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SUBJECT: LTR 3 ENCL 40
FORWARDING LIC NOS DPR-39, 47 & 55 APPL FOR AMEND: TECH SPEC PROPOSED
CHANGE CONCERNING REVISION TO THE CORE PROTECTION SAFETY LIMITS AND
PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS, IN SUPPORT OF OPERATION FOR
FULL RATED PWR DURING CYCLE 4...W/ATT R _ BAW-1491, Oconee Unit 2, cycle 4
50-270

PLANT NAME: OCONEE - UNIT 1 REVIEWER INITIAL: XJM
OCONEE - UNIT 2 DISTRIBUTER INITIAL: RTW
OCONEE - UNIT 3

****************** DISTRIBUTION OF THIS MATERIAL IS AS FOLLOWS ******************

NOTES:
1. M. CUNNINGHAM -- ALL AMENDMENTS TO FSAR AND CHANGES TO TECH SPECS

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DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

PP

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

September 18, 1978

TELEPHONE: AREA 704
373-4083

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. R. Reid, Chief
Operating Reactors Branch #4

Reference: Oconee Nuclear Station
Docket No. 50-269, -270, -287

Dear Sir:

Pursuant to 10CFR 50, §50.90, please find attached proposed changes to the Oconee Nuclear Station Technical Specifications. These changes are required to support the operation of Oconee Unit 2 at full rated power during Cycle 4. This proposal consist of the following changes. The Core Protection Safety Limits and Protective System Maximum Allowable Setpoints contained in Specifications 2.1 and 2.3 respectively, have been revised. The bases of Specification 2.1 has been revised to reflect the proposed removal of orifice rod assemblies and the subsequent reduction of $F^N_{\Delta H}$ to 1.71. Specification 3.2 has been revised to reflect the boron volume requirements upon startup of Unit 2 for Cycle 4. The bases and Table 3.5-1 of Specification 3.5 have been revised to reflect the new steady state tilt limit of 5%. The proposed Technical Specifications changes also include a change in the xenon reactivity specification (Specification 3.5.2.6) applicable to all three Oconee units. This change in the xenon reactivity specification is based on the consideration of the effect of the transient xenon on power peaking calculated for Oconee 1 Cycle 5, Oconee 2 Cycle 4, and Oconee 3 Cycle 4 and based on confirmation that the proposed limiting condition for xenon reactivity conservatively ensures acceptable power peaks during xenon transients. The applicable data for Oconee 1 Cycle 5 were provided to the NRC in my letter of August 28, 1978. The data for Oconee 2 and 3 are similar and consistent with the Oconee 1 data.

Also attached is Babcock and Wilcox Report BAW-1491, "Oconee Unit 2, Cycle 4 Reload Report." This report includes a summary of Cycle 4 operating parameters and contains the safety analyses supporting the operation of Oconee 2, Cycle 4 core at rated power in accordance with the Technical Specifications provided.

A001
5/3/40
File(2)

On June 8, 1978, Duke Power Company filed a request for exemption from 10CFR50, §50.46 for Oconee Unit 3 based on a staff request. It continues to be our position that the Oconee ECCS is currently acceptable and wholly in conformance with 10CFR50, §50.46. Improvements to the ECCS prior to the startup of Unit 2 following the forthcoming refueling are neither necessary nor feasible. However, in response to a staff requirement and pursuant to 10CFR50, §50.12, it is hereby requested that an exemption be granted to the provision of 10CFR 50, §50.46 and that Oconee Unit 2 be licensed to operate at full rated core thermal power (2568 MW_t).

This proposed revision to the Oconee Nuclear Station Technical Specifications is considered to involve a single safety issue with no significant hazards involved and two duplicate amendments for the identical Oconee units. Accordingly, a check in the amount of $4,800 is attached, consisting of the licensing fees for one Class III amendment and two Class I amendments. Three signed originals and thirty-seven copies are provided in this submittal.

Very truly yours,

William O. Parker, Jr.

RLG:scs
Attachments (40)

Mr. Harold R. Denton, Director
Page Three
September 18, 1978

WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker, Jr.
William O. Parker, Jr., Vice President

Subscribed and sworn to before me this 18th day of September, 1978.

Vivian B. Hopkins
Notary Public

My Commission Expires:

February 15, 1982

Attachment 1

Proposed Technical Specification

Revision Pages

Bases - Unit 2

The safety limits presented for Oconee Unit 2 have been generated using BAW-2 critical heat flux correlation[1] and the Reactor Coolant System flow rate of 106.5 percent of the design flow (design flow is 352,000 gpm for four-pump operation). The flow rate utilized is conservative compared to the actual measured flow rate[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 1.30. A DNBR of 1.30 corresponds to a 95 percent probability at a 95 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 represents the conditions at which a minimum DNBR of 1.30 is predictecd for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 374,880 gpm). This curve is based on the following nuclear power peaking factors with potential fuel densification and fuel rod bowing effects:

$F_q^N = 2.565$; $F_{\Delta H}^N = 1.71$[3] $F_z^N = 1.50$

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

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

1. The 1.30 DNBR limit produced by the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 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 for Unit 2.

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, and 3 of Figure 2.1-2B correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

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

The maximum thermal power for three-pump operation is 85.3 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.055 = 78.8 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar manner.

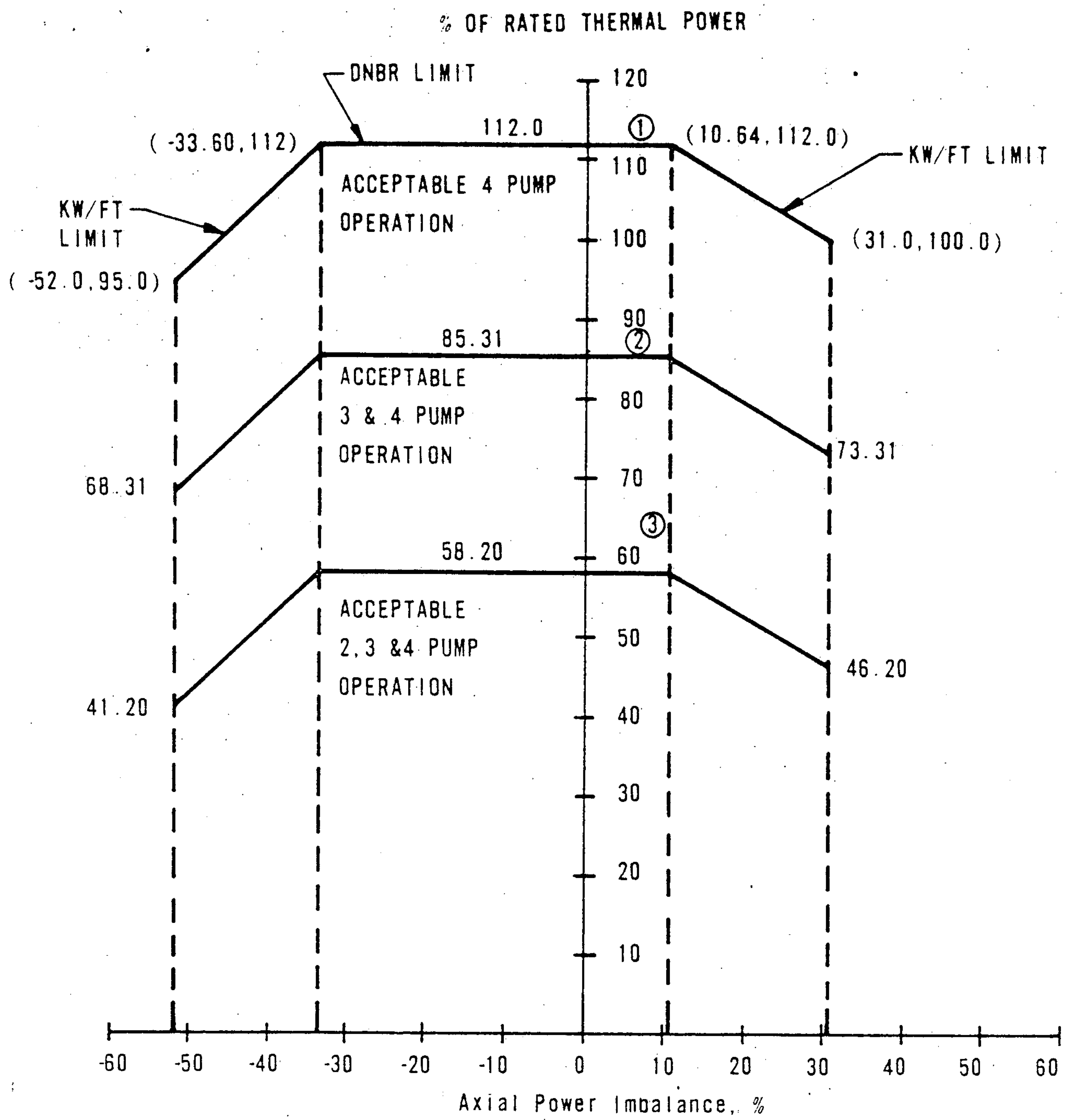
For each curve of Figure 2.1-3B, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The 1.30 DNBR curve for four-pump operation 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 2, Cycle 3 - Reload Report - BAW-1452, April, 1977.

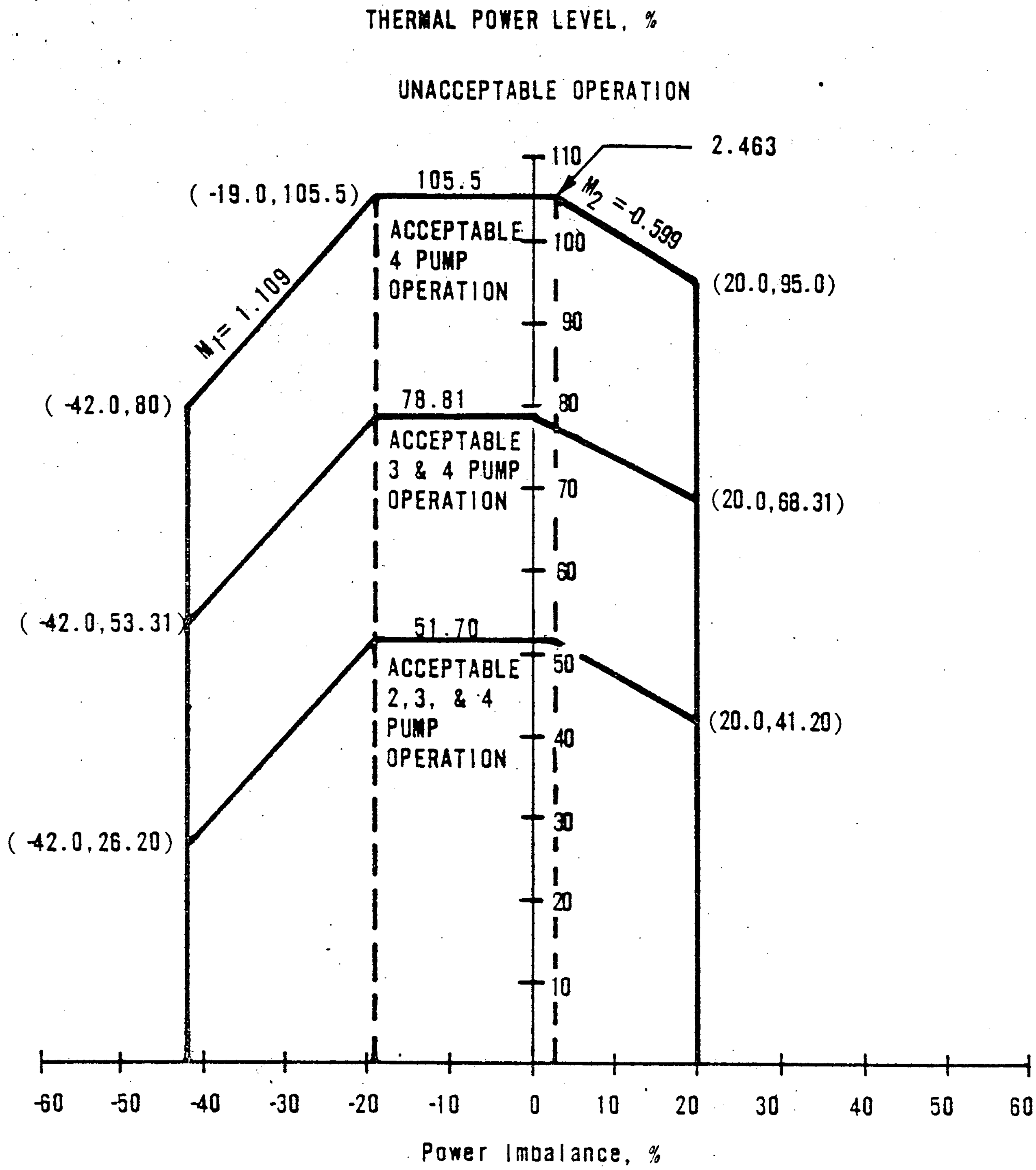
(3) Oconee 2, Cycle 4 - Reload Report - BAW-1491, August, 1978.



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

CORE PROTECTION
SAFETY LIMITS
UNIT 2

OCONEE NUCLEAR STATION
Figure 2.1-2B



PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 2

OCONEE NUCLEAR STATION

Figure 2.3-2B

3.2 HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the high pressure injection and the chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

Specification

The reactor shall not be critical unless the following conditions are met:

3.2.1 Two high pressure injection pumps per unit are operable except as specified in 3.3.

3.2.2 One source per unit of concentrated soluble boric acid in addition to the borated water storage tank is available and operable.

This source will be the concentrated boric acid storage tank containing at least the equivalent of 995 ft^3 of 8700 ppm boron as boric acid solution with a temperature at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the high pressure injection system shall be operable and shall have the same temperature requirement as the concentrated boric acid storage tank. At least one channel of heat tracing capable of meeting the above temperature requirement shall be in operation. One associated boric acid pump shall be operable.

If the concentrated boric acid storage tank with its associated flowpath is unavailable, but the borated water storage tank is available and operable, the concentrated boric acid storage tank shall be restored to operability within 72 hours or the reactor shall be placed in a hot shutdown condition and be borated to a shutdown margin equivalent to 1% $\Delta k/k$ at 200°F within the next twelve hours; if the concentrated boric acid storage tank has not been restored to operability within the next 7 days the reactor shall be placed in a cold shutdown condition within an additional 30 hours.

If the concentrated boric acid storage tank is available but the borated water storage tank is neither available nor operable, the borated water storage tank shall be restored to operability within one hour or the reactor shall be placed in a hot shutdown condition within 6 hours and in a cold shutdown condition within an additional 30 hours.

Bases

The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration.(1) This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or the concentrated boric acid storage tank. An alternate method of boration will be the use of the high pressure injection pumps taking suction directly from the borated water storage tank.(2)

The quantity of boric acid in storage in the concentrated boric acid storage tank or the borated water storage tank is sufficient to borate th reactor coolant system to a 1% Δk/k subcritical margin at cold conditions (70°F) with the maximum worth stuck rod and no credit for zenon at the worst time in core life. The current cycles for each unit, Oconee 1 Cycle 5, Oconee 2, Cycle 4, and Oconee 3, Cycle 4 were analyzed with the most limiting case selected as the basis for all three units. Since only the present cycles were analyzed, the specifications will be re-evaluated with each reload. A minimum of 995 ft^3 of 8,700 ppm boric acid in the concentrated boric acid storage tank, or a minimum of 350,000 gallongs of 1800 ppm boric acid in the borated water storage tank (3) will satisfy the requirements. The volume requirements include a 10% margin and in addition allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition. The required amount of boric acid can be added in several ways. Using only one 10 gpm boric acid pump taking suction from the concentrated boric acid storage tank would require approximately 12.25 hours to inject the required boron. An alternate method of addition is to inject boric acid from the borated water storage tank using the makeup pumps. The required boric acid can be injected in less than six hours using only one of the makeup pumps.

The concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept at least 10°F above the crystallization temperature for the concentration present. The boric acid concentration of 8,700 ppm in the concentrated boric acid storage tank corresponds to a crystallization temperature of 77°F and therefore a temperature requirement of 87°F. Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

(1) FSAR, Section 9.1; 9.2
(2) FSAR, Figure 7.2
(3) Technical Specification 3.3

3.5.2.5 Control Rod Positions

a. Technical Specification 3.1.3.5 does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

b. Except for physics test, operating rod group overlap shall be 25% ± 5% between two sequential groups. If this limit is exceeded, corrective measures shall be taken immediately to achieve an acceptable overlap. Acceptable overlap shall be attained within two hours or the reactor shall be placed in a hot shutdown condition within an additional 12 hours.

