2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

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

Objective

To maintain the integrity of the fuel cladding.

Specification

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

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

Bases - Unit 1

The safety limits presented for Oconee Unit 1 have been generated using the BAW-2 & BWC critical heat flux (CHF) correlations[1,3]. The BAW-2 correlation applies to fuel batches 9B and 10C while the BWC correlation applies to batches 10B and 11. The reactor coolant system flow rate utilized is 106.5 percent of the design flow (131.32 x 10^6 lbs/hr) based on four-pump operation.[2]

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure

can be related to DNB through the use of the CHF correlations ([1],[3]). The BAW-2 and BWC correlations have 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 (BAW-2) or 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) 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 the minimum allowable DNBR is predicted to occur for the limiting combination of thermal power and reactor coolant pump configuration. The curve is based upon the design nuclear power peaking factors including potential effects of fuel densification.

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 combination of the radial peak, axial peak and position of the axial peak that yields no less than the CHF correlation limit.

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 9B, and 10C, and 11 Batches of fuel and 17.6 kw/ft for the 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.

A B&W topical report discussing the mechanisms and resulting effects of fuel rod bow has been approved by the NRC ([4]). The report concludes that the DNBR penalty due to rod bow is insignificant and unnecessary because the power production capability of the fuel decreases with irradiation. Therefore, no rod bow DNBR penalty needs to be considered for thermal-hydraulic analyses.

The maximum thermal power for three-pump operation is 88.8 percent due to a power level trip produced by the flux-flow ratio (74.7 percent flow x 1.08 = 80.67 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 the CHF correlation limit 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.

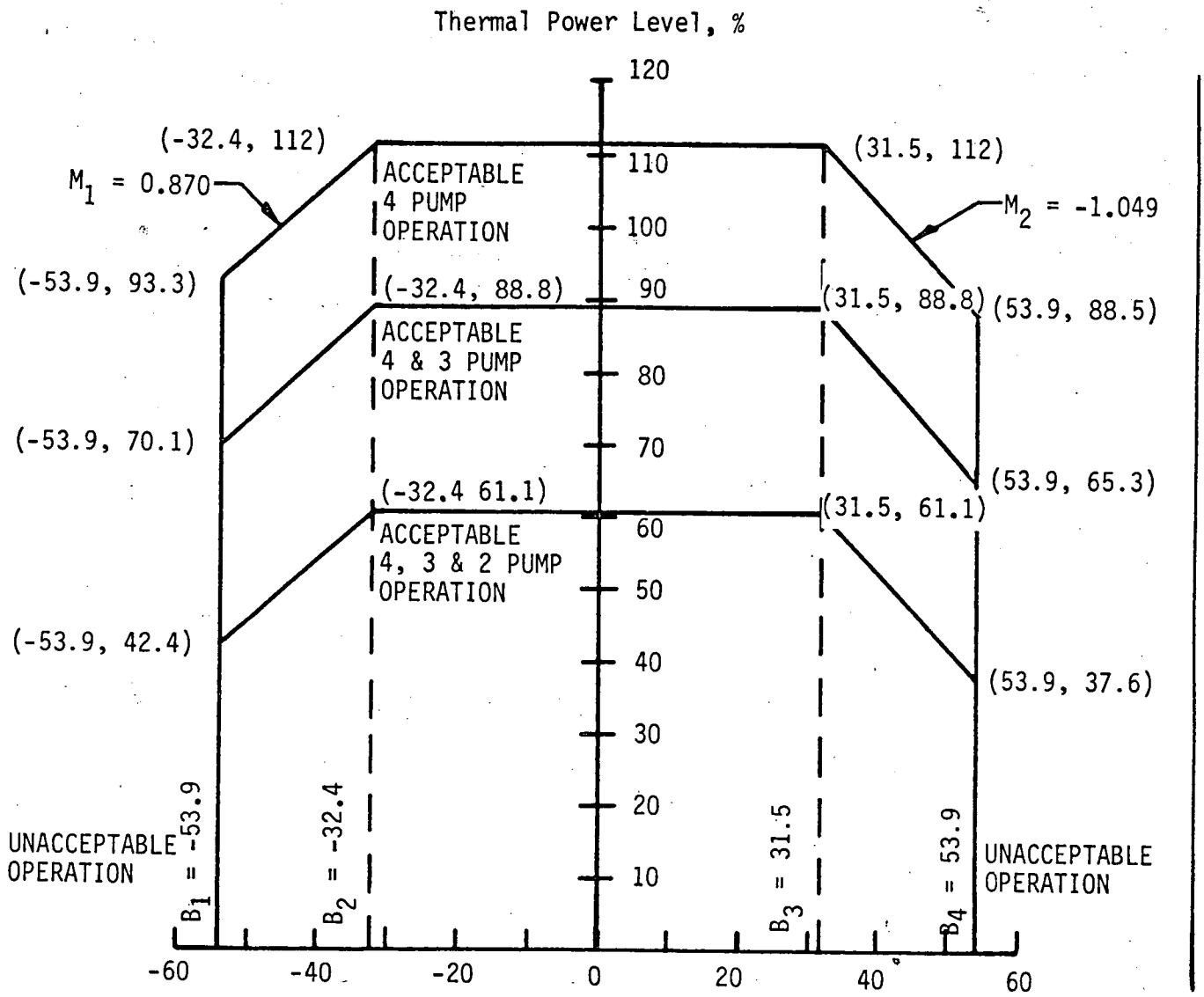
References

(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) Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143P, Part 2, Babcock & Wilcox, Lynchburg, Virginia, August 1981.

(4) Fuel Rod Bowing in Babcock & Wilcox Fuel Designs, BAW-10147P-A, Rev. 1, Babcock & Wilcox, May 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

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

Overpower Trip Based on Flow and Imbalance

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

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

1. Trip would occur when four reactor coolant pumps are operating if power is 108% and reactor flow rate is 100%, or flow rate is 92.59% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 80.67% and reactor flow rate is 74.7% or flow rate is 69.44% and power level is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is 52.92% and reactor flow rate is 49.0% or flow rate is 45.37% and the power level is 49%.

   The analyses to determine the flux-to-flow ratios account for calibration and instrument errors and the maximum variation in RC flow in such a manner as to ensure a conservative setpoint. A Monte-Carlo simulation technique is used to determine the combined effects of calibration and instrument uncertainties with the final string uncertainties used in the analyses corresponding to the 95/95 tolerance limits.

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

level trip and associated reactor power/reactor power-imbalance boundaries by 1.08% - Unit 1 for 1% flow reduction.
1.07% - Unit 2
1.08% - Unit 3

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below the minimum allowable value by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure setpoint is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Figure 2.3-1A - Unit 1
2.3-1B - Unit 2
2.3-1C - Unit 3
for high reactor coolant system pressure (2300 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient. (1)

The low pressure (1800) psig and variable low pressure (11.14 T_{out}-4706) trip
(1800) psig (11.14 T_{out}-4706)
(1800) psig (11.14 T_{out}-4706)
setpoints shown in Figure 2.3-1A have been established to maintain to DNB
2.3-1B
2.3-1C
ratio greater than or equal to the minimum allowable value for those design accidents that result in a pressure reduction. (2,3)

