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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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POWER BUILDING

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

A. C. THIES
SENIOR VICE PRESIDENT
PRODUCTION AND TRANSMISSION

P. O. Box 2178

RECEIVED NOV 8 1974 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

October 31, 1974

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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11442

Mr. Angelo Giambusso
Page 2
October 31, 1974

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 Company 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

PROPOSED TECHNICAL SPECIFICATIONS

Rev. 1. 10/28/74

## 2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

### 2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

The combination of the reactor system pressure and coolant temperature shall not exceed the safety limit as defined by the locus of points established in Figure 2.1-1A-Unit 1. If the actual pressure/temperature point is below
2.1-1B-Unit 2
2.1-1C-Unit 3
and to the right of the line, the safety limit is exceeded.

The combination of reactor thermal power and reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of the rated power) shall not exceed the safety limit as defined by the locus of points (solid line) for the specified flow set forth in Figure 2.1-2A-Unit 1. If the actual reactor-thermal-power/power
2.1-2B-Unit 2
2.1-2C-Unit 3
imbalance point is above the line for the specified flow, the safety limit is exceeded.

Bases - Unit 1

The safety limits presented for Oconee Unit 1 have been generated using BAW-2 critical heat flux (CHF) correlation(1) and the actual measured flow rate at Oconee Unit 1 (2). This development is discussed in the Oconee 1, Cycle 2-Reload Report, reference (2). The flow rate utilized is 107.6 percent of the design flow (131.32 x $10^6$ lbs/hr) based on four-pump operation.(2)

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the BAW-2 correlation (1). The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.32. A DNBR of 1.32 corresponds to a 95 percent probability at a 99 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setponts to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1A represents the conditions at which a minimum DNBR of 1.32 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 107.6 percent of 131.3 x $10^6$ lbs/hr.). This curve is based on the combination of nuclear power peaking factors, with potential fuel densification effects, which result in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2A are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The 1.32 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.32 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.15 kw/ft for Unit 1.

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, 3 and 4 of Figure 2.1-2A correspond to the expected minimum flow rates with four pumps, three pumps, one pump in each loop and two pumps in one loop, respectively.

The curve of Figure 2.1-1A is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3A. (because the four-pump pressure - temperature restriction is known to be more limiting than the 3 and 2 pump combinations, only the four pump limit has been shown on Figure 2.1-3A).

The maximum thermal power for three-pump operation is 87 percent due to a power level trip produced by the flux-flow ratio 75 percent flow x 1.08 = 81 percent power, plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

For Figure 2.1-3A, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.32. The 1.32 DNBR curve for four-pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

References


(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March, 1970.

(2) Oconee 1, Cycle 2 - Reload Report - BAW-1409, Sepetmeber, 1974.

Bases - Units 2 and 3

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the W-3 correlation.(1) The W-3 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.3. A DNBR of 1.3 corresponds to a 94.3 percent probability at a 99 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1B
2.1-1C
represents the conditions at which a minimum DNBR of 1.3 is predicted for the maximum possible thermal power (112%) when four reactor coolant pumps are operating (minimum reactor coolant flow is 131.3 x $10^6$ lbs/hr). This curve is based on the following nuclear power peaking factors(2) with potential fuel densification effects:

$$F_q^N = 2.67; \; F_{\Delta H}^N = 1.78; F_z^N = 1.50$$

The design peaking combination results in a more conservative DNBR than any other shape that exists during normal operation.

The curves of Figure 2.1-2B
2.1-2C
are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The 1.3 DNBR limit produced by a nuclear power peaking factor of $F_q^N = 2.67$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than 1.3 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 19.8 kw/ft - Unit 2
19.8 kw/ft - Unit 3

Power peaking is not a directly observable quantity and therefore limits have been established on the bases of the reactor power imbalance produced by the power peaking.

The specified flow rates for Curves 1, 2, 3, and 4 of Figure 2.1-2B correspond
2.1-2C
to the expected minimum flow rates with four pumps, three pumps, one pump in each loop and two pumps in one loop, respectively.

The curve of Figure 2.1-1B is the most restrictive of all possible reactor
2.1-1C
coolant pump-maximum thermal power combinations shown in Figure 2.1-3B.
2.1-3C

The curves of Figure 2.1-3B represent the conditions at which a minimum DNBR
2.1-3C
of 1.3 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 15%,(3) whichever condition is more restrictive.

Using a local quality limit of 15 percent at the point of minimum DNBR as a basis for Curves 2 and 4 of Figure 2.1-3B is a conservative criterion even
2.1-3C
though the quality of the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the W-3 correlation continually increases from point of minimum DNBR, so that the exit DNBR is 1.7 or higher, depending on the pressure. Extrapolation of the W-3 correlation beyond its published quality range of +15 percent is justified on the basis of experimental data.(4)

The maximum thermal power for three pump operation is 86% - Unit 2
86% - Unit 3
due to a power level trip produced by the flux-flow ratio 75% flow x 1.07 = 80%
1.07 = 80%
power
plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

For each curve of Figure 2.1-3B, a pressure-temperature point above and to the
2.1-3C
left of the curve would result in a DNBR greater than 1.3 or a local quality at the point of minimum DNBR less than 15 percent for that particular reactor coolant pump situation. The 1.3 DNBR curve for four-pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four-pump curve will be above and to the left of the other curves.

