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DESCRIPTION:

Ltr submitting changes to their 9-20-74 request for change to tech specs.......

PLANT NAME: Oconee #1

ENCLOSURES:

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DUKE POWER COMPANY

POWER BUILDING

422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28201

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. Box 2178

October 31, 1974

RECEIVED NOV 8 1974 ATOMIC ENERGY COMMISSION Regulatory Mail Section

Mr. Angelo Giambusso
Deputy Director for Reactor Projects
Directorate of Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Re: Oconee Unit 1
Docket No. 50-269

Dear Mr. Giambusso:

On September 20, 1974, Duke Power Company submitted proposed changes to the Oconee Nuclear Station Technical Specifications and the Babcock & Wilcox Report BAW-1409, "Oconee 1, Cycle 2 Reload Report," to support operation of Oconee 1, Cycle 2 at rated power.

In subsequent conversations with Mr. Leo McDonough and Mr. Larry Chandler of your staff, we have been advised of several questions and comments which were raised during their review of these proposed changes to the Technical Specifications and the supporting B&W report. Please find attached revised pages for the proposed Technical Specifications and BAW-1409 which respond to these questions and comments.

Furthermore, we wish to make these additional comments:

1. Figure 3.5.2-1A1 of proposed Technical Specification 3.5.2 satisfies both the Interim Acceptance Criteria and Appendix K to 10CFR50. This figure shows the control rod group withdrawal limits for four pump operation for the first 250 full power days of operation. Figure 3.5.2-1A2, which shows the withdrawal limits in effect after 250 full power days of operation, satisfies only Appendix K to 10CFR50. In the unlikely event that the B&W ECCS evaluation model is not approved prior to completion of 250 effective full power days of Cycle 2 operation (approximately August, 1975), Technical Specification 3.5.2 will be appropriately modified to meet both the Interim Acceptance Criteria and Appendix K.

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2. Although the boron reactivity worth has decreased from 85 ppm/%Δk/k for Cycle 1 to 97 ppm/%Δk/k for Cycle 2, the Oconee 1 boron injection system continues to meet the applicable Design Criteria for Reactivity Control System Redundancy and Capability (10CFR50, Appendix A).

3. Table 2-3 of BAW-1409 has been clarified by deletion of the hot, full power (HFP) rod worths and by explicitly presenting the worth reduction factors. The HFP worths have been deleted because the shutdown margin is determined at the hot, zero power (HZP) conditions. Thus, only HZP worths should be considered. No changes have been made in the values of any of the parameters shown in the original shutdown margin calculations.

Very truly yours,

A. C. Thies

A. C. Thies

A. C. THIES, being duly sworn, states that he is Senior Vice President of Duke Power Company; that he is authorized on the part of said Compan to sign and file with the Atomic Energy Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

A. C. Thies

A. C. Thies, Senior Vice President

ATTEST:

John C. Goodman, Jr.

John C. Goodman, Jr.
Assistant Secretary

Subscribed and sworn to before me this 31st day of October, 1974.

Edna B. Farmer

Notary Public

My Commission Expires:

October 24, 1977

REPLACEMENT PAGES FOR

BABCOCK AND WILCOX REPORT 1409

The differential boron worths and total xenon worths for Cycle 2 are lower than for Cycle 1 due to depletion of the fuel and the associated buildup of fission products.

2.4 Core Loading - Batch 4 Fuel

The Batch 4 fuel assemblies will be loaded as shown in Figure 2-1. As-built data have been used to ensure eighth core symmetry in U-235 loading. Also, fuel assemblies with the highest U-235 loadings will be placed in locations of low power density in order to minimize power peaking.

As stated in the Nuclear Analysis section of this report, a fuel melt limit of 20.15 kw/ft has been employed in calculating the reactor protection system (RPS) setpoints. This value is the same as that used in the Cycle 1 analysis. Based on the as-built data, all Batch 4 assemblies meet or exceed the 20.15 kw/ft fuel melt criterion with the exception of three assemblies which have been assigned a maximum allowable linear heat rate to fuel melt of 20.02 kw/ft.