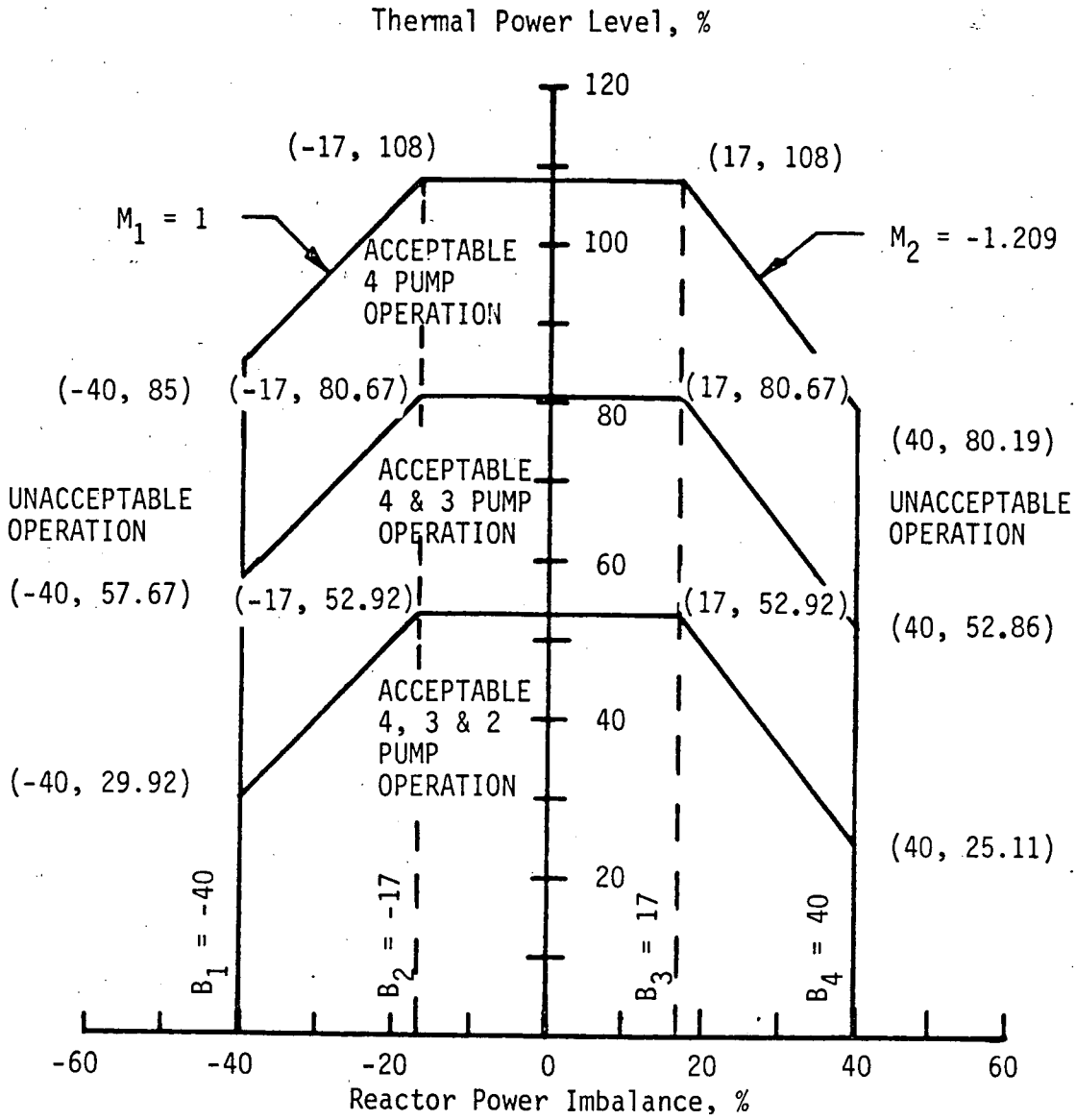
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of (11.14 T_{out} - 4746)
(11.14 T_{out} - 4746)
(11.14 T_{out} - 4746)

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618°F) shown in Figure 2.3-1A has been established to prevent excessive core coolant
2.3-1B
2.3-1C
temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620°F.

Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.



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

Table 2.3-1A
Unit 1

Reactor Protective System Trip Setting Limits

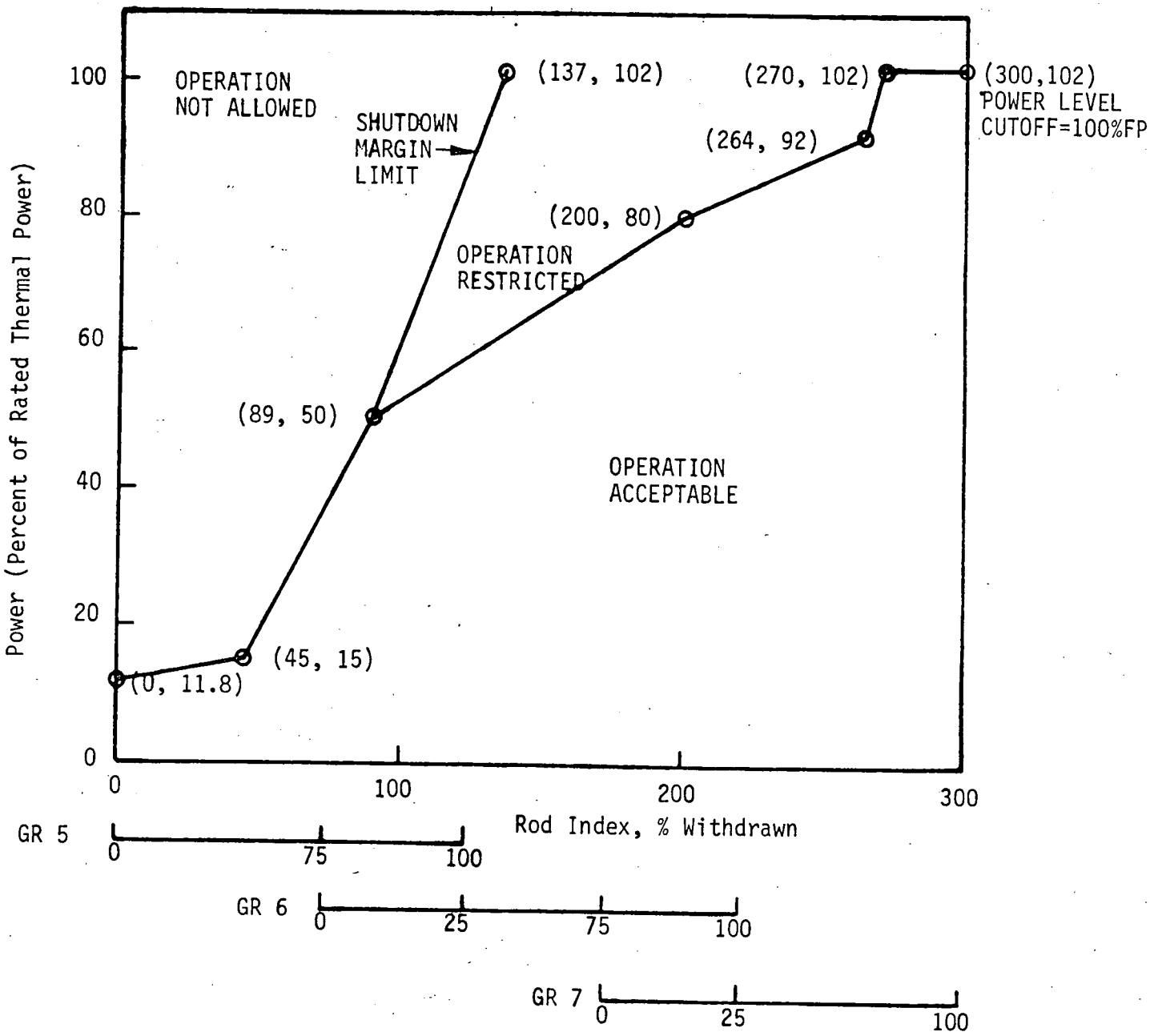
	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power 100% Rated	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated	One Reactor Coolant Pump Operating In Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imblance, (% Rated)	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	1.08 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Bases on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	$(11.14\ T_{out} - 4706)$(1)	Bypassed
7.	Reactor Coolant Temp. F., Max.	618	618	618	618
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

(1) T_{out} is in degrees Fahrenheit (°F).

(2) Reactor Coolant System Flow, %.

(3) Administratively controlled reduction set only during reactor shutdown.

