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Revision 1
(January 1978)

THREE MILE ISLAND UNIT 1
CYCLE 4 RELOAD REPORT

1565 304

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1565 305

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the Three Mile Island Nuclear Station Unit 1 (TMI-1) (cycle 4) at a 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 4 operation of the TMI-1, this report employs analytical techniques and design bases established in reports that have been submitted and received technical approval by the USNRC (see references).

The design for cycle 4 is based on operation in the feed-and-bleed or rods-out mode. All nuclear parameters pertinent to accident analyses have been calculated consistent with this mode of operation. Section 5.3 describes the change to feed-and-bleed operation.

Cycle 4 reactor parameters that are related to power capability are summarized in this report and referenced to cycle 3. All the accidents analyzed in the FSAR have been reviewed for cycle 4 operation, and in cases where cycle 4 characteristics proved to be conservative with respect to those analyzed previously, no new analysis was performed.

The Technical Specifications have been reviewed, and the modifications required for cycle 4 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 for Emergency Core Cooling Systems (ECCS), it has been concluded that TMI-1, cycle 4 can be safely operated at the rated core power level of 2535 MWt.

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2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of the Three Mile Island, Unit 1 is the operating cycle 3. Cycle 3 achieved criticality on May 13, 1977, and after zero power testing attained 100% power on May 20, 1977. No control rod interchange was planned for cycle 3, which is scheduled for completion in early March after 270 ± 10 EFPD. No operating anomalies occurred during the first three cycles that would adversely affect fuel performance during the fourth cycle. The operation of cycle 4 is scheduled to begin in April 1978. The design cycle length is 265 ± 15 EFPD, and no control rod interchanges are planned.

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3. GENERAL DESCRIPTION

The TMI-1 reactor core is described in detail in Section 3 of the Final Safety Analysis Report for that Unit.[1] The cycle 4 core consists of 177 fuel assemblies (FAs), 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 undensified nominal active lengths of the fuel rods are 144 inches for batches 1c and 2b, 142.6 inches for batch 4, and 142.25 inches for batches 5 and 6. All fuel assemblies in cycle 4 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 (see Table 4-2 for data).

Figure 3-1 is the core loading diagram for TMI-1, cycle 4. The initial enrichments of batches 1c, 2b, and 4 were 2.06, 2.75, and 2.64 wt % uranium-235, respectively. Batches 5 and 6 have a 2.85 wt % uranium-235 enrichment. All the batch 3 assemblies will be discharged at the end of cycle 3, and the batch 4 and 5 assemblies will be shuffled to new locations. The batch 6 assemblies will occupy the periphery of the core. The 13 batch 1c and the eight batch 2b assemblies will occupy interior core locations. Note that the designations 1c and 2b are used to identify assemblies from the original batch 1 and batch 2 fuel. The 1c and 2b assemblies were removed from the core at the end of cycle 1 and cycle 2, respectively (see Table 4-1). They are being reinserted into the cycle 4 core to lower feed batch size requirements and spent fuel storage, thereby producing a more efficient fuel cycle. It should be noted that the assemblies referred to as batch 1a in the cycle 3 Reload Report (reference 2) are now designated as 1b; batch 1a is now the remainder of batch 1 assemblies which have not been scheduled for reinsertion. Figure 3-2 is an eighth-core map showing each assembly's burnup

at the beginning of cycle 4 and its initial enrichment.

Cycle 4 will be operated in a rods-out, feed-and-bleed mode. The core reactivity control will be supplied mainly by soluble boron and supplemented by 61 full-length, Ag-In-Cd control rod assemblies (CRAs). In addition to the full-length control rods, eight axial power shaping rods (APSRs) are provided for additional control of axial power distribution. The cycle 4 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The core locations of the 69 control rods for cycle 4 are identical to those of the reference cycle 3. However, the group designations differ between cycle 4 and the reference cycle to minimize power peaking. No control rod interchanges or burnable poison rods are necessary for cycle 4. The nominal system pressure is 2200 psia, and the core average densified nominal linear heat rate is 5.72 kW/ft at the rated core power of 2535 MWt. The heat rate is slightly higher than in the reference cycle 3 (5.71 kW/ft) due to the shorter stack height of batch 6 relative to the discharged batch 3.

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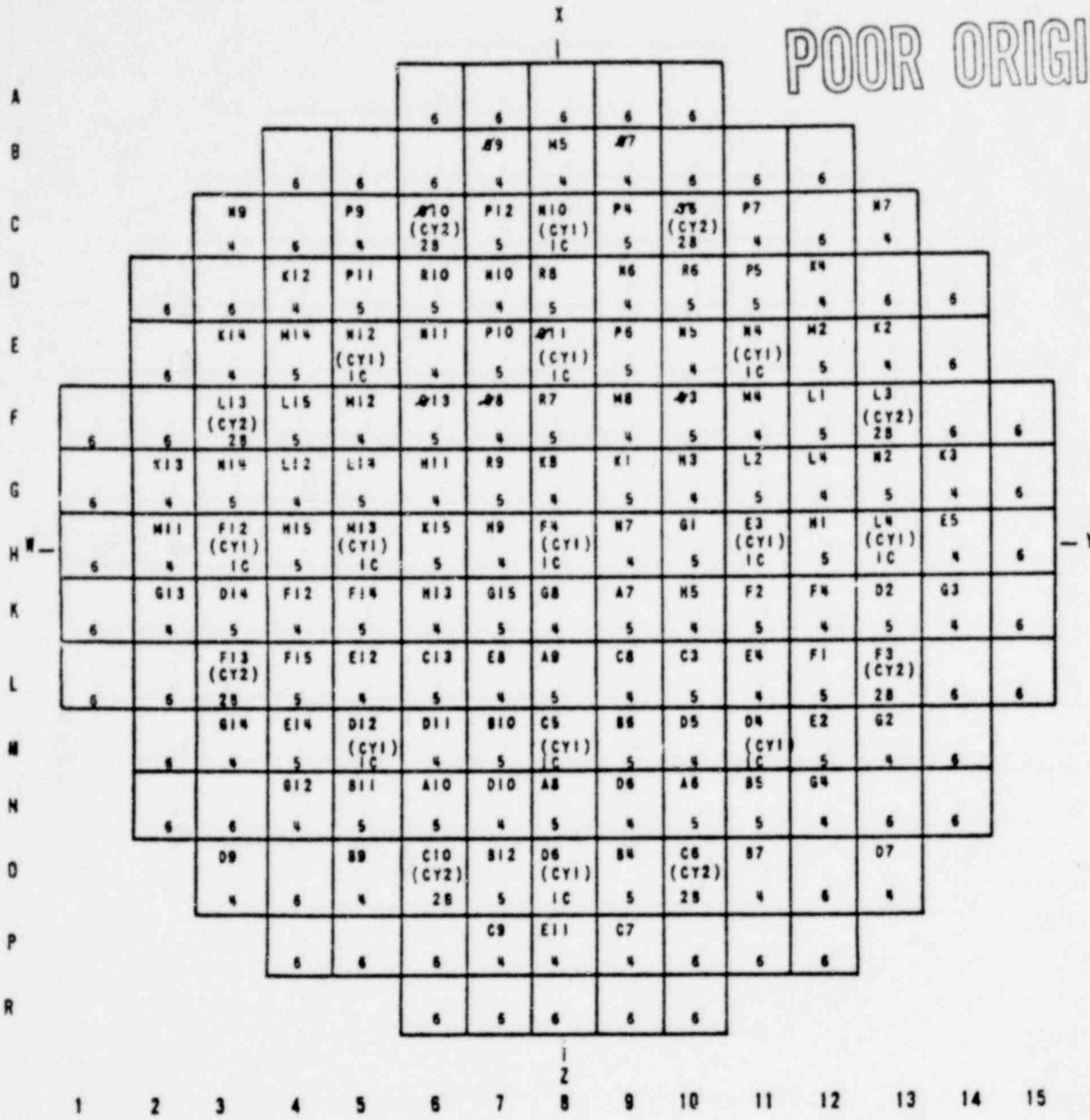
Figure 3-1. Core Loading Diagram for TMI-1, Cycle 4



POOR ORIGINAL

XXX	Previous Cycle Location (except as noted)
X	Batch

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Figure 3-2. Enrichment and Burnup Distribution for TMI-1, Cycle 4

	8	9	10	11	12	13	14	15
H	2.06 14263	2.64 16182	2.85 7647	2.06 13600	2.85 7056	2.06 14263	2.64 16071	2.85 0
K		2.85 7636	2.64 18217	2.85 10723	2.64 14918	2.85 5176	2.64 19704	2.85 0
L			2.95 4792	2.64 14740	2.85 6690	2.75 23049	2.85 0	2.85 0
M				2.06 11718	2.85 8304	2.64 16979	2.85 0	
N					2.64 15685	2.85 0	2.85 0	
O						2.64 15689		
P								
R								