REFERENCES

(1) FSAR, Section 3.2.3.1.1
(2) FSAR, Section 3.2.3.1.1.c
(3) FSAR, Section 3.2.3.1.1.k

(4) The following papers which were presented at the Winter Annual Meeting, ASME, November 18, 1969, during the "Two-phase Flow and Heat Transfer in Rod Bundles Symposium:"

   (a) Wilson, et al.
       "Critical Heat Flux in Non-Uniform Heater Rod Bundles"

   (b) Gellerstedt, et al.
       "Correlation of a Critical Heat Flux in a Bundle Cooled by Pressurized Water"

## 2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

### Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

### Objective

To provide automatic protective action to prevent any combination of process variables from exceeding a safety limit.

### Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1A - Unit 1 and
2.3-1B - Unit 2
2.3-1C - Unit 3

Figure 2.3-2A1 } Unit 1
2.3-2A2
2.3-2B - Unit 2
2.3-2C - Unit 3

The pump monitors shall produce a reactor trip for the following conditions:

a. Loss of two pumps and reactor power level is greater than 55% (0.0% for Unit 1) of rated power.

b. Loss of two pumps in one reactor coolant loop and reactor power level is greater than 0.0% of rated power. (Power/RC pump trip setpoint is reset to 55% of rated power for single loop operation. Power/RC pump trip setpoint is reset to 55% for all modes of 2 pump operation for Unit 1.)

c. Loss of one or two pumps during two-pump operation.

### Bases

The reactor protective system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protective system instrumentation are listed in Table 2.3-1A - Unit 1. The safety analysis has been based upon these protective
2.3-1B - Unit 2
2.3-1C - Unit 3
system instrumentation trip set points plus calibration and instrumentation errors.

### Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is more conservative than the value used in the safety analysis.(4)

Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.3 should a low flow condition exist due to any electrical malfunction.

The power level trip set point produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip set point produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.3-1A are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 108% and reactor flow rate is 100%, or flow rate is 93% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 81.0% and reactor flow rate is 74.7% or flow rate is 69% and power level is 75%.

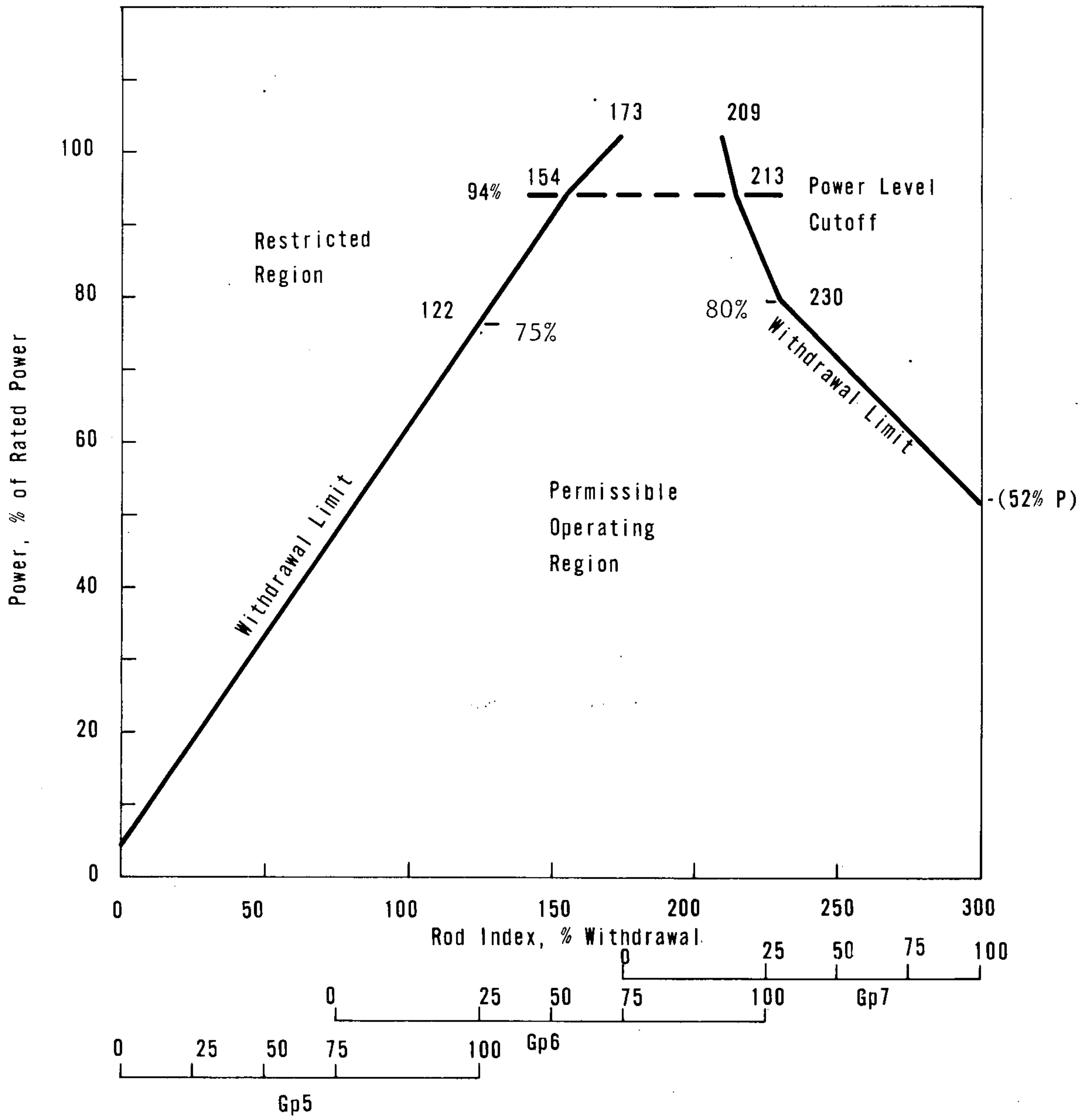
3. Trip would occur when two reactor coolant pumps are operating in a single loop if power is 59% and the operating loop flow rate is 54.5% or flow rate is 43% and power level is 46%.

4. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 53% and reactor flow rate is 49.0% or flow rate is 45% and the power level is 49%.

For safety calculations the maximum calibration and instrumentation errors for the power level trip were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in the top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio such that the boundaries of Figure 2.3-2A1 }Unit 1 are produced. The power-to-flow ratio reduces the power
2.3-2A2
2.3-2B - Unit 2
2.3-2C - Unit 3

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.



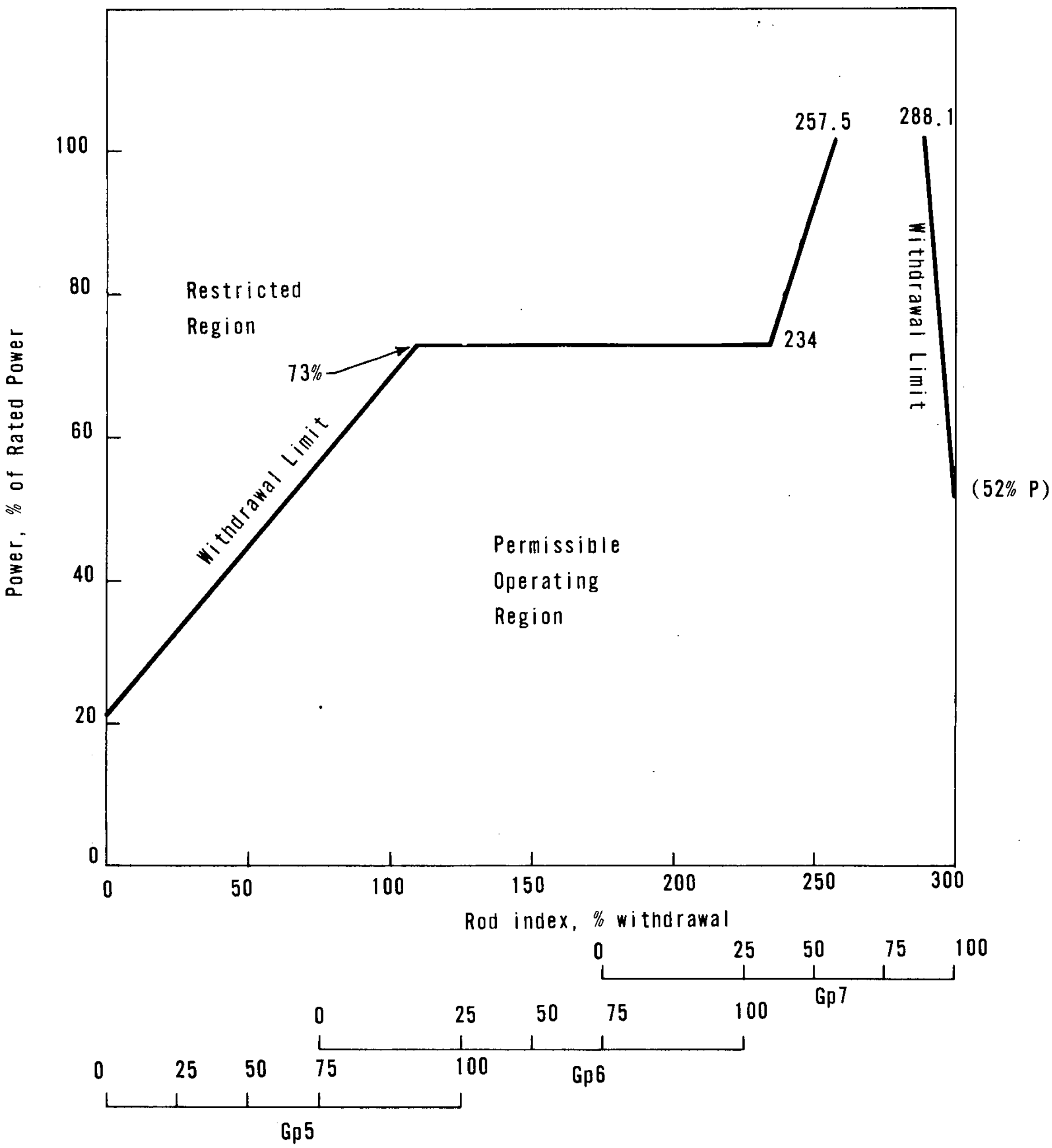

CONTROL ROD GROUP WITHDRAWAL LIMITS FOR 4 PUMP OPERATION

UNIT 1



OCONEE NUCLEAR STATION
Figure 3.5.2-1A1

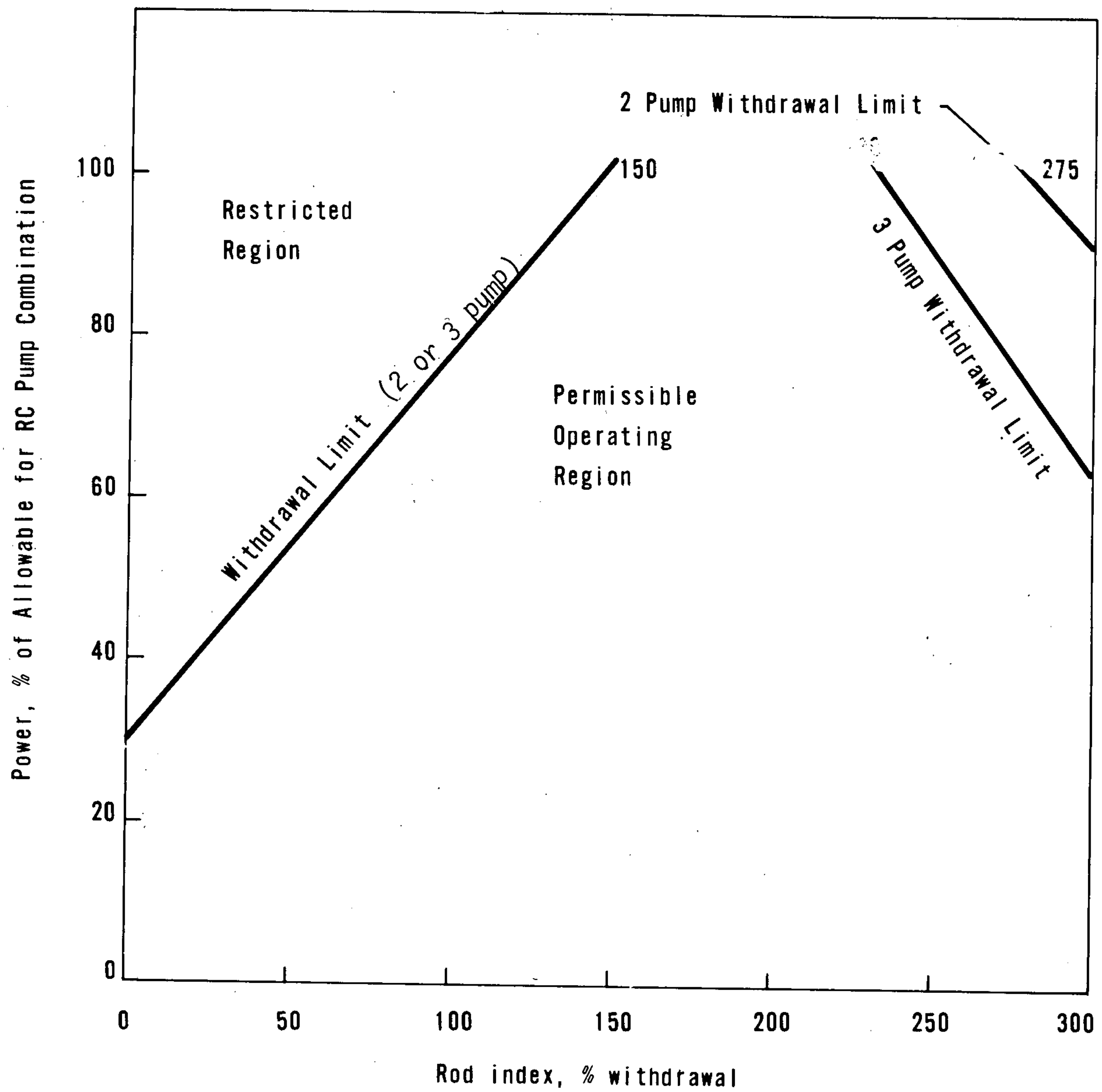
1. Rod Index is the percentage sum of the withdrawal of the operating groups.
2. The withdrawal limits are in effect after 250 ± 5 full power days of operation. (The applicable power level cutoff is 100% power)




