Cycle 14 values for a rated thermal power of 2772 MWt without the Caldon PLU and for a thermal power of 2820 MWt analyzed for the Caldon PLU are shown in Table 6-1.

The thermal-hydraulic design-based analyses were performed with NRC approved methodologies and, where applicable show that all design criteria are met.

Table 6-1. Limiting Thermal-Hydraulic Design Conditions, Cycles 13 and 14

	Cycle 13	Cycle 14 (2772 MWt)	Cycle 14 (2820 MWt)
Design power level, MWt	2772	2772	2820
Nominal core exit pressure, psia	2200	2200	2200
Minimum core exit pressure, psia	2135	2135	2135
Reactor coolant flow, gpm	380,000	380,000	380,000
Core bypass flow, %	5.3[a]	5.3[a]	5.7[a]
DNBR modeling	SCD	SCD	SCD
Reference design (radial x local) power peaking factor	1.795	1.795	1.800
Reference design axial flux shape	1.65 chopped cosine	1.65 chopped cosine	1.65 chopped cosine
Hot channel factors			
Enthalpy rise	1.015	1.015	1.015
Heat flux	N/A[b]	N/A[b]	N/A[b]
Flow Area	0.97	0.97	0.97
Active fuel length, in.	140.6[c]	140.6[c]	140.6[c]
Avg heat flux at 100% power, 10^5 Btu/h-ft^2	1.89	1.89	1.92
Max heat flux at 100% power, 10^5 Btu/h-ft^2	5.60	5.60	5.71
CHF Correlation	BWC	BWC	BWC
CHF Correlation DNB limit	1.40 TDL[d]	1.40 TDL[d]	1.40 TDL[d]
Minimum DNBR			
at 102% power (2772 MWt)	2.02	2.02	---
at 112% power (2772 MWt)	1.79	1.79	---
at 100.37% power (2820 MWt)[e]	---	---	1.95
at 110.37% power (2820 MWt)[e]	---	---	1.72

(a) Used in the analysis.
(b) The hot channel factor for heat flux is no longer applicable in DNB calculations as allowed by Reference 1.
(c) Value used is conservative for DNB analysis relative to the 143.0 in. batch 16 active fuel length.
(d) Thermal Design Limit
(e) Analysis was performed at a conservative thermal power of 2820 MWt. The Caldon PLU Rated Thermal Power is 2817 MWt.

7.0 ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each USAR accident analysis has been examined with respect to changes in the cycle 14 parameters to determine the effects of the cycle 14 reload and to ensure that thermal performance during hypothetical transients is not degraded.

The radiological dose consequences of the USAR Chapter 15 accidents have been evaluated using conservative radionuclide source terms that bound the cycle specific source terms for Davis-Besse cycle 14. None of the accident doses are adversely impacted by the cycle 14 design and remain below the respective acceptance criteria values as documented in the USAR. The cycle 14 doses also remain below the NUREG-0800 (Reference 16) acceptance criteria.

7.2 Accident Evaluation

A comparison of the key kinetics parameters from the USAR and cycle 14 is provided in Table 7-1.

The EOC moderator temperature coefficient listed in Table 7-1 for cycle 14 is the 3-D, hot full power (HFP) temperature coefficient. An evaluation was performed to verify the acceptability of the cycle 14 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 combined effects of moderator density, boron worth, control rod worth degradation and Doppler reactivity, over the temperature range from 532°F to 510°F. The combined EOC temperature coefficient for cycle 14 is shown in Tables 5-1 and 7-1 as -2.77×10^{-2} %Δk/k/°F. Since the safety analysis value for the EOC temperature coefficient is more negative than the cycle 14 value, the steam line break analysis remains bounding for cycle 14.

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-B10M, Mark-B10K, and Mark-B12 fuel types. With implementation of the Caldon LEFM CheckPlus™ system at Davis-Besse Unit 1, the power level uncertainty will be reduced from 2 percent to 0.37 percent, with a corresponding increase in the rated thermal power. The LOCA LHR limits are analyzed at a power level that includes the power level uncertainty. Therefore, the LOCA LHR limits are applicable to a cycle 14 rated thermal power of 2772 MWt without the Caldon system operational and 2817 MWt with the Caldon system in operation. In either case, the LOCA LHR limits were analyzed at a power of at least 2827 MWt.

The RELAP5/MOD2-B&W ECCS Evaluation Model techniques and assumptions, as described in BAW-10192PA (Reference 17), were used in the Mark-B10M, Mark-B10K, and Mark-B12 analyses. These

assemblies were analyzed at 3025 MWt to support a future power uprate. The CRAFT2-based ECCS Evaluation Model, as described in BAW-10104PA, Rev. 5 (Reference 18), was used in the Mark-B8A analyses. The Mark-B8A assembly was analyzed at 2827 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 7-2 shows the maximum allowable LOCA linear heat rate limits for the different types of fuel in the Davis-Besse Unit 1 cycle 14 core as functions of burnup. Sensitivity studies performed at power levels below 3025 MWt at 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-B10M, Mark-B10K, and Mark-B12 LHR limits is necessary for application in the maneuvering analysis for some core power levels less than 3025 MWt to ensure that the LHRs determined at 3025 MWt remain limiting. The Mark-B8A LHR limits were analyzed at an initial core power level of 2827 MWt and do not require further adjustment based on reduced power levels.

For the batch 9H Mark-B8A assembly, linear interpolation between the elevation-specific linear heat rate limits at 24,500 MWd/mtU and the LHR limit of 12.0 kW/ft at 52,000 MWd/mtU was justified for cycle 14. The LHR limit for any burnup beyond 52,000 MWd/mtU can be interpolated between 12.0 kW/ft at 52,000 MWd/mtU and 10.2 kW/ft at 60,000 MWd/mtU. The LHR limits for the Mark-B8A fuel were converted to a basis that is consistent with that reported for the Mark-B10M, Mark-B10K, and Mark-B12 assemblies analyzed based on the RELAP5/MOD2-B&W ECCS Evaluation Model.

For the batch 14 UO_2 fuel, the cycle-specific fuel rod performance data and predicted radial peaks for cycle 14 were found to be bounded by the fuel data used in the Mark-B10M (B9A fuel rods) LOCA analyses, which were calculated using the TACO3 fuel performance code (Reference 4). At high fuel burnups, the limits for batch 14 are reduced in order to maintain the internal fuel rod pressure consistent with (less than or equal to) the limit based on the NRC-approved fuel rod gas pressure criterion (Reference 9).

The maximum allowable LOCA linear heat rate limits for batch 15 Mark-B10K UO_2 fuel were determined using material properties for M5™ cladding (Reference 6). The effect of the M5™ spacer grids occupying the uppermost two intermediate grid locations in the batch 15E M5™ structural assemblies was evaluated for Davis-Besse Unit 1 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 batch 16 Mark-B12 assembly design is based on the Mark-B10K design, but with M5™ guide tubes and a slightly longer shoulder gap height that results in a slightly reduced fuel rod plenum volume. Analyses and evaluations were performed for the Mark-B12 design to justify applicability of the Mark-B10K LOCA LHR limits to the Mark-B12.

The linear heat rate limits for batch 14, 15, and 16 fuel rods containing gadolinia, which are based on fuel rod performance data from the GDTACO (Reference 8) fuel rod performance code, were determined for evaluation in the subsequent power distribution analysis. The 8 wt% Gd fuel pellets in batch 16 that were found to have a negative resintering density after testing were evaluated and do not affect the LOCA analyses. The LHR limits for the Gd fuel are based on percentages of the UO_2 LHR limits to account for the reduction in thermal conductivity in the fuel rod. The UO_2 LHR limits are reduced between 95 percent and 85 percent for Gd concentrations between 2 and 8 wt%.

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 14 was compared to the resulting allowable MTC curve 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 the EOC T_{avg} reduction maneuver could be performed. The results of the analysis showed that operation for an analyzed EOC T_{avg} reduction of up to 12°F, based on a nominal T_{avg} of 582°F, and an MTC more negative than -10 pcm/°F provides LOCA results that are bounded by the nominal T_{avg} LOCA results. The allowable indicated EOC T_{avg} reduction is smaller than the analyzed reduction of 12°F to account for measurement uncertainty. The cycle 14 EOC MTC values are significantly more negative than the -10 pcm/°F allowed.

The effect of the inclusion of 25 stainless steel rods on the maximum predicted pin peaks and LOCA LHR limits for all batches was evaluated according to Reference 15. It was confirmed that no changes to the LOCA LHR limits or additional evaluations were required as a result of inclusion of these stainless steel rods in cycle 14. Additionally, the closure head replacement did not affect the LOCA analyses and evaluations applicable to cycle 14.

The continued validity of the non-LOCA USAR analyses was assessed for cycle 14 operation both prior to the implementation of the Caldon power level uprate, when the rated thermal power will be 2772 MWt, and after the Caldon power level uprate, when the rated thermal power will be 2817 MWt. It was determined that the non-LOCA USAR analyses remain valid for Davis-Besse Unit 1 cycle 14 operation both before and after implementation of the Caldon power level uprate.

The continued validity of the non-LOCA USAR analyses was assessed for a withdrawal of the APSRs and an actual reduction in T_{avg} of as much as 12°F near the end of cycle 14. It was determined that the non-LOCA USAR analyses remain valid for the APSR withdrawal and an actual T_{avg} reduction of as much as 12°F near EOC.

It is concluded by the examination of cycle 14 core thermal, thermal-hydraulic, and kinetics properties, with respect to acceptable previous cycle values, that the cycle 14 core reload will not adversely affect the

ability to safely operate the Davis-Besse plant during cycle 14. The previously accepted analysis basis for Davis-Besse consists of the analyses presented in the USAR. The analysis basis was developed using kinetics parameter values that were shown to bound the corresponding cycle 14 parameter values. Consequently, it is concluded that the existing analysis basis is bounding for cycle 14.

Table 7-1. Comparison of Key Parameters for Accident Analysis

Parameter	USAR Value	Cycle 14 Value	Bounding Value is:
BOL[a] Doppler coeff, 10^{-3} %Δk/k/°F	-1.28	-1.61	Less Negative
EOL[a,i] Doppler coeff, 10^{-3} %Δk/k/°F	-1.45	-1.64	Less Negative
EOL Doppler coeff, 10^{-3} %Δk/k/°F	-1.77[b]	-1.78	More Negative[e]
BOL HFP moderator coeff, 10^{-2} %Δk/k/°F	+0.13	-0.37	Less Negative/ More Positive
BOL HZP moderator coeff, 10^{-2} %Δk/k/°F	+0.90	+0.27	Less Negative/ More Positive
EOL HFP moderator coeff, 10^{-2} %Δk/k/°F	-4.0	-3.57	More Negative
EOL temperature coeff (532 to 510°F), 10^{-2} %Δk/k/°F	-3.10	-2.77	More Negative
BOL All rod group worth (HZP), %Δk/k	10.0	5.863	Larger[f]
Boron reactivity worth (HFP), ppm/%Δk/k	100	172	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	2133[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. The value shown in Table 7.1 is the value given in the USAR. The actual value for cycle 14 is 0.19%Δk/k.

(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 14 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 more negative than, and therefore bounding for, the cycle 14 temperature coefficient.

(f) For the analysis to remain bounding, the cycle-specific value must be $\leq$ 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) Davis-Besse-specific dropped rod accident analyses performed subsequent to the issuance of the Davis-Besse USAR determined that the acceptance criteria for this event are met for dropped rod worths of $\leq$0.28 %Δk/k, which also bounds the cycle 14 value.

(i) Two cycle-specific EOC Doppler coefficients are provided: a maximum value (-1.64×10^{-3} %Δk/k/F for CRG 7 at 60 percent withdrawn) and a minimum value (-1.78×10^{-3} %Δk/k/F for CRG 1-8 fully withdrawn).

Table 7-2. Bounding Values for Allowable LOCA Peak Linear Heat Rates

Mark-B8A Fuel Type as Evaluated at 2827 MWt [1]
Allowable Peak UO_2 LHR for Specified Burnup, kW/ft

Core Elevation, ft	24,500 MWd/mtU	52,000 MWd/mtU	60,000 MWd/mtU
0	16.3	12.0	10.2
2	16.3	12.0	10.2
4	15.1	12.0	10.2
6	15.5	12.0	10.2
8	16.1	12.0	10.2
10	15.2	12.0	10.2
12	14.3	12.0	10.2

Mark-B10M Fuel Type as Analyzed at 3025 MWt [1,2,3]
Allowable Peak UO_2 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 and Mark-B12 Fuel Types as Analyzed at 3025 MWt [1,2,3]
Allowable Peak UO_2 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

[1] Linear interpolation between burnup points and elevations is permitted to calculate the Allowable LHR.
[2] These LHR limits must be reduced by up to 0.2 kW/ft for power levels less than 3025 MWt.
[3] The LHR limits for the gadolinia fuel are based on percentages of the UO_2 LHR limits to account for the reduction in fuel thermal conductivity.

8.0 PROPOSED CORE OPERATING LIMITS REPORT

The Core Operating Limits Report (COLR) has been revised for cycle 14 operation to accommodate the influence of the cycle 14 core design, which includes replacement of the reactor vessel closure head and the effects of stainless steel replacement rods (SSRs), on power peaking, reactivity, and control rod worth. 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 14 specific power distribution analysis using NRC approved methodology provided in the references of Technical Specification 6.9.1.7.

A cycle 14 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 1. The regulating group position operating limits, axial power imbalance operating limits, quadrant power tilt limits, and APSR position operating limits are provided for the COLR. The rated thermal power level for the base cycle 14 design is 2817 MWt. In addition, the effects of long-term operation at a licensed thermal power level of 2772 MWt before implementation of the Caldon LEFM CheckPlus™ power level uprate to 2817 MWt were evaluated and determined to have no detrimental effect on the limits for core protection and operation. Therefore, the power uprate from 2772 MWt to 2817 MWt can be implemented at any time during cycle 14. The analysis incorporates DNB maximum allowable peaking limits based on the allowable increase in design (radial x local) peaking provided by the statistical core design methodology described in Reference 14. The effects of control rod group 7 and gray APSR repositioning were included explicitly in the analysis. The analysis determined that the cycle 14 core operating limits provide protection for the overpower condition that could occur during an overcooling transient because of nuclear instrumentation errors. The cycle 14 analysis also determined that the core safety limits are not violated in the event of a dropped or misaligned control rod assembly initiated from within the limits of normal operation.

The analysis verified that the end of cycle (EOC) hot full power maneuver is bounded by the safety analysis assumptions, and is accommodated in the core RPS protective limits and trip allowable values. The maneuver consists of an APSR withdrawal designed to occur at 654 $\pm$ 10 EFPD and a T_{avg} reduction of up to 12°F (analyzed) to extend HFP operation. An evaluation was performed for the early side of the cycle 14 APSR pull being outside of the operating window basis used in the Nuclear Analysis portion of the 12°F (analyzed) T_{avg} reduction task. The evaluation determined that there was no impact to the conclusions in the Nuclear Analysis evaluation of EOC maneuvers. The xenon stability index after APSR withdrawal was determined to be -0.0332 h^{-1} which demonstrates the axial stability of the core during operation with the APSRs fully withdrawn.

Separate tables and figures are provided for those COLR parameters that contain differences for operation at 2772 MWt Rated Thermal Power prior to implementation of the Caldon power level uprate and at 2817 MWt Rated Thermal Power after implementation of the uprate. The tables and figures applicable to operation at the power uprate conditions are denoted with a suffix (A) in the table or figure number. In addition, the applicable rated thermal power level value is noted on the table or figure, as appropriate.

