Duke Power Company
Oconee Nuclear Station

Oconee Unit 3, Cycle 8
Reload Report

DPC-RD-2003

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OCONEE UNIT 3, CYCLE 8

- Reload Report -

DPC - RD - 2003

February 1984

Duke Power Company
Nuclear Production Department
P. O. Box 33189
Charlotte, North Carolina 28242

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the eighth cycle of Oconee Nuclear Station, Unit 3, at the rated core power of 2568 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 8 operation of Oconee Unit 3, this report employs analytical techniques and design bases established in reports that were previously submitted and accepted by the USNRC and its predecessor (see references).

A brief summary of cycle 7 and 8 reactor parameters related to power capability is included in section 5 of this report. All of the accidents analyzed in the FSAR[1] have been reviewed for cycle 8 operation. In those cases where cycle 8 characteristics were conservative compared to those analyzed for previous cycles, no new accident analyses were performed.

The Technical Specifications have been reviewed, and the modifications required for cycle 8 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, it has been concluded that Oconee Unit 3 can be operated safely for cycle 8 at the rated power level of 2568 MWt.

2. OPERATING HISTORY

The referenced fuel cycle for the nuclear and thermal-hydraulic analyses of Oconee Unit 3, cycle 8, is the currently operating cycle 7. Cycle 6 was terminated after 349 EFPD of operation. Cycle 7 achieved initial criticality on October 1, 1982 and power escalation commenced on October 2, 1981. The fuel cycle design length for cycle 8 - 400 EFPD - is based on a nominal cycle 7 length of 440 EFPD. No operating anomalies occurred during previous cycle operations that would adversely affect fuel performance in cycle 8.

Cycle 8 will operate in a feed-and-bleed mode for its entire design length, as did cycle 7.

3. GENERAL DESCRIPTION

The Oconee Unit 3 reactor core and fuel design basis are described in detail in Chapter 3, of the FSAR.[1] The cycle 8 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 consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The fuel assemblies in all batches have an average nominal fuel loading of 463.6 kg uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Table 4-1.

Figure 3-1 is the core loading diagram for Oconee 3, cycle 8. Thirty-one of the batch 8 assemblies will be discharged at the end of cycle 7 along with batch 7B. The remaining 37 batch 8 assemblies, designated "8B," and the fresh batch 10 FAs - with initial enrichments of 3.07 and 3.28 wt % ^{235}U, respectively - will be loaded into the central portion of the core. Batch 9, with an initial enrichment of 3.18 wt % ^{235}U, will occupy primarily the core periphery. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of cycle 8.

Cycle 8 will operate in a rods-out, feed-and-bleed mode. Core reactivity control is supplied mainly by soluble boron and supplemented by 61 full-length Ag-In-Cd control rods and 60 burnable poison rod assemblies (BPRAs). In addition to the full-length control rods, eight partial-length axial power shaping rods (APSRs) are provided for additional control of axial power distribution. The cycle 8 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The cycle 8 locations and enrichments of the BPRAs are shown in Figure 3-4.

FIGURE 3.1 CORE LOADING FOR OCONEE 3 CYCLE 8

X

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	
A						L03 9	K02 9	L07 9	K14 9	L13 9						
B				N03 9	10	M02 9	10	K08 9	10	M14 9	10	N13 9				
C			L09 9	10	L05 9	10	K04 9	10	K12 9	10	L11 9	10	K06 9			
D		C12 9	10	L15 8B	10	A07 8B	10	P10 8B	10	A09 8B	10	R06 8B	10	C04 9		
E		10	E10 9	10	O13 8B	10	N02 8B	10	N14 8B	10	O03 8B	10	E06 9	10		
F	C10 9	B11 9	10	G01 8B	10	R10 8B	M04 9	R08 8B	M12 9	L01 8B	10	G15 8B	10	B05 9	C06 9	
G	B09 9	10	D09 9	10	B12 8B	D11 9	H13 9	10	O08 9	D05 9	B04 8B	10	D07 9	10	B07 9	
H W —	K10 9	H09 9	10	F14 8B	10	H15 8B	10	L14 8B	10	H01 8B	10	L02 8B	10	H07 9	G06 9	— Y
K	P09 9	10	N09 9	10	P12 8B	N11 9	C08 9	10	H03 9	N05 9	P04 8B	10	N07 9	10	P07 9	
L	O10 9	P11 9	10	K01 8B	10	F15 8B	E04 9	A08 8B	E12 9	A06 8B	10	K15 8B	10	P05 9	O06 9	
M		10	M10 9	10	C13 8B	10	D02 8B	10	D14 8B	10	C03 8B	10	M06 9	10		
N		O12 9	10	A10 8B	10	R07 8B	10	B06 8B	10	R09 8B	10	F01 8B	10	O04 9		
O			G10 9	10	F05 9	10	G04 9	10	G12 9	10	F11 9	10	F07 9			
P				D03 9	10	E02 9	10	G08 9	10	E14 9	10	D13 9				
R						F03 9	G02 9	F09 9	G14 9	F13 9						
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	

Z

XX X	PREVIOUS CYCLE LOCATION BATCH NO.

FIGURE 3.2 ENRICHMENT & BURNUP FOR OCONEE 3, CYCLE 8

	8	9	10	11	12	13	14	15
H	3.07 25624	3.28 0	3.07 21033	3.28 0	3.07 25631	3.28 0	3.18 16764	3.18 17471
K	3.28 0	3.18 17502	3.18 17137	3.07 16231	3.28 0	3.18 17511	3.28 0	3.18 14910
L	3.07 21031	3.18 17132	3.07 16605	3.28 0	3.07 19930	3.28 0	3.18 11225	3.18 16842
M	3.28 0	3.07 16229	3.28 0	3.07 20417	3.28 0	3.18 17505	3.28 0	
N	3.07 25631	3.28 0	3.07 19932	3.28 0	3.07 16617	3.28 0	3.18 13360	
O	3.28 0	3.18 17502	3.28 0	3.18 17480	3.28 0	3.18 17438		
P	3.18 16735	3.28 0	3.18 11221	3.28 0	3.18 13350			
R	3.18 17471	3.18 14906	3.18 16837					

X.XX	INITIAL ENRICHMENT, wt% ^{235}U
XXXXX	BOC BURNUP, MWd/mtU

FIGURE 3.3 CONTROL ROD LOCATIONS FOR OCONEE 3, CYCLE 8

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

X (top of column 8); Z (bottom of column 8)

X	GROUP NO.

GROUP	NO. of RODS	FUNCTION
1	8	SAFETY
2	8	SAFETY
3	8	SAFETY
4	9	SAFETY
5	8	CONTROL
6	8	CONTROL
7	12	CONTROL
8	8	APSRs
TOTAL	69	

FIGURE 3.4 BPRA ENRICHMENT & DISTRIBUTION FOR OCONEE 3, CYCLE 8

	8	9	10	11	12	13	14	15
H		1.1		1.1		1.1		
K	1.1				1.1		0.2	
L				1.1		0.8		
M	1.1		1.1		1.1			
N		1.1		1.1		0.2		
O	1.1		0.8		0.2			
P		0.2						
R								

X.X	BPRA CONCENTRATION, wt % B_4C IN Al_2O_3

4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for Oconee 3, Cycle 8, are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. Two regenerative neutron sources will be used in MK B5 fuel assemblies.

The batch 10 Mark B5 fuel assemblies have redesigned upper end fittings which provide a positive holddown of BPRAs. Reference is made to the Oconee 3 Cycle 7 reload report (5), for a design description of this end fitting. All 60 BPRAs will be inserted into batch 10 fuel assemblies.

Other results presented in the FSAR[1] fuel assembly mechanical discussions and in previous reload reports, are applicable to the reload fuel assemblies. Duke has performed generic mechanical analyses, as described below, which envelope the cycle 8 design. All methods are consistent with the approved methodologies of Reference 11.

4.2 Fuel Rod Design

The mechanical evaluation of the fuel rod is discussed below.

4.2.1 Cladding Collapse

The fuel of batch 8B is more limiting than other batches due to its longer previous incore exposure time. The batch 8B assembly power histories were analyzed, and the most limiting assembly was used to perform the creep collapse analysis using the CROV computer code and procedures described in topical report BAW-10084, Rev. 2.[2] The TACO2[3] code was used to calculate internal pin pressures and clad temperatures used as input to CROV. The collapse time for the most limiting assembly was conservatively determined to be more than 31,400 EFPH, which is greater than the maximum projected residence time of cycle 8 fuel (Table 4-1).

4.2.2 Cladding Stress

As described in Reference 11, Duke has performed a generic and conservative fuel rod cladding stress analysis in accordance with the guidelines set forth in Section III, Division 1 - Subsection NB, of the ASME Boiler and Pressure Vessel Code. Compliance with ASME Code criteria verify the structural integrity of the cladding throughout the most limiting design conditions.

The following conservatisms exist in the generic cladding stress calculation:

- high external cladding pressure (110% of design system pressure)
- low internal pressure (HZP - min. specified pre-pressure)
- maximum possible radial temperature gradient through clad (fuel melt conditions)
- conservative cladding dimensions with regards to stress

4.2.3 Cladding Strain

Duke has performed a cladding strain calculation using TACO[4] in accordance with the approved methodology.[11] This analysis demonstrated that for transient conditions, the uniform, circumferential strain of the cladding was within the 1.0% limit.

4.3 Thermal Design

All fuel in the cycle 8 core is thermally similar. The fresh batch 10 fuel inserted for cycle 8 operation introduces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. The linear heat rate to melt capability based on centerline fuel melt was determined separately for each batch of fuel using the TACO[4] computer code for batches 8B and 9 and the recently approved TACO2[3] computer code for batch 10 fuel. The individual fuel parameters used to determine the fuel melt limits are shown in Table 4-2.

The input shown includes the following conservatisms:

1. Lower Tolerance Limit (LTL) initial density.
2. LTL initial pellet diameter.
3. A maximum gap based on as-fabricated pellet and cladding data.
4. Maximum incore densification based on resinter test results.

The burnup dependent linear heat rate (LHR) capability and the average fuel temperature for each batch are shown in Table 4-2.

The maximum assembly average burnup is predicted to be 38358 MWD/mtU, and the maximum fuel rod burnup is predicted to be 40839 MWD/mtU. Fuel rod internal pressure has been evaluated using TACO[4] with a conservative pin power history, and the maximum pressure is less than the nominal reactor coolant (RC) system pressure of 2200 psia.

4.4 Material Design

The batch 10 fuel assemblies are not unique in concept, nor do they utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 10 fuel assemblies is identical to those of the present fuel.

Table 4-1

Fuel Design Parameters and Dimensions

	Batch No.		
	8B	9	10
FA type	Mark B4	Mark B5	Mark B5
No. of FAs	37	72	68
Fuel rod OD, in.	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377
Flex spacers, type	Spring	Spring	Spring
Rigid Spacers, type	Zr-4	Zr-4	Zr-4
Undensif. active fuel length, in.	141.8	141.8	141.8
Fuel pellet OD (mean spec.), in.	0.3686	0.3686	0.3686
Fuel pellet initial density (mean spec.), %TD	95.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	3.07	3.18	3.28
Est residence time, EOC 8, EFPH	28,560	20,160	9,600
Cladding collapse time, EFPH	>31,400	>31,400	>31,400

Table 4-2

Linear Heat Rate to Melt Analysis

	Batch No.		
	8B	9	10
Initial density, % TD	94.66(a)	94.58	94.79
Max. In-reactor densification, % TD	1.25(a)	1.85	2.13
Burnup corresponding to max. densification, MWd/mtU	2784(a)	3317	2896
Average linear heat rate @ 100% of 2568 MW, kW/ft	5.74	5.74	5.74
Linear heat rate capability(c) from 0-10,000 MWD/mtU, kW/ft	20.5	20.5	20.5
Linear heat rate capability(c) >10,000 MWD/mtU, kW/ft	21.5	21.5	21.5
Average fuel temp. @ nominal linear heat rate, °F	1240(b)	1240	1240

(a) Basis: Batch specific pellet resinter data

(b) Basis: TACO, 96.5% TD @ 4000 MWD/mtU, nominal pellet and cladding dimensions

(c) These values are utilized as fuel design limits for Cycle 8.

5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design cycles 7 and 8; the values for both cycles 7 and 8 were generated by Duke Power Company using methods described in Reference 11. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 8 at full power with equilibrium xenon and nominal rod positions.

The initial BPRA loading, fewer fresh fuel assemblies, shorter cycle length, different shuffle pattern, and different control rod pattern for cycle 8 make it difficult to compare the physics parameters with those of cycle 7. The control rod worths differ between cycles primarily due to changes in control rod patterns. 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. All safety criteria associated with these rod worths are met. The adequacy of the shutdown margin is demonstrated in Table 5-2. The following conservatisms were applied for the shutdown calculations:

1. Poison material depletion allowance.
2. 10% uncertainty on net rod worth.

Flux redistribution was explicitly accounted for since the shutdown analysis was calculated using a three-dimensional model.

5.2 Analytical Input

The cycle 8 incore measurement calculation constants to be used to compute core power distributions were obtained in a manner identical to that used for Oconee 2 Cycle 7[12].

5.3 Changes in Nuclear Design

There are no changes in design methodology between Oconee 3 Cycle 8 and Oconee 3 Cycle 7.

Table 5-1. Oconee 3 Physics Parameters[a]

	Cycle 7[b]	Cycle 8[c]
Cycle length, EFPD	440	400
Cycle burnup, MWd/mtU	13,752	12,307
Average core burnup, EOC, MWd/mtU	21,608	22,818
Initial core loading, mtU	82.1	82.1
Critical boron - BOC (no xenon), ppm		
HZP, group 7 at 100% WD, 8 at 37.5% WD	1623	1550
HFP, group 7 at 87% WD, 8 at 25% WD	1440	1358
Critical boron - EOC (equilibrium xenon), ppm		
HZP, group 7 at 100% WD, 8 at 37.5% WD	396	404
HFP, group 7 at 87% WD, 8 at 25% WD	11	13
Control rod worths - HFP, BOC, % Δk/k		
Group 6	1.22	1.19
Group 7	1.47	1.41
Group 8 (25% to 100% WD)	0.33	0.33
Control rod worths - HFP, EOC, % Δk/k		
Group 7	1.65	1.57
Group 8 (25% to 100% WD)	0.29	0.30
Max ejected rod worth - HZP, % Δk/k		
BOC, groups 5-8 inserted	0.74	0.42
EOC, groups 5-8 inserted	0.82	0.51
Max stuck rod worth - HZP, % Δk/k		
BOC	1.50	1.12
EOC	2.06	1.76
Power deficit, HZP to HFP, % Δk/k		
BOC	1.76	1.83
EOC	3.12	3.12
Doppler coeff - BOC, 10^{-5} (Δk/k-°F)		
100% power (no xenon)	-1.32	-1.37
Doppler coeff - EOC, 10^{-5} (Δk/k-°F)		
100% power (equilibrium xenon)	-1.68	-1.68

Table 5-1. (Cont'd)

	Cycle 7[b]	Cycle 8[c]
Moderator coeff - HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (no xenon, group 8 ins.)	-0.34	-0.62
EOC (equilibrium xenon, group 8 ins.)	-2.85	-2.92
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC	121	122
EOC	108	110
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 days)	2.49	2.48
EOC (equilibrium)	2.70	2.68
Effective delayed neutron fraction - HFP		
BOC	0.00628	0.00617
EOC	0.00518	0.00522

[a] Cycle 8 data are for the conditions stated in this report. The cycle 7 core conditions are identified in reference 5.

[b] Based on a 350-EFPD cycle 6. (Actual cycle 6 length 349 EFPD).

[c] Based on 440-EFPD cycle 7.

Table 5-2. Shutdown Margin Calculation for Oconee 3, Cycle 8

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP	8.42	9.34
Worth reduction due to poison burnup	-0.42	-0.42
Maximum stuck rod, HZP	-1.12	-1.76
Net worth	6.88	7.16
Less 10% uncertainty	-0.69	-0.72
Total available worth	6.19	6.44
Required Rod Worth		
Power deficit, HFP to HZP	1.83	3.12
Max inserted rod worth, HFP	0.22	0.59
Total required worth	2.05	3.71
Shutdown Margin		
Total available worth minus total required worth	4.14	2.73

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

FIGURE 5.1
OCONEE 3 CYCLE 8
TWO DIMENSIONAL
RELATIVE POWER DISTRIBUTION

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	0.960	1.226	0.975	1.264	1.002	1.298	0.998	0.517
K		1.078	1.050	1.124	1.279	1.177	1.234	0.535
L			1.079	1.263	0.995	1.278	1.009	0.407
M				1.054	1.213	1.031	0.914	
N					1.001	1.019	0.486	
O						0.498		
P								
R								

SL

6. THERMAL-HYDRAULIC DESIGN

The incoming batch 10 fuel assemblies are hydraulically and geometrically similar to the fuel remaining in the core from previous cycles. The generic thermal-hydraulic design analysis supporting cycle 8 operation was performed by Duke Power Company and employed the methods and models described in references 1, 5, 6 and 11.

The maximum core bypass flow for cycle 7 was 7.6% of the total system flow. For cycle 8 operation, 60 BPRAs will be inserted, and two assemblies will contain regenerative neutron sources, leaving 46 FAs with open guide tubes. This results in a core bypass flow of 7.9% of the total system flow. The bypass flow is less than that assumed in the generic thermal-hydraulic design analysis and the consequent increase in core flow establishes the generic analysis as conservative for cycle 8 operation. The cycle 7 and 8 maximum design conditions are summarized in Table 6-1.

A B&W topical report discussing the mechanisms and resulting effects of fuel rod bow has been approved by the NRC (reference 7). The report concludes that the DNBR penalty due to rod bow is insignificant and unnecessary because the power production capability of the fuel decreases with irradiation. Therefore, no rod bow DNBR penalty needs to be considered for cycle 8 thermal-hydraulic analyses.

For cycle 8 operation a flux to flow setpoint of 1.08 is maintained. The minimum DNBR value determined by the flux to flow setpoint analysis is greater than the BAW-2 CHF correlation limit of 1.30. All other plant operating limits based on DNBR criteria include a minimum of 10.2% DNBR margin from the design limit of 1.30.

Table 6-1. Thermal Hydraulic Design Conditions

	Cycle 7	Cycle 8
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	106.5
Core bypass flow, % total flow[a]	7.6	7.9
Vessel inlet/outlet coolant temp at 100% power, °F	555.6/602.4	555.6/602.4
Ref design radial-local power peaking factor	1.71	1.71
Ref design axial flux shape	1.5 cosine	1.5 cosine
Hot channel factors: Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Active fuel length, in.	141.8	141.8
Avg heat flux at 100% power, 10^3 Btu/h-ft^2	176[b]	176[b]
CHF correlation	BAW-2	BAW-2
Min DNBR with densification penalty	2.05	>2.05

[a] Generic analysis based on 8.2% core bypass flow.

[b] Heat flux based on densified length of 140.3 in., which is a conservative minimum value.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 7 parameters to determine the effect of the cycle 8 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the FSAR accident results has been evaluated and are reported in reference 6. Since batch 10 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in reference 6, the conclusions in that reference are still valid.

No new dose calculations were performed for the reload report. The dose considerations in reference 10 are characteristic for Oconee 3 cycle 8 based upon comparisons of key parameters which determine radionuclide inventories.

7.2 Accident Evaluations

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.

Fuel thermal analysis parameters for each batch in cycle 8 are given in Table 4-2. Table 6-1 compares the cycle 7 and 8 thermal-hydraulic maximum design conditions. Table 7-1 compares the key kinetics parameters from the FSAR and cycle 8.

A generic LOCA analysis for the 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. 3.[8] The analysis in BAW-10103 is generic since the limiting values of key parameters for all plants in this category were used. Furthermore, the combination of average fuel temperature as a

function of LHR and the lifetime pin pressure data used in the BAW-10103 LOCA limits analysis[8,9] is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in BAW-10103 provide conservative results for the operation of Oconee 3, cycle 8 fuel.

Table 7-2 shows the bounding values for allowable LOCA peak LHRs for Oconee 3 cycle 8 fuel after 2600 MWd/Mtu. The LOCA kW/ft limits have been reduced for low burnups to ensure conservative limits based upon an interim bounding analytical assessment of NUREG-0630 on LOCA and operating kW/ft limits performed by Babcock & Wilcox[13]. The limits for the first 0-2600 MWd/Mtu are shown in Table 7-3.

From the examination of cycle 8 core thermal properties and kinetics properties with respect to acceptable previous cycle values, it is concluded that this core reload will not adversely affect the safe operation of the Oconee 3 plant during cycle 8. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 8 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 8 are bounded by the FSAR and/or the fuel densification report.[6]

Table 7.1. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR[1] value	Predicted cycle 8 value
BOC Doppler coeff, 10^{-5}, Δk/k/°F	-1.17	-1.37
EOC Doppler coeff, 10^{-5} Δk/k/°F	-1.33[(a)]	-1.68
BOC moderator coeff, 10^{-4}, Δk/k°F	+0.5[(b)]	-0.62
EOC moderator coeff, 10^{-4}, Δk/k/°F	-3.0	-2.92
All rod bank worth, HZP, % Δk/k	10.0	9.34
Boron reactivity worth, 70°F ppm/1% Δk/k	75	87
Max. ejected rod worth, HFP, % Δk/k	0.65	0.18
Dropped rod worth, HFP, % Δk/k	0.46	0.12
Initial boron conc, HFP, ppm	1400	1358

(a) -1.2×10^{-5} Δk/k/F was used for steam-line analysis.
-1.3×10^{-5} Δk/k/F was used for cold water accident (pump start-up).

(b) $+0.94 \times 10^{-4}$ Δk/k/F was used for the moderator dilution accident.

Table 7-2. LOCA Limits, Oconee 3, Cycle 8, After 2600 MWd/Mtu(b)

Elevation, ft	LHR limits, kW/ft
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

Table 7-3. LOCA Limits, Oconee 3, Cycle 8

Elevation, ft	LHR Limits, kW/ft	
	0-1000 MWd/Mtu[a]	1000-2600 MWd/Mtu[b]
2	13.5	15.0
4	16.1	16.6
6	17.5	18.0
8	17.0	17.0
10	16.0	16.0

(a) 1000 MWd/Mtu corresponds to approximately 25 EFPD for the most limiting assembly.

(b) 2600 MWd/Mtu corresponds to approximately 65 EFPD for the most limiting assembly.

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 8 operation in accordance with the methods of reference 11 to account for minor changes in power peaking and control rod worths. Cycle 8 Technical Specifications were also generated in accordance with the methods described in Reference 11. Revised Reactor Protective System (RPS) instrumentation string errors, as discussed in reference 12, were used in the anaylses of the power imbalance trip setpoints.

Based on the Technical Specifications derived from the analyses presented in this report, The Final Acceptance Criteria ECCS limits will not be exceeded, nor will the thermal design criteria be violated. Figures 8-1 through 8-15 are revisions to previous Technical Specification limits.

Figure 8-1

Core Protection Safety Power-Imbalance Limits

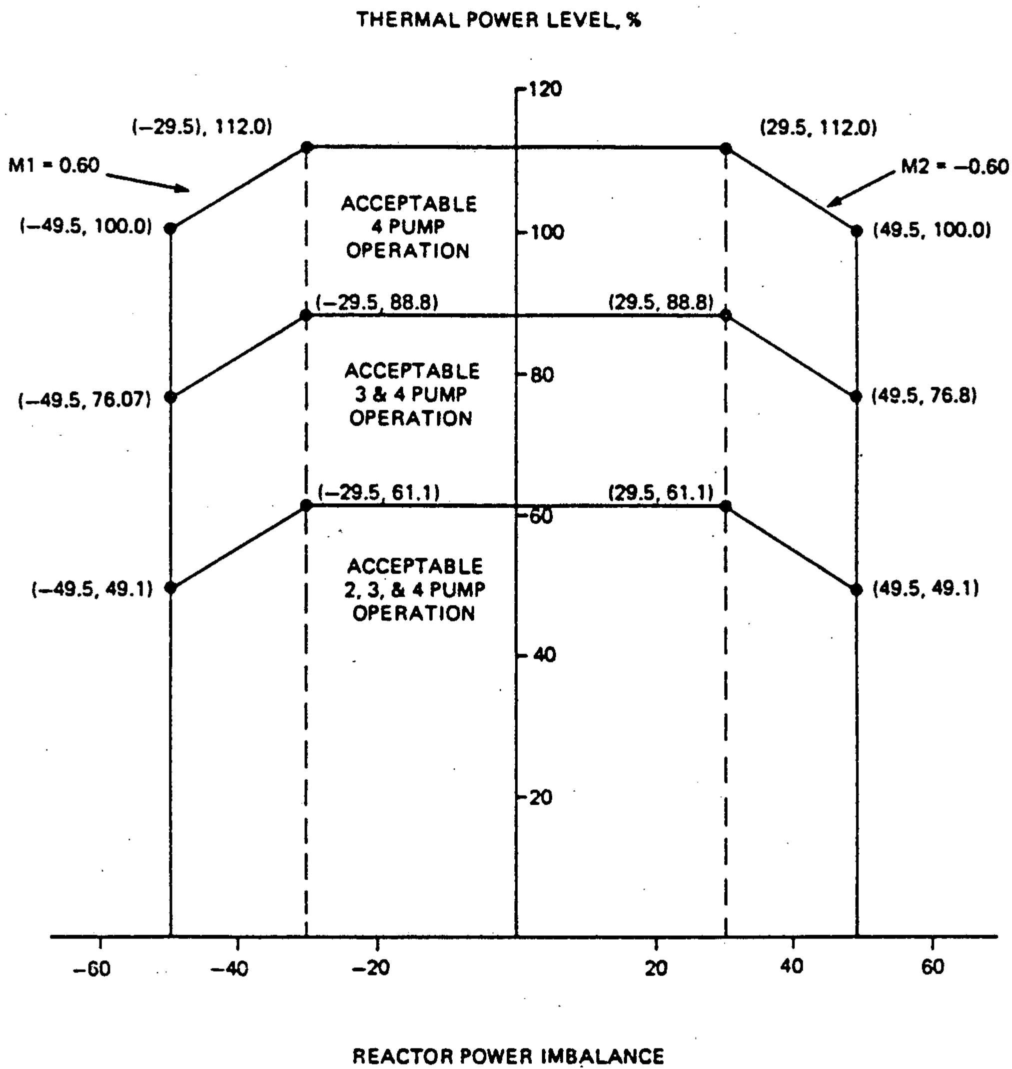


Figure 8-2

Maximum Allowable Power-Imbalance Setpoints

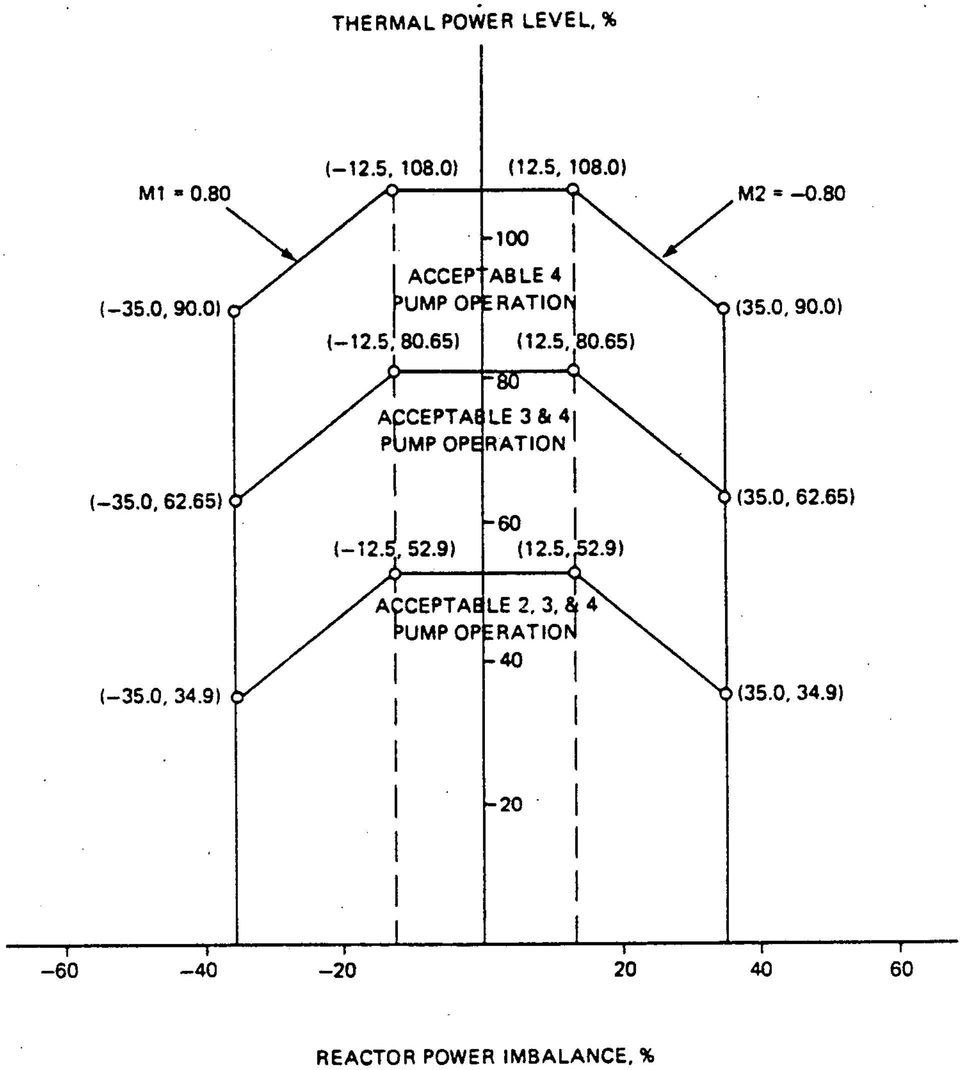


Figure 8-3

Operational Power-Imbalance Limits, 0-200 ± 10 EFPD

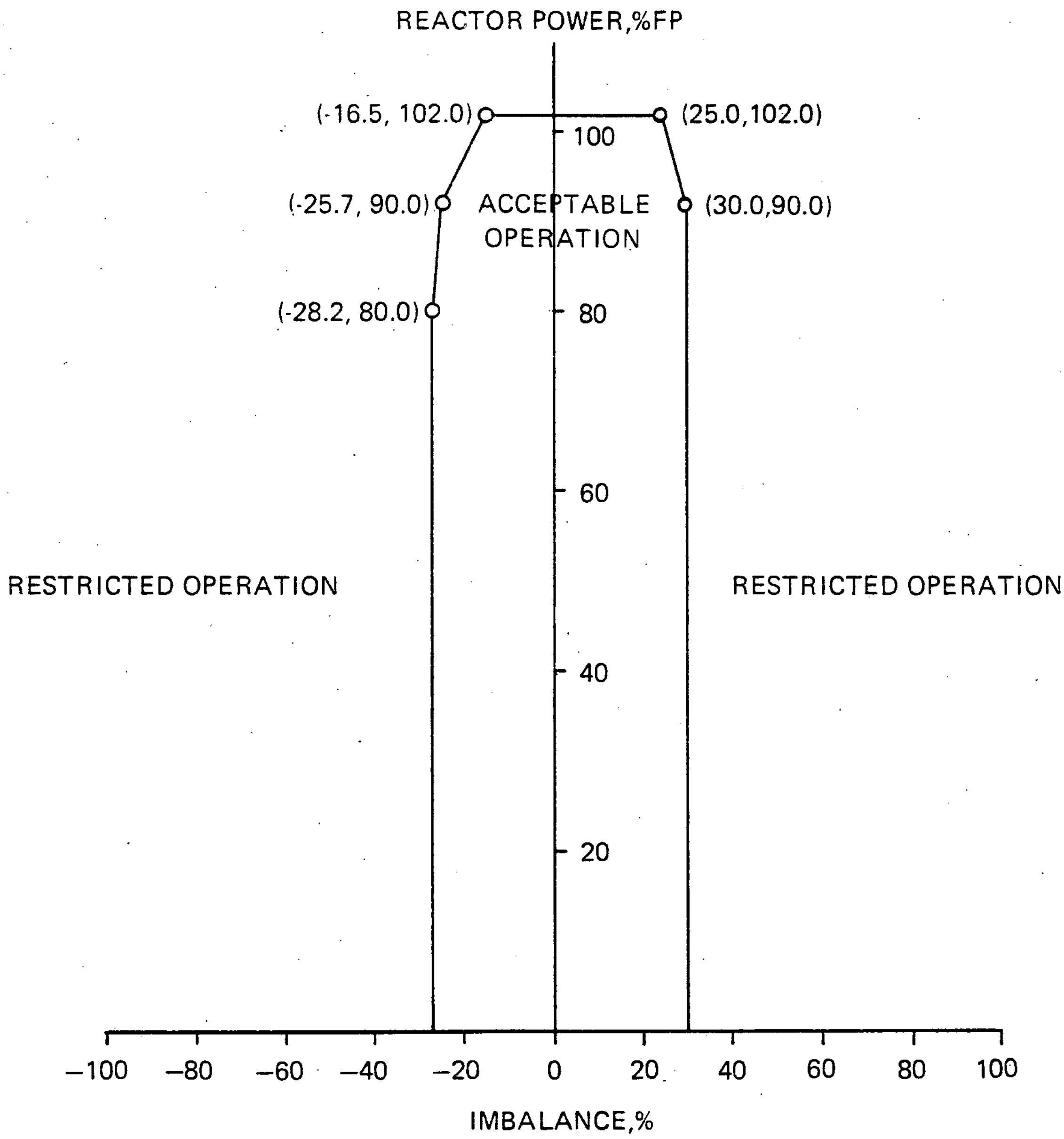


Figure 8-4

Operational Power-Imbalance Limits After 200 ± 10 EFPD

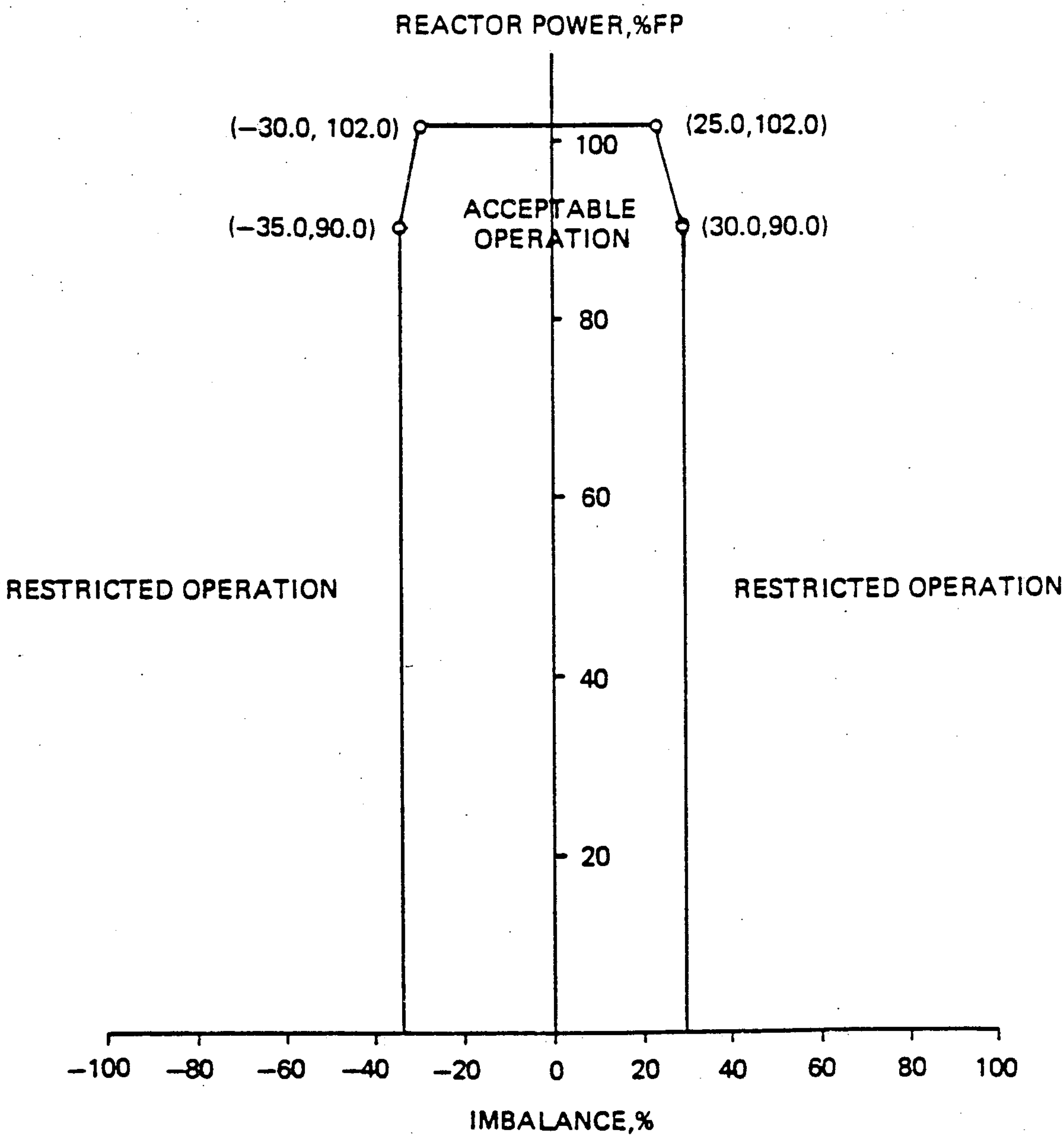


Figure 8-5

Control Rod Position Limits, 4 Pumps, 0-25 $\pm\,^{10}_{0}$ EFPD

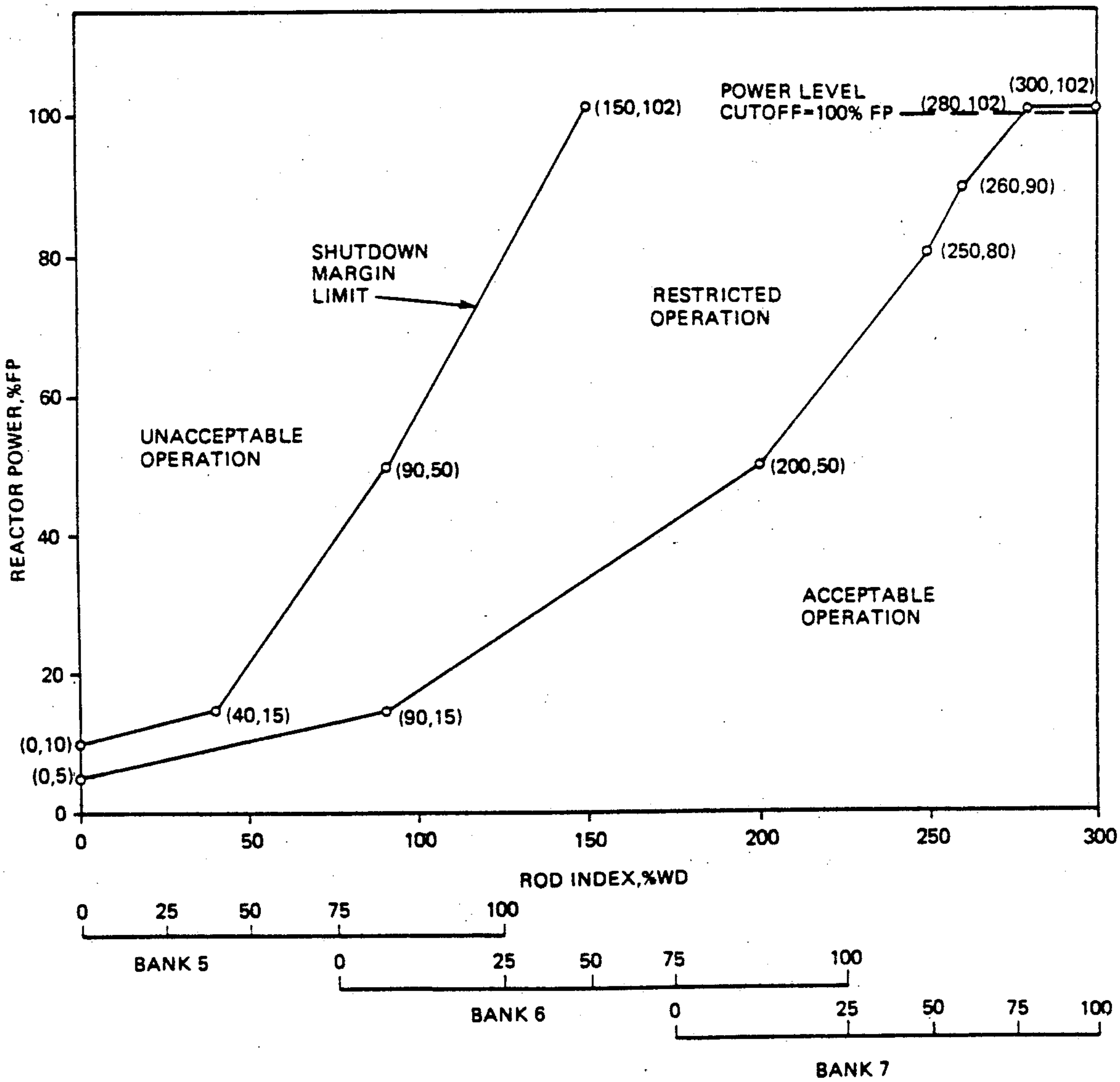


Figure 8-6

Control Rod Position Limits, 4 Pumps, $25 \pm {}^{10}_{0}$ - 200 ± 10 EFPD

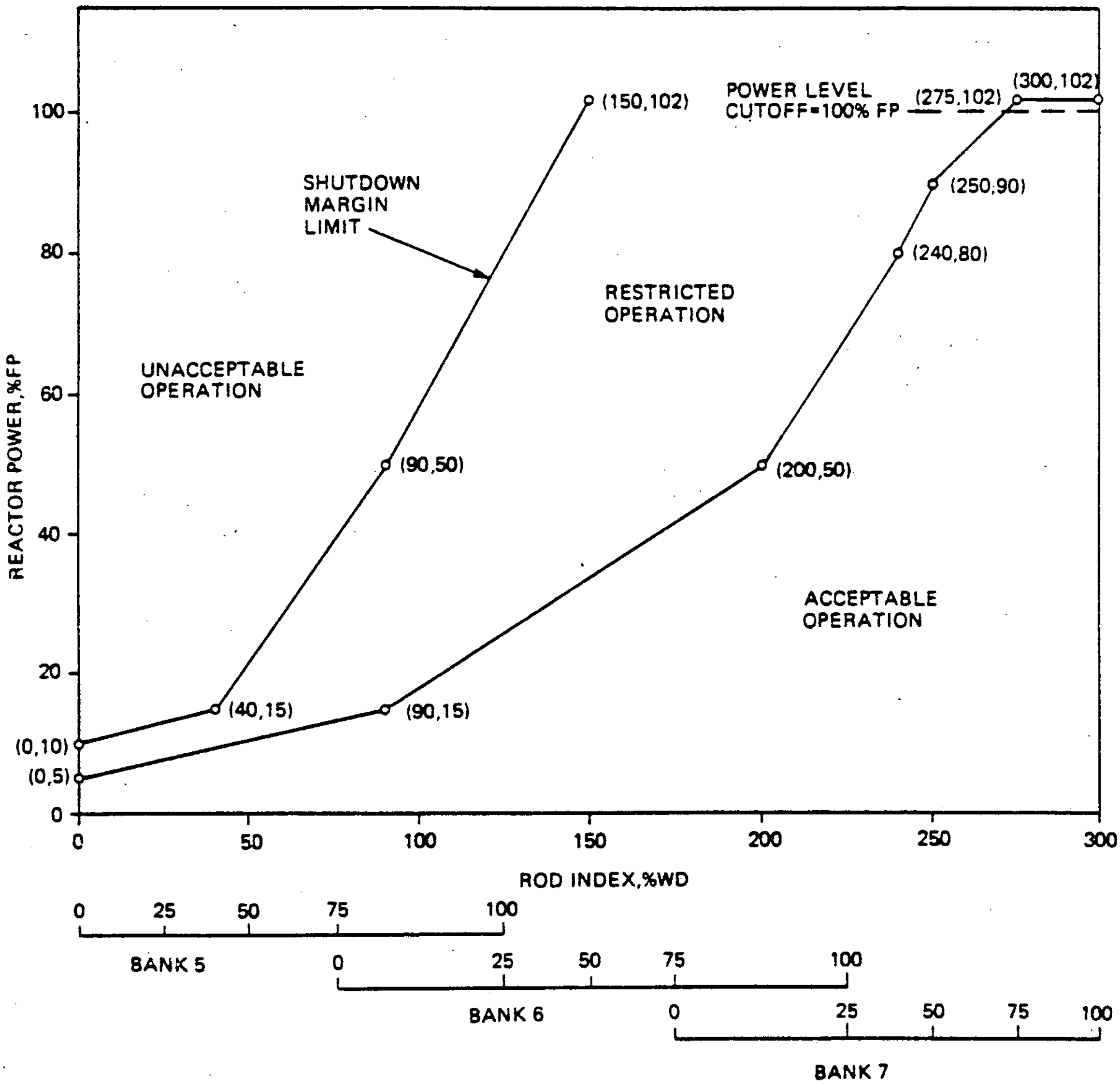


Figure 8-7

Control Rod Position Limits, 4 Pumps, After 200 $\pm$ 10 EFPD

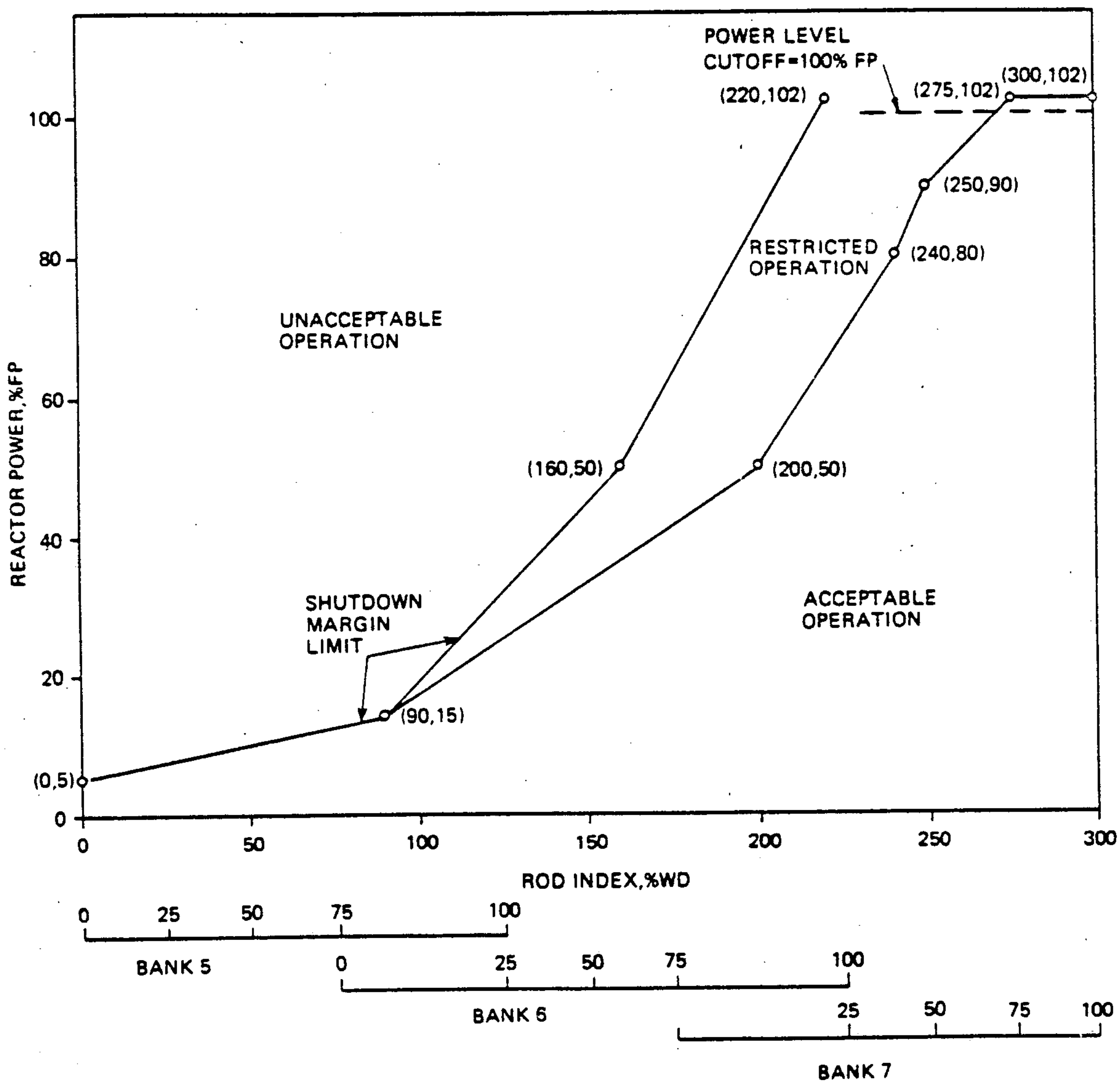


Figure 8-8

Control Rod Position Limits, 3 Pumps, 0-25 $^{+10}_{-0}$ EFPD

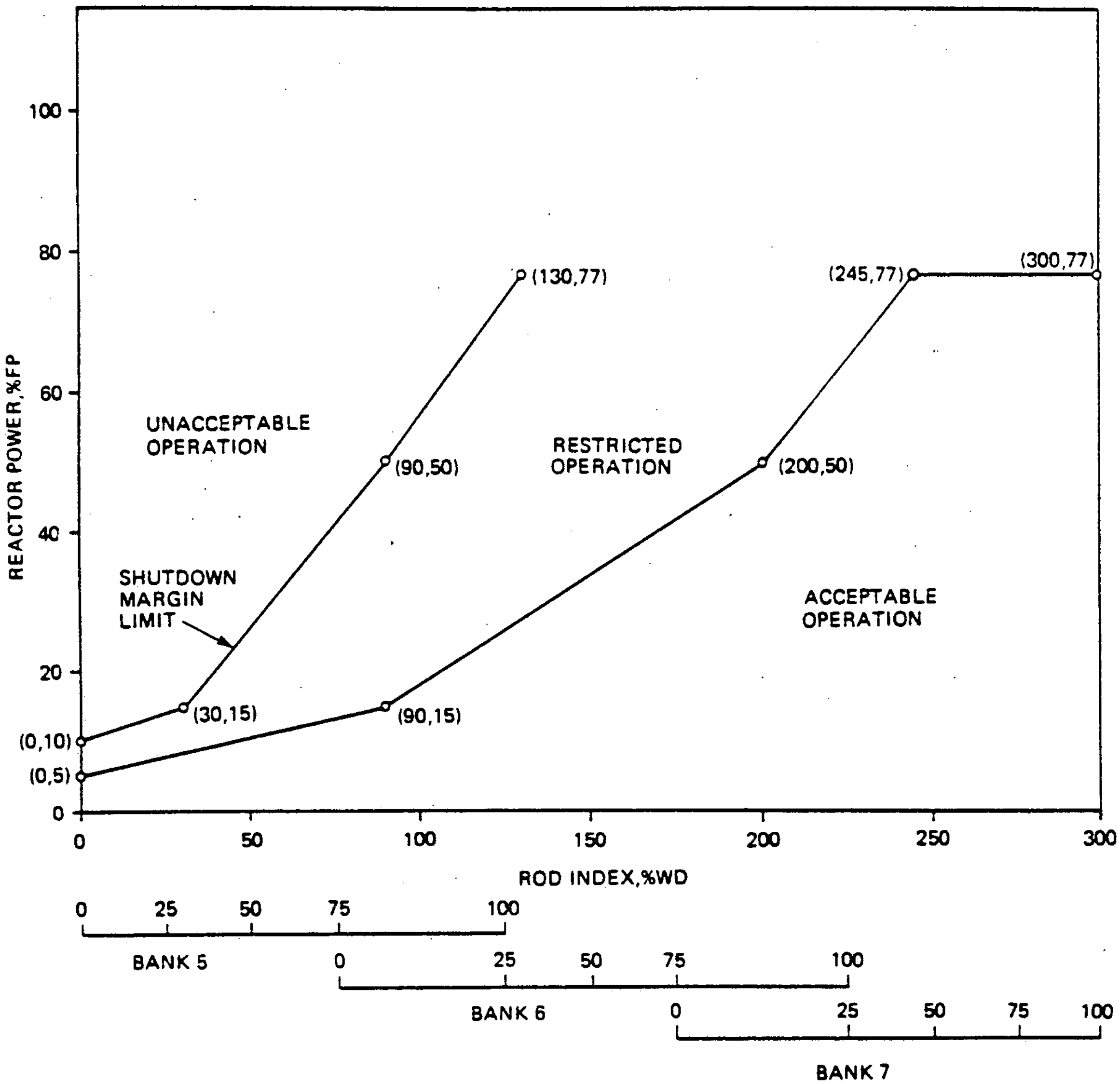


Figure 8-9

Control Rod Position Limits, 3 Pumps, 25 $^{+10}_{-0}$ - 200 $\pm$ 10 EFPD

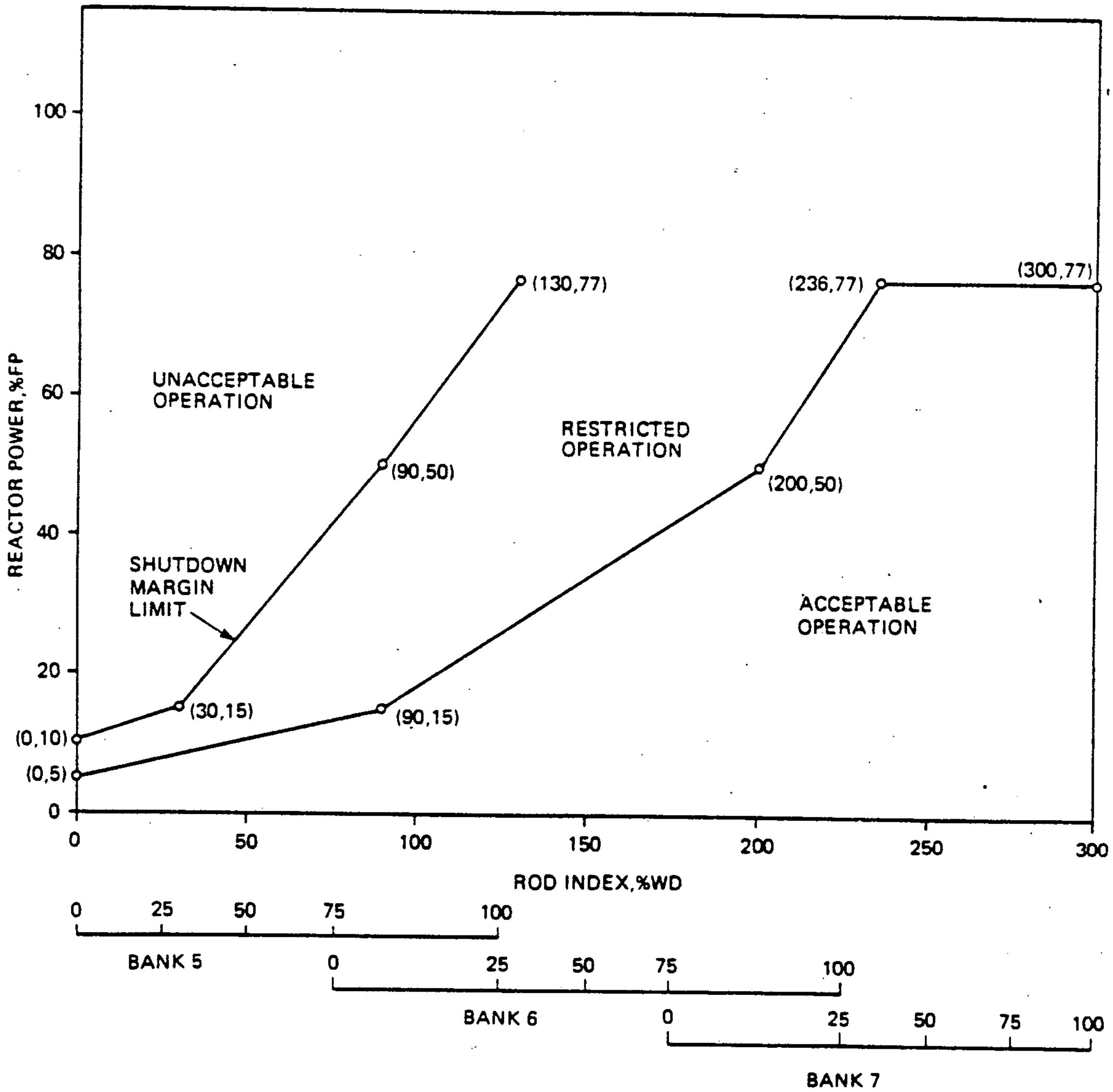


Figure 8-10

Control Rod Position Limits, 3 Pumps, After 200 ± 10 EFPD

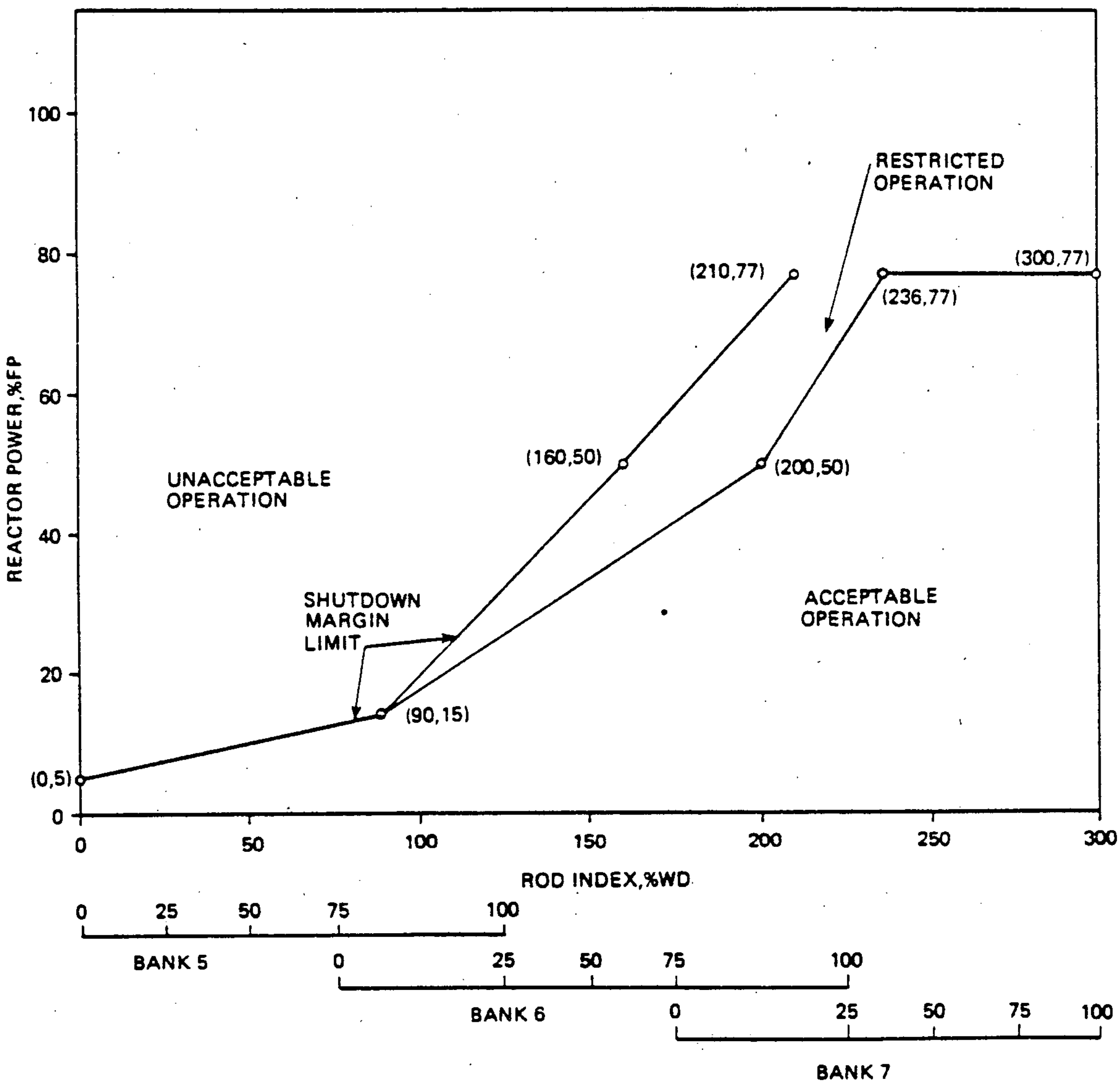


Figure 8-11

Control Rod Position Limits, 2 Pumps, 0-25 $^{+10}_{-0}$ EFPD

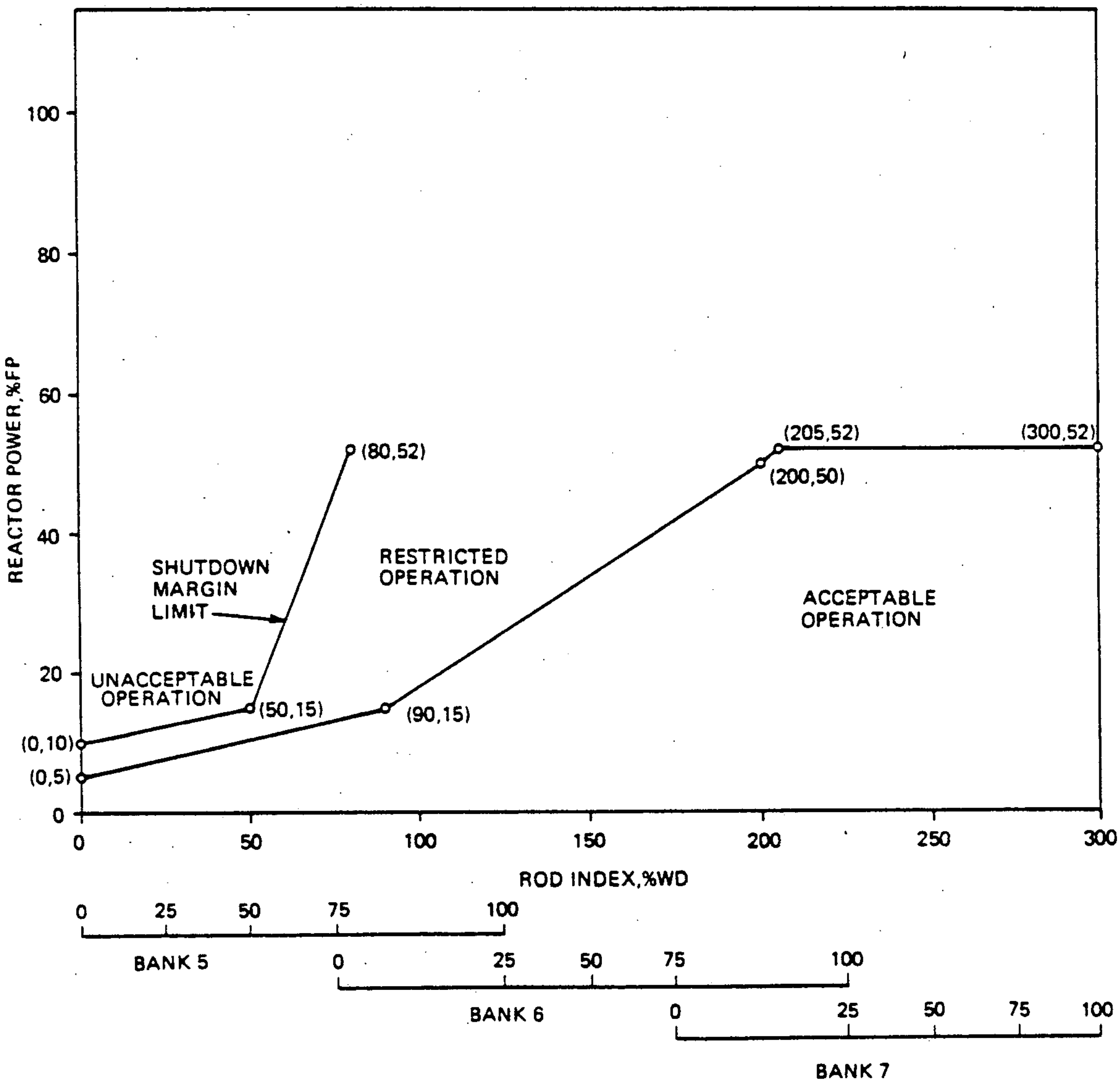


Figure 8-12

Control Rod Position Limits, 2 Pumps, $25 \pm {}^{10}_{0}$ - 200 ± 10 EFPD

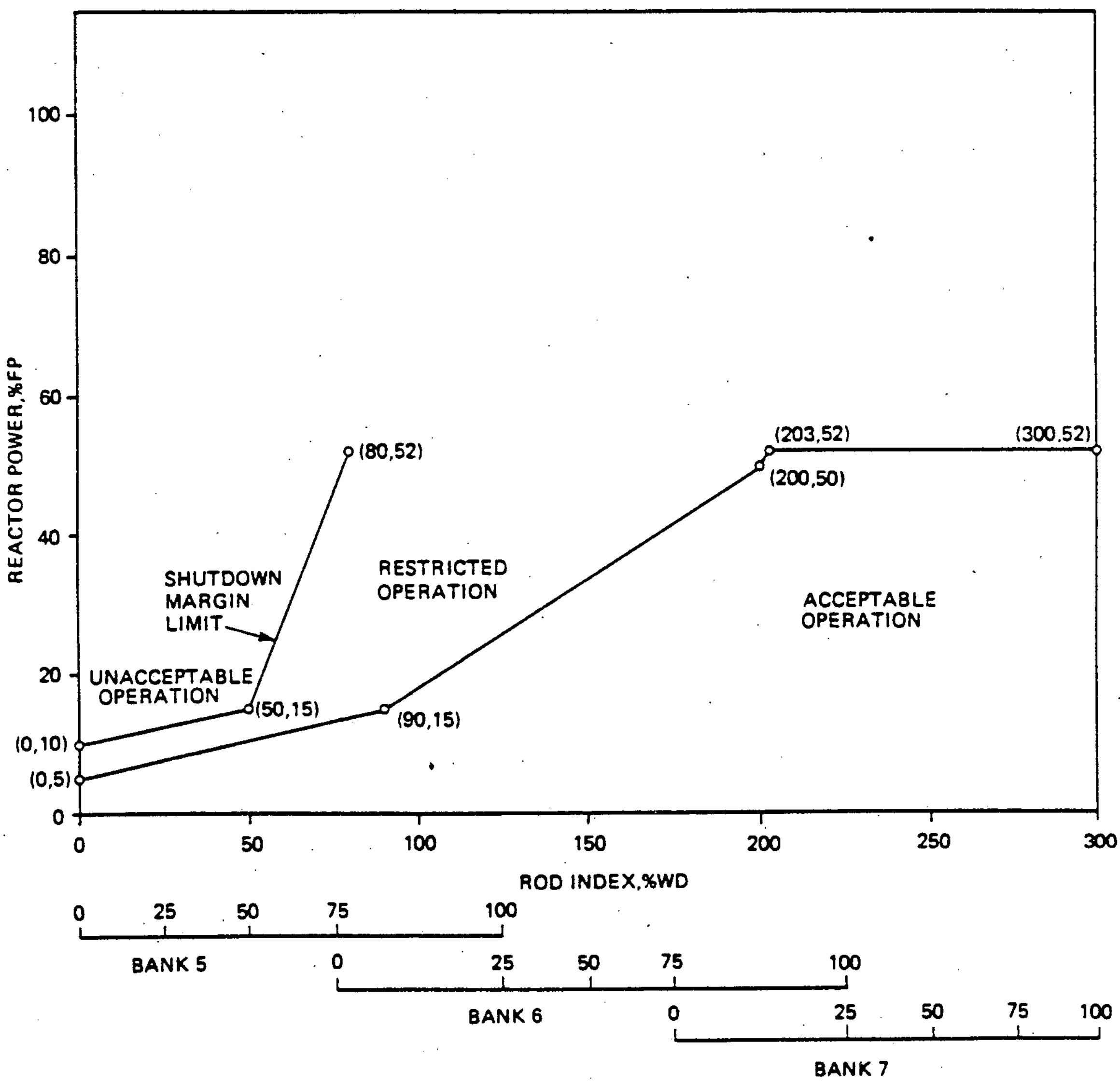


Figure 8-13

Control Rod Position Limits, 2 Pumps, After 200 $\pm$ 10 EFPD

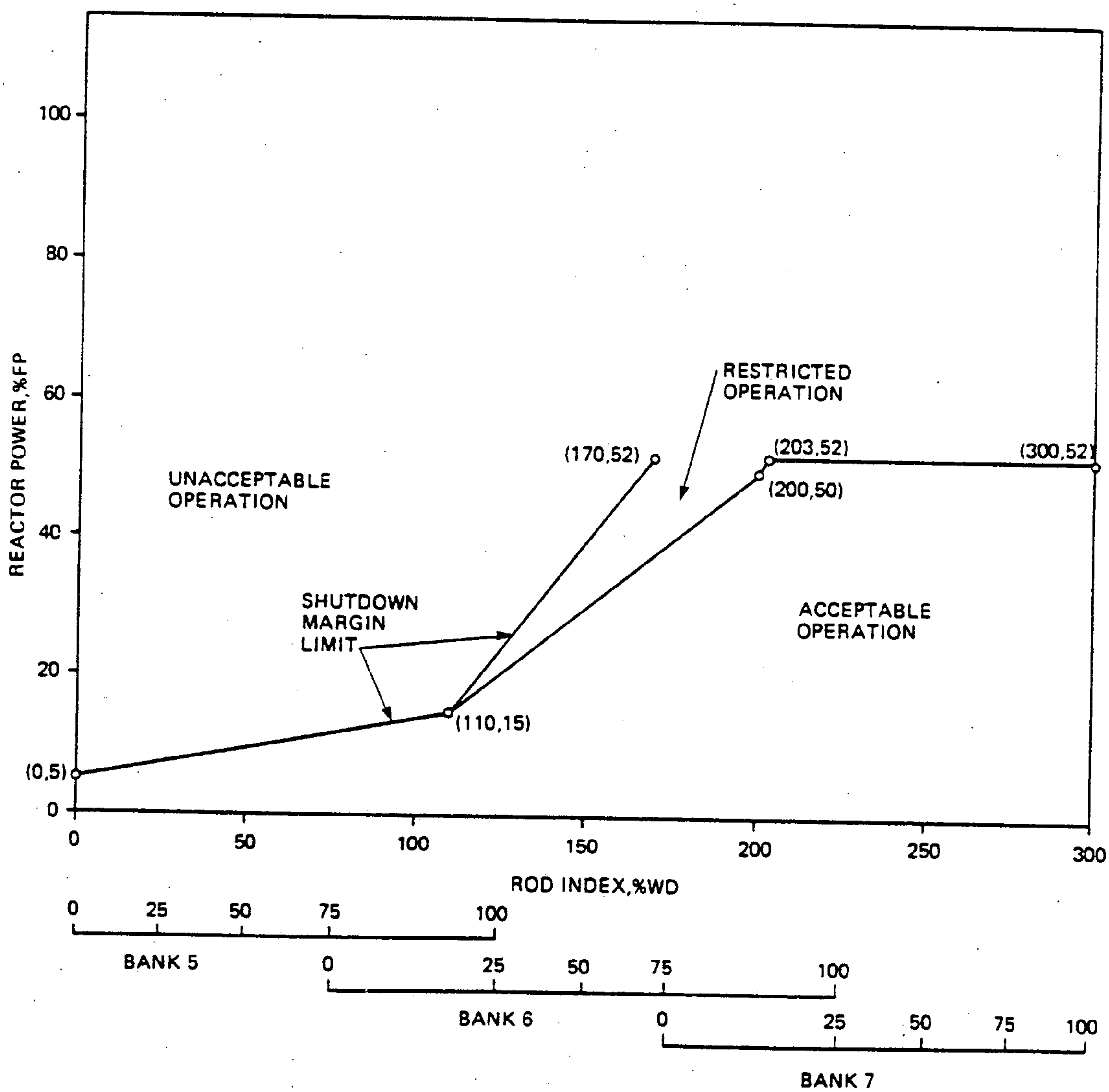


Figure 8-14

APSR Position Limits, 0-200 ± 10 EFPD

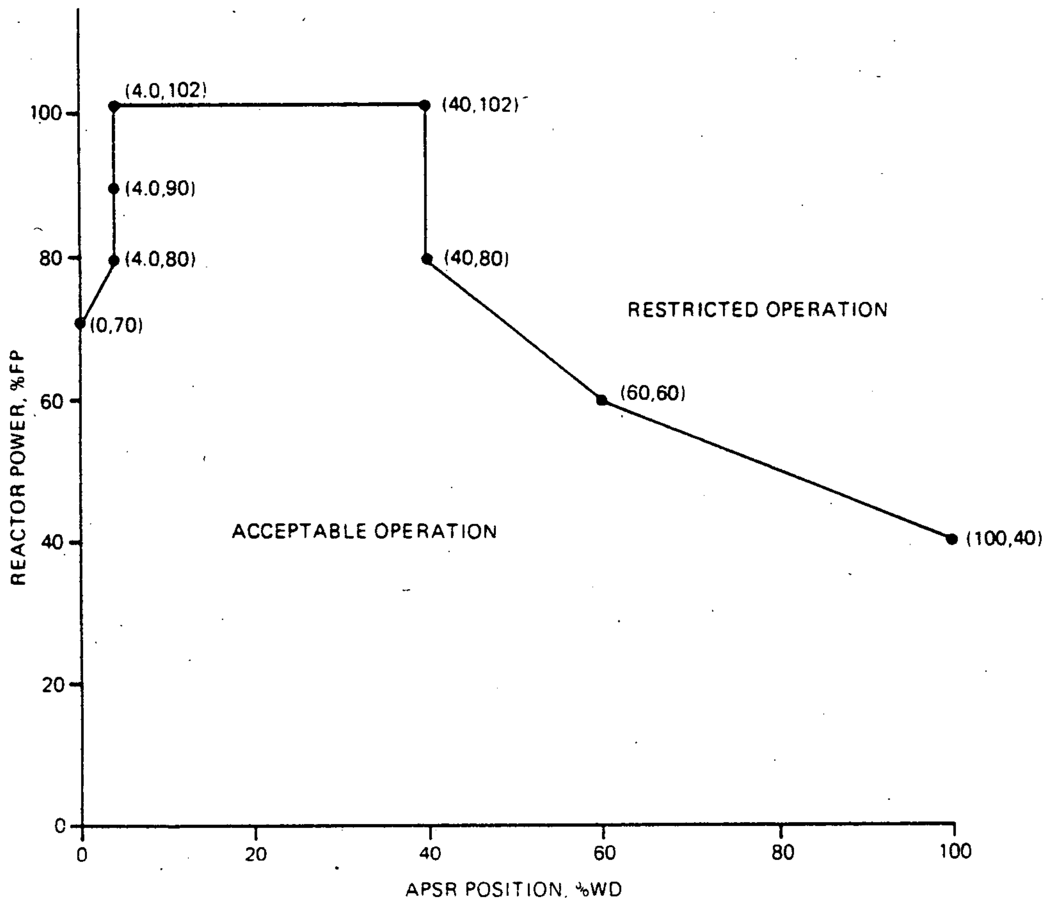
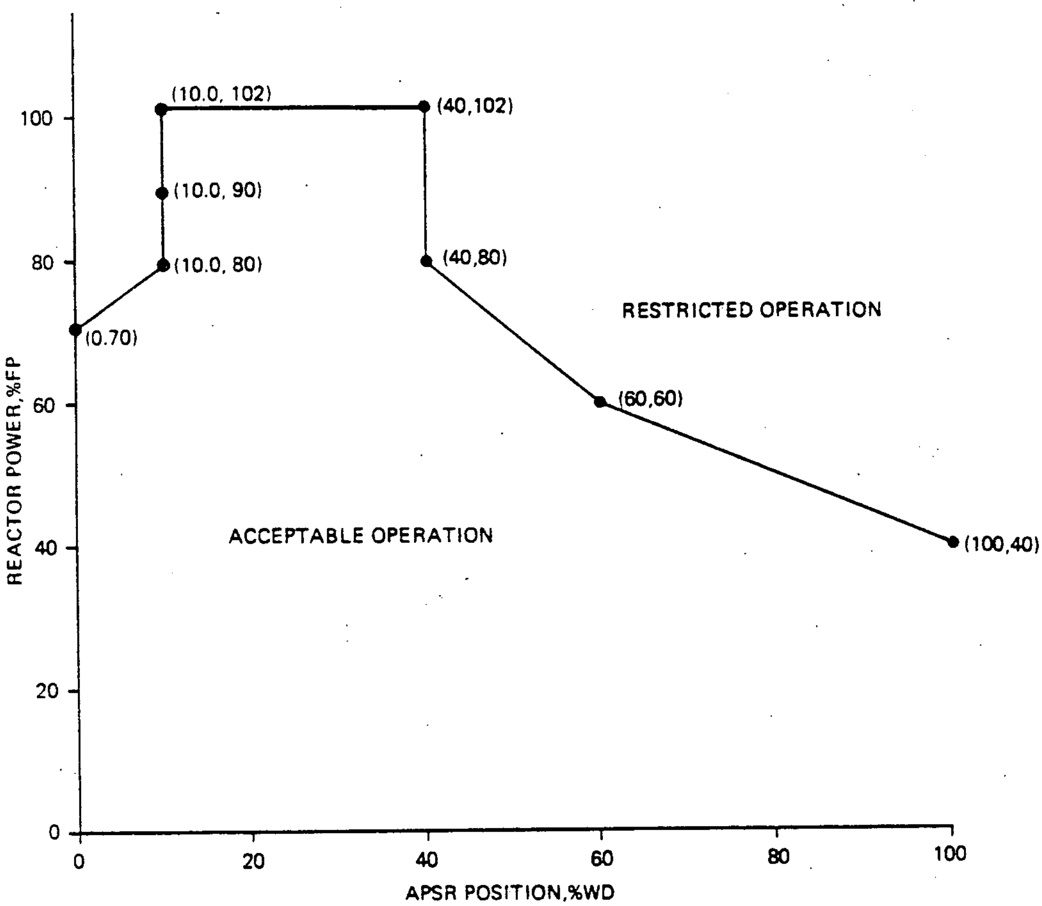


Figure 8-15

APSR Position Limits, 200 ± 10 EFPD to EOC



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2 Program to Determine In-reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084A, Rev. 2, Babcock & Wilcox Co., Lynchburg, Virginia, October 1978.

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4 TACO - Fuel Performance Analysis, BAW-10087A, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, August 1977.

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