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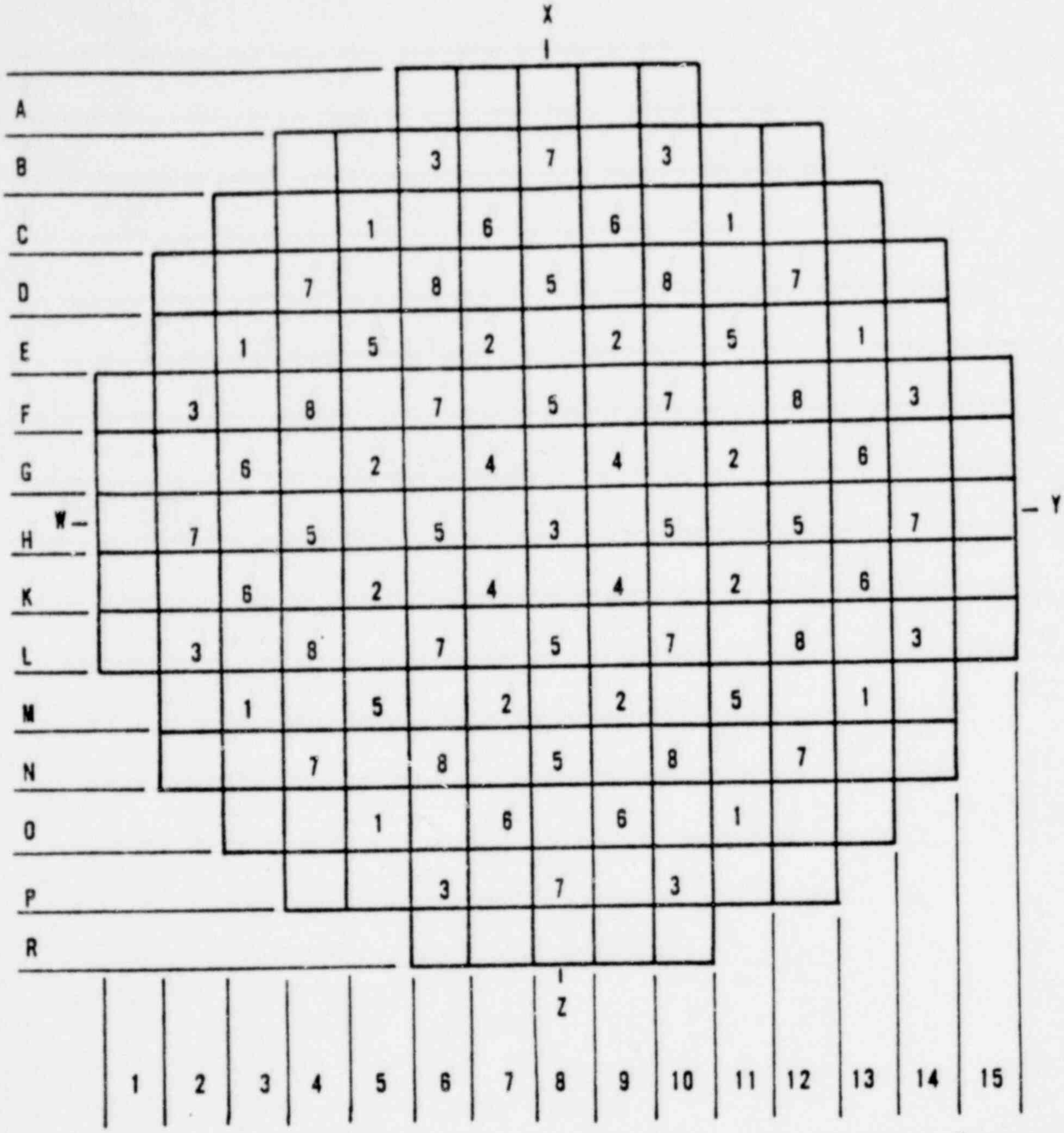
X.XX	Initial Enrichment
XXXXX	BOC Burnup

Figure 3-3. Control Rod Locations for TMI-1, Cycle 4



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

total # 69

X Group Number

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4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for TMI-1, cycle 4 are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. All results, references, and identified conservatisms presented in section 4.1 of reference 2 are applicable to the cycle 4 reload core.

4.2. Fuel Rod Design

The mechanical evaluation of the fuel rod is discussed below.

4.2.1. Cladding Collapse

Creep collapse analyses were performed for three cycle assembly power histories. The batch 2b reinserted fuel is more limiting than the other batches due to lower prepressurization, lower pellet density, and/or longer previous incore exposure time. The batch 2b assembly power histories were analyzed, and the most limiting assembly was determined.

The power history for the most limiting assembly was used to calculate the fast neutron flux level for the energy range above 1 MeV. The collapse time for the most limiting assembly was conservatively determined to be more than 30,000 EFPH (effective full power hours), which is greater than the maximum projected residence time (Table 4-1). The creep collapse analysis was performed based on the conditions set forth in references 2 and 3.

4.2.2. Cladding Stress

The batch 1c and 2b reinserted fuel is the most limiting for cladding stress. The results presented in reference 4 are applicable.

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% on cladding plastic circumferential strain. The pellet design is established for a plastic cladding strain

of less than 1% at values of maximum design local pellet burnup and heat generation rate, which are considerably higher than the values that the TMI-1 fuel is expected to see. This will result in an even greater margin than the analysis demonstrated. The strain analysis is also based on the maximum allowable value for the fuel pellet diameter and density and the lowest permitted tolerance for the cladding ID.

4.3. Thermal Design

The incoming batch 6 fuel is thermally and geometrically similar to the batch 5 fuel of cycle 3. The TAFY[5] fuel pin analysis performed for batch 5 fuel also applies to batch 6. An analysis was also performed for batch 6 using the fuel performance code, TACO[6]. Where differences occurred between corresponding calculated values of each code, the more conservative values were chosen for batch 6 design.

Thermal analysis of the fuel rods assumed in-reactor densification to 96.5% TD. The average fuel temperatures (Table 4-2) for batches 1 through 5 are taken from the TAFY analyses which define the linear heat rate (LHR) capability for each batch. The average temperature shown for batch 6 was taken from an average pin analysis using the TACO code. The value shown represents the BOL (100 MWd/mtU) average fuel temperature at 5.80 kW/ft. The average temperature decreases with burnup to a value of 1120F at 38,000 MWd/mtU.

Linear heat rate capabilities are based on centerline fuel melt. Batch 6 linear heat rate capability was determined based on the lower tolerance limit (LTL) of the fuel density specification. The design LHR capability used for batch 6 was 20.15 kW/ft, which was calculated by the TAFY code and is the same as that for batches 4 and 5. The TACO analysis for batch 6 fuel gives a higher LHR capability. Therefore, the more conservative TAFY LHR capability was used in the design of batch 6.

4.4. Material Design

The chemical compatibility of all possible fuel-cladding-coolant assembly interactions for the batch 6 fuel assemblies is identical to that of the present fuel.

4.5. Operating Experience

B&W's operating experience with Mark B, 15-by-15 fuel assembly design has verified the adequacy of the fuel assembly design. As of August 31, 1977, the following operating experience has been collected for the seven B&W 177-FA plants using the Mark B fuel assembly:

Reactor	Current cycle	Current cycle max assembly burnup, MWd/mtU	Cumulative net electrical output, MWh
Oconee 1	3	26,300	18,134,699
Oconee 2	3	23,400	13,475,779
Oconee 3	2	25,700	13,907,914
TMI Unit 1	3	26,700	15,259,750
ANO Unit 1	2	24,294	12,044,505
Rancho Seco	1	19,664	8,328,383
Crystal River 3	1	4,400	1,881,750

Table 4-1. Fuel Design Parameters and Dimensions

	Twice-burned assys		Once-burned assys		Fresh fuel assemblies,
	Batch 2b	Batch 4	Batch 1c	Batch 5	Batch 6
Fuel assembly type	Mark B3	Mark B4	Mark B2	Mark B4	Mark B4
No. of assemblies	8	56	13	48	52
Fuel rod OD, in.	0.430	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377	0.377
Flexible spacers, type	Corr'd	Spring	Corr'd	Spring	Spring
Rigid spacers, type	ZrO_2	Zr-4	ZrO_2	Zr-4	Zr-4
Undensified active fuel length, in.	144.0	142.6	144.0	142.25	142.25
Fuel pellet OD (mean specified), in.	0.3700	0.3700	0.3700	0.3695	0.3695
Fuel pellet initial Density, % TD	92.5	93.5	92.5	94.0	94.0
Initial fuel enrichment wt % ^{235}U	2.75	2.64	2.06	2.85	2.85
Burnup (BOC) MWd/mtU	23,049	16,625	13,276	7,410	0
Cladding collapse Time, EFPH	>30,000	>30,000	>30,000	>30,000	>30,000
Residence Time, EFPH	23,976	19,272	17,736	19,200	19,080

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Table 4-2. Fuel Thermal Analysis Parameters

Densified fuel parameters[a]	Batch 1c	Batch 2b	Batch 4	Batch 5	Batch 6
Pellet diameter, in.	0.3640	0.3640	0.3645	0.3646	0.3646
Fuel stack height, in.	141.12	141.12	140.46	140.47	140.47
Nominal LHR at 2568 MWt, kW/ft	5.77	5.77	5.80	5.80	5.80
Avg fuel temp at nominal LHR, °F (BOL)	1335	1335	1320	1315	1400
LHR to C_L fuel melt, kW/ft	19.6	19.6	20.15	20.15	20.15

[a] Densification to 96.5% TD assumed.

