MAY 15 1984

DMB-016

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

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Mr. Hal B. Tucker
Vice President - Steam Production
Duke Power Company
Post Office Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos.129 , 129 , and 126 to Facility Operating 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 February 13, 1984.

These amendments revise the TSs to allow full power operation of Oconee Unit 3 during fuel Cycle 8. We have also revised the administrative numbering of the figures in TS 3.5.2.

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

Sincerely,

"ORIGINAL SIGNED BY:"

Helen Nicolaras, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No.129 to DPR-38
2. Amendment No.129 to DPR-47
3. Amendment No.126 to DPR-55
4. Safety Evaluation

cc: w/enclosures
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	OELD	AD/OR:DL
RIngram	HNicolaras;cf	JStolz		GLainas
05/10/84	05/10/84	05/11/84	05/11/84	05/14/84

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. 129
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 February 13, 1984, 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 <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 129 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

George W Knighton, for

John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 15, 1984

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 129
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 February 13, 1984, 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.129 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, Chief
Operating Reactors Branch No. 4
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: May 15, 1984

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.126
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 February 13, 1984, 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.126 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

George W Rivenbark

John F. Stolz, Chief
Operating Reactors Branch No. 4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 15, 1984

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 129 TO DPR-38

AMENDMENT NO. 129 TO DPR-47

AMENDMENT NO. 126 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
iii	iii
vi	vi
viii	viii
ix	ix
x	x
2.1-3d	2.1-3d
2.1-9	2.1-9
2.3-2	2.3-2
2.3-10	2.3-10
3.5-9	3.5-9
3.5-10	3.5-10
3.5-11	3.5-11
3.5-15	3.5-15 (3 pages)
3.5-15a	--
3.5-15b	--
3.5-16	3.6-16 (3 pages)
3.5-16a	--
3.5-16b	--
3.5-17	3.5-17 (3 pages)
3.5-17a	--
3.5-17b	--
3.5-18	3.5-18 (3 pages)
3.5-18a	--
3.5-18b	--
3.5-18c	--
3.5-18d	--
3.5-18e	--
3.5-19	3.5-19 (3 pages)
3.5-19a	--
3.5-19b	--
3.5-19c	--
3.5-19d	--
3.5-19e	--

Remove Pages	Insert Pages
3.5-20	3.5-20 (3 pages)
3.5-20a	--
3.5-20b	--
3.5-20c	--
3.5-20d	--
3.5-20e	--
3.5-21	3.5-21 (3 pages)
3.5-21a	--
3.5-21b	--
3.5-22	3.5-22 (3 pages)
3.5-22a	--
3.5-23	3.5-23 (3 pages)
3.5-23a	--
3.5-23b	--
3.5-24	3.5-24 (3 pages)
3.5-24a	--
3.5-24b	--
3.5-25	3.5-25 (2 pages)
3.5-26	3.5-26 (2 pages)
3.5-26a	--
3.5-26b	--
3.5-27	3.5-27 (3 pages)
3.5-28	3.5-28
3.5-29	3.5-29 (2 pages)
3.5-30	3.5-30
3.5-31	3.5-31
3.5-32	3.5-32
3.5-33	3.5-33
3.5-34	3.5-34
3.5-35	3.5-35
3.5-36	3.5-36
3.5-37	3.5-37
3.5-38	3.5-38
3.5-39	3.5-39
3.5-40	3.5-40
--	3.5-41
--	3.5-42
--	3.5-43

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

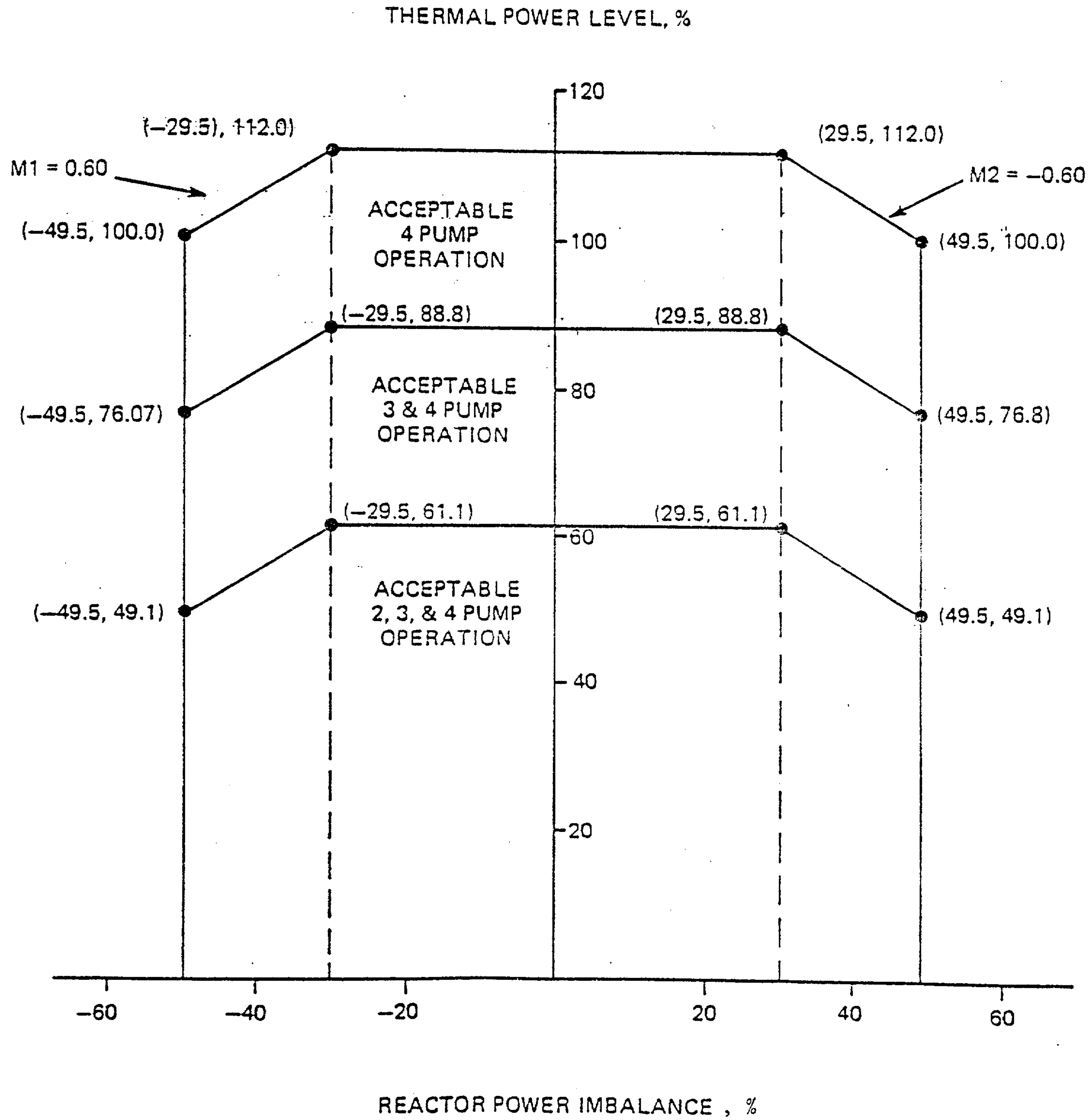
(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000, March 1970.

(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:

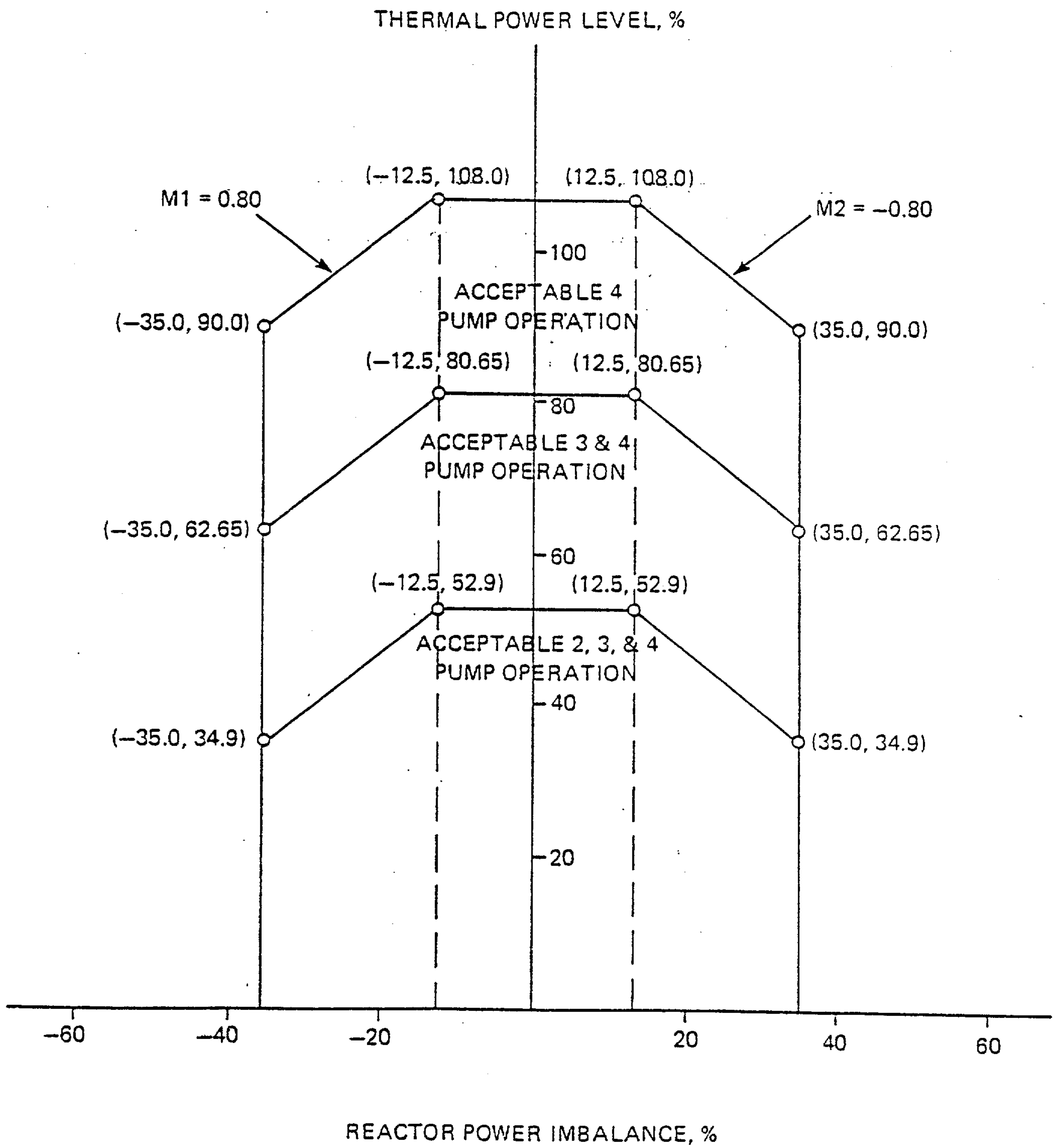
1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor flow rate is 100%, or flow rate is 93.46% and power level is 100%.

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 <u>not</u> 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.

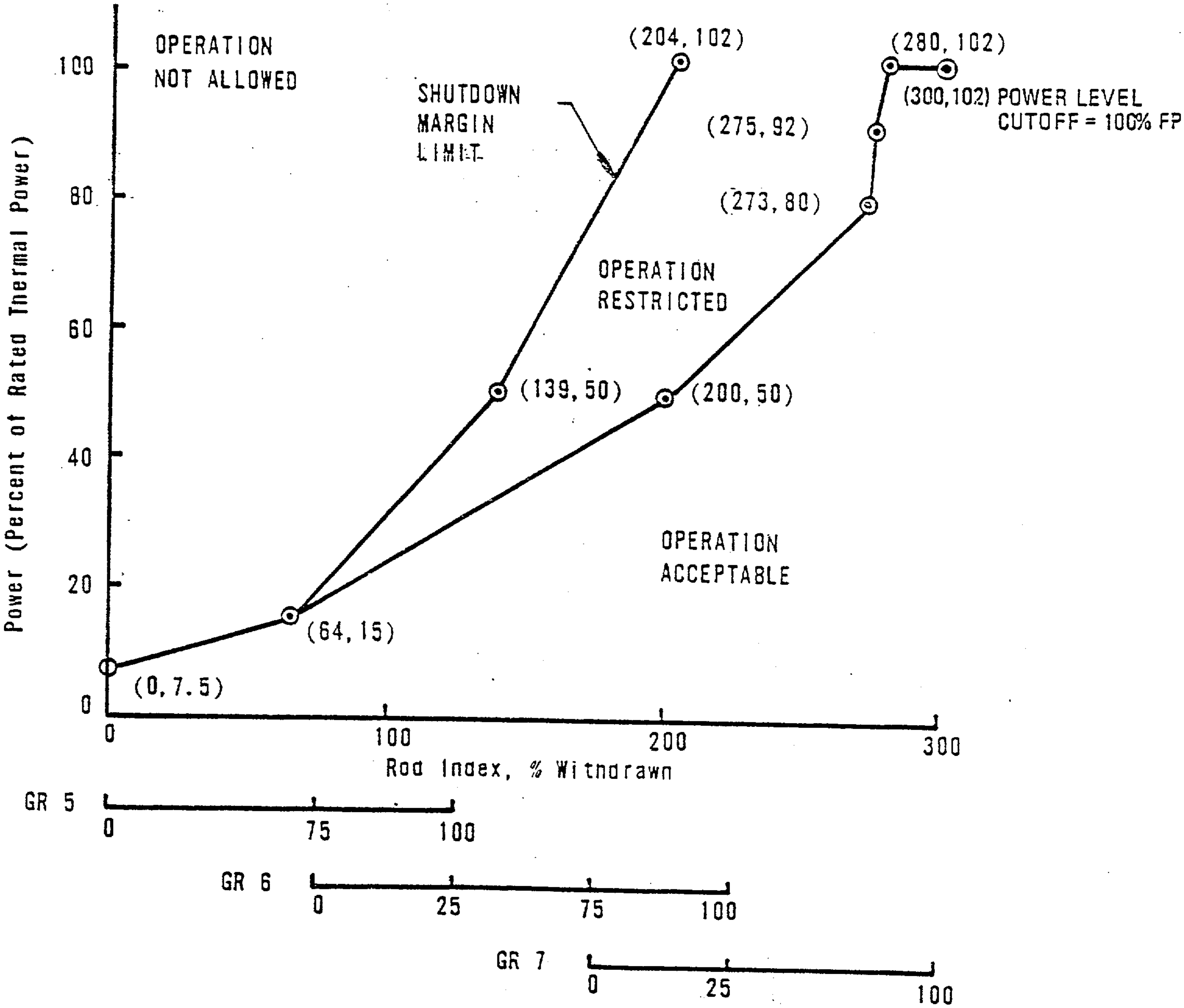
The power-imbalance envelope defined in Figures 3.5.2-10 (Unit 1)
3.5.2-11 (Unit 2)
3.5.2-12 (Unit 3)
is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-16) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures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

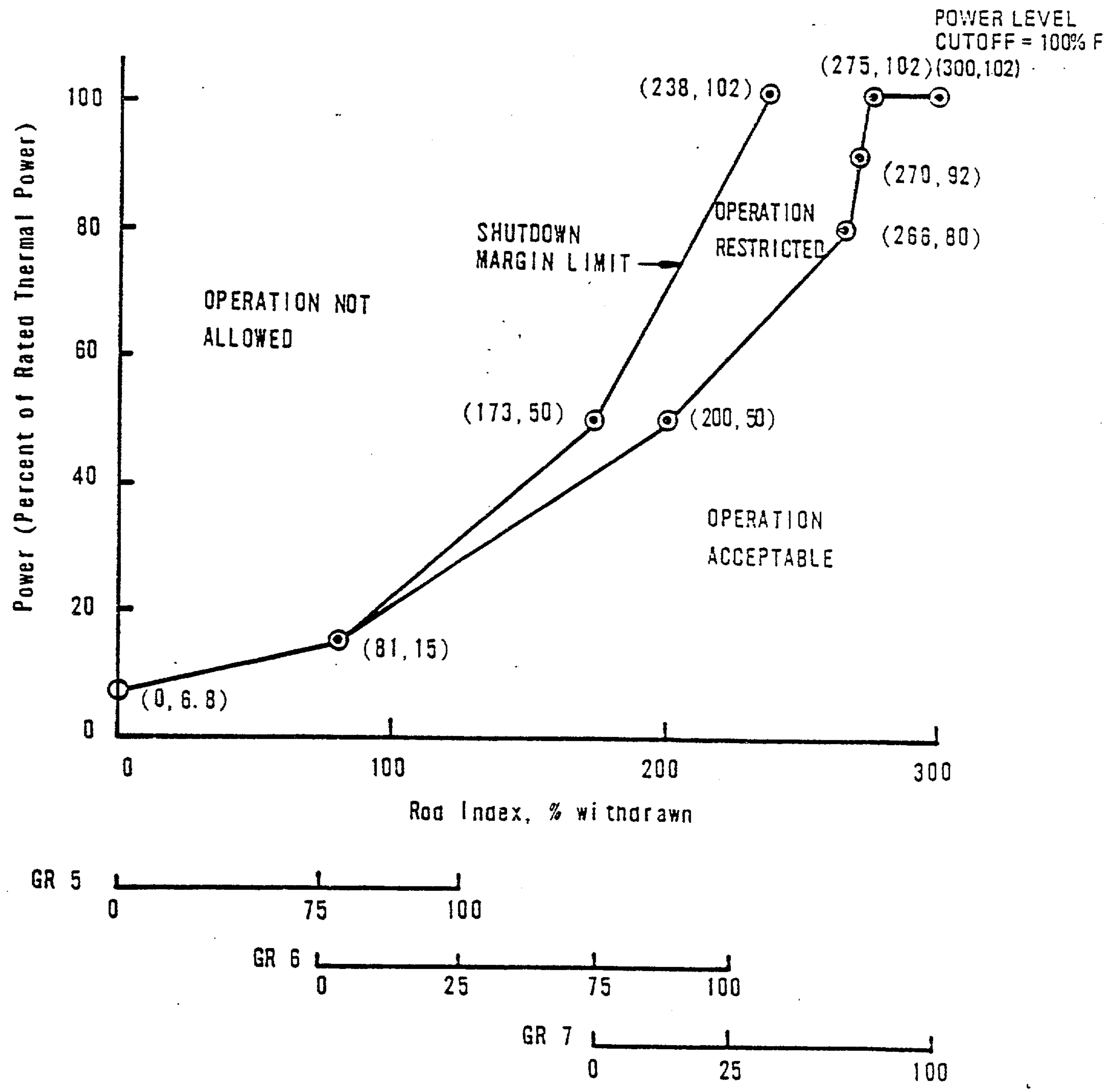
The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

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.



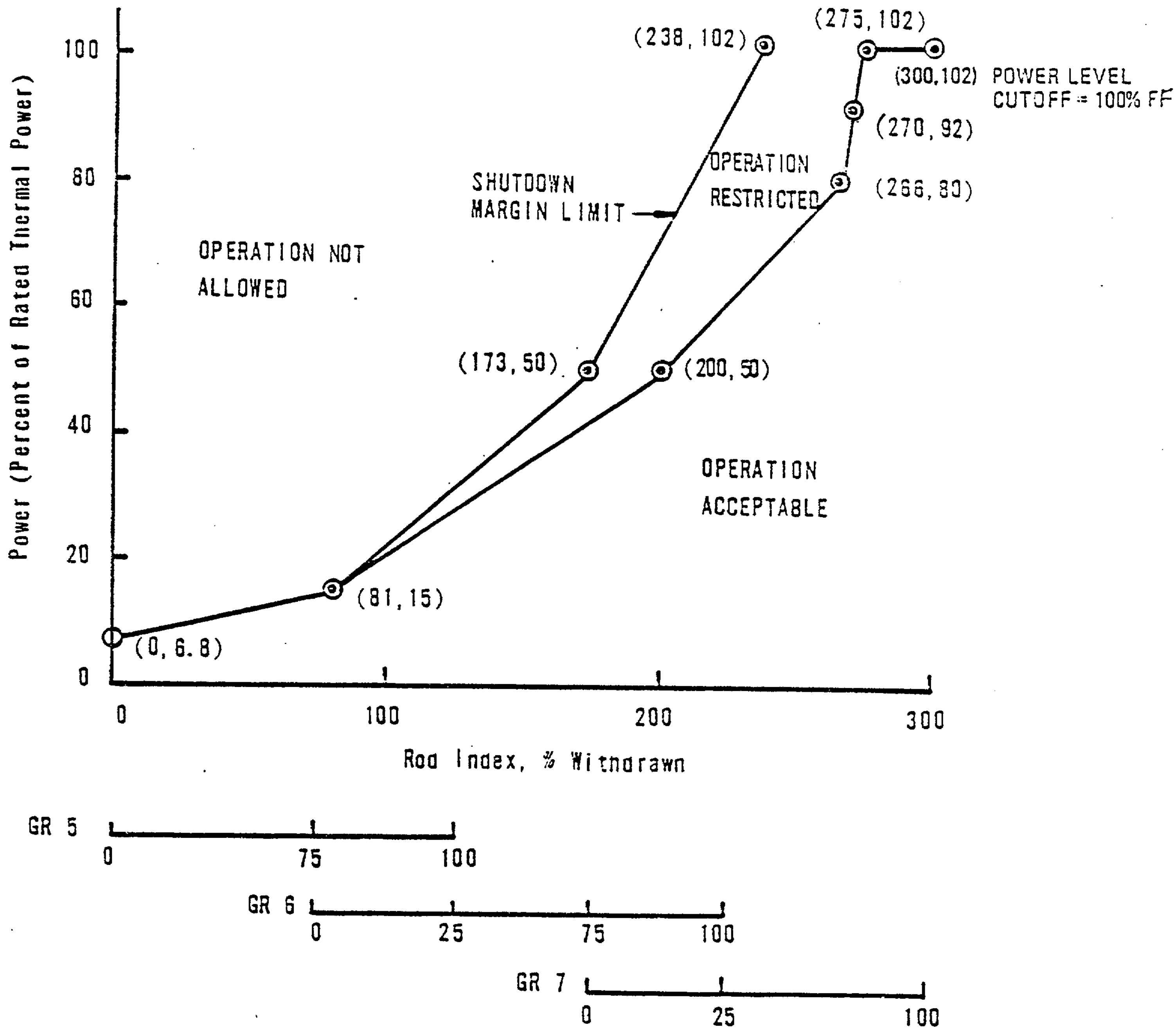
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 26 +10/-0 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-1
(1 of 3)



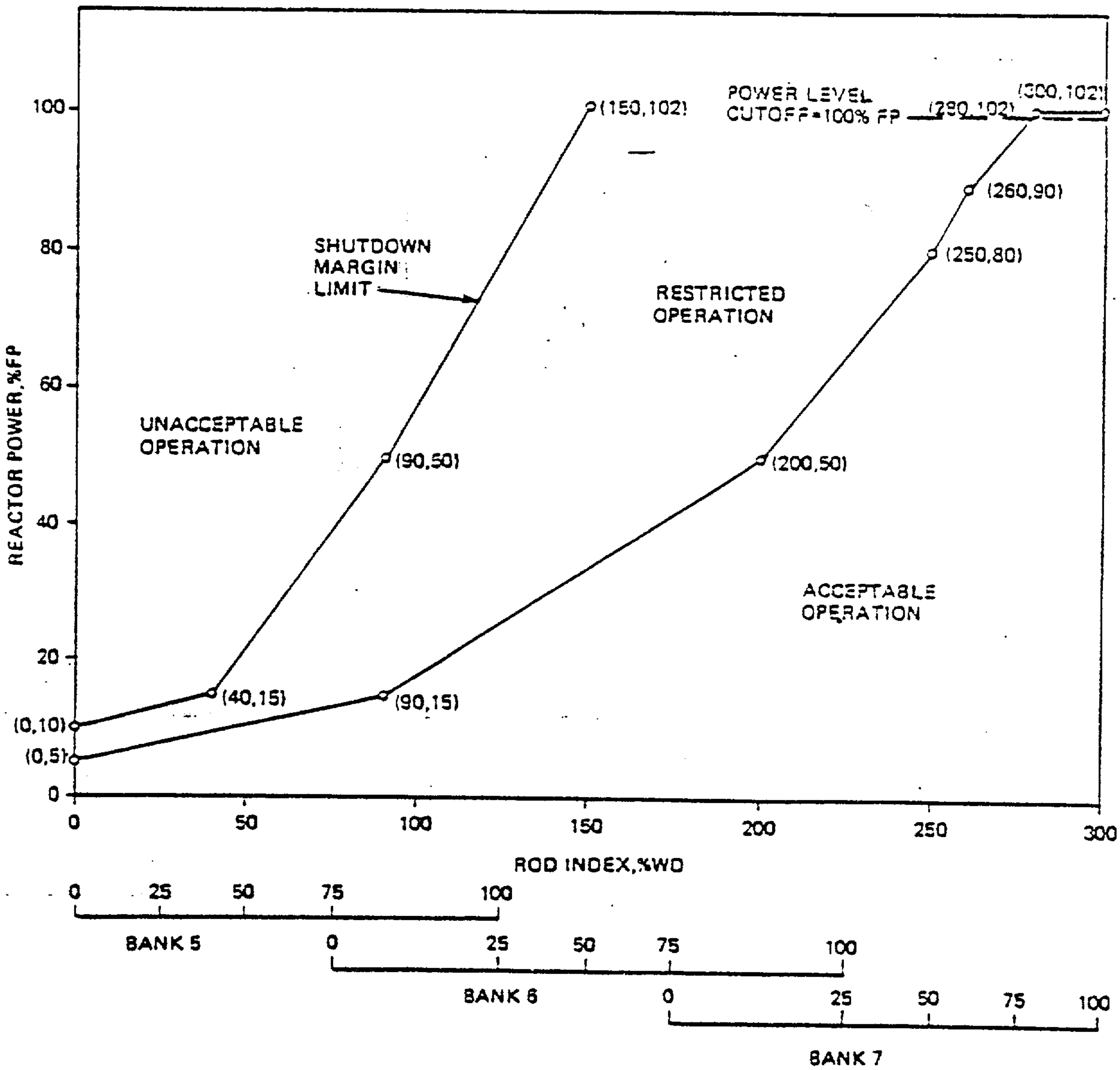
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-1
(2 of 3)

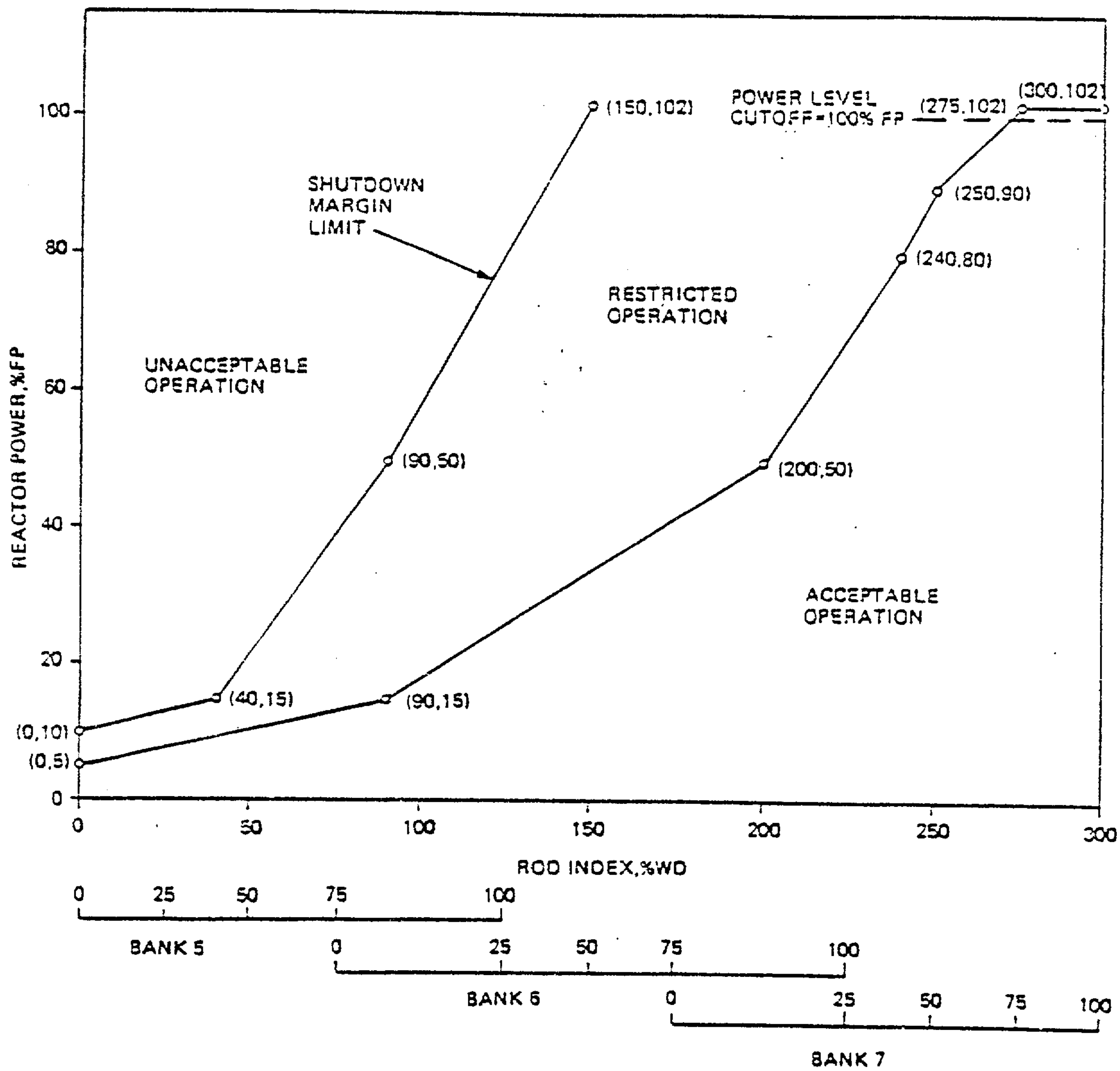


ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
AFTER 200 ± 10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-1
(3 of 3)



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

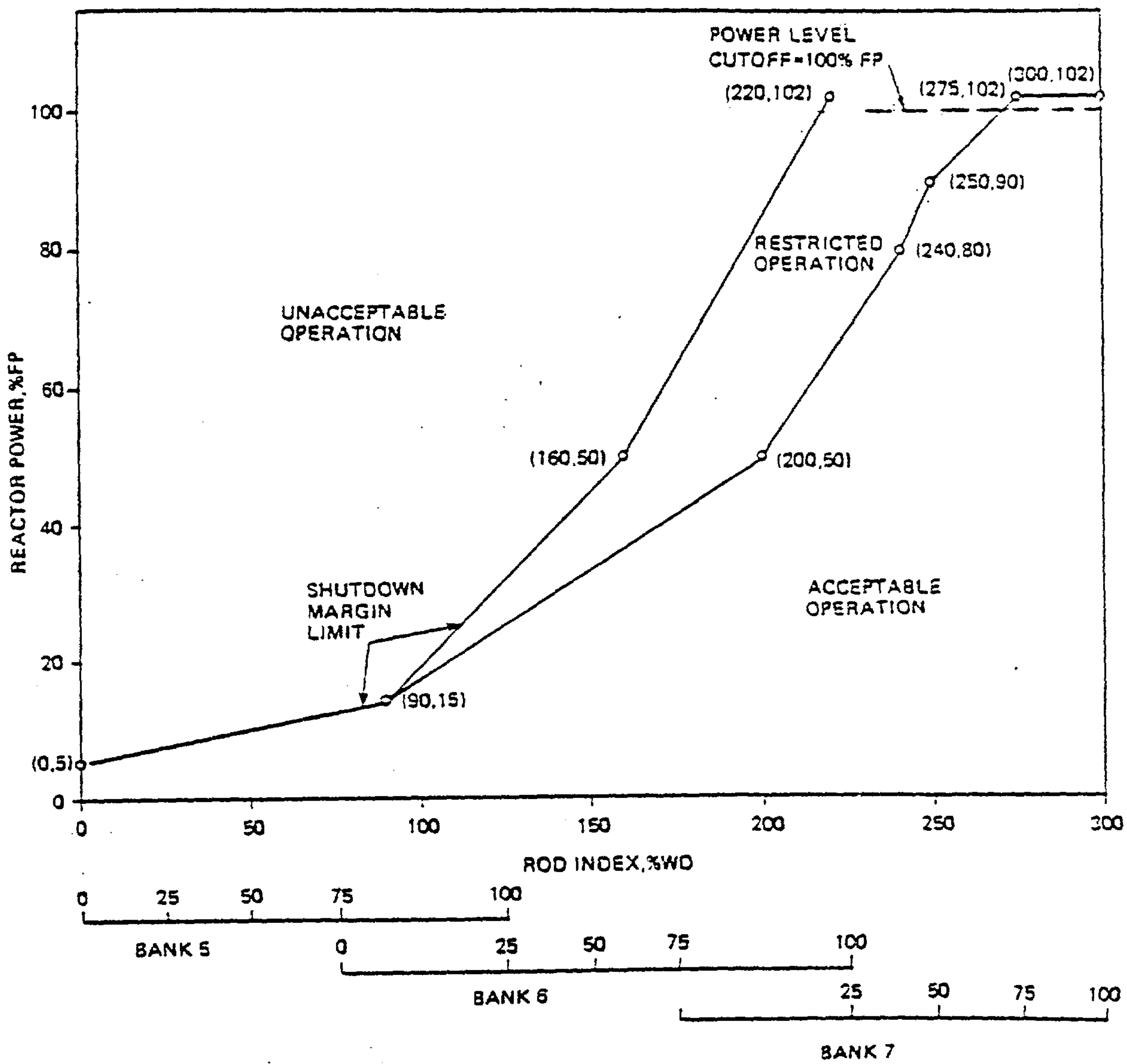
OCONEE NUCLEAR STATION
Figure 3.5.2-2
(1 of 3)



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

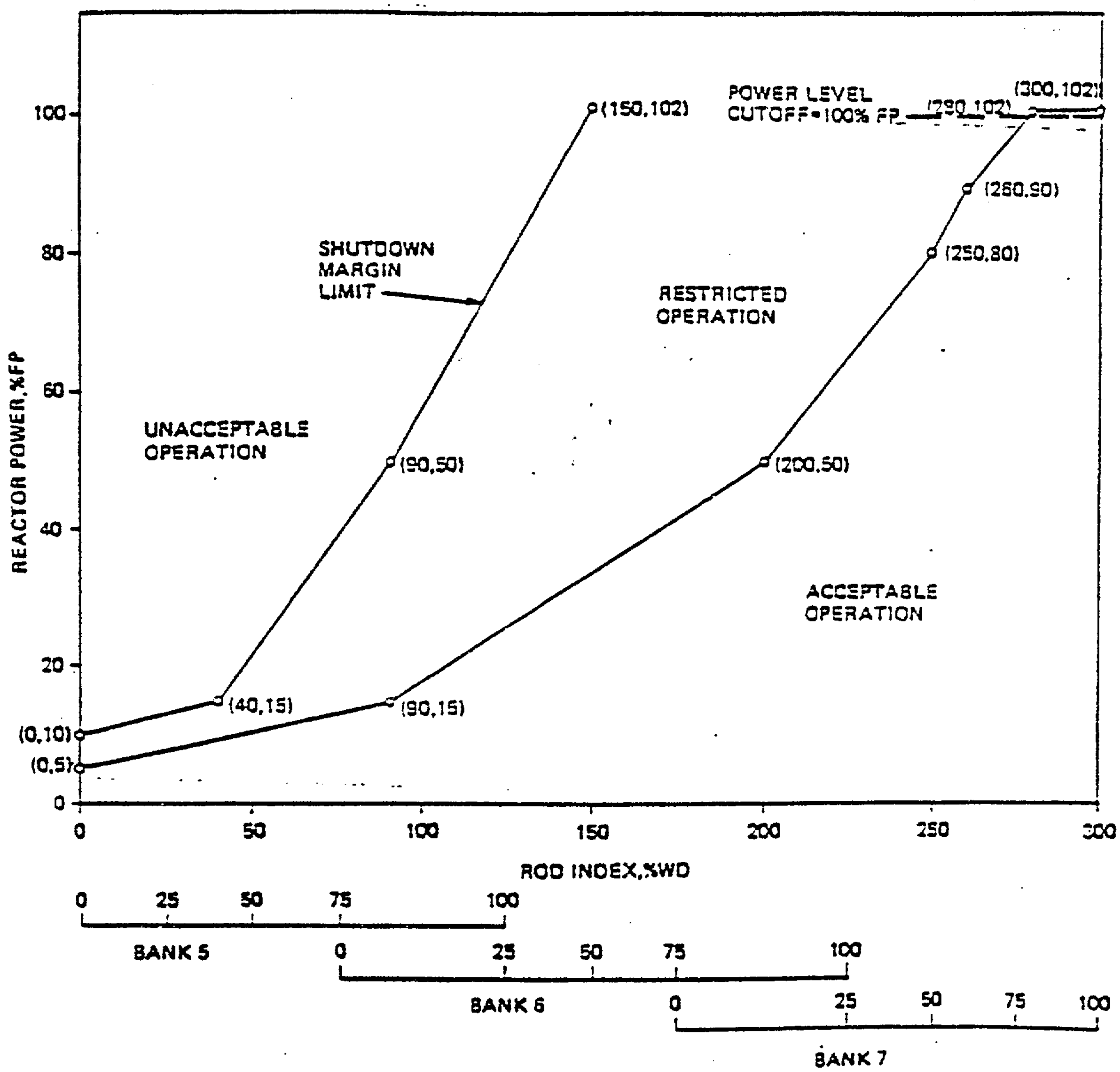
OCONEE NUCLEAR STATION

Figure 3.5.2-2
(2 of 3)

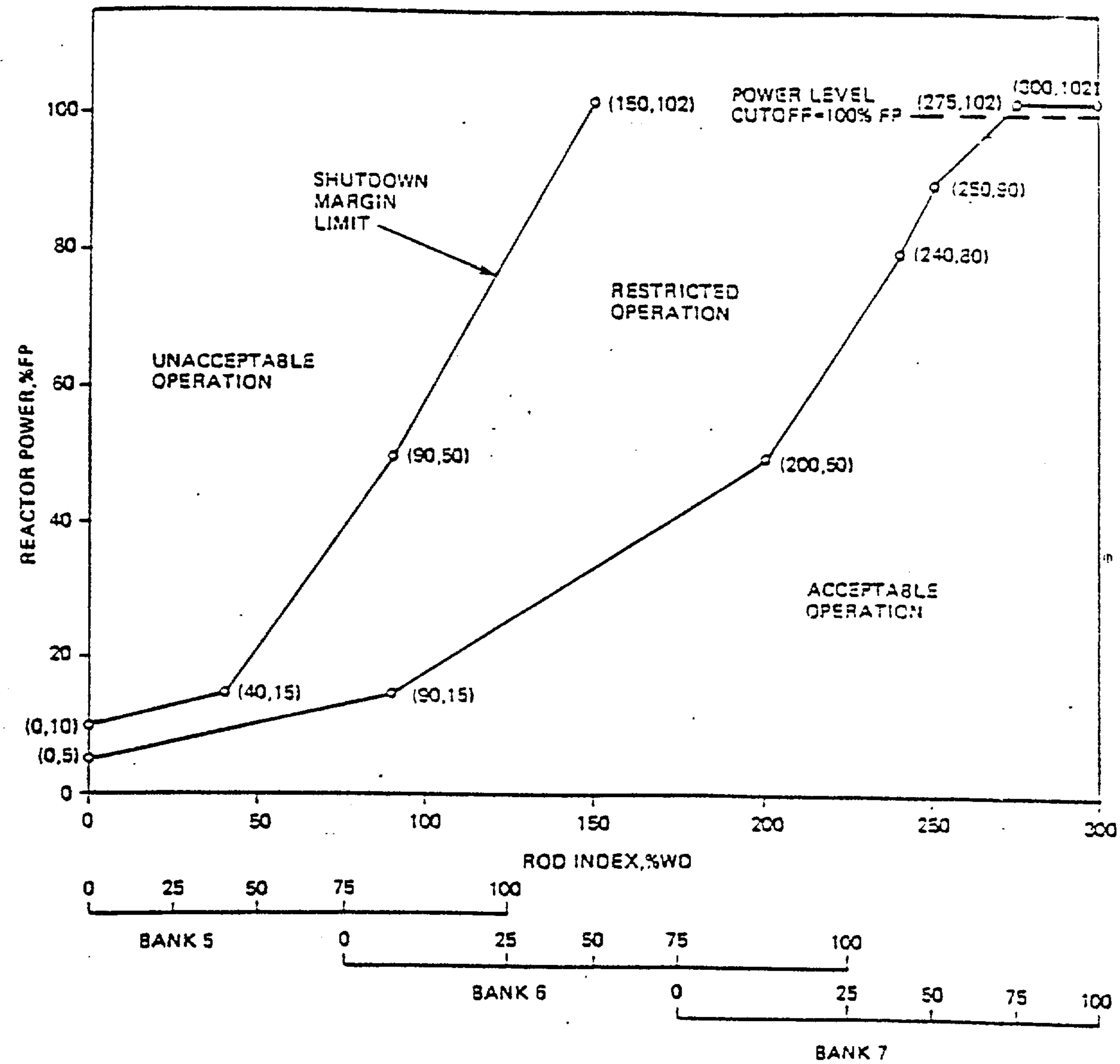


ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 200 ± 10 EFPD
UNIT 2
OCONEE NUCLEAR STATION

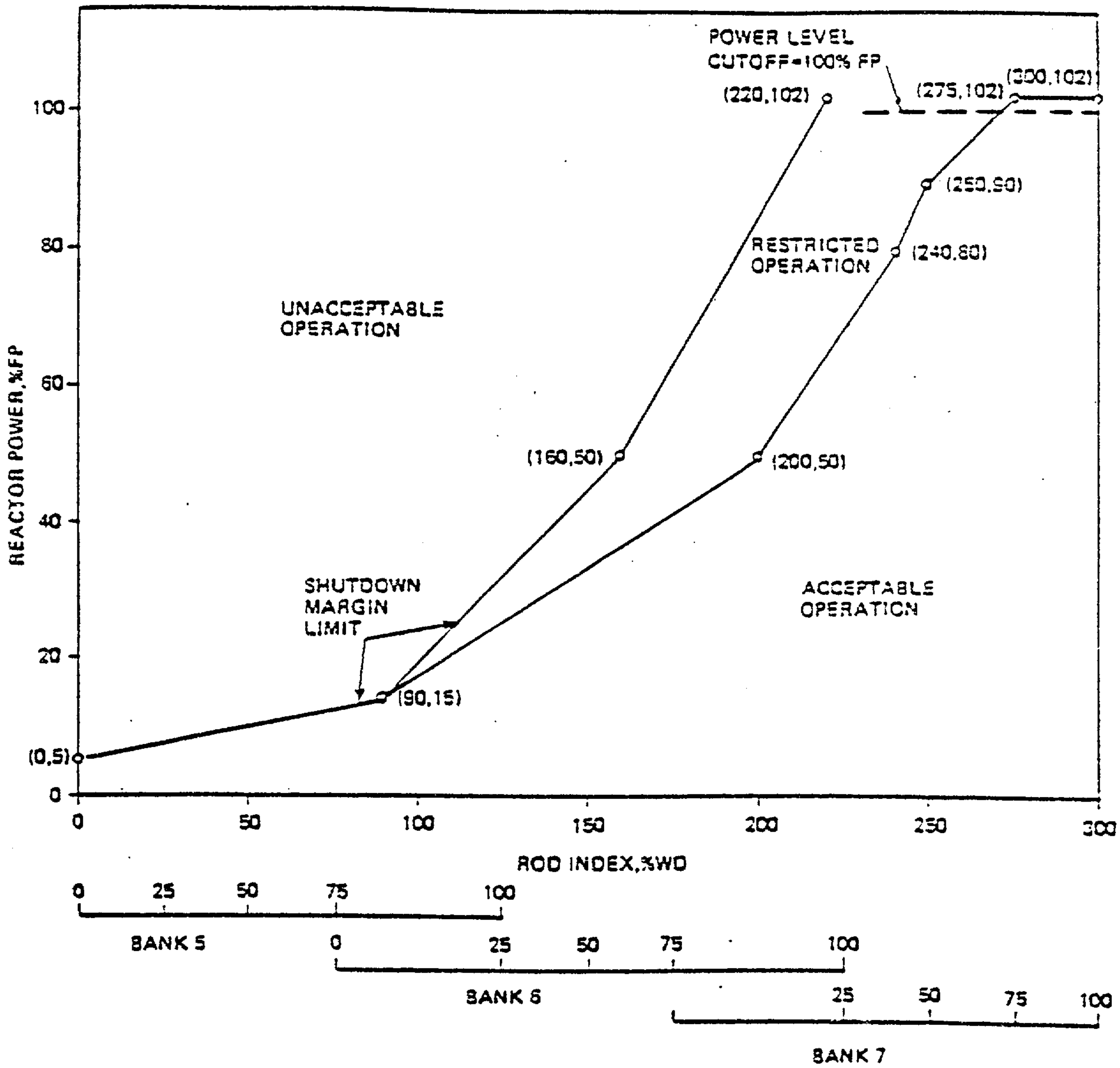
Figure 3.5.2-2
(3 of 3)



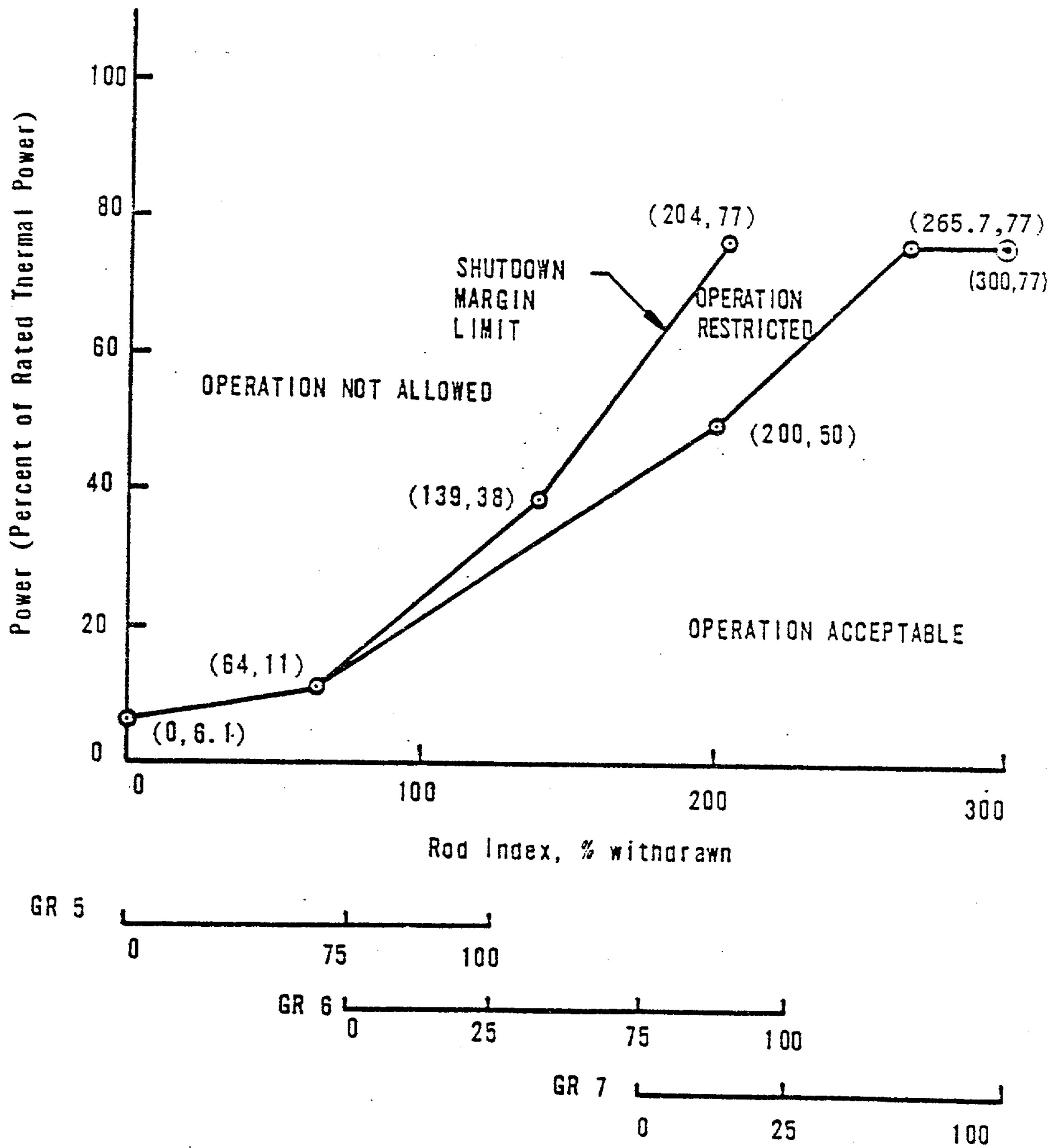
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 0 TO 25 +10/-0 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3
(1 of 3)



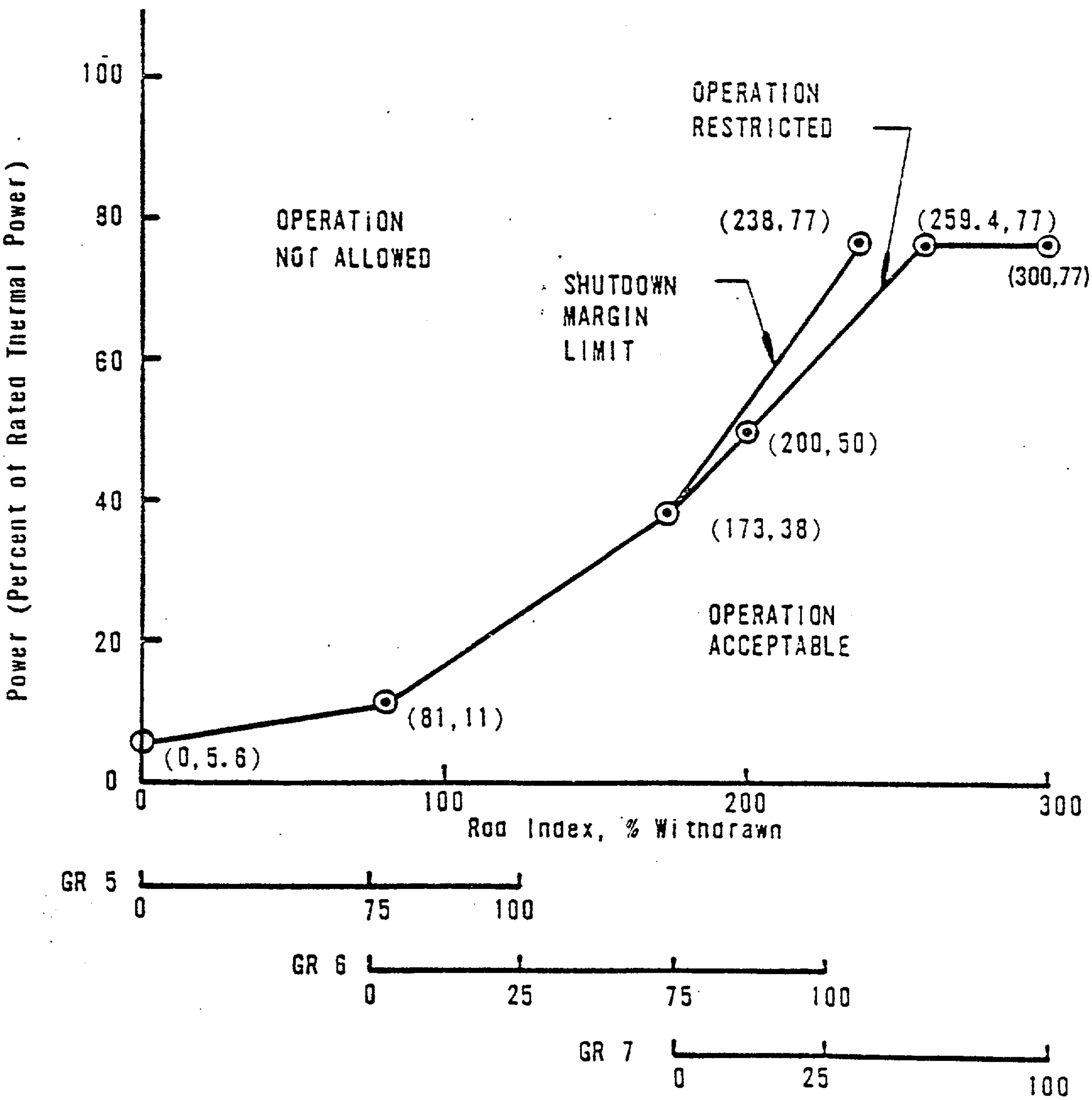
ROD POSITION LIMITS
FOR FOUR-PUMP OPERATION
FROM 25 +10/-0 TO 200 +10/-10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-3
(2 of 3)



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)



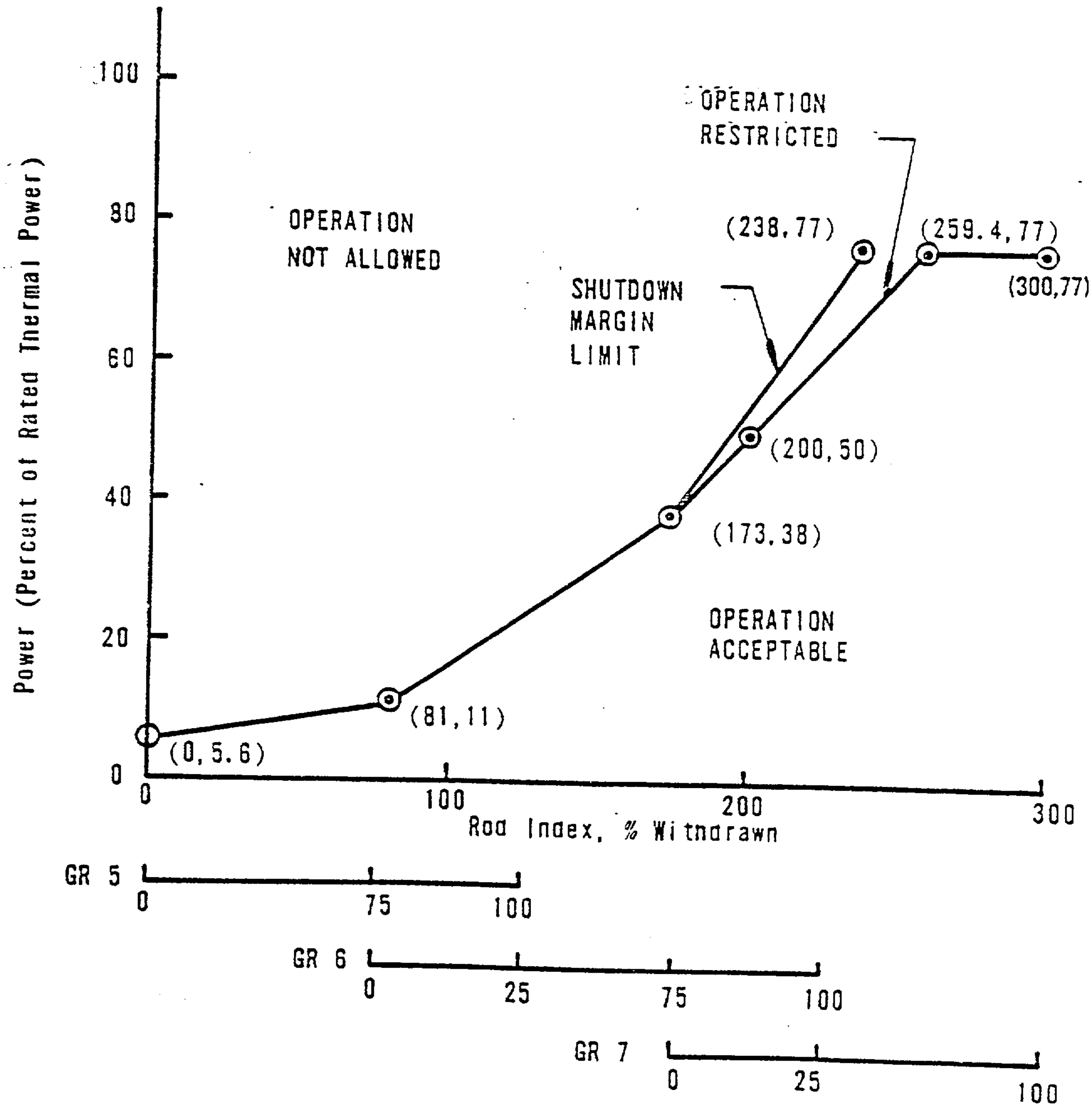
ROD POSITION LIMITS
FOR THREE-PUMP OPERATION
FROM 0 TO 26 +10/-0 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-4
(1 of 3)



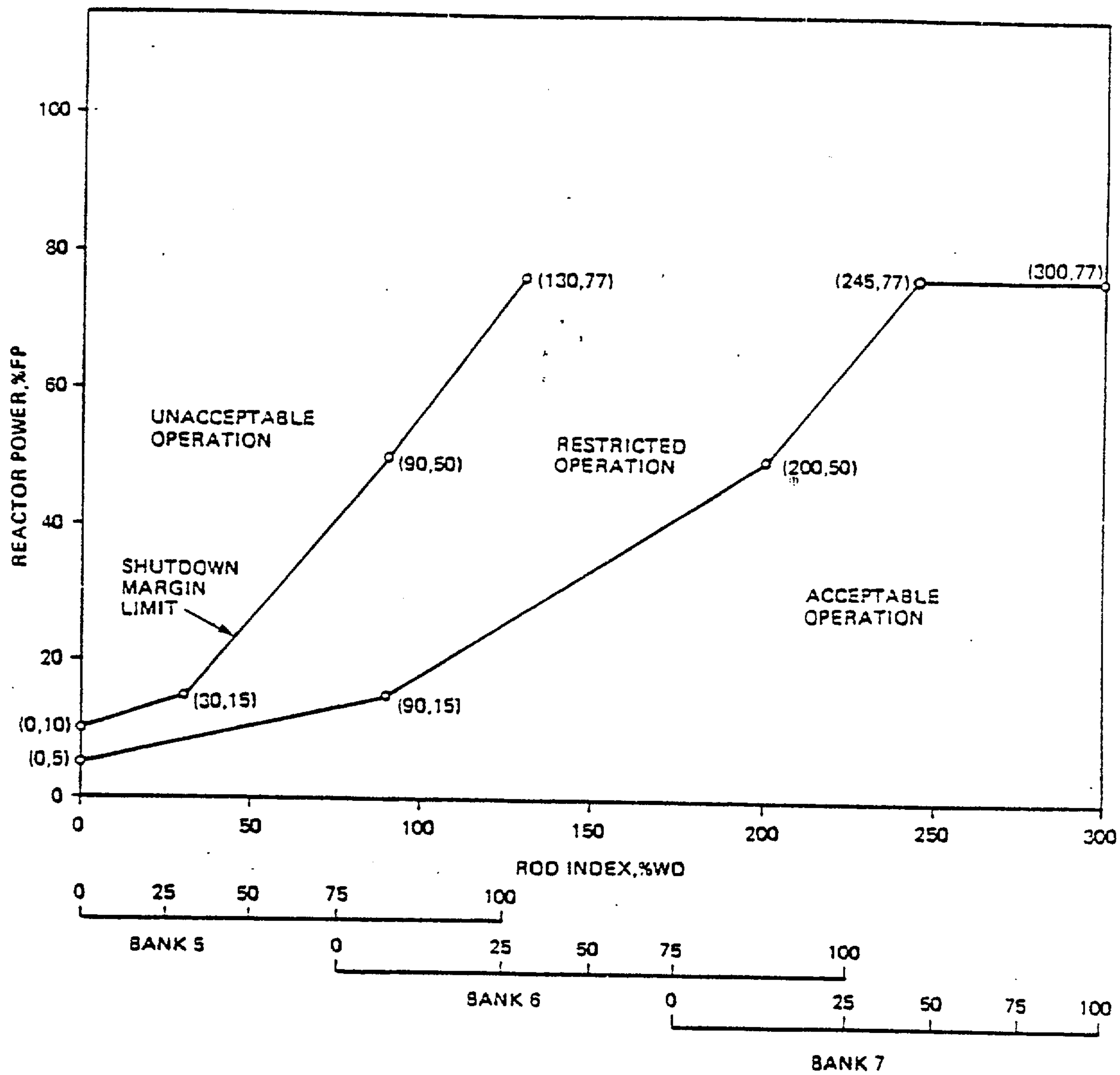
ROD POSITION LIMITS
FOR THREE-PUMP OPERATION
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-4
(2 of 3)



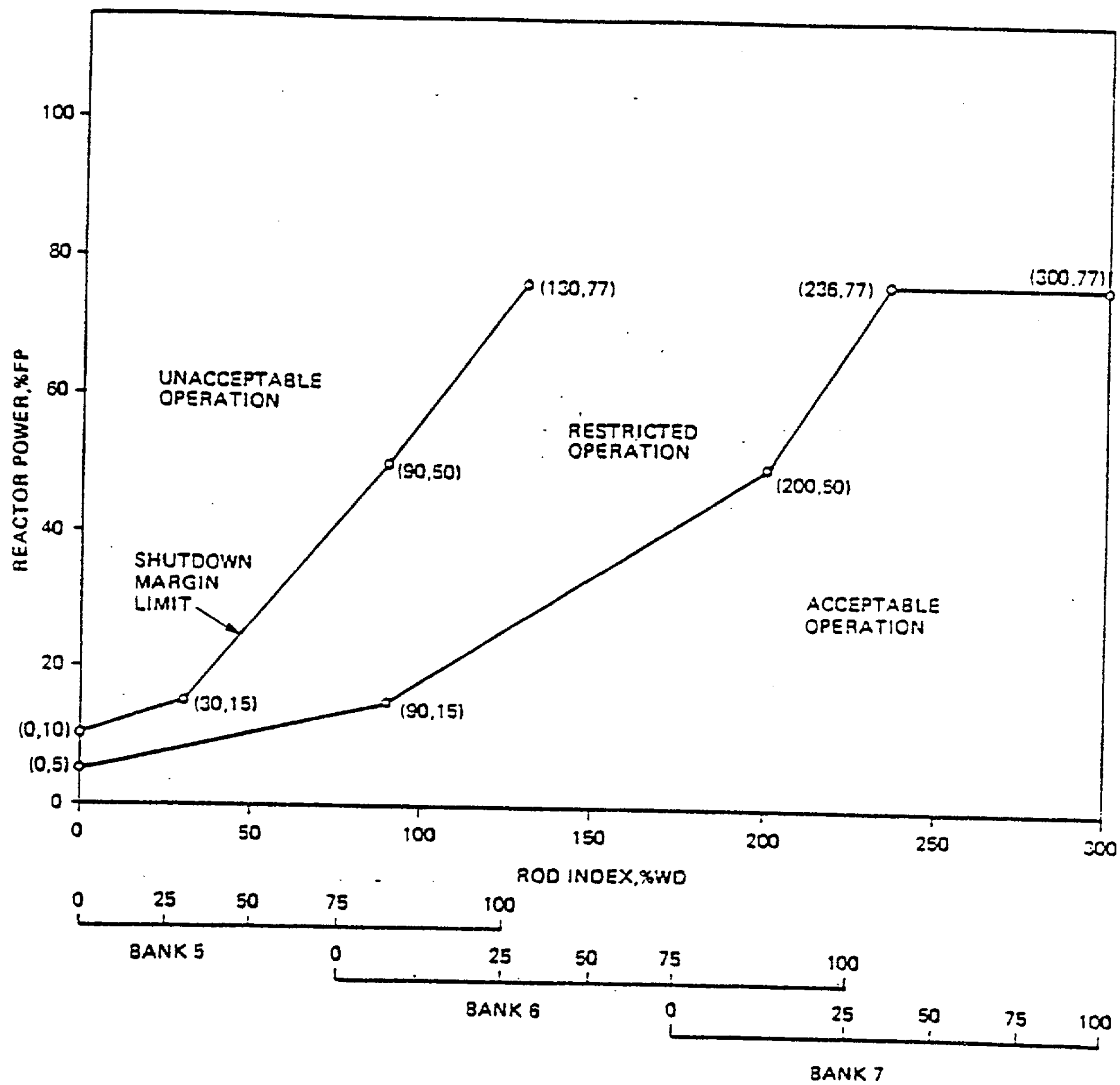
ROD POSITION LIMITS
FOR THREE-PUMP OPERATION
AFTER 200 ±10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-4
(3 of 3)



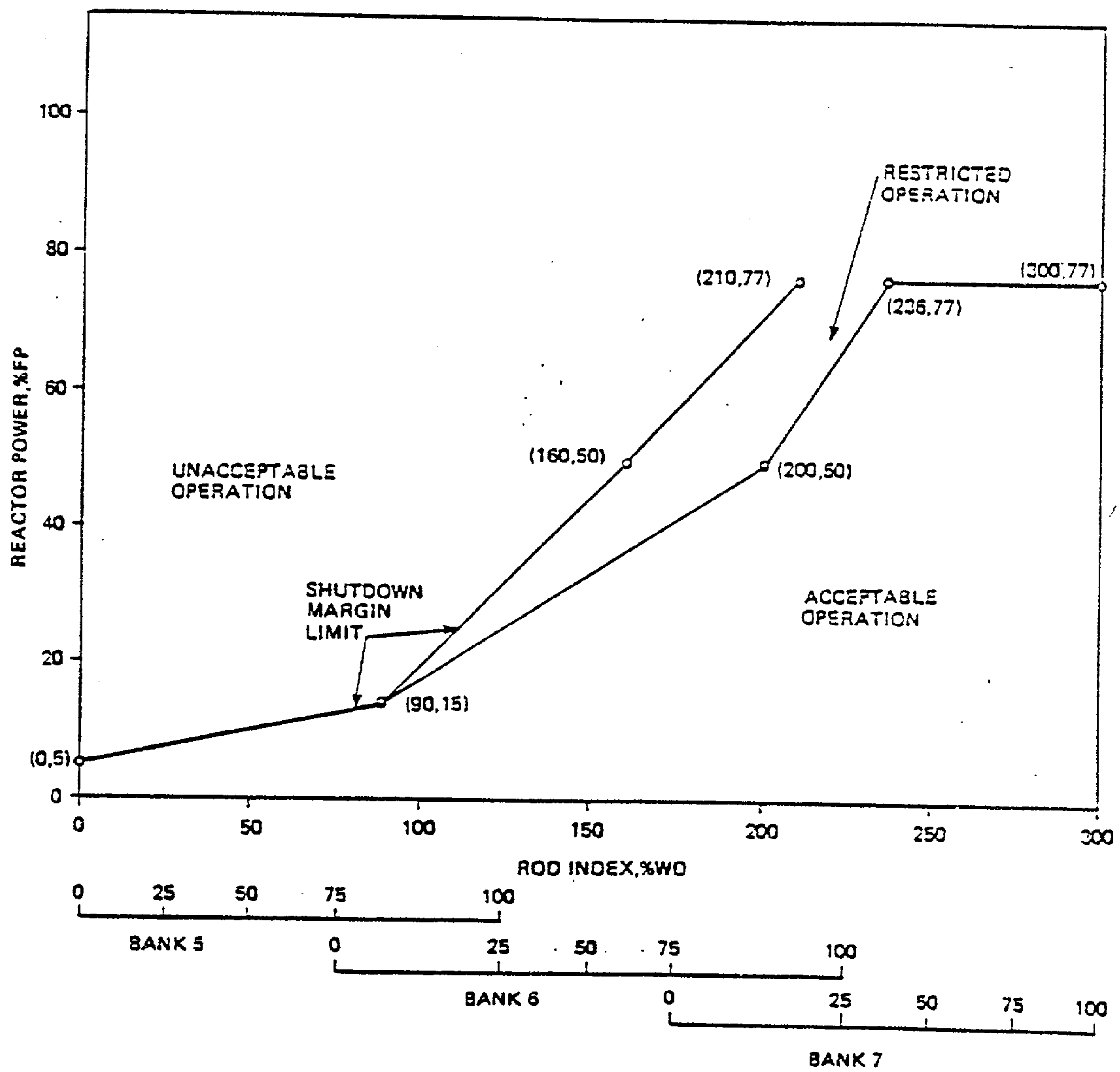
ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 TO 25 (+10, -0) EFPD
UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-5
(1 of 3)

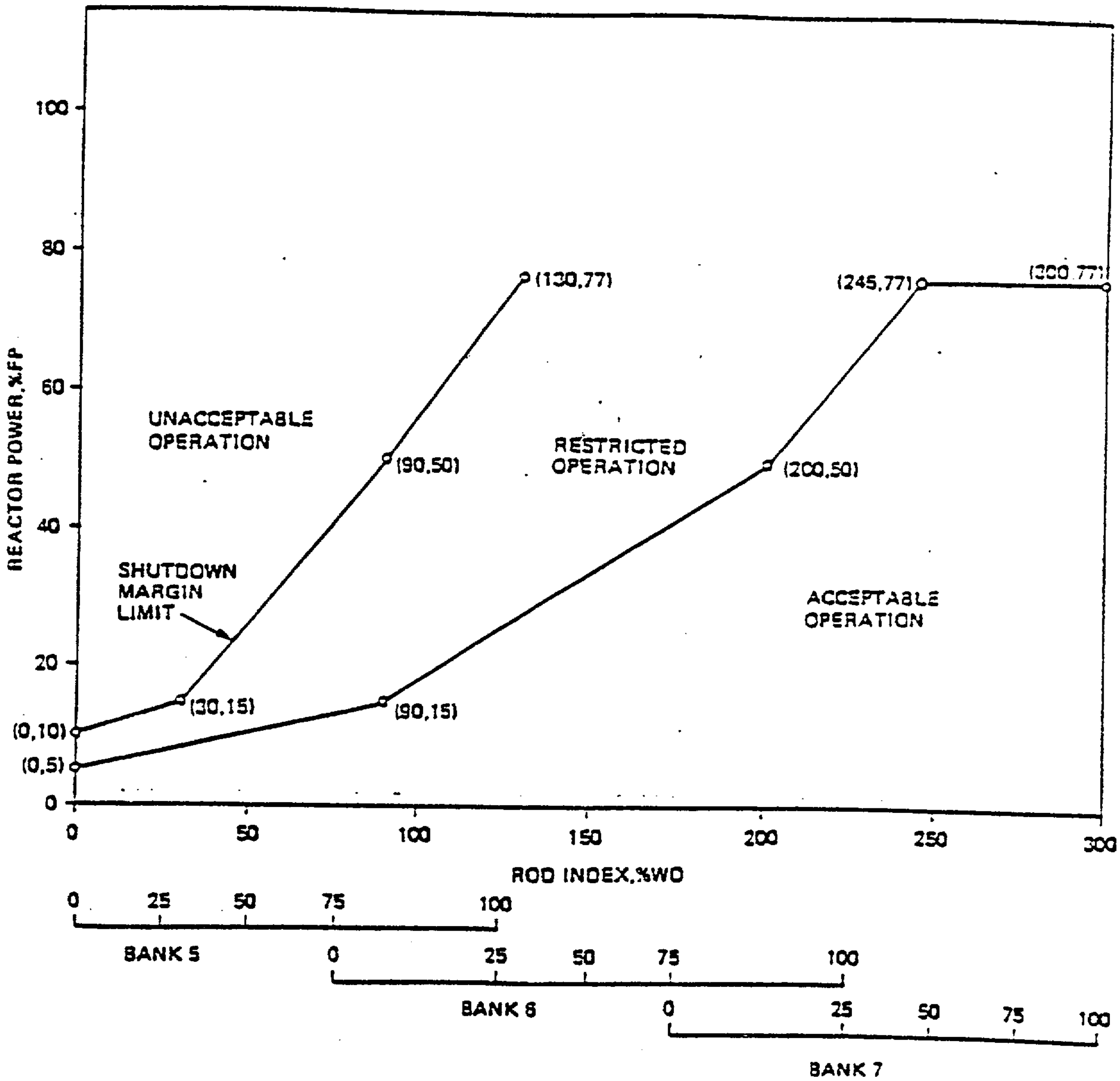


ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 25 (+10, -0) TO 200 ±10 EFPD
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-5
(2 of 3)

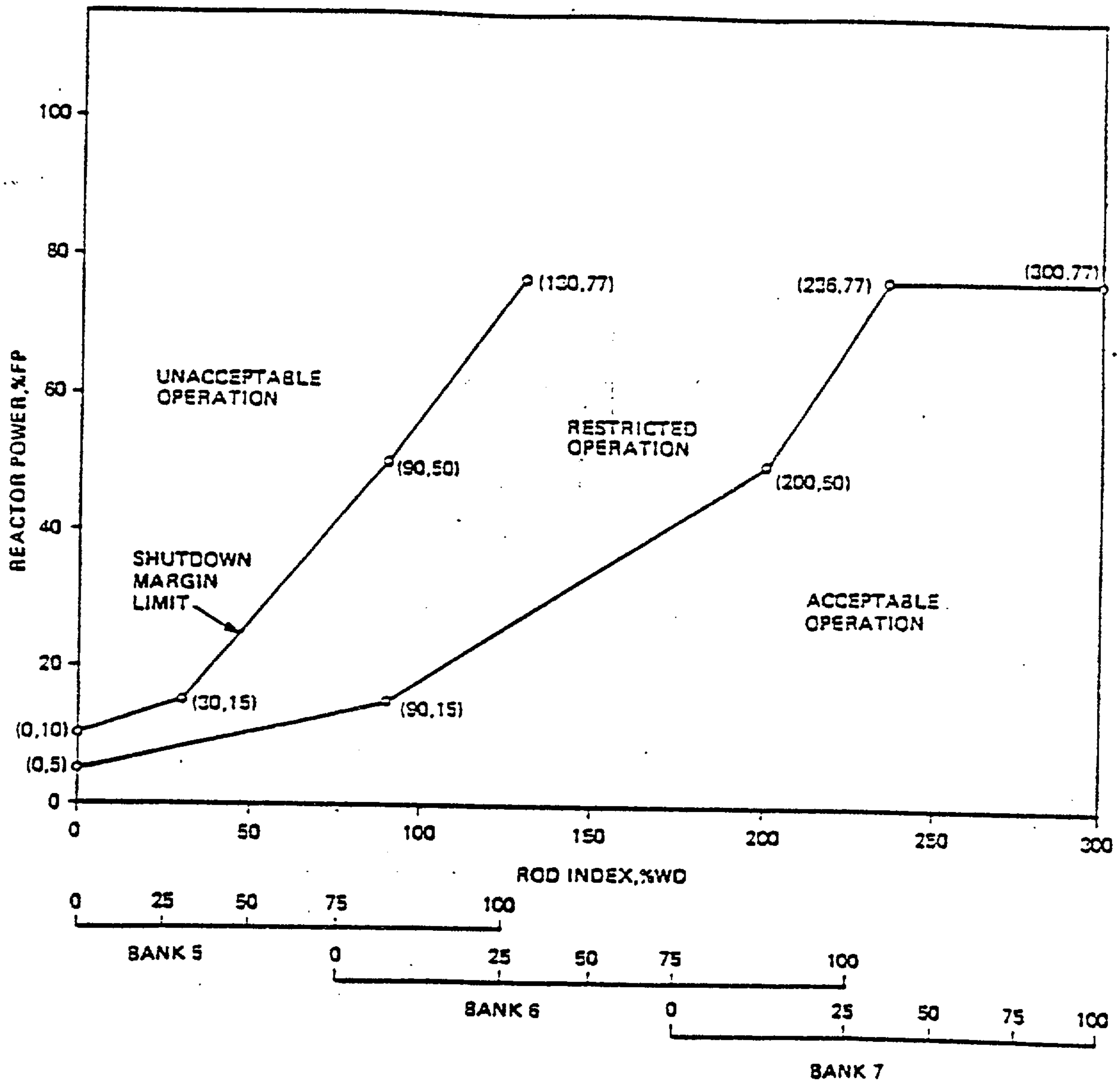


ROD POSITION LIMITS
FOR THREE PUMP OPERATION
AFTER 200 ±10 EFPD
UNIT 2

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

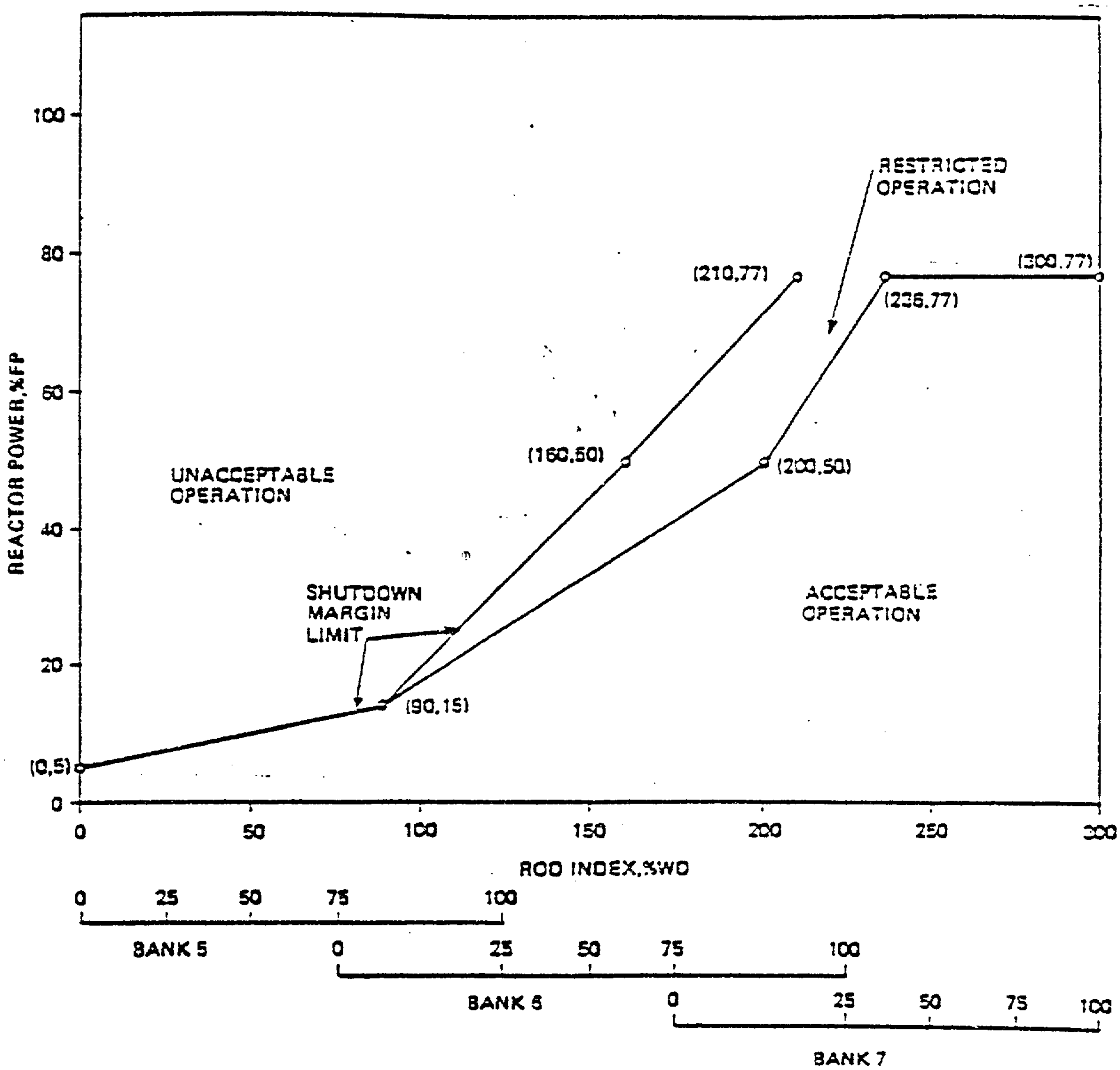


ROD POSITION LIMITS
FOR THREE PUMP OPERATION
FROM 0 TO 25 +10/-0 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-6
(1 of 3)



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

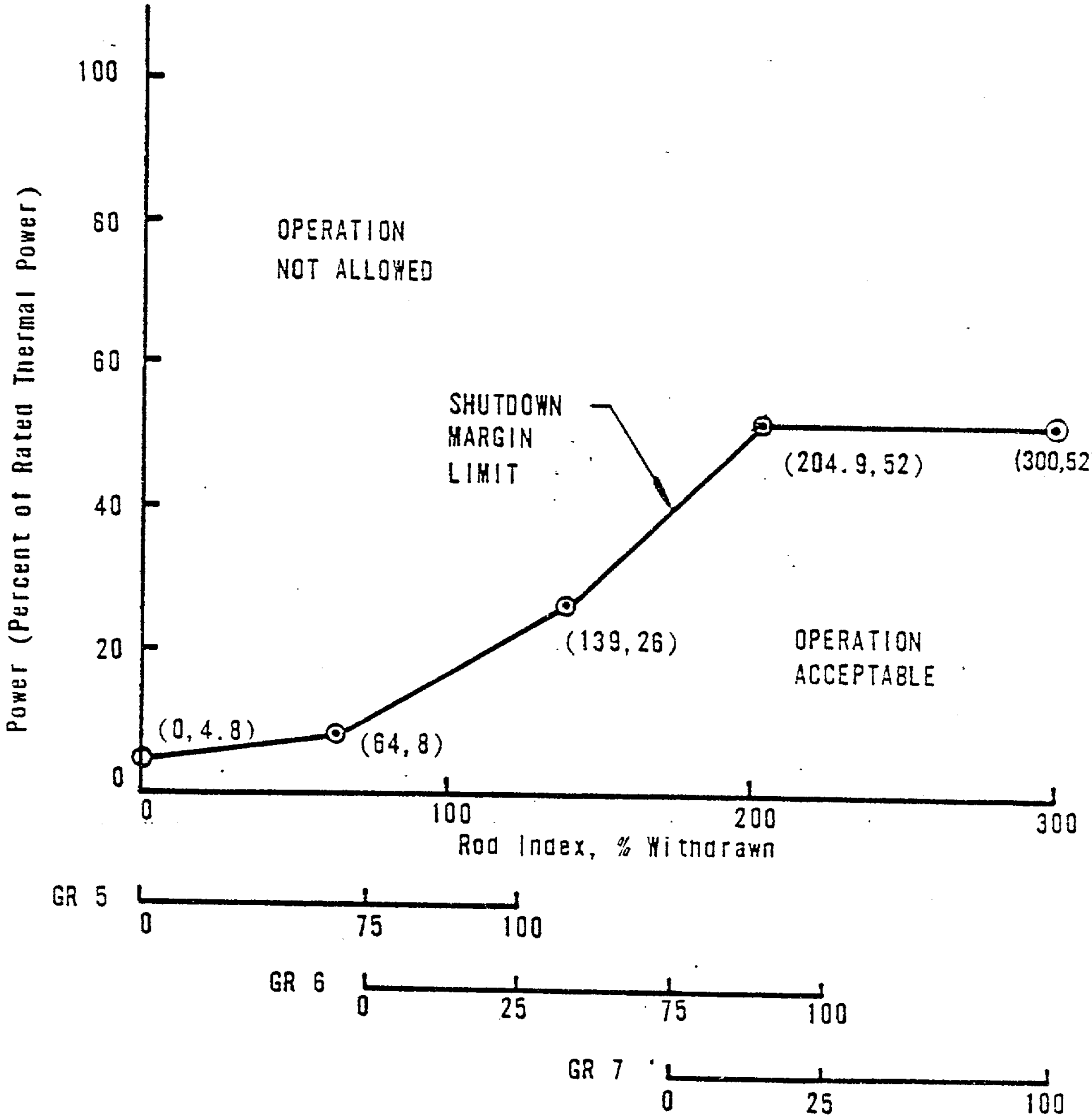
Figure 3.5.2-6
(2 of 3)



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

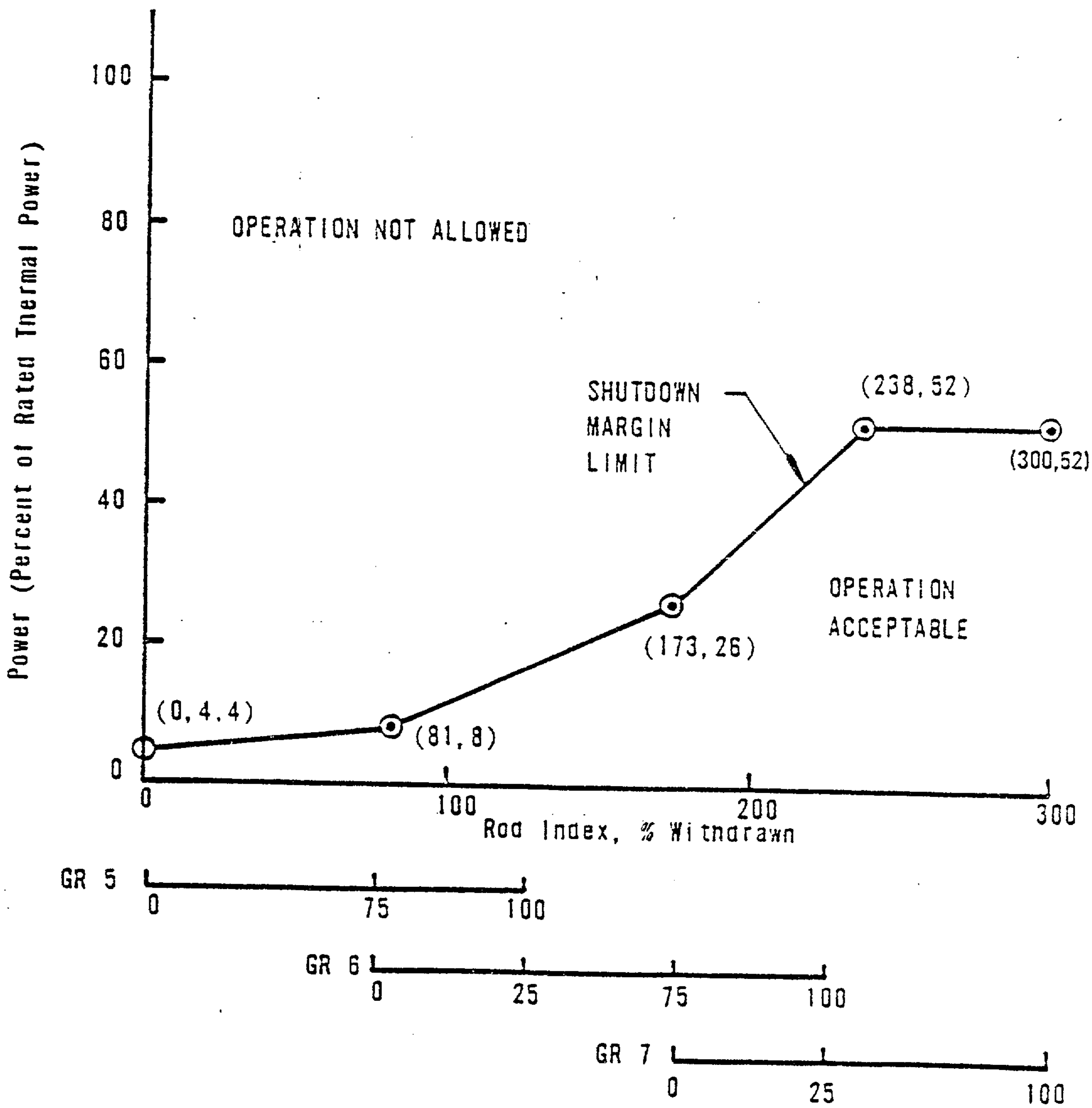
OCONEE NUCLEAR STATION

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

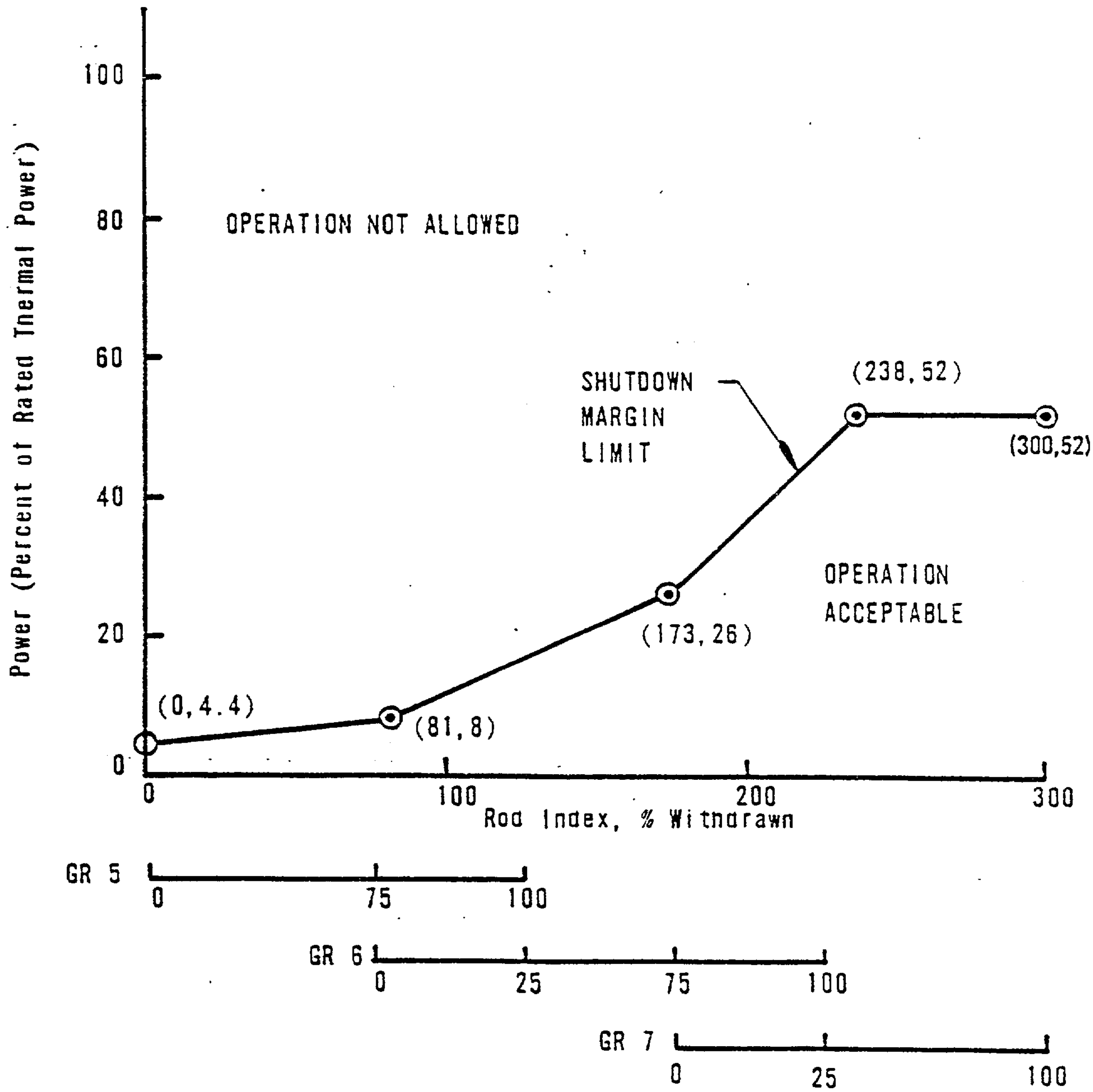
Figure 3.5.2-7
(1 of 3)



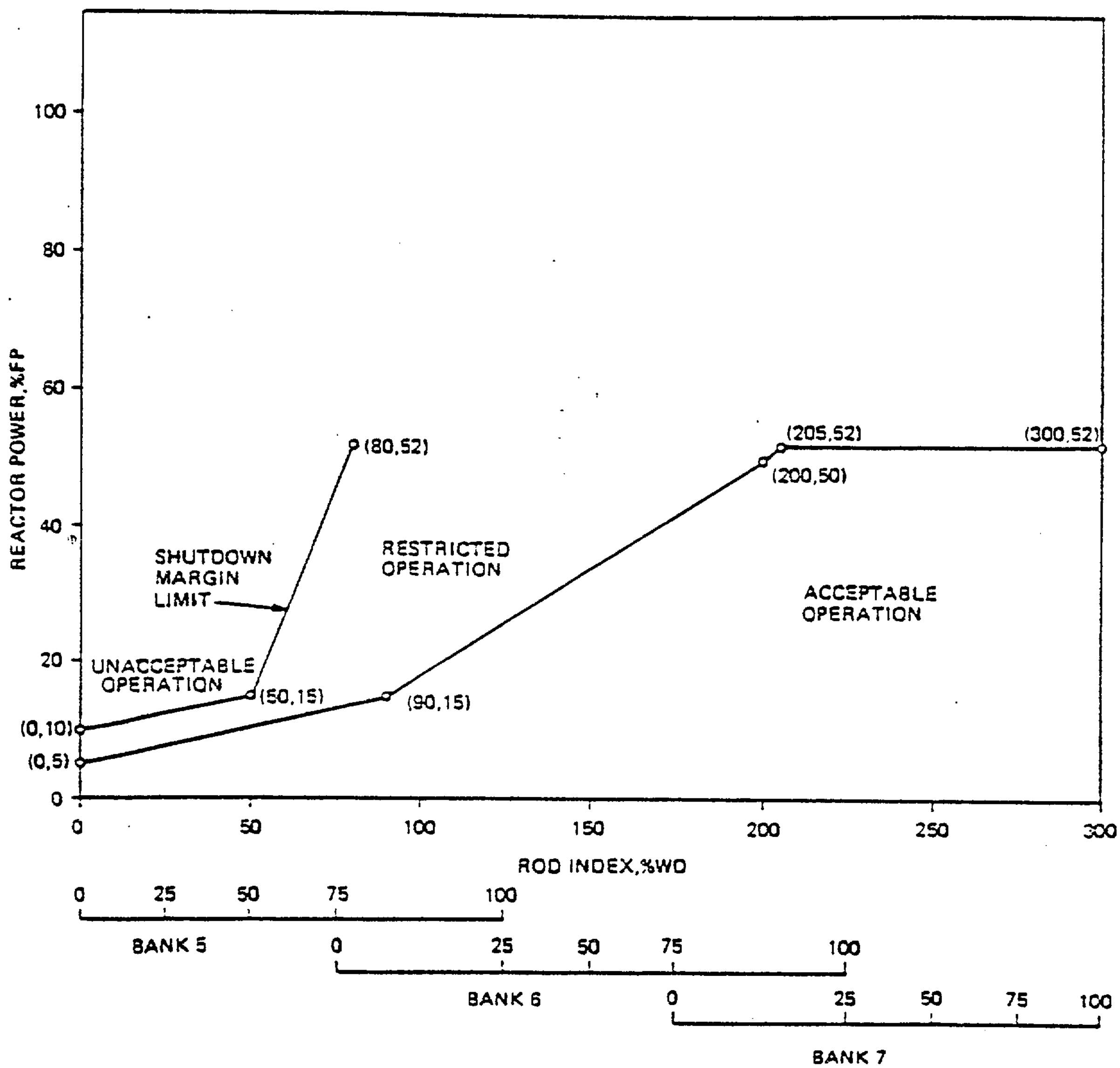
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1

OCONEE NUCLEAR STATION

Figure 3.5.2-7
(2 of 3)

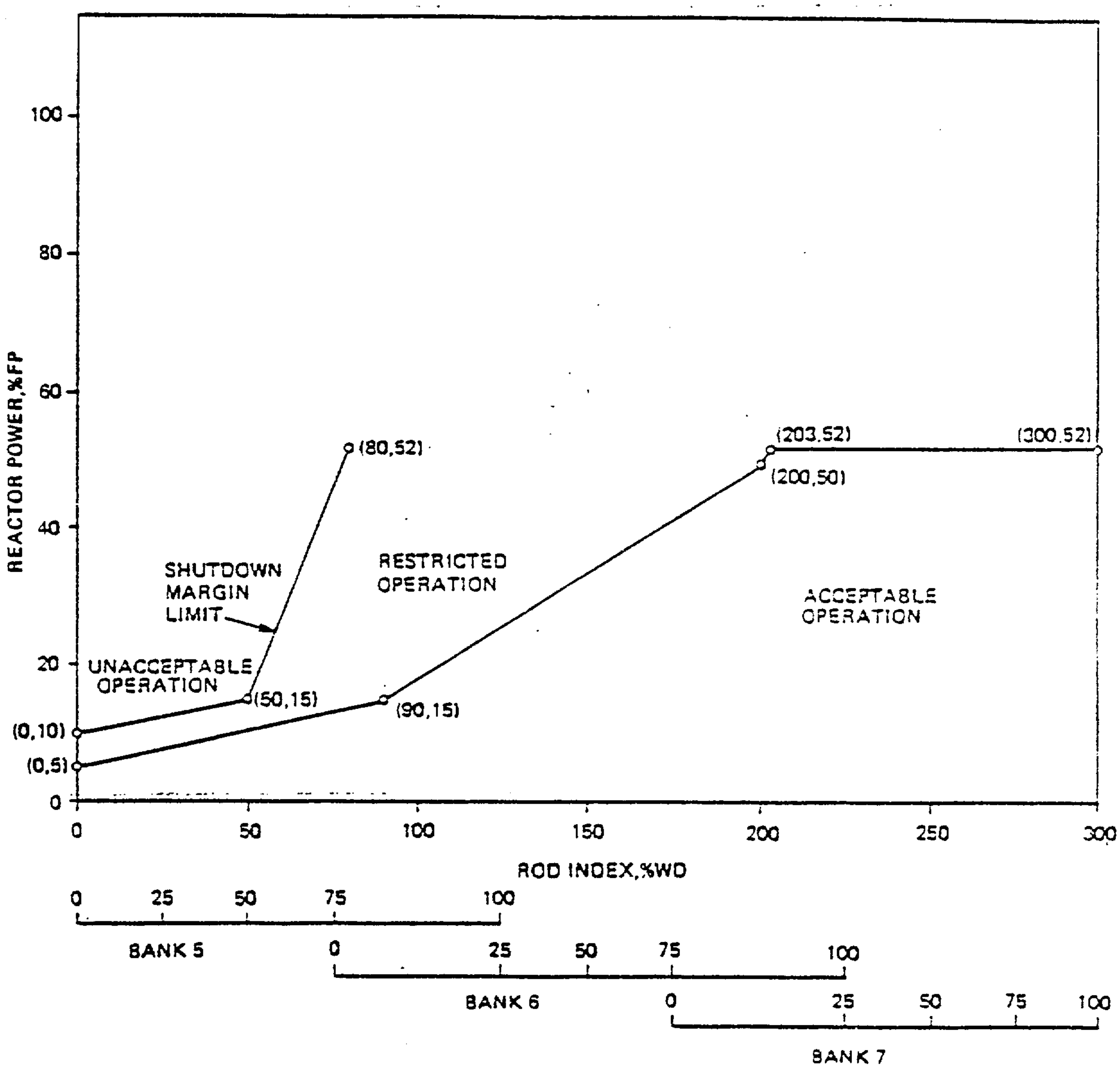


ROD POSITION LIMITS
FOR TWO PUMP OPERATION
AFTER 200 ±10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-7
(3 of 3)



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

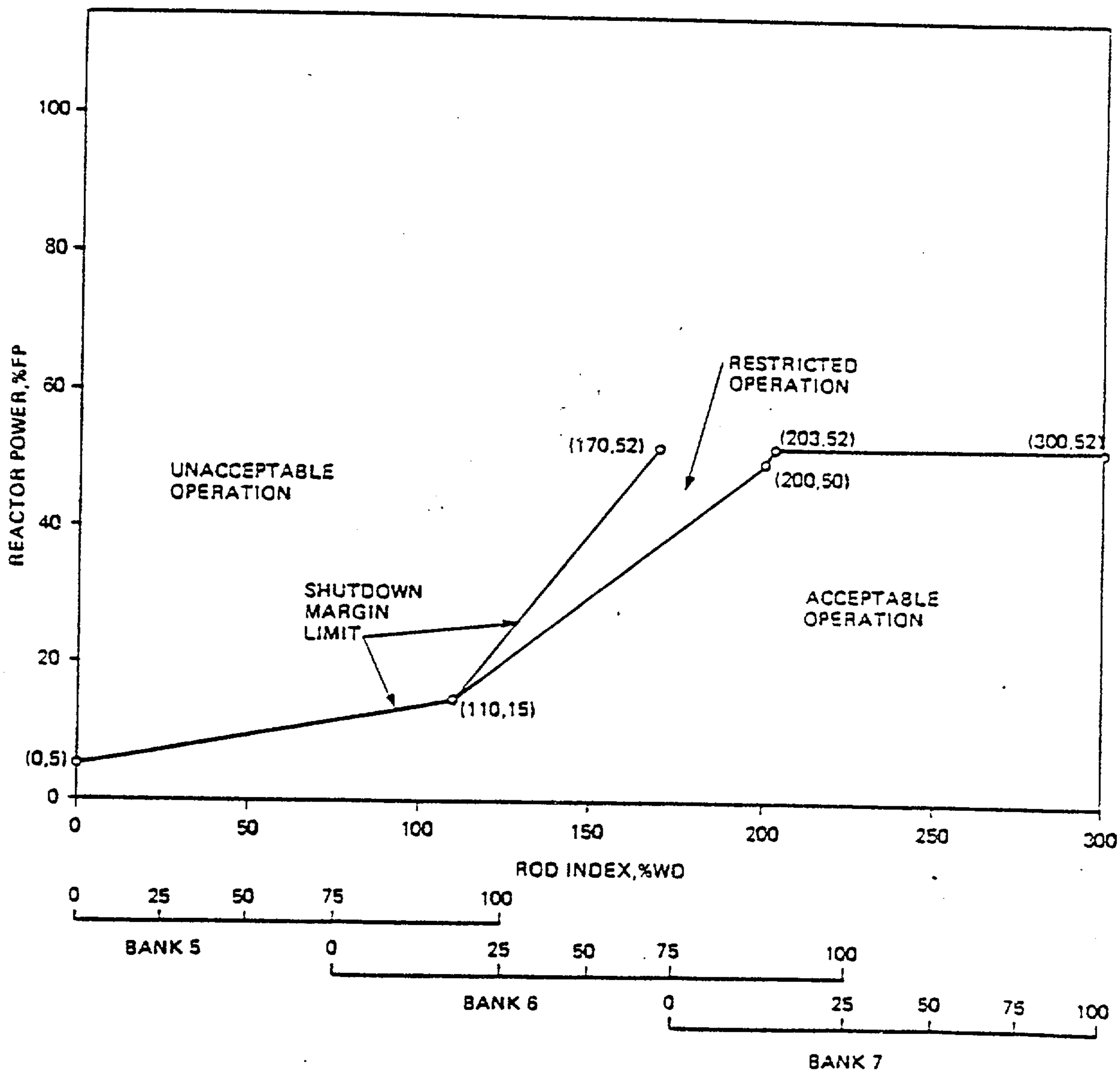
OCONEE NUCLEAR STATION
Figure 3.5.2-8
(1 of 3)



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

DUKE POWER

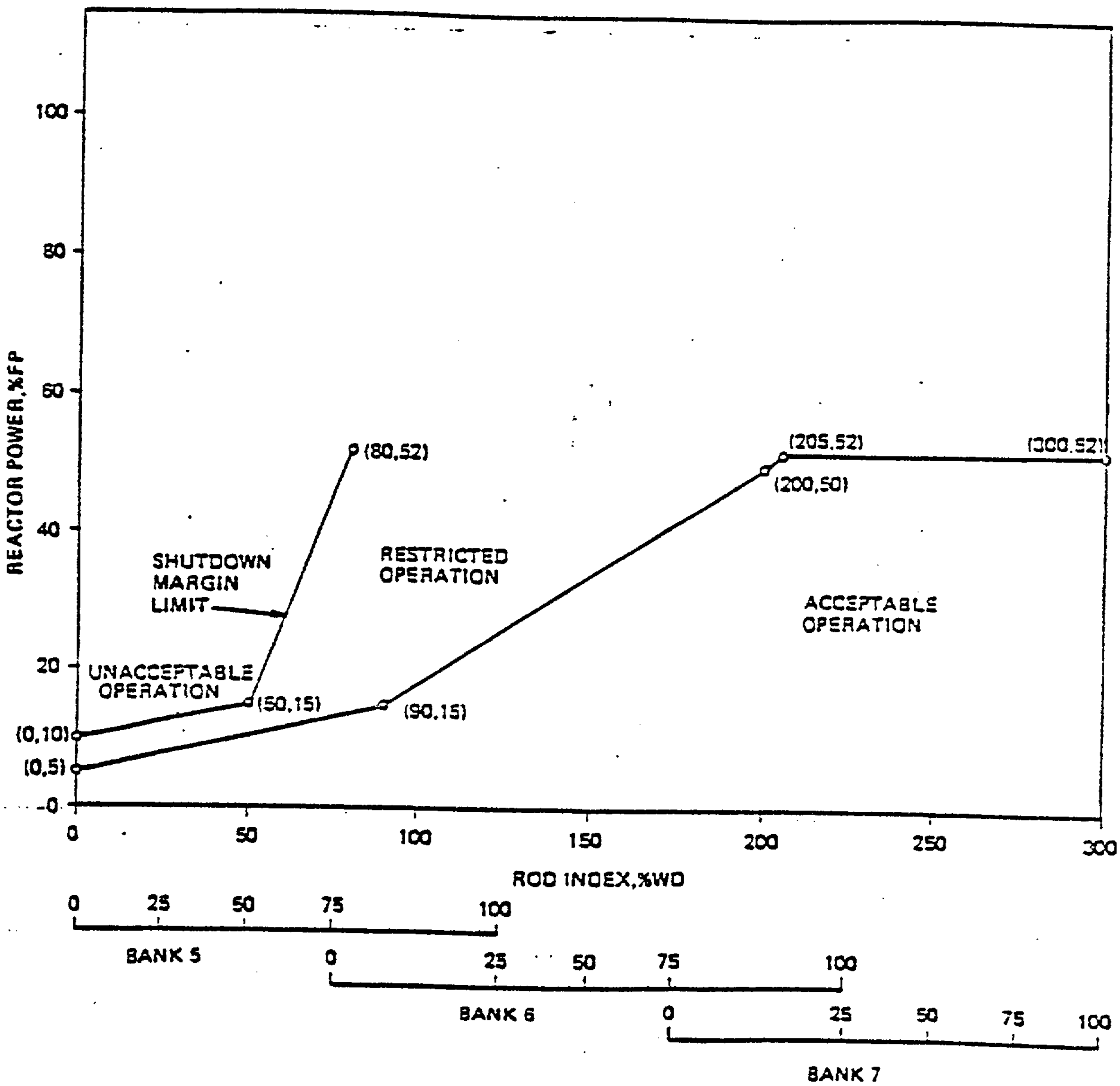
OCONEE NUCLEAR STATION
Figure 3.5.2-8
(2 of 3)



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

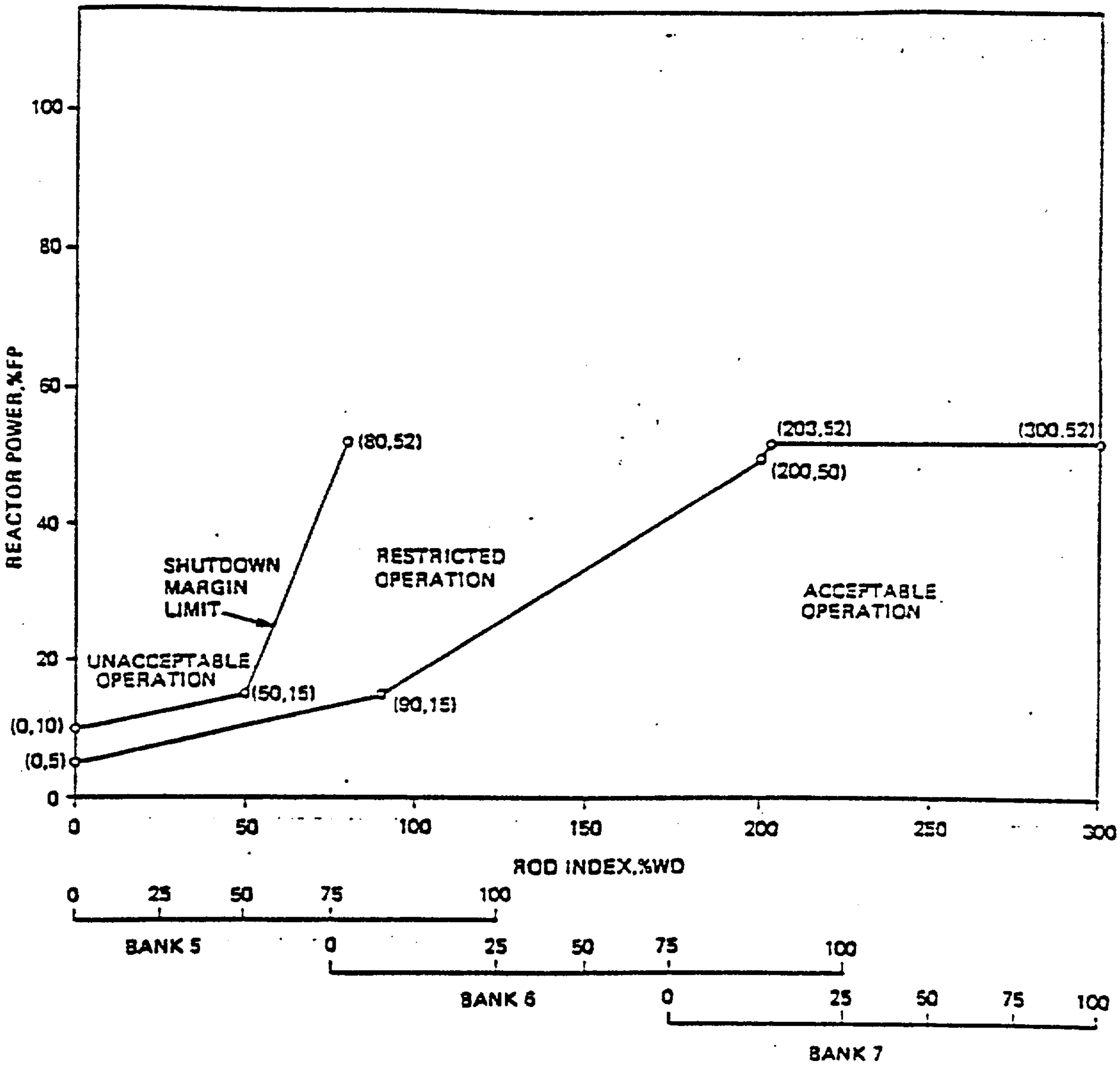
OCONEE NUCLEAR STATION

Figure 3.5.2-8
(3 of 3)

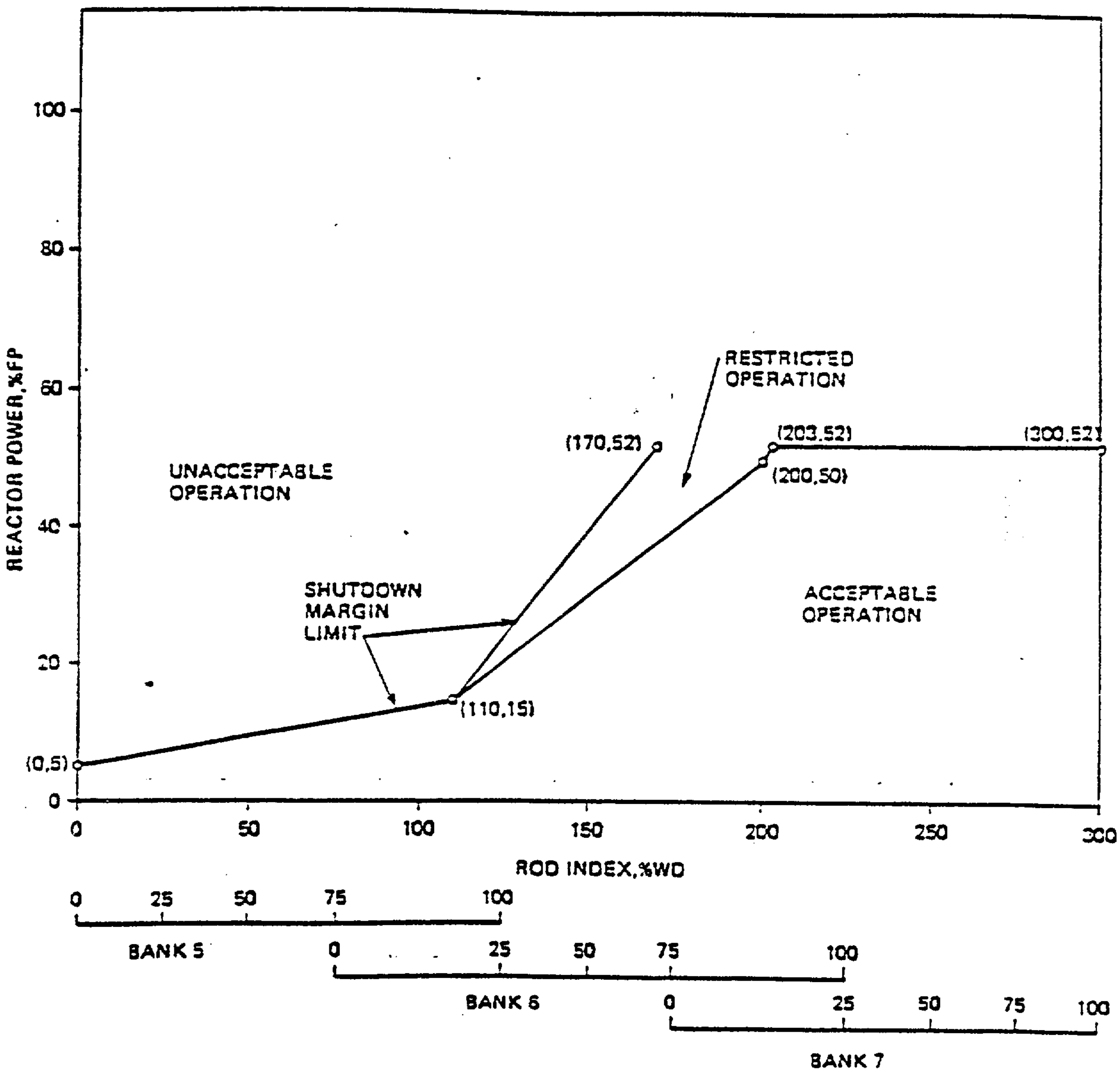


ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 0 TO 25 +10/-0 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-9
(1 of 3)

Amendments Nos. 129 , 129 , & 126



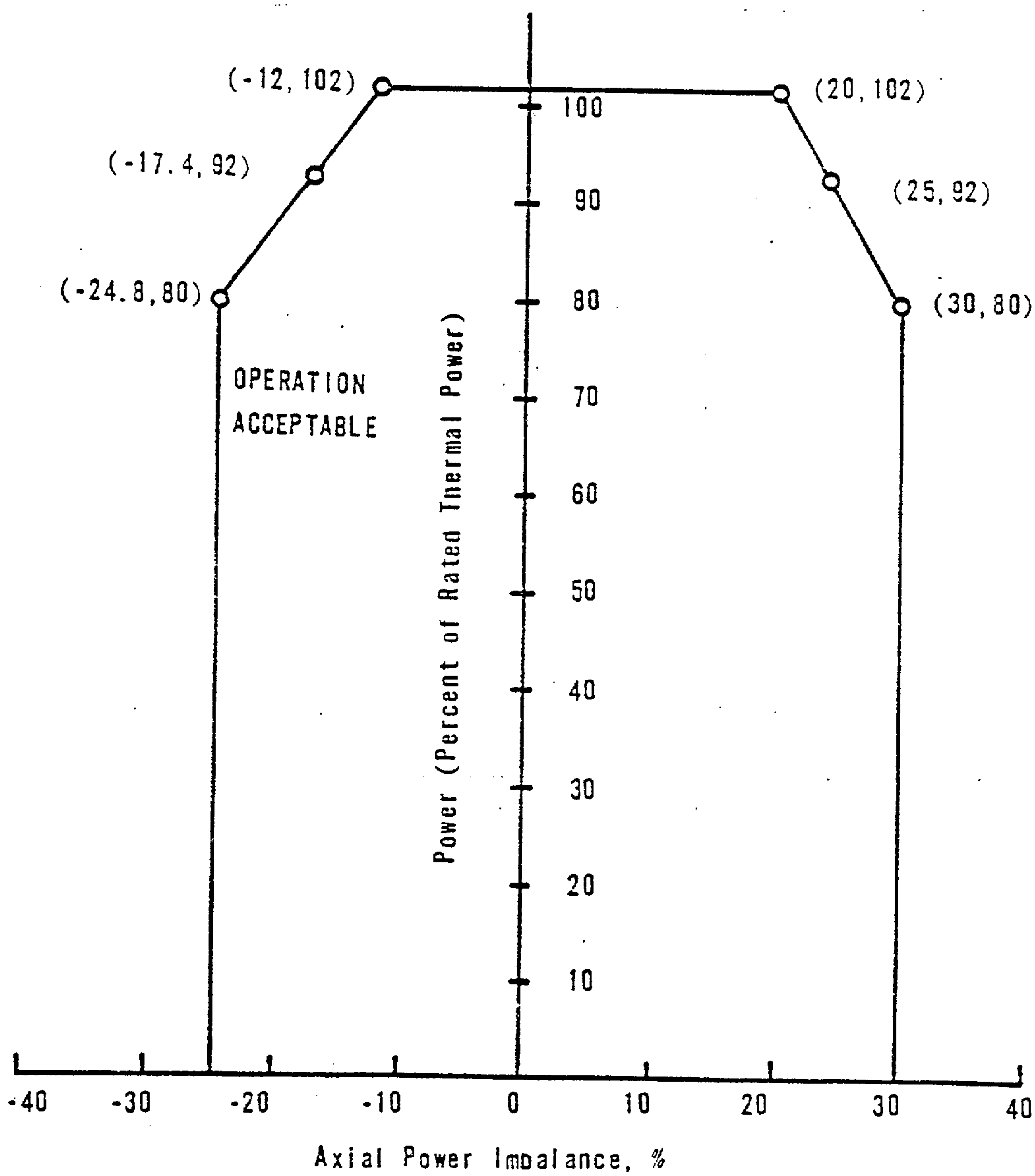
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
FROM 25 +10/-0 TO 200 +10/-10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-9
(2 of 3)



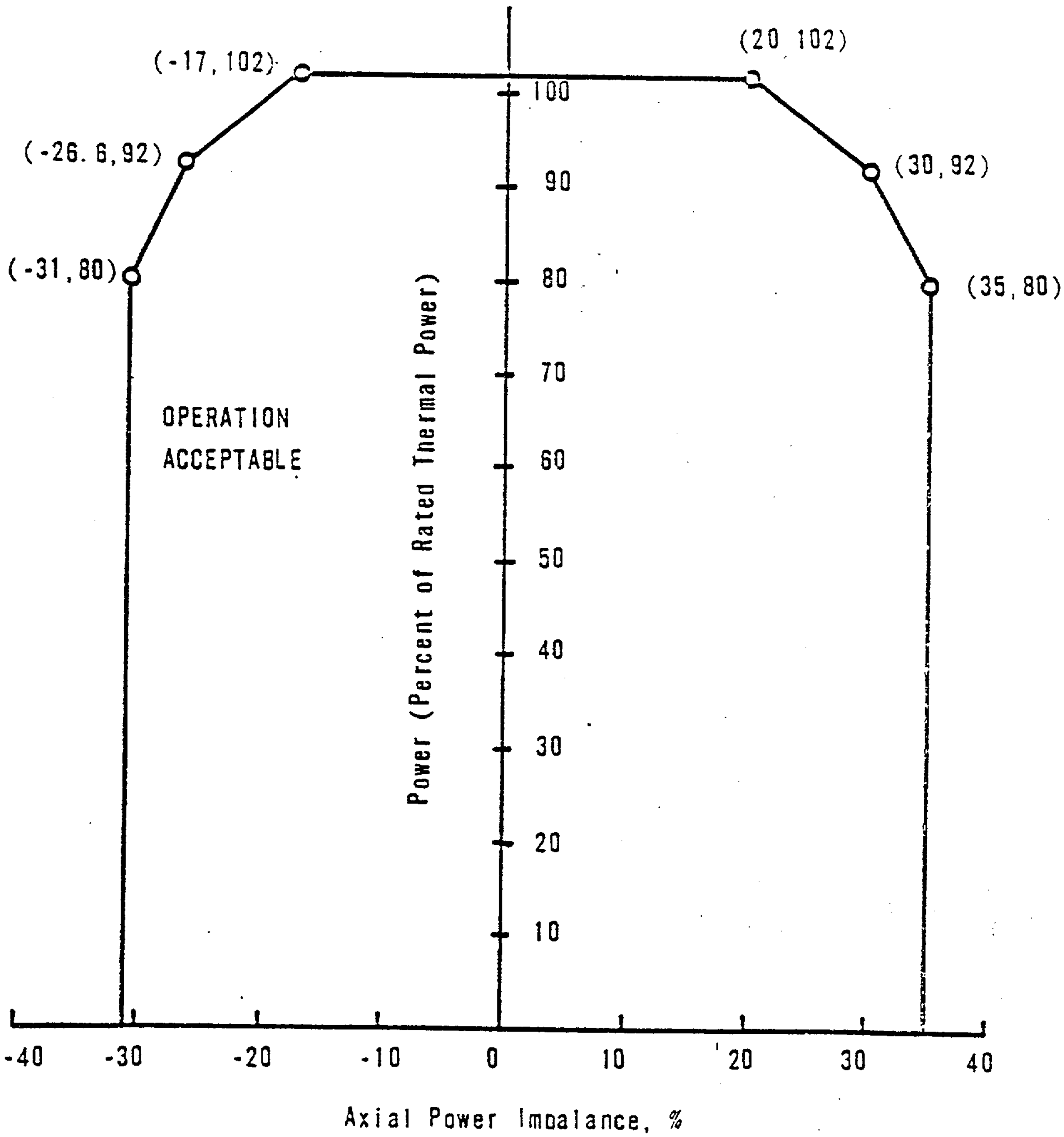
ROD POSITION LIMITS
FOR TWO PUMP OPERATION
AFTER 200 +10/-10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-9
(3 of 3)



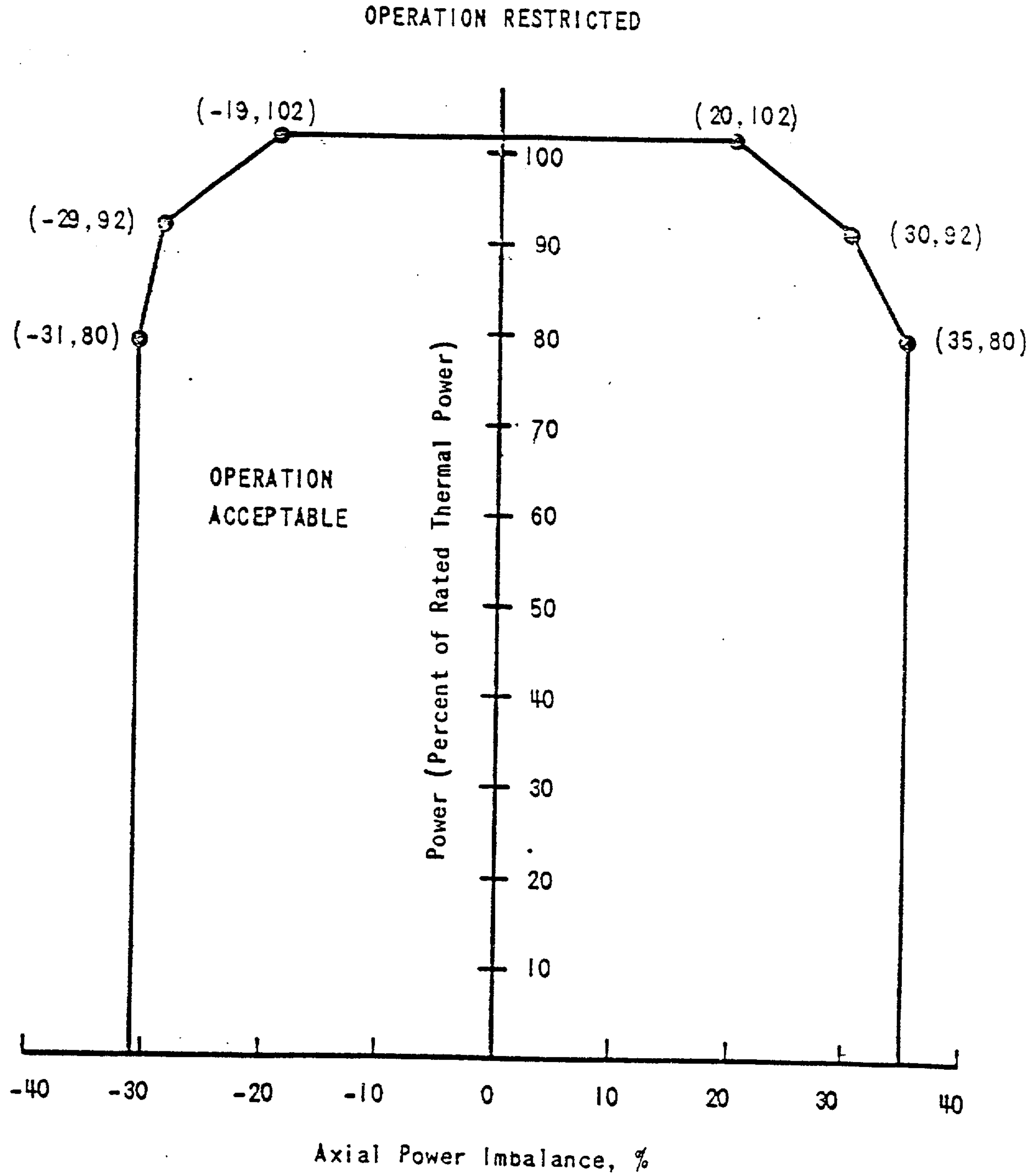
OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 TO 26 +10/-0 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-10
(1 of 3)



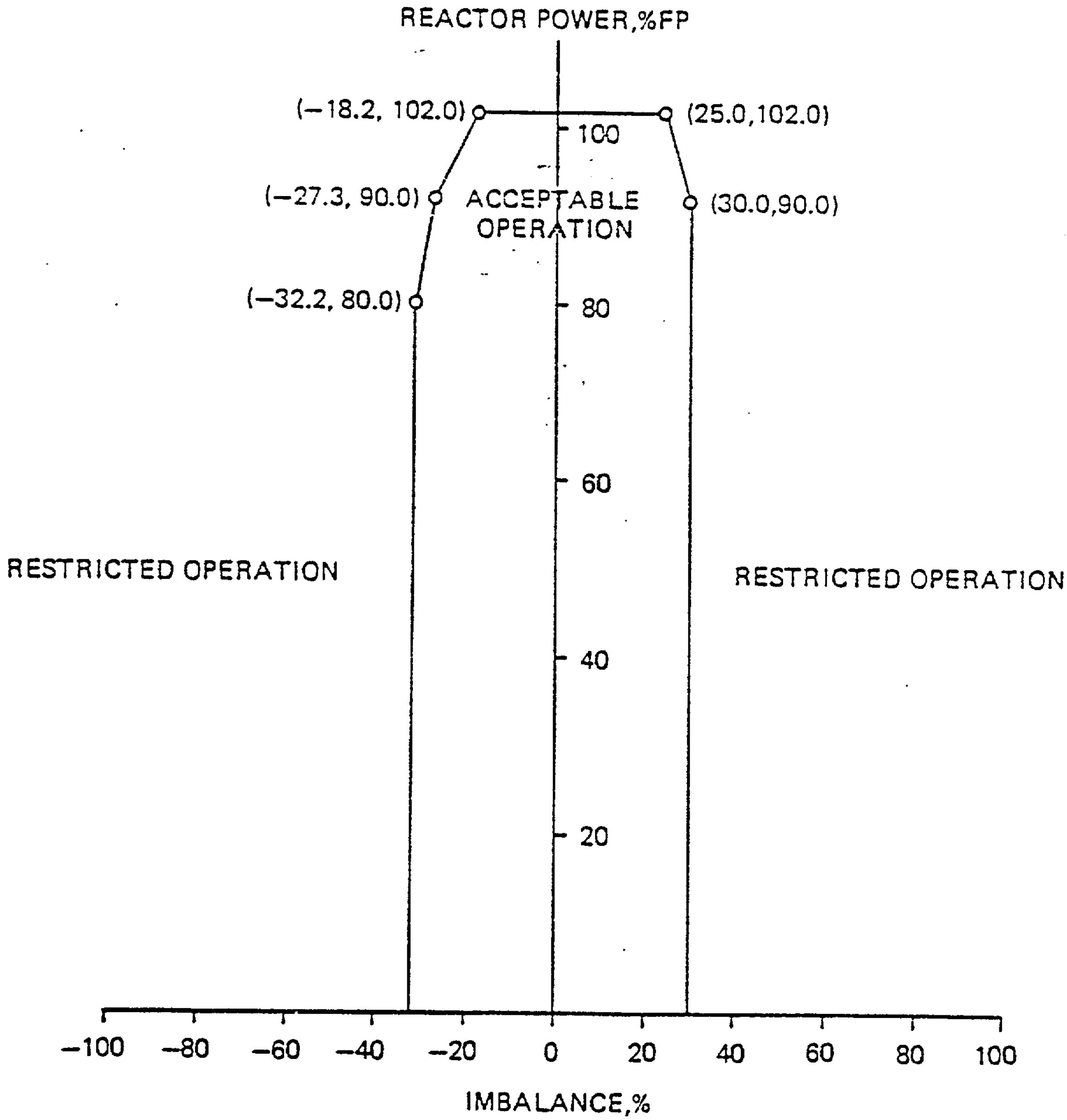
OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1

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)

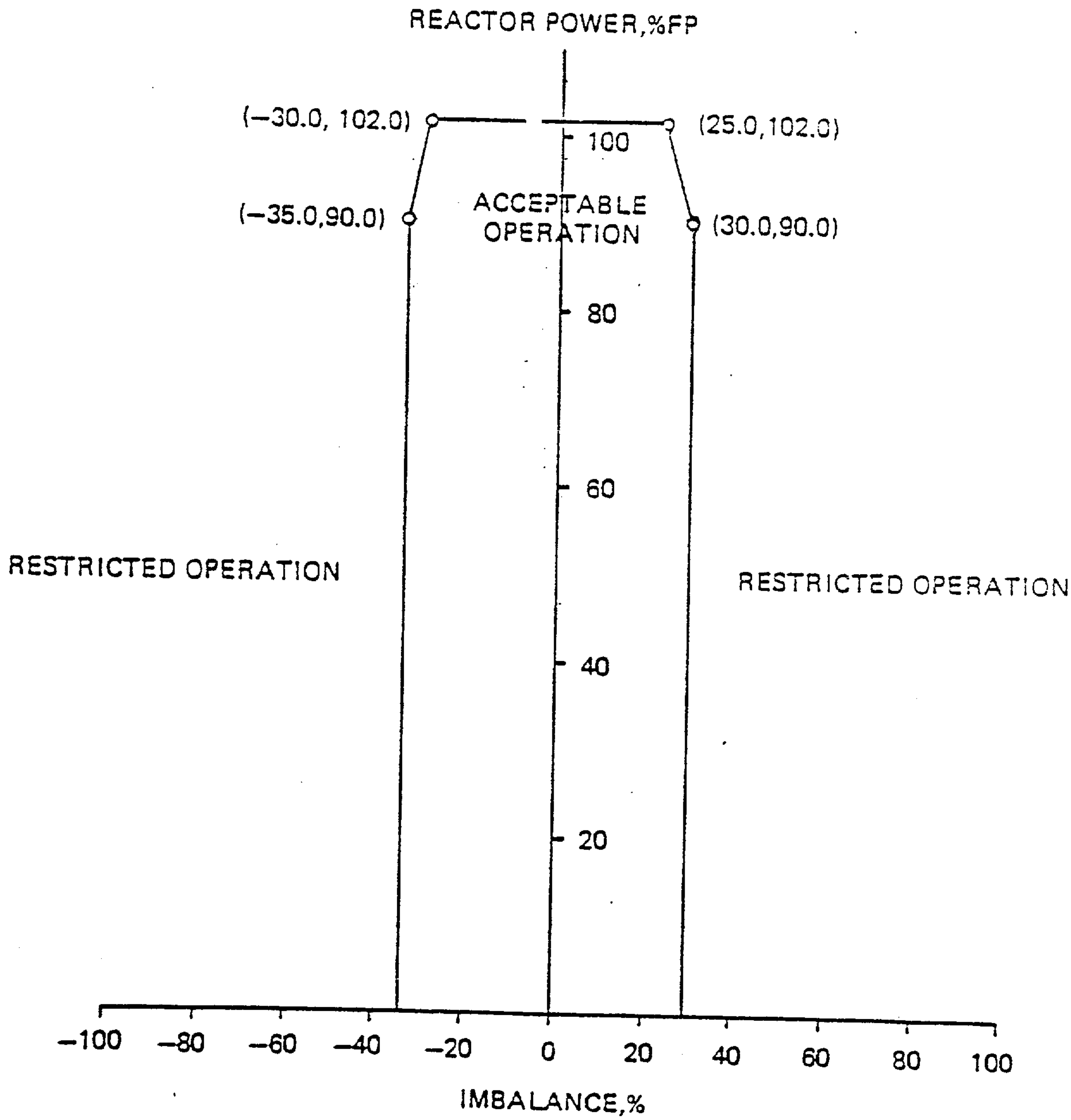


OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 TO 25 (+10, -0) EFPD
UNIT 2

DUKE POWER

OCONEE NUCLEAR STATION

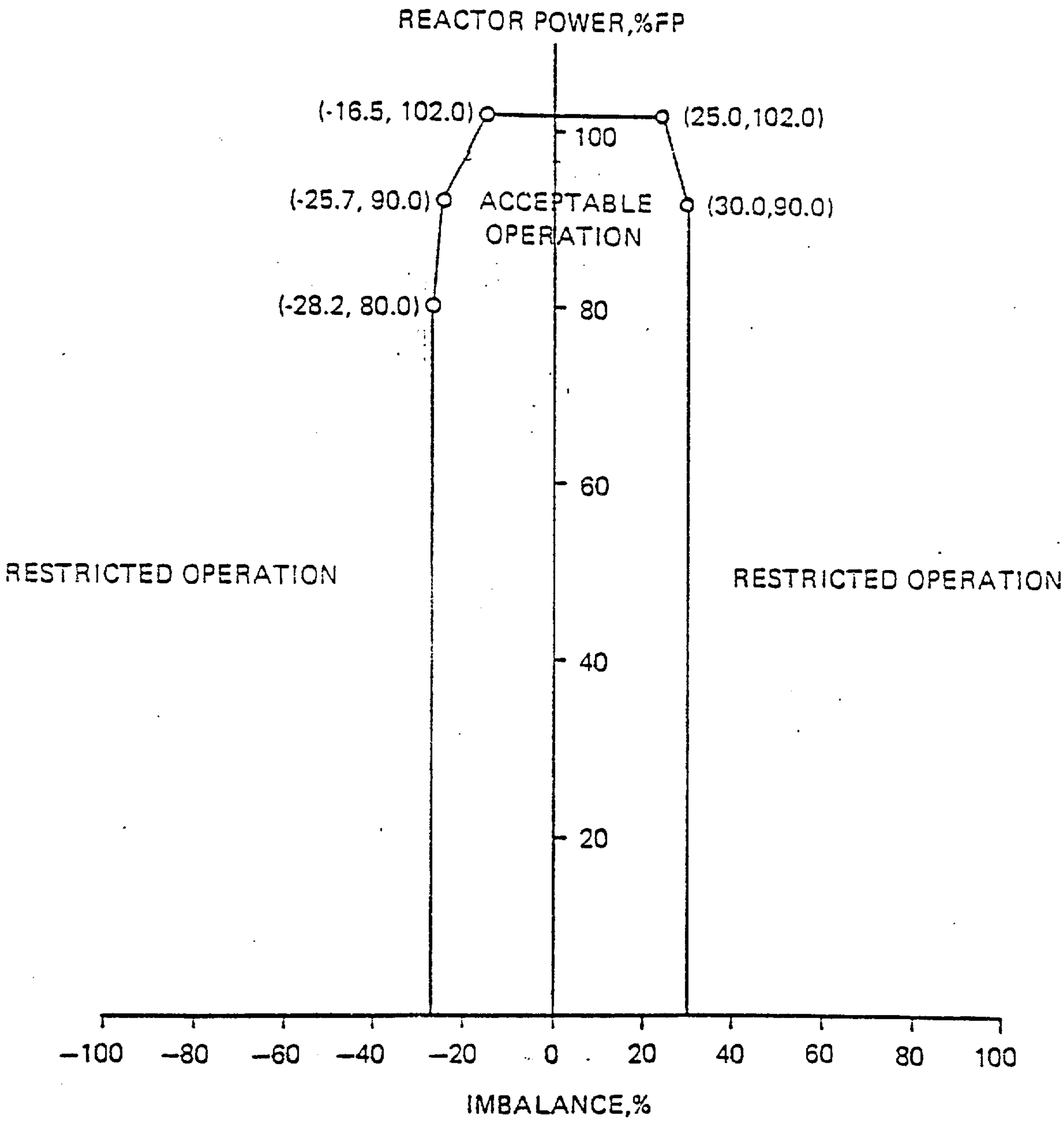
Figure 3.5.2-11
(1 of 2)



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

OCONEE NUCLEAR STATION

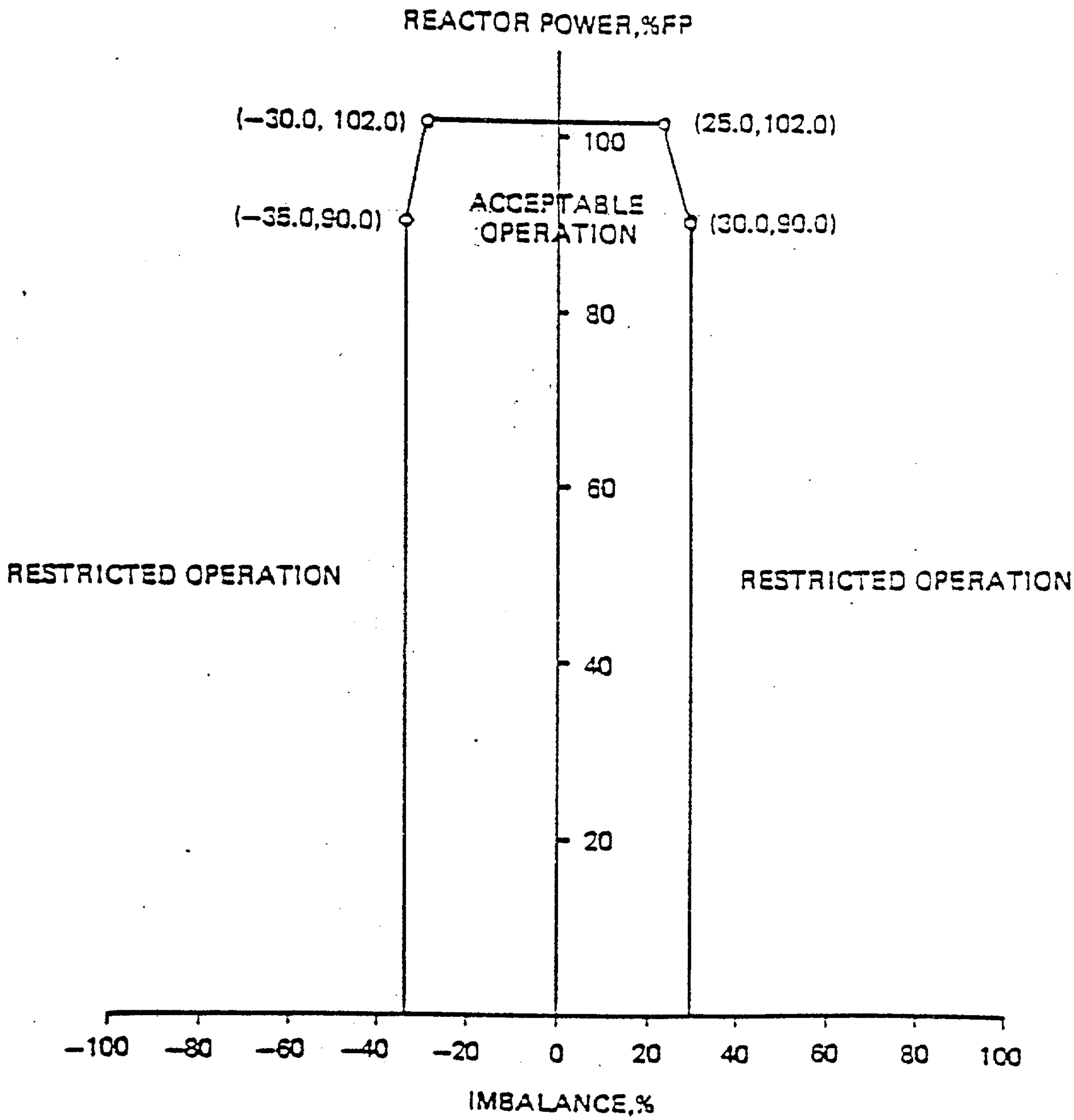
Figure 3.5.2-11
(2 of 2)



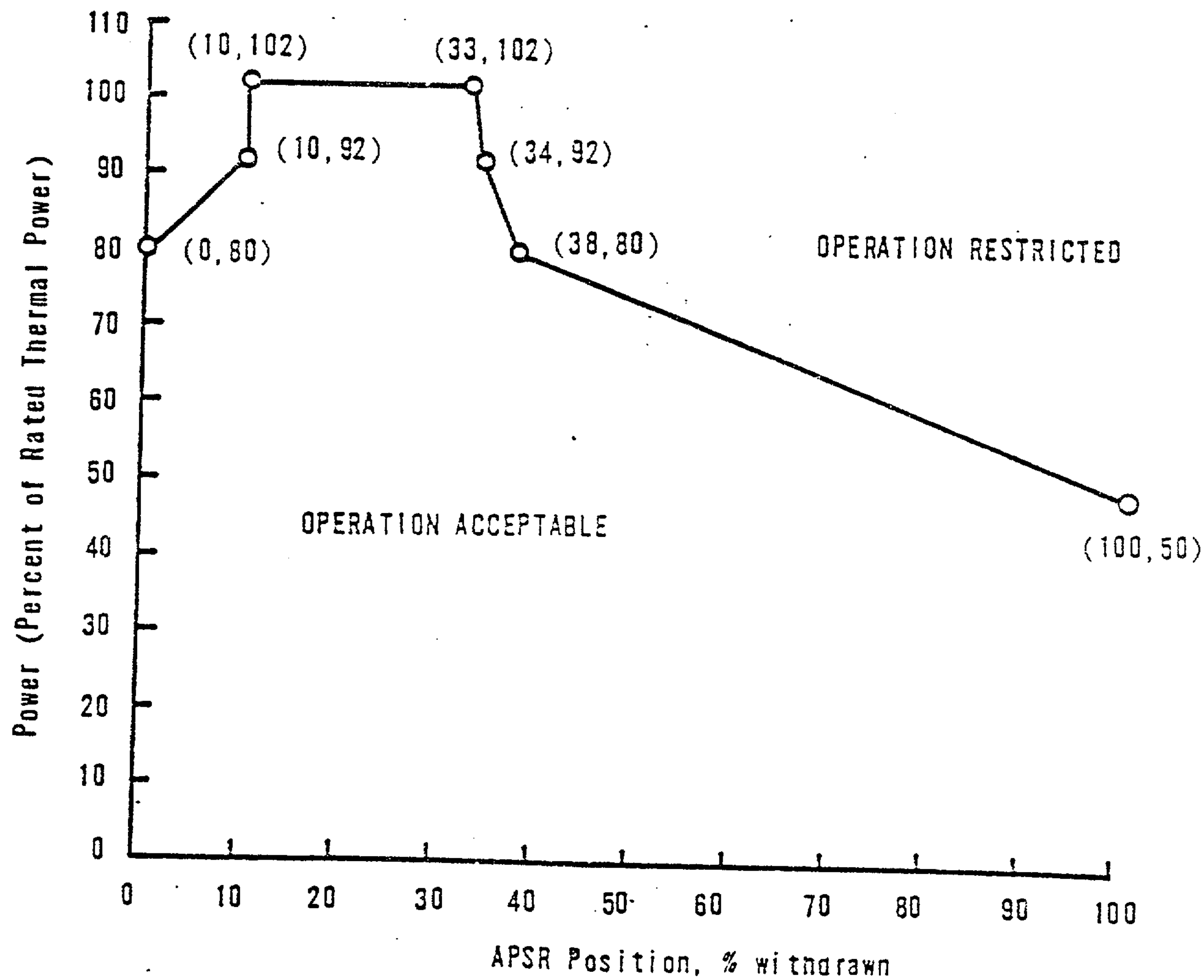
OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 TO 200 +10/-10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

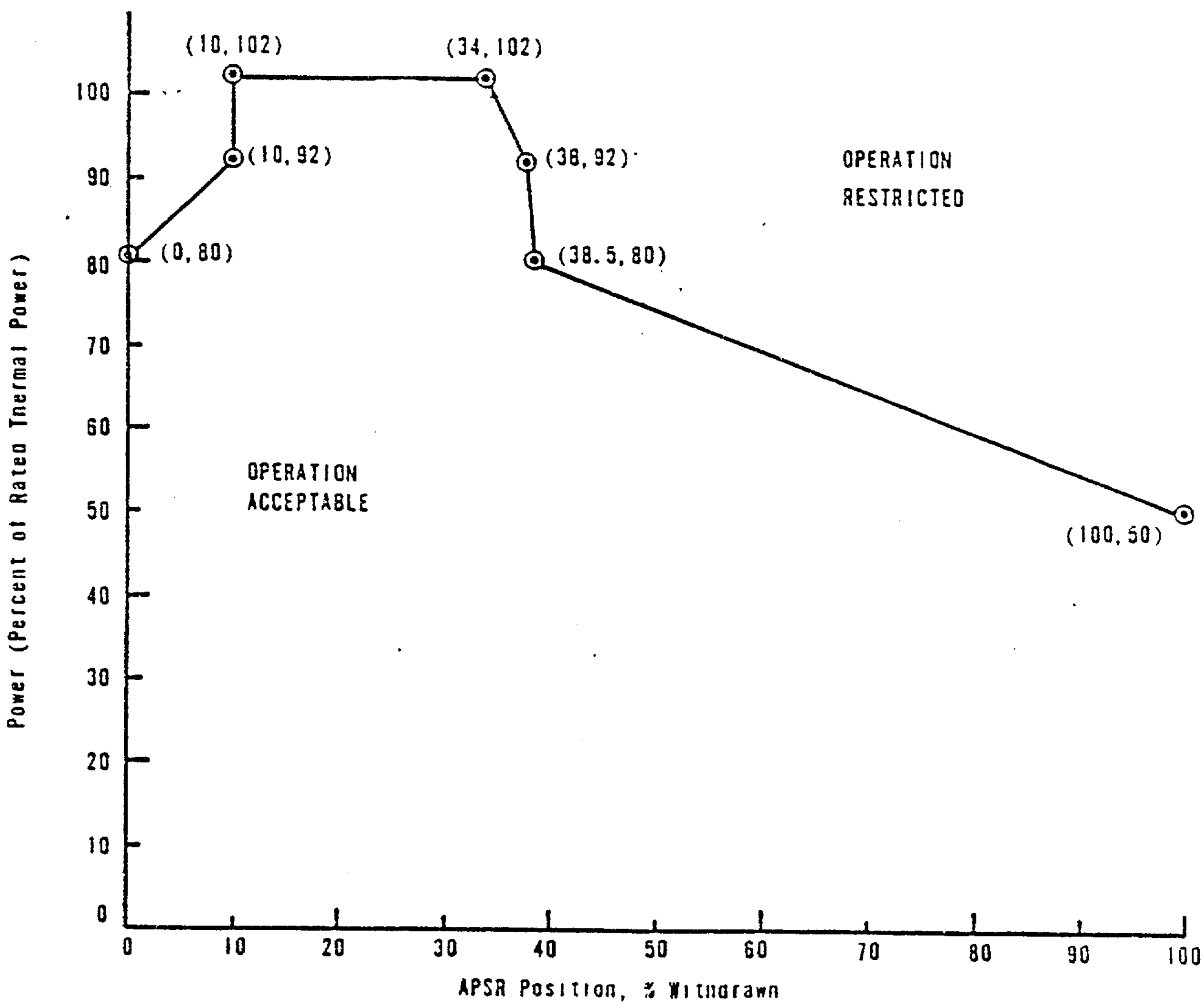
Figure 3.5.2-12
(1 of 2)



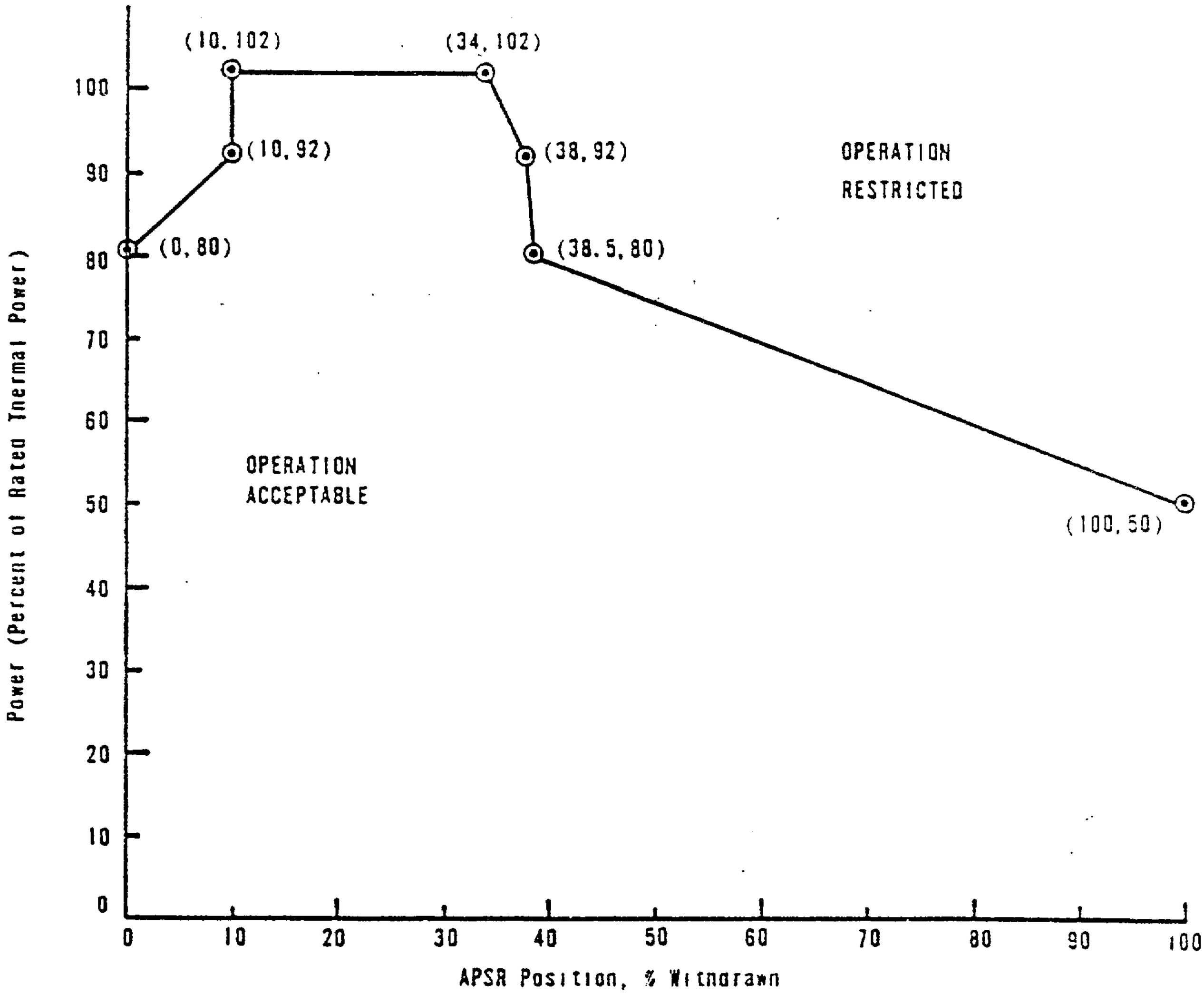
OPERATIONAL POWER
IMBALANCE ENVELOPE
AFTER 200 +10/-10 EFPD
UNIT 3
OCONEE NUCLEAR STATION
Figure 3.5.2-12
(2 of 2)



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)

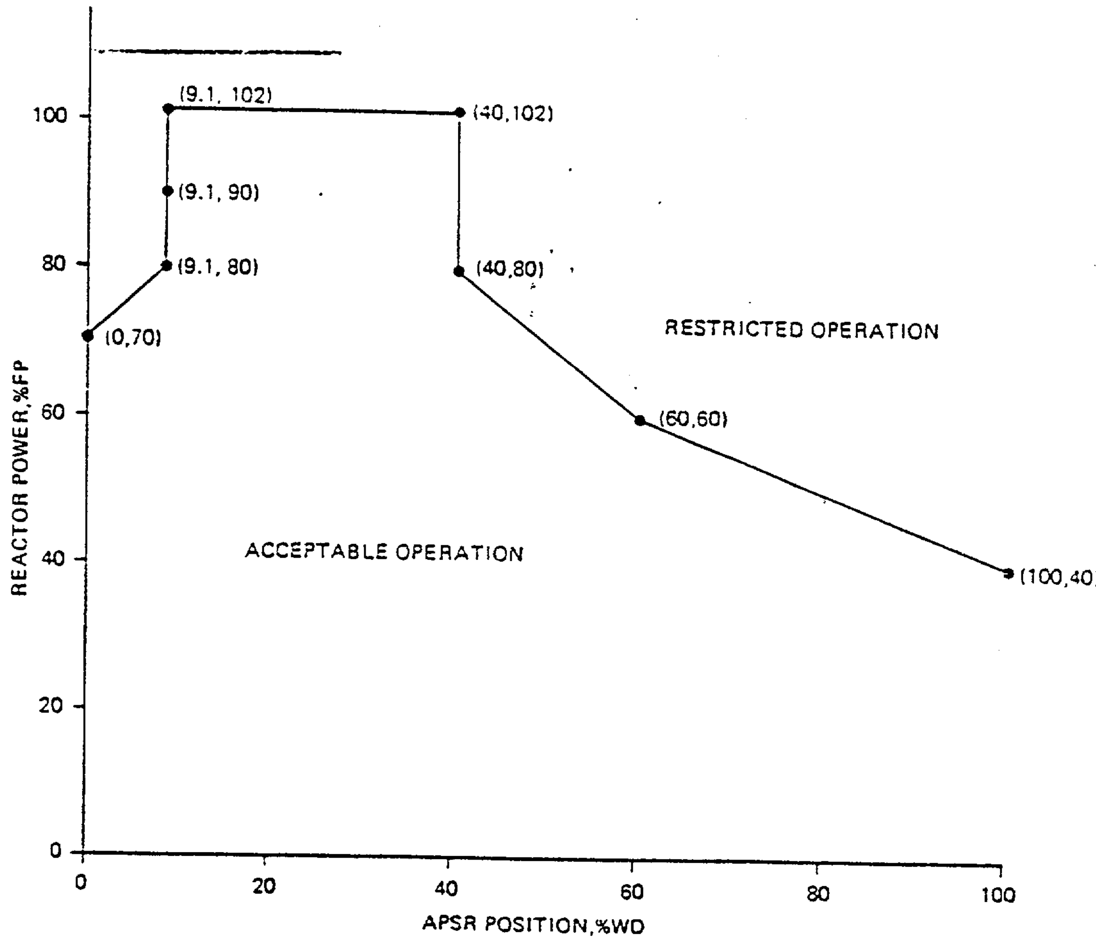


APSR POSITION LIMITS
FOR OPERATION
FROM 26 +10/-0 TO 200 ±10 EFPD
UNIT 1
OCONEE NUCLEAR STATION
Figure 3.5.2-13
(2 of 3)

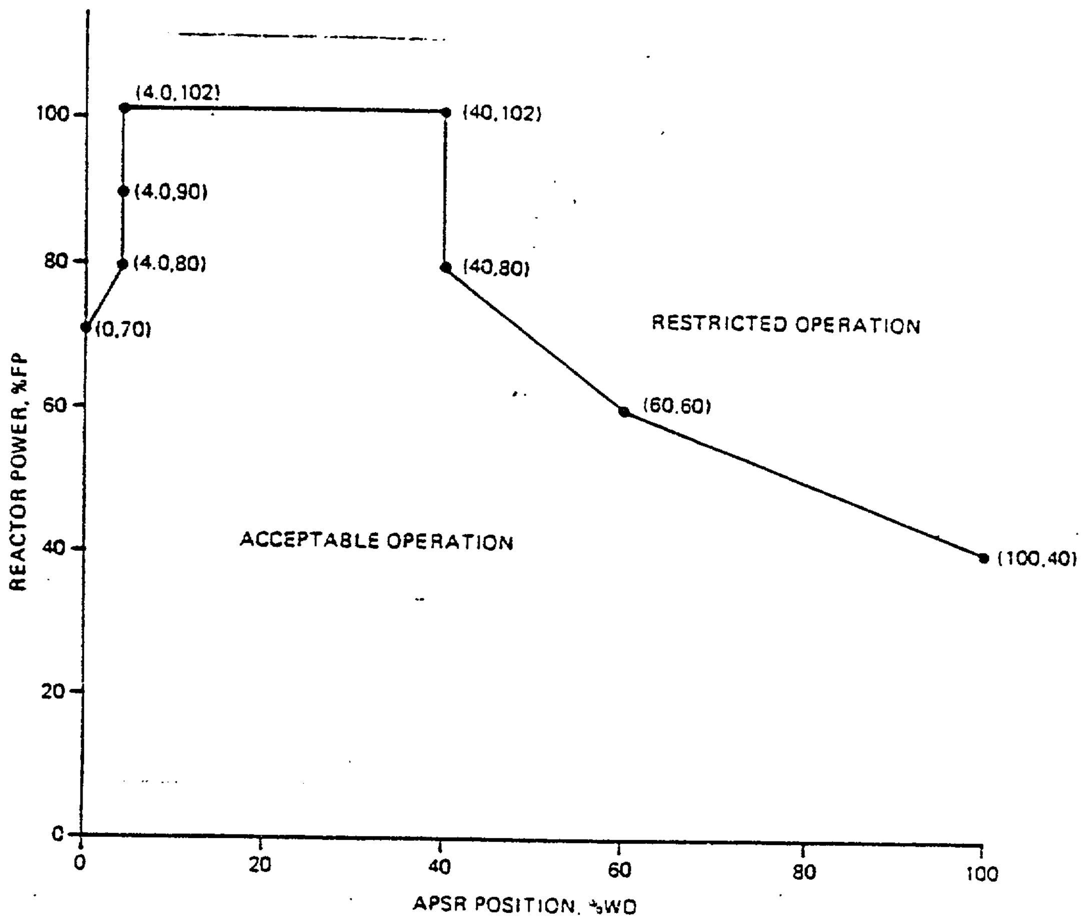


APSR POSITION LIMITS
FOR OPERATION
AFTER 200 ±10 EFPD
UNIT 1

OCONEE NUCLEAR STATION
Figure 3.5.2-13
(3 of 3)

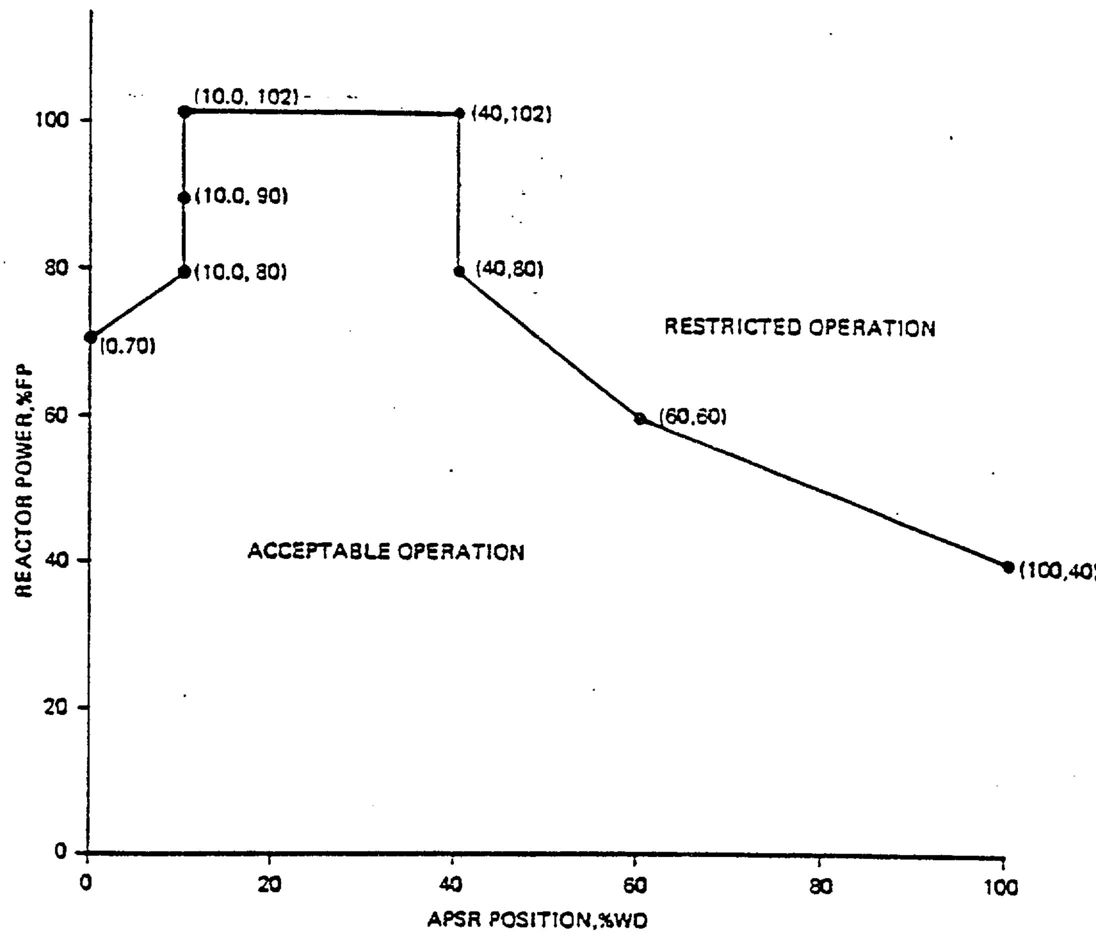


APSR POSITION LIMITS
FOR OPERATION
FROM 0 EFPD TO EOC
UNIT 2
OCONEE NUCLEAR STATION
Figure 3.5.2-14
(1 of 1)



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

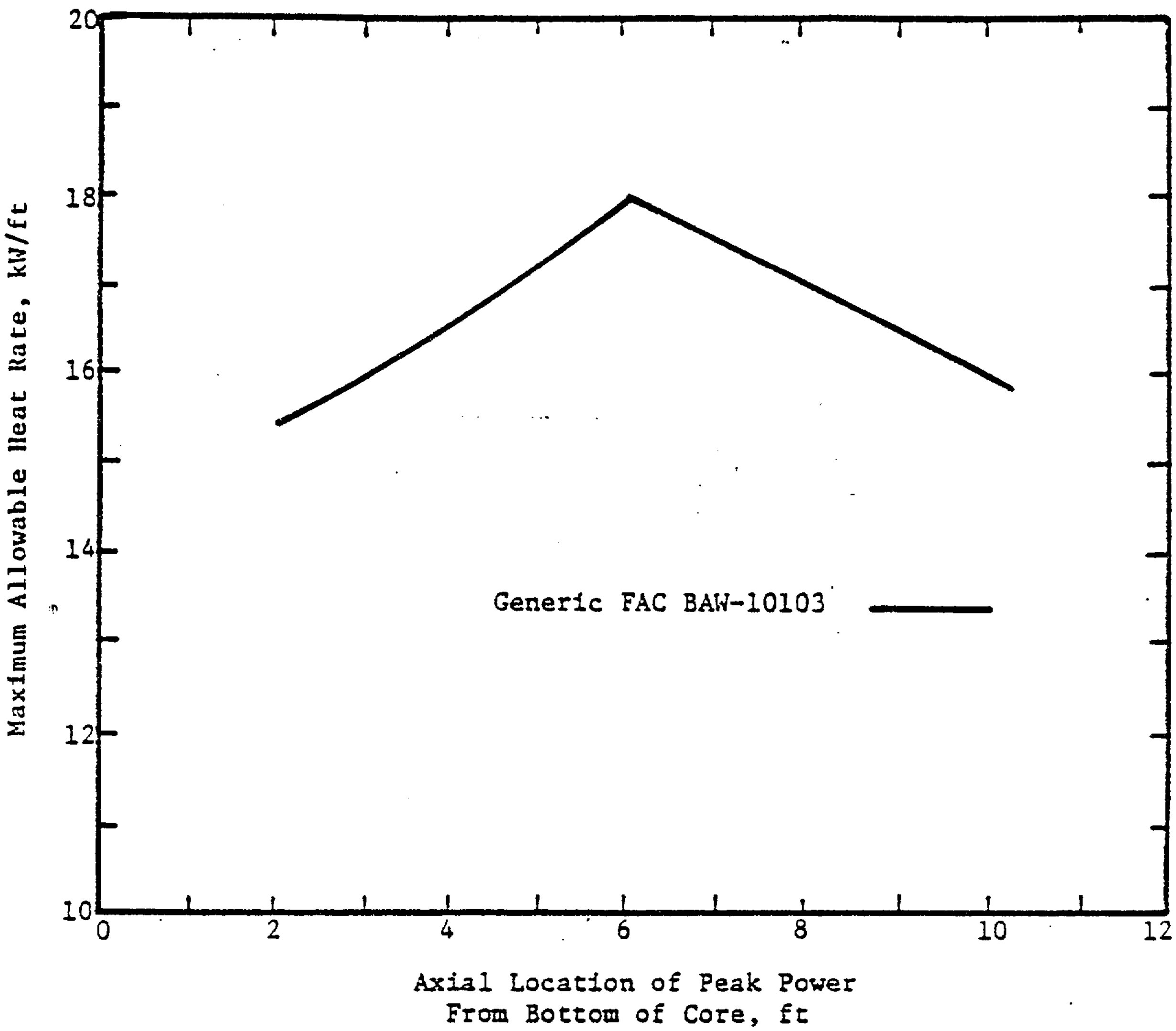
OCONEE NUCLEAR STATION
Figure 3.5.2-15
(1 of 2)



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

DUKE POWER

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.

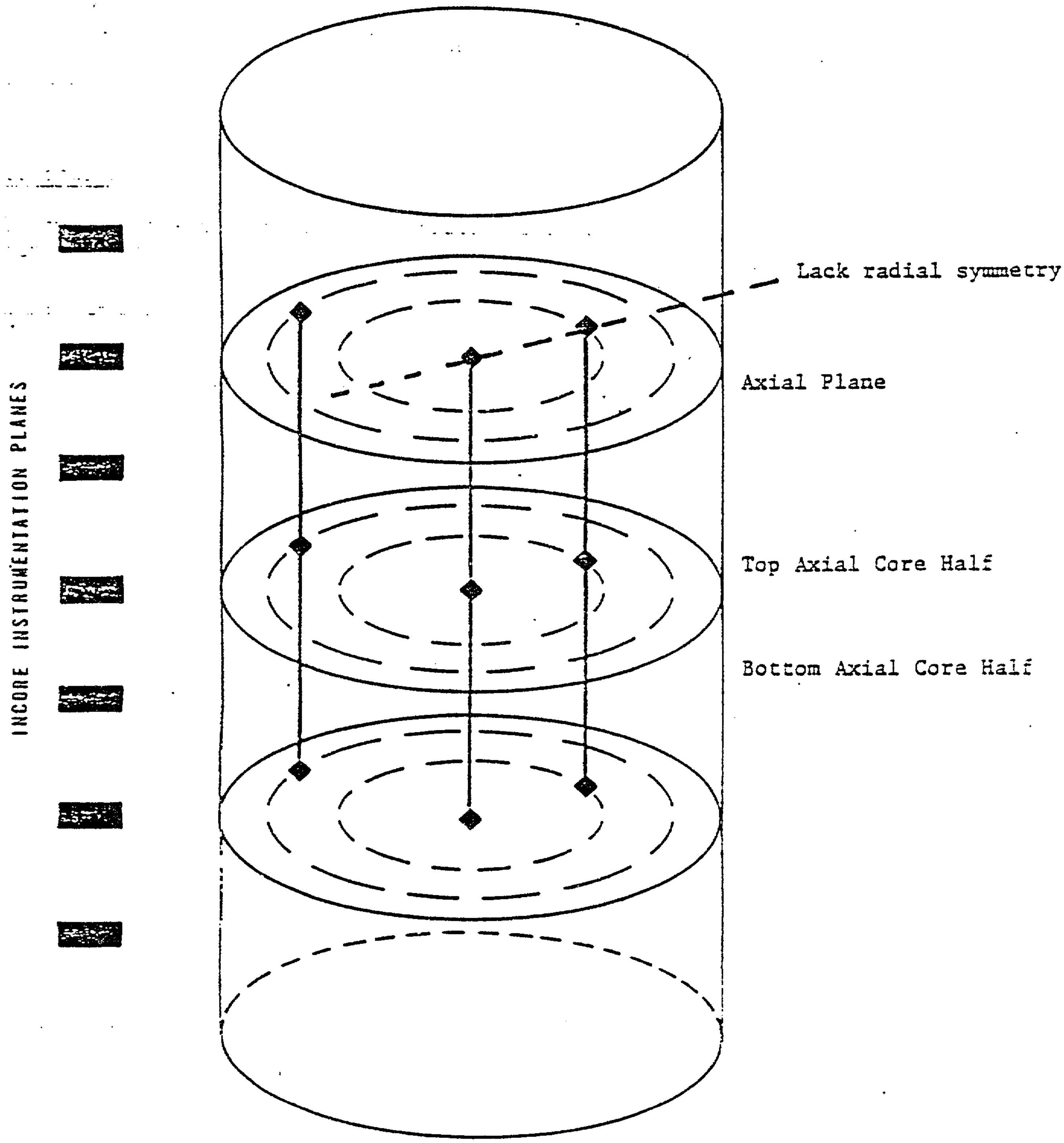
3.5.4.2 If requirements of 3.5.4.1 are not met, power shall be reduced below 80 percent of the power allowable for the existing reactor coolant pump combination within eight hours and incore detector measurements shall not be used to determine axial imbalance or quadrant power tilt.

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.

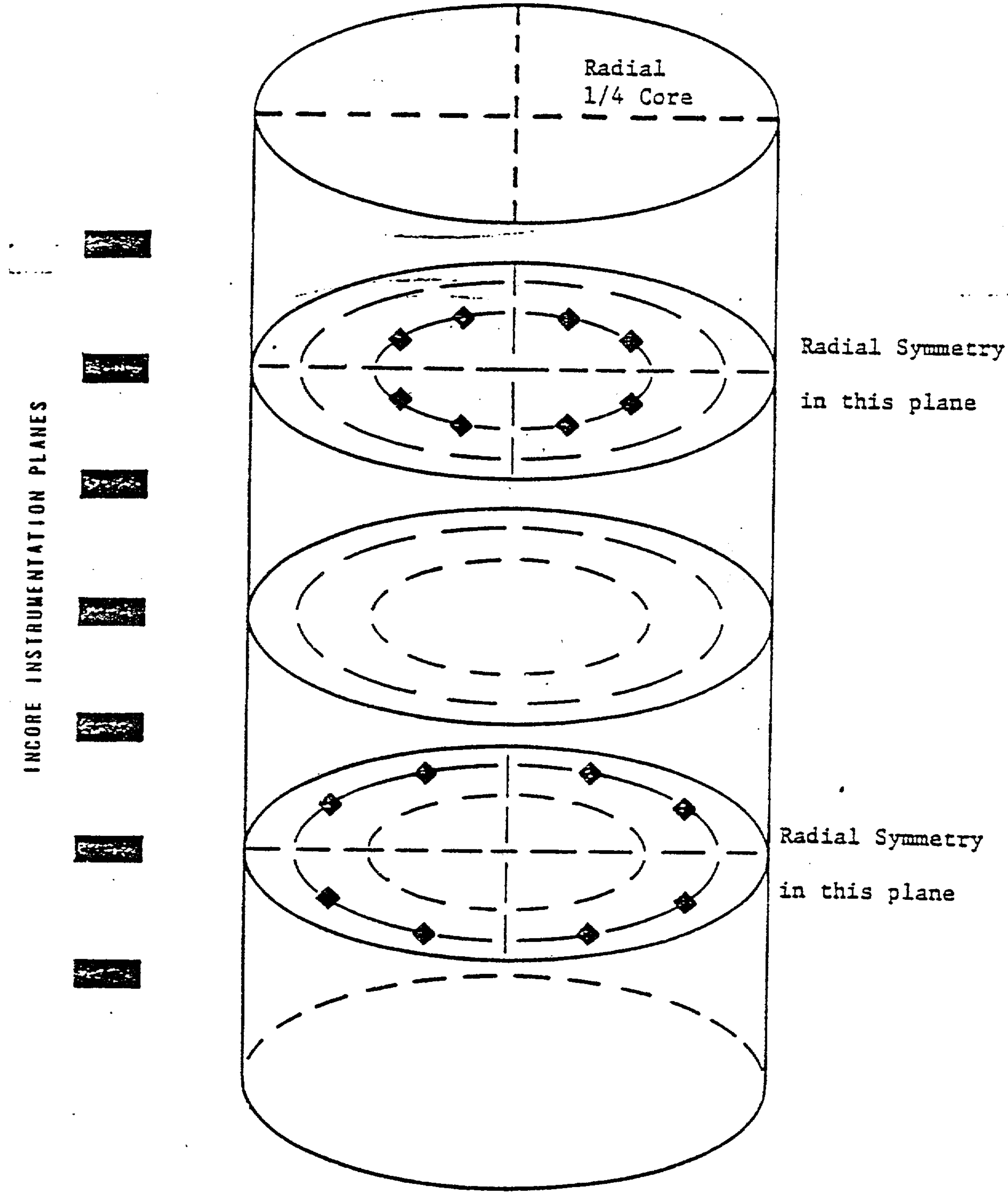
The safety of reactor operation at or below 80 percent of the power allowable for the reactor coolant pump combination(1) without the axial imbalance trip system has been determined by extensive 3-D calculations, and was verified during the physics startup testing program.

(1) FSAR, Section 5.1.2.3



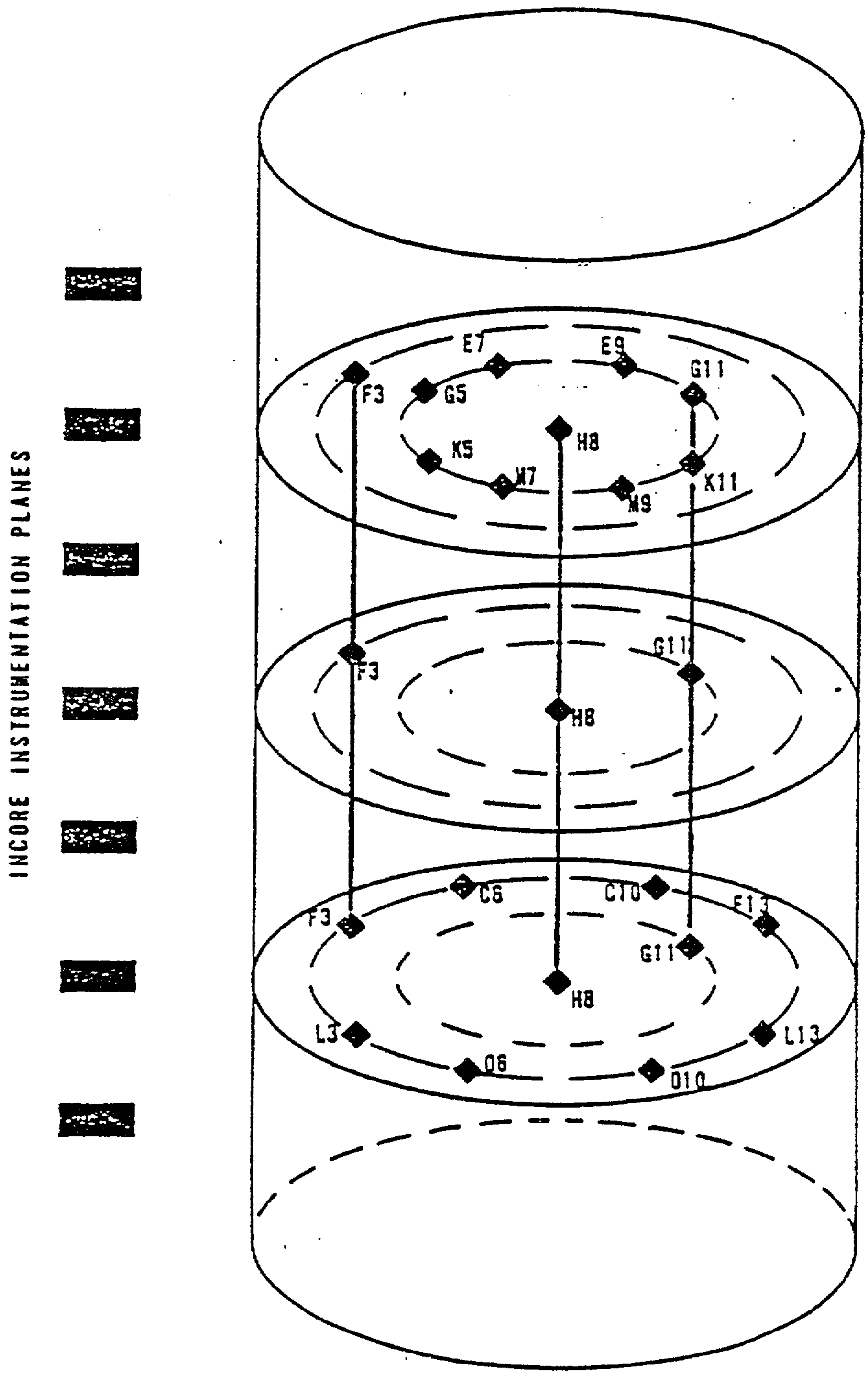
INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION

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 analyzed 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 analyzed 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-38
AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-47
AMENDMENT NO. 126 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 February 13, 1984, (Ref. 1) Duke Power Company (the licensee) proposed changes to the Technical Specifications (TSs) of Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments would consist of changes to the Station's common TSs.

These amendments would authorize proposed changes to the Oconee Nuclear Station (ONS) Technical Specifications (TSs) which are required to support the operation of Oconee Unit 3 at full rated power during Cycle 8. The proposed changes include the core protection safety limits (TS section 2.1), the protective system maximum allowable setpoints (TS section 2.3), and the rod position limits (TS section 3.5.2), as well as the administrative renumbering of the figures in the TS section 3.5.2. To support the application, the licensee submitted report DPC-RD-2003, "Oconee Unit 3, Cycle 8 Reload Report" (Ref. 2) as an attachment to Reference 1.

The Cycle 8 core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 8 is to have a length of approximately 400 effective full power days (EFPD) of operation. As has been the case for Cycle 7, Cycle 8 will be operated in a rods-out, feed-and-bleed mode. Specific aspects of the Oconee Unit 3 Cycle 8 reload are discussed in the following sections.

2.0 EVALUATION OF FUEL SYSTEM DESIGN

The analytical methods used in the safety analysis of the proposed eighth cycle of operation at Oconee 3 are described in the Duke Power Company Oconee Nuclear Station Reload Design Methodology Report (Ref. 3) which has been reviewed and approved by the NRC staff (Ref. 4). Although the methodology report continues to rely on a number of analytical methods developed by the fuel vendor, Babcock and Wilcox, this is not always the case. Where methods used in the Cycle 8 analysis are unchanged from those described in the Methodology Report, we have concluded that additional review is unnecessary for Cycle 8 operation. Also, where conditions are identical or limited by the analysis of a previous cycle of operation, the evaluation of that cycle continues to apply.

2.1 Fuel Assembly Design

Although all batches in the Oconee 3 Cycle 8 core will utilize the same Babcock and Wilcox 15x15 fuel design, the Batch 9 and 10 assemblies will be of the Mark B5 fuel design. The Mark B5 fuel assembly is identical to the Mark B4 except its upper end fitting has been redesigned to provide a positive holddown of fixed control components such as burnable poison rod assemblies (BPRAs), neutron source rod assemblies, and orifice rod assemblies. Two regenerative neutron sources will be used in Mark B4 and B5 assemblies. The Mark B4 and B5 assembly designs were reviewed and found acceptable for previous B&W 177 FA reloads and are therefore acceptable for Oconee 3 Cycle 8.

2.2 Fuel Rod Design

The Oconee 3 Cycle 8 core contains both Mark B4 and Mark B5 fuel assemblies and the fuel rods used in both of these assemblies are virtually identical. While the results of the linear-heat-rate-to-melt analysis (Table 4.2 of Reference 2) indicate some variation in densification characteristics for the different batches, the resulting linear heat rate values are the same for all batches in the Cycle 8 core.

2.2.1 Cladding Collapse

The licensee has stated that the cladding collapse time for the most limiting Cycle 8 assembly was conservatively determined to be greater than the maximum projected residence time for any Cycle 8 assembly. The creep collapse analysis used the CROV computer code (Ref. 5) using input conditions from TACO-2 (Ref. 6) in a manner described in the Reload Methodology Report (Ref. 3). All of these methods have been reviewed and approved by the NRC staff. We conclude that cladding collapse has been appropriately considered for Cycle 8 operation.

2.2.2 Cladding Strain

The licensee has performed a cladding strain calculation using TACO-1 (Ref. 9) in accordance with methods and limits given in the Reload Methodology Report (Ref. 3). It is concluded that the analysis of cladding strain has been appropriately considered for Cycle 8 operation.

2.2.3 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (Ref. 8) was previously cited as a source of acceptance criteria used to establish the design bases and evaluation of the fuel system. Among those criteria which may affect the operation of the fuel rod is the internal pressure limit. The pressure criterion (SRP 4.2, Section II.S.1(f)) is a requirement that the fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. Based on a TACO-1 analysis, the licensee has stated that fuel rod internal pressure will not exceed system pressure during normal operation for Cycle 8. We find this acceptable and conclude that the rod internal pressure limits have been adequately considered for Cycle 8 operation.

2.3 Fuel Thermal Design

There are no major changes in the physical characteristics of the Cycle 8 core which would result in altered thermal conditions. As pointed out in Section 2.2 of this report, the linear-heat-rate-to-melt for all batches in the Cycle 8 core is the same. The linear-heat-rate-to-melt capability was determined separately for Batches 8B and 9 using TACO-1 and for Batch 10 using TACO-2. The centerline melt limits are generated at both low and high burnup conditions and the linear heat rate capability is both batch and burnup dependent. All values given in Table 4.2 of the reload report (Ref. 2) are higher (less limiting) than those used in the Oconee 3 Cycle 7 reload. These values have been incorporated into the proposed Technical Specifications and we find them acceptable.

2.3.1 LOCA Initial Conditions

A combination of TAFY and TACO-2 analyses were used to generate the LOCA limits as described in Tables 7-2 and 7-3 of Reference 2. Three sets of bounding values for allowable LOCA peak linear heat rates are given as a function of core height. These limits apply during the periods 0-25 EFPD, 25-65 EFPD and 65 EFPD to end-of-cycle. These limits have been incorporated into the Technical Specifications for Cycle 8 through the operating limits on rod index and axial power imbalance. It is concluded that the initial thermal conditions for LOCA analysis have been appropriately considered for Cycle 8 operation.

2.3.2 Fuel Rod Bowing

The licensee has determined a fuel rod bowing gap closure correlation for use in the calculation of the rod bowing penalty as described in Reference 10. It is concluded that this correlation adequately accounts for gap closure as a function of burnup in the Mark B fuel design. The rod bowing penalty is discussed in the Thermal Hydraulic Design section of this report.

2.4 Operating Experience

Babcock and Wilcox has accumulated operating experience with the Mark B4 15x15 fuel assembly at all of the eight operating B&W 177-fuel assembly plants and Mark B5 experience during Cycle 7 of Oconee 3. A summary of this operating experience is provided as part of our fuel operating experience report (Ref. 7).

2.4.1 Holddown Spring Failures

It has been noted during previous Oconee reload reviews (e.g., Ref. 11) that a small number of holddown spring failures are continuing to occur at the Oconee station. These springs are contained in the upper end fitting of the Mark B4 fuel assembly and are used to accommodate length changes due to thermal expansion and irradiation growth while providing a positive holddown force for the assembly. On May 14, 1980, a failed holddown spring was discovered by remote video inspection at Davis-Besse Unit 1. Further examination ultimately identified a total of 19 failed springs at that plant. Subsequent examination of spent fuel assemblies at other B&W reactors, including the Oconee station, revealed a small number of similar failures.

An inspection (Ref. 12) of all Oconee Unit 3 Cycle 6 assemblies revealed broken holddown springs in two assemblies due to be discharged. Another inspection (Ref. 13) revealed one broken holddown spring in Unit 1 Batch 4 fuel and three broken holddown springs in Oconee Unit 2 Batch 7 fuel.

More recently, four additional broken holddown springs were found in Unit 1 (Ref. 14). In all cases, the fuel was due to be discharged or the holddown springs were replaced prior to reinsertion. It has been concluded that a continuing program of detection and discharge/replacement of failed holddown springs is necessary to minimize the probability of operating with broken holddown springs for the Mark B fuel design.

2.5 Conclusions

We have reviewed those sections of the reload report for Oconee Unit 3 Cycle 8 dealing with the fuel system design and find those portions of the application acceptable.

3.0 EVALUATION OF NUCLEAR DESIGN

The nuclear design parameters characterizing the operation of Oconee Unit 3 Cycle 8 have been obtained with the Duke Power physics calculational methods (Ref. 3). These methods have been approved for use in reload design calculations (Ref. 4) and were used previously in deriving the Cycle 7 nuclear design parameters. The Cycle 8 core will contain 68 fresh assemblies with a U-235 initial enrichment of 3.28%. In addition to the 68 fresh assemblies, there are two batches of exposed assemblies: a batch of 37 assemblies having an initial U-235 enrichment of 3.07% and a batch of 68 assemblies having an initial enrichment of 3.18%. Four fresh assemblies are located in the central core region with the remaining fresh assemblies distributed in a checkerboard pattern in the surrounding annular region. No fresh assemblies are loaded in the outermost peripheral ring. This is characteristic of all current extended burnup PWR reloads. The excess reactivity is controlled by soluble boron which is supplemented by 61 full-length Ag-In-Cd control rods and 60 BPRAs. Furthermore, eight partial length axial power shaping rods (APSRs) are provided for additional control of axial power distribution. All safety criteria are met. Shutdown margin values at beginning and end of cycle are 4.14% and 2.73% $\Delta k/k$, respectively, compared to the minimum required value of 1.0 percent. Beginning of cycle radial power distributions show acceptable margins to limits. 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 Cycle 8 is acceptable.

4.0 EVALUATION OF THERMAL-HYDRAULIC DESIGN

The objective of this review is to confirm that the thermal-hydraulic design of the reload core has been accomplished using acceptable methods and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal and anticipated operational transients. The reload design methodology is described in Reference 3 and has been approved (Ref. 4). Discussion of the main differences between Cycle 8 and Cycle 7 follows.

4.1 Core Bypass Flow

The incoming Batch 10 fuel is hydraulically and geometrically similar to the fuel remaining in the core from the previous cycles. For Cycle 8 operation, 60 BPRAs will be inserted, and two assemblies will contain regenerative neutron sources, leaving 46 open assemblies, resulting in an increase in calculated maximum core bypass flow of 7.9 percent compared with 7.6 percent for Cycle 7. The bypass flow of 7.9 percent is less than the 8.2 percent assumed in the generic thermal-hydraulic design analysis (Ref. 2), and the consequent increase in Cycle 8 core flow relative to the generic analysis value establishes the generic analysis as conservative for Cycle 8 operation.

4.2 DNBR Penalty Due to Rod Bow

A B&W topical report (Ref. 10) discussing the mechanisms and resulting effects of bowing in B&W fuel has been reviewed and approved (Ref. 15). The report concludes that the DNBR penalty due to rod bow need not be imposed for those assemblies with significant bow because the power production capability of the fuel decreases sufficiently with irradiation to offset the effects of bowing. Therefore, no rod bow penalty needed to be considered for Cycle 8 operation. We conclude that the available margin for Cycle 8 has been demonstrated and that the thermal-hydraulic design is, therefore, acceptable.

4.3 Conclusions

The pertinent thermal-hydraulic parameters summarized in Table 6-1 of the submittal are identical except for the core bypass flow of 7.9 percent of total flow for Cycle 8 as compared to 7.6 percent for Cycle 7 and 8.2 percent for the generic analysis. The decrease of bypass flow, relative to the generic analysis value, resulting in a net increase in core flow indicates that, with other parameters unchanged, the safety margin for Cycle 8 is comparable to that of the generic analysis. The reload design methodology for Cycle 8 included in Reference 3 has been approved as indicated in Reference 4. We conclude from the examination of the Cycle 8 core thermal-hydraulic design, with respect to the FSAR values, that the core reload will not adversely affect the capability to operate Oconee Unit 3 safely during Cycle 8 and that the proposed changes to Technical Specifications discussed in Section 6.0 of the submittal are acceptable.

5.0 ACCIDENT ANALYSIS

The important kinetics parameters for Cycle 8 are compared to the values used in the FSAR in Table 7.1 of the reload submittal (Ref. 2). For the parameters quoted, the Cycle 8 values are bounded by those used previously. The licensee has also determined that the initial conditions of the transients in Cycle 8 are bounded by the FSAR and/or the fuel densification report (Ref. 16). Since the Batch 10 reload fuel contains rods with a theoretical density higher than that considered in the densification report, the conclusions in Ref. 16 are still valid. These analyses have been previously accepted by the NRC.

The licensee's Reload Methodology Technical Report (Ref. 3), which has been accepted by the NRC staff (Ref. 4), was examined vis-a-vis the Accident Analysis Review process. Of the items contained in the "Key Safety Parameter Checklist" (Table 8-1, Ref. 3), virtually all are addressed in Table 7-1 and other tables in the submittal. It should be noted, however, that the "Minimum Tripped Rod Worth" available in case of a steamline break is not given and therefore cannot be compared to the value assumed in the FSAR analysis. However, the total available worth and the shutdown margin for Cycle 8 presented in Table 5-2 of Ref. 2 are larger, and hence more conservative, than the corresponding values for Cycle 7. In addition, the values in Table 5-1 for the effective delayed neutron fraction are lower than the nominal values assumed in the FSAR analysis of the rod ejection accident (REA). While this would tend to increase the maximum fuel enthalpy associated with a postulated REA event, the maximum ejected rod worth at HFP for Cycle 8 is so much lower than that assumed in the FSAR analysis that it offsets this non-conservatism.

Three sets of bounding values for allowable LOCA peak linear heat rates are given as a function of core height. These limits apply during the periods 0 - 1000 MWD/MTU, 1000 - 2600 MWD/MTU, and for the balance of the cycle. These results are based upon a bounding analytical assessment of NUREG-0630 on LOCA and operating linear heat rate limits performed by Babcock & Wilcox (Ref. 17). The B&W analyses have been approved by the NRC staff and the three sets of limits were accepted in conjunction with the review of the Oconee Unit 2 Cycle 7 reload submittal (Refs. 18 and 11).

New dose calculations were not performed for Oconee 3 Cycle 8. The licensee has determined that the dose considerations for Oconee 1 Cycle 8 (Ref. 19) are characteristic for Oconee 3 Cycle 8 based on comparisons of key parameters which determine radionuclide inventories. Therefore, it is acceptable.

6.0 TECHNICAL SPECIFICATION MODIFICATIONS

Oconee Unit 3 Cycle 8 Technical Specifications have been modified to account for (i) minor changes in power peaking and control rod worths during Cycle 8 operation, (ii) incorporation of NUREG-0630 data (Ref. 17) in the LOCA analysis and (iii) employment of a Monte Carlo simulation technique in determining instrument string errors (Refs. 18 and 11). We have reviewed the proposed Specification revisions for Cycle 8. These changes concern the (1) Core Protection Safety Limits of Specification 2.1, (2) Protective System Maximum Allowable Setpoints of Specification 2.3 and (3) Rod Position Limits of

Specification 3.5.2. The limiting safety system settings and the limiting conditions for operation have been established by approved methods. Changes which reflect the core thermal-hydraulic response continue to maintain the safety limit DNBR criterion of 1.30. The control rod withdrawal limits for the various pump combinations and times in core life are presented as well as part length axial power shaping rod position limits. On the basis that previously approved methods were used to obtain the limits, we find these Technical Specification modifications acceptable.

Selected Technical Specification changes for Oconee Unit 1 and 2 were also included in the Oconee Unit 3 Cycle 8 submittal. These changes are administrative only, i.e., figure and page numbering changes, and are, therefore, acceptable.

7.0 START-UP TESTING

The startup testing program for Oconee Unit 3 Cycle 8 will be carried out in accordance with approved methods and procedures.

8.0 EVALUATION FINDINGS

We have reviewed the fuels, physics, thermal-hydraulic and transient information presented in the Oconee 3 Cycle 8 reload report. We find the proposed reload and the associated modified Technical Specifications acceptable.

9.0 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.

10.0 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: May 15, 1984

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