Duke Power Company
Oconee Nuclear Station

Oconee Unit 3, Cycle 7 Reload Report

DPC-RD-2001

Revision 1

July 1982

OCONEE UNIT 3, CYCLE 7

- Reload Report -

DPC - RD - 2001

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the seventh 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 7 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 6 and 7 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 7 operation. In those cases where cycle 7 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 7 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 7 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 7, is the currently operating cycle 6. Cycle 5 was terminated after 309 EFPD of operation. Cycle 6 achieved initial criticality on March 12, 1981 and power escalation commenced on March 14, 1981. The fuel cycle design length for cycle 7 - 440 EFPD - is based on cycle 6 length of 350 EFPD. No operating anomalies occurred during previous cycle operations that would adversely affect fuel performance in cycle 7.

Cycle 7 will operate in a feed-and-bleed mode for its entire design length, as did cycle 6.

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 7 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 7. Nineteen of the batch 7 assemblies will be discharged at the end of cycle 6 along with batches 5B, and 6. The remaining 37 batch 7 assemblies, designated "7B," and the fresh batch 9 FAs - with initial enrichments of 2.80 and 3.18 wt % ^{235}U, respectively - will be loaded into the central portion of the core. Batch 8, with an initial enrichment of 3.07 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 7.

Cycle 7 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 64 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 7 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The cycle 7 locations and enrichments of the BPRAs are shown in Figure 3-4.

FIGURE 3-1. CORE LOADING DIAGRAM FOR OCONEE 3 CYCLE 7

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						N03 8	G14 8	N05 8	G02 8	N13 8					
B				M02 8	9	F11 8	9	M04 8	9	F05 8	9	M14 8			
C			K08 8	9	F13 8	9	K04 8	9	K12 8	9	F03 8	9	H07 8		
D		B11 8	9	H13 8	9	A07 7B	9	M08 8	9	A09 7B	9	O08 8	9	B05 8	
E		9	O06 8	9	C03 7B	9	A06 7B	N02 7B	A10 7B	9	C13 7B	9	O10 8	9	
F	C12 8	M06 8	9	G01 7B	9	H01 7B	9	P04 7B	9	A08 7B	9	G15 7B	9	M10 8	C04 8
G	P07 8	9	D09 8	9	F01 7B	9	F02 7B	9	B10 7B	9	F15 7B	9	D07 8	9	P09 8
H W—	M12 8	N11 8	9	H11 8	P12 7B	N14 7B	9	P10 7B	9	D02 7B	B04 7B	H05 8	9	D05 8	E04 8
K	B07 8	9	N09 8	9	L01 7B	9	P06 7B	9	L14 7B	9	L15 7B	9	N07 8	9	B09 8
L	O12 8	E06 8	9	K01 7B	9	R08 7B	9	B12 7B	9	H15 7B	9	K15 7B	9	E10 8	O04 8
M		9	C06 8	9	O03 7B	9	R06 7B	D14 7B	R10 7B	9	O13 7B	9	C10 8	9	
N		P11 8	9	C08 8	9	R07 7B	9	E08 8	9	R09 7B	9	H03 8	9	P05 8	
O			H09 8	9	L13 8	9	G04 8	9	G12 8	9	L03 8	9	G08 8		
P				E02 8	9	L11 8	9	E12 8	9	L05 8	9	E14 8			
R						D03 8	K14 8	D11 8	K02 8	D13 8					

X (axis above column 8) — Y (axis right of row H) — Z (axis below column 8)

XX X	PREVIOUS CYCLE LOCATION BATCH NO.

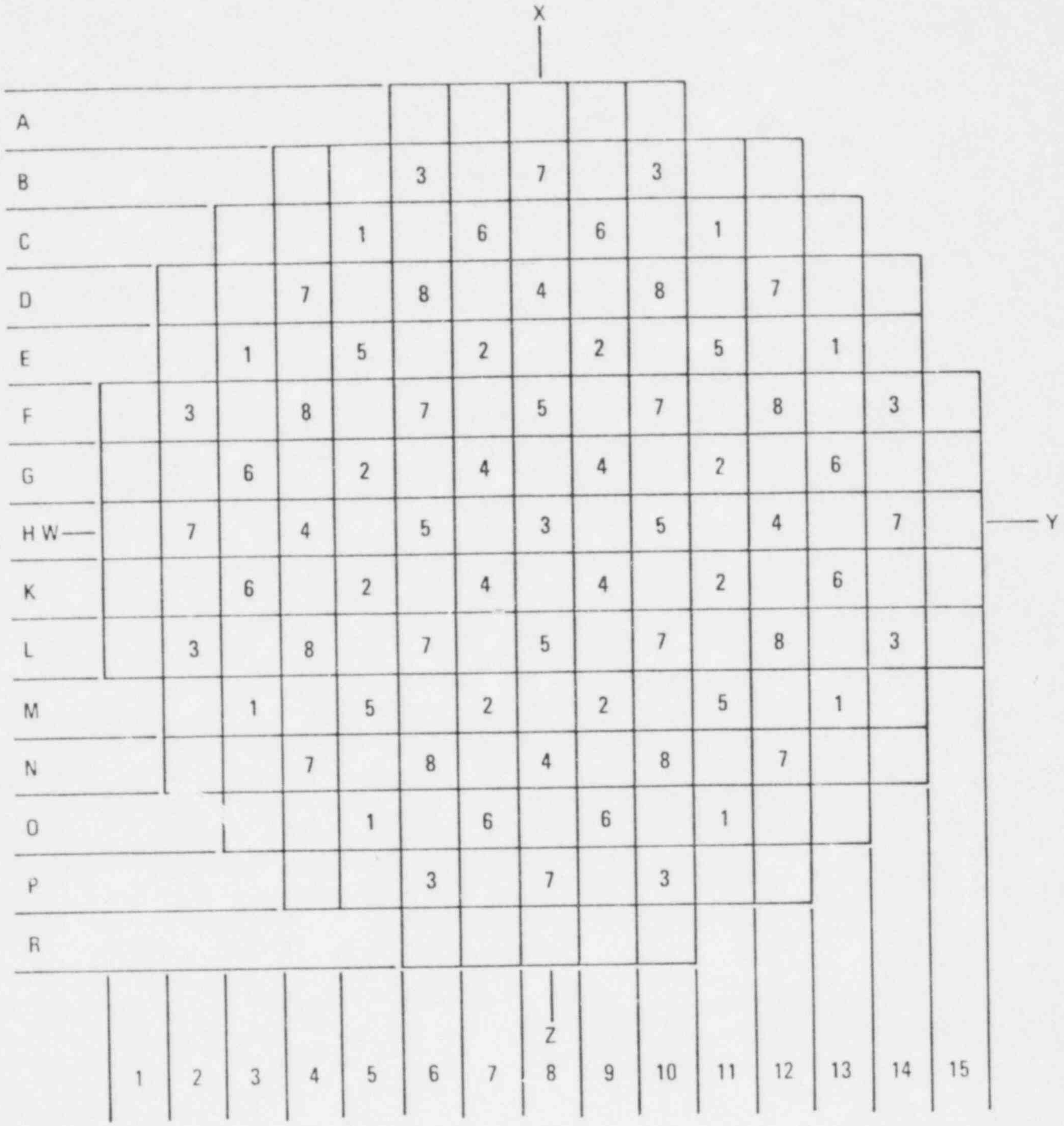
FIGURE 3-2. ENRICHMENT AND BURNUP DISTRIBUTION FOR OCONEE 3, CYCLE 7

	8	9	10	11	12	13	14	15
H	2.80 20132	3.18 0	2.80 11639	2.80 11637	3.07 13659	3.18 0	3.07 13692	3.07 13628
K		2.80 20128	3.18 0	2.80 11702	3.18 0	3.07 14069	3.18 0	3.07 12671
L			2.80 14935	3.18 0	2.80 14780	3.18 0	3.07 13637	3.07 11003
M				2.80 12995	3.18 0	3.07 13949	3.18 0	
N					3.07 14105	3.18 0	3.07 9631	
O						3.07 13259		
P								
R								

S_L

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

FIGURE 3-3. CONTROL ROD LOCATIONS FOR OCONEE 3, CYCLE 7



$\boxed{X}$ GROUP NO.

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

FIGURE 3-4. BPRA ENRICHMENT AND DISTRIBUTION FOR OCONEE 3, CYCLE 7

	8	9	10	11	12	13	14	15
H		1.4				1.4		
K			1.4		1.4		0.2	
L				1.4		0.5		
M					1.0			
N						0.2		
O								
P								
R								

S_L

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 7, are listed in Table 4-1. All fuel assemblies are identical in concept and are mechanically interchangeable. Four regenerative neutron sources will be used: two will be contained in MK B5 fuel assemblies and two in MK B4 assemblies. Retainer assemblies will be used on the two MK B4 FAs containing the regenerative neutron sources. The justification for the design and use of the BPRA retainers is described in reference 3 and 21, which is also applicable to the RNS retainers of Oconee 3, cycle 7.

The batch 9 Mark B5 fuel assemblies have redesigned upper end fittings which provide a positive holddown of BPRAs. Section 4.1.1 describes the design features of this end fitting. All 64 BPRAs will be inserted into batch 9 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, [illegible]ribed below, which envelope the cycle 7 design. All methods are consi[illegible] with the approved methodologies of Reference 16 except where specifically stated.

4.1.1 Mark B5 Fuel Assembly

Batch 9 fuel assemblies are Babcock & Wilcox Mark B5 fuel assemblies (FA's). The Mark B5 assembly is identical to the Mark B4 except that its upper end fitting has been developed to provide a positive holdown of fixed control components such as burnable poison rod assemblies, neutron source rod assemblies, and orifice rod assemblies (should reinsertion of orifice rod assemblies be desirable to minimize core bypass flow). The B4 and B5 FA's function identically with existing handling equipment and movable control components, such as control rod assemblies and axial power shaping rod assemblies.

A spring loaded retainer assembly, references 3 and 21, is used with the Mark B4 FA design to insure positive holddown of the fixed control components at all design flow conditions. A locking-ball coupling attaches the control components to the FA.

The Mark B5 upper end fitting, Figure 4-1, provides four open slots that align and allow designed movement of the holddown spring retainer, Figure 4-2, and the B5 fixed control component spider, Figure 4-3 and 4-4. The holddown spring used in the B5 FA will provide positive holddown capability, with or without a fixed control component installed, for all design flow conditions. The holddown spring is preloaded through a stop pin, welded to an ear on each side of the upper end fitting. In core, the spider feet are captured between the holddown spring retainer and the upper grid pads on the reactor internals as shown in Figure 4-5. This arrangement retains the B5 fixed control components at all design flow conditions.

Mark B5 fixed control component assemblies are not compatible with B4 FA's for in core operation and vice versa. Cycle 7 has been designed to preclude mixing of control component designs and this will be verified by video prior to plenum installation.

It has been determined that no special treatment of the B5 assembly is required for core reload design analyses. The upper end fitting form loss coefficient remains significantly unchanged, and the fuel rod design remains unchanged. Therefore, the thermal-hydraulic and fuel rod mechanical analyses are unaffected.

4.2 Fuel Rod Design

The mechanical evaluation of the fuel rod is discussed below.

4.2.1 Cladding Collapse

The fuel of batch 7B is more limiting than other batches due to its longer previous incore exposure time. The batch 7B 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 TACO[4] 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 35,000 EFPH, which is greater than the maximum projected residence time of cycle 7 fuel (Table 4-1). 1

4.2.2 Cladding Stress

Duke has performed a generic and conservative fuel rod cladding stress analysis. This analysis is consistent with the methodology described in Reference 16 with the following exception: the fuel rod total stress (primary plus secondary) was permitted to exceed the unirradiated yield strength. Two times the minimum unirradiated yield strength (2.0 Sy) has been used as a criterion for the total stress calculation, as permitted by Section III, Article NB-3000 of the ASME Boiler and Pressure Vessel Code. Approximately 0.35 Sy margin remains in this total stress calculation.

Primary membrane plus primary bending stresses are limited to 1.0 Sy, and
primary membrane stress is limited to 2/3 Sy. Substantial margin exists in 1
both of these evaluations.

The following conservatisms exist in the generic cladding stress calculation:

- Specification cladding dimensions which result in highest stress
- a low internal pressure (HZP);
- a high external pressure (110 percent of design pressure);
- a large through wall cladding temperature gradient (fuel melt conditions), and
- BOL grid loads for worst grid cell type (based on as-built cladding diameter and spacer grid cell size)

4.2.3 Cladding Strain

Duke has performed a cladding strain calculation using TACO in accordance with the approved methodology.[16] This analysis demonstrated that the uniform, circumferential strain of the cladding was within 1.0%.

4.3. Thermal Design

All fuel in the cycle 7 core is thermally similar. The fresh batch 9 fuel inserted for cycle 7 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 computer code. 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. LTL initial density 1
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.

Reference 16, Section 4.6, states that "no credit is taken for fuel relocation in LHRTM analyses". This is an error. Fuel relocation is assumed in these analyses in that relocation is an integral part of the TACO model. However, credit for restructuring is not assumed in these analyses, in accordance with Reference 4.

Fuel rod internal pressure has been evaluated using TACO 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 9 fuel assemblies are not unique in concept (excluding the upper end fitting design modification of the Mark B5 fuel assembly), nor do they utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 9 fuel assemblies is identical to those of the present fuel.

Table 4-1. Fuel Design Parameters and Dimensions

	Batch No. 7B	Batch No. 8	Batch No. 9
FA type	Mark B4	Mark B4	Mark B5
No. of FAs	37	68	72
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.	142.2	141.8	141.8
Fuel pellet OD (mean spec), in.	0.3695	0.3686	0.3686
Fuel pellet initial density (mean spec), %TD	94.0	95.0	95.0
Initial fuel enrichment, wt % ^{235}U	2.80	3.07	3.18
Est residence time, EOC 7, EFPH	26,376	18,960	10,560
Cladding collapse time, EFPH	>35,000	>35,000	>35,000

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No.		
	7B	8	9
Initial density, % TD	93.50(a)	94.66	94.58
Max. In-reactor densification, % TD	2.65(a)	1.25	1.85
Burnup corresponding to max. densification, MWd/mtU	3964(a)	2784	3317
Initial pellet diameter, in.	0.3691(b)	0.3683	0.3682
Initial clad ID, in.	0.3770(b)	0.3776	0.3773
Initial clad OD, in.	0.4300(b)	0.4306	0.4303
Average linear heat rate @ 100% of 2568 MW, kW/ft	5.73	5.74	5.74
Linear heat rate capability(d) from 0-10,000 MWD/MTU, kW/ft	20.5	20.5	20.5
Linear heat rate capability(d) >10,000 MWD/MTU, kW/ft	21.5	21.5	21.5
Average fuel temp. @ nominal linear heat rate, °F	1250(c)	1240	1240

(a) Basis: Batch specific pellet resinter data

(b) Basis: Pellet and cladding as-fabricated dimensions (95/95 tolerances)

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

(d) These values are utilized as fuel design limits for Cycle 7.

