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OCONEE UNIT 1, CYCLE 5

— Reload Report —

Babcock & Wilcox

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1. INTRODUCTION

This report justifies operation of the Oconee Nuclear Station, Unit 1, cycle 5 at a rated core power of 2568 MWt. The required analyses are included as outlined in the USNRC document, "Guidance for Proposed License Amendments Relating to Refueling," June 1975. This report uses the analytical techniques and design bases documented in several reports that have been submitted to and approved by the USNRC.

Cycle 5 reactor and fuel parameters related to power capability are summarized in this report and compared to those of cycle 4. All accidents analyzed in the Oconee FSAR have been reviewed for cycle 5 operation; a detailed comparison of cycle 5 characteristics to the FSAR analyses showed that no new analyses were necessary since cycle 5 parameters are conservative.

The Technical Specifications have been reviewed and modified where required for cycle 5 operation. Based on the analyses performed and taking into account the ECCS Final Acceptance Criteria and postulated fuel densification effects, it is concluded that Oconee 1, cycle 5 can be safely operated at its licensed core power level of 2568 MWt.

Five fuel assemblies from batch 4 will be irradiated for a fourth cycle as part of a joint Duke Power/B&W/Dept. of Energy program to demonstrate reliable fuel performance at extended burnups and to obtain post-irradiation data. These assemblies will not adversely affect cycle 5 operation.

2. OPERATING HISTORY

The reference cycle for the nuclear and thermal-hydraulic analyses of Oconee 1, cycle 5 is the currently operating cycle 4. This cycle 5 design is based on a planned cycle 4 length of 235 EFPD rather than the design length of 292 EFPD.

Cycle 5 will operate in a feed-and-bleed mode for its entire design length of 330 EFPD. Initial cycle 4 operation was in a rodded mode. However, a quadrant power tilt was detected during cycle 4 power escalation[1], and the mode of operation was converted to feed-and-bleed to provide a larger margin for cycle 4 operation.[2] The shuffle pattern for cycle 5 was designed to minimize the effects of any power tilts present in cycle 4. No control rod interchange is planned during cycle 5.

3. GENERAL DESCRIPTION

The Oconee Unit 1 reactor core and fuel design basis are described in detail in section 3 of the Final Safety Analysis Report[3] for Oconee Nuclear Station, Unit 1. The cycle 5 core contains 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one in-core 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 of uranium. The undensified nominal active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in Tables 4-1 and 4-2.

Figure 3-1 is the core loading diagram for Oconee 1, cycle 5. The initial enrichment of the fresh batch 7 fuel is 3.02 wt % ^{235}U. The remaining batches 4D, 5, and 6 were initially enriched to 3.20, 2.75, and 2.795 wt % ^{235}U, respectively. All the batch 4A and all but five batch 4B assemblies will be discharged at the end of cycle 4. The five remaining batch 4B assemblies will be retained in cycle 5 and are redesignated as batch 4D. The batch 4D, 5, and 6 assemblies will be shuffled to new locations at the beginning of cycle 5. The fresh batch 7 assemblies will occupy the periphery of the core and eight interior locations. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of cycle 5.

Reactivity is controlled by 61 full-length Ag-In-Cd control rods and by soluble boron shim. In addition to the full-length control rods, eight axial power shaping rods are provided for additional control of the axial power distribution. The cycle 5 locations of the 69 control rods and the group designations are indicated in Figure 3-3. The core locations of the total pattern (69 control rods) for cycle 5 are identical to those of the reference cycle indicated in the Oconee 1, cycle 4 reload report.[4] The group designations, however, differ between cycle 5 and the reference cycle in order to minimize power peaking. Neither control rod interchange nor burnable poison rods are necessary for cycle 5.

Figure 3-1. Oconee 1, Cycle 5 — Full Core Loading Diagram

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						7	7	7	7	7					
B				7	7	7	E7 5	R8 6	E9 5	7	7	7			
C			7	M2 6	C6 5	F7 5	L2 6	M5 5	L14 6	F9 5	C10 5	M14 6	7		
D		7	B11 6	O13 6	L1 6	N3 6	D5 5	7	D11 5	N13 6	L15 6	O3 6	B5 6	7	
E		7	F3 5	A10 6	O9 5	K1 6	N2 6	O5* 4D	N14 6	N15 6	K3 5	A6 6	F13 5	7	
F	7	7	G6 5	C12 6	A9 6	K13 5	D6 5	N8 5	D10 5	O7 5	A7 6	C4 6	G10 5	7	7
G	7	G5 5	B10 6	E4 5	B12 6	F4 5	7	K8 5	7	F12 5	B4 6	E12 5	B6 6	G11 5	7
H	7	H15 6	M11 5	7	M13* 4D	H12 5	H9 5	K14* 4D	B7 5	H4 5	E3* 4D	7	E5 5	H1 6	7
K	7	K5 5	P10 6	M4 5	P12 6	L4 5	7	G8 5	7	L12 5	P4 6	M12 5	P6 6	K11 5	7
L	7	7	K6 5	C12 6	R9 6	C9 5	N6 5	D8 5	N10 5	G3 5	R7 6	O4 6	K10 5	7	7
M		7	L5 5	R10 6	O13 5	C1 6	D2 6	C11* 4D	D14 6	O15 6	C7 5	R6 6	L13 5	7	
N		7	P11 6	C13 6	F1 6	D3 6	N5 5	7	N11 5	D13 6	F15 6	C3 6	P5 6	7	
O			7	E2 6	O6 5	L7 5	F2 6	E11 5	F14 6	L9 5	O10 5	E14 6	7		
P				7	7	7	M7 5	A8 6	M9 5	7	7	7			
R						7	7	7	7	7					

*Location of thrice-burned batch 4D assemblies.

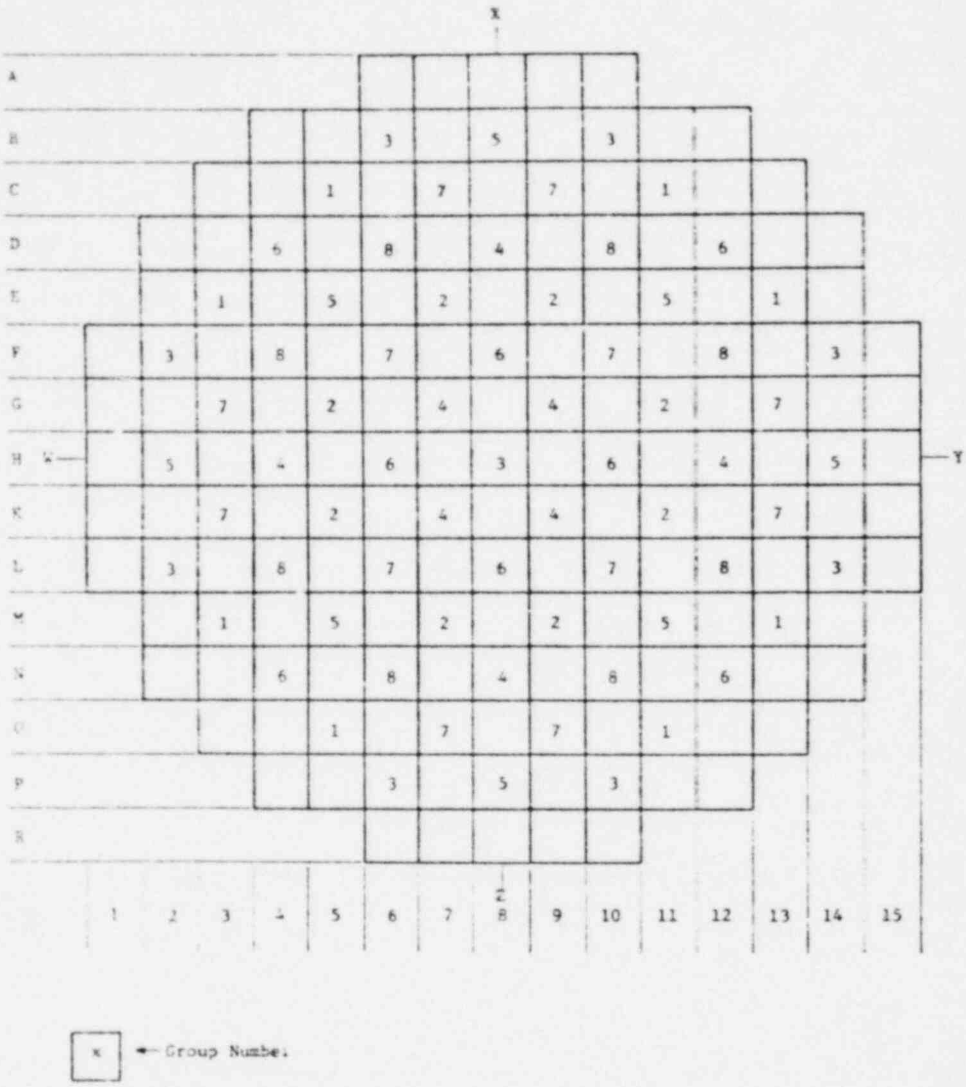
xxx	Previous cycle location.
x	Batch No.

