THREE MILE ISLAND UNIT 1

CYCLE 2 RELOAD REPORT (Based on a Cycle 1 Burnup of 440$\mp$ 10 EFPD)

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1487 227

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CONTENTS

1487 228

CONTENTS (Continued)

List of Tables

List of Figures

1487 229

List of Figures (Continued)

1487 230

1. INTRODUCTION AND SUMMARY

This report justifies the operation of the Three Mile Island-Unit 1 Nuclear Station, Cycle 2 at the rated core power of 2535 MWt. Included are the required analyses, as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling" June 1975.

To support Cycle 2 operation of the Three Mile Island Unit 1 Nuclear Station, this report employs analytical techniques and design bases established in reports which were previously submitted and accepted by the USNRC (see References).

A brief summary of Cycle 1 and 2 reactor parameters that are related to power capability is included in this report. All of the accidents analyzed in the FSAR have been reviewed for Cycle 2 operation. In those cases where Cycle 2 characteristics proved to be conservative with respect to those analyzed for Cycle 1 operation, no new analysis was performed.

The Technical Specifications have been reviewed and the modifications required for Cycle 2 operation are justified in this report.

Based on the analyses performed, which take into account the postulated effects of fuel densification and the Final Acceptance Criteria, it has been concluded that Three Mile Island-Unit 1, Cycle 2 can be safely operated at the rated core power level of 2535 MWt.

1487 231

2. OPERATING HISTORY

Three Mile Island-Unit 1 achieved initial criticality on June 5, 1974, and power generation commenced on June 15, 1974. The 100% power level of 2535 MWt was reached on August 3, 1974. A control rod interchange was performed at 256 effective full power days (EFPD). The fuel cycle is scheduled for completion February 14, 1976 after 440 $\pm$ 10 EFPD. No operating anomalies occurred during the first cycle which would adversely affect the fuel performance during the second cycle.

Operation of Cycle 2 is scheduled to begin in late April 1976. The design cycle length is 296 EFPD and no control rod interchanges are planned.

1487 232

3. GENERAL DESCRIPTION

The TMI-1 reactor core is described in detail in Section 3 of the Three Mile Island-Unit 1 Nuclear Station, Final Safety Analysis Report (Reference 1).

The Cycle 2 core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel rods in batches 2 and 3 of Core I have an undensified nominal active length of 144 inches while batch 4 has an undensified length of 142.6 inches. All fuel assemblies in Cycle 2 maintain a constant nominal fuel loading of 463.6 kg of uranium. The cladding is cold-worked Zircaloy-4 with an OD of 0.430 inch and a wall thickness of 0.0265 inch. The fuel consists of dished end, cylindrical pellets of uranium dioxide which are 0.700 inch in length and 0.370 inch in diameter. (See Table 4.1-1 for additional data.)

Figure 3-1 is the core loading diagram for TMI-1, Cycle 2. The initial enrichments of batches 2 and 3 were 2.75 and 3.05 wt % ^{235}U, respectively. The 56 batch 4 assemblies are enriched to 2.64 wt % ^{235}U. All of the batch 1 assemblies will be discharged at the end of Cycle 1. The batch 2 and 3 assemblies will be shuffled to new locations. The batch 4 assemblies will occupy primarily the periphery of the core and eight locations interior to the core. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of Cycle 2.

Reactivity control is supplied by 61 full-length Ag-In-Cd control rods and soluble boron shim. In addition to the full-length control rods, eight axial power shaping rods are provided for additional control of axial power distribution. The Cycle 2 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The core locations of the total pattern (69 control rods) for Cycle 2 are identical to that of the reference cycle indicated in Section 3 of the FSAR. The group designations, however, differ between Cycle 2 and the reference cycle in order to minimize power peaking. No control rod interchange and no burnable poison rods are necessary during Cycle 2.

1487 233

The nominal system pressure is 2200 psia, and the densified nominal heat rate is 5.78 kW/ft at the rated core power of 2535 MWt. The heat rate is slightly higher than in Cycle 1 due to the shorter stack height of batch 4.

1487 234

Figure 3-1 TMI-1 Cycle 2 - Core Loading Diagram

FUEL TRANSFER CANAL →

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

XX	Batch
XX	Previous Core Location

1487 235

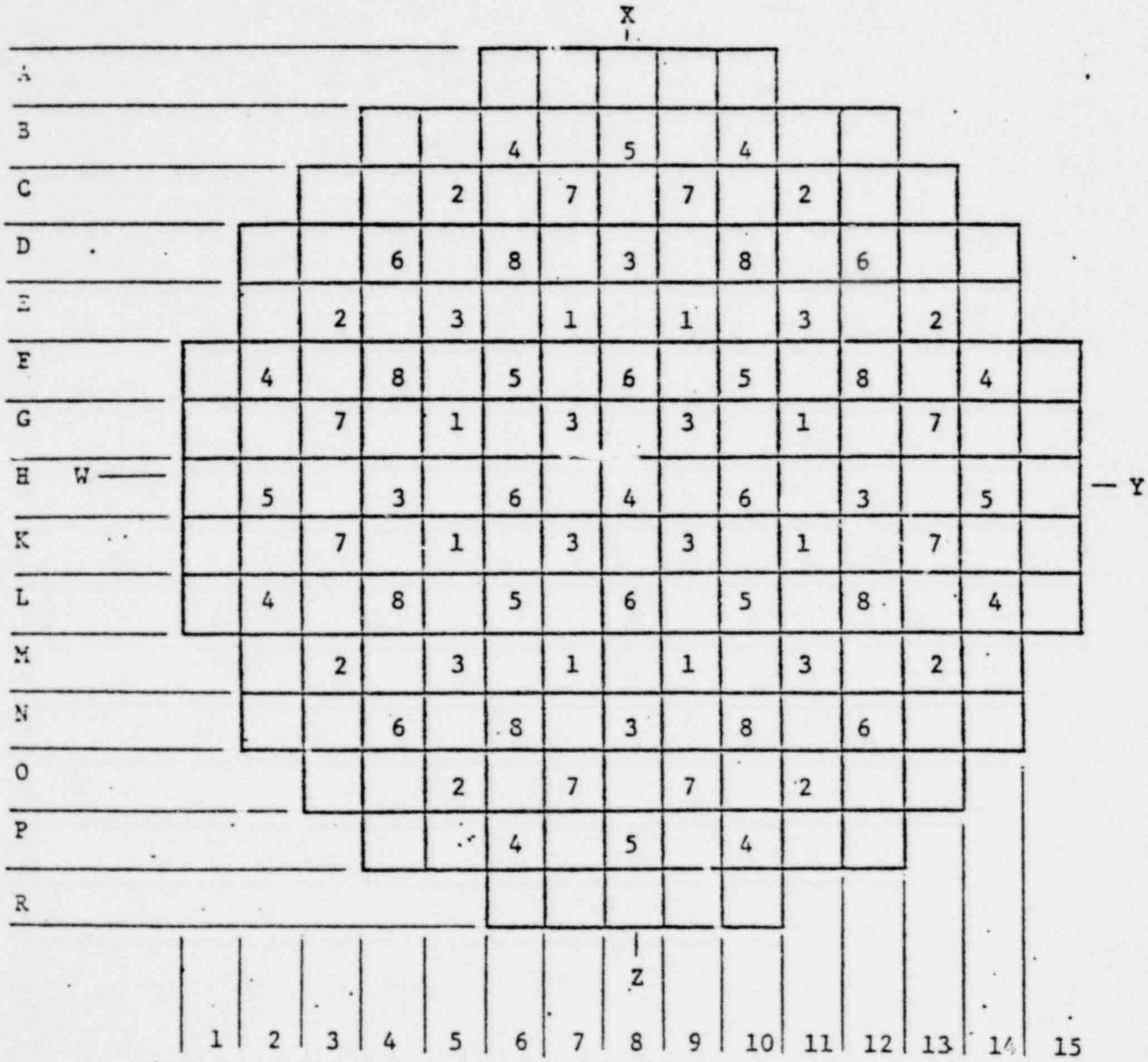
Figure 3-2 TMI-1 Enrichment and Burnup Distribution for Cycle 2

	8	9	10	11	12	13	14	15
H	2.75 11748	2.75 16541	2.75 16106	2.75 16787	3.05 11508	3.05 13815	3.05 10552	2.64 0
K		3.05 11508	3.05 11825	2.75 13318	2.64 0	2.75 14689	2.75 17987	2.64 0
L			2.75 16787	2.75 17056	3.05 11096	2.75 14515	2.64 0	2.64 0
M				3.05 8207	3.05 10132	3.05 7778	2.64 0	
N					2.75 18057	3.05 7727	2.64 0	
O						2.64 0		
P								
R								

xxx	Initial Enrichment
xxxx	BOC Burnup (MWD/MTU)

1487 236

Figure 3-3 TMI-1 Cycle 2 - Control Rod Locations



X	Group Number

Group	Number of Rods	Function
1	8	Safety
2	8	Safety
3	12	Safety
4	9	Safety
5	8	Control
6	8	Control
7	8	Control
8	8	APSR's
Total	69	

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

Pertinent fuel design parameters are listed in Table 4.1-1. All fuel assemblies are identical in concept and are mechanically interchangeable. The new fuel assemblies incorporate minor modifications to the end fittings, primarily to reduce fuel assembly pressure drop and to increase holddown margin. All other results presented in the FSAR fuel assembly mechanical discussion are applicable to the reload fuel assemblies.

4.2. Fuel Rod Design

Pertinent fuel rod dimensions for residual and new fuel are listed in Table 4.2-1. The mechanical evaluation of the fuel rod is discussed below.

Cladding Collapse:

Creep collapse analyses were performed for three-cycle assembly power histories for TMI-1. The batch 3 fuel is more limiting than batch 4 fuel due to the lower prepressurization and lower pellet density. A summary of the batch 3 and batch 4 fuel rod designs are contained in Table 4.2-2. The batch 3 assembly power histories were analyzed and the most limiting assembly determined. Actual operating history was used where it was available. This included data through ∿ 7000 EFPH operation and included the initial power operation at 40% and 80% core power. The predicted assembly power history for the most limiting assembly was used to determine the most limiting collapse time as described in BAW-10084P-A, (Reference 2).

The following conditions were analyzed for the worst assembly power history. In all cases, the 2000 hr densification assumption described in Reference 2 was used since it was found to be the most severe case. The coolant and cladding temperatures, and fast flux values were calculated at axial locations corresponding to the conditions listed below.

1. Assembly outlet conditions
2. Axial power peak of 1.0 in the upper part of core
3. Maximum axial peak in the upper part of the core averaged over three cycles of operation.

1487 238

The third condition above was found to be the most limiting.

In addition to the above, the worst batch 2 assembly was analyzed using the same procedure described above.

The conservatisms in the analytical procedure are summarized below.

1. The CROV computer code was used to predict the time to collapse. CROV conservatively predicts collapse times, as demonstrated in Reference 2.
2. No credit is taken for fission gas release. Therefore, the net differential pressures used in the analysis are conservatively high.
3. The cladding thickness used was the LTL (lower tolerance limit) of the as-built measurements. The initial ovality of the cladding used was the UTL (upper tolerance limit) of the as-built measurements. These values were taken from a statistical sampling of the cladding.

The most limiting assembly, Batch 2, was found to have a collapse time greater than the maximum projected Cycle 2 life of 19,000 hours. This analysis was performed using the assumptions on densification described in Reference 2.

Cladding Stress:

Since the Batch 3 fuel is the most limiting from a cladding stress point of view due to the low prepressurization and low density, the calculations performed in the TMI-1 Fuel Densification Report, BAW-1389, June 1973 (Reference 3), are the most limiting.

Fuel Pellet Irradiation Swelling:

The fuel design criteria specify a limit of 1.0% on cladding circumferential plastic strain. The pellet design is set such that the plastic cladding strain is less than 1% at 55,000 MWD/MTU. The conservatisms in this analysis are listed below.

1. The maximum specification value for the fuel pellet diameter was used.
2. The maximum specification value for the fuel pellet density was used.

1487 239

3. The cladding ID used was the lowest permitted specification tolerance.

4. The maximum expected 3 cycle local pellet burnup is less than 55,000 MWD/MTU.

4.3. Thermal Design

The core loading for Cycle 2 operation is shown in Figure 3-1. There are 56 fresh (batch 4) fuel assemblies and 121 once-burned (batches 2 and 3) fuel assemblies. These assemblies are thermally and geometrically similar. However, the batch 4 fuel has a higher initial theoretical density (TDI), and a correspondingly higher linear heat rate capability (20.15 kW/ft vs 19.6 kW/ft) than does the batch 2 and 3 fuel. These linear heat rate limitations were established utilizing the TAFY-3 (Reference 4) code with full fuel densification penalties.

Power Spike Model

The power spike model utilized in this analysis is identical to that presented in BAW-10055 (Reference 5) except for two modifications. The modifications have been applied to F_g and F_k as described in Reference 6. These probabilities have been changed to reflect additional data from operating reactors that support a somewhat different approach and yield less severe penalties due to power spikes. F_g was changed from 1.0 to 0.5. F_k was changed from a Gaussian distribution to a linear distribution, which reflects a decreasing frequency with increasing gap size.

The maximum gap size versus axial position is shown in Figure 4.3-1, and the power spike factor versus axial position is shown in Figure 4.3-2. The calculated power spike and gap size were based upon 92.5% TDI and an enrichment of 3 wt % ^{235}U. The corresponding values for batch 4 fuel would be smaller because of the increased density and lower enrichment of this fuel.

Fuel Temperature Analysis

Thermal analysis of the fuel rods assumed in-reactor fuel densification to 96.5% theoretical density, TDF. The basis for the analysis utilized is given in References 4 and 5 with the following modifications:

1487 240

1. The option in the code for no restructuring of fuel has been used in the analysis presented here in accordance with the NRC's interim evaluation of TAFY.
2. The calculated gap conductance was reduced by 25% in accordance with the NRC's interim evaluation of TAFY.