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits are specified on figures 2.5.2-1A1 and 3.5.2-1A2 (Unit 1); 3.5.2-1B1, 3.5.2-1B2 and 3.5.2-1B3 (Unit 2); 3.5.2-1C1, 3.5.2-1C2 and 3.5.2-1C3 (Unit 3) for four pump operation, and on figures 3.5.2-2A1 and 3.5.2-2A2 (Unit 1); 3.5.3-2B1, 3.5.2-2B2 and 3.5.2-2B3 (Unit 2); 3.5.2-2C1, 3.5.2-2C2 and 3.5.2-2C3 (Unit 3) for two or three pump operation. Also, excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on figures 3.5.2-4A1, and 3.5.2-4A2 (Unit 1); 3.5.2-4B1, 3.5.2-4B2, and 3.5.2-4B3 (Unit 2); 3.5.2-4C1, 3.5.2-4C2, and 3.5.2-4C3 (Unit 3).

   If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. An acceptable control rod position shall then be attained within two hours. The minimum shutdown margin required by Specification 3.5.2.1 shall be maintained at all times.

3.5.2.6 Xenon Reactivity

Except for physics tests, reactor power shall not be increased above the power-level-cutoff shown in Figures 3.5.2-1A1, and 3.5.2-1A2 for Unit 1; Figures 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3 for Unit 2; and Figures 3.5.2-1C1, 3.5.2-1C2, and 3.5.2-1C3 for Unit 3 unless one of the following conditions is satisfied:

1. Xenon reactivity did not deviate more than 10 percent from the equilibrium value for operation at steady state power.

2. Xenon reactivity deviated more than 10 percent but is now within 10 percent of the equilibrium value for operation at steady state rated power and has passed its final maximum or minimum peak during is approach to its equilibrium value for operation at the power level cutoff.

3. Except for xenon free startup (when 2. applies), the reactor has operated within a range of 87 to 92 percent of rated thermal power for a period exceeding 2 hours.

3.5.2.7 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2, and 3.5.2-3C3. If the imbalance is not within the envelope defined by these figures, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.8 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate.

Bases

The power-imbalance envelope defined in Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3B3, 3.5.2-3C1, 3.5.2-3C2 and 3.5.2-3C3 is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification power spike factors (Units 1 and 2 only)
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing power spike factors

The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Xenon transient override
8	APSR (axial power shaping bank)

** Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument calibration errors. The method used to define the operating limits is defined in plant operating procedures.

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position(1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2,3,4,5) of hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Groups 6 and 7 to be partially inserted.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding 7.50% for Unit 1. The limits shown in Specification 3.5.2.4
7.50% for Unit 2
7.50% for Unit 3
are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

The quadrant tilt and axial imbalance monitoring in Specification 3.5.2.4 and 3.5.2.6, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Operating restrictions are included in Technical Specification 3.5.2.6 to prevent excessive power peaking by transient xenon. For Unit 1, a 5% peaking increase is applied to calculated peaks at equilibrium conditions for powers above the power level cutoff. For Units 2 and 3, an 8% peaking increase is applied. These values conservatively bound the peaking effects of transient xenon once the applicable requirement of 3.5.2.6 has been satisfied.

REFERENCES

[1]FSAR, Section 3.2.2.1.2

[2]FSAR, Section 14.2.2.2

[3]FSAR, SUPPLEMENT 9

[4]B&W FUEL DENSIFICATION REPORT

BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

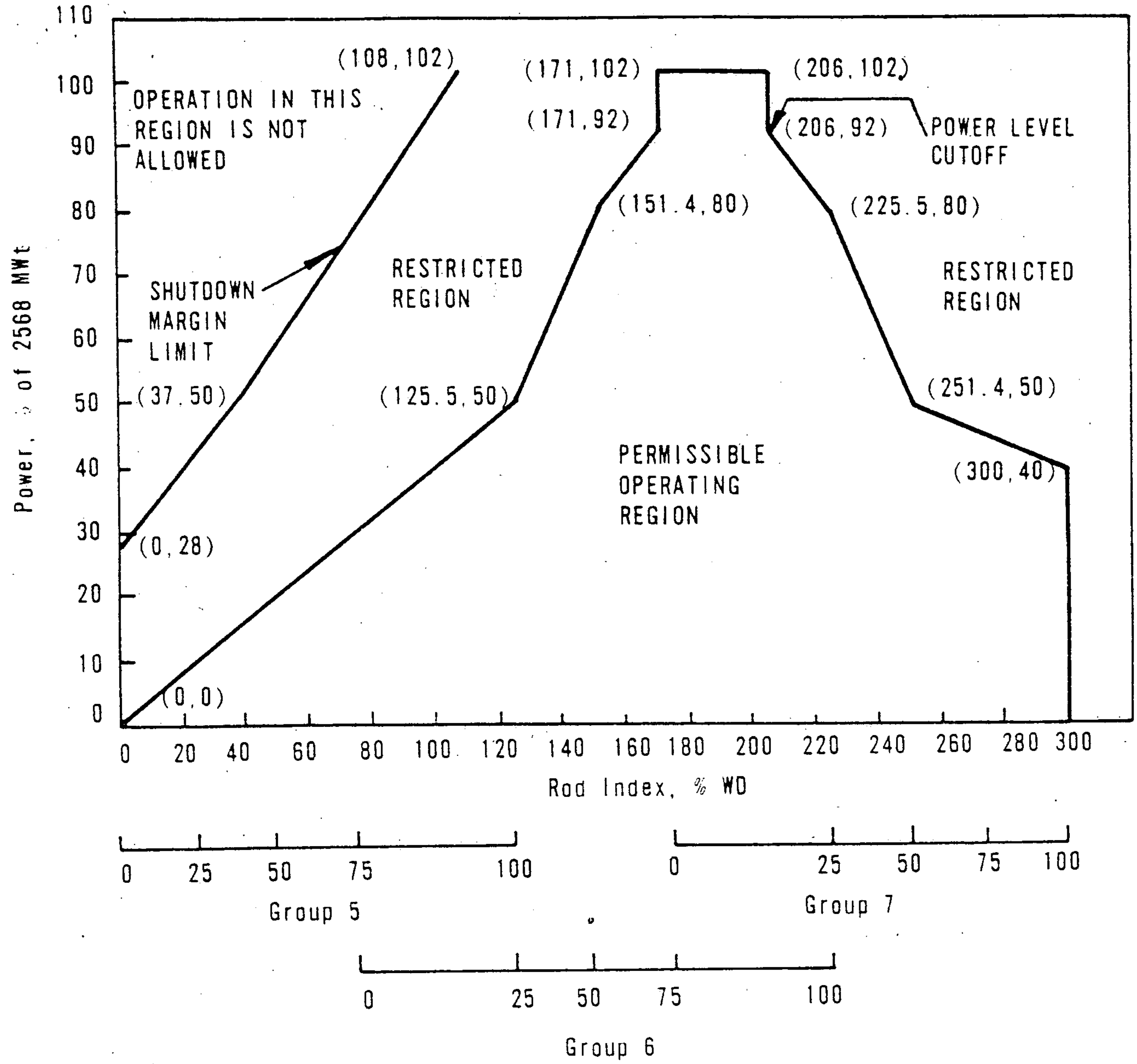
BAW-1400 (UNIT 3)

[5]Oconee 1, Cycle 4 - Reload Report - BAW 1447, March 1977, Section 7.11

TABLE 3.5-1

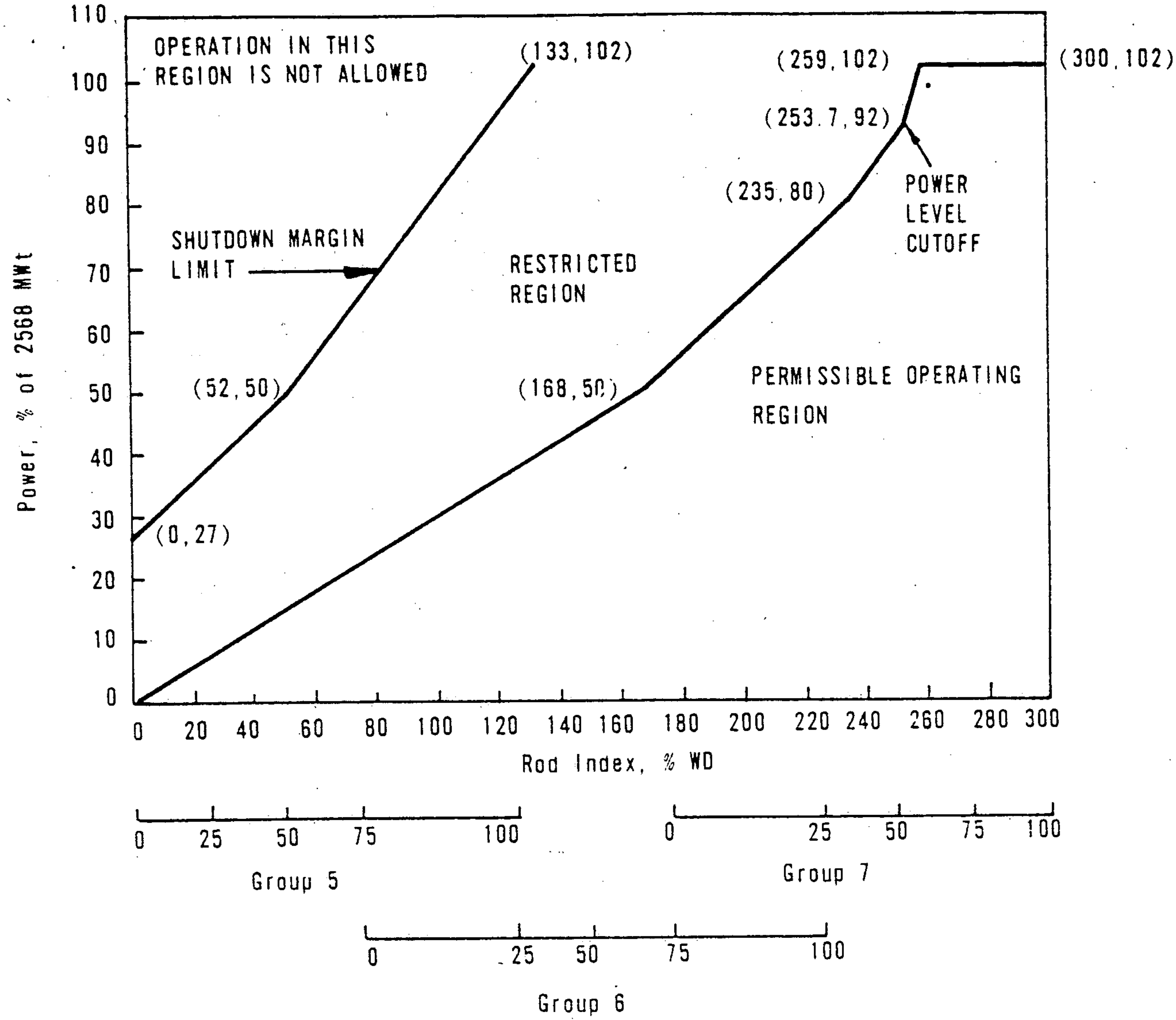
Quadrant Power Tilt Limits

	Steady State Limit	Transient Limit	Maximum Limit
Unit 1	5.00	9.44	20.0
Unit 2	5.00	9.44	20.0
Unit 3	5.00	9.44	20.0



ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 250 $\pm$ EFPD
OCONEE 2

OCONEE NUCLEAR STATION
Figure 3.5.2-1B1

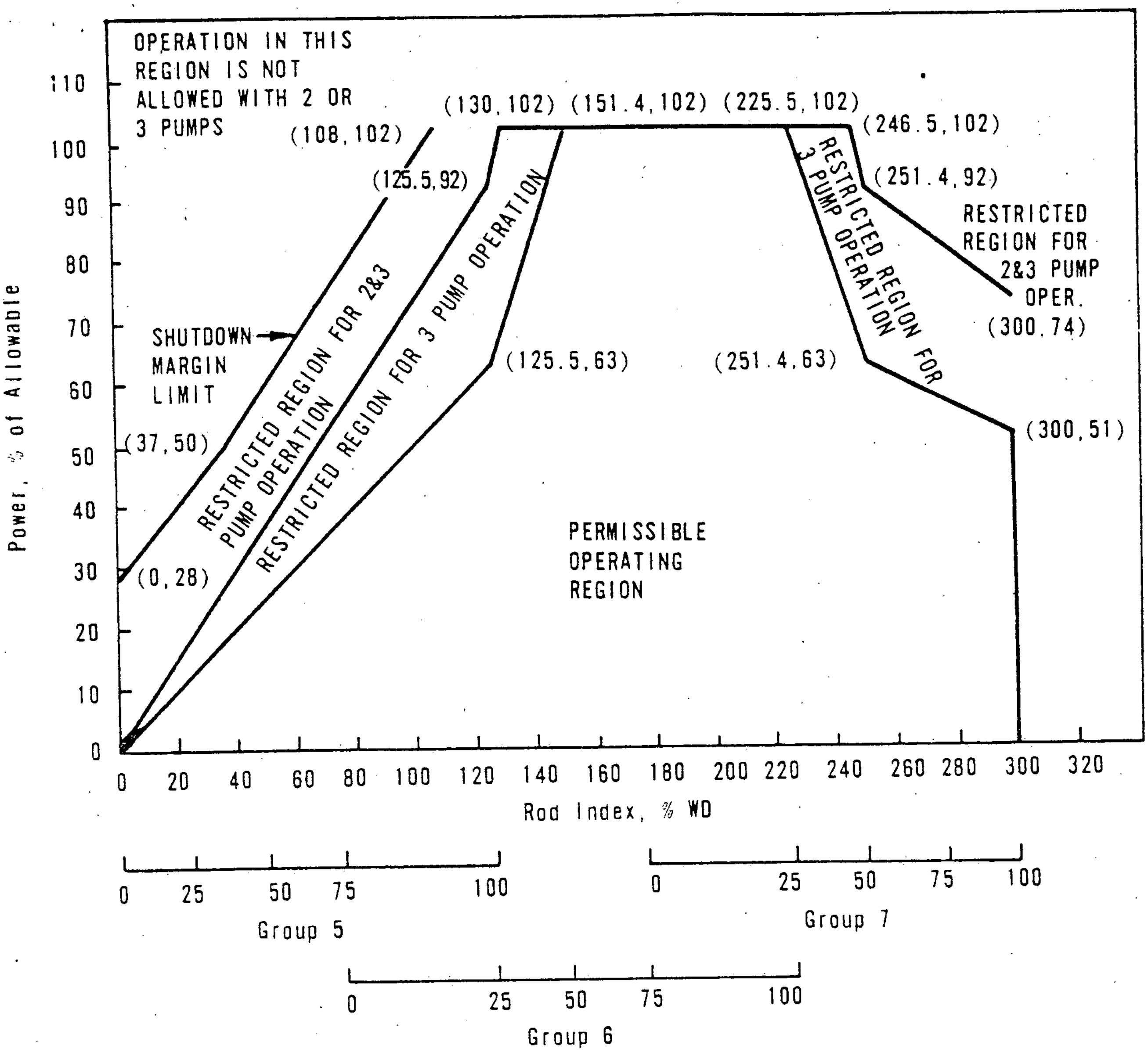


ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
AFTER 250 $\pm$ EFPD
OCONEE 2

