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Docket Nos. 50-269
50-270
and 50-287

November 21, 1977

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President - Steam Production
422 South Church Street
P. O. Box 2178
Charlotte, North Carolina 28242

Gentlemen:

The Commission has issued the enclosed Amendment Nos. 52, 52 and 49 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Unit Nos. 1, 2 and 3. These amendments consist of changes to the Station's common Technical Specifications and are in response to your request dated September 6, 1977.

These amendments revise the Technical Specifications to establish operating limits for Unit 3 Cycle 3 operation.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 52 to DPR-38
2. Amendment No. 52 to DPR-47
3. Amendment No. 49 to DPR-55
4. Safety Evaluation
5. Notice of Issuance

cc w/enclosures:
See next page

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OFFICE➤	DOR:ORB#1	OELD	DOR:ORB#1			
SURNAME➤	DNeighbors:tsb		ASchwencer			
DATE➤	11/17/77	11/ /77	11/ /77			

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

Docket Nos. 50-269
50-270
and 50-287

November 21, 1977

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Vice President - Steam Production
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A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

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1. Amendment No. 52 to DPR-38
2. Amendment No. 52 to DPR-47
3. Amendment No. 49 to DPR-55
4. Safety Evaluation
5. Notice of Issuance

cc w/enclosures:
See next page

cc: Mr. William L. Porter
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

J. Micheal McGarry, III, Esquire
DeBevoise & Liberman
700 Shoreham Building
806-15th Street, NW.,
Washington, D.C. 20005

Oconee Public Library
201 South Spring Street
Walhalla, South Carolina 29691

Honorable James M. Phinney
County Supervisor of Oconee County
Walhalla, South Carolina 29621

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Coutland Street, N. E.
Atlanta, Georgia 30308

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. 52
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 6, 1977, 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 License No. DPR-38 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 52, 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 21, 1977

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. 52
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 6, 1977, 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 License No. DPR-47 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 52, 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 21, 1977

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. 49
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 6, 1977, 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 License No. DPR-55 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49, 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 21, 1977

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 52 TO DPR-38

AMENDMENT NO. 52 TO DPR-47

AMENDMENT NO. 49 TO DPR-55

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

Revise Appendix A as follows:

1. Remove the following pages and replace with identically numbered pages.

2.1-3c	3.5-10
2.1-3d	3.5-11
2.1-6	3.5-16
2.1-9	3.5-16a
2.1-12	3.5-17
2.3-1	3.5-20
2.3-3	3.5-20a
2.3-4	3.5-20b
2.3-7	3.5-23
2.3-10	3.5-23a
2.3-13	3.5-23b
3.5-9	4.1-9

2. Add pages 3.5-23i, 3.5-23j and 3.5-23k

Bases - Unit 3

The safety limits presented for Oconee Unit 3 have been generated using BAW-2 critical heat flux correlation(1) and the Reactor Coolant System flow rate of 106.5 percent of the design flow (131.32 x 10^6 lbs/hr for four-pump operation). The flow rate utilized is conservative compared to the actual measured flow rate.(2)

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of the BAW-2 correlation(1). The BAW-2 correlation has been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30. A DNBR of 1.30 corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure has been considered in determining the core protection safety limits. The difference in these two pressures is nominally 45 psi; however, only a 30 psi drop was assumed in reducing the pressure trip setpoints to correspond to the elevated location where the pressure is actually measured.

The curve presented in Figure 2.1-1C represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power (112 percent) when four reactor coolant pumps are operating (minimum reactor coolant flow is 139.86x 10^6 lbs/hr.). This curve is based on the following nuclear power peaking factors with potential fuel densification and fuel rod bowing effects: $F_q^N = 2.67$; $F_{\Delta H}^N = 1.78$; $F_z^N = 1.50$. The design peaking combination results in a more conservative DNBR than any other power shape that exists during normal operation.

The curves of Figure 2.1-2C are based on the more restrictive of two thermal limits and include the effects of potential fuel densification and fuel rod bowing.

1. The 1.30 DNBR limit produced by a nuclear peaking factor of $F_q^N = 2.67$ or 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 20.15 kw/ft for Unit 3.

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 for Curves 1, 2 and 3 of Figure 2.1-2C correspond to the expected minimum flow rates with four pumps, three pumps and one pump in each loop, respectively.

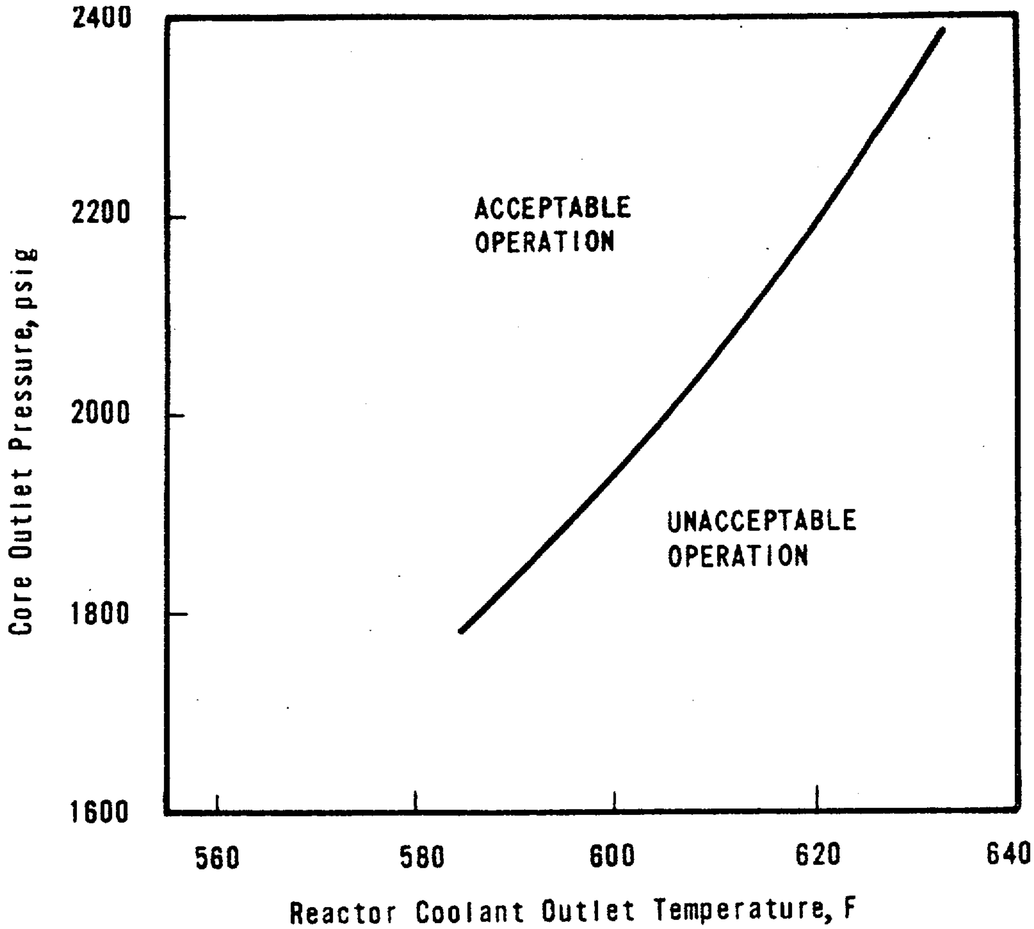
The maximum thermal power for three-pump operation is 85.3 percent due to a power level trip produced by the flux-flow ratio 74.7 percent flow x 1.055= 78.8 percent power plus the maximum calibration and instrument error. The maximum thermal power for other coolant pump conditions are produced in a similar 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.

<u>References</u>

(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.

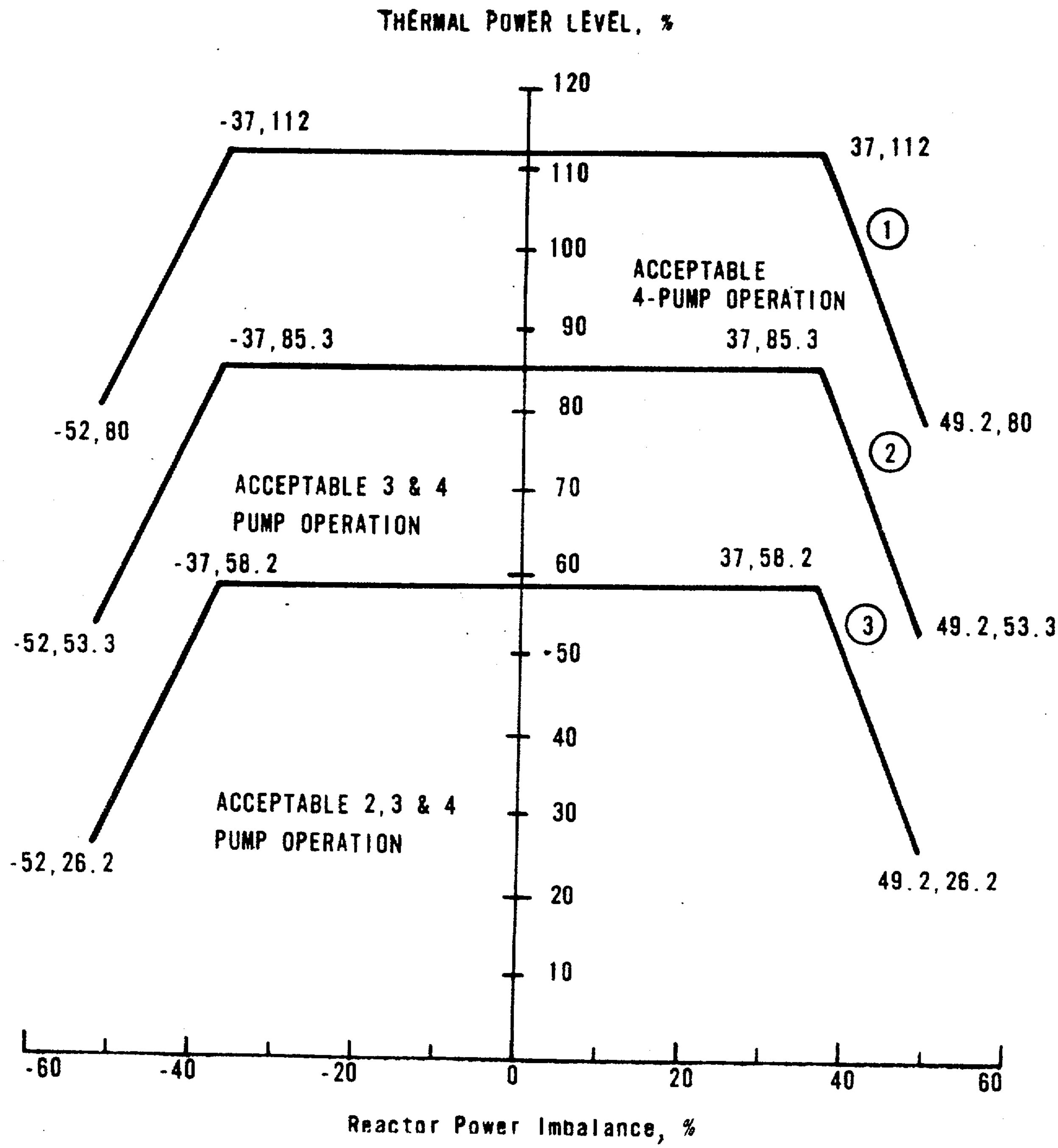


CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.1-1C

Amendment Nos. 52, 52 & 49



Curve	Reactor Coolant Flow, gpm
1	374,880 (100%)*
2	280,035 (74.7%)
3	183,690 (49.0%)