During Cycle 2 operation the highest relative assembly power levels occur in batch 3 fuel (See Figures 3-1 and 5.1-1). Fuel temperature analysis for this fuel is documented in the TMI-1 Fuel Densification Report (Reference 3) for BOC. The maximum hot spot centerline fuel temperature is predicted on the basis of reference design peaking conditions, as shown in Table 4.3-1. Although batch 4 fuel has a reduced active fuel length, and a correspondingly higher average linear heat rate, the maximum predicted centerline temperature of this fuel would be lower than that of batch 3 fuel, even with the same peaking factors applied. This is primarily due to the higher initial density of the batch 4 fuel. Therefore, the analysis performed for Cycle 1 is also applicable for Cycle 2.

4.4. Materials Design

The batch 4 fuel assemblies are not new in concept and they do not utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 4 fuel assemblies are identical to those of the present fuel.

4.5. Operating Experience

B&W's operating experiences with comparable fuel assembly design have been demonstrated in the operation of six 177 fuel assembly plants utilizing this fuel assembly design. The most recent application for fuel assemblies of similar design is described in the Oconee 1, Cycle 2 Reload Report (Reference 7).

1487 241

Table 4.1-1. Fu-l Design Parameters

		Residual Fuel Assembly		New Fuel Assembly
		Batch 2	Batch 3	Batch 4
1.	Fuel Assembly Type	Mk-B3	Mk-B3	Mk-B4
2.	Number	61	60	56
3.	Initial Fuel Enrichment	2.75	3.05	2.64
4.	Initial Fuel Density % Theoretical	92.5	92.5	93.5
5.	Initial Fill Gas Pressure, psia	proprietary data to be supplied under separate cover.		
6.	Batch Bu nup, BOL, MWD/MTU	15,892	10,180	0
7.	Clad Collapse Time Effective Full Power Hours	>19,000	>19,000	>19,000

Table 4.2-1. Fuel Rod Dimensions

	Component	Residual Fuel Batches 2 & 3	New Fuel Batch 4
1.	Fuel Rods		
	O.D. inches	.430	.430
	I.D. inches	.377	.377
2.	Fuel Pellet		
	O.D. inches	.370	.370
	Density, % Theoretical	92.5	93.5
3.	Undensified Active Fuel Length, inches	144	142.6
4.	Flexible Spacers, Type	Corrugated Spacer	Spring
5.	Solid Spacers, Material	ZrO_2	Zr-4

1487 242

Table 4.2-2. Input Summary for Cladding Creep Collapse Calculations

	Batches 2 & 3	Batch 4
Pellet OD (mean specified), in.	.3700	.3700
Pellet Density (mean Specified), % TDI	92.5	93.5
Densified Pellet OD, in.	.3650	.3663
Cladding ID (mean specified), in.	.377	.377
Reactor System Pressure, psia	2200	2200
Stack Height (undensified), in.	144	142.6

1487 243

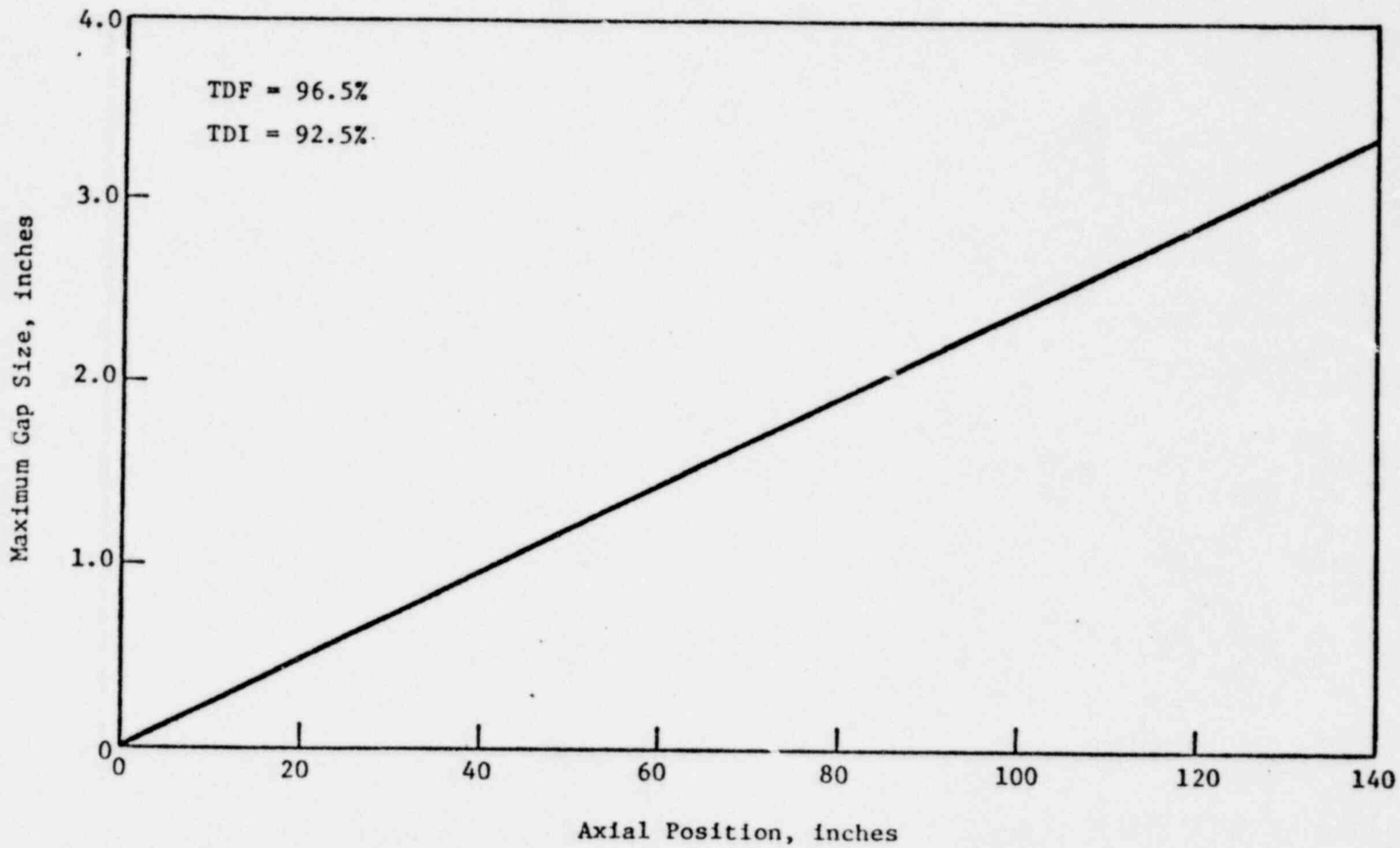
Table 4.3-1. Fuel Temperature Analysis Parameters for Cycle 2**

Reactor Core Power Level, MWt	2568
System Pressure, psia	2200
Reactor Vessel Average Coolant Temperature, F	579
Fraction of Heat Generated in Fuel and Cladding	0.973
$F_{\Delta H}^{N}$	1.78
F_{Z}^{N}	1.70
Fq (Nuclear)	3.03
Fq (Nuclear and Mechanical)	3.12
Average Thermal Output kW/ft - Batches 2, 3	5.77
Batch 4	5.80*
Average Fuel Temperature, F	1335
Maximum Fuel Centerline Temperature at Hot Spot, F	4780
Densified Active Fuel Length, In. - Batches 2, 3	141.2
Batch 4	140.4*
Linear Heat Rate to Central Fuel Melt, kW/ft - Batches 2, 3	19.6
Batch 4	20.15*
Initial Theoretical Density (TDI) - Batches 2, 3	92.5
Batch 4	93.5*

* Batch 4 fuel not used for analysis because batch 2,3 fuel is more limiting.
**Reference 3, BAW-1389.

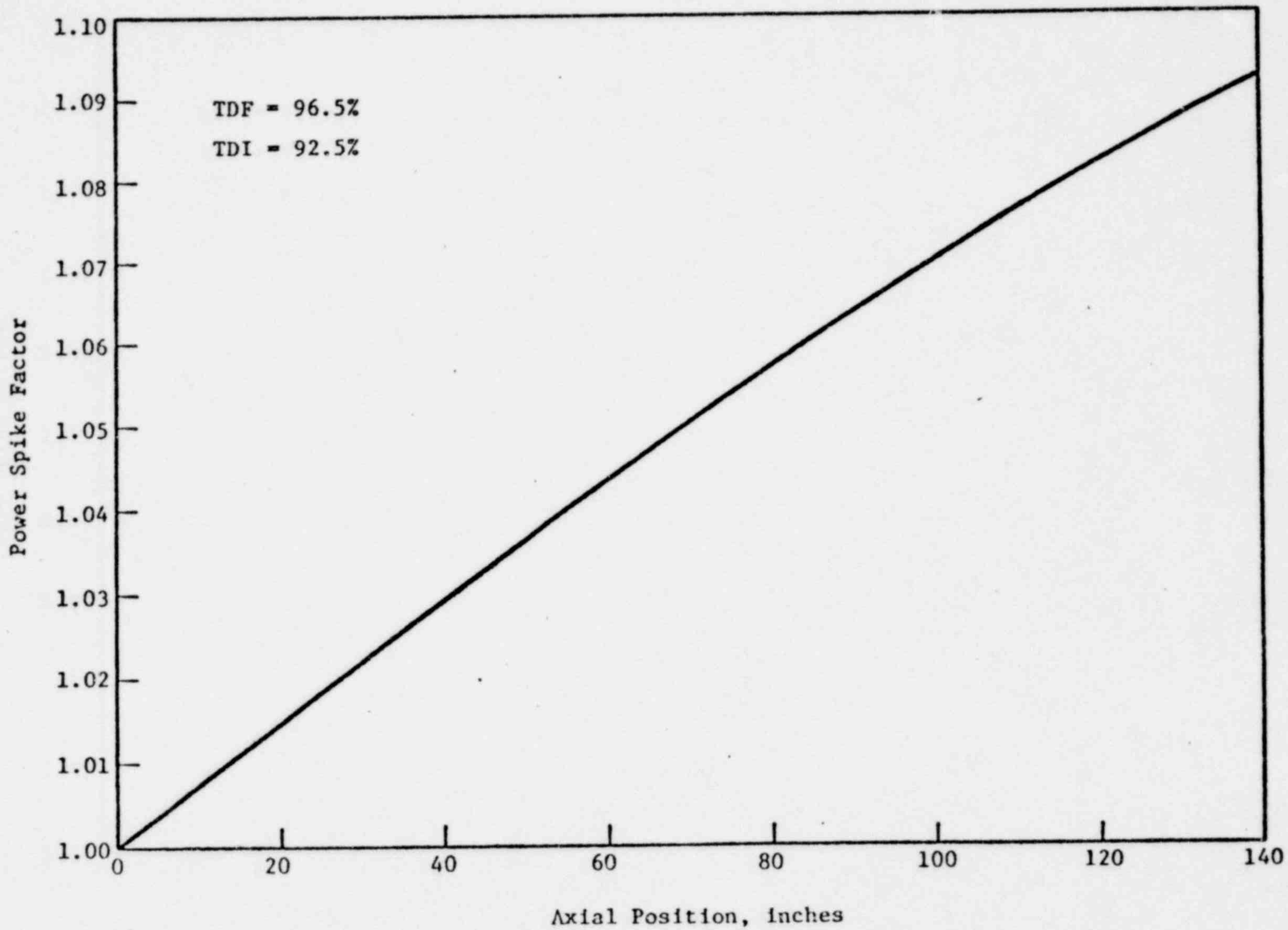
1487 244

Figure 4.3-1. Maximum Gap Size Vs Axial Position



1487 245

Figure 4.3-2. Power Spike Factor Vs Axial Position



1487 246

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5.1-1 compares the core physics parameters of cycles 1 and 2. The values for both cycles were generated using PDQ07. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles.

The shorter cycle 2 will produce a smaller cycle differential burnup than that for cycle 1. The accumulated average core burnup will be higher in cycle 2 than in cycle 1 because of the presence of the once-burned batch 2 and 3 fuel. Figure 5.1-1 illustrates a representative relative power distribution for the beginning of the second cycle at full power with equilibrium xenon and normal rod positions.

The critical boron concentrations for cycle 2 are lower in all cases than for the cycle 1. The control rod worths for hot full power (due to changes in radial flux distribution and isotopics) are somewhat less than those for cycle 1 (at BOL), although they are sufficient to maintain the required shutdown margin, as indicated in Table 5.1-2. Reference fuel cycle shutdown margin is presented in the TMI-1 FSAR Table 3-5. The stuck rod and ejected rod worths are considerably lower for cycle 2 than for cycle 1 except for the stuck rod worth at the end of cycle. Cycle 2 is only slightly higher for this case than cycle 1 and no adverse safety implications are associated with this higher worth. The adequacy of the shutdown margin with cycle 2 stuck rod worths is demonstrated in Table 5.1-2. For the shutdown calculations the following conservatisms were applied:

1) Poison material depletion allowance
2) 10% uncertainty on net rod worth
3) Flux redistirbution penalty

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The shutdown calculation at the end of cycle 2 is analyzed at approximately 275 EFPD's. This is the latest

1487 247

time (± 10 days) in core life in which the transient bank is nearly fully inserted. After 275 EFPD's the transient bank will be almost fully withdrawn thus increasing the potential shutdown margin.

The cycle 2 power deficits from hot zero power to hot full power are higher than those for cycle 1 due to a more negative moderator coefficient in cycle 2. The differential boron worths and total xenon worths for cycle 2 are lower than for cycle 1 due to depletion of the fuel and the associated buildup of fission products. Effective delayed neutron fractions show a similar trend with burnup.