FIGURE 4-1
MARK B5 UPPER END FITTING (SIDE VIEW)

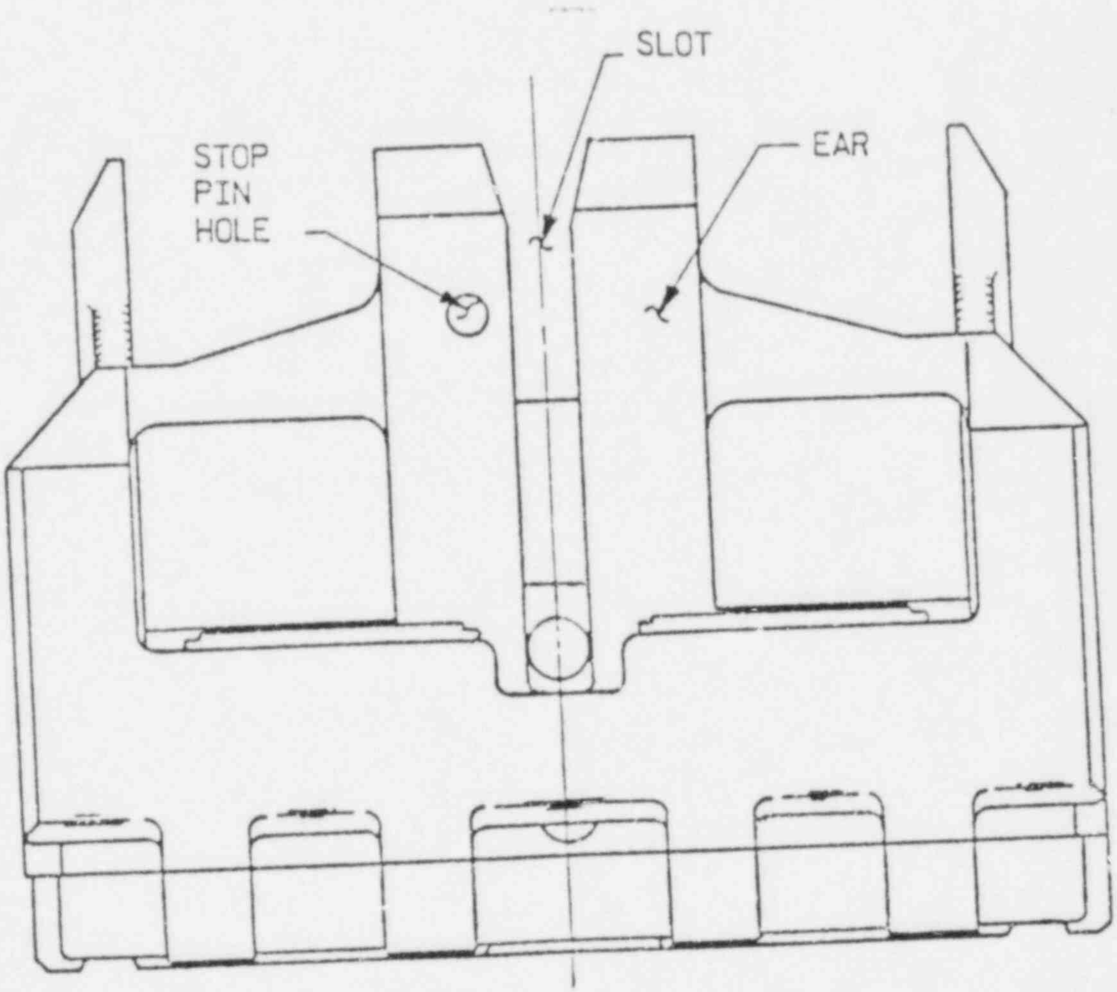


FIGURE 4-2
MARK B5 HOLDDOWN SPRING RETAINER

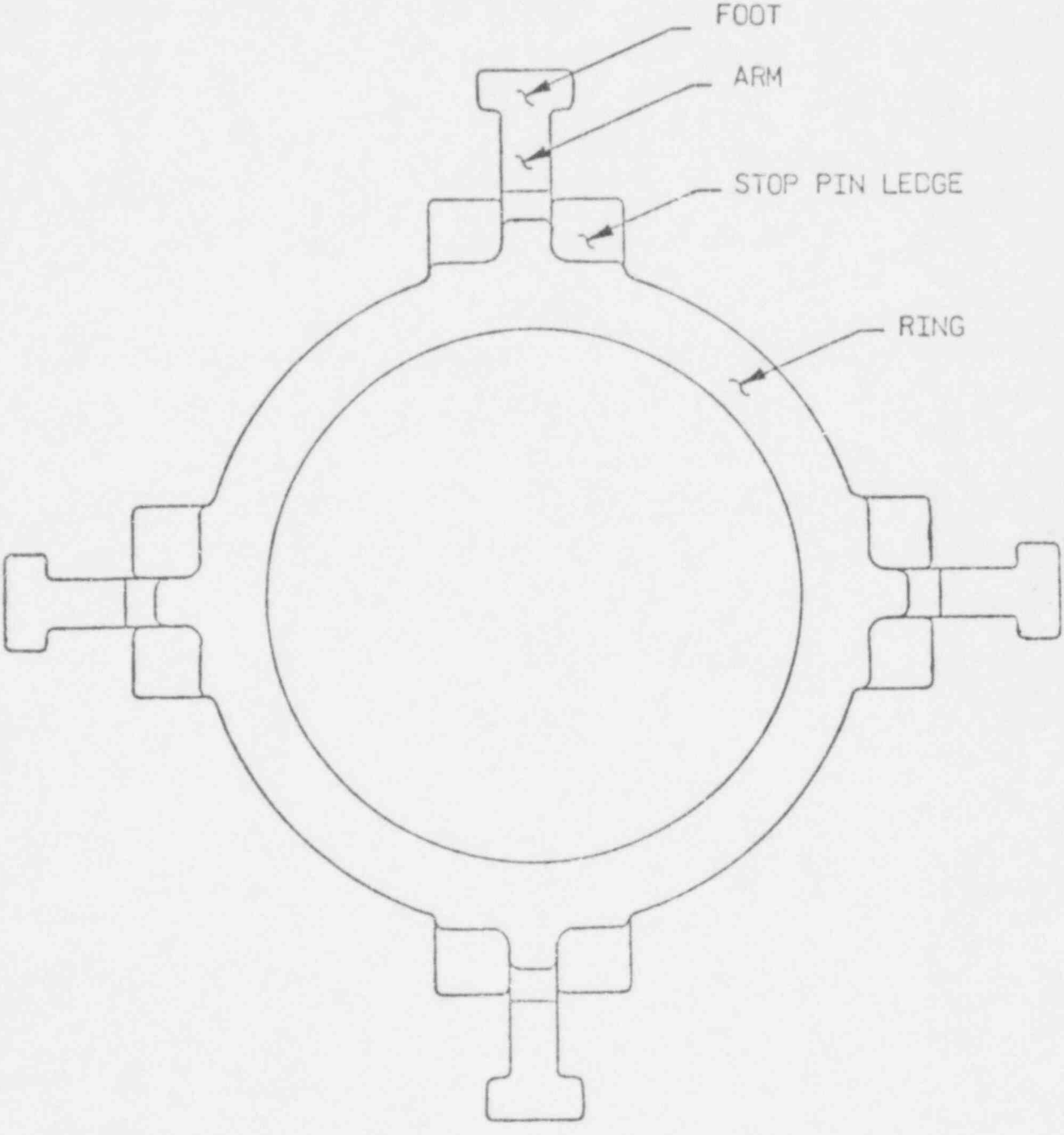


FIGURE 4-3
MARK B5 FIXED CONTROL COMPONENT SPIDER
(TOP VIEW)

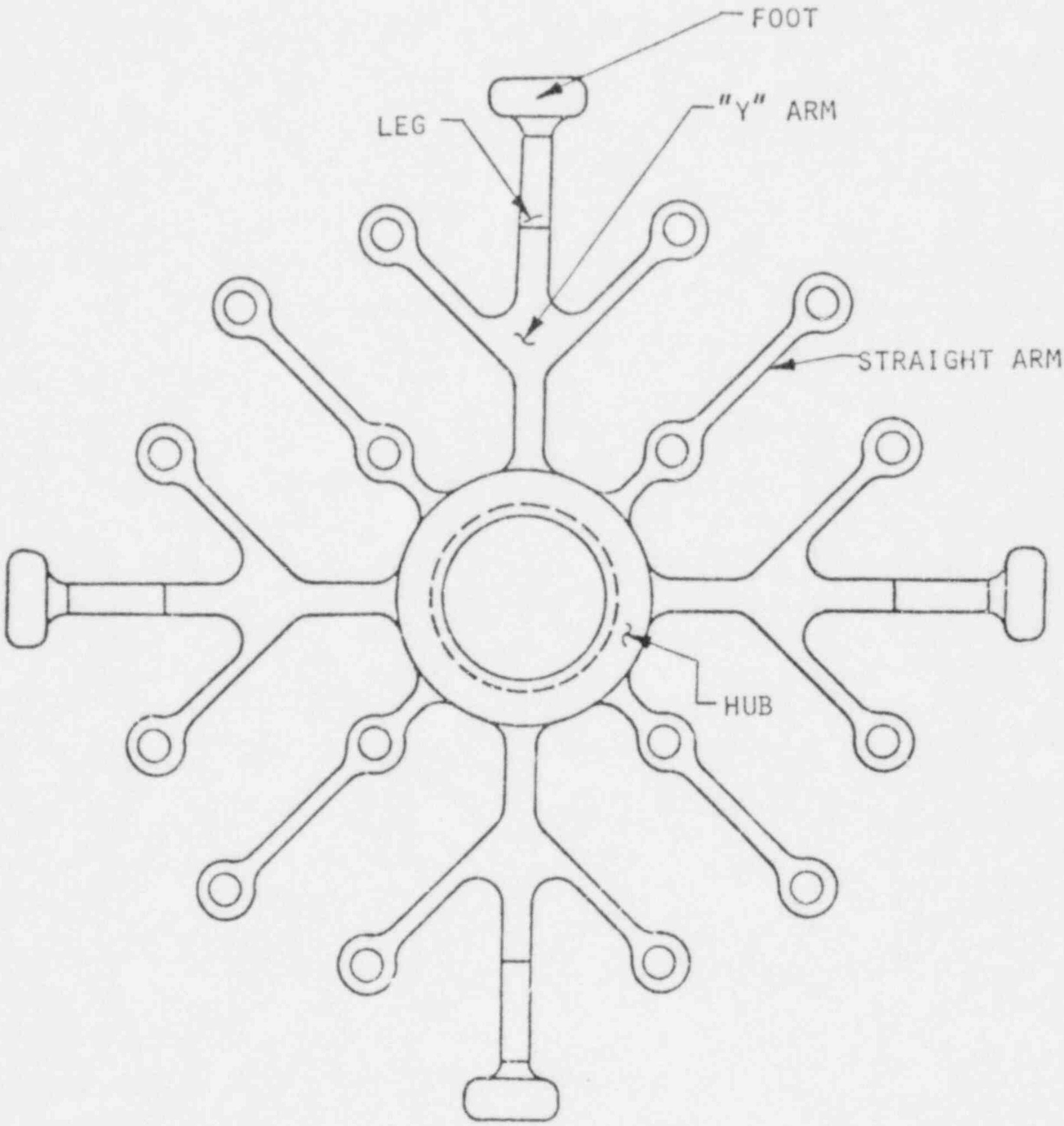


FIGURE 4-4
MARK B5 COUPLING - SPIDER ASSY - SIDE VIEW (SECTION)

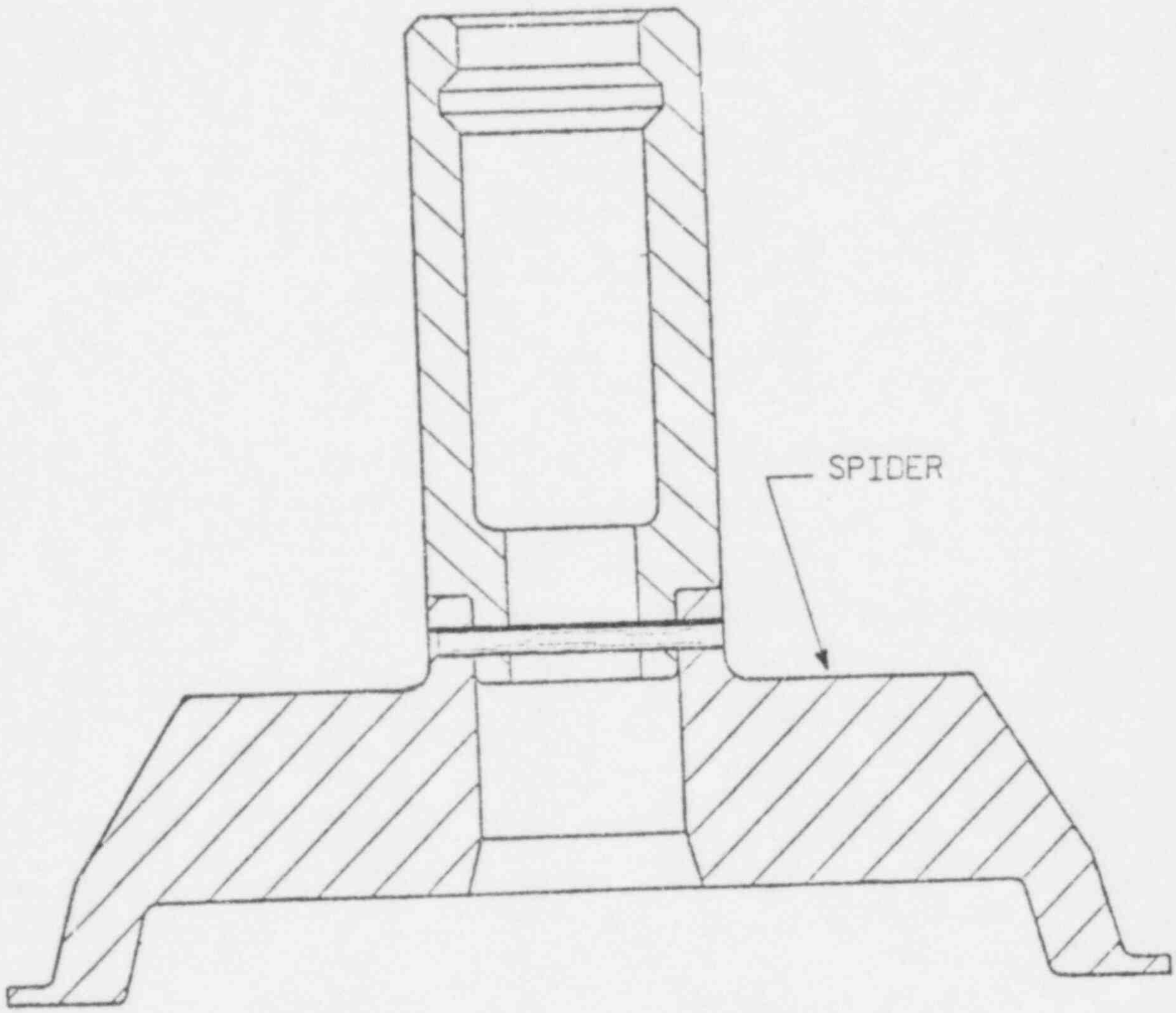
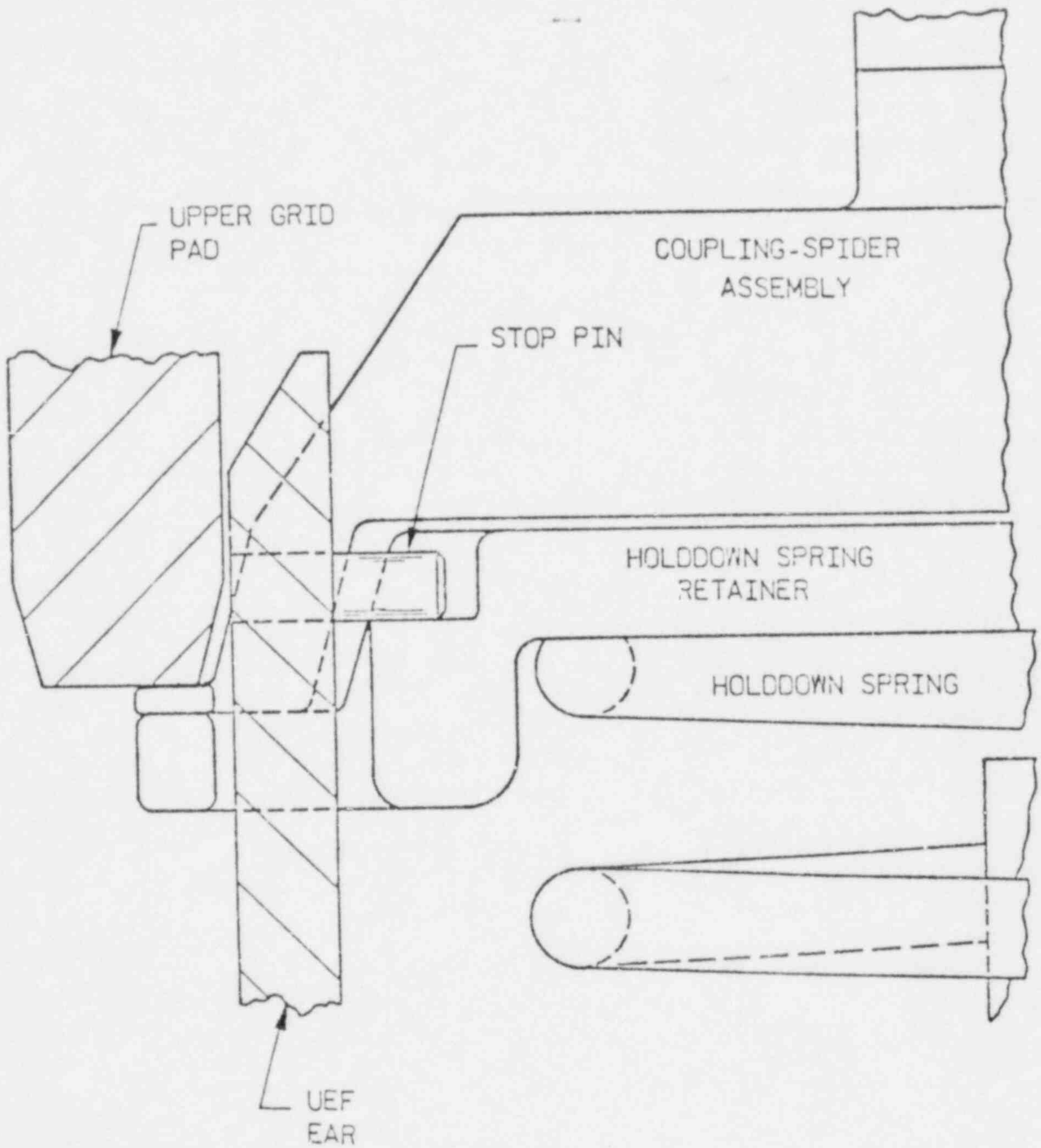


FIGURE 4-5
MARK B5 FIXED CONTROL COMPONENT SPIDER/UPPER
END FITTING INTERACTION



5. NUCLEAR DESIGN

5.1 Physics Characteristics

Table 5-1 compares the core physics parameters of design cycles 6 and 7; the values for cycle 6 were generated by B&W [6, 7, 8, 13, 15] using PDQ07 while the values for cycle 7 were generated by Duke Power Company using methods described in Reference 16. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. The longer cycle 7 will produce a higher cycle burnup than that for the design cycle 6. Figure 5-1 illustrates a representative relative power distribution for the beginning of the seventh cycle at full power with equilibrium xenon and normal rod positions.

The initial BPRA loading, longer design life, different shuffle pattern, and different control rod pattern for cycle 7 make it difficult to compare the physics parameters with those of cycle 6. The BOC critical boron concentrations for cycle 7 are higher because the additional reactivity necessary for the longer cycle is not completely offset by burnable poison. 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 with cycle 7 stuck worths 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. The reference fuel cycle shutdown margin is presented in the Oconee 3, cycle 6 reload report.[5]

The cycle 7 power deficits, differential boron worths, and effective delayed neutron fractions differ from those of cycle 6 because of the longer cycle length and differences in core loading.

5.2 Analytical Input

The cycle 7 incore measurement calculation constants to be used to compute core power distributions were obtained in a similar manner for cycle 7 as for the reference cycle, however, CASMO[17] was used to verify the F-factors derived from B&W's codes.

5.3 Changes in Nuclear Design

There are only two significant core design changes between the reference cycle and the reload cycle. The cycle lifetime is increased to 440 EFPD requiring an increase in the number of fresh fuel assemblies and BPRAs. 1
Duke Power calculational methods[16] are used to obtain the important nuclear design parameters for this cycle.

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

	Cycle 6[b]	Cycle 7[c]
Cycle length, EFPD	376	440
Cycle burnup, MWd/mtU	11,766	13,752
Average core burnup, EOC, MWd/mtU	20,231	21,608
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	1471	1623
HFP, group 7 at 87% WD, 8 at 25% WD	1282	1440
Critical boron - EOC (equilibrium xenon), ppm		
HZP, group 7 at 100% WD, 8 at 37.5% WD	385	396
HFP, group 7 at 87% WD, 8 at 25% WD	78	11
Control rod worths - HFP, BOC, % Δk/k		
Group 6	0.98	1.22
Group 7	1.36	1.47
Group 8 (25% to 100% WD)	0.50	0.33
Control rod worths - HFP, EOC, % Δk/k		
Group 7	1.48	1.65
Group 8 (25% to 100% WD)	0.54	0.29
Max ejected rod worth - HZP, % Δk/k		
BOC, (N12) groups 5-8 inserted	0.38	0.74
EOC, (N12) groups 5-8 inserted	0.51	0.82
Max stuck rod worth - HZP, % Δk/k		
BOC (N12)	1.39	1.50
EOC (N12)	1.52	2.06
Power deficit, HZP to HFP, % Δk/k		
BOC	1.39	1.76
EOC	2.22	3.12
Doppler coeff - BOC, 10^{-5} (Δk/k-°F)		
100% power (no xenon)	-1.49	-1.32
Doppler coeff - EOC, 10^{-5} (Δk/k-°F)		
100% power (equilibrium xenon)	-1.62	-1.68

Table 5-1. (Cont'd)

	Cycle 6[b]	Cycle 7[c]
Moderator coeff - HFP, 10^{-4} ($\Delta k/k$-°F)		
BOC (no xenon, 1370 ppm, group 8 ins.)	-0.65	-0.34
EOC (equilibrium xenon, 15 ppm, group 8 ins.)	-2.82	-2.85
Boron worth - HFP, ppm/% $\Delta k/k$		
BOC (1058 ppm)	116	121
EOC (50 ppm)	102	108
Xenon worth - HFP, % $\Delta k/k$		
BOC (4 days)	2.61	2.49
EOC (equilibrium)	2.74	2.70
Effective delayed neutron fraction - HFP		
BOC	0.00628	0.00628
EOC	0.00526	0.00518

[a] Cycle 7 data are for the conditions stated in this report. The cycle 6 core conditions are identified in reference 5.

[b] Based on a 299-EFPD cycle 5. (Actual cycle 5 length 309 EFPD).

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

Table 5-2. Shutdown Margin Calculation for Oconee 3, Cycle 7

	BOC, % Δk/k	EOC, % Δk/k
Available Rod Worth		
Total rod worth, HZP	8.21	9.07
Worth reduction due to poison burnup	-0.42	-0.42
Maximum stuck rod, HZP	-1.50	-2.06
Net worth	6.29	6.59
Less 10% uncertainty	-0.63	-0.66
Total available worth	5.66	5.93
Required Rod Worth		
Power deficit, HFP to HZP	1.76	3.12
Max inserted rod worth, HFP	0.23	0.55
Total required worth	1.99	3.67
Shutdown Margin		
Total available worth minus total required worth	3.67	2.26

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

FIGURE 5-1
OCONEE 3 CYCLE 7
TWO DIMENSIONAL
RELATIVE POWER DISTRIBUTION

HFP, 004 EFPD, EQXE
NOMINAL ROD POSITIONS

	8	9	10	11	12	13	14	15
H	0.814	1.025	1.031	1.074	1.174	1.307	1.039	0.561
K		0.854	1.129	1.094	1.260	1.217	1.232	0.556
L			0.986	1.178	1.003	1.309	0.944	0.433
M				1.098	1.240	1.082	0.888	
N					1.082	1.065	0.507	
O						0.542		
P								
R								

S_L

6. THERMAL-HYDRAULIC DESIGN

The incoming batch 9 fuel is hydraulically and geometrically similar to the fuel remaining in the core from previous cycles. The thermal-hydraulic design analysis supporting cycle 7 operation was performed by Duke Power Company and employed the methods and models described in references 1, 5, 9 and 16.

The maximum core bypass flow for cycle 6 was 8.1% of the total system flow. For cycle 7 operation, 64 BPRAs will be inserted, and four assemblies contain regenerative neutron sources. The number of open assemblies is 40, and the maximum core bypass flow is reduced to 7.6.%. The cycle 6 and 7 maximum design conditions are summarized in Table 6-1.

A net rod bow DNBR penalty of 0.0% was calculated for cycle 7, taking credit for the flow area reduction hot channel factor used in all DNBR calculations. The penalty was based on the highest batch 9 assembly burnup, 17,500 MWD/MTU.

An analysis was performed to conservatively determine the minimum allowable reduction in pin peak as a function of burnup required to offset rod bow DNBR penalty, reference 18. The result was used to demonstrate that the increase in DNBR associated with the lower pin peaks (relative to the limiting batch 9 assembly) for the limiting batch 7 and 8 assemblies more than offsets the increased rod bow DNBR penalty that would be calculated for the higher 1
assembly burnups of batch 7 or 8 fuel.

The reduction in pin peak (relative to the limiting batch 9 assembly) and the minimum allowable reduction in pin peak are given in Tables 6-2 and 6-3 for batch 7 and 8 fuel, respectively. The required minimum reduction in pin peak was determined for the limiting assembly burnup as a function of time rather than on the basis of the maximum EOC assembly burnup which is overly conservative.

For cycle 7 operation a flux to flow setpoint of 1.08 is maintained. The minimum DNBR value determined by the flux to flow setpoint analysis is above the design minimum DNBR of 1.30. All other plant operating limits based on DNBR criteria include a minimum of 10.2% DNBR margin from the B&W-2 correlation design limit of 1.30.

Table 6-1. Thermal Hydraulic Design Conditions

	Cycle 6	Cycle 7
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	8.1	7.6
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.	(a)	(a)
Avg heat flux at 100% power, 10^3 Btu/h-ft^2 [a]	176[b]	176[b]
CHF correlation	BAW-2	BAW-2
Min DNBR with densification penalty	2.05	>2.05

[a] See Table 4-1.

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

Table 6-2. Rod Bow DNBR Penalty Justification - Batch 7

Max. Peak Pin Power	1.43	
Max. Assembly Burnup	17.5	GWD/MTU

EFPD	Peak Pin Power	Assembly BU	ΔBU GWD/MTU	ΔPP %	Min. Allowable ΔPP, %
0	1.20	20.1	2.60	16.1	0.60
4	1.20	20.2	2.70	16.1	0.62
12	1.20	20.4	2.90	16.1	0.67
25	1.20	20.8	3.30	16.1	0.76
50	1.21	21.5	4.00	15.4	0.92
100	1.22	22.9	5.40	14.7	1.24
150	1.24	24.4	6.90	13.3	1.59
200	1.25	26.0	8.50	12.6	1.96
250	1.27	27.6	10.1	11.2	2.32
300	1.26	29.4	11.9	11.9	2.74
350	1.24	31.1	13.6	13.3	3.13
400	1.21	32.8	15.3	15.4	3.52
421	1.20	33.5	16.0	16.1	3.68
440	1.19	34.1	16.6	16.8	3.82

ΔBU = the change in fuel assembly burnup relative to the maximum batch 9 assembly burnup

ΔPP = the percent reduction in peak pin power relative to the maximum batch 9 peak pin power

Min. Allowable ΔPP = the minimum permissible reduction in peak pin power
= 0.23 x ΔBU (GWD/MTU), reference 18

Table 6-3. Rod Bow DNBR Penalty Justification - Batch 8

Max. Peak Pin Power	1.43
Max. Assembly Burnup	17.5 GWD/MTU

EFPD	Peak Pin Power	Assembly BU	ΔBU GWD/MTU	ΔPP %	Min. Allowable ΔPP, %
0	1.40	14.1	——*		——
4	1.39	14.3	——		——
12	1.38	14.6	——		——
25	1.37	15.1	——		——
50	1.36	16.1	——		——
100	1.33	18.0	0.50	6.99	0.12
150	1.31	19.9	2.40	8.39	0.55
200	1.29	21.7	4.20	9.79	0.97
250	1.27	23.5	6.00	11.2	1.38
300	1.25	25.2	7.70	12.6	1.77
350	1.23	27.0	9.50	14.0	2.19
400	1.21	28.7	11.2	15.4	2.58
421	1.21	29.5	12.0	15.4	2.76
440	1.20	30.2	12.7	16.1	2.92

ΔBU = the change in fuel assembly burnup relative to the maximum batch 9 assembly burnup

ΔPP = the percent reduction in peak pin power relative to the maximum batch 9 peak pin power

Min. Allowable ΔPP = the minimum permissible reduction in peak pin power
= 0.23 x ΔBU (GWD/MTU), reference 18

* The limiting assembly burnup is less than 17.5 GWD/MTU, the max. batch 9 assembly burnup. The rod bow DNBR penalty is therefore less than that for batch 9.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1 General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 6 parameters to determine the effect of the cycle 7 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 9. Since batch 9 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in reference 9, the conclusions in that reference are still valid.

No new dose calculations were performed for the reload report. The dose considerations in reference 20 are conservative for Oconee 3 cycle 7 based upon comparisons of core average burnup for the two cycles.

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

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. 1.[11] 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[11,12] 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 7 fuel.

Table 7-2 shows the bounding values for allowable LOCA peak LHRs for Oconee 3 cycle 7 fuel after 50 EFPD. The LOCA kW/ft limits have been reduced for the first 50 EFPDs. The reduction will ensure that conservative limits are maintained while a transition is being made in the fuel performance codes that provide input to the ECCS analysis[19] in order to account for mechanistic fuel densification. The limits for the first 50 EFPD are shown in Table 7-3.

From the examination of cycle 7 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 7. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 7 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 7 are bounded by the FSAR and/or the fuel densification report.[9]

Table 7.1. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR[1] value	Predicted cycle 7 value
BOC Doppler coeff, 10^{-5}, $\Delta k/k/°F$	-1.17	-1.32
EOC Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.33[a]	-1.68
BOC moderator coeff, 10^{-4}, $\Delta k/k°F$	+0.5[b]	-0.34
EOC moderator coeff, 10^{-4}, $\Delta k/k/°F$--	-3.0	-2.85
All rod bank worth, HZP, % $\Delta k/k$	10.0	9.07
Boron reactivity worth, 70°F ppm/1% $\Delta k/k$	75	83
Max. ejected rod worth, HFP, % $\Delta k/k$	0.65	0.20
Dropped rod worth, HFP, % $\Delta k/k$	0.46	0.12
Initial boron conc, HFP, ppm	1400	1440[c]

[a] -1.2×10^{-5} $\Delta k/k/F$ was used for steam-line analysis.

-1.3×10^{-5} $\Delta k/k/F$ was used for cold water accident (pump start-up).

[b] $+0.94 \times 10^{-4}$ $\Delta k/k/F$ was used for the moderator dilution accident.

[c] The combined effect of boron concentration and boron worth is conservative for Cycle 7.

Table 7-2. LOCA Limits, Oconee 3, Cycle 7, After 50 EFPD

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 7 0-50 EFPD

Elevation, ft	LHR Limits, kW/ft
2	14.5
4	16.1
6	17.5
8	17.0
10	16.0

8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 7 operation in accordance with the methods of reference 16 to account for minor changes in power peaking and control rod worths inherent with a transition to 18-month, lumped burnable poison cycles. Cycle 6 Technical Specifications were generated in accordance with the methods described in Reference 14.

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 8-18 are revisions to previous Technical Specification limits.

Figure 8-1

Core Protection Safety Power-Imbalance Limits

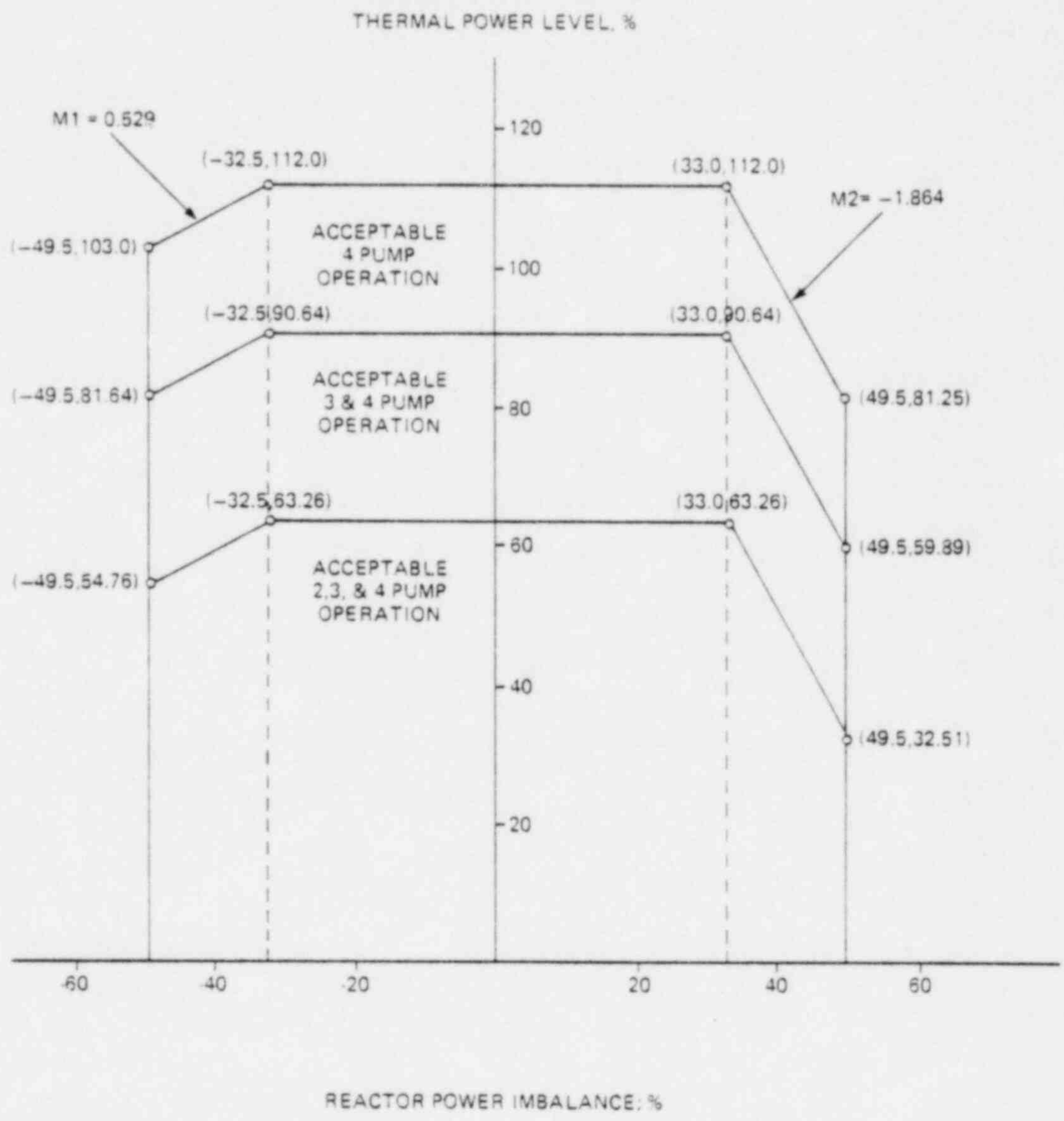


Figure 8-2
Core Protection Safety Pressure-Temperature Limits

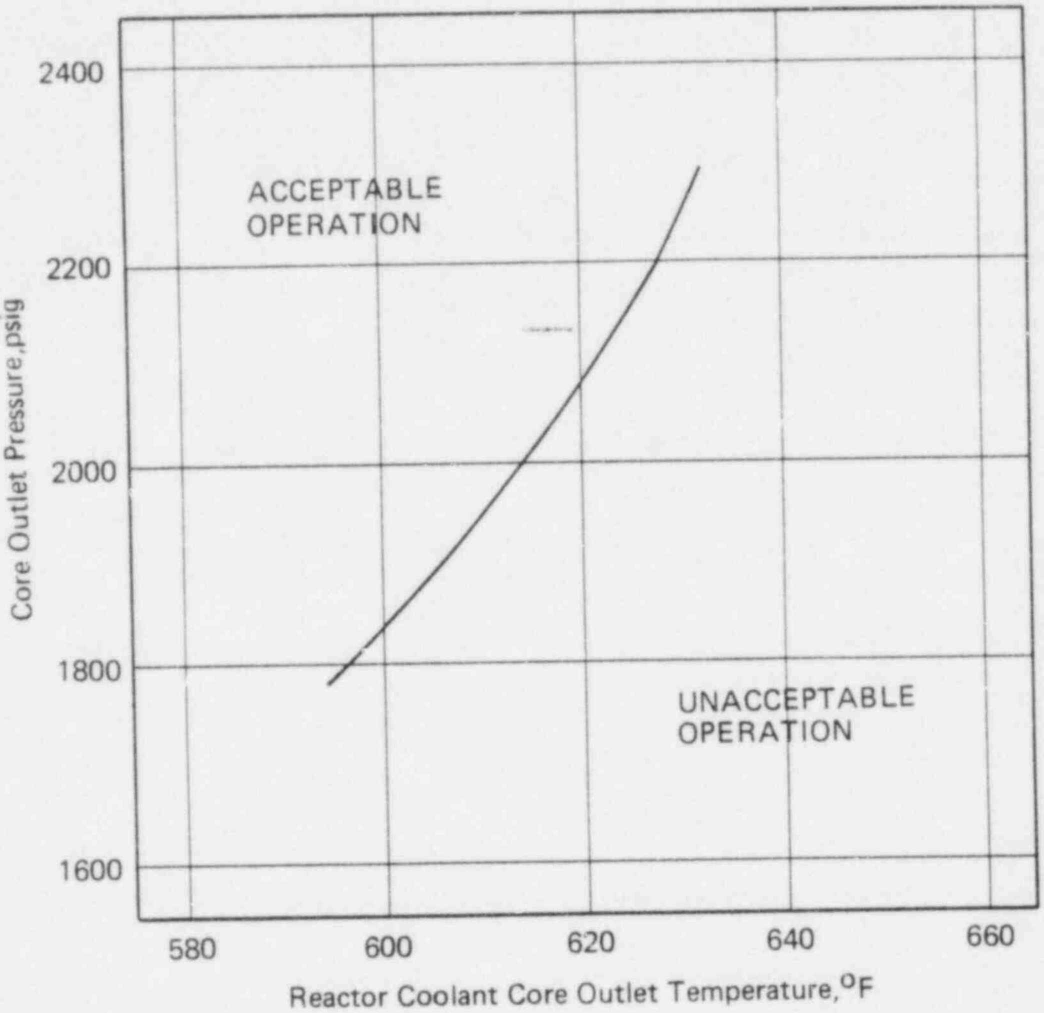
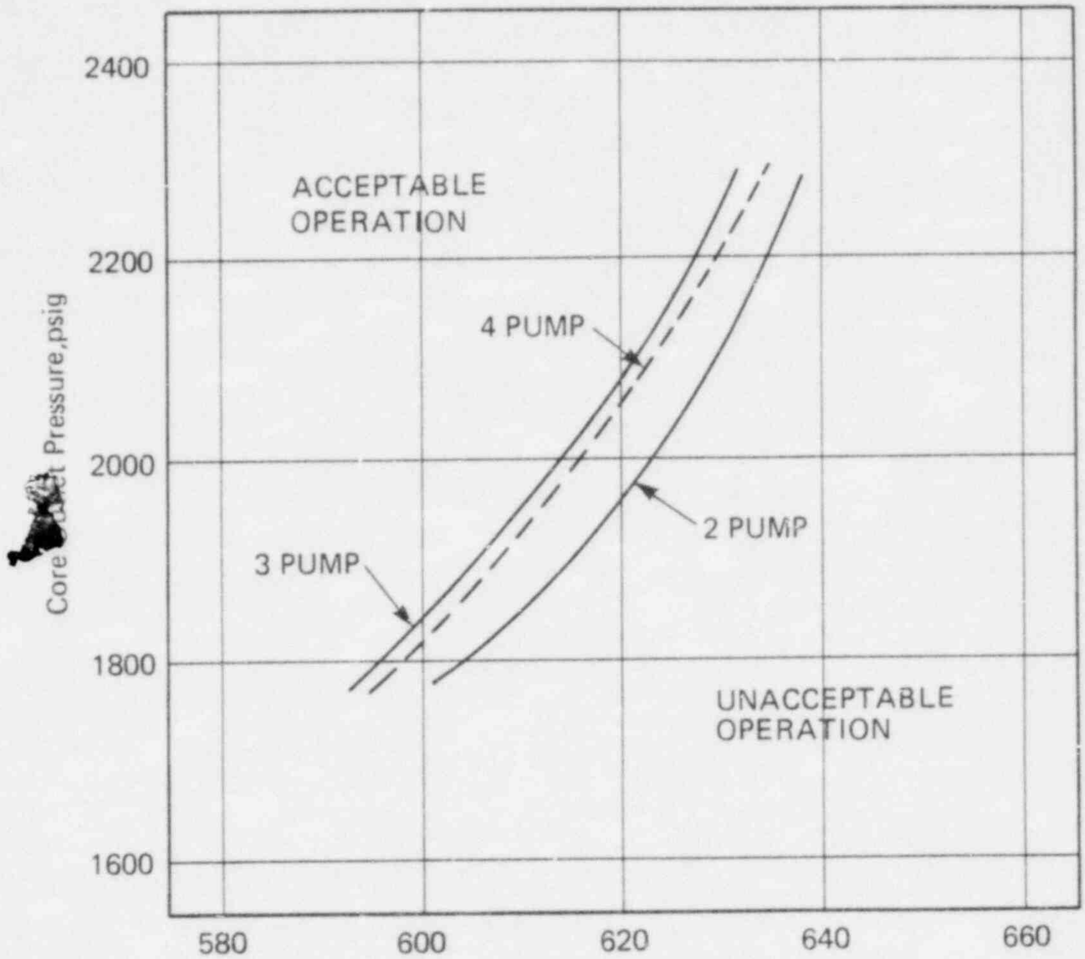


Figure 8-3

Core Protection Pressure-Temperature Limits



PUMPS OPERATING	COOLANT FLOW (GPM)	POWER (% FP)	TYPE OF LIMIT
4	374,880(100%)	112.0	DNBR
3	280,035(74.7%)	90.7	DNBR
2	183,690(49.0%)	63.63	DNBR/QUALITY

Figure 8-4

Maximum Allowable Power-Imbalance Setpoints

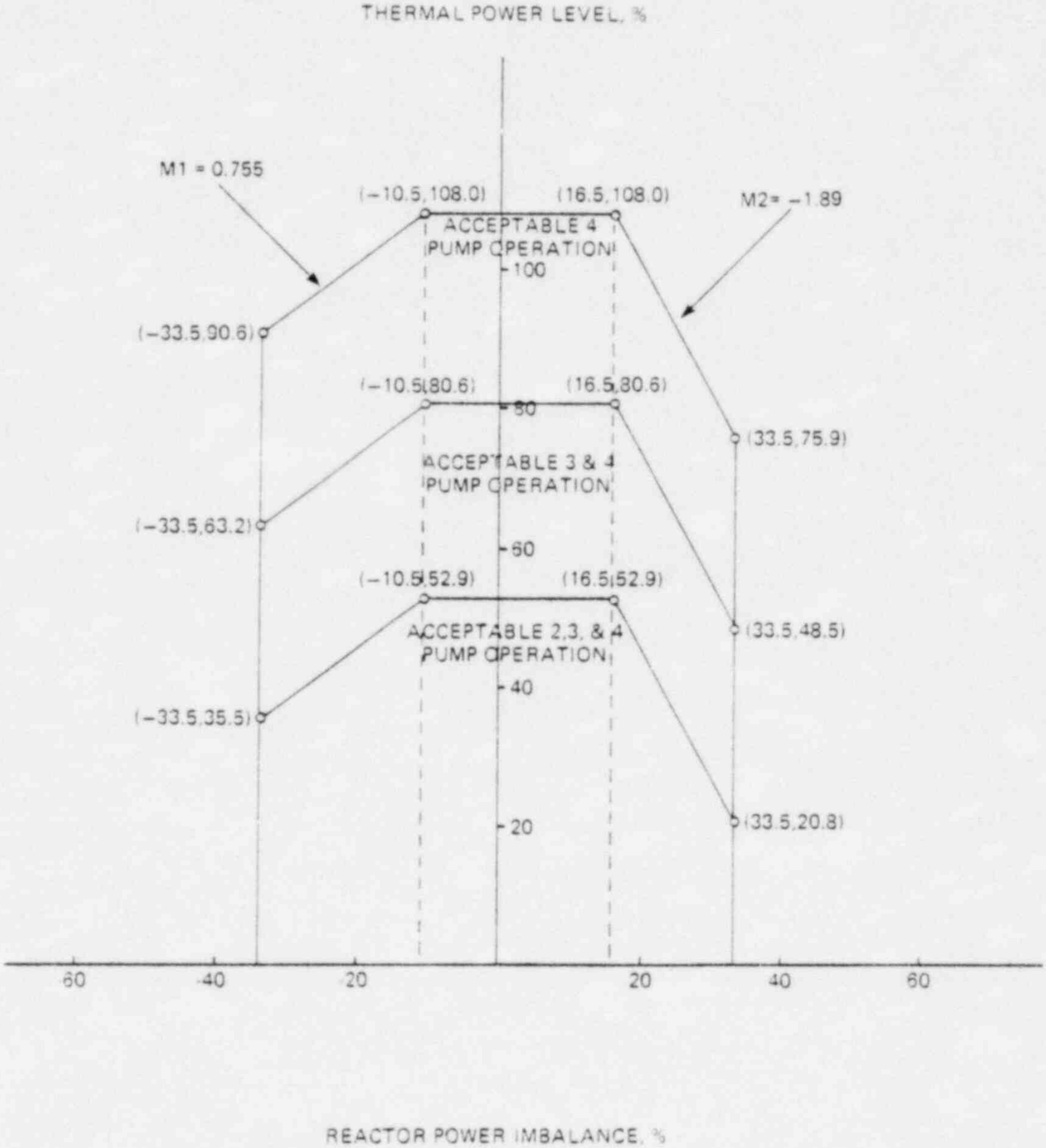


Figure 8-5

Operational Power-Imbalance Limits 0-50 $^{+10}_{-0}$ EFPD

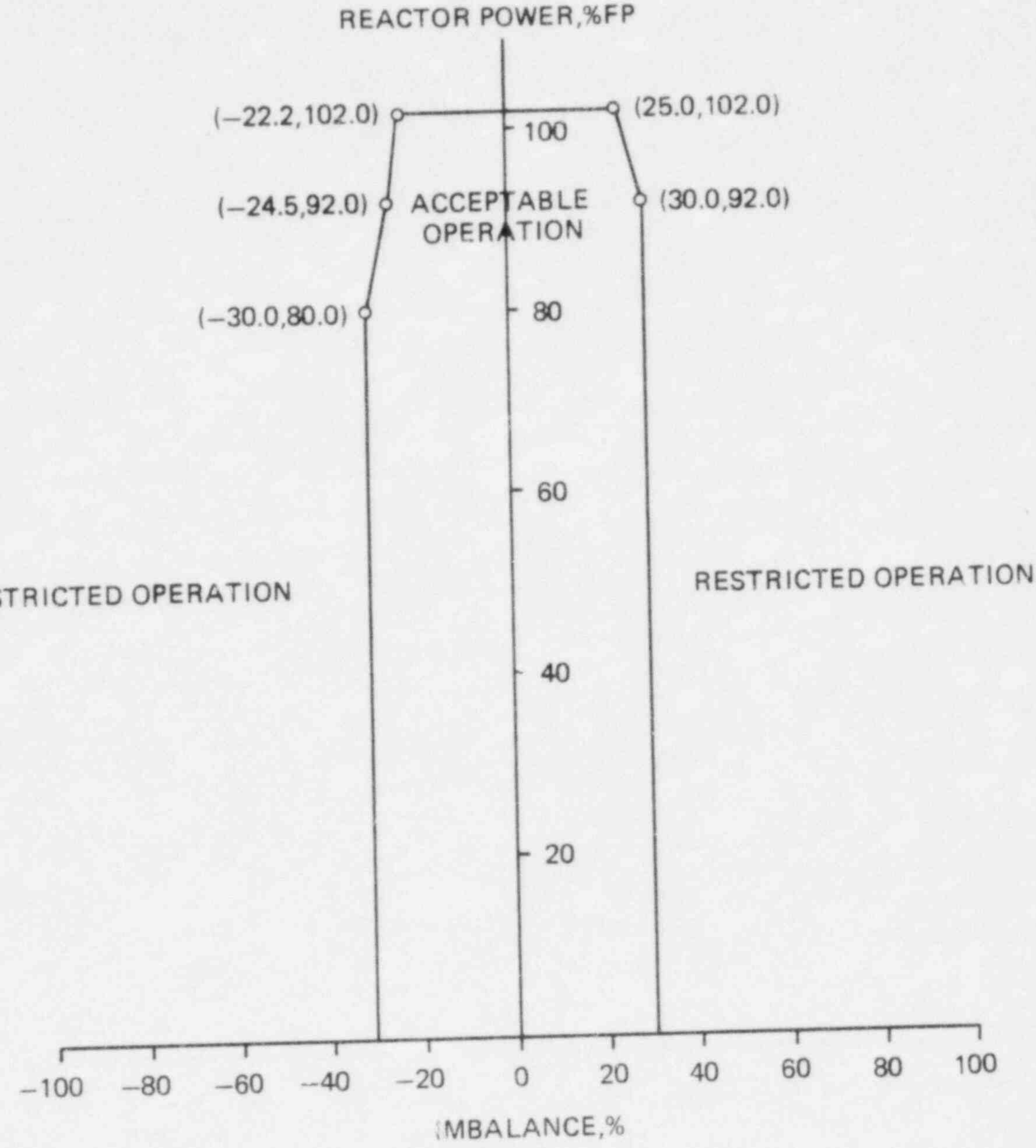


Figure 8-6

Operational Power-Imbalance Limits $50 \pm {}^{10}_{0} - 200 \pm {}^{10}_{0}$ EFPD

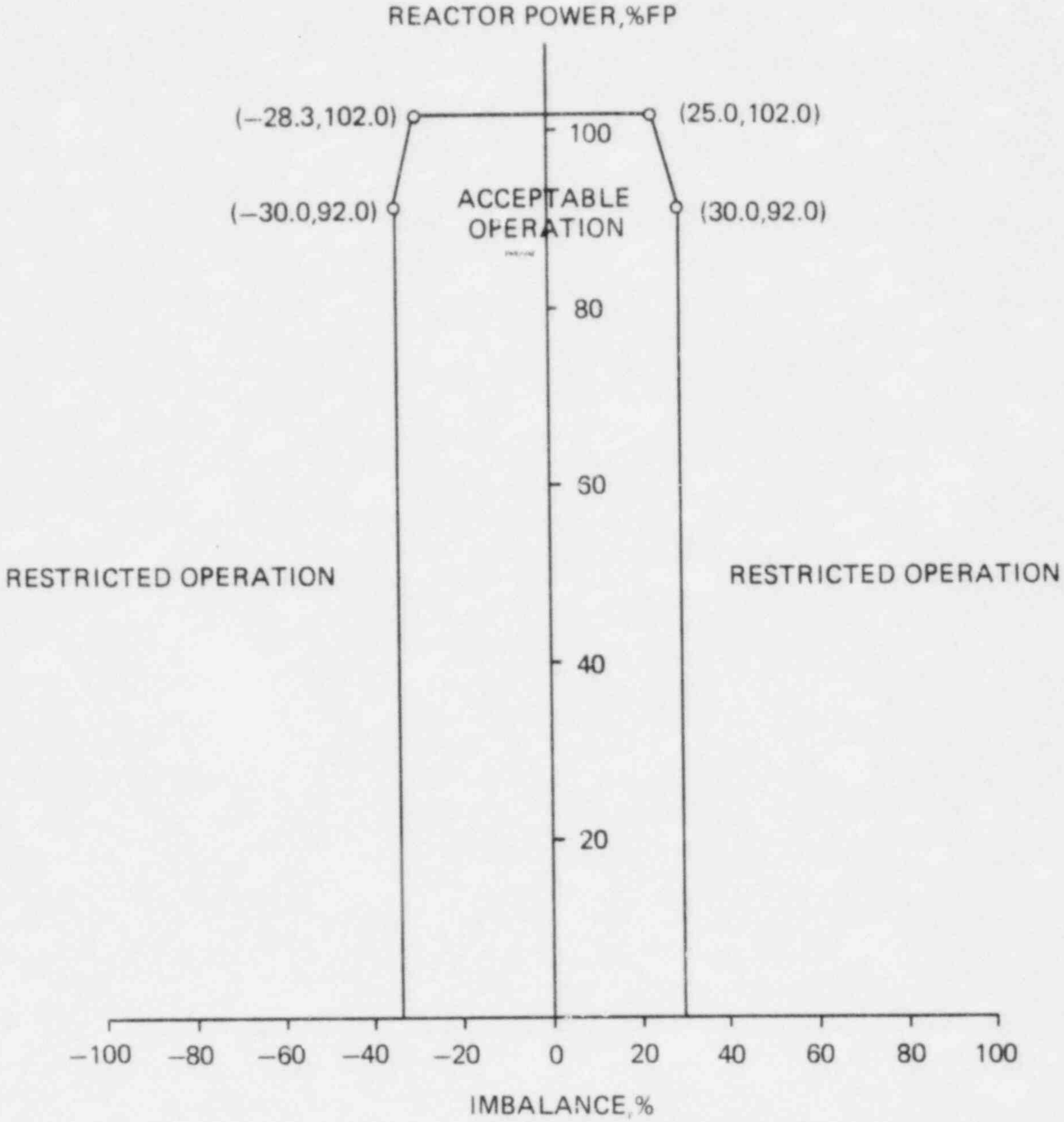


Figure 8-7

Operational Power-Imbalance Limits After 200 ± 10 EFPD

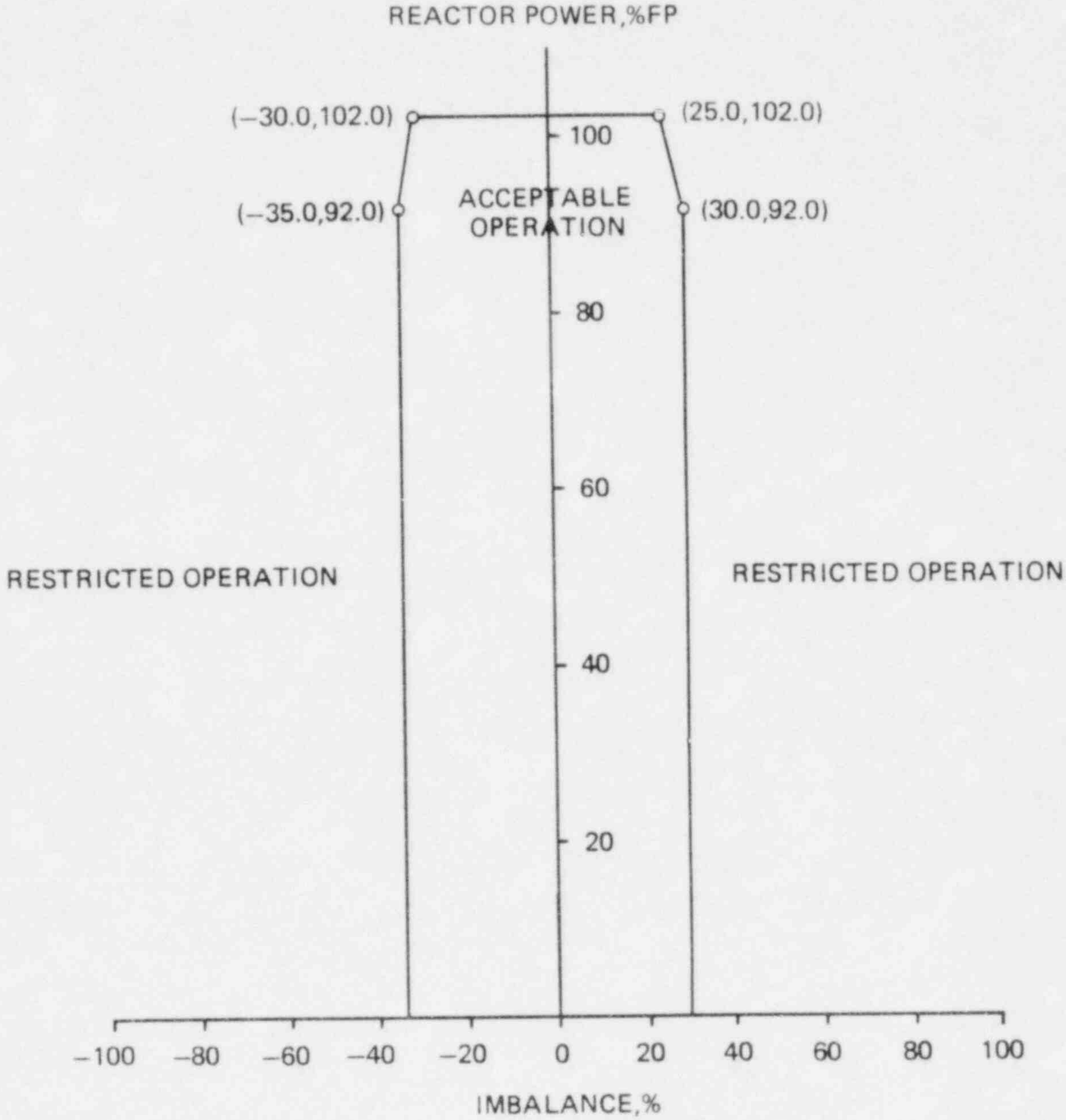


Figure 8-8

Control Rod Position Limits, 4 Pumps, 0-50 $\pm$ 10 EFPD

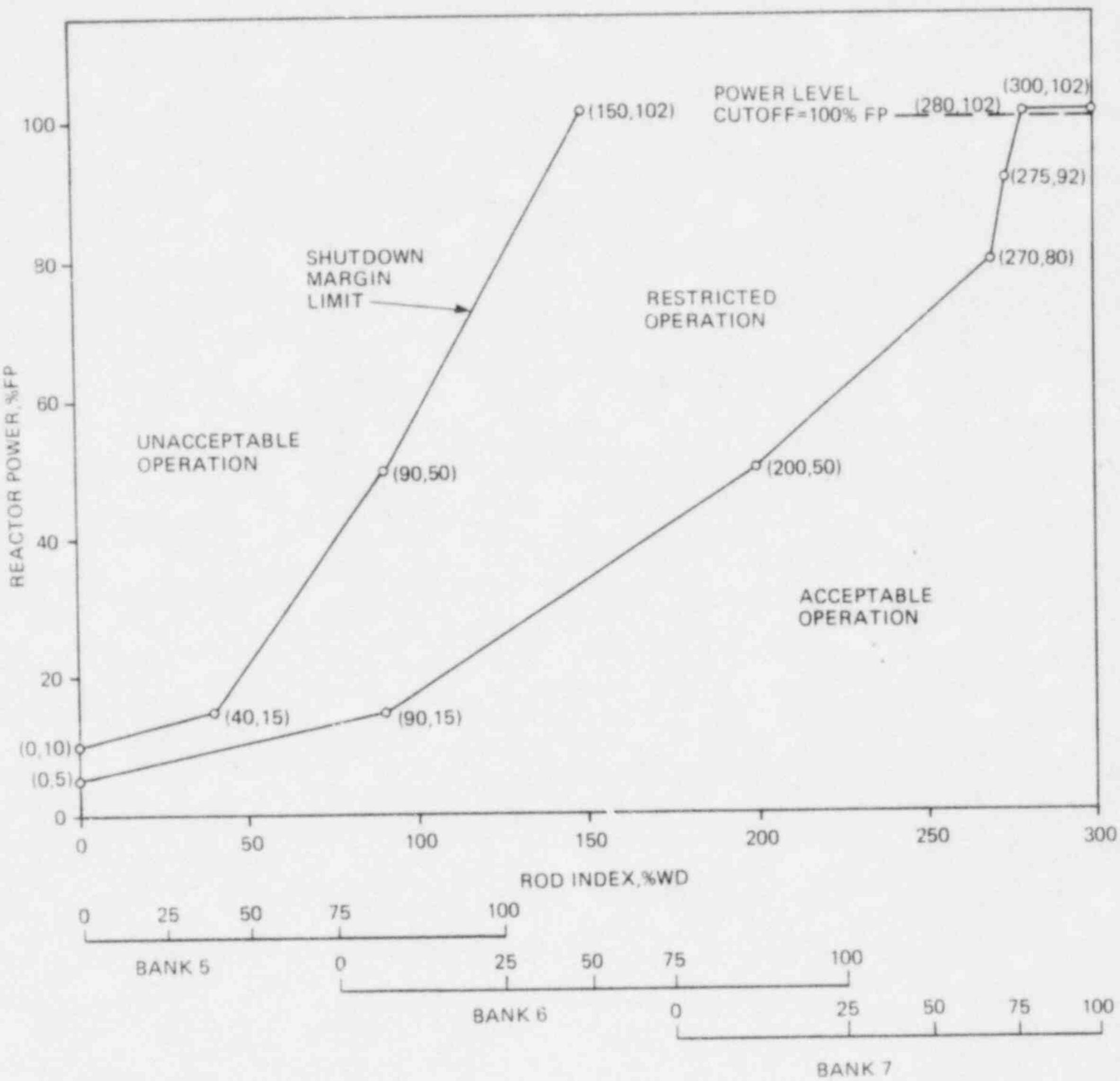


Figure 8-9

Control Rod Position Limits, 4 Pumps, 50 $\pm$ 10 - 200 $\pm$ 10 EFPD

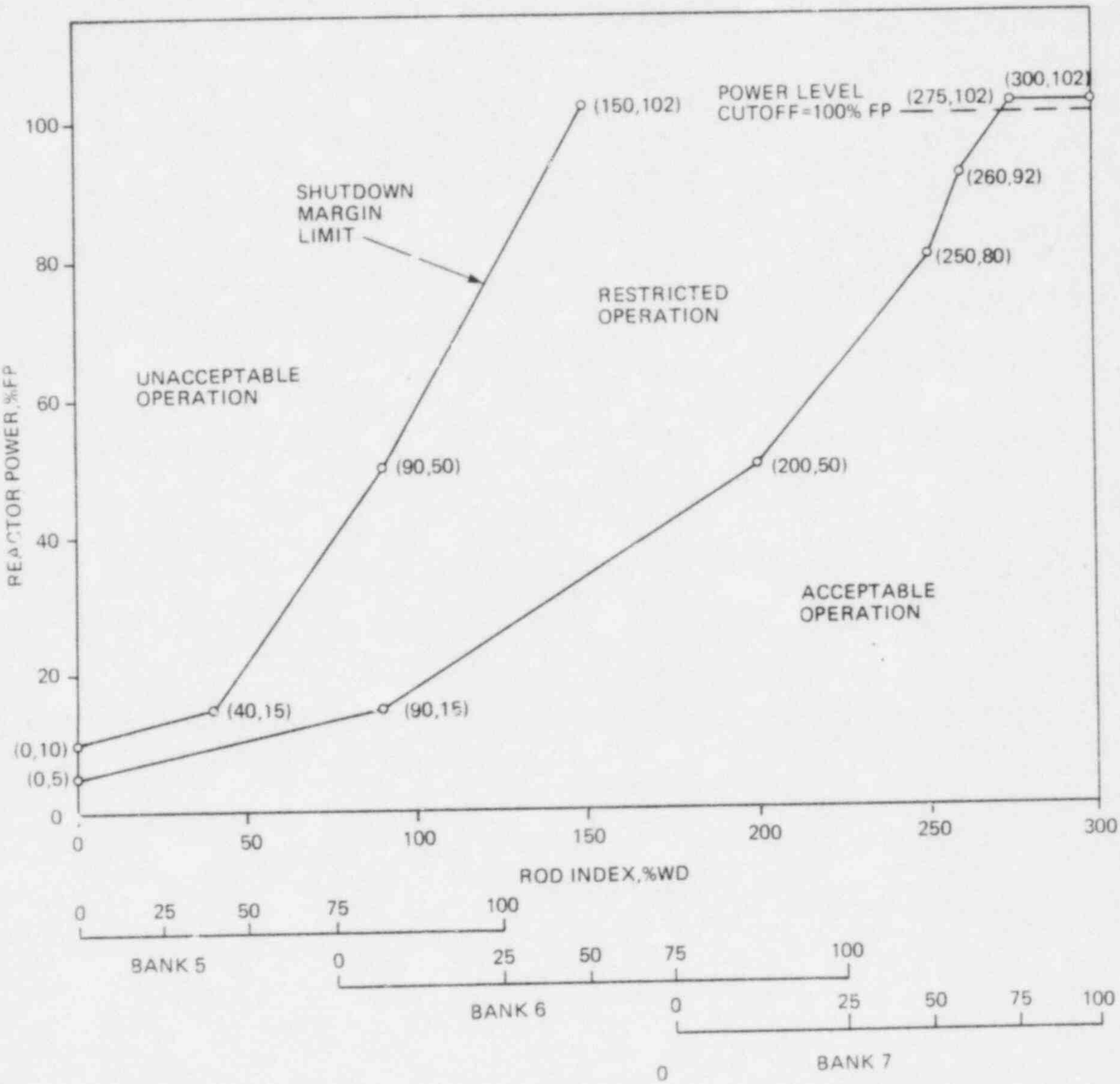


Figure 8-10

Control Rod Position Limits, 4 Pumps, After 200 ± 10 EFPD

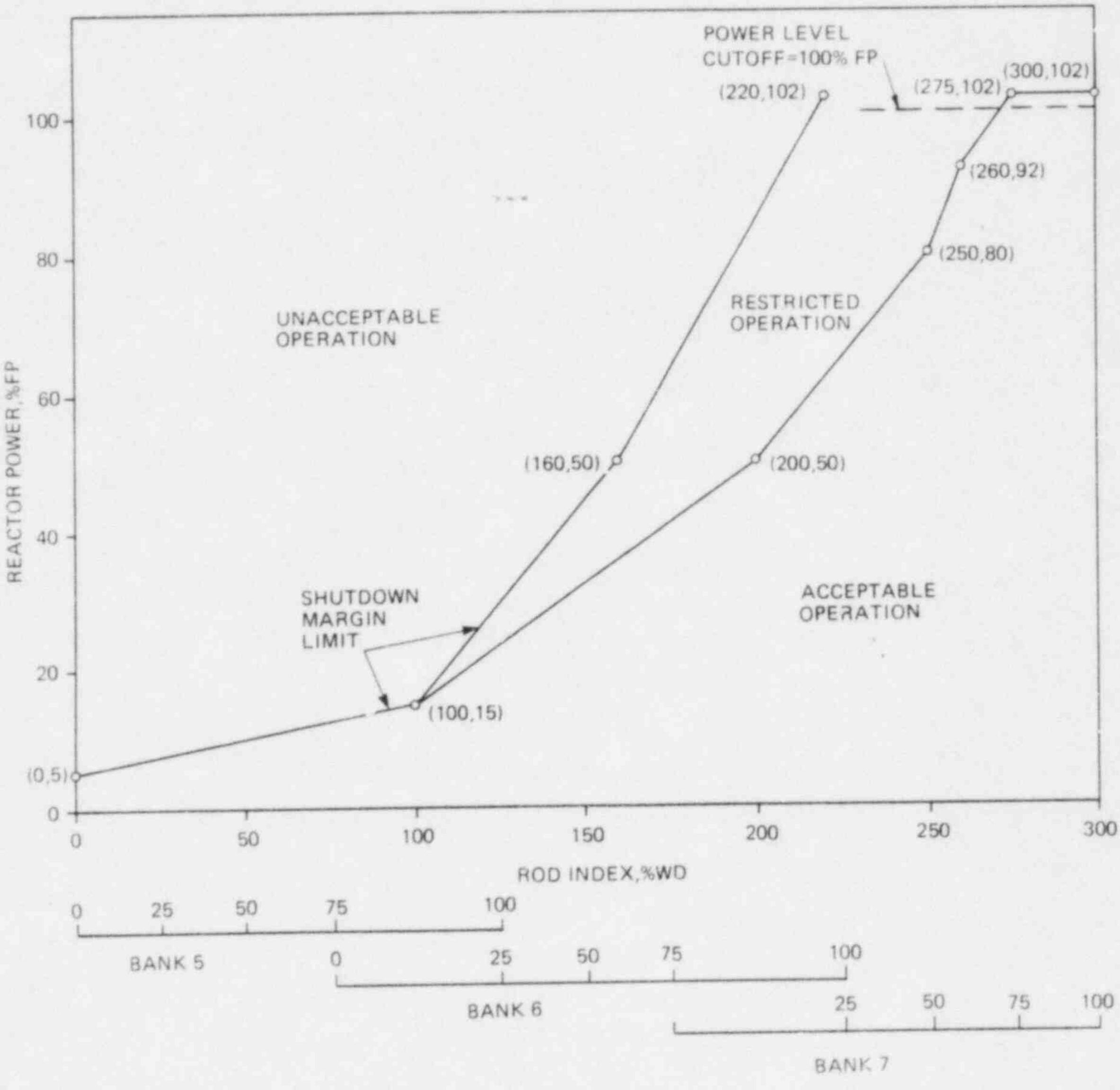


Figure 8-11

Control Rod Position Limits, 3 Pumps, 0-50 $\pm\, {}^{10}_{0}$ EFPD

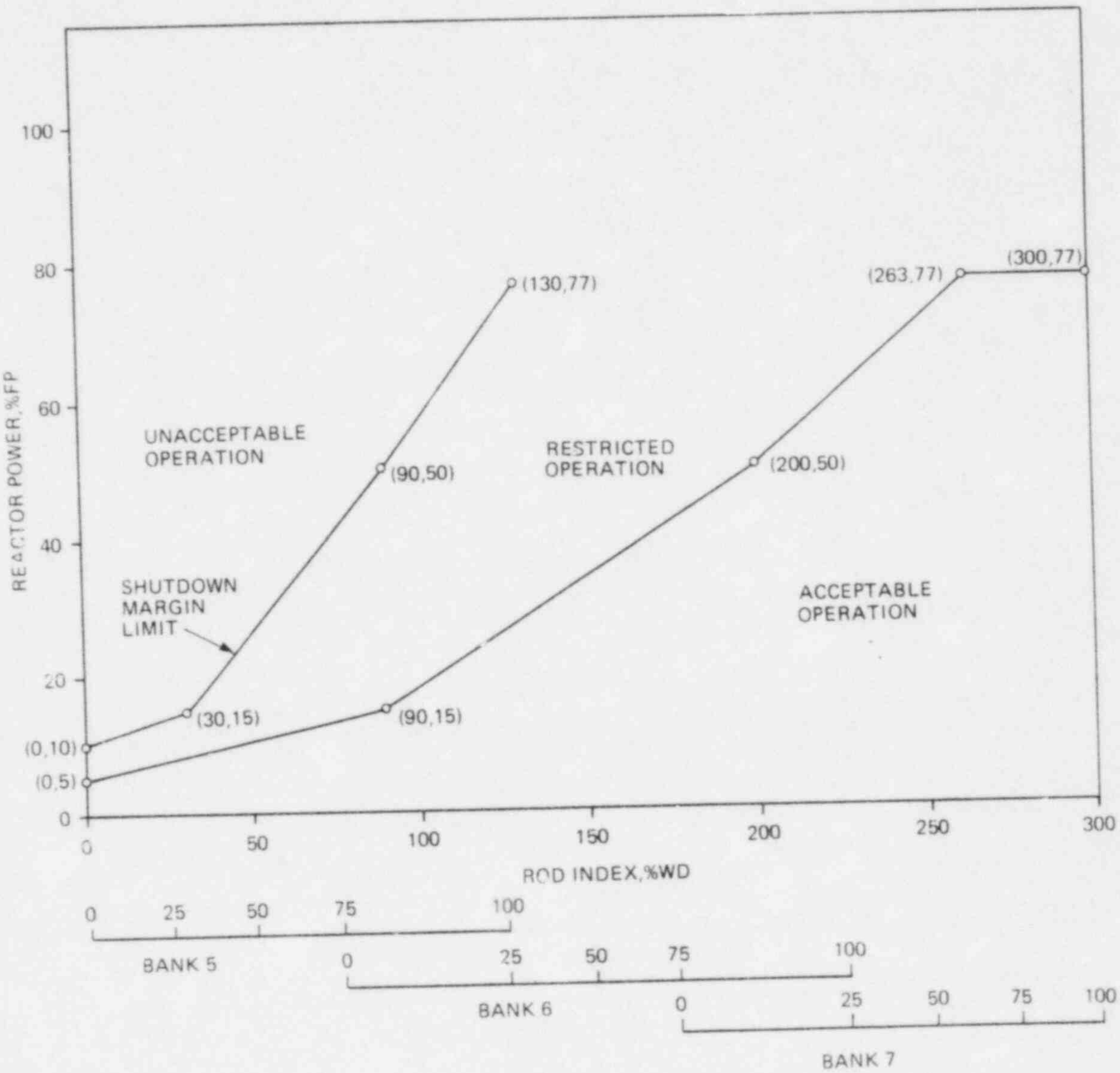


Figure 8-12

Control Rod Position Limits, 3 Pumps, 50 $\pm$ $^{10}_{0}$ - 200 $\pm$ 10 EFPD

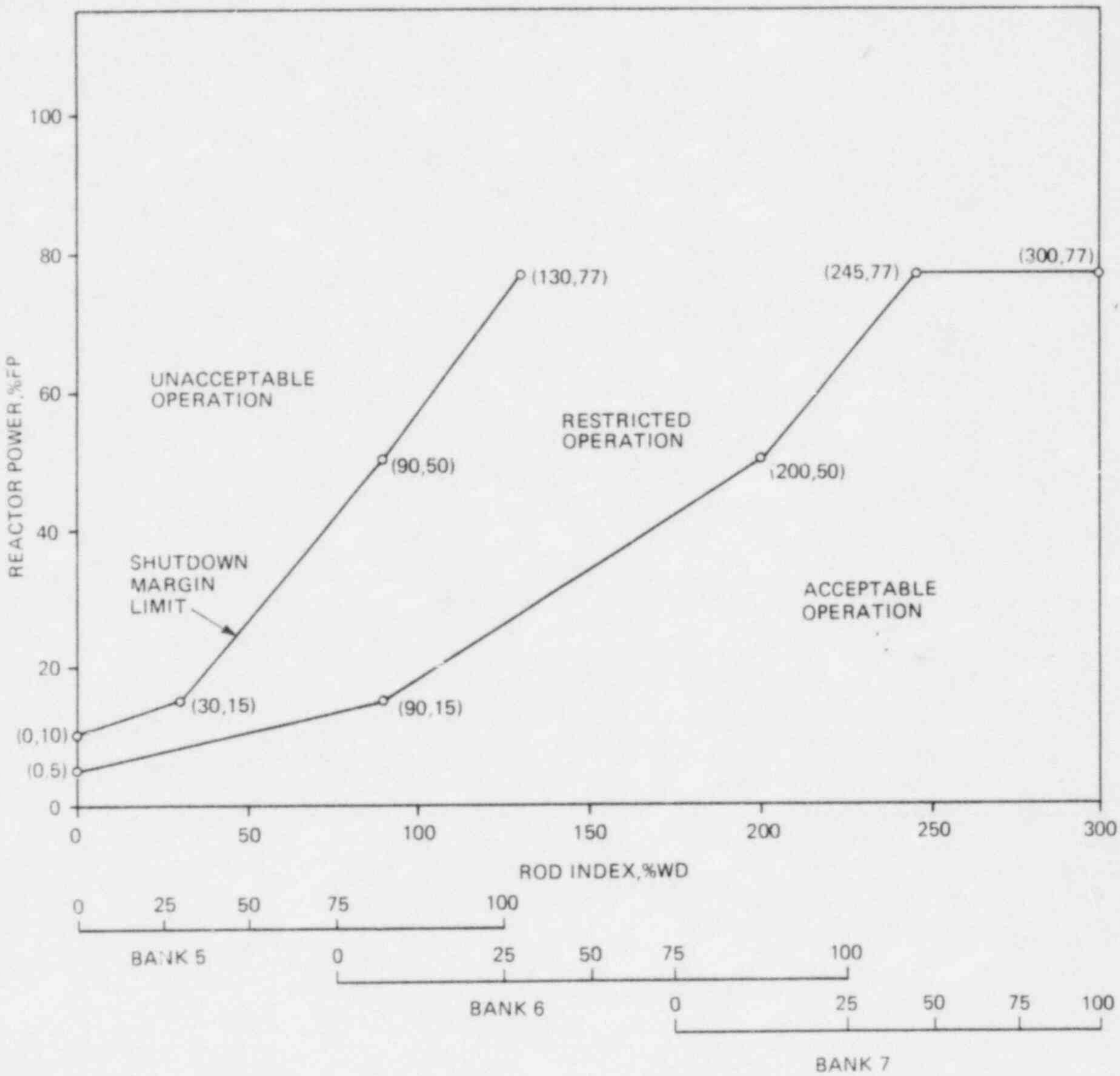


Figure 8-13

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

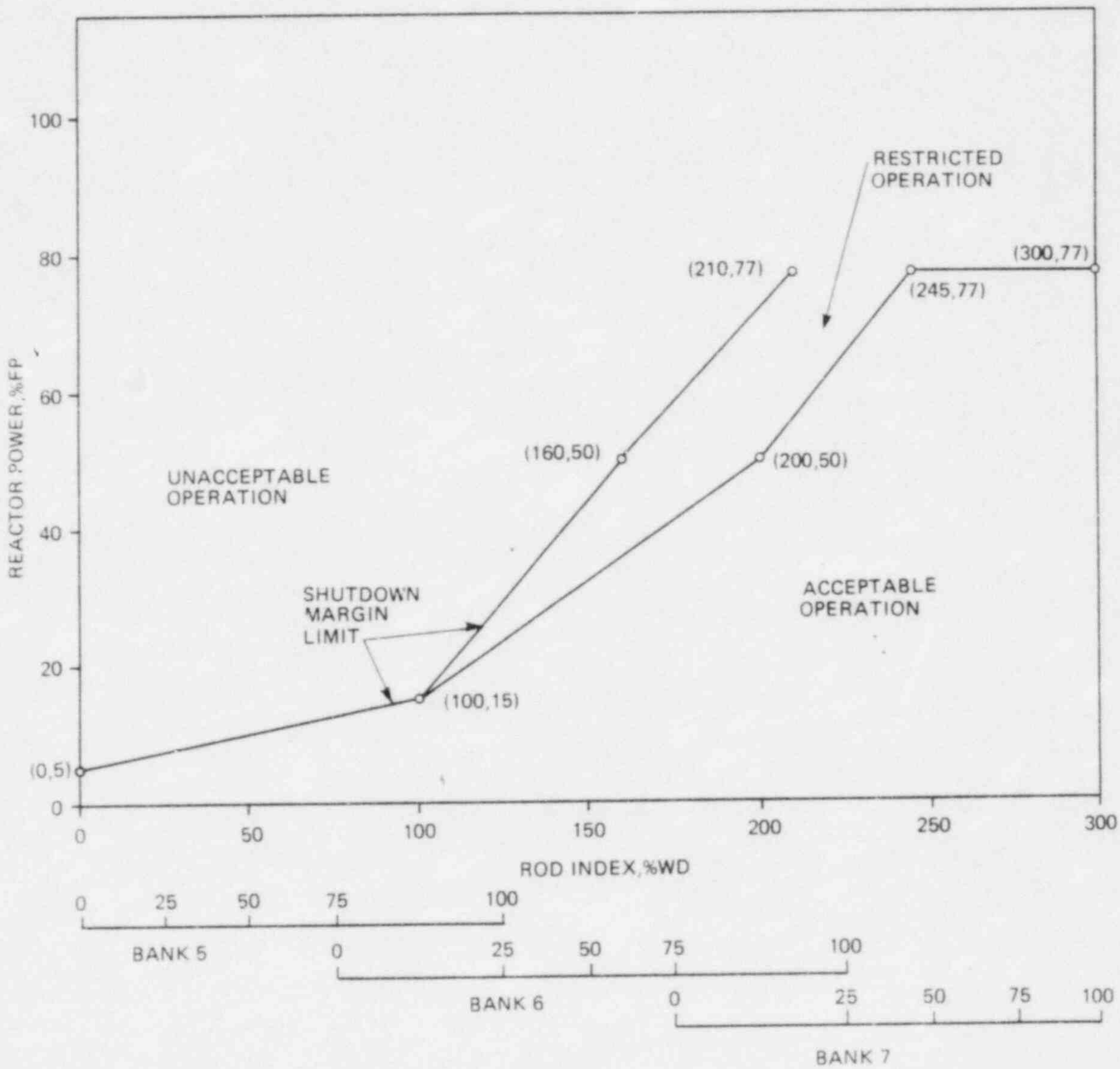


Figure 8-14

Control Rod Position Limits, 2 Pumps, 0-50 $\pm\, ^{10}_{0}$ EFPD

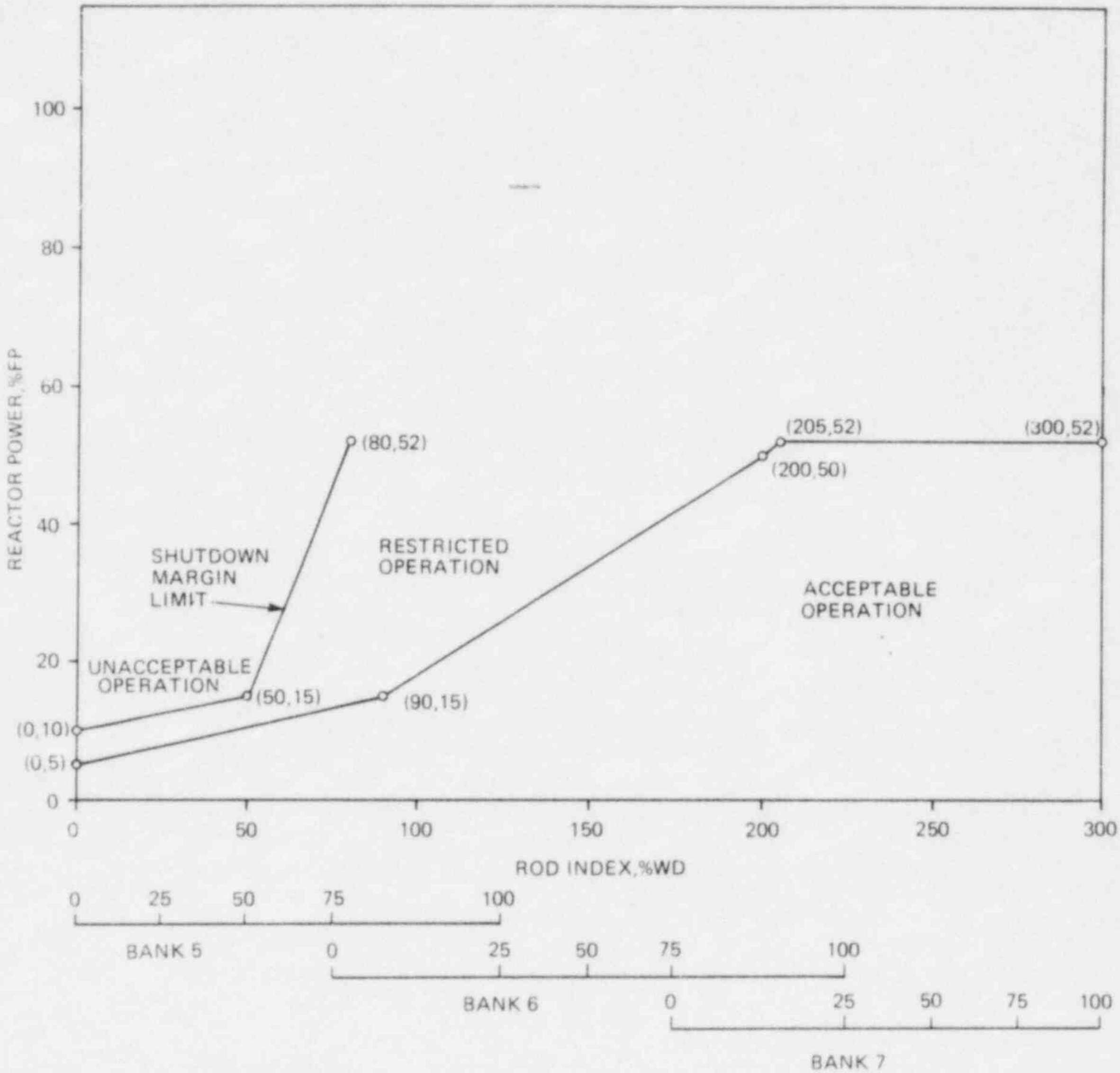


Figure 8-15

Control Rod Position Limits, 2 Pumps, 50 ± 10 - 200 ± 10 EFPD

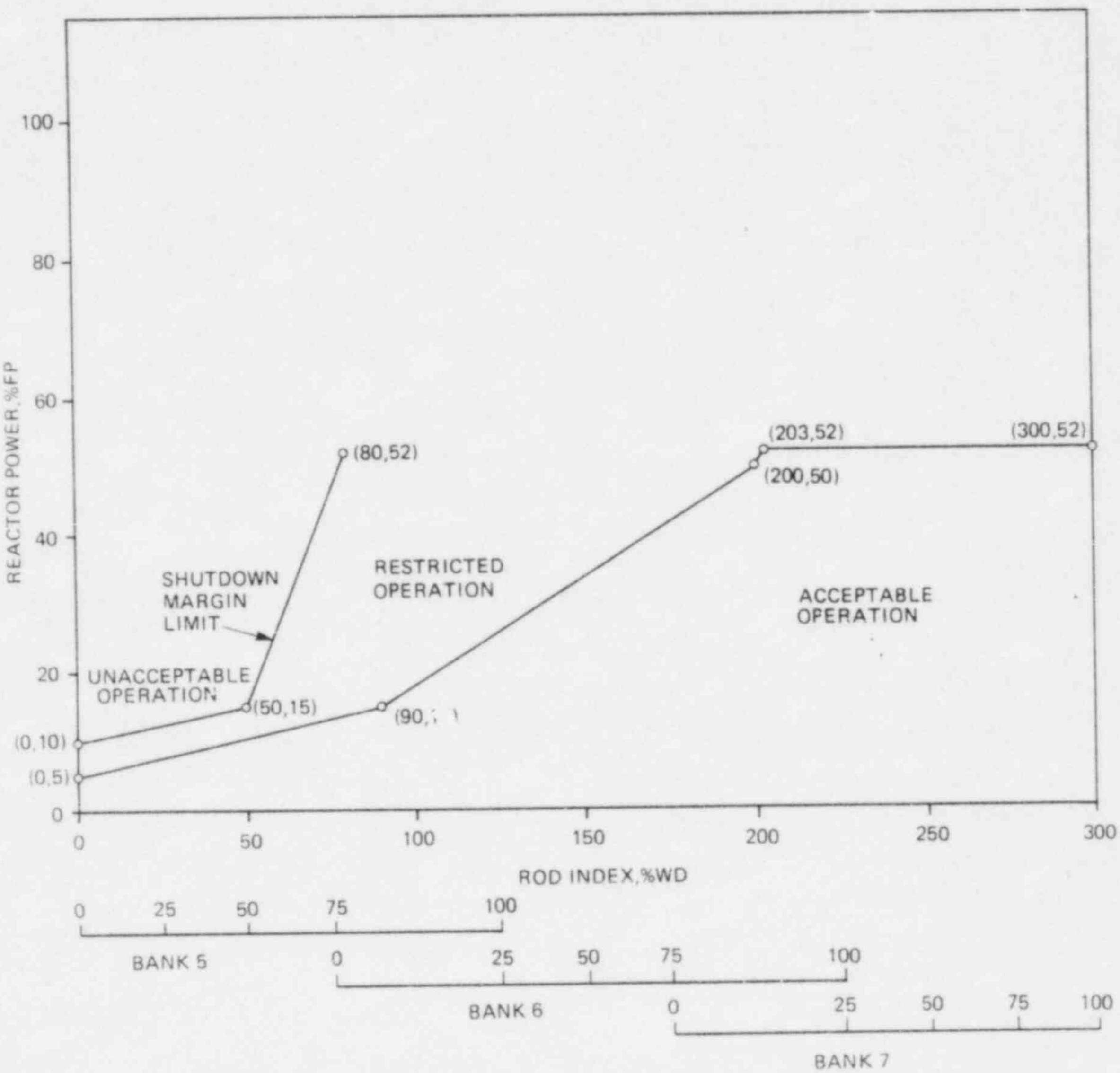


Figure 8-16

Control Rod Position Limits, 2 Pumps, After 200 ± 10 EFPD

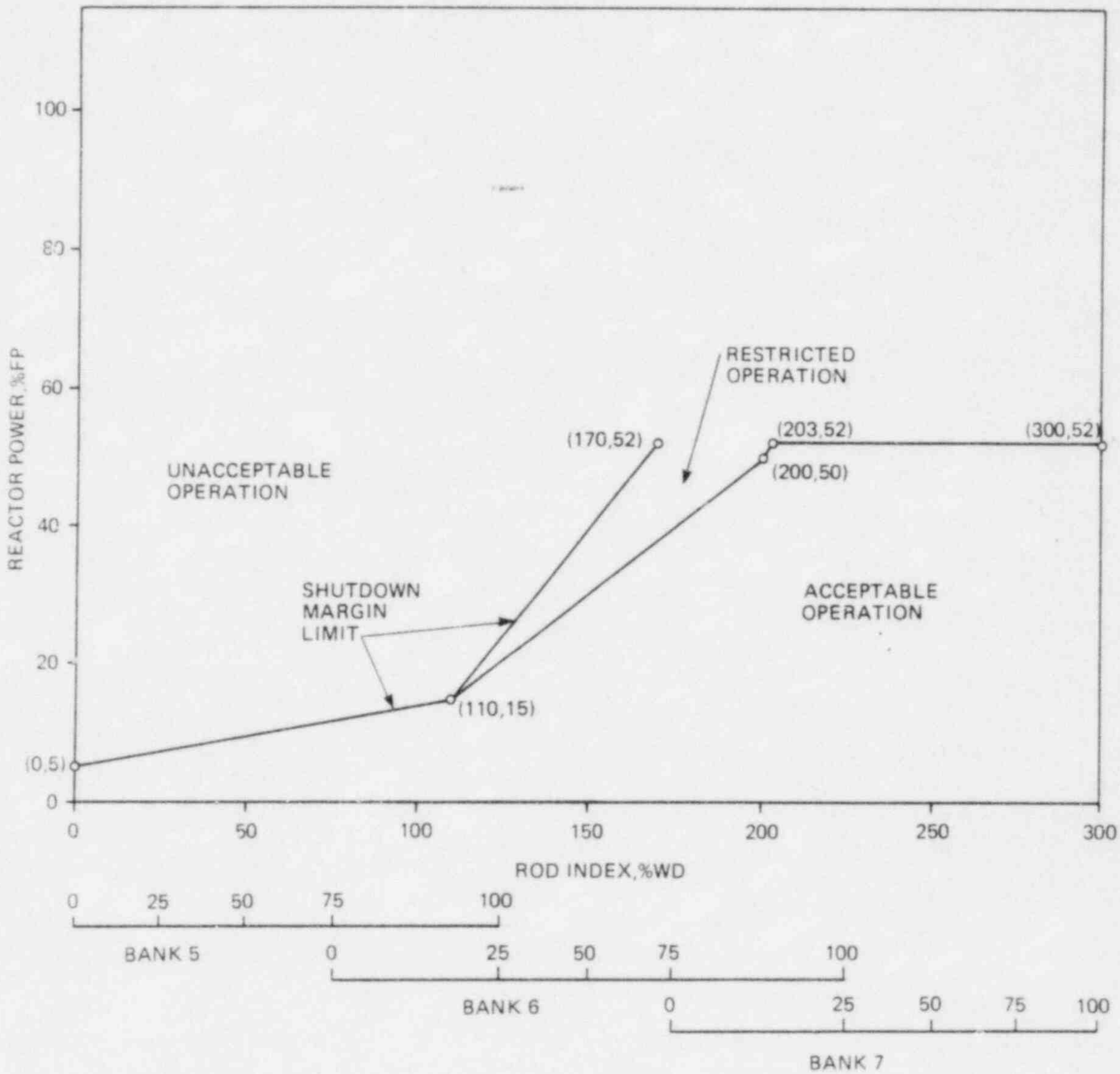


Figure 8-17

APSR Position Limits, 0-200 $\pm$ 10 EFPD

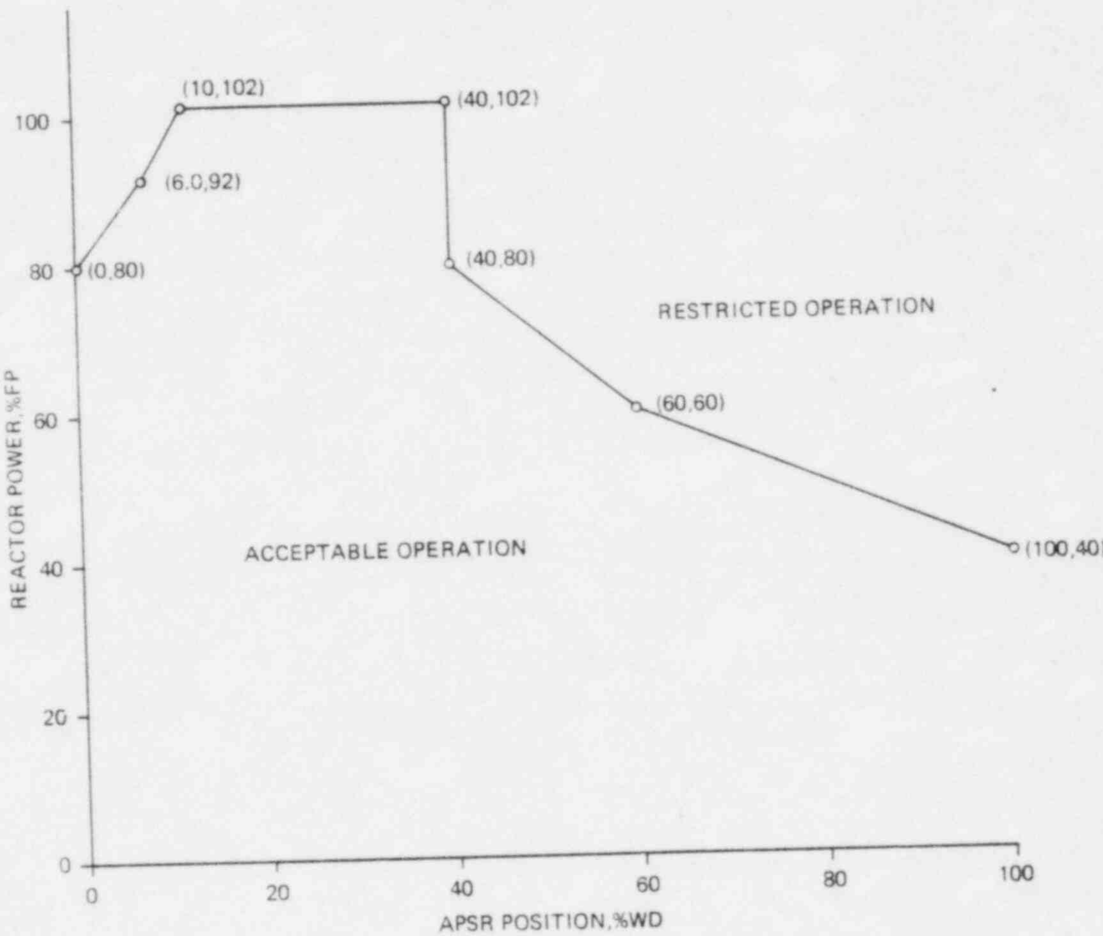
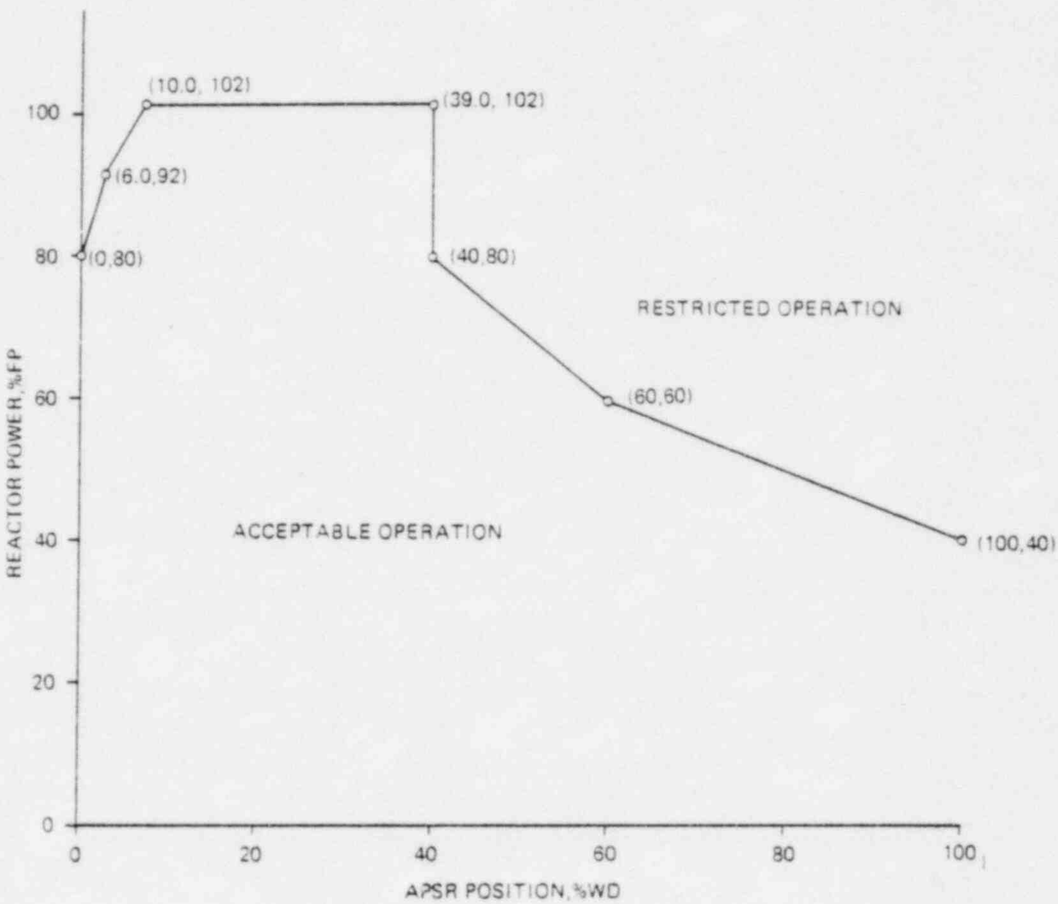


Figure 8-18

APSR Position Limits, After 200 ± 10 EFPD



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