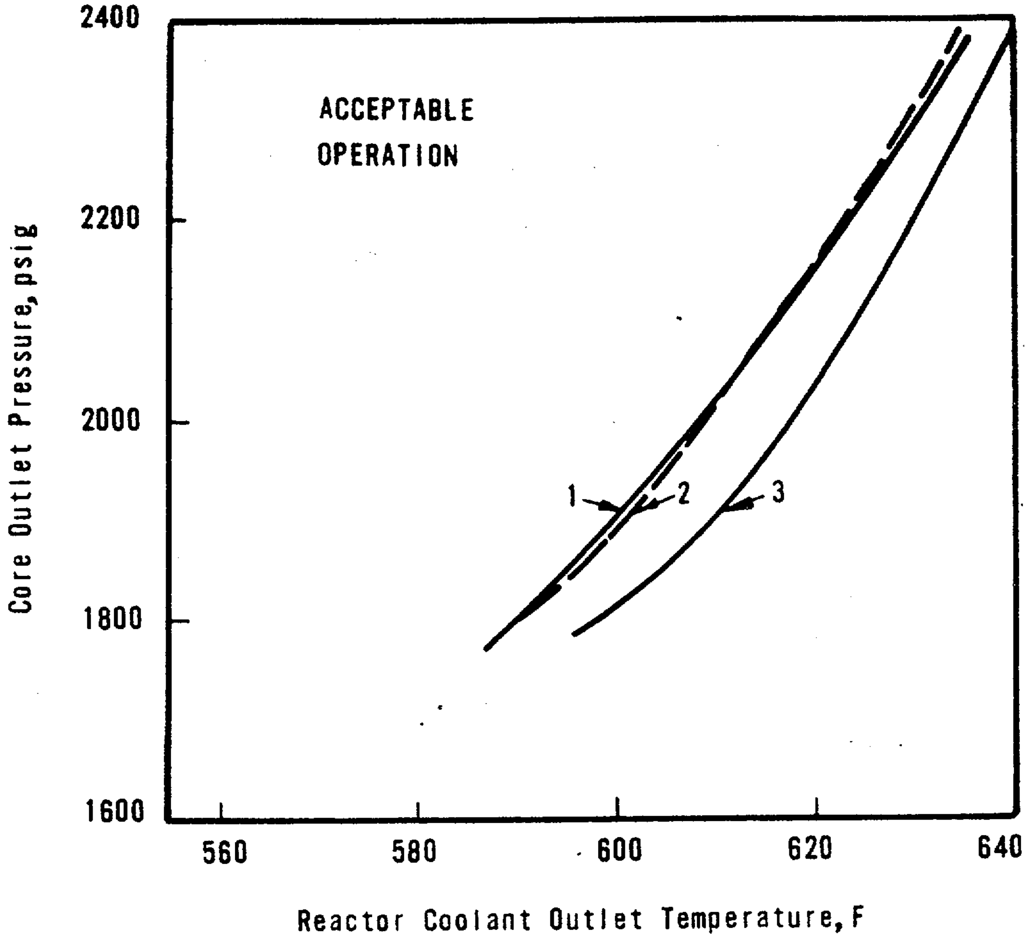
*106.5% of first-core design flow.

CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.1-2C

Amendment Nos. 52, 52 & 49



Curve	Coolant Flow, gpm	Power, %	Pumps Operating	Type of Limit
1	374,880 (100%)*	112	4	DNBR
2	280,035 (74.7%)	86.7	3	DNBR
3	183,690 (49.0%)	59.0	2	Quality

*106.5% of first-core design flow.

CORE PROTECTION SAFETY LIMITS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.1-3C

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protective action to prevent any combination of process variables from exceeding a safety limit.

Specification

The reactor protective system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3.1A-Unit 1
and 2.3-1B-Unit 2
2.3-1C-Unit 3

Figure 2.3-2A-Unit 1
2.3-2B-Unit 2
2.3-2C-Unit 3

The pump monitors shall produce a reactor trip for the following conditions:

a. Loss of one pump during four-pump operation if power level is greater than 80% of rated power

b. Loss of two pumps and reactor power level is greater than 55% of rated power. (Power/RC pump trip setpoint is reset to 55% for operation with one pump in each loop).

c. Loss of two pumps in one reactor coolant loop and reactor power level is greater than 0.0% of rated power.

d. Loss of one or two pumps during two-pump operation.

Bases

The reactor protective system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protective system instrumentation are listed in Table 2.3-1A-Unit 1. The safety analysis has been based upon these protective
2.3-1B-Unit 2
2.3-1C-Unit 3
system instrumentation trip setpoints plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

level trip and associated reactor power/reactor power-imbalance boundaries by 1.055% for a 1% flow reduction.

The power-to-flow reduction ratio is 0.949 during single loop operation.

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 set point is reached before the nuclear over-power trip set point. 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 (2355 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 (619 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 set point 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.

Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1A. Two conditions are imposed when
2.3-1B
2.3-1C

the bypass is used:

1. By administrative control the nuclear overpower trip set point must be reduced to a value $\leq$ 5.0% of rated power during reactor shutdown.
2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The over power trip set point of $\leq$ 5.0% prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0% of rated power if none of the reactor coolant pumps were operating.

Two Pump Operation

A. Two Loop Operation

Operation with one pump in each loop will be allowed only following reactor shutdown. After shutdown has occurred, reset the pump contact monitor power level trip setpoint to 55.0%.

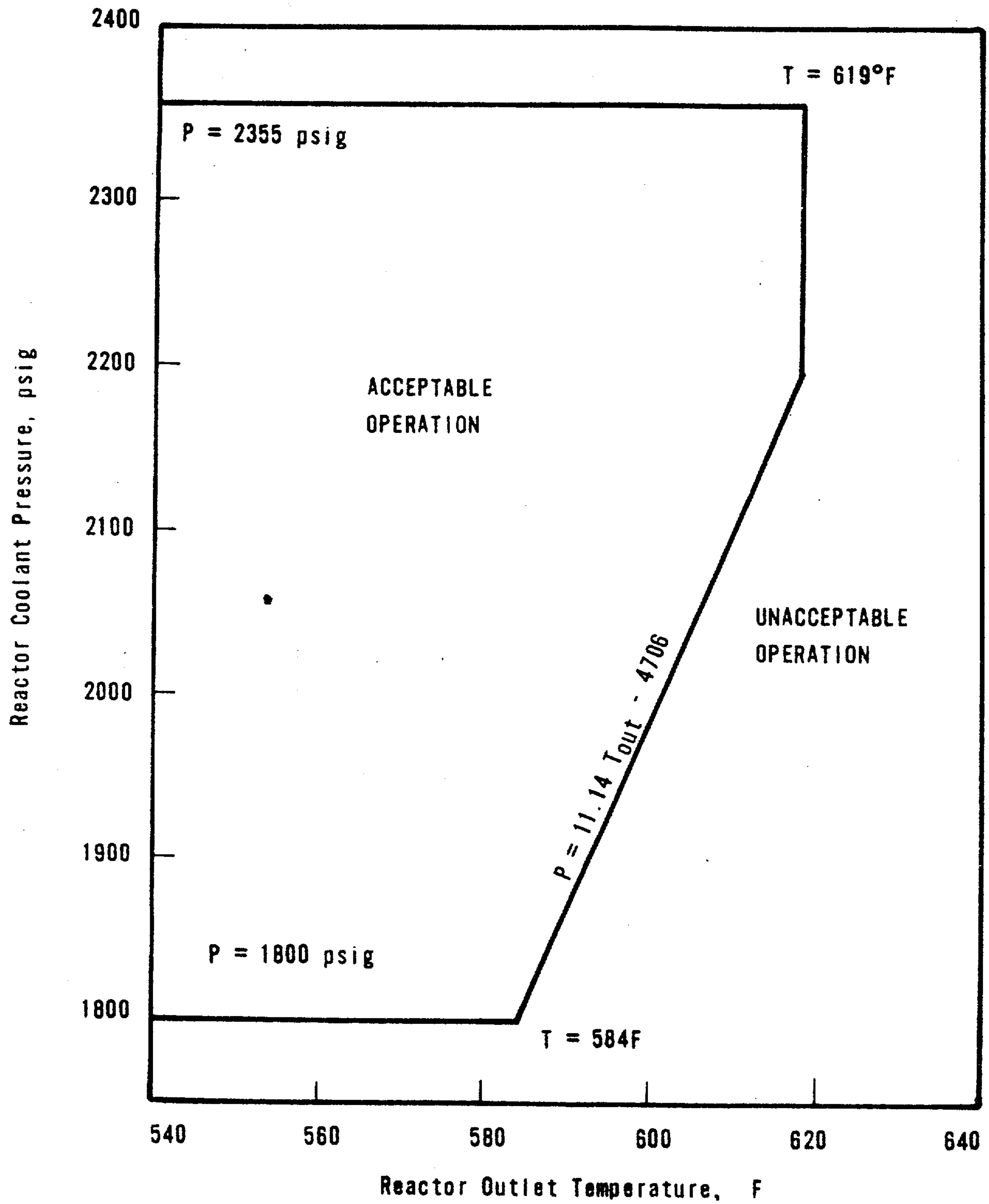
B. Single Loop Operation

Single loop operation is permitted only after the reactor has been tripped. After the pump contact monitor trip has occurred, the following actions will permit single loop operation:

1. Reset the pump contact monitor power level trip setpoint to 55.0%.
2. Trip one of the two protective channels receiving outlet temperature information from sensors in the Idle Loop.
3. Reset flux-flow setpoint to 0.949.

REFERENCES

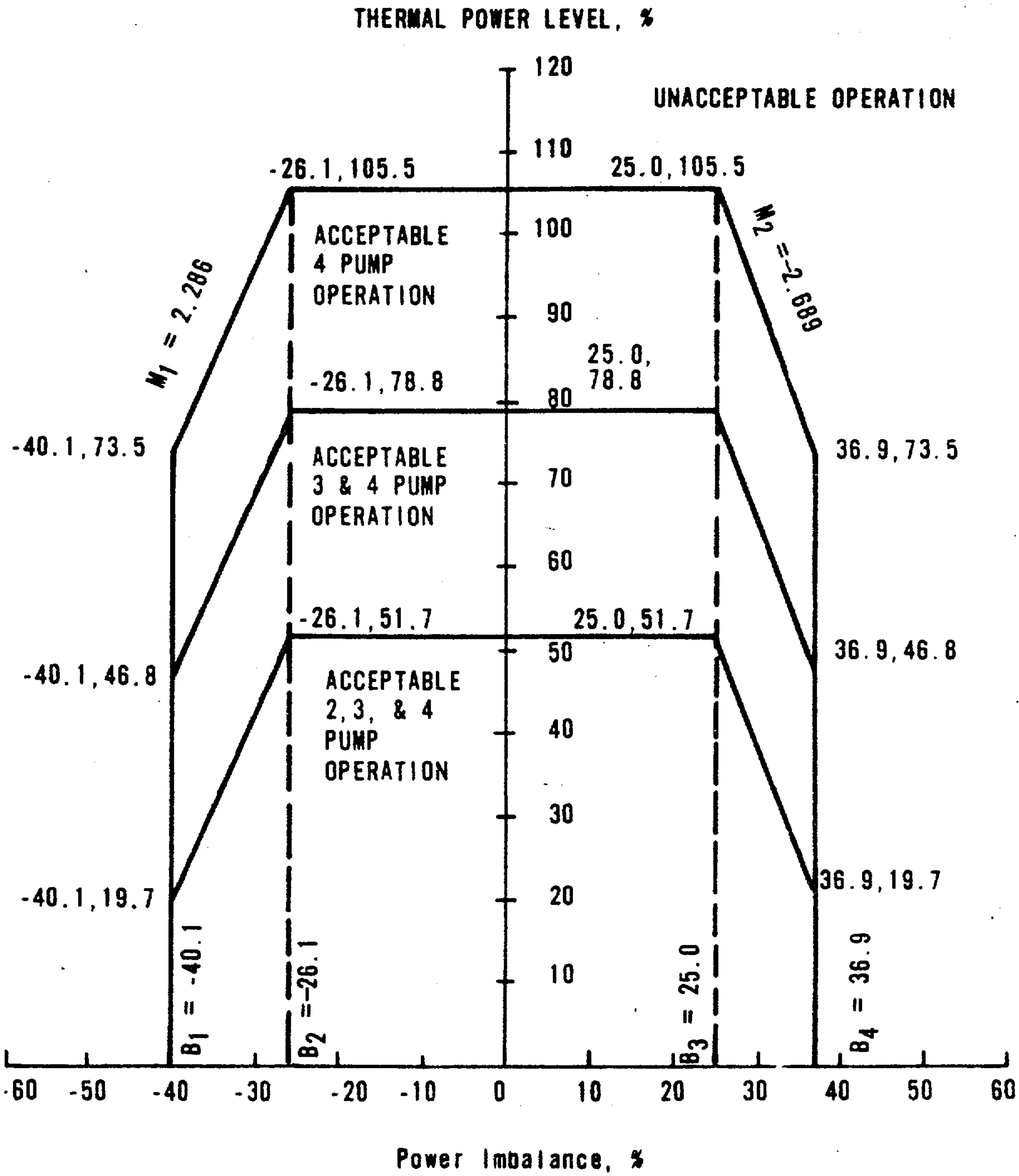
(1) FSAR, Section 14.1.2.2
(2) FSAR, Section 14.1.2.7
(3) FSAR, Section 14.1.2.8
(4) FSAR, Section 14.1.2.3
(5) FSAR, Section 14.1.2.6



