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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DO NOT REMOVE

August 3, 1983

Posted
Amdt. 122
to DPR-47

Dockets Nos. 50-269, 50-270
and 50-287

Mr. H. B. Tucker
Vice President - Steam Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 122 , 122 , and 119 to Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated May 19, 1983, as supplemented by letter dated July 13, 1983.

These amendments revise the TSs to allow full power operation of Oconee Unit 1 during fuel Cycle 8. We have also reviewed a modified version of the Oconee Nuclear Station Generic Startup Physics Test Program submitted as part of the reload package and find the modified version acceptable.

Our review also included an evaluation of the effect of NUREG-0630 cladding models on the Loss of Coolant Accident (LOCA) analysis, and we find that this issue has been resolved for all three units at the Oconee Station.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Monthly Notice.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 122 to DPR-38
2. Amendment No. 122 to DPR-47
3. Amendment No. 119 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

Duke Power Company

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT NO.1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 122
License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated May 19, 1983, as supplemented July 13, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 122 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 122
License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated May 19, 1983, as supplemented July 13, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 122 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 119
License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Duke Power Company (the licensee) dated May 19, 1983, as supplemented July 13, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B of Facility Operating License No. DPR-55 is hereby amended to read as follows:

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.119 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 3, 1983

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 122 TO DPR-38

AMENDMENT NO. 122 TO DPR-47

AMENDMENT NO. 119 TO DPR-55

DOCKETS NOS. 50-269, 50-270 AND 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment numbers and contain vertical lines indicating the area of change.

Remove pages	Insert Pages
2.1-2	2.1-2
2.1-3	2.1-3
2.1-7	2.1-7
2.3-8	2.3-8
3.5-15	3.5-15
3.5-15a	3.5-15a
3.5-15b	3.5-15b
3.5-18	3.5-18
3.5-18a	3.5-18a
3.5-18b	3.5-18b
3.5-18c	3.5-18c
3.5-18d	3.5-18d
3.5-18e	3.5-18e
3.5-21	3.5-21
3.5-21a	3.5-21a
3.5-21b	3.5-21b
3.5-24	3.5-24
3.5-24a	3.5-24a
3.5-24b	3.5-24b

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 2.1-1A represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 106.5 percent of 131.3×10^6 lbs/hr). This curve is based on the combination of nuclear power peaking factors, with potential effects of fuel densification and rod bowing, 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 and 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 20.5 kw/ft for 8C, 9 and 10C Batches of fuel and 17.6 kw/ft for the 10A, 10B gadolinia fuel Batch 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 of Figure 2.1-3A correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

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

The magnitude of the rod bow penalty applied to each fuel cycle is equal to or greater than the necessary burnup dependent DNBR rod bow penalty for the applicable cycle minus a credit of 1% for the flow area reduction factor used in the hot channel analysis. All plant operating limits are based on a minimum DNBR criteria of 1.30 plus the amount necessary to offset the reduction in DNBR due to fuel rod bow. (3)

The maximum thermal po r for three-pump opertion is .899 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.929 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions is produced in a similar manner.

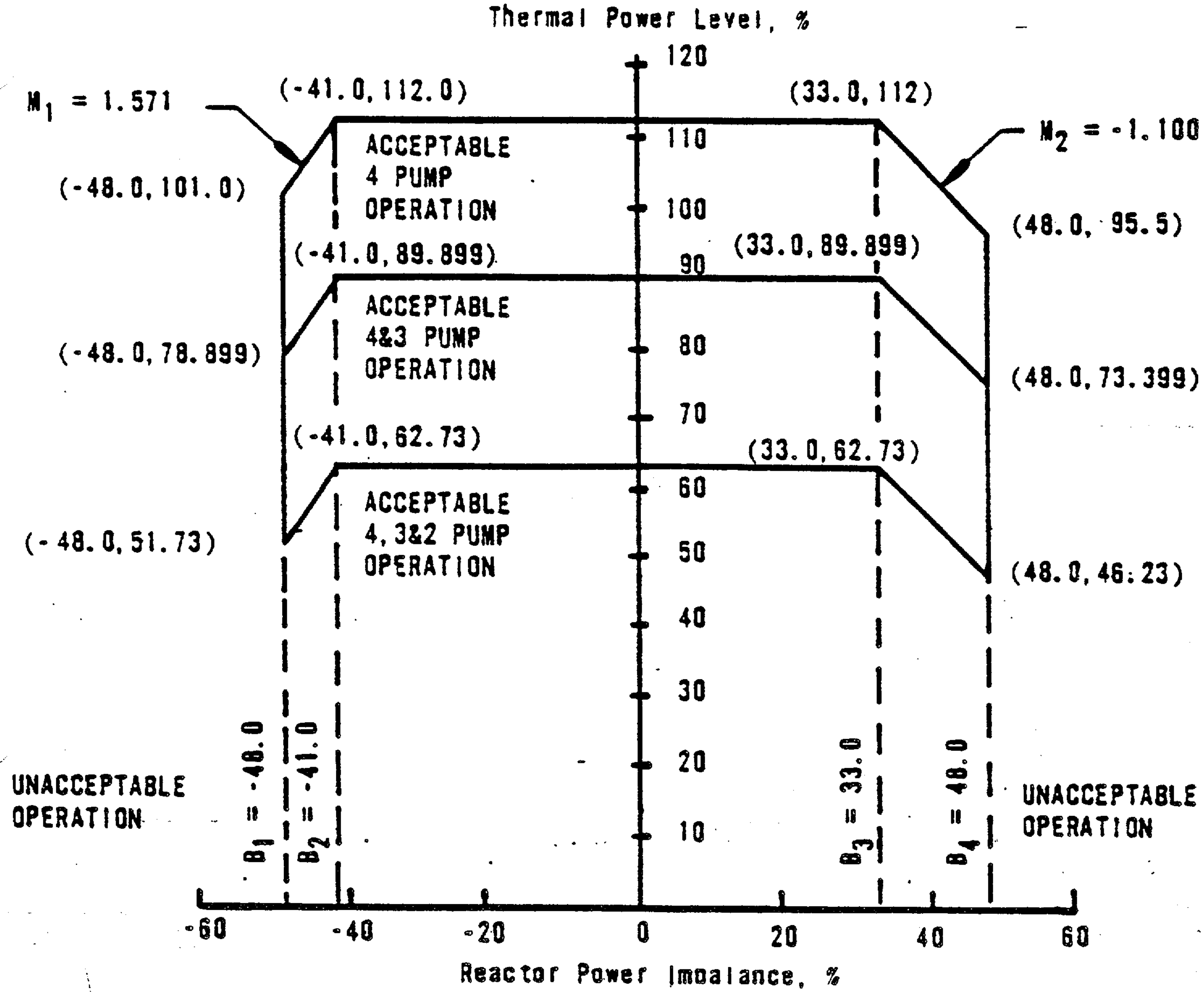
For each curve of Figure 2.1-3A a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The curve of Figure 2.1-1A is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in Figure 2.1-3A.

References

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

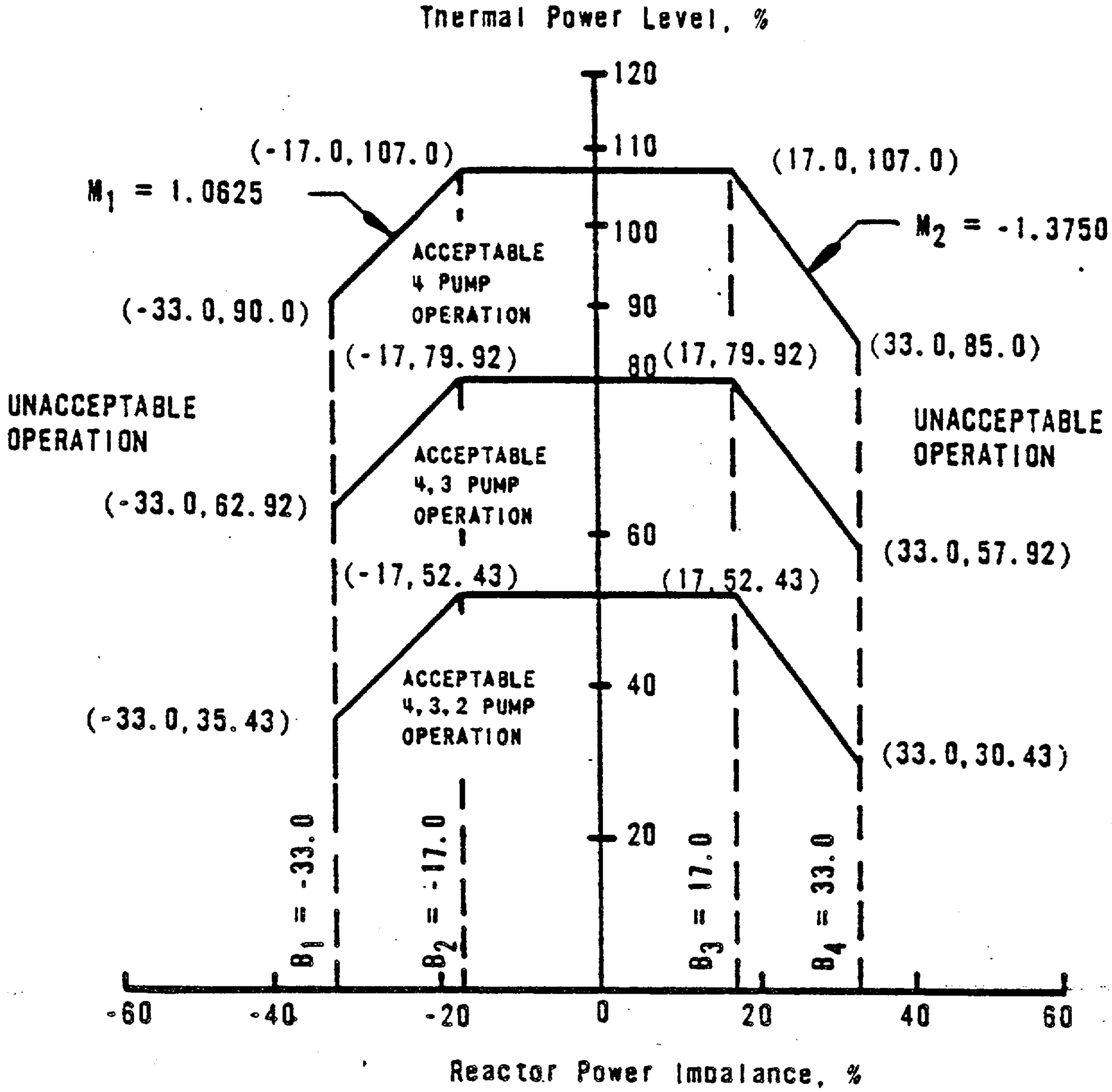
(2) Oconee 1, Cycle 4 - Reload Report - BAW-1447, March, 1977.

(3) Oconee 1, Cycle 8 - Reload Report - BAW-1774, February, 1983.