5.2 Analytical Input

The cycle 2 incore measurement calculation constants used for computing core power distributions were prepared in the same manner as the reference cycle.

5.3 Changes in Nuclear Design

There were no relevant changes in core design between the reference and reload cycles. The same calculational methods and design information were used to obtain the important nuclear design parameters. In addition, no significant operational procedure changes exist from the reference cycle with regard to axial or radial power shape control, xenon control, or tilt control. The operational limits (Technical Specifications changes) for the reload cycle are shown in Section 8.

1487 248

Table 5.1-1 TMI-1, Cycle 2 Physics Parameters

	Cycle 2**	Cycle 1*
Cycle length, EFPD	296	466
Cycle burnup, MWd/mtU	9144	14,396
Average core burnup - EOC, MWd/mtU	18,072	14,396
Initial core loading, mtU	82.1	82.1
Critical boron - BOC, ppm		
HZP - all rods out	1415	1634
HZP - groups 7 and 8 inserted	1252	1494
HFP - groups 7 and 8 inserted	1066	1382
Critical boron - EOC, ppm		
HZP - all rods out	440	480
HFP - group 8 (37.5% withdrawn, equil. Xe)	96	180
Control rod worths - HFP, BOC, %Δk/k		
Group 6	1.20	1.58
Group 7	0.96	0.99
Group 8 (37.5% wd)	0.54	0.44
Control rod worths - HFP, EOC, %Δk/k		
Group 7	1.33	1.37
Group 8 (37.5% wd)	0.51	0.26
Ejected rod worth - HZP, %Δk/k		
BOC	0.59+	0.48++
EOC	0.58+	0.72++
Stuck rod worth - HZP, %Δk/k		
BOC	2.16	4.27
EOC	2.22	2.69
Power deficit, HZP to HFP, %Δk/k		
BOC	-1.65	-1.32
EOC	-2.49	-2.10
Doppler coeff - BOC, 10^{-5} ($\Delta k/k/^{o}F$)		
100% power (0 Xe)	-1.51	-1.51
Doppler coeff - EOC, 10^{-5} ($\Delta k/k/^{o}F$)		
100% power (equil Xe)	-1.55	-1.67
Moderator coeff - HFP, 10^{-4} ($\Delta k/k/^{o}F$)		
BOC (0 Xe, 1000 ppm, groups 7 and 8 inserted)	-1.03	-0.23
EOC (equil Xe, 17 ppm, group 8 inserted)	-2.60	-2.70

1487 249

Table 5.1-1 (Continued)

	Cycle 2**	Cycle 1*
Boron worth - HFP, ppm/%Δk/k		
BOC (1000 ppm)	109	98
EOC (17 ppm)	101	95
Xenon worth - HFP, %Δk/k		
BOC (4 days)	2.60	2.71
EOC (equilibrium)	2.66	2.65
Effective delayed neutron fraction (HFP)		
BOC	.00577	.00690
EOC	.00516	.00514

*For Cycle 1 length of 466 EFPD
**Based on Cycle 1 length of 440 EFPD
+Ejected rod value for group 5, 6, 7 and 8 inserted
++Ejected rod value for Group 6, 7, and 8 inserted

1467 250

Table 5.1-2 Shutdown Margin Calculation

TMI-1, Cycle 2

	BOC, % Δk/k	EOC*, % Δk/k
I. Available Rod Worth		
a. Total rod worth, HZP**	9.77	9.80
b. Worth reduction due to burnup of poison material	-0.19	-0.30
c. Maximum stuck rod, HZP	-2.16	-2.22
d. Net worth	7.42	7.28
e. Less 10% uncertainty	-0.74	-0.73
f. Total available worth	6.68	6.55
II. Required Rod Worth		
a. Power deficit, HFP to HZP	1.65	2.49
b. Max. allowable inserted rod worth	1.07	1.27
c. Flux redistribution	0.40	1.00
d. Total required worth	3.12	4.76
III. Shutdown Margin		
(I.f. minus II.d.)	3.56	1.79

Note: Required shutdown margin is 1.00% Δk/k

*For shutdown margin calculations this is defined as ~ 275 EFPD, the latest time in core life in which the transient bank is nearly full in.

**HZP denotes Hot Zero Power/HFP denotes Hot Full Power.

1487 251

Fig[illegible]-1 BOC (4 EFPD), Cycle 2 Two-dimensional Relative Power Distribution - Full Power, Equilibrium Xenon, Normal Rod Positions (Groups 7 and 8 Inserted)

	8	9	10	11	12	13	14	15
H	1.40	1.27	1.22	1.17	1.21	.85	.77	.60
K	1.27	1.41	1.37	1.23	1.23	7 .57	.69	.59
L	1.22	1.37	1.18	1.13	8 1.12	.95	.95	.52
M	1.17	1.23	1.13	1.35	1.29	1.24	.92	
N	1.21	1.23	8 1.12	1.29	1.10	1.09	.66	
O	.85	7 .57	.95	1.24	1.09	.72		
P	.77	.69	.95	.92	.66			
R	.60	.59	.52					

x	Inserted Rod Group No.
x.xx	Relative Power Density

1487 252

6. ERMAL-HYDRAULIC DESIGN

6.1 Thermal-Hydraulic Design Calculations

Thermal-hydraulic design calculations for support of cycle 2 operation utilized predominantly the same analytical methods previously documented in references 1 and 3. Adjustments to these calculations were made to account for the introduction of the Mk-B4 assemblies in batch 4, to consider the minimum actual Reactor Coolant system flow rate as measured during first cycle operation, and to incorporate the B&W-2 critical heat flux correlation in place of the previously used W-3 correlation.

Introduction of Mark B-4 Assemblies

As discussed in section 4.1, the Mk-B4 assemblies differ from the Mk-B3 assemblies primarily in the end fittings. This difference causes a slight reduction in the flow resistance of the B4 assemblies relative to the B3's. Since the B4 assembiles are loaded primarily on the periphery of the core, the hottest (highest power) assembly is a B3 assembly (see Figures 3-1 and 5.1-1). In order to conservatively account for the introduction of the B-4 assemblies, the thermal-hydraulic model utilized the cycle 3 configuration (117 B4, 60 B3 assemblies), but retained the B3 assemblies in the hottest core locations. This assumption increases the conservatism of the cycle 2 design by reducing the calculated hot assembly flow rate.

Increased RC System Flow

RC flow data obtained during Cycle 1 operation verified that the system flow was greater than the design flow. The minimum measured value was 108 percent of design. For the Cycle 2 thermal-hydraulic design analysis the increase in system flow was conservatively chosen to be 106.5 percent of Cycle 1 design flow. Therefore Cycle 2 100% flow is 139.8×10^6 lb/hr.

B&W-2 DNB Correlation

The B&W-2 DNB correlation is a more realistic prediction of the burnout phenomena (as described in references 8 and 9) and has been reviewed and approved for use with the Mark B fuel assembly design.

In the application of the B&W-2 DNB correlation to the TMI-1, Cycle 2 core, two modifications in the use of the correlation have been instituted.

1487 253

1. The limiting design DNBR of 1.30 was used representing a 95 percent confidence level for 95 percent population protection. A limiting DNBR of 1.32 which has been used for this correlation in previous design analyses represented a 99 percent confidence level for a 95 percent protection. This change is consistent with industry practice and the statistical standards associated with limiting design DNBR values accepted by the NRC Staff and ACRS.

2. The pressure range applicable to the correlation has been extended downward from 2000 psia to 1750 psia. This revision is based on a review of rod bundle CHF data taken at pressures below 2000 psia which shows that the B&W-2 correlation conservatively predicts the data in this range.

The use of this correlation in conjunction with increased system flow for the Cycle 2 analysis indicates that the calculated margin to DNB is higher than had been predicted for first core operation as shown in Table 6.2-1.

6.2 DNBR Analysis

In addition to the items discussed above, the maximum design conditions considered in the FSAR and generic fuel assembly geometry based on total Mark B as-built data were taken into account. This resulted in a minimum DNBR of 1.919 at 112 percent power for undensified fuel.

The effects due to densification can be divided into two categories: (1) the result of reduced stack height and (2) the power spike resulting from densification induced gaps in the fuel column. The active length was calculated to be 141.2 inches, a reduction from the undensified length of 144.0 inches. These densified and undensified lengths are based on fuel Batches 2 and 3 as discussed in Section 4.3.1.

The axial flux shape which resulted in the maximum change in DNBR from the original design value was an outlet peak with a core offset of +11.8%. The spike magnitude and the maximum gap size are discussed in Section 4.3 and the values used in the analysis are 1.087 and 3.1 inches, respectively. The results of the two effects are -1.88 and -1.06 percent change in minimum hot channel DNBR and peaking margin, respectively. The changes in these margins are summarized in Table 6.2-1 which includes comparisons of other pertinent Cycle 1 and Cycle 2 data.

Table 6.2-1 Cycle 1 and Cycle 2 Design Conditions

	Cycle 1*	Cycle 2
Power Level, MWt	2568	2568
System Pressure, psia	2200	2200
Reactor Coolant Flow, % Design Flow	100.0	106.5
Vessel Inlet Coolant Temperature - 100% Power, °F	554.0	555.6
Vessel Outlet Coolant Temperature- 100% Power, °F	603.8	602.4
Ref. Design Radial - Local Power Peaking Factor	1.78	1.78
Ref. Design Axial Flux Shape	1.5 cosine	1.5 cosine
Active Fuel Length, in. (undensified)	144	144+
Average Heat Flux (100% Power), Btu/h-ft^2	171,470	174,870***
Maximum Heat Flux (100% Power), Btu/h-ft^2 (for DNBR Calc.)	457,825	466,903***
CHF Correlation	W-3	B&W-2
Minimum DNBR (Max. Design Conditions, At overpower indicated) no densification penalty.	1.55 (114% power)	1.919 (112% power)
Hot Channel Factors		
Enthalpy Rise	1.011	1.011
Heat Flux	1.014	1.014
Flow Area	0.98	0.98
Densification Effects		
Change in DNBR Margin, %	-6.08**	-1.88
Change in Power Peaking Margin, %	-2.82**	-1.06

*Reference 1. Three Mile Island, Unit 1 FSAR, Docket No. 50-289.
**Reference 3, "Three Mile Island, Unit 1 Fuel Densifiction Report," Babcock & Wilcox Report BAW-1389 (Proprietary), June 1973.
*** For densified fuel
+ Batches 2 & 3 limiting assembly. For batch 4; 142.6 inches.

1487 255

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each FSAR (Reference 1) accident analysis has been examined with respect to changes in Cycle 2 parameters to determine the effects of the Cycle 2 reload and assures that thermal performance during hypothetical transients is not degraded.

Core thermal parameters used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. Comparison of first core values (FSAR values) of core thermal parameters with parameters used in Cycle 2 analysis are given in Table 6.2-1. These are parameters common to all of the accident analyses presented herein. For each accident of the FSAR, a discussion of the accident and the key parameters are provided. A comparison of the key parameters (See Table 7.1-1) from the FSAR and Cycle 2 is provided with the accident discussions to show that the initial conditions for the transients are bounded by the FSAR analysis

The effects of fuel densification on the FSAR accident results have been evaluated and are reported in Reference 3. Since Batch 4 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in Reference 3, the conclusions in that reference are still valid.

Calculational techniques and methods remain consistent for Cycle 2 analysis with those used for the FSAR. Additional DNBR margin is shown for Cycle 2 by the use of the B&W-2 CHF correlation rather than the W-3 CHF correlation.

No new dose calculations were performed for the reload report. The dose considerations in the FSAR were based on maximum peaking and burnup for all core cycles and therefore the dose considerations are independent of the reload batch.

7.2 Rod Withdrawal Accidents

This accident is defined as uncontrolled reactivity addition to the core from withdrawal of control rods during startup conditions or from rated power conditions. Both types of incidents were analyzed in the FSAR.

The important parameters during a rod withdrawal accident are Doppler coefficient, moderator temperature coefficient and the rate at which reactivity is added to the core. Only high pressure and high flux trips are accounted for

in the FSAR analysis, ignoring multiple alarms, interlocks and trips that normally precludes this type of incident.

For positive reactivity addition indicative of these events, the most severe results occur for BOL conditions. The FSAR values of the key parameters for BOL conditions were -1.17×10^{-5} ($\Delta k/k/F$) for the Doppler coefficient, 0.0 ($\Delta k/k/F$) for the moderator temperature coefficient and rod group worths up to and including a 10% $\Delta k/k$ rod group worth. Comparable Cycle 2 parametric values are -1.51×10^{-5} ($\Delta k/k/F$) for the Doppler coefficient, -1.03×10^{-4} ($\Delta k/k/F$) for the moderator temperature coefficient, and maximum rod group worth of 9.8% $\frac{\Delta k}{k}$. Therefore, Cycle 2 parameters are bounded by design values assumed for the FSAR analysis. Thus, for the rod withdrawal transients, the consequences will be no more severe than those presented in Reference 1 or 3.

7.3 Moderator Dilution Accident

Boron in the form of boric acid is utilized to control excess reactivity. The boron content of the reactor coolant is periodically reduced to compensate for fuel burnup and transient xenon effects with dilution water supplied by the makeup and purification system. The moderator dilution transients considered are the pumping of water with zero boron concentration from the makeup tank to the reactor coolant system under conditions of full power operation, hot shutdown and during refueling.

The key parameters in this analysis are the initial boron concentration, boron reactivity worth, and the moderator temperature coefficient for power cases.

For positive reactivity addition of this type, the most severe results occur at BOC conditions. The FSAR values of the key parameters for BOC conditions were 1200 ppm for the initial boron concentration, 75 ppm/1% $\Delta k/k$ boron reactivity worth and 0.5×10^{-4} $\Delta k/k/F$ for the moderator temperature coefficient.