Figure 3-2. Enrichment and Burnup Distribution for Oconee 1, Cycle 5

	8	9	10	11	12	13	14	15
H	3.20 28,479	2.75 20,488	2.75 16,053	3.20 31,135	3.02 0	2.75 15,903	2.79 5,889	3.02 0
K		3.02 0	2.75 14,270	2.79 5,138	2.75 19,206	2.79 8,537	2.75 16,345	3.02 0
L			2.75 17,336	2.79 5,853	2.79 8,262	2.75 15,846	3.02 0	3.02 0
M				2.75 17,341	2.79 5,011	2.75 18,348	3.02 0	
N					2.79 5,846	2.79 7,092	3.02 0	
O						3.02 0		
P								
R								

x.xx	Initial Enrichment
xxxxx	BOC Burnup, MWd/mtU

Figure 3-3. Control Rod Locations for Oconee 1, Cycle 5



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	

4. FUEL SYSTEM DESIGN

4.1. Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters and dimensions for Oconee 1, cycle 5 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 the Oconee 1, cycle 4 reload report[4] are applicable to the cycle 5 reload core.

Five batch 4D Mark-B3 assemblies are remaining in the core for their fourth cycle of irradiation and will experience burnups up to approximately 41,000 MWd/mtU as part of a joint Duke Power/B&W/Dept. of Energy program to demonstrate extended burnup feasibility in LWRs. The Mark-B fuel assembly mechanical design will maintain its structural integrity with those burnups. Analyses of post-irradiation examination (PIE) data from two cycles of operation in the Oconee 1 reactor show that all parameters measured indicate that extended operation is quite feasible. The parameters investigated include fuel rod and assembly growth, fuel swelling, and holddown spring force. The intended peak burnups of batch 4D fuel are within the original mechanical peak design limits reported in the Oconee FSAR.[3] Design parameters can be affected by burnup, effective full power time, or calendar residence time. Those parameters affected most by the amount of irradiation are fuel rod and assembly growth and fuel swelling. Since burnup is within conservative design limits, growth will be acceptable. Section 4.2.3 discusses fuel swelling as it relates to cladding strain. The holddown spring force is affected by residence time as well as burnup. Evaluation of the PIE data indicates that the holddown spring will meet performance requirements through the fourth cycle of irradiation.

4.2. Fuel Rod Design

4.2.1. Cladding Collapse

Creep collapse analyses were performed for three-cycle assembly power histories as well as for batch 4D's four-cycle assembly power histories. For cycle 5, the batch 5 fuel is more limiting than all other batches except for 4D because of its previous incore exposure time. The batch 5 and 4D assembly power histories were analyzed, and the most limiting assembly from each batch was determined.

The power histories for the most limiting assemblies were used to calculate the fast neutron flux level for the energy range above 1 MeV. The collapse time for the most limiting assembly from each batch was conservatively determined to be more than 30,000 effective full-power hours (EFPH), which is longer than the maximum projected batch 5 residence time of 21,456 EFPH (three cycles) and the maximum projected batch 4D residence time of 28,469 EFPH (four cycles). The creep collapse analyses were performed based on the conditions set forth in references 4 and 5.

4.2.2. Cladding Stress

The Oconee 1 stress parameters are enveloped by a conservative fuel rod stress analysis. Since worst-case stress conditions are at BOL, the batch 4D fuel is also bounded by the fuel rod stress analysis. For design evaluation, the primary membrane stress must be less than two-thirds of the minimum specified unirradiated yield strength, and all stresses (primary and secondary) must be less than the minimum specified unirradiated yield strength. The margin is in excess of 30% in all cases. With respect to Oconee 1 fuel, the following conservatisms were used in the analysis:

1. Low post-densification internal pressure.
2. Low initial pellet density.
3. High system pressure.
4. High thermal gradient across the cladding.

The stresses reported in reference 6 for core 1 fuel represent conservative values with respect to the cycle 5 core.

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% on cladding circumferential plastic strain. The pellet design is established for plastic cladding strain of less than 1% at maximum design local pellet burnup (55,000 MWd/mtU) and heat generation rate (20.15 kW/ft) values that are higher than the values the Oconee 1 fuel is expected to see, including batch 4D. The strain analysis is also based on the maximum Specification value for the fuel pellet diameter and density and the lowest permitted Specification tolerance for the cladding ID.

4.3. Thermal Design

All fuel assemblies in this core are thermally similar. The fresh batch 7 fuel inserted for cycle 5 operation introduces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. The design minimum linear heat rate (LHR) capacity and the average fuel temperature for each batch in cycle 5 are shown in Table 4-2. LHR capabilities are based on centerline fuel melt and were established using the TAFY-3 code[7] with fuel densification to 96.5% of theoretical density. The five batch 4D fuel assemblies have an EOC burnup of about 41,000 MWd/mtU. The EOL maximum pin pressure for these assemblies is well below the system pressure of 2200 psia.

4.4. Material Design

The batch 7 fuel assemblies are not new in concept, nor do they utilize different component materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 7 fuel assemblies are identical to those of the present fuel.

4.5. Operating Experience

Babcock & Wilcox operating experience with the Mark-B, 15 by 15 fuel assembly has verified the adequacy of its design. As of February 28, 1978, the experience described below has been accumulated for the eight operating B&W 177-fuel assembly plants using the Mark-B fuel assembly. In addition, Three Mile Island Unit 2 achieved initial criticality on March 28, 1978, and is currently in the startup testing phase that precedes commercial operation.

Reactor	Current cycle	Max assembly burnup, MWd/mtU		Cumulative net elect. output, mWh
		Incore	Disch.	
Oconee 1	4	27,200	25,300	20,385,249
Oconee 2	3	26,700	26,800	15,248,595
Oconee 3	3	27,140	27,200	16,182,813

Reactor	Current cycle	Max assembly burnup, MWd/mtU Incore	Disch.	Cumulative net elect. output, mWh
TMI-1	3	31,720	25,860	18,430,506
ANO-1	2	28,290	17,650	14,575,320
Rancho Seco	2	22,3[illegible]	17,170	10,297,637
Crystal River 3	1	10,430	--	4,936,412
Davis-Besse 1	1	2,490	--	1,009,741

Table 4-1. Fuel Design Parameters and Dimensions

	Thrice-burned FAs, Batch 4D	Twice-burned FAs, Batch 5	Once-burned FAs, Batch 6	Fresh FAs, Batch 7
FA type	Mark-B3	Mark-B4	Mark-B4	Mark-B4
No. of FAs	5	60	56	56
Fuel rod OD, in.	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.377
Flex. spacers, type	Spring	Spring	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4	Zr-4
Undensif active fuel length (nom), in.	142.0	142.6	142.25	142.25
Fuel pellet initial density (nom), % TD	>94.5	93.5	94.0	94.0
Fuel pellet OD (mean specif), in.	0.3685	0.3700	0.3695	0.3695
Initial fuel enrich., wt % ^{235}U	3.20	2.75	2.79	3.02
BOC burnup (avg), MWd/mtU	30,604	17,011	6,539	0
Cladding collapse time, EFPH	>30,000	>30,000	>30,000	>30,000
Estimated residence time (max), EFPH	28,469	21,456	22,440	26,496

Table 4-2. Fuel Thermal Analysis Parameters

	Batch			
	4D[a]	5[a]	6[a]	7
No. of assemblies	5	60	56	56
Nominal pellet density, % TD	95.5	93.5	94.0	94.0
Pellet diameter, in.	0.3685	0.3700	0.3695	0.3695
Stack height, in.	141.0[b]	142.6	142.25	142.25
Densified Fuel Parameters[c]				
Pellet diameter, in.	0.3640	0.3645	0.3646	0.3646
Fuel stack height, in.	140.30	140.46	140.47	140.47
Nominal LHR at 2568 MWt, kW/ft	5.80	5.80	5.80	5.80
Avg fuel temp at nominal LHR, F	1320	1320	1320	1320
LHR to ℄ fuel melt, kW/ft	20.15	20.15	20.15	20.15

[a] Data from reference 4.
[b] Conservative calculational parameter.
[c] 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 design cycle 5 with those of reference cycle 4. The values for both cycles were generated using PDQ07. The average cycle burnup will be higher in cycle 5 than in the design cycle 4 because of the longer cycle 5 length. Figure 5-1 illustrates a representative relative power distribution for the beginning of cycle 5 at full power with equilibrium xenon and normal rod positions.