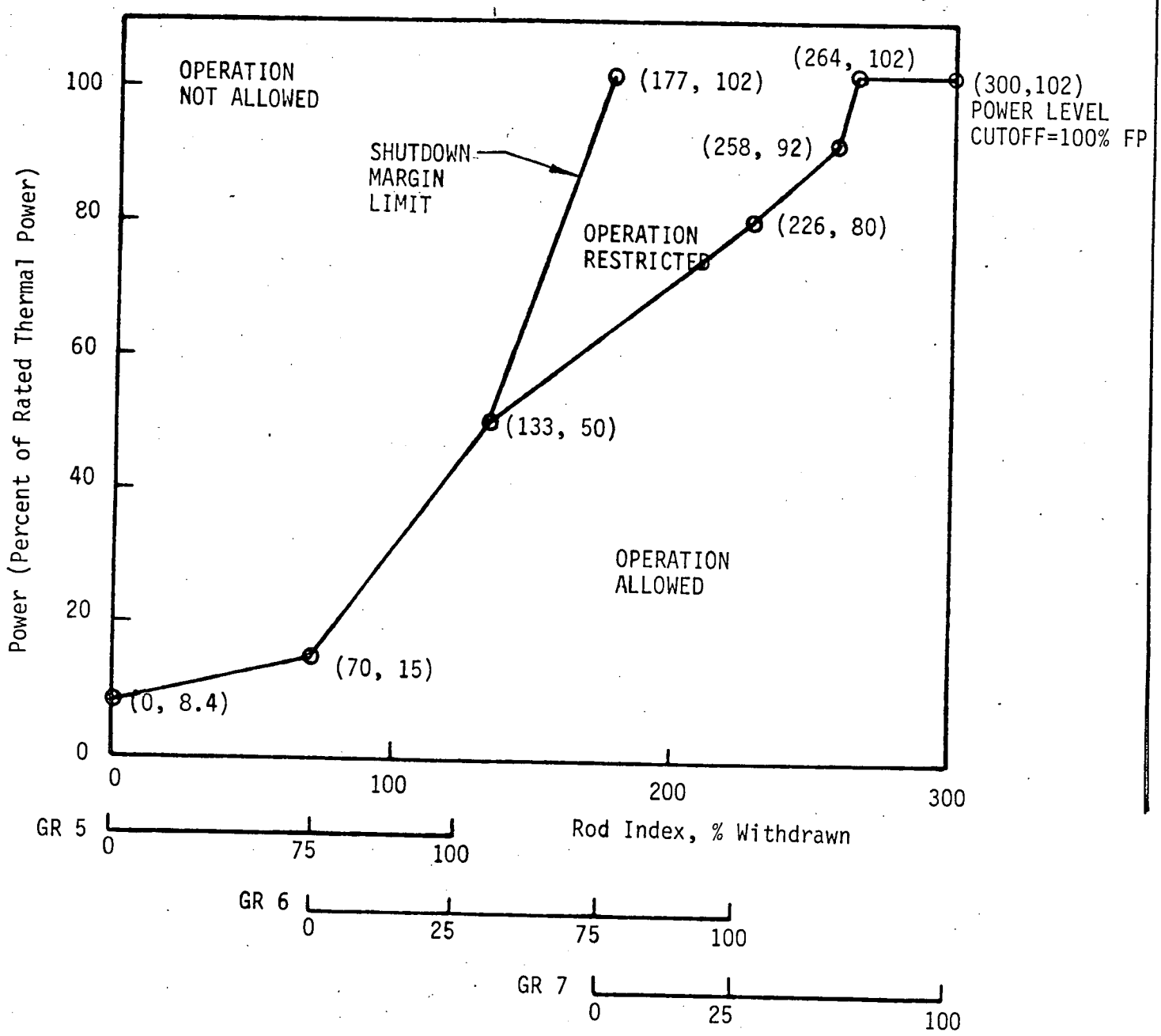
(4) Automatically set when other segments of the RPS are bypassed.



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 TO 30 + 10/0 EFPD
UNIT 1

OCONEE NUCLEAR STATION

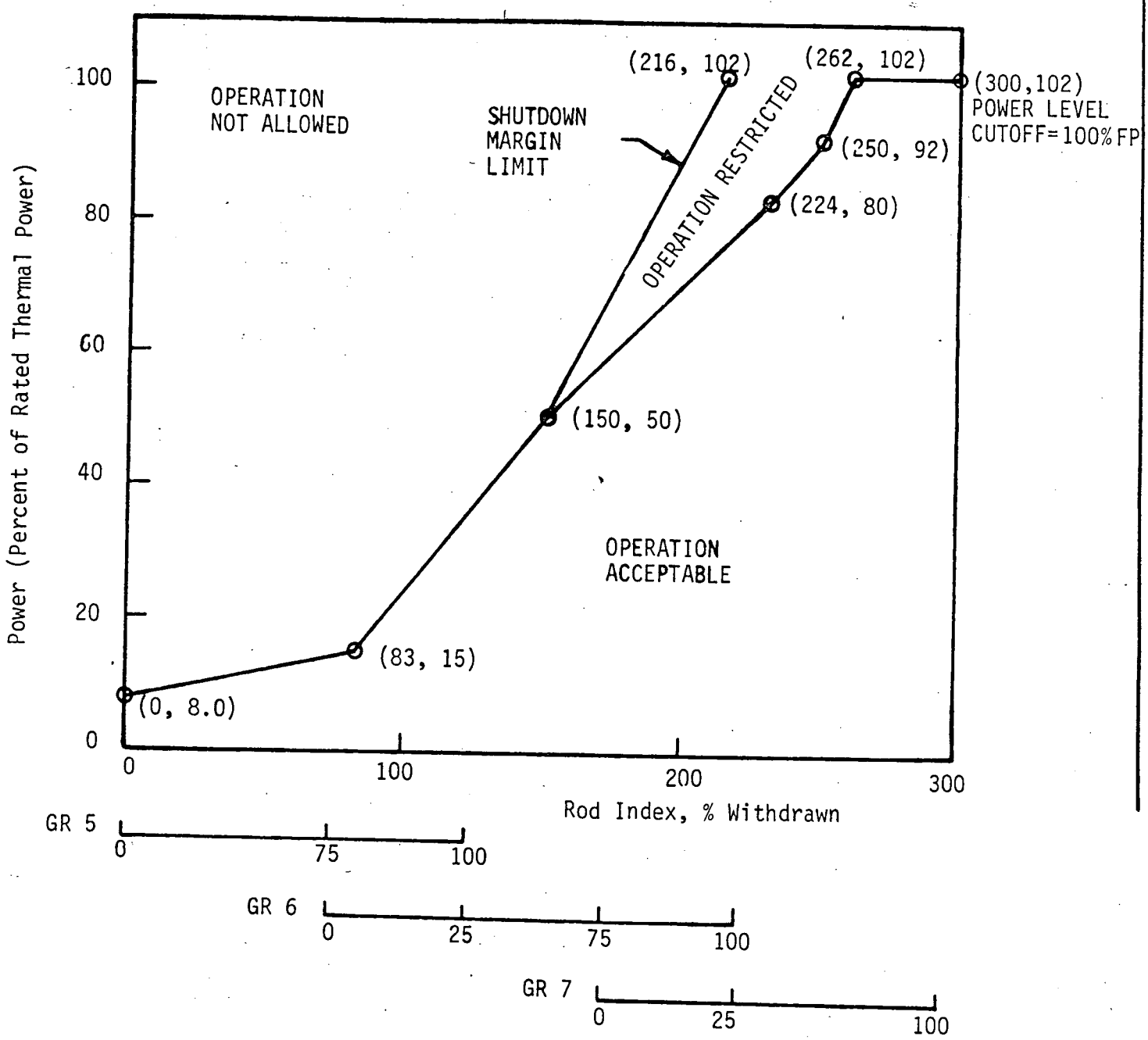
Figure 3.5.2-1
(1 of 3)



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 30 + 10/0 TO 250 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-1
(2 of 3)

