Attachment 1
Duke Power Company
Oconee Nuclear Station
Proposed Technical Specification Revision
03C8

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Proposed Technical Specification Revision
03C8 (Cont'd)

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LIST OF TABLES

LIST OF FIGURES (CONT'D)

LIST OF FIGURES (CONT'D)

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2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limits for Unit 3 are 20.5 kw/ft for fuel rod burn-up less than or equal to 10,000 MWD/MTU and 21.5 kw/ft - after 10,000 MWD/MTU.

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

Due to the reduced power production capability of the fuel with increasing irradiation, the DNBR penalty for rod bow has been determined to be insignificant and unnecessary. (4,5)

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 = 80.7 percent power plus the maximum calibration and instrument error (Reference 4). The maximum thermal power for other coolant pump conditions are produced in a similar manner.

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

References

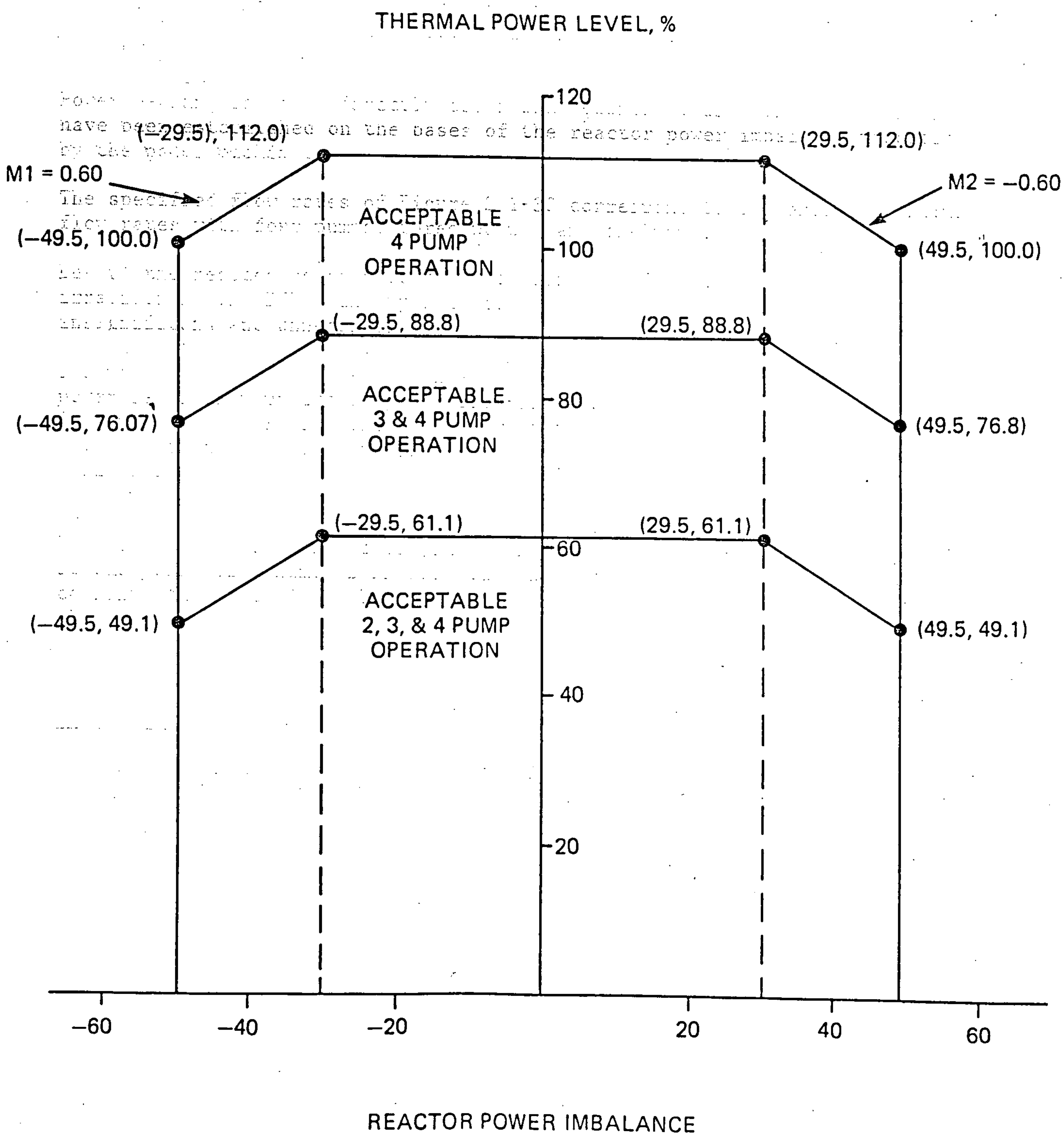
(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March 1970.

(2) Oconee 3, Cycle 3 - Reload Report - BAW-1453, August 1977.

(3) Amendment 1 - Oconee 3, Cycle 4 - Reload Report - BAW-1486, June 12, 1978.

(4) Oconee 3, Cycle 8 - Reload Report - DPC-RD-2003, February 1984.

(5) Fuel Rod Bowing in Babcock & Wilcox Fuel Designs, BAW-10147P-A, Rev. 1, Babcock & Wilcox, May 1983.



CORE PROTECTION
SAFETY LIMITS
UNIT 3
OCONEE NUCLEAR STATION
Figure 2.1-2C

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

Overpower Trip Based on Flow and Imbalance

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

The power level trip 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:

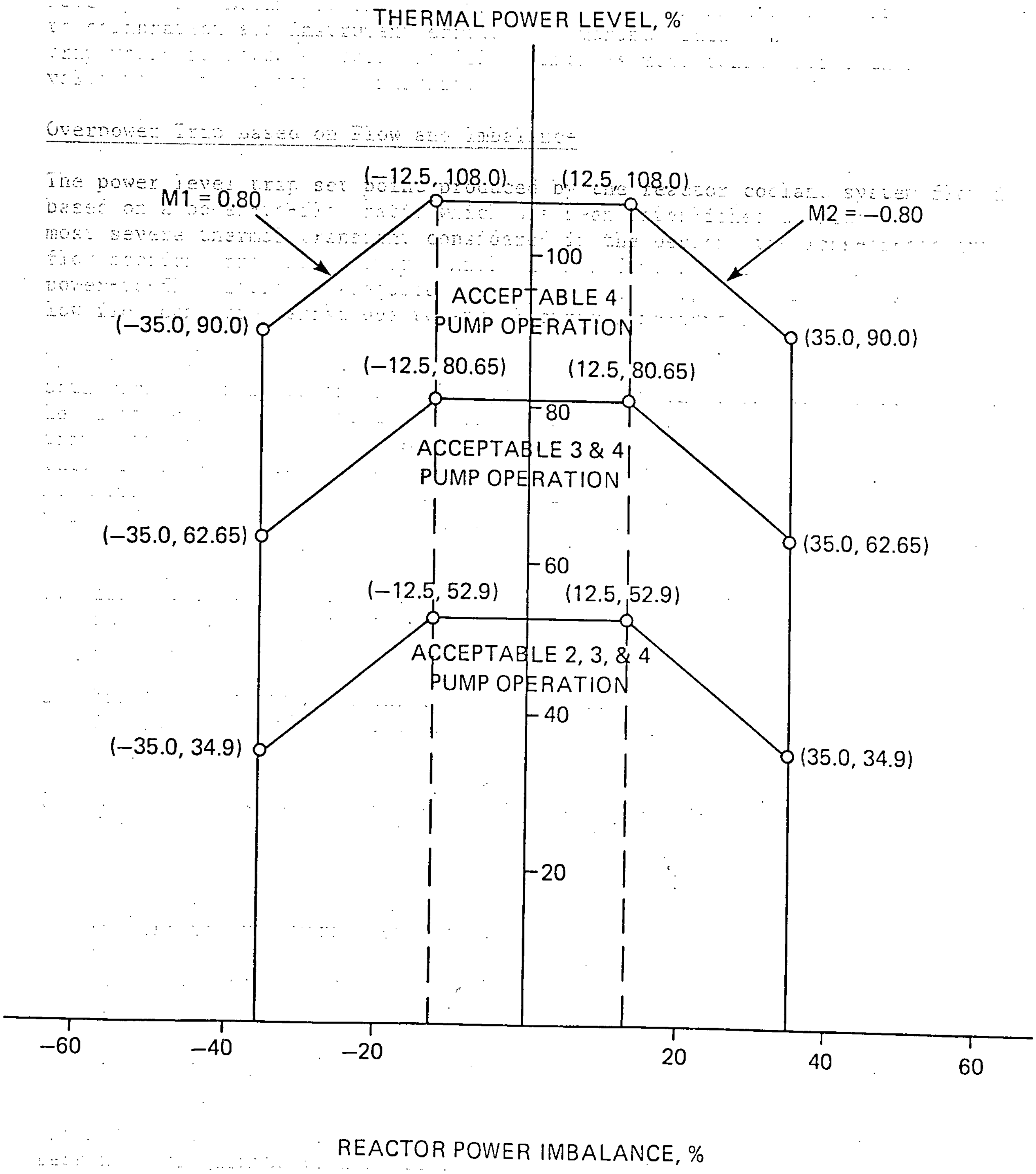
1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor flow rate is 100%, or flow rate is 93.46% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.92% and reactor flow rate is 74.7% or flow rate is 70.09% 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 52.43% and reactor flow rate is 49.0% or flow rate is 45.79% and the power level is 49%.

   The flux-to-flow ratios account for calibration and instrument errors and the maximum variation from the RC flow signal in such a manner that the reactor protective system receives a conservative indication of RC flow. For unit 1, the maximum calibration and instrument errors are algebraically summed to determine the string errors in the safety calculations. Units 2 and 3 employ a Monte-Carlo simulation technique with final string errors 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



PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE
SETPOINTS - UNIT 3
OCONEE NUCLEAR STATION
Figure 2.3-2C

f. If the maximum positive quadrant power tilt exceeds the Maximum Limit of Table 3.5-1, the reactor shall be shut down within 4 hours. Subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the Nuclear Overpower Trip Setpoints allowable for the reactor coolant pump combination are restricted by a reduction of 2% of thermal power for each 1% tilt for the maximum tilt observed prior to shutdown.

g. Quadrant power tilt shall be monitored on a minimum frequency of once every 2 hours during power operation above 15% full power.

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 tests, 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 3.5.2-1 (Unit 1) for four
3.5.2-2 (Unit 2)
3.5.2-3 (Unit 3)

pump operation, on figures 3.5.2-4 (Unit 1) for three
3.5.2-5 (Unit 2)
3.5.2-6 (Unit 3)

pump operation, and on figures 3.5.2-7 (Unit 1) for two
3.5.2-8 (Unit 2)
3.5.2-9 (Unit 3)

pump operation. Also, excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on figures 3.5.2-13 (Unit 1)
3.5.2-14 (Unit 2)
3.5.2-15 (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 3.5.2-1 (Unit 1) unless one of the following
3.5.2-2 (Unit 2)
3.5.2-3 (Unit 3)
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 its 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 3.5.2-10 (Unit 1). If the imbalance
3.5.2-11 (Unit 2)
3.5.2-12 (Unit 3)

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.

3.5.2.9 The operational limit curves of Technical Specifications 3.5.2.5.c and 3.5.2.7 are valid for a nominal design cycle length, as defined in the Safety Evaluation Report for the appropriate unit and cycle. Operation beyond the nominal design cycle length is permitted provided that an evaluation is performed to verify that the operational limit curves are valid for extended operation. If the operational limit curves are not valid for the extended period of the operation, appropriate limits will be established and the Technical Specification curves will be modified as required.

Bases

Operation at power with an inoperable control rod is permitted within the limits provided. These limits assure that an acceptable power distribution is maintained and that the potential effects of rod misalignment on associated accident analyses are minimized. For a rod declared inoperable due to misalignment, the rod with the greatest misalignment shall be evaluated first. Additionally, the position of the rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments. When a control rod is declared inoperable, boration may be initiated to achieve the existence of 1% $\Delta k/k$ hot shutdown margin.

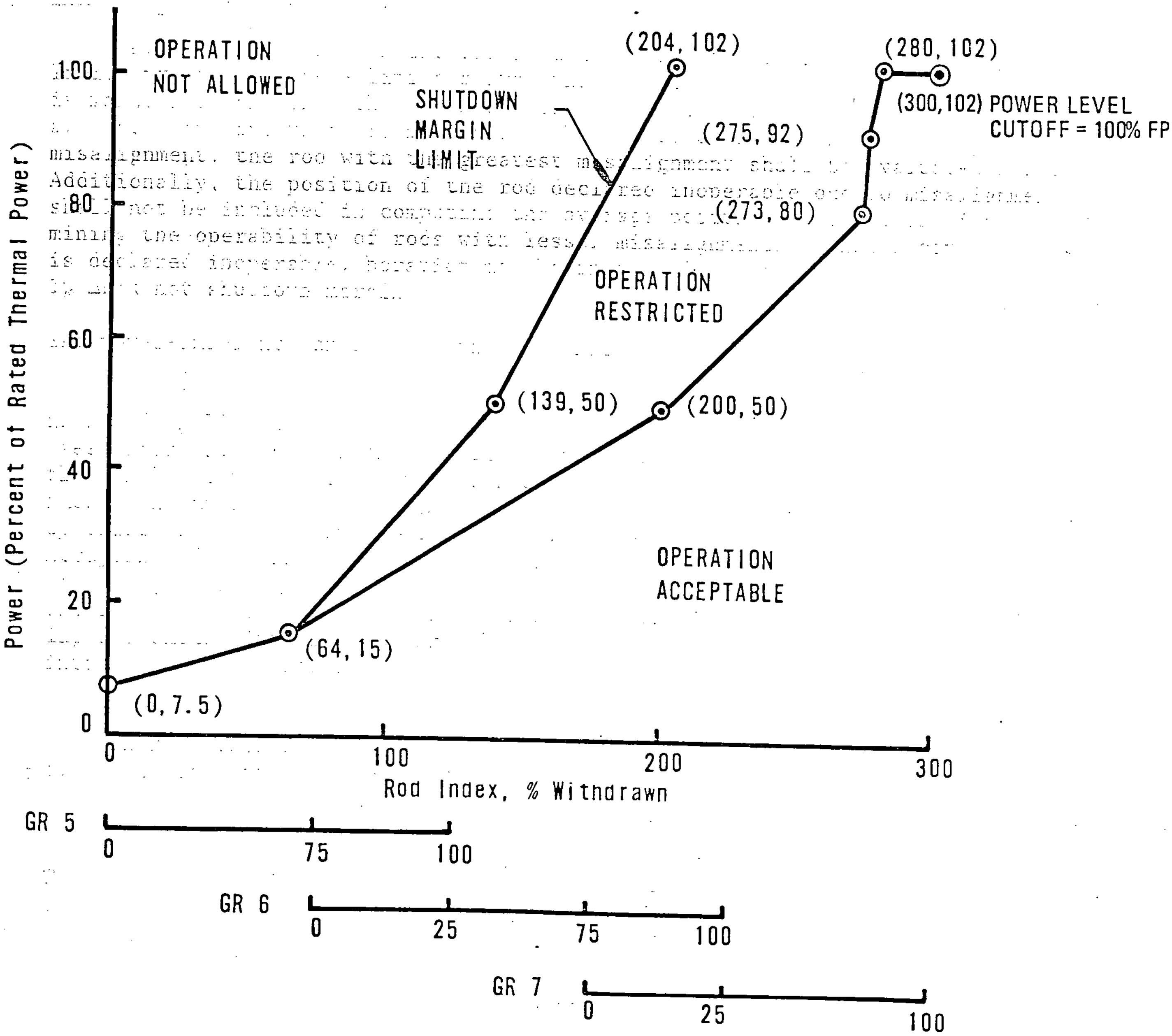
The power-imbalance envelope defined in Figures 3.5.2-10 (Unit 1)
3.5.2-11 (Unit 2)
3.5.2-12 (Unit 3)
is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-16) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures 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

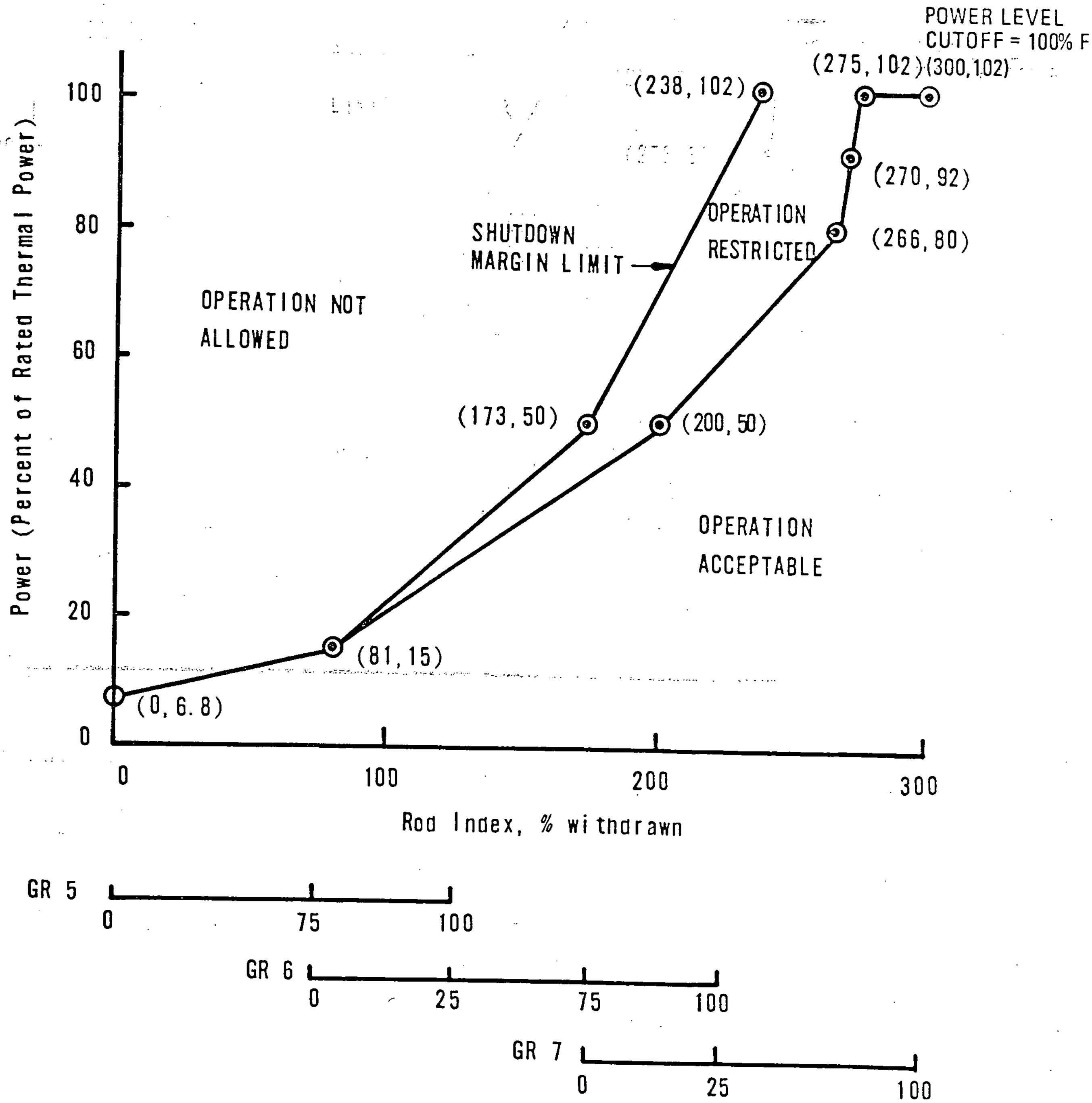
The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

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.



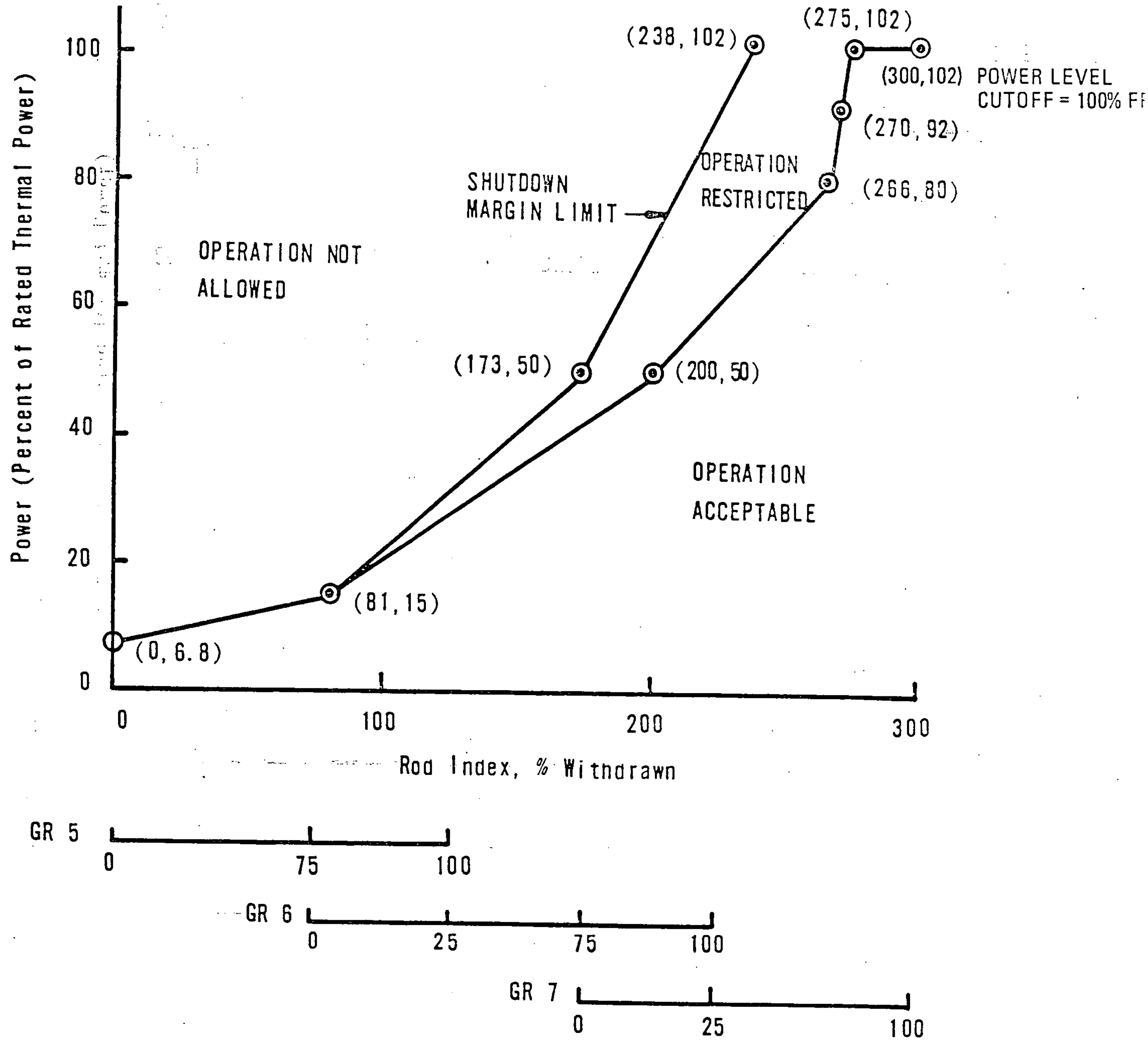
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 26 +10/-0 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-1
(1 of 3)



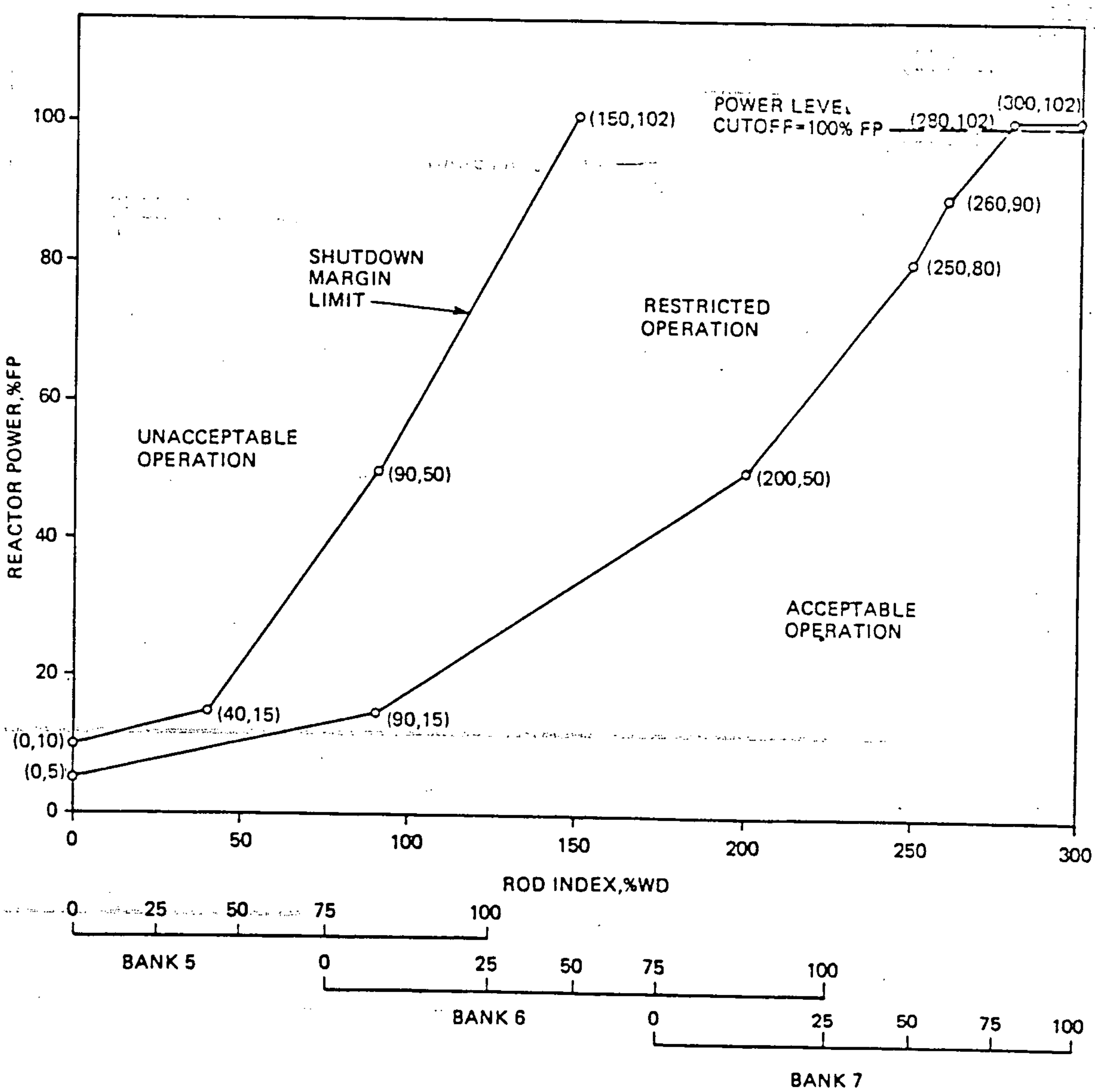
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-1
(2 of 3)



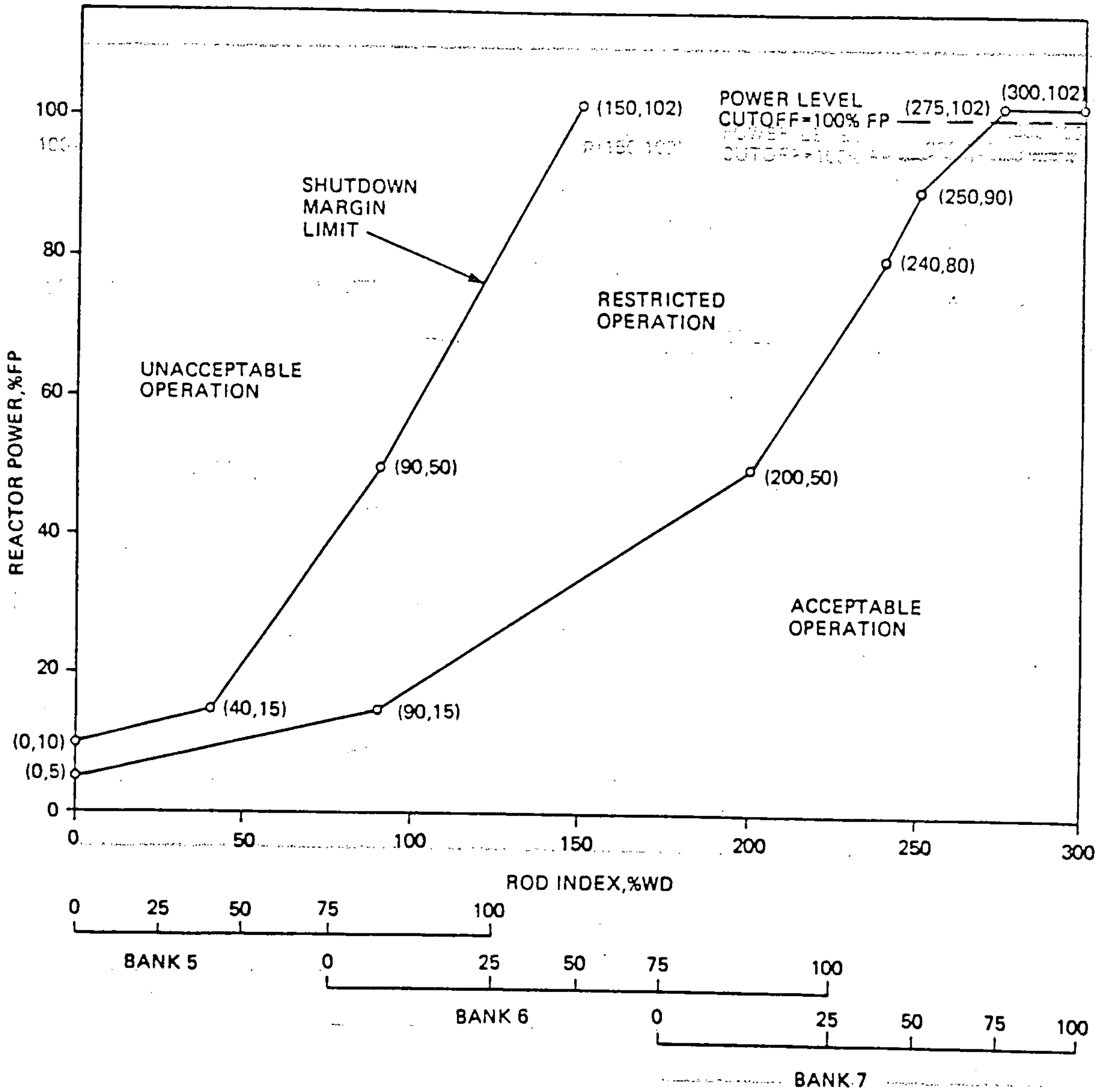
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
AFTER 200 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-1
(3 of 3)



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 TO 25 (+10, -0) EFPD
UNIT 2

OCONEE NUCLEAR STATION

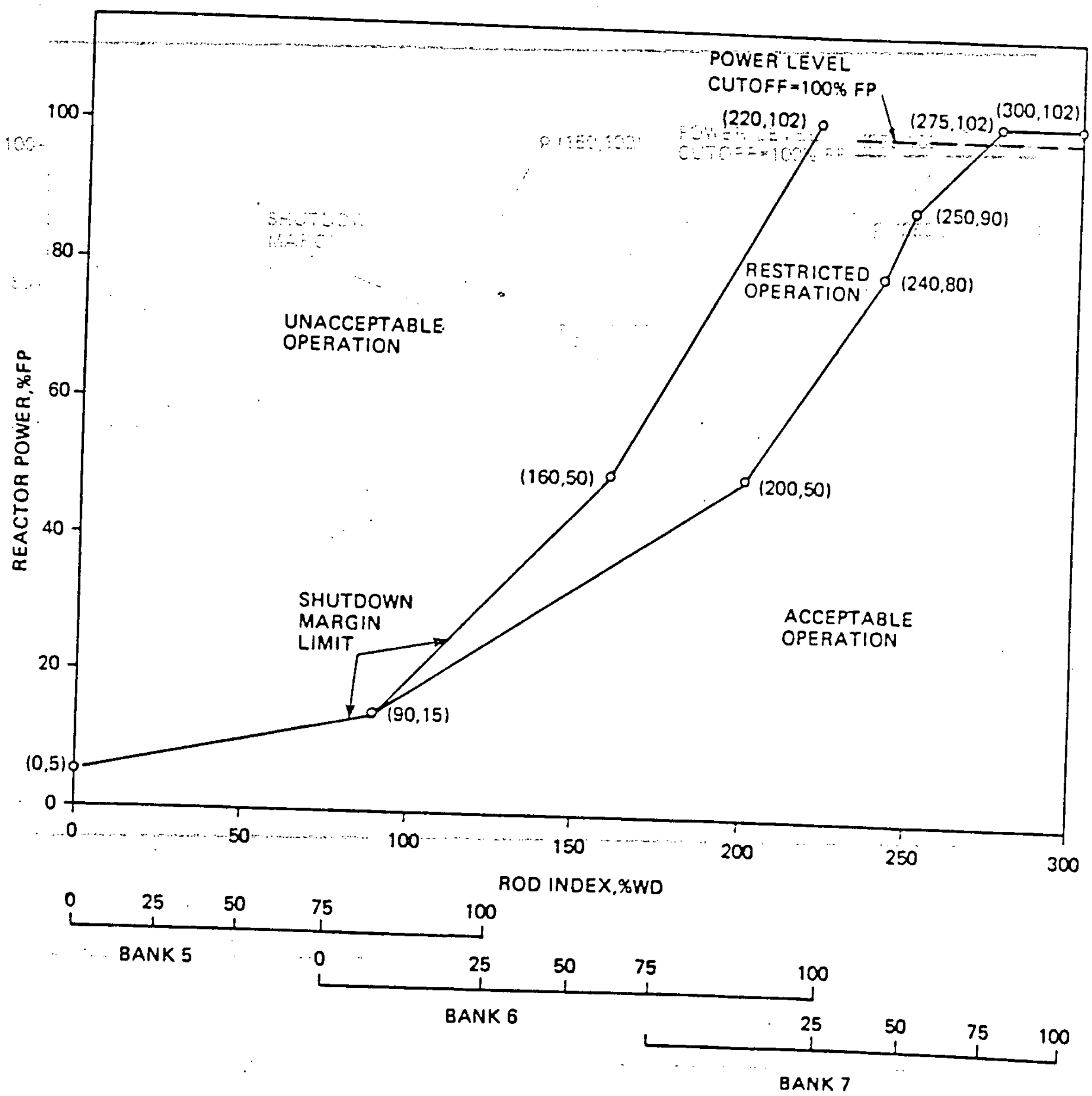
Figure 3.5.2-2
(1 of 3)



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 25 (+10, -0) TO 200 ± 10 EFPD
UNIT 2

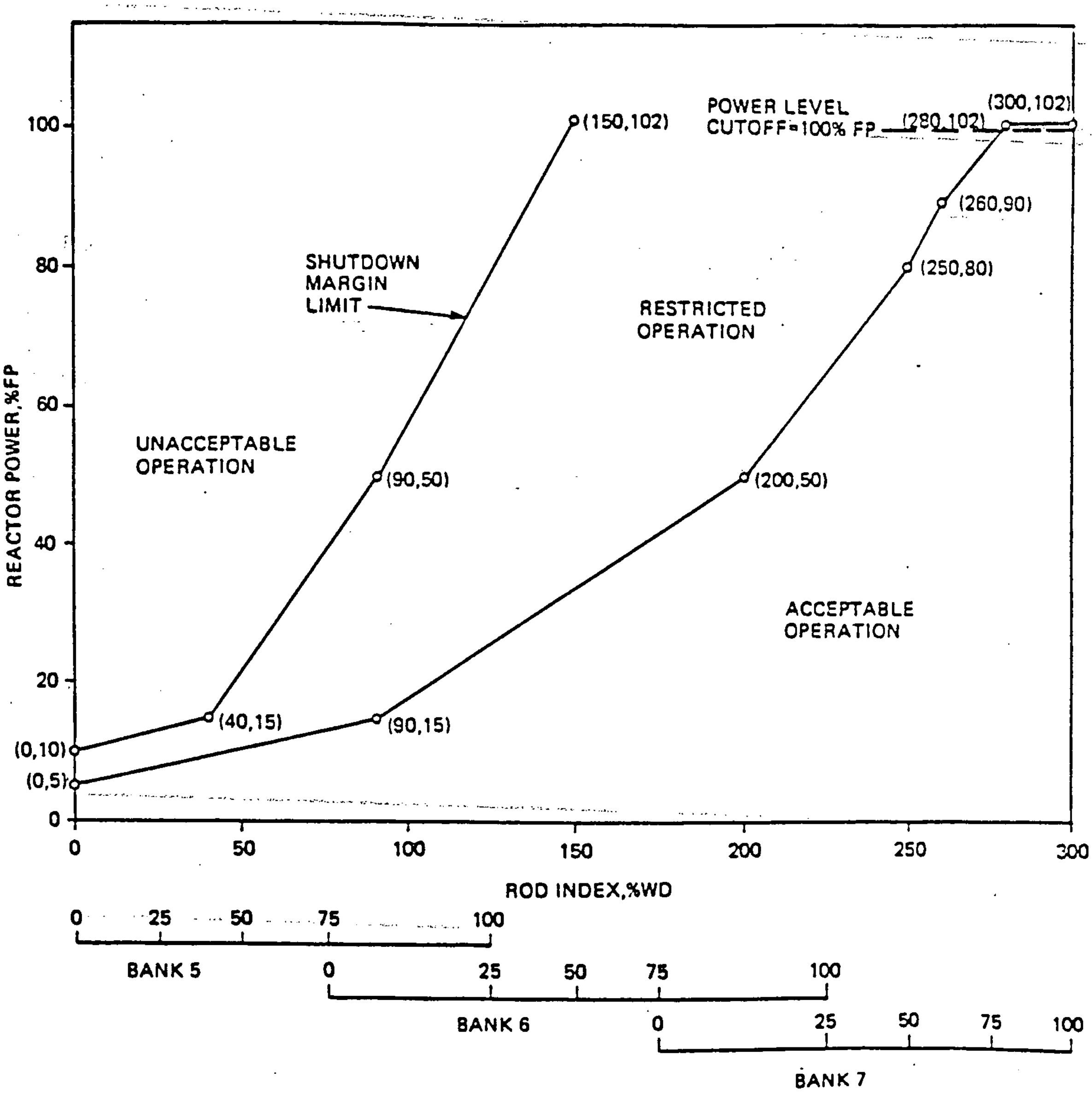
OCONEE NUCLEAR STATION

Figure 3.5.2-2
(2 of 3)

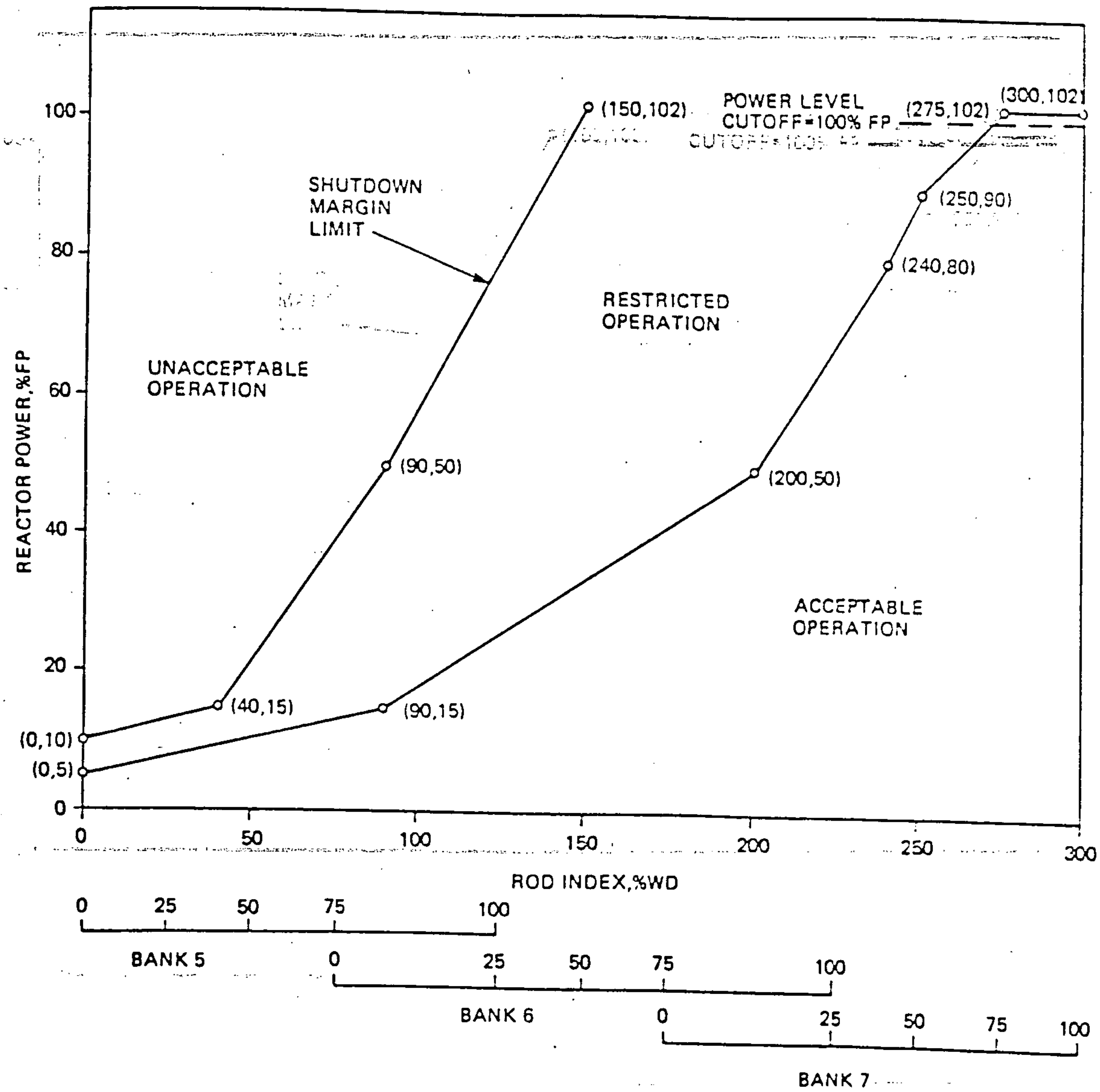


ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 200 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION

Figure 3.5.2-2
(3 of 3)



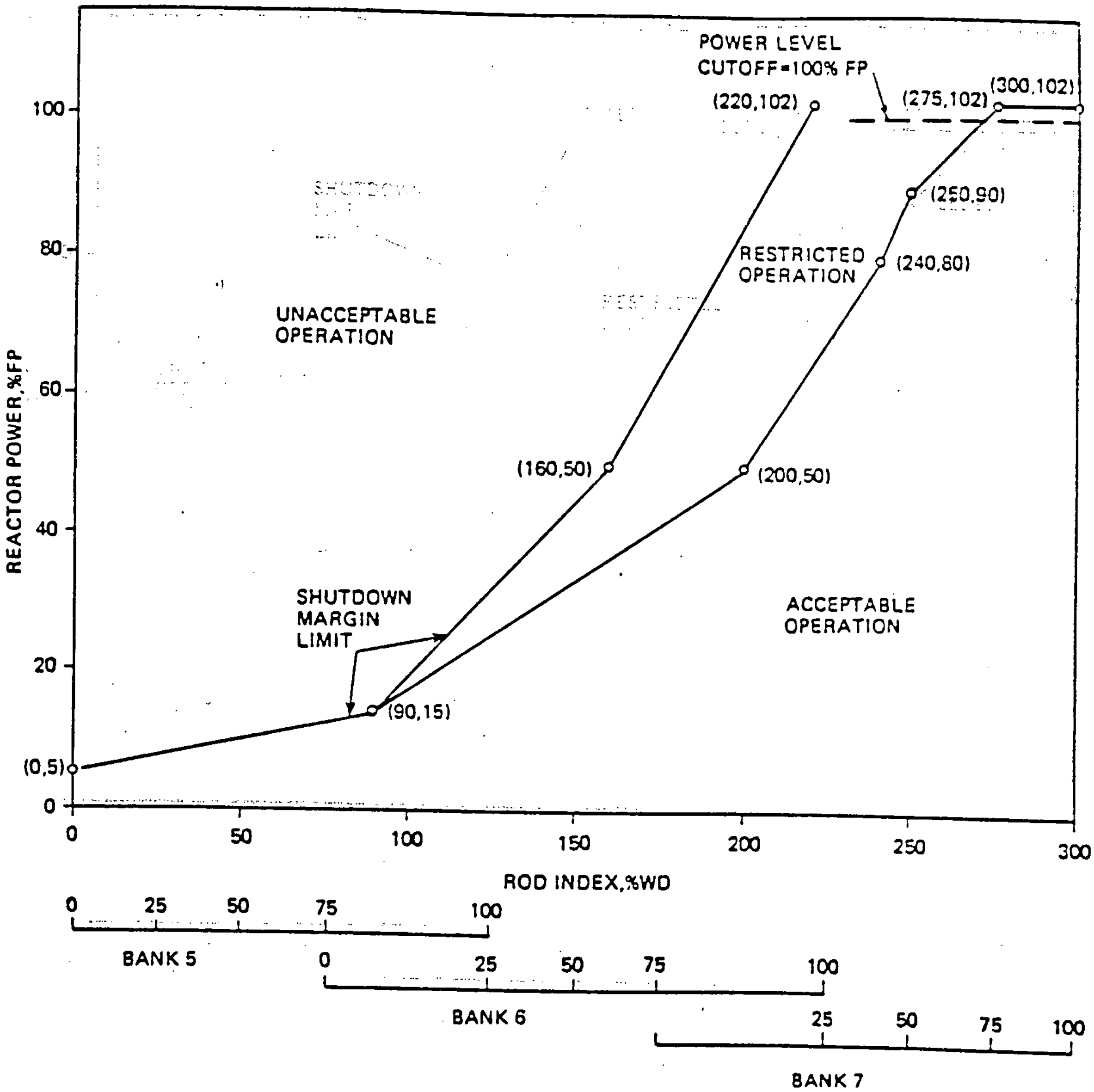
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 25 +10/-0 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3
(1 of 3)



ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 25 +10/-0 TO 200 +10/-10 EFPD
UNIT 3

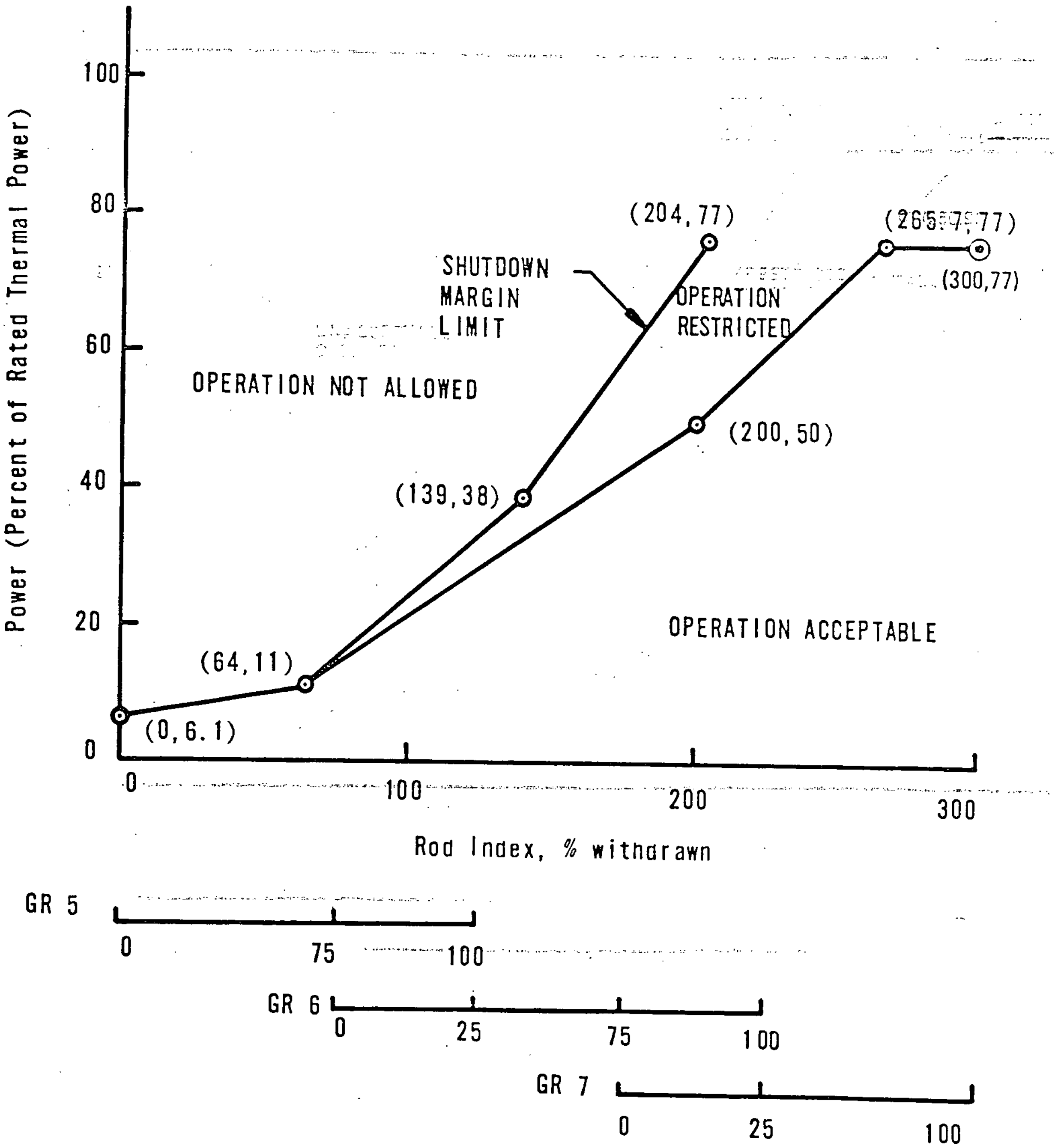
OCONEE NUCLEAR STATION

Figure 3.5.2-3
(2 of 3)

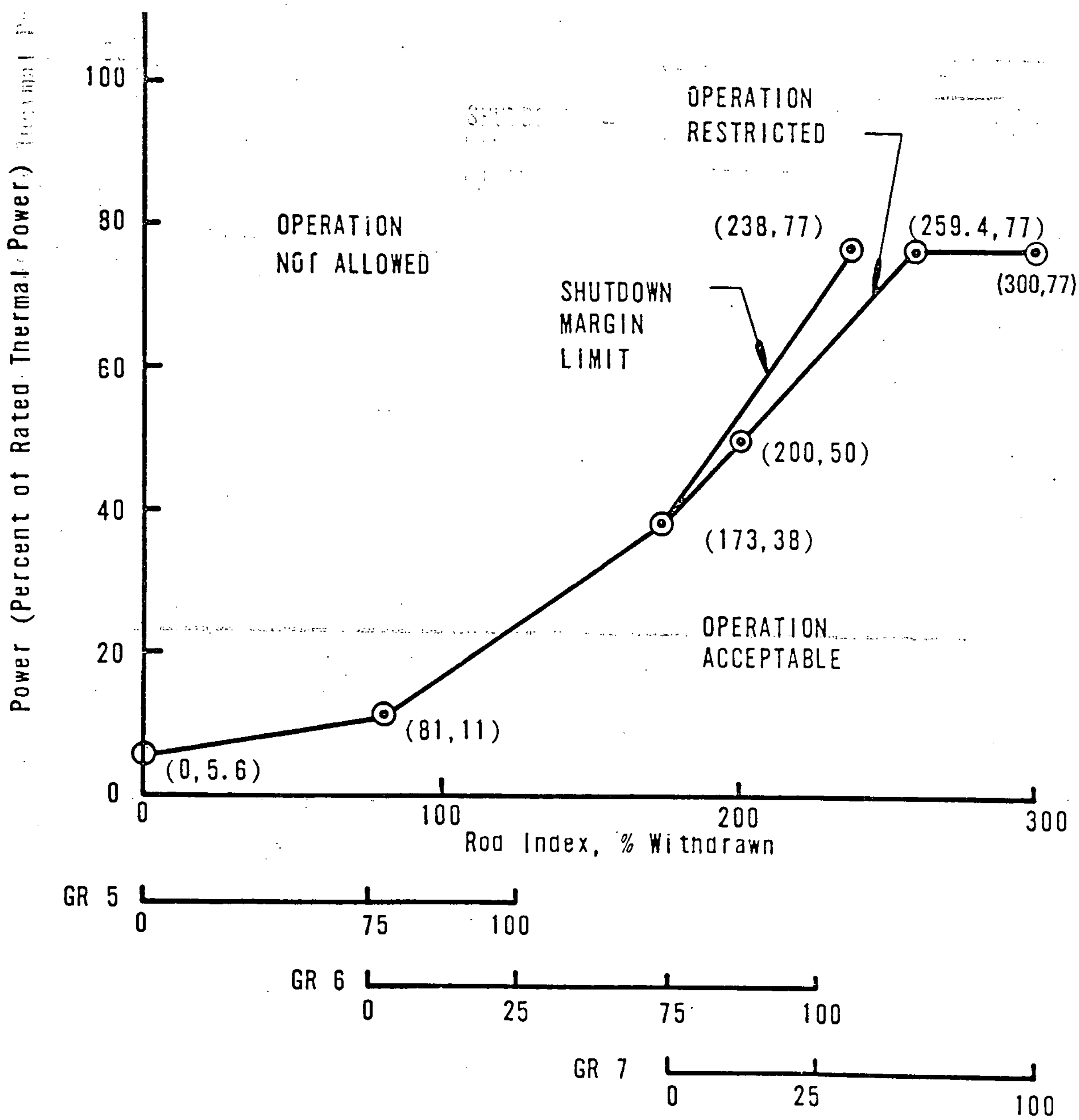


ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
AFTER 200 +10/-10 EFPD
UNIT 3

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



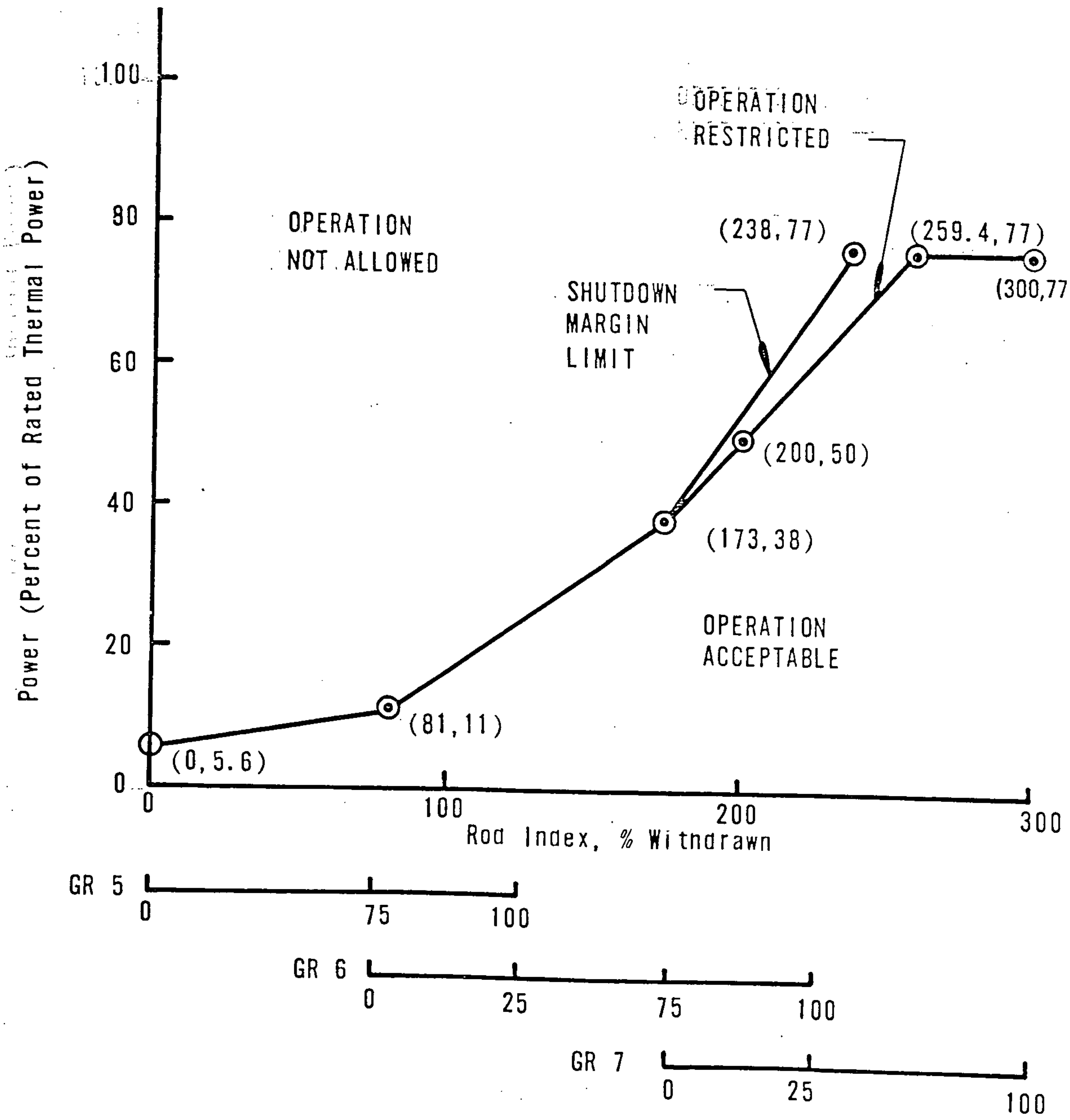
ROD POSITION LIMITS
FOR THREE-PUMP OPERATION
FROM 0 TO 26 +10/-0 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-4
(1 of 3)



ROD POSITION LIMITS
FOR THREE-PUMP OPERATION
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1

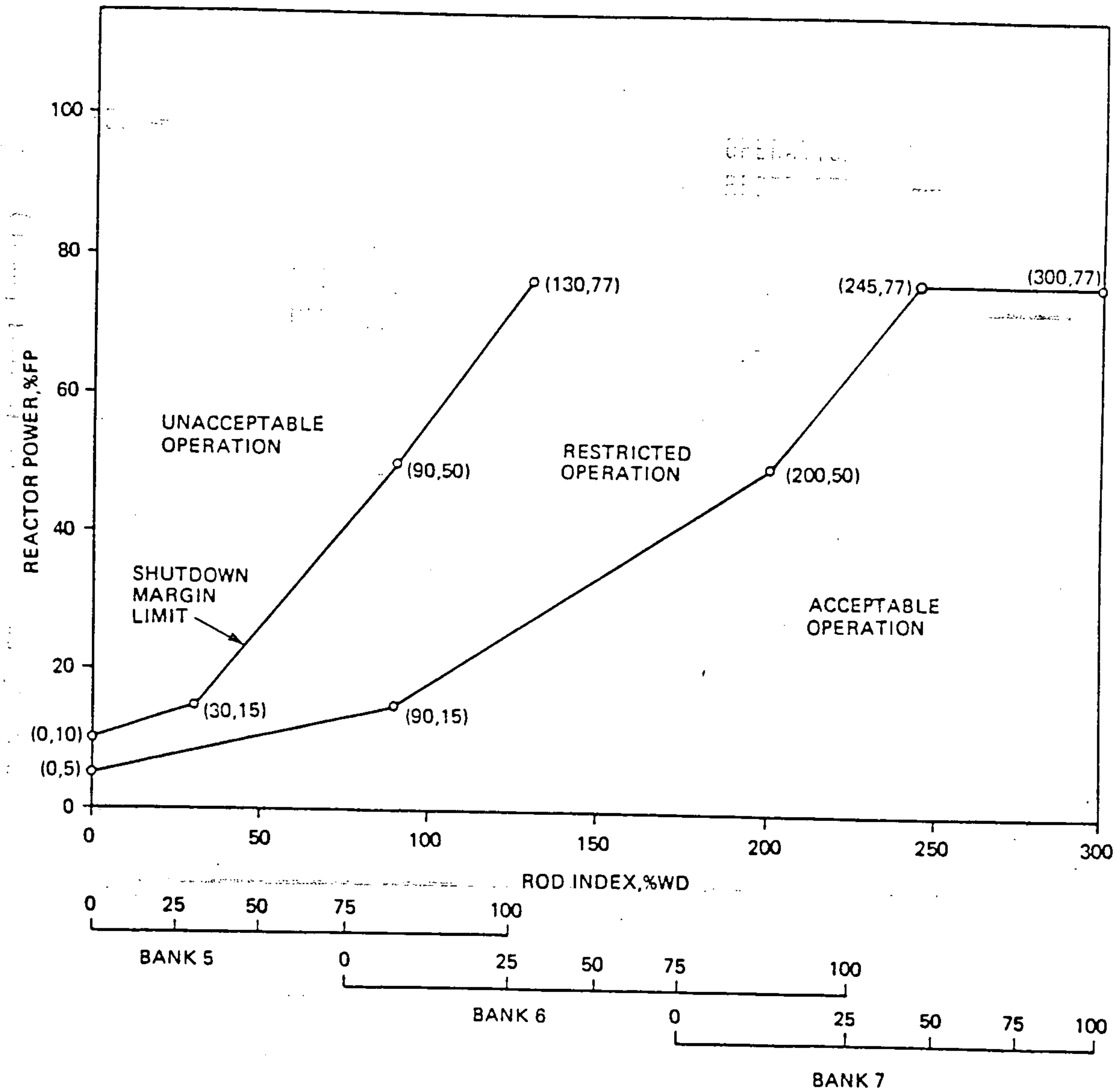
OCONEE NUCLEAR STATION

Figure 3.5.2-4
(2 of 3)



ROD POSITION LIMITS
FOR THREE-PUMP OPERATION
AFTER 200 ±10 EFPD
UNIT 1

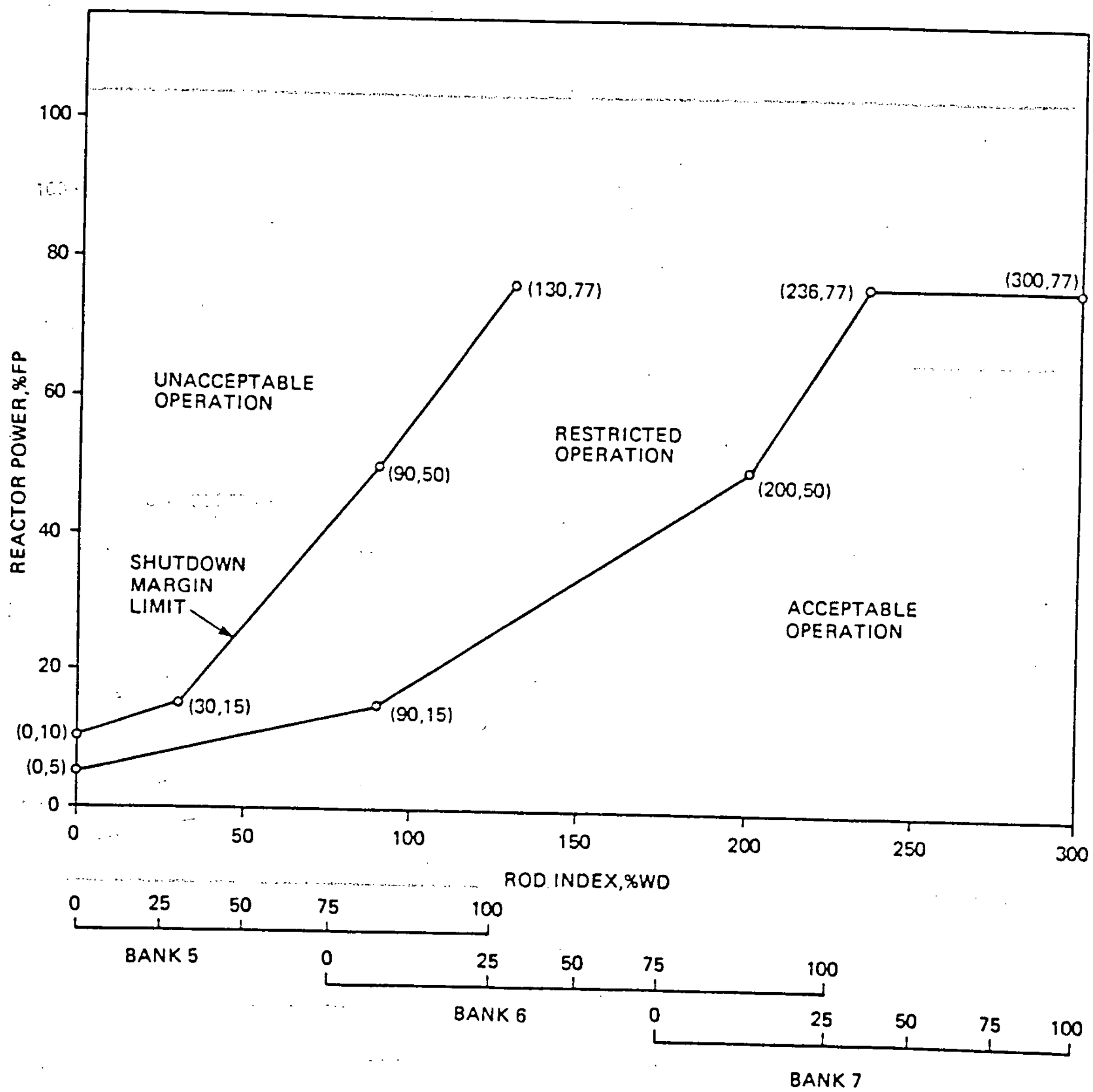
OCONEE NUCLEAR STATION
Figure 3.5.2-4
(3 of 3)



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 TO 25 (+10, -0) EFPD
UNIT 2

OCONEE NUCLEAR STATION

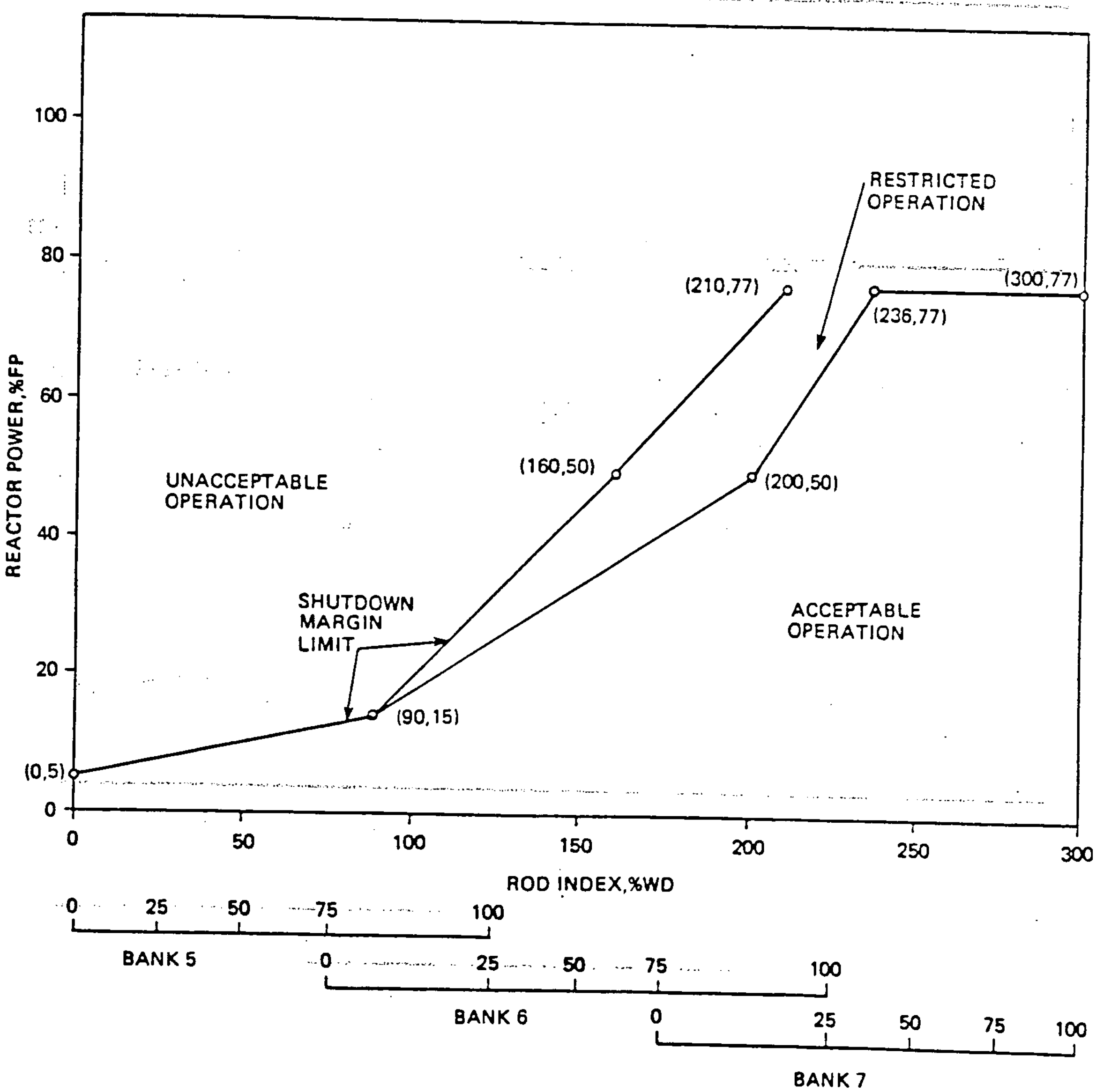
Figure 3.5.2-5
(1 of 3)



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 25 (+10, -0) TO 200 ±10 EFPD
UNIT 2

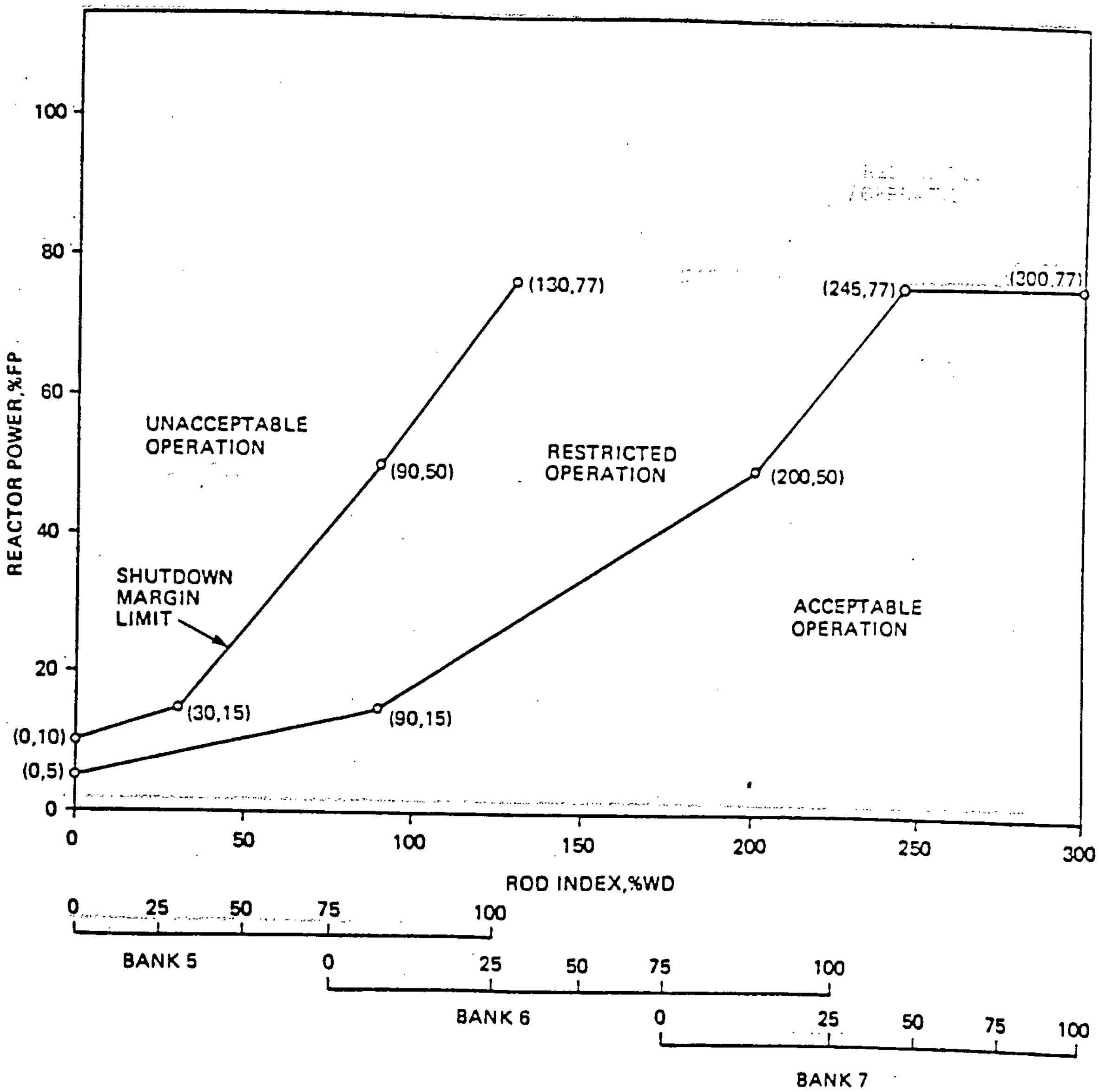
OCONEE NUCLEAR STATION

Figure 3.5.2-5
(2 of 3)



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
AFTER 200 ±10 EFPD
UNIT 2

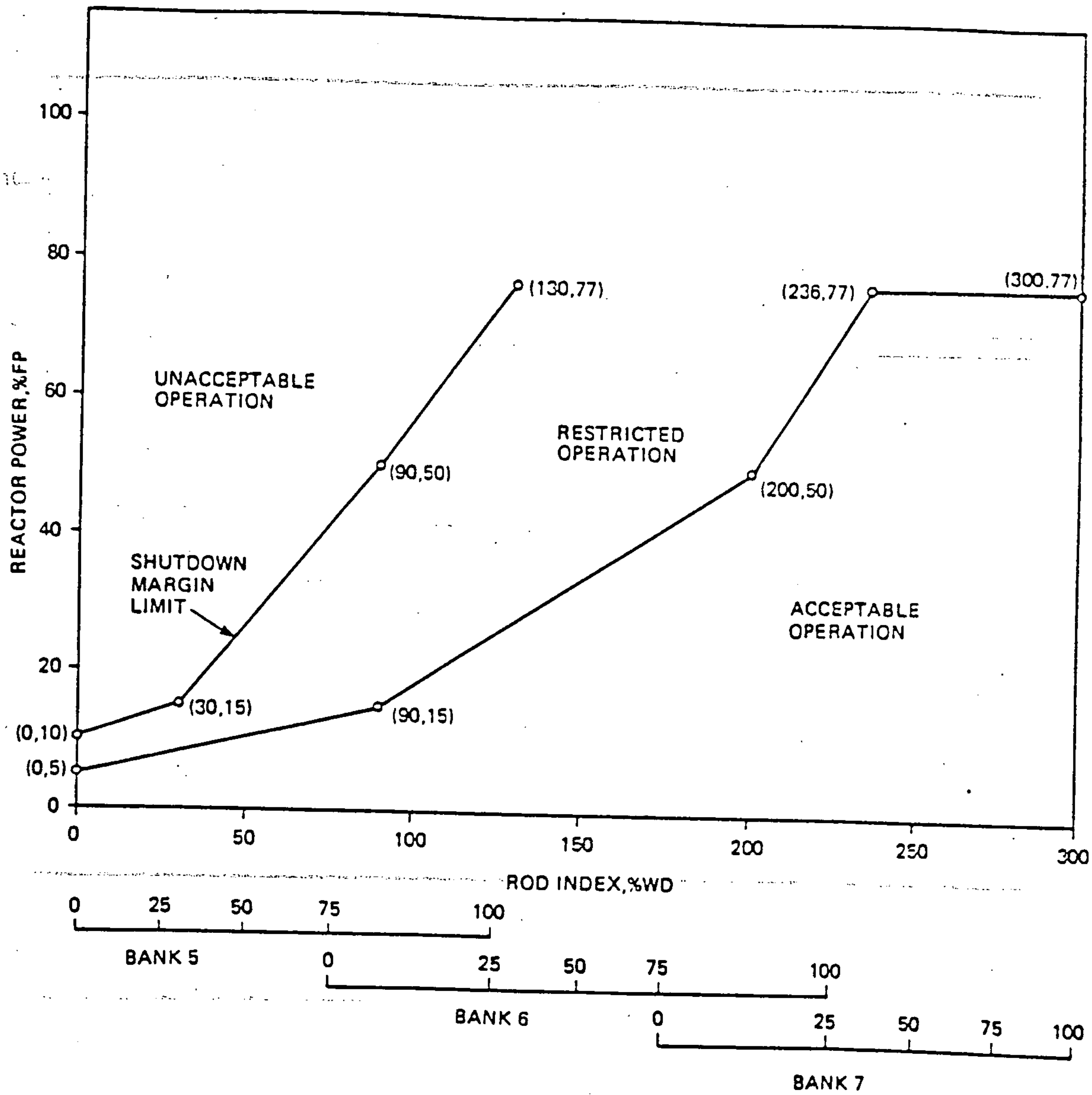
OCONEE NUCLEAR STATION
Figure 3.5.2-5
(3 of 3)



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 TO 25 +10/-0 EFPD
UNIT 3

OCONEE NUCLEAR STATION

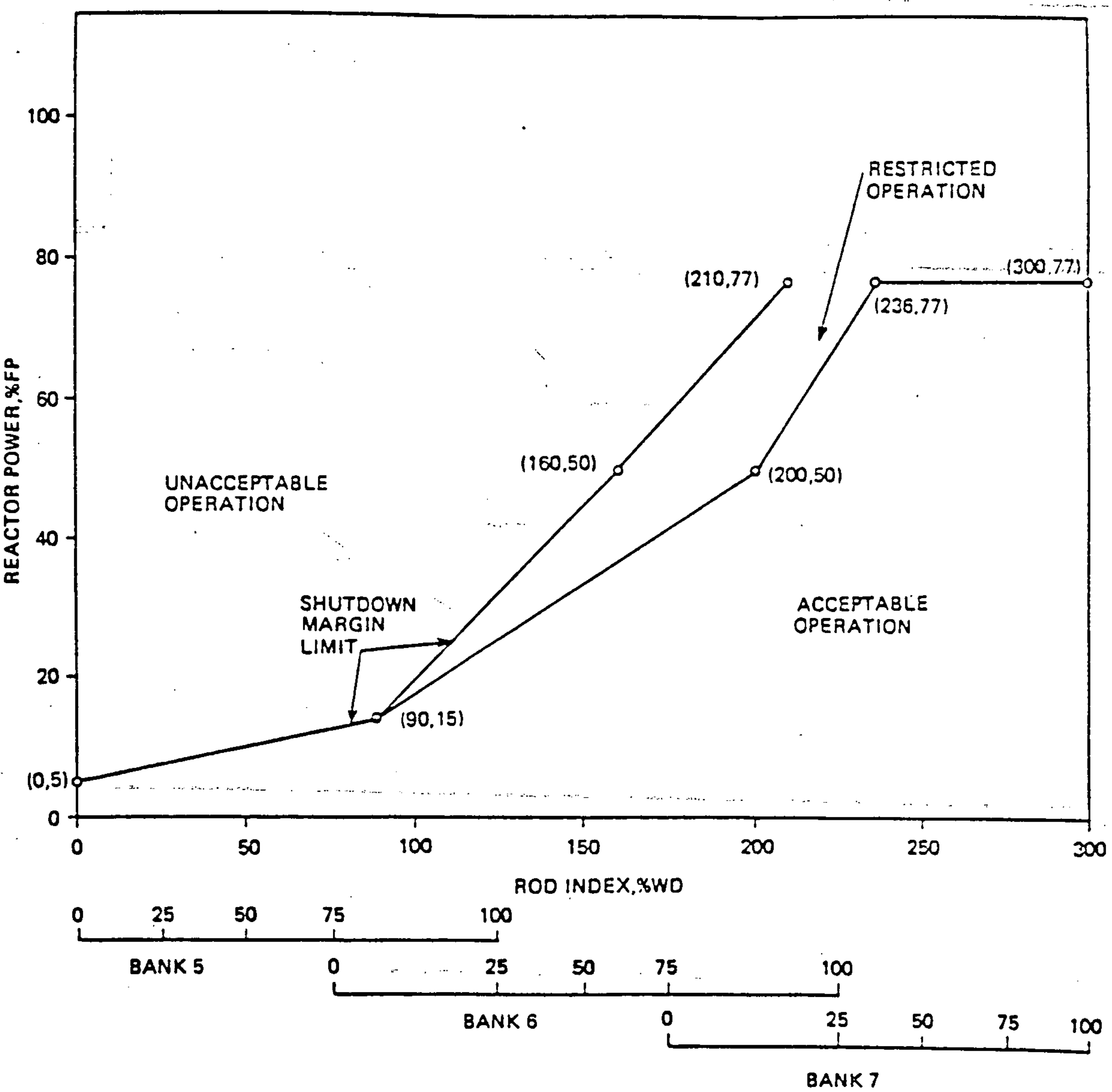
Figure 3.5.2-6
(1 of 3)



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 25 +10/-0 TO 200 +10/-10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

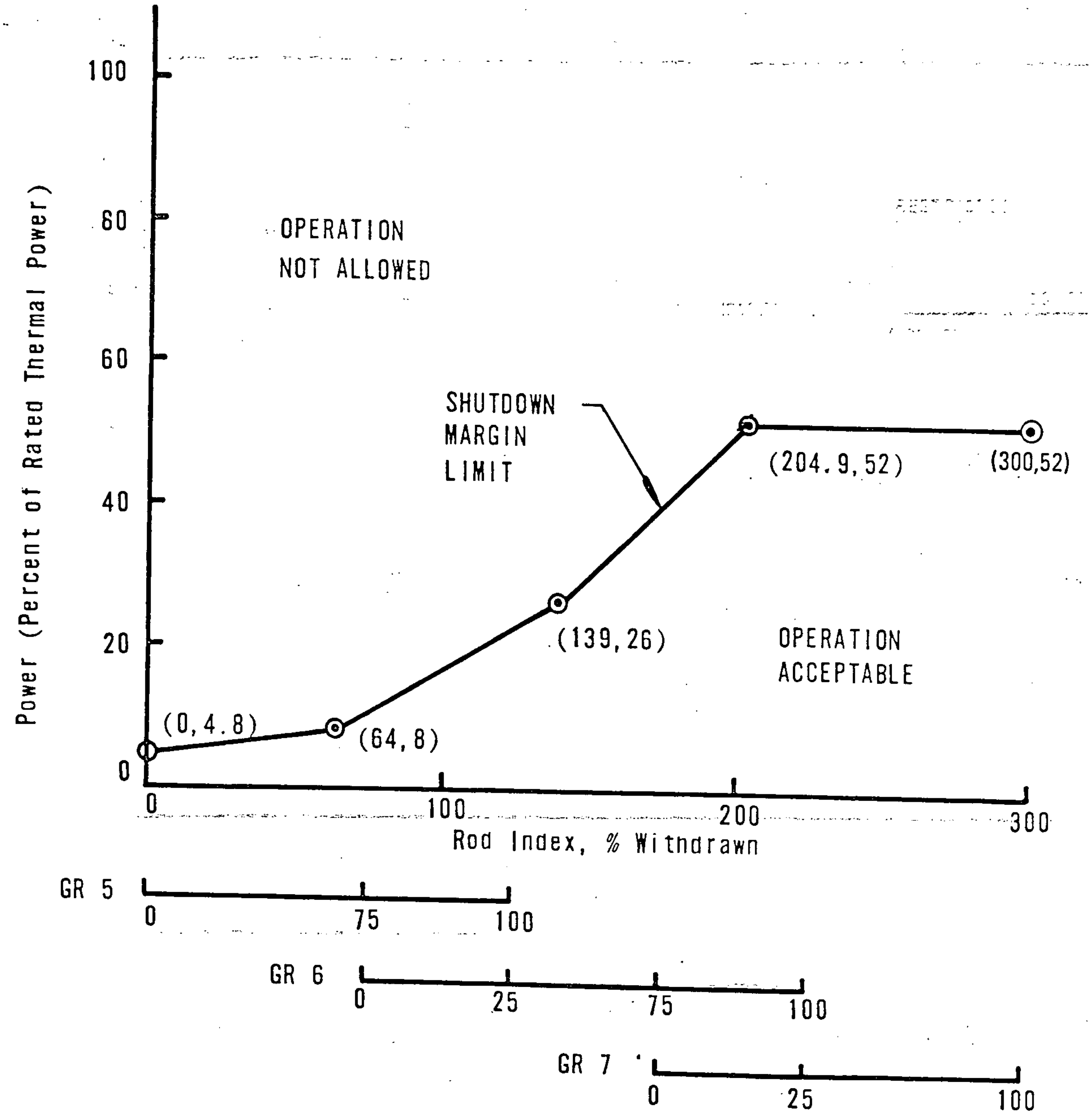
Figure 3.5.2-6
(2 of 3)



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
AFTER 200 +10/-10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

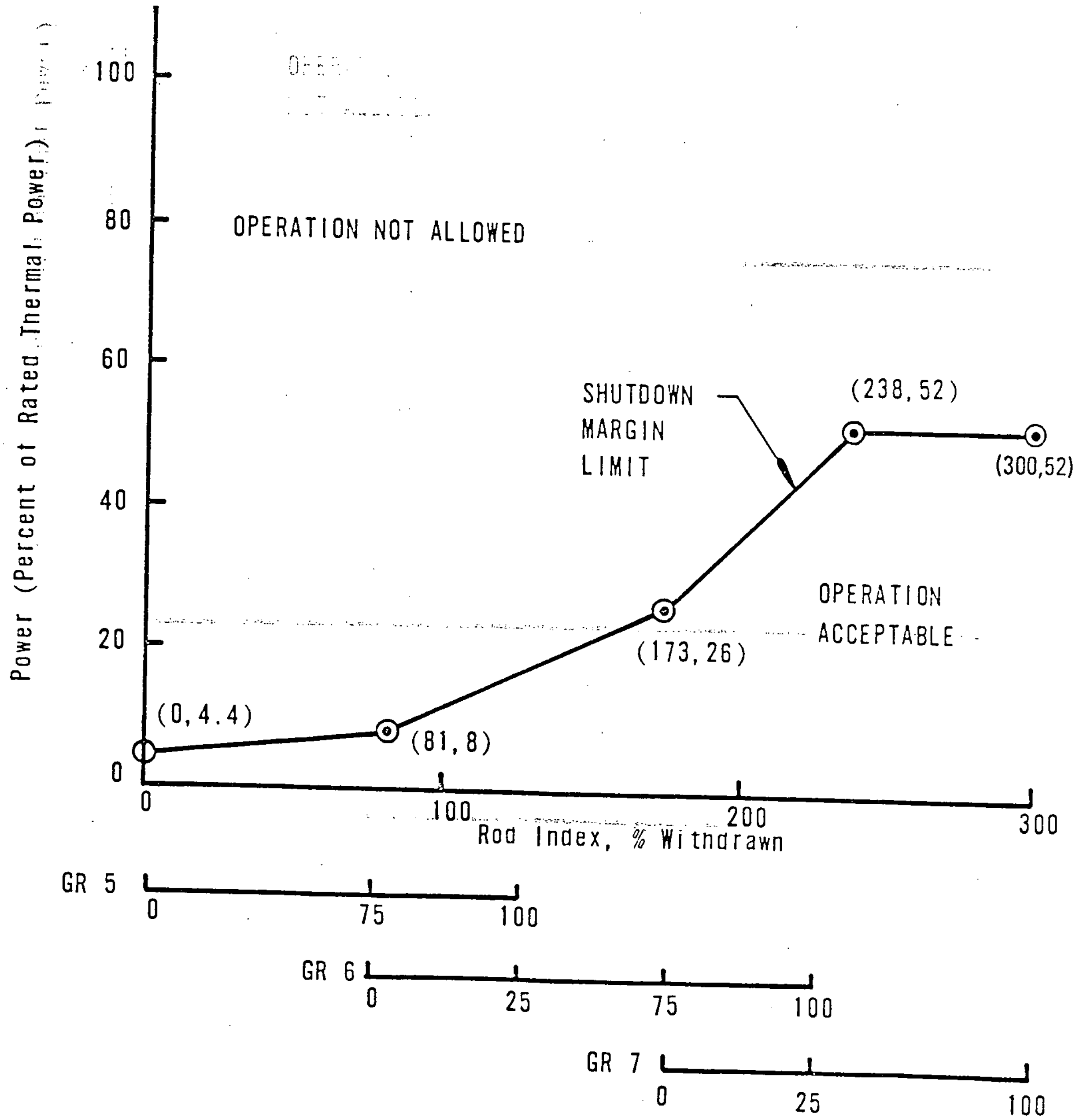
Figure 3.5.2-6
(3 of 3)



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 TO 26 +10/-0 EFPD
UNIT 1

OCONEE NUCLEAR STATION

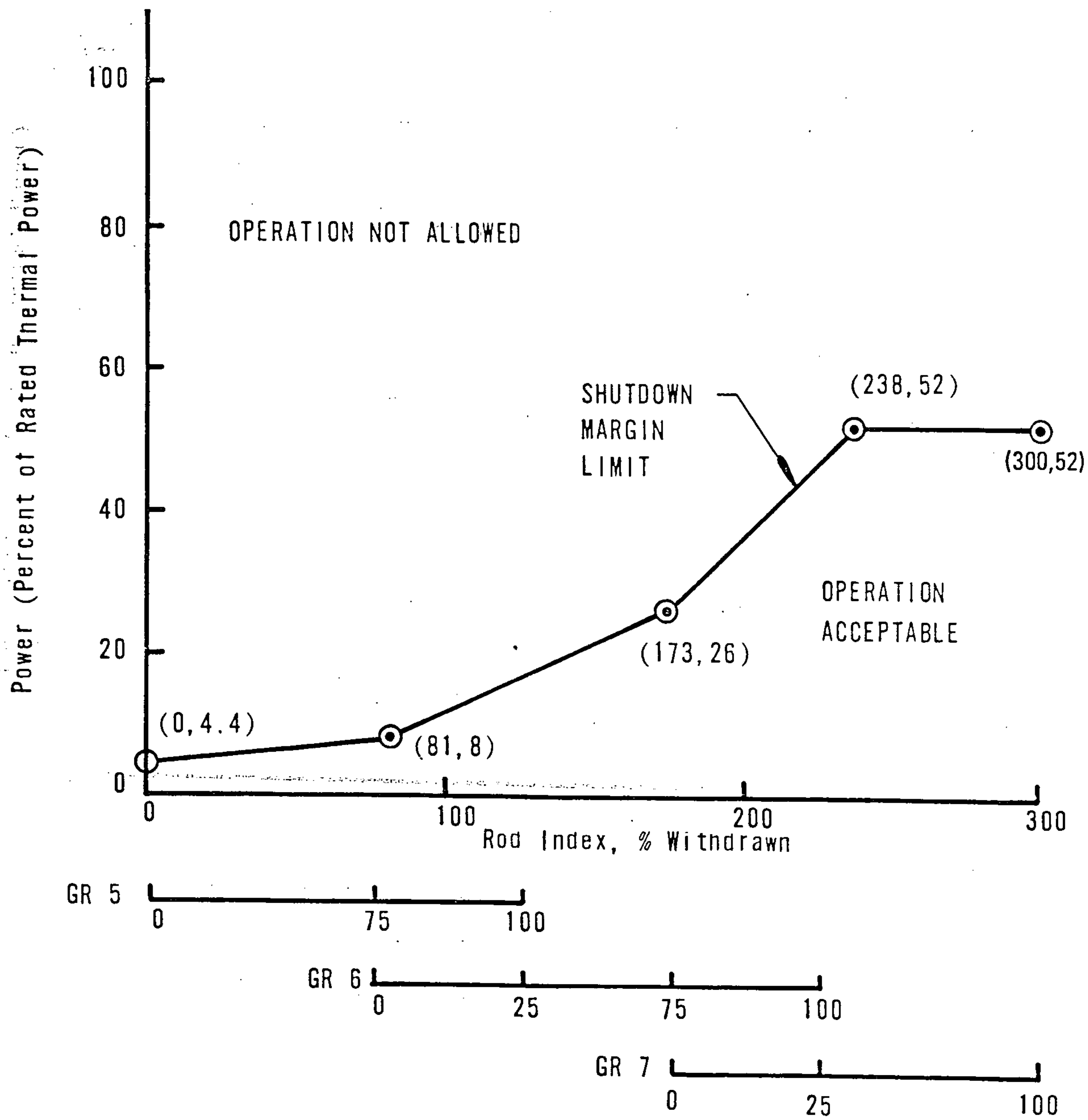
Figure 3.5.2-7
(1 of 3)



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1

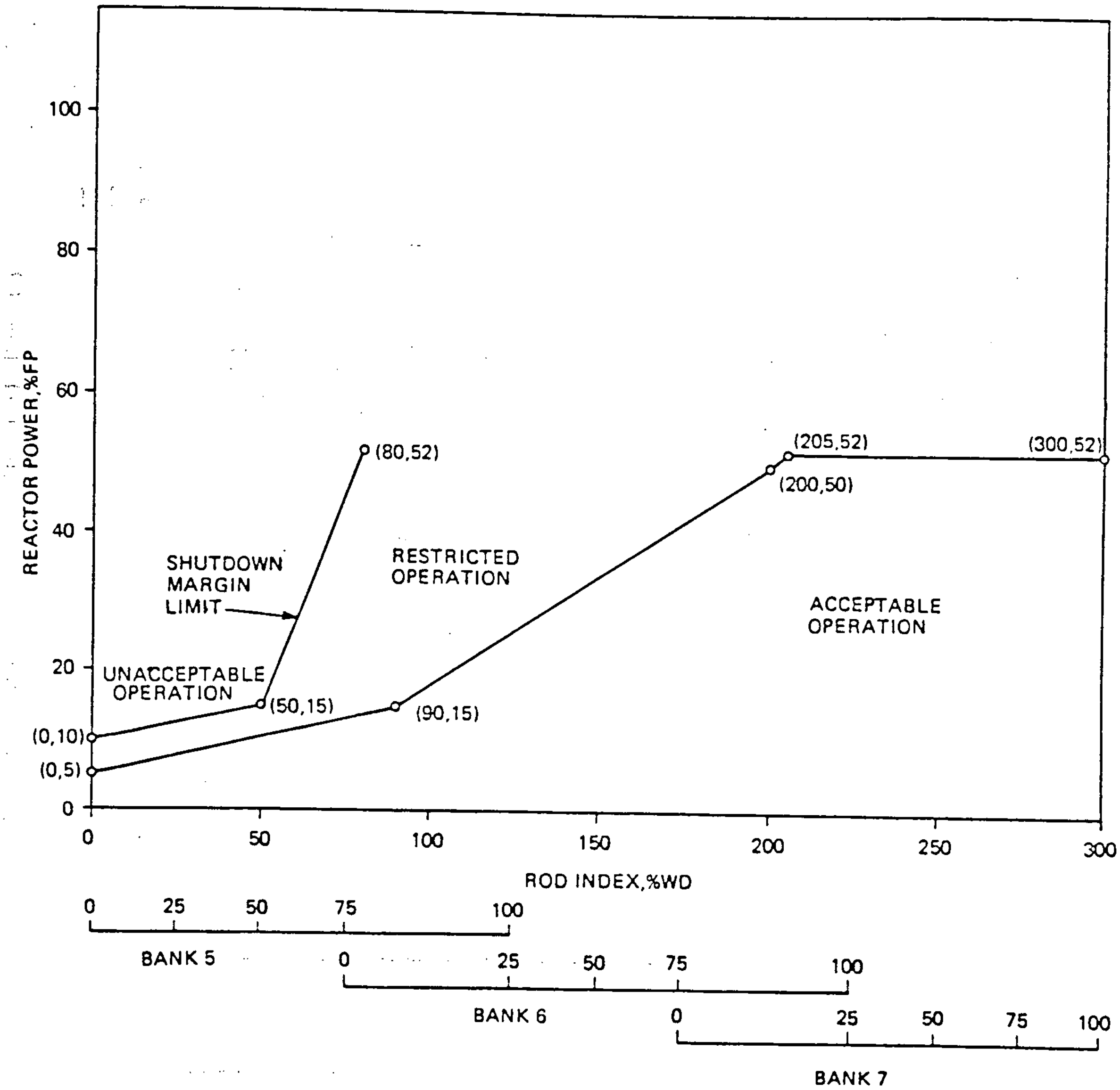
OCONEE NUCLEAR STATION

Figure 3.5.2-7
(2 of 3)



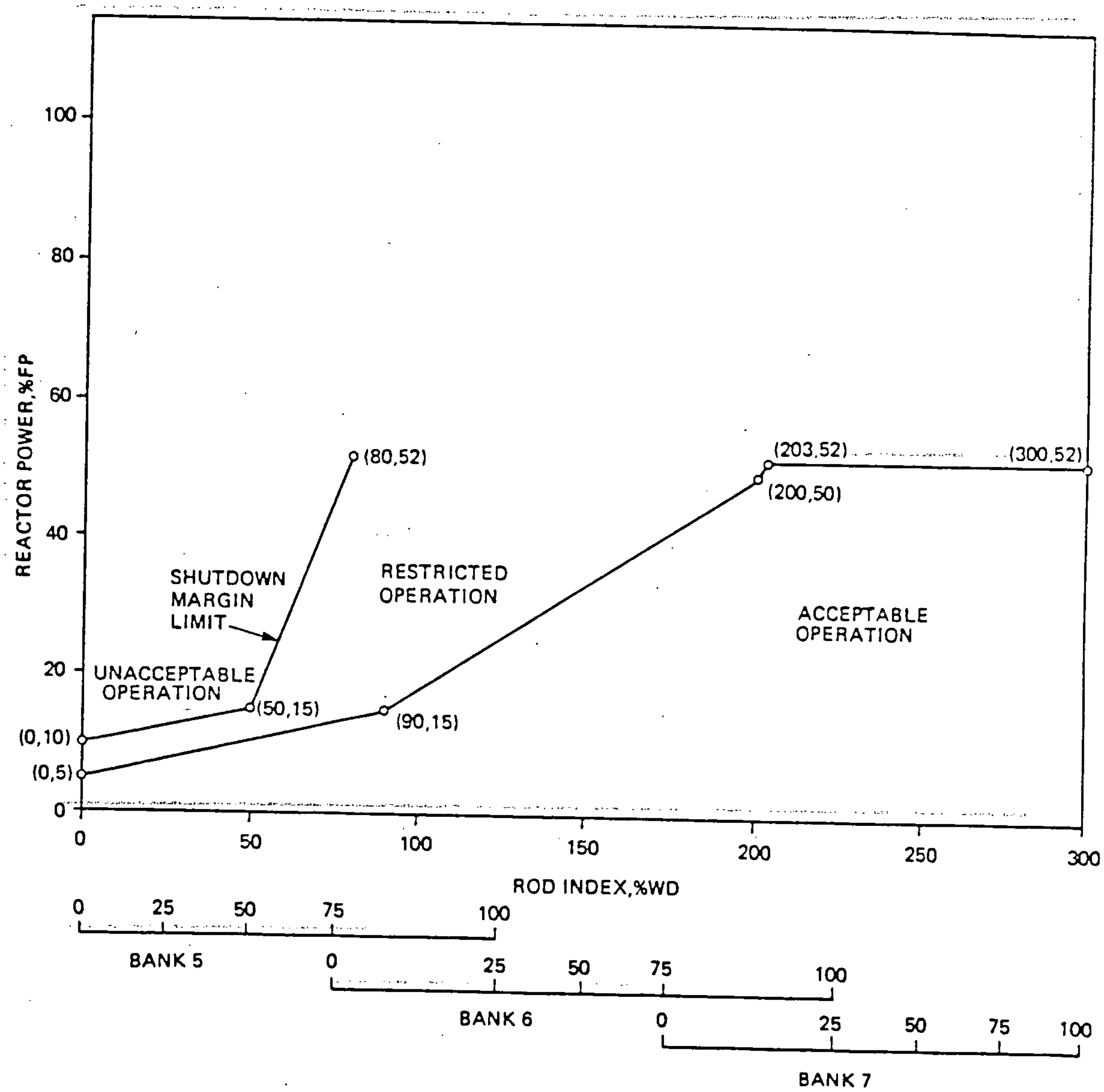
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
AFTER 200 ±10 EFPD
UNIT 1

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



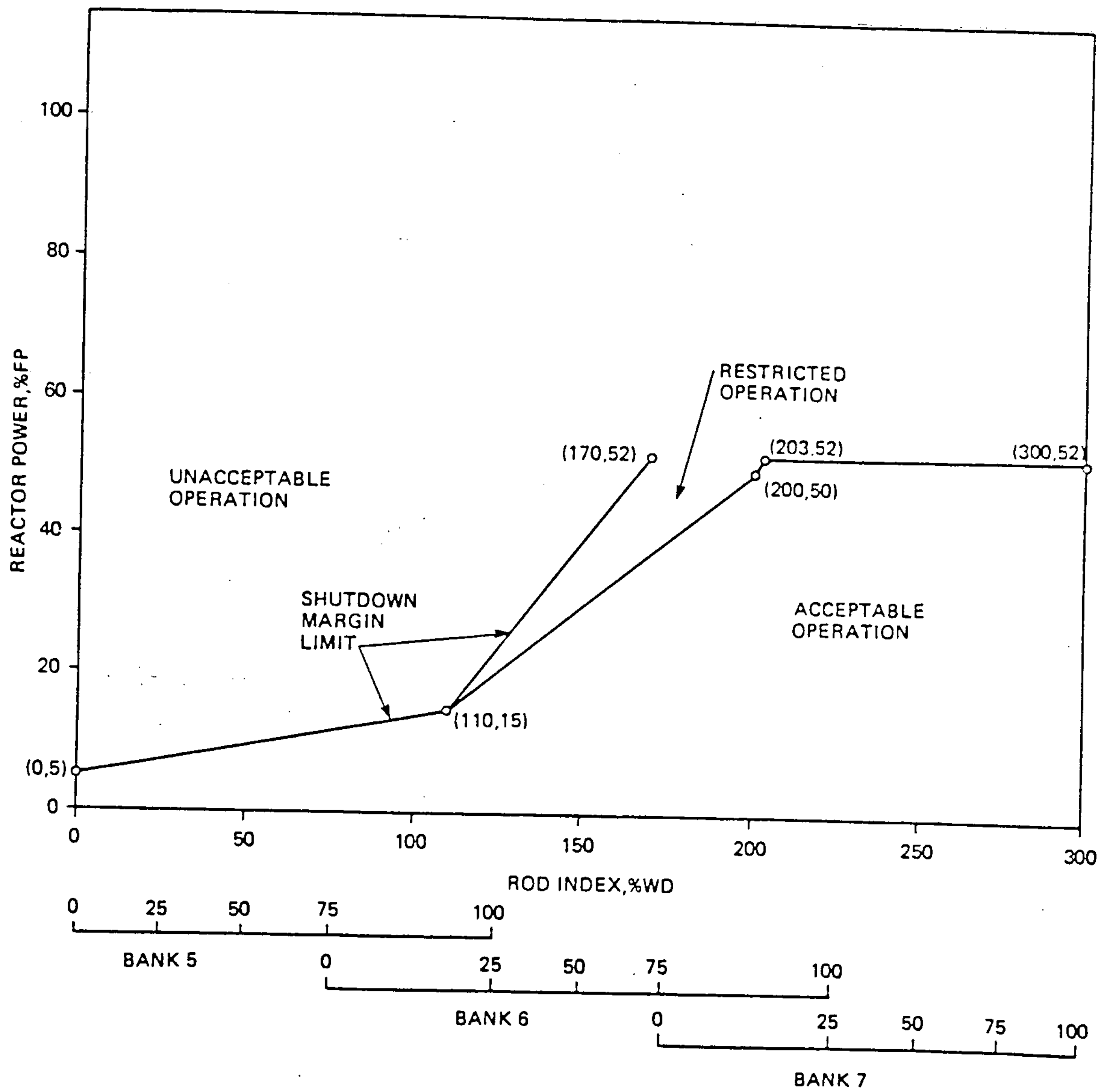
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 TO 25 (+10, -0) EFPD
UNIT 2

OCONEE NUCLEAR STATION
Figure 3.5.2-8
(1 of 3)



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 25 (+10,-0) TO 200 ±10 EFPD
UNIT 2

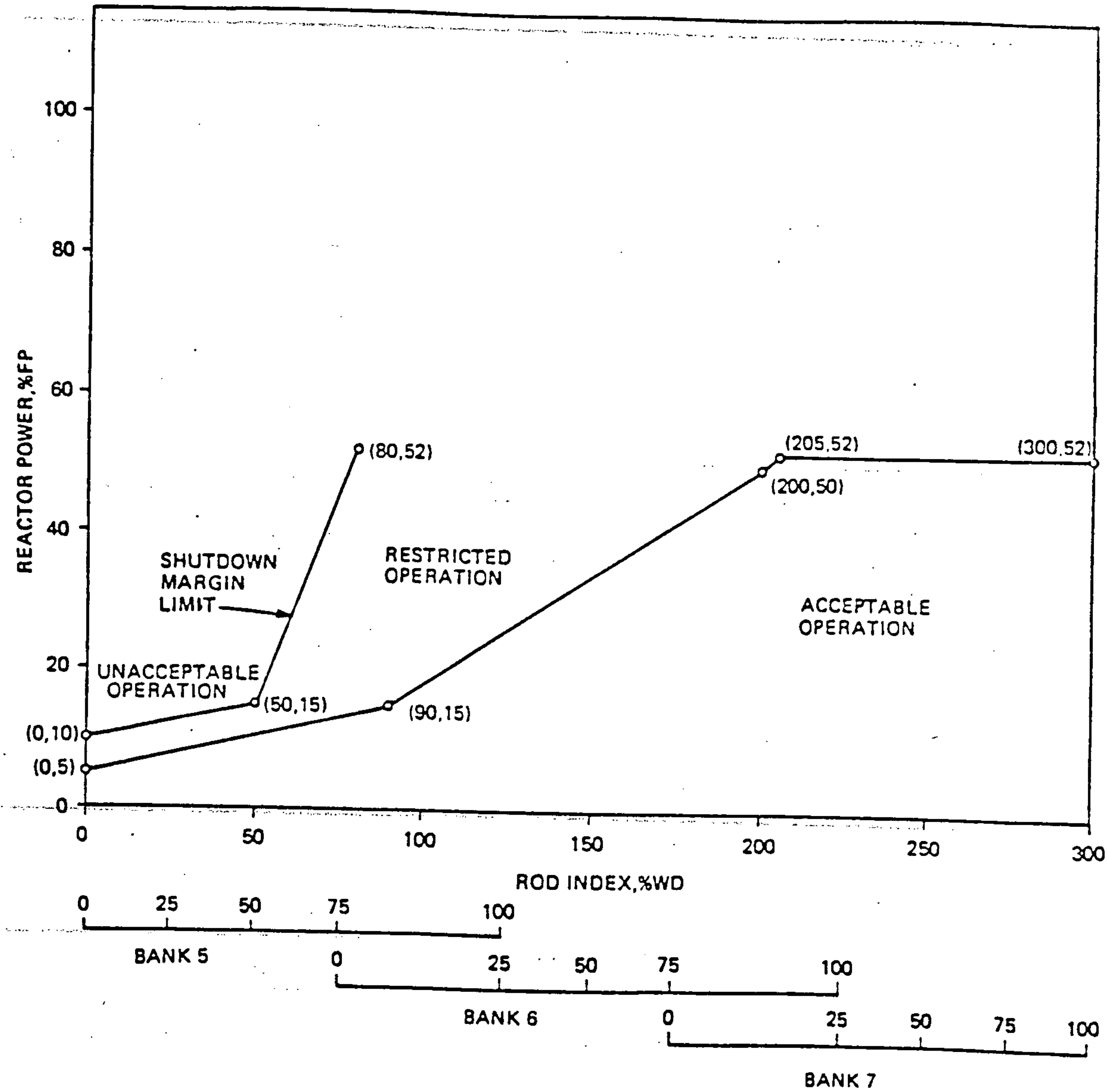
OCONEE NUCLEAR STATION
Figure 3.5.2-8
(2 of 3)



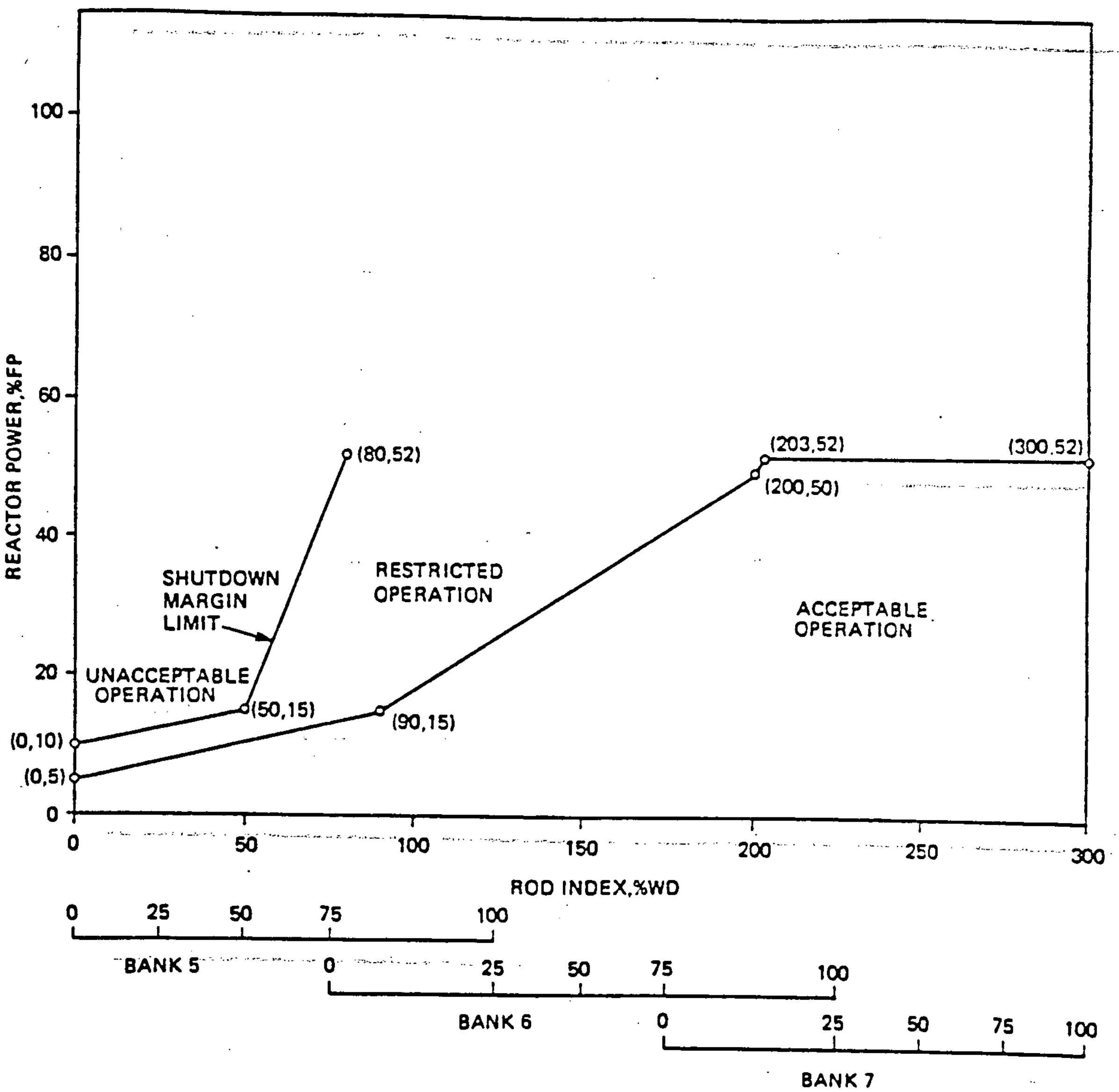
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
AFTER 200 ±10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

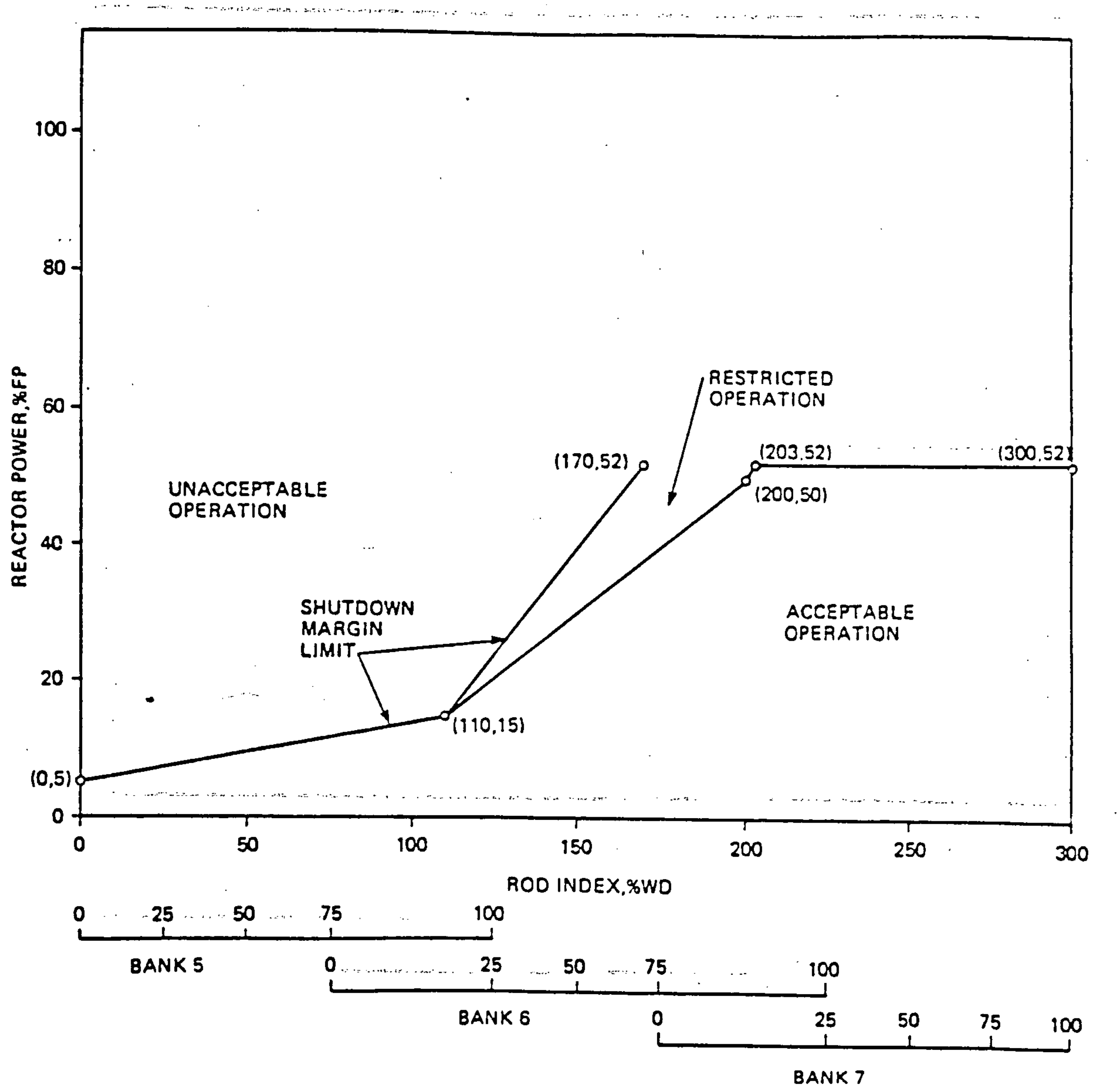
Figure 3.5.2-8
(3 of 3)



ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 TO 25 +10/-0 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-9
(1 of 3)



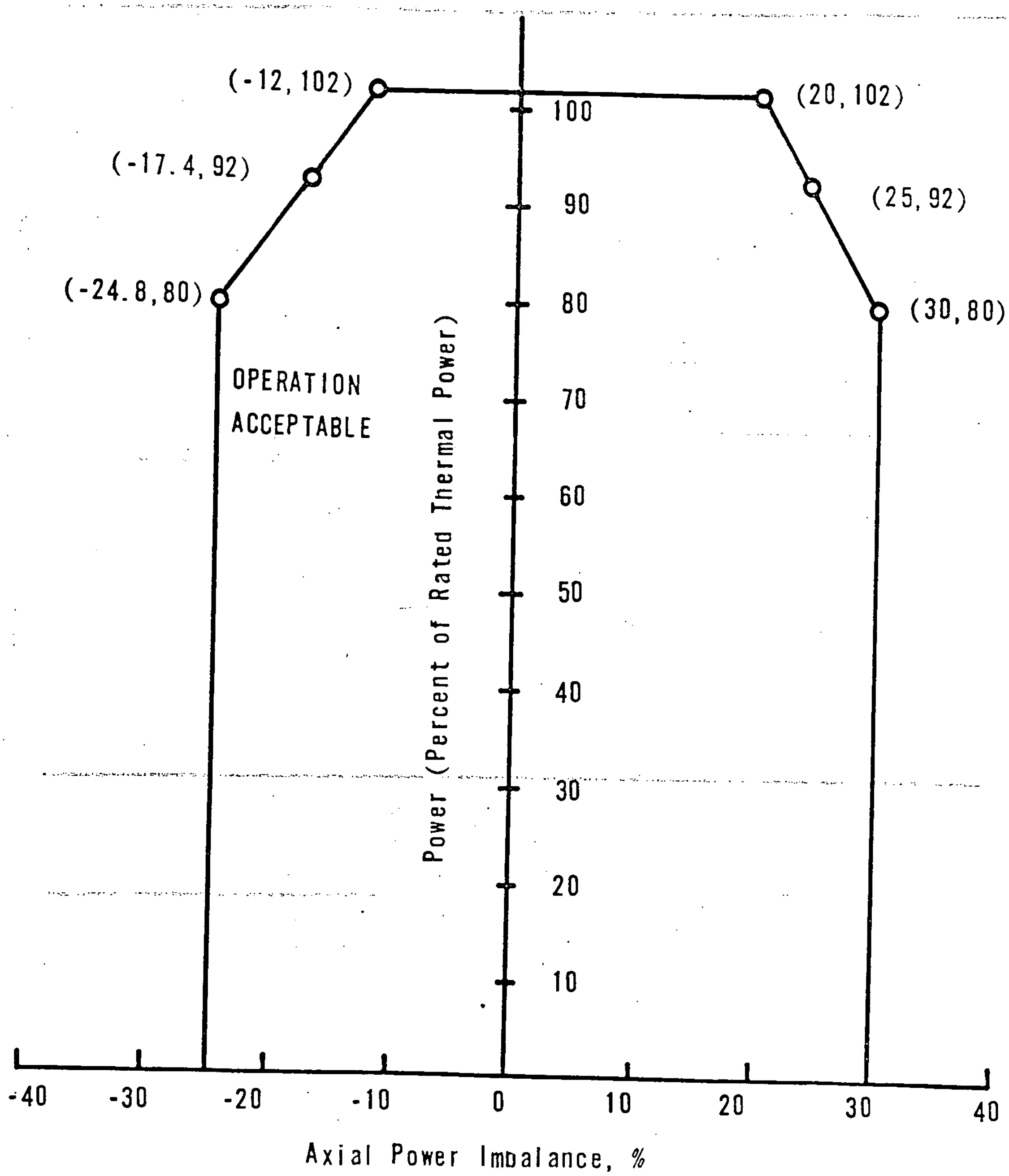
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 25 +10/-0 TO 200 +10/-10 EFPI
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-9
(2 of 3)



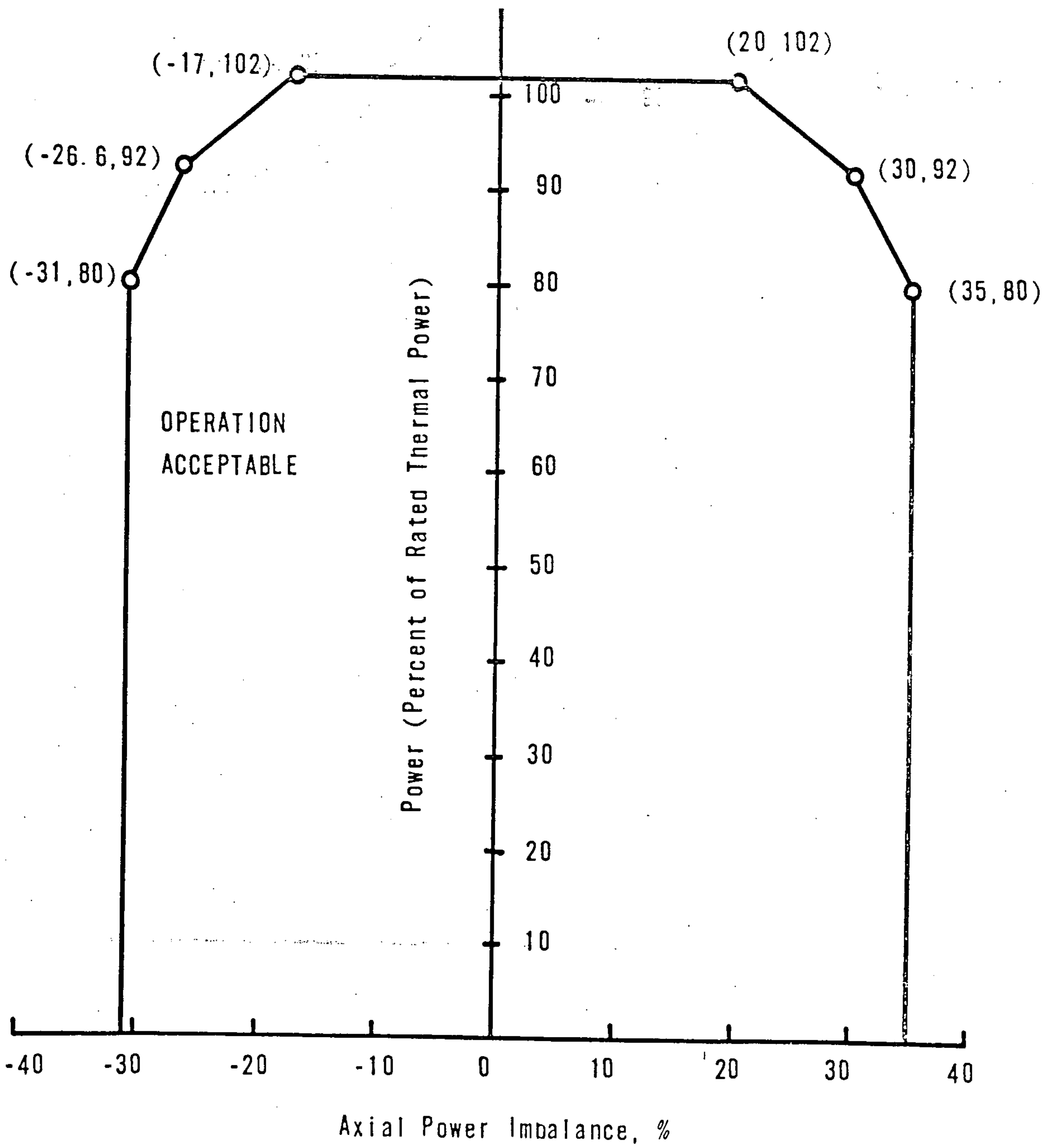
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
AFTER 200 +10/-10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-9
(3 of 3)



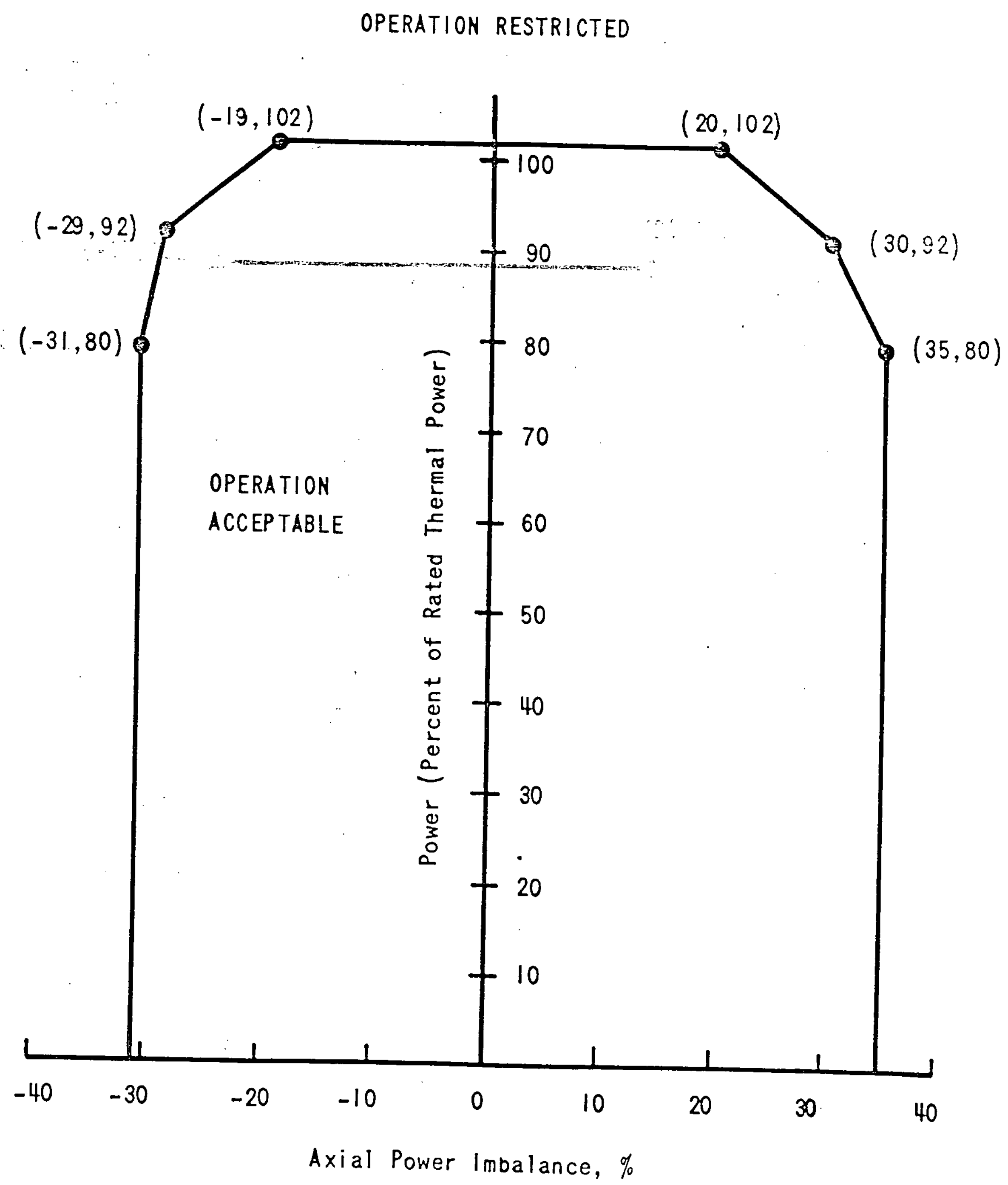
OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 TO 26 +10/-0 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-10
(1 of 3)



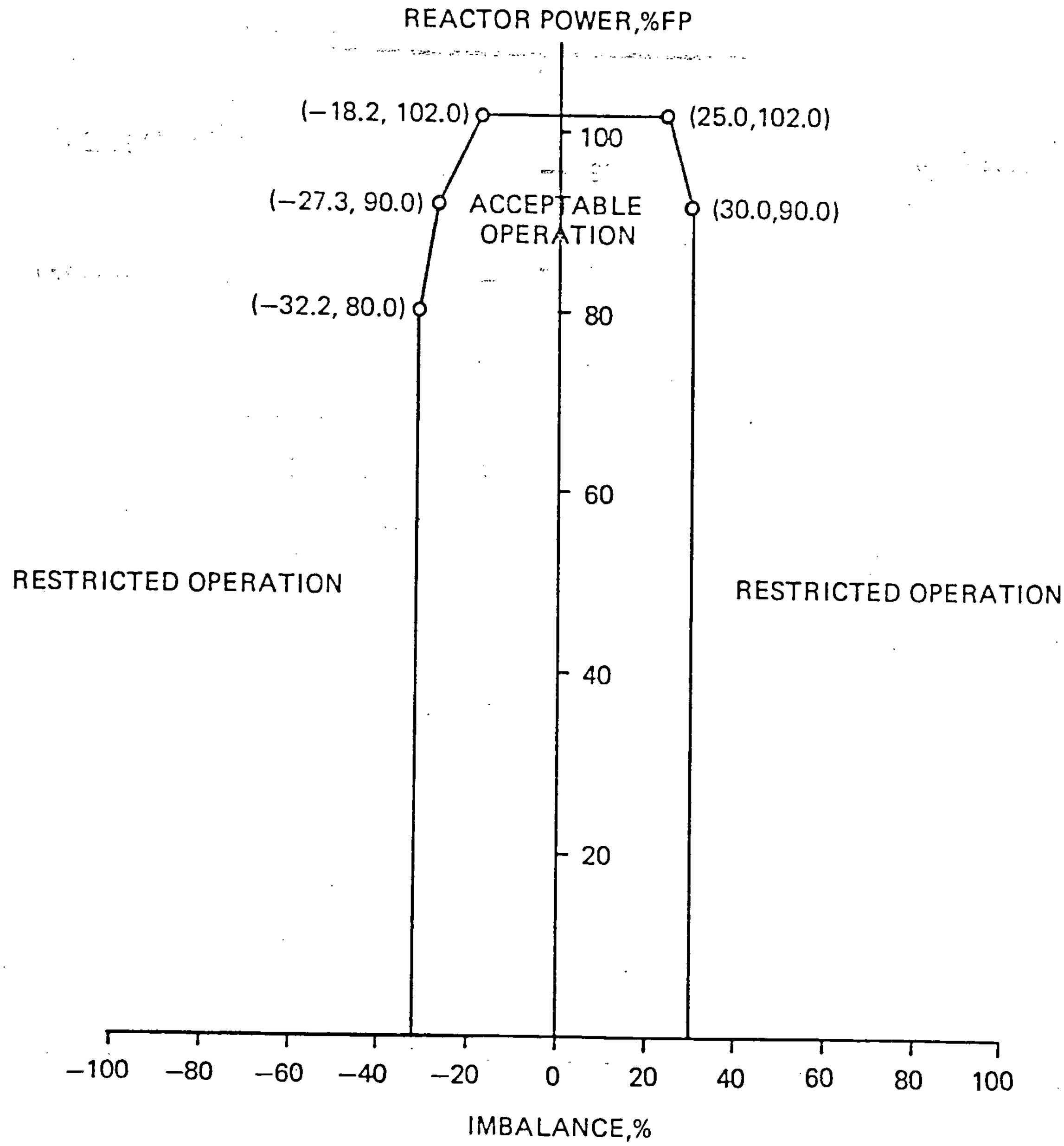
OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-10
(2 of 3)



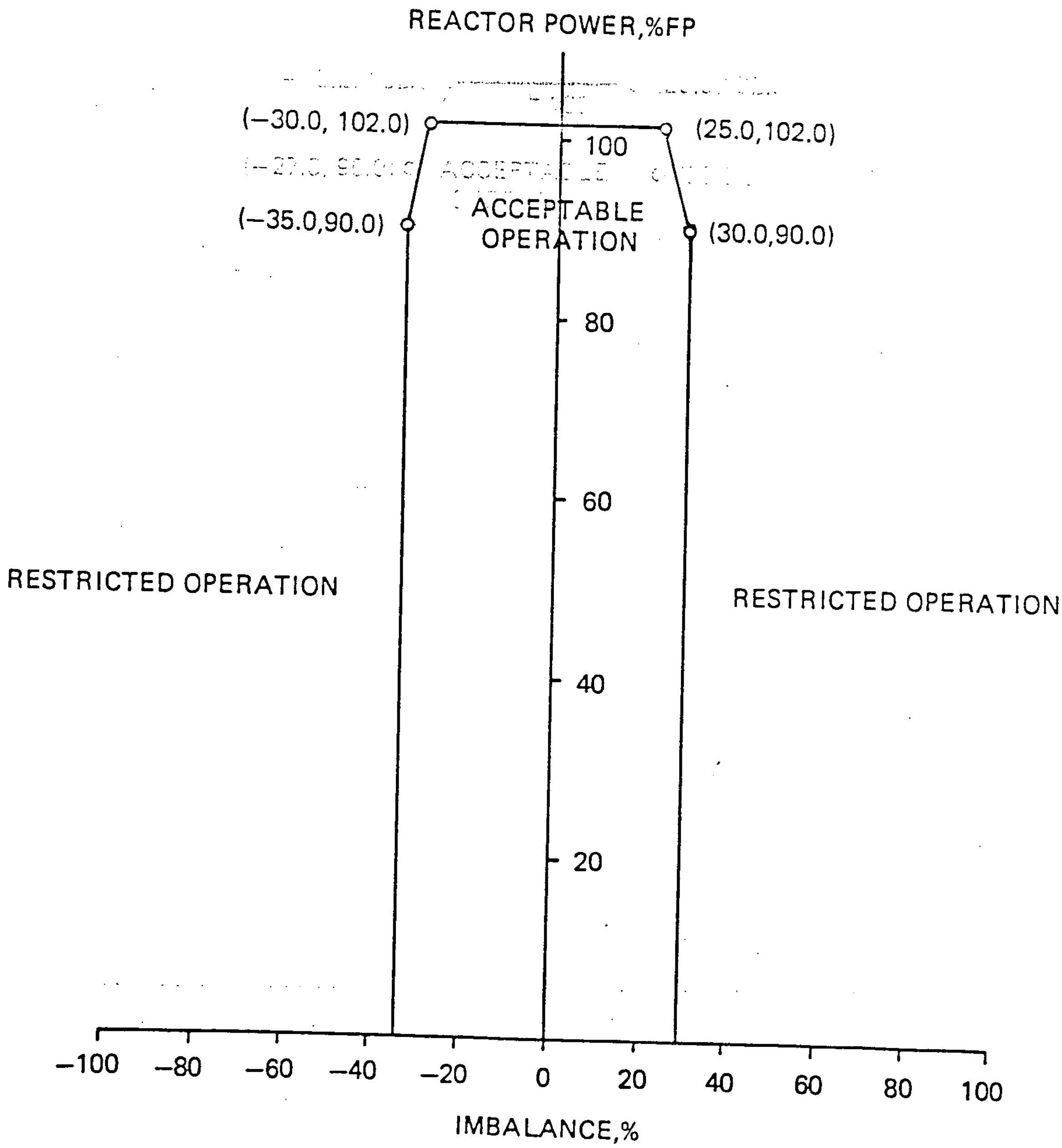
OPERATIONAL POWER
IMBALANCE ENVELOPE
AFTER 200 ±10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-10
(3 of 3)



OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 TO 25 (+10, -0) EFPD
UNIT 2

OCONEE NUCLEAR STATION

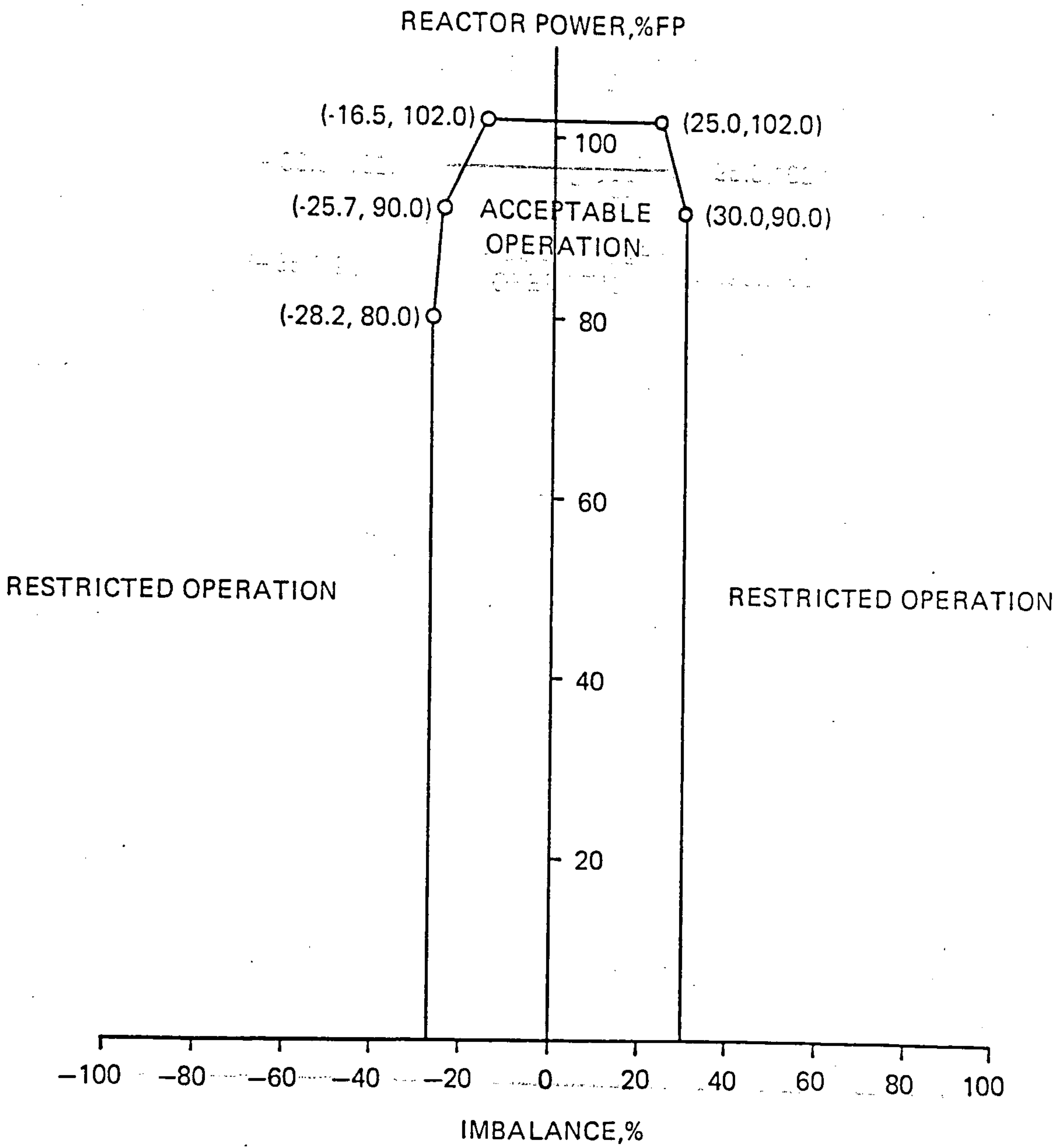
Figure 3.5.2-11
(1 of 2)



OPERATIONAL POWER
IMBALANCE ENVELOPE
AFTER 25 (+10, -0) EFPD
UNIT 2

OCONEE NUCLEAR STATION

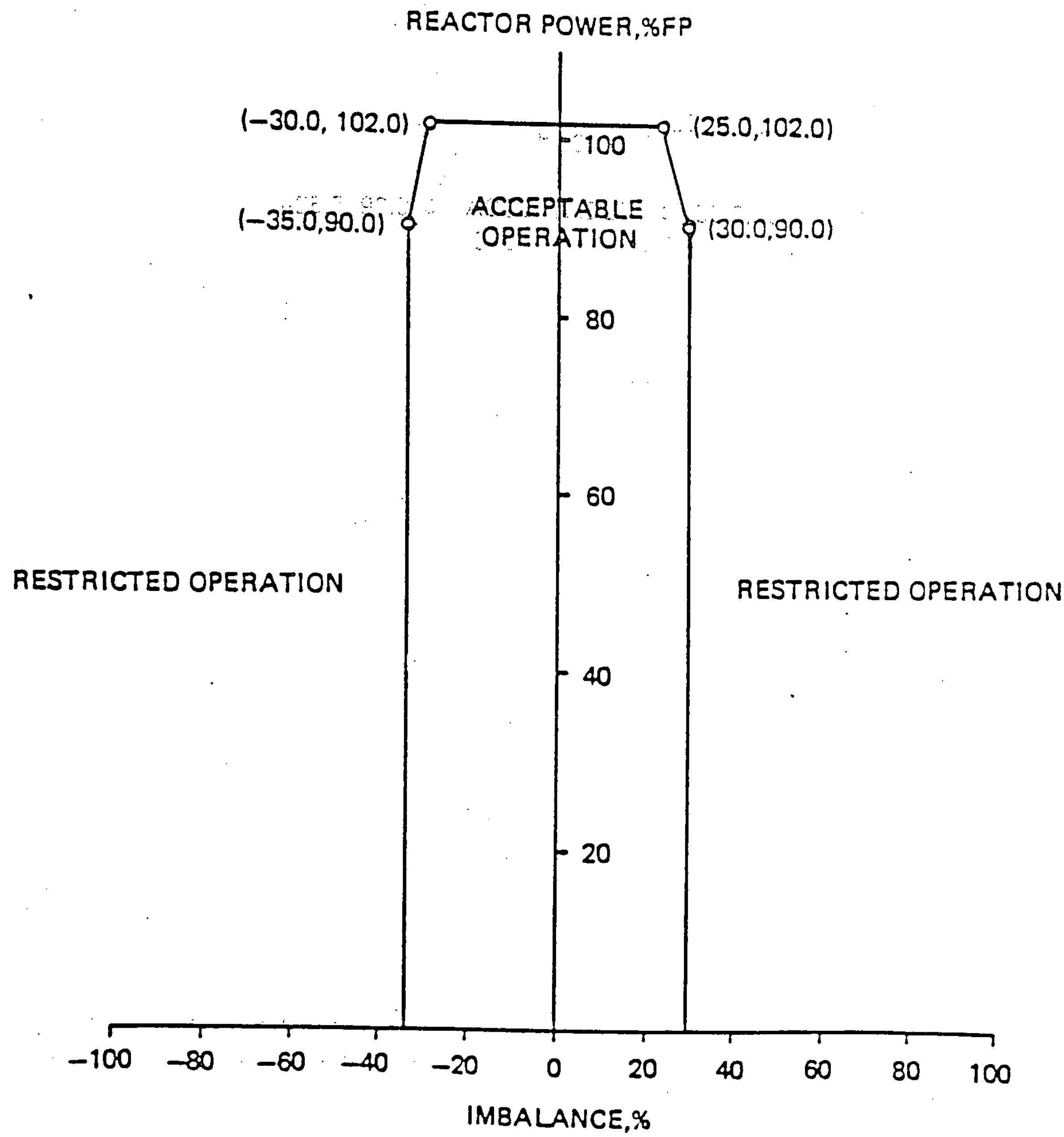
Figure 3.5.2-11
(2 of 2)



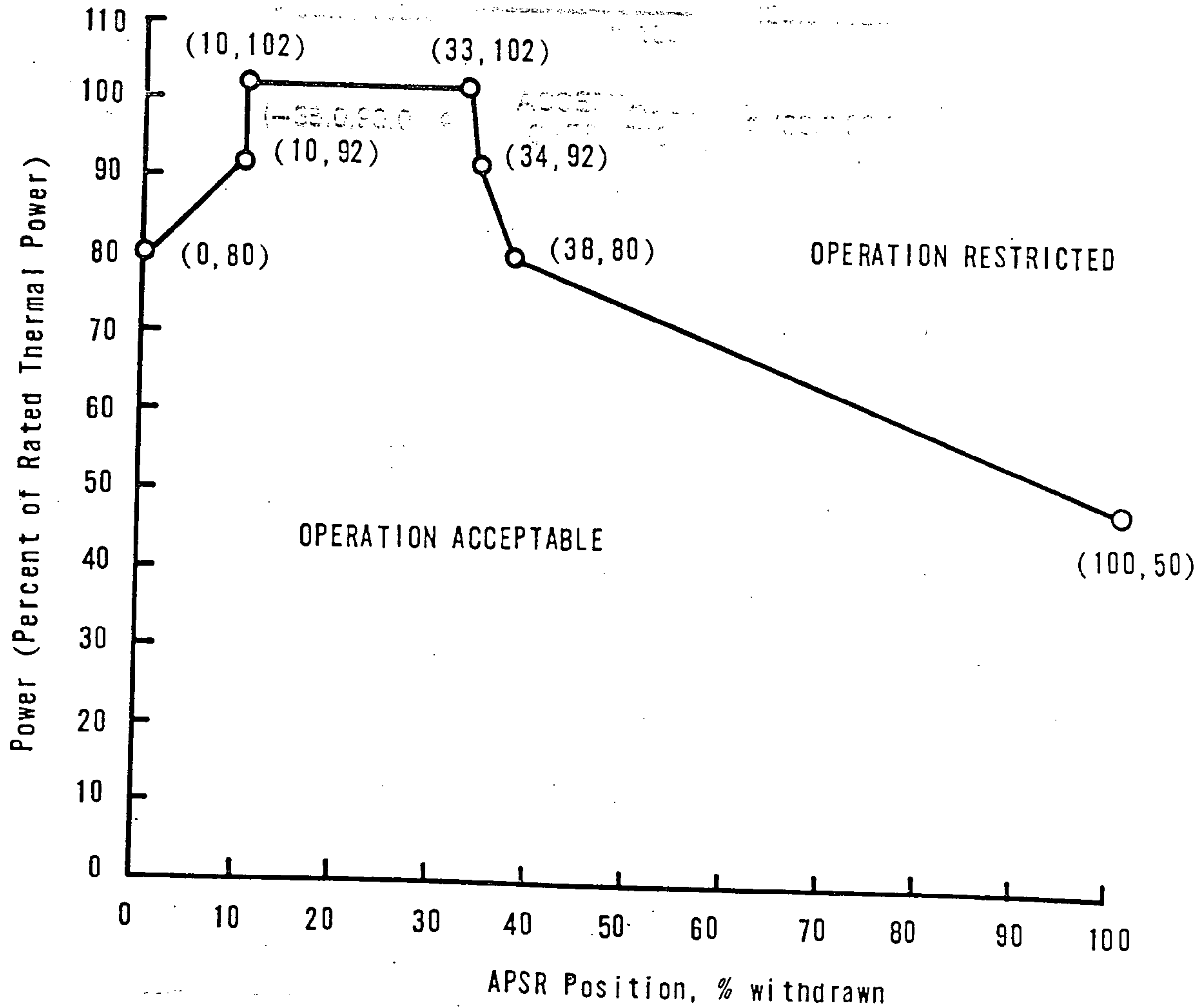
OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 TO 200 +10/-10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-12
(1 of 2)



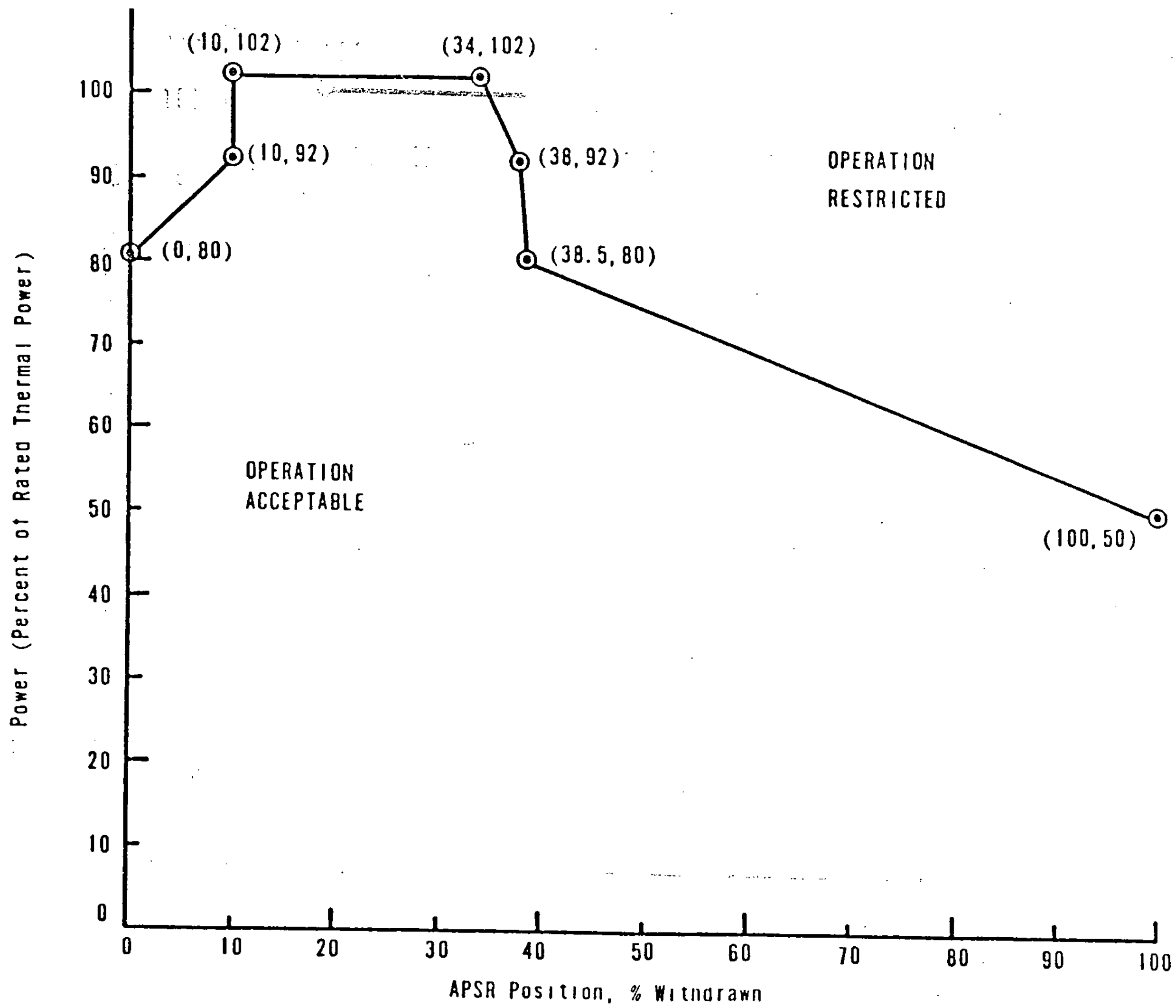
OPERATIONAL POWER
IMBALANCE ENVELOPE
AFTER 200 +10/-10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-12
(2 of 2)



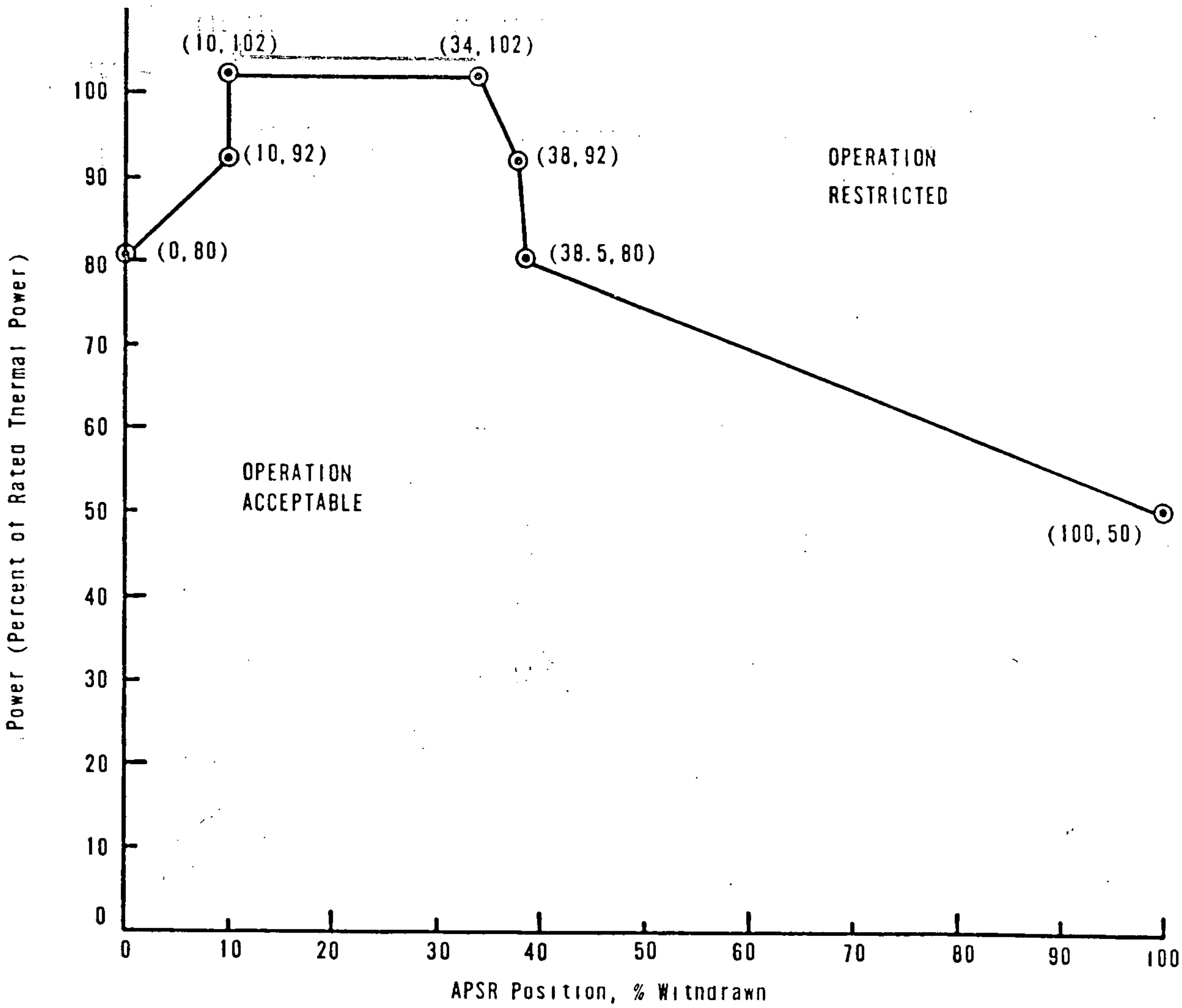
APSR POSITION LIMITS
FOR OPERATION
FROM 0 TO 26 +10/-0 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-13
(1 of 3)



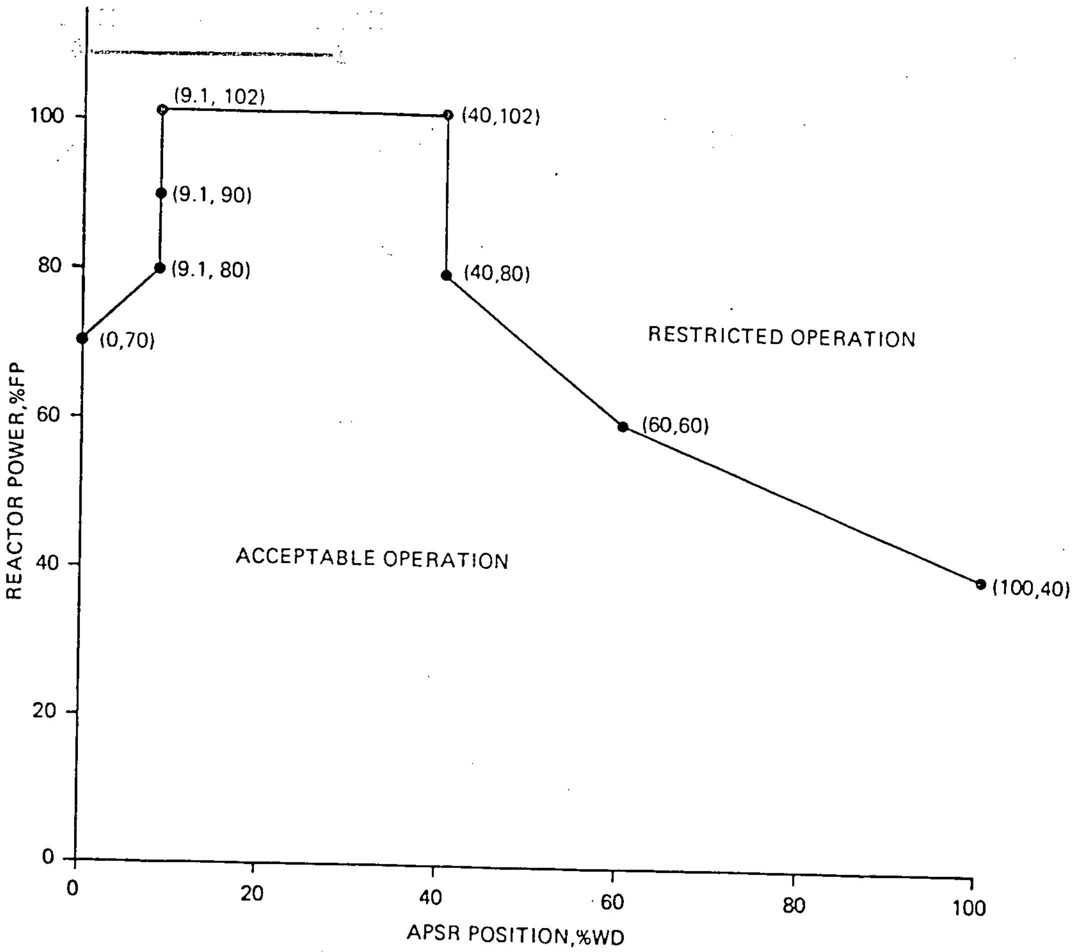
APSR POSITION LIMITS
FOR OPERATION
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-13
(2 of 3)



APSR POSITION LIMITS
FOR OPERATION
AFTER 200 ±10 EFPD
UNIT 1

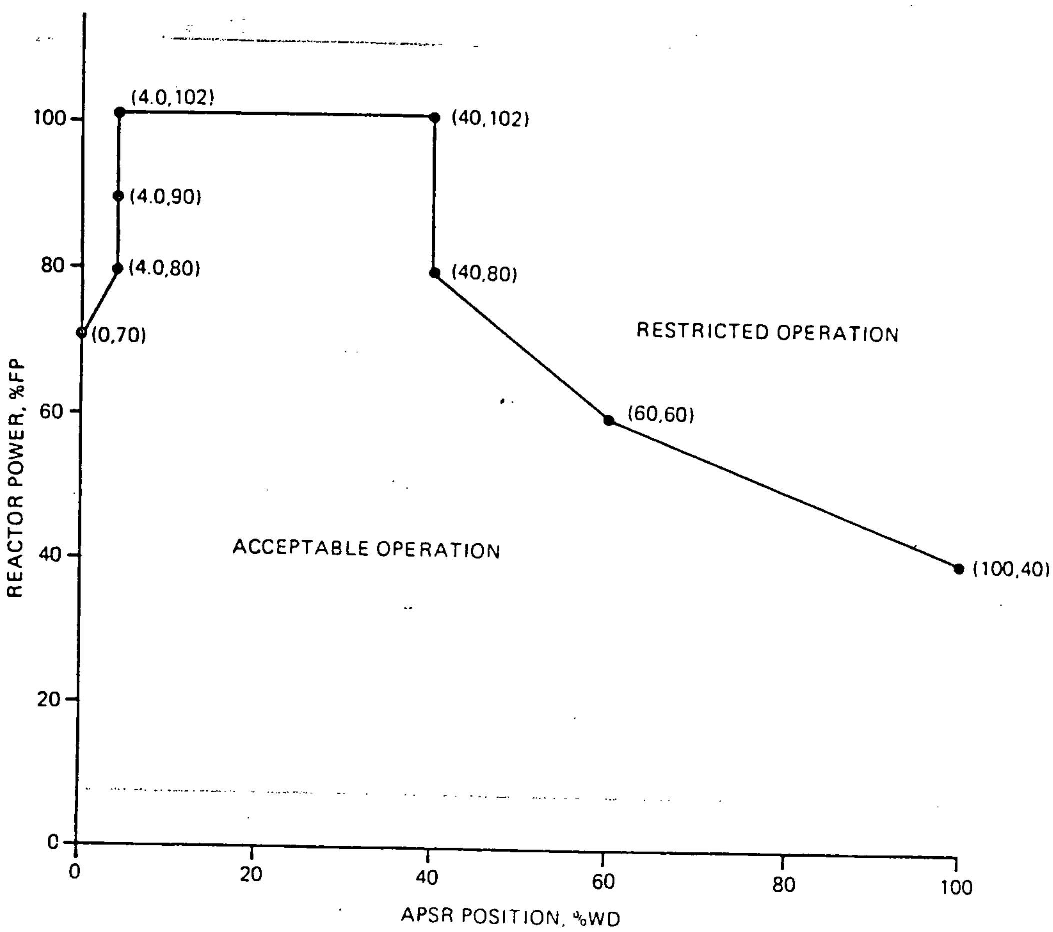
OCONEE NUCLEAR STATION

Figure 3.5.2-13
(3 of 3)



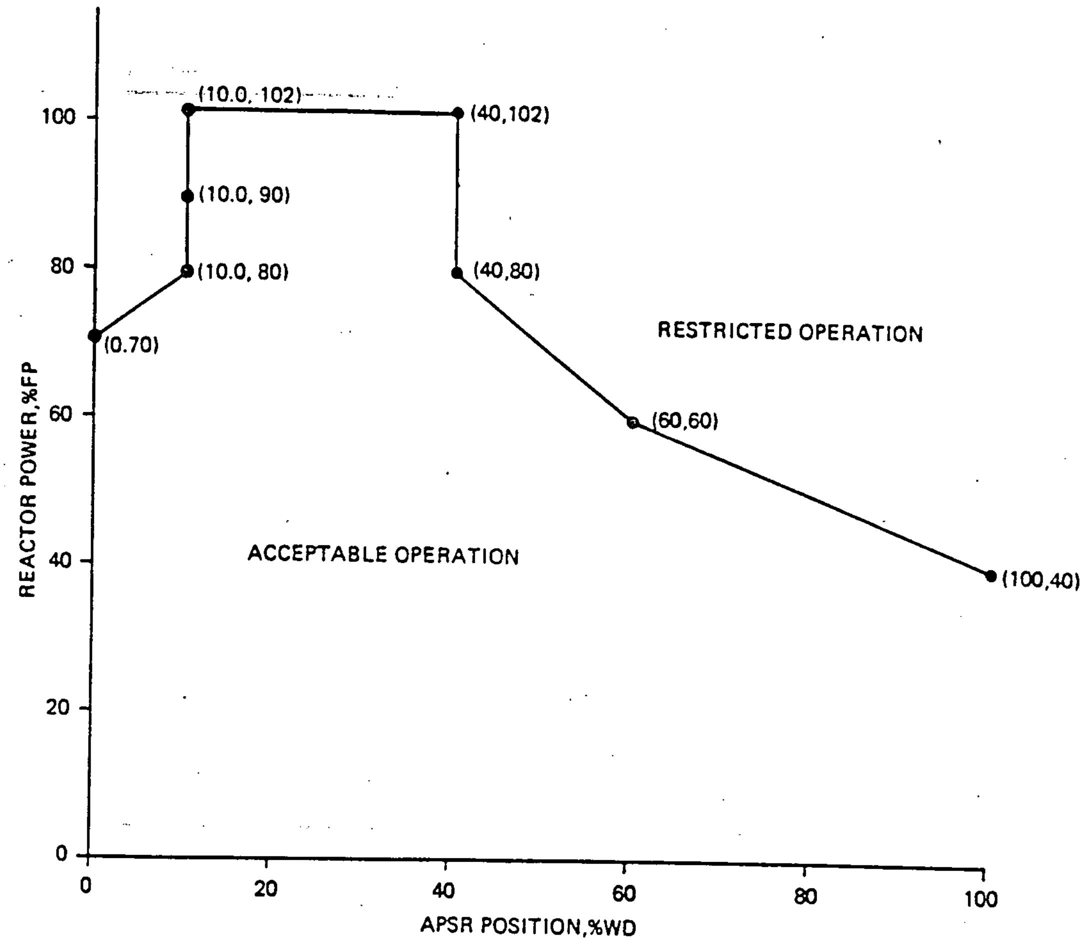
APSR POSITION LIMITS
FOR OPERATION
FROM 0 EFPD TO EOC
UNIT 2

OCONEE NUCLEAR STATION
Figure 3.5.2-14
(1 of 1)



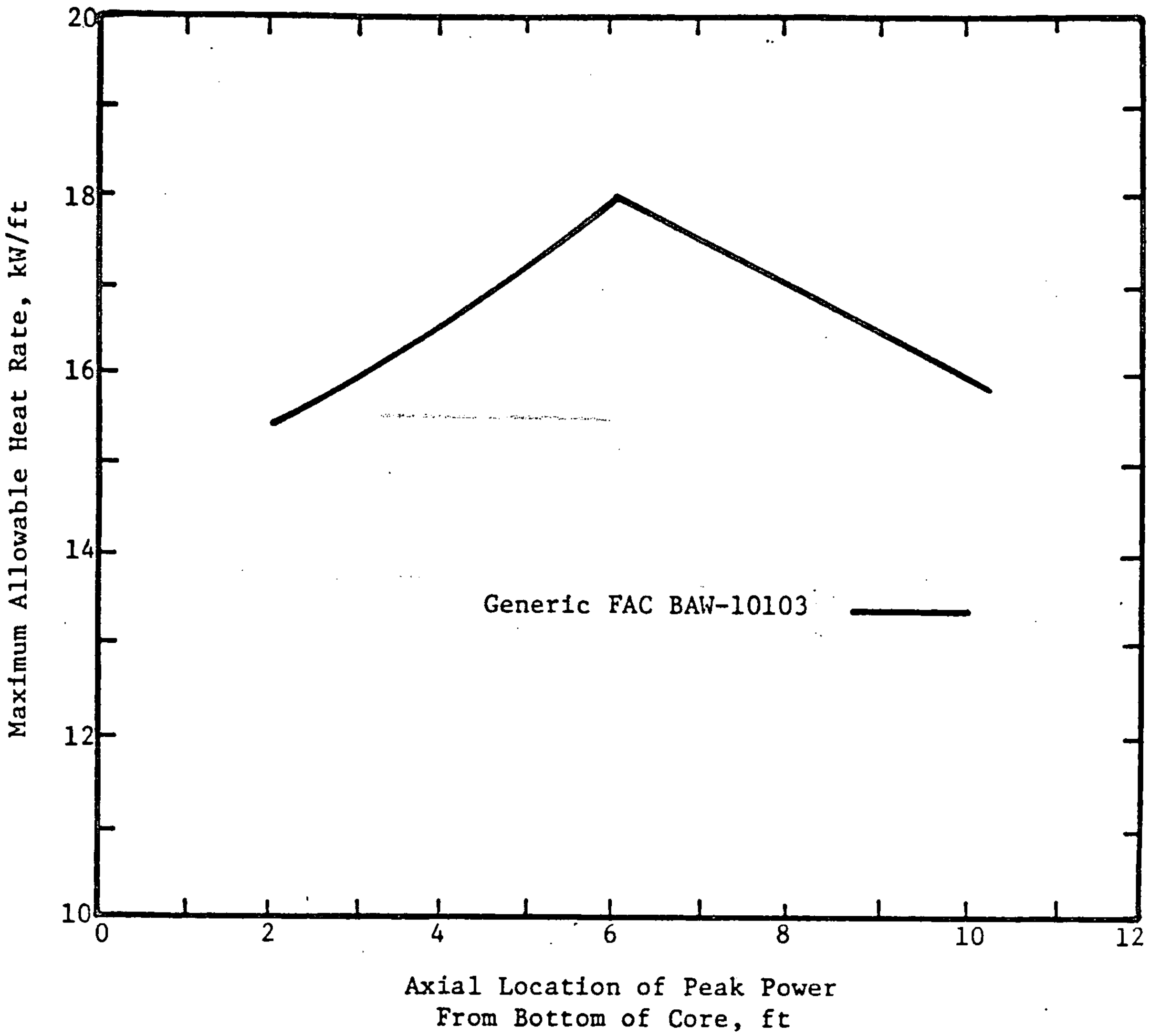
APSR POSITION LIMITS
FOR OPERATION
FROM 0 TO 200 +10/-10 EFPD
UNIT 3

OCONEE NUCLEAR STATION
Figure 3.5.2-15
(1 of 2)



APSR POSITION LIMITS
FOR OPERATION
AFTER 200 +10/-10 EFPD
UNIT 3

OCONEE NUCLEAR STATION
Figure 3.5.2-15
(2 of 2)



LOCA-LIMITED MAXIMUM ALLOWABLE
LINEAR HEAT

OCONEE NUCLEAR STATION

Figure 3.5.2-16

3.5.3 Engineered Safety Features Protective System Actuation Setpoints

Applicability

This specification applies to the engineered safety features protective system actuation setpoints.

Objective

To provide for automatic initiation of the engineered safety features protective system in the event of a breach of RCS integrity.

Specification

The engineered safety features protective actuation setpoints and permissible bypasses shall be as follows:

Functional Unit	Action	Setpoint
High Reactor Building Pressure	Reactor Building Spray	≦30 psig
	High-Pressure Injection	≦4 psig
	Low-Pressure Injection	≦4 psig
	Start Reactor Building Cooling & Reactor Building Isolation (Essential and Non-essential Systems)	≦4 psig
	Penetration Room Ventilation	≦4 psig
Lower Reactor Coolant System Pressure	High Pressure Injection(1) & Reactor Building Isolation (Non-essential systems)	≧1500 psig
	Low Pressure Injection(2)	≧500 psig

(1) May be bypassed below 1750 psig and is automatically reinstated aboved 1750 psig.

(2) May be bypassed below 900 psig and is automatically reinstated above 900 psig.

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached immediately in the event of a DBA, cover the entire spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1500 psig low reactor coolant pressure setpoint for high pressure injection initiation and 500 psig for low pressure injection is to

establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation.(1)

REFERENCE

(1) FSAR, Section 15.14.

3.5.4 Incore Instrumentation

Applicability

Applies to the operability of the incore instrumentation system.

Objective

To specify the functional and operational requirements of the incore instrumentation system.

Specification

3.5.4.1 At or above 80 percent of the power allowable for the existing reactor coolant pump operating combination, incore detectors shall be operable as necessary to meet the following:

a. For axial imbalance measurements:

At least three detectors in each of at least three strings shall lie in the same axial plane, with one plane in each axial core half. The axial planes in each core half shall be symmetrical about the core mid-plane. The detector strings shall not have radial symmetry.

b. For quadrant power tilt measurements:

At least two sets of at least four detectors shall lie in each axial core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane. Detectors in the same plane shall have quarter core radial symmetry.

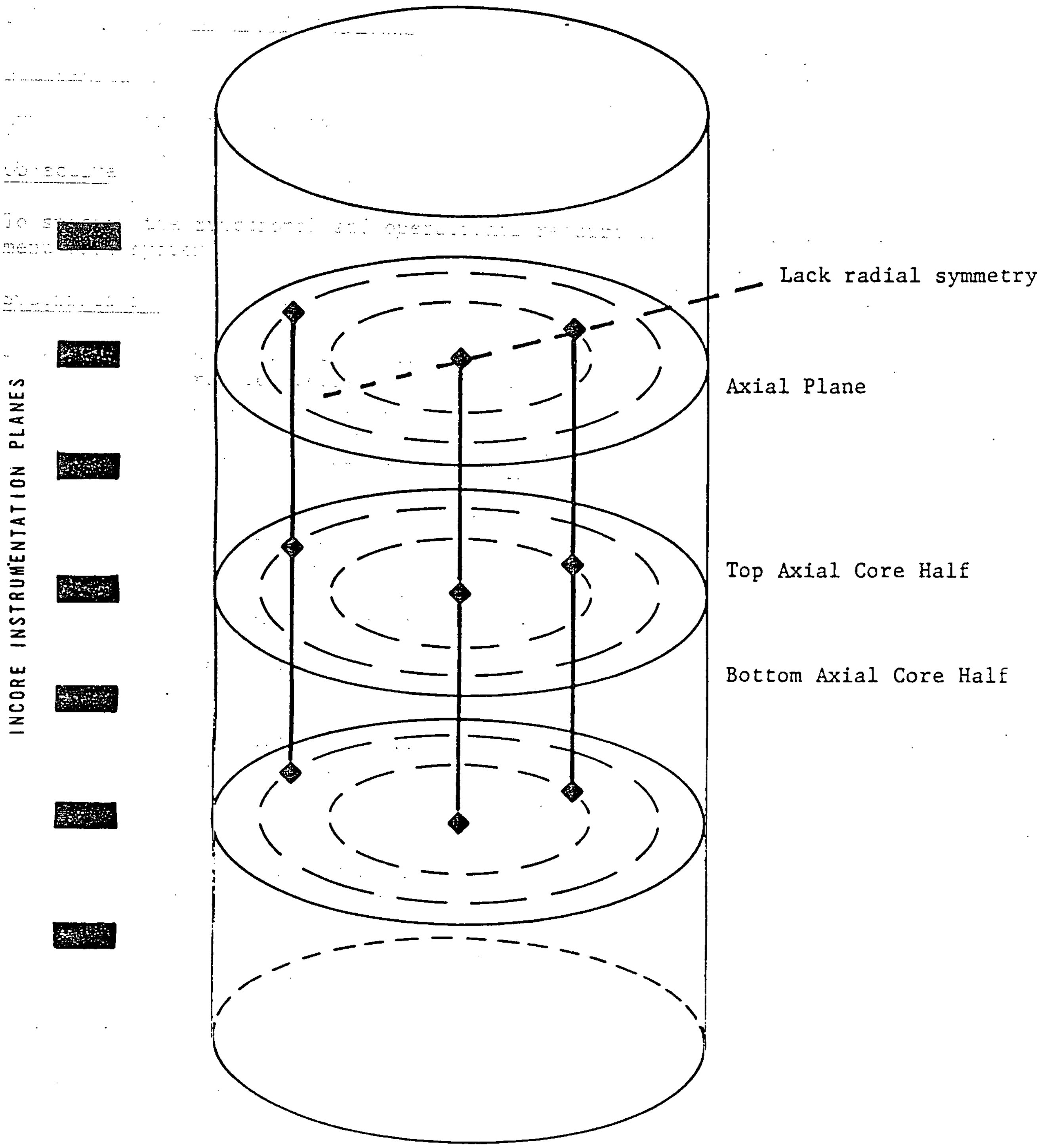
3.5.4.2 If requirements of 3.5.4.1 are not met, power shall be reduced below 80 percent of the power allowable for the existing reactor coolant pump combination within eight hours and incore detector measurements shall not be used to determine axial imbalance or quadrant power tilt.

Bases

The operability of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Figures 3.5.4-1, 3.5.4-2, and 3.5.4-3 for satisfactory incore detector arrangements.

The safety of reactor operation at or below 80 percent of the power allowable for the reactor coolant pump combination(1) without the axial imbalance trip system has been determined by extensive 3-D calculations, and was verified during the physics startup testing program.

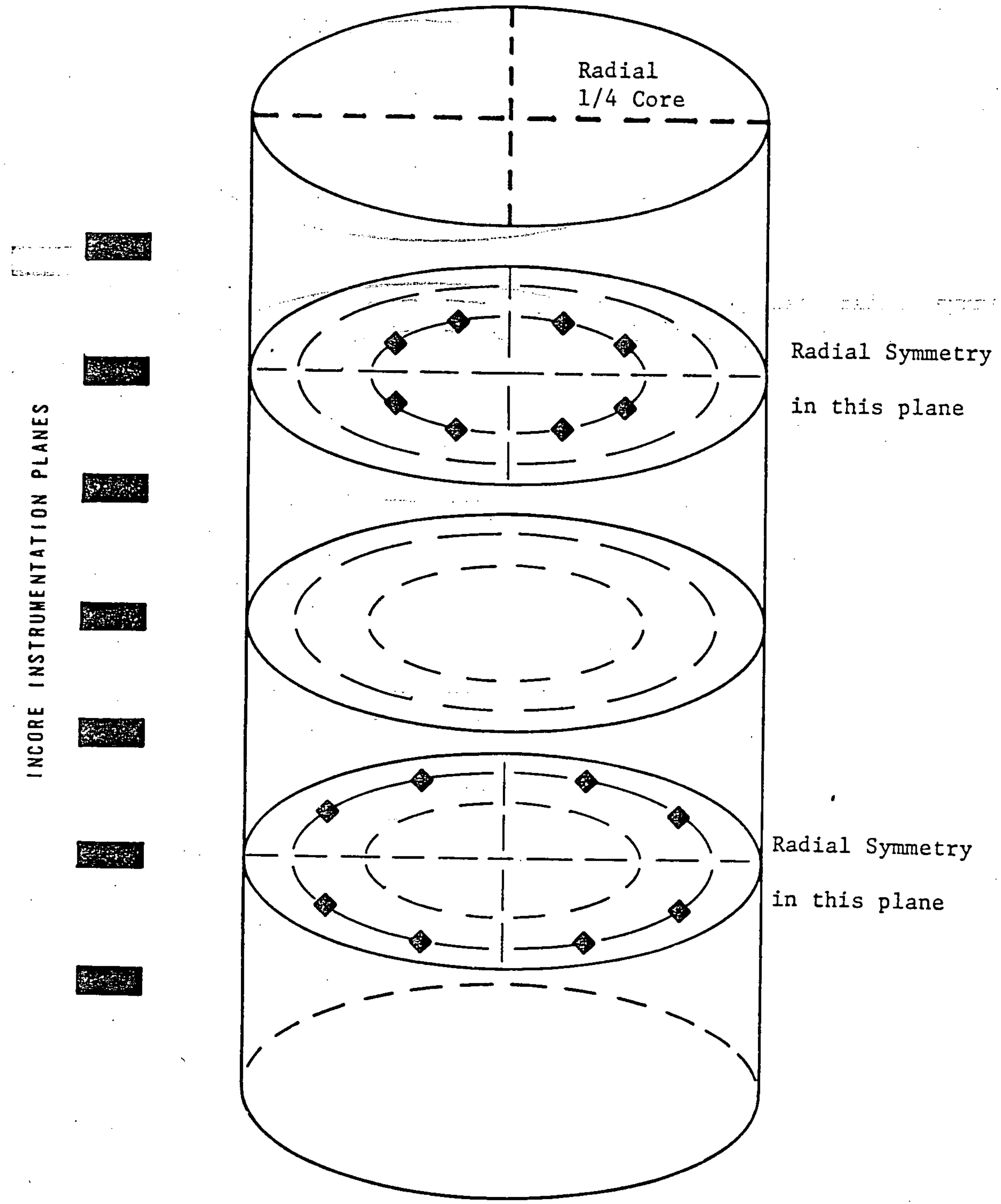
(1) FSAR, Section 5.1.2.3



INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION

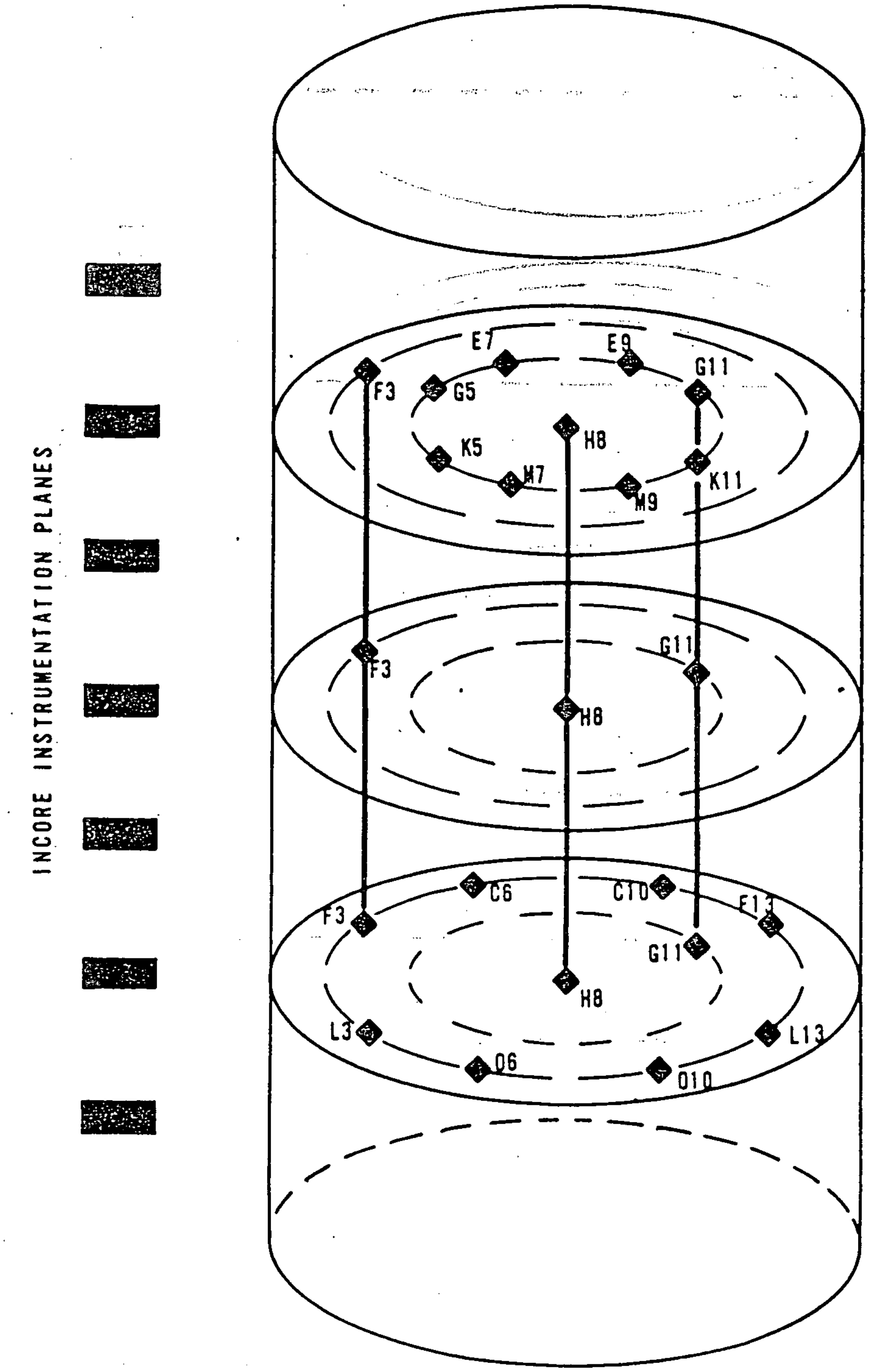
INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION

OCONEE NUCLEAR STATION
Figure 3.5.4-1



INCORE INSTRUMENTATION SPECIFICATION
RADIAL FLUX TILT INDICATION

INCORE INSTRUMENTATION
SPECIFICATION
RADIAL FLUX TILT INDICATION
OCONEE NUCLEAR STATION
Figure 3.5.4-2



INCORE INSTRUMENTATION SPECIFICATION

INCORE INSTRUMENTATION
SPECIFICATION

OCONEE NUCLEAR STATION
Figure 3.5.4-3

3.5.5 Radioactive Effluent Monitoring Instrumentation

Applicability

Applies to radioactive liquid effluent, gaseous effluent, and gaseous process monitoring instrumentation.

Specifications

3.5.5.1 Liquid Effluents

a. The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.5.5-1 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.9.1 are not exceeded.

b. If a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

c. In the event that the number of operable radioactive liquid effluent monitoring instrumentation channels falls below the limit given under Table 3.5.5-1, Column A, action shall be as shown in Column B. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.5.2 Gaseous Process and Effluents

a. The radioactive gaseous process and effluent monitoring instrumentation channels shown in Table 3.5.5-2 shall be operable with their alarm/trip setpoints set to ensure that the limits of Specification 3.10.1 are not exceeded.

b. If a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel or declare the channel inoperable, or change the setpoint so it is acceptably conservative.

c. In the event that the number of radioactive gaseous process or effluent monitoring instrumentation channels falls below the limit given under Table 3.5.5-2, Column A, action shall be taken as shown in Column B. Exert best efforts to return the instruments to operable status within 30 days and, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

3.5.5.3 Setpoints

The setpoints shall be determined in accordance with the methodology described in the ODCM and shall be recorded. Setpoint correction may be permitted without declaring the channel inoperable.

3.5.5.4 The provisions of Technical Specification 3.0 do not apply.

Bases

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to assure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance with NRC approved methods in the ODCM to assure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentration of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

Table 3.5.5-1
LIQUID EFFLUENT MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
1. Monitors Providing Automatic Termination of Release			
Liquid Radwaste Effluent Line Monitors			
1 RIA-33	1	*	(a)
Turbine Building Sump			
1 RIA-54 (Units 1 & 2)	1	*	(b)
3 RIA-54 (Unit 3)	1	*	(b)
2. Monitors not Providing Automatic Termination of Release			
Low Pressure Service Water			
1 RIA-35	1	*	(d)
2 RIA-35	1	*	(d)
3 RIA-35	1	*	(d)
3. Flow Rate Measuring Devices			
Liquid Radwaste Effluent Line	1	*	(c)
Keowee Hydroelectric Station Tailrace Discharge **	NA	NA	NA
4. Continuous Composite Sampler			
#3 Chemical Treatment Pond Composite Sampler and Sampler Flow Monitor (Turbine Building Sumps Effluent)	1	*	(d)

*At all times.
**Flow determined from number of hydro units operating; if hydro is not operating, leakage flow, which is measured periodically, is used.

Table 3.5.5-1 NOTES

(a) Effluent releases may continue provided that prior to initiating a release:

1. Two independent samples are analyzed in accordance with Specification 3.9 and;

2. Two independent data entry checks for release rate calculations and valve lineups of the effluent pathway are conducted.

Otherwise, suspend release of radioactive effluents by this pathway.

(b) Effluent releases may continue provided that prior to each discrete release of the sump, grab samples are collected and analyzed for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml.

(c) Effluent releases may continue provided flow rate is estimated at least once per four hours during actual releases.

(d) Effluent releases may continue provided that grab samples are collected and analyzed for gross radioactivity (beta and/or gamma) at a lower limit of detection of at least 10^{-7} μCi/ml every 12 hours.

Table 3.5.5-2
GASEOUS PROCESS AND EFFLUENT
MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
1. Waste Gas Holdup Tanks			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination Of Release (RIA-37, - 38)	1	**	(a)
b. Effluent Flow Rate Monitor (Waste Gas Discharge Flow)	1	**	(b)
2. Unit Vent Monitoring System			
a. Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Containment Purge Release (RIA - 45)	1	*	(a)
b. Iodine Sampler	1	*	(d)
c. Particulate Sampler	1	*	(d)
d. Effluent Flow Rate Monitor (Unit Vent Flow)	1	*	(b)
e. Sampler Flow Rate Monitor	1	*	(e)
f. Effluent Flow Rate Monitor (Containment Purge)	1	**	(b)
3. Interim Radwaste Building Ventilation Monitoring System			
a. Noble Gas Activity Monitor (RIA - 53)	1	*	(c)
b. Iodine Sampler#	1	*	(d)
c. Particulate Sampler#	1	*	(d)

Table 3.5.5-2 (Cont'd)
GASEOUS PROCESS AND EFFLUENT
MONITORING INSTRUMENTATION
OPERATING CONDITIONS

INSTRUMENT	A MINIMUM OPERABLE CHANNELS (PER RELEASE PATH)	APPLICABILITY	B OPERATOR ACTION IF MINIMUM NUMBER OF OPERABLE CHANNELS IS NOT MET
d. Effluent Flow Rate Monitor (Interim Radwaste Exhaust)#	1	*	(b)
e. Sampler Flow Rate Monitor#	1	*	(e)
4. Hot Machine Shop Ventilation Monitoring System			
a. Iodine Sampler#	1	*	(d)
b. Particulate Sampler#	1	*	(d)
c. Effluent Flow Rate Monitor (Hot Machine Shop Exhaust)#	1	*	(b)
d. Sampler Flow Rate Monitor#	1	*	(e)

* At all times.
** During waste gas holdup tank releases and/or containment purge operation.
\# Effective upon installation of equipment.

Table 3.5.5-2 NOTES

(a) Effluent releases from waste gas tanks or containment purges may continue provided that prior to initiating a release:

1. Two independent samples are analyzed and;

2. Two independent data entry checks for release rate calculations and valve lineups of the effluent pathway are conducted and;

Effluent release from ventilation system or condenser air ejectors may continue provided that grab samples are taken once per 8 hours and these samples are analayzed for gross activity (beta and/or gamma) within 24 hours, or continuously monitor through the unit vent. Otherwise, suspend release of radioactive effluents via this pathway.

(b) Effluent releases may continue provided the flow rate is estimated at least once per 4 hours.

(c) Effluent releases may continue provided grab samples are taken once per 8 hours and these samples are analayzed for gross activity (beta and/or gamma) within 24 hours.

(d) Effluent releases may continue provided samples are continuously collected with auxiliary sampling equipment for periods not to exceed 7 days and analyzed within 48 hours of the end of sample collection.

(e) Alarms indicating low flow may be substituted for flow measuring devices.

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Attachment 2

No Significant Hazards Consideration Evaluation

No Significant Hazards Consideration Evaluation
for Oconee 3 Cycle 8 Reload

Duke Power has made the determination that this amendment request involves no significant hazards under the Commission's regulations in 10 CFR 50.92. 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.

Guidance has been supplied by the Commission concerning the application of these standards by providing certain examples (48 FR 14870). Example (iii) of the types of amendments not likely to involve significant hazards considerations applies in this case as the reload is for a nuclear power reactor. This assumes:

(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; and

(2) no significant changes are made to the acceptance criteria for the Technical Specifications; and

(3) that the analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed; and

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

This reload does not involve fuel assemblies significantly different from those found previously acceptable to the NRC. In this reload all of the 177 fuel assemblies to be inserted into the core are similar to fuel assemblies previously used and found acceptable by the NRC.

There are no significant changes to the acceptance criteria for the Technical Specifications. The Technical Specification revisions required for Cycle 8 operation were made in accordance with methods and procedures found acceptable through previously submitted reloads. The analytical methods used are consistent with the approved methodologies of the Oconee Nuclear Station Reload Design Methodology Technical Report (NFS-1001, Rev. 4., Duke Power Company, April 1979).

The Oconee 3 Cycle 8 reload report (Attachment 3) justifies the operation of the eighth cycle at the rated core power of 2568 MWt. Included in the report 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 that were previously submitted and accepted by the USNRC and its predecessor. They are listed in the Reload Report references.

The following evaluation demonstrates by reference to previously performed analysis, that when measured against the three significant safety hazards consideration standards in 10 CFR 50.92, the circumstances of this reload amendment would not involve nor create the conditions described.

First Standard

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 7 parameters to determine the effect of the Cycle 8 reload and to ensure that thermal performance during hypothetical transients is not degraded. The transient evaluation of Cycle 8 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

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 analysis found 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

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
3. Thermal-Hydraulic Design considerations

Section 4, 5, and 6 of the Oconee 3, Cycle 8 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. The evaulations are summarized below:

Fuel System Design Consideration

The Fuel System Design consideration is described in Section 4 of Attachment 3. The most limited fuel assemblies, from the collapse of the clad standpoint are the batch 8B assemblies because of their longer incore exposure time. The most limiting fuel assembly was compared to the generic MK-B creep collapse analysis and was found to be enveloped. The analysis performed is based on the methods and procedures described in the B&W report, "Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse" (BAW-10084A, Rev. 2). The generic analysis predicts a collapse time of more than 31,400 EFPH which exceeds the maximum projected residence time of 28,560 EFPH.

The stress parameters for the Oconee 3 fuel rods were conservatively analyzed in accordance with the guidelines set forth in Section III, Division 1 - subsection NB of the ASME Boiler pressure vessel code. Compliance with ASME code criteria verify the structural integrity of the cladding throughout the most limiting design conditions.

The fuel design criterion specified for the cladding uniform circumferential strain is not to exceed 1.0% strain, for transient conditions.

The batch 10 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 10 fuel assemblies is acceptable.

All fuel in the cycle 8 core is thermally similar. The incoming fresh batch 10 fuel inserted for cycle 8 operation introduces no significant differences in fuel thermal performance relative to the fuel remaining in the core. The burnup dependent linear heat rate (LHR) capability and the average fuel temperature for each batch are shown in Table 4-2 of Attachment 3. The maximum assembly average burnup is predicted to be 38358 MWd/MTu and the maximum fuel rod burnup is predicted to be 40839 MWd/MTu. The maximum fuel rod internal pressure has been conservatively evaluated to be less than the nominal reactor coolant system pressure of 2200 psia.

Thermal-Hydraulic Design Consideration

The incoming batch 10 fuel is hydraulically and geometrically similar to the fuel remaining in the core from previous cycles. Thermal-hydraulic design evaluation supporting cycle 8 operation used the methods and models described in section 6 of Attachment 3, which have been found to be acceptable by the NRC through previously submitted reloads. For cycle 8 operation, the flux to flow setpoint is maintained at 1.08. The minimum DNBR value has been determined to be greater than the BAW-2 CHF correlation limit of 1.30. All other plant operating limits based on DNBR criteria include a minimum of 10.2% DNBR margin from the design limit of 1.30.

Nuclear Design Consideration

The Nuclear Design Analysis was performed using similar methods and procedures that were employed for previous reload reports that have been submitted and accepted by the NRC. Section 5 of Attachment 3 describes the core physics parameters of cycle 8. The required shutdown margin is 1.00% $\Delta k/K$, the shutdown margin at the beginning of the cycle and at the end of the cycle is 4.14% $\Delta k/K$ and 2.73% $\Delta k/K$ respectively. The calculated ejected rod worths and their adherence to criteria are considered at all times in life and at all power levels in the development of the Rod Position Limits. All safety criteria associated with these rod worths are met. Thus, the analysis shows that the cycle 8 design meets all criteria including those applicable to radial power peaking, ejected rod worths and shutdown margin.

One can conclude from the examination of these sections, that the cycle 8 core thermal and kinetics properties, (with respect to previous cycle values), that this core reload will not significantly reduce the ability of the Oconee 3 unit to operate safely during cycle 8.

In summary, Duke has determined and submits that the proposed reload described herein does not involve a significant safety hazard.