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 3

OCONEE NUCLEAR STATION

Figure 2.3-1C



PROTECTIVE SYSTEM MAXIMUM
ALLOWABLE SETPOINTS
UNIT 3

OCONEE NUCLEAR STATION
Figure 2.3-2C

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 -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)	1.055 times flow minus reduction due to imbalance	1.055 times flow minus reduction due to imbalance	1.055 times flow minus reduction due to imbalance	Bypassed
3. Nuclear Power Max. Based on Pump Monitors, (% Rated)	NA	80%	55%	Bypassed
4. High Reactor Coolant System Pressure, psig, Max.	2355	2355	2355	$1720^{(4)}$
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.	619	619	619	619
8. High Reactor Building Pressure, psig, Max.	4	4	4	4

---

(1) T_{out} is in degrees Fahrenheit (^{o}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.

pump operation. Also, excepting physics tests 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, 3.5.2-4B2, and 3.5.2-4B3 (Unit 2), and 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.

d. Except for physics tests, power shall not be increased above the power level cutoff as shown on Figures 3.5.2-1A1, 3.5.2-1A2 (Unit 1), 3.5.2-1B1, 3.5.2-1B2, and 3.5.2-1B3 (Unit 2), and 3.5.2-1C1, 3.5.2-1C2, 3.5.2-1C3 (Unit 3), unless the following requirements are met.

   (1) The xenon reactivity shall be within 10 percent of the value for operation at steady-state rated power.

   (2) The xenon reactivity worth has passed its final maximum or minimum peak during its approach to its equilibrium value for operation at the power level cutoff.

3.5.2.6 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, 3.5.2-3B3, 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.

3.5.2.7 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.

3.5.2.8 For Oconee Unit 1, in the event Specifications 3.5.2.4.a or 3.5.2.5.c are not met, operation shall be restricted as follows:

a. The core thermal power shall be limited to 75 percent full power.

b. The nuclear power maximum setpoint shall be 84 percent full power.

c. The quadrant tilt shall not exceed 6.03 percent.

d. The regulating control rod insertion/withdrawal limits are specified on Figure 3.5.2-6A1

If any of the above provisions are not met within two hours, the reactor shall be in the hot shutdown condition within an additional 4 hours.

Within 25 EFPD of the date of issuance of this Specification, provide a report and analysis of the quadrant flux tilt observed and projections for the next 25 EFPD. Operation above 75% is not authorized if flux tilt is above 3.41% unless an amendment request is submitted accompanied by detailed evaluation and justification

Bases

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, 3.5.2-3B3, 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)

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2, 3, 4, 5) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

**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.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Groups 6 and 7 to be partially inserted.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding 5.10% for Unit 1. The limits shown in Specification 3.5.2.4
5.10% for Unit 2
5.10% for Unit 3
are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

The quadrant tilt and axial imbalance monitoring in Specification 3.5.2.4 and 3.5.2.6, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Operating restrictions are included in Technical Specification 3.5.2.5d to prevent excessive power peaking by transient xenon. The xenon reactivity must be beyond its final maximum or minimum peak and approaching its equilibrium value at the power level cutoff.

REFERENCES

[1]FSAR, Section 3.2.2.1.2

[2]FSAR, Section 14.2.2.2

[3]FSAR, SUPPLEMENT 9

[4]B&W FUEL DENSIFICATION REPORT

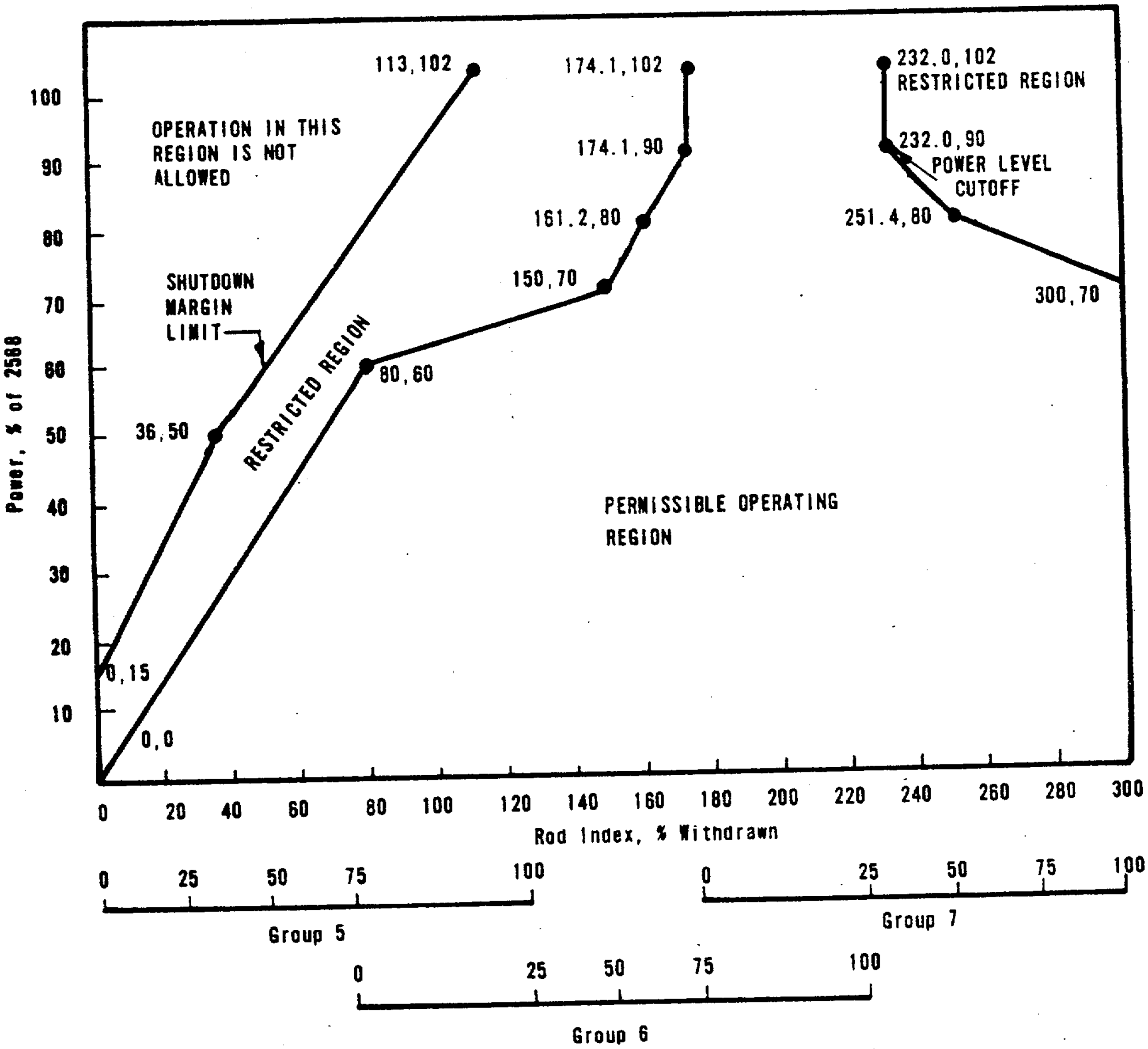
BAW-1409 (UNIT 1)

BAW-1396 (UNIT 2)

BAW-1400 (UNIT 3)

[5]OCONEE UNIT 1, CYCLE 4 RELOAD REPORT,

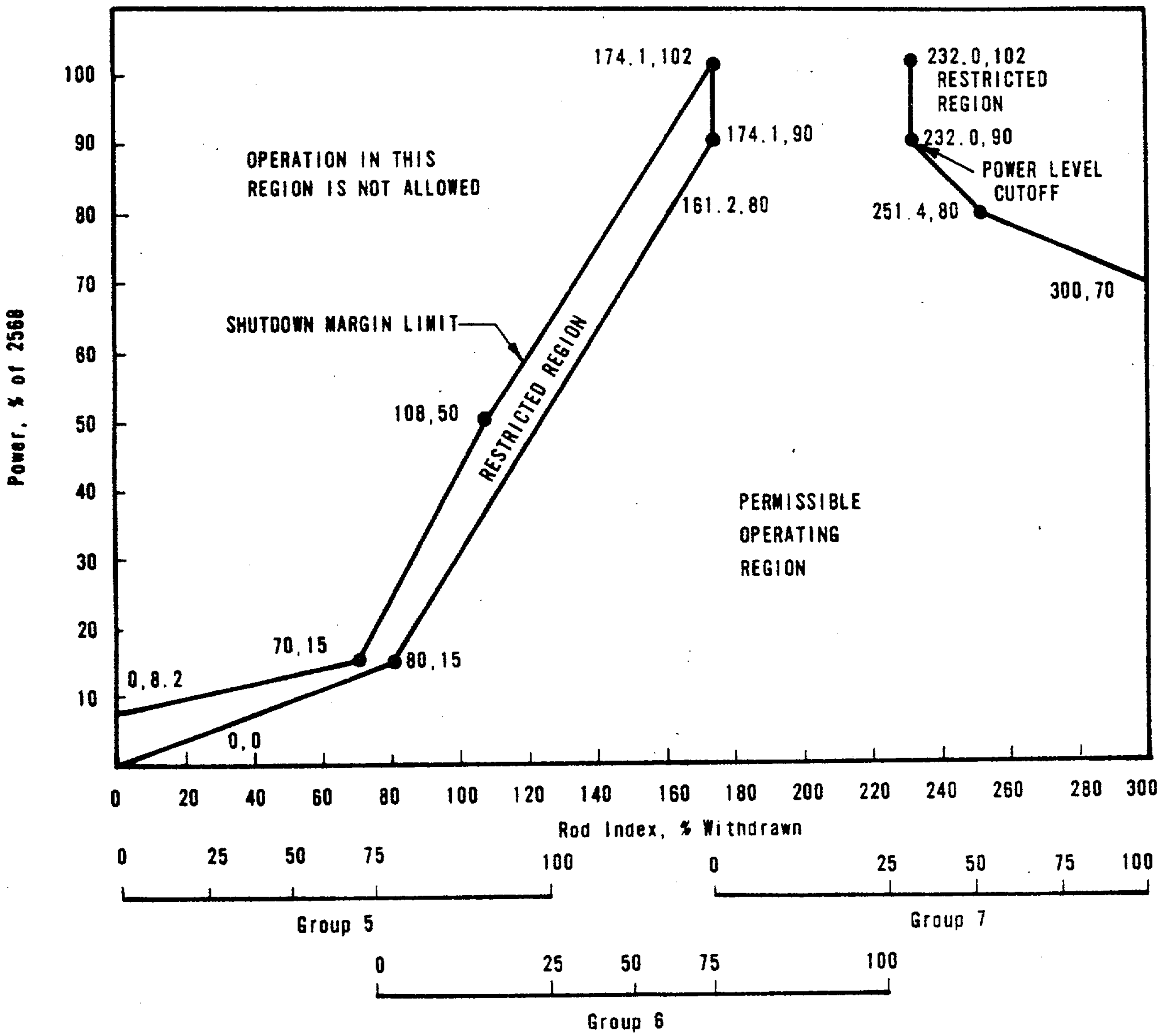
BAW-1447, March, 1977 Section 7.11



ROD POSITION LIMITS FOR
FOUR PUMP OPERATION FROM
0 TO 100 ($\pm$ 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