OCONEE NUCLEAR STATION

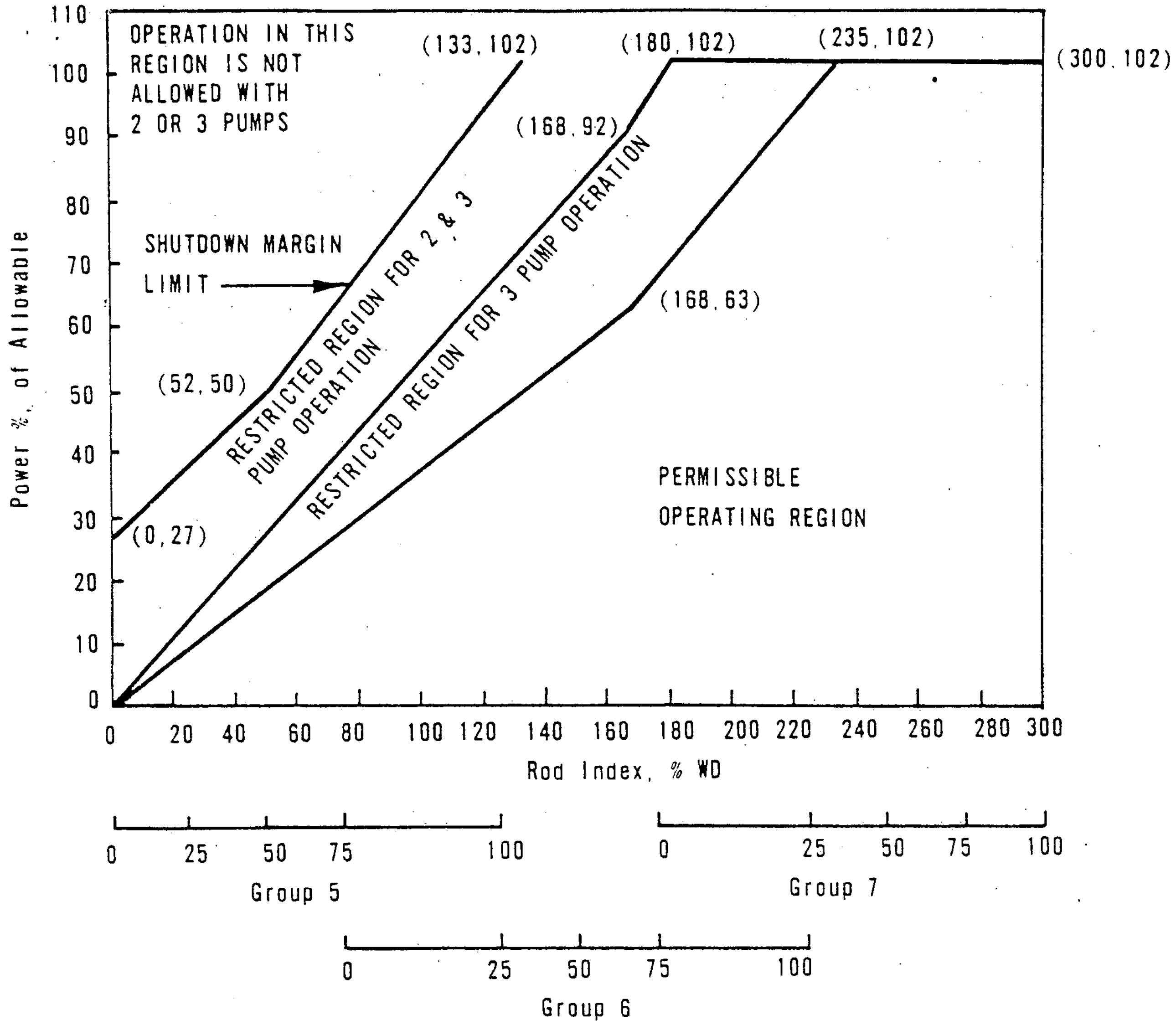
Figure 3.5.2-1B2

Figure 3.5.2-1B3
Deleted During Oconee Unit 2, Cycle 4 Operation



ROD POSITION LIMITS
FOR TWO AND THREE PUMP OPERATION
FROM 0 TO 250 ± 10 EFPD
OCONEE 2

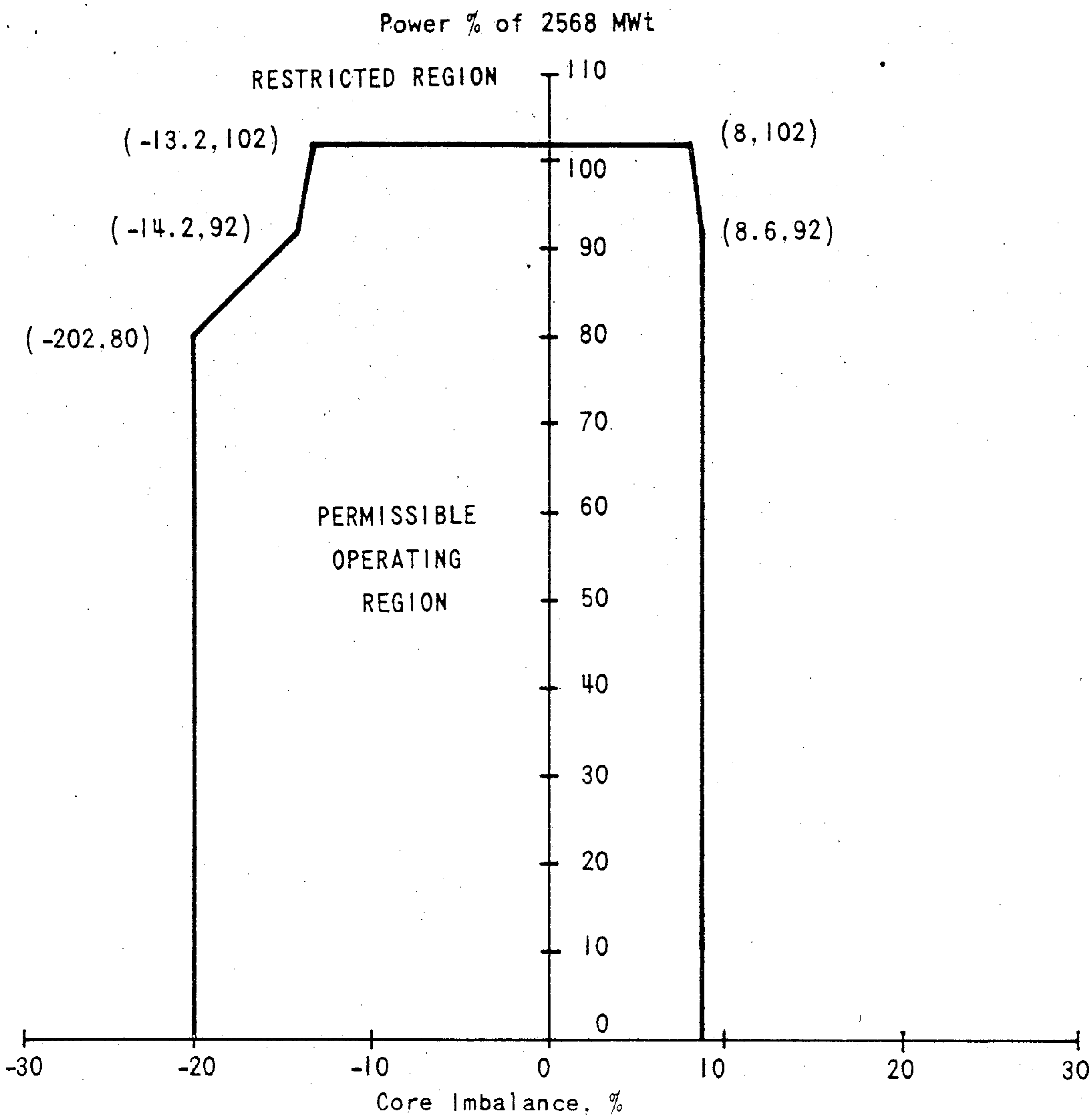
OCONEE NUCLEAR STATION
Figure 3.5.2-2B1



ROD POSITION LIMITS
FOR TWO AND THREE PUMP OPERATION
AFTER 250 $\pm$ 10 EFPD
OCONEE 2

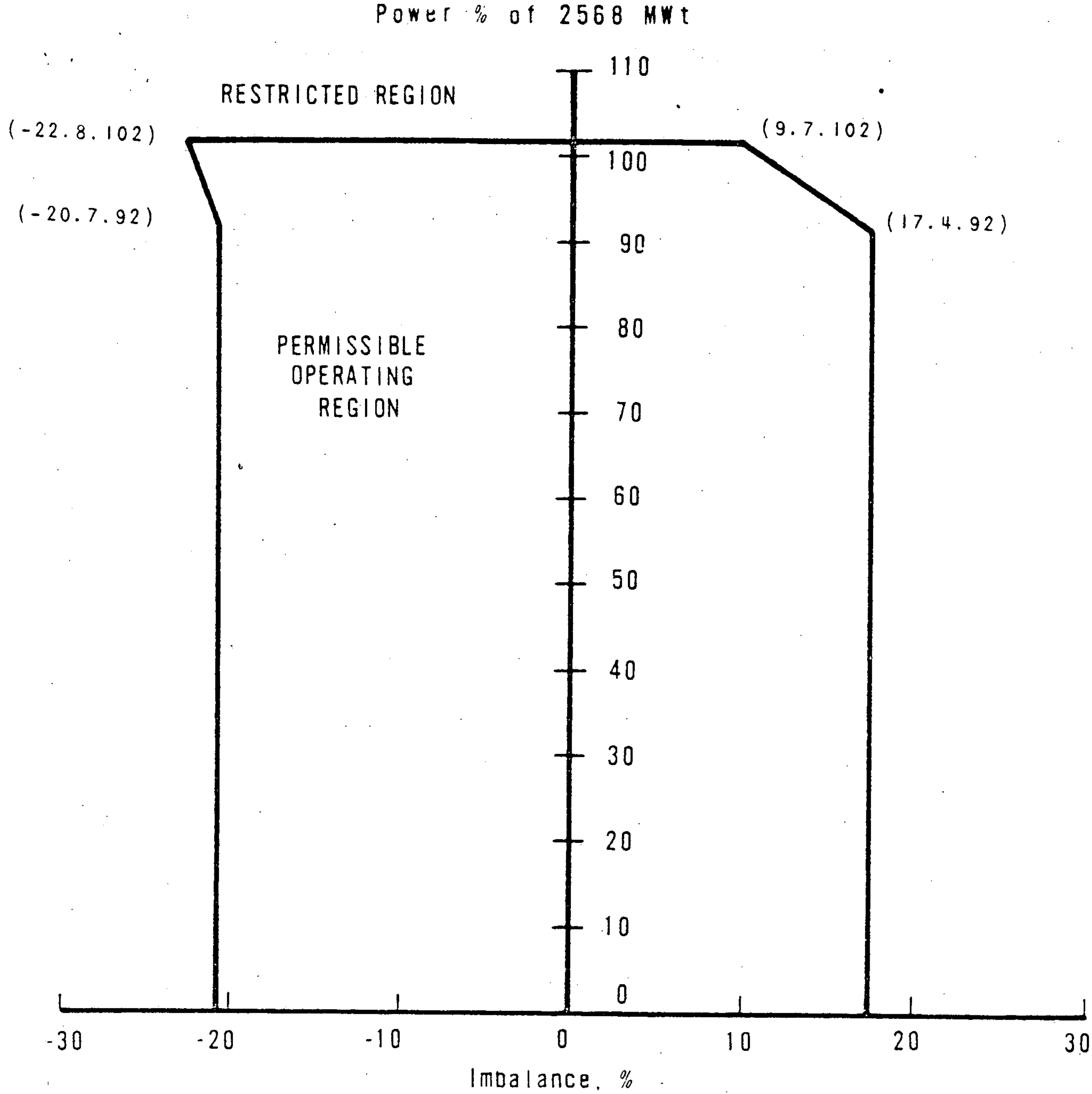
OCONEE NUCLEAR STATION
Figure 3.5.2-2B2

Figure 3.5.2-2B3
Deleted during Oconee Unit 2, Cycle 4 Operation



OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION FROM 0 TO 250 ± 10 EFPD
OCONEE 2