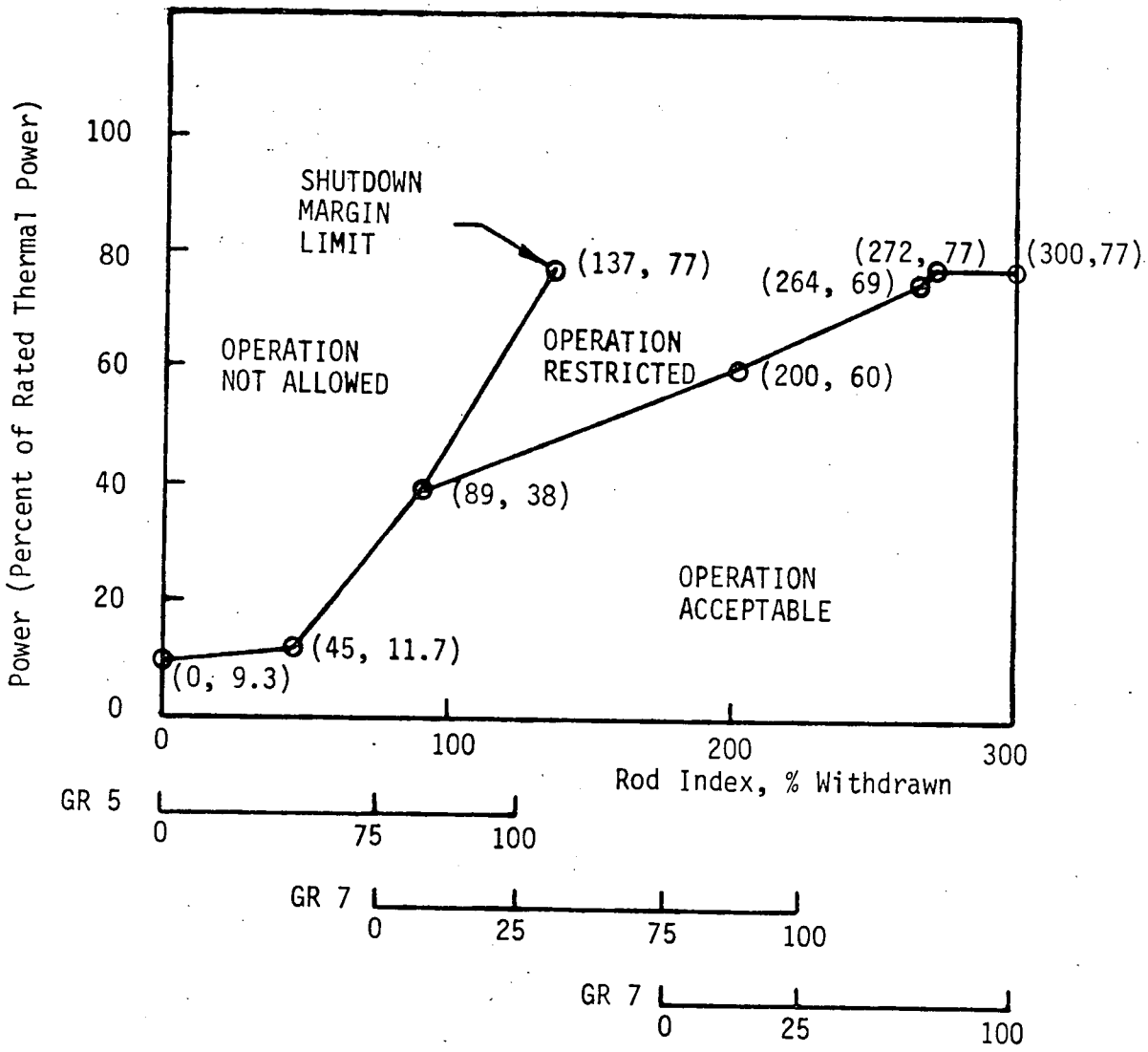


ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 250 ± 10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

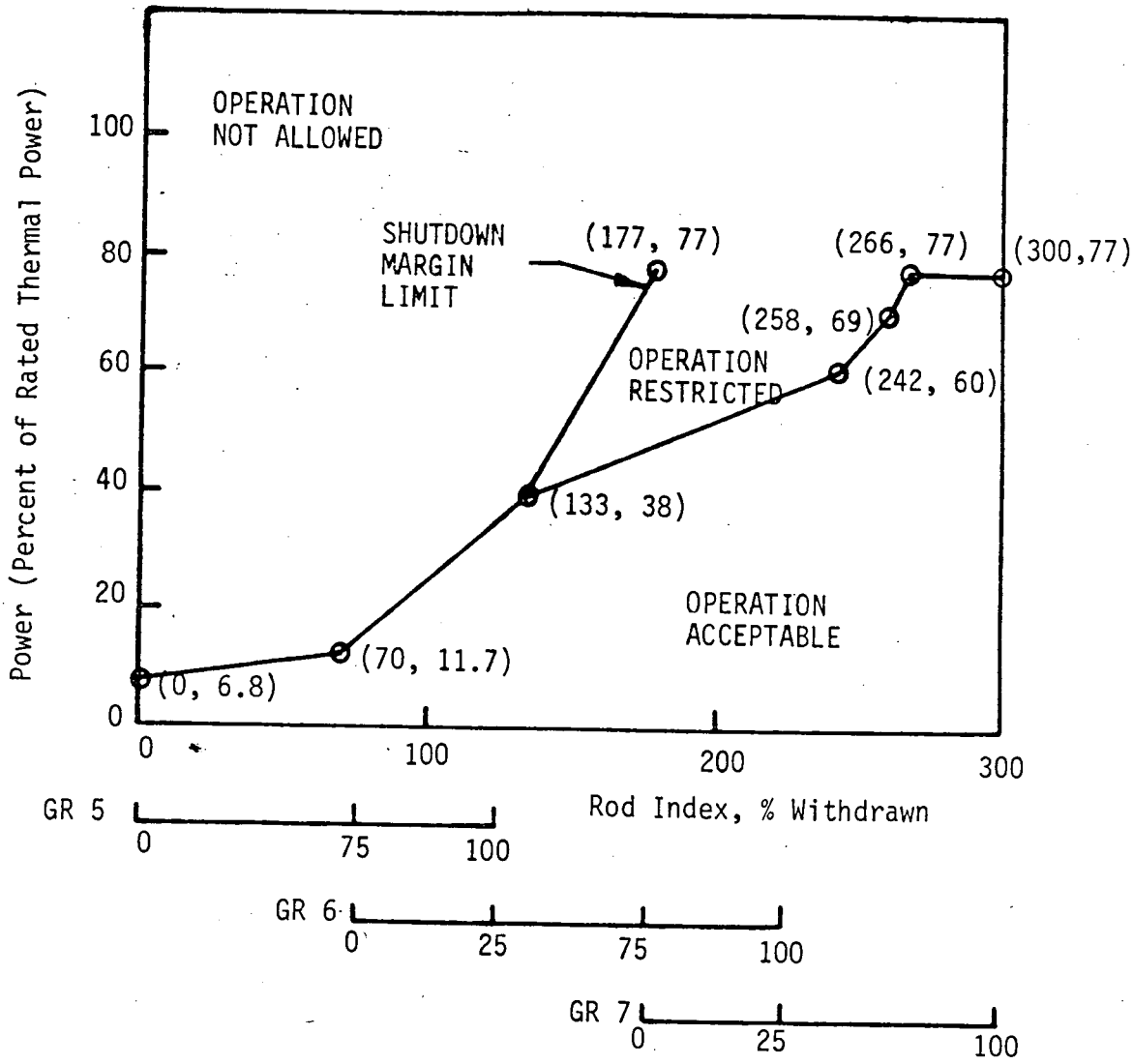
Figure 3.5.2-1

(3 of 3)