The maximum allowable LOCA linear heat rate limits used in the analysis are based on the ECCS analysis described in Section 7.2. The LBLOCA analyses were based on the approved methods listed in Reference 1, and the SBLOCA analyses were based on Reference 1 and the recently approved revision 4 of BAW-10164P-A (Reference 19). Tables 8-4/8-4A provide the burnup- and elevation-dependent LOCA linear heat rate limits for each incore segment for input to the Nuclear Applications Software (NAS). Tables 8-5/8-5A provide 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 8-4/8-4A and 8-5/8-5A are reduced by 0.2 kW/ft compared to those provided in Section 7.2 (Table 7-2). 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 14 operation. The linear heat rate limits provided in Tables 8-4/8-4A and 8-5/8-5A 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™ structural fuel assemblies was performed. The gadolinia rods and the M5™ structural fuel 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 14 analysis were error adjusted to generate operating limits for power operation. Figures 8-1/8-1A through 8-4/8-4A and Figures 8-7/8-7A through 8-12/8-12A 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 14. Figures 8-6/8-6A provide the APSR position operating limits for cycle 14. Figures 8-13/8-13A and 8-14/8-14A are the cycle 14-specific core protective limits and RPS imbalance trip allowable values. Limiting nuclear instrumentation scaled difference amplifier gains of 2.0 and 5.0 were used, as appropriate, to establish these limits. 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 positive axial power imbalance operating limits provided in Figures 8-10/8-10A through 8-12/8-12A are based on IC-DNB allowable peaking limits, which bound the power level

dependent LOCA linear heat rate limits at maximum allowable power conditions. Table 8-1 presents the power-dependent quadrant power tilt limits for cycle 14, Table 8-2 provides the negative moderator temperature coefficient limit for cycle 14, and Table 8-3 provides minimum linear heat rate to melt (kW/ft) limits. Tables 8-4/8-4A and 8-5/8-5A 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 both NAS and FIDMS applications reflect the three different active fuel lengths among the four different fuel assembly types and their respective allowable linear heat rate limits. The allowable linear heat rate limits for the NAS application are provided as functions of incore segment (core elevation) and burnup, whereas the limits for the FIDMS application are provided as functions of core elevation (feet) and burnup. 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 reactor coolant pump operation.

Davis-Besse Unit 1 cycle 14 will be operated in accordance with the safety analysis and applicable power calorimetric measurement uncertainty analysis. After the license has been amended and the Caldon power uprate is implemented, the core may be operated at a nominal core power of up to 2817 MWt when the LEFM CheckPlus™ system is in operation. The Davis-Besse Unit 1 Technical Requirements Manual (TRM) and procedures will specify the appropriate actions to be taken when the LEFM CheckPlus™ system is not available. The procedural guidance will contain instructions on the power level reductions that are required under circumstances when the LEFM CheckPlus™ system is not available or not in service.

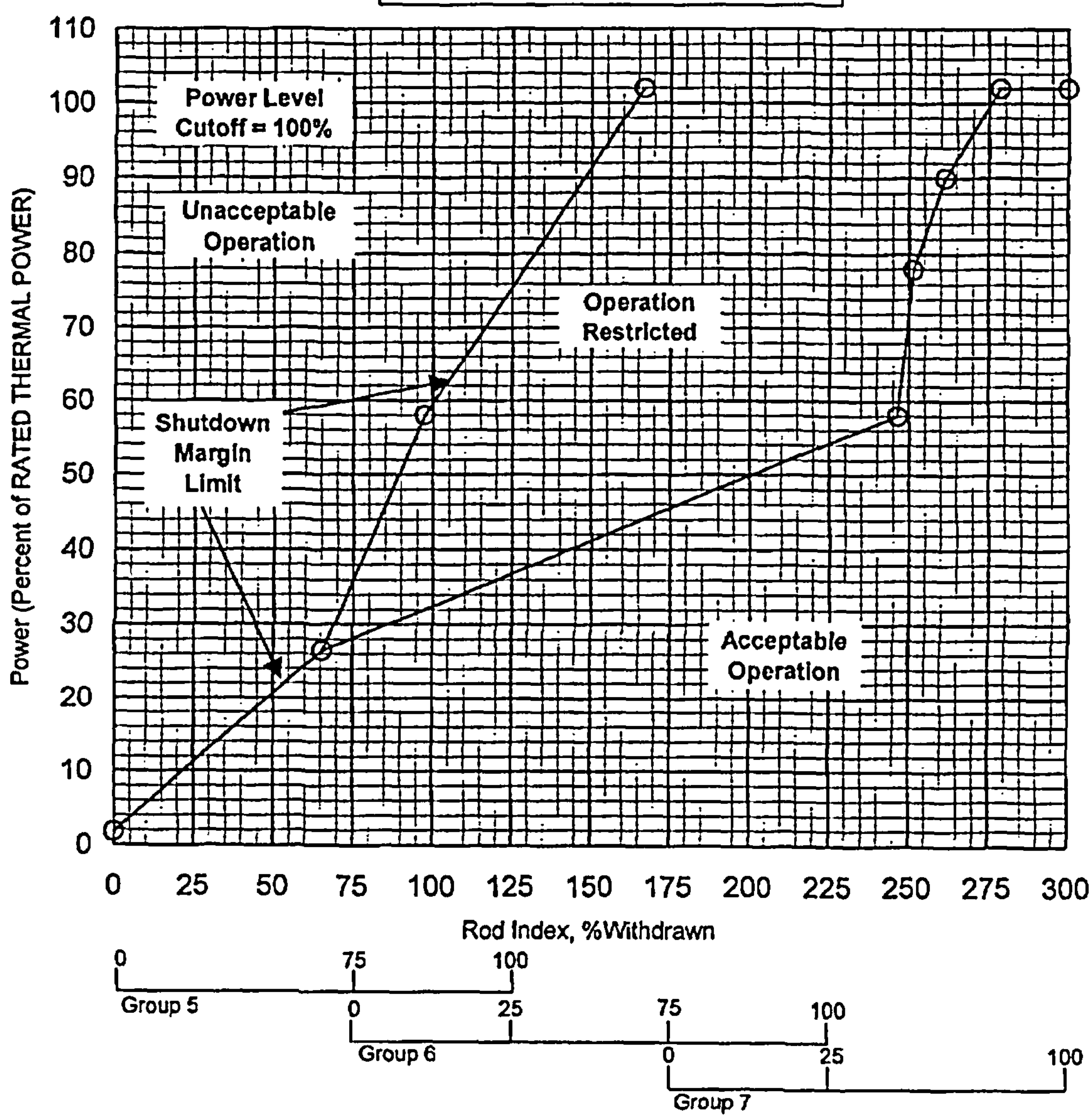
Boric acid volume storage for the boric acid addition system (BAAS) required by Technical Requirements Manual 3/4.1.2.8, 3/4.1.2.9, and Figure 3.1-1 were verified to be acceptable for cycle 14. In addition, the minimum boron concentration requirements and boric acid volume storage for the borated water storage tank (BWST) given in Technical Specifications 3/4.5.4 and Technical Requirements Manual 3/4.1.2.8 were verified to be acceptable for cycle 14 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 within the constraints specified in the COLR.

Figure 8-1

Figure Regulating Group Position Operating Limits
0 to 400 ±10 EFPD, Four RC Pumps -- 2772 MWt RTP
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

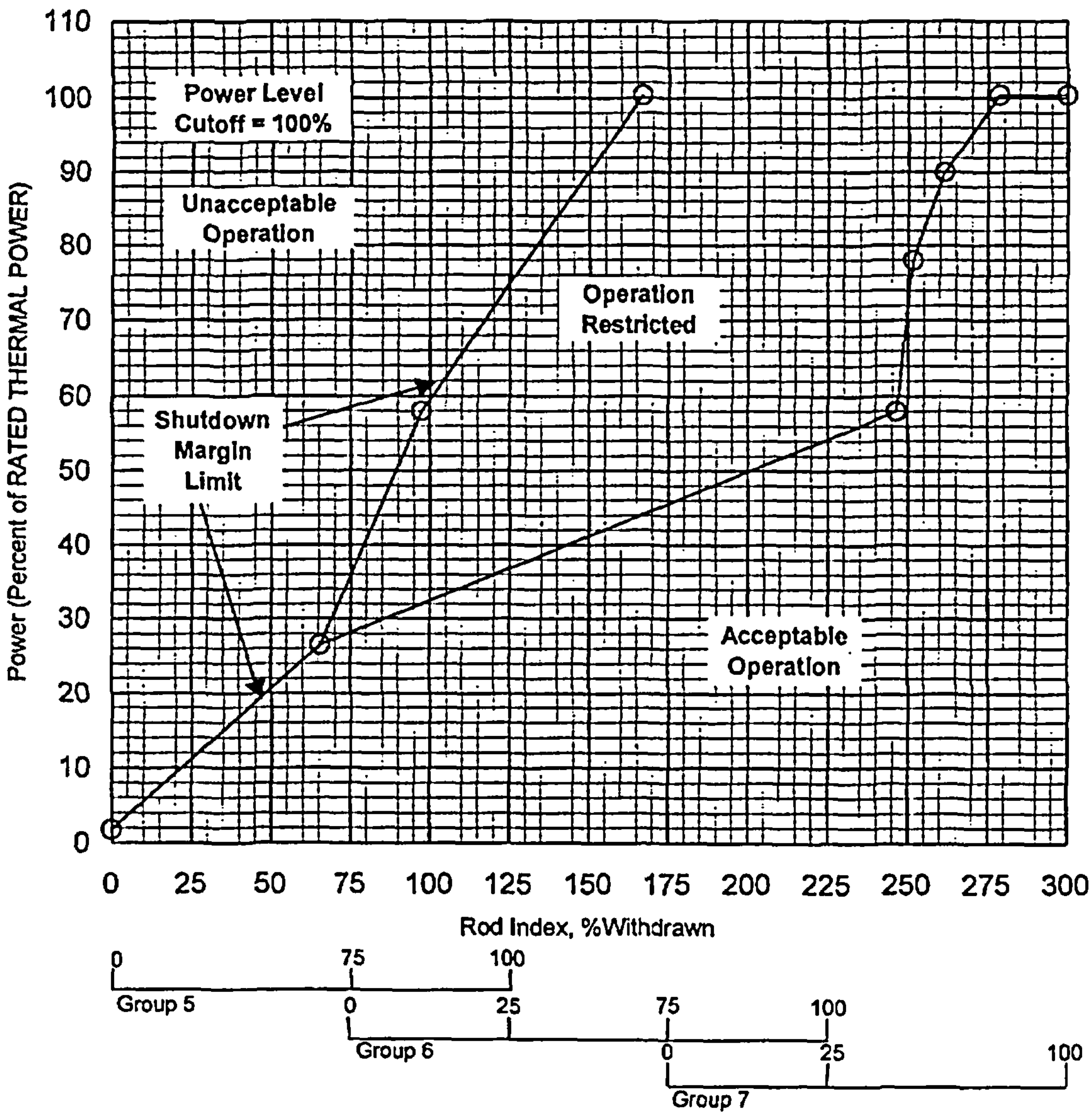


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-1A

Figure Regulating Group Position Operating Limits
0 to 400 ±10 EFPD, Four RC Pumps -- **2817 MWt RTP**
Davis-Besse 1, Cycle 14

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.

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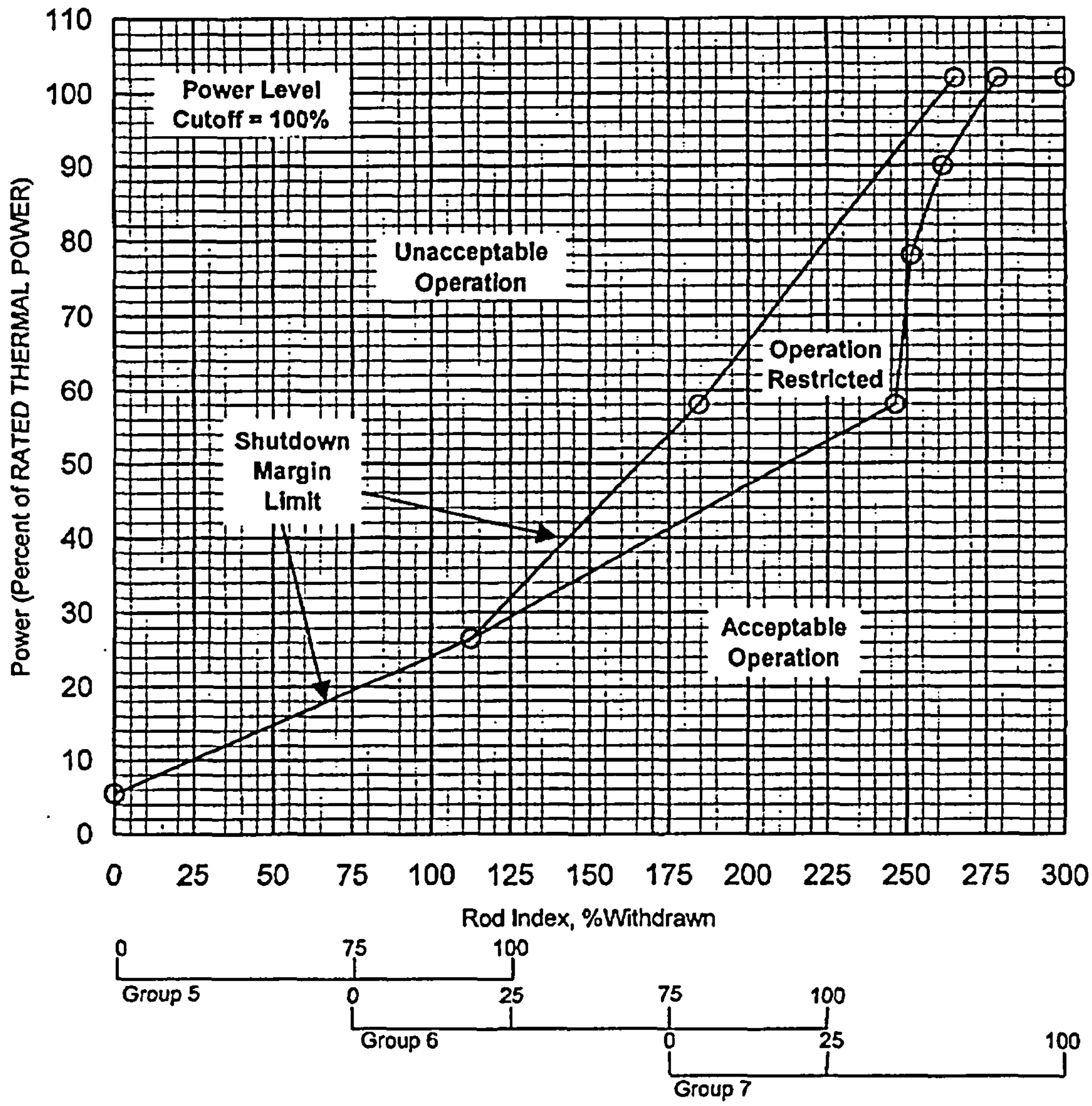
Figure 8-2

Figure Regulating Group Position Operating Limits
After 400 ±10 EFPD, Four RC Pumps -- **2772 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