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5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of cycles 3 and 4; these values were generated using PDQ07 for both cycles. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles.

The longer design life of cycle 4 will produce a corresponding larger cycle differential burnup than designed for the reference cycle 3. The lower accumulated average core burnup at the end of cycle 4 is mainly due to the discharge at the end of cycle 3 of batch 3 fuel which had a high burnup history. Figure 5-1 illustrates a representative relative power distribution for the beginning of the cycle 4 at full power with equilibrium xenon and group 8 inserted.

The critical boron concentrations are approximately the same as those of reference cycle 3. The hot, full-power control rod worths are similar in both cycles except for group 7, which is significantly higher in cycle 4, being composed of 12 control rod assemblies rather than eight as in cycle 3. Control rod worths are sufficient to maintain the required shutdown margin as indicated in Table 5-2. The differences in the parameters between cycles 3 and 4 are due to changes in radial flux distribution, isotopics, and the difference in cycle lengths. The ejected rod worth in Table 5-1 are the maximum calculated values. It is difficult to compare values between cycles or between rod patterns since neither the rod patterns from which the CRA is ejected nor the isotopic distributions are identical. Calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the rod position limits presented in section 8. The maximum stuck rod worth at the end of cycle 4 is similar to that for the reference cycle 3 but is lower at the beginning of the cycle.

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The following conservatisms were applied for the shutdown calculations:

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

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The shutdown calculation was analyzed at 277 EFPD. The maximum allowable inserted rod worth is smaller in cycle 4 than in cycle 3 due to the operation of this cycle in a feed-and-bleed mode in which control rod group 7 is only partially inserted in the core during the entire cycle.

The cycle 4 power deficits from hot zero power to hot full power are lower than those for cycle 3 due to the less negative moderator coefficients in cycle 4. The differential boron and xenon worths are similar for cycles 3 and 4. The effective delayed neutron fractions for cycle 4 show a decrease with burnup (also reported in the reference cycle 3).

5.2. Analytical Input

The cycle 4 incore measurement calculation constants used for computing core power distributions were prepared in the same manner as for the reference cycle.

5.3. Changes in Nuclear Design

Cycle 4 is designed to operate in a feed-and-bleed mode in contrast to the rodded operation of cycles 1, 2, and 3. The major difference in operational modes during equilibrium, steady-state conditions is that no full-length control rods are inserted into the core. (A small, bite insertion, approximately 10%, of one regulating bank is maintained to allow discrete changes in soluble boron and to accommodate small temperature and load demand changes.) During load follow operation the regulating bank is inserted into the core only to offset power Doppler reactivity changes. Transient xenon reactivity effects are compensated by changing the soluble boron concentration.

The same calculational methods and design information used in reference cycle 3 were used to obtain the important nuclear design parameters in cycle 4. Additional calculations were performed for soluble boron control, shutdown, reactivity control, and radiation analyses due to the modification in the mode of operation. As in cycle 3, both APSRA and CRA position limits, as well as power imbalance limits,

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will be specified based on LOCA analyses. These operational limits and the RPS limits (Technical Specification changes) for cycle 4 are presented in section 8.

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Table 5-1. TMI-1, Cycle 4 Physics Parameters[a]

	Cycle 3[b]	Cycle 4
Cycle length, EFPD	270	277
Cycle burnup, MWd/mtU	8341	8557
Average core burnup — EOC, MWd/mtU	18,352	17,844
Initial core loading, mtU	82.1	82.1
Critical boron — BOC, ppm (no Xe)		
HZP[c], group 8 (37.5% wd)	1317	1250
HZP, groups 7 and 8 inserted	1155	1115
HFP, group 8 inserted	998	1084
Critical boron — EOC, ppm (eq Xe)		
HZP group 8 (37.5% wd, eq Xe)	380	308
HFP group 8 (37.5% wd, eq Xe)	84	47
Control rod worths — HFP, BOC, % Δk/k		
Group 6	1.18	1.10
Group 7	0.84	1.48
Group 8 (37.5% wd)	0.54	0.46
Control rod worths — HFP, EOC,[d] % Δk/k		
Group 7	1.11	1.57
Group 8 (37.5% wd)	0.50	0.50
Max ejected rod worth — HZP, % Δk/k[e]		
BOC	0.34	0.81
EOC	0.77	0.81
Max stuck rod worth — HZP, % Δk/k		
BOC	2.42	1.95
EOC	2.06	2.03
Power deficit, HZP to HFP, % Δk/k		
BOC	-1.58	-1.28
EOC	-2.15	-2.05
Doppler coeff — BOC, 10^{-5} (Δk/k/°F)		
100% power (0 Xe)	-1.47	-1.49
Doppler coeff — EOC, 10^{-5} (Δk/k/°F)		
100% power (eq Xe)	-1.51	-1.59
Moderator coeff — HFP, 10^{-4} (Δk/k/°F)		
BOC (0 Xe, 1084 ppm, group 8 ins)	-0.91	-0.63
EOC (eq Xe, 17 ppm, group 8 ins)	-2.54	-2.52
Boron worth — HFP, ppm/% Δk/k		
BOC (1000 ppm)	107	105
EOC (17 ppm)	97	95
Xenon worth — HFP, % Δk/k		
BOC (4 EFPD)	2.59	2.63
EOC (equilibrium)	2.64	2.73

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Table 5-1. (Cont'd)

	Cycle 3[a]	Cycle 4
Effective delayed neutron fraction — HFP		
BOC	0.00584	0.000586
EOC	0.00524	0.00522

[a] Cycle 4 data are for the conditions stated in this report. The cycle 3 core conditions are identified in reference 2.

[b] Based on 253 EFPD at 2535 MWt, cycle 2.

[c] HZP denotes hot zero power (532F T_{avg}); HFP denotes hot full power (579F T_{avg}).

[d] 246 EFPD in cycle 3; 277 EFPD in cycle 4.

[e] Ejected rod worth for groups 5 through 8 inserted.

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Table 5-2. Shutdown Margin Calculation for TMI-1, Cycle 4

	BOC, % Δk/k	EOC,[a] % Δk/k
Available Rod Worth		
Total rod worth, HZP[b]	8.71	8.81
Worth reduction due to burnup of poison material	-0.37	-0.46
Maximum stuck rod, HZP	-1.95	-2.03
Net worth	6.39	6.32
Less 10% uncertainty	-0.64	-0.63
Total available worth	5.75	5.69
Required Rod Worth		
Power deficit, HFP to HZP	1.28	2.05
Max allowable inserted rod worth	0.40	0.42
Flux redistribution	0.40	0.90
Total required worth	2.08	3.37
Shutdown Margin		
Total available - total required	3.67	2.32

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

[a] 277 EFPD.

[b] HZP denotes hot zero power (532F T_{avg}); HFP denotes hot full power (579F T_{avg}).

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Figure 5-1. BOC (4 EFPD), Cycle 4 Two-Dimensional Relative Power Distribution — Full Power, Equilibrium Xenon, APSRs Inserted

	8	9	10	11	12	13	14	15
H	.99	1.14	1.32	1.01	1.25	.91	.81	.76
K		1.33	1.13	1.21	1.08	1.17	.83	.76
L			1.37	1.09	8 1.07	.90	1.13	.65
M				1.01	1.15	.93	.96	
N					1.00	1.10	.67	
O						.52		
P								
R								

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X — INSERTED ROD GROUP NUMBER

X.XX — RELATIVE POWER DENSITY

6. THERMAL-HYDRAULIC DESIGN

The incoming batch 6 fuel is hydraulically and geometrically similar to batch 5 fuel. The only difference between cycles 3 and 4 is the core configuration. The cycle 2 DNBR analysis was used for reference cycle 3[2]; this analysis is also valid for cycle 4 as discussed below. The core configuration used for cycle 2 analysis consisted of 60 Mark B3 assemblies and 117 Mark B4 assemblies with the most limiting (hot) assembly being a B3 assembly. The cycle 4 configuration consists of 13 Mark B2, 8 March B3, and 156 Mark B4 assemblies with the most limiting assembly being a B4. Both the Mark B2 and B3 assemblies have a higher resistance to flow than the Mark B4 assembly.