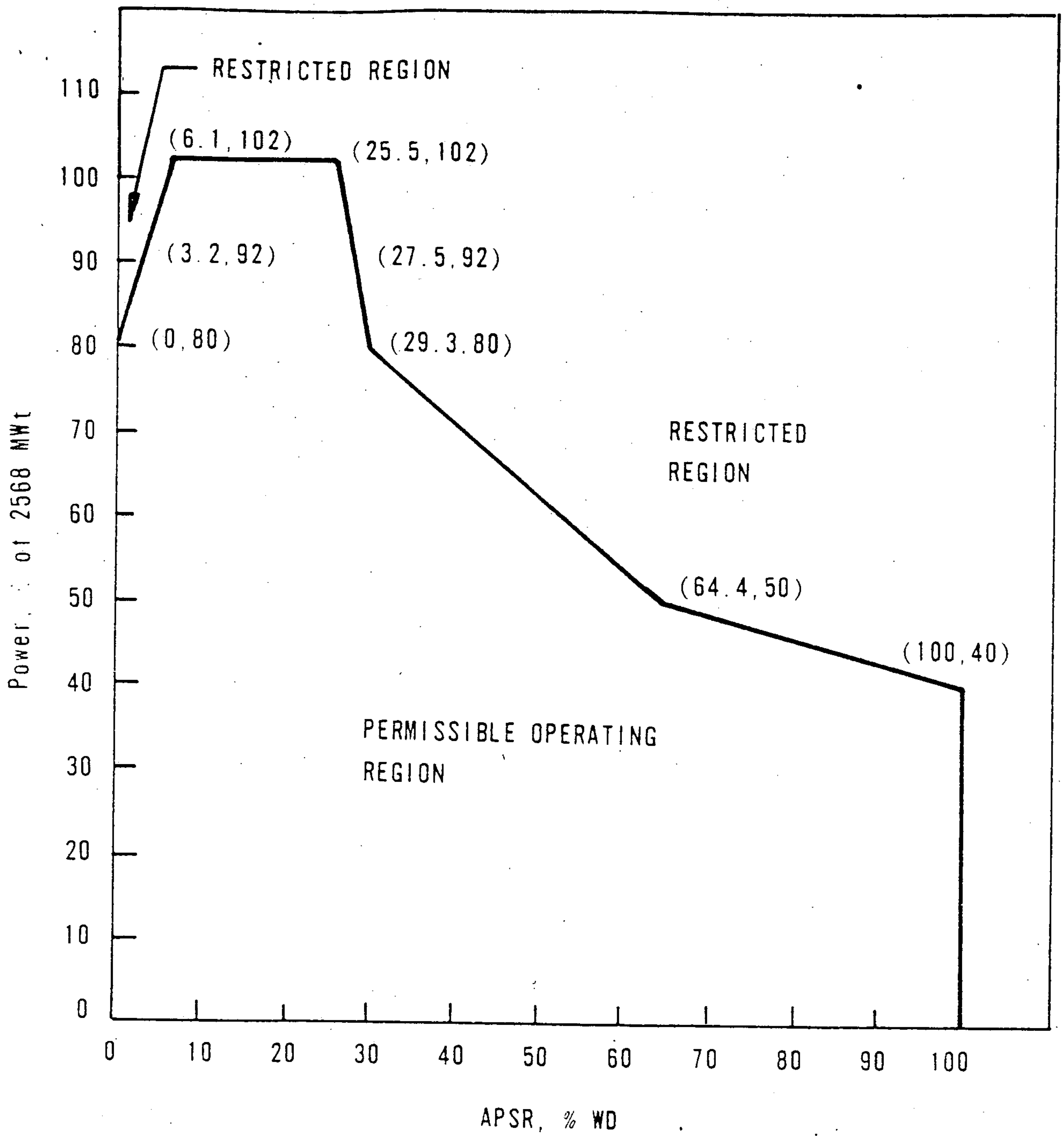
OCONEE NUCLEAR STATION
Figure 3.5.2-3B1



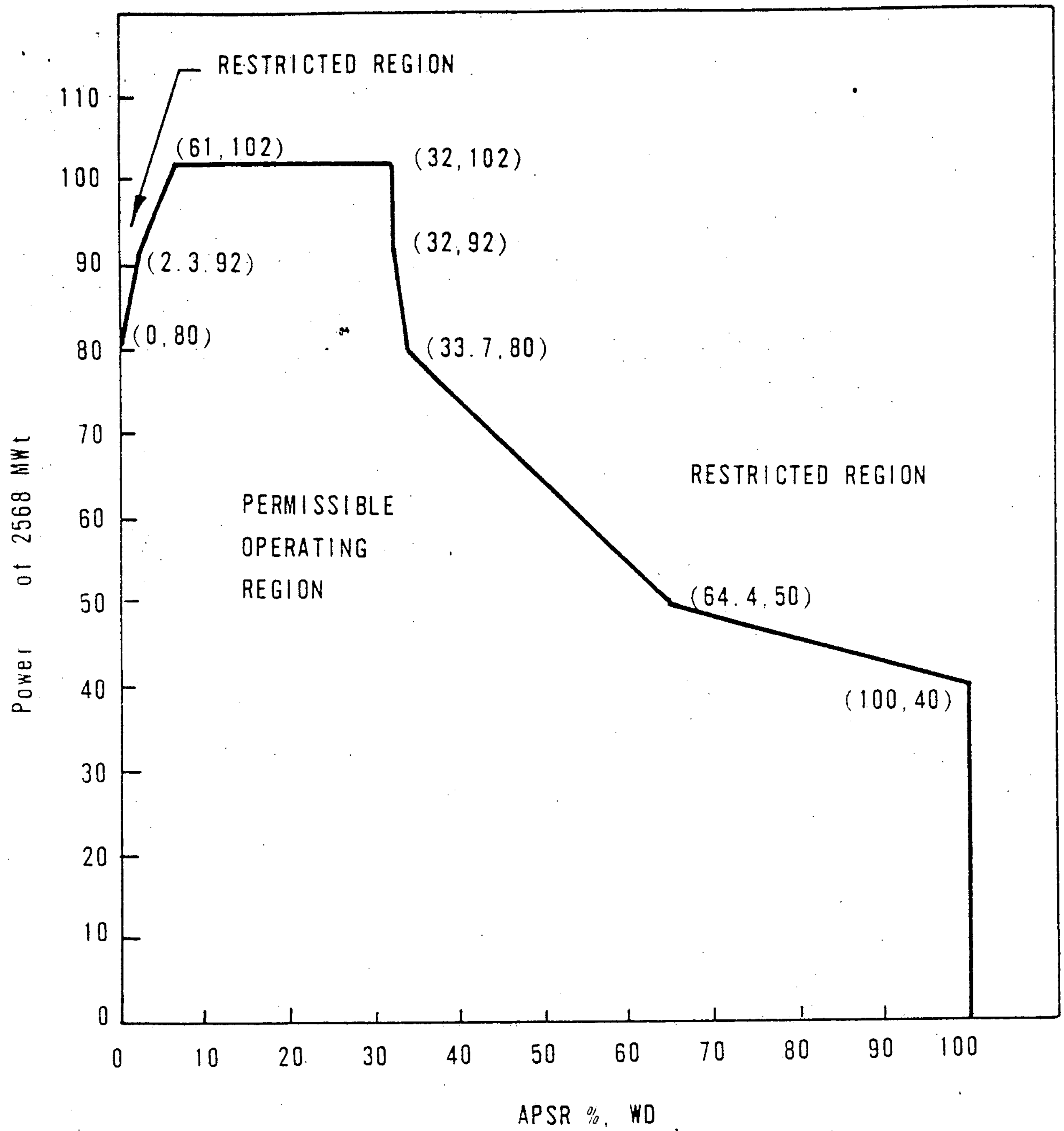
OPERATIONAL POWER IMBALANCE ENVELOPE
FOR OPERATION AFTER 250 ± 10 EFPD
OCONEE 2

OCONEE NUCLEAR STATION
Figure 3.5.2-3B2

Figure 3.5.2-3B3
Deleted During Oconee Unit 2, Cycle 4 Operation



APSR POSITION LIMITS
FOR OPERATION
FROM 0 TO 250 ± 10 EFPD
OCONEE 2
OCONEE NUCLEAR STATION
Figure 3.5.2-4B1



APSR POSITION LIMITS FOR OPERATION
AFTER 250 ± 10 EFPD
OCONEE 2
OCONEE NUCLEAR STATION
Figure 3.5.2-4B2

Figure 3.5.2-4B3
Deleted during Oconee Unit 2, Cycle 4 Operation

Attachment 2

Oconee Unit 2, Cycle 4

Reload Report

BAW-1491

BAW-1491

August 1978

OCONEE UNIT 2, CYCLE 4

— Reload Report —

Babcock & Wilcox

BAW-1491
August 1978

OCONEE UNIT 2, CYCLE 4

— Reload Report —

BABCOCK & WILCOX
Power Generation Group
Nuclear Power Generation Division
P. O. Box 1260
Lynchburg, Virginia 24505

Babcock & Wilcox

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1. INTRODUCTION AND SUMMARY

This report justifies the operation of the fourth cycle of Oconee Nuclear Station, Unit 2, 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 4 operation of Oconee Unit 2, 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 3 and 4 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 4 operation. In those cases where cycle 4 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 4 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 2 can be operated safely for cycle 4 at the rated power level of 2568 MWt.

Because of performance anomalies observed at other B&W plants, orifice rod assemblies will not be used in Oconee 2, cycle 4. This change from normal practice has been accounted for in the analyses performed for cycle 4. In addition, retainer assemblies will be installed on two fresh batch 6 fuel assemblies containing regenerative neutron sources.

2. OPERATING HISTORY

The reference fuel cycle for the nuclear and thermal-hydraulic analyses of the fourth cycle of Oconee Nuclear Station Unit 2 is the currently operating cycle 3. Cycle 3 achieved initial criticality on August 26, 1977, and was escalated to 40% power on August 28, 1977. The 100% power level of 2568 MWt was reached on January 18, 1978. No operating anomalies occurred during cycle 3 operation that would adversely affect the fuel performance in cycle 4 during the design length of 292 EFPD. No control rod interchanges are planned for cycle 4. Control rod group 7 will be withdrawn at 250 (±10) EFPD of operation.

3. GENERAL DESCRIPTION

The Oconee 2 reactor core is described in detail in Chapter 3 of the FSAR.[1] The core consists of 177 fuel assemblies, 173 of which have 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 other four FAs in cycle 4 are demonstration 17 by 17 FAs — two Mark C and two Mark CR. All fuel assemblies in cycle 4 except the 17 × 17 demonstration assemblies maintain a constant 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 included in Tables 4-1 and 4-2 of this report.

Figure 3-1 is the core loading diagram for Oconee 2, cycle 4. Nine once-burned batch 1 assemblies with an initial enrichment of 2.06 wt % ^{235}U will be reloaded into the core. Batches 4, 5, and 5A with initial enrichments of 2.64, 3.03, and 2.53 wt % ^{235}U, respectively, will be shuffled to new locations. Batch 6, with an initial enrichment of 2.91 wt % ^{235}U, will occupy the core periphery and eight interior locations. Figure 3-2 is an eighth-core map showing the assembly burnup and enrichment distribution at the beginning of cycle 4.

Reactivity control is supplied by 61 full-length, Ag-In-Cd control rods and soluble boron shim. 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 4 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 4 are identical to those of the initial cycle described in Chapter 3 of the FSAR.[1] However, the group designations differ between cycle 4 and the first cycle[1] to minimize power peaking.

The nominal system pressure is 2200 psia, and the densified nominal heat rate is 5.80 kW/ft at the rated core power of 2568 MWt.

Figure 3-1. Core Loading Diagram — Oconee 2, Cycle 4

X   Fuel Transfer Canal →

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						6	6	6	6	6					
B				6	6	6	D6 4	D7 4	D10 4	6	6	6			
C			6	R10 5	G15 5	M9 4	N3 5	L8 4	N13 5	M7 4	G1 5	R6 5	6		
D		6	L15 5	CY 1 L12 1B	M10 4	N2 5	C7 4	6	C9 4	N14 5	M6 4	CY 1 N6 1B	L1 5	6	
E		6	R7 5	L11 4	H15 5	P5 5	L2 5	CY 1 05 1B	L14 5	P11 5	R8 5	L5 4	R9 5	6	
F	6	6	K11 4	B12 5	E14 5	M11 4	F7 4	H4 5A	F9 4	M5 4	MK CR E2 5	B4 5	K5 4	6	6
G	6	F4 4	C12 5	G3 4	B10 5	G6 4	6	03 4	6	G10 4	B6 5	G13 4	C4 5	F12 4	6
H (W — — Y)	6	K9 4	H10 4	6	CY 1 011 1B	D8 5A	MKC 013 4	CY 1 M3 1B	MKC C3 4	N8 5A	CY 1 C5 1B	6	H6 4	G7 4	6
K	6	L4 4	012 5	K3 4	P10 5	K6 4	6	C13 4	6	K10 4	P6 5	K13 4	04 5	L12 4	6
L	6	6	G11 4	P12 5	MK CR M14 5	E11 4	L7 4	H12 5A	L9 4	E5 4	M2 5	P4 5	G5 4	6	6
M		6	A7 5	F11 4	A8 5	B5 5	F2 5	CY 1 C11 1B	F14 5	B11 5	H1 5	F5 4	A9 5	6	
N		6	F15 5	CY 1 D10 1B	E10 4	D2 5	07 4	6	09 4	D14 5	E6 4	CY 1 F4 1B	F1 5	6	
O			6	A10 5	K15 5	E9 4	D3 5	F8 4	D13 5	E7 4	K1 5	A6 5	6		
P				6	6	6	N6 4	G9 4	N10 4	6	6	6			
R						6	6	6	6	6					

Z

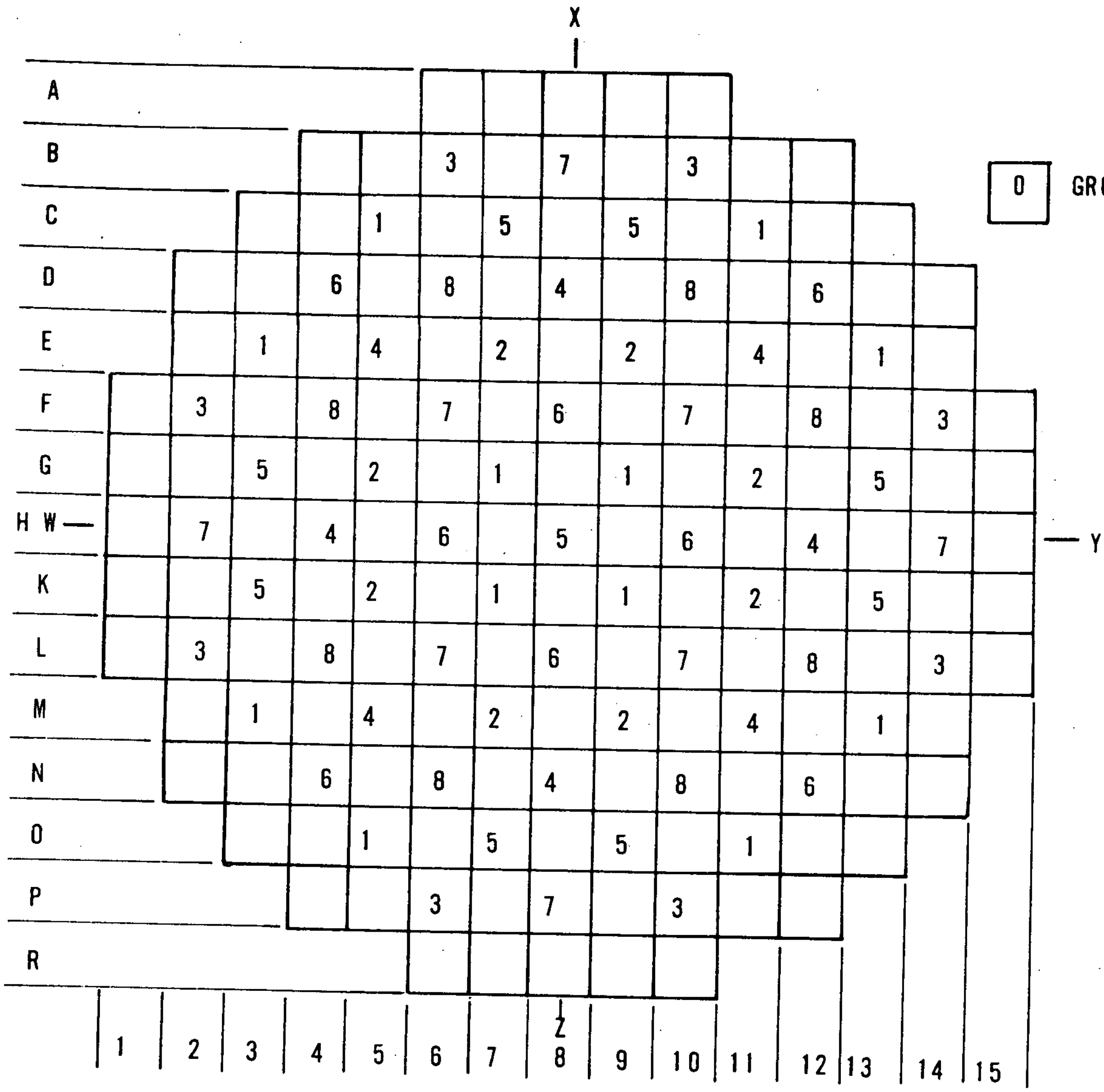
xx	Cycle 3 Location Unluess Otherwise Noted
x	Batch

Figure 3-2. Oconee 2, Cycle 4 Enrichment and Burnup Distribution

	8	9	10	11	12	13	14	15
H	2.06 12,965	2.64 12,152	2.53 12,188	2.06 12,965	2.91 0	2.64 18,296	2.64 18,057	2.91 0
K		2.91 0	2.64 21,460	3.03 11,214	2.64 18,505	3.03 11,222	2.64 17,448	2.91 0
L			2.64 16,482	3.03 9,769	3.03 7,325	2.64 17,514	2.91 0	2.91 0
M				3.03 7,326	2.64 17,027	3.03 7,844	2.91 0	
N					2.06 13,724	3.03 7,082	2.91 0	
O						2.91 0		
P								
R								

x.xx	Initial Enrichment, wt % ^{235}U
xx,xxx	BOC Burnup, MWd/mtU

Figure 3-3. Control Rod Locations for Oconee 2, Cycle 4



GROUP	NO. OF RODS	FUNCTION
1	12	SAFETY
2	8	SAFETY
3	8	SAFETY
4	8	SAFETY
5	9	CONTROL
6	8	CONTROL
7	8	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 2, cycle 4 are listed in Table 4-1. All Mark B fuel assemblies are identical in concept and are mechanically interchangeable. The Mark CR demonstration assemblies of batch 5 are mechanically identical in function to the Mark C demonstration assemblies of batch 4. All results, references, and identified conservatisms presented in section 4.1 of the previous Oconee 2 reload report[2] are applicable to the cycle 4 reload core.

4.2. Fuel Rod Design

4.2.1. Cladding Collapse

Creep collapse analyses were performed for three-cycle assembly power histories. The batch 4 fuel is more limiting than the other batches becauses of its previous incore exposure time. The batch 4 assembly power histories were analyzed and the most limiting Mark B and Mark C assemblies determined. The power histories of these assemblies were used to calculate the fast neutron flux level for the energy range above 1 MeV. Both Mark B and Mark C creep collapse times were determined to be more than 30,000 EFPH (effective full-power hours), which is longer than their maximum projected incore residence time (see Table 4-1). The creep collapse analyses were performed based on the conditions set forth in references 2 and 3.

4.2.2. Cladding Stress

The batch 1 reinserted fuel is the most limiting batch from a cladding stress viewpoint owing to its lower prepressurization and density. The batch 1 fuel has been analyzed and documented in the Oconee 2 Fuel Densification Report.[4]

4.2.3. Cladding Strain

The fuel design criteria specify a limit of 1.0% on cladding plastic circumferential strain. The pellet design is established for cladding plastic strain

of less than 1% at maximum values of design local pellet burnup and heat generation rate, which are considerably higher than the values the Oconee 2 fuel is expected to see. 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 6 fuel inserted for cycle 4 operation introduces no significant differences in fuel thermal performance relative to the batch 3 fuel discharged at the end of cycle 3. The design minimum linear heat rate (LHR) capacity and the average fuel temperature for each batch in cycle 4 are shown in Table 4-2. LHR limitations were established using the TAFY code[5] with fuel densification to 96.5% of theoretical density.

4.4. Material Design

The batch 6 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 6 fuel assemblies are identical to those of the present fuel.

4.5. Operating Experience

Babcock & Wilcox operating experience with the Mark B 15 × 15 fuel assembly has verified the adequacy of its design. As of April 30, 1978, the experience described by Table 4-3 has been accumulated for the eight operating B&W 177-fuel assembly plants using the Mark B fuel assembly.

Table 4-1. Fuel Design Parameters and Dimensions

	Twice-burned FAs		Once-burned FAs			
	Batch 4	Mark C demo	Batch 5/5A	Batch 1B	Mark CR demo	Fresh FAs, batch 6
Fuel assembly type	Mark B-4	17×17 array	Mark B-4	Mark B-2	17×17 array	Mark B-4
No. of assemblies	54	2	50/4	9	2	56
Fuel rod OD, in.	0.430	0.379	0.430	0.430	0.379	0.430
Fuel rod ID, in.	0.377	0.332	0.377	0.377	0.332	0.377
Flexible spacers, type	Spring	Spring	Spring	Corrugated	Spring	Spring
Rigid spacers, type	Zr-4	Zr-4	Zr-4	ZrO_2	Zr-4	Zr-4
Undensified active fuel length, in.	142.6	143.0	142.25	144.0	143.0	142.25
Fuel pellet OD (mean specified), in.	0.370	0.324	0.3695	0.370	0.324	0.3695
Fuel pellet initial density, % TD	93.5	94.0	94.0	92.5	94.0	94.0
Initial fuel enrichment, wt % ^{235}U	2.64	2.64	3.03/2.53	2.06	3.03	2.91
BOC burnup, MWd/mtU	17,778	12,152	8,941/12,188	13,302	9,769	0
Cladding collapse time, EFPH	>30,000	>30,000	>30,000	>30,000	>30,000	>30,000
Estimated residence time (max), EFPH	21,576	21,576	24,576	17,808	24,576	27,000

Table 4-2. Fuel Thermal Analysis Parameters

	Batch 1B	Batch 4	Batch 5	Batch 5A	Batch 6
No. of assemblies	9	56[(a)]	52[(b)]	4	56
Initial density, % TD	92.5	93.5	94.0	94.0	94.0
Pellet diameter, in.	0.370	0.370	0.3695	0.3695	0.3695
Stack height, in.	144	142.6	142.25	142.25	142.25
Densified Fuel Parameters[(c)]					
Pellet diameter, in.	0.3632	0.3645	0.3646	0.3646	0.3646
Fuel stack height, in.	141.1	140.5	140.5	140.5	140.5
Nominal LHR at 2568 MWt, kW/ft	5.77	5.80	5.80	5.80	5.80
Avg fuel temp at nominal LHR, F	1335	1320	1320	1320	1320
LHR capability (centerline fuel melt), kW/ft	19.8	20.15	20.15	20.15	20.15

(a) Includes two Mark C (17×17) demonstration assemblies.

(b) Includes two Mark CR (17×17) demonstration assemblies.

(c) Densification to 96.5% TD assumed.

Table 4-3. Operating Experience Using Mark B Fuel Assembly

Reactor	Current cycle	Max assembly burnup, MWd/mtU		Cumulative net electrical output, MWh
		Incore	Discharged	
Oconee 1	4	28,500	25,300	21,413,802
Oconee 2	3	28,000	26,800	16,081,241
Oconee 3	3	27,800	27,200	17,415,480
TMI-1	4	24,700	32,200	18,710,670
ANO-1	3	24,400	28,300	14,705,349
Rancho Seco	2	23,700	17,170	11,088,121
Crystal River 3	1	10,700	--	4,978,690
Davis Besse 1	1	4,000	--	1,734,732

5. NUCLEAR DESIGN

5.1. Physics Characteristics

Table 5-1 compares the core physics parameters of cycles 3 and 4. The values for both cycles were generated using PDQ07. (The similarity between the two is to be expected since the core has nearly reached an equilibrium cycle.)

The accumulated average core burnup will be lower in cycle 4 than in cycle 3 because of the shorter cycle lifetime. Figure 5-1 illustrates a representative relative power distribution for the beginning of the fourth cycle at full power with equilibrium xenon and normal rod position.

The critical boron concentrations for the beginning of cycle 4 are very close to those of cycle 3. End-of-cycle (EOC) conditions vary between cycles 3 and 4, causing higher critical boron concentrations. As indicated in Table 5-2, the control rod worths are sufficient to maintain the required shutdown margin. However, due to changes in isotopics and the radial flux distribution, the hot, full-power control rod worths will be less than those for cycle 3. The cycle 4 ejected rod worths for the same number of regulating banks inserted are lower than those in cycle 3. It is difficult to compare values between cycles or between rod patterns since neither the rod patterns from which the CRA is assumed to be ejected nor the isotopic distributions are identical. 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 insertion limits presented in section 8. The maximum stuck rod worths for cycle 4 are similar to those in cycle 3. The adequacy of the shutdown margin with cycle 4 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 shutdown calculation at the end of cycle 4 is analyzed at approximately 250 EFPD (±10). This is the latest time (±10 days) in core life at which the transient bank is nearly full inserted. After 250 EFPD, the transient bank will be almost fully withdrawn, thus increasing the available shutdown margin. The reference fuel cycle shutdown margin is presentd in BAW-1452.[2]

The cycle 4 power deficits from hot zero power to hot full power are similar to but slightly lower than those for cycle 3. Doppler coefficients, moderator coefficients, differential boron worth, and xenon worths are similar for the two cycles. The effective delayed neutron fractions for both cycles show a decrease with burnup.

5.2. Analytical Input

The cycle 4 incore measurement calculation constants used for computing core power distributions were prepared in the same manner as 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 for cycles 3 and 4. The operational limits (Technical Specification changes) for the reload cycle are shown in section 8. The FLAME code was used in setting the Technical Specification limits.

The nuclear characteristics of the two batch 5 Mark CR (17 × 17) fuel assemblies are nearly identical to the Mark B (15 × 15) assemblies that make up the balance of batch 5. The two batch 4 Mark C demonstration assemblies are also comparable to batch 4 Mark B assemblies. Therefore, the presence of the 17 × 17 demonstration FAs (two Mark C and two Mark CR) will not discernibly affect overall core reactivity coefficients or performance. However, since the two Mark C and two Mark CR fuel assemblies are demonstration assemblies, standard practice dictates their placement in nonlimiting core locations during cycle 4.

Table 5-1. Oconee 2, Cycle 3 and 4 Physics Parameters

	Cycle 3[(a)]	Cycle 4[(b)]
Cycle length, EFPD	300	292
Cycle burnup, MWd/mtU	9392	9138
Average core burnup — EOC, MWd/mtU	18,554	18,341
Initial core loading, mtU	82.0	82.0
Critical boron — BOC (no Xe), ppm		
HZP, group 8 37.5% wd[(c)]	1335	1340
HZP, groups 7 and 8 inserted	1250	1271
HFP, groups 7 and 8 inserted	1070	1078
Critical boron, EOC (eq Xe), ppm		
HZP, group 8 37.5% wd, eq Xe	290	324
HFP, group 8 37.5% wd, eq Xe	35	55
Control rod worth — HFP, BOC, % Δk/k		
Group 6	1.12	0.87
Group 7	0.88	0.74
Group 8 37.5% wd	0.33	0.47
Control rod worth — HFP (250 EFPD), % Δk/k		
Group 7	1.12	1.03
Group 8 37.5% wd	0.40	0.48
Max ejected rod worth — HZP, % Δk/k[(d)]		
BOC	0.49	0.38
250 EFPD	0.47	0.41
Max stuck rod worth — HZP, % Δk/k		
BOC	1.94	2.22
250 EFPD	2.04	2.09
Power deficit — HZP to HFP, % Δk/k		
BOC	-1.63	-1.52
250 EFPD	-2.17	-1.99
Doppler coeff — BOC, 10^{-5} Δk/k-°F		
100% power (group 8 in, no Xe, critical boron)	-1.47	-1.47
Doppler coeff — EOC, 10^{-5} Δk/k-°F		
100% power (group 8 in, eq Xe, 17 ppm boron)	-1.54	-1.59
Moderator coeff — HFP, 10^{-4} Δk/k-°F		
BOC (group 8 in, no Xe, critical boron)	-0.49	-0.63
EOC (group 8 in, eq Xe, 17 ppm boron)	-2.58	-2.58
Boron worth — HFP, ppm/%(Δk/k)		
BOC (1000 ppm boron)	106	107
EOC (17 ppm boron)	96	96
Xenon worth — HFP, % Δk/k		
BOC (4 EFPD)	2.65	2.65
EOC (equilibrium)	2.75	2.75
Effective delayed neutron fraction — HFP		
BOC	0.00596	0.00591
EOC	0.00522	0.00517

(a) Based on cycle 2 length of 277 EFPD.

(b) Cycle 4 data are for the condition stated and are based on a cycle 3 operating length of 300 EFPD; cycle 3 data may not be at the same conditions as cycle 4.

(c) HZP: hot zero power, HFP: hot full power.

(d) Ejected rod value for groups 5, 6, 7, and 8 inserted.

Table 5-2. Shutdown Margin Calculation for Oconee 2, Cycle 4[a]

	BOC, % Δk/k	EOL, % Δk/k[b]
Available Rod Worth		
Total rod worth, HZP[c]	8.86	8.86
Worth reduction due to burnup of poison material	-0.26	-0.31
Maximum stuck rod, HZP	-2.22	-2.09
Net worth	6.38	6.46
Less 10% uncertainty	-0.64	-0.65
Total available worth	5.74	5.81
Required Rod Worth		
Power deficit, HFP to HZP	1.52	1.99
Max allowable inserted rod worth	0.88	1.48
Flux redistribution	0.47	0.89
Total required worth	2.87	4.36
Shutdown Margin		
Total available worth minus total required worth	2.87	1.45

[a] Required shutdown margin is 1.00% Δk/k.

[b] For shutdown margin calculations, this is defined as approximately 250 EFPD — the latest time in core life at which the transient bank is nearly fully in.

[c] HZP: hot zero power, HFP: hot full power.

Figure 5-1. BOC (4 EFPD) Cycle 4 Two-Dimensional Relative Power Distribution — Full Power, Equilibrium Xenon, Normal Rod Positions, Groups 7 and 8 Inserted

	8	9	10	11	12	13	14	15
H	1.04	1.16	1.00	0.93	1.37	0.84	7 0.43	0.62
K		1.38	0.86	1.11	0.98	1.06	0.78	0.71
L			7 0.59	1.15	8 1.13	1.04	1.26	0.70
M				1.29	1.05	1.33	1.16	
N					0.98	1.18	0.80	
O						0.84		
P								
R								

Legend	
0 0.00	Inserted Rod Group No. Relative Power Density

6. THERMAL-HYDRAULIC DESIGN

The incoming batch 6 fuel is hydraulically and geometrically similar to the fuel remaining in the core from previous cycles. The thermal-hydraulic design evaluation supporting cycle 4 operation used the methods and models described in references 2, 4, and 6 with two exceptions. The exceptions are the core bypass flow and reference design radial × local peaking used in the analysis.

Fuel assemblies not containing control rods or neutron sources usually have orifice rod assemblies (ORAs) installed in the guide tubes to minimize core bypass flow. Including the two neutron sources, there are a total of 108 possible locations for ORAs. During cycle 3 operation, 70 ORAs were installed, leaving 36 vacant fuel assemblies. The maximum core bypass used for cycle 3 analysis was 8.34% based on an assumed 44 ORAs removed. For cycle 4 operation, all ORAs will be removed, leaving 106 vacant fuel assemblies and a maximum core bypass flow of 10.4%.

To offset the effect of the increased core bypass flow in cycle 4 on thermal-hydraulic design, the reference design radial × local peaking factor ($F_{\Delta H}$) has been reduced from 1.78 to 1.71. This reduction in $F_{\Delta H}$ is fully supported by the cycle 4 nuclear design, for which the maximum predicted radial × local peaking factor is 1.57. Reactor core safety limits have been re-evaluated based on the reduced $F_{\Delta H}$ and the increased core bypass flow. The cycle 3 and 4 maximum design conditions and significant parameters are shown in Table 6-1.

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 penalty was calculated from the following equation:

$$\frac{\Delta C}{C_o} = 0.065 + 0.001449\sqrt{BU} \qquad (1)$$

where

ΔC = rod bow magnitude, mils,

C_o = initial gap (138 mils),

BU = maximum assembly burnup, MWd/mtU.

An 11.2% rod bow penalty based on an assumed burnup of 33,000 MWd/mtU is applied to all analyses that define plant operating limits and to design transients. No fuel assembly will achieve a burnup as high as 33,000 MWd/mtU during cycle 4 operation. A thermal margin credit equivalent to 1% DNBR to offset the rod bow penalty has been used as a result of the flow area (pitch) reduction factor included in all thermal-hydraulic analyses. The 1% DNBR is the only credit applied to offset the rod bow penalty.

The flux/flow trip setpoint was determined by analyzing an assumed two-pump coastdown starting from an initial power level (indicated) of 102%. A flux/flow trip setpont of 1.055 is established for cycle 4 based on a minimum DNBR of 1.30 plus a suitable margin to offset the 11.2% rod bow penalty.

The DNBR analysis has been based on a core configuration consisting of 177 Mark B (15 × 15) fuel assemblies. Comparative analyses have been performed to show that the insertion of the Mark C (17 × 17) demonstration assemblies will increase the MDNBR in the hot assembly. The demonstration assemblies have been placed in low-power-producing core locations to ensure that they will not be limiting and to provide minimum impact on the hot assembly performance. Therefore, the presence of the Mark C demonstration assemblies in cycle 4 will not discernibly affect the thermal-hydraulic character of the reactor.

Table 6-1. Thermal-Hydraulic Design Conditions

	Cycle 3[2]	Cycle 4
Power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % of design flow	106.5	106.5
Vessel inlet coolant temp at 100% power, F	555.6	555.6
Vessel outlet coolant temp at 100% power, F	602.4	602.4
Reference design radial-local power peaking factor	1.78	1.71
Reference design axial flux shape	1.5 cos	1.5 cos
Active fuel length, in.	140.5	140.5
Avg heat flux, 100% power, 10^3 Btu/h-ft^2	176[(a)]	176[(a)]
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
MDNBR with densification penalty	1.91	1.98

(a) Based on densified length of 140.3 inches.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[1] accident analysis has been examined with respect to changes in cycle 4 parameters to determine the effect of the cycle 4 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 4. Since batch 6 reload fuel assemblies contain fuel rods with a theoretical density higher than those considered in reference 4, the conclusions in that reference are still valid.

No new dose calculations were performed for the reload report. The dose considerations in the FSAR were based on maximum peaking and burnup for all core cycles; therefore, the dose considerations are independent of the reload batch.

7.2. Accident Evaluation

The key parameters that have the greatest effect on 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 4 are compared in Table 4-2. A comparison of the cycle 4 thermal-hydraulic maximum design conditions to the previous cycle 3 values is presented in Table 6-1. The key kinetics parameters from the FSAR and cycle 4 are compared in Table 7-1.

A generic LOCA analysis has been performed for the B&W 177-FA lowered-loop NSS using the Final Acceptance Criteria ECCS evaluation model reported in reference 7. This analysis is generic in nature since the limiting values of the key parameters for all plants in this category were used. Furthermore, the combination of the average fuel temperature as a function of linear heat rate and the

lifetime pin pressure data used in the LOCA limits analysis[7] is conservative compared to those calculated for this reload. Thus, the analysis and the LOCA limits reported in reference 5 provided conservative results for the operation of Oconee 2, cycle 4 fuel. Table 7-2 shows the bounding values for allowable LOCA peak linear heat rates for Oconee 2, cycle 4 fuel.

From examinations of cycle 4 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 ability to operate the Oconee 2 plant safely during cycle 4. Considering the previously accepted design basis used in the FSAR and subsequent cycles, the transient evaluation of cycle 4 is considered to be bounded by previously accepted analyses. The initial conditions of the transients in cycle 4 are bounded by the FSAR, the fuel densification report[4] and/or subsequent cycle analyses.

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

Parameter	FSAR, densif value	Predicted cycle 4 value
BOL Doppler coeff, 10^{-5} Δk/k-°F	-1.17[a]	-1.47
EOL Doppler coeff, 10^{-5} Δk/k-°F	-1.33	-1.59
BOL moderator coeff, 10^{-4} Δk/k-°F	+0.5[b]	-0.63
EOL moderator coeff, 10^{-4} Δk/k-°F	-3.0	-2.58
All rod bank worth (HZP), % Δk/k	10.0	8.86
Boron reactivity worth @ 70F, ppm/1% (Δk/k)	75	75
Max ejected rod worth (HFP), % Δk/k	0.65	0.41
Dropped rod worth (HFP), % Δk/k	0.46	0.20
Initial boron conc (HFP), ppm	1400	1078

[a] -1.2×10^{-5} Δk/k-°F was used for steam line failure analysis; -1.3×10^{-5} Δk/k -°F was used for cold water analysis.

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

Table 7-2. Bounding Values for Allowable LOCA Peak Linear Heat Rates

Core elevation, ft	Allowable peak LHR, kW/ft
2	15.5
4	16.6
6	18.0
8	17.0
10	16.0

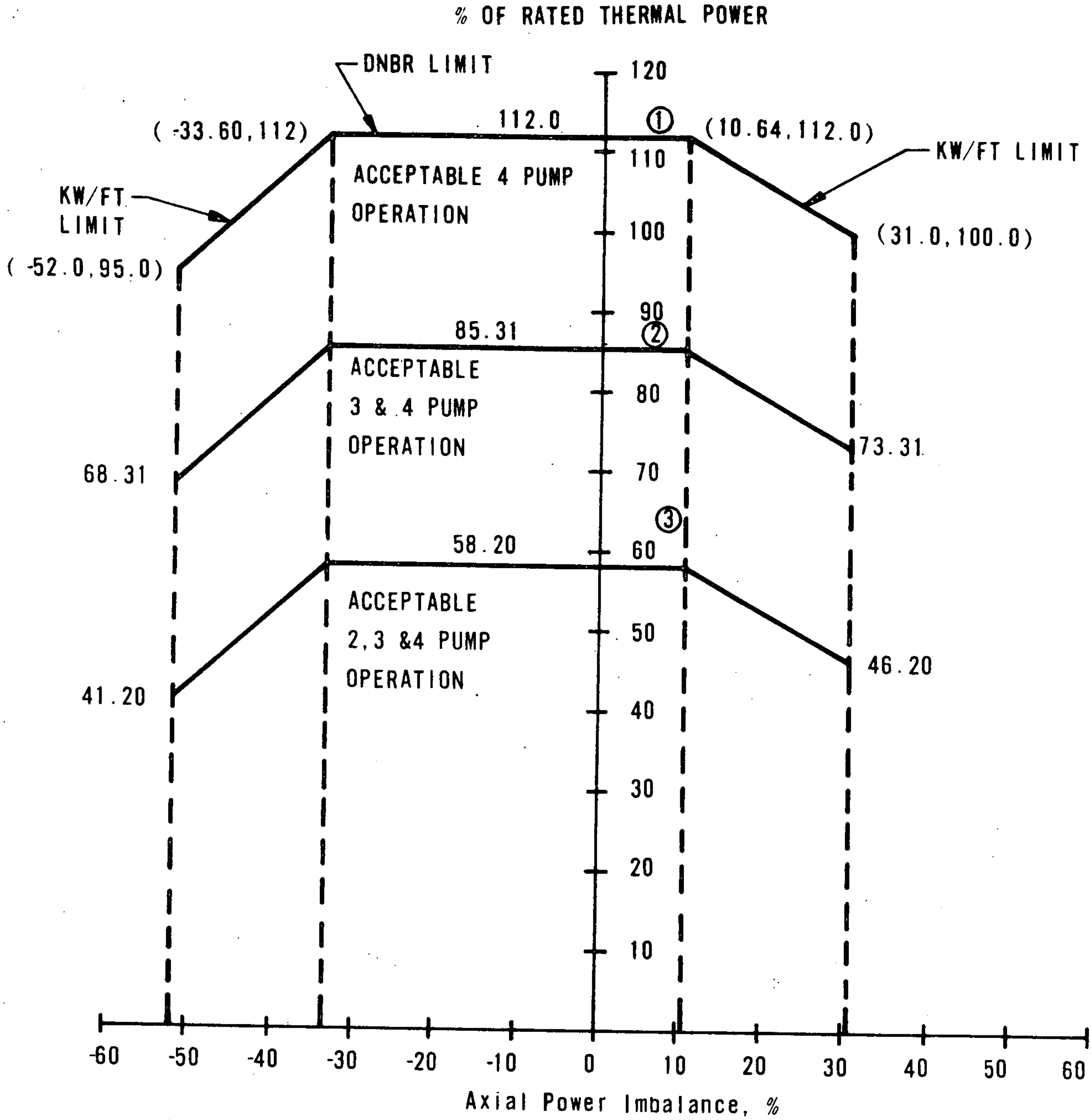
8. PROPOSED MODIFICATIONS TO TECHNICAL SPECIFICATIONS

The Technical Specifications have been revised for cycle 4 operation to account for minor changes in power peaking and control rod worths inherent with non-equilibrium cycles. In addition, changes were the result 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 8.
2. Per reference 9, 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 4 is 5.0%. A power peaking penalty (1.075) appropriate for this quadrant tilt limit was used to set imbalance and rod insertion limits.

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



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

Figure 8-2. Protective System Maximum Allowable Setpoints

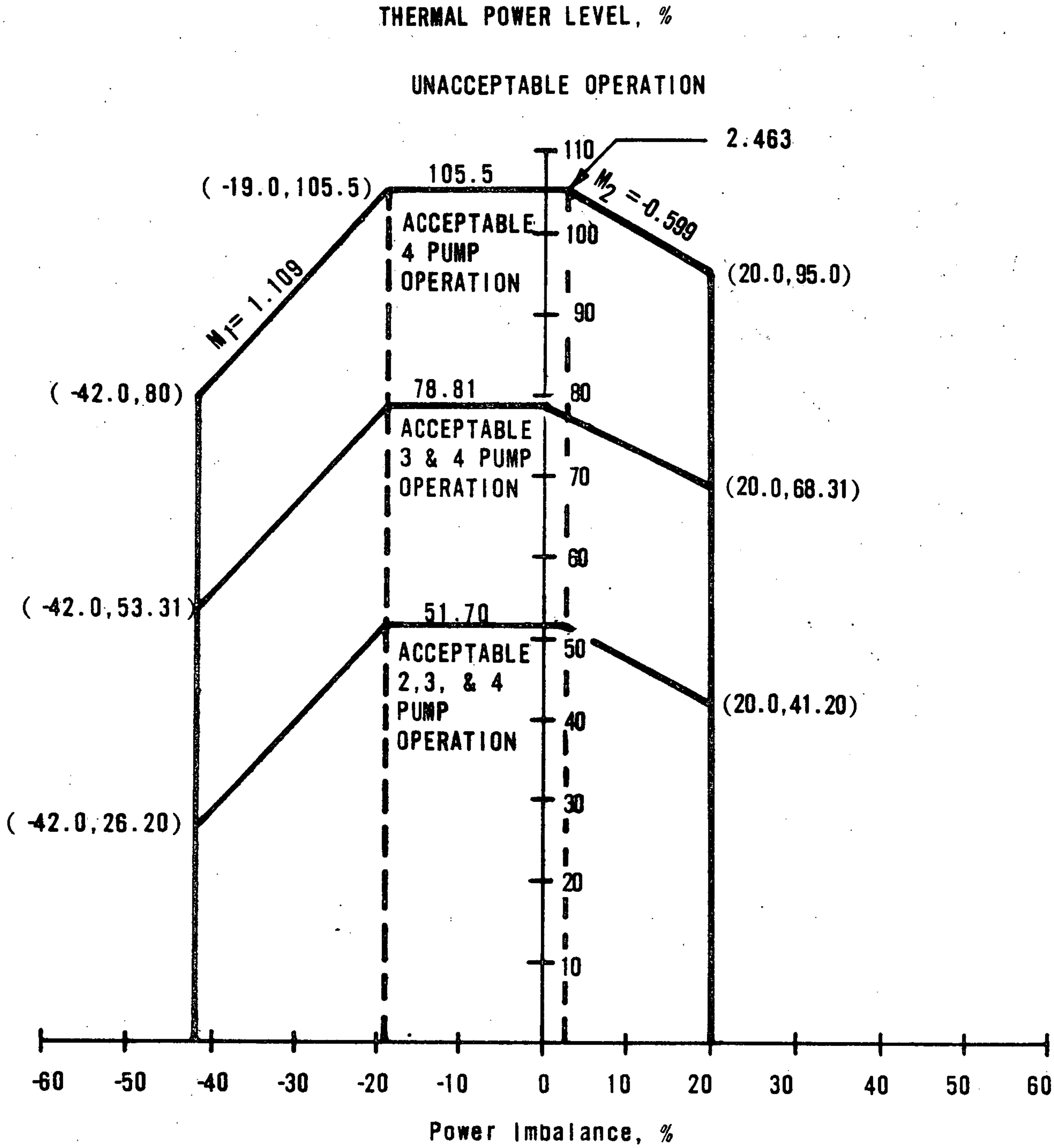


Figure 8-3. Rod Position Limits for Four-Pump Operation
From 0 to 250 ± 10 EFPD — Oconee 2, Cycle 4

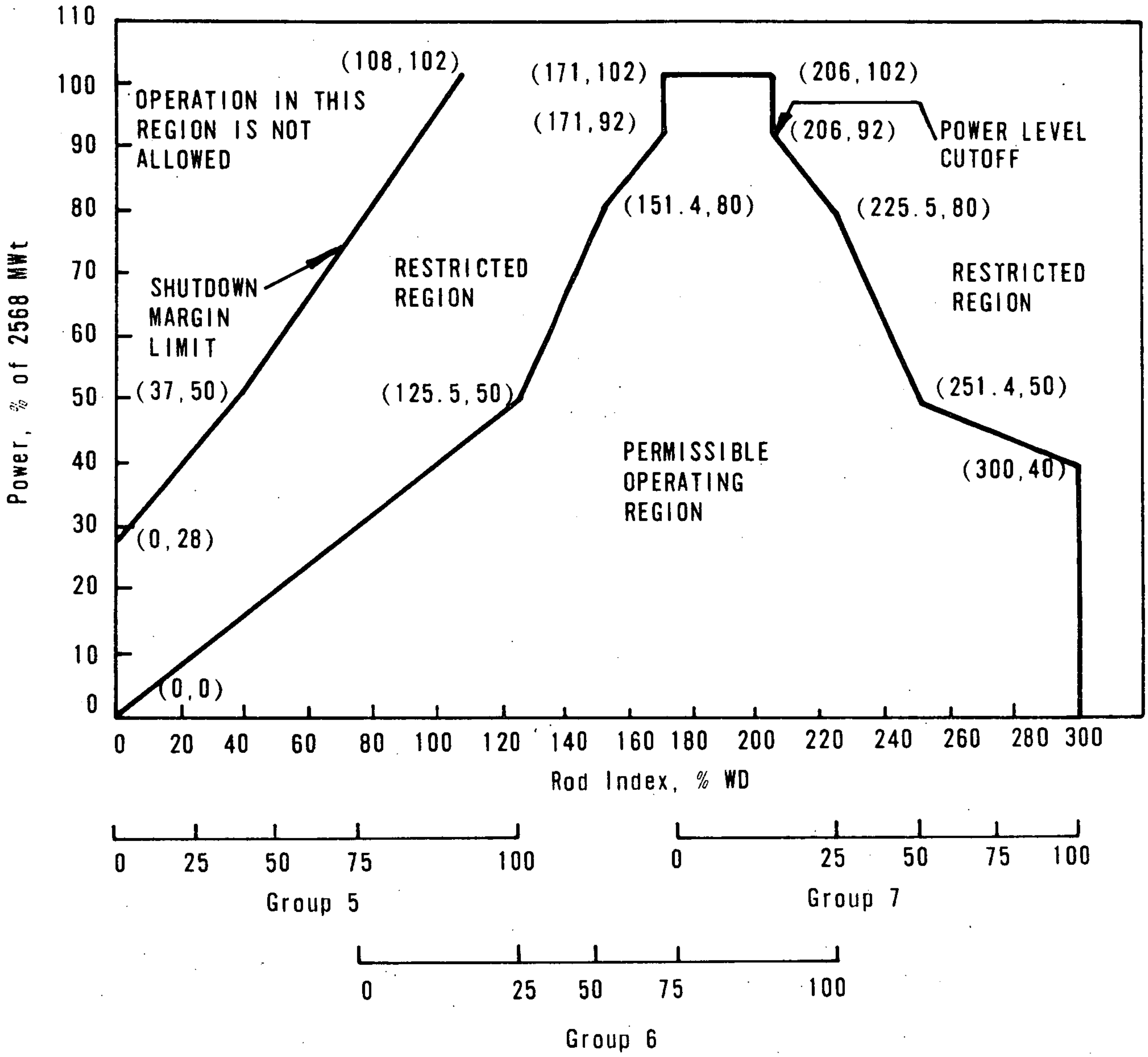


Figure 8-4. Rod Position Limits for Four-Pump Operation After 250 ± 10 EFPD — Oconee 2, Cycle 4

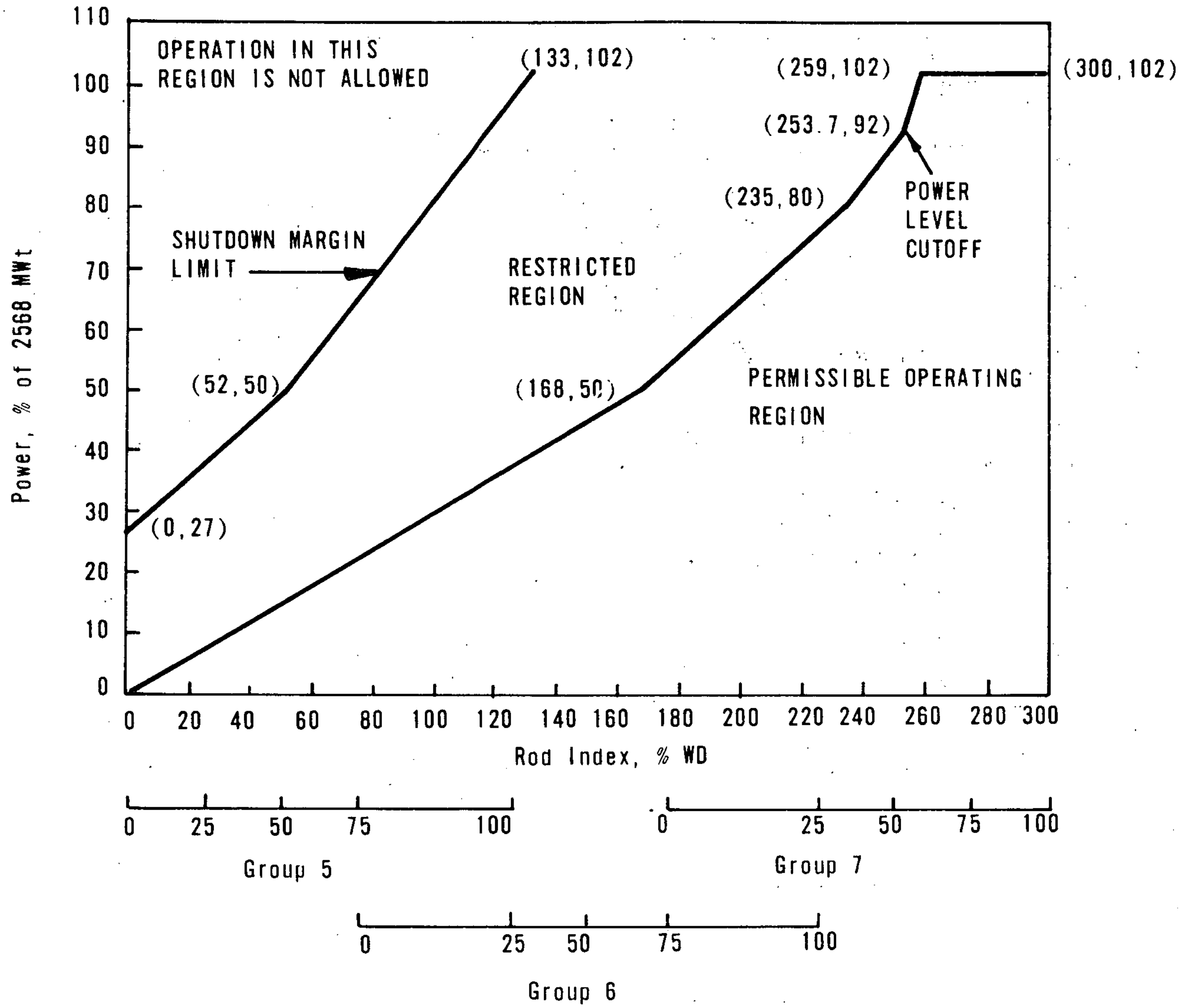


Figure 8-5. Rod Position Limits for Two- and Three-Pump Operation From 0 to 250 ± 10 EFPD — Oconee 2, Cycle 4

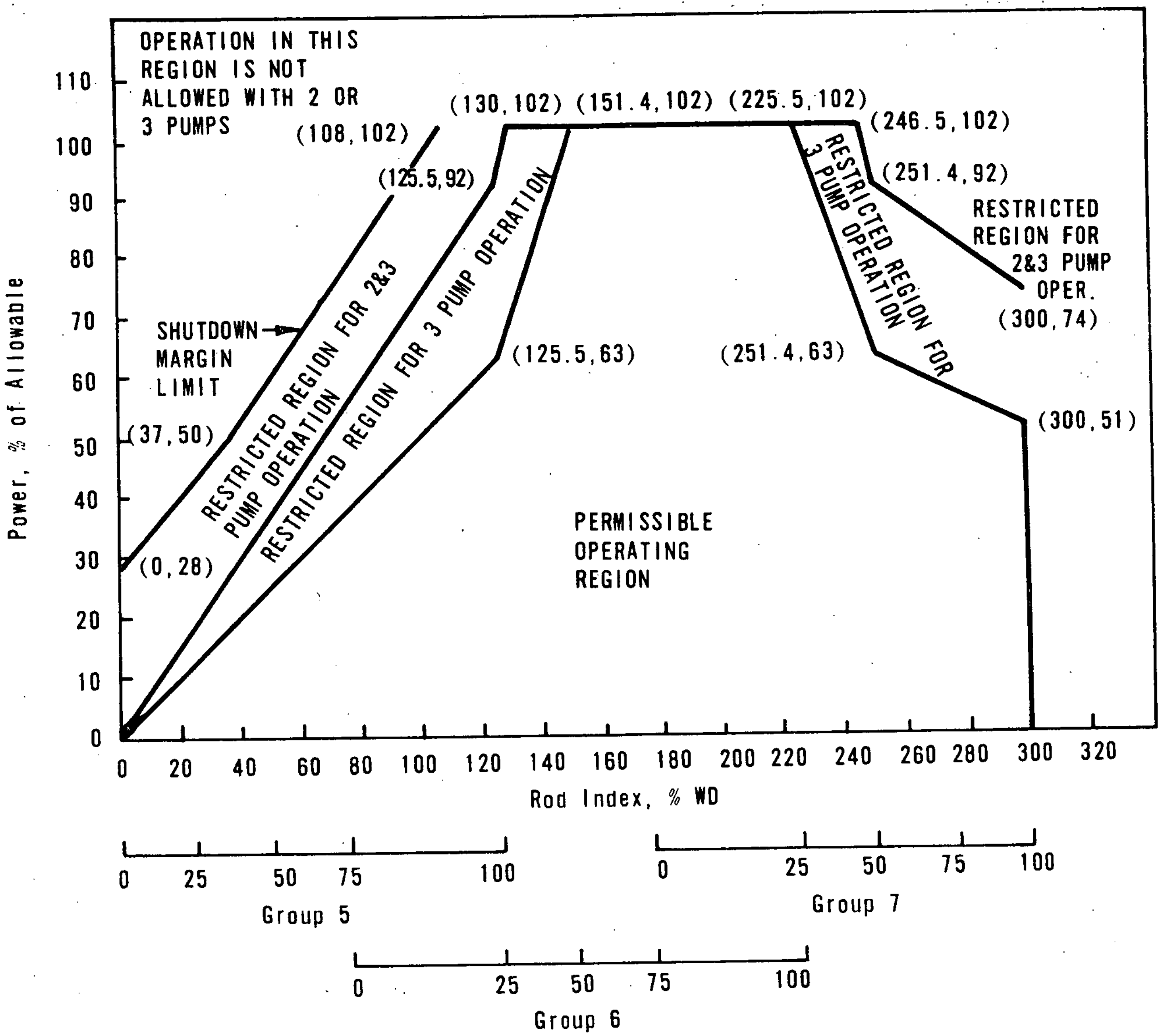


Figure 8-6. Rod Position Limits for Two- and Three-Pump Operation After 250 ± 10 EFPD — Oconee 2, Cycle 4

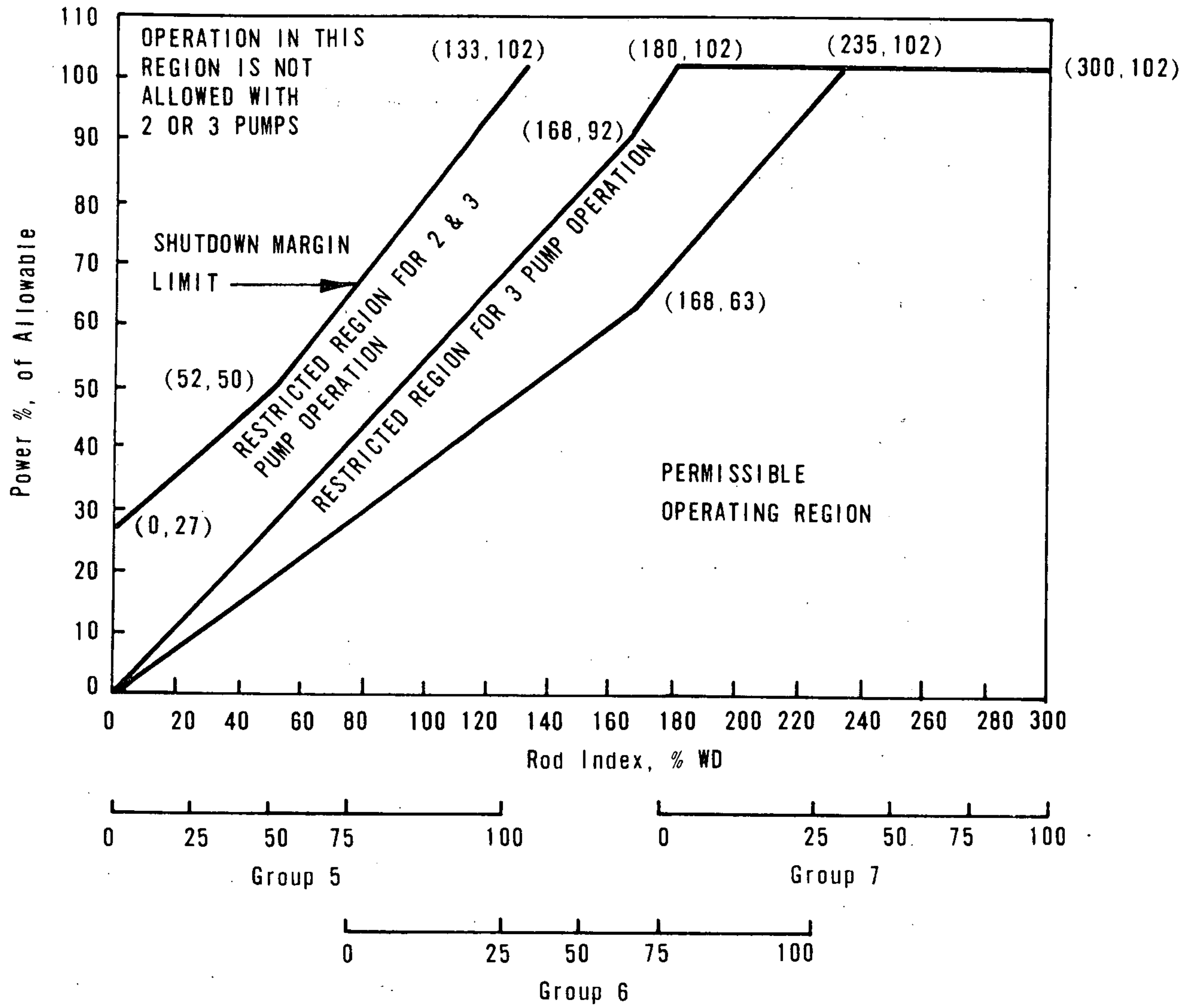


Figure 8-7. Operational Power Imbalance Envelope for Operation From 0 to 250 ± 10 EFPD — Oconee 2, Cycle 4

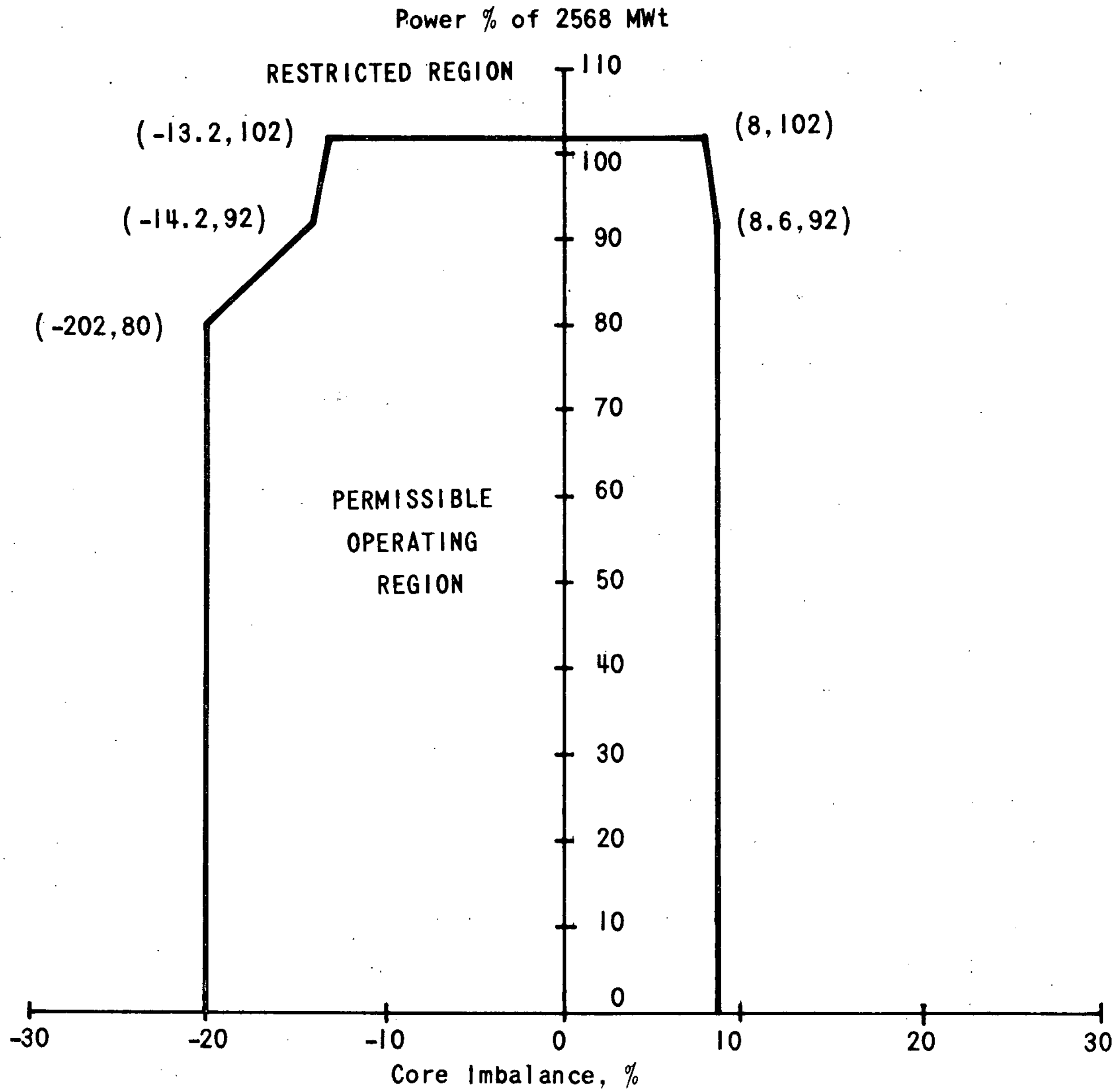


Figure 8-8. Operational Power Imbalance Envelope for Operation After 250 ± 10 EFPD — Oconee 2, Cycle 4

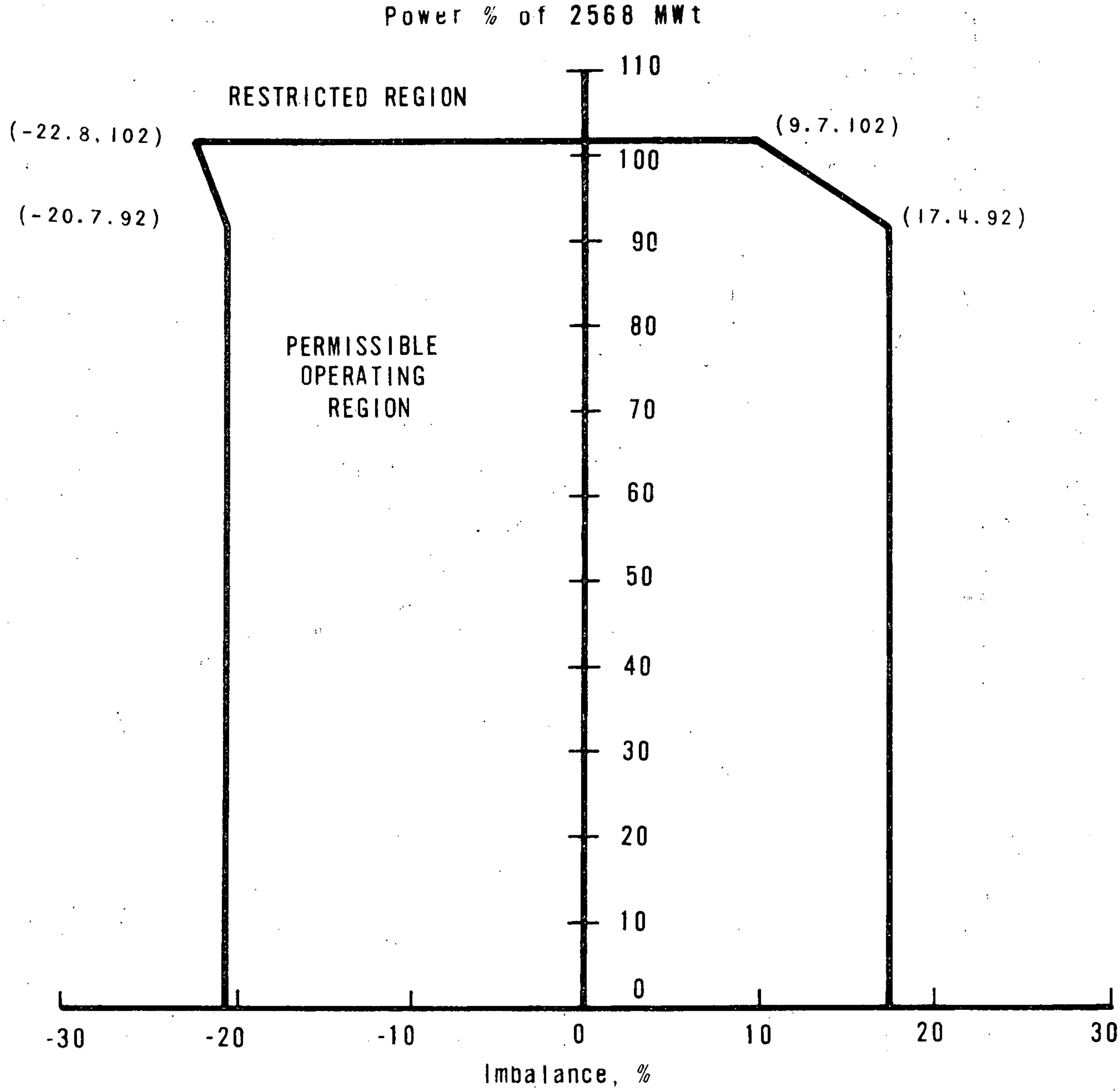


Figure 8-9. APSR Position Limits for Operation From 0 to 250 ± 10 EFPD — Oconee 2, Cycle 4