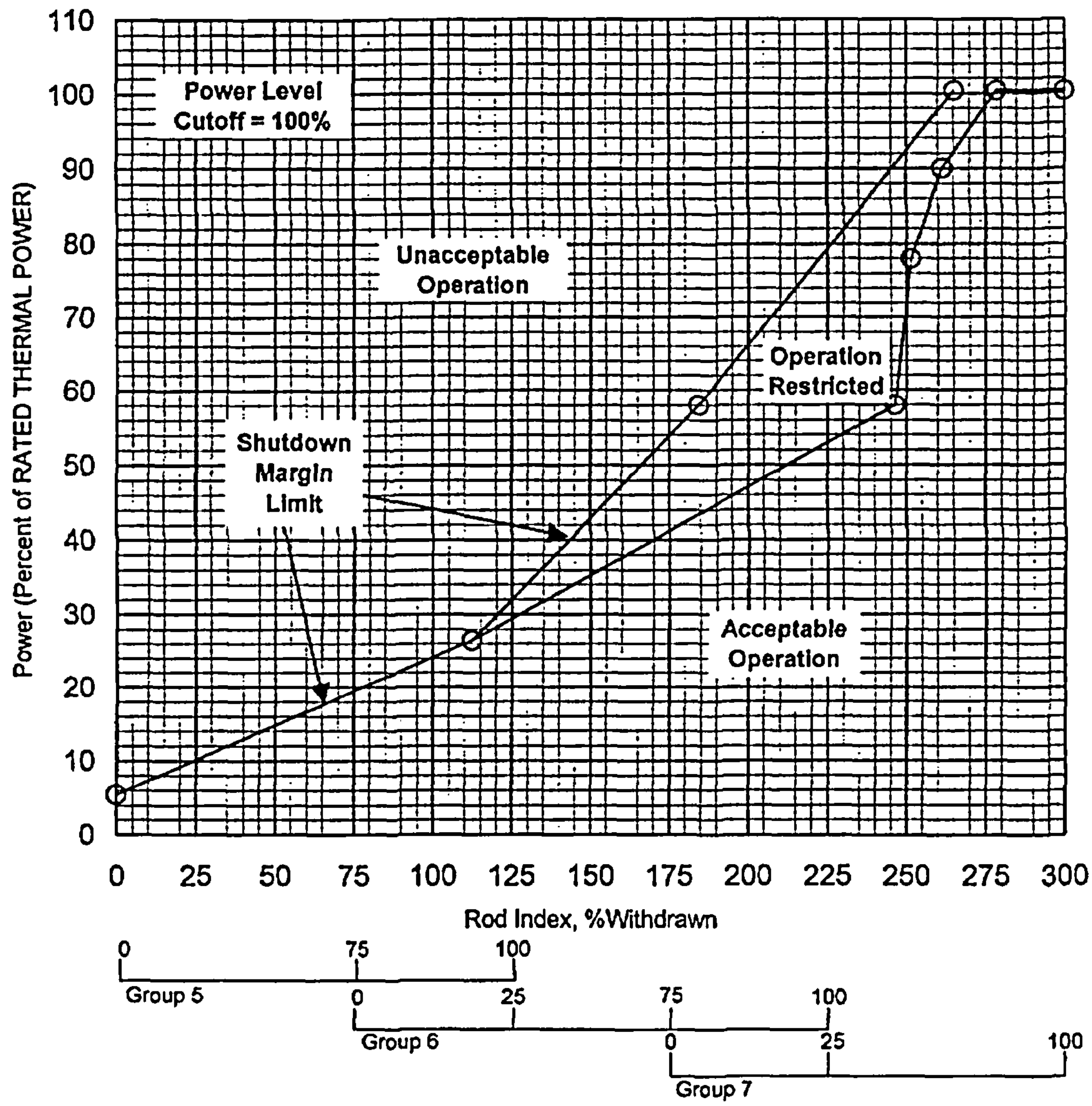


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-2A

Figure Regulating Group Position Operating Limits
After 400 ±10 EFPD, Four RC Pumps -- 2817 MWt RTP
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

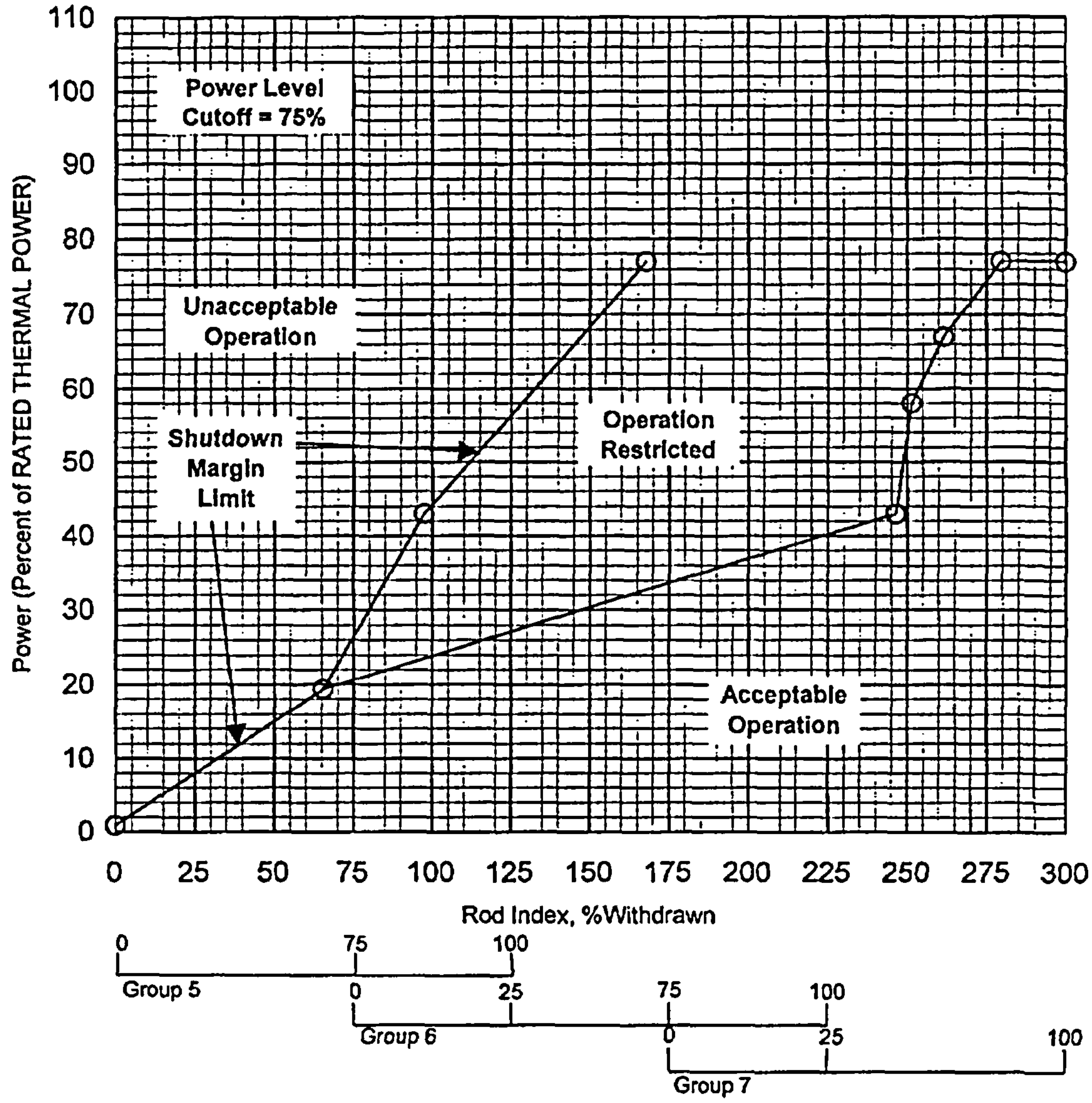


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-3

Figure Regulating Group Position Operating Limits
0 to 400 ±10 EFPD, Three RC Pumps -- 2772 MWt RTP
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

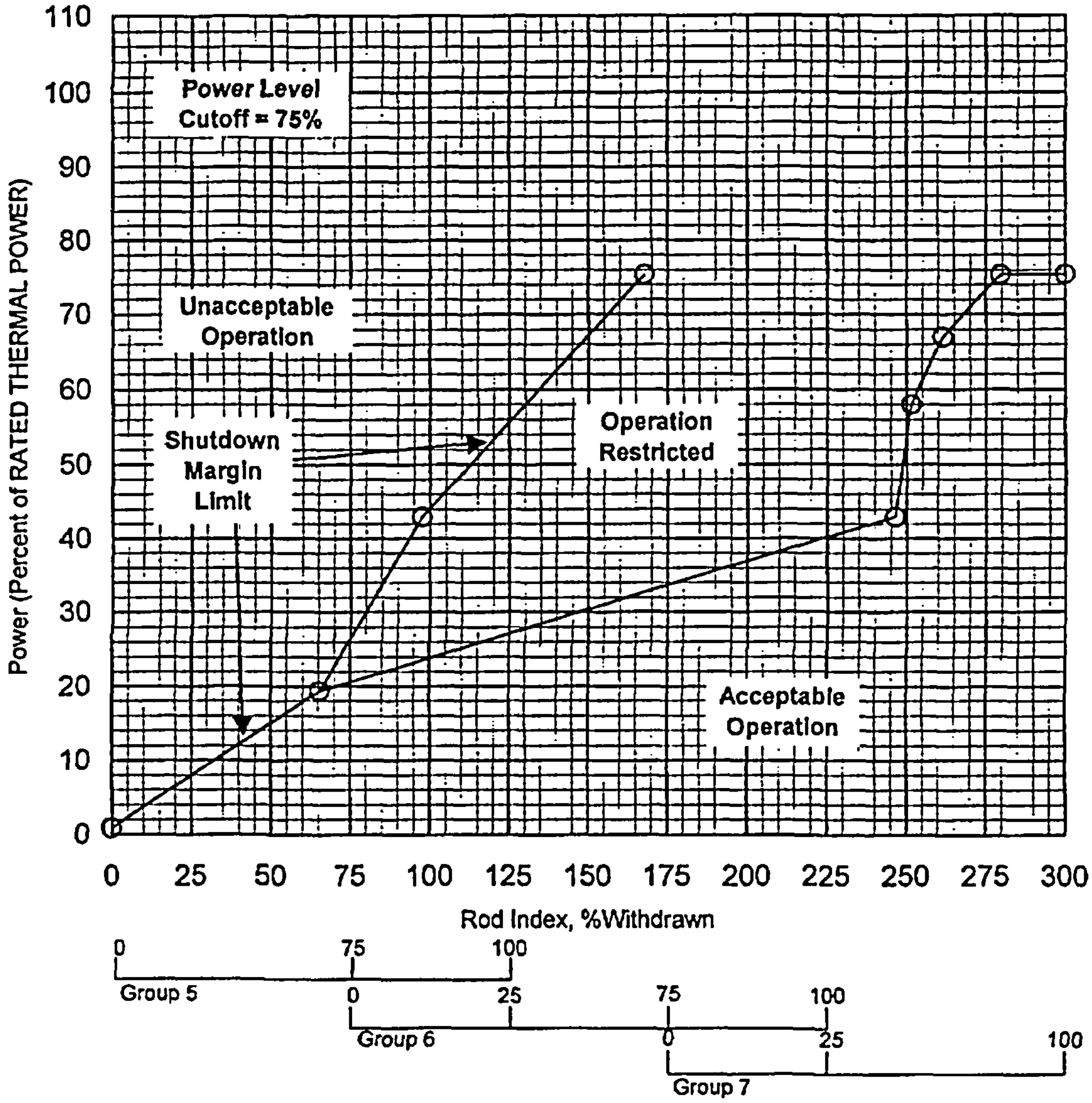


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-3A

Figure Regulating Group Position Operating Limits
0 to 400 ±10 EFPD, Three RC Pumps -- 2817 MWt RTP
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

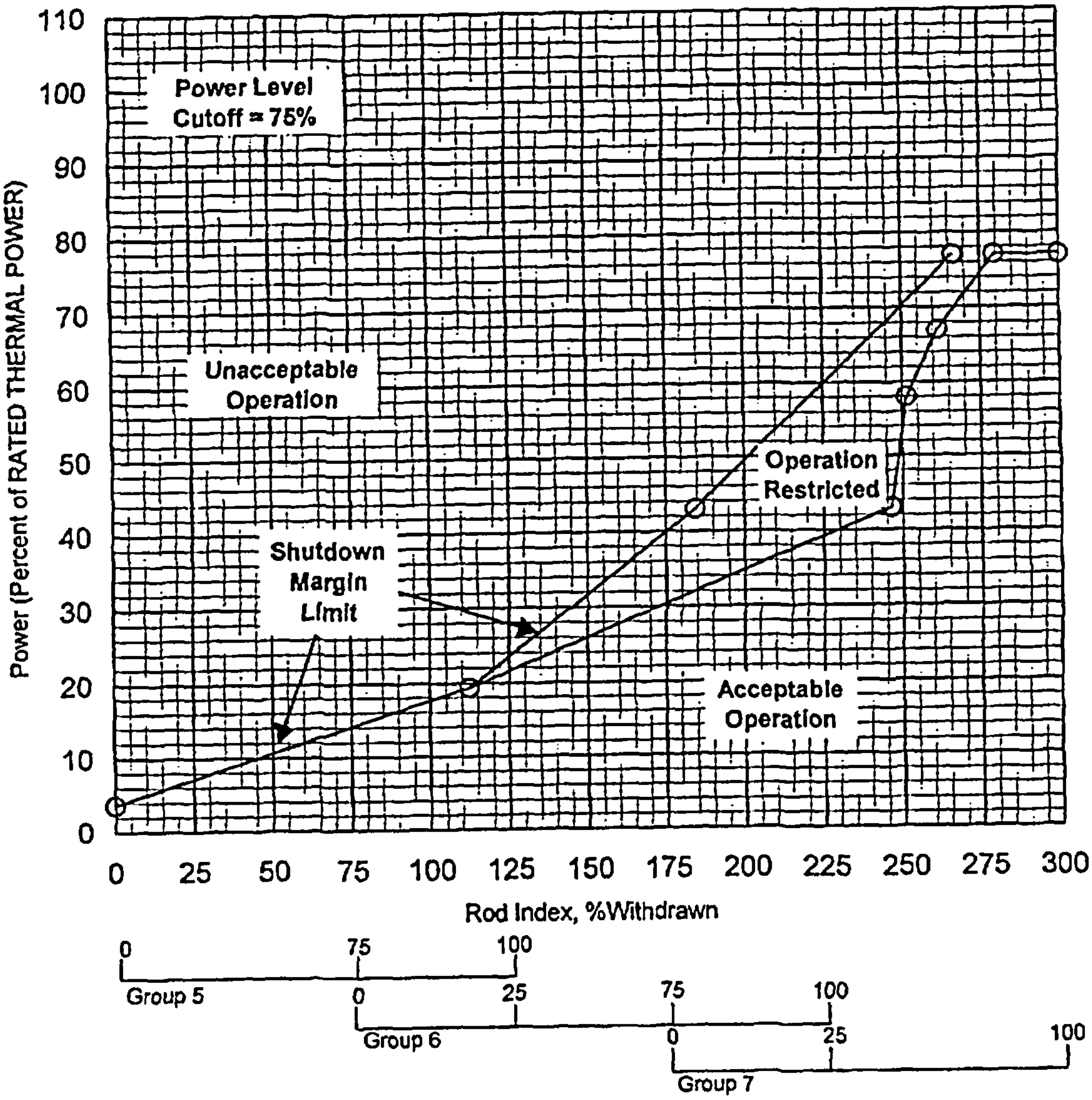


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-4

Figure Regulating Group Position Operating Limits
After 400 ±10 EFPD, Three RC Pumps -- **2772 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specifications 3.1.3.6 and 3.1.3.8

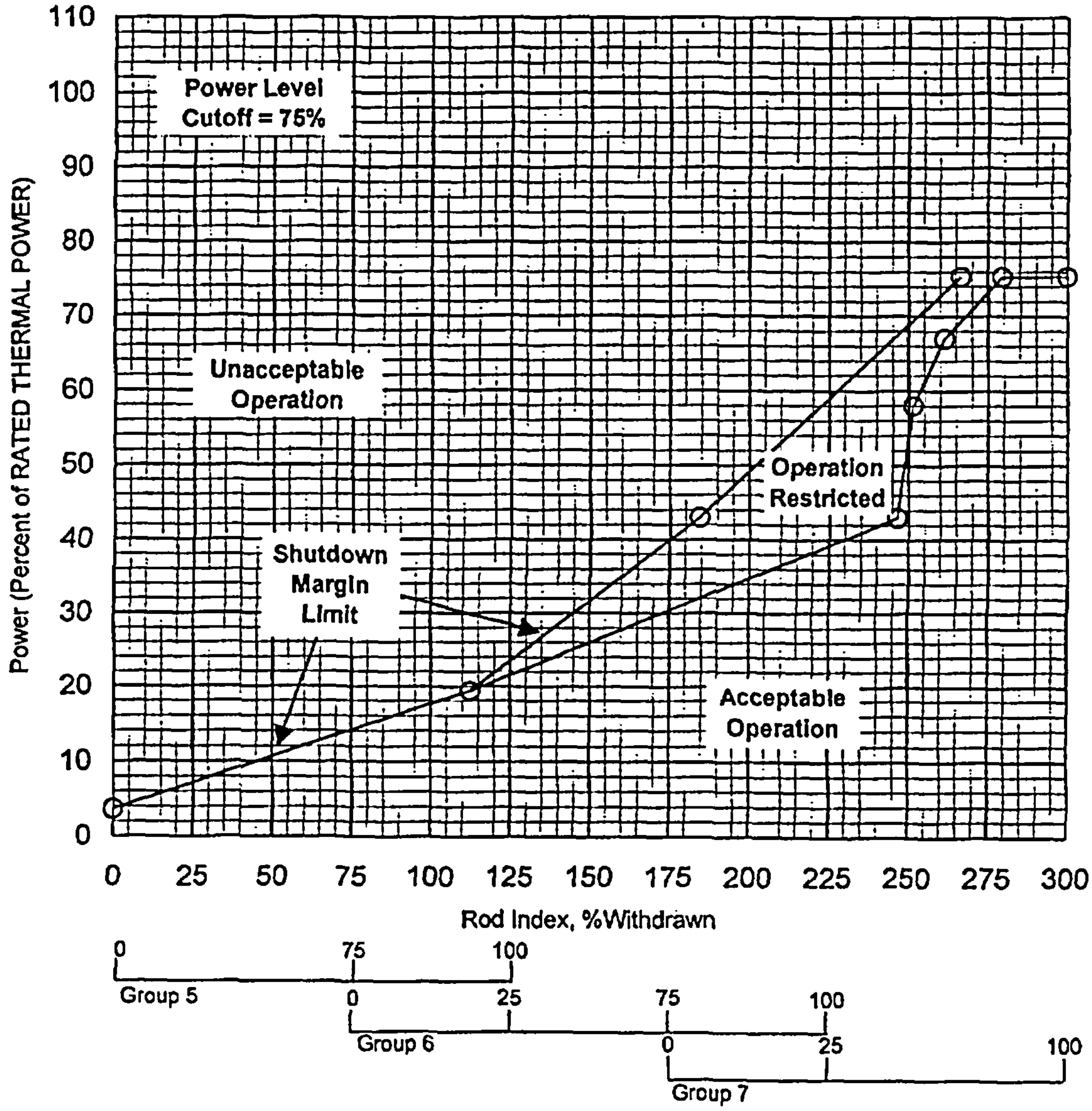


Note 1: A Rod Group overlap of 25 ±5% between sequential withdrawn groups 5 and 6, and 6 and 7, shall be maintained.
Note 2: Instrument error is accounted for in these Operating Limits.