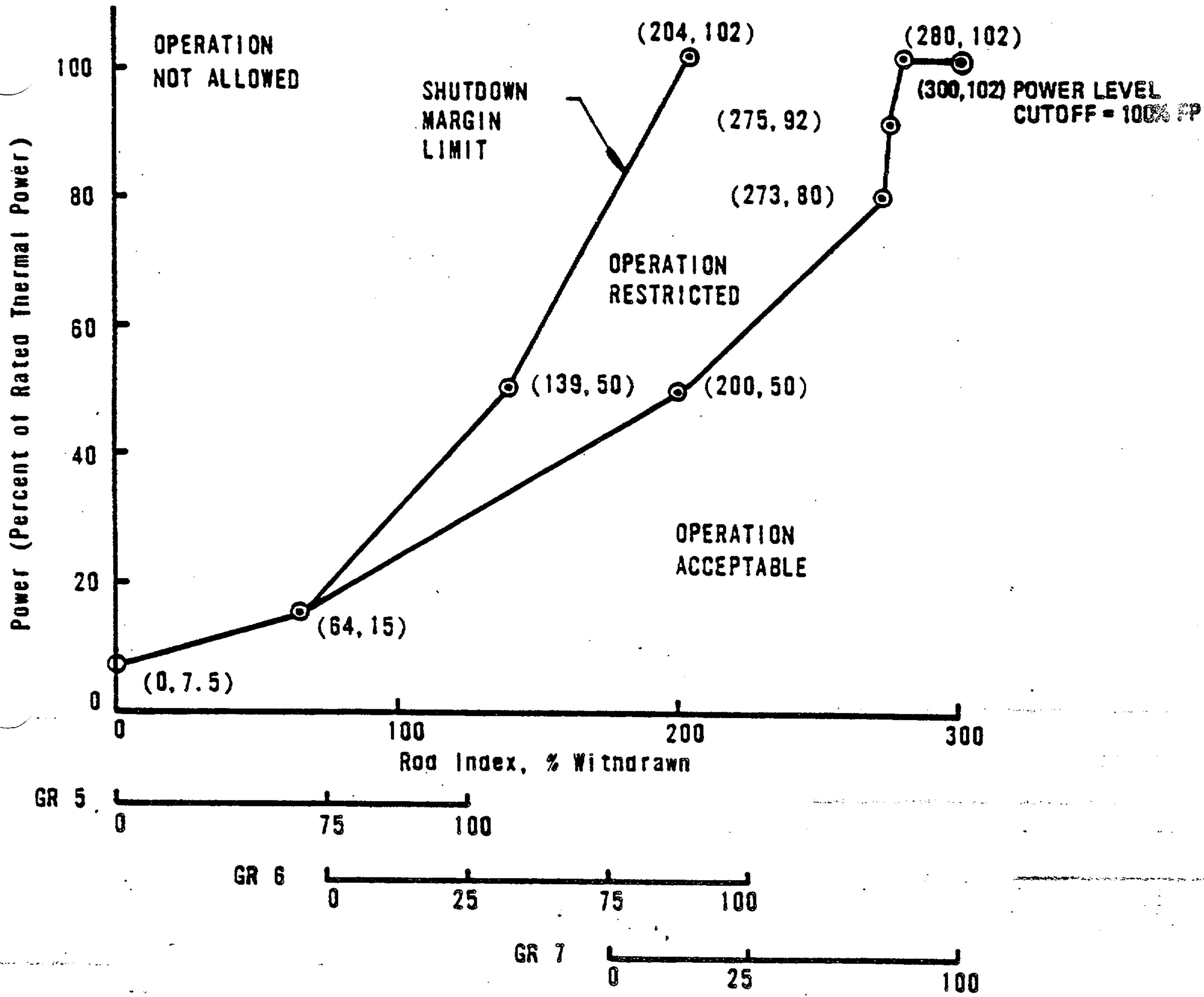


CURVE	RC FLOW (GPM)
1	374,880
2	280,035
3	183,690

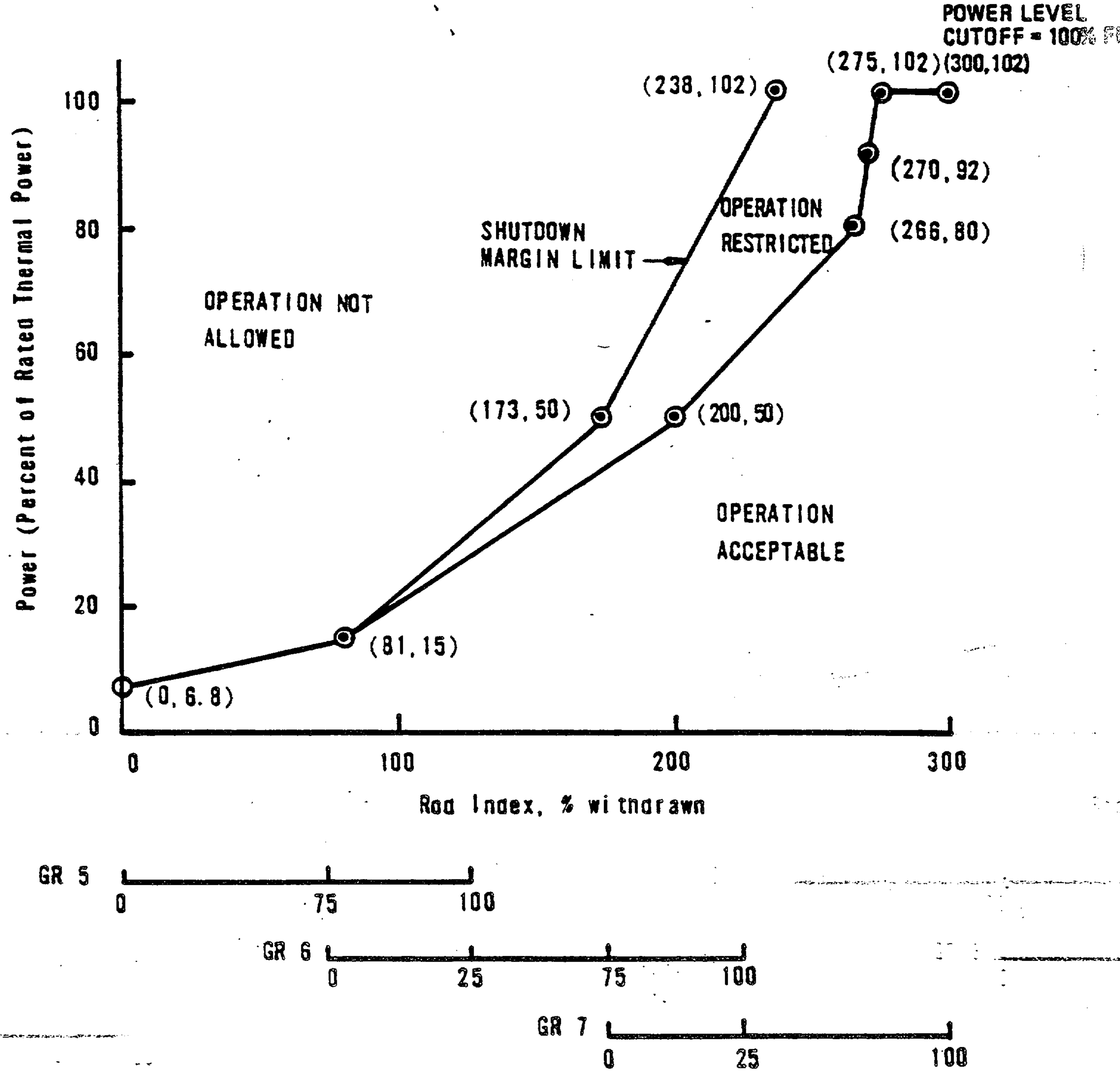
CORE PROTECTION SAFETY LIMITS
UNIT 1
OCONEE NUCLEAR STATION
Figure 2.1-2A



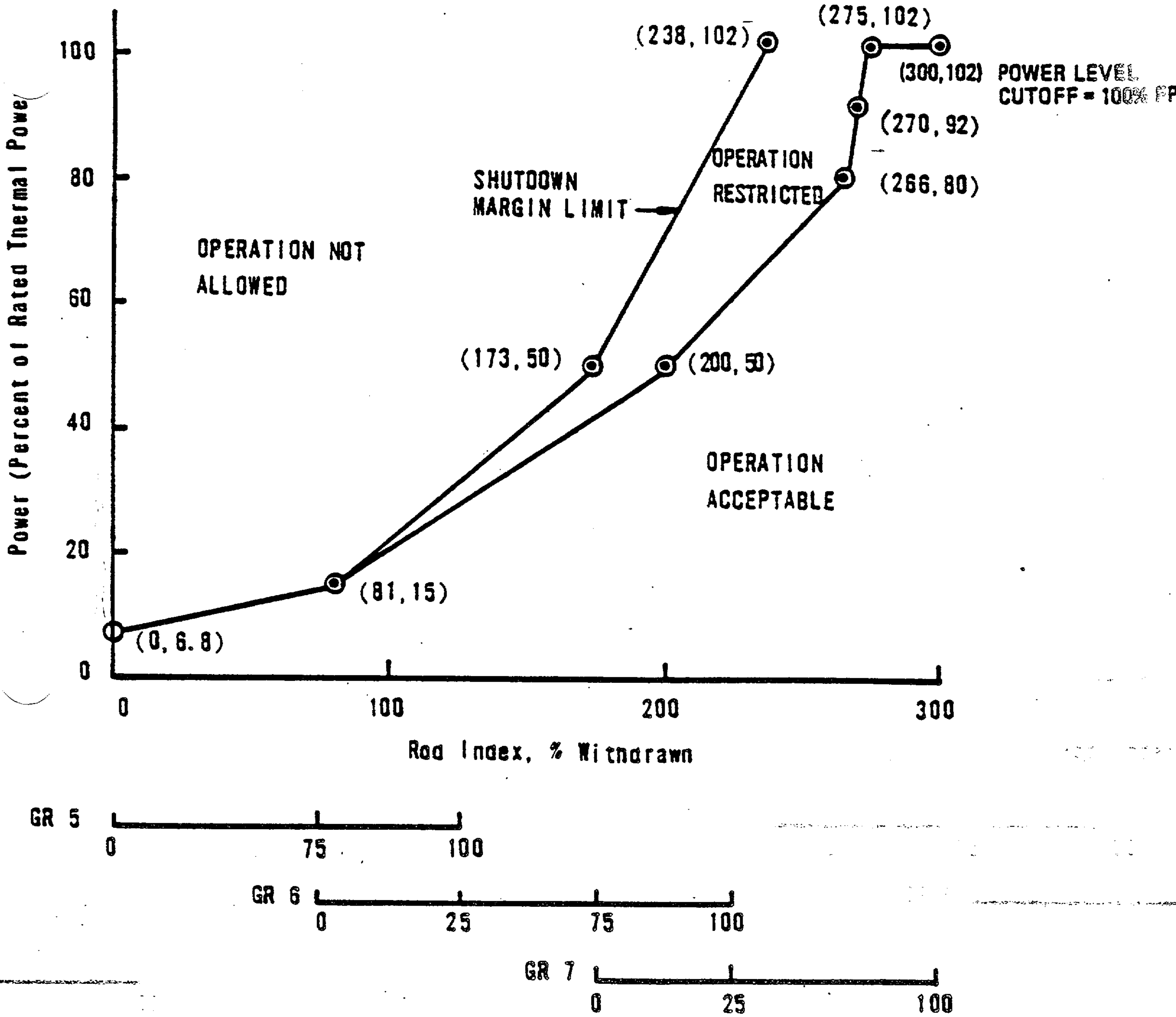
PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 1
OCONEE NUCLEAR STATION
Figure 2.3-2A