Comparable Cycle 2 values are 1066 ppm for the initial boron concentration, 84 ppm/1% $\Delta k/k$ boron reactivity worth and -1.03×10^{-4} $(\Delta k/k)/F$ for the moderator temperature coefficient. The FSAR shows that the core and RCS are adequately protected during this event. Sufficient time for operator action to terminate this transient is also shown in the FSAR even with maximum dilution and minimum shutdown margin. The predicted Cycle 2 parameters of importance to moderator dilution transient are bounded by the FSAR design values, thus, the analysis in the FSAR is valid.

7.4 Cold Water (Pump Startup) Accident

The NSS does not contain any check or isolation valves in the reactor coolant system piping, therefore, the classical cold water accident is not possible. However, when the reactor is operated with one or more pumps not running, and these pumps are then started, the increased flow rate will cause the average core temperature to decrease. If the moderator temperature coefficient is negative, reactivity will be added to the core and a power increase will occur.

Protective interlocks exist and administrative procedures are imposed to prevent the starting of idle pumps if the reactor power is above 30%. However, these restrictions were not assumed and two pump startup from 50% power was analyzed as the most severe transient.

To maximize reactivity addition, the FSAR analysis used the most negative moderator temperature coefficient of -3.0×10^{-4} $\Delta k/k/F$ and least negative Doppler coefficient of -1.17×10^{-5} $\Delta k/k/F$. The corresponding most negative moderator temperature coefficient and least negative Doppler coefficient predicted by Cycle 2 are -2.60×10^{-4} $\Delta k/k/F$ and $-1\ \ 1\times10^{-5}$ $\Delta k/k/F$, respectively. As the predicted Cycle 2 moderator temperature coefficient is less negative and the Doppler coefficient is more negative than the values used in the FSAR, the transient results would be less severe than those reported in the FSAR.

7.5 Loss of Coolant Flow

A reduction in the reactor coolant flow rate can occur from mechanical failures or from a loss of electrical power to the pumps. With four independent pumps available, a mechanical failure in one pump will not affect operation of the others. With the reactor at power, the effect of loss of coolant flow is a rapid increase in coolant temperature due to reduction of heat removal capability. This temperature increase could result in DNB if corrective action were not taken immediately. The key parameters for 4 pump coastdown or locked rotor incident are the flow rate, flow coastdown characteristics, Doppler coefficient, moderator temperature coefficient, and hot channel DNB peaking factors. The conservative initial conditions assumed for the densification report (Reference 3) were: FSAR values of flow and coastdown, -1.17×10^{-5} $\Delta k/k/F$ Doppler coefficient, $+0.5\times10^{-4}$ $\Delta k/k/F$ moderator temperature coefficient, with densified fuel power spike and peaking. The results showed the DNBR remained above 1.3 (W-3) for the 4-pump coastdown and the fuel cladding temperature remained below criteria limits for the locked rotor transient.

The predicted values for Cycle 2 are -1.51×10^{-5} Δk/k/F Doppler coefficient, -1.03×10^{-4} Δk/k/F moderator temperature coefficient and peaking factors as shown in Table 6.2-1. Since the B&W-2 CHF correlation was used for Cycle 2 and the predicted Cycle 2 values are bounded by those used in the Cycle 1 densification report, the results of the Cycle 1 analysis represents the most severe consequences from a loss of flow incident.

7.6 Stuck-Out, Stuck-In, or Dropped Control Rod Accident

If a control rod is dropped into the core while operating, a rapid decrease in neutron power would occur, accompanied by a decrease in core average coolant temperature. In addition, the power distribution may be distorted due to a new control rod pattern. Therefore, under these conditions a return to rated power may lead to localized power densities and heat fluxes in excess of design limitations.

The key parameters for this transient are moderator temperature coefficient, worth of dropped rod, and local peaking factors. The FSAR analysis was based on 0.46% Δk/k and 0.36% Δk/k rod worths with a moderator temperature coefficient of -3.0×10^{-4} Δk/k/F. For Cycle 2, the maximum worth rod at power is 0.20% Δk/k and a moderator temperature coefficient of -2.60×10^{-4} Δk/k/F. Since the predicted rod worth is less and the moderator temperature coefficient more positive, the consequences of this transient are less severe than the results presented in the FSAR.

7.7 Loss of Electric Power

Two types of power losses were considered in the FSAR: i) a loss of load condition, caused by separation of the unit from the transmission system, and ii) a hypothetical condition which results in a complete loss of all system and unit power except the unit batteries.

The FSAR analysis evaluated the loss of load both with and without turbine runback. When there is no runback, a reactor trip occurs on high reactor coolant pressure or temperature. This case resulted in a non-limiting accident. The largest offsite dose occurs for the second case, i.e., loss of all electrical power except unit batteries, and assuming operation with failed fuel and steam generator tube leakage. These results are independent of core loading, and therefore, the results of the FSAR are applicable for any reload.

7.8 Steam Line Failure

A steam line failure is defined as a rupture of any of the steam lines from the steam generators. Upon initiation of the rupture, both steam generators start to blowdown, causing a sudden decrease in primary system temperature, pressure and pressurizer level. The temperature reduction leads to positive reactivity

insertion (at EOL, the moderator temperatures coefficient is negative) and the reactor trips on high flux or low RC pressure. The FSAR has identified a double-ended rupture of the steam line between the steam generator and steam stop valve as the worst case situation, at end-of-life conditions.

The key parameter for the core response is the moderator temperature coefficient which was in the FSAR assumed to be -3.0×10^{-4} $\Delta k/k/F$. The Cycle 2 predicted value of moderator temperature coefficient is -2.60×10^{-4} $\Delta k/k/F$. This value is bounded by those used in the FSAR analysis and hence, the results in the FSAR represent the worst situation.

7.9 Steam Generator Tube Failure.

A rupture or leak in a steam generator tube allows reactor coolant and associated activity to pass to the secondary system. The FSAR analysis is based on complete severence of a steam generator tube. The primary concern for this incident is the potential radiological release, which is independent of core loading. Hence, the FSAR results are applicable to this reload.

7.10 Fuel Handling Accident

The mechanical damage type of accident is considered the maximum potential source of activity release during fuel handling activity. The primary concern is radiological releases which are independent of core loading and, therefore, the results of the FSAR are applicable to all reloads.

7.11 Rod Ejection Accident

For reactivity to be added to the core at a more rapid rate than by uncontrolled rod withdrawal, physical failure of a pressure barrier component in the control rod drive assembly must occur. Such a failure could cause a pressure differential to act on a control rod assembly and rapidly eject the assembly from the core. This incident represents the most rapid reactivity insertion that can be reasonably postulated. The values used in the FSAR and densification report at BOL conditions of -1.17×10^{-5} $\Delta k/k/F$ Doppler coefficient, $+0.5 \times 10^{-4}$ $\Delta k/k/F$ moderator temperature coefficient, and ejected rod worth of 0.65% $\Delta k/k$ represent the maximum possible transient. The corresponding Cycle 2 parametric values of -1.51×10^{-5} $\Delta k/k/F$ Doppler, -1.03×10^{-4} $\Delta k/k/F$ moderator temperature coefficient, are both more negative than used in Reference 3 and a maximum predicted ejected rod worth of 0.18% $\Delta k/k$ assure that the results will be less severe than those presented in the FSAR and densification report (References 1 and 3)

1487 260

7.12 Maximum Hypothetical Accident

There is no postulated mechanism whereby this accident can occur, since this would require a multitude of failures in the engineered safeguards. The hypothetical accident is based solely on a gross release of radioactivity to the reactor building and is independent of core loading. Therefore, the results reported in the FSAR are applicable for all reloads.

7.13 Waste Gas Tank Rupture

The waste gas tank was assumed to contain the gaseous activity evolved from degassing all of the reactor coolant following operation with 1% defective fuel. Rupture of the tank would result in the release of its radioactive contents to the plant ventilation system and to the atmosphere through the unit vent. The consequences of this incident are independent of core loading, and therefore, the results reported the FSAR are applicable to any reload.

7.14 LOCA Analysis

A generic LOCA analysis for B&W 177 FA lowered-loop NSS has been performed using the Final Acceptance Criteria ECCS Evaluation Model. This study is reported in BAW-10103, Rev. 1 (Reference 10). The analysis in Reference 10 is generic in nature since the limiting values of key parameters for all plants in this category were used. Furthermore, the average fuel temperature as a function of the linear heat rate and the lifetime pin pressure data used in the BAW-10103, Rev. 1 LOCA limits analysis are conservative and/or identical compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in Reference 10, provide conservative results for the operation of TMI-1, Cycle 2.

Table 7.14-1 shows the bounding values for allowable LOCA peak linear heat rates for TMI-1 Cycle 2 fuel.

1487 261

Table 7.1-1 COMPARISON OF :Y PARAMETERS FOR ACCIDENT ANA IS (440 EFPD)

PARAMETER	FSAR & DENSIFICATION VALUE	PREDICTED CYCLE 2 VALUE
Doppler Coefficient, BOL	-1.17×10^{-5} $(\Delta k/k)/F$	-1.51×10^{-5} $(\Delta k/k)/F$
, EOL	-1.33×10^{-5} $(\Delta k/k)/F$	-1.55×10^{-5} $(\Delta k/k)/F$
Moderator Coefficient, BOL	$+0.5\times10^{-4}$ $(\Delta k/k)/F$	-1.03×10^{-4} $(\Delta k/k)/F$
,EOL	-3.0×10^{-4} $(\Delta k/k)/F$	-2.60×10^{-4} $(\Delta k/k)/F$
All Rod Group Worth	10%$\Delta k/k$	9.8%$\Delta k/k$
Initial Boron Concentration	1200 ppm	1066 ppm
Boron Reactivity Worth, cold	75 ppm/1% $\Delta k/k$	84 ppm/1% $\Delta k/k$
Maximum Ejected Rod Worth	0.65% $\Delta k/k$	0.18% $\Delta k/k$
Dropped Rod Worth, HFP	0.46% $\Delta k/k$	0.20% $\Delta k/k$

1487 262

Table 7.14-1 ALLOWABLE LOCA PEAK LINEAR HEAT RATE

Core Elevation, ft.	Allowable Peak Linear Heat Rate, kW/ft
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

1487 263

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for Cycle 2 operation. The changes made are as a result of:

(1) Using the B&W-2 CHF correlation rather than W-3 as discussed in Section 6.1,

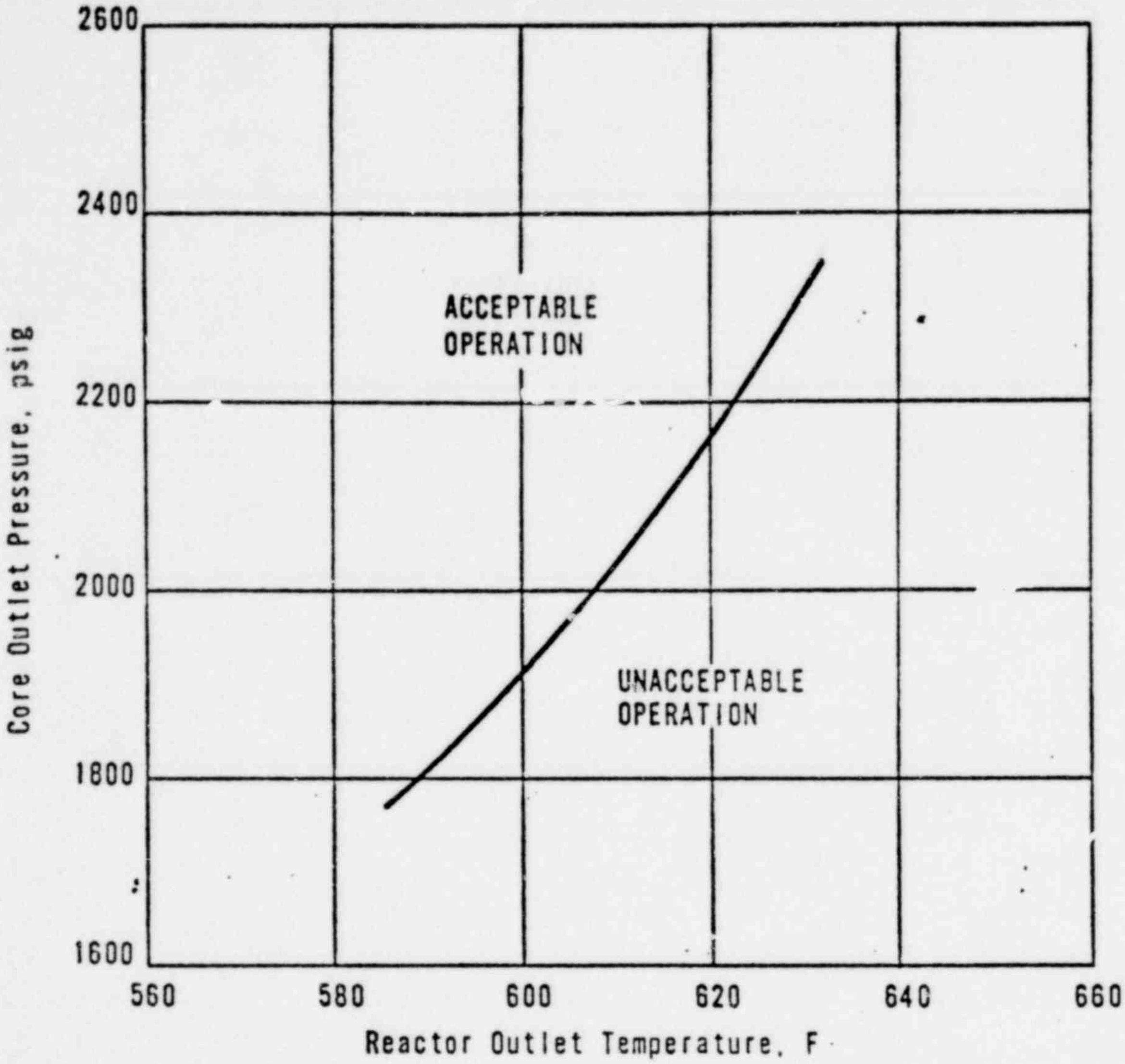
(2) The use of a 95/95 confidence level rather than 99/95 as discussed in Section 6.1,

(3) The use of 106.5% of design flow rather than 100% as discussed in Section 6.1,

(4) The use of the Final Acceptance Criteria LOCA analyses for restricting peaks during operation as discussed in Section 7.14.