Figure 8-4A

Figure Regulating Group Position Operating Limits
After 400 ±10 EFPD, Three RC Pumps -- **2817 MWt RTP**
Davis-Besse 1, Cycle 14

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.

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Figure 8-5

Figure Control Rod Core Locations
and Group Assignments
Davis-Besse 1, Cycle 14

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)

$\boxed{X}$ 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

2772 MWt RTP

Figure APSR Position Operating Limits

This Figure is referred to by Technical Specification 3.1.3.9

Before APSR Pull: 0 EFPD to 654 ±10 EFPD,
Three or Four RC pumps operation*

Lower Limit: 0 %WD

Upper Limit: 100 %WD

After APSR Pull: 654 ± 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**

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Figure 8-6A

2817 MWt RTP

Figure APSR Position Operating Limits

This Figure is referred to by Technical Specification 3.1.3.9

Before APSR Pull: 0 EFPD to 654 ±10 EFPD,
Three or Four RC pumps operation*

Lower Limit: 0 %WD

Upper Limit: 100 %WD

After APSR Pull: 654 ± 10 EFPD to End-of-Cycle
Three or Four RC pumps operation*

Insertion Prohibited (maintain ≥ 99% WD)

* **Power restricted to 75.37% for 3 pump operation**

Figure 8-7

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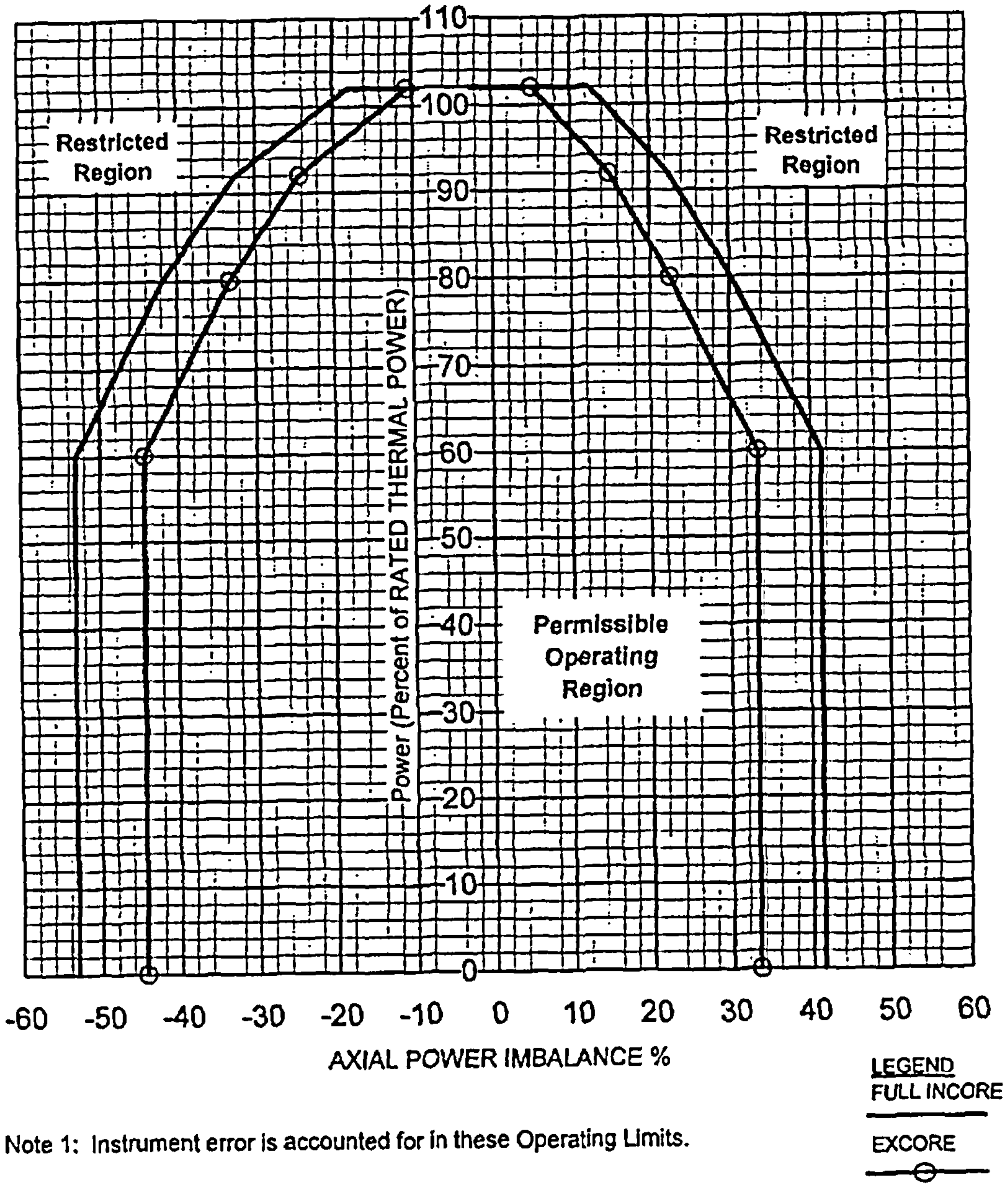
Figure AXIAL POWER IMBALANCE Operating Limits
0 to 300 ±10 EFPD, Four RC Pumps -- 2772 MWt RTP
Davis-Besse 1, Cycle 14

This Figure is referred to by
Technical Specification 3.2.1



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

Figure 8-7A

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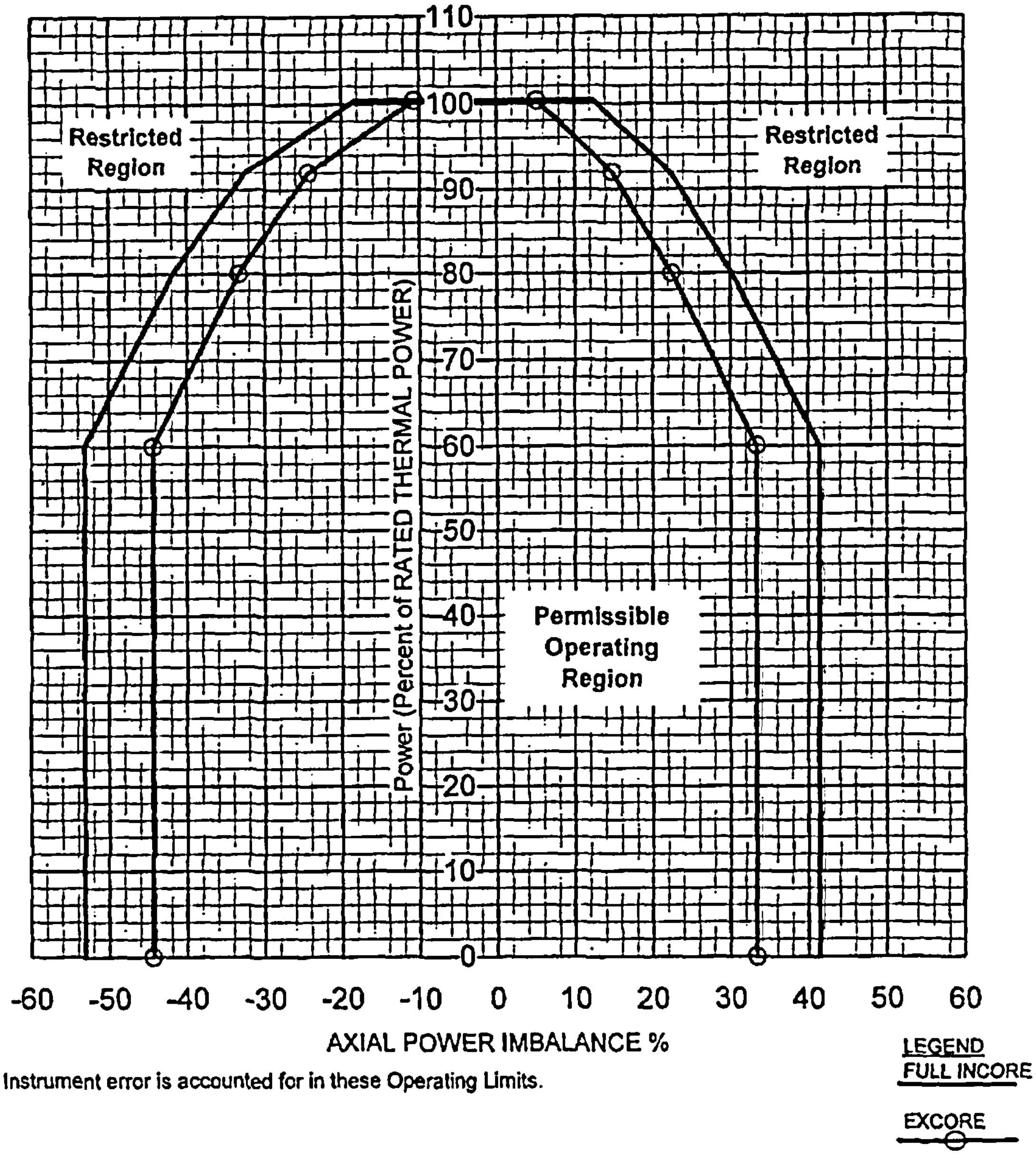
Figure AXIAL POWER IMBALANCE Operating Limits
0 to 300 ±10 EFPD, Four RC Pumps – 2817 MWt RTP
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specification 3.2.1

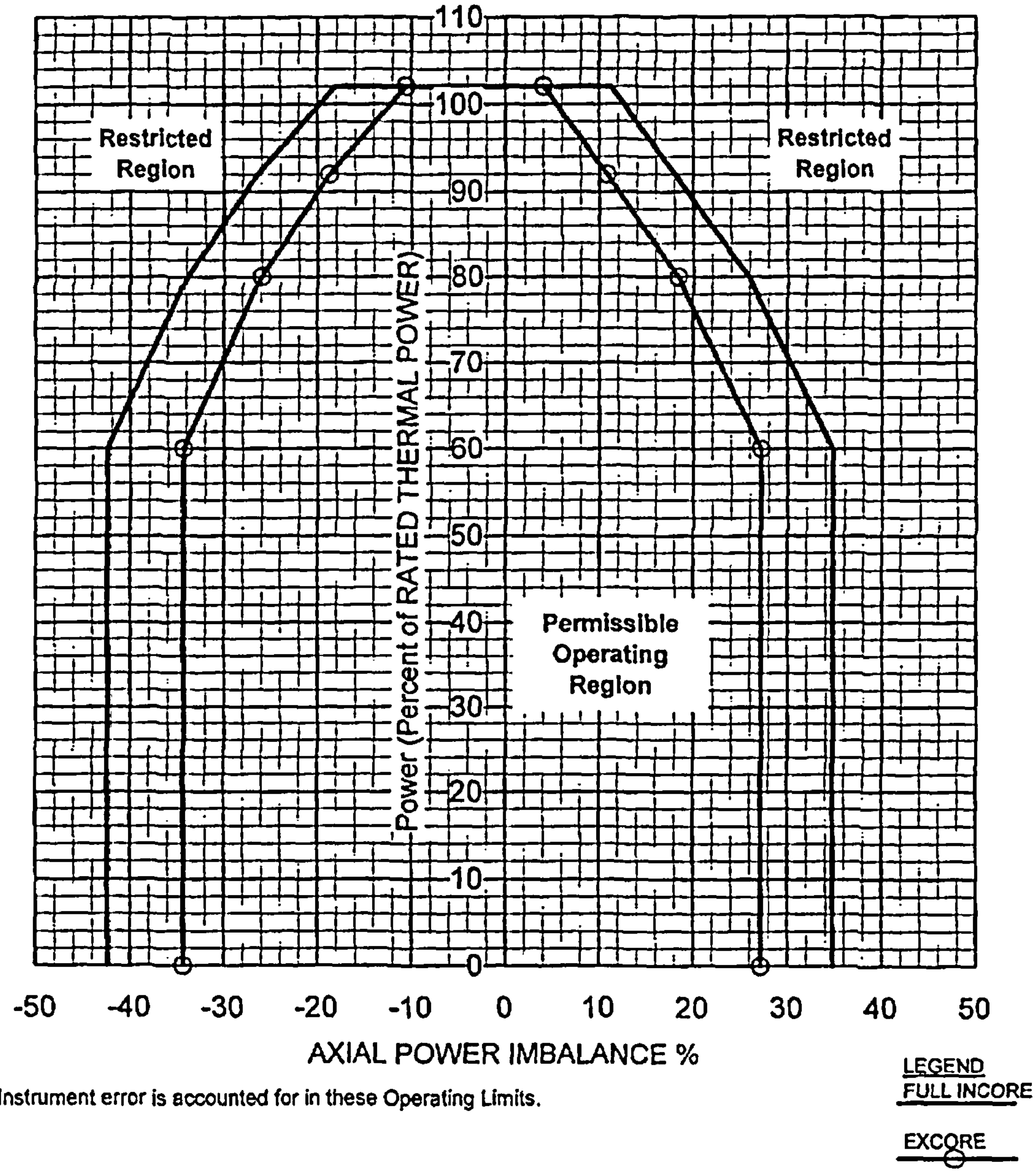


Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-8

Figure AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 654 ±10 EFPD, Four RC Pumps -- **2772 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specification 3.2.1