The critical boron concentrations for cycle 5 are comparable to those of the design cycle 4. The control rod worths for hot full power differ between cycles due to changes in group designations as well as changes in radial flux distributions and isotopics. The ejected rod worths in Table 5-1 are the maximum calculated values within the allowable rod insertion limits. 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 for cycle 5 is greater than that for the design cycle 4 at BOC and approximately the same at EOC. All safety criteria associated with these worths are met. The adequacy of the shutdown margin with cycle 5 stuck rod 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.
3. Flux redistribution penalty.

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The reference fuel cycle shutdown margin is presented in the Oconee 1, cycle 4 reload report.[4]

The cycle 5 power deficits from hot zero power to hot full power differ from those for the design cycle 4 because of the longer cycle 5 design length. The differential boron worths and total xenon worths for cycle 5 are greater than or equal to those for the design cycle 4 because of fuel depletion and the associated buildup of fission products. Effective delayed neutron fractions for both cycles show a decrease with burnup.

5.2. Analytical Input

The cycle 5 incore measurement calculation constants to be used for computing core power distributions were prepared in the same manner as those for the reference cycle.

5.3. Changes in Nuclear Design

There were no relevant changes in core design between the reference and reload cycles. The same calculational methods and design information were used to obtain the important nuclear design parameters. The only significant operational procedure change from the reference cycle is the operation in a feed-and-bleed mode. The reference cycle began operation in the rodded mode but was subsequently modified for operation in the feed-and-bleed mode. Therefore, since nearly the entire reference cycle 4 was cperated in the feed-and-bleed mode, this is not actually a new mode of operation.

Table 5-1. Oconee 1, Cycle 5 Physics Parameters[(a)]

	Cycle 4[(b)]	Cycle 5[(c)]
Cycle length, EFPD	292	330
Cycle burnup, MWd/mtU	9,136	10,327
Average core burnup, EOC, MWd/mtU	19,034	19,027
Initial core loading, mtU	82.1	82.1
Critical boron, BOC (no Xe), ppm		
HZP, group 8 37.5% wd[(d)]	1415	1458
HZP, groups 7 and 8 inserted	1335	1324
HFP, group 8 inserted	1145	1276
Critical boron, EOC (eq Xe), ppm		
HZP, group 8 37.5% wd	373	343
HFP, group 8 37.5% wd	88	44
Control rod worths, HFP, BOC, % Δk/k		
Group 6	1.07	1.21
Group 7	0.93	1.45
Group 8 37.5% wd	0.50	0.43
Control rod worths, HFP, EOC, % Δk/k		
Group 7	1.16	1.53
Group 8 37.5% wd	0.47	0.48

Table 5-1. (Cont'd)

	Cycle 4[b]	Cycle 5[c]
Max ejected rod worth, HZP, % Δk/k[e]		
BOC (N-12)	0.68	0.57
EOC (N-12)	0.61	0.70
Max stuck rod worth, HZP, % Δk/k		
BOC (N-12)	1.74	2.17
EOC (N-12)	2.02	2.01
Power deficit, HZP to HFP, % Δk/k		
BOC	1.49	1.31
EOC	2.07	2.12
Doppler coeff, 10^{-5}(Δk/k-°F)		
BOC, 100% power, no Xe	-1.45	-1.45
EOC, 100% power, eq Xe	-1.55	-1.62
Moderator coeff, HFP, 10^{-4}(Δk/k-°F)		
BOC (0 Xe, crit ppm, gp 8 ins)	-1.00	-0.45
EOC (eq Xe, 17 ppm, gp 8 ins)	-2.55	-2.64
Boron worth, HFP, ppm/% Δk/k		
BOC (1150 ppm)	109	109
EOC (17 ppm)	101	97
Xenon worth, HFP, % Δk/k		
BOC (4 EFPD)	2.60	2.62
EOC (equilibrium)	2.61	2.73
Eff delayed neutron fraction, HFP		
BOC	0.00593	0.00598
EOC	0.00530	0.00521

(a) Cycle 5 data are for the conditions stated in this report. The cycle 4 core conditions are identified in reference 4.

(b) Based on 292 EFPD at 2568 MWt, cycle 3.

(c) Cycle 5 data are based on a "planned" cycle 4 length of 235 EFPD; the cycle 4 "design" lifetime is 292 EFPD.

(d) HZP denotes hot zero power (532F T_{avg}), HFP denotes hot full power (579F T_{avg}).

(e) Ejected rod worth for groups 5 through 8 inserted.

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

	BOC, % Δk/k	EOC, % Δk/k
Available rod worth		
Total rod worth, HZP	8.91	8.79
Worth reduction due to burnup of poison material	-0.36	-0.42
Maximum stuck rod, HZP	-2.17	-2.01
Net worth	6.38	6.36
Less 10% uncertainty	-0.64	-0.64
Total available worth	5.74	5.72
Required rod worth		
Power deficit, HFP to HZP	1.31	2.12
Max allowable inserted rod worth	0.40	0.60
Flux redistribution	0.59	1.20
Total required worth	2.30	3.92
Shutdown margin (total available worth minus total required worth)	3.44	1.80

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

Figure 5-1. BOC (4 EFPD), Cycle 5 Two-Dimensional Relative Power Distribution — Full Power, Equilibrium Xenon, Normal Rod Positions (Group 8 Inserted)

	8	9	10	11	12	13	14	15
H	0.83	0.93	0.96	0.90	1.37	1.03	1.09	0.87
K		1.35	1.07	1.21	0.98	7 1.09	0.93	0.83
L			7 1.05	1.25	8 1.03	0.95	1.15	0.67
M				1.09	1.23	0.89	0.91	
N					1.21	0.94	0.61	
O						0.70		
P								
R								

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

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6. THERMAL-HYDRAULIC DESIGN

The thermal-hydraulic design evaluation supporting cycle 5 operation utilized the methods and models described in references 3, 4, and 6. The fresh batch 7 fuel is hydraulically and geometrically similar to batch 6 fuel. The cycle 4 and 5 maximum design conditions and significant parameters are shown in Table 6-1. The minimum DNBR shown at the design overpower is unchanged for cycle 5 and is based on 106.5% of RC design flow and on the Mark-B4 fuel assembly and includes the effects of incore fuel densification.

The potential effect of fuel rod bow on DNBR was considered by incorporating suitable margins into DNB-limited core safety limits and RPS setpoints. The maximum rod bow was calculated from the equation

$$\frac{\Delta C}{C_o} = 0.065 + 0.001449\sqrt{BU}$$

where

ΔC = rod bow magnitude, mils,

C_o = initial gap (138 mils),

BU = maximum assembly burnup, MWd/mtU.

The fuel cycle design calculations show that the maximum radial-local peak during cycle 5 is always located in the batch 7 fuel assembly with the maximum burnup. This maximum peak (1.527) is 17% below the 1.78 reference design peak. Since this fuel assembly is limiting for DNBR analysis, the rod bow penalty associated with batch 7 is applied to cycle 5 operation. This method for calculating the maximum core rod bow penalty has been reviewed and approved for acceptability by the USNRC.[8] The Oconee 1, cycle 5 calculated rod bow penalty is 8.0% based on the maximum burnup in batch 7, 13,667 MWd/mtU. No credit is claimed for the difference between calculated cycle 5 peaking and the reference design peaking used for the analysis. An 11.2% rod bow penalty is conservatively applied to all analyses that define plant operating limits and to design transients.

The pressure-temperature limit curve shown in Figure 2.1-3A of the Oconee Technical Specifications provides the basis for the variable low-pressure trip setpoint. The curves shown for four- and three-pump operation represent a locus of points for which the calculated minimum DNBR is equal to 1.30 (BAW-2) plus a suitable margin to offset the DNBR reduction due to rod bow (discussed in the previous paragraph).

The flux/flow trip setpoint was determined on the basis that the Oconee 1 plant has the pump monitor trip function set to trip the reactor upon loss of one pump during four-pump operation if the indicated reactor power is greater than 80% of full power.[9] The flux/flow trip setpoint of 1.055 established for cycle 5 yields a minimum DNBR of 1.68 and a 30% DNBR credit to offset the rod bow penalty.

Table 6-1. Thermal-Hydraulic Design Conditions

	Cycle 4[4]	Cycle 5
Power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	106.5
Vessel inlet coolant temp, 100% power, F	555.6	555.6
Vessel outlet coolant temp, 100% power, F	602.4	602.4
Ref design radial-local power peaking factor	1.783	1.783
Ref design axial flux shape	1.5 cos	1.5 cos
Active fuel length, in.	(a)	(a)
Average heat flux, 100% power, 10^3 Btu/h-ft^2	176[(b)]	176[(b)]
CHF correlation	BAW-2	BAW-2
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Minimum DNBR with densif'n penalty	1.91	1.91

[(a)]See Table 4-2.

[(b)]Based on densified length of 140.3 inches.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[3] accident analysis has been examined with respect to changes in cycle 5 parameters to determine the effect of the cycle 5 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 6. Since batch 7 reload fuel assemblies contain fuel rods whose theoretical density is higher than those considered in the reference 6 report, the conclusions in that reference are still valid.

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. Fuel thermal analysis values for each batch in cycle 5 are compared in Table 4-2. The cycle 5 thermal-hydraulic maximum design conditions are compared to the previous cycle 4 values[4] 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 5 is provided in Table 7-1.

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.[10] 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 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 1 cycle 5 fuel.

Table 7-2 shows the bounding values for allowable LOCA peak LHRs for Oconee 1, cycle 5 fuel.

It is concluded from the examination of cycle 5 core thermal and kinetics properties, with respect to acceptable previous cycle values, that this core reload will not adversely affect the Oconee 1 plant's ability to operate safely during cycle 5. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 5 is considered to be bounded by previously accepted analyses. The initial conditions for the transients in cycle 5 are bounded by the FSAR[3], the fuel densification report[6], and/or subsequent cycle analyses.

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

Parameter	FSAR and densification report value	Predicted cycle 5 value
Doppler coeff, $\Delta k/k/°F$		
BOC	-1.17×10^{-5}	-1.45×10^{-5}
EOC	-1.33×10^{-5}	-1.62×10^{-5}
Moderator coeff, $\Delta k/k/°F$		
BOC	$+0.5 \times 10^{-4}$	-0.45×10^{-4}
EOC	-3.0×10^{-4}	-2.64×10^{-4}
All-rod group worth, HZP % $\Delta k/k$	10	8.91
Initial boron conc'n, HFP, ppm	1400	1276
Boron reactivity worth at 70F, ppm/1% $\Delta k/k$	75	76
Max ejected rod worth, HFP, % $\Delta k/k$	0.65	0.25
Dropped rod worth (HFP), % $\Delta k/k$	0.46	0.20

Table 7-2. LOCA Limits, Oconee 1, Cycle 5

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

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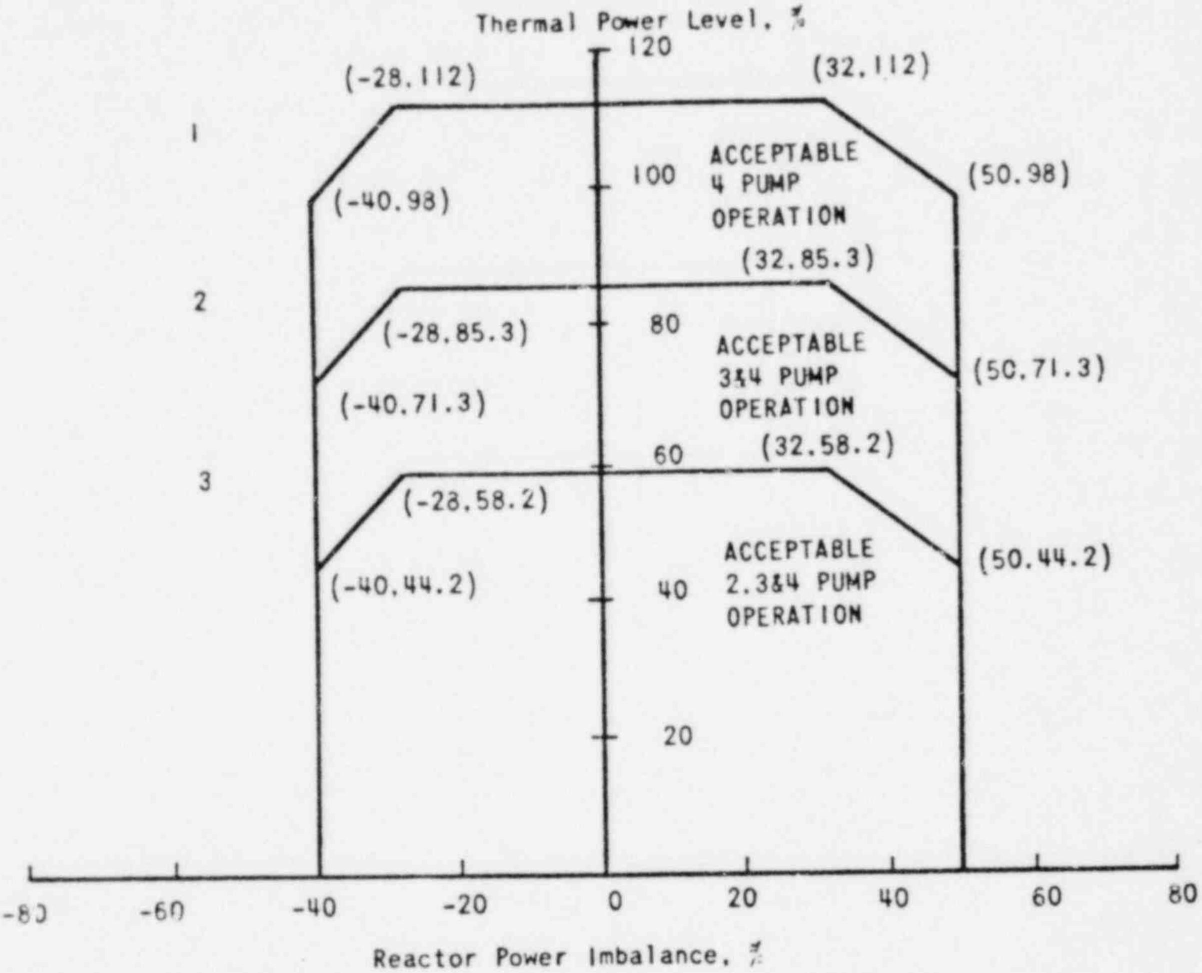
8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 5 operation. Changes were the results of the following:

1. 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 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 11.
2. Per reference 12, the power spike penalty due to fuel densification was not used in setting the DNBR- and ECCS-dependent Technical Specification limits.
3. The allowable quadrant tilt limit for cycle 5 is 5.0%.

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 8-10 illustrate revisions to previous Technical Specification limits.

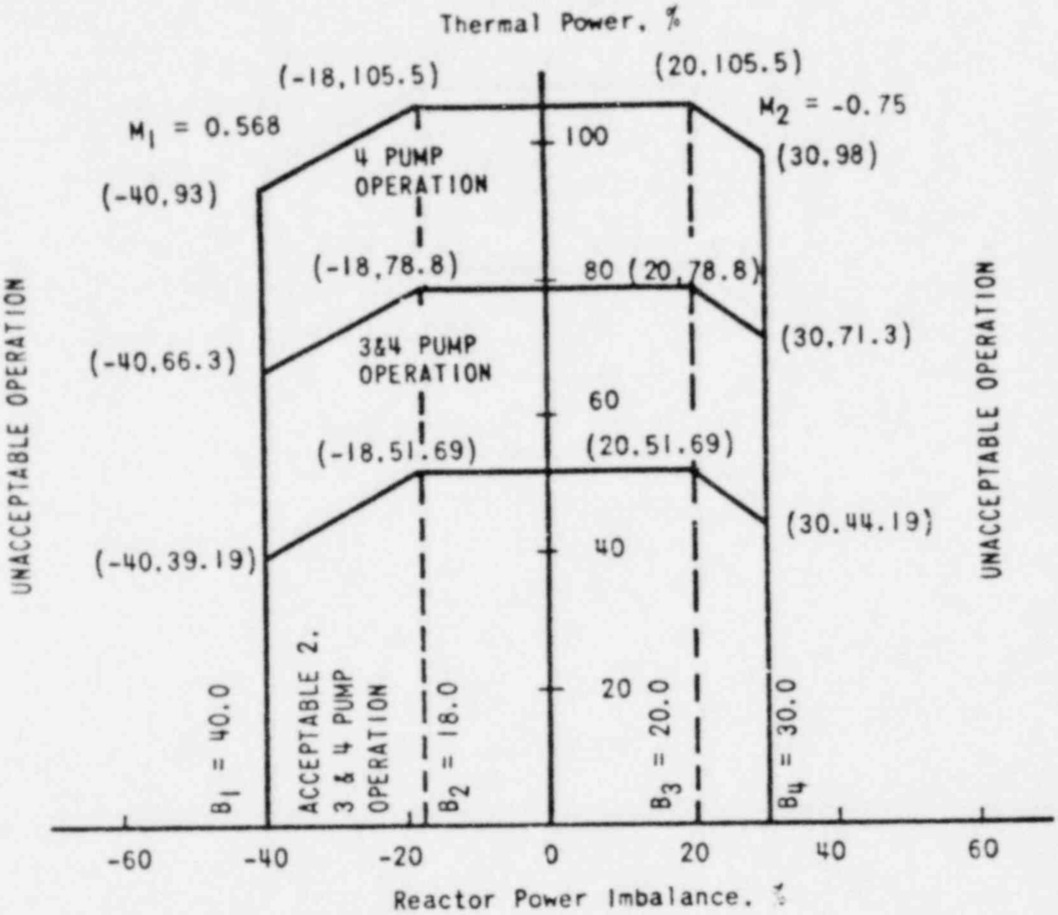
Figure 8-1. Core Protection Safety Limits, Oconee Unit 1



CURVE	RC FLOW (GPM)
1	374,880
2	280,035
3	183,690

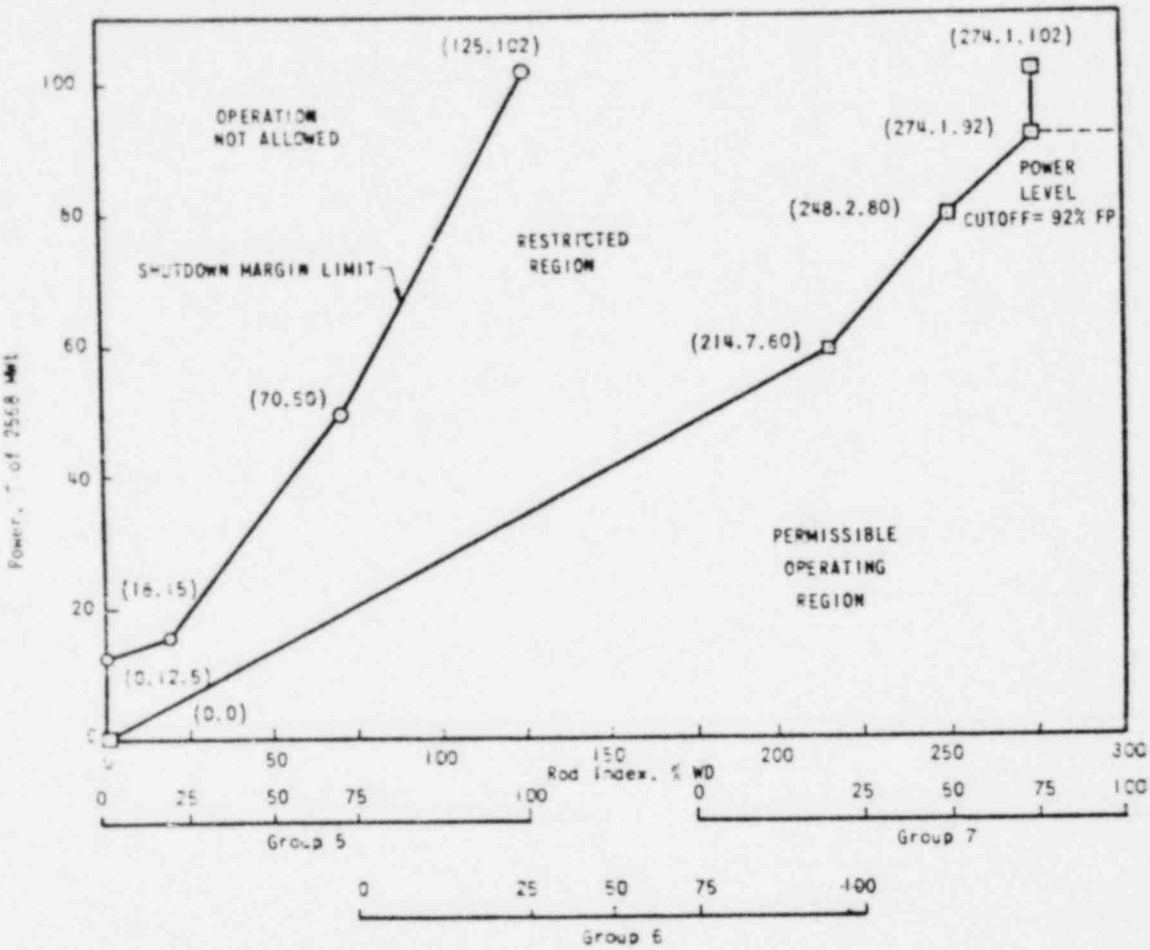
This is proposed new Technical Specification Figure 2.1-2A.

Figure 8-2. Protective System Maximum Allowable Setpoints, Oconee Unit 1



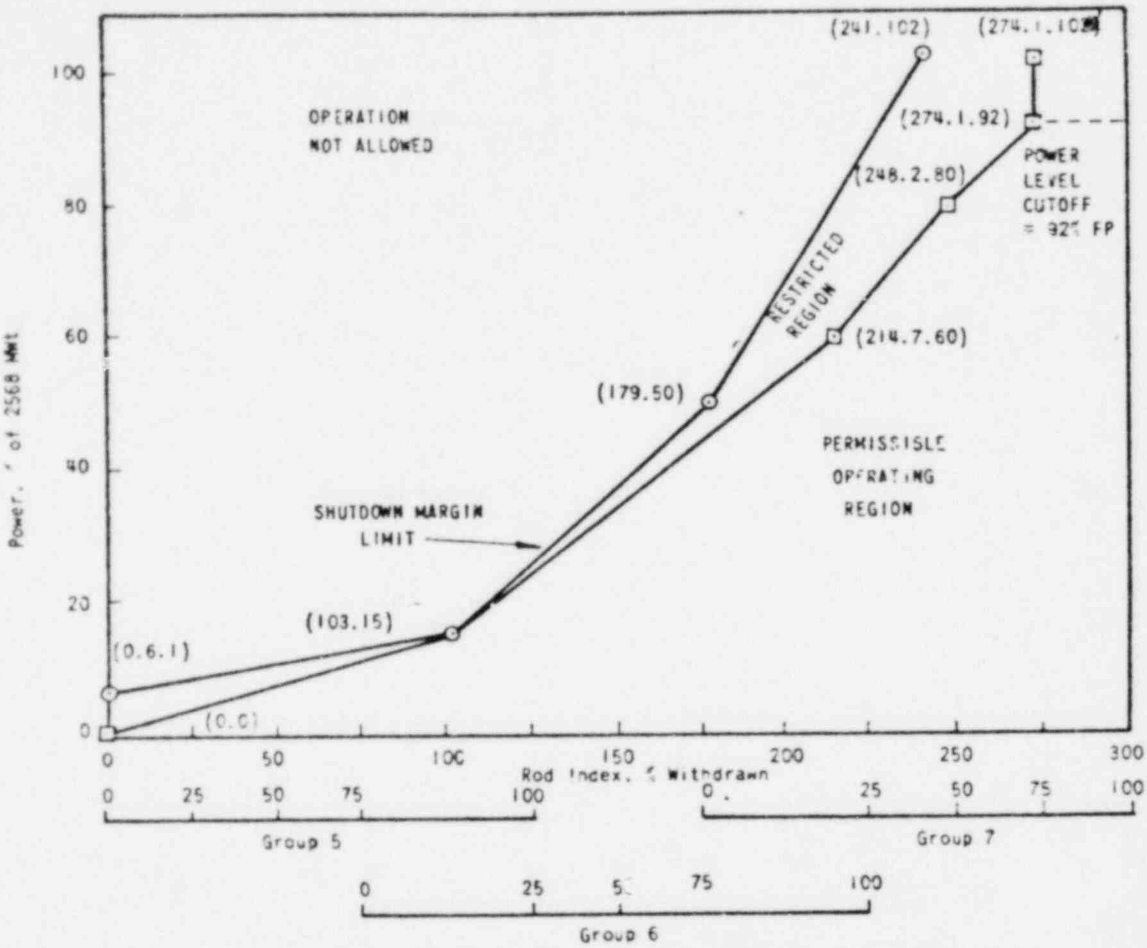
This is proposed new Technical Specification Figure 2.3-2A.

Figure 8-3. Rod Position Limits for Four-Pump Operation, Oconee Unit 1 (0 to 100 ± 10 EFPD)



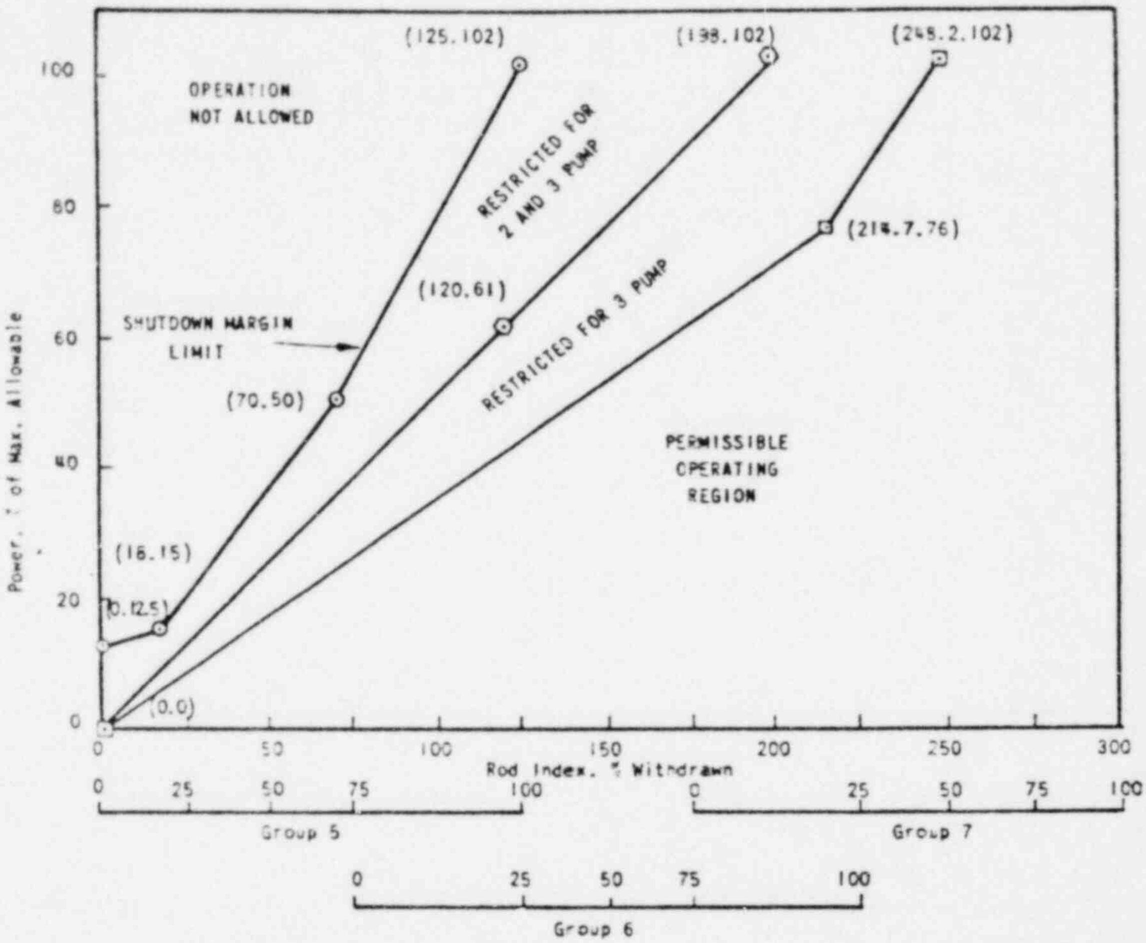
This is proposed new Technical Specification Figure 3.5.2-1A1.

Figure 8-4. Rod Position Limits for Four-Pump Operation, Oconee Unit 1 (After 100 ± 10 EFPD)



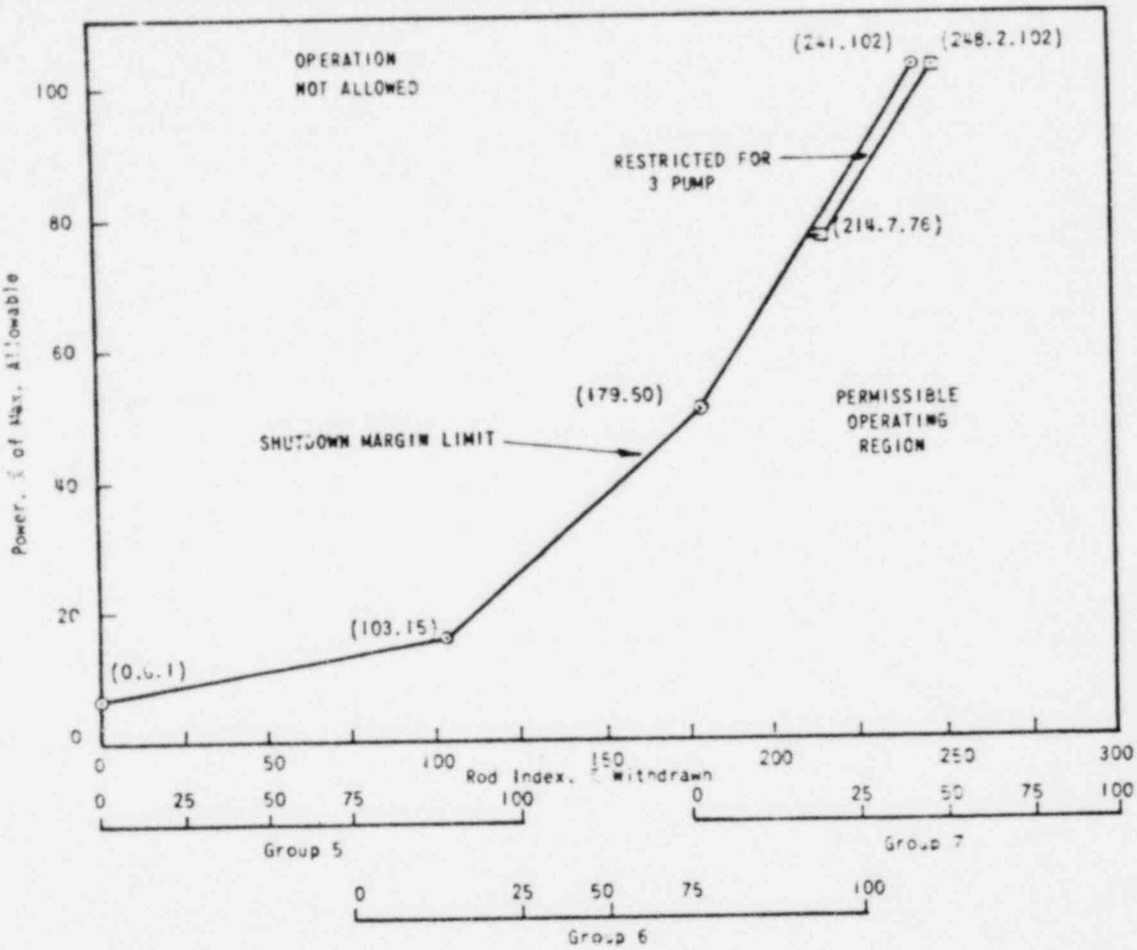
This is proposed new Technical Specification Figure 3.5.2-1A2.

Figure 8-5. Rod Position Limits for Two- and Three-Pump Operation, Oconee Unit 1 (0 to 100 ± 10 EFPD)



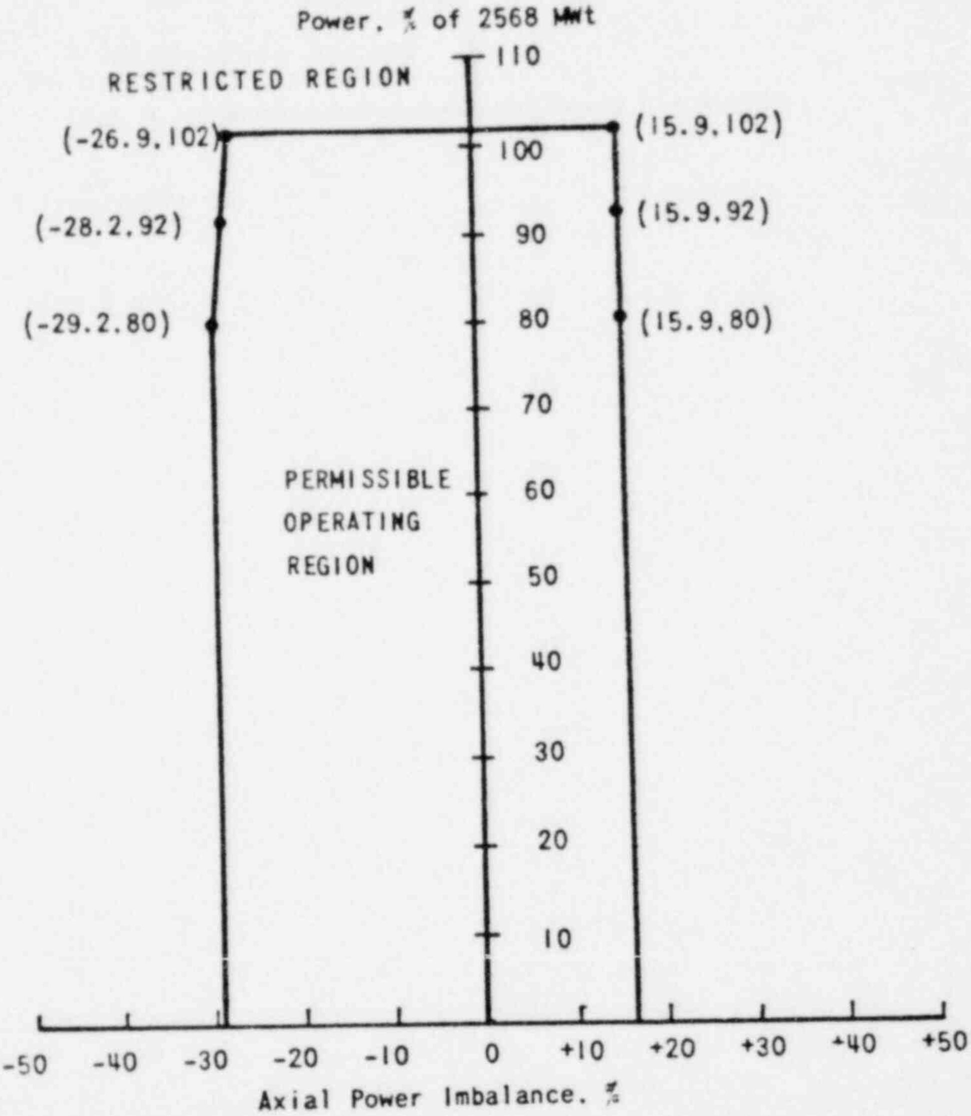
This is proposed new Technical Specification Figure 3.5.2-2A1.

Figure 8-6. Rod Position Limits for Two- and Three-Pump Operation, Oconee Unit 1 (After 100 ± 10 EFPD)



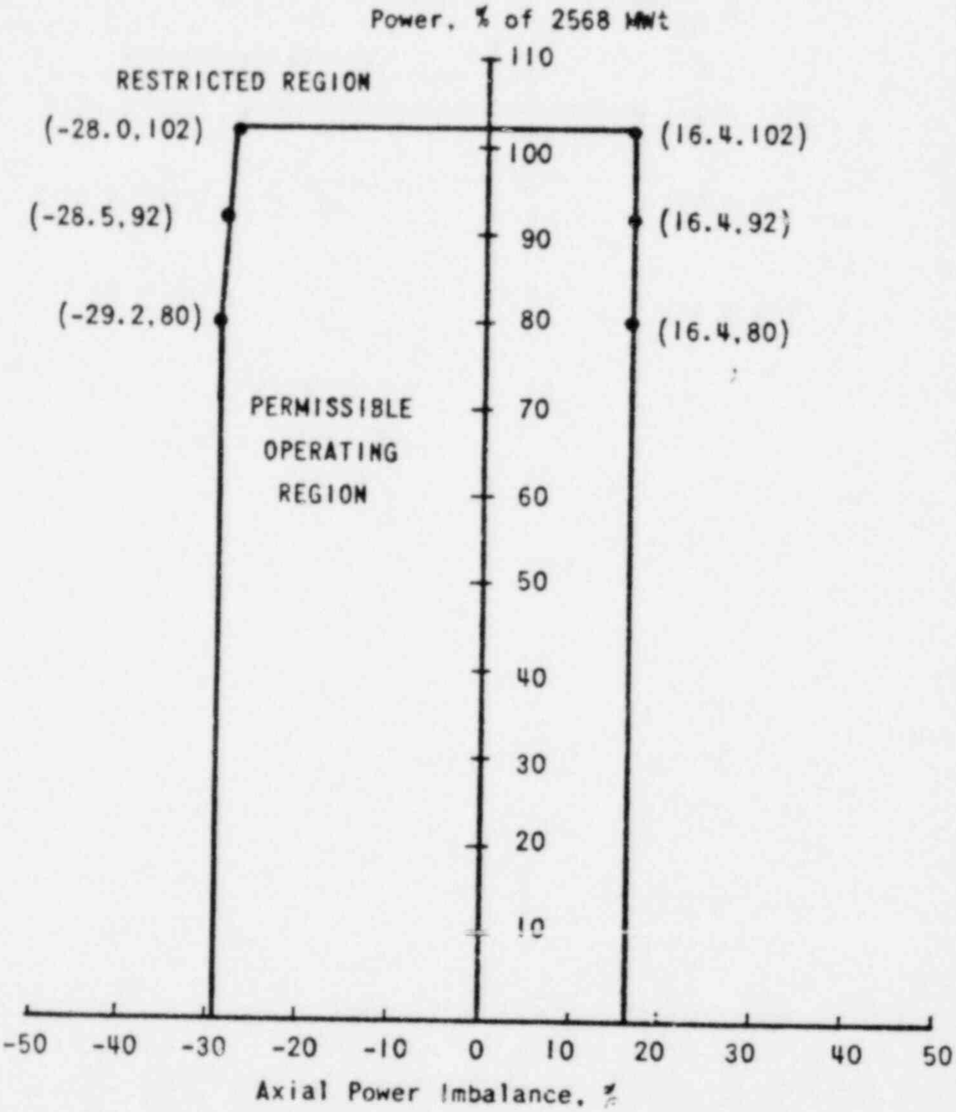
This is proposed new Technical Specification Figure 3.5.2-2A2.

Figure 8-7. Power Imbalance Limits, Oconee Unit 1
(0 to 100 ± 10 EFPD)



This is proposed new Technical Specification Figure 3.5.2-3A1.

Figure 8-8. Power Imbalance Limits, Oconee Unit 1
(After 100 ± 10 EFPD)



This is proposed new Technical Specification Figure 3.5.2-3A2.

Figure 8-9. APSR Position Limits, Oconee Unit 1
(From 0 to 100 ± 10 EFPD)

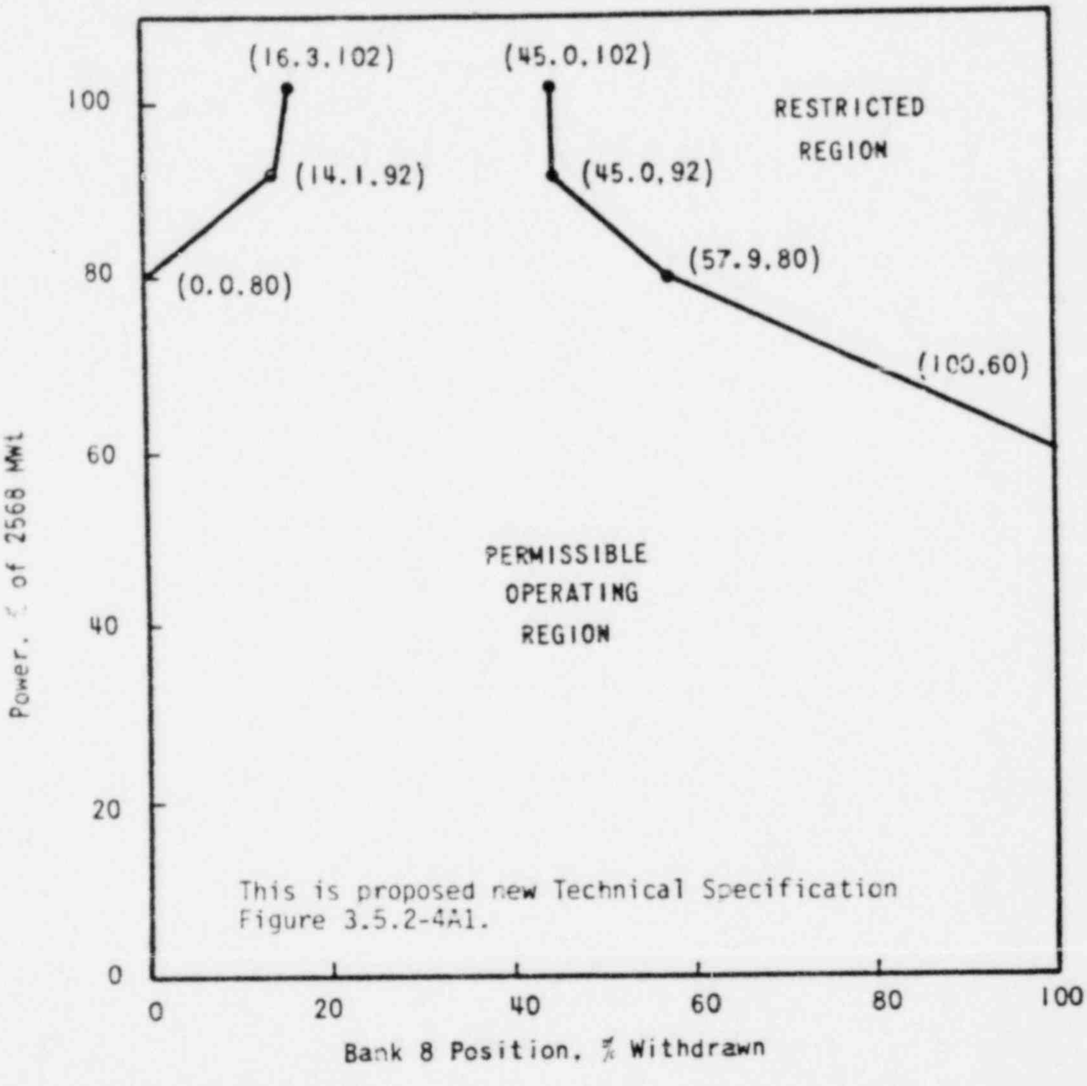
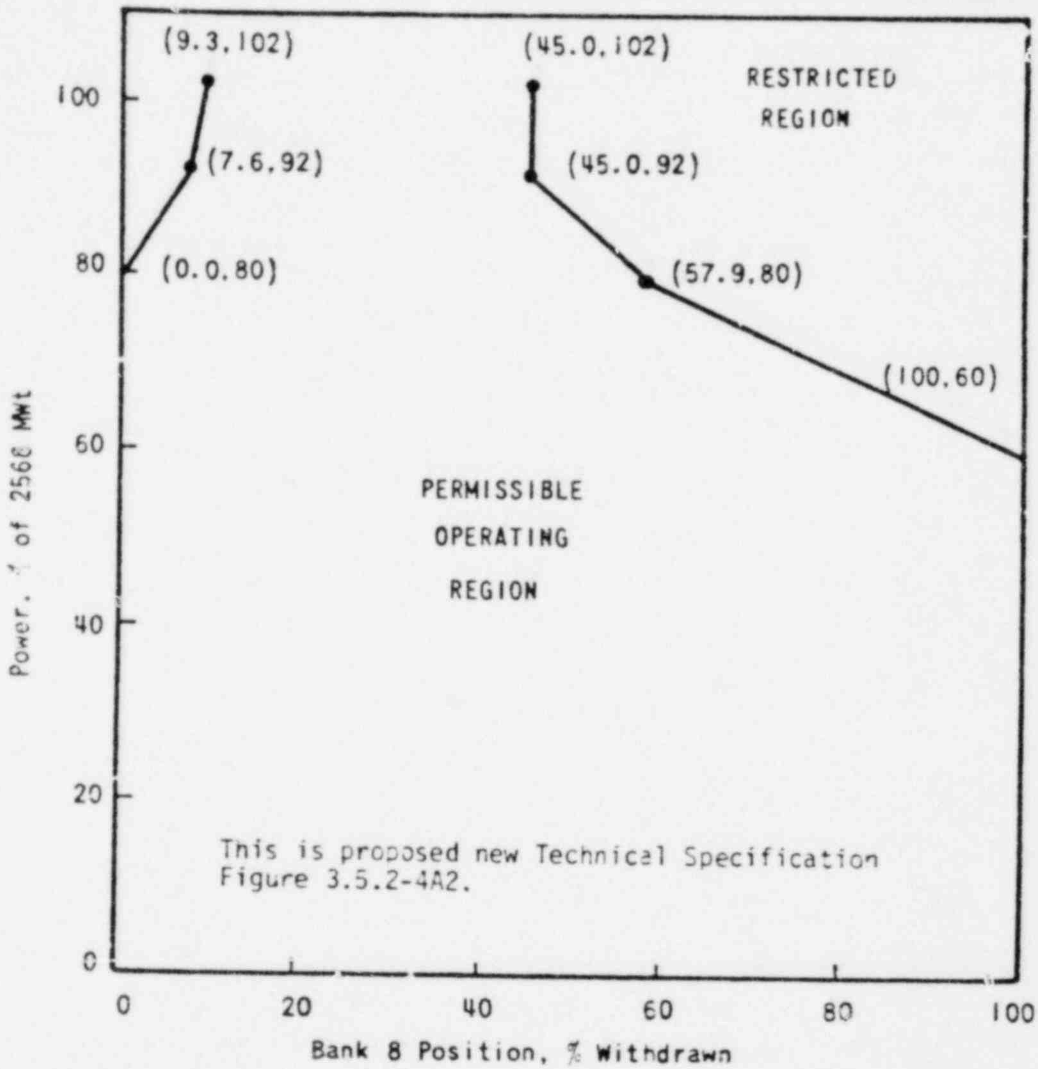


Figure 8-10. APSR Position Limits, Oconee Unit 1
(After 100 ± 10 EFPD)



This is proposed new Technical Specification Figure 3.5.2-4A2.

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. Incore versus out-of-core detector imbalance correlation verification at less than full power.
3. Power Doppler reactivity coefficient at approximately 100% full power.
4. Temperature reactivity coefficient at approximately 100% full power.

REFERENCES

1 Oconee 1, Cycle 4 Quadrant Flux Tilt, BAW-1477, Babcock & Wilcox, Lynchburg, Virginia, January 1978.

2 A. C. Thies (Duke Power Co.) to Edson G. Case (USNRC), Letter, October 26, 1977, Docket No. 50-269.

3 Oconee Nuclear Station, Units 1, 2, and 3 — Final Safety Analysis Reports, Docket Nos. 50-269, 50-270, and 50-287, Duke Power Company.

4 Oconee Unit 1, Cycle 4 Reload Report, BAW-1447, Babcock & Wilcox, Lynchburg, Virginia, March 1977.

5 Program to Determine In-Reactor Performance of B&W Fuels — Cladding Creep Collapse, BAW-10084, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, December 1976.

6 Oconee 1 Fuel Densification Report, BAW-1388, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, July 1973.

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

8 Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment 14 to Facility Operating License No. DPR-54, Rancho Seco Nuclear Generation Station, Sacramento Municipal Utility District, Docket No. 50-312.

9 Duke Power Company to E. G. Case (Acting Director, Office of Nuclear Reactor Regulation), Letter, "Revision of Oconee Nuclear Station Tech. Spec. to Modify Pump Monitor Trip Setpoint," September 14, 1977.

10 ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, September 1975.

[11] S. A. Varga (USNRC) to J. H. Taylor (B&W), Letter, "Comments on B&W's Submittal on Combination of Peaking Factors," May 13, 1977.

[12] S. A. Varga (USNRC) to J. H. Taylor (B&W), Letter, "Update of BAW-10055 — Fuel Densification Report," December 5, 1977.