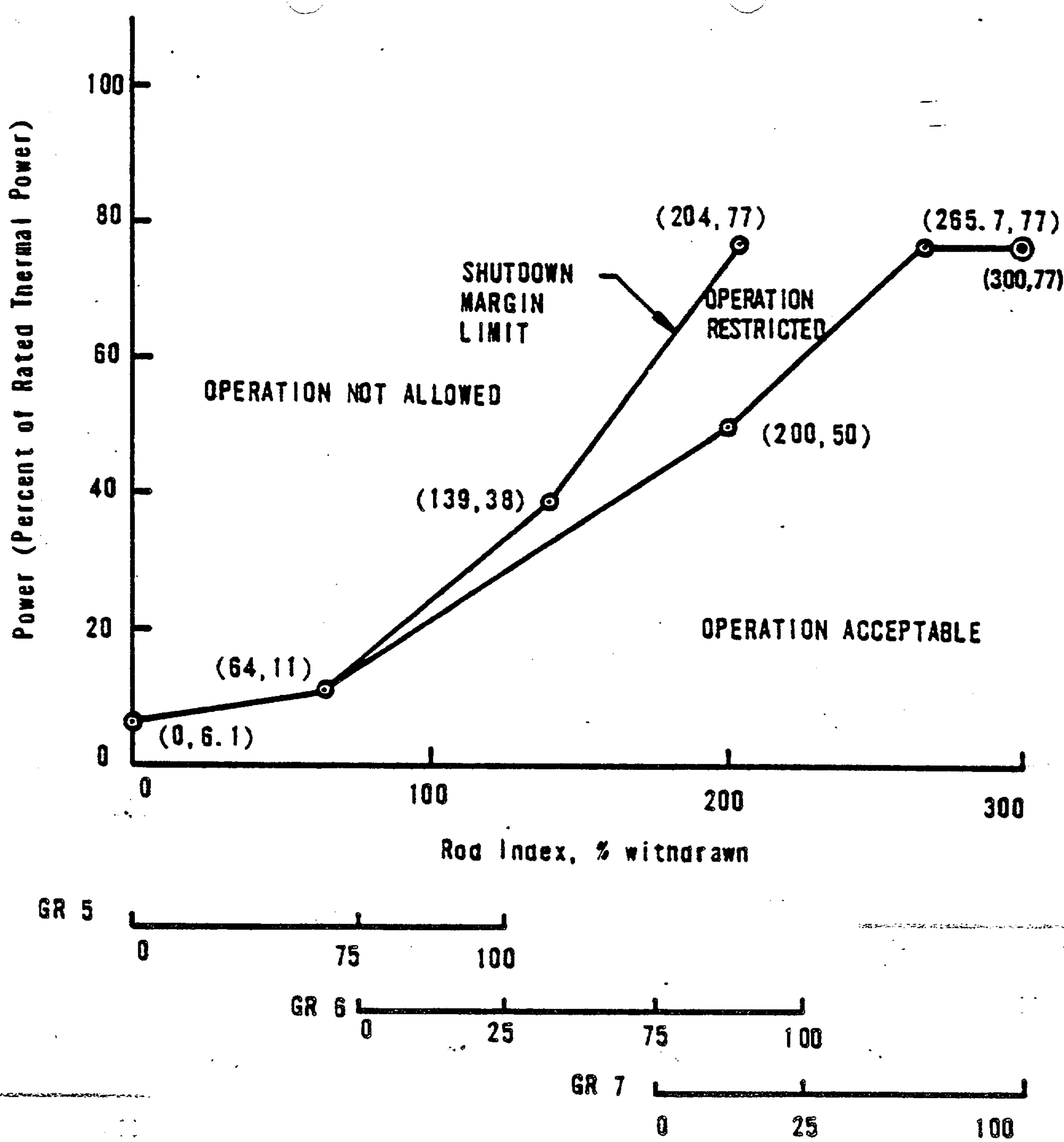
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION,
0 to 26 +10/-0 EFPD,
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-1A1



ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION,
26 +10/-0 TO 200 ±10 EFPD
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-1A2

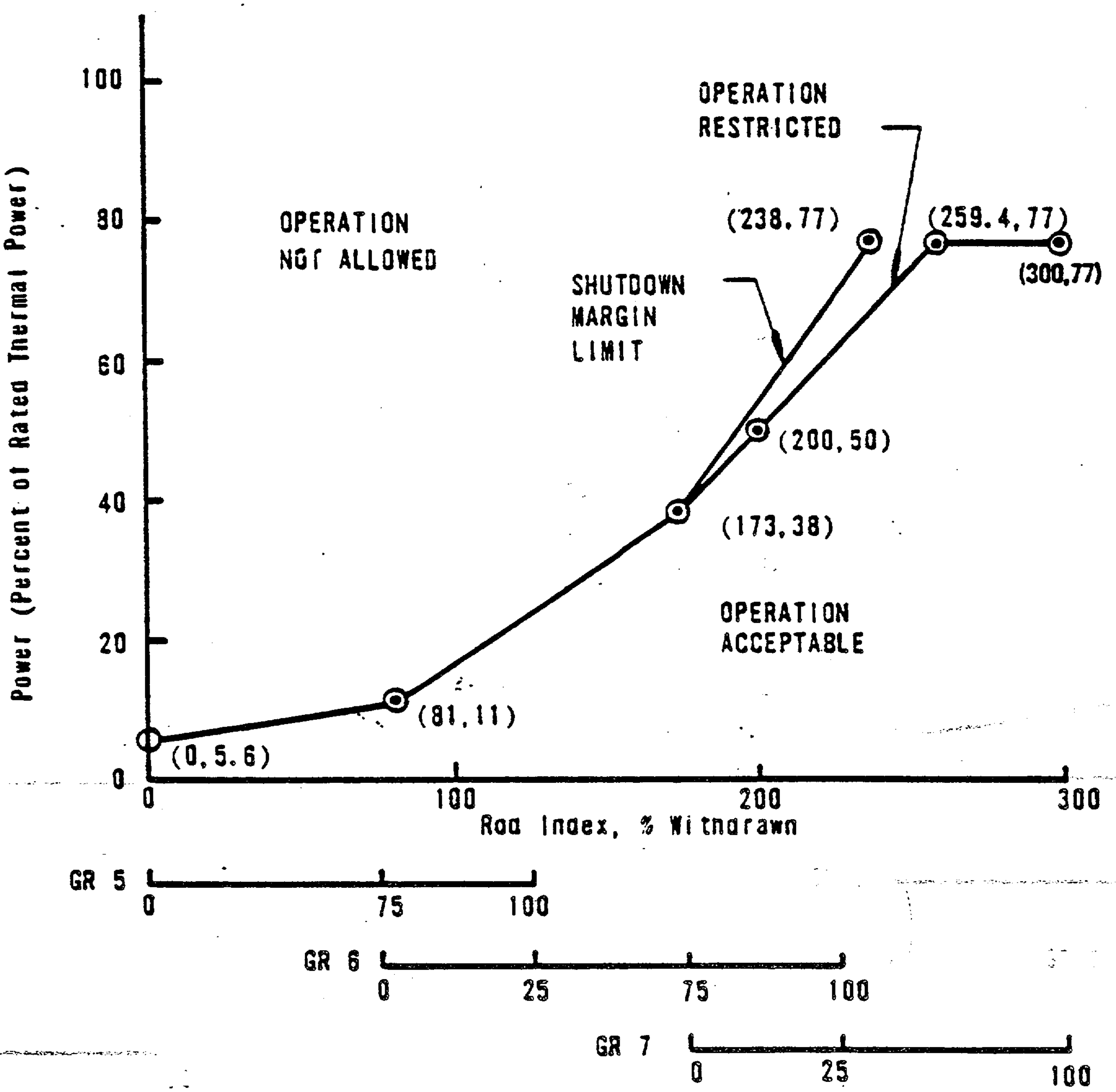


ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION,
AFTER 200 ±10 EFPD,
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-1A3



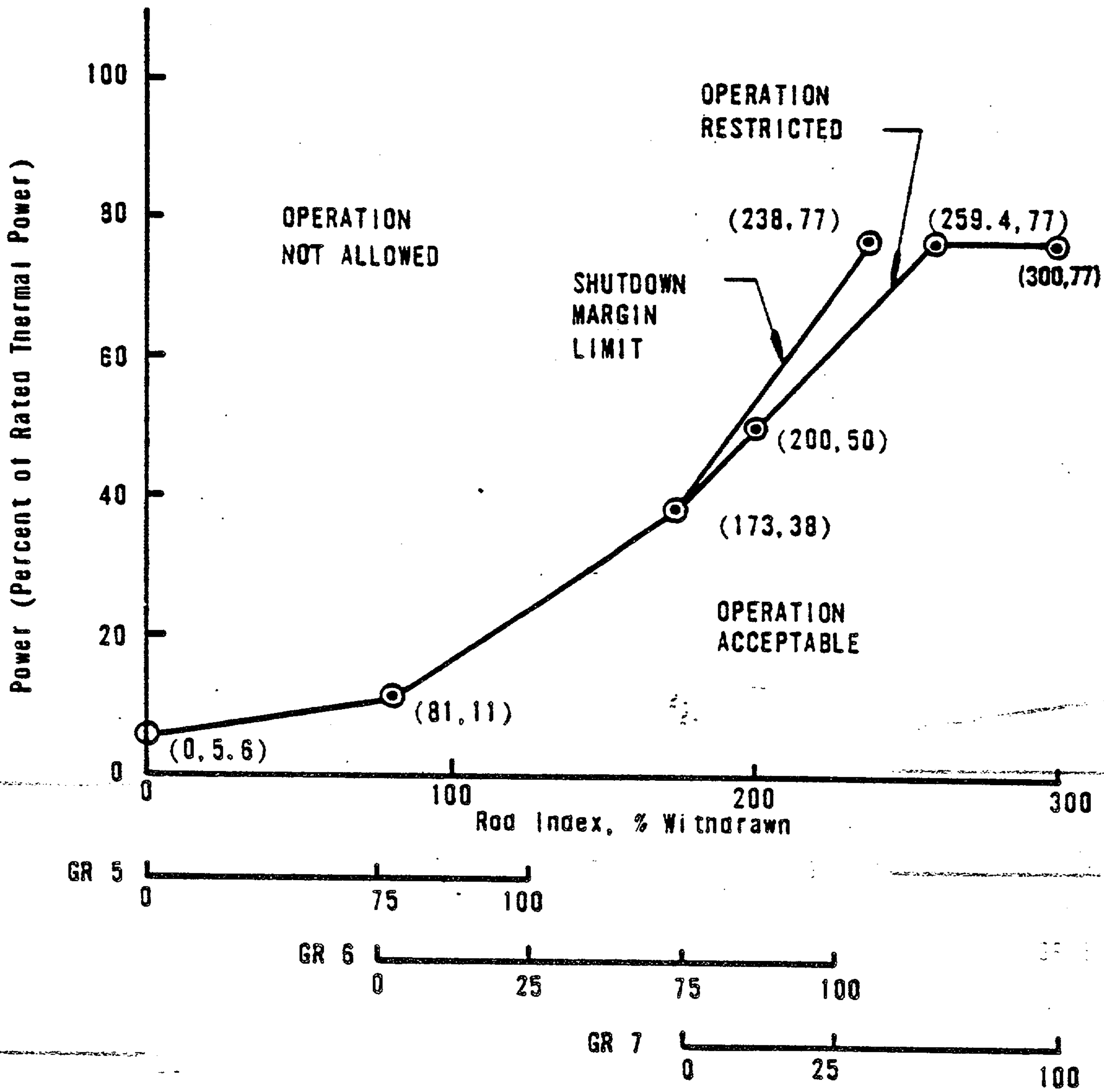
ROD POSITION LIMITS
FOR THREE-PUMP OPERATION,
0 to 26 +10/-0 EFPD,
OCONEE 1, CYCLE 8