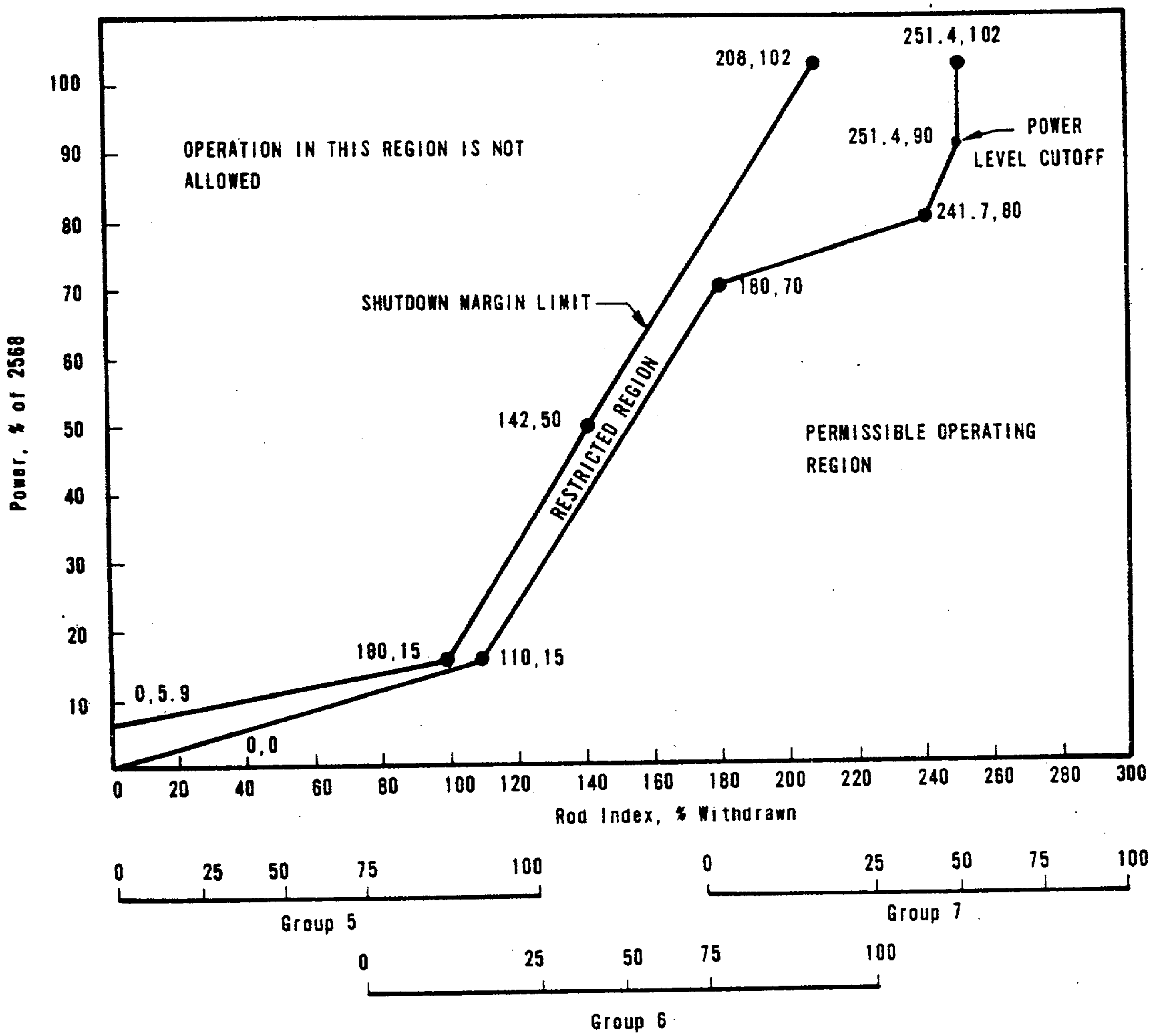
Figure 3.5.2-1C1



ROD POSITION LIMITS FOR FOUR
PUMP OPERATION FROM 100 (± 10)
EFPD TO 235 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION
Figure 3.5.2-1C2