The maximum allowable linear heat rate of each fuel assembly in Batch 4 is assigned on the basis of the lowest maximum allowable linear heat rate of any fuel pellet lot used to build that fuel assembly. The maximum allowable heat rate of each fuel pellet lot is determined from the lower tolerance limit on the density and diameter of the fuel in that lot. The maximum allowable heat rate values were obtained from the results of studies conducted by B&W which determined the relationship between density-diameter combinations and maximum allowable heat rate to fuel melt. The three Batch 4 assemblies which were assigned the lower heat rating contain some fuel pellets which are part of a pellet lot which had a pellet density-diameter combination which corresponds to a maximum allowable heat rate of 20.02 kw/ft. Therefore, these three assemblies were assigned a 20.02 kw/ft rating even though they contained other pellets capable of linear heat rates greater than or equal to 20.15 kw/ft.

1. Analyses have been conducted by B&W which demonstrate that the limiting criteria which determines the capability of a core to operate at a specific power level is centerline fuel melt. These analyses were conducted under the AEC guidelines established in "Technical Report on Densification of Light Water Reactor Fuels," November 14, 1972. The sensitivity of DNB, LOCA, centerline fuel melt and the design thermal transients (such as the ejected rod accident), to the maximum linear heat rate, were investigated. The results showed that centerline fuel melt was the limiting criteria. Therefore, this is the criteria which must be considered in selective fuel placement.

The current design tool for nuclear core performance analysis is PDQ07. Both steady-state and transient analyses have been performed on the second cycle of Oconee 1 in 3-D and 2-D representations. From these analyses, maximum total peaks and radial power distributions have been determined as a basis for the calculation of operational limits for the second cycle. By comparing the maximum expected power density in each assembly to that of the hottest assembly, it is possible to assign a maximum expected linear heat rate to each assembly location. Since the assembly or assemblies with the highest power density will operate below the limiting heat rate, the lower power density assemblies will approach the limiting heat rates by no more than the ratio of their maximum respective power densities. Thus, by placing the three 20.02 kw/ft assemblies

in low power density locations, a more than sufficient design margin can be
maintained. The locations chosen for these assemblies (core locations A-10,
1. L-15, and R-6) will experience a maximum linear heat rate of 15.3 kw/ft in
Cycle 2. Cycle 3 has also been investigated and it has been determined that
after the fuel has been shuffled to the Cycle 3 core locations, they will not
experience greater than 19.8 kw/ft through the two cycles. Thus, a sufficient
fuel melt margin will be maintained through that cycle also.

In addition, it should be noted that assembly 1D61 will be placed in core location D-14 in conjunction with B&W's continuing program to evaluate fuel performance. Contained in one fuel rod of assembly 1D61 are three ceramic spacers which simulate fuel densification gaps. The proposal to insert this special assembly into Oconee Unit 1 has been described in a letter (6/18/74) to Angelo Giambusso, USAEC.

Table 2-2. Oconee 1 Cycle 2 Physics Paramters

	Cycle 2	Cycle 1
Cvc e length, EFPD	290	310
Cycle burnup, MWd/mtU	9,000	9,600
Average :ore burnup - EOC, MWd/mtU	14,550	9,600
Initial core loading, mtU	82.6	82.9
Critical boron — BOC, ppm		
HZP - all rods out	1,285	1,476
HZP - banks 7 and 8 inserted	1,159	1,335
HFP - banks 7 and 8 inserted	1,028	1,230
Critical boron — EOC, ppm		
HZP - all rods out	285	405
HFP - bank 8 (37.5% wd, equil Xe)	75	210
Control rod worths — HFP, BOC, %Δk/k		
Banks 1-7 (bank 8, 37.5% wd)	10.30	11.41
Bank 6	1.12	1.17
Bank 7	1.14	1.18
Bank 8 (37.5% wd)	0.36	0.55
Control rod worths — HFP, EOC, %Δk/k		
Banks 1-7	11.20	10.13
Bank 7	1.97	1.24
Bank 8 (37.5% wd)	0.41	0.49
Ejected rod worth — HFP, %Δk/k		
BOC	0.35	0.32
EOC	0.25	0.23
Stuck rod worth — HZP, %Δk/k		
BOC	2.55	2.20
EOC	1.96	1.69
Power deficit, HZP to HFP, %Δk/k		
BOC	-1.34	-1.09
EOC	-1.99	-1.78
Power Doppler coeff - BOC, 10^{-4} (Δk/k-% power)		
100% power (0 Xe)	-0.99	-0.99
95% power (0 Xe)	-1.01	-1.00
75% power (0 Xe)	-1.03	-1.05
40% power (0 Xe)	-1.03	-1.14
Power Doppler coeff - EOC, 10^{-4} (Δk/k-% power)		
95% power (equil Xe)	-1.15	-1.14
Moderator coeff — HFP, 10^{-4} (Δk/k-°F)		
BOC (0 Xe, 1000 ppm)	-0.79	-0.12
EOC (equil Xe, 17 ppm)	-2.35	-2.27
Boron worth — HFP, ppm/%Δk/k		
BOC (1000 ppm)	97.0	84.0
EOC (17 ppm)	91.0	82.0
Xenon worth — HFP, %Δk/k		
BOC (4 days)	2.64	2.77
EOC (equilibrium)	2.69	2.74