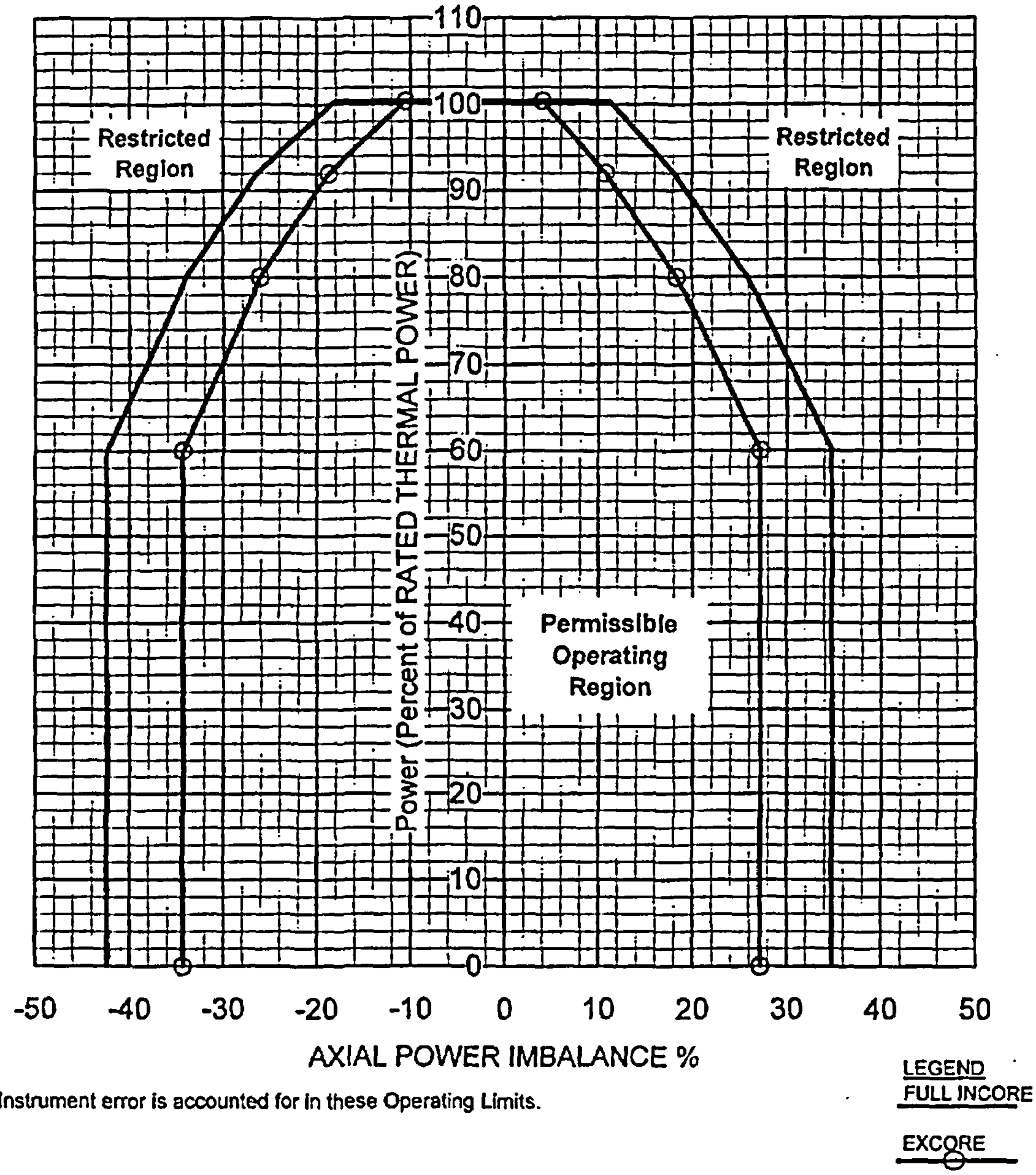
Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-8A

Rev. 1
3/03

Figure AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 654±10 EFPD, Four RC Pumps – **2817 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specification 3.2.1



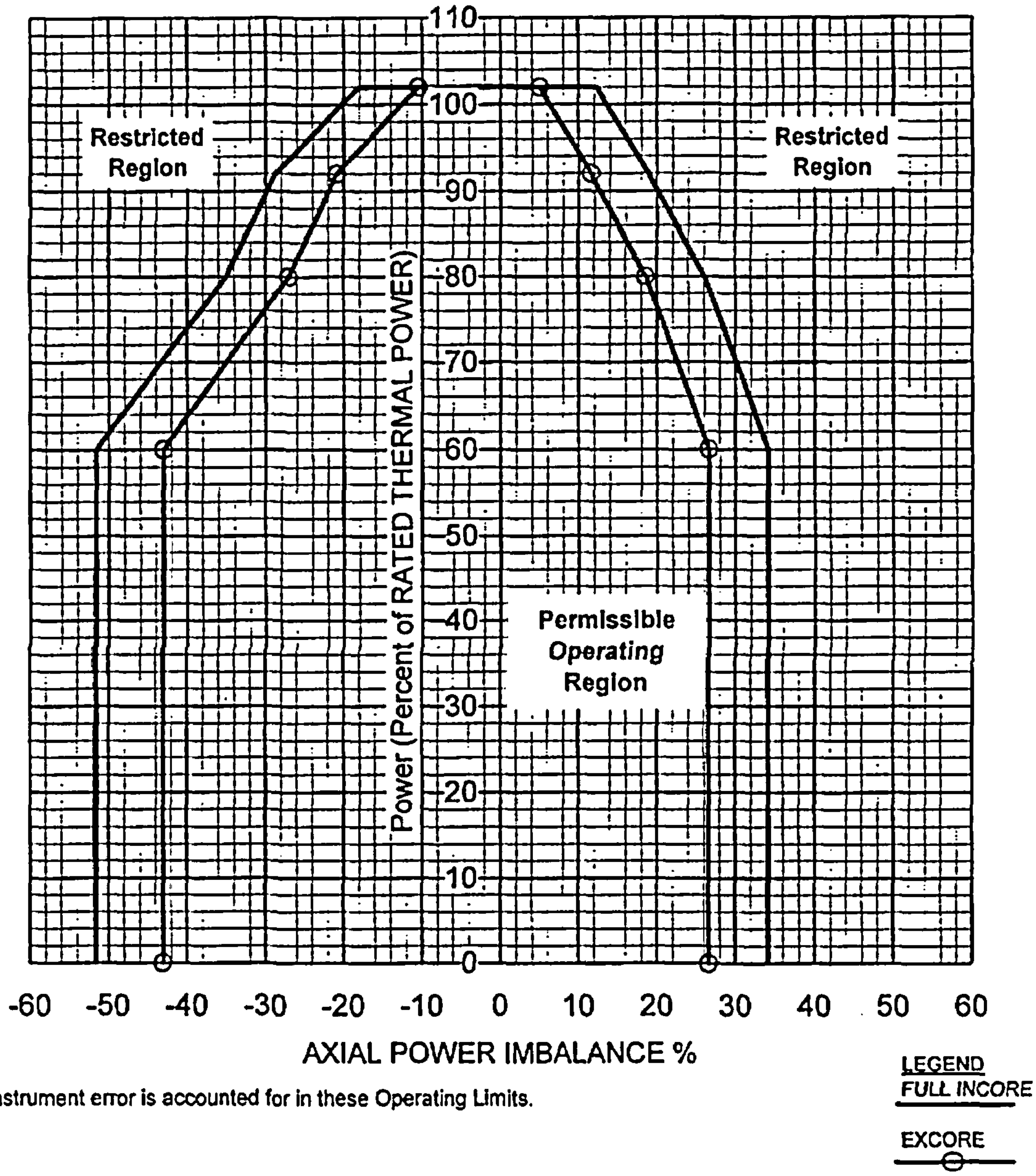
Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-9

Rev. 1
3/03

Figure AXIAL POWER IMBALANCE Operating Limits
After 654 ±10 EFPD, Four RC Pumps -- **2772 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by
Technical Specification 3.2.1



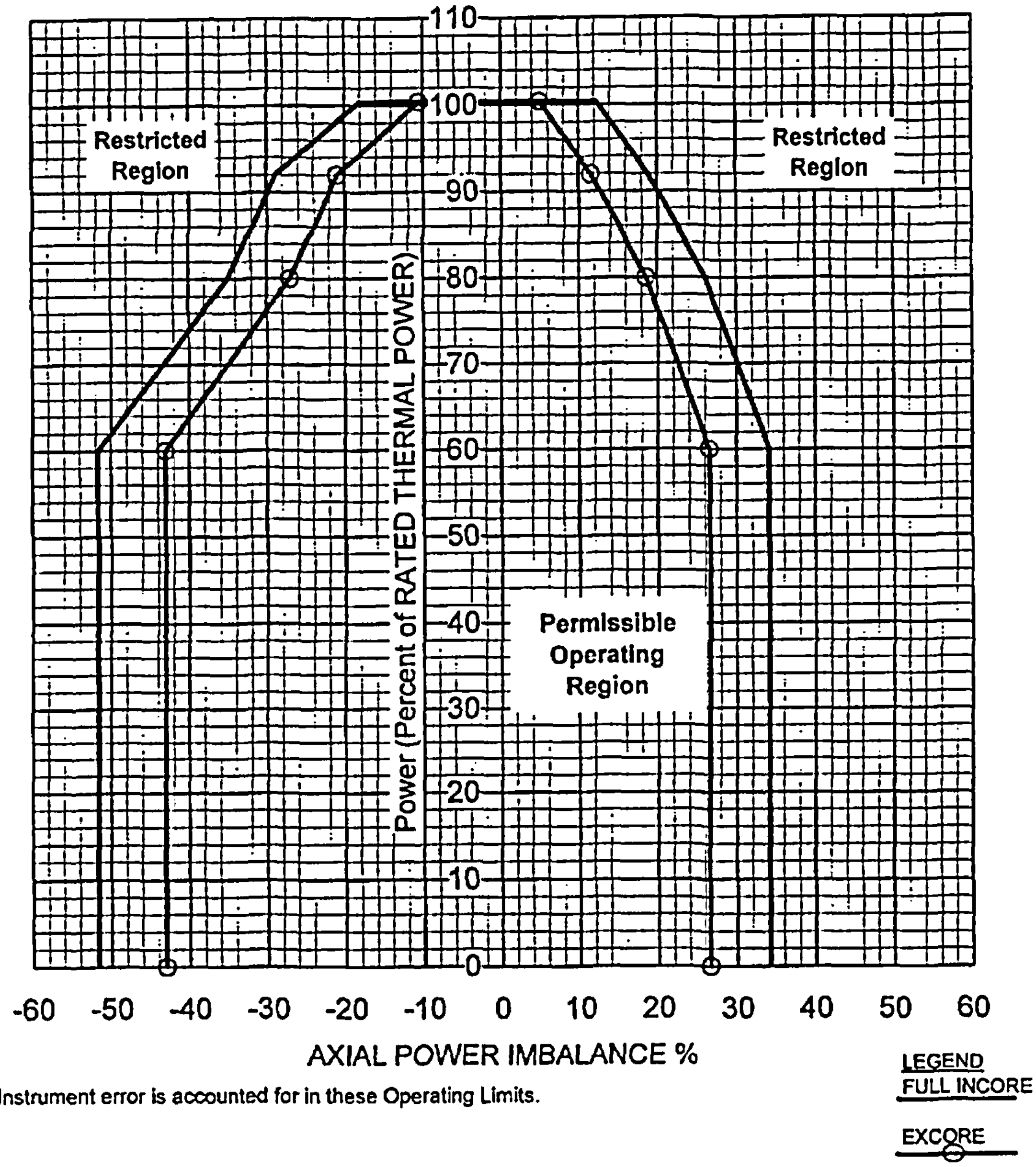
Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-9A

Rev. 1
3/03

Figure AXIAL POWER IMBALANCE Operating Limits
After 654 ±10 EFPD, Four RC Pumps -- **2817 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by
Technical Specification 3.2.1



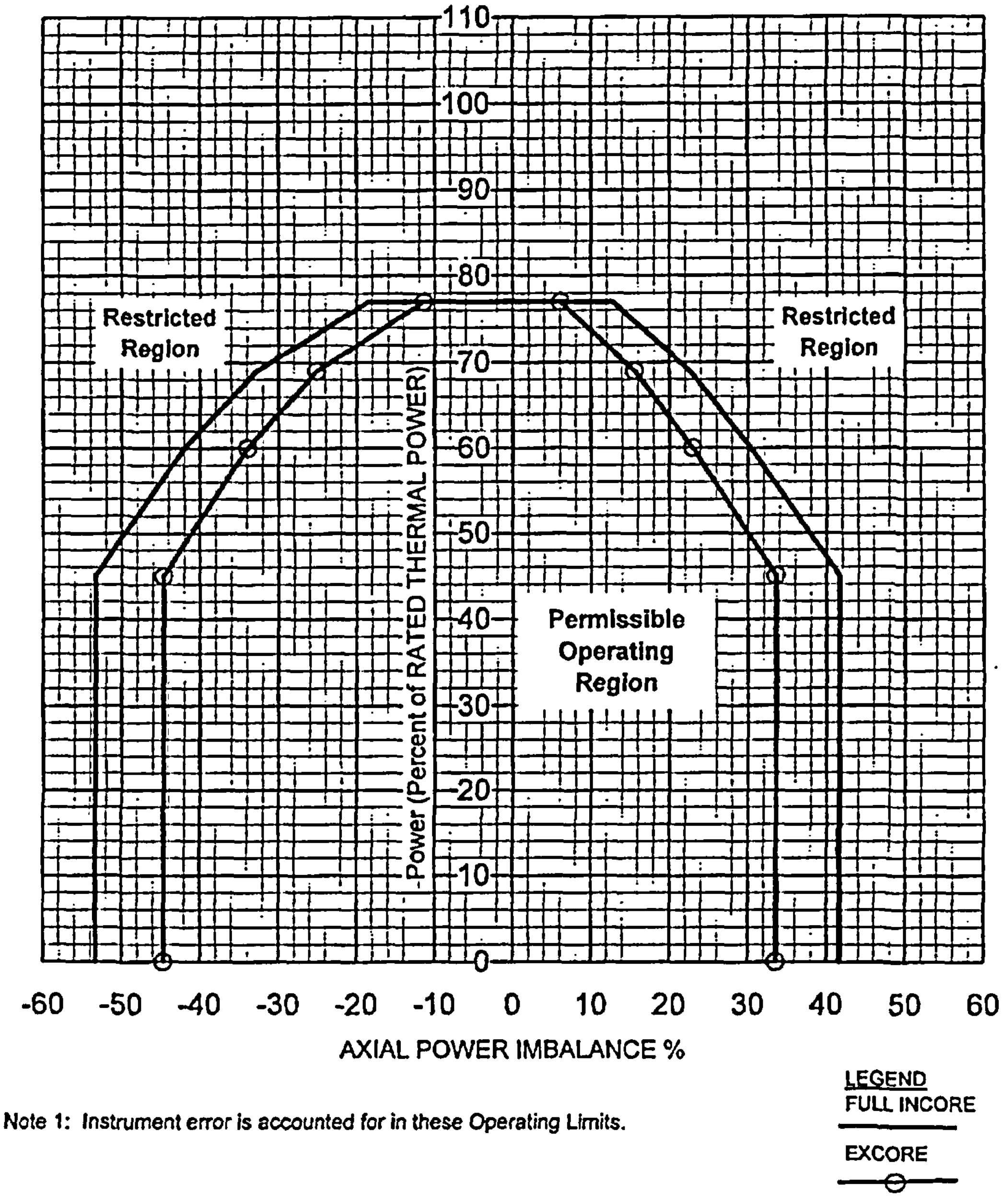
Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-10

Rev. 1
3/03

Figure AXIAL POWER IMBALANCE Operating Limits
0 to 300 ±10 EFPD, Three RC Pumps -- **2772 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by
Technical Specification 3.2.1