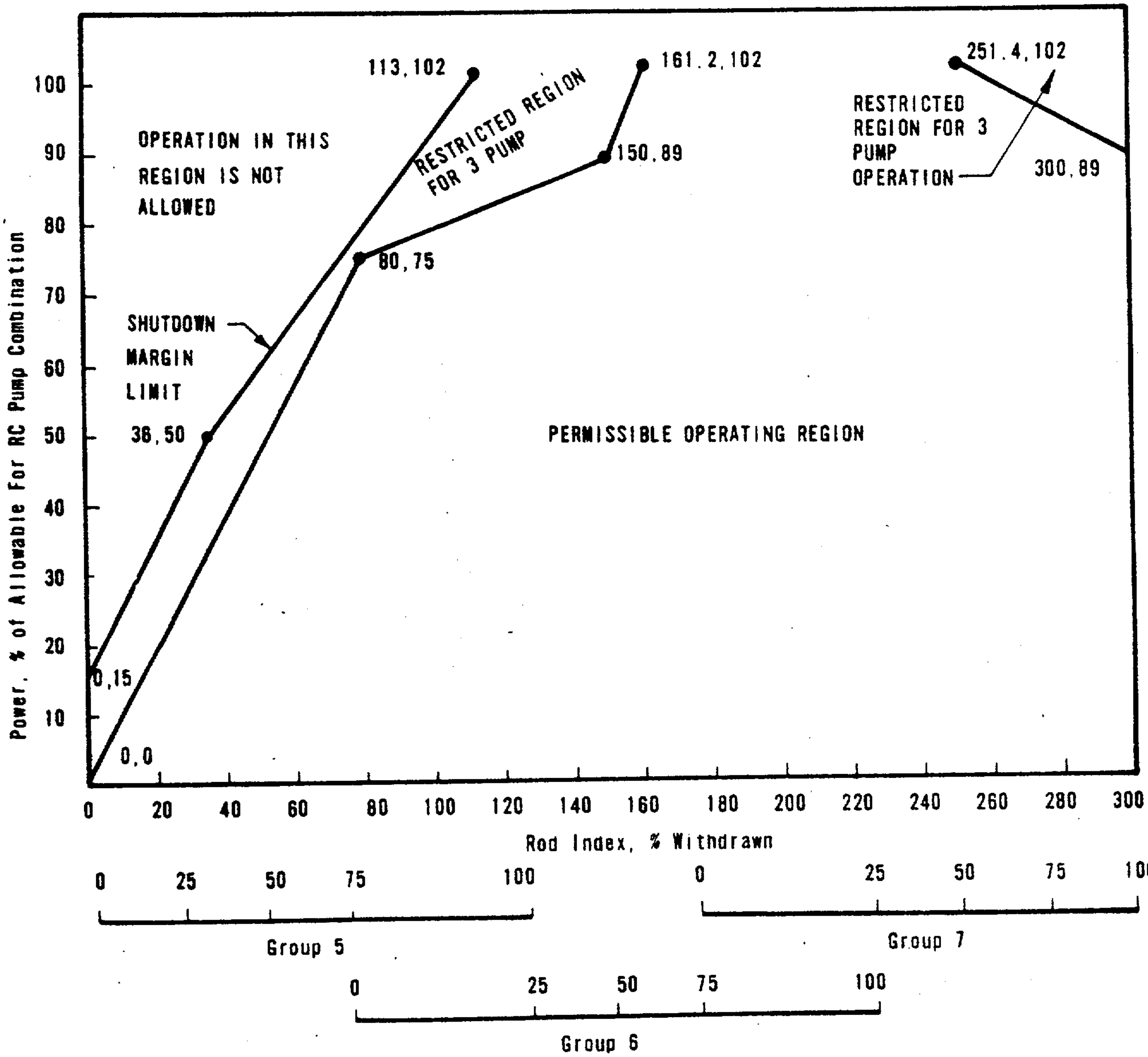
Amendment Nos. 52, 52 & 49



ROD POSITION LIMITS FOR FOUR
PUMP OPERATION AFTER 235
(+ 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-1C3

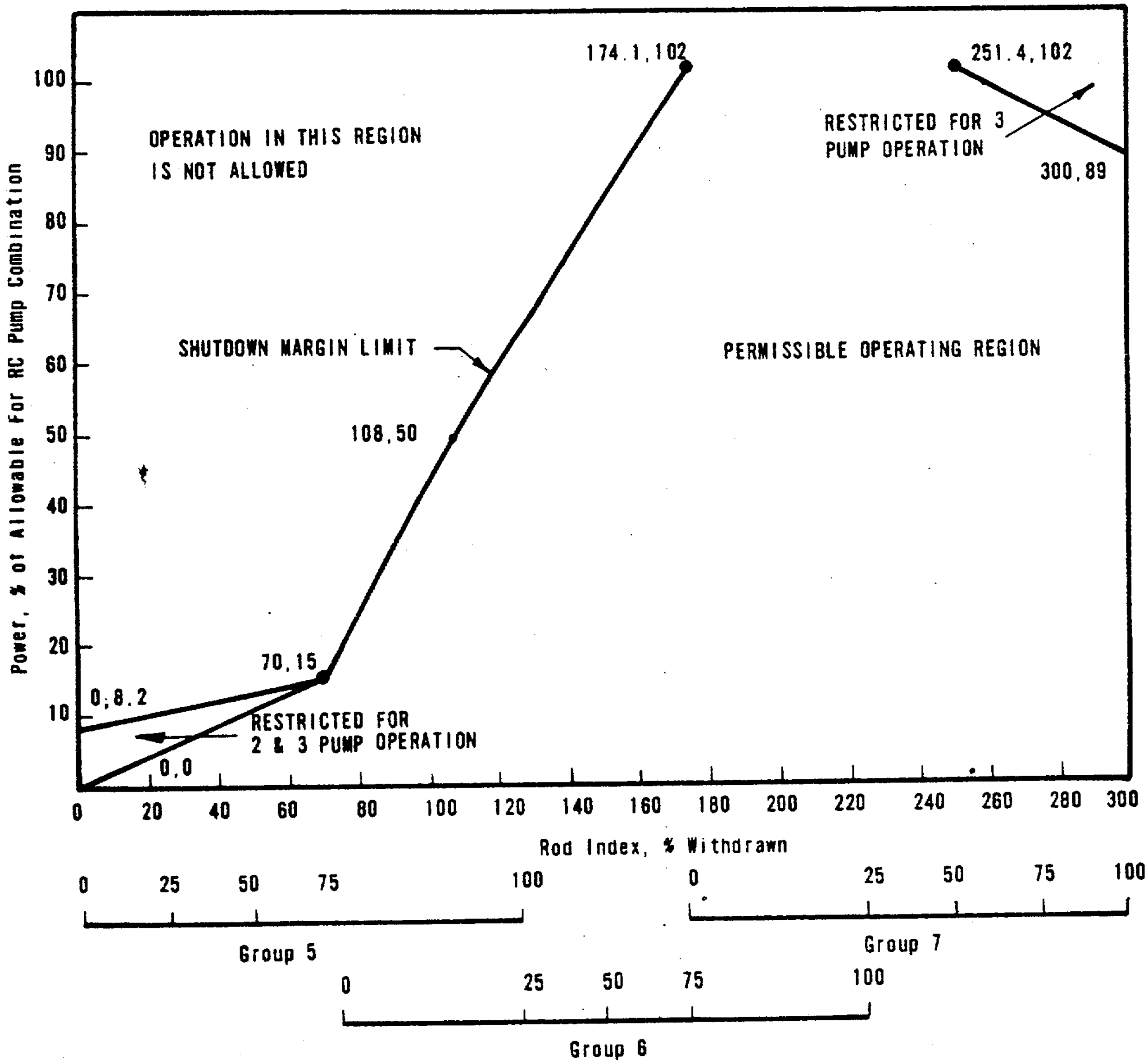


ROD POSITION LIMITS FOR TWO-
AND THREE-PUMP OPERATION
FROM 0 TO 100 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-2C1

Amendment Nos. 52, 52 & 49

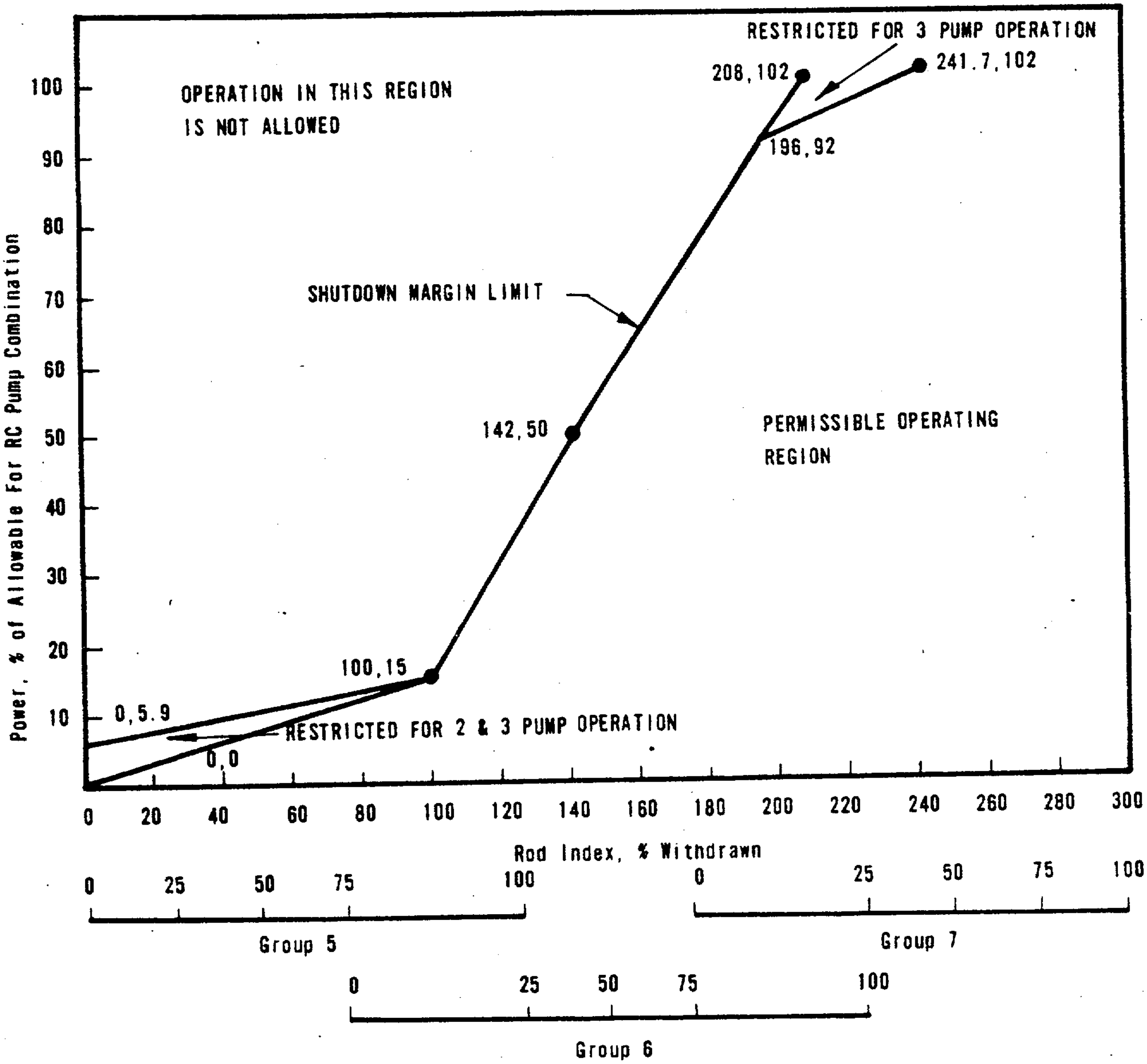


ROD POSITION LIMITS FOR TWO-
AND THREE-PUMP OPERATION FROM
100 (± 10) TO 235 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-2C2

Amendment Nos. 52, 52 & 49

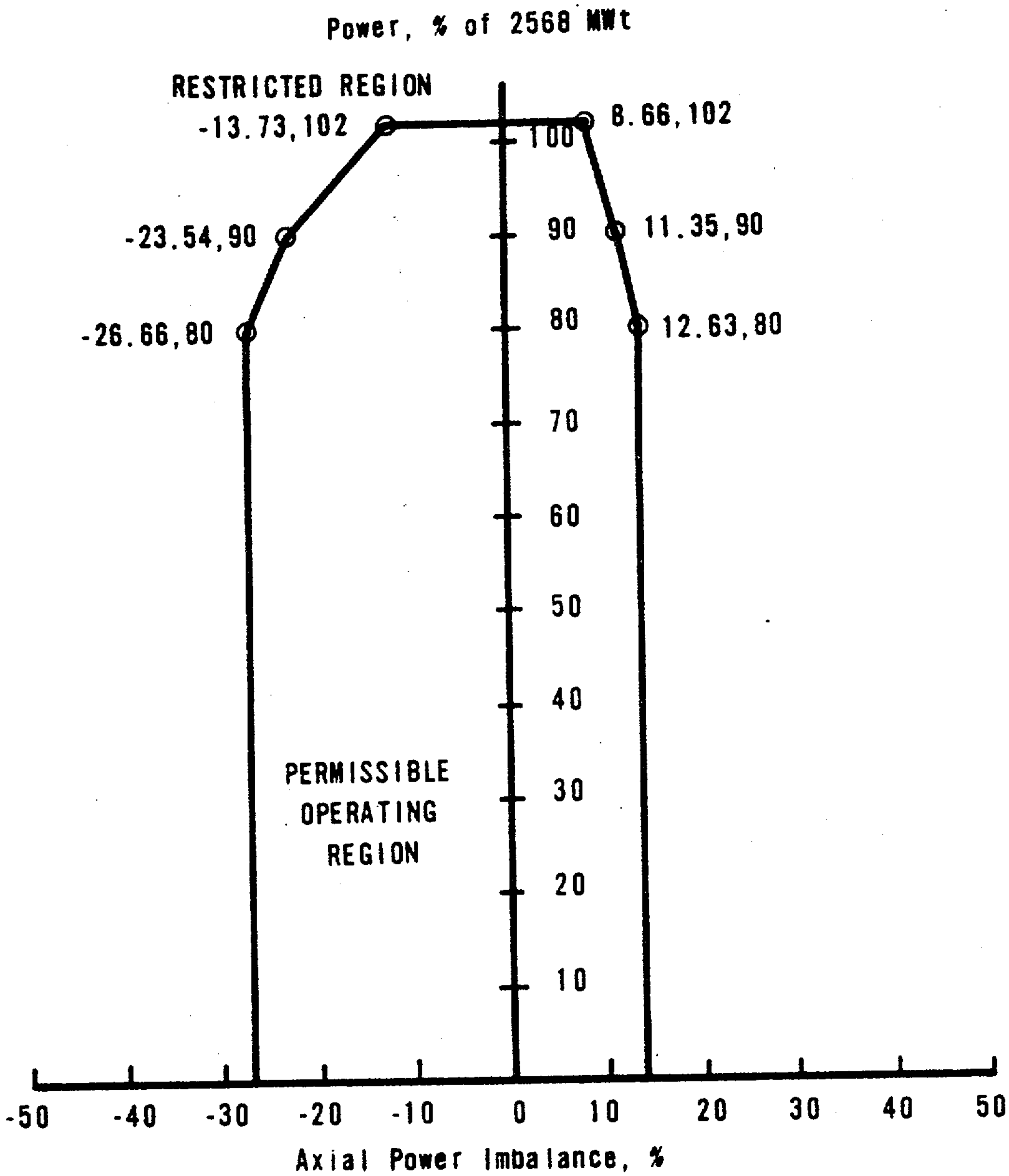


ROD POSITION LIMITS FOR TWO-
AND THREE-PUMP OPERATION
AFTER 235 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-2C3