Rev.1. 10/28/74

Table 2-3
Shutdown Margin Calculation - Oconee 1, Cycle 2

	BOC, %Δk/k	EOC,** %Δk/k
1. Available Rod Worth		
Total Rod Worth, HZP*	8.67	9.30
Worth Reductions		
Burnup of poison material	-.14	-.36
Most Reactive Stuck Rod Worth	-2.55	-1.96
Net	5.98	6.98
10% Uncertainty	-.60	-.70
a. Total Available Rod Worth	5.38	6.28
2. Required Rod Worth		
Power Deficit, HFP to HZP	1.34	1.99
Inserted Rod Worth, HZP	1.05	1.89
Flux Redistribution	.40	1.00
a. Total Required Worth	2.79	4.88
Shutdown Martin (1a-2a)	2.59	1.40

*HZP denotes hot, zero power; HFP denotes hot, full power

**For shutdown margin calculations, the end of cycle 2 is defined as ~265 EFPD, the time at which the transient control rod group (Group 7) begins to be withdrawn from the core.

Rev. 1. 10/28/74

2. The option in the code for no restructuring of fuel has been used in the analysis presented here in accordance with AEC's interim evaluation of TAFY.[8]

3. The calculated gap conductance is reduced by 25% by the code, also in accordance with AEC's interim evaluation of TAFY.[8]

All fuel lots were inspected for average and LTL density and diameter values. Each lot was then evaluated as to its limiting linear heat rate in accordance with reference 4. As a result, three assemblies will be selectively loaded as described in section 2.4.

3.1.5. Summary

This analysis assumes that densification and associated phenomena will affect the hot channel, which has the most limiting thermal-hydraulic characteristics in the core. In addition, the power spike is assumed to be located at the hot channel position that minimizes the DNBR. The resultant 5.4% DNBR loss, or 3.0% reduction in power peaking margin, will be compensated by changes in the Technical Specifications, so that the plant can function at rated power without violating the initial design criteria for DNBR and/or fuel melting.

Table 3-1 compares thermal-hydraulic operating conditions for cycles 2 and 1.

3.2. Nuclear Analysis

The RPS power/imbalance limits (DNBR and centerline fuel melt protection) and the operational limits (administrative LOCA kW/ft controls) have been established for cycle 2 operation according to the methods and procedures described in BAW-10079[3] and BAW-1388[2]. Following is a summary of cycle 2 design parameters utilized in the analysis:

Parameter	Cycle 2 value
Fuel melt limit, kW/ft	20.15
DNB peaking margin penalty due to densification, %	-3.0
Overpower, % of 2568 MWt	112
Densified nominal heat rate at 100% power, kW/ft	5.80
Power spike factor	Figure 3-2
Nuclear power peaking uncertainty	1.075
LOCA limit, kW/ft	Figure 3-8

The maximum power peaks resulting from Cycle 2 operation occur in fresh (Batch
4) fuel assemblies throughout the cycle. The largest total power peak in the
once-burned fuel is always at least 20 percent lower than the power peak in
1. the fresh fuel. Thus, the effect on power peaking of burnup gradients across
the once-burned assemblies is insignificant with regard to operational limits.
The plant can operate at rated power without exceeding DNBR, fuel melt, and
ECCS criteria by adhering to the limits specified in figures 3-3 through 3-7.

3.3 Safety Analysis

3.3.1 General Safety Analysis

The safety analysis presented in the Oconee FSAR covered a range of physics
parameters for BOL and EOL situations. The spectrum of accidents analyzed in
the FSAR were considered using the Cycle 2 physics parameters. The Cycle 2
physics parameters that affect the safety analysis are bounded by those
1. utilized in the FSAR. Hence, the limiting transients are the same as those
established in BAW-1382[2] where the significant effects of fuel densification
were identified, and the effects on the safety analysis reported. It was
established in BAW-1388[2] that the limiting transients were the rod ejection
and loss of coolant flow.