OCONEE NUCLEAR STATION
Figure 3.5.2-2A1

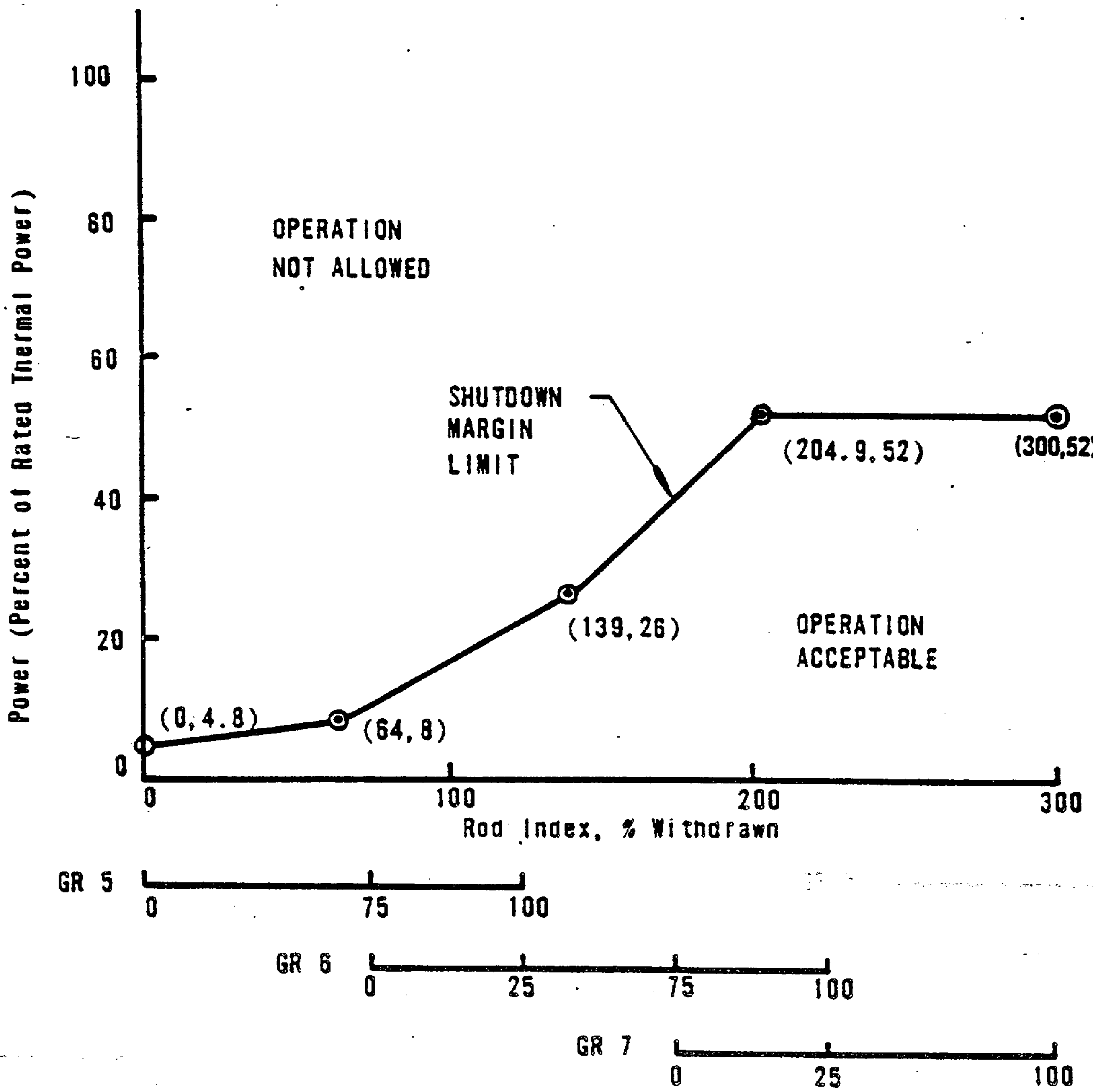


ROD POSITION LIMITS
FOR THREE-PUMP OPERATION,
26 +10/-0 TO 200 ±10 EFPD,
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-2A2

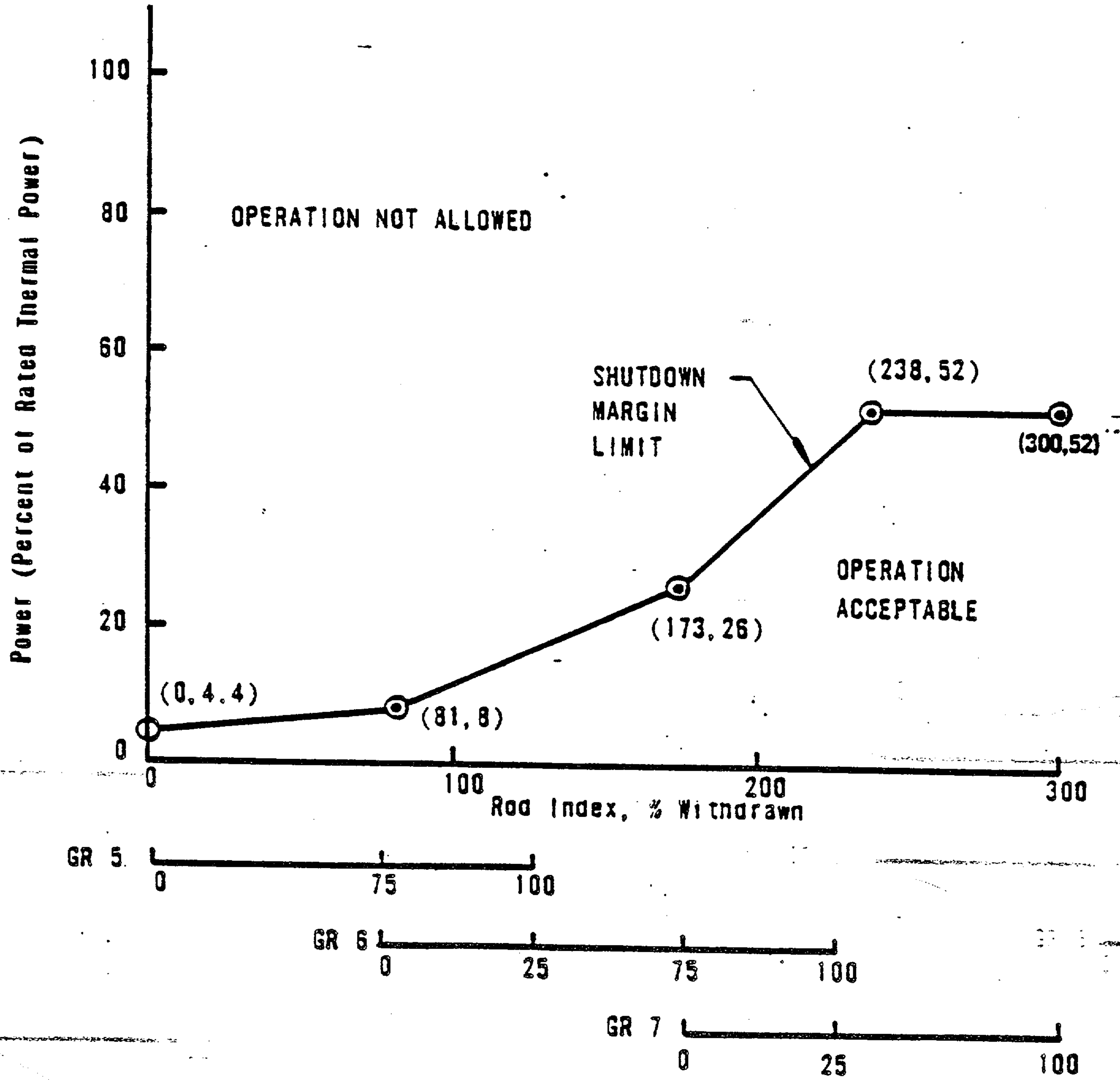
Amendment Nos. 122, 122 & 119



ROD POSITION LIMITS
FOR THREE-PUMP OPERATION,
AFTER 200 ±10 EFPD,
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-2A3

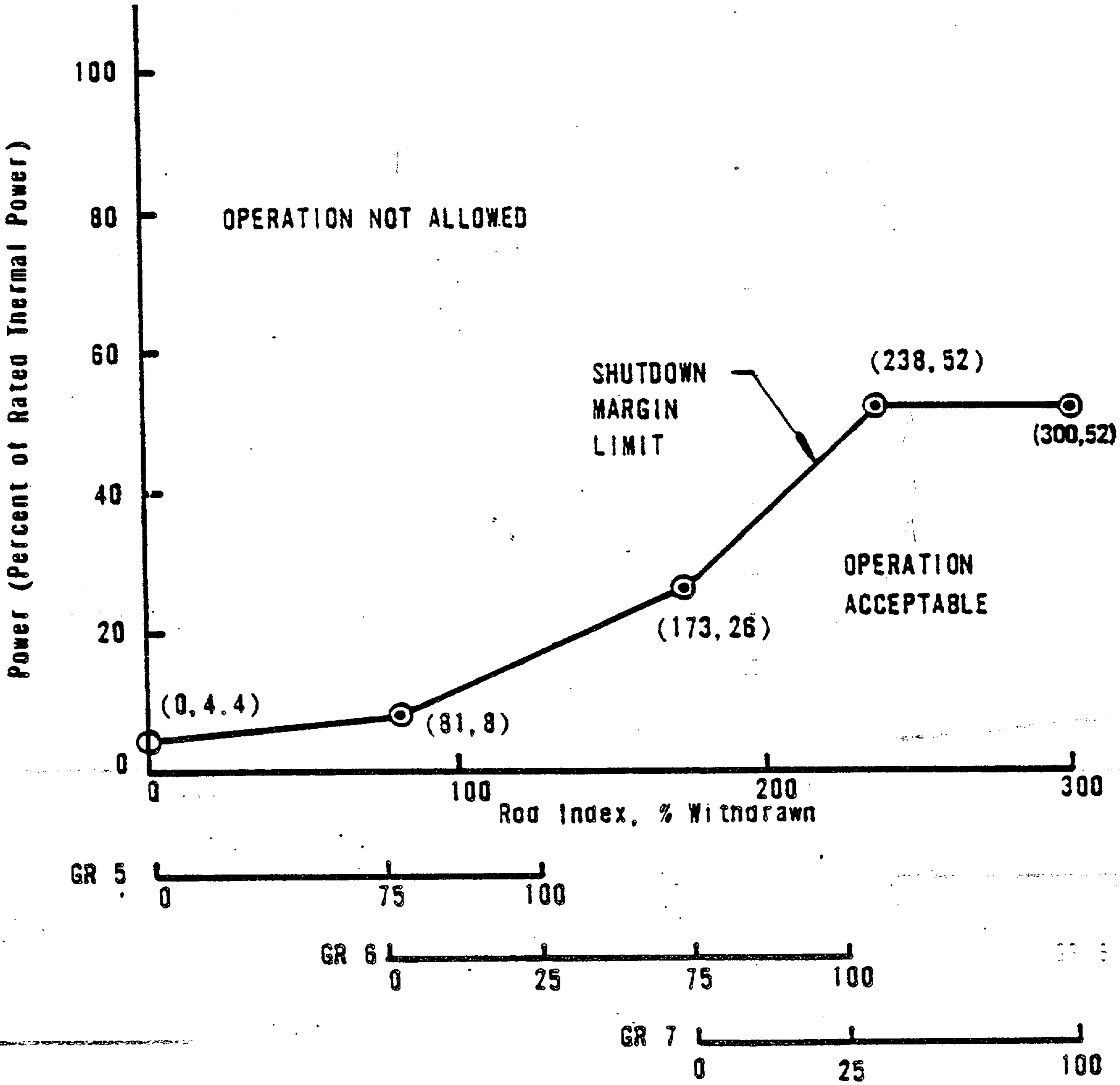


ROD POSITION LIMITS
FOR TWO-PUMP OPERATION,
0 to 26 +10/-0 EFPD,
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-2A4

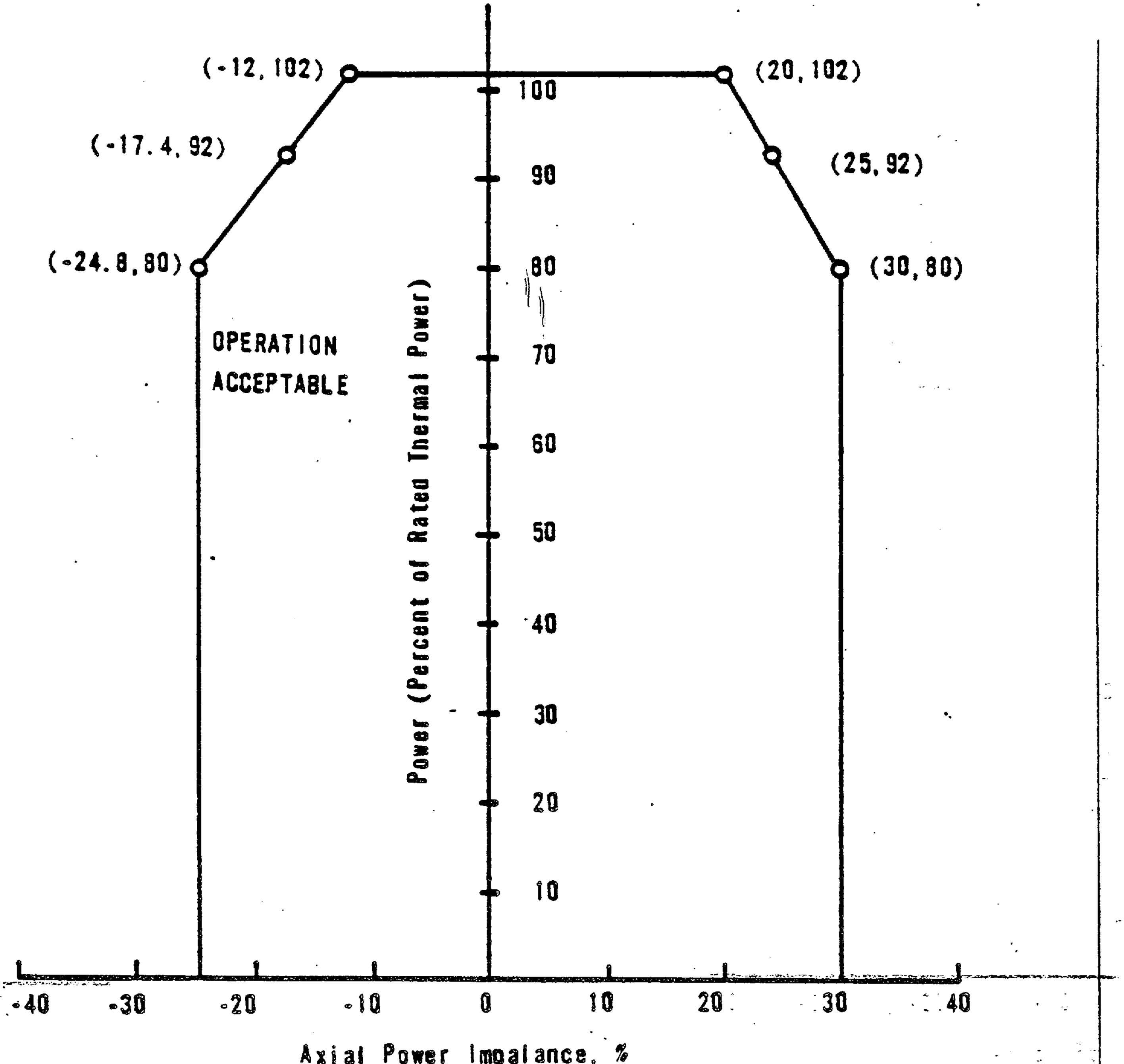


ROD POSITION LIMITS
FOR TWO-PUMP OPERATION,
26 +10/-0 TO 200 EFPD,
OCONEE 1, CYCLE 8

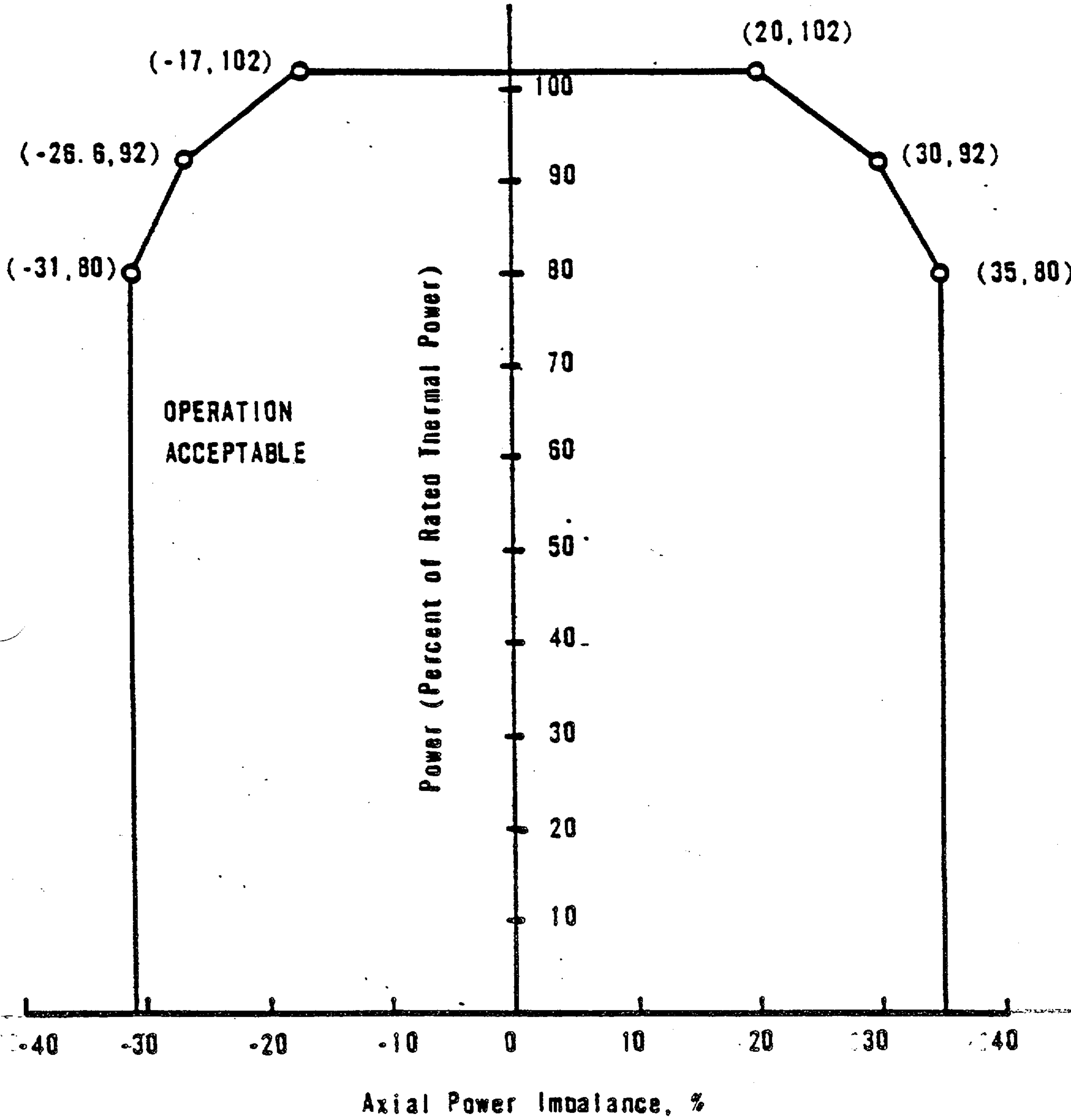
OCONEE NUCLEAR STATION
Figure 3.5.2-2A5



ROD POSITION LIMITS
FOR TWO-PUMP OPERATION,
AFTER 200 ±10 EFPD,
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-2A6



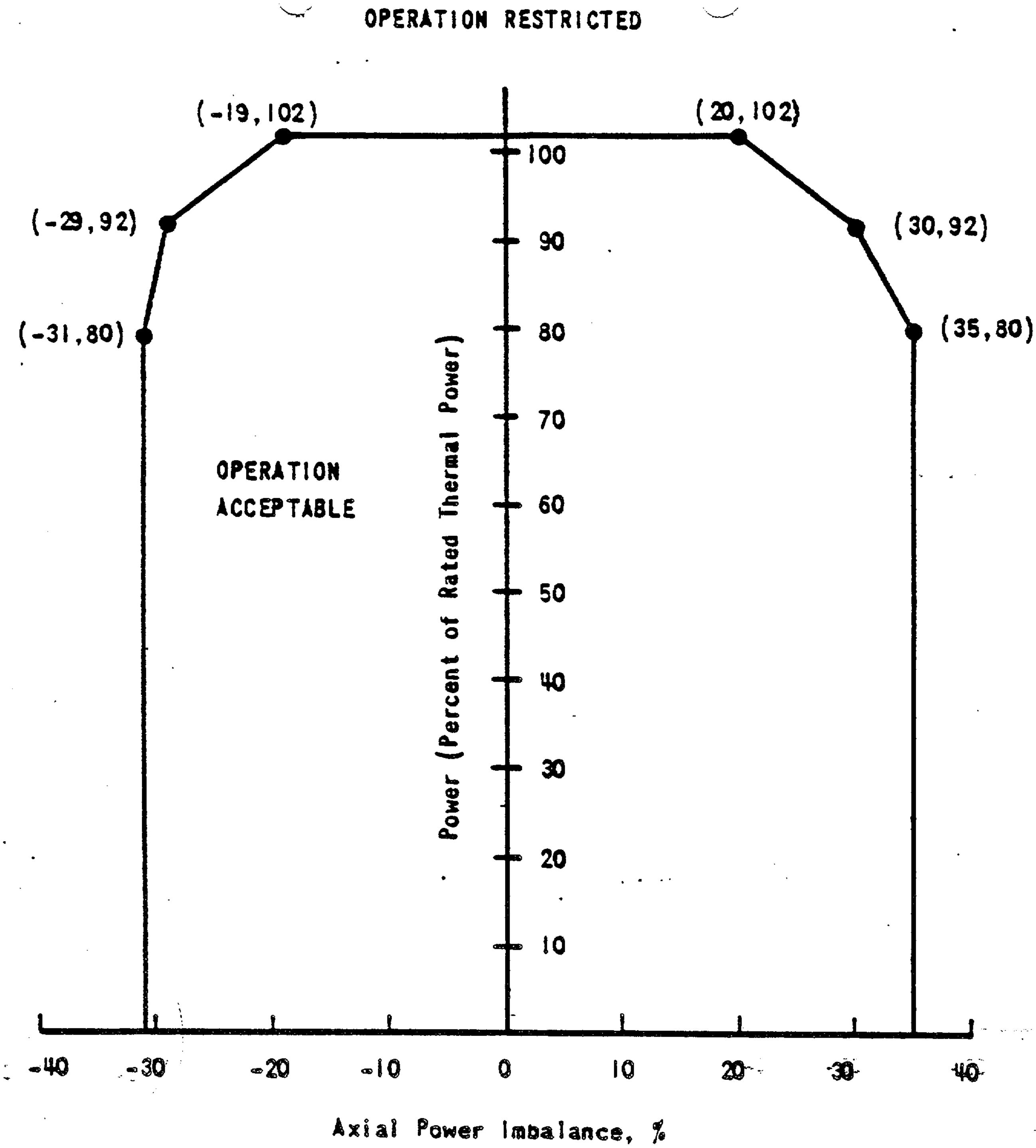
POWER IMBALANCE LIMITS
FOR 0 to 26 +10/-0 EFPD,
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-3A1



POWER IMBALANCE LIMITS,
26 +10/-0 to 200 ±10 EFPD
OCONEE 1, CYCLE 8

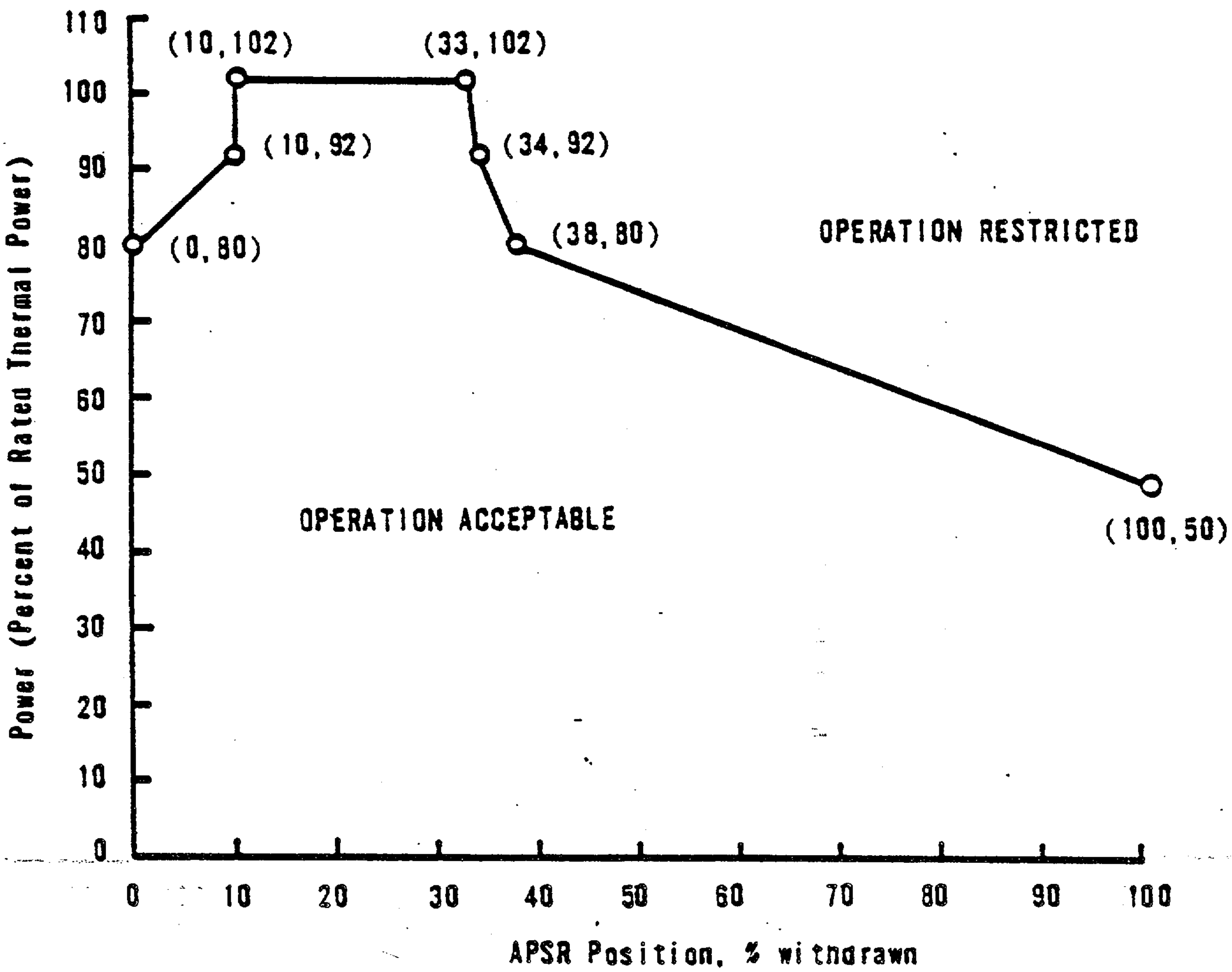
OCONEE NUCLEAR STATION
Figure 3.5.2-3A2

Amendment Nos. 122, 122 & 119

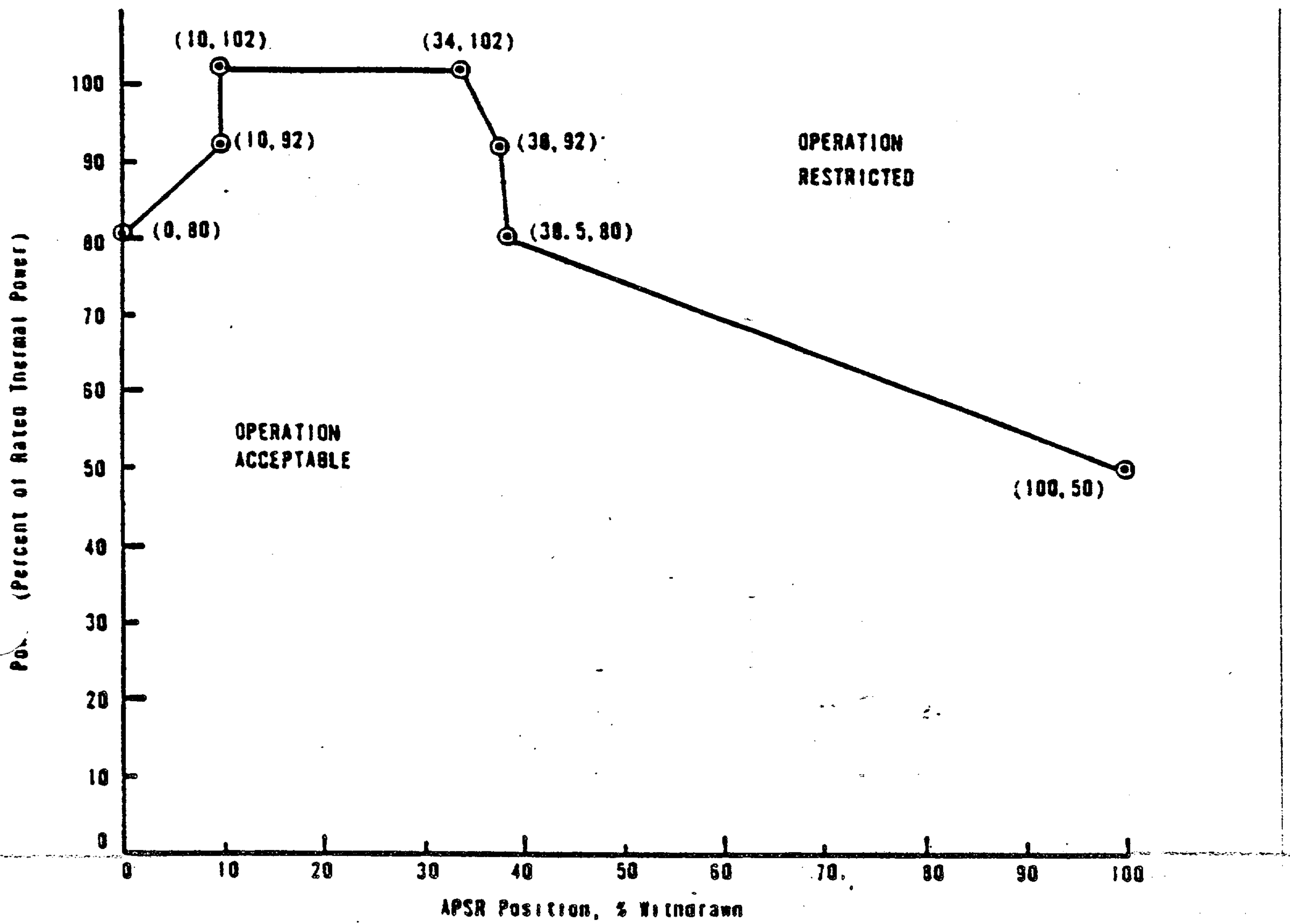


POWER IMBALANCE LIMITS
AFTER 200 ±10 EFPD
OCONEE 1, CYCLE 8

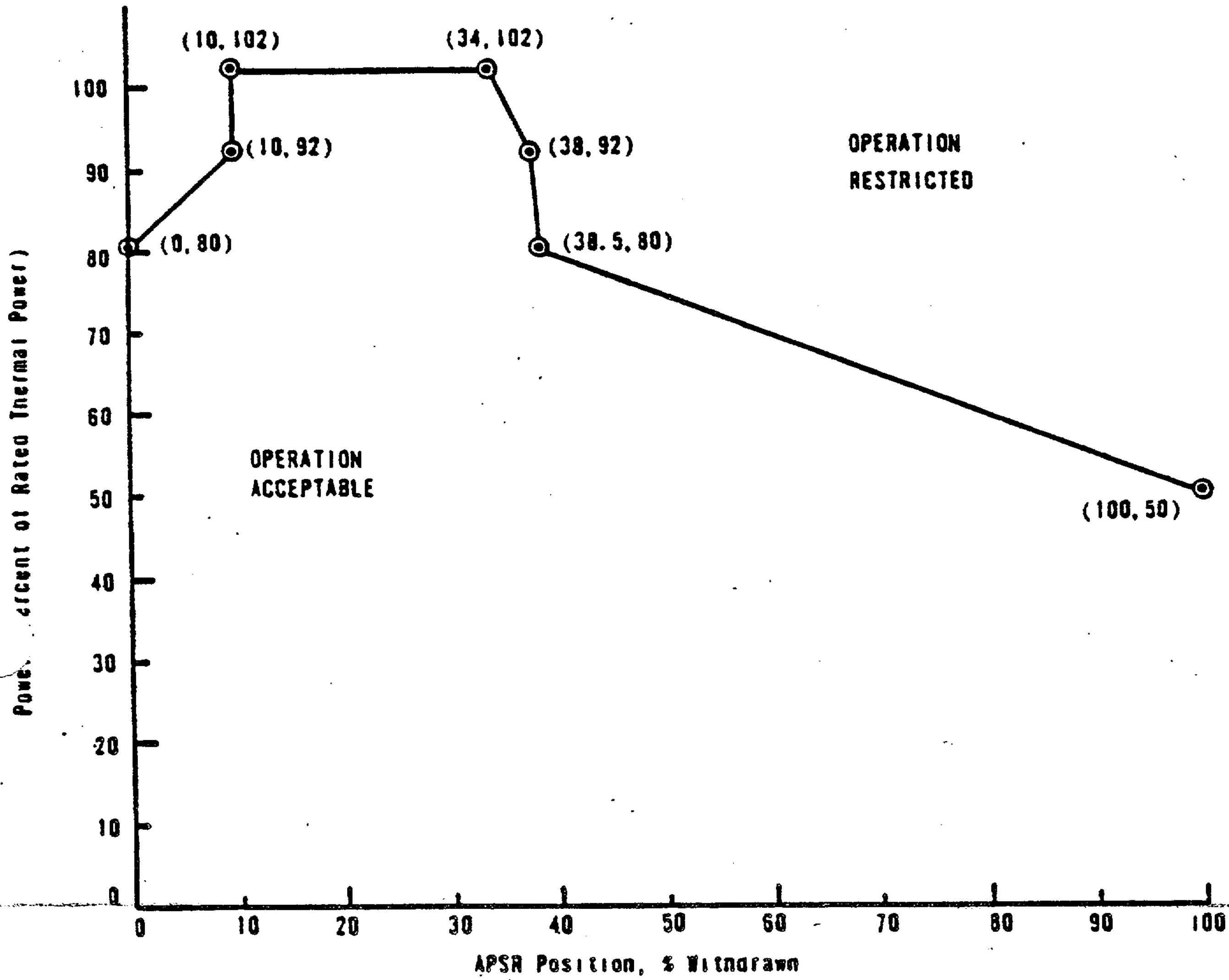
OCONEE NUCLEAR STATION
Figure 3.5.2-3A3



APSR POSITION LIMITS
FOR 0 to 26 +10/-0 EFPD,
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-4A1



APSR POSITION LIMITS,
26 +10/-0 TO 200 ±10 EFPD
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-4A2



APSR POSITION LIMITS
AFTER 200 ±10 EFPD
OCONEE 1, CYCLE 8
OCONEE NUCLEAR STATION
Figure 3.5.2-4A3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 122 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 122 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 119 TO FACILITY OPERATING LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS NOS. 1, 2 AND 3

DOCKETS NOS. 50-269, 50-270 AND 50-287

1.0 Introduction

By letter dated May 19, 1983 (Ref. 1), Duke Power Company (Duke or the licensee) made application to modify the Oconee Nuclear Station Technical Specifications in support of Cycle 8 operation of Unit 1. The analysis performed and the resulting modifications to the Station's common Technical Specifications are described in the Unit 1 Cycle 8 Reload Report (Ref. 2). The application also includes a modified version of the Oconee Nuclear Station Generic Startup Physics Test Program Report (Ref. 3).

The safety analysis for the previous seventh cycle of operation at Oconee Unit 1 is being used by the licensee as a reference for the proposed eighth cycle of operation. Where conditions are identified as limiting in the seventh cycle analysis, our previous evaluation (Ref. 4) of that cycle continues to apply.

1.1 Description of the Cycle 8 Core

The Oconee Unit 1 Cycle 8 core will consist of 177 fuel assemblies, each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 8 will operate in a bleed-and-feed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 61 full length control rod assemblies (CRAs) and 60 burnable poison rod assemblies (BPRAs). In addition, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution.

The length of Cycle 8 is expected to be 410 effective full power days (EFPD) of operation, somewhat lower than the 427 EFPD accumulated during Cycle 7. Due to the shorter cycle length, the average burnup for Cycle 8 will also be lower than the previous cycle, 12,858 MWd/MtU as compared to 13,363 MWd/MtU. The licensed core full power level remains at 2568 MWt.

2.0 Evaluation of the Fuel System Design

2.1 Fuel Assembly Mechanical Design

The 65 Babcock and Wilcox (B&W) Mark-B4 fuel assemblies loaded as Batch 10 at end of Cycle 7 (EOC 7) are mechanically interchangeable with Batch 8 and 9 fuel assemblies loaded previously at Oconee Unit 1. The Mark-B4 fuel assembly has been previously approved (Ref. 4) by the NRC staff and is utilized in other B&W nuclear steam supply systems. The oldest fuel in the core (designated Batch 8C) consists of 44 assemblies of the standard Mark-B4 design. Batch 9 consists of 64 assemblies of the standard design (designated Batch 9A) and also includes four once-burned Mark-BZ fuel assemblies (designated Batch 9B). The Mark-BZ design is similar to the standard Mark-B4 except that six imtermediate Inconel spacer grids have been replaced with Zircaloy grids. The design (Ref. 5) of these demonstration assemblies was reviewed and approved by the NRC staff (Ref. 4) for the previous cycle (Cycle 7) of operation at Oconee 1. We continue to find the use of these demonstration assemblies acceptable.

We are aware of a number of other recent changes to the B&W 15x15 fuel assembly design (e.g., a larger fuel assembly holddown spring, fuel pellets manufactured by an alternate supplier). These changes have been approved for use in other operating B&W 177-fuel-assembly plants on a limited basis and may be incorporated into future cycles of operation at Oconee Unit 1. However, for the current cycle of operation, the licensee has identified no other changes in the fuel assembly mechanical design. We find this acceptable.

2.2 Fuel Rod Design

Although the 65 fresh fuel assemblies in Batch 10 are all of the Mark-B4 design (and externally similar), five assemblies (denoted Batch 10A and 10B) will contain fuel pellets containing both urania (UO2) and gadolinia (Gd2O3) as described in Reference 6. These five lead test assemblies (LTAs) are part of a joint Duke Power/Babcock & Wilcox/Department of Energy program to develop and demonstrate an advanced fuel design incorporating UO2-Gd2O3 for extended burnup in pressurized water reactors (PWRs).

Gadolinium is a so-called burnable poison. That is, it contains isotopes which have large absorption cross sections and are converted to isotopes of low absorption cross section as the result of neutron absorption. Thus the increase in reactivity accompanying the burnup of the poison compensates to some extent for the decrease in reactivity due to fuel burnup and the accumulation of fission product poisons. Gadolinia is commonly used in boiling water reactor fuel designs (Ref. 7-9), but its use in PWRs has been limited. In addition to changing the neutronic properties (e.g., radial power distribution) of the fuel, the introduction of gadolinia is known to affect the physical properties (e.g., thermal conductivity, melting point) as well. The effects of gadolinia on irradiation properties (e.g., fuel swelling and fission gas release), particularly at the higher concentrations proposed for PWR designs, is not well known.

Under the provisions of 10 CFR 50.59, a licensee may conduct tests or experiments (i.e., incorporate LTAs in a reload core design) without prior NRC notification or approval. One of these provisions is that the proposed test or experiment does not involve a change in the plant Technical Specifications. In the case of Oconee Unit 1, the power-to-incipient-centerline melt limit (Page 2.1-2 of the Technical Specifications) has been modified to account for the lower power-to-melt values calculated for the gadolinia rods. However, a limited number of gadolinia-bearing rods are present in each LTA and the power peaking in those rods (for Cycle 8) is significantly less than the expected power peaking of gadolinia-free rods in either the LTAs or any other fuel assembly in the core. Thus, the licensee has concluded (Ref. 6) that the loading of the five extended-burnup lead test assemblies in the Oconee 1 Cycle 8 core will not adversely affect either the nuclear, mechanical, or thermal-hydraulic character of the reactor, or the existing safety anaylsis. Since the addition of gadolinia to the Oconee 1 Cycle 8 core appears to change only the design, as opposed to operating limits in the Technical Specifications, it is debatable whether formal review of this application is required. Nevertheless, we have reviewed this application and find it acceptable for the reasons discussed below.

We have, however, examined the licensee's report (BAW-1772 - Ref. 6) and noted that the evaluation performed includes all fuel system, nuclear and thermal-hydraulic design analyses, as well as the transient and accident evaluations, considered in a standard reload safety analysis. With the exception of the gadolinia properties (i.e., physical, neutronic and irradiation behavior), which were not covered in the report, the evaluation utilized methods previously reviewed and approved by the NRC. Because of the exception, we are unable to confirm the licensee's findings. However, the results appear similar to those submitted by other manufacturers of gadolinia-bearing fuel and the report provides a reasonable basis for our approval of the LTA irradiations. Our approval is limited to Cycle 8 (as the analyses presented were limited to Cycle 8) and should not be construed as an approval of this design for full-scale applications (because of the use of unreviewed properties and other information). We will pursue the issue of the continued irradiation of these lead test assemblies at the time the Oconee 1 Cycle 9 reload application is made.

The cladding stress, strain and collapse analyses for the standard fuel designs in the core are bounded by conditions previously analyzed for Oconee Unit 1 or were analyzed specifically for Cycle 8 using methods and limits previously reviewed and approved by the NRC. We find that no further review of these areas is necessary.

2.2.1 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (SRP) (Ref. 10) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Among those which may affect the operation of the fuel rod is the internal pressure limit. The acceptance criterion (SRP 4.2, Section II.A.1(f)) is that the fuel rod internal gas pressure should remain below normal system pressure unless otherwise justified.

The licensee has stated that the fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 8. This analysis is based on the use of the B&W TAFY-3 code (Ref. 11) rather than one of the newer B&W codes, TACO-1 (Ref. 12) and TACO-2 (Ref. 13). Although all of these codes have been approved for use in safety analysis, we believe (Ref. 14) that only the newer TACO series of codes are capable of correctly calculating fission gas release (and therefore rod pressure) at very high burnups. Babcock & Wilcox has responded (Ref. 15) to this concern with an analytical comparison between the TAFY-3 and TACO-1 codes. In this response, they have stated that the fuel rod internal pressure predicted by TACO-1 is lower than that predicted by TAFY-3 for fuel rod exposures of up to 42,000 MWd/MtU. Although we have not examined this comparison, we note that the analyses exceed the maximum expected exposure (40,238 MWd/MtU) for all fuel rods in the Oconee Unit 1 core at the end of Cycle 8. We conclude that the rod internal pressure limits have been adequately considered for Cycle 8 operation.

2.3 Fuel Thermal Design

The thermal behavior of all fuel in the Cycle 8 core, with the exception of the gadolinia-bearing lead test assemblies, is virtually identical. In general, the thermal analysis was performed with the approved version of TACO-2. We find this acceptable.

For the Loss of Coolant Accident (LOCA) analysis (Section 7.2 of the Reload Report), the average fuel temperature as a function of linar heat rate and the lifetime pin pressure data were calculated with the older TAFY-3 code. The licensee has stated that the fuel temperature and pin pressure data used in the generic LOCA analysis are conservative compared with those calculated for Cycle 8 at Oconee Unit 1.

As mentioned previously, B&W currently has several fuel performance codes which are approved and could be used to calculate LOCA initial conditions. The older TAFY-3 code was used for the generic LOCA analysis cited in the Oconee Unit 1 Cycle 8 Reload Report. Information obtained by the NRC staff (Ref. 16) indicates that the TAFY-3 code predictions do not produce higher calculated peak cladding temperatures in the generic LOCA analysis than the newer TACO-1 or TACO-2 codes as suggested by the licensee. The issue involves excessive fuel densification and lowered fuel rod internal gas pressures at beginning of life. Babcock and Wilcox has proposed a method of resolving this issue which has been adopted by Duke Power Company (Ref. 17). The method relies on reduced peak linear heat rate (PLHR) limits at low core elevations for the first 26 EFPD of operation based on comparison of TAFY-3 and TACO-2 calculated LOCA initial conditions. The method is similar to an older TAFY-3/TACO-1 comparison (Ref. 2) used in the original Oconee Unit 1 Cycle 8 safety analysis. However, the resulting PLHR reduction is different for each code.

In addition to the issue of initial fuel temperatures and rod internal pressures used in the LOCA analysis, a second issue involving cladding swelling and rupture models has affected the proposed Cycle 8 operating limits for Oconee 1.

In late 1979, the NRC staff reviewed Emergency Core Cooling System fuel cladding models in light of new data. Adequacy of the models then in use was questioned and new models, developed as Appendix K acceptance criteria, were presented in NUREG-0630 (Ref. 18). Each fuel vendor was then asked to show how, in light of the new models, the plants analyzed with their analytical methods continued to meet the applicable LOCA limits. The B&W response (Ref. 19) concluded that the impact of the NRC models was small and did not result in analytical results in excess of the LOCA limits.

A more recent B&W calculation (Ref. 20), however, found that the cladding swelling and rupture models presented by the staff have a non-trival effect on LOCA peak cladding temperatures in B&W 177 fuel assembly plants. Because this calculation was applicable to all B&W plants, the licensee was requested (Ref. 21) to provide supplemental calculations for Oconee Unit 1 similar to those provided in Reference 20. The licensee's responses (Refs. 22, 23 and 24) culminated in the supplemental calculation (Ref. 17) cited previously. This calculation, which considers both fuel densification (TAFY-3/TACO-2) and cladding swelling and rupture effects, results in low core elevation PLHR limits which are more restrictive than those which consider only fuel densification. The licensee has proposed (Ref. 17) modification to the Oconee Station Technical Specifications which account for these reduced PLHR limits.

In general, the supplemental calculation utilizes previously approved methods except for the substitution of the NRC cladding models. However, there are segments of the analysis (e.g. THETA1-B - Ref. 25) that are currently undergoing NRC review. The licensee has also presented results from a calculation using a new FLECTSET heat transfer correlation (Refs. 26 and 27). This correlation appears to offset the NUREG-0630 penalties. The licensee has not yet claimed these FLECTSET benefits, however, because the benchmarking and other final evaluations of FLECTSET have not been completed and provided to the NRC for review.

Considering the above, we conclude that the licensee's proposed Technical Specification changes are both appropriate and necessary. Since these operating limits are more restrictive than those previously used at the Oconee Station, since they are only needed for a brief time period, and since potential but unused compensating benefits may exist, we, therefore, conclude that the operating restrictions imposed on an interim basis are acceptable for incorporating the NUREG-0630 penalties until our final evaluation of FLECSET is completed.

3.0 Evaluation of the Nuclear Design

3.1 Physics Characteristics

The nuclear characteristics of the Oconee 1 Cycle 8 core have been computed by methods previously used and approved for B&W reactors. Comparisons are made between the physics parameters for Cycles 7 and 8. The differences that exist between the parameters are due to the decreased cycle length and the higher average burnable poison enrichment which tends to decrease values of critical boron concentrations. Changes in the radial flux and burnup distributions between cycles also account for the differences in control rod worths, including ejected and stuck rod worths. All safety criteria are still met. Beginning-of-cycle radial power distributions show acceptable margins to limits.

Shutdown margin calculations for Cycle 8 include the effects of poison material depletion, a 10% calculational uncertainty, and flux redistribution as well as a maximum worth stuck rod. Beginning and end-of-cycle shutdown margins show adequate reactivity worth exists above the total required worth during the cycle. The required shutdown margin is 1.00% $\Delta k/k$, the shutdown margin at the beginning and end-of-cycle is 3.11% $\Delta k/k$ and 2.28% $\Delta k/k$, respectively.

Based on our review, we conclude that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of Oconee 1 Cycle 8 is acceptable.

3.2 Gadolinia Lead Test Assemblies

Four of the unirradiated Mark-GdB fuel assemblies in the Oconee 1 Cycle 8 core will have an initial enrichment of 4.0 weight percent U-235. Storage of this maximum enrichment in the Oconee Unit spent fuel storage racks has been reviewed and approved by the staff (Ref. 28). The effect of these higher enriched fuel assemblies on the nuclear design has been taken into account for Cycle 8 and the design continues to meet all criteria including those applicable to radial power peaking, ejected rod worths, and shutdown margin.

3.3 Startup Physics Test Program

A modified version (Ref. 3) of a report entitled "Oconee Nuclear Station Generic Startup Physics Test Program" has been reviewed. The modifications consist of:

1. Changing the names of the 40% FP, 75% FP and 100% FP Power Distribution Tests to low, intermediate and full power core mapping.

2. Establishing power ranges for this mapping and for the core symmetry test.

3. Changes to the critical boron concentration tests.

Establishing power ranges for the mapping permits operating flexibility with no loss of information from the tests. Establishing a lower power range for the symmetry test takes advantage of a planned change to the computer software and will allow earlier identification of core power distribution problems. The changes to the critical boron concentration tests are minor procedural changes.

We have reviewed these changes and find them acceptable.

4.0 Evaluation of the Thermal-Hydraulic Design

In Section 6 of the B&W report BAW-1774 (Ref. 2), the licensee has described the thermal-hydraulic design. Cycle 7 is used as the reference cycle for the thermal-hydraulic evaluation. Table 1 shows a comparison of the maximum design conditions for Cycles 7 and 8. As seen from the table, the maximum design conditions are unchanged from Cycle 7.

Cycle 8 fuel includes four Mark BZ-demonstration assemblies and five LTA demonstration assemblies. These assemblies have a design peak of 1.61 (6% peaking reduction) to ensure that they are not thermally limiting. All other assemblies have a 1.71 design radial times local peak. We find the incorporation of these four Mark-BZ and five LTA demonstration assemblies acceptable since their design peak is at least 6% less than the remaining assemblies.

A rod bow penalty was calculated using an approved (Ref. 29), interim procedure for calculating departure from nucleate boiling ratio (DNBR) reduction due to rod bow. The licensee used the maximum fuel assembly burnup of the batch that contains the maximum radial-local peak. For Cycle 8, that burnup is 17,511 MWd/MtU in a Batch 10C assembly. The resultant net rod bow penalty, after inclusion of the 1% flow area reduction credit, is 0.2% reduction in DNBR. We find this acceptable since the thermal-hydraulic design for Cycle 8 includes a margin greater than 0.2% above the minimum DNBR of 1.30.

Table 1. Thermal-Hydraulic Design Conditions

	Cycle 7	Cycle 8
Power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	106.5
Vessel inlet coolant temp, 100% power, F	555.6	555.6
Vessel outlet coolant temp, 100% power, F	602.4	602.4
Ref design axial flux shape	1.5 cos	1.5 cos
Ref design radial-local power peaking factor	1.71	1.71
Active fuel length, in.	(a)	(a)
Average heat flux, 100% power, 10^3 Btu/h-ft^2	176[b]	176[b]
CHF correlation	BAW-2	BAW-2
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Minimum DNBR with densification penalty	2.05	2.05

(a) See Table 4-2 of Reference 2

(b) Based on densified length of 140.3 in.

5.0 Evaluation of Accident and Transient Analysis

The key kinetics parameters for Oconee 1 Cycle 8 have been compared to the values used in the Final Safety Analysis Report (FSAR) and densification report. It is shown that in all cases Cycle 8 values are bounded by those previously used. We conclude that the FSAR or previous reference cycle transient and accident analyses are valid.

Three sets of bounding values for allowable LOCA peak linear heat rates are given as a function of core height (Ref. 17). These limits apply during the periods 0-26 EFPD, 26-200 EFPD, and 200 to end-of-cycle. The limits are satisfactorily incorporated into the Technical Specifications for Cycle 8 through the operating limits on rod index, axial power shaping rod limits, and axial power imbalance.

6.0 Evaluation of Technical Specification Modifications

We have reviewed the proposed Technical Specifications for Cycle 8. The limiting conditions for operation have been established by previously used and approved methods. The rod withdrawal limits for the various pump combinations and times in life are presented. On the basis that previously approved methods were used to obtain the limits, we find them acceptable.

7.0 Summary

We conclude from the examination of Cycle 8 core thermal and kinetic properties with respect to acceptable previous cycle values and with respect to the FSAR values, that this core reload will not adversely affect the Oconee Nuclear Station's ability to operate safely during Cycle 8 of Unit 1.

8.0 Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

9.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 3, 1983

The following NRC staff personnel have contributed to this Safety Evaluation: J. Suermann, J. Vogelwede, L. Kopp, A. Gill, M. Chatterton.

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