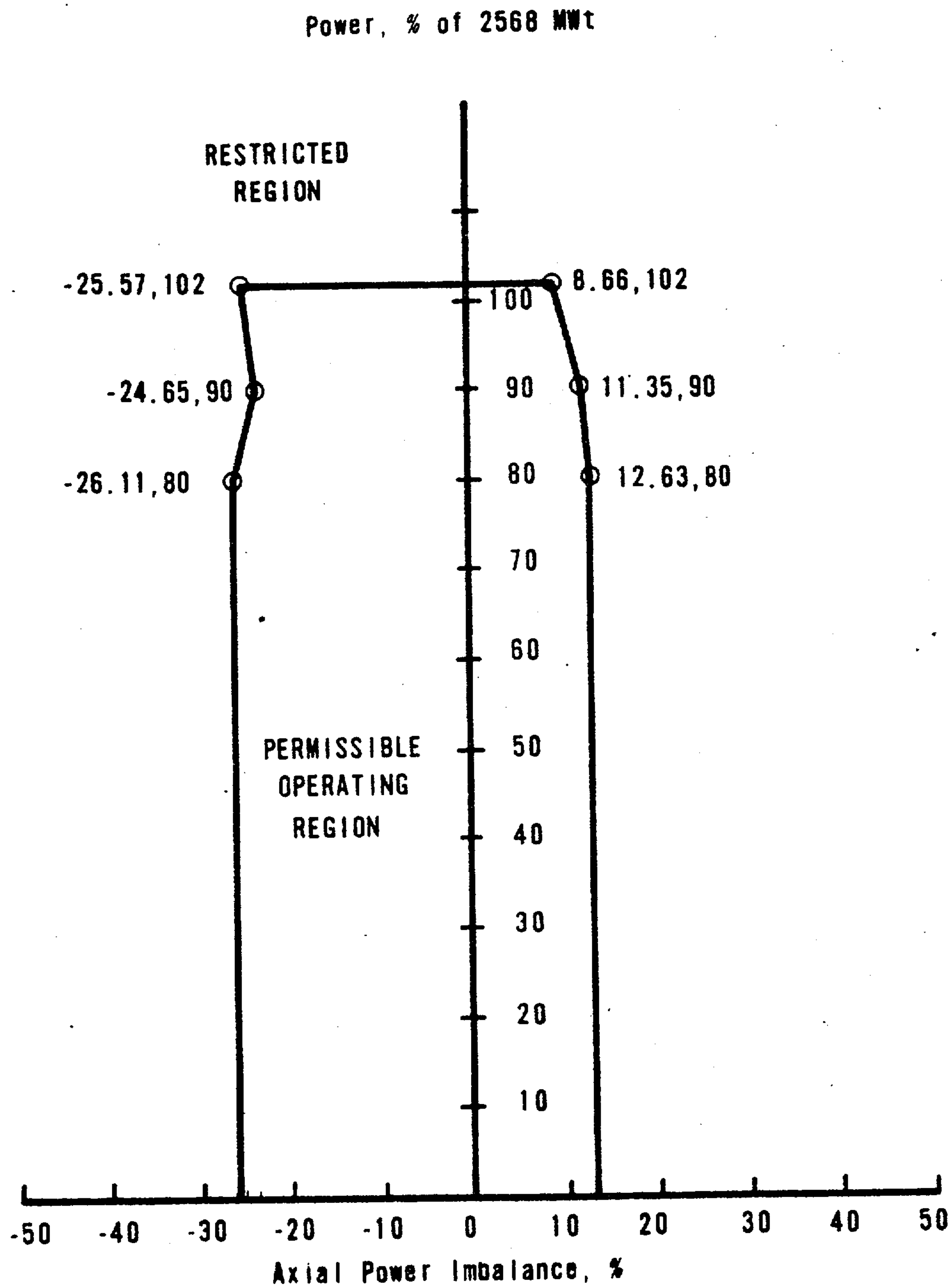
Amendment Nos. 52, 52 & 49



OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION FROM
0 TO 100 (± 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-3C1

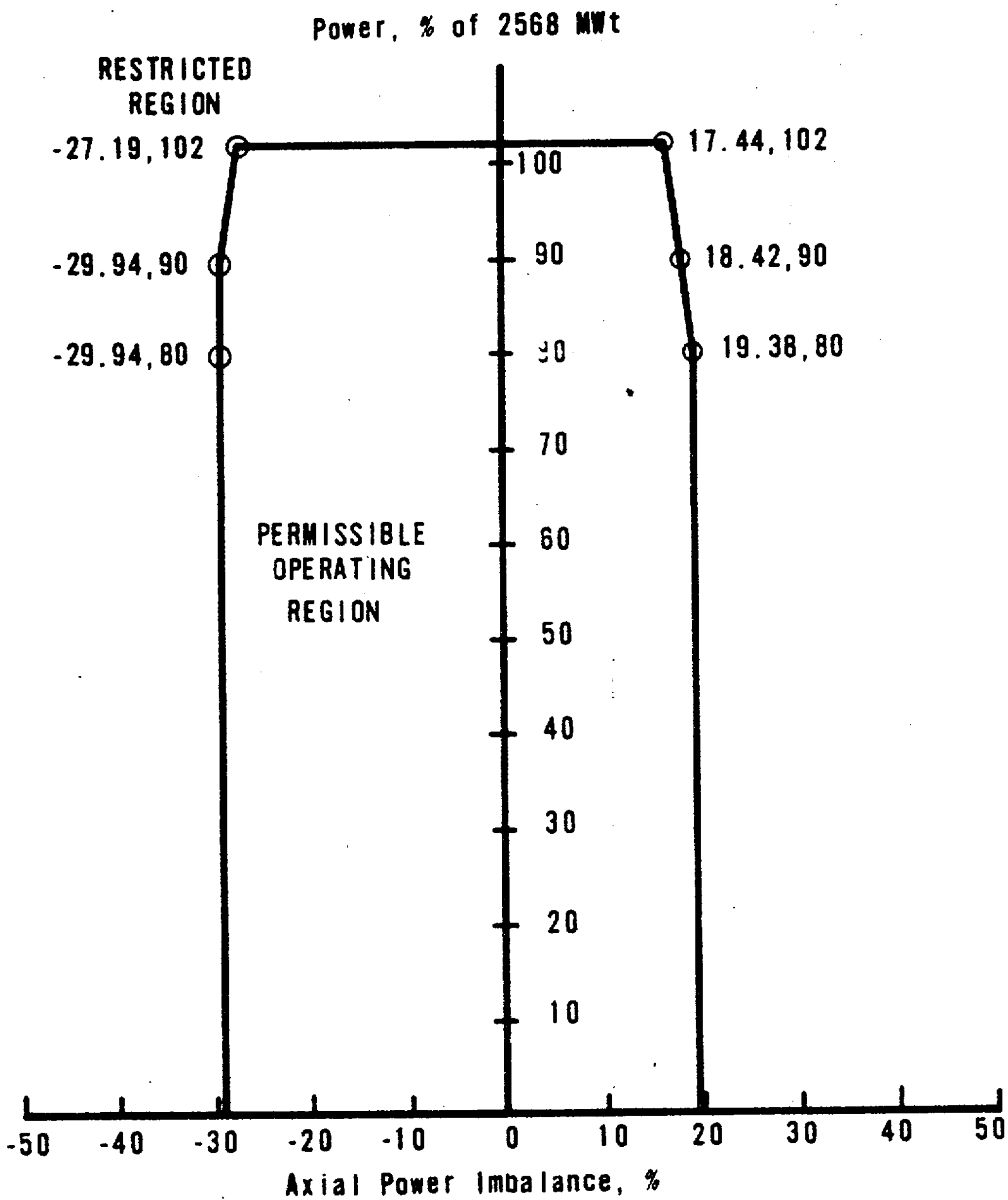


OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 100 (+) 10 EFPD TO 235 (± 10) EFPD UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-3C2

Amendment Nos. 52, 52 & 49

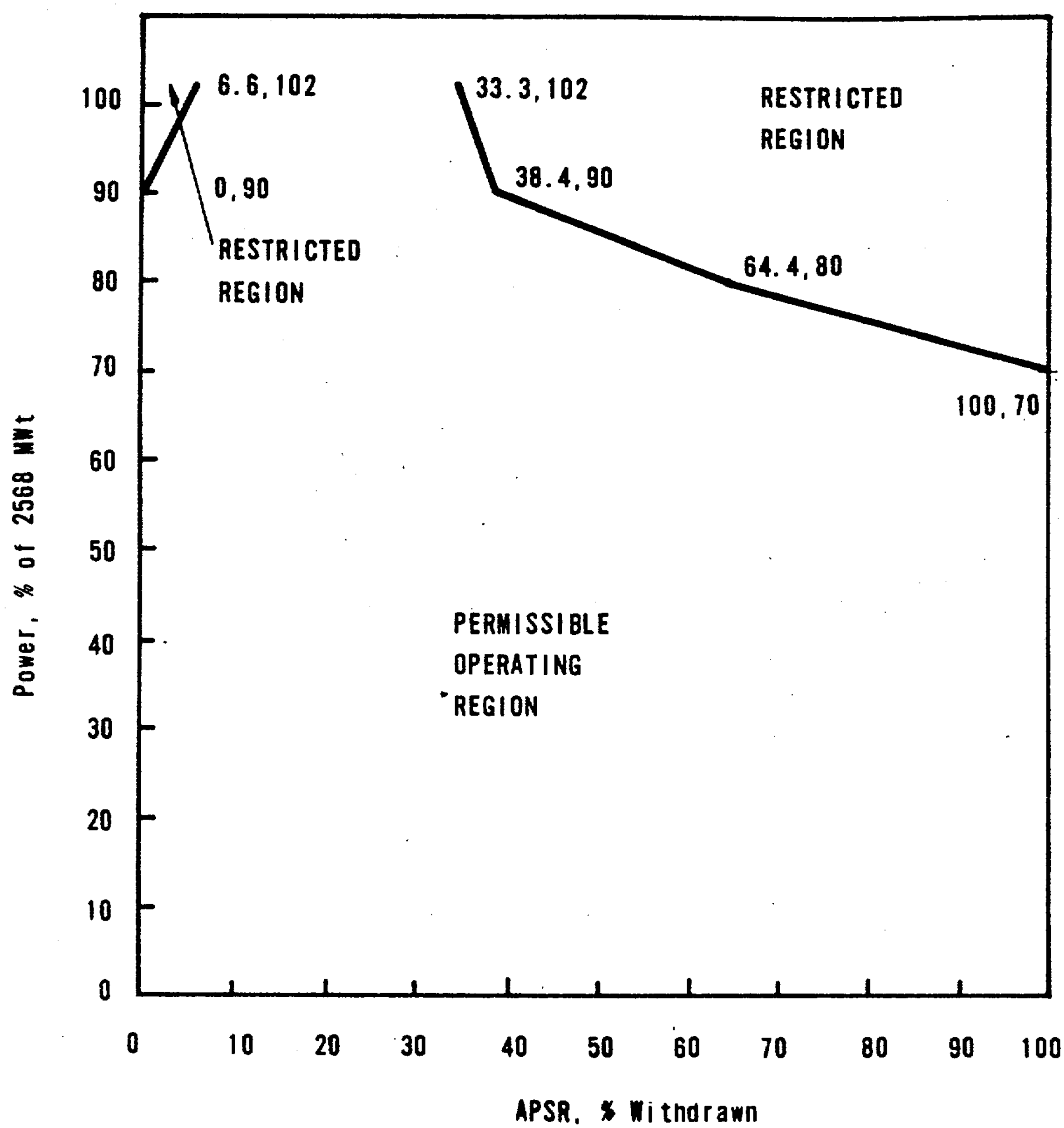


OPERATIONAL POWER IMBALANCE
ENVELOPE FOR OPERATION
AFTER 235 ($\pm$ 10) EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-3C3