Based upon the Technical Specifications derived from the analyses presented in this report, the Final Acceptance Criteria ECCS limits will not be exceeded and the thermal design criteria will not be violated.

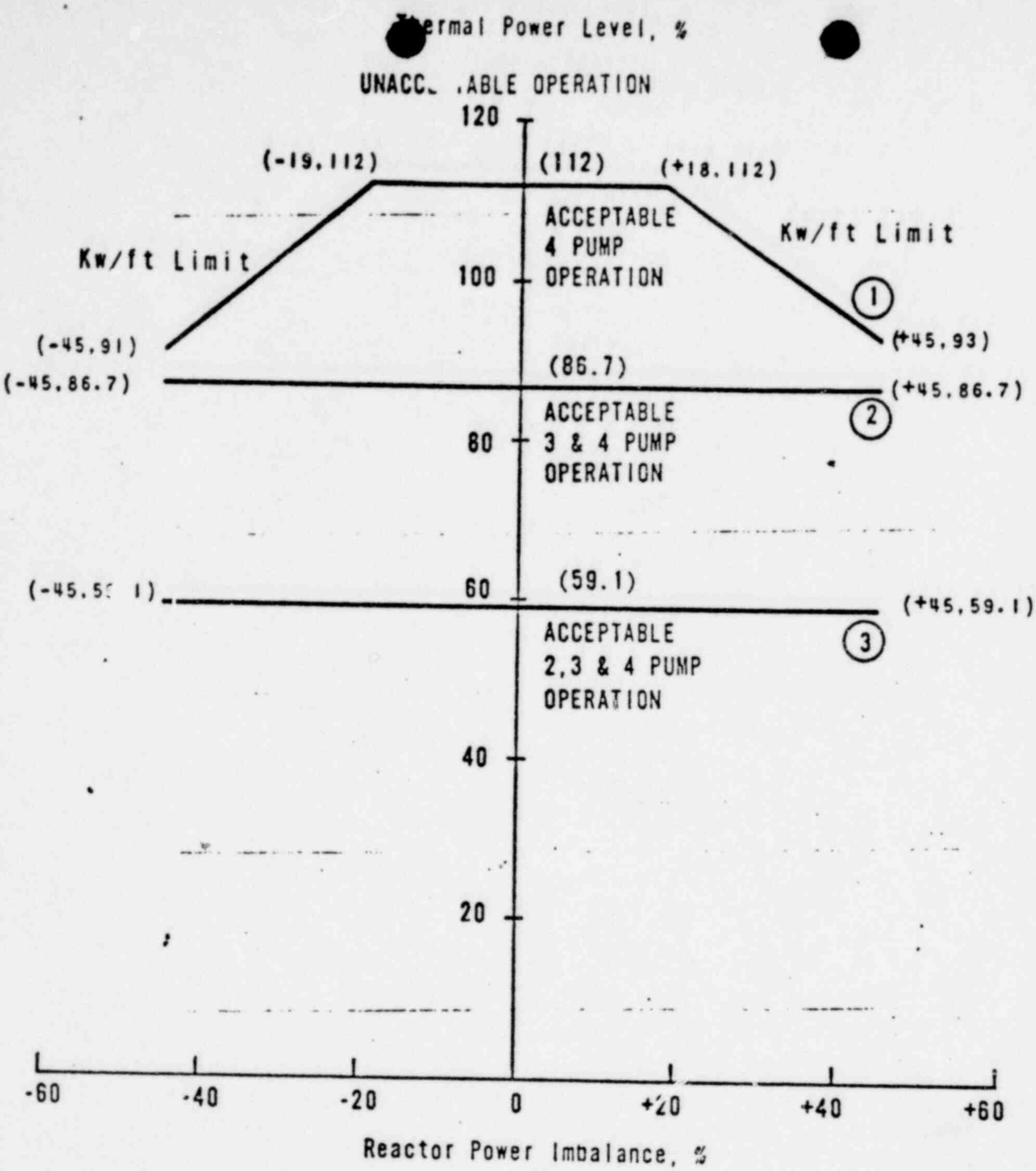
1487 264



CORE PROTECTION SAFETY LIMIT

Figure 8-1

1487 265

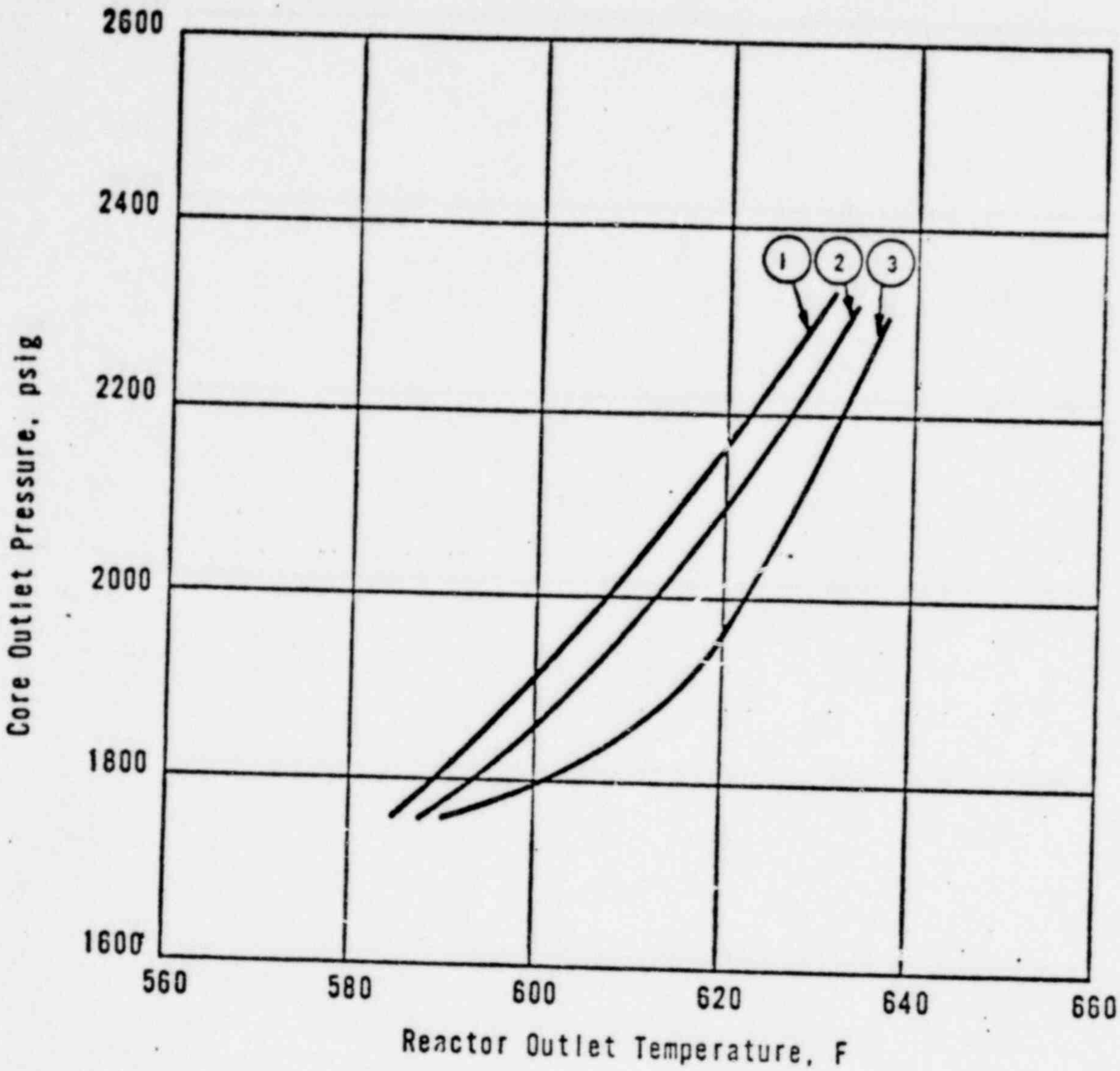


CURVE	REACTOR COOLANT FLOW (lb/hr)
1	139.8×10^6
2	104.5×10^6
3	68.8×10^6

1487 266

CORE PROTECTION SAFETY LIMITS

Figure 8-2



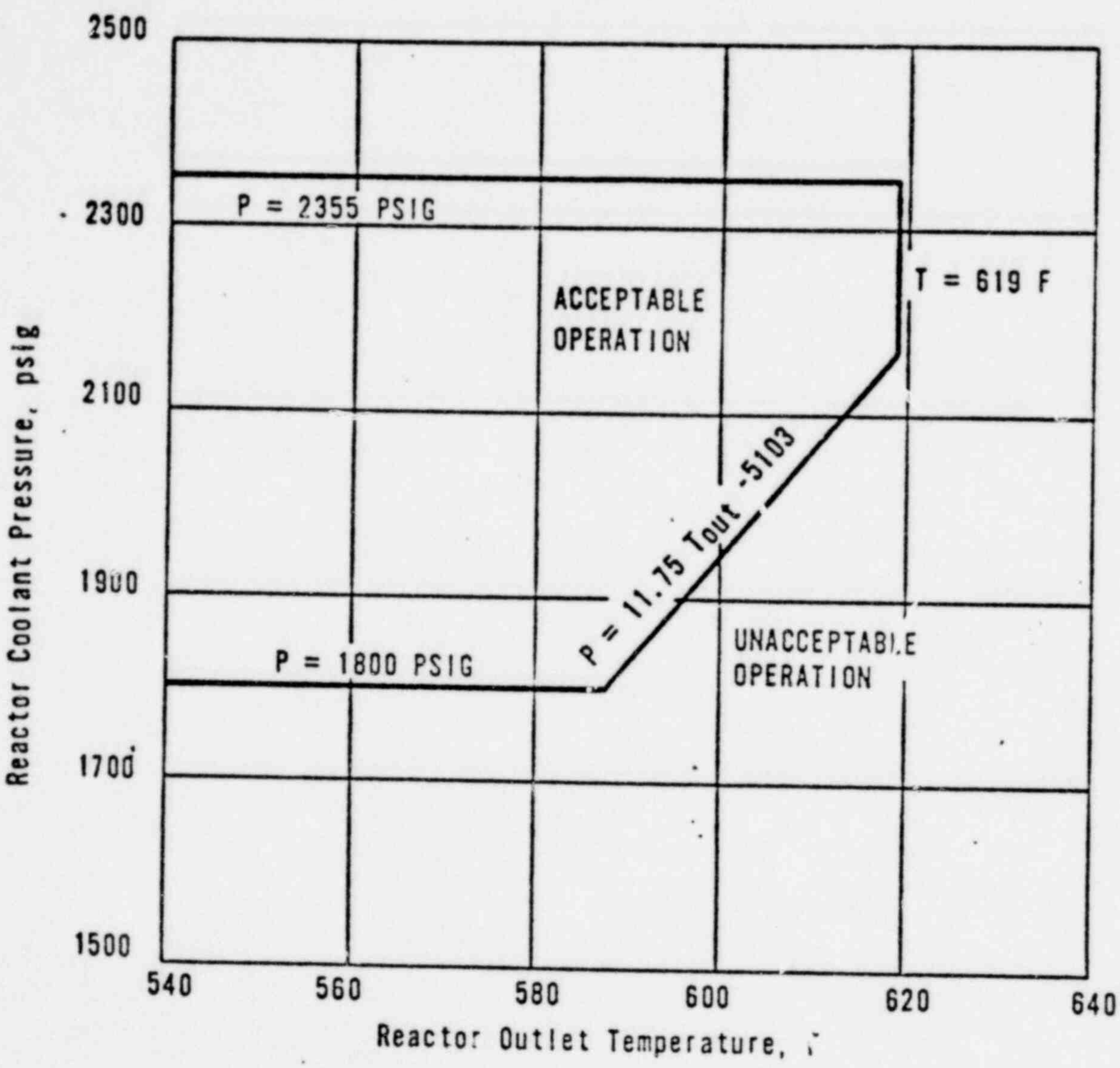
CURVE	REACTOR COOLANT FLOW (LBS/HR)	POWER	PUMPS OPERATING (TYPE OF LIMIT)
1	139.8×10^6 (100%)*	112%	Four Pumps (DNBR Limit)
2	104.5×10^6 (74.7%)	86.7%	Three Pumps (DNBR Limit)
3	68.8×10^6 (49.2%)	59.1%	One Pump in Each Loop (Quality Limit)

* 106.5% of Cycle 1 Design Flow.

CORE PROTECTION SAFETY

BASES Figure 8-3

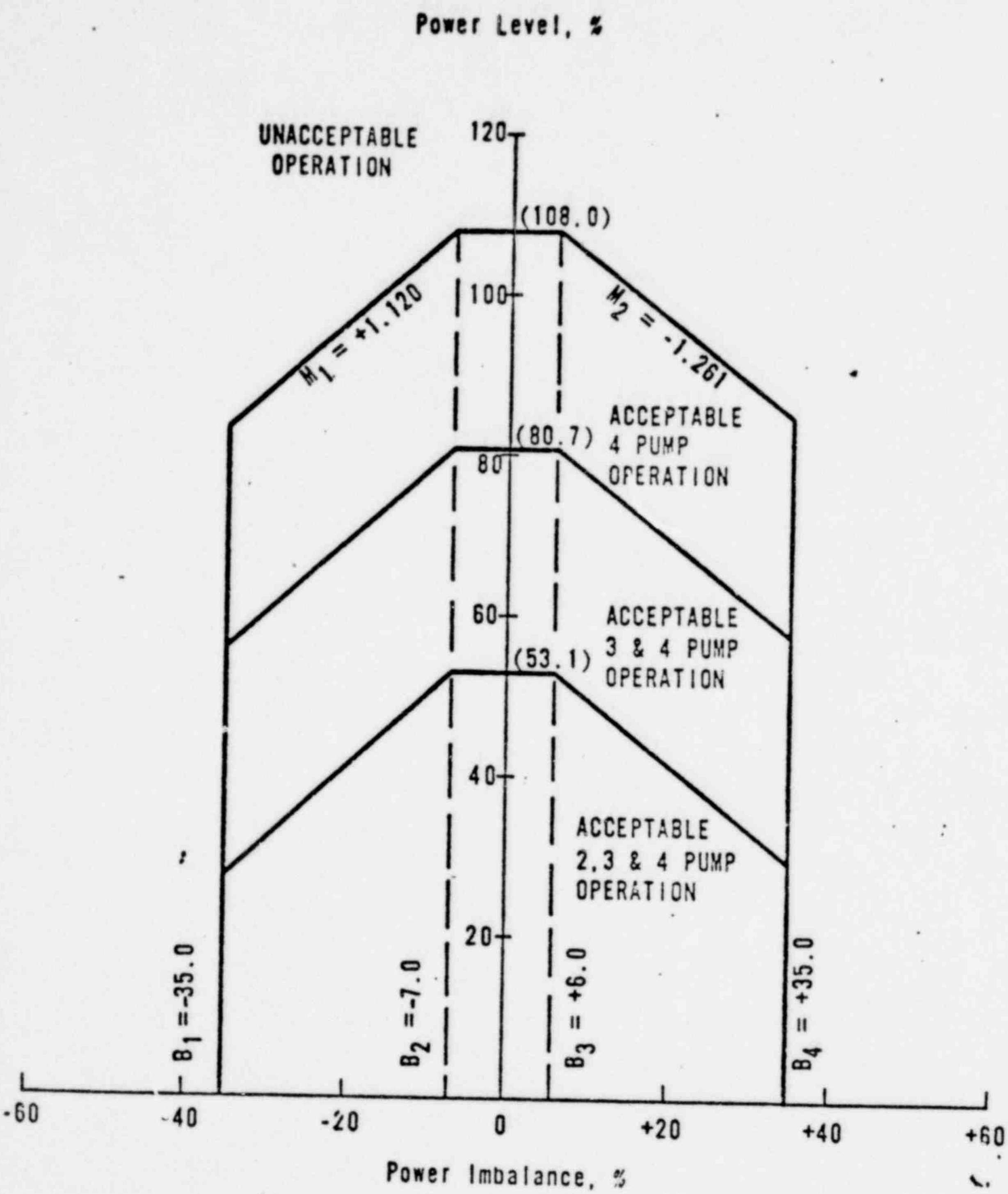
1487 267



PROTECTION SYSTEM MAXIMUM
ALLOWABLE SET POINTS

Figure 8-4

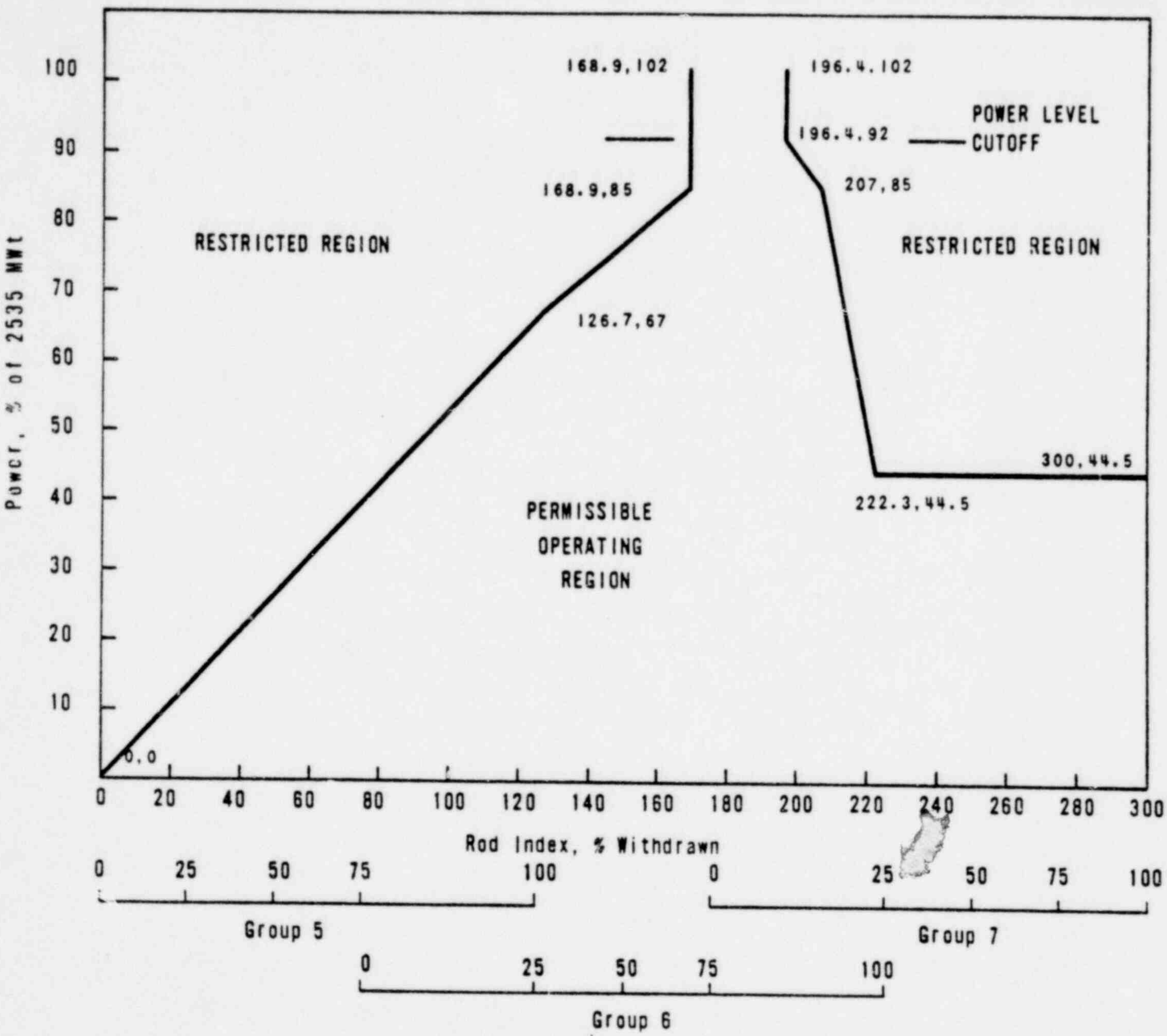
1487 268



PROTECTION SYSTEM MAXIMUM ALLOWABLE
SET POINTS

Figure 8-5

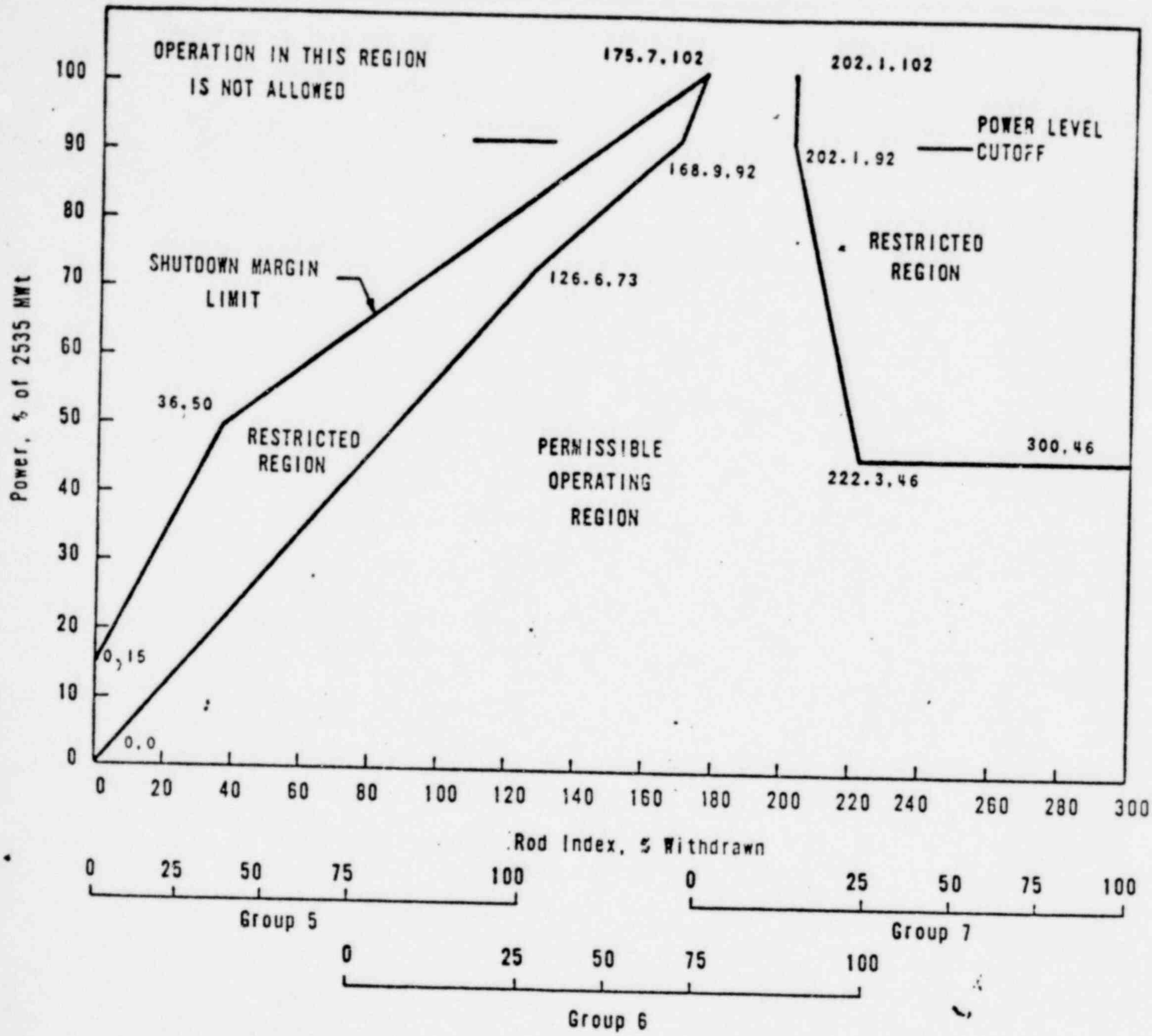
1487 269



ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 0 TO 152 ± 10 EFPD; CYCLE 2

Figure 8-6

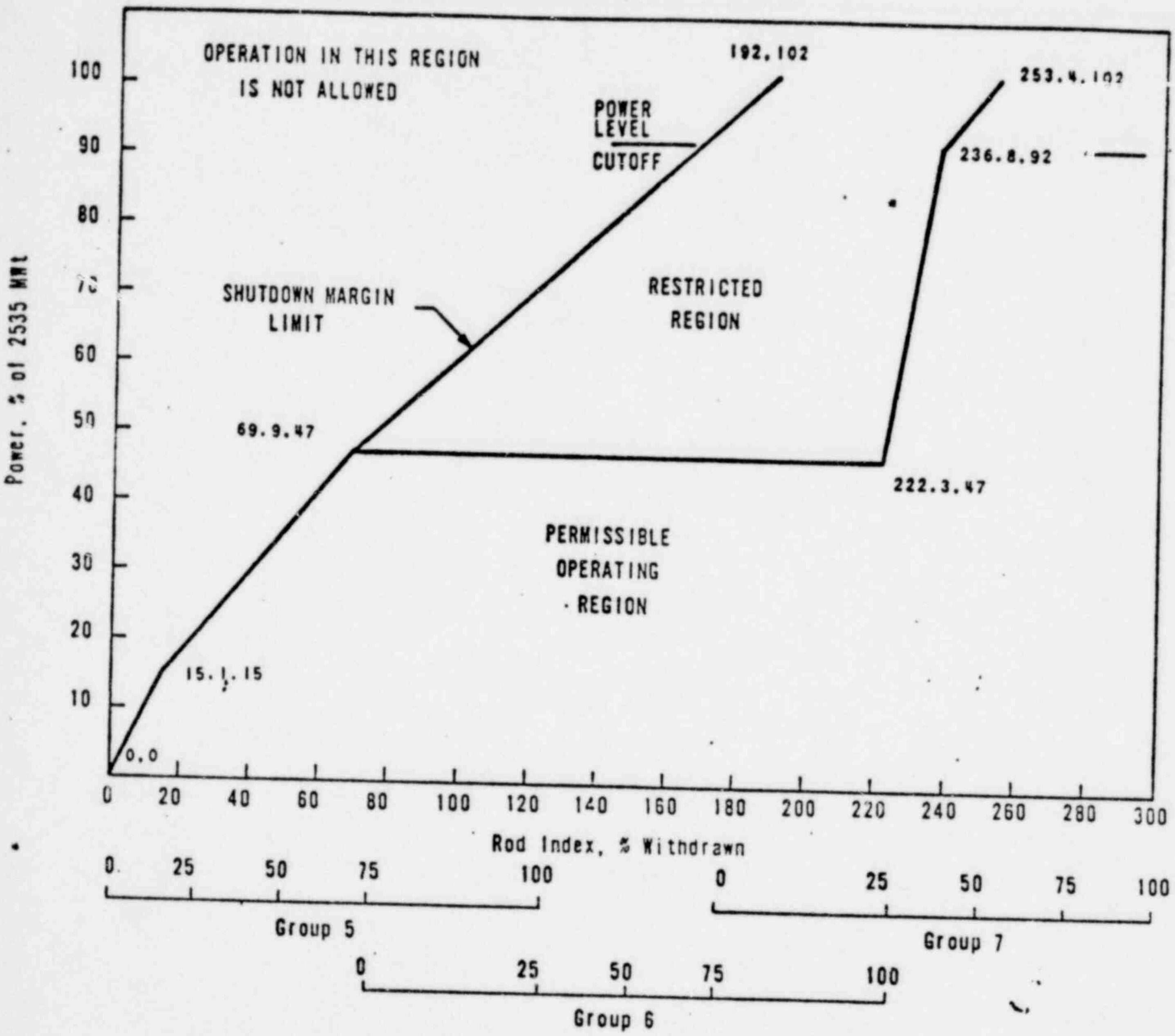
1487 270



ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 152 ± 10 TO 275 ± 10 EFPD; CYCLE 2