Table 2-2 shows that the ejected rod worth for Cycle 2 (0.35%) will be much less than the rod worth used in BAW-1388 (0.50%). In addition, the moderator and Doppler coefficients of reactivity are more favorable than those used in the previous analysis. Therefore, it can be concluded that the rod ejection accident will result in conditions no more severe than previously reported.

The loss-of-coolant-flow type accidents will be less severe than previously reported since the initial DNBR will be higher. As shown in Table 3-1, the initial DNBR at the overpower of 114 percent of rated power for Cycle 2, Batch 4, is much higher using the measured flow of 107.6 percent and the BAW-2 correlation.[5,6] Thus, the transient results for Cycle 2 fuel will be less severe than or equal to the results reported previously.

The peaking values are consistent with the discussion presented in Section 3 of BAW-1388.[2]

3.3.2 LOCA Analysis

A generic LOCA analysis for B&W 177-fuel assembly nuclear steam systems with lowered steam generators has been performed using the Final Acceptance Criteria ECCS Evaluation Model and is reported in BAW-10091.[7] That analysis is generic in nature since the limiting values of key parameters for all plants in this category were used. Thus, the analysis provides conservative results for operation of Oconee 1.

Figure 3-1. Maximum Gap Size Vs Axial Position

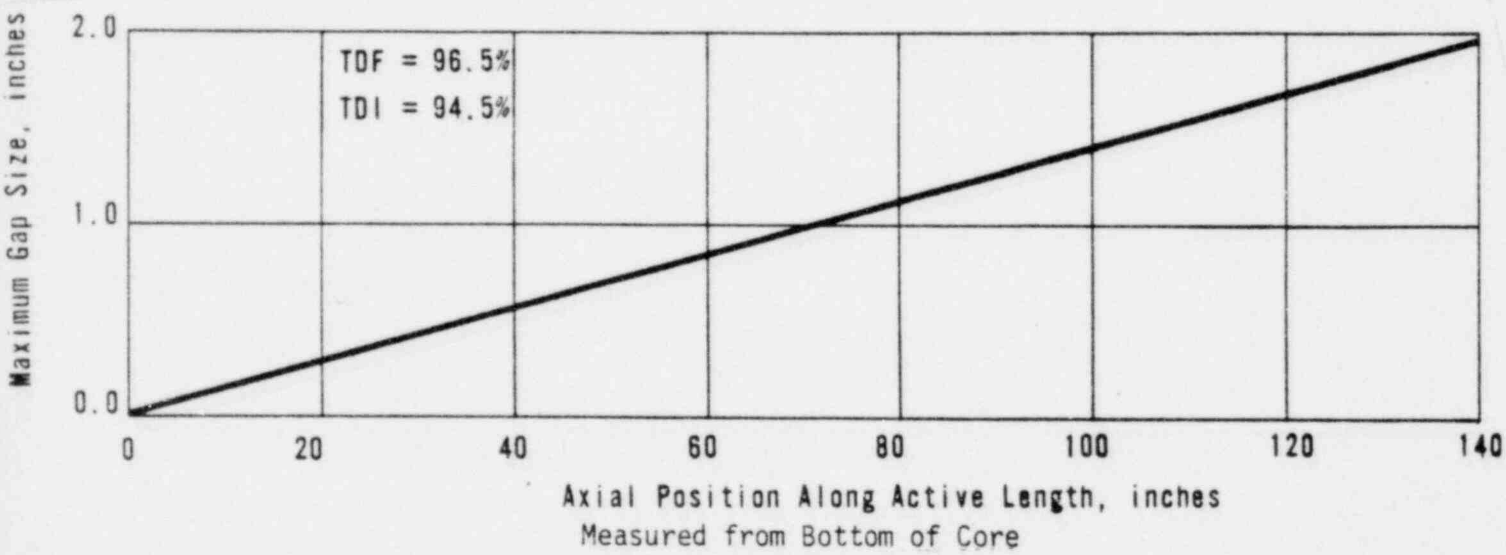

Figure 3-2. Power Spike Factor Vs Axial Position

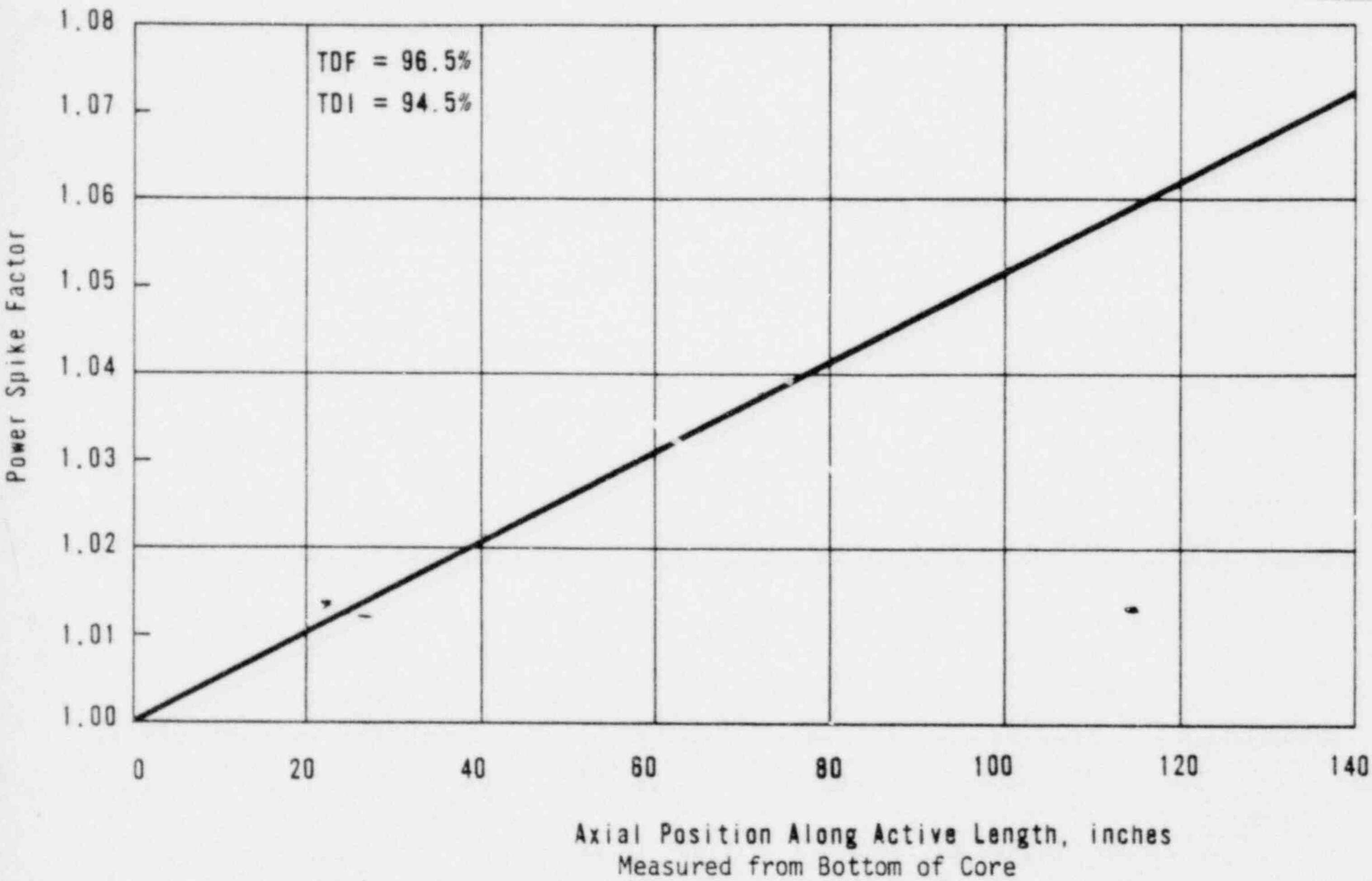

Figure 3-5 Control Rod Group Withdrawal Limits
for Four Pump Operation

1. Rod index is the percentage sum of the withdrawal of the operating groups.

2. The withdrawal limits are modified after 250 ± 5 full power days of operation.

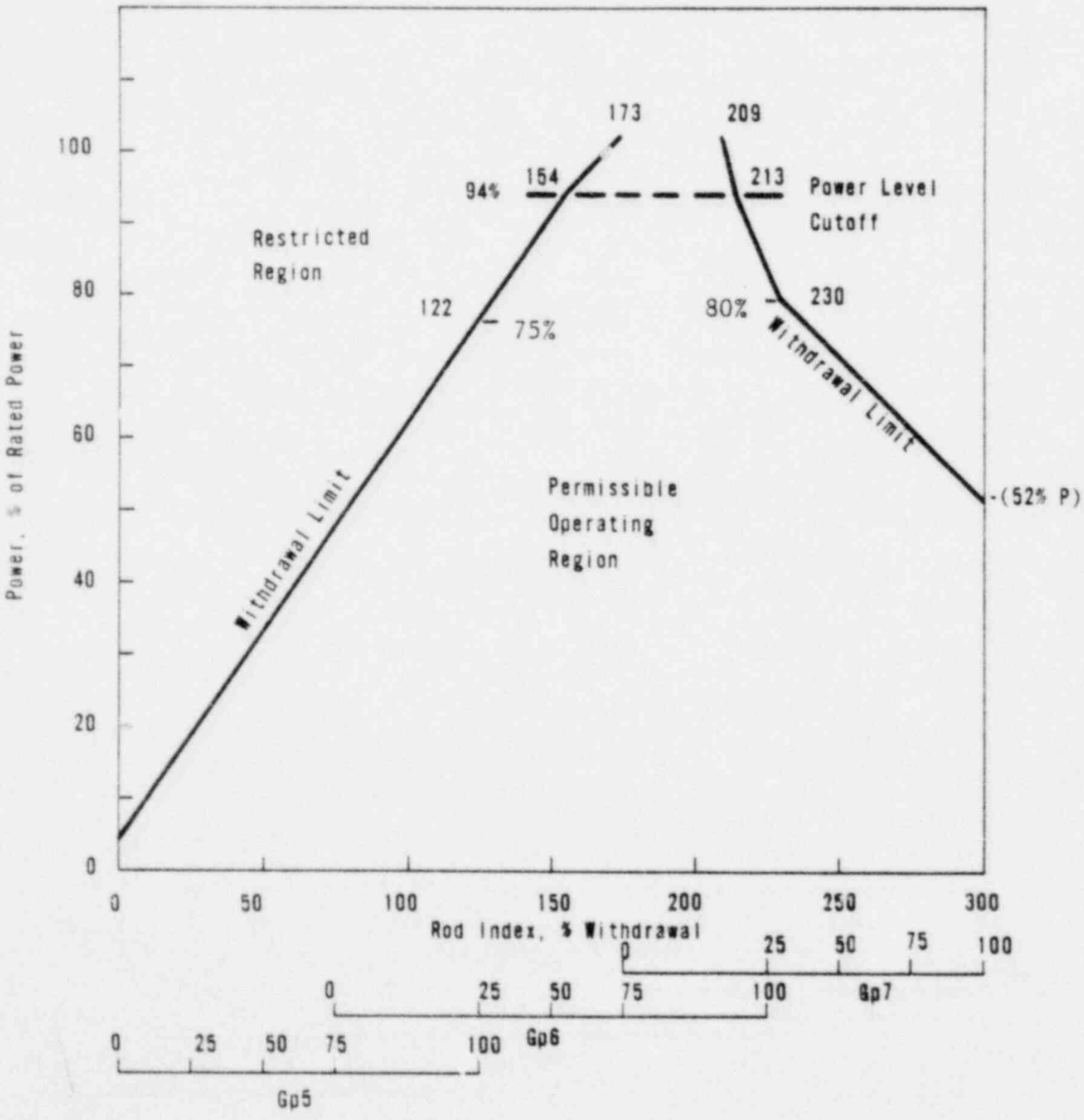

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Figure 3-6 Control Rod Group Withdrawal Limits for Four Pump Operation

1. Rod index is the percentage sum of the withdrawal of the operating groups.
2. The withdrawal limits are in effect after 250 ± 5 full power days of operation. (The applicable power level cutoff is 100% power)

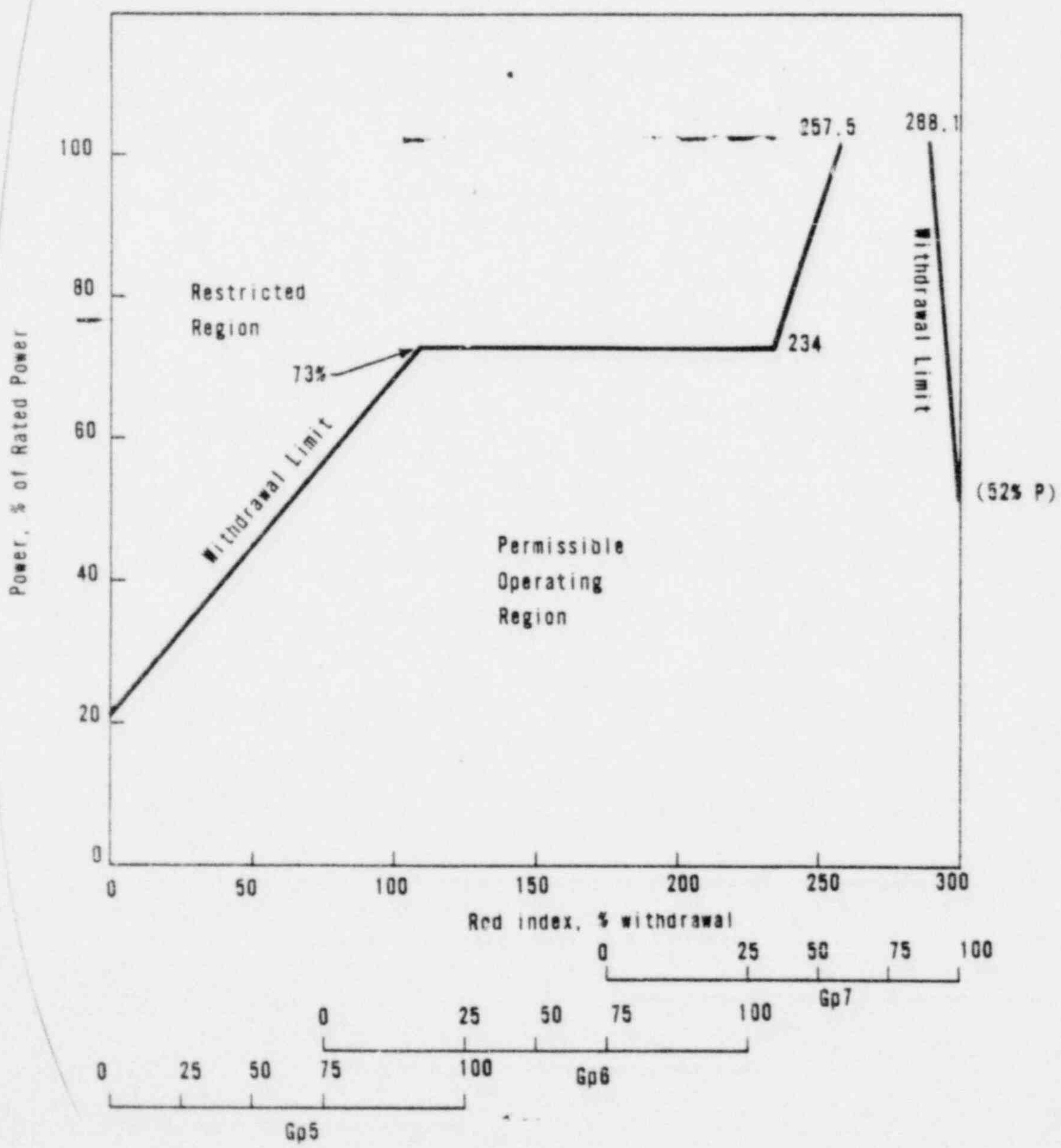
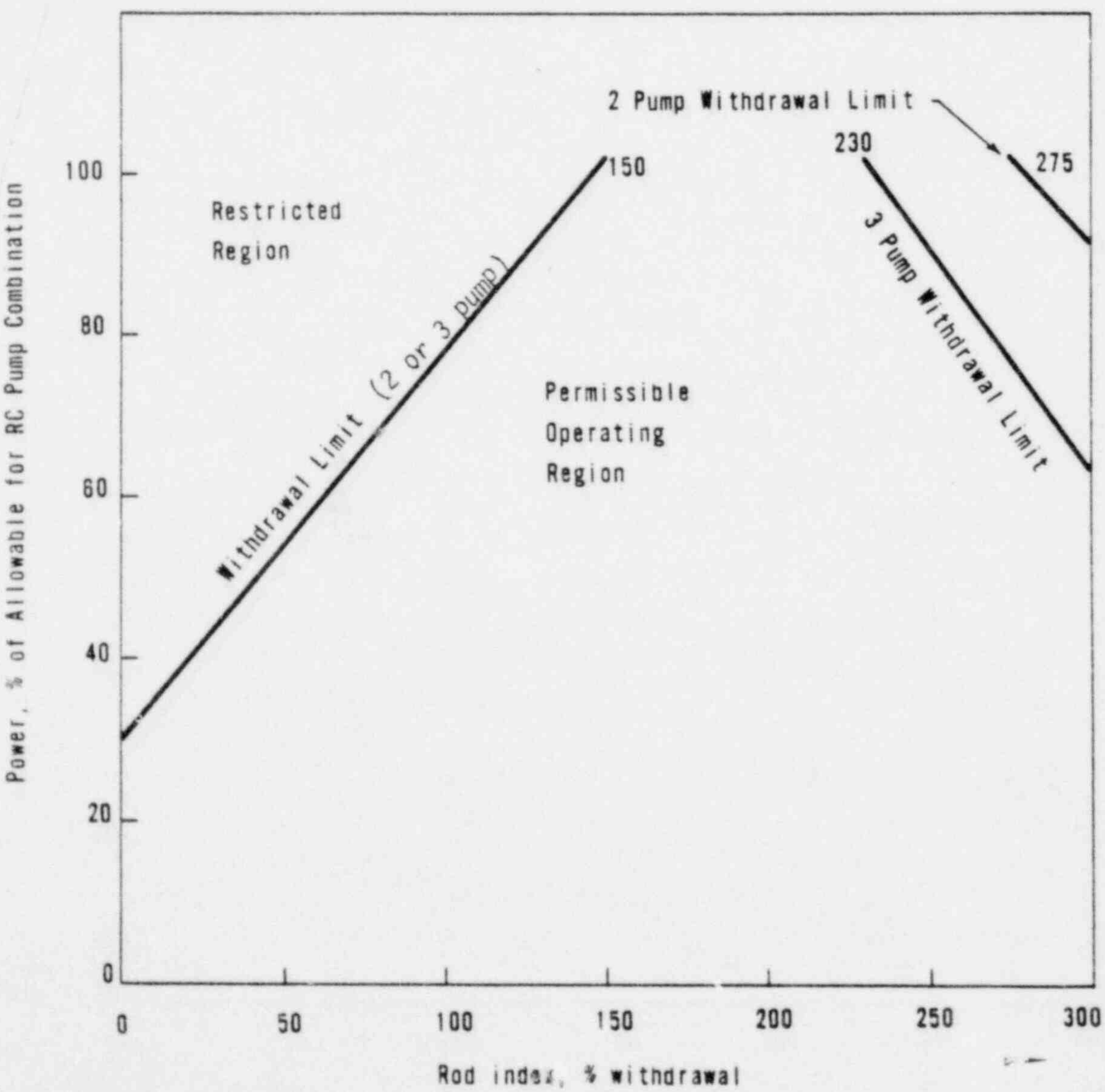


Figure 3-7. Control Rod Group Withdrawal Limits for Three- and Two-Pump Operation

1. Rod index is the percentage sum of the withdrawal of the operating groups. (The applicable power level cutoff is 100% power)



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[1] Fuel Densification Report, BAW-10055, Rev. 1, Babcock & Wilcox, July 1973.

[2] Oconee 1 Fuel Densification Report, BAW-1388, Rev. 1, Babcock & Wilcox, July 1973.

[3] Operational Parameters for B&W Rodded Plants, BAW-10079, Babcock & Wilcox, October 1973.

1.

[5] Oconee 1 Startup Report, Duke Power Co., November 16, 1973.

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[7] "Densification Kinetics and Power Spike Model," Meeting With USAEC, July 3, 1974; J. F. Harrison (B&W) to R. Lobel (USAEC), Telecon, "Power Spike Factor," July 18, 1974.

[8] C. D. Morgan, and H. S. Kao, TAFY — Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock & Wilcox, May 1972.

[9] B. M. Dunn, et al., B&W's ECCS Evaluation Model Report With Specific Application to 177-Fuel Assembly Plants With Lowered-Loop Arrangement, BAW-10091, Babcock & Wilcox, August 1974.

[10] A. F. J. Eckert, H. W. Wilson, and K. E. Yoon, Program to Determine In-Reactor Performance of B&W Fuels — Cladding Creep Collapse, BAW-10084, Babcock & Wilcox, May 1974.