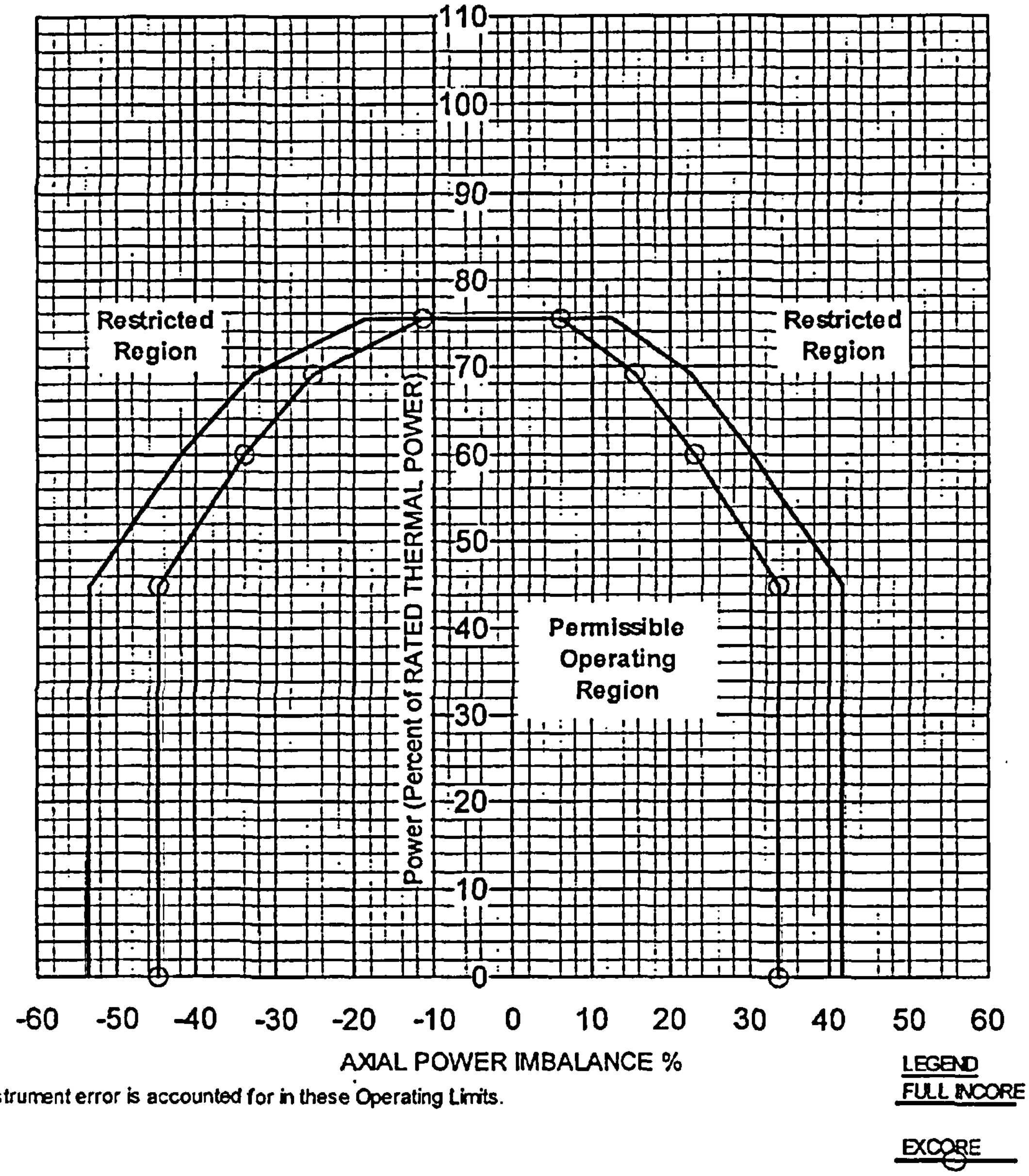
Note 1: *Instrument error is accounted for in these Operating Limits.*

Figure 8-10A

Rev. 1
3/03

Figure AXIAL POWER IMBALANCE Operating Limits
0 to 300 ±10 EFPD, Three RC Pumps – 2817 MWt RTP
Davis-Besse 1, Cycle 14

This Figure is referred to by
Technical Specification 3.2.1



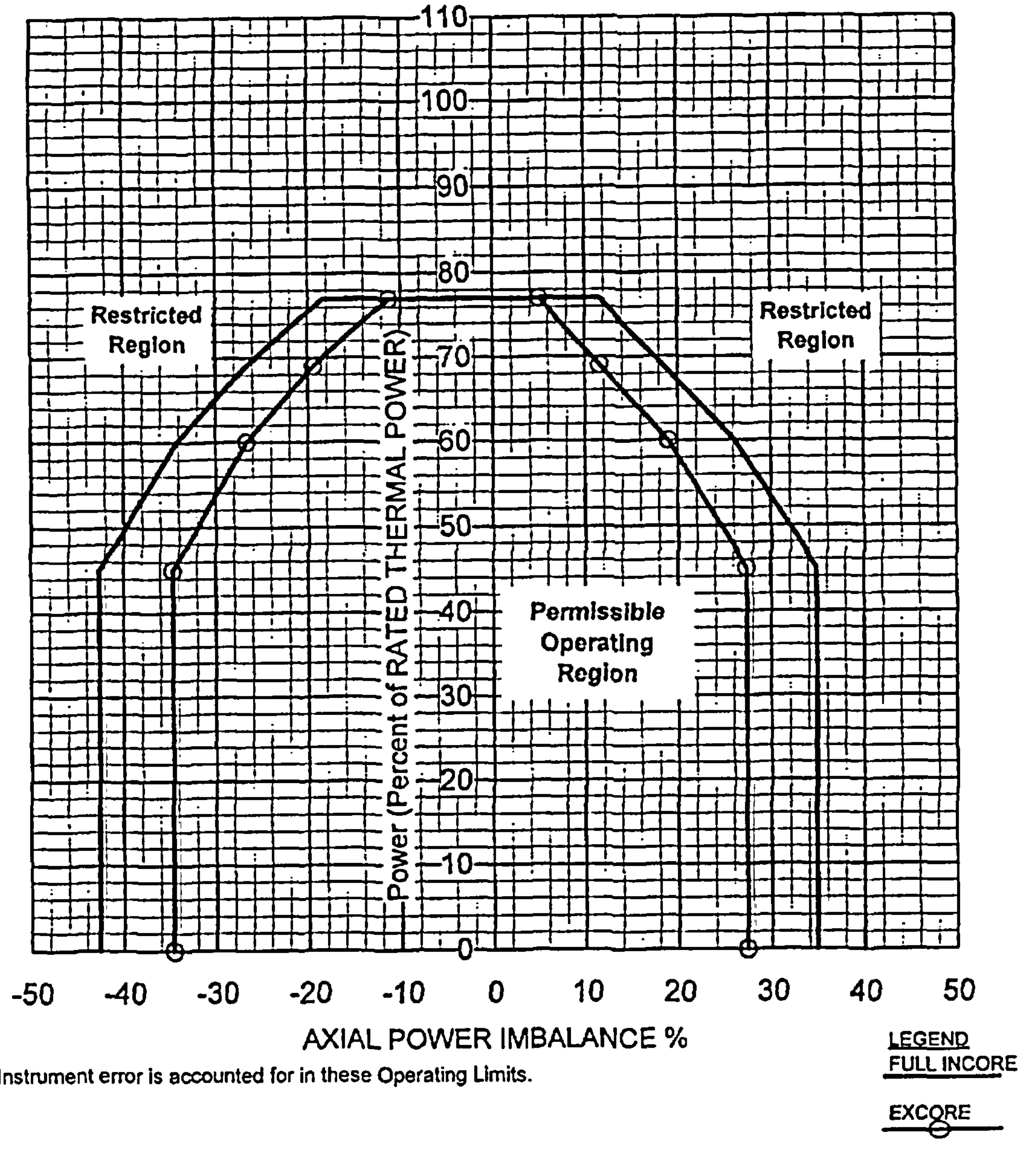
Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-11

Rev. 1
3/03

Figure AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 654 ±10 EFPD, Three RC Pumps – **2772 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by
Technical Specification 3.2.1



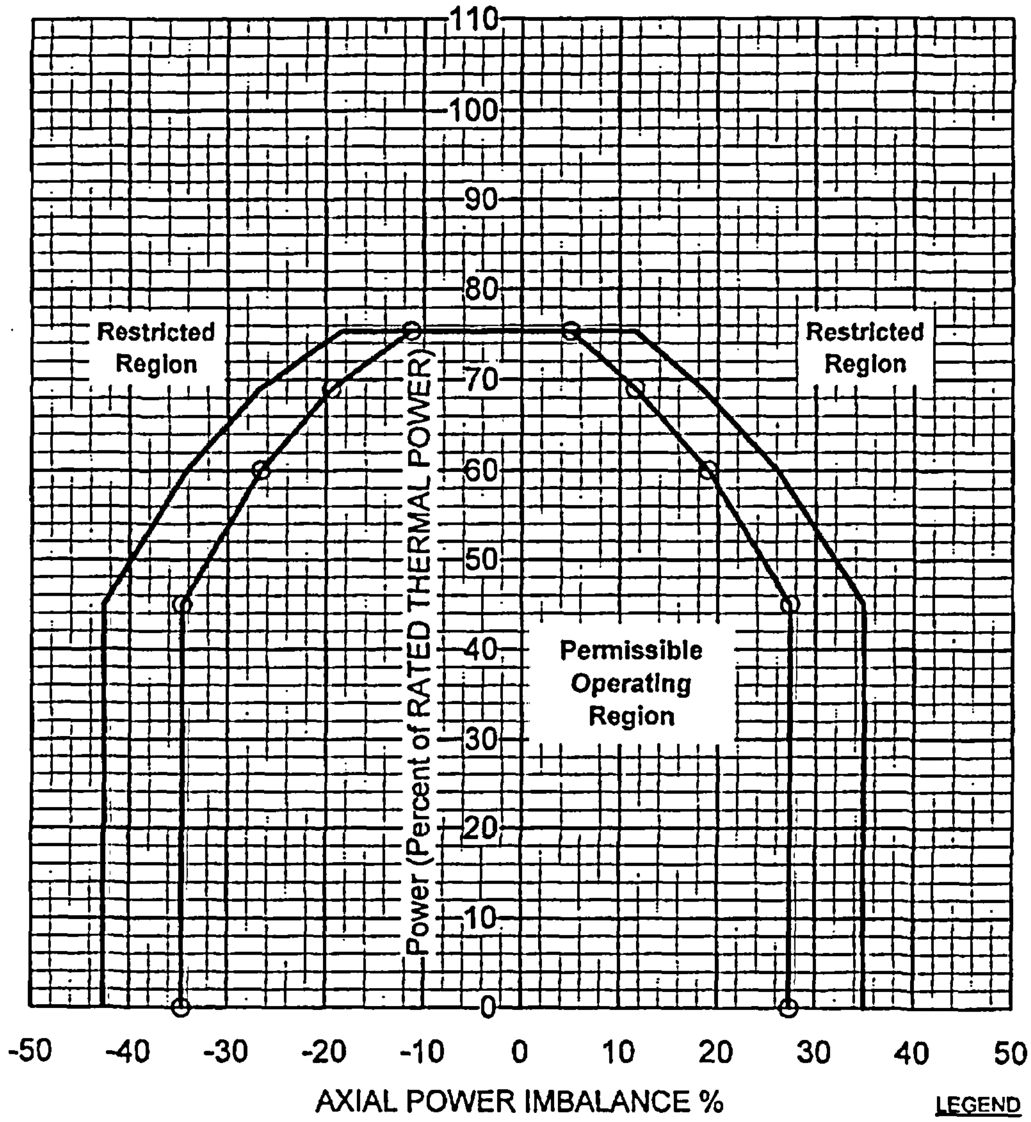
Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-11A

Rev. 1
3/03

Figure AXIAL POWER IMBALANCE Operating Limits
300 ±10 to 654 ±10 EFPD, Three RC Pumps – **2817 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specification 3.2.1



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

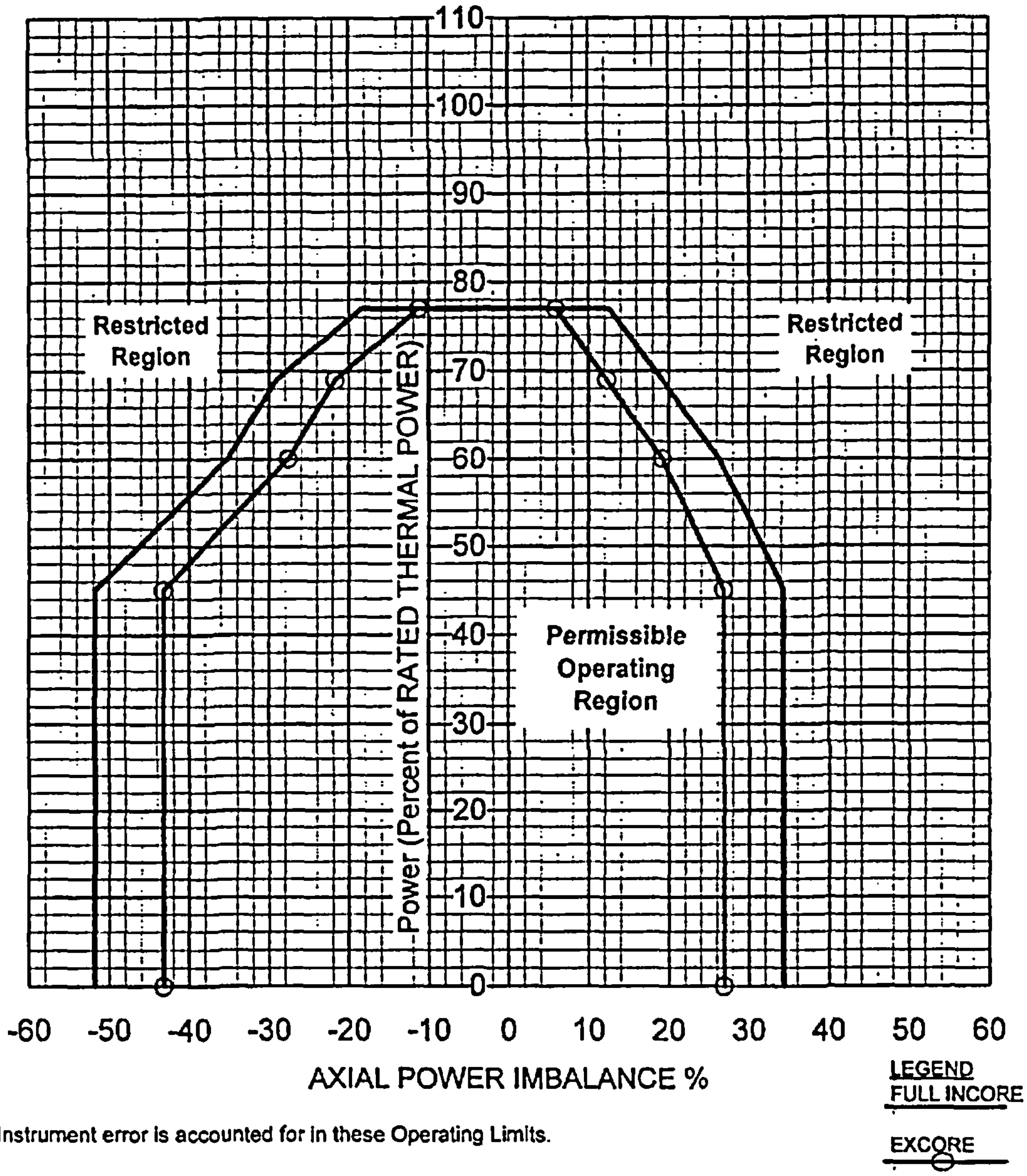
LEGEND
FULL INCORE
EXCORE

Figure 8-12

Rev. 1
3/03

Figure AXIAL POWER IMBALANCE Operating Limits
After 654 ±10 EFPD, Three RC Pumps – **2772 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specification 3.2.1



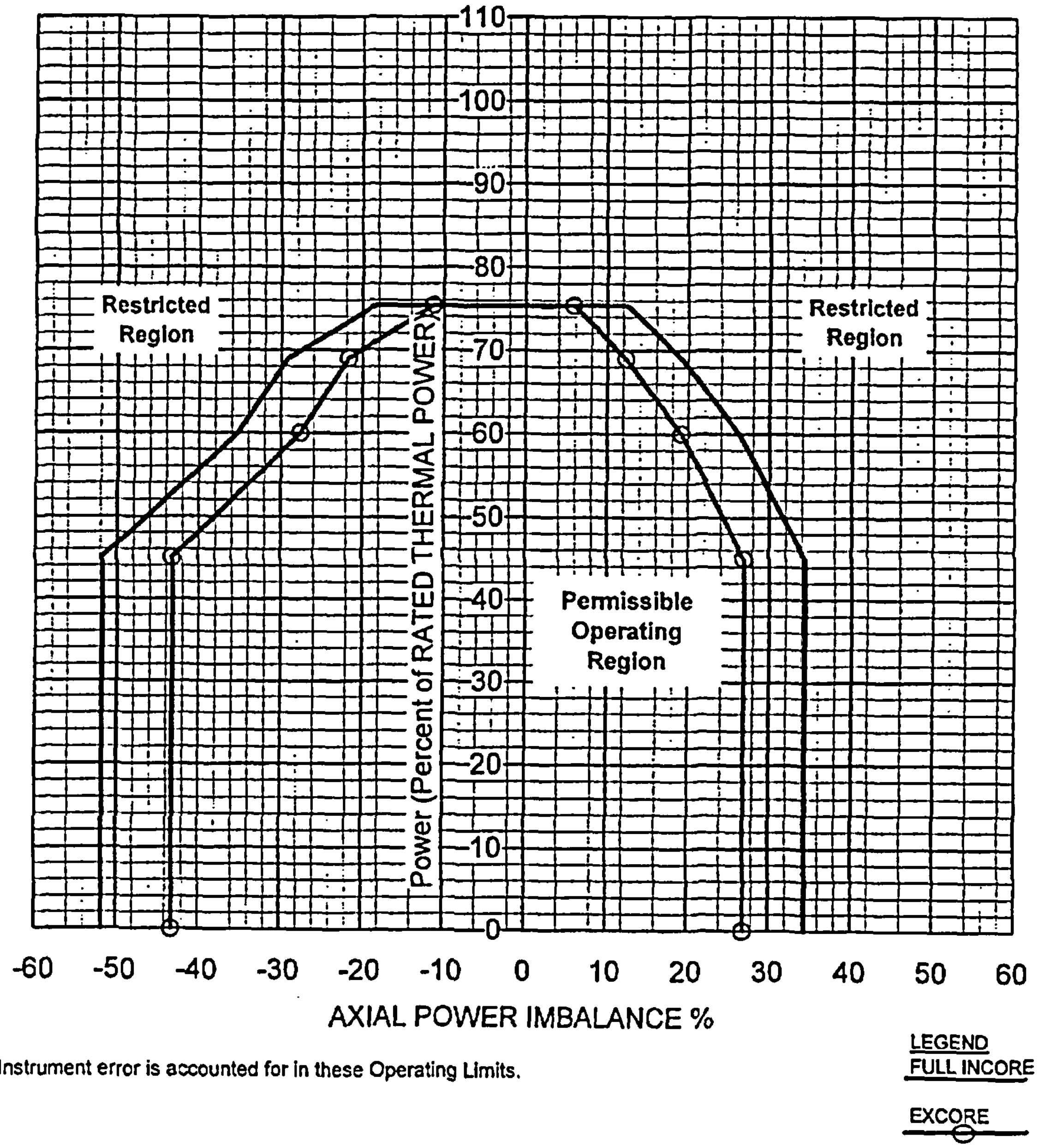
Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-12A

Rev. 1
3/03

Figure AXIAL POWER IMBALANCE Operating Limits
After 654 ±10 EFPD, Three RC Pumps -- **2817 MWt RTP**
Davis-Besse 1, Cycle 14

This Figure is referred to by Technical Specification 3.2.1

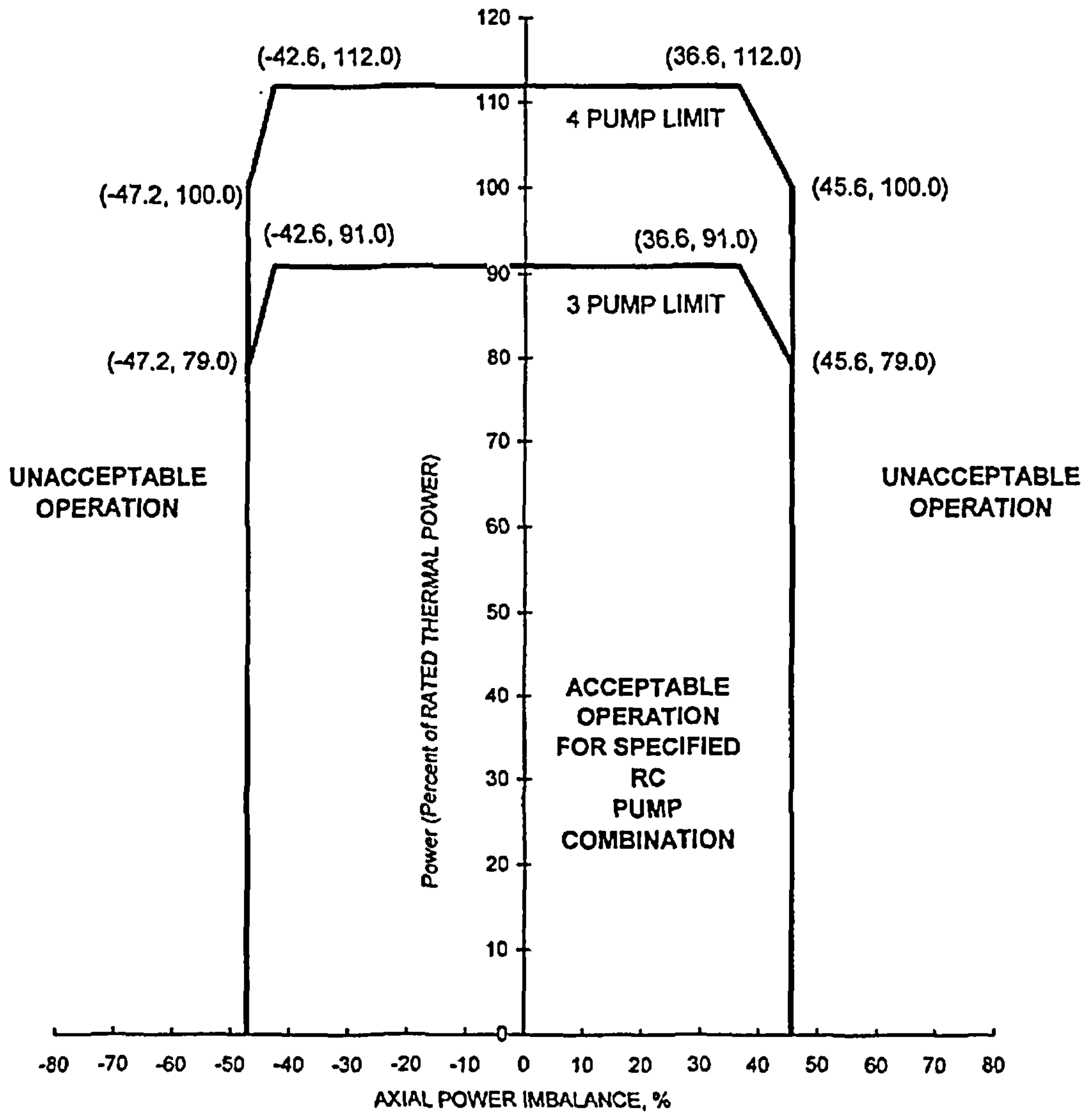


Note 1: Instrument error is accounted for in these Operating Limits.

Figure 8-13

Figure AXIAL POWER IMBALANCE Protective Limits
2772 MWt RTP

This Figure is referred to by
Technical Specification 2.1.2

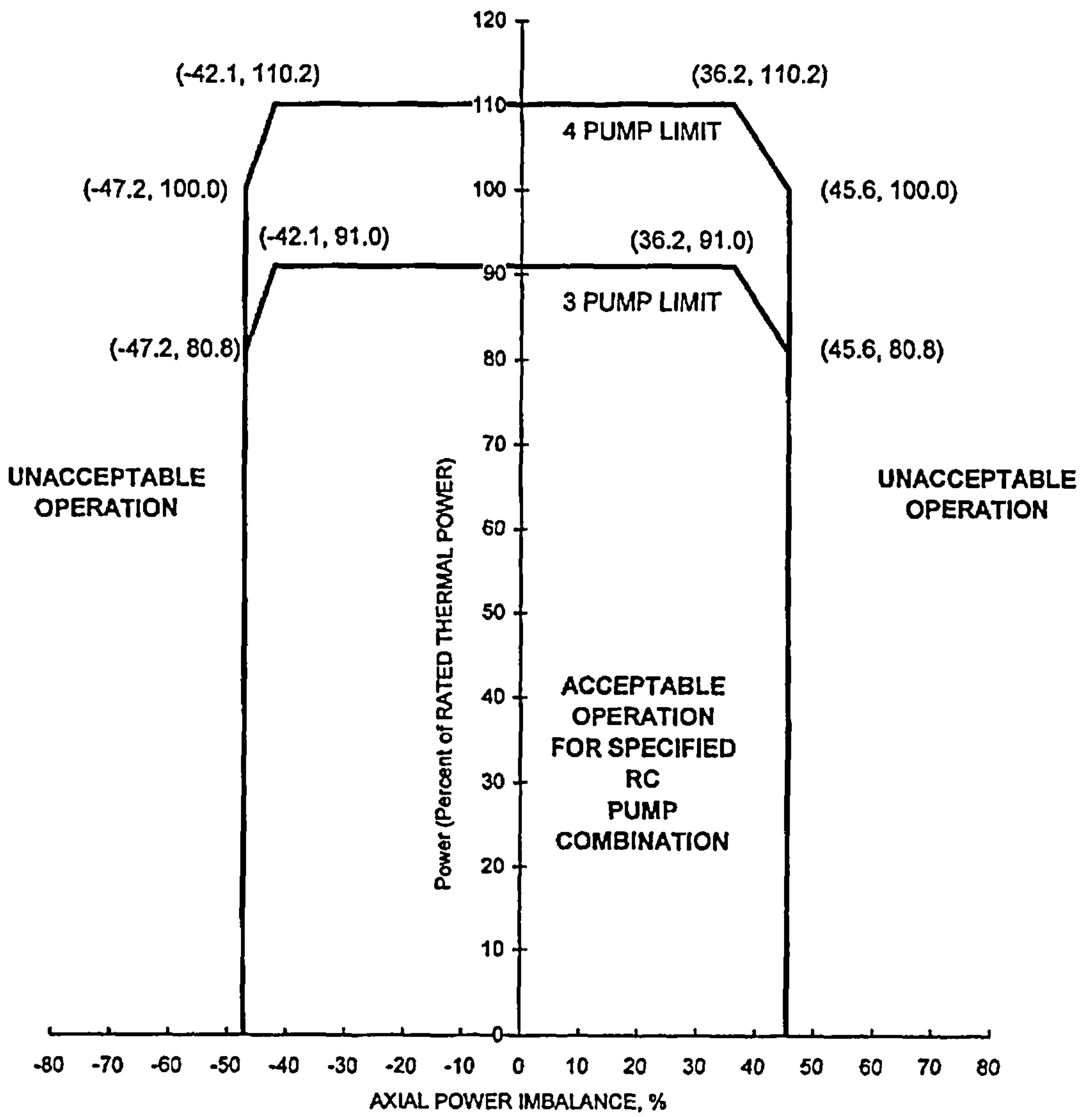


Pumps Operating	Reactor Coolant Flow, gpm	Required Measured Flow to Ensure Compliance, gpm
4	380,000	389,500
3	283,860	290,957

Figure 8-13A

Figure AXIAL POWER IMBALANCE Protective Limits
2817 MWt RTP

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 - 2772 MWt RTP

This Figure is referred to by
Technical Specification 2.2.1

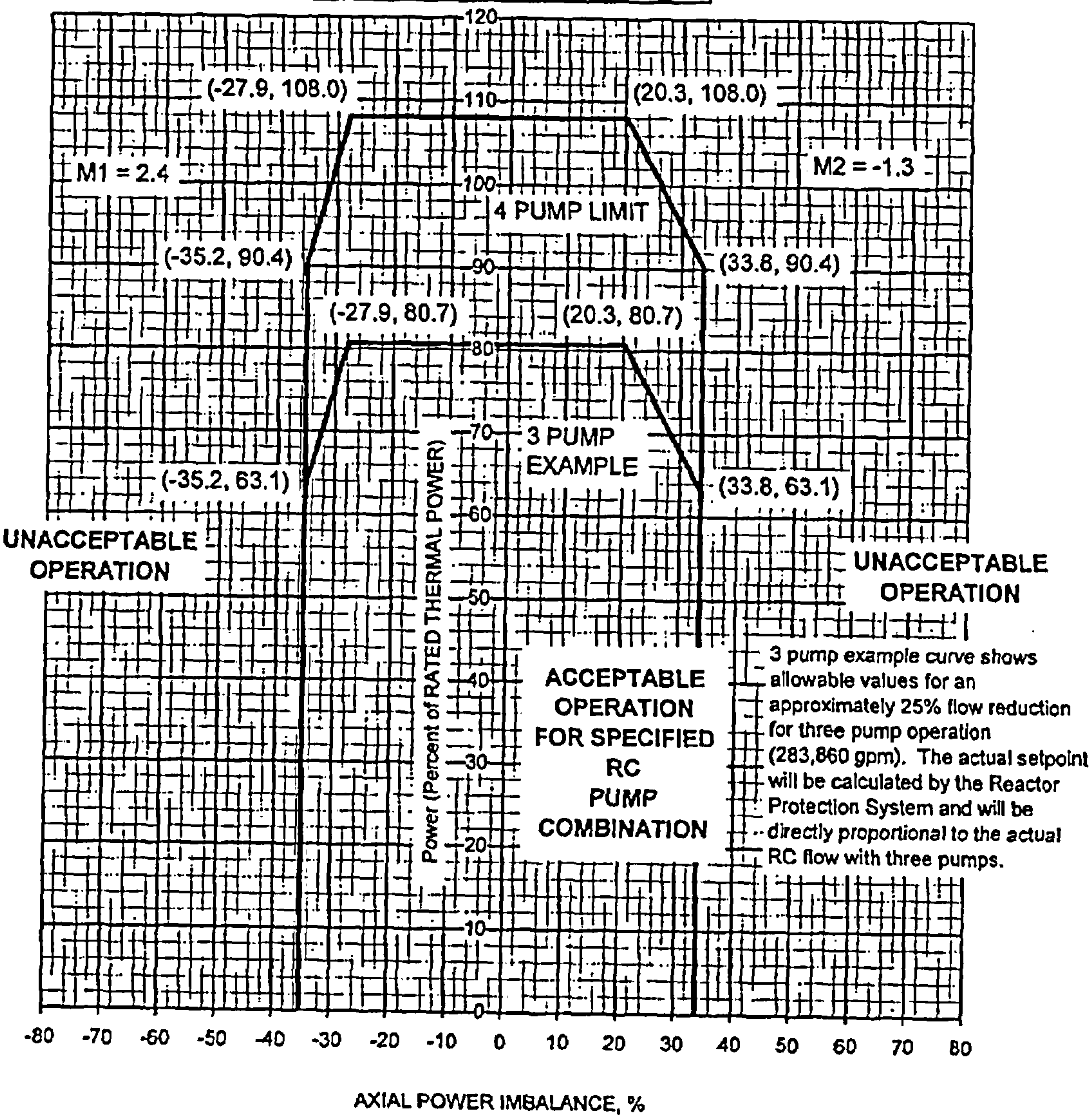


Figure 8-14A

Figure

Flux--ΔFlux-Flow
(or Power/Imbalance/Flow)
Allowable Values - 2817 MWt RTP

This Figure is referred to by
Technical Specification 2.2.1

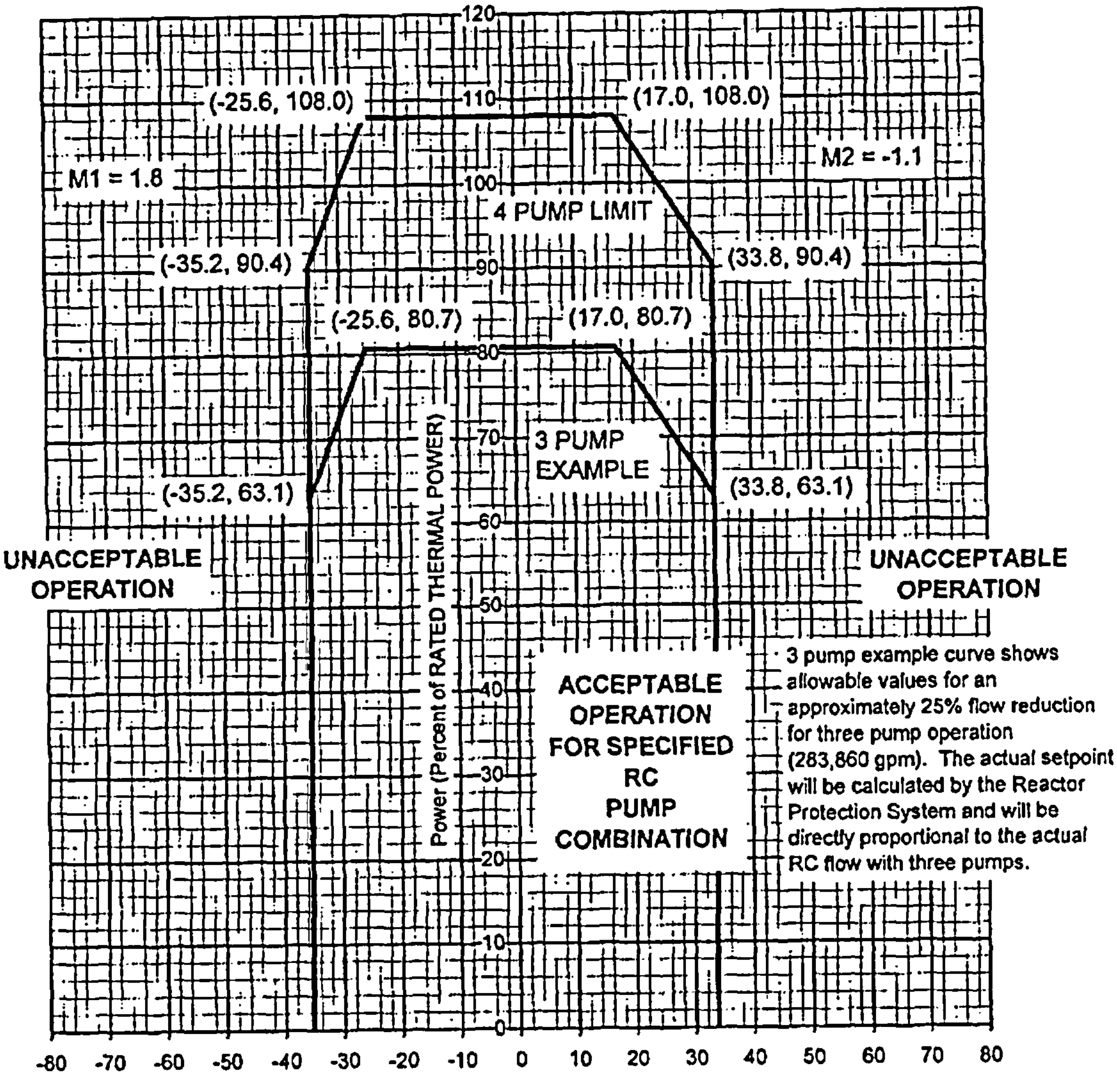


Table 8-1

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-14			
	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)	-3.83×10^{-4} Δk/k/°F

Table 8-3

Table Power To Melt Limits

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

	Batch 9H	Batch 14	Batch 15	Batch 16
Fuel Assembly Type	Mark-B8A	Mark-B10M	Mark-B10K	Mark-B12
Minimum linear heat rate to melt, kW/ft	20.5	22.3 (20.8)[a] (19.9)[b]	22.1 (21.1)[c] (20.7)[d] (19.3)[e]	22.1 (20.3)[f] (19.3)[g]

[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
[f] Limit for 4 wt% Gd rods - Batch 16
[g] Limit for 8 wt% Gd rods - Batch 16

Table 8-4

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

This Table is referred to by Technical Specification 3.2.2

Heat Flux Hot Channel Factor F_Q **2772 MWt RTP**

F_Q shall be limited by the following relationships:

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

$LHR^{ALLOW}(Bu)$: See Tables below

LHR^{AVG} = 6.3095 kW/ft for Mark-B8A fuel

LHR^{AVG} = 6.4201 kW/ft for Mark-B10M fuel

LHR^{AVG} = 6.3183 kW/ft for Mark-B10K fuel

LHR^{AVG} = 6.3183 kW/ft for Mark-B12 fuel

P = ratio of THERMAL POWER/RATED THERMAL POWER

Bu = Fuel Burnup (MWd/mtU)

Batch 9H (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	16.1	16.1	12.0	10.2
2	15.8	15.8	12.0	10.2
3	15.0	15.0	12.0	10.2
4	15.0	15.0	12.0	10.2
5	15.4	15.4	12.0	10.2
6	15.4	15.4	12.0	10.2
7	14.6	14.6	12.0	10.2
8	14.3	14.3	12.0	10.2

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

Table 8-4, continued

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

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	14.7	14.4	14.1	13.5
2	17.5	16.7	14.7	14.4	14.1	13.5
3	17.0	15.6	14.6	14.4	14.1	13.5
4	16.6	15.3	14.4	14.4	14.1	13.5
5	16.0	15.3	14.2	14.2	14.1	13.5
6	15.3	15.3	13.8	13.7	13.6	13.5
7	14.7	14.7	13.3	13.2	13.1	13.0
8	14.5	14.5	13.1	13.0	12.9	12.8

Batch 16 (Mark-B12) LHR^{ALLOW} kW/ft[a]

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	14.7	14.4	14.1	13.5
2	17.5	16.7	14.7	14.4	14.1	13.5
3	17.0	15.6	14.6	14.4	14.1	13.5
4	16.6	15.3	14.4	14.4	14.1	13.5
5	16.0	15.3	14.2	14.2	14.1	13.5
6	15.3	15.3	13.8	13.7	13.6	13.5
7	14.7	14.7	13.3	13.2	13.1	13.0
8	14.5	14.5	13.1	13.0	12.9	12.8

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

Table 8-4A

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

This Table is referred to by Technical Specification 3.2.2

Heat Flux Hot Channel Factor F_Q **2817 MWt RTP**

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} = 6.4119 kW/ft for Mark-B8A fuel

LHR^{AVG} = 6.5243 kW/ft for Mark-B10M fuel

LHR^{AVG} = 6.4209 kW/ft for Mark-B10K fuel

LHR^{AVG} = 6.4209 kW/ft for Mark-B12 fuel

P = ratio of THERMAL POWER/RATED THERMAL POWER

Bu = Fuel Burnup (MWd/mtU)

Batch 9H (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	16.1	16.1	12.0	10.2
2	15.8	15.8	12.0	10.2
3	15.0	15.0	12.0	10.2
4	15.0	15.0	12.0	10.2
5	15.4	15.4	12.0	10.2
6	15.4	15.4	12.0	10.2
7	14.6	14.6	12.0	10.2
8	14.3	14.3	12.0	10.2

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

Table 8-4A continued

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

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	14.7	14.4	14.1	13.5
2	17.5	16.7	14.7	14.4	14.1	13.5
3	17.0	15.6	14.6	14.4	14.1	13.5
4	16.6	15.3	14.4	14.4	14.1	13.5
5	16.0	15.3	14.2	14.2	14.1	13.5
6	15.3	15.3	13.8	13.7	13.6	13.5
7	14.7	14.7	13.3	13.2	13.1	13.0
8	14.5	14.5	13.1	13.0	12.9	12.8

Batch 16 (Mark-B12) LHR^{ALLOW} kW/ft[(a)]

Axial Segment	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
1	17.6	16.8	14.7	14.4	14.1	13.5
2	17.5	16.7	14.7	14.4	14.1	13.5
3	17.0	15.6	14.6	14.4	14.1	13.5
4	16.6	15.3	14.4	14.4	14.1	13.5
5	16.0	15.3	14.2	14.2	14.1	13.5
6	15.3	15.3	13.8	13.7	13.6	13.5
7	14.7	14.7	13.3	13.2	13.1	13.0
8	14.5	14.5	13.1	13.0	12.9	12.8

(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

Heat Flux Hot Channel Factor F_Q **2772 MWt RTP**

F_Q shall be limited by the following relationships:

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

$LHR^{ALLOW}(Bu)$: See the Tables below

LHR^{AVG} = 6.3095 kW/ft for Mark-B8A fuel

LHR^{AVG} = 6.4201 kW/ft for Mark-B10M fuel

LHR^{AVG} = 6.3183 kW/ft for Mark-B10K fuel

LHR^{AVG} = 6.3183 kW/ft for Mark-B12 fuel

P = ratio of THERMAL POWER/RATED THERMAL POWER

Bu = Fuel Burnup (MWd/mtU)

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

Core Elevation (feet)	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU	60,000 MWd/mtU
0.000	16.3	16.3	12.0	10.2
2.506	15.9	15.9	12.0	10.2
4.264	15.1	15.1	12.0	10.2
6.021	15.5	15.5	12.0	10.2
7.779	16.0	16.0	12.0	10.2
9.536	15.4	15.4	12.0	10.2
12.000	14.3	14.3	12.0	10.2

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

Core Elevation (feet)	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

Table 8-5, continued

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

Core Elevation (feet)	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	14.7	14.4	14.1	13.5
2.506	17.6	16.8	14.7	14.4	14.1	13.5
4.264	17.1	15.7	14.7	14.4	14.1	13.5
6.021	16.6	15.3	14.4	14.4	14.1	13.5
7.779	16.0	15.8	14.2	14.2	14.1	13.5
9.536	15.3	15.3	13.8	13.7	13.6	13.5
12.000	14.5	14.5	13.1	13.0	12.9	12.8

Batch 16 (Mark-B12) LHR^{ALLOW} kW/ft [a]

Core Elevation (feet)	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	14.7	14.4	14.1	13.5
2.506	17.6	16.8	14.7	14.4	14.1	13.5
4.264	17.1	15.7	14.7	14.4	14.1	13.5
6.021	16.6	15.3	14.4	14.4	14.1	13.5
7.779	16.0	15.8	14.2	14.2	14.1	13.5
9.536	15.3	15.3	13.8	13.7	13.6	13.5
12.000	14.5	14.5	13.1	13.0	12.9	12.8

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

Table 8-5A

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

This Table is referred to by Technical Specification 3.2.2

Heat Flux Hot Channel Factor F_Q **2817 MWt RTP**

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 the Tables below

LHR^{AVG} = 6.4119 kW/ft for Mark-B8A fuel

LHR^{AVG} = 6.5243 kW/ft for Mark-B10M fuel

LHR^{AVG} = 6.4209 kW/ft for Mark-B10K fuel

LHR^{AVG} = 6.4209 kW/ft for Mark-B12 fuel

P = ratio of THERMAL POWER/RATED THERMAL POWER

Bu = Fuel Burnup (MWd/mtU)

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

Core Elevation (feet)	0 MWd/mtU	24,500 MWd/mtU	52,000 MWd/mtU	60,000 MWd/mtU
0.000	16.3	16.3	12.0	10.2
2.506	15.9	15.9	12.0	10.2
4.264	15.1	15.1	12.0	10.2
6.021	15.5	15.5	12.0	10.2
7.779	16.0	16.0	12.0	10.2
9.536	15.4	15.4	12.0	10.2
12.000	14.3	14.3	12.0	10.2

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

Core Elevation (feet)	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

Table 8-5A continued

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

Core Elevation (feet)	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	14.7	14.4	14.1	13.5
2.506	17.6	16.8	14.7	14.4	14.1	13.5
4.264	17.1	15.7	14.7	14.4	14.1	13.5
6.021	16.6	15.3	14.4	14.4	14.1	13.5
7.779	16.0	15.8	14.2	14.2	14.1	13.5
9.536	15.3	15.3	13.8	13.7	13.6	13.5
12.000	14.5	14.5	13.1	13.0	12.9	12.8

Batch 16 (Mark-B12) LHR^{ALLOW} kW/ft [a]

Core Elevation (feet)	0 MWd/mtU	35,000 MWd/mtU	58,000 MWd/mtU	59,000 MWd/mtU	60,000 MWd/mtU	62,000 MWd/mtU
0.000	17.6	16.8	14.7	14.4	14.1	13.5
2.506	17.6	16.8	14.7	14.4	14.1	13.5
4.264	17.1	15.7	14.7	14.4	14.1	13.5
6.021	16.6	15.3	14.4	14.4	14.1	13.5
7.779	16.0	15.8	14.2	14.2	14.1	13.5
9.536	15.3	15.3	13.8	13.7	13.6	13.5
12.000	14.5	14.5	13.1	13.0	12.9	12.8

[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

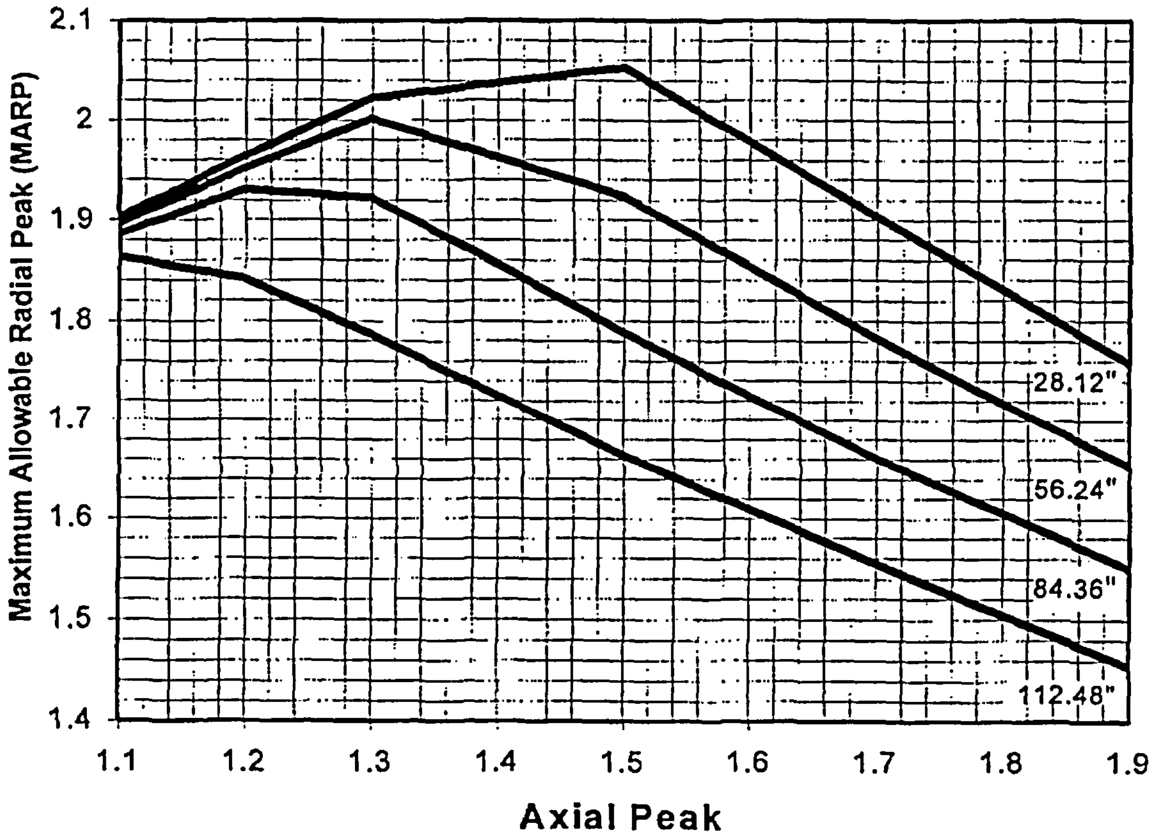
P = THERMAL POWER/RATED THERMAL POWER and P $\leq$ 1.0

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.

9.0 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. 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 $\pm 0.2 \times 10^{-2}$ %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:

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

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

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

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:

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

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:

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

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

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

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 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 ANP 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 recommended before continued testing at a higher power plateau. This evaluation is similar to that performed to address failure of an acceptance criterion.

10.0 REFERENCES

1. Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, BAW-10179P-A, Rev. 4, Framatome ANP, Lynchburg, Virginia, dated August 2001.

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

3. Davis-Besse Nuclear Power Station Unit 1, Cycle 13 Reload Report, BAW-2368 , Framatome Cogema Fuels, Lynchburg, Virginia, dated March 2000.

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

5. 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.

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

7. Extended Burnup Evaluation, BAW-10186P-A, Rev. 1, Framatome Cogema Fuels, Lynchburg, Virginia, dated April 2000.

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

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

10. 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).

11. Mark-C FA LOCA-Seismic Analyses, BAW-10133P-A, Rev. 1, Addendum 1 and Addendum 2, Framatome Cogema Fuels, Lynchburg, Virginia, dated October 2000.

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

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

14. Statistical Core Design for B&W-Designed 177-FA Plants, BAW-10187P-A, B&W Fuel Company,

Lynchburg, Virginia, dated March 1994.

15. Evaluation of Replacement Rods in BWFC Fuel Assemblies, BAW-2149-A, B&W Fuel Company, Lynchburg, Virginia, September 1993.

16. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Ed., USNRC.

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

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

19. RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, BAW-10164P-A, Rev. 4, Framatome Technologies, Inc., Lynchburg, Virginia, November 2002.

Docket Number 50-346
License Number NPF-3
Serial Number 2991
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 (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	DUE DATE
None	N/A