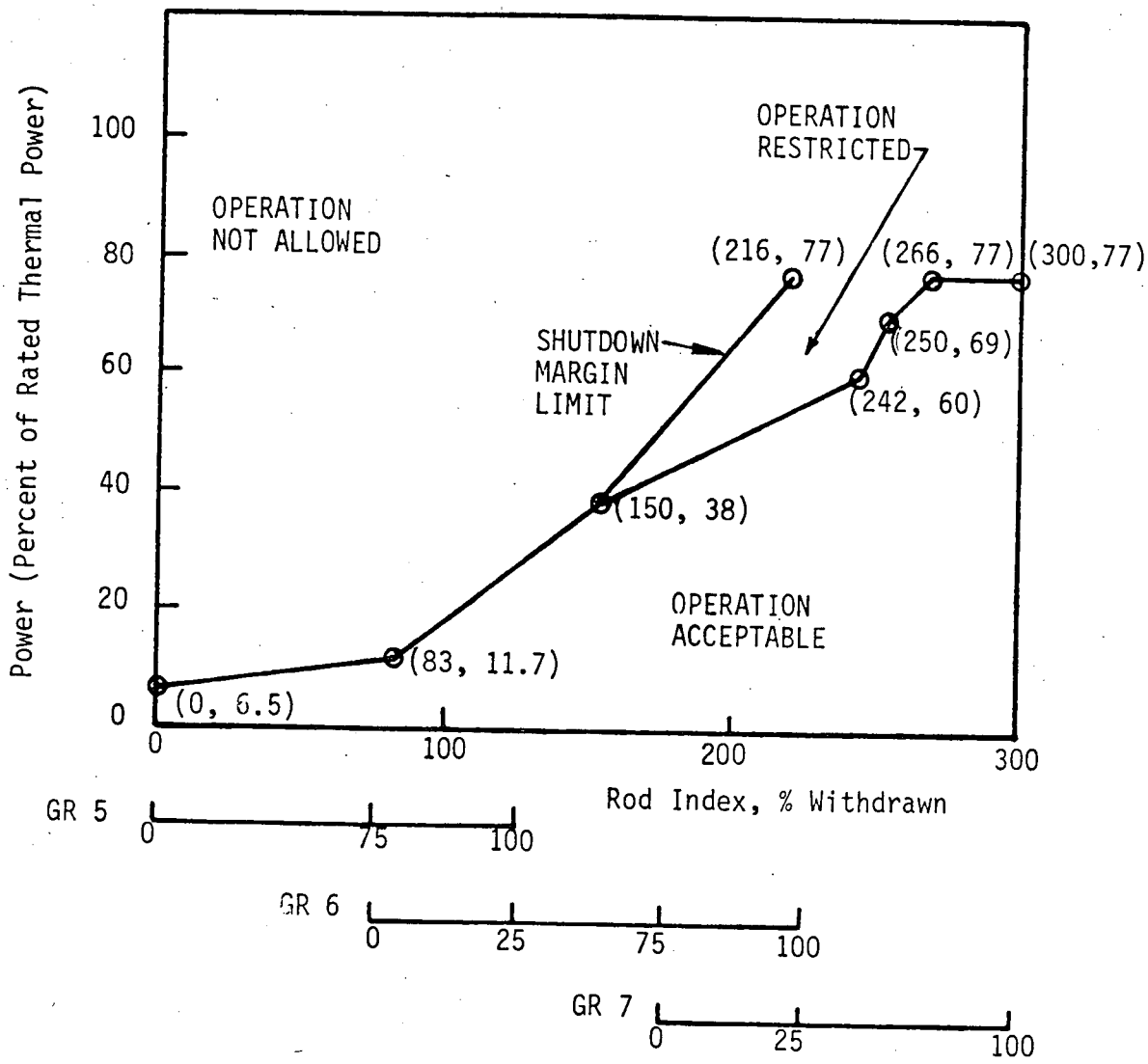


ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 TO 30 + 10/-0 EFPD
UNIT 1

OCONEE NUCLEAR STATION
Figure 3.5.2-4
(1 of 3)



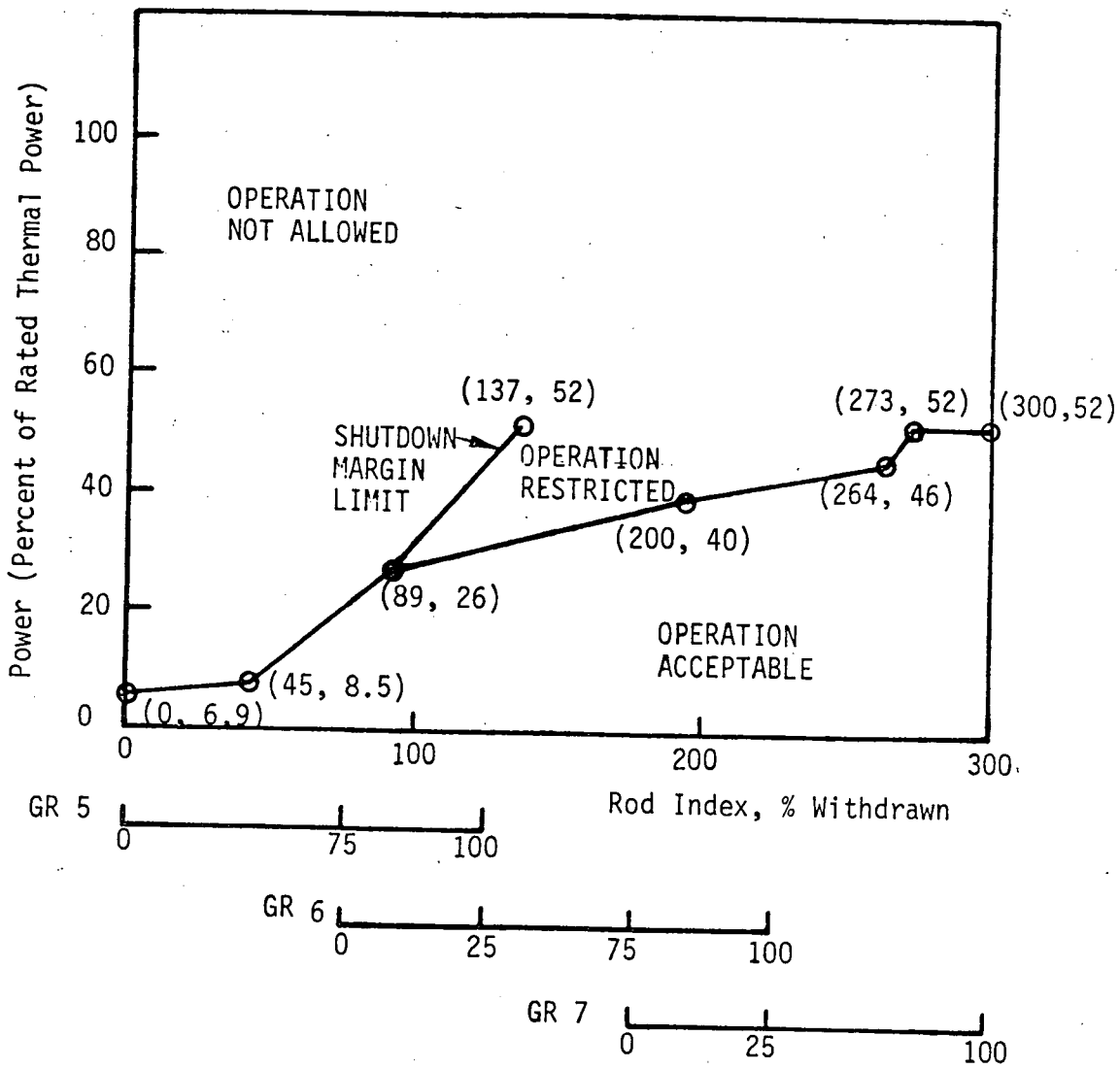
ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 30 + 10/-0 TO 250 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-4
(2 of 3)



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
AFTER 250 ± 10 EFPD
UNIT 1

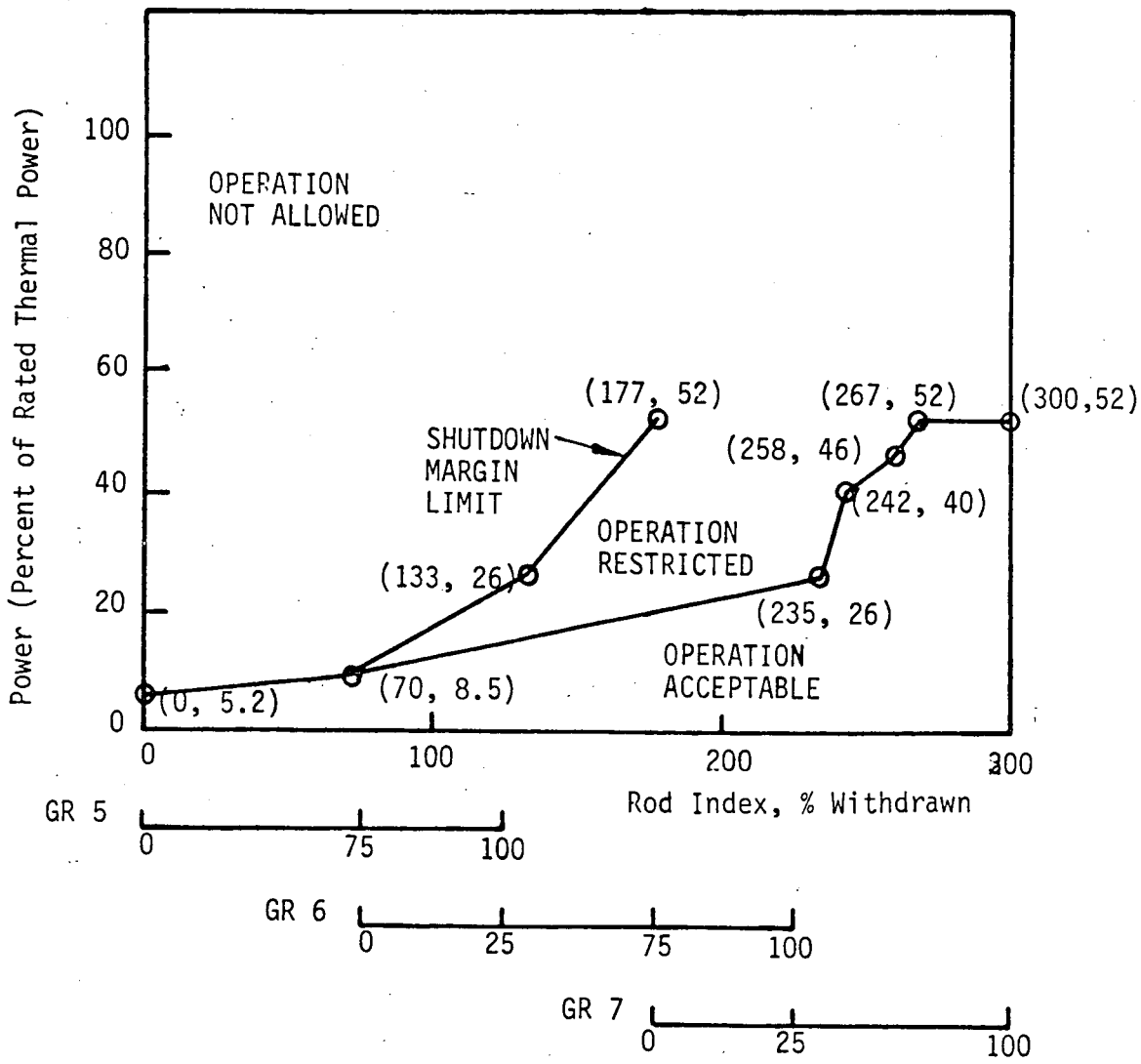
OCONEE NUCLEAR STATION

Figure 3.5.2-4
(3 of 3)



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 TO 30 + 10/-0 EFPD
UNIT 1

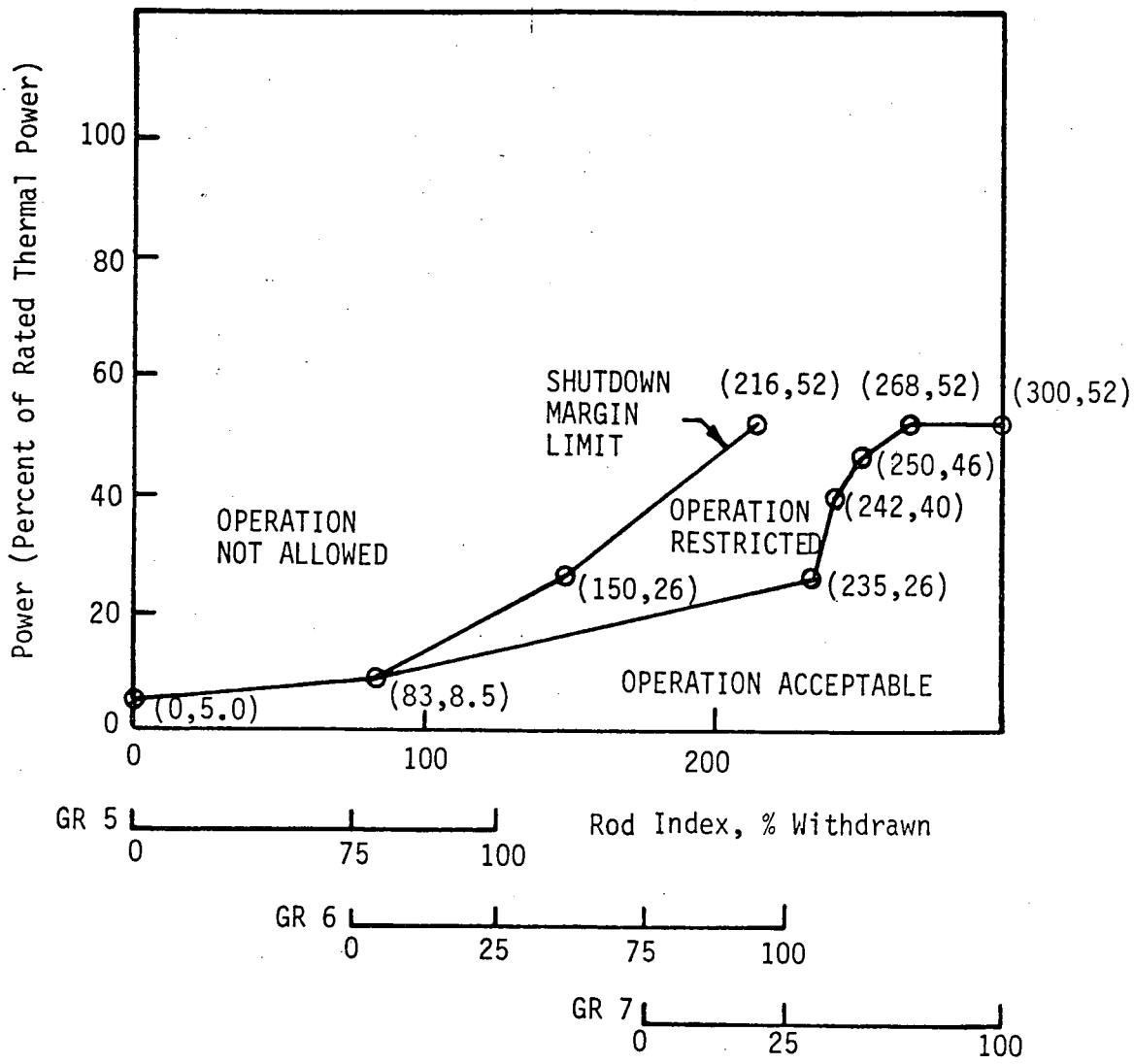
OCONEE NUCLEAR STATION
Figure 3.5.2-7
(1 of 3)



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 30 + 10/-0 TO 250 ± 10 EFPD
UNIT 1

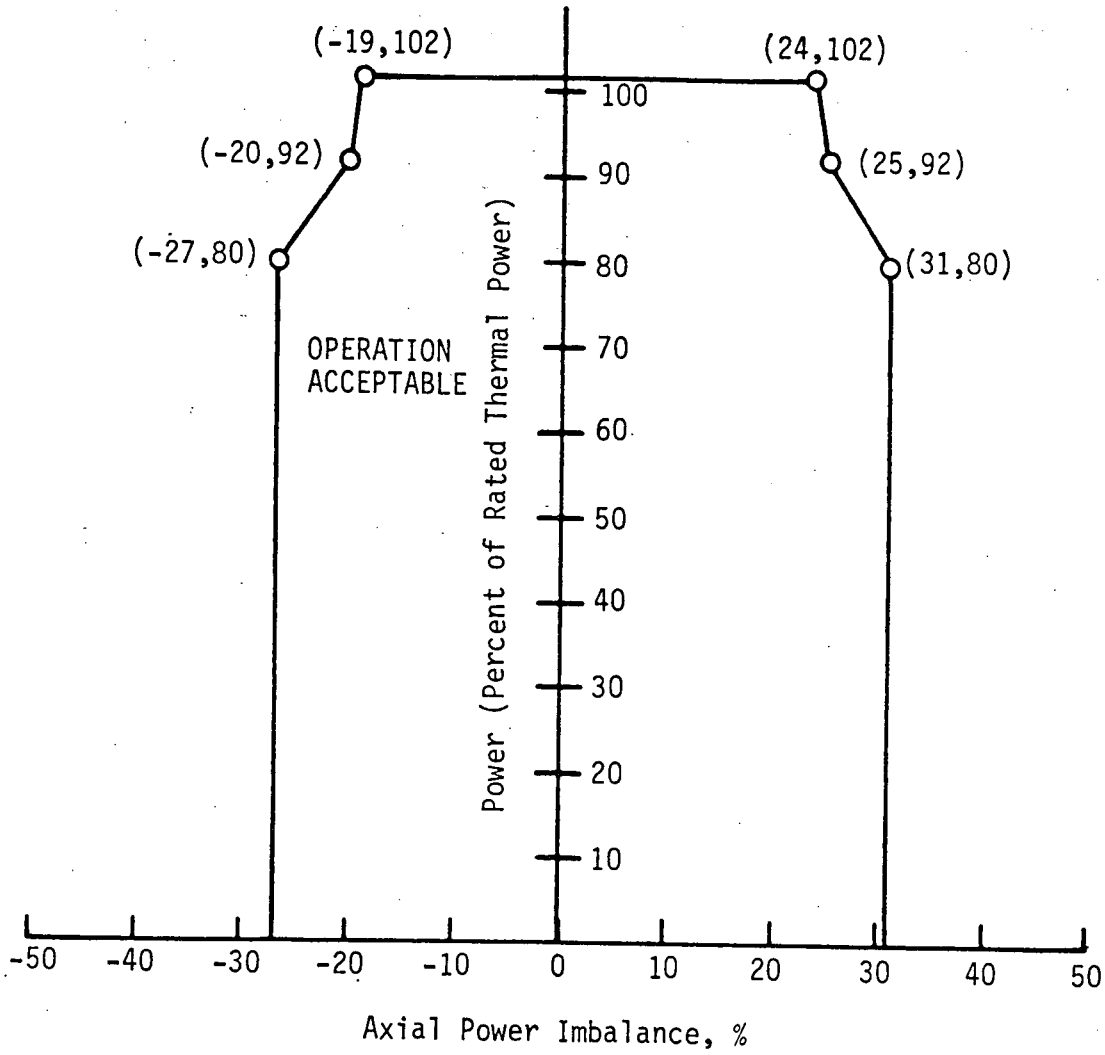
OCONEE NUCLEAR STATION

Figure 3.5.2-7
(2 of 3)



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
AFTER 250 ± 10 EFPD
UNIT 1

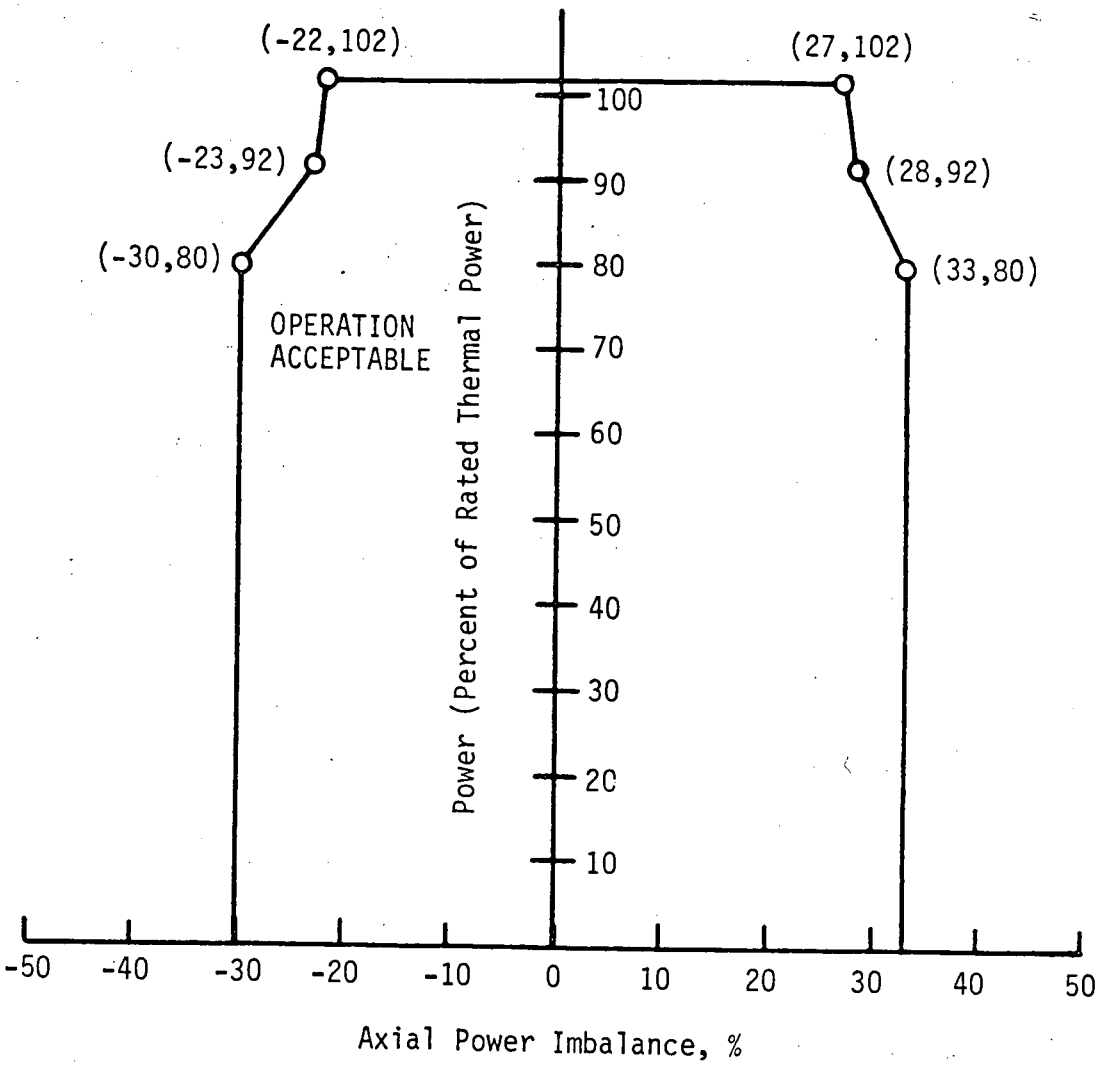
OCONEE NUCLEAR STATION
Figure 3.5.2-7
(3 of 3)



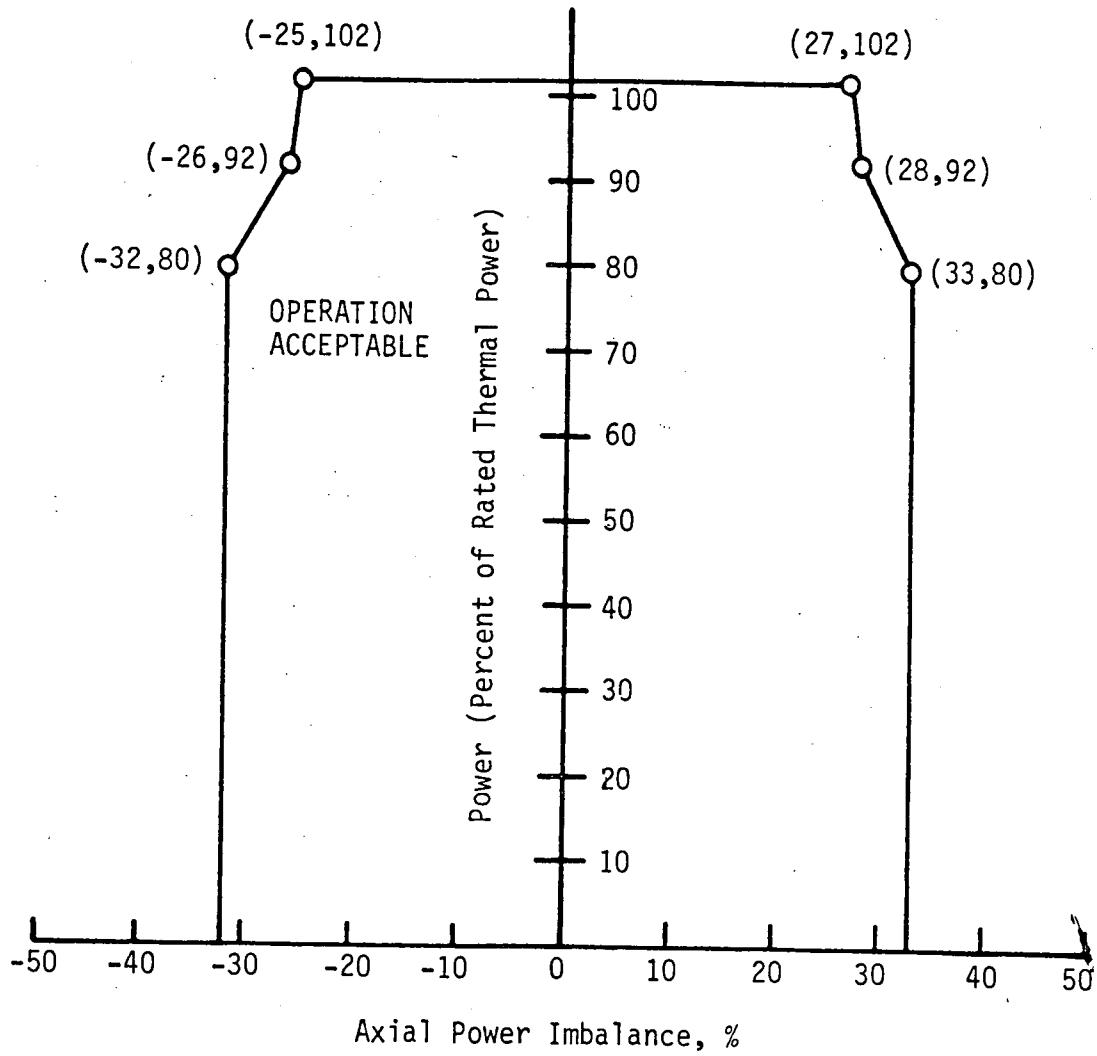
OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 TO 30 + 10/-0 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-10
(1 of 3)



OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 30 + 10/-0 TO 250 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-10
(2 of 3)



OPERATIONAL POWER
IMBALANCE ENVELOPE
AFTER 250 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION

Figure 3.5.2-10
(3 of 3)

Figure 3.5.2-13
(Deleted)

[Note that no rod position limits exist for Unit 1 axial power shaping rods.]

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Attachment 2

No Significant Hazards Consideration Evaluation

No Significant Hazards Consideration Evaluation
for Oconee Unit 1, Cycle 9 Reload

Duke Power has determined that the present amendment request poses no significant hazards as defined by NRC regulations in 10CFR50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

The commission has provided guidelines pertaining to the application of the three standards by listing specific examples in 48 FR 14870. Example (iii) and Example (vi) of the types of amendments not likely to involve significant hazards considerations expressly apply, inasmuch as the proposed amendment involves a nuclear power reactor core reload with a fuel assembly design slightly different from those previously found acceptable by the NRC.

Example (vi) of the types of amendments not likely to involve significant hazards considerations is applicable to the utilization of the mark BZ fuel assemblies for this reload. Example (vi) of amendments <u>not</u> likely to involve significant hazards considerations reads as follows:

> "A change which either may result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the standard review plan."

The results and conclusions of the Design report for the Mark BZ Fuel Assembly (BAW-1781 P) support the view that the introduction of the Zircaloy grid constitutes this type of change. For the transition cycle in which a substantial portion of the core is made up of the Mark B fuel assembly design, the thermal margin is slightly reduced. However, a small reduction in allowable power peaking limits compensates for the reduced thermal margin in such a manner that positive safety margins are assured relative to the acceptance criteria specified in the Standard Review Plan.

The BWC CHF correlation was used to predict the DNB behavior of the Mark BZ fuel assemblies. The Babcock and Wilcox (B&W) Topical Report BAW-10143P, Part 2 entitled "Correlation of 15 x 15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation", August 1981, was submitted by B&W in a letter dated September 22, 1981. By letter dated June 1, 1984, NRC found that this report was acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation.

Within the June 1, 1984 letter, the staff concluded that before the BWC correlation could be used with a subchannel analysis code other than LYNX-2, it must

be appropriately qualified by data comparisons. The subchannel code used for this reload was performed with the CHATA and TEMP core thermal-hydraulic analysis codes. By August 2, 1984, B&W provided the data comparison which demonstrated the conservatism of the CHATA/TEMP codes relative to the LYNX-2 code.

The Mark BZ verification program demonstrated the acceptability of the Zirc grid design from a mechanical standpoint. The structural integrity and corrosion properties were investigated and found to be within acceptable limits. The impact of the Zircaloy grids on the accident analysis was evaluated and shown to produce acceptable results.

The change to Zircaloy grids will not violate any of the three standards given in 10CFR50.92. The analyses and results presented in BAW-1781P show that this is true on a generic basis. Duke submitted BAW-1781P for NRC Review and approval by an October 7, 1983 letter.

The Mark BZ fuel assembly has been analyzed to ensure conformance with the standard review plan acceptance criteria. The design has also been reviewed in light of the standards given in 10CFR50.92. The analysis and review establish that there is no significant hazards consideration involved in the change to the Mark BZ design.

Example (iii) of amendments <u>not</u> likely to involve a significant hazards consideration concerns a core reload, assuming that:

(1) no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved,

(2) no significant changes are made to the acceptance criteria for the technical specifications,

(3) the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and

(4) the NRC has previously found such methods acceptable.

The Mark BZ is an improved version of the Mark B fuel assembly, in that the six Inconel Intermediate Spacer grids are replaced with Zircaloy grids. The Mark BZ spacer grids have the same functional design as the Mark B grids, but are slightly different dimensionally to accommodate Zircaloy's material properties. The other fuel assembly components such as the fuel rods, end grid, end fittings, and guide tubes are the same for both designs. The interfaces with the control rods and fuel handling equipment are unchanged, ensuring compatibility with the present reactor site operational procedures. Therefore, on the whole, the Mark BZ fuel assembly is a minor design change from the Mark B -- with the Zircaloy spacer grids being the only major difference. Each of the 113 fuel assemblies to be reinserted into the core are similar to fuel assemblies previously used and found acceptable by the NRC.

The Cycle 9 control rods differ from those of Cycle 8 in that gray (less-absorbing) axial power shaping rods (APSRs) are to be utilized instead of the previously used black (highly-absorbing) APSRs. The gray APSRs, which have a greater absorber length than the APSRs used in previous reloads and utilize an Inconel absorber instead of the Ag-In-Cd alloy, will not adversely affect Cycle 9 operation, according to analyses described in Attachment 3.

The present reload involves no significant changes to the acceptance criteria for the Technical Specifications. Revisions of the Technical Specifications required for Cycle 9 operation were made in accordance with methods and procedures found acceptable in connection with previous reloads. The final acceptance criteria of the ECCS limits will not be exceeded, and thermal design criteria will be satisfied.

The Oconee Unit 1, Cycle 9 Reload Report (Attachment 3) justifies the operation of the ninth cycle 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. The Reload Report employs analytical techniques and design bases established in reports submitted for previous reloads which were accepted by USNRC and its predecessor. These techniques are described in the Reload Report references.

With supporting reference to previously performed analyses, the following evaluation measures aspects of the Unit 1, Cycle 9 reload against the §50.92(c) requirements to demonstrate that all three standards are satisfied.

First Standard

> (Amendment would not) involve a significant increase in the probability or consequences of an accident previously evaluated.

Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to changes in Cycle 8 parameters to determine the effect of the Cycle 9 reload and to ensure that thermal performance during hypothetical transients is not degraded. The transient evaluation of Cycle 9 is considered to be bound by previously accepted analyses. Section 7 of the Reload Report addresses "Accident and Transient Analysis" for this core reload. This analysis ensures that the proposed reload will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

> (Amendment would not) create the possibility of a new or different kind of accident from any accident previously evaluated.

The analyses performed in support of this reload are in accordance with the USNRC document "Guidance for Proposed License Amendments Relating to Refueling", June 1975. The conclusion of the overall analysis is that the proposed reload does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

(Amendment would not) involve a significant reduction in a margin of safety.

The issue of margin of safety for a reload modification involves the following areas:

1. Fuel System Design considerations,

2. Nuclear Design considerations, and

3. Thermal-Hydraulic Design considerations.

Sections 4, 5, and 6 of the Oconee Unit 1, Cycle 9 Reload Report address the above areas, respectively. The value limits and margins discussed in these areas are well within the allowable limits and requirements, and reflect no significant reductions to any margins of safety. One can conclude from the examination of these sections, and the Cycle 9 core thermal and kinetic properties (with respect to previous cycle values), that this core reload will not significantly reduce the ability of Oconee Unit 1 to operate safely during Cycle 9.

With specific regard to the Mark BZ fuel assemblies, the results of the BAW-1781P report show, generically, that the change to Zircaloy grids will not violate any of the three standards set forth above. It is concluded, on the basis of the overall analysis and review of BAW-1781P and the Reload Report, that the inclusion of the Mark BZ assemblies poses no significant hazards.

The above evaluation, with its accompanying references, shows that the three §50.92(c) standards are satisfied. In summary, Duke has determined and submits that the proposed reload described herein does not represent any significant hazards.