CONTROL ROD GROUP WITHDRAWAL LIMIT FOR
4 PUMP OPERATION

UNIT 1

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




CONTROL ROD GROUP WITHDRAWAL LIMITS FOR
3 AND 2 PUMP OPERATION

UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-2A

REPLACEMENT PAGES FOR

BABCOCK AND WILCOX REPORT 1409

Rev. 1. 10/28/74

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 |
|---|---|---|
| Cycle length, EFPD | 290 | 310 |
| Cycle burnup, MWd/mtU | 9,000 | 9,600 |
| Average core 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 |

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

| | BOC, $\frac{\%\Delta k}{k}$ | EOC,** $\frac{\%\Delta k}{k}$ |
|---|---|---|
| 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.

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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

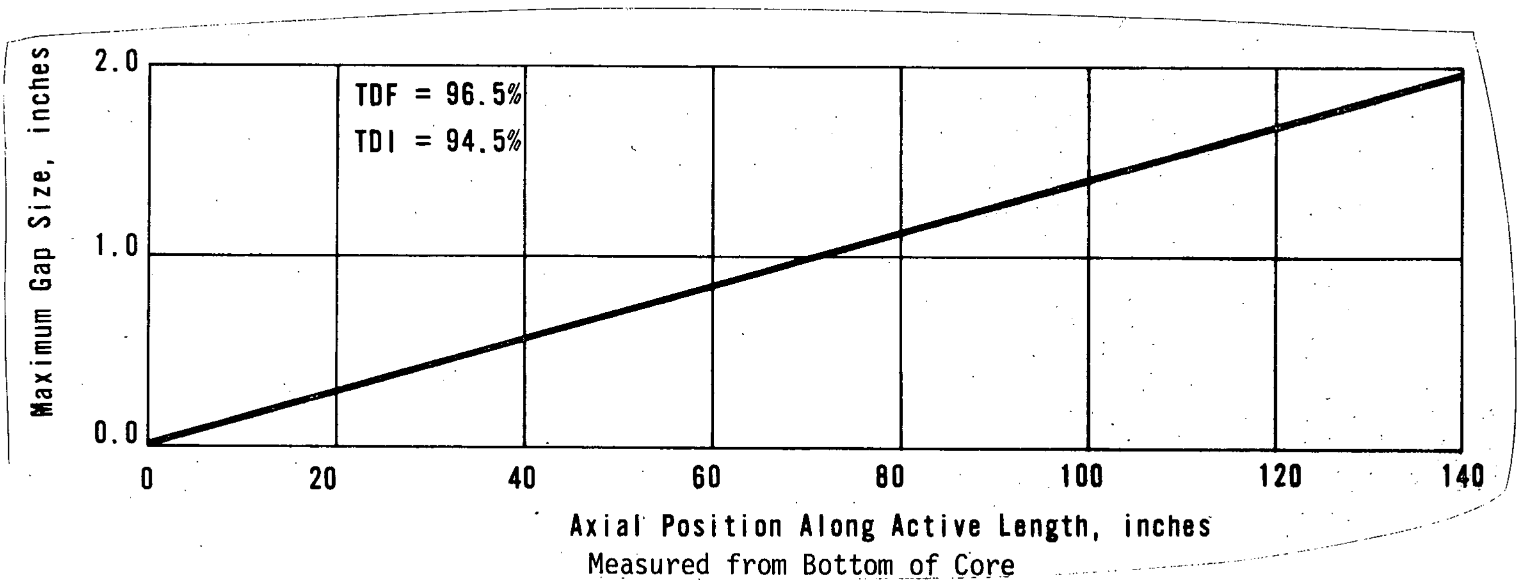

Figure 3-2. Power Spike Factor Vs Axial Position

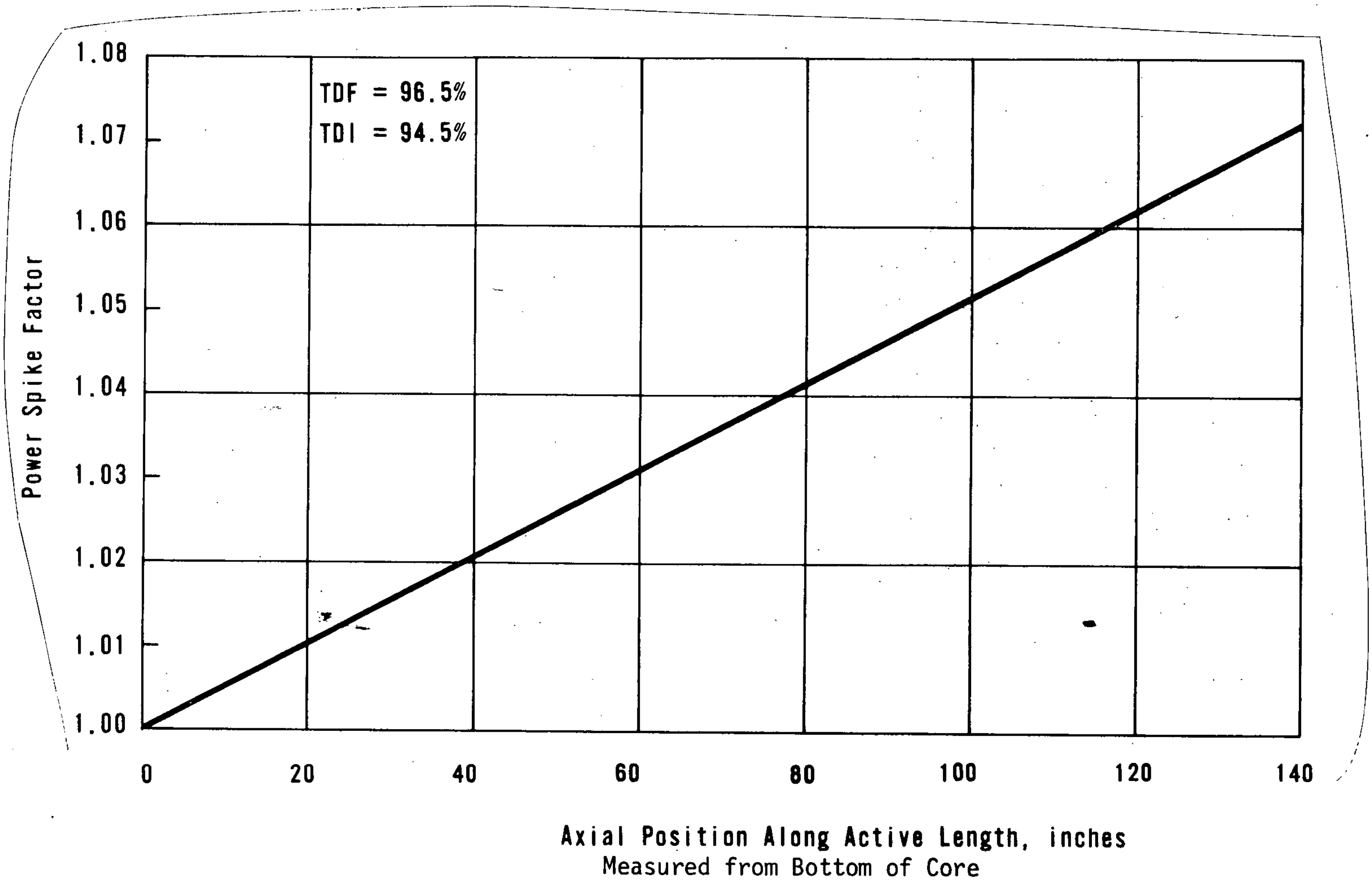

Figure 3-5 Control Rod Group Withdrawal Limits for Four Pump Operation

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.

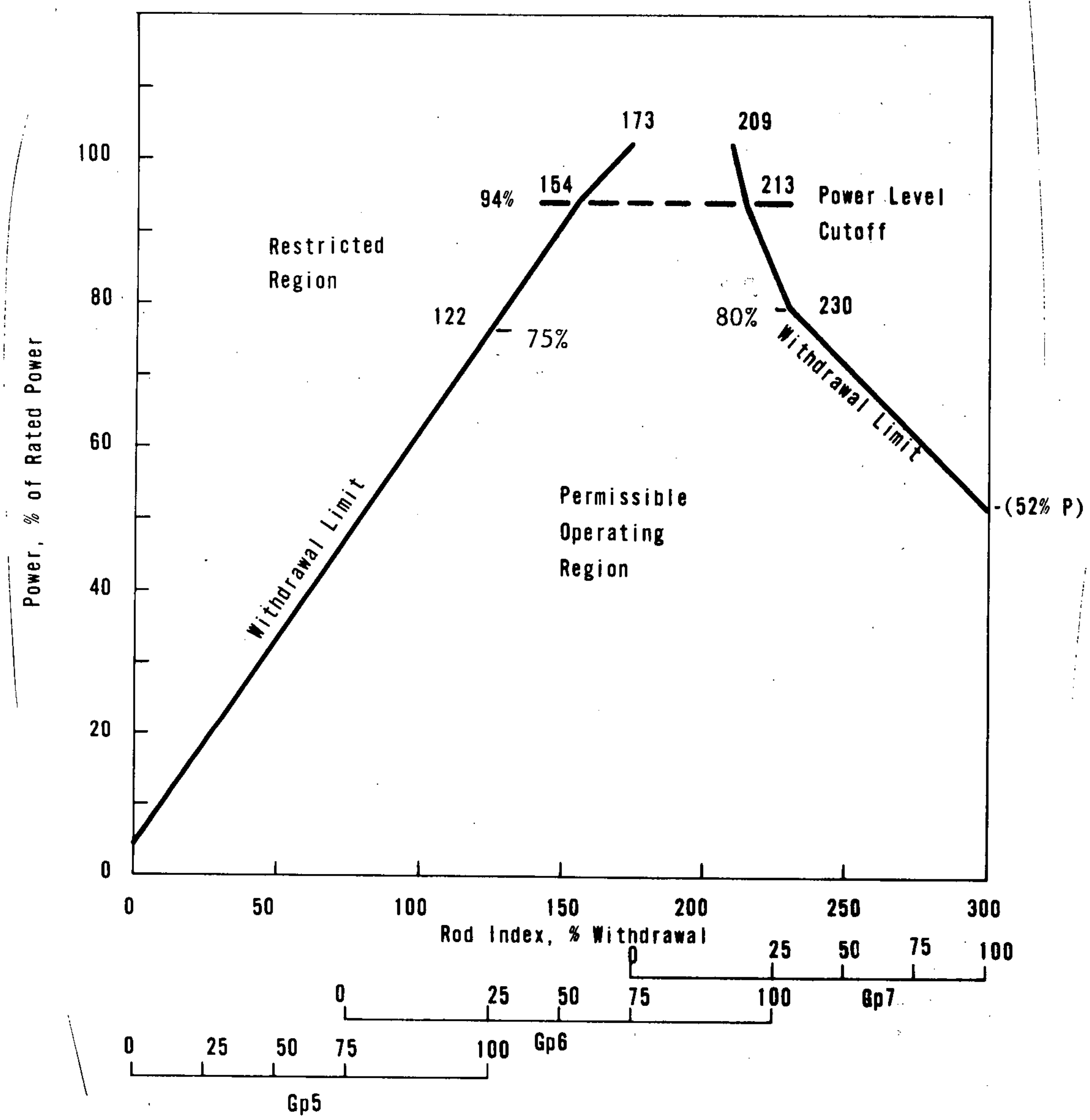




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Figure 3-6 Control Rod Group Withdrawal Limits for Four Pump Operation

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)

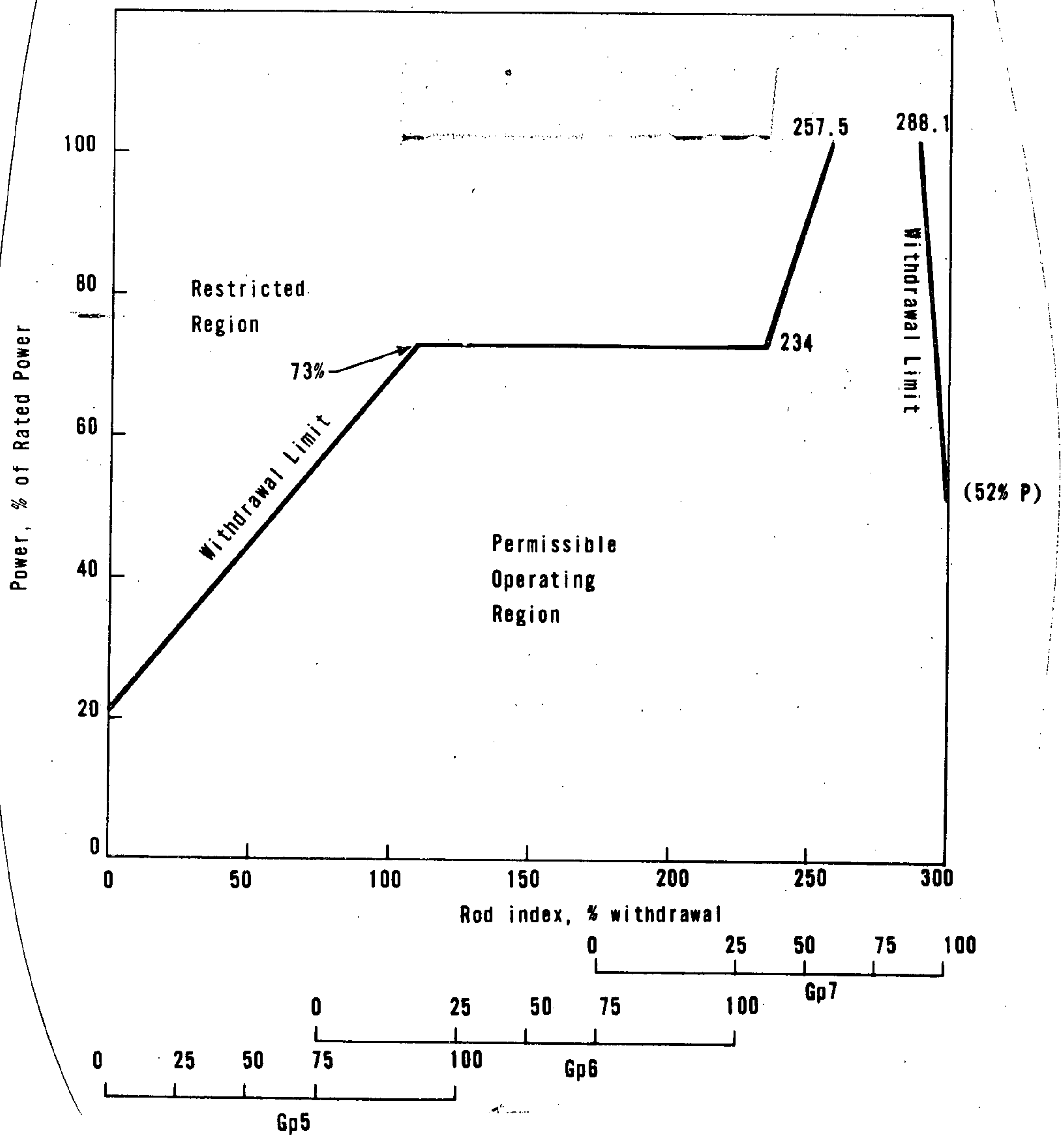




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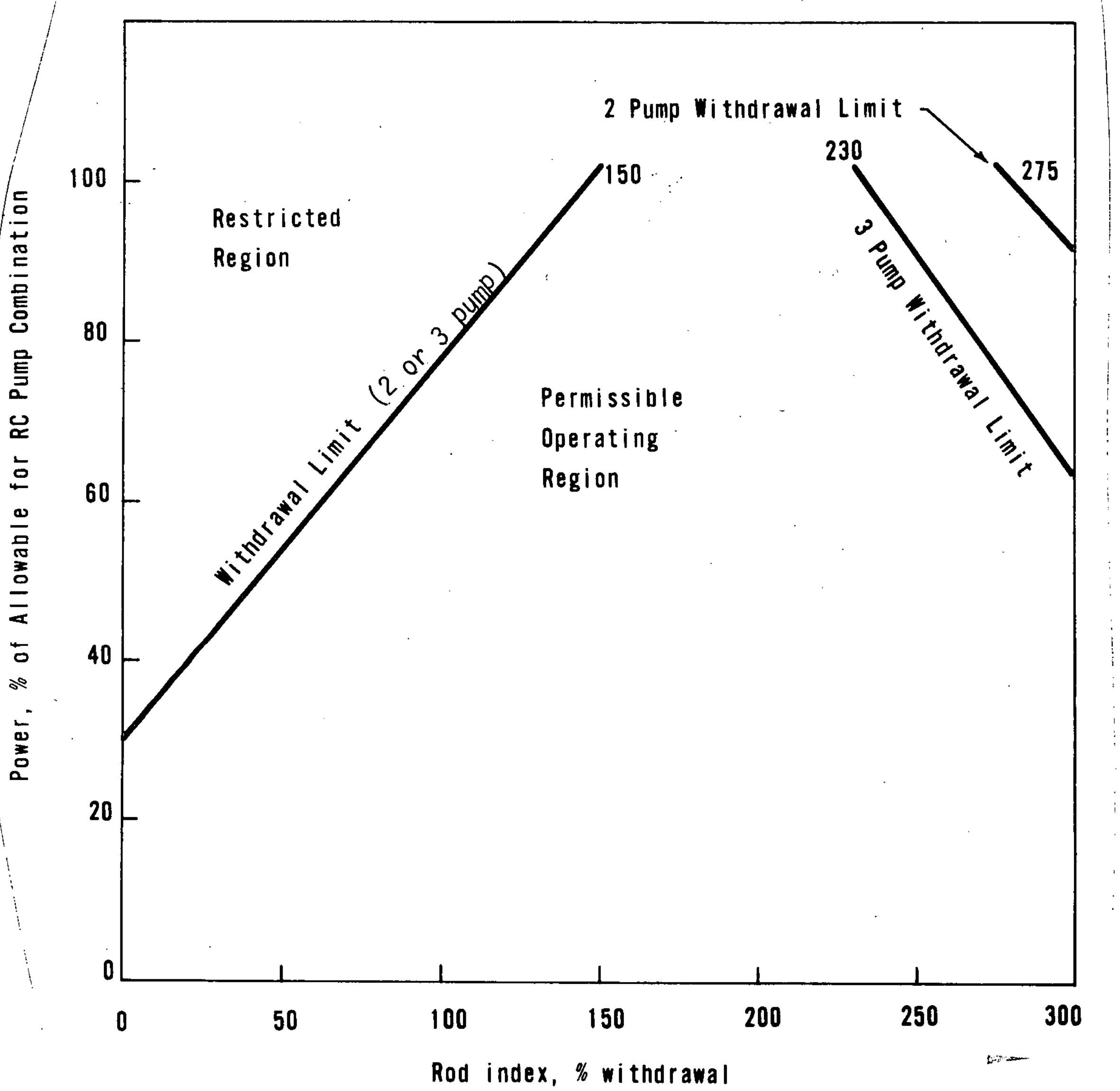
Figure 3-7. Control Rod Group Withdrawal Limits for Three- and Two-Pump Operation

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.

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