Amendment Nos. 52, 52 & 49



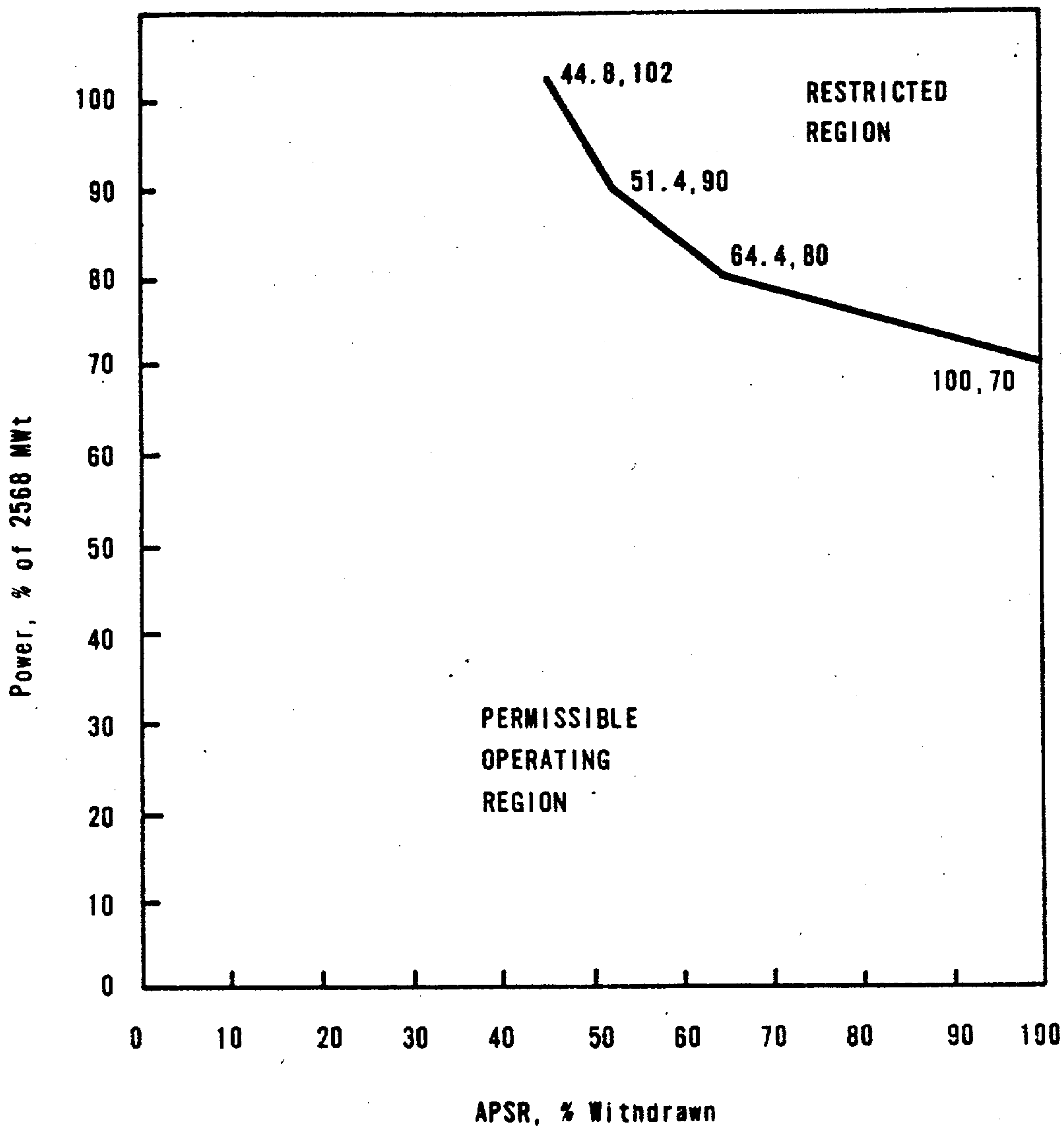
APSR POSITION LIMITS FOR OPERATION FROM 0 TO 100 $\pm$ 10 EFPD
UNIT 3

OCONEE NUCLEAR STATION

Figure 3.5.2-4C1

Amendment Nos. 52, 52 & 49

Figure 8-16. APSR Position Limits for Operation From 100 ± 10 to 235 ± 10 EFPD — Oconee 3, Cycle 3



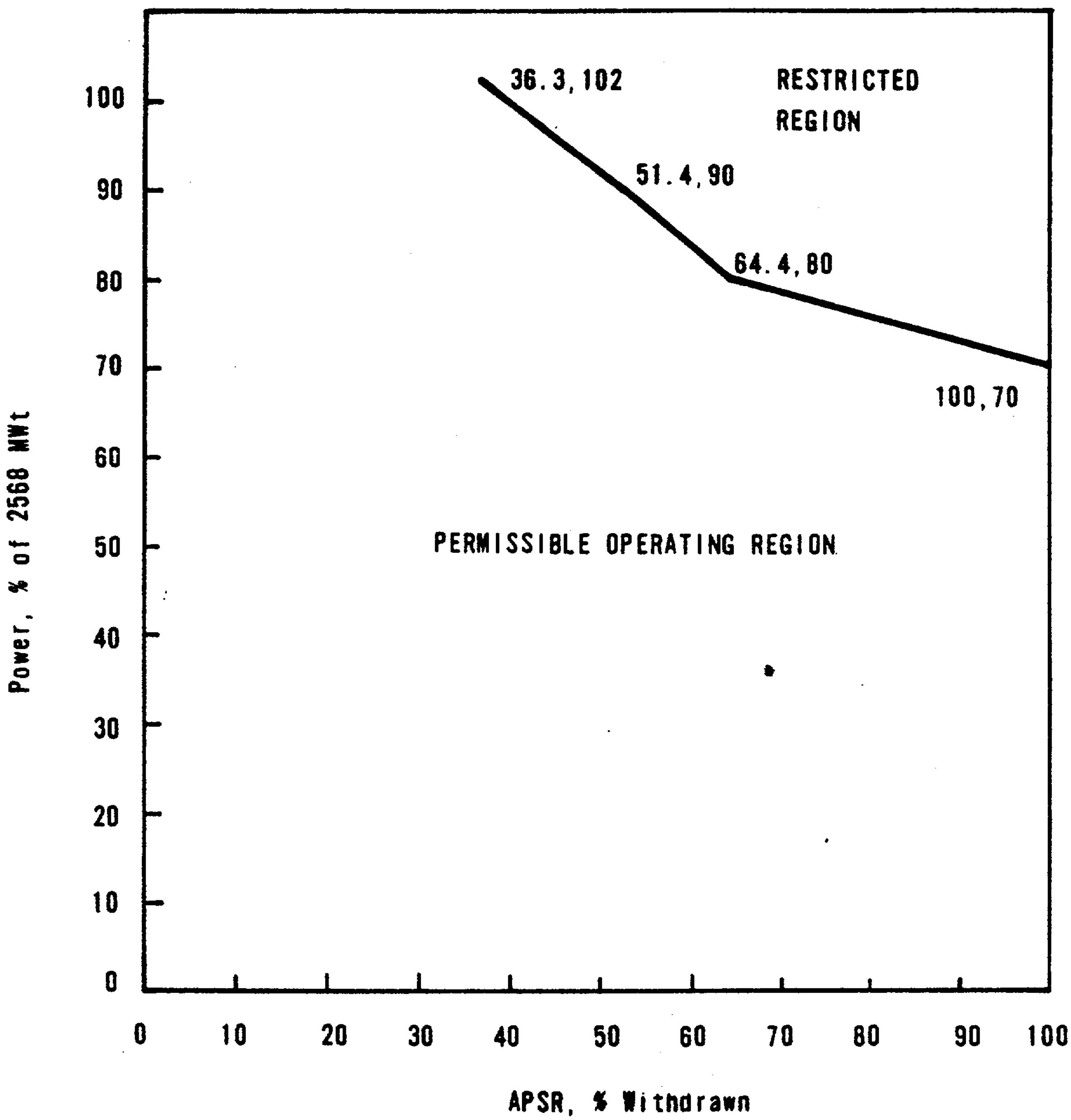
APSR POSITION LIMITS FOR OPERATION FROM 100 ± 10 TO 235 ± 10 EFPD
UNIT 3

3.5-23j

OCONEE NUCLEAR STATION

3.5.2-4C2

Figure 8-17. APSR Position Limits for Operation After 235 ± 10 EFPD — Oconee 3, Cycle 3



APSR POSITION LIMITS FOR OPERATION AFTER 235 ± 10 EFPD UNIT 3

3.5-23k

OCONEE NUCLEAR STATION

Figure 3.5.2-4C3

Table 4.1-2

MINIMUM EQUIPMENT TEST FREQUENCY

Item	Test	Frequency
1. Control Rod Movement (1)	Movement of Each Rod	Bi-Weekly
2. Pressurizer Safety Valves	Setpoint	50% Annually
3. Main Steam Safety Valves	Setpoint	25% Annually
4. Refueling System Interlocks	Functional	Prior to Refueling
5. Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly
6. Reactor Coolant System (2) Leakage	Evaluate	Daily
7. Condenser Cooling Water System Gravity Flow Test	Functional	Annually
8. High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
9. Spent Fuel Cooling System	Functional	Prior to Refueling
10. High Pressure and Low (3) Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing

(1) Applicable only when the reactor is critical.

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

(3) Operating pumps excluded.

Amendment Nos. 52, 52 & 49

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 52 TO LICENSE NO. DPR-38

AMENDMENT NO. 52 TO LICENSE NO. DPR-47

AMENDMENT NO. 49 TO LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

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

Introduction

By letter dated September 6, 1977[1] Duke Power Company (the licensee) requested changes to the Technical Specifications appended to the Oconee Unit 3 operating License for Cycle 3 operation.

Evaluation

The Oconee Unit 3 reactor core consists of 177 fuel assemblies. All of the Batch 2 fuel assemblies will be discharged at the end of Cycle 2. Five once-burned Batch 1 fuel assemblies, with an initial enrichment of 2.01 wt% ^{235}U, will be reloaded into the central portion of the core. Batches 3, 4 and 4A with initial enrichments of 3.00, 2.53 and 2.64 wt% ^{235}U, respectively, will be shuffled to new locations. Batch 5, with an initial enrichment of 3.02 wt% ^{235}U, will occupy primarily the core periphery and eight interior locations.

Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters and dimensions for Oconee 3, Cycle 3 are listed in Table 4-1 of the attachment to reference 1. Batches 3, 4 and 4A fuel are essentially the same as Batch 1 fuel. The Mark B4 fresh fuel assemblies (Batch 5) incorporate minor design modifications to end fittings and spacer grid corner cells. The latter change reduced spacer grid interaction during handling. In addition, improved dynamic impact testing methods show that the spacer grids have a higher seismic capability and thus increased safety margin.[1]

The Batch 5, 15 x 15 (Mark B-4), fuel assembly design and the Batch 1, 15 x 15 (Mark B-3), fuel assembly design have been previously reviewed and accepted by us for use in Oconee Unit 3. Also, these types of fuel assemblies are currently operating in Oconee Unit 3. The reload fuel assemblies, therefore, do not represent any unreviewed change in mechanical design from the reference cycle.

Each fuel assembly design has been taken into account in the various mechanical analyses. The Batch 3 fuel is generally limiting, because of its relatively low initial fuel pellet density, and previous incore exposure. The results of these analyses have shown that the mechanical design differences between fuels for Cycle 2 and Cycle 3 are negligible and are acceptable.

Creep collapse analyses were performed for three-cycle fuel assembly power histories. Batches 3 and 4 were analyzed using as-built data. The Batch 3 fuel is more limiting for cladding collapse due to its previous incore exposure time. The creep collapse analyses were performed based on the conditions set forth in reference 2 which have been previously found acceptable.[3] The collapse time for the most limiting assembly was conservatively determined to be more than 30,000 EFPH (effective full-power hours), which is longer than the maximum design exposure for the total of three cycles.

The Oconee 3 stress parameters were enveloped by a conservative fuel rod stress analysis. The following conservatisms with respect to Oconee 3 fuel were used in the analysis: lower post-densification internal pressure, lower initial pellet density, higher system pressure, and higher thermal gradient across the cladding.

For design evaluation, the primary stress must be less than two-thirds of the minimum specified unirradiated yield strength, and all stresses must be less than the minimum specified unirradiated yield strength. In all cases, the margin is in excess of 30%.

The fuel design criteria specify a limit to the cladding plastic circumferential strain of 1.0%. The pellet design is established for plastic cladding strain of less than 1% at maximum design local pellet burnup and heat generation rate values, which are considerably higher than the values for Oconee 3 fuel. This will result in an even greater margin than the analysis demonstrated. The strain analysis is also based on the maximum manufacturing specifications values for the fuel pellet diameter and density and the lowest permitted manufacturing specifications tolerance for the cladding internal diameter.

The linear heat generation rate (LHGR) capabilities are based on centerline fuel melt and were established using the TAFY-3 code[4] with fuel densification to 96.5% of theoretical density. Two of the Batch 5 fuel assemblies were loaded with fuel pellets which have a different nominal density (91% TD) and diameter than the Batch 5 fuel assemblies. Based on these characteristics a LHGR of 19.74 KW/ft has been established for these two fuel assemblies. These assemblies will be placed in non-limiting locations during their entire core life. For Cycle 3, the licensee has specified that if these fuel assemblies and placed in locations M-14 and E-2, they will not experience LHGR's greater than 19.15 KW/ft.