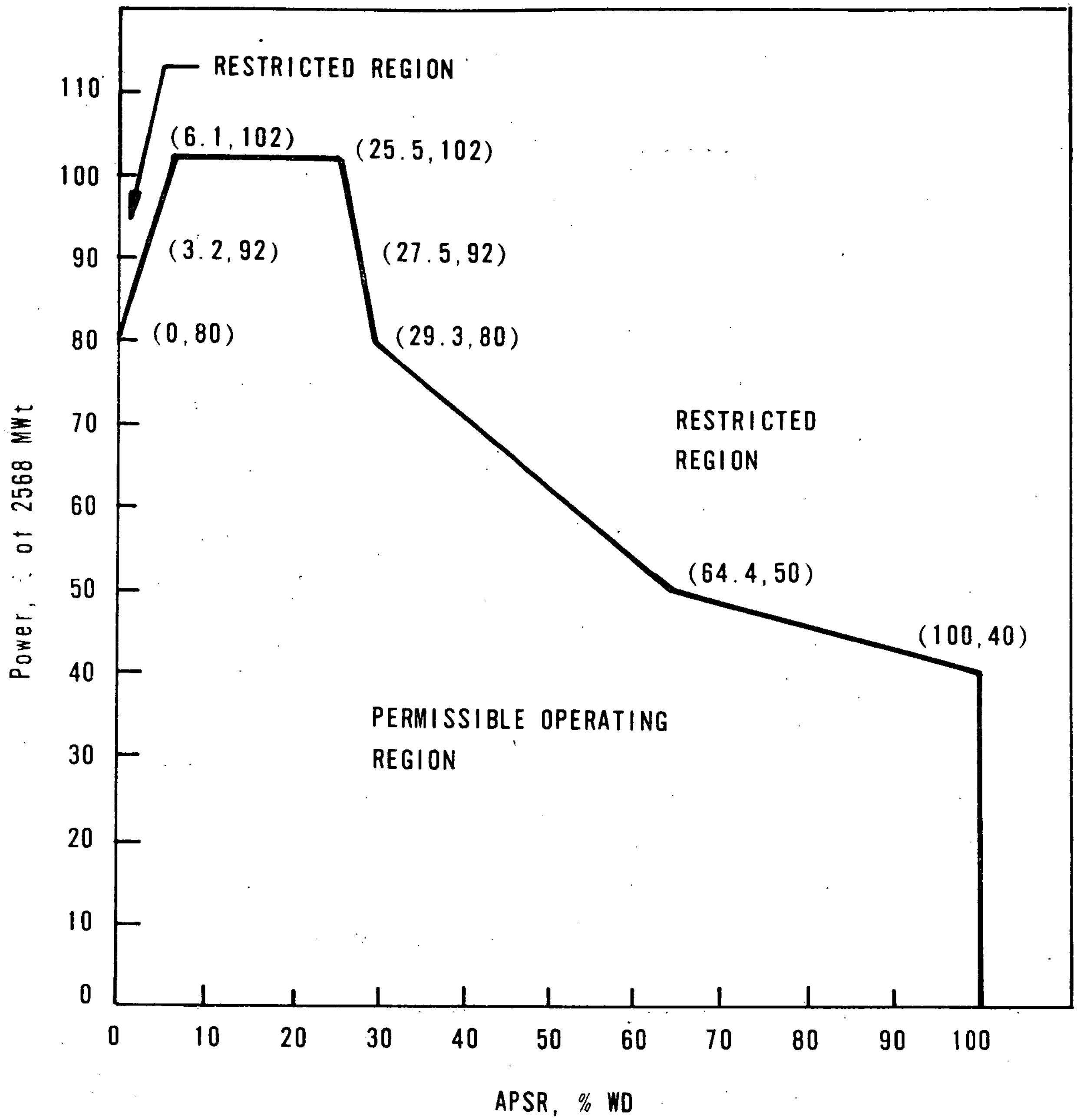
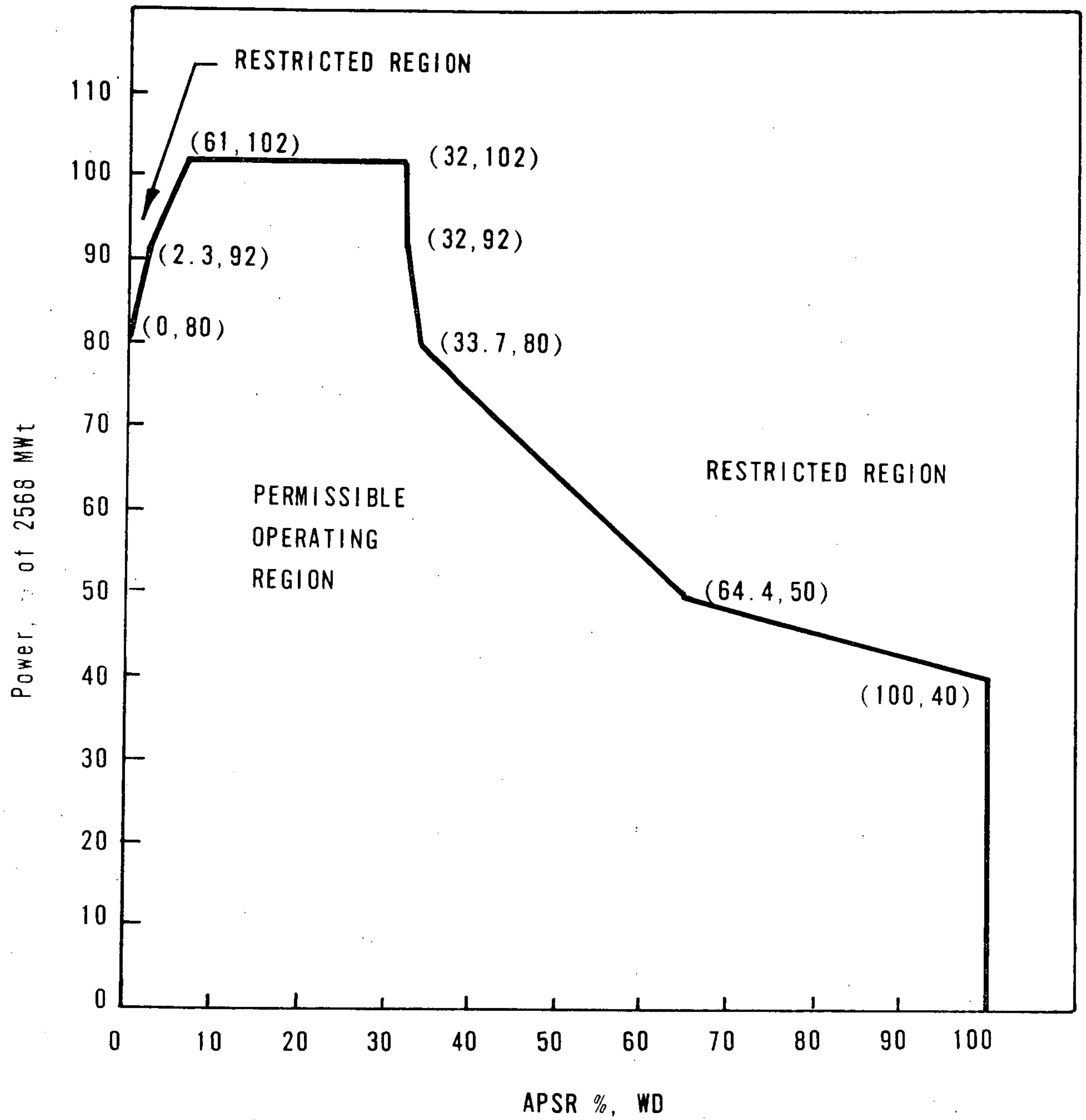


Figure 8-10. APSR Position Limits for Operation After 250 ± 10 EFPD — Oconee 2, Cycle 4



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 confirmation for continued safe operation of the unit.

9.1. Precritical Tests

9.1.1. Control Rod Trip Test

Precritical control rod drop times are recorded for all control rods at hot full flow conditions before zero power physics testing begins. Acceptance criteria state that the rod drop time from fully withdrawn to 3/4-inserted shall be less than 1.66 seconds at the conditions above.

It should be noted that safety analysis calculations are based on a rod drop time of 1.40 seconds from fully withdrawn to 2/3-inserted. Since the most accurate position indication is obtained from the zone reference switch at the 3/4-inserted position, this position is used instead of the 2/3-inserted position for data gathering. The acceptance criterion of 1.40 seconds corrected to a 3/4-inserted position (by rod insertion versus time correlation) is 1.66 seconds.

9.1.2. Reactor Coolant Flow

RC flow with four RC pumps running will be measured at hot zero power, steady-state conditions. Acceptance criteria require that the measured flow be within allowable limits.

9.1.3. RC Flow Coastdown

The coastdown of RC flow from the tripping of the most limiting RC pump combination from four RC pumps running will be measured at hot zero power conditions. The coastdown of RC flow versus time will then be compared to the required value to determine whether acceptance is met.

9.2. Zero Power Physics Tests

9.2.1. Critical Boron Concentration

Criticality is obtained by deboration at a constant dilution rate. Once criticality is achieved, equilibrium boron is obtained and the critical boron concentration determined. The critical boron concentration is calculated by correcting for any rod withdrawal required in achieving equilibrium boron. The acceptance criterion placed on critical boron concentration is that the actual boron concentration must be within ±100 ppm of predicted.

9.2.2. Temperature Reactivity Coefficient

The isothermal temperature coefficient is measured at approximately the all-rods-out configuration and at the hot zero power rod insertion limit. The average coolant temperature is varied by first decreasing then increasing temperature by 5F. During the change in temperature reactivity feedback is compensated by discrete change in rod motion; the change in reactivity is then calculated by the summation of reactivity (obtained from reactivity calculation on strip chart recorder) associated with the temperature change.

Acceptance criteria state that the measured value shall not differ from the predicted value by more than $\pm 0.4 \times 10^{-4}$ $(\Delta k/k)/°F$ (the predicted value is obtained from Physics Test Manual curves).

The moderator coefficient of reactivity is calculated in conjunction with the temperature coefficient measurement. After the temperature coefficient has been measured, a predicted value of fuel Doppler coefficient of reactivity is added to obtain moderator coefficient. This value must not be in excess of the acceptance criteria limit of $+0.5 \times 10^{-4}$ $(\Delta k/k)/°F$.

9.2.3. Control Rod Group Reactivity Worth

Control bank group reactivity worths (groups 5, 6, and 7) are measured at hot zero power conditions using the boron/rod swap method. This method consists of establishing a deboration rate in the RC system and compensating for the reactivity changes of this deboration by inserting control rod groups 7, 6, and 5 in incremental steps. The reactivity changes that occur during these measurements are calculated based on Reactimeter data, and differential rod worths are obtained from the measured reactivity worth versus the change in rod group position. The differential rod worths of the controlling groups are then summed to obtain integral rod group worths.

The acceptance criteria for the control bank group worths are as follows:

1. Individual bank 5, 6, 7 worth:

$$\left|\frac{\text{predicted value} - \text{measured value}}{\text{measured}} \times 100\right| \leq 15.$$

2. Sum of groups 5, 6, and 7:

$$\left|\frac{\text{predicted} - \text{measured}}{\text{measured}} \times 100\right| \leq 10.$$

9.2.4. Ejected Control Rod Reactivity Worth

After the CRA groups have been positioned at the rod insertion limit, the ejected rod is borated to 100% withdrawn and the worth obtained by adding the incremental changes in reactivity by boration.

After the ejected rod has been borated to 100% withdrawn and equilibrium boron established, the ejected rod is then swapped in versus the controlling rod group and the worth determined by the change in controlling rod group position. The boron swap and rod swap values are used to determine ejected rod worth.

Acceptance criteria for the ejected rod worth test are as follows:

1. $$\left|\frac{\text{predicted value} - \text{measured value}}{\text{measured value}} \times 100\right| \leq 20.$$

2. Measured value (error adjusted) $\leq$ 1.0% $\Delta k/k$.

The predicted ejected rod worth is given in the Physics Test Manual.

9.3. Power Escalation Tests

9.3.1. Core Power Distribution Verification at ~40, 75, and 100% FP With Nominal Control Rod Group Configuration

Core power distribution tests are performed at 40, 75, and 100% FP. The test at 40% FP is essentially a check on power distribution in the core to bring attention to any abnormalities before escalating to the 75% FP plateau. Rod index is established at a nominal full power configuration, at which the core power distribution calculations are performed. APSR position is established to provide a core power imbalance corresponding to the imbalance at which the core power distribution calculations are performed.

The following acceptance criteria are placed on the 40% FP test:

1. The worst-case maximum linear heat rate must be less than the LOCA limit.

2. The minimum DNBR must be greater than 1.30.

3. The value obtained from the extrapolation of the minimum DNBR to the next power plateau overpower trip setpoint must be greater than 1.30 or fall outside the RPS power/imbalance trip envelope.

4. The value obtained from the extrapolation of the worst-case maximum linear heat rate to the next power plateau overpower trip setpoint must be less than the fuel melt limit or fall outside the RPS power/imbalance trip envelope.

5. The quadrant power tilt shall not exceed the limits specified in the Technical Specifications.

6. The highest measured radial peak and the highest predicted radial peak shall be within the following limits:

$$\left| \frac{\text{predicted} - \text{measured}}{\text{measured}} \times 100 \right| \leq 8$$

7. The highest measured total peak and the highest predicted total peak shall be within the following limits:

$$\left| \frac{\text{predicted} - \text{measured}}{\text{measured}} \times 100 \right| \leq 12$$

Items 1, 2, 5, 6, and 7 above are established to verify core nuclear and thermal calculation models, thereby verifying the acceptability of data from these models for input to safety evaluations.

Items 3 and 4 establish the criteria whereby escalation to the next power plateau may be accomplished without exceeding any safety limits specified by the safety analysis with regard to DNBR and linear heat rate.

The power distribution tests performed at 75 and 100% FP are identical to the 40% FP test except that core equilibrium xenon is established prior to the 75 and 100% FP tests. Accordingly, the 75 and 100% FP measured peak acceptance criteria are as follows:

1. The highest measured radial peak and the highest predicted radial peak shall be within the following limits:

$$\left| \frac{\text{predicted} - \text{measured}}{\text{measured}} \times 100 \right| \leq 5$$

2. The highest measured total peak and the highest predicted total peak shall be within the following limits:

$$\left|\frac{\text{predicted} - \text{measured}}{\text{measured}} \times 100\right| \leq 7.5$$

9.3.2. Incore Vs Excore Detector Imbalance Correlation Verification at ~40% FP

Imbalances are set up in the core by control rod positioning and are read simultaneously on the incore detectors and excore power range detectors. The imbalances from the excore detectors must exceed those on the incore detectors by a factor of 1.25. If the ratio of excore to incore detector imbalance is less than 1.25, gain amplifiers in the excore detector signal processing equipment are adjusted to give the needed gain.

9.3.3. Temperature Reactivity Coefficient at ~100% FP

The average RC temperature is decreased and then increased by about 5F at constant reactor power. The reactivity associated with each temperature change is obtained from the change in the controlling rod group position. Controlling rod group worth is measured by the fast insert/withdrawal method. The temperature reactivity coefficient is calculated from the measured reactivity and temperature changes.

Acceptance criteria are that the moderator temperature coefficient shall be negative.

9.3.4. Power Doppler Reactivity Coefficient at 100% FP

Reactor power is decreased and then increased by about 5% FP. The reactivity change is obtained from the change in controlling rod group position. Control rod group worth is measured using the fast insert/withdrawal method. Reactivity corrections are made for changes in xenon and RC temperature that occur during the measurement. The power Doppler reactivity coefficient is calculated from the measured reactivity change, adjusted as stated above, and the measured power change. The predicted value of the power Doppler reactivity coefficient is given in the Physics Test Manual. Acceptance criteria state that the measured value shall be more negative than -0.55×10^{-4} $(\Delta k/k)/\%$ FP.

9.4. Procedure for Use When Acceptance Criteria Are Not Met

An evaluation is performed before the test program is continued if acceptance criteria for any test are not met. This evaluation is performed by site test

personnel, with participation by Babcock & Wilcox technical personnel as required. Further specific actions depend on evaluation results. These actions can include retesting with more detailed attention to test prerequisites, added tests to search for anomalies, or detailed analysis (by design personnel) of potential safety problems because of parameter deviation. The plant is not escalated in power until evaluation shows that plant safety will not be compromised by such escalation.

REFERENCES

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

2 Oconee 2, Cycle 3 Reload Report, BAW-1452, Babcock & Wilcox, Lynchburg, Va., April 1977.

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

4 Oconee 2 Fuel Densification Report, BAW-1395, Babcock & Wilcox, Lynchburg, Va., June 1973.

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

6 Amendment to Oconee 2, Cycle 3 Reload Report (BAW-1452), Babcock & Wilcox, Lynchburg, Va., June 15, 1977.

7 R. C. Jones, J. R. Biller, and B. M. Dunn, ECCS Analysis of B&W's 177-FA Lowered Loop NSS, BAW-10103A, Rev. 3, Babcock & Wilcox, Lynchburg, Va., October 1977.

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

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