Figure 8-7

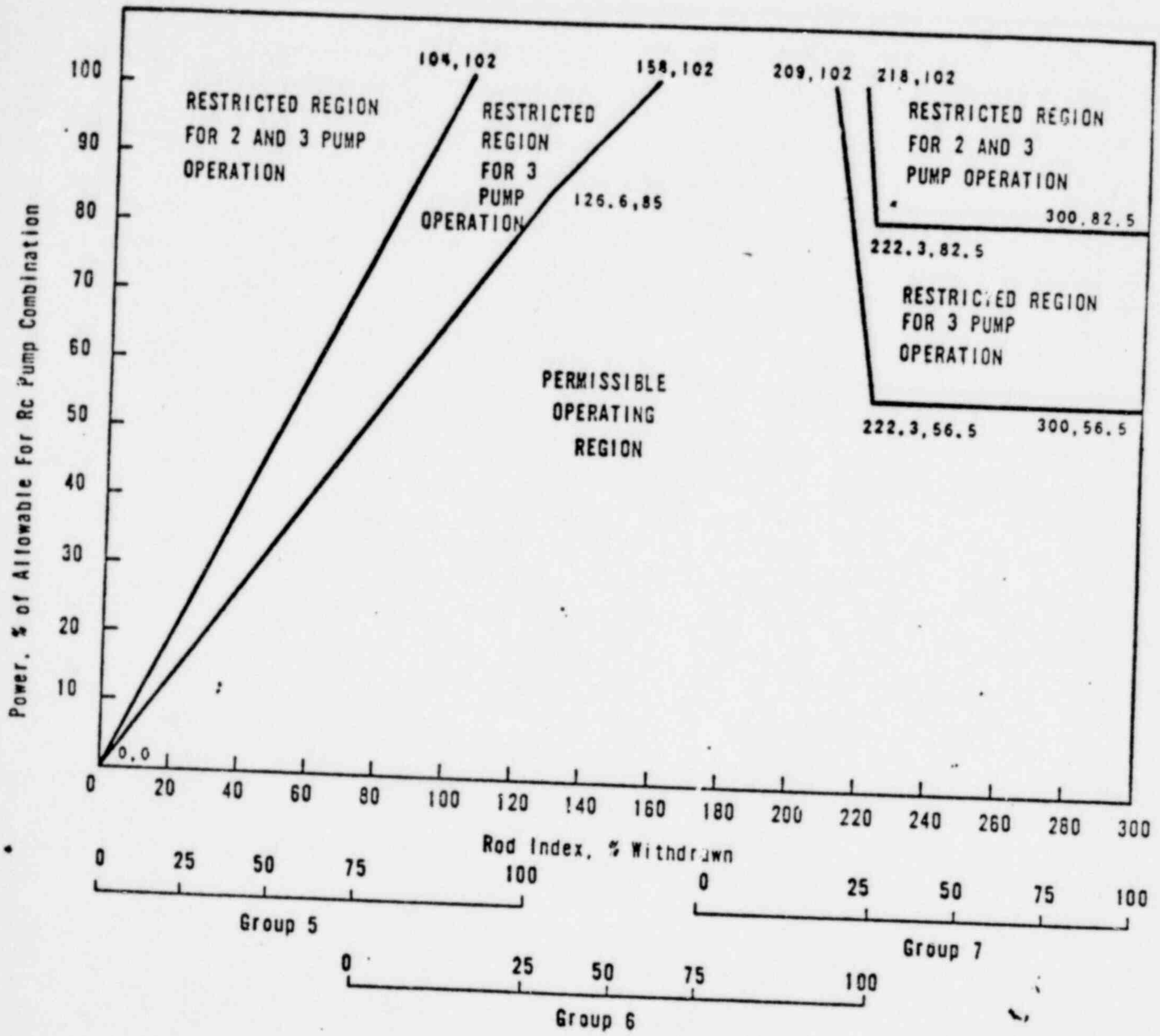
1487 271



ROD POSITION LIMITS FOR 4 PUMP OPERATION APPLICABLE DURING THE PERIOD AFTER 275 ± 10 EFPD; CYCLE 2

Figure 8-8

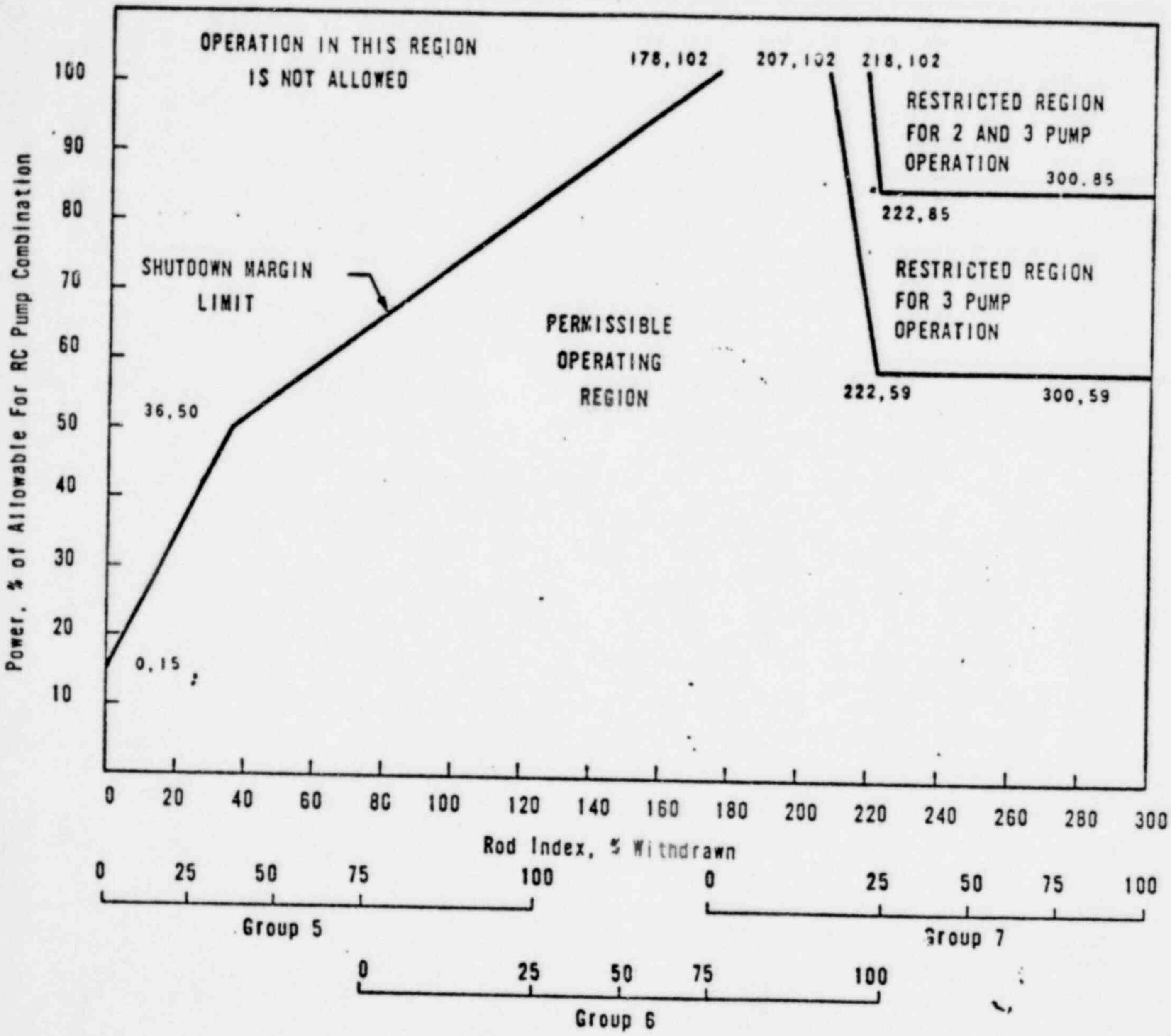
1487 272



ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 0 TO 152 ± 10 EFPD; CYCLE 2

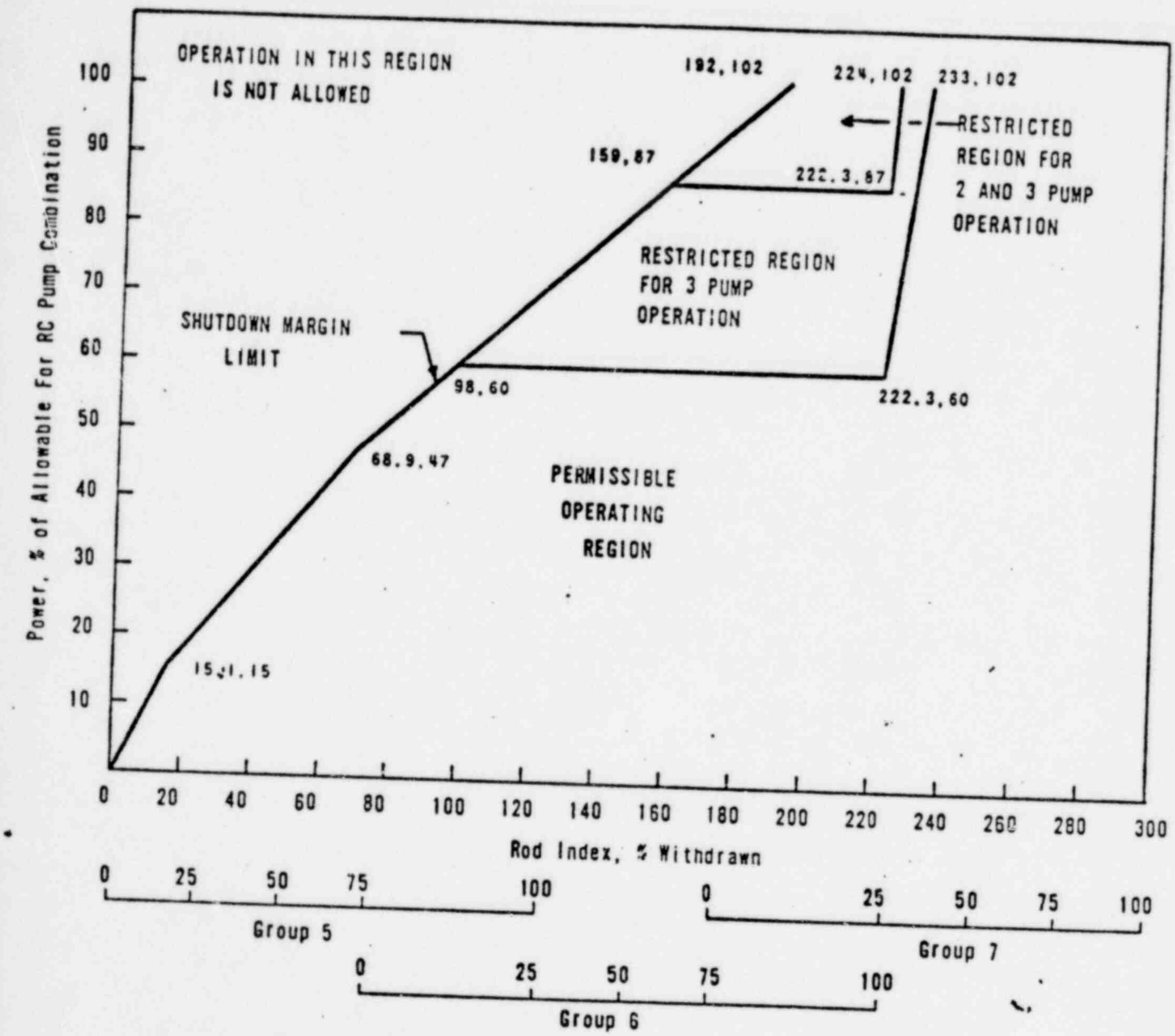
Figure 8-9

1487 273



ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE DURING THE PERIOD FROM 152 ± 10 TO 275 ± 10 EFPD; CYCLE 2