All the other fuel assemblies in the Cycle 3 core are thermally similar. The fresh Batch 5 fuel inserted for Cycle 3 operation introduces no significant difference in fuel thermal performance relative to the other fuel remaining in the core, and their LHGR limit has been established as 20.15 KW/ft.

The power spike model used for Cycle 3 analyses is the same as that used for Cycle 2. The power spike factor and gap size were based on unirradiated Batch 4 and Batch 5 fuel (94.0% TD) with an assumed enrichment of 3.0 wt% ^{235}U. These values are conservatively high for Batch 1 and Batch 3 fuel.

The thermal analysis of the fuel rods assumed an in-reactor densification to 96.5% of theoretical density. The analytical methods used are the same as those for Cycle 2.[5] These analyses were based on the lower tolerance limit of the fuel density specification and assumed isotropic diametral shrinkage and anisotropic axial shrinkage resulting from fuel densification.

The Batch 5 fuel assemblies are not new in concept, nor do they use different materials. Therefore, the chemical compatibility of all possible fuel-cladding-coolant assembly interactions for the Batch 5 fuel assemblies are identical to those of the present fuel.

This fuel as proposed for reload in Oconee 3 has had considerable operating experience. Because of this experience, the similarity of the Batches 1 and 5 fuel and because the fuel assemblies for Cycle 3 operation will not exceed ay design life limits, we conclude that the fuel mechanical design and the fuel thermal design for Cycle 3 operation is acceptable.

Nuclear Analyses

Table 5-1 of the attachment to reference 1 compares the core physics parameters of Cycles 2 and 3. The values for both cycles were generated using PDQ07. Since the core has not yet reached an equilibrium cycle, differences in core physics parameters are to be expected between the cycles. The shorter Cycle 2 produced a smaller cycle differential burnup than is expected for Cycle 3. The accumulated average core burnup will be higher in Cycle 3 than in Cycle 2 because of the presence of the once-burned fuel of Batches 1, 3, 4 and 4A.

The critical boron concentrations for Cycle 3 are higher than for Cycle 2 because of a higher fuel enrichment in Cycle 3. The control rod worths are sufficient to maintain the required shutdown margin for Cycle 3. The maximum stuck rod worths for Cycle 3 are less than those in Cycle 2. The adequacy of the shutdown margin with Cycle 3 rod worths has been demonstrated analytically. The shutdown calculations conservatively used a poison material depletion allowance and 10% uncertainty on net rod worth.

The same calculational methods and design information were used to obtain the nuclear design parameters for Cycles 2 and 3. In addition, for Cycle 3 there are no significant operational procedure changes from the reference cycle procedures with regard to axial or radial power shape control, xenon control or tilt control.

In view of the above and the fact that startup tests (to be conducted prior to power operation) will verify that the significant aspects of the core performance are within the assumptions of the safety analysis, we find the licensee's nuclear analyses for Cycle 3 to be acceptable.

Thermal-Hydraulic Analyses

The major acceptance criteria which are used for the thermal-hydraulic design are specified in Standard Review Plan (SRP) 4.4. These criteria establish acceptable limits on departure from nucleate boiling (DNB). The thermal-hydraulic analyses for Oconee Unit 3 Cycle 3 reload were made using previously approved models and methods. Certain aspects of the thermal-hydraulic design are new for the Cycle 3 core and are discussed below.

The thermal-hydraulic design evaluation in support of Cycle 3 operation used the methods and models described in references 5, 6 and 7. Cycle 3 analyses and resulting setpoints have been based on 106.5% of the design reactor coolant (RC) system flow rate. Cycle 2 analyses used 107.6% of design flow based on a measured flow value of 110.0%. The reduced flow rate has been selected for Cycle 3 analyses to provide consistency with Oconee Units 1 and 2.

The core configuration for Cycle 3 differs slightly from that of Cycle 2 in that the Batch 2 fuel removed at the end of Cycle 2 is the Mark B-3 fuel assembly design, and the fresh Batch 5 fuel inserted for Cycle 3 is the Mark B-4 assembly design. Mark B-4 assemblies differ from the Mark B-3 primarily in the design of the end fitting, which results in a slight reduction in flow resistance for the B-4 design. No credit was taken in the analyses for the increased flow to the Mark B-4 assemblies, located in the hottest core locations, as a result of slight changes in the core flow distribution or for the increase in the system flow resulting from the reduction in total core pressure drop. However, the slight reduction in flow rate of the Mark B-3 assemblies (because of the lower flow resistance of the B-4 assemblies) was considered.

The BAW-2 CHF correlation[8] was used for thermal-hydraulic analysis of Cycle 3. This correlation has been reviewed and approved for use with the Mark B fuel assembly design.[9]

The effect of fuel densification on minimum DNBR is primarily a result of the reduction in active fuel length, which increases the average heat flux. The Cycle 3 DNBR analysis was based on a cold densified active length of 140.2 inches, a value selected to apply generically to a number of B&W plants. This is a conservative method of applying the densification effect since all the fuel assemblies in Cycle 3 have longer densified lengths and because no credit is taken for axial thermal expansion of the fuel column.

The potential effect of fuel rod bow on DNBR can be considered by incorporating suitable margins into DNB-dominated core safety limits and reactor protection system setpoints. The maximum rod bow magnitude would be calculated from the equation $\sigma_b = 11.5 + 0.069\sqrt{BU}$, where σ_b is the rod bow magnitude in mils and BU is the burnup is MWD/mtU. The resultant DNBR penalty based on the maximum predicted assembly burnup at the end of Cycle 3 is approximately 6.0%. However, since this rod bow model has not yet been found acceptable, the maximum rod bow magnitude was calculated using the NRC approved interim model, $\Delta C/C_0 = 0.065 + 0.001449\sqrt{BU}$ where ΔC is the rod bow magnitude (in mils) and C_0 is the initial gap. The resultant DNBR penalty, based on the maximum predicted assembly burnup at end of Cycle 3, is 11.2%.

The pressure-temperature limit curve shown in Figure 2.1-1C of the Technical Specifications provides the basis for the variable low-pressure trip setpoint. The curve shows for all modes of reactor operation locus of points for which the calculated minimum DNBR is equal to 1.30 (BAW-2) plus the margin required to offset an 11.2% DNBR reduction due to rod bow. The specific credits used in this analysis to account for rod bow are as follows:

		% DNBR credit
Credit for rod bow penalty already included in analysis	=	10.2
Credit for flowarea reduction factor in analysis	=	1.0
Credit for plant excess flow (3.5% available)	=	none claimed
Total		11.2

The flux/flow trip setpoint was determined by analyzing an assumed two-pump coastdown starting from an initial indicated power level of 102% plus flux measurement and heat balance errors (equal to 108% full power in core). The specific credits used in this analysis to account for rod bow are as follows:

		% DNBR credit
Credit for rod bow penalty already included in analysis	=	5.8
Credit for flow area reduction factor in analysis	=	1.0
Credit for 2% (3.5% available) excess RC flow	=	4.4
Total		11.2

In summary, a reactor coolant flow rate based on actual measured flow with uncertainties and conservatisms was used in the Oconee Unit 3 Cycle 3 thermal hydraulic analyses. The licensee has also assured us that there will be sufficient margin in the reactor coolant flow rate (at least 108.5% of design) to compensate for the difference between the approved and the not yet approved rod bow models. Based on our review, we find that the licensee has included appropriate conservatisms in the analyses and that the proposed Technical Specifications provide assurance that the criteria of SRP 4.4 will be met. Therefore, we conclude that the thermal-hydraulic analyses are acceptable.

Accident and Transient Analyses

The accident and transient analyses as provided by the licensee demonstrate that the Oconee FSAR analyses conservatively bound the predicted conditions of the Oconee Unit 3 Cycle 3 core and are, therefore acceptable. The licensee has stated that each FSAR accident analysis has been examined, with respect to changes in Cycle 3 parameters, to determine the effects of the reload and to ensure that performance is not degraded during hypothetical transients. The core thermal parameters used in the FSAR accident analyses were design operating values based on calculated values plus uncertainties. FSAR values of core thermal parameters were compared with those calculated in the Cycle 3 analyses. For each accident of the FSAR, a discussion and comparison of the key parameters from the FSAR and Cycle 3 was provided by the licensee to show that the initial conditions of the transient are bounded by the FSAR analysis. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in the Oconee Unit 3 fuel densification report.(5) Since Cycle 3 reload fuel assemblies contain fuel rods with theoretical density higher than those considered there, the conclusions derived in that report are valid for Oconee Unit 3 Cycle 3. Calculational techniques and methods for Cycle 3 analyses remain consistent with those used for the FSAR. No new dose calculations were performed for the reload report. The dose considerations in the FSAR are based on maximum peaking and burnup for all core cycles; therefore, the dose considerations are independent of the reload batch.

A review of the ECCS U-baffle pressure drop error has been performed and documented in reference 10. The review considered a reanalysis of the reactor coolant system pressure loss characteristics and the effects and ECCS performance. The review found the current ECCS performance analysis acceptable for all three Oconee units. Reference 10 also found that a new surveillance testing program of the reactor internals vent valves is acceptable for all three Oconee units. The review considered the impact of these changes on ECCS performance and the adequacy of the surveillance techniques.

Startup Tests

A startup program will be conducted to verify that the core performance is within the assumptions of the safety analyses and provide the necessary data for continued plant operation. The startup test program is similar to that previously approved for Cycle 2 operation. Within 90 days following completion of physics testing the licensee will provide a summary of the test program results. This startup test program and reporting schedule are acceptable.

Environmental Consideration

We have determined that these 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 these 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, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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.

Date: November 21, 1977

REFERENCES

1. Letter from W. O. Parker, Jr., (Duke Power Company) to Edson G. Case, (NRC) dated September 6, 1977.

2. Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084, Rev. 1, Babcock & Wilcox, November 1976.

3. Letter from A. Schwencer (NRC) to J. F. Mallary (B&W) dated Jan. 29, 1975

4. C. D. Morgan and H. S. Kao, TAFY - Fuel Pin Temperature and Gas Pressure Analysis, BAW-10044, Babcock & Wilcox, May 1972.

5. Oconee 3 Fuel Densification Report, BAW-1399, Babcock & Wilcox November 1973.

6. B. J. Buescher and J. W. Pergram, Babcock & Wilcox Model for Predicting In-Reactor Densification, BAW-10083P, Rev. 1, Babcock & Wilcox, November 1976.

7. Oconee Nuclear Station, Units 1, 2, and 3, Final Safety Analysis Report, Docket Nos. 50-269, 50-270, and 50-287.

8. Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, Babcock & Wilcox, June 1976.

9. Letter from J. Stolz (NRC) to K. E. Surke (B&W), dated April 15, 1976.

10. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment Nos. 45, 45 and 42 to Facility License Nos. DPR-38, DPR-47 and DPR-55, Duke Power Company, Oconee Nuclear Station Unit Nos. 1, 2 and 3, July 29, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 52 , 52 and 49 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, issued to Duke Power Company which revised Technical Specifications for operation of the Oconee Nuclear Station Unit Nos. 1, 2 and 3, located in Oconee County, South Carolina. The amendments are effective as of their date of issuance.

The amendments revise the Technical Specifications to establish operating limits for Unit 3 Cycle 3 operation.

The application for these amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated September 6, 1977, (2) Amendment Nos. 52 , 52 and 49 to Licenses Nos. DPR-38, DPR-47 and DPR-55, respectively, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 and at the Oconee County Library, 201 South Spring, Walhalla, South Carolina 29691. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 21st day of November 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors