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

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

Dear Mr. Tucker:

The Commission has issued the enclosed Amendments Nos. 124, 124, and 121 to Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications (TSs) in response to your request dated September 1, 1983, as supplemented by letter dated September 14, 1983.

These amendments revise the TSs to allow full power operation of Oconee Unit 2 during fuel Cycle 7. These revisions involve core protection safety limits, protective system maximum allowable setpoints, and rod position limits.

We are handling separately the necessary actions on your September 14, 1983, request for a Unit 3 TS change to the Axial Power Shaping Rod Position Limits.

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

Sincerely,

ORIGINAL SIGNED BY
JOHN F. STOLZ

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

Enclosures:
1. Amendment No. 124 to DPR-38
2. Amendment No. 124 to DPR-47
3. Amendment No. 121 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

ORB#4:DL	ORB#4:DL	ORB#4:DL	AD:OR:DL	OELD
RIngram	JSuermann;cf	JStolz	GLainas	
11/21/83	11/21/83	11/21/83	11/21/83	11/22/83

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PDR ADOCK 05000269
P PDR

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. 124
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 September 1, 1983, as supplemented September 14, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 124 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1983

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124
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 September 1, 1983, as supplemented September 14, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 124 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1983

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
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 September 1, 1983, as supplemented September 14, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 121 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1983

ATTACHMENTS TO LICENSE AMENDMENTS

AMENDMENT NO. 124 TO DPR-38

AMENDMENT NO. 124 TO DPR-47

AMENDMENT NO. 121 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
viii	viii
ix	ix
x	x
-	xi
2.1-3b	2.1-3b
2.1-5	2.1-5
2.1-8	2.1-8
2.1-11	2.1-11
2.3-2	2.3-2
2.3-3	2.3-3
2.3-9	2.3-9
2.3-11	2.3-11
2.3-12	2.3-12
2.3-13	2.3-13
3.5-9	3.5-9
3.5-10	3.5-10
3.5-11	3.5-11
3.5-16	3.5-16
3.5-16a	3.5-16a
3.5-16b	3.5-16b
3.5-19	3.5-19
3.5-19a	3.5-19a
3.5-19b	3.5-19b
3.5-19c	3.5-19c
3.5-19d	3.5-19d
3.5-19e	3.5-19e
3.5-22	3.5-22
3.5-22a	3.5-22a
3.5-22b	Deleted
3.5-25	3.5-25
3.5-25a	Deleted
3.5-25b	Deleted

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3.5.2-2C6	Rod Position Limits for 2 Pump Operation - Unit 3	3.5-20e
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3.5.2-3A3	Operational Power Imbalance Envelope - Unit 1	3.5-21b
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INTRODUCTION

These Technical Specifications apply to the Oconee Nuclear Station, Units 1, 2, and 3 and are in accordance with the requirements of 10CFR50, Section 50.36. The bases, which provide technical support or reference the pertinent FSAR section for technical support of the individual specifications,are included for informational purposes and to clarify the intent of the specification. These bases are not part of the Technical Specifications, and they do not constitute limitations or requirements for the licensee. The Technical Specifications while applying to Units 1, 2, and 3 are written on a single unit basis; exceptions to this are identified.

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

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.4 kw/ft for fuel rod burnup less than or equal to 10,000 MWD/MTU and 21.2 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 2.1-3B correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

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

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

The maximum thermal power for three-pump operation is 88.07 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.07 = 79.92 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions is produced in a similar manner.

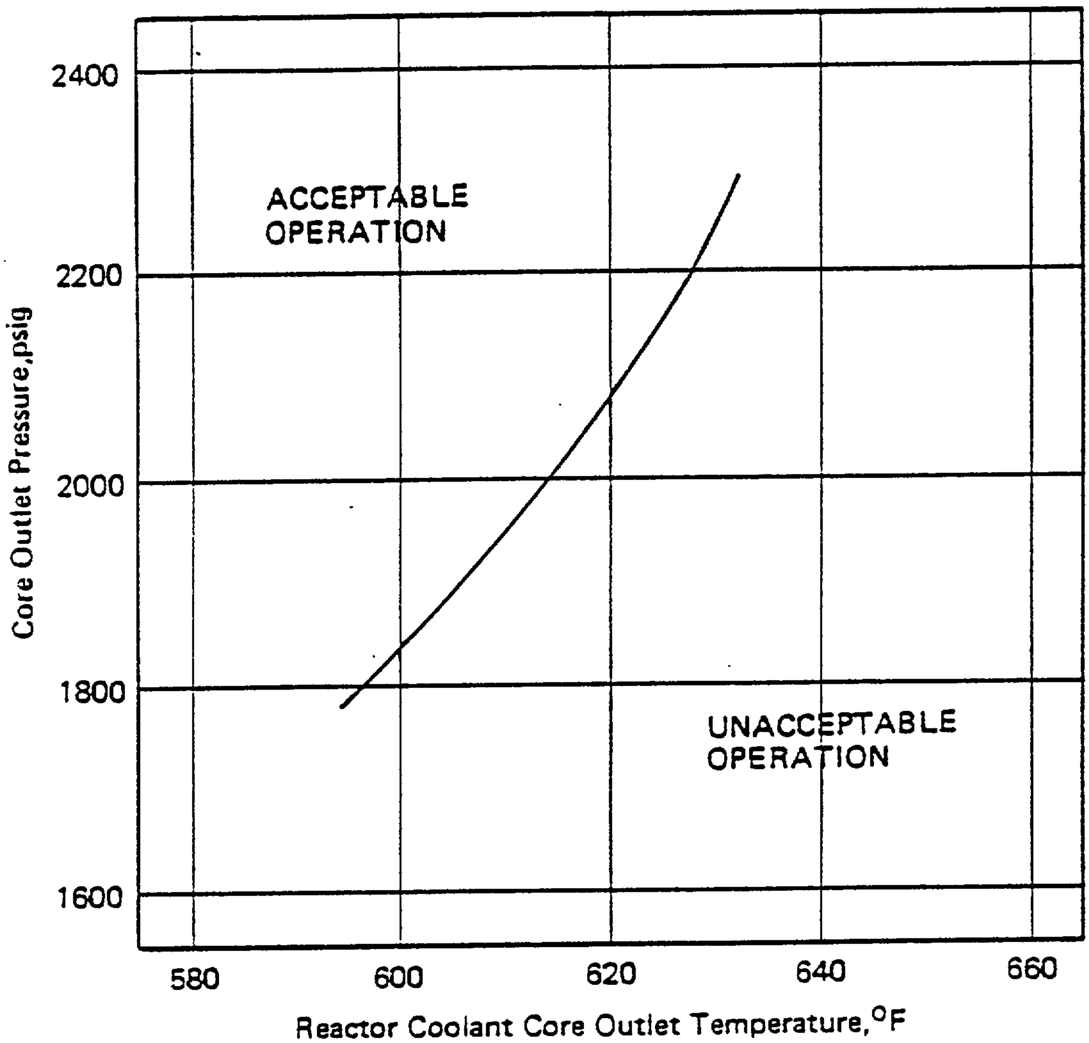
For each curve of Figure 2.1-3B, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 22 percent for that particular reactor coolant pump situation. The curve of Figure 2.1-1B is the most restrictive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

References

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

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

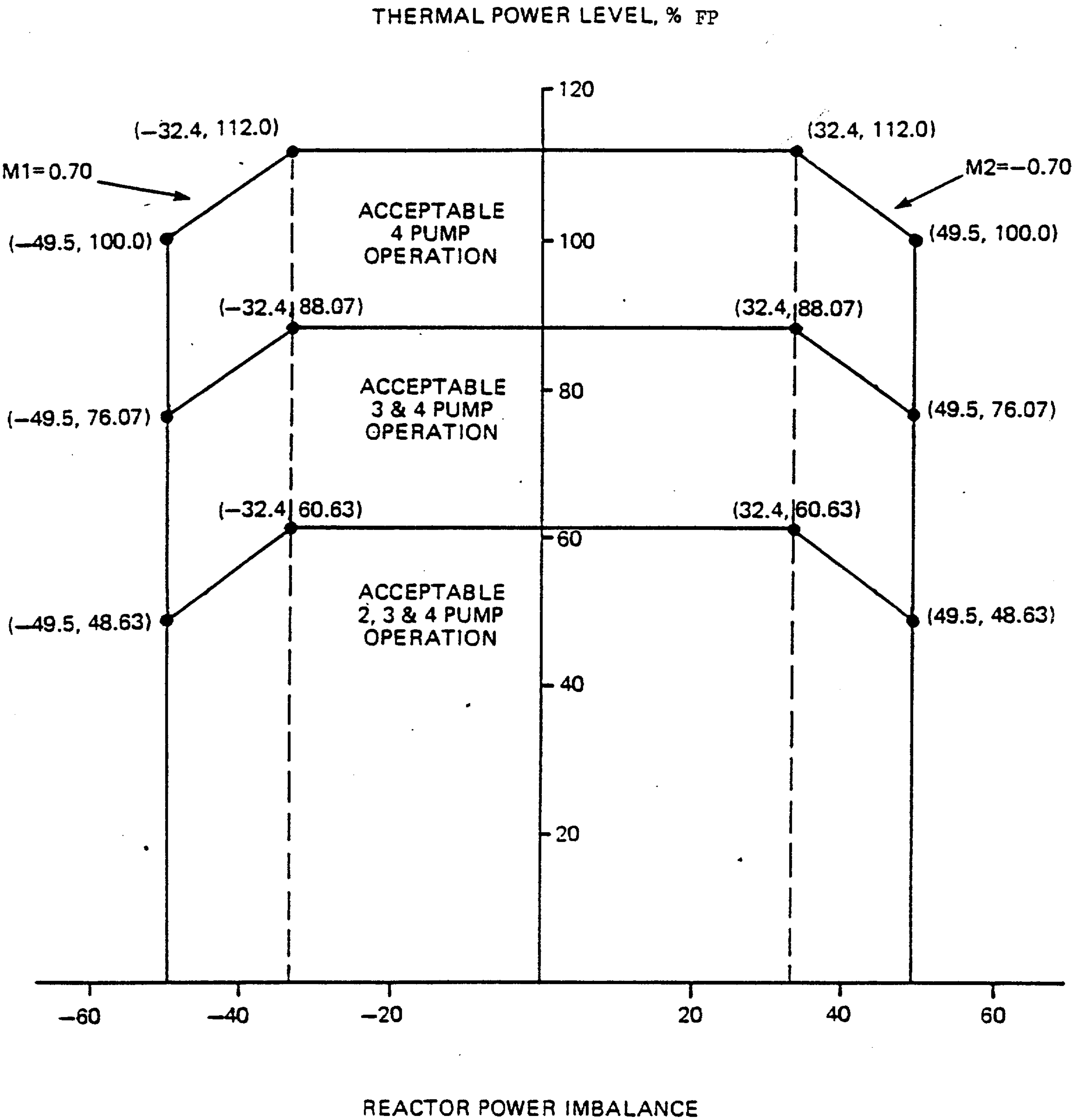
(3) Oconee 2, Cycle 7 - Reload Report, DPC-RD-2002, September 1983.



CORE PROTECTION SAFETY LIMITS
UNIT 2

OCONEE NUCLEAR STATION

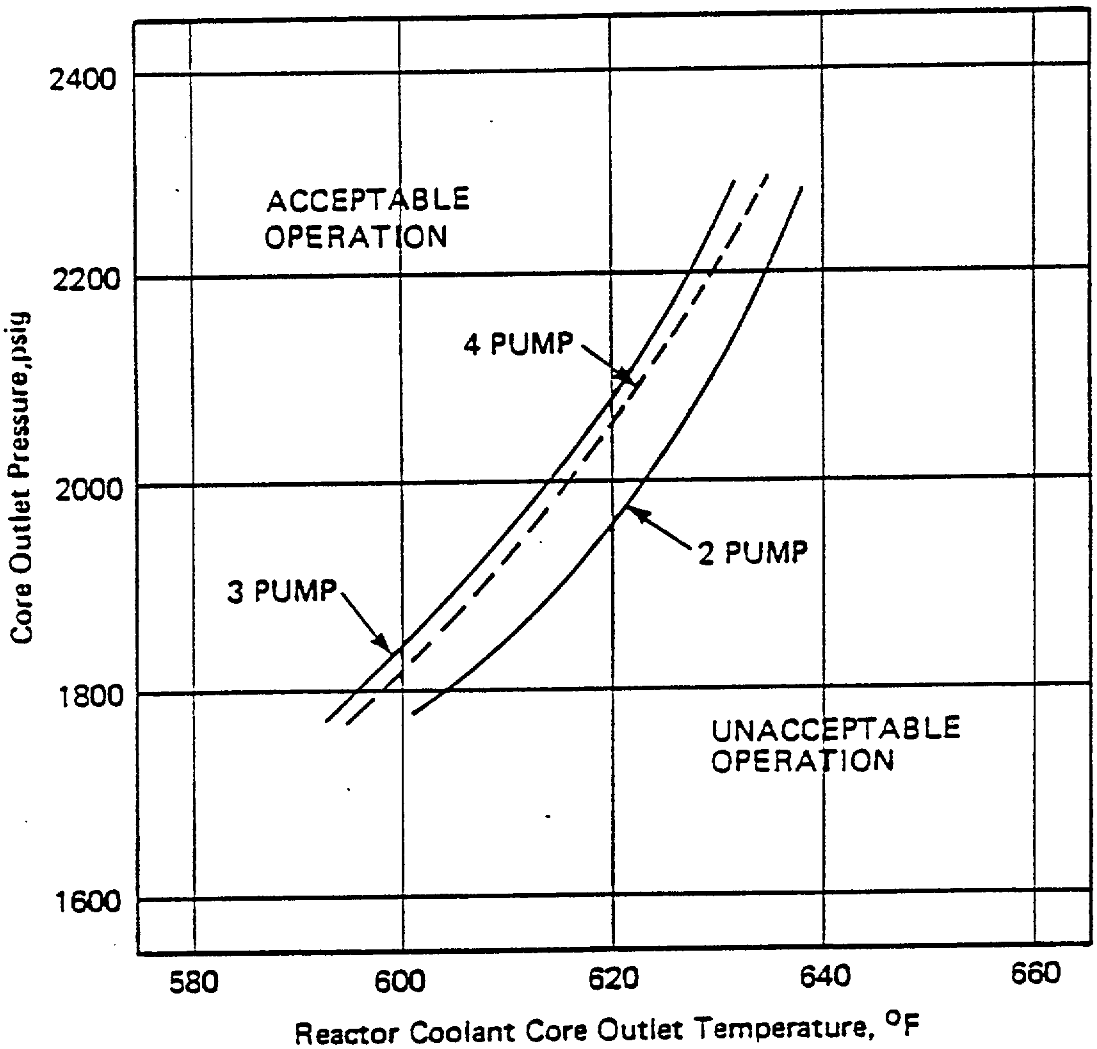
Figure 2.1-1B



CORE PROTECTION SAFETY LIMITS
UNIT 2

OCONEE NUCLEAR STATION

Figure 2.1-2B



PUMPS OPERATING	COOLANT FLOW (GPM)	POWER (% FP)	TYPE OF LIMIT
4	374,880(100%)	112.0	DNBR
3	280,035(74.7%)	90.7	DNBR
2	183,690(49.0%)	63.63	DNBR/QUALITY

CORE PROTECTION SAFETY LIMITS
UNIT 2

OCONEE NUCLEAR STATION

Figure 2.1-3B

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 units 1 and 3, the maximum calibration and instrument errors are algebraically summed to determine the string errors in the safety calculations. Unit 2 employs 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

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

Pump Monitors

The pump monitors prevent the minimum core DNBR from decreasing below 1.3 by tripping the reactor due to the loss of reactor coolant pump(s). The circuitry monitoring pump operational status provides redundant trip protection for DNB by tripping the reactor on a signal diverse from that of the power-to-flow ratio. The pump monitors also restrict the power level for the number of pumps in operation.

Reactor Coolant System Pressure

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

The low pressure (1800) psig and variable low pressure ($11.14\ T_{out}-4706$) trip
(1800) psig ($11.14\ T_{out}-4706$)
(1800) psig ($11.14\ T_{out}-4706$)
setpoints shown in Figure 2.3-1A have been established to maintain the DNB
2.3-1B
2.3-1C
ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

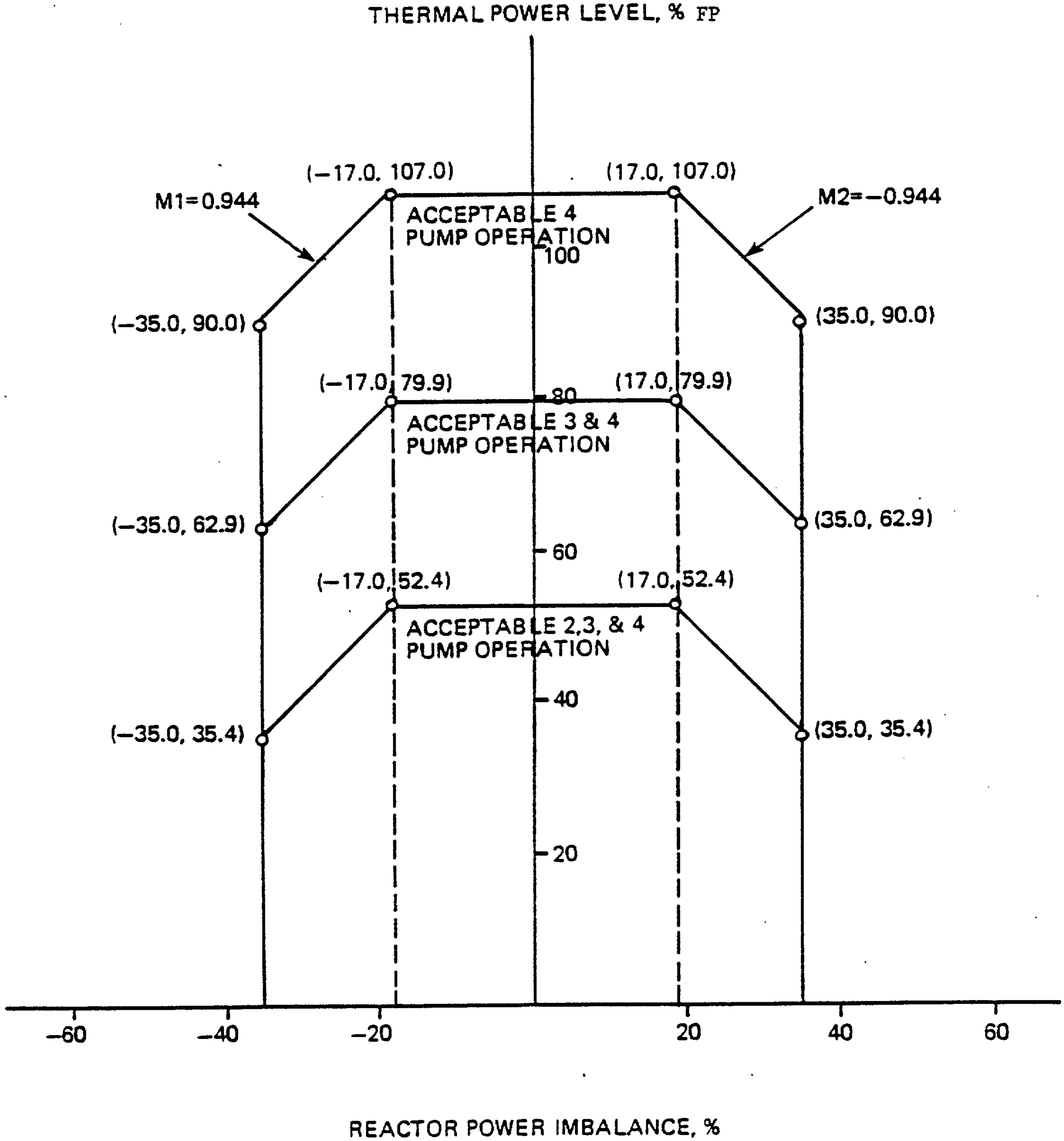
Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($11.14\ T_{out} - 4746$)
($11.14\ T_{out} - 4746$)
($11.14\ T_{out} - 4746$)

Coolant Outlet Temperature

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

Reactor Building Pressure

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



PROTECTIVE SYSTEM
MAXIMUM ALLOWABLE SETPOINTS
UNIT 2

OCONEE NUCLEAR STATION

Figure 2.3-2B

Table 2.3-1A
Unit 1

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating In Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0[3]
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720[4]
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)$[1]	$(11.14\ T_{out} - 4706)$[1]	$(11.14\ T_{out} - 4706)$[1]	Bypassed
7.	Reactor Coolant Temp. F., Max.	618	618	618	618
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

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

(2) Reactor Coolant System Flow, %.

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

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

Table 2.3-1B
Unit 2

Reactor Protective System Trip Setting Limits

	RPS Segment	Four Reactor Coolant Pumps Operating (Operating Power -100% Rated)	Three Reactor Coolant Pumps Operating (Operating Power -75% Rated)	One Reactor Coolant Pump Operating in Each Loop (Operating Power -49% Rated)	Shutdown Bypass
1.	Nuclear Power Max. (% Rated)	105.5	105.5	105.5	5.0(3)
2.	Nuclear Power Max. Based on Flow (2) and Imbalance, (% Rated)	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	1.07 times flow minus reduction due to imbalance	Bypassed
3.	Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	NA	55%	Bypassed
4.	High Reactor Coolant System Pressure, psig, Max.	2300	2300	2300	1720(4)
5.	Low Reactor Coolant System Pressure, psig, Min.	1800	1800	1800	Bypassed
6.	Variable Low Reactor Coolant System Pressure psig, Min.	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	$(11.14\ T_{out} - 4706)^{(1)}$	Bypassed
7.	Reactor Coolant Temp. F., Max.	618	618	618	618
8.	High Reactor Building Pressure, psig, Max.	4	4	4	4

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

(2) Reactor Coolant System Flow, %.

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

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

Table 2.3-1C
Unit 3

Reactor Protective System Trip Setting Limits

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

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

(2) Reactor Coolant System Flow, %.

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

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

Amendments Nos. 124, 124, & 121

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 3.5.2-1A1, 3.5.2-1A2, and 3.5.2-1A3 (Unit 1); 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3 (Unit 2); 3.5.2-1C1, 3.5.2-1C2, and 3.5.2-1C3 (Unit 3) for four pump operation, on figures 3.5.2-2A1, 3.5.2-2A2, and 3.5.2-2A3 (Unit 1); 3.5.2-2B1, 3.5.2-2B2, and 3.5.2-2B3 (Unit 2); figures 3.5.2-2C1, 3.5.2-2C2, and 3.5.2-2C3 (Unit 3) for three pump operation, and on figures 3.5.2-2A4, 3.5.2-2A5, and 3.5.2-2A6 (Unit 1); 3.5.2-2B4, 3.5.2-2B5, and 3.5.2-2B6 (Unit 2); figures 3.5.2-2C4, 3.5.2-2C5, and 3.5.2-2C6 (Unit 3) for two pump operation. Also, excepting physics tests or exercising control rods, the axial power shaping control rod insertion/withdrawal limits are specified on figures 3.5.2-4A1, 3.5.2-4A2, and 3.5.2-4A3 (Unit 1); 3.5.2-4B1 (Unit 2); 3.5.2-4C1, 3.5.2-4C2, and 3.5.2-4C3 (Unit 3).

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

3.5.2.6 Xenon Reactivity

Except for physics tests, reactor power shall not be increased above the power-level-cutoff shown in Figures 3.5.2-1A1, 3.5.2-1A2, and 3.5.2-1A3 for Unit 1; Figures 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3, for Unit 2; and Figures 3.5.2-1C1, 3.5.2-1C2, and 3.5.2-1C3 for Unit 3 unless one of the following conditions is satisfied:

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

2. Xenon reactivity deviated more than 10 percent but is now within 10 percent of the equilibrium value for operation at steady state rated power and has passed its final maximum or minimum peak during 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 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3A3, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3C1, 3.5.2-3C2, and 3.5.2-3C3. If the imbalance is not within the envelope defined by these figures, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

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

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. Operational 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.

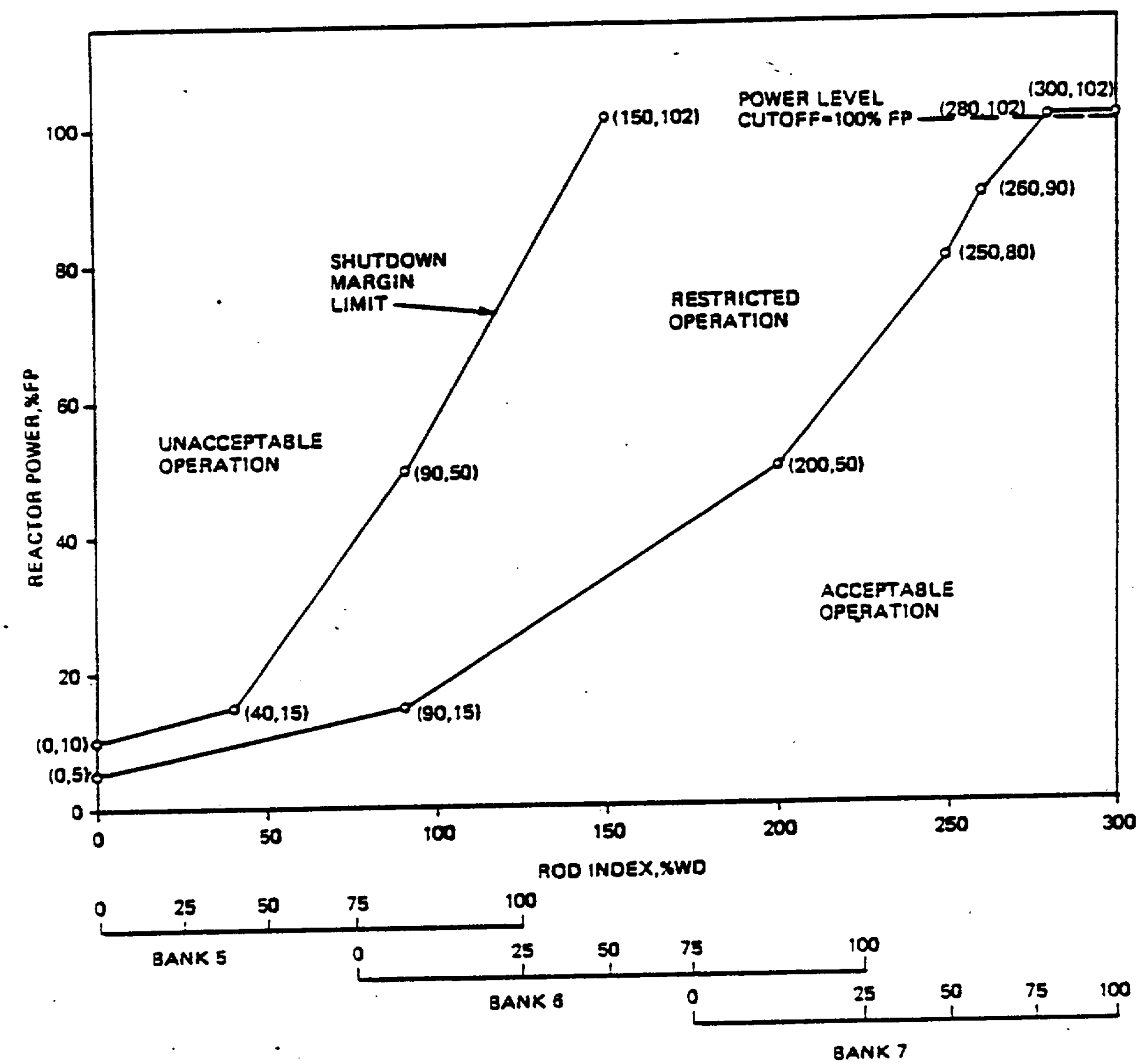
The power-imbalance envelope defined in Figures 3.5.2-3A1, 3.5.2-3A2, 3.5.2-3A3, 3.5.2-3B1, 3.5.2-3B2, 3.5.2-3C1, 3.5.2-3C2 and 3.5.2-3C3 is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

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

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

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

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

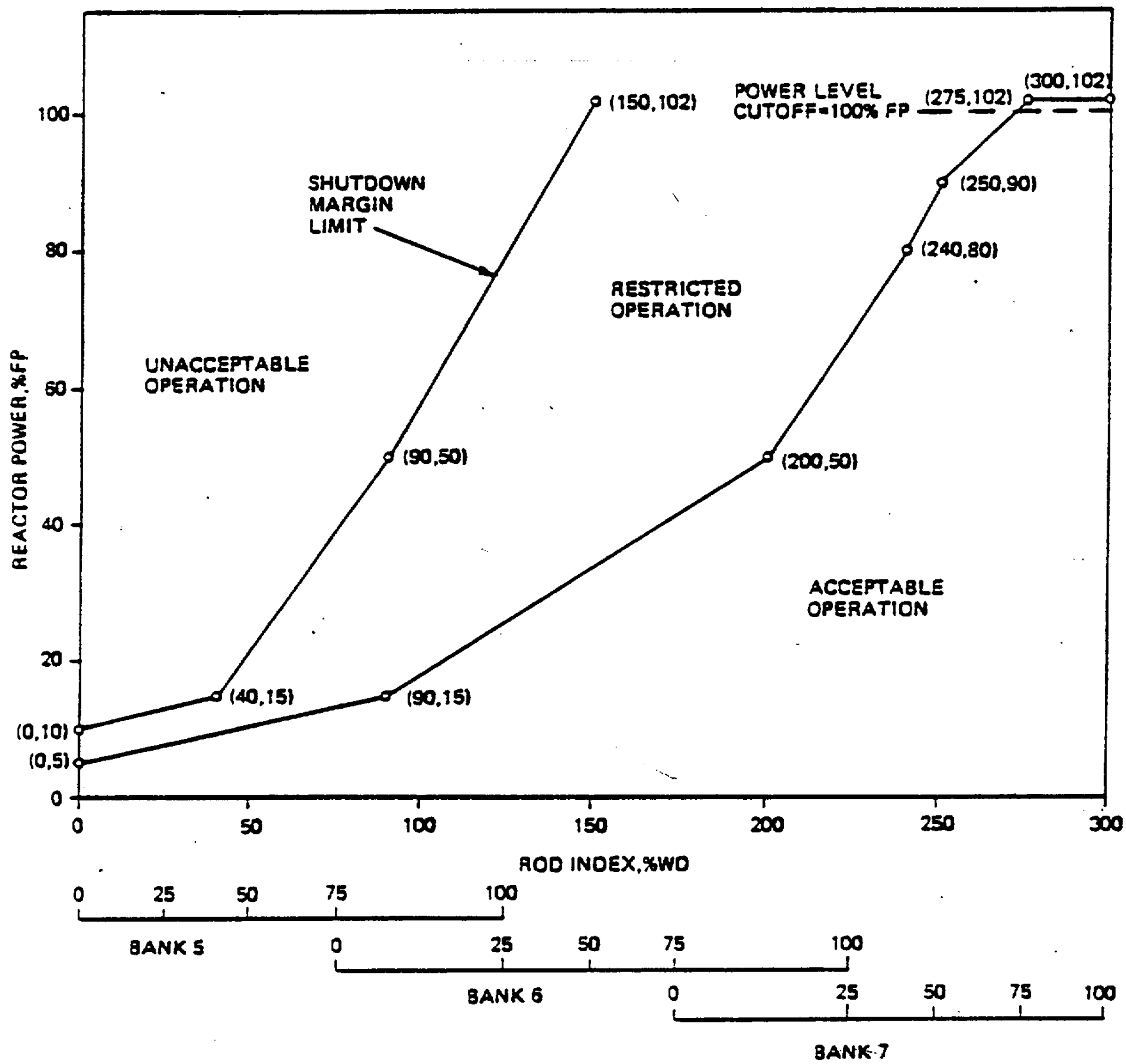


ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 0 to 25 (+10, -0) EFPD
UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-1B1

Amendments Nos. 124 , 124 , & 121

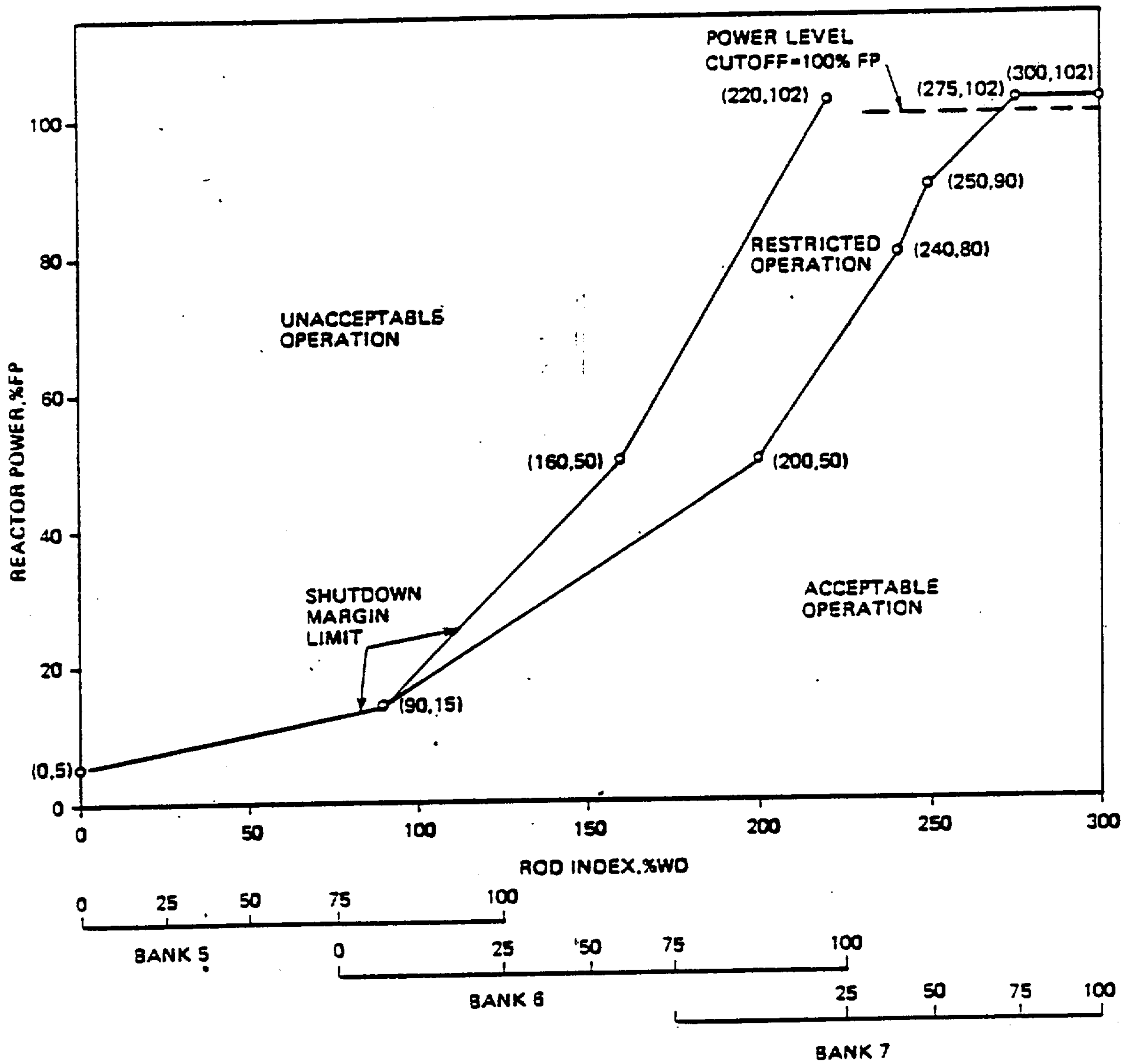


ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
FROM 25 (+10, -0) to 200 ± 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-1B2

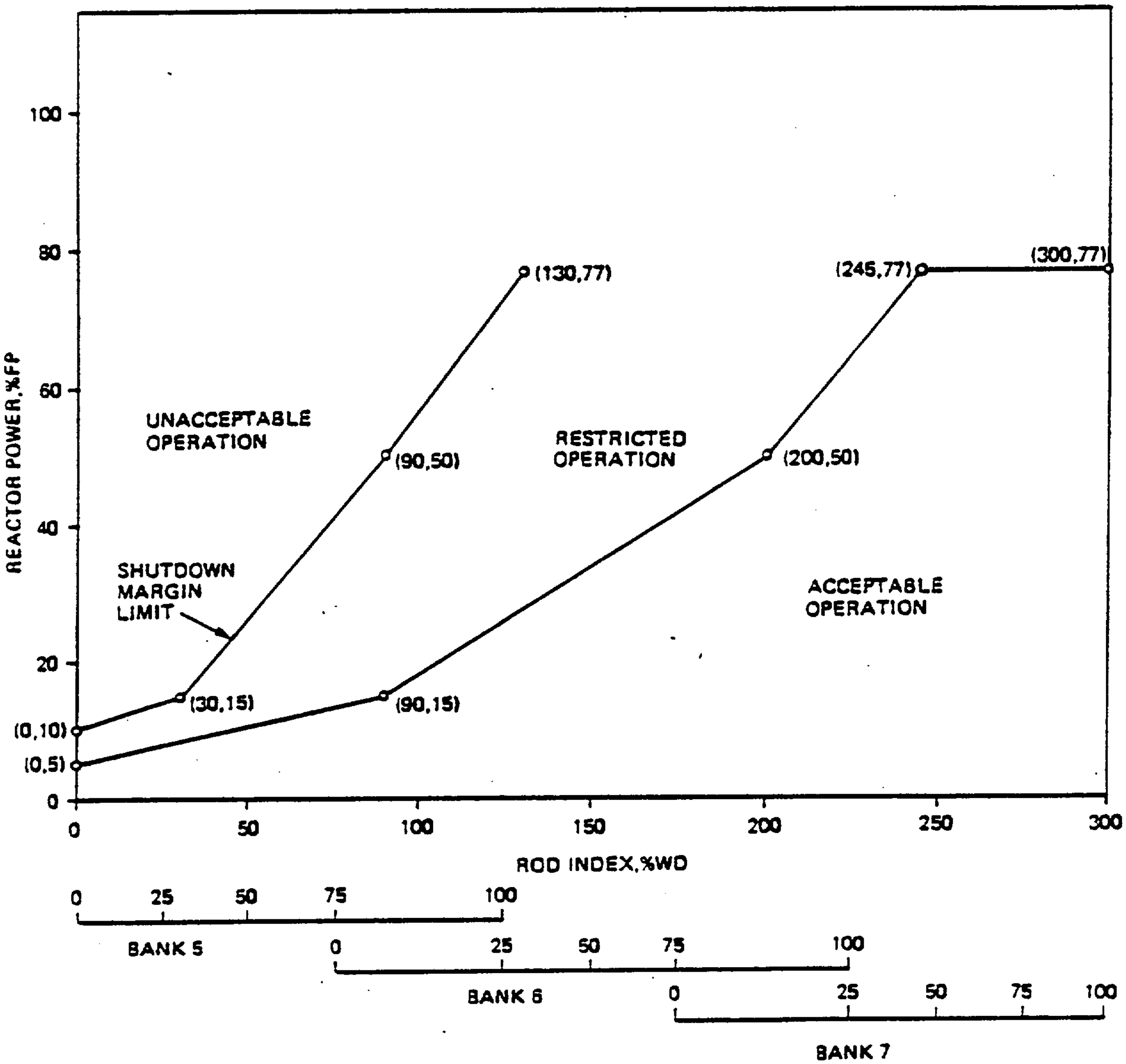
Amendments Nos. 124 , 124 , & 121



ROD POSITION LIMITS
FOR FOUR PUMP OPERATION
AFTER 200 $\pm$ 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

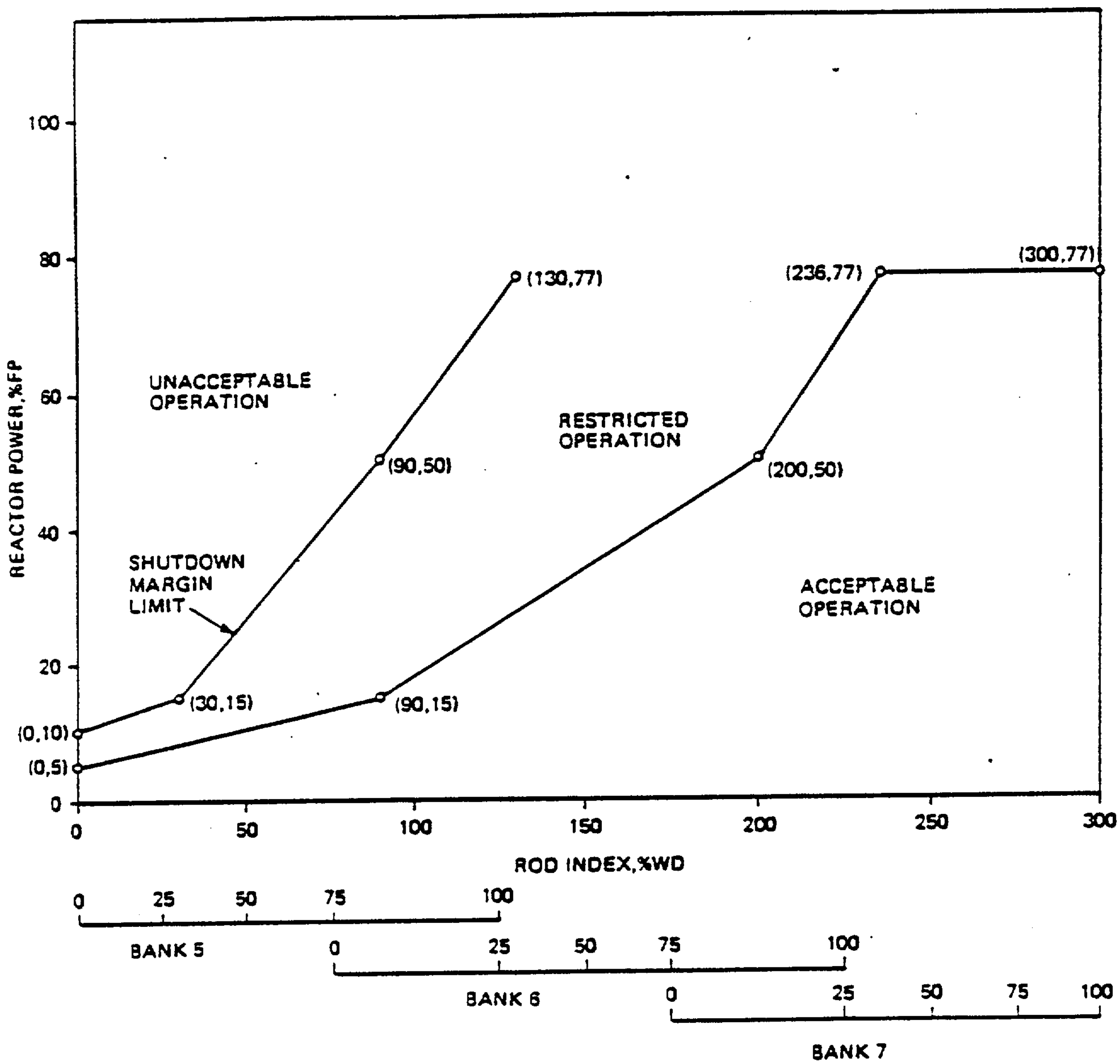
Figure 3.5.2-1B3



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

OCONEE NUCLEAR STATION

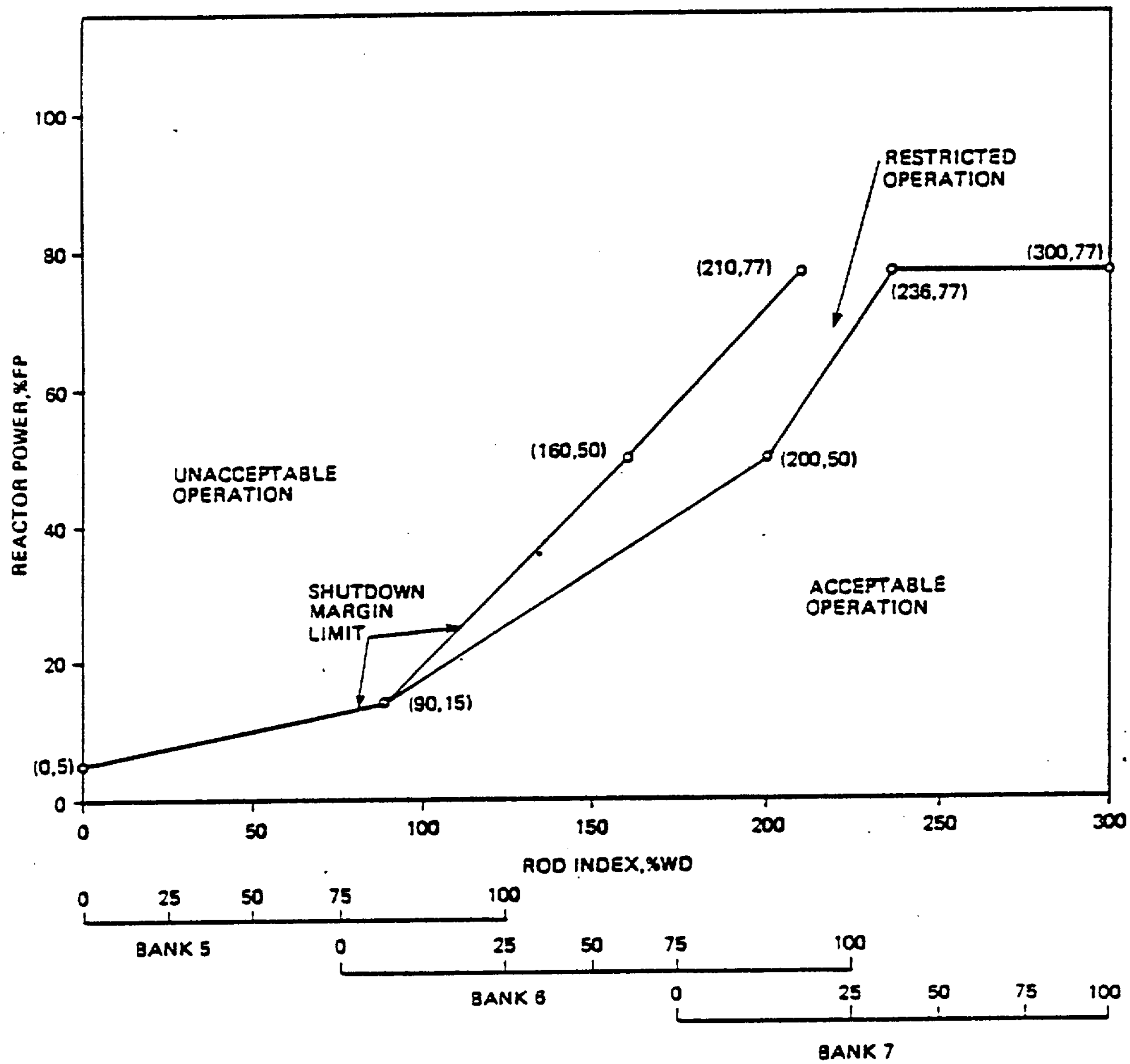
Figure 3.5.2-2B1



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

OCONEE NUCLEAR STATION

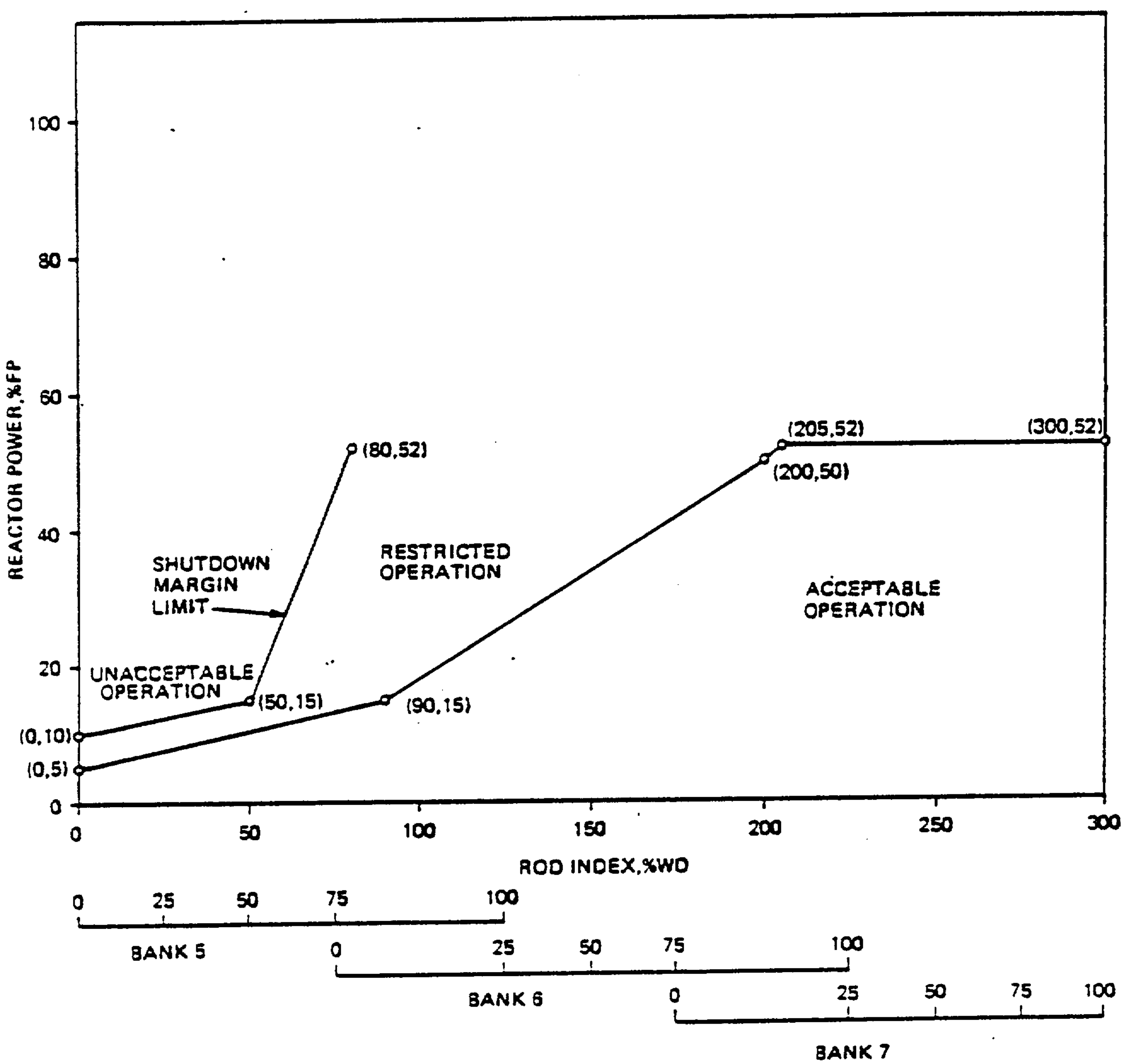
Figure 3.5.2-2B2



ROD POSITION LIMITS
FOR THREE PUMP OPERATION
AFTER 200 $\pm$ 10 EFPD
UNIT 2

OCONEE NUCLEAR STATION

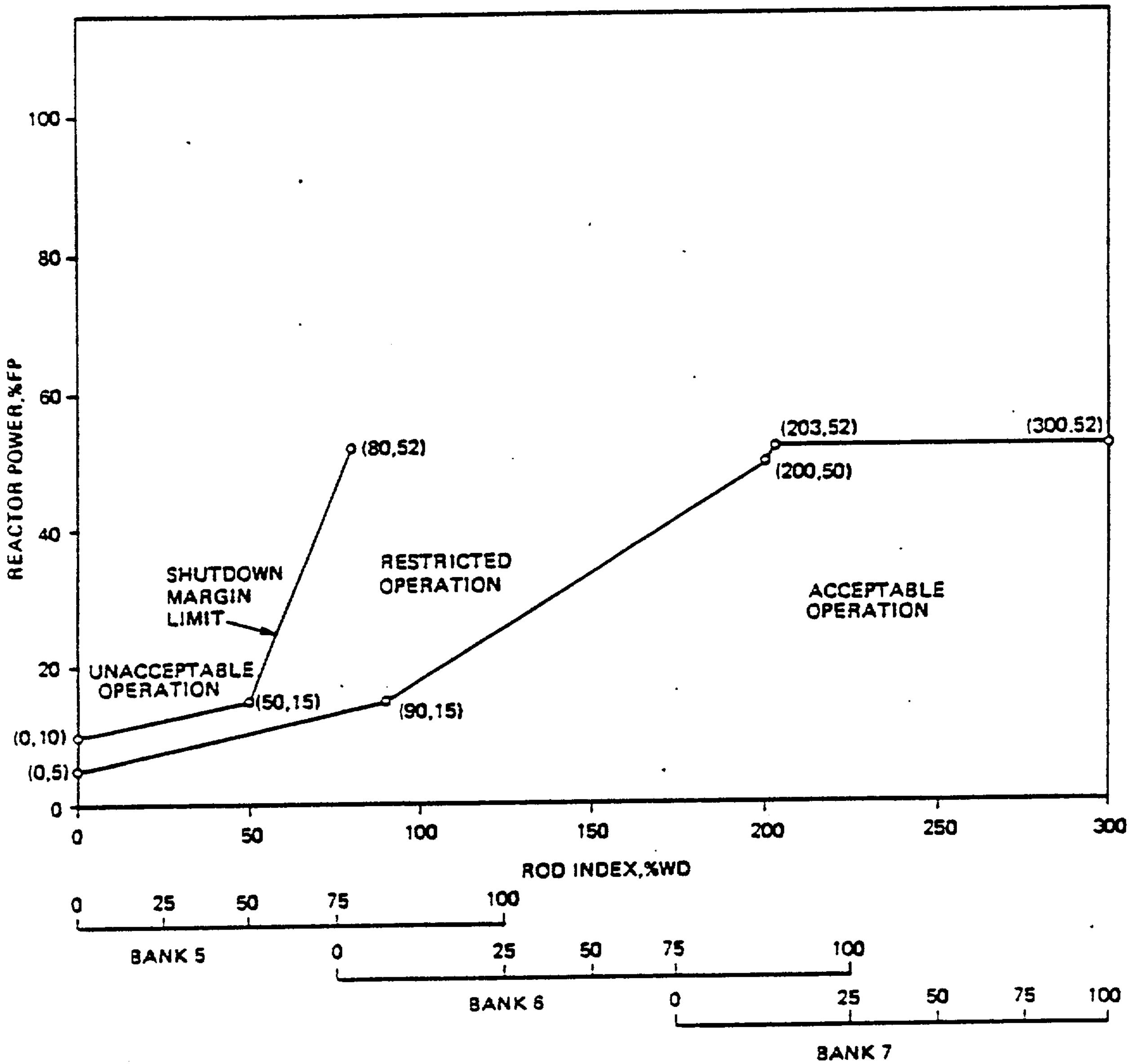
Figure 3.5.2-2B3



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

OCONEE NUCLEAR STATION

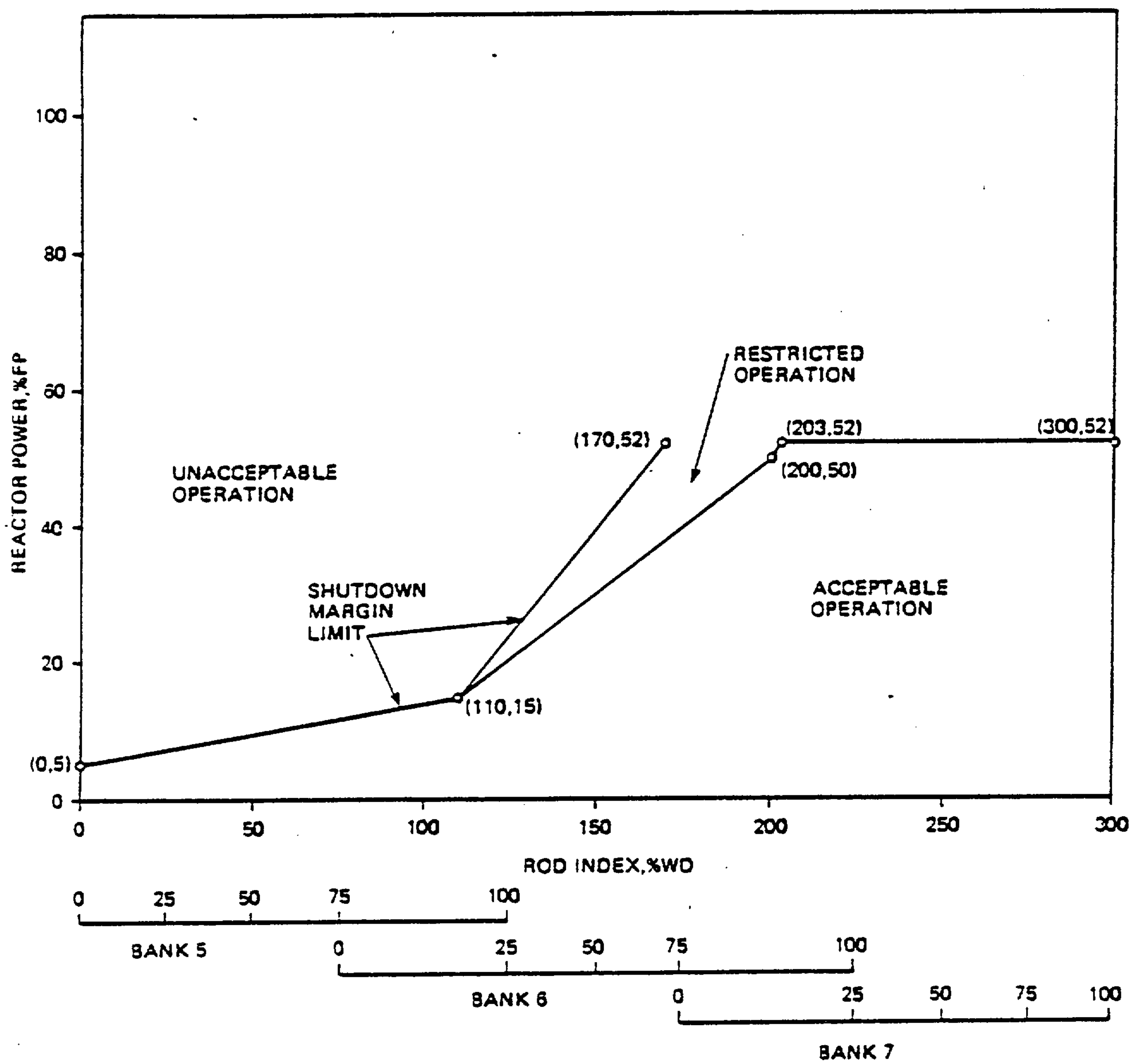
Figure 3.5.2-2B4



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

OCONEE NUCLEAR STATION

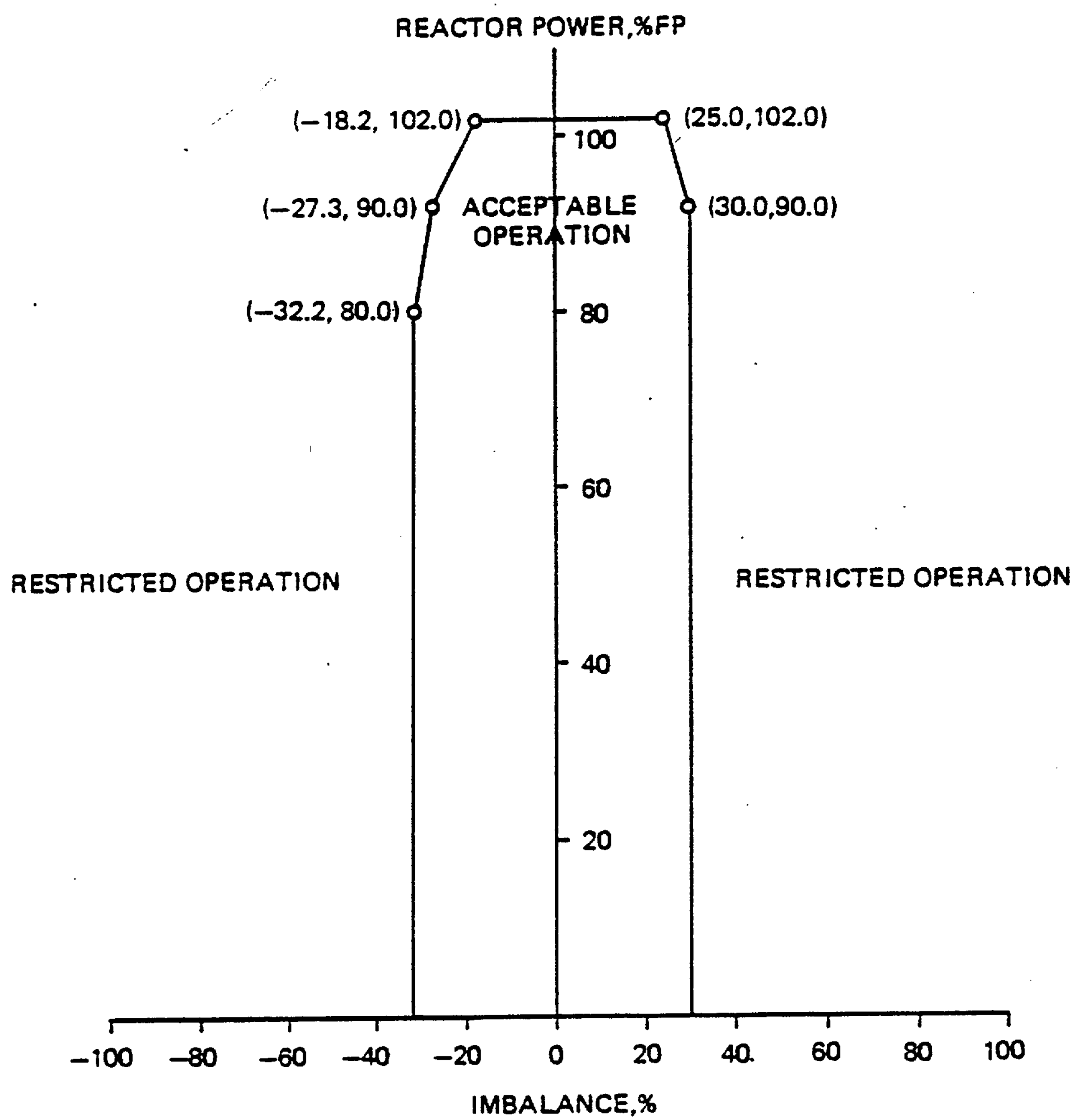
Figure 3.5.2-2B5



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

OCONEE NUCLEAR STATION

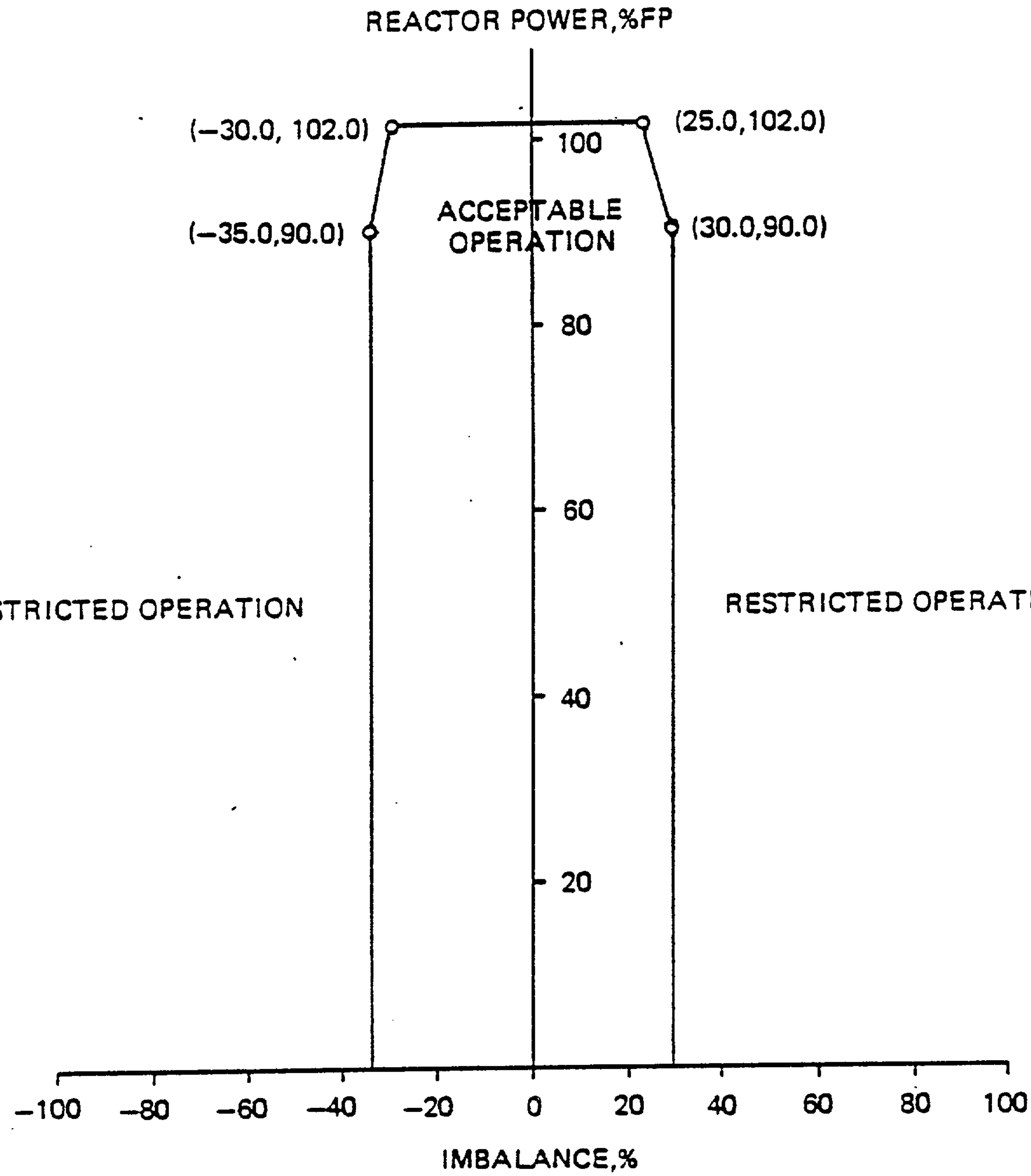
Figure 3.5.2-2B6



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

OCONEE NUCLEAR STATION

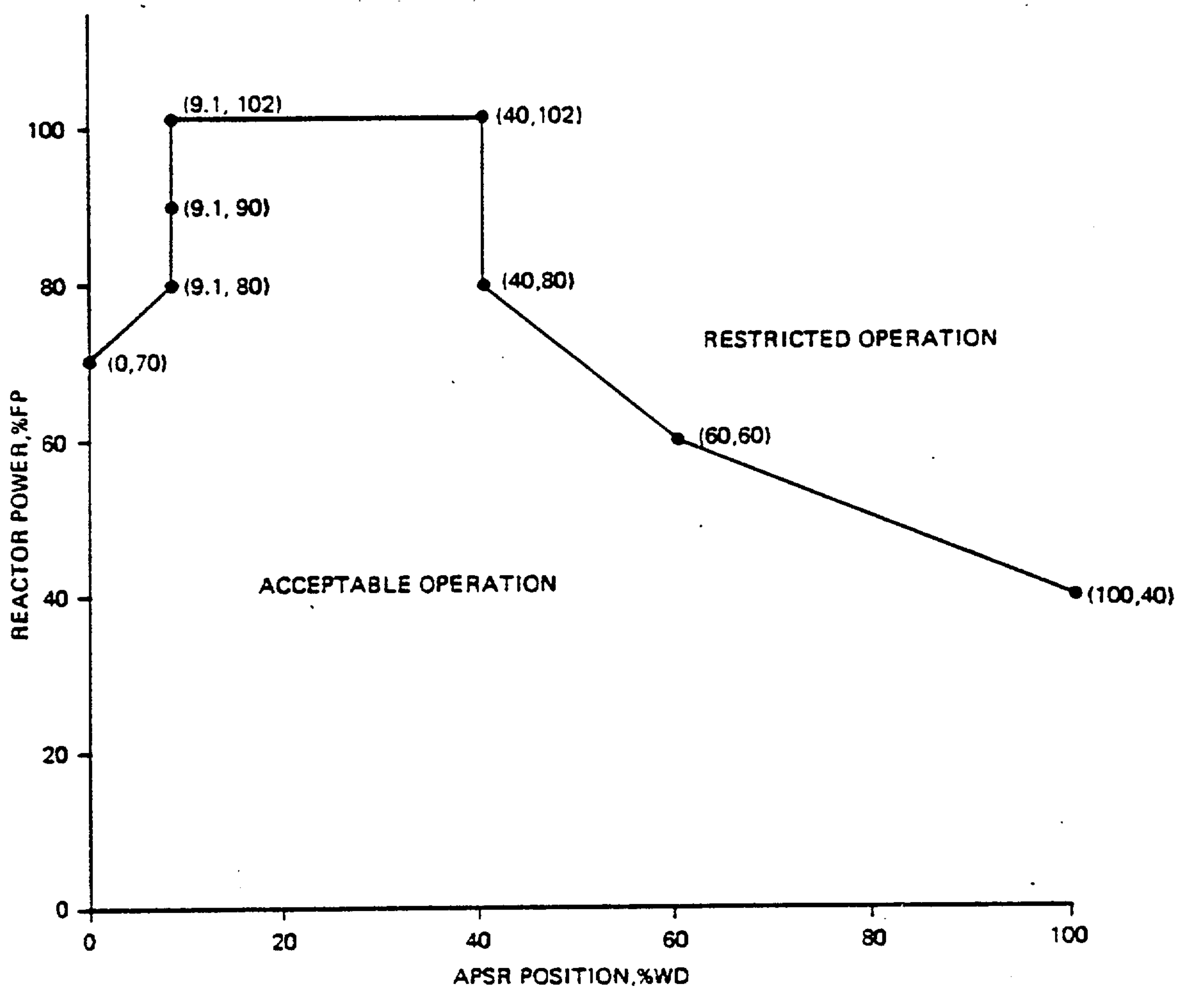
Figure 3.5.2-3B1



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

OCONEE NUCLEAR STATION

Figure 3.5.2-3B2



APSR POSITION LIMITS
FOR OPERATION
0 EFPD to EOC
UNIT 2

OCONEE NUCLEAR STATION

Figure 3.5.2-4B1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 124 TO FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 121 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 letters dated September 1, 1983 (Ref. 1) and September 14, 1983 (Ref. 2), Duke Power Company made application to modify the Oconee Nuclear Station Technical Specifications in support of Cycle 7 operation of Unit 2. The analysis performed and the resulting modifications to the Station's common Technical Specifications are described in the Unit 2 Cycle 7 reload report (Ref. 3).

The safety analysis for the previous sixth cycle of operation at Oconee Unit 2 is being used by the licensee as a reference for the proposed seventh cycle of operation. Where conditions are identified as limiting in the sixth cycle analysis, our previous evaluation (Ref. 4) of that cycle continues to apply. Our evaluation of the most recent reload submittal from the Oconee Station (Oconee Unit 1, Cycle 8 - Ref. 5) is also used as a basis for the findings described in the following sections.

1.1 Description of the Cycle 7 Core

The Oconee Unit 2 Cycle 7 core will consist of 177 fuel assemblies, each of which is a 15X15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 7 will operate in bleed-and-feed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 61 full length control rod assemblies (CRAs) and 64 burnable poison rod assemblies (BPRAs). In addition, 8 axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. Seventy-two fresh assemblies having an initial enrichment of 3.24 weight percent U-235 will be loaded. The length of Cycle 7 is expected to be 421 effective full power days (EFPD) compared with 400 EFPD accumulated during Cycle 6. The licensed core full power level remains at 2568 MWt.

2.0 EVALUATION OF THE FUEL SYSTEM DESIGN

2.1 Fuel Assembly Mechanical Design

The 72 Babcock & Wilcox (B&W) Mark-B4 fuel assemblies loaded as Batch 9 at end of Cycle 6 (EOC 6) are mechanically interchangeable with the Batches 6C, 7C and 8 fuel assemblies loaded previously at Oconee Unit 2. The Mark-B4 fuel assembly has been previously approved (Ref. 4) by the staff and utilized in other B&W nuclear steam supply systems. Batch 9 will contain one Advanced Cladding Pathfinder (ACP) assembly which, unlike the standard Mark-B4 design, is reconstitutable. This assembly contains 12 rods with special cladding, 6 zirconium lined and 6 beta quenched, which are expected to provide improved resistance to water-side corrosion and pellet-cladding interaction (PCI). In other respects, the ACP assembly is nearly identical to the standard Mark-B4 design and is not limiting for Cycle 7 operation. Because the ACP assembly does not, in itself, result in any Technical Specification changes for the Cycle 7 core, and because the licensee has determined that its inclusion does not result in any unreviewed safety question, this assembly may be incorporated into the Cycle 7 reload without NRC approval.

We are aware of a number of other recent changes to the B&W 15X15 fuel assembly design (e.g., a larger fuel assembly holddown spring, fuel pellets manufactured by an alternate supplier). These changes have been approved for use in other operating B&W 177-fuel-assembly plants on a limited basis and may be incorporated into future cycles of operation at Oconee Unit 2. However, for the current cycle of operation, the licensee has identified no other changes in the fuel assembly mechanical design. We find this acceptable.

In the course of our review, we have noted 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.

This issue was previously considered in our safety evaluation of the Oconee Unit 3 Cycle 6 reload (Ref. 6). In that evaluation, we concluded that the holddown spring issue had been correctly analyzed and did not result in a safety concern for Unit 3 Cycle 6 operation. An inspection (Ref. 7) of all Unit 3 Cycle 6 assemblies revealed broken holddown springs in two assemblies due to be discharged. Another inspection (Ref. 8) 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, 4 additional broken holddown springs were found in Unit 1 (Ref. 9). In all cases, the fuel was due to be discharged or the holddown springs were replaced prior to insertion. We now conclude that a continuing program of detection and discharge/replacement of failed holddown springs is no longer adequate for the Mark-B4 fuel design. Because our concern involves the use of the Mark-B4 design in operating reactors in addition to the Oconee Station, we have reinitiated discussions of this problem with the fuel vendor, B&W.

2.2 Fuel Rod Design

The cladding stress, strain and collapse analyses for the fuel in the Cycle 7 core are generally bounded by conditions previously analyzed for Oconee Unit 2 or were analyzed specifically for Cycle 7 using methods and limits developed and used by the fuel vendor (B&W) and reviewed and approved by the NRC. In the case of Cycle 7, however, these same analyses were performed by the licensee rather than by the fuel vendor using a reload methodology (Ref. 10), which has also been approved (Ref. 11) by the NRC. Duke Power Company previously applied this reload methodology, for the first time, in the Oconee Unit 3 Cycle 7 analysis (Ref. 12).

An exception to the general methodology has been identified by the licensee in the cladding stress analysis. The fuel rod total stress is not permitted to exceed the unirradiated yield strength (the previous limit) of the cladding. Two times the minimum unirradiated yield strength is now used as a criterion for the total stress calculation, as suggested by the ASME Boiler and Pressure Vessel Code (Ref. 13). Primary membrane plus primary bending stresses continue to be limited to the unirradiated yield strength, and primary membrane stress continues to be limited to two-thirds of this value.

We have previously noted (Ref. 12) that the ASME Boiler and Pressure Vessel Code does not apply to fuel rod cladding, specifically Zircaloy cladding. The application of this code to the mechanical analysis of fuel rods is suggested by the NRC Standard Review Plan (Ref. 14), but only as general guidance. Stress limits similar to those proposed by the licensee have been accepted elsewhere, but these are usually based on both yield and ultimate tensile strength limits. Because the cladding stress analysis is not, and has not been, limiting for operation at Oconee, we accept the analysis as submitted.

The fuel thermal and material design analyses, including fuel rod internal pressure limits, continue to be analyzed with previously approved methods. The licensee has continued to rely upon fuel thermal performance from several B&W codes: TAFY-3 (Ref. 15), TACO-1 (Ref. 16), and TACO-2 (Ref. 17). A combination of TAFY-3 and TACO-2 analyses were used to generate the LOCA limits as described in Tables 7-2 and 7-3 of Reference 3. 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. We have determined (see Section 3.0) that these limits have been satisfactorily incorporated in to the Technical Specifications for Cycle 7 through the operating limits on rod index and axial power imbalance.

3.0 EVALUATION OF NUCLEAR DESIGN

The nuclear characteristics of the core have been computed by methods previously approved for the Oconee Nuclear Station (Ref. 10). Comparisons are made between the physics parameters for Cycles 6 and 7. The differences that exist between the nuclear characteristics are due to the increased cycle length, i.e., an estimated 421 EFPD for Cycle 7 vs an actual 400 EFPD for Cycle 6 (or an estimated 390 EFPD for Cycle 6). In addition the peripheral assembly locations are occupied by once burned assemblies in order to minimize neutron leakage and increase cycle length. The critical boron concentration for beginning of Cvcle 7 is higher at hot zero power, no xenon conditions, than at Cycle 6. Changes in the radial flux and burnup distribution (peaked in the inner fuel assemblies and suppressed in the peripheral assemblies) accounts for the differences in control rod worths. For example, Group 7 control rod worth at the beginning of cycle (hot full power) is 1.51 percent $\Delta k/k$ vs 1.46 percent $\Delta k/k$ for Cycle 6 and at the end of Cycle 7 is estimated as 1.64 percent $\Delta k/k$ vs 1.53 percent $\Delta k/k$ at the end of Cycle 6. Safety criteria are met for stuck and ejected rod worths. Shutdown margin values are 3.88 percent $\Delta k/k$ at beginning of cycle and 2.68 percent $\Delta k/k$ at end of Cycle 7 compared to the required 1.0 percent $\Delta k/k$. The effective delayed neutron fraction both at beginning and end of Cycle 7 is practically unchanged compared to the corresponding values of Cycle 6. The values in Cycle 6 and those in Cycle 7 are bounded by those calculated in the FSAR. Based on this review, we conclude that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of Cycle 7 is acceptable.

3.1 Evaluation of Accident and Transient Analysis

The key kinetics parameters for Cycle 7 have been compared to the values used in the FSAR and those calculated in Cycle 6. It is shown that in all cases Cycle 7 values are bounded by those used previously. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in Reference 19. However, the fuel assemblies of this reload contain fuel rods with a theoretical density higher than those considered in Reference 19, hence the conclusions are still valid. Considering the previously reviewed and approved design used in the FSAR and the results of the densification report (Ref. 18), we conclude that the transients in Cycle 7 are bounded by the FSAR and the densification report and, hence, are acceptable.

4.0 THERMAL-HYDRAULIC

Duke Power Company performed the thermal-hydraulic design analysis supporting Cycle 7 operation using the Oconee Reload Design Methodology (Ref. 10) that was previously reviewed and approved by the staff and employing the B&W approved methodology (Refs. 19, 20 and 21).

Cycles 6 and 7 are hydraulically and geometrically similar as shown in Table 1. The maximum core bypass flow for Cycle 6 was 7.6 percent of the total system flow versus 7.8 percent for Cycle 7. This value is less than the bypass flow value (8.2 percent) that is assumed in the generic thermal-hydraulic design analysis. Therefore, we find core bypass flow of 7.8 percent of the total system flow to be conservative for Cycle 7 operation.

6.0 SUMMARY

We conclude that the Oconee 2 Cycle 7 will not adversely affect the capability to operate the plant safely. We also conclude that the proposed changes to the Technical Specifications discussed above for Oconee 2 Cycle 7 are acceptable.

7.0 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.

8.0 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: November 23, 1983

The following NRC staff personnel have contributed to this Safety Evaluation: L. Lois, A. Gill, J. Voglewede, J. Suermann.

TABLE 1

THERMAL HYDRAULIC DESIGN CONDITIONS
OCONEE UNIT 2, CYCLE 7

	Cycle 6	Cycle 7
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design flow	106.5	106.5
Core bypass flow, % total flow	7.6	7.8
Vessel inlet/outlet coolant temp at 100% power, °F	555.6/602.4	555.6/602.4
Ref design radial-local power peaking factor	1.71	1.71
Ref design axial flux shape	1.5 cosine	1.5 cosine
Hot channel factors: Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Active fuel length, in.	(a)	(a)
Avg heat flux at 100% power, 10^3 Btu/h-ft^2 (a)	176 (b)	176 (b)
CHF correlation	BAW-2	BAW-2
Min DNBR with densification penalty	2.05	>2.05

(a) 142.2 inches for Batches 6C and 7C, 141.8 for Batches 8 and 9.

(b) Heat flux based on a conservative minimum densified length of 140.3 in.

REFERENCES

1. H. B. Tucker (Duke) letter to H. R. Denton (NRC) on "Oconee Nuclear Station Unit 2" dated September 1, 1983.

2. H. B. Tucker (Duke) letter to H. R. Denton (NRC) on "Oconee Nuclear Station Unit 2" dated September 14, 1983.

3. "Oconee Unit 2, Cycle 7 Reload Report," Duke Power Company Report DPC-RD-2002, September 1983. Attachment 3 to References 1 and (revised) 2 above.

4. L. S. Rubenstein (NRC) memorandum for T. Novak (NRC) on "SER - Oconee Unit 2 Reload for Cycle 6" dated February 19, 1982.

5. J. F. Stolz (NRC) letter to H. B. Tucker (Duke) on Oconee Unit 1 Cycle 8 reload dated August 3, 1983 and transmitting Amendments No. 122, 122 and 119 to Facility Operating Licenses No. DPR-38, DPR-47 and DPR-55, respectively.

6. L. S. Rubenstein (NRC) memorandum for T. M. Novak (NRC) on "Oconee Unit 3 Cycle 6 Reload" dated February 2, 1981.

7. W. O. Parker (Duke) letter to J. P. O'Reilly (NRC) dated July 23, 1982.

8. W. O. Parker (Duke) letter to J. P. O'Reilly (NRC) dated February 16, 1982.

9. H. B. Tucker (Duke) letter to J. P. O'Reilly (NRC) dated July 21, 1983.

10. "Oconee Nuclear Station Reload Design Methodology" Technical Report, Duke Power Company Report NFS-1001, Revision 4, April 1979.

11. L. S. Rubenstein (NRC) memorandum for T. M. Novak (NRC) on "Duke Reload Design Methodology Technical Report Evaluation" dated May 26, 1981.

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20. "Oconee Nuclear Station, Units 1, 2 and 3, Final Safety Analysis Report," Docket Nos. 50-269, 50-270 and 50-287, Duke Power Company, Charlotte, North Carolina.

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23. W. O. Parker, Jr. (Duke), letter to H. R. Denton (NRC), October 16, 1981.