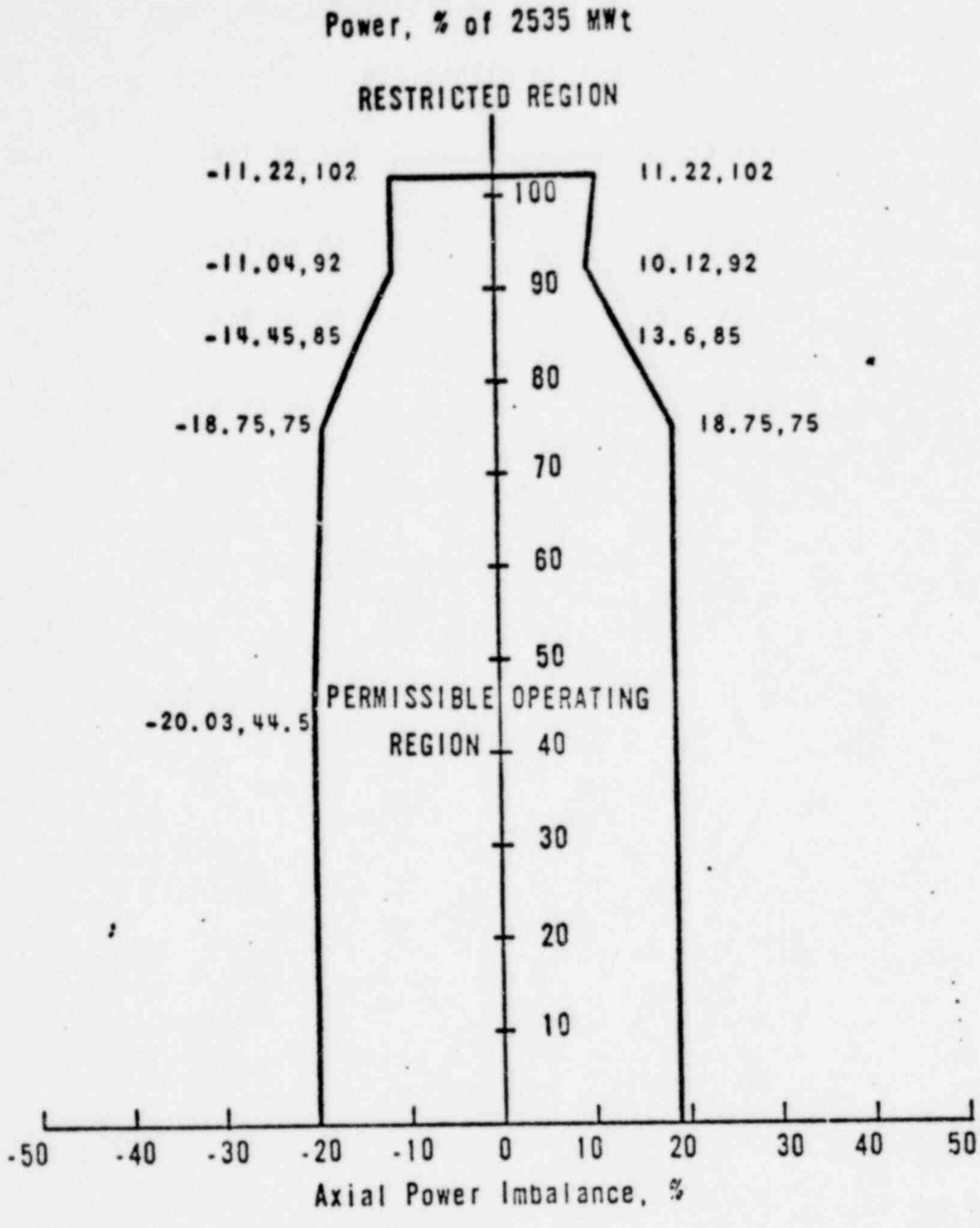
Figure 8-10



ROD POSITION LIMITS FOR 2 AND 3 PUMP OPERATION APPLICABLE DURING THE PERIOD AFTER 275 ± 10 EFPD; CYCLE 2

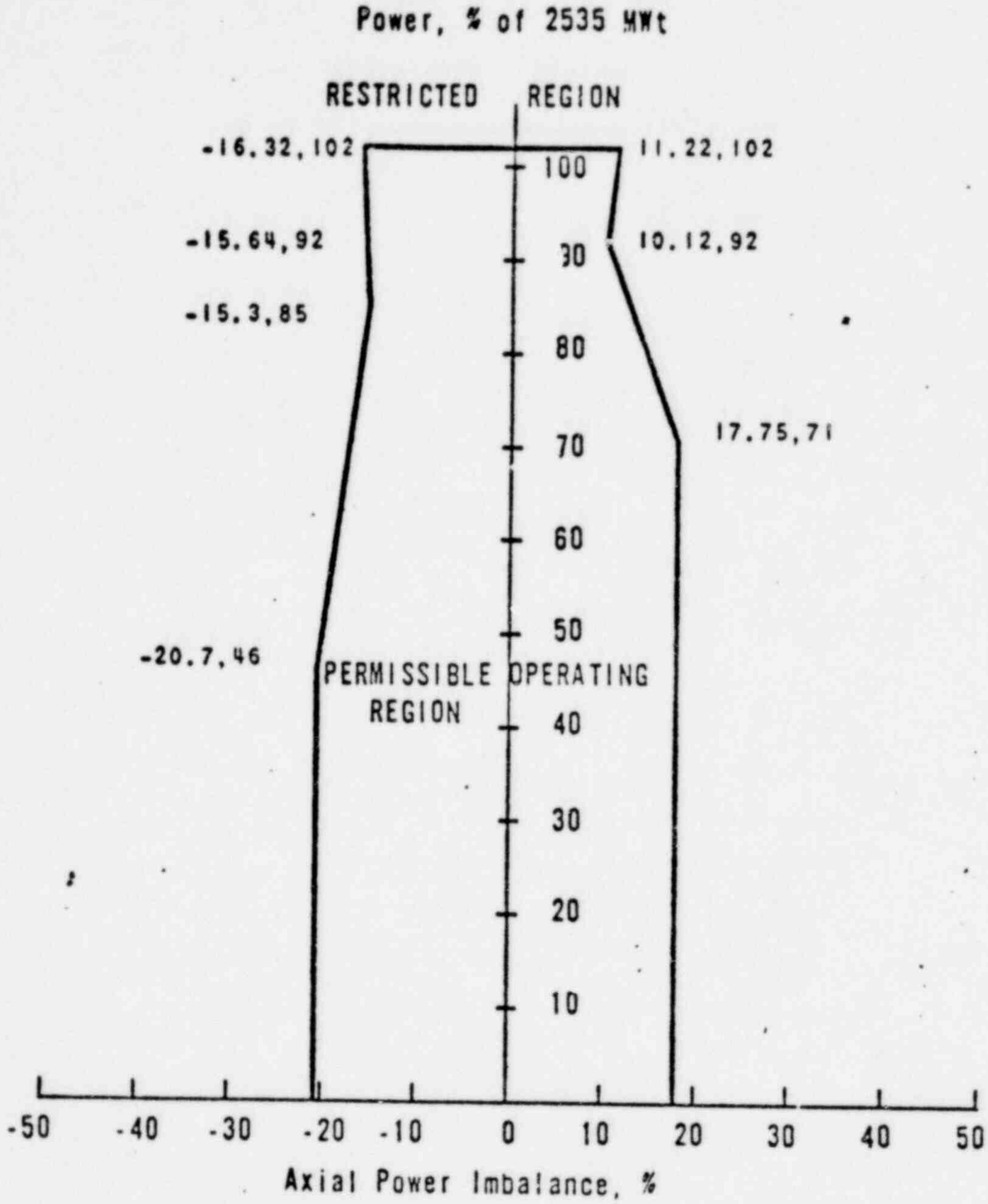
Figure 8-11

1487 275



OPERATIONAL POWER IMBALANCE ENVELOPE
APPLICABLE TO OPERATION FROM 0 TO 152
± 10 EFPD- CYCLE 2

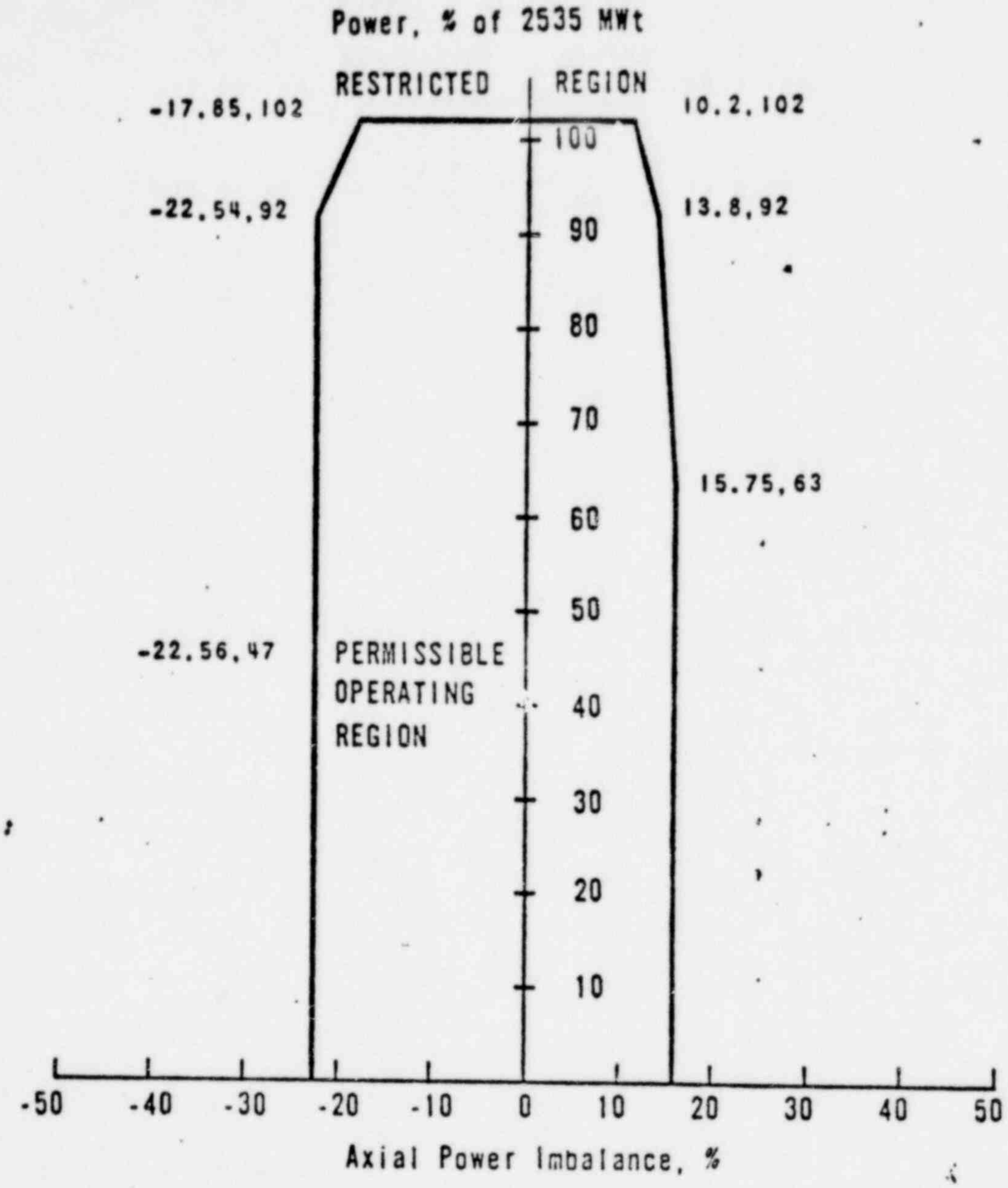
Figure 8-12



OPERATIONAL POWER IMBALANCE ENVELOPE APPLICABLE TO OPERATION FROM 152 ± 10 TO 275 ± 10 EFPD; CYCLE 2

Figure 8-13

1487 277



OPERATIONAL POWER IMBALANCE ENVELOPE APPLICABLE TO OPERATION AFTER 275 ± 10 EFPD; CYCLE 2

Figure 8-14

9. STARTUP PROGRAM

The planned startup testing associated with core performance are provided below. These tests verify that core performance is within the assumptions of the safety analysis and provide the necessary data for continued safe plant operation.

Zero Power Tests

1. Critical Boron Concentration
2. Temperature Reactivity Coefficient
3. Control Rod Group Worth
4. Ejected Rod Worth

Power Tests

1. Core Power Distribution Verification at Approximately 40, 75, and 100% FP Normal Control Rod Group Configuration
2. Core Power Distribution Verification at Approximately 40% FP With Worst Case Dropped Rod Fully Inserted
3. Incore/Out-of-Core Detector Imbalance Correlation Verification at Approximately 75% FP
4. Power Doppler Reactivity Coefficient at Approximately 100% FP
5. Temperature Reactivity Coefficient at Approximately 100% FP

1487 279

10. REFERENCES

1. Three Mile Island-Unit 1 Nuclear Station, Final Safety Analysis Report, Docket No. 50-289.
2. A.F.J. Eckert, H. W. Wilson, and K. E. Yoon, "Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse," BAW-10084P-A, Babcock & Wilcox, January 1975.
3. Three Mile Island Unit 1 Fuel Densification Report, BAW-1389, Babcock & Wilcox, June 1973.
4. C. D. Morgan and H. S. Kao, "TAFY - Fuel Pin Temperature and Gas Pressure Analysis," BAW-10044, Babcock & Wilcox, May 1972.
5. Fuel Densification Report, BAW-10055, Rev. 1, Babcock & Wilcox, June 1973.
6. "Densification Kinetics and Power Spike Model," Meeting with USAEC, July 3, 1974; J. F. Harrison (B&W) to R. Lobel (USAEC) Telecon, "Power Spike Factor," July 18, 1974.
7. Oconee 1 Cycle 2 - Reload Report, BAW-1409 (Rev. 1), Babcock & Wilcox, October 10, 1974.
8. "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," BAW-10000, Babcock & Wilcox, March 1970.
9. "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," BAW-10036, Babcock & Wilcox, February 1972.
10. "ECCS Analysis of B&W's 177 FA Lowered-Loop NSS," BAW-10103, Rev. 1, September 1975.

1487 280