The minimum DNBR calculated from the cycle 2 analysis was compared to the minimum DNBR obtained from an analysis of an all B4 core. The cycle 2 analysis provided the more restrictive minimum DNBR. For cycle 4 the addition of the higher resistance Mark B2 and B3 assemblies provides additional DNBR margin. The higher resistance Mark B2 and B3 assemblies will tend to increase flow through the limiting Mark B4 assembly. Therefore, the cycle 2 DNBR analysis bounds that for cycle 4.

This conservatism is amplified when peaking factors are considered. The minimum DNBR calculated from the cycle 2 analysis is based on a radial-local peak of 1.783. The maximum radial-local peak calculated for cycle 4 operation, including 8% nuclear uncertainity, is 1.637 at BOL. This decreases to 1.421 at the end of cycle 4. This provides an 8.2% margin to the design peak at BOC and a 20.3% margin at EOC. This margin ensures that the design conditions shown in Table 6-1 will not be exceeded during cycle 4 operation.

All other thermal hydraulic analyses which were applicable to cycle 3 remain applicable to cycle 4.

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Table 6-1. Thermal-Hydraulic Design Conditions

	Cycle 1	Cycles 2, 3, and 4
Power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design	100.0	106.5
Vessel inlet coolant temperature at 100% power, F	554.0	555.6
Vessel outlet coolant temperature at 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.	Table 4-2	Table 4-2
Avg heat flux (100% power), Btu/h-ft^2	171,470	174,870
Maximum heat flux (100% power), Btu/h-ft^2 (for DNBR calculation)	457,825	466,903
CHF correlation	W-3	B&W-2
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Minimum DNBR (densified fuel)	1.46 (114% power)	1.877 (112% power)

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7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 4 parameters to determine the effect of the cycle 4 reload and to ensure that thermal performance during hypothetical transients is not degraded.

The effects of fuel densification on the FSAR accident results have been evaluated and are reported in reference 4. Since batch 6 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in reference 4, the conclusions in that reference are still valid.

The dose evaluations in the FSAR were based on conservative values for fuel burnup and power peaking; however, improved fuel utilization and improved calculational methods have led to a higher plutonium-to-uranium fission ratio. Since plutonium has a higher iodine fission yield than uranium, more iodine activity will be produced and thus the thyroid doses will be slightly higher than reported in the FSAR.

A comparison has been made between the 2-hour thyroid doses associated with the major accidents in Chapter 14 of the FSAR[1] with the 2-hour thyroid doses that would result from the cycle 4 iodine activity inventory. The results show that although the thyroid doses for cycle 4 increase by 8 to 15% over the FSAR, the cycle 4 doses are still only a very small fraction of 10 CFR 100 limits.

7.2. Accident Evaluation

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

Core thermal properties used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. Comparisons of

first core values (FSAR values) of core thermal parameters and subsequent fuel batches to parameters used in cycle 4 analyses are given in Table 4-2. A comparison of the cycle 4 thermal-hydraulic maximum design conditions to the previous cycle values is presented in Table 6-1. These parameters are common to all the accidents considered in this report. A comparison of the key kinetics parameters from the FSAR and cycle 4 is provided in Table 7-1.

A generic LOCA analysis has been performed for the B&W 177-FA lowered loop NSS using the Final Acceptance Criteria ECCS evaluation model reported in reference 7. This analysis is generic in nature since the limiting values of the key parameters for all plants in this category were used. Furthermore, the combination of the average fuel temperature as a function of linear heat rate and the lifetime pin pressure data used in the LOCA limits analysis (reference 7) is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in reference 7 provide conservative results for the operation of TMI-1, cycle 4 fuel. A tabulation showing the bounding values for allowable LOCA peak linear heat rates for TMI-1, cycle 4 fuel are provided in Table 7-2.

It is concluded by examination of cycle 4 core thermal properties and kinetics properties with respect to acceptable previous cycle values that this core reload will not adversely affect the ability to safely operate the TMI-1 plant during cycle 4. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 4 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 4 are bounded by the FSAR and/or the fuel densification report[4] and/or subsequent cycle analyses.

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

Parameter	FSAR and densif'n report value	Predicted value
Doppler coeff (BOC), Δk/k/°F	-1.17×10^{-5}	-1.49×10^{-5}
Doppler coeff (EOC), Δk/k/°F	-1.33×10^{-5}	-1.59×10^{-5}
Moderator coeff (BOC), Δk/k/°F	$+0.5 \times 10^{-4}$	-0.63×10^{-4}
Moderator coeff (EOC), Δk/k/°F	-3.0×10^{-4}	-2.52×10^{-4}
All rod group worth (HZP) % Δk/k	10.0	8.71
Initial boron conc (HZP), ppm	1200	1084
Boron reactivity worth (70°F), ppm/1% Δk/k	75	74
Max ejected rod worth (HFP), % Δk/k	0.65	0.28
Dropped rod worth (HFP), % Δk/k	0.46	0.20

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

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

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8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 4 operation. The changes were made as a result of the following:

1. TMI-1 will be changed from a rodded to a feed-bleed mode of operation for cycle 4. This change is not regarded as a major change in the operating mode since TMI-1 was operated in essentially a rods-out configuration during the latter part of the previous cycles. Imbalance limits, control rod position limits, and APSR position limits are utilized to control power peaking and linear heat rates. A power level cutoff of 92% full power is used to control power peaking due to transient xenon effects.
2. The quadrant tilt limit will be changed from a maximum actual core tilt of 3.41% to a maximum actual core tilt limit of 4.92%. A maximum actual core tilt limit of 4.92% was used in cycles 1 and 2, whereas cycle 3 used a maximum actual core tilt limit of 3.41%. The larger tilt limit was used for cycle 4 due to the larger operating windows for feed-bleed operation.
3. The Technical Specification limits based on DNBR and LHR criteria include appropriate allowances for projected fuel rod bow penalties, i.e., potential reduction in DNBR and an increase in power peaks. A statistical combination of the nuclear uncertainty factor, engineering hot channel factor, and rod bow peaking penalty was used in evaluating LHR criteria, as approved in reference 8.
4. Per reference 9, the power spike penalty due to fuel densification was not used in setting the DNBR- and ECCS-dependent Technical Specification limits.

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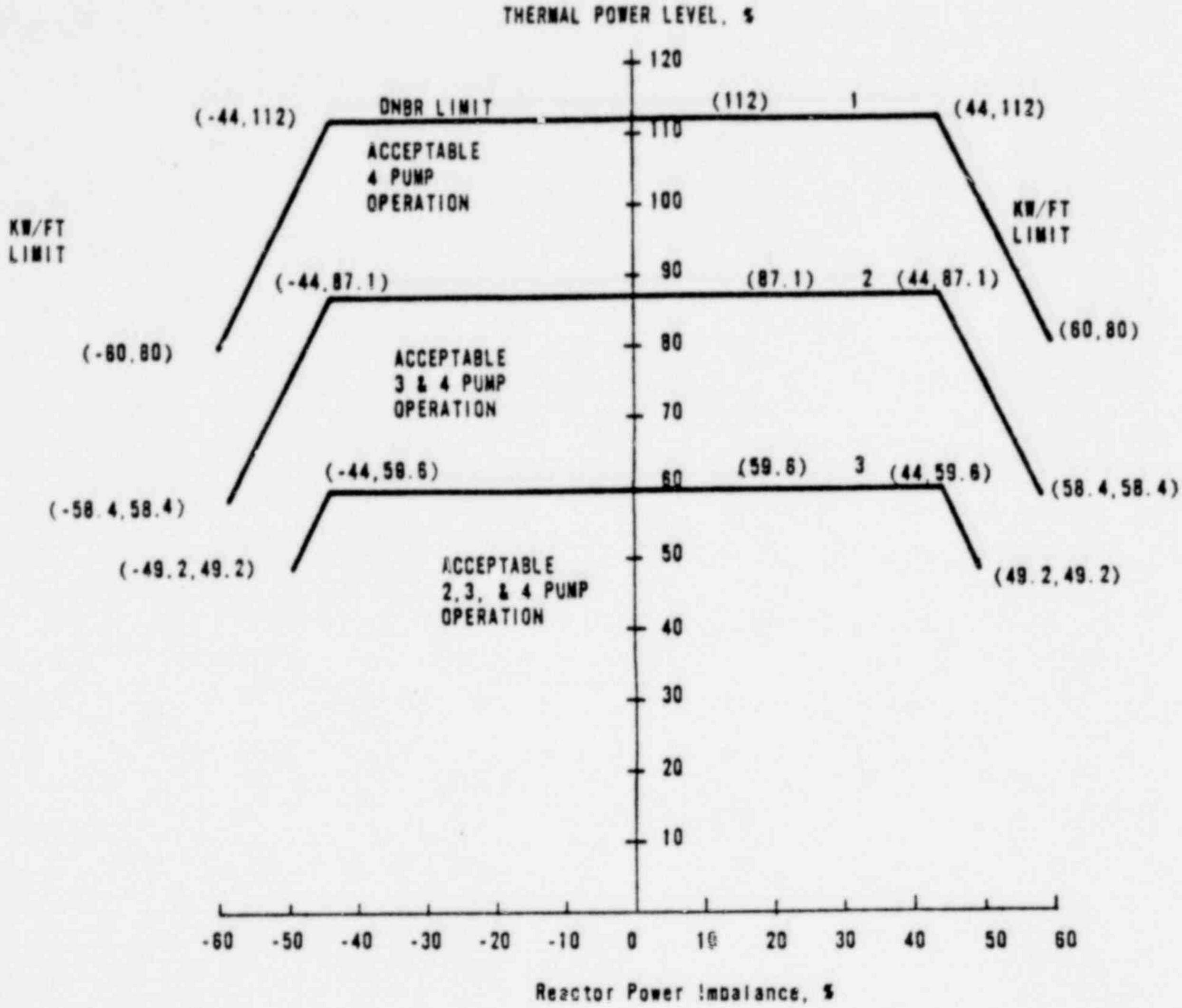
Figure 8-1. Core Protection Safety Limits



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

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Figure 8-2. Protection System Maximum Allowable Setpoints for Reactor Power Imbalance

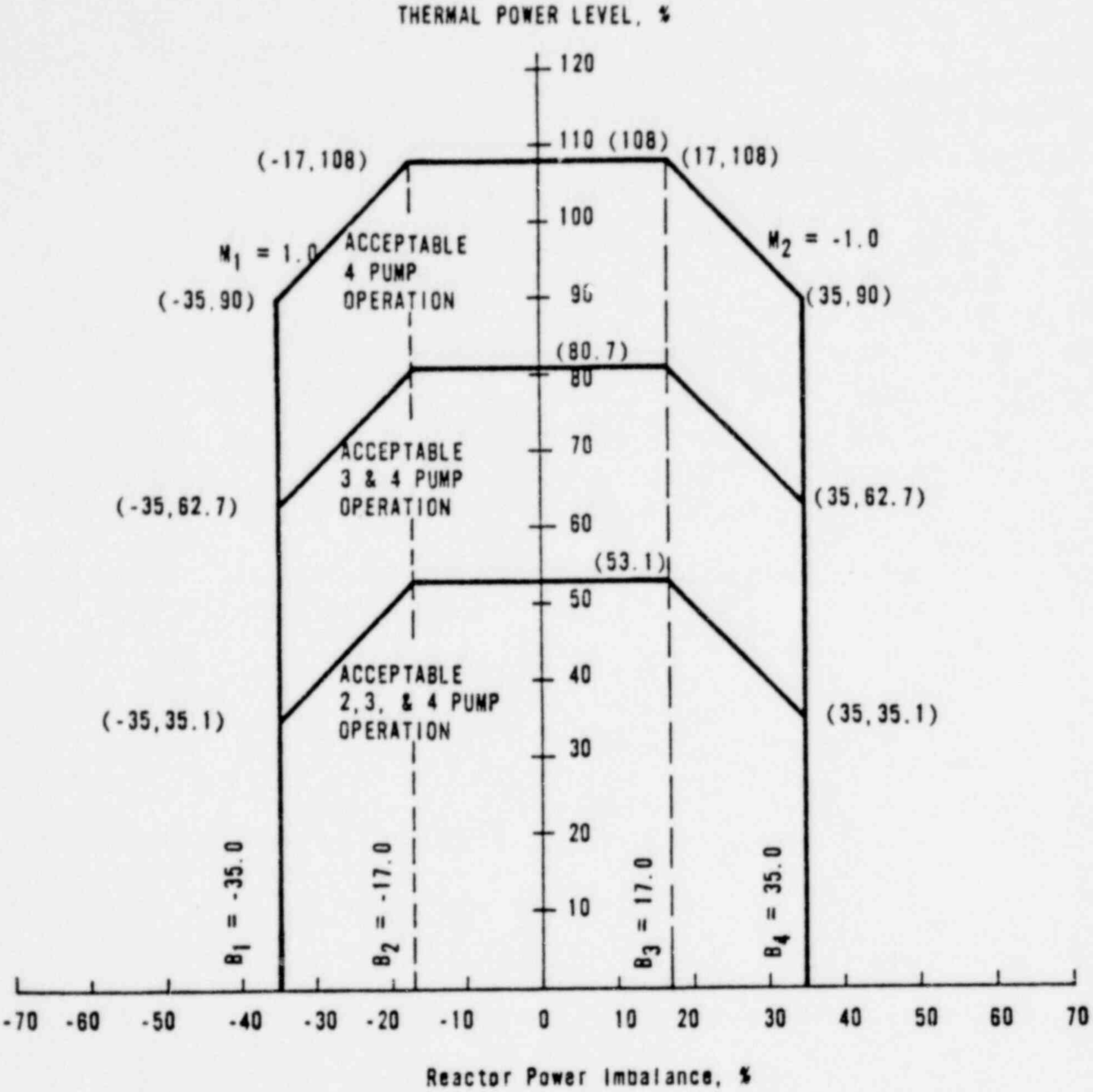

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Figure 8-3. Rod Position Limits for Four-Pump Operation From 0 to 125 ± 5 EFPD — TMI-1, Cycle 4

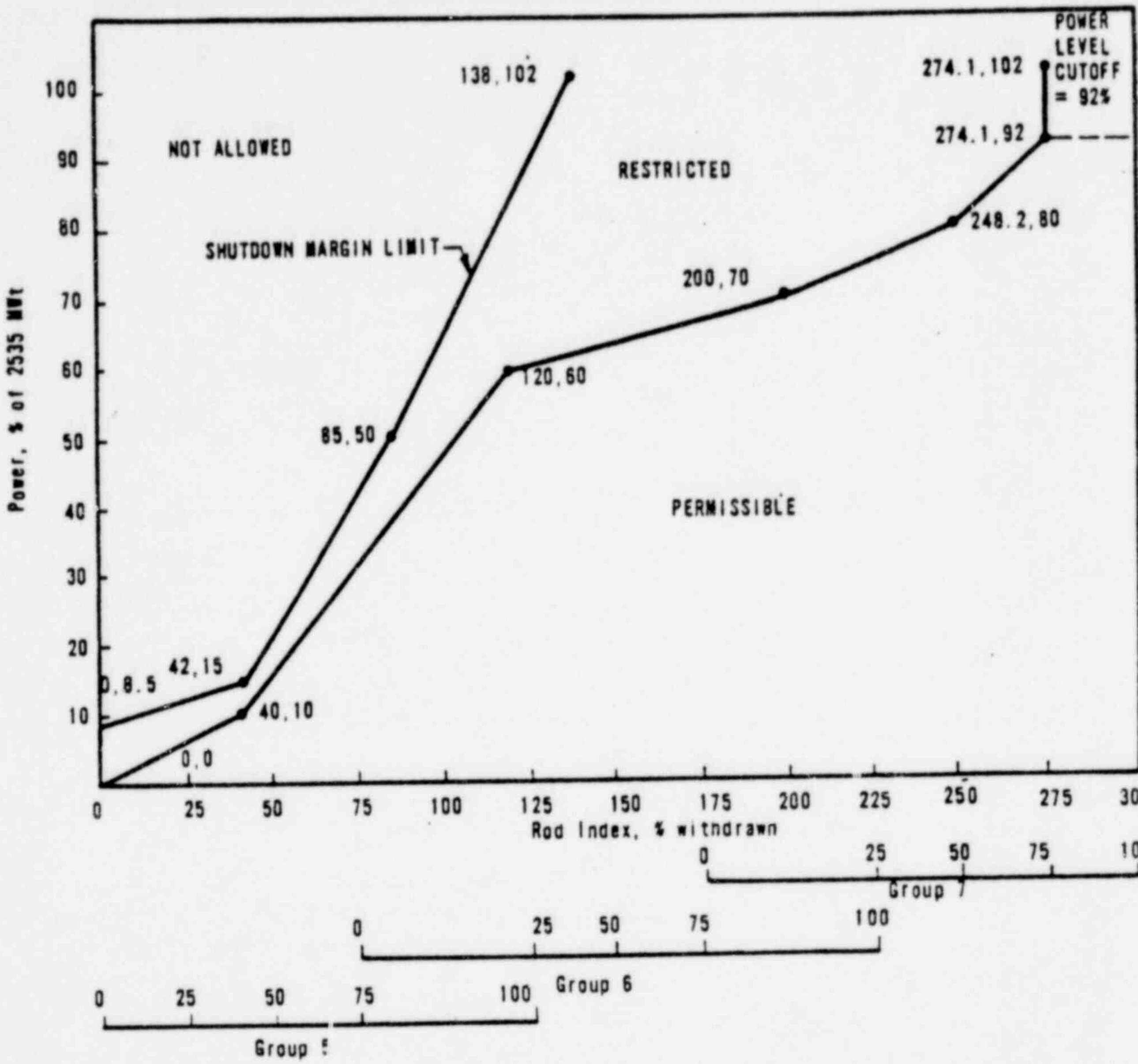

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Figure 8-4. Rod Position Limits for Four-Pump Operation From 125 ± 5 EFPD to 265 ± 15 EFPD — TMI-1, Cycle 4

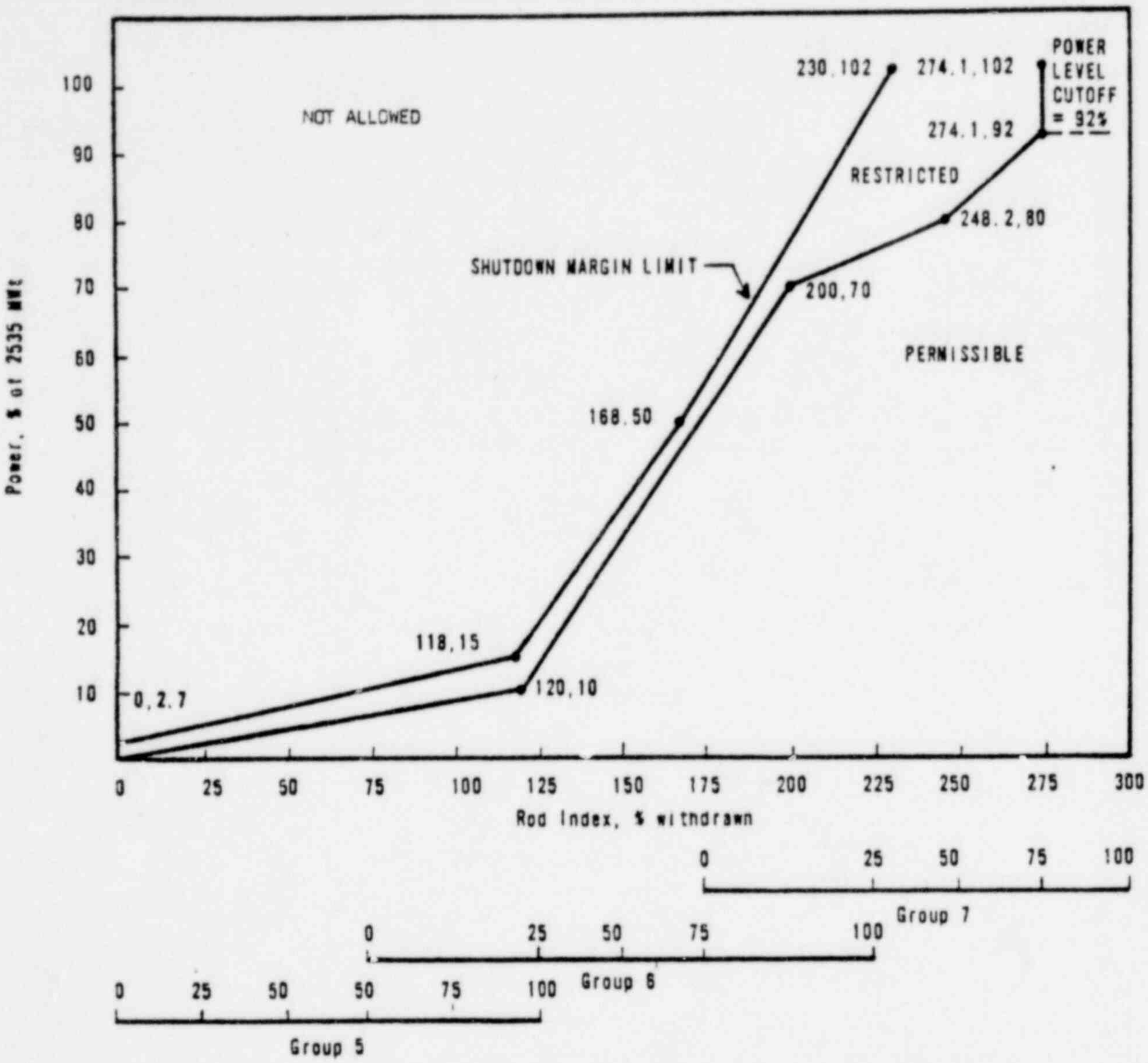

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Figure 8-5. Rod Position Limits for Two- and Three-Pump Operation From 0 to 125 ± 5 EFPD — TMI-1, Cycle 4

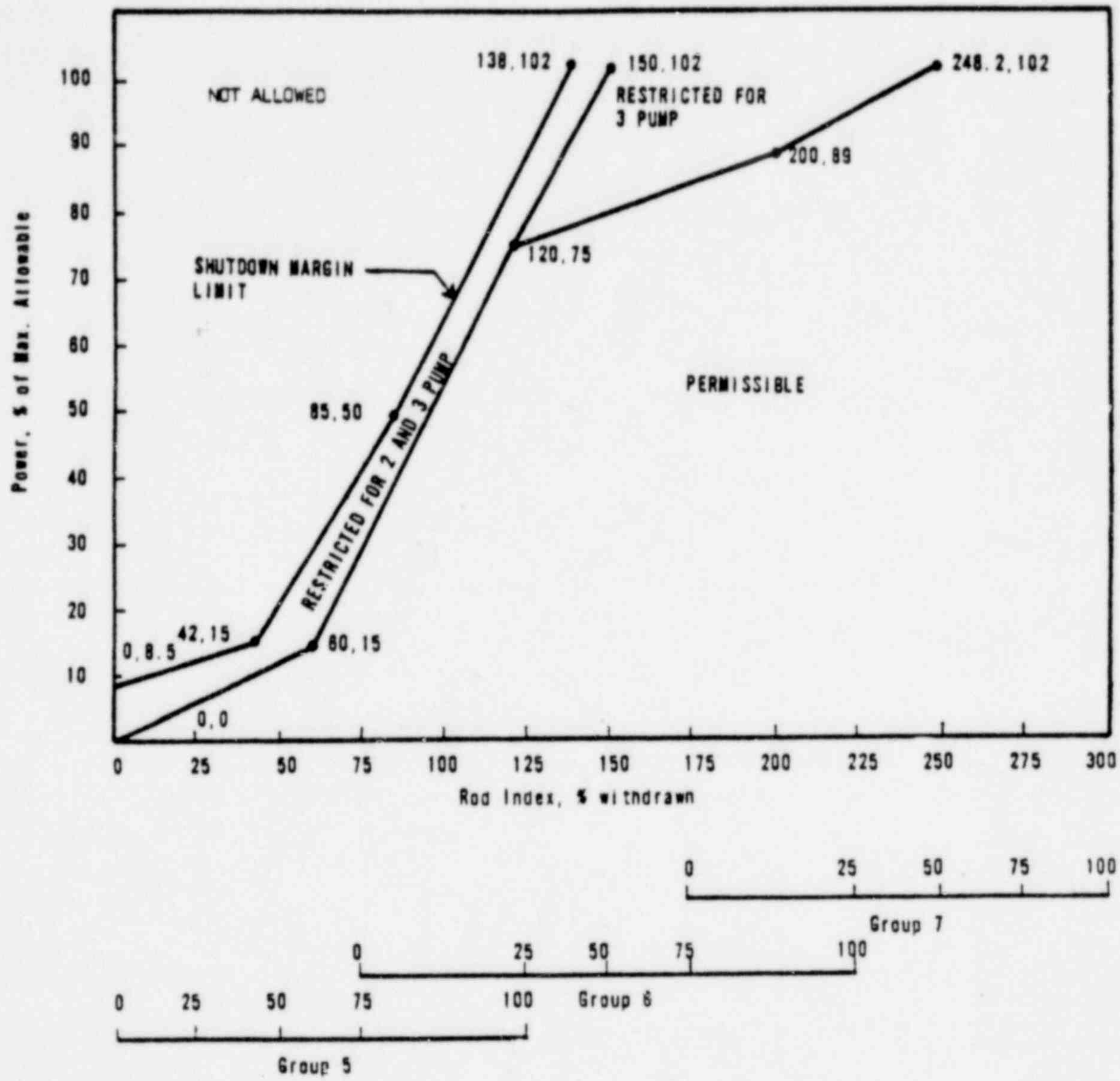

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Figure 8-6. Rod Position Limits for Two- and Three-Pump Operation From 125 ± 5 to 265 ± 15 EFPD — TMI-1, Cycle 4

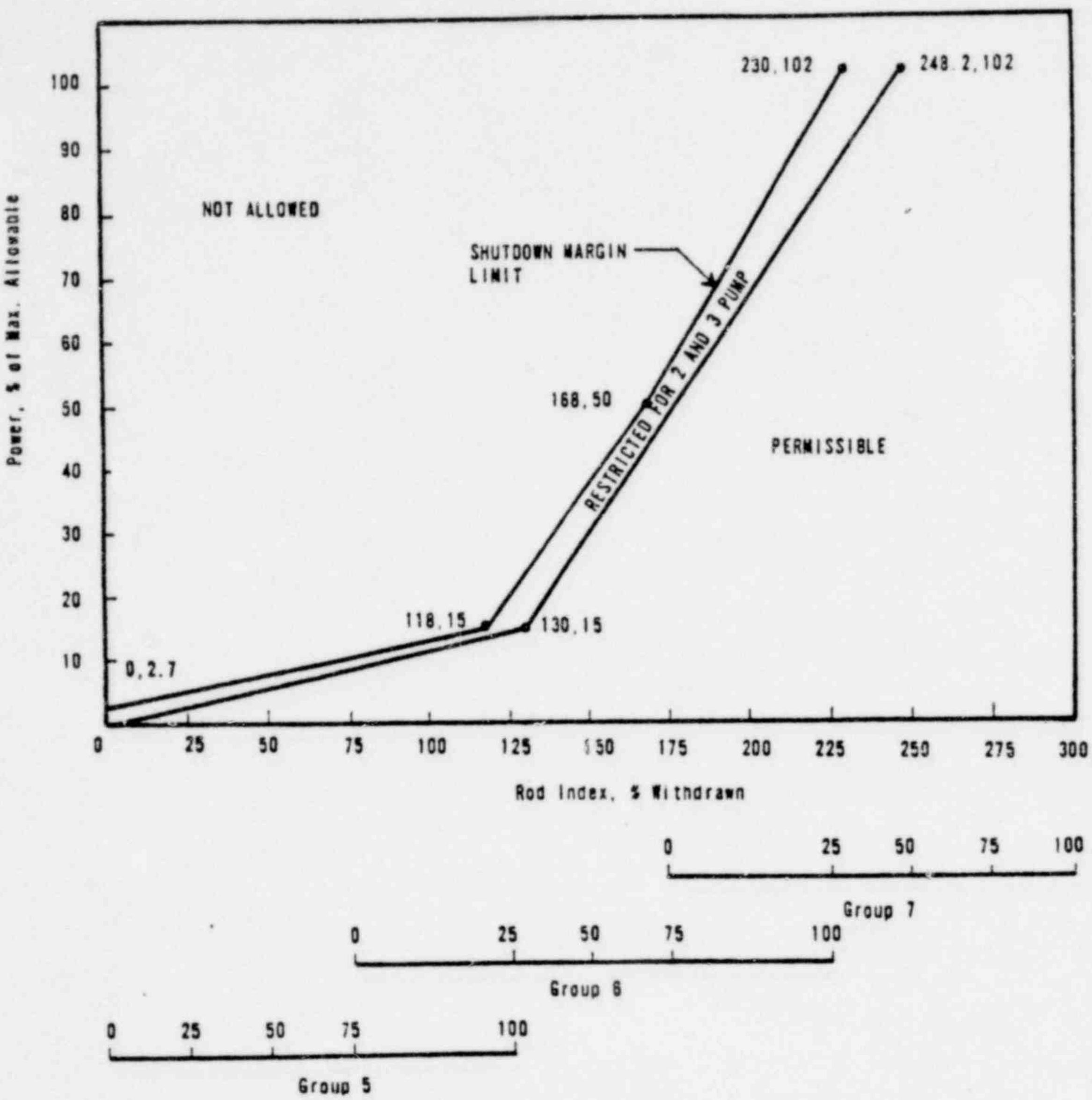

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Figure 8-7. Power Imbalance Envelope for Operation From 0 to 125 ± 5 EFPD

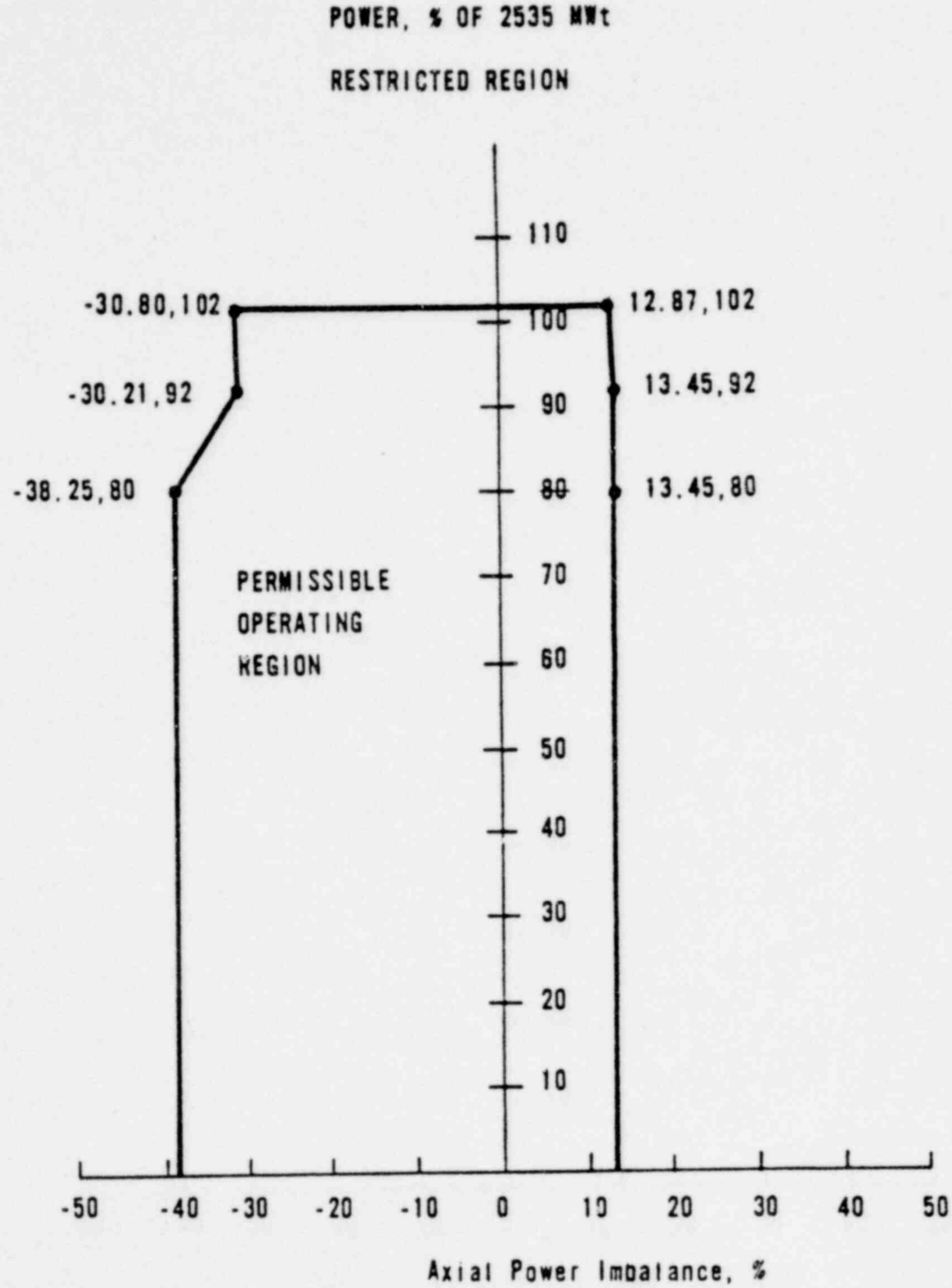

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Figure 8-8. Power Imbalance Envelope for Operation
From 125 ± 5 to 265 ± 15 EFPD

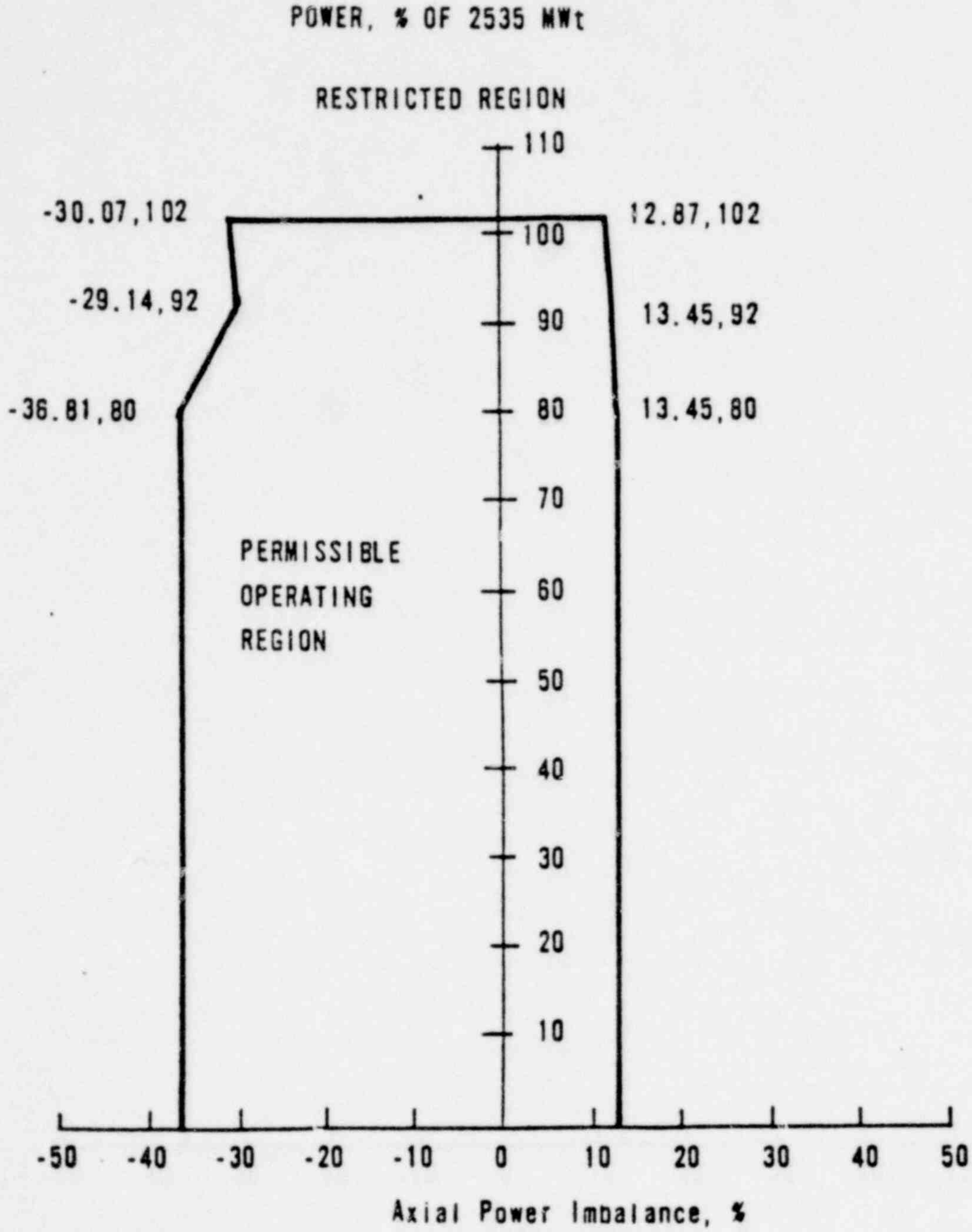

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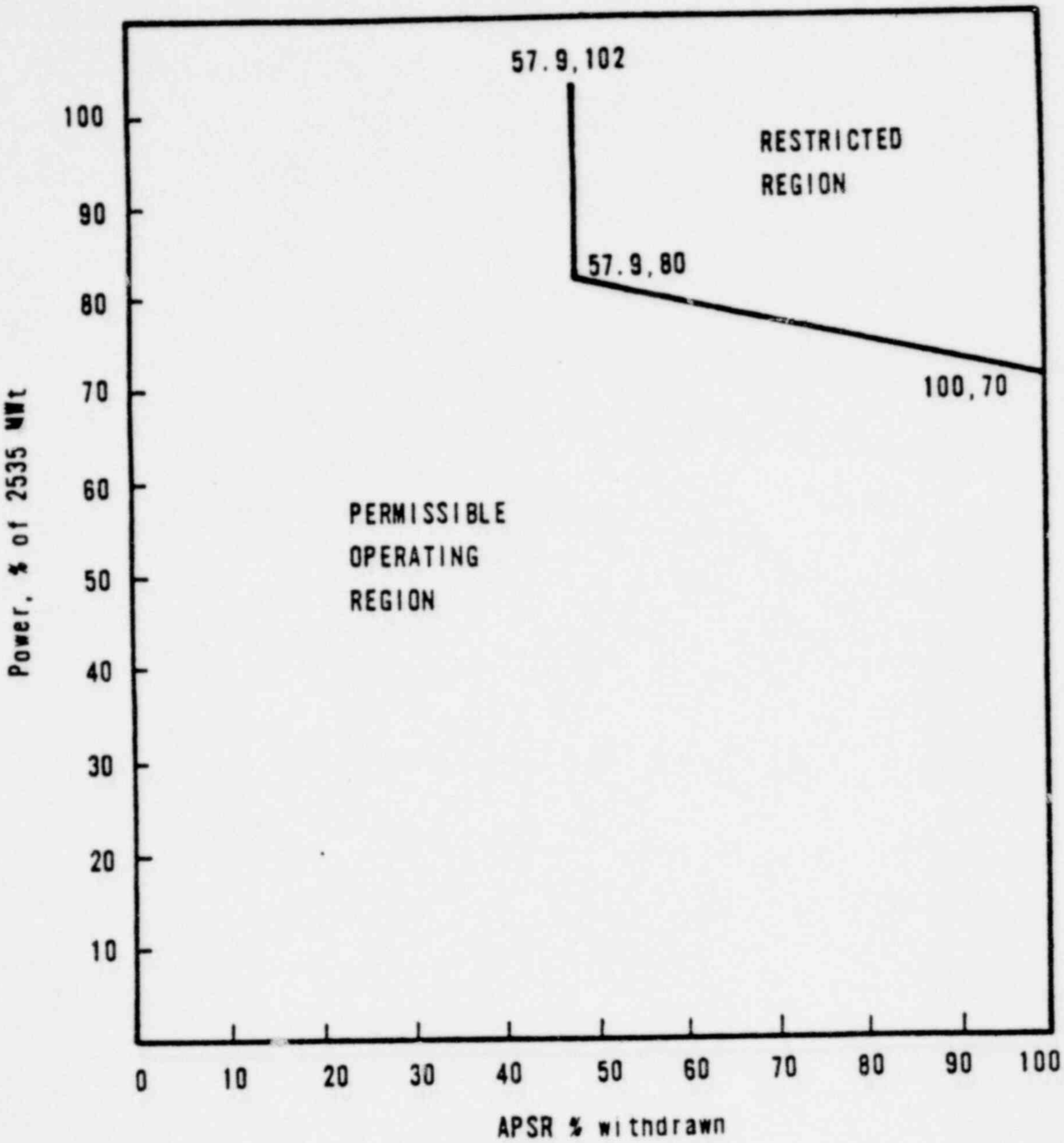
Figure 8-9. APSR Position Limits for Operation From 0 to 265 ± 15 EFPD



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9. STARTUP PROGRAM — PHYSICS TESTING

The planned Startup Test Program associated with core performance is outlined below. These tests verify that core performance is within the assumptions of the safety analysis and provide the necessary data for continued safe operation.

Precritical Tests

1. Control rod trip test.

Zero Power Physics Tests

1. Critical boron concentration.
2. Temperature reactivity coefficient.
    a. All rods out; group 8 in.
    b. Groups 5 through 8 inserted; groups 1 through 4 out.
3. Control rod group reactivity worth.
4. Ejected control rod reactivity worth.

Power Tests

1. Core power distribution verification at approximately 40, 75, and 100% full power with normal control rod group configuration.
2. In-core versus out-of-core detector imbalance correlation verification at approximately 75% full power.
3. Power Doppler reactivity coefficient at approximately 100% FP.
4. Temperature reactivity coefficient at approximately 100% FP.
5. Dropped control rod core power distribution verification at approximately 40% FP.

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10. REFERENCES

1 Three Mile Island Nuclear Station, Unit 1, Final Safety Analysis Report, USNRC Docket No. 50-289.

2 Three Mile Island Unit 1, Cycle 3 Reload Report, BAW-1442, Babcock & Wilcox, Lynchburg, Virginia (1976).

3 A. J. Echert, *et al.*, Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, November 1976.

4 TMI-1 Fuel Densification Report, BAW-1389, Babcock & Wilcox, Lynchburg, Virginia, June 1973.

5 C. D. Morgan and H. S. Kao, TAFY - Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock & Wilcox, Lynchburg, Virginia, May 1972.

6 R. H. Stoudt, *et al.*, TACO - Fuel Performance Analysis, BAW-10087P, Rev 1, Babcock & Wilcox, Lynchburg, Virginia, May 1976.

7 R. C. Jones, *et al.*, ECCS Analysis of B&W's 177-FA Lowered Loop NSS, BAW-10103, Rev 2, Babcock & Wilcox, Lynchburg, Virginia, April 1976.

8 S. A. Varga to J. H. Taylor, Letter, "Comments on B&W's Submittal on Combination of Peaking Factors," May 13, 1977.

9 S. A. Varga to J. H. Taylor, Letter, "Update of BAW-10055, 'Fuel Densification Report